APPENDIX 1.A
THERMAL-HYDRAULICS ANALYSIS OF SPENT FUEL POOL HEATUP

1.0 INTRODUCTION

Spent fuel heatup analyses involving postulated loss of coolant or loss of cooling accidents were performed to support the decommissioning rulemaking effort. The staff developed spent fuel heatup models for boiling water reactors (BWRs) and pressurized water reactors (PWRs) that take into account the decay power and configuration of the fuel in the spent fuel pool (SFP) and the air flow in the building surrounding the SFP. Discussions in this Appendix include an explanation of how the thermal-hydraulic estimates of heatup times are used in the rest of the report. Important assumptions in the analysis (e.g., oxidation rates, ventilation rates, ignition temperatures, and fuel burnup), are discussed in the context of estimating the heatup and boiloff times for the SFP inventory, and heatup and ignition timing of fuel cladding to a zirconium fire. Limitations in the state-of-the-art in performing these calculations are also discussed. Technical bases are given in Appendix 1.B. for spent fuel cladding temperature criteria used to estimate when significant fission product releases occur in decommissioning plant SFP accidents.

The time it takes to uncover spent fuel because of pool inventory boiloff is an input to the risk assessment. This information is used to estimate human error rates and repair time available for fuel handlers faced with inoperative equipment. Calculations for heatup and boiloff of SFP inventory involve heatup of the pool water to boiling followed by boil down of the inventory to within 3 feet of the top of the spent fuel. The time it takes to heatup fuel cladding to zirconium fire/fission product release temperatures is important in the establishment of consequence estimates regarding how evacuations would proceed following either loss of inventory or cooling to an SFP.

1.1 SFP Inventory Heatup and Boiloff

The staff conducted a thermal-hydraulic assessment of the SFP for various scenarios involving loss of pool cooling and loss of inventory in support of the risk assessment. These calculations resulted in the estimates of heatup and boiloff times for an SFP, which are displayed in Table A1-1. These estimates are straight forward energy balance calculations based on representative decay heat levels, representative volumes in the SFPs, and standard water properties. The end state used for these accident sequences was an SFP water level 3 feet above the top of the fuel. This simplified end state was used because recovery below this level, given failure to recover before reaching this level, was judged to be unlikely given the significant radiation field in and around the SFP at lowered water levels. The simplified end state provides a slightly conservative, but adequate measure to determine time frames important to human error and recovery estimates. It also greatly simplifies the analysis by eliminating the need to accurately model the complex heat transfer mechanisms and chemical reactions that are occurring in the fuel assemblies as they are being slowly uncovered.

Assumptions

For the purpose of the boiloff analysis, the fuel burnup is assumed to be 62.5 GWD/MTU with a 2 year cycle time. The decay heat at this value of burnup is an extrapolation of the decay heat

tables from NUREG/CR-5625 (Ref. 1). The BWR pool is assumed to hold 4,200 9x9 fuel assemblies. The BWR pool surface area is assumed to be 105.7 square meters. The PWR pool is assumed to hold 965 17x17 fuel assemblies. The PWR pool surface area is assumed to be 61.3 square meters. The pools are assumed to have a water depth of 11.54 meters and are assumed to be at an initial temperature of 30 °C. An estimated water volume fraction of 0.5 of water in the racks and assemblies is used in the calculations. The specific heat of water is assumed to be constant at 4,200 J/kg for the heatup calculation. Temperature dependent properties were used for steel, zircaloy, and UO_2. The enthalpy change because of vaporization used in the boiloff calculation is 2,257 KJ/kg. The results of these calculations are shown in Table A1-1. The results show that the progression of heatup and boiloff accidents take place on a very long time scale.

Table A1-1 Heatup and Boiloff Times from Normal Pool Level to Three Feet Above Active Fuel

Decay Time	**Boiloff Time (hours)**	
	PWR	**BWR**
60 days	100	145
1 year	195	253
2 years	272	337
5 years	400	459
10 years	476	532

1.2 Spent Fuel Heatup Analyses

Once the spent fuel is uncovered (partially or fully), it would begin to heat up. The only significant heat source initially would be the decay heat. Later, at high cladding temperatures, additional heat is added by the exothermic oxidation of the zirconium fuel rod cladding. The staff's review of previous analysis of spent fuel heatup in air concluded that changes to SFP storage practices indicated a need for significant revisions to previous analysis assumptions. Accordingly, new analyses were performed for this study to predict fuel rod heatup, in order to better represent current decommissioning plant operation and SFP storage practices. These analyses had two basic objectives:

1. Determine the heatup time of fuel cladding from 30 °C to 900 °C (the temperature at which the onset of significant fission product release is expected).

2. Determine a generic critical decay time (the time after shutdown that a release of fission products is no longer possible).

The analyses were to include these considerations:

1. Partial draindown concerns (i.e., cases where the fuel is only partially uncovered).

2. Study of the global flow pattern in the SFP building to determine the applicability of approximations used in previous calculations.

3. Determine the effect of detailed pool loading assumptions on critical decay times.

The staff quantified the heatup time of the fuel after uncovery as a function of the decay time since final shutdown. The heatup time (displayed in Figure A1-1 below) was defined as the time to heat the fuel from 30 °C to 900 °C. The heatup time of the fuel depends on the amount of decay heat in the fuel, the oxidation heat input, and the amount of heat removal available from the fuel. The amount of decay heat is dependent on the burnup. The amount of heat removal is dependent on several variables that are difficult to represent generically without making a number of assumptions that may be difficult to confirm on a plant- and event-specific basis. For example, the air flow path is a critical parameter in the analyses. This in turn depends upon the fuel assembly geometry, rack configuration, and loading. However, the rack configuration and fuel assembly geometry are not only plant specific, they are subject to unpredictable changes when subjected to a severe seismic event or cask drop accident. For this analyses the staff initially assumed an undamaged fuel assembly and fuel rack geometry. Variations in air flow were then used to model the effects of potential flow blockage because of damage, as plant-specific design variations.

The staff used a specially modified version of TRAC-M to estimate the heatup time. TRAC-M was used because it is robust, has flexible modeling capabilities, and runs fast enough to perform sensitivity studies. Modifications were made to the wall drag, the wall heat transfer, and the oxidation models so that they would be applicable to the SFP heatup problem. The transfer of heat between high powered bundles and low powered bundles was not modeled, and only the fuel and fuel rack heat structures were modeled so the heatup time estimates should be conservative if the rack geometry is intact after the pool draining.

For the calculations, the staff used a decay heat per assembly and divided it equally among the pins. It assumed a 9X9 assembly for the BWRs and a 17x17 assembly for the PWRs. Decay heats were computed using an extrapolation of the decay power tables in NUREG/CR-5625 (Ref. 1). The decay heat in NUREG/CR-5625 is based on ORIGEN code calculations. The tables used in ORIGEN for the decay heat extend to burnups of 50 GWD/MTU for PWRs and 45 GWD/MTU for BWRs. The staff recognizes that the decay heat is only valid for values up to the maximum values in the tables, but staff ORIGEN calculations of the decay power, with respect to burnup for values in the table, indicate that extrapolation provides a reasonable and slightly conservative estimate of the decay heat for burnup values beyond the limits of the tables. Current peak bundle average burnups are approximately 50 GWD/MTU for BWRs and 55 GWD/MTU for PWRs. The BWR decay heat was calculated using a specific power of 26.2 MW/MTU. The PWR decay heat was calculated using a specific power of 37.5 MW/MTU. Both the PWR and BWR decay heats were calculated for a burnup of 60 GWD/MTU and include an uncertainty factor of 6 percent. The pool is divided into 10 flow channels. The downcomer flow area around the periphery of the pool is one channel. The last core offloaded is represented by 3 flow channels with each representing 1/3 of a core. The three previous 1/3 core offloads are modeled separately. The other three channels each model a full core for

a PWR and slightly more than a full core for a BWR. Only an average fuel rod is modeled for each channel. The model does not allow heat transfer between different powered flow channels.

Figure 1A-1 shows that for the configuration modeled, and for decay times of less than about 2 years for PWRs and 1.5 years for BWRs (assuming burnup of 60 GWD/MTU), it would take less than 10 hours for a zirconium fire to start or for significant fission product releases to begin once the fuel was fully uncovered and the fuel was cooled by an air flow of about two building volumes per hour. The figure also shows that after 4 years, PWR fuel could reach the point of fission product release in about 24 hours.

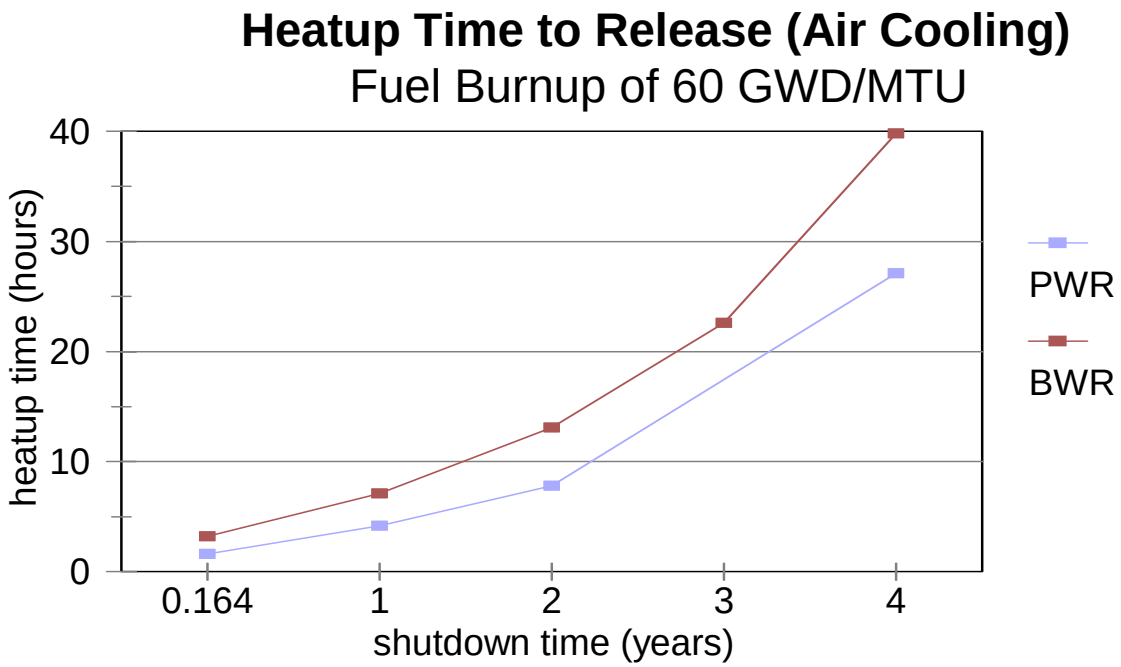

Figure 1A-1 Heatup time from 30 °C to 900 °C

The calculations also indicate that about 4-5 years decay is needed before air cooling is sufficient to preclude a zirconium fire.

The staff considered a number of sequences where the spent fuel might be only partially uncovered, such as in the case of a rapid partial draindown to a level at or below the top of active fuel with a slow boiloff of water after the draindown. This could occur if a large breach occurred in the liner at or below the top of active fuel. The staff has reasoned that for partial draindown or other cases, the lack of air cooling because of flow blockage combined with the geometry of the fuel racks result in the decay heat in the fuel rods effectively heating up the spent fuel in a near adiabatic manner. For these cases all the heat generated is retained in the fuel, its cladding, and the SFP rack structures. Knowledge of the specific heat of the cladding and the fuel allow for simple estimation of the time it would take to reach temperatures at which significant zirconium oxidation would begin or significant fission product releases would occur.

Figure 1A-2 shows a comparison of the air-cooled calculation to an adiabatic heatup calculation for a PWR at a burnup of 60 GWD/MTU. The timing of the event is important because of the infinite number of configurations that are possible after a dynamic event. The results show that the air-cooled heatup times are shorter than the adiabatic heatup times for times up to 2 years after shutdown. This is because the air cooling heatup rate is close to the adiabatic heatup rate and the oxidation heat source becomes a significant contributor to the total power at temperatures of approximately 600 °C. An adiabatic heatup calculation that included the oxidation heat source would have heatup times shorter than the air-cooled heatup times. The results show a heatup time to fission product release of 4 hours at 1 year after shutdown for a PWR with 60 GWD/MTU fuel burnup even with unobstructed airflow. At 5 years after shutdown the release of fission products may occur approximately 24 hours after the accident even with obstructed airflow. The air-cooled calculations used the parabolic oxidation rate equation recommended in Appendix 1B. This oxidation model leads to the fastest heatup times of the oxidation models that were examined.

The staff attempted to calculate a generic critical decay time necessary to ensure that air cooling was adequate to prevent the clad temperature from reaching the temperature of self-sustaining zirconium oxidation. The staff determined that it was not feasible absent setting stringent requirements or restrictions on plant fuel rack configurations, fuel burnup, and building ventilation to calculate a generic critical decay time. The staff examined the impact of burnup, oxidation models, building ventilation volumetric flow rate, and downcomer flow on the critical decay time. Burnup and variation in oxidation models are of secondary importance relative to air flow. As noted above air flow is dependent upon factors which can vary widely from pool to pool and even within an SFP. Therefore, the staff used an unobstructed flow model for one bound, and an adiabatic model as a second bound for its analyses. As seen in Appendix 1.B, the maximum clad temperature that is used for the definition of the critical decay time is dependent on the time after shutdown. The maximum clad temperature allowed is 600 °C for times less than 5 years after shutdown and 800 °C for times greater than 5 years after shutdown. Ultimately, the time differences resulting from these temperatures are not significant because of the already short time available in the early years where the source term is changing significantly because of ruthenium decay.

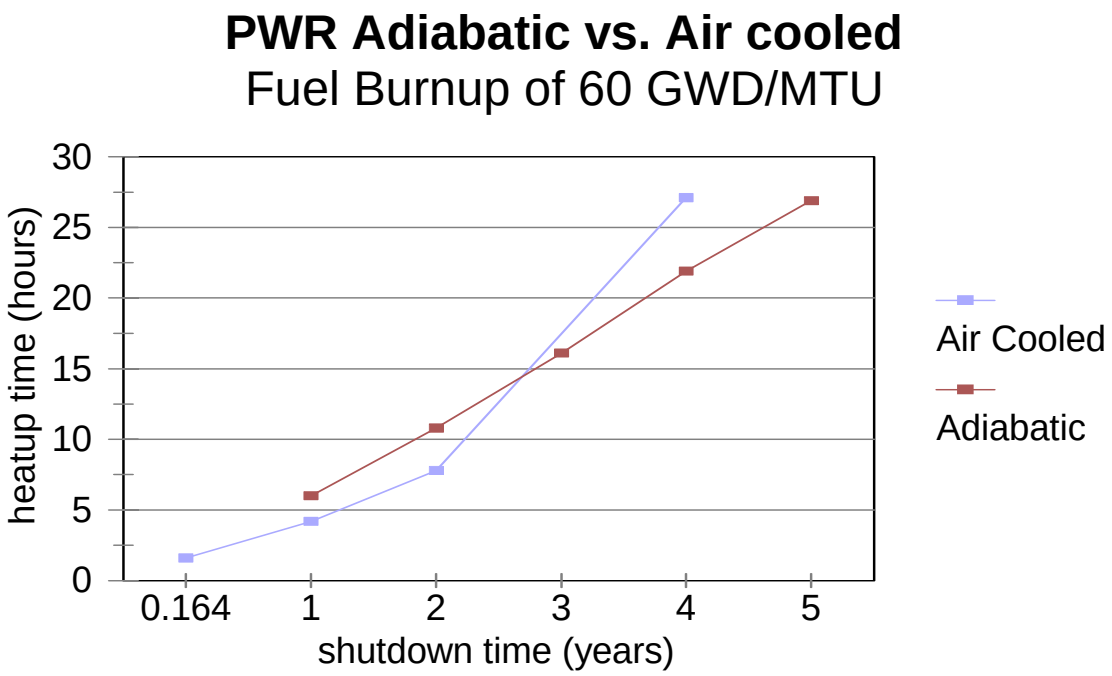

Figure 1A-2. Comparison of Adiabatic and Air-Cooled Heatup Times

Reference

1 Hermann, et.al., “Technical Support for a Proposed Decay Heat Guide Using SAS2H/ORIGEN-S Data,” NUREG/CR-5625, September 1994.

APPENDIX 1.B
TEMPERATURE CRITERIA FOR SPENT FUEL POOL ANALYSIS

1. BACKGROUND

The engineering analyses performed to address spent fuel pool (SFP) performance during various accidents have, in the past, used a temperature criterion to evaluate the potential for significant fuel damage. This temperature was intended as an acceptance criterion beyond which one would expect the onset of significant, global, fuel damage and substantial release of fission products (e.g., 50-100% of inventory of volatiles) associated with such damage. Further, the temperature criterion cited (generally about 900°C) has been selected on the basis that it represented a threshold for self sustained oxidation (Ref. 1) of cladding in air and on that basis it has been argued that if cooling of the spent fuel could limit fuel temperatures in equilibrium below this threshold then large releases of fission products need not be considered. Self sustaining reaction in this sense means the reaction rate and thus heat generation rate is sufficient, to roughly balance heat losses for given cooling mechanisms, resulting in an isothermal condition. Once the fuel temperature exceeds this threshold temperature (alternatively identified as an ignition or autoignition temperature) it was presumed that subsequent heat up and further increases in reaction rates would be escalating and rapid and that serious fuel damage would ensue. The temperature escalation associated with oxidation in this regime would not be balanced by any reasonable cooling afforded by natural circulation of air. While it was not expected that fission product releases associated with core melt accidents would immediately emerge at this temperature (based on reactor research in various steam and hydrogen environments) it was recognized that the time window for subsequent fuel heating would be relatively small once oxidation escalated. This also did not preclude gap type releases associated with fuel failures below the threshold temperature but these generally were not considered to be significant compared to the releases associated with higher fuel temperatures and significant fuel damage.

In the report, “Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants,” February 2000, the temperature criterion selected, 800 °C, was used in two ways. First, it was used to determine the decay heat level and corresponding time at which heat generation and losses for complete and instantaneous draining of the pool would lead to heating of the fuel (to 800 °C) after 10 hours. This time period would allow for the implementation of effective emergency response without the full compliment of regulatory requirements associated with operating reactors. Secondly, the temperature criterion was also used to evaluate the decay heat level and time (“critical decay time”) at which heat generation and losses for a fully drained fuel pool would result in an equilibrium temperature of 800 °C (typically this critical decay time has been on the order of 5 years). On that basis it was reasoned that since serious overheating of the fuel had not occurred, the fission product release associated with core melt need not be considered.

2. AIR OXIDATION AND TEMPERATURE ESCALATION

The NRC has received a number of comments related to the use of this temperature criterion and has reassessed the appropriateness of such a value for both its intended purposes. At the outset RES acknowledges that an ignition temperature, or more precisely in this case a temperature for incipient temperature escalation is dependent on heat generation and losses

which in turn is dependent on system geometry and configuration. In fact, much of the data on oxidation is produced in isothermal tests up to near the melting temperature of zirconium. In examining an appropriate criterion, it is useful to consider the range of available data including core degradation testing in steam environments, since it is likely that many SFP accidents may involve some initial period during which steam oxidation kinetics controls the initial oxidation, heatup, and release of fission products. In various experimental programs around the world (e.g., PBF-SFD, ACRR, CORA, NSRR, PHEBUS and QUENCH) repeatable phenomena have been observed for the early phase of core degradation (in steam) which proceeds initially at temperature increase rates associated with decay heat (at levels characteristic of reactor accidents) until cladding oxidation becomes dominant and a more rapid temperature escalation occurs. The point at which the escalation occurs, which does vary between tests, has been attributed to heat losses (Ref. 2) characteristic of the facility and to phase changes of ZrO_2 over a temperature range. The threshold at which temperature escalation occurred has been reported to vary from approximately 1100 °C to 1600 °C. In a CORA test performed with a lower initial heatup rate (to simulate reduced decay heat during shutdown conditions) it was reported that uncontrolled temperature escalation did not occur, raising the prospect that heating rate may be a factor. (This is probably because of the formation of a thicker oxidation layer built up over the protracted time at lower temperature such that when higher temperatures are attained, the thicker scale results in a lower oxidation rate relative to a thinner scale at the same temperature.) In more recent QUENCH tests (Quench 04 and 05) the effect of preoxidation was evaluated for its effect on hydrogen generation and temperature escalation. In Quench 04 temperature escalation was reported to occur at 1300 °C; in Quench 05 with approximately 200µm preoxidation temperature escalation was reported to be delayed until the fuel rod temperature reached 1620 °C.

Because of interest in air ingression phenomena for reactor accidents, recent severe accident research has also examined oxidation in air environments. Publication of results from the DRESSMAN and CODEX test programs (Ref. 3) has provided much of the transient data on fuel rod and rod bundle behavior for air kinetics as well as data on fuel oxidation and volatility. Early studies of zirconium oxidation in air (Refs. 4 and 5) were performed by comparing isothermal oxidation and scaling of fresh samples to determine the influence of different atmospheres and materials as well as to examine potential for fire hazards. The general observation is that, at least at higher temperatures(>1000 °C), the oxidation rate is higher in air than in steam. Another observation of the early studies was, under the same conditions, oxidation in an air environment produced an oxide layer or scale less protective than that for steam owing to the possible instability of a nitride layer beneath the outer oxide layer leading to scale cracking and a breakaway in the oxidation rate. The onset of this breakaway in the oxidation rate occurred at about 800 °C after a time period of 10 hours in the studies performed by Evans et al (Ref. 4) and after a period of approximately 4 hours in studies by Leistikow (Ref. 9). The Leistikow studies were performed on fresh cladding, however, and it is expected that breakaway would occur after a longer time delay with preoxidized cladding. As breakaway oxidation occurs the oxidation behavior observed no longer reflects a parabolic rate dependence but takes on a linear rate dependence. Also, at lower temperatures the kinetics of reaction indicate near cubic rate dependence thus the representation of the oxidation behavior at both high and low temperatures with a parabolic rate dependence may introduce unnecessary simplification and an understatement of the low temperature behavior. Breakaway scaling in an isothermal test may not translate to similar behavior under transient heatup conditions where initial oxidation occurs at lower temperatures and may involve steam oxidation. The presence of hydrides in the cladding may also increase the potential for

exfoliation and a breakaway in the oxidation; the effect of this has, however, been seen more clearly in testing conducted with steam and high hydrogen concentrations. Also, zirconium hydride will be dissolved at 700 °C and above, thus its contribution to exfoliation and breakaway will be minimal.

Autoignition is known to occur in zirconium alloys and zirconium hydride, especially when clean metal or hydride is suddenly exposed to air. The temperature of ignition is highly dependent on the ratio of surface area to volume and the degree of surface cleanliness. Generally, spent fuel rod cladding is covered with a relatively thick oxide layer (20-100µm), therefore, unless ballooning and burst occur in the cladding during heatup, clean high-temperature Zircaloy metal will not be exposed to air in an SFP accident. However, if there is cladding failure by ballooning and burst (expected to occur over a temperature range of 700-850 °C), hot oxide-free clean metal will be abruptly exposed to air. Zirconium hydride is expected to dissolve into the metal matrix during the slow heatup to these temperatures. At the moment of burst, some clean surface area of Zr metal will be exposed to air in the location of the rupture. Although data applicable to this situation is quite limited, considering the relatively small surface-to-volume ratio of the exposed metal, likelihood of ignition and subsequent propagation of the burning front of Zr metal is believed to be small (Ref. 8).

In the CODEX tests annular cladded fuel (in a 9 rod bundle) were heated with an inner tungsten heater rod to examine fuel degradation, with preoxidized cladding, in an air environment. Zircaloy oxidation kinetics were evaluated as well as the oxidation of the fuel. In the CODEX AIT-1 test the early phase of the test involved creating a preoxidation using an argon-oxygen mixture. The intent was to achieve a controlled preoxidation at a temperature of 900-950 °C, but it was reported (Ref. 3) that preoxidation was started at a slightly higher temperature than planned. What subsequently occurred was an uncontrollable temperature escalation up to approximately 2200 °C before it was cooled with cold argon flow. After restabilization of the rods at 900 °C air injection was started, electrical heatup commenced, and a second temperature escalation occurred. In the CODEX AIT-2 test, designed to proceed to a more damaged state, the preoxidation phase was conducted in an argon/steam mixture at 820 °C and 950 °C (a malfunction occurred during the preoxidation phase resulting in the admission of a small air flow as well). No temperature escalation was seen during the preoxidation phase. Following the restabilization of the fuel rods, a linear power increase was started and a temperature excursion subsequently occurred.

In addition to examining relevant test data RES also looked at determining a temperature based threshold for temperature escalation in an air environment by determining equivalent heat generation from steam transient tests. In this exercise we posited that at equivalent heat generation rates, i.e., accommodating different reaction rates and different heats of reaction for air and steam, we should be able to predict the corresponding temperature for escalation in air based on temperature escalations seen in severe fuel damage tests conducted in steam. Using this approach, the heat generation rate was estimated, assuming parabolic kinetics, and the following equation for a rate constant in air:

$$k_p = 52.67 \exp(-17597/T) \quad \{kg/m^2\}^2 \cdot sec^{-1} \qquad \text{[rate constant of } O_2 \text{ mass produced]}$$

It was predicted that based on an escalation temperature of 1200 °C in steam (observed in many of the steam tests), the equivalent heat generation rate in air would produce a temperature escalation at approximately 925 °C. The above equation for air kinetics was

identified in Reference 3 as the best fit for the CODEX AIT test data, i.e., it provided the best agreement to the temperature transient in the peak position. For steam kinetics, the rate equation used in MELCOR was selected for calculating the heat generation rate. The prediction of an escalation temperature in air using this approach seems to conform quite well with the observed behavior in the transient CODEX tests and lends further credence to the relative effect of oxidation in air with respect temperature escalation. The assumption of parabolic kinetics is routine in oxidation calculations and has been shown to provide a good match with a wide spectrum of experimental data even though, over select temperature ranges, deviations from that formulation have been observed. At temperatures above 900 °C, the reaction rate in air is high, regardless of whether parabolic or linear kinetics is assumed at that point and distinguishing between the rates of escalation is unimportant for our purposes.

3. ACCEPTANCE CRITERIA

In assessing a temperature criterion for escalation of the oxidation process and subsequent temperature escalation it is necessary to reconsider the intended uses of the criterion: 1) to evaluate the decay time after which the fuel heatup, in the case of complete fuel uncovery, leads to reaching that temperature at 10 hours and 2) to evaluate the decay time after which the fuel heatup, in the case of complete uncovery will never exceed the temperature criterion.

On balance it appears that a reasonable criteria for the threshold of temperature escalation in an air environment is a value of approximately 900 °C. This value is supported both by limited experimental data as well as by inference from the more abundant steam testing data. While certain weight gain data indicate the onset of a break away in the oxidation rate at lower temperatures after a period of 10 hours, this additional time period then exceeds the time interval for which the first use of the criterion is intended. With regard to the second use of the criterion, determination of the point at which severe fuel heatup is precluded, the onset of breakaway indicated in certain tests indicates that the temperature criterion should be lowered to 800 °C. It is important to stress that, in both instances, the temperature criteria should be used together with a thermal-hydraulic analysis that considers heat generation (i.e., decay heat and zircaloy reactions) and heat losses. For the second use of the criterion, i.e., establishing a threshold for precluding escalation, the analysis must demonstrate that heat losses, through convection, conduction and radiation, are sufficient to stabilize the temperature at the value selected.

In the case of slow, complete draining of the pool, or partial draining of the pool it is appropriate to consider use of a higher temperature criterion for escalation, perhaps as high as 1100 to 1200 °C. This would be appropriate if the primary oxidation reaction was with steam. Such a temperature criterion is relevant for the first intended use of the criterion, determining the point at which the temperature is not exceeded for 10 hours, however it is not appropriate for use as a long-term equilibrium temperature since over long intervals at such high temperature, one might reasonably expect significant fission product releases.

In addition to comments on the selection of an ignition temperature, the staff received comments related to the effect of intermetallic reactions and eutectic reactions. With respect to intermetallic reactions, the melting temperature of aluminum, which is a constituent in BORAL poison plates in some types of spent fuel storage racks, is approximately 640 °C. Molten aluminum can dissolve stainless steel and zirconium in an exothermic reaction forming intermetallic compounds. In the SFP configuration, zircaloy cladding will be covered with an

oxide layer and unless significant fresh metal surface is exposed through exfoliation there will be no opportunity to interact metallic zircaloy with aluminum (which similarly will be oxidized). Aluminum and steel will form an intermetallic compound at a temperature of 1150 °C, (Ref. 5) which is above the temperature criterion selected for fuel damage.

Besides intermetallic compounds, eutectic reactions may take place between pairs of various reactor materials, e.g., Zr-Inconel (937 °C), Zr-steel (937°C), Zr-Ag-In-Cd (1200 °C), Zr-B_4C (1627 °C), steel-B_4C (1150 °C), etc. (Ref. 6). Consideration of eutectics and intermetallics is important from the standpoint of heat addition as well as assuring the structural integrity of the storage racks and maintaining a coolable configuration. Noting the eutectic and intermetallic reaction temperatures, however it does not appear that formation of these compounds imposes any additional temperature limit on the degradation of cladding in an air environment.

Since the temperature criterion is also a surrogate of sorts for the subsequent release of fission products it is useful to consider the temperature threshold versus temperatures at which cladding may fail and fission products be released. Cladding is likely to fail by ballooning and burst in the temperature range of 700-850 °C, resulting in the release of fission products and fuel fines. At burst, clean Zircalloy metal will also be exposed, leading to an increase in oxidation although the total amount of metal involved will be limited. Creep failure of the cladding at or above 600 °C is also a possibility. This temperature limit is roughly associated with the 10 hour creep rupture time (565 °C) which has been used as a regulatory limit. While failure of the cladding at these lower temperatures will lead to fission product release, such release is considerably smaller than that assumed for the cases where the temperature criterion is exceeded and significant fuel heatup and damage occurs. Low temperature cladding failures might be expected to produce releases similar to those associated with dry cask accident conditions as represented in Interim Staff Guidance (ISG)-5. This NRC guidance document prescribes release fractions for failed fuel ($2x10^{-4}$ for cesium and ruthenium and $3x10^{-5}$ for fuel fines). Use of these release fractions would reduce the estimated offsite consequences dramatically from the fuel melt cases, early fatalities would be eliminated and latent cancer fatalities would be reduced by a factor of 100. As the temperature limit is increased from 600 °C to 900 °C there is some evidence that ruthenium releases would be increased based on ORNL test data from unclad pellets. Canadian data indicate though, that in the case of clad fuel the ruthenium release did not commence until virtually all of the cladding had oxidized. By this point it might be surmised that the fuel configuration would more closely resemble a debris bed than intact fuel rods. Selection of a temperature criterion for fuel pool damage also depends on the intended use, i.e., whether it is intended as the criterion for the 10 hour delay before the onset of fission product release or whether it is being used as a threshold for long-term fission product release. If the criteria is being used to judge when 10 hours are available for evacuation, then it may be argued that a higher temperature could be adopted, one associated with the significant release of fission products,1200 °C, since the release of fission products at lower temperatures will likely be small. However, in air it may be that the oxidation rate above 900 °C is sufficient to reduce the additional time gained to reach 1200 °C to a relatively small amount. Selection of a temperature criterion for long-term fuel pool integrity needs to consider that ruthenium release rates, in air, become significant at approximately 600-800 °C, based on the data of Parker et al. (Ref. 7).

Selection of an acceptance criterion for precluding significant offsite release after roughly 5 years, should also consider that ruthenium with a 1 year half life will be substantially decayed and that at 5 years cesium (and perhaps fuel fines such as plutonium) will dominate the dose

calculation. For these reasons RES believes that the long-term viability of the pool in a completely drained condition (air environment), if it concerns time periods of approximately 5 years, pool degradation should be assessed for a temperature of approximately 800 °C. Again, an analysis needs to be performed to demonstrate that at that temperature an equilibrium condition can be established. While this would result in an offsite release, there would be substantial time available to take corrective action after a 5 year decay time for the most recently loaded fuel. If shorter decay time periods are proposed for achieving the long-term equilibrium temperature criterion, then the impact of ruthenium releases would dictate reconsideration of this value.

4. SUMMARY

In summary, we conclude that for assessing the onset of fission product release under transient conditions (to establish the critical decay time for determining availability of 10 hours to evacuate) it is acceptable to use a temperature of 900 °C if fuel and cladding oxidation occurs in air. If steam kinetics dominate the transient heatup case, as it would in many boildown and draindown scenarios, then a suitable temperature criterion would be around 1200 °C. For establishing long-term equilibrium conditions for fuel pool integrity during SFP accidents which preclude significant fission product release it is necessary to limit temperatures to values of 600 °C to 800 °C. If the critical decay time is sufficiently long (>5 yrs) that ruthenium inventories have substantially decayed then it would be appropriate to consider the use of a higher temperature, 800 °C, otherwise fission product releases should be assumed to commence at 600 °C. These cases are marked by substantial time for corrective action to restore cooling and prevent smaller gap type releases associated with early cladding failures. A tabulated summary of the suggested criteria is listed below.

	Adequacy of 10 hrs for Evacuation	Precluding Large Release Fuel <5yrs	Precluding Large Release Fuel >5yrs
Dominant Air Environment	900 °C	600 °C	800 °C
Dominant Steam Environment	1200 °C	N/A	N/A

The degradation of fuel during SFP accidents is an area of uncertainty since most research on severe fuel degradation has focused on reactor accidents in steam environments. Because of this uncertainty, we have tended to rely on the selection of conservative criterion for predicting the global behavior of the SFP. It is our recommendation that the modeling of SFP accidents be performed with codes capable of calculating the heat generation and losses associated with the range of accidents, including phenomena associated with both water boiloff and air circulation. Further, the calculation of critical decay times for establishing both the validity of ad hoc evacuation and precluding fission product release must also include consideration of the exothermic energy of reactions (i.e., reactions with air and steam) with cladding, or alternatively demonstrate that such energy contribution is negligible in comparison to decay heat at that point. Severe accident codes, such as MELCOR, developed for modeling the degradation of reactor cores, would seem to be a reasonable approach for analysis of integral behavior and

would possess the general capabilities for modeling liquid levels and vapor generation, air circulation, cladding oxidation and fission product release. Use of a severe accident code also facilitates the use of self consistent modeling and assumptions for the analysis. The proper calculation of fission product releases depends in large part on the prediction of thermal-hydraulic conditions. More detailed CFD modeling would improve the calculation of boundary conditions for air circulation and could be used in conjunction with integral codes to better evaluate convective cooling. The kinetics of cladding reactions should be confirmed with experiments designed to simulate the range of conditions of interest under steady state and transient heating. The experimental database on ruthenium releases under conditions applicable to SFP accidents is inadequate and we are currently extrapolating data from conditions which tend to maximize such releases.

While there is uncertainty in the analysis of spent fuel degradation, especially for the conditions of air ingression, it is also true that elements of the analysis contain conservatism. The assumption of 75-100 percent release of ruthenium initiated at lower temperatures is based in large part on tests with bare fuel pellets, testing of cladded fuel indicates that the cladding acts as a getter of oxygen limiting release of ruthenium until virtually all of the cladding has oxidized. Further, before significant ruthenium release occurs (in its more volatile oxide form) the surrounding fuel matrix must be oxidized. During transient heatup of an SFP with temperature escalation one would expect the ruthenium release to follow the oxidation of the cladding at which point the fuel would more likely resemble a debris bed (the seismic event may also contribute in that regard) limiting the release fraction. The competition between formation of hyperstoichiometric UO_2 and U_3O_8 may also limit the release fraction below that seen in the data. The use of a temperature criterion of 600 °C to preclude significant fission product releases is conservative in that it is based in large part on data that discounts the effect of cladding to limit releases. The cladding failures at low temperatures will still allow substantial retention of fuel fines and the presence of unoxidized zircalloy will prevent formation of volatile forms of ruthenium. More prototypic experimental data on releases under these kinds of conditions may reveal that the onset of significant releases, especially ruthenium, would not occur under SFP accident conditions until fuel rod temperatures reached much higher temperatures associated with complete oxidation of the cladding.

Use of the hottest fuel assemblies to predict global release of fission products from the entire spent fuel inventory is a significant conservatism as well. Transient fuel damage testing indicates that at the time of local temperature escalation not all of the rod bundle undergoes rapid heating, cooler regions can avoid the oxidation transient. Prediction of the propagation of the temperature escalation to the cooler regions of the pool needs to be carefully examined to see if significant benefit can be gained, at a minimum it will lengthen the period of fission product release reducing the concentration of activity in the plume of fission products for offsite consequence analysis.

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APPENDIX 2
ASSESSMENT OF SPENT FUEL POOL RISK AT DECOMMISSIONING PLANTS

1.0 INTRODUCTION

As the number of decommissioning plants increases, the ability to address generic regulatory issues has become more important. After a nuclear power plant is permanently shut down and the reactor is defueled, most of the accident sequences that normally dominate operating reactor risk are no longer applicable. The predominant source of risk remaining at permanently shutdown plants involves accidents associated with spent fuel stored in the spent fuel pool (SFP). Previously, requests for relief from regulatory requirements that are less safety significant for decommissioning plants than operating reactors were granted on a plant-specific basis. This is not the best use of resources and led to differing requirements among decommissioning plants. The NRC Commission urged its staff to develop a risk-informed basis for making decisions on exemption requests and to develop a technical basis for rulemaking for decommissioning reactors in the areas of emergency preparedness, indemnification, and security. This study is one part of that basis.

The staff's assessment found that the frequency of spent fuel uncovery leading to a zirconium fire at decommissioning SFPs is less than $5x10^{-6}$ per year (using the Lawrence Livermore National Laboratory (LLNL) seismic hazard estimates (Ref. 1) for nuclear power plant sites) when a utility follows certain industry commitments and certain of our recommendations. The estimate drops to less than $1x10^{-6}$ per year if the EPRI site-specific seismic hazard estimates are used. These frequencies are made up of contributors from a detailed risk assessment of initiators ($3.4x10^{-7}$ per year), both internal and external, and a quasi-probabilistic contribution from seismic events ($<5x10^{-6}$ per year using the LLNL hazard estimates or $<6x10^{-7}$ per year using the EPRI hazard estimates [Ref.2]) that have ground motions many times larger than individual site design-basis earthquake ground motions (and higher uncertainty). It was also determined that if these commitments and recommendations are ignored, the estimated frequency of a zirconium fire could be significantly higher. Section 4 of this study discusses the steps necessary to assure that a decommissioning plant operates within the bounds assumed in the risk assessment.

Previous NRC-sponsored studies have evaluated some severe accident scenarios for SFPs at operating reactors that involved draining the SFP of its coolant and shielding water. Because of the significant configuration and staffing differences between operating and decommissioning plants, the staff performed this assessment to examine the risk associated with decommissioning reactor SFPs.

First, the staff examined whether or not it was possible from a deterministic view point for a zirconium cladding fire to occur. Zirconium fires were chosen as the key factor because radionuclides require an energetic source to transport them offsite if they are to have a significant health effect on local (first few miles outside the exclusion area) and more distant populations. Deterministic evaluations in the staff's preliminary draft risk assessment indicated that zirconium cladding fires could not be ruled out for loss of SFP cooling for fuel that has been shut down and removed from an operating reactor within approximately 5 years. The consequence analysis indicated that zirconium cladding fires could give offsite doses that the NRC would consider unacceptable. To assess the risk during the period of vulnerability to

zirconium cladding fires, the staff initially performed a broad preliminary risk assessment, which modeled many internal and external initiating events. The preliminary risk assessment was made publicly available early in the process (June 1999 [Ref. 3]) so that the public and the nuclear industry could track the NRC's evaluation and provide comments. In addition, the preliminary risk assessment was subjected to a technical review and requantification by the Idaho National Engineering and Environmental Laboratory (INEEL). The NRC continued to refine its estimates, putting particular emphasis on improving the human reliability assessment (HRA), which is central to the analysis given the long periods required for lowering the water in the SFP for most initiators. The staff identified those characteristics that a decommissioning plant and its utility should have to assure that the risks driven by fuel handler error and institutional mistakes are maintained at an acceptable level. In conjunction with the staff's HRA effort and ongoing reassessment of risk, the nuclear industry through NEI developed a list of commitments (NEI letter dated November 12, 1999 [Ref. 4]) that provide boundaries within which the risk assessment's assumptions have been refined. The staff released a draft risk assessment in February 2000, which updated the June 1999 preliminary study. The assessment reflects the commitments made by industry, the additional requirements we have developed to ensure the assumptions in the assessment remain valid, the technical review by INEEL, the staff's ongoing efforts to improve the assessment, and input from stakeholders. The study provides a technical basis for determining the acceptability of exemption requests and future rulemaking on decommissioning plant risk.

The staff looked at the broad aspects of the issue. A wide range of initiators (internal and external events including loss of inventory events, fires, seismic, aircraft, and tornadoes) was considered. The staff modeled a decommissioning plant's SFP cooling system based on the sled-mounted systems that are used at many current decommissioning plants. One representative SFP configuration (see Appendix 2A, Figure 2.1) was chosen for the evaluation except for seismic events, where the PWR and BWR SFP designs (i.e., the difference in location of the pools in PWRs and BWRs) were specifically considered. Information about existing decommissioning plants was gathered from decommissioning plant project managers and during visits to four sites covering all four major nuclear steam supply system vendors (General Electric, Westinghouse, Babcock & Wilcox, and Combustion Engineering). Plant visits gathered information on the as-operated, as-modified SFPs, their cooling systems, and other support systems.

From the perspective of offsite consequences, the staff focused on the zirconium fire end state, because there has to be an energetic source (e.g., a large high temperature fire) to transport the fission products offsite in order to have potentially significant offsite consequences. The staff chose the timing of when the SFP inventory is drained to within three feet of the top of the spent fuel as a surrogate for onset of the zirconium fire because once the fuel is uncovered, the dose rates at the edge of the pool would be in the tens of thousands of rem per hour, because it is unclear whether hydrides could cause ignition at lower cladding temperatures than previously predicted, because of the differences in configurations, and because there was great difficulty in modeling the heat transfer rate as the fuel was uncovered. In addition, from the point of view of estimation of human error rates, since for initiating events (other than seismic and heavy load drop) would take many days to uncover the top of the fuel, it was considered of small numerical benefit (and significant analytical effort) if the potential additional 2 days until the zirconium fire began were added to the timing.

After the preliminary draft risk assessment was released in June 1999, the staff sent the

assessment to INEEL for review and held public meetings and a workshop to assure that models appropriately accounted for the way decommissioning plants operate today and to help determine if some of the assumptions we made in the preliminary draft risk assessment needed improvement. Following a workshop, NEI provided a list of general commitments (see Appendix 5) that proved instrumental in refining the assumptions and models in the draft final risk assessment. Working with several PRA experts, the staff subsequently developed improved HRA estimates for events that lasted for extended periods.

This appendix describes how the risk assessment was performed for beyond design bases internal event accident sequences (i.e., sequences of equipment failures or operator errors that could lead to a zirconium cladding fire and release of radionuclides offsite). Event trees and fault trees were developed that model the initiating events and system or component failures that lead to fuel uncovery (these trees are provided in Appendix 2A).

APPENDIX 2A
DETAILED ASSESSMENT OF RISK FROM DECOMMISSIONING PLANT SPENT FUEL POOLS

1.0 INTRODUCTION

In Reference 1, the NRC performed a preliminary study of spent fuel pool risk at decommissioning plants to: examine the full scope of potentially risk-significant issues; identify credible accident scenarios; document the assessment for public review; and to elicit feedback from all stakeholders regarding analysis assumptions and design and operational features expected at decommissioning plants. In the February 2000 draft risk assessment, the staff updated the June 1999 preliminary draft risk assessment to include industry commitments. In this current analysis, the February 2000 draft was updated based on:

- stakeholder feedback
- additional thermal-hydraulic calculations

This updated PRA addresses the following initiating events:

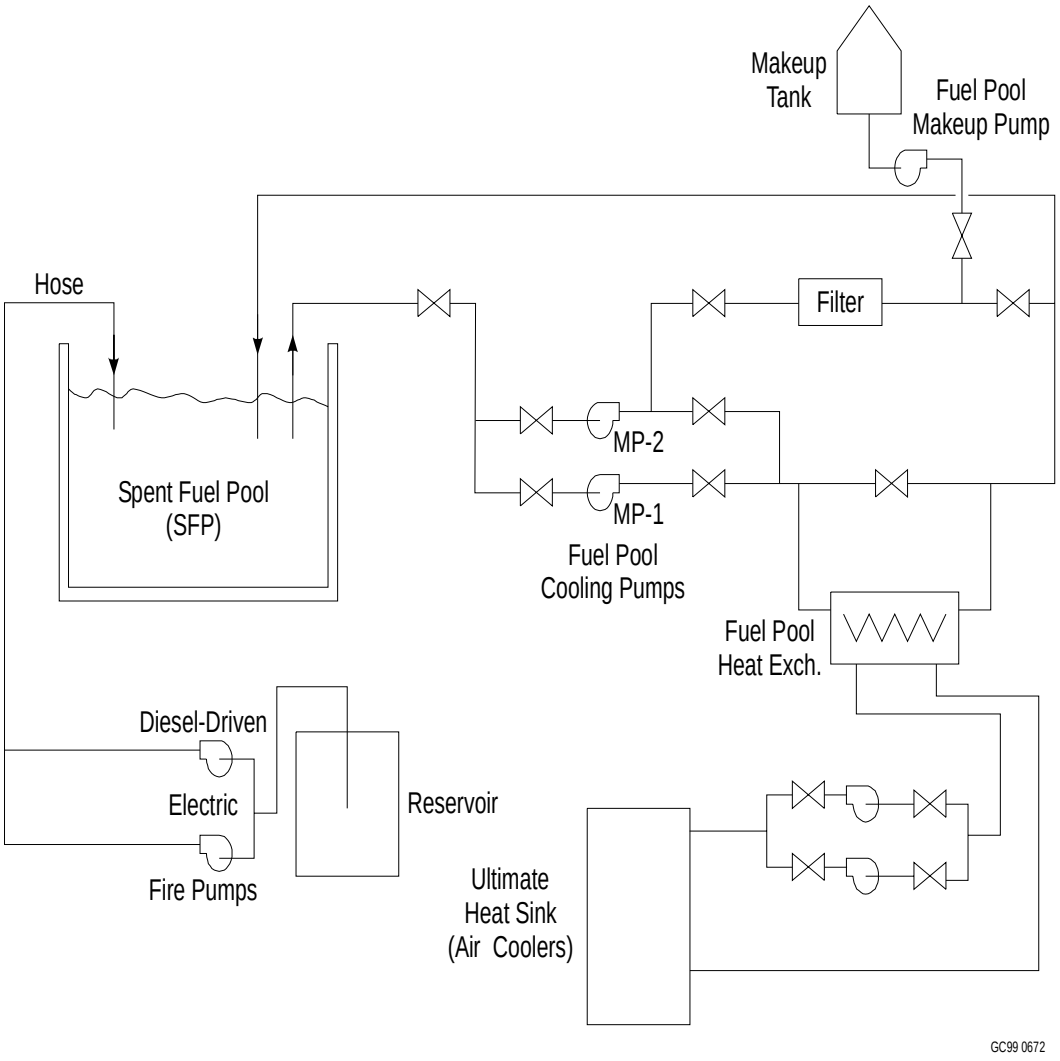
- loss of SFP cooling
- fire leading to loss of SFP cooling
- loss of offsite power because of plant centered and grid related causes
- loss of offsite power because of severe weather
- non-catastrophic loss of SFP inventory

External events such as earthquakes, aircraft crashes, heavy load drops, and tornado strikes that could lead to catastrophic pool failure are dealt with elsewhere in this study. The analysis is based on the following input. The assumed system configuration is typical of the sled-mounted systems that are used at many current decommissioned plants. Information about existing decommissioned plants was gathered from project managers (NRC Staff) of decommissioning plants, and during visits to four sites covering all four major nuclear steam supply system vendors (General Electric, Westinghouse, Babcock & Wilcox, and Combustion Engineering). The assumptions made about the operation of the facility are based in part on a set of commitments made by NEI (Ref. 2), supplemented by an interpretation of how some of those commitments might be applied.

2.0 SYSTEM DESCRIPTION

Figure 2.1 is a simplified drawing of the system assumed for the development of the model. The spent fuel pool cooling (SFPC) system is located in the SFP area and consists of motor-driven

Figure 2.1 Simplified Diagram of Spent Fuel Pool Cooling and Inventory Makeup Systems



pumps, a heat exchanger, an ultimate heat sink, a makeup tank, filtration system and isolation valves. Suction is taken via one of the two pumps on the primary side from the SFP and is passed through the heat exchanger and returned back to the pool. One of the two pumps on the secondary side rejects the heat to the ultimate heat sink. A small amount of water is diverted to the filtration process and is returned to the discharge line. A regular makeup system supplements the small losses because of evaporation. In the case of prolonged loss of SFPC system or loss of inventory events, the inventory in the pool can be made up using the firewater system. There are two firewater pumps, one motor-driven (electric) and the other diesel-driven, which provide firewater throughout the plant. A firewater hose station is provided in the SFP area. The firewater pumps are assumed to be located in a separate structure.

3.0 METHODOLOGY

3.1 Logic Model

This section summarizes the SFP PRA model developed in this study. The description of the modeling approach and key assumptions is intended to provide a basis for interpreting the results in Sections 4 and 5. The event trees and fault trees presented in this study are meant to be generic enough to apply to many different configurations. The fault trees are documented in Attachment A to this appendix. An example of the HRA worksheet used for this analysis is presented in Attachment B.

The endstate for this analysis is defined as loss of coolant inventory to the point of fuel uncovery from either leakage or boil-off. Dose calculations (Ref. 5) show that when there is less than 3 feet of water above the top of the fuel, an environment that is rapidly lethal to anyone at the edge of the pool can result. Therefore, 3 feet has been adopted as an effective limit for recovery purposes. In other words, the endstate for this analysis is effectively defined as loss of coolant inventory to a point 3 feet above the top of the fuel. One of the NEI commitments is that there should be a provision for remote alignment of the makeup source to the pool, which would make this assumption conservative. However, the impact of this conservatism on the conclusions of this analysis is minor.

The event tree and fault tree models were developed and quantified using Version 6 of the SAPHIRE software package (Ref. 6), using a fault tree linking approach. Event trees were developed for each of the initiators identified in Section 1.

3.2 HRA Methodology

3.2.1 Introduction

One of the key issues in performing a probabilistic risk assessment (PRA) for the SFP during the decommissioning phase of a nuclear power plant's life cycle is how much credit can be given to the operating staff to respond to an incident that impacts the SFP that would, if not attended to, lead to a loss of cooling of the spent fuel and eventually to a zirconium fire.

The objective of the HRA analysis in this PRA is to assess whether the design features and operational practices assumed can be argued to suggest that the non-response probabilities should be low. The design features include the physical plant characteristics (e.g., nature and number of alarms, available mitigation equipment) and the operational practices include

operational and management practices (including crew structure and individual responsibilities), procedures, contingency plans, and training. Since the details will vary from plant to plant, the focus is on general design features and operational practices that can support low non-response probabilities.

Section 3.2.2 discusses the differences between the full power and decommissioning modes of operation as they impact human reliability analysis, and the issues that need to be addressed in the analysis of the decommissioning mode are identified. Section 3.2.3 discusses the factors that recent studies have shown to be significant in establishing adequacy of human performance.

3.2.2 Analysis Approach

The HRA approaches that have been developed over the past few years have primarily been for use in PRAs of nuclear power plants at full power. Methods have been developed for assessing the likelihood of errors associated with routine processes such as restoration of systems to operation following maintenance, and those errors in responding to plant transients or accidents from full power. For SFP operation during the decommissioning phase, there are unique conditions not typical of those found during full-power operation. Thus the human reliability methods developed for full power operation PRAs, and their associated error probabilities, are not directly applicable. However, some of the methods can be adapted to provide insights into the likelihood of failures in operator performance for the SFP analysis by accommodating the differences in conditions that might impact operating crew performance in the full power and decommissioning phases. There are both positive and negative aspects of the difference in conditions with respect to the reliability of human performance.

Examples of the positive aspects are:

- For most scenarios, the time-scale for changes to plant condition to become significant are protracted. This is in contrast to full power transients or accidents in which response is required in a relatively short time, ranging from a few minutes to a few hours. In the staff's analysis, times ranging from 100 to greater than 220 hours were estimated for heat up and boil off following loss of SFP cooling. Thus, there are many opportunities for different plant personnel to recognize off-normal conditions, and a long time to take corrective action, such as making repairs, hooking up alternate cooling or inventory makeup systems, or even bringing in help from offsite.

- There is only one function to be maintained, namely decay heat removal, and the systems available to perform this function are relatively simple. By contrast, in the full power case there are several functions that have to be maintained, including criticality, pressure control, heat removal, containment integrity.

- With respect to the last point, it is also expected that the number of controls and indications that are required in the control room are considerably fewer than for an operating plant, and therefore, there is less cause for confusion or distraction.

Examples of the negative aspects are:

- The plant operation is not as constrained by regulatory tools (technical specifications are not as comprehensive and restrictive as they are for operating plants), and there is no

requirement for emergency procedures.

- Because the back-up systems are not automatically initiated, operator action is essential to successfully respond to failures of the cooling function.

- There is expected to be little or no redundancy in the onsite mitigating capability as compared with the operating plant mode of operation. (In the staff's initial evaluation, because little redundant onsite equipment was assumed to be available, the failure to bring on offsite equipment was one of the most important contributors.) This implies that repair of failed functions is relatively more significant in the risk analysis for the SFP case.

In choosing an approach for developing the estimates documented in this study, the following issues were considered to be important:

- Because of the long time scales, it is essential to address the potential for recovery of failures on the part of one crew or individual by other plant staff, including subsequent shifts.

- Potential sources of dependency that could lead to a failure of the organization as a whole to respond adequately should be taken into account.

- The approach should be consistent with current understanding of human performance issues (Refs. 7, 8, and 9).

- Those factors that the industry has suggested that will help ensure adequate response (instrumentation, monitoring strategies, procedures, contingency plans) should be addressed (Ref. 4).

- Where possible, any evaluations of human error probabilities (HEPs) should be calibrated against currently acceptable ranges for HEPs.

- The reasoning behind the assumptions made should be transparent.

3.2.3 Human Performance Issues

In order to be successful in coping with an incident at the facility, there are three basic functions that are required of the operating staff, and these are either explicit (awareness) or implicit (situation assessment and response planning and response implementation) in the definitions of the human failure events in the PRA model.

- Plant personnel must be able to detect and recognize when the spent fuel cooling function is deteriorating or pool inventory is being lost (Awareness).

- Plant personnel must be able to interpret the indications (identify the source of the problem) and formulate a plan that would mitigate the situation (Situation Assessment and Response Planning).

- Plant personnel must be able to perform the actions required to maintain cooling of and/or add water to the SFP (Response Implementation).

In the following sections, factors that are relevant to determining effective operator responses are discussed. While not minimizing the importance of such factors as the establishment of a safety culture and effective intra-crew communication, the focus is on factors which can be determined to be present on a relatively objective basis. A review of LERs associated with human performance problems involved in response to loss of fuel pool cooling revealed a variety of contributing factors, including crew inexperience, poor communication, and inadequate administrative controls. In addition, there were some instances of design peculiarities that made operator response more complex than necessary.

The factors discussed below were used to identify additional assumptions made in the analysis that the staff considered would provide for an effective implementation of the NEI commitments.

3.2.3.1 Awareness/Detection of Deviant Conditions

There are two types of monitoring that can be expected to be used in alerting the plant staff to deviant conditions: a) passive monitoring in which alarms and annunciators are used to alert operators; b) active monitoring in which operators, on a routine basis, make observations to detect off-normal behavior. In practice both would probably be used to some extent. The amount of credit that can be assumed depends on the detailed design and application of the monitoring scheme.

In assessing the effectiveness of alarms there are several factors that could be taken into account, for example:

- alarms (including control room indications) are maintained and checked/calibrated on a regular basis
- the instruments that activate instruments and alarms measure, as directly as possible, the parameters they purport to measure
- alarm set-point is not too sensitive, so that there are few false alarms
- alarms cannot be permanently canceled without taking action to clear the signal
- alarms have multiple set-points corresponding to increasing degradation
- the importance of responding to the alarms is stressed in plant operating procedures and training
- the existence of independent alarms that measure different primary parameters (e.g., level, temperature, airborne radiation), or provide indirect evidence (sump pump alarms, secondary side cooling system trouble alarms)

The first and last of these factors may be reflected in the reliability assumed for the alarm and in the structure of the logic model (fault tree) for the event tree function control room alarms (CRA), respectively. The other factors may be taken into account in assessing the reliability of the operator response.

For active monitoring, examples of the factors used in assessing the effectiveness of the monitoring include:

- scheduled walk-downs required within areas of concern, with specific items to check (particularly to look for indications not annunciated in, or monitored from, the control room, for example, indications of leakage, operation of sump pumps if not monitored, steaming over the pool, humidity level)

- plant operating procedures that require the active measurement of parameters (e.g., temperature, level) rather than simply observing the condition of the pool

- requirement to log, check, and trend results of monitoring

- alert levels specified and noted on measurement devices

These factors can all be regarded as performance shaping factors (PSFs) that affect the reliability of the operators.

An important factor that should mitigate against not noticing a deteriorating condition is the time scale of development, which allows the opportunity for several shifts to notice the problem. The requirement for a formal shift turnover meeting should be considered.

3.2.3.2 Situation Assessment and Response Planning

The principal operator aids for situation assessment and response planning are procedures and training in their use.

The types of procedures that might be available are:

- annunciator/alarm response procedure that is explicit in pointing towards potential problems

- detailed procedures for use of alternate systems indicating primary and back-up sources, recovery of power, etc.

The response procedures may have features that enhance the likelihood of success, for example:

- inclusion of guidance for early action to establish contingency plans (e.g., alerting offsite agencies such as fire brigades) in parallel with a primary response such as carrying out repairs or lining up an onsite alternate system.

- clearly and unambiguously written, with an understanding of a variety of different scenarios and their timing.

In addition:

- training for plant staff to provide an awareness of the time scales of heat up to boiling and fuel uncovery as a function of the age of the fuel would enhance the likelihood of

successful response.

3.2.3.3 Response Implementation

Successful implementation of planned responses may be influenced by several factors, for example:

- accessibility/availability of equipment
- staffing levels that are adequate for conducting each task and any parallel contingency plans, or plans to bring in additional staff
- training
- timely feedback on corrective action

3.2.4 Quantification Method

Three HRA quantification methods were applied, and each is briefly described below.

- The Technique for Human Error Prediction (THERP, Ref. 10). This method was used to quantify the initial recognition of the problem. Specifically, the annunciator response model (Table 20-23) was used for response to alarms. The THERP approach was also used to assess the likelihood of failure to detect a deviant condition during a walk-down, and also the failure to respond to a fire. While this method was developed over 20 years ago, it is still regarded as an appropriate method for the types of HEPs for which it is being used in this analysis.
- The Exponential Repair Model (while not strictly a human reliability model) was applied to calculate the probability of failure associated with the repair of systems and components in this analysis. This method is described in the main body of the study. In cases where dependency exists with prior repair tasks, the dependency model used in THERP was used to assess the impact of that dependency.
- The Simplified Plant Analysis Risk Human Error Analysis Method (SPAR HRA, Ref. 11) was employed for all other HEPs. This model was chosen because it includes an appropriate level of detail in terms of performance shaping factors and error modes (cognition and execution) given the lack of detailed knowledge about expected plant practices and designs. The PSFs used in the model allowed the impact of the NEI commitments and additional staff assumptions to be incorporated explicitly into the evaluation.

3.3 Other Inputs to the Risk Model

A variety of other inputs were required for this PRA, including generic configuration data used in the fault tree models, radiological calculations, and timing calculations. Initiating event frequencies and generic reliability data were derived from other studies sponsored by the NRC. The times available for operator actions are based on calculations of the time it would take for bulk boiling to begin in the pool, or on the time it takes for the level in the pool to fall to the level

of the fuel pool cooling system suction, or to a height of approximately 3 ft above the fuel, as appropriate to the definition of the corresponding human failure event.

It takes a relatively long time to uncover the fuel if the initiating event does not involve a catastrophic failure of the pool. This is because of the large amount of water in an SFP, the large specific heat of water, and the large latent heat of vaporization for water. Calculations that were used in the June 1999 and February 2000 study for a typical-sized SFP yield the results in Table 3.1. Subsequently the staff determined that it had made a mistake in the assumed heat load on the generic SFP. Current estimates of time to bulk boiling are actually about twice those given in Table 3.1 below. The staff debated whether to redo the human reliability analysis estimates assuming the longer periods. It was determined that the credit given for fuel handler recovery was already so great that it would be difficult to estimate the numerical benefit of the additional time. Rather, it can be inferred that the uncertainties of whether the absolute value of the recovery estimates are really so large have been reduced. In addition, the numerical estimates for the sequences that are affected by these longer recovery times are already so low that they contribute very little to the overall risk estimates, which are dominated by seismic events, heavy load drop, and loss of offsite power because of extreme weather that are not as strongly affected by fuel handler error.

The bulk boiling and boil-off results are based on the following assumptions:

- no heat losses
- atmospheric pressure
- Heat of vaporization $h_{fg} \approx 2258$ kJ/kg
- base pool heat load for a full pool of 2 MW
- core thermal power of 3293 MW
- typical pool size (based on Tables 2.1 and 2.2 of NUREG/CR-4982, Ref. 10)
  - typical BWR pool is 40' deep by 26' by 39'
  - typical PWR pool is 43' deep by 22' by 40'

Table 3.1 Time to Bulk Boiling, and Boil-off Rates

Time after discharge (days)	Decay power from last core (MW)	Total heat load (MW)	Time to bulk boiling (hr)	Boil-off rate (gpm)	Level decrease (ft/hr)[1]
2	16.4	18.4	5.6	130	1.0
10	8.6	10.6	9.8	74	0.6
30	5.5	7.5	14	52	0.42
60	3.8	5.8	18	41	0.33
90	3.0	5.0	21	35	0.28
180	1.9	3.9	27	27	0.22
365	1.1	3.1	33	22	0.18 ≈ 0.2

Notes: (1) using typical pool sizes, it is estimated that for BWRs, we have 1040 ft^3/ft depth, and for PWRs, we have 957 ft^3/ft depth. Assume ≈ 1000 ft^3/ft depth for level decreases resulting from boil-off.

In an SFP, the depth of water above the fuel is typically 23 to 25 feet. Subtracting 3 feet to account for shielding requirements, it is estimated that approximately 20 feet of water will have to boil-off before the start of fuel uncovery. Therefore, using the above table, the available time for operator actions for the loss of cooling type accidents is estimated as follows:

For one-year-old fuel, the total boiloff time available equals the time to bulk boiling plus the time to boildown to 3 ft above the top of the fuel. Therefore, the total time available for operator action is as follows:

Total Time = 33 hr + (20 ft)/(0.2 ft/hr)
= 133 hours

It is assumed that the operator will not use alternate systems (e.g., firewater) until after bulk boiling begins and the level drops to below the suction of the cooling system. It is assumed that the suction of the cooling system is 2 ft below the nominal pool level. Therefore, if bulk boiling begins at 33 hours, and the boil-off rate is 0.2 ft/hr, then the total boiloff time available to provide make-up using the firewater system to prevent fuel uncovery is as follows:

133 hrs -(Time to Bulk Boiling + Time for Boil-off) = $133 - (33hrs + \frac{2\,ft}{0.2\,ft/hr}) = 133 - 43hrs$ = 90hrs

3.4 General Assumptions

This analysis is based on the assumption that the commitments for procedures and equipment proposed by NEI in their November 12, 1999, letter to Richard J. Barrett (Ref. 4) are adopted. These are reproduced below:

1. Cask drop analyses will be performed or single failure-proof cranes will be in use for handling of heavy loads, (i.e., phase II of NUREG-0612 (Ref. 13) will be implemented).

2. Procedures and training of personnel will be in place to ensure that onsite and offsite resources can be brought to bear during an event.

3. Procedures will be in place to establish communication between onsite and offsite organizations during severe weather and seismic events.

4. An offsite resource plan will be developed which will include access to portable pumps and emergency power to supplement onsite resources. The plan would principally identify organizations or suppliers where offsite resources could be obtained in a timely manner.

5. SFP instrumentation will include readouts and alarms in the control room (or where personnel are stationed) for SFP temperature, water level, and area radiation levels.

6. SFP boundary seals that could cause leakage leading to fuel uncovery in the event of seal failure shall be self limiting to leakage or otherwise engineered so that drainage cannot occur.

7. Procedures or administrative controls to reduce the likelihood of rapid draindown events will include (1) prohibitions on the use of pumps that lack adequate siphon protection; or (2) controls for pump suction and discharge points. The functionality of anti-siphon devices will be periodically verified.

8. An onsite restoration plan will be in place to provide for repair of the SFP cooling systems or to provide access for makeup water to the SFP. The plan will provide for remote

alignment of the makeup source to the SFP without requiring entry to the refuel floor.

9. Procedures will be in place to control SFP operations that have the potential to rapidly decrease SFP inventory. These administrative controls may require additional operations or administrative limitations such as restrictions on heavy load movements.

10. Routine testing of the alternative fuel pool makeup system components will be performed and administrative controls for equipment out of service will be implemented to provide added assurance that the components would be available if needed.

Since the commitments are stated at a relatively high level, additional assumptions have been made as detailed below.

- It is assumed that the operators (through procedures and training) are aware of the available backup sources that can be used to replenish the SFP inventory (i.e., the fire protection pumps, or offsite sources such as from fire engines). Arrangements have been made in advance with fire stations including what is required from the fire department including equipment and tasks.

- The site has two operable firewater pumps, one diesel-driven and one electrically driven from offsite power.

- The makeup capability (with respect to volumetric flow) is assumed as follows:

Makeup pump:	20 - 30 gpm
Firewater pump:	100 - 200 gpm
Fire engine:	100 - 250 gpm [depending on hose size: 1-½" (100 gpm) or 2-½" (250 gpm)]

- It is therefore assumed that, for the larger loss of coolant inventory accidents, make-up through the makeup pumps is not feasible unless the source of inventory loss can be isolated.

- The operators perform walk-downs of the SFP area once per shift (8- to 12-hour shifts). A different crew member is assumed for the next shift. It is also assumed that the SFP water is clear and pool level is observable via a measuring stick in the pool that can alert operators to level changes.

- Requirements for fire detection and suppression may be reduced (when compared to those for an operating plant) and it is assumed that automatic detection and suppression capability may not be present.

- All equipment, including external sources (fire department), are available and in good working order.

- The emergency diesel generators and support systems such as residual heat removal and service water (that could provide SFP cooling or make-up before the plant being decommissioned) have been removed from service.

- The SFP cooling system, its support systems, and the electric driven fire protection pump are fed off the same electrical bus.

- Procedures exist to mitigate small leaks from the SFP or for loss of the SFP cooling system.

- The only significant technical specification applicable to SFPs is the requirement for radiation monitors to be operable when fuel is being moved. There are no technical specifications requirements for the cooling pumps, makeup pumps, firewater pumps, or any of the support systems.

- There are multiple sources of water for make-up via the firewater pumps or fire engine.

- Generic industry data were used for initiating event frequencies for the loss of offsite power, the loss of pool cooling, and the loss of coolant inventory.

- Instrumentation that measures SFP temperature and level measures these parameters directly.

- For the purposes of timing, the transfer of the last fuel from the reactor to the SFP is assumed to have occurred one year previously.

4.0 MODEL DEVELOPMENT

This section describes the risk models that were developed to assess the likelihood of fuel uncovery from SFP loss of cooling events, fire events, loss of offsite power, and loss of inventory events.

4.1 Loss of Cooling Event Tree

This event tree (Figure 4.1) models generic loss of cooling events (i.e., those not related to other causes such as fire or loss of power, which are modeled in later sections). The top events and the supporting functional fault trees are discussed in the following sections.

4.1.1 Initiating Event LOC – Loss of Cooling

4.1.1.1 Event Description

This initiating event includes conditions arising from loss of coolant system flow because of the failure of the operating pumps or valves, from piping failures, from an ineffective heat sink (e.g., loss of heat exchangers), or from a local loss of power (e.g., failure of electrical connections).

4.1.1.2 Quantification

This initiating event is modeled by a single basic event, IE-LOC. An initiation frequency of 3.0E-3/yr is taken from NUREG-1275 Volume 12 (Ref. 14). This represents the frequency of loss of cooling events in which temperatures rise more than 20 °F.

4.1.2 Top Event CRA – Control Room Alarms

4.1.2.1 Event Description and Timing

This event represents a failure to respond to conditions in the pool that are sufficient to trigger an alarm. Failure could be because of operator error (failure to respond), or loss of indication because of equipment faults. Success for this event is defined as the operator recognizing the alarm and understanding the need to investigate its cause. This event is quantified by fault tree LOC-CRA and includes hardware and human failures basic events that represent failure of control room instrumentation to alarm given that SFP cooling has been lost, and the operators fail to respond to the alarm, respectively.

4.1.2.2 Relevant Assumptions

- Within 8 to 12 hours of the loss of cooling, one or more alarms or indications will reflect an out-of-tolerance condition to the operators in the control room (there may be level indication available locally or remotely, but any change in level is not likely to be significant until later in the sequence of events).

- The SFP has at least one water temperature measuring device, with an alarm and a readout in the control room (NEI commitment no. 5). There could also be indications or alarms associated with pump flow and pressure, but no credit is taken here.

- The instrumentation is tested on a routine basis and maintained operable.

- Procedures are available to guide the operators in their response to off-normal conditions, and the operators are trained on the use of these procedures (NEI commitment no. 2).

4.1.2.3 Quantification

<u>Human Error Probabilities</u>

The basic event HEP-DIAG-ALARM models operator failure to respond to an indication in the control room and diagnose a loss of cooling event. Such an alarm would likely be the first indication of trouble, so the operator would not be under any heightened state of alertness. On the other hand, it is not likely that any other signals or alarms for any other conditions would be present to distract the operator. The error rate is taken from THERP (Table 20-23).

<u>Hardware Failure Probabilities</u>

The value used for local faults leading to alarm channel failure (event SPC-LVL-LOP, 2.0E-3) was estimated based on information in Reference 14. This event includes failure of instrumentation and local electrical faults.

4.1.2.4 Basic Event Probabilities

Basic Event	Basic Event Probability
HEP-DIAG-ALARM	3.0E-4

SPC-LVL-LOP	2.0E-3

4.1.3 Top Event IND – Other Indications of Loss of Cooling

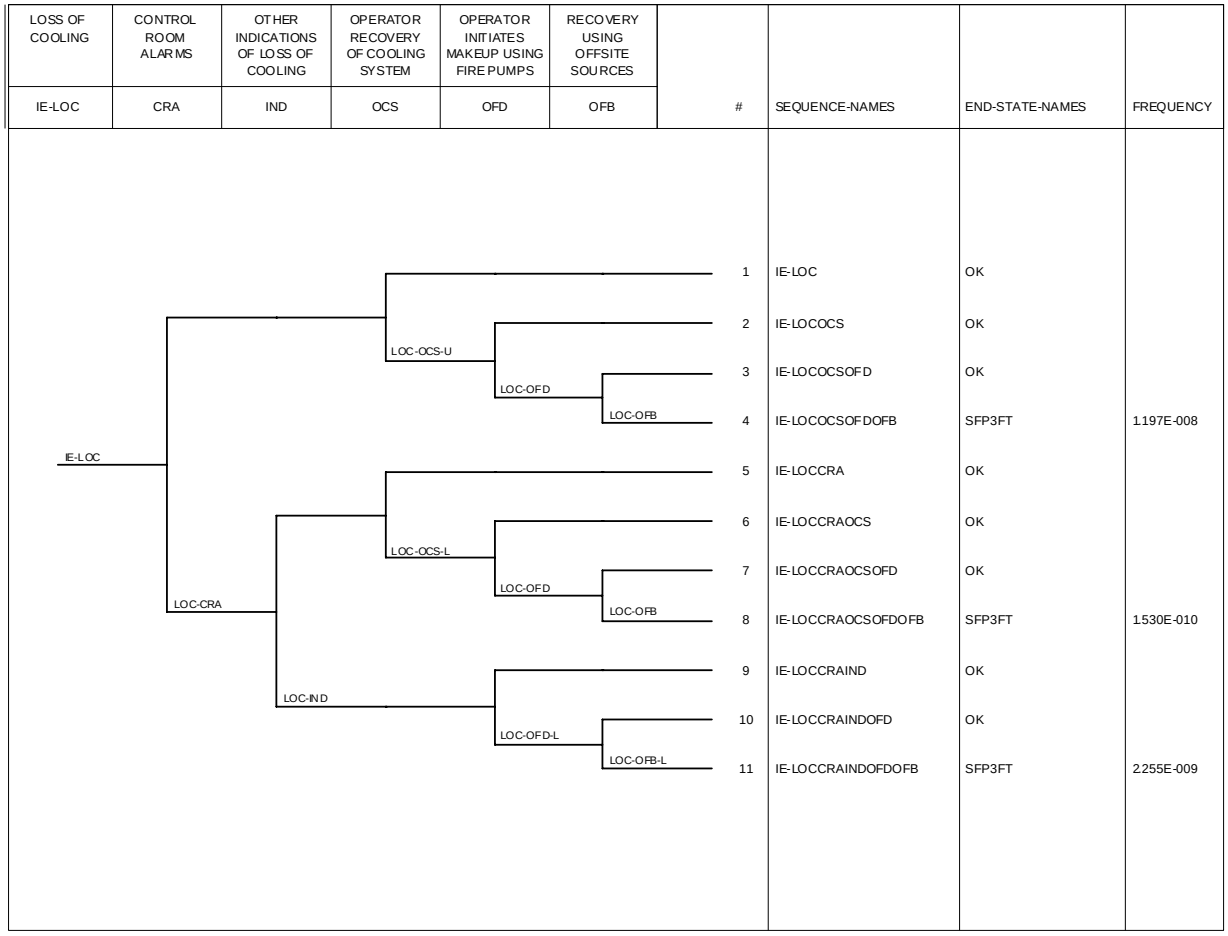
4.1.3.1 Event Description and Timing

This top event models subsequent operator failures to recognize the loss of cooling during walk-downs over multiple shifts. Indications available to the operators include: temperature readouts in the control room (NEI commitment no. 5), local temperature measurements, and eventually, increasing area temperature and humidity, low water level from boil-off, and local alarms. Success for this event is defined as the operator recognizing the abnormal condition and understanding the need to investigate its cause, leaving sufficient time to attempt to correct the problem before the pool level drops below the SFP cooling system suction. The event is modeled by fault tree LOC-IND.

4.1.3.2 Relevant Assumptions

- The loss of cooling may not be noticeable during the first two shifts but conditions are assumed to be sufficient to trigger high temperature alarms locally and in the control room.
- Operators perform walk-downs and control room readouts once per shift (every 8 to 12 hours) and document observations in a log.
- Regular test and maintenance is performed on instrumentation (NEI commitment no. 10).
- During walk-downs, level changes in the SFP can be observed on a large, graduated level indicator in the pool.
- 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)

Figure 4.1 Loss of SFP cooling system event tree



4.1.3.3 Quantification

Human Error Probabilities

The functional fault trees include two human failure events, depending on whether the control room alarms have failed, or whether there was a failure to respond to the initial alarm (it is assumed that the alarm was canceled). If the operator failed to respond to control room alarms, then event HEP-WLKDWN-DEPEN models subsequent operating crews' failures to recognize the loss of cooling during walk-downs, taking into account the dependence on event HEP-DIAG-ALARM. A specific mechanism for dependence can only be identified on a plant and event specific basis, but could result, for example, from an organizational failure that leads to poor adherence to plant procedures. Because this is considered unlikely, and because the conditions in the pool area change significantly over the time scale defined by the success criterion for this event, the degree of dependence is assumed to be low.

If the alarms failed, then event HEP-WLKDWN-LSFPC models subsequent crews' failures to recognize the loss of cooling during walk-downs, with no dependence on previous HEPs. However, because the control room readouts could share a dependency with the alarms, the assumption of local temperature measurements becomes important. The failure probabilities for these events were developed using THERP, and are based upon three individual failures: failure to carry out an inspection, missing a step in a written procedure, and misreading a measuring device. Because there are on the order of 33 - 43 hours before the SFP cooling system becomes irrecoverable without pool make-up, it is assumed that multiple crews would have to fail. Assuming that the crews are totally independent would give a very low probability. However, a low level of dependence is assumed and the probability is truncated at 1E-05.

4.1.3.4 Basic Event Probabilities

Basic Event	**Basic Event Probability**
HEP-WLKDWN-LSFPC	1.0E-5
HEP-WLKDWN-DEPEN	5.0E-2

4.1.4 Top Event OCS – Operator Recovery of Cooling System

4.1.4.1 Event Description and Timing

Once the operators recognize loss of SFP cooling, they will likely focus their attention on recovery of the SFP cooling system. It is assumed that only after bulk boiling begins and the water level drops below the cooling system suction that the operator will inject water from other makeup systems (e.g., firewater). Therefore, the time available to recover the SFP cooling system could be as long as 43 hours, given an immediate response to an alarm. However, it has been assumed that the operating staff has only until shortly after bulk boiling begins (assumed to be 33 hours) to restore the SFP cooling system. This assumption is based on concerns about volume reduction because of cooling and whether the makeup system capacity is sufficient to overcome that volume reduction.

The initial cause of the loss of cooling could be the failure of a running pump in either the primary or the secondary system, in which case the response required is simply to start the

redundant pump. However, it could also be a more significant failure, such as a pipe break or a heat exchanger blockage. To simplify the model, it has been assumed that a repair is necessary. While this is conservative, it does not unduly bias the conclusions of the overall study.

If the loss of cooling was detected via the control room alarms, the staff has the full 33 hours in which to repair the system. Assuming that it takes at least 16 hours before parts and technical help arrive, then the operators have 17 hours (33 hours less 16 hours) to repair the system. Failure to repair the SFPC system event is modeled as HEP-COOL-REP-E. This case is modeled by fault tree LOC-OCS-U.

If the loss of cooling was discovered during walk-downs, it has been conservatively assumed the operator has only 9 hours available (allowing 24 hours before loss of cooling was noticed). Since it is assumed that it takes at least 16 hours before technical help and parts arrive, it is not possible that the SFPC system can be repaired before the bulk boiling would begin. Failure to repair the SFPC system event is modeled as HEP-COOL-REP-L. This case is modeled by fault tree LOC-OCS-L.

4.1.4.2 Relevant Assumptions

- The operators will avoid using raw water (e.g., water not chemically controlled) if possible. Therefore, the operators are assumed to focus solely on restoration of the SFP cooling system in the initial stages of the event.
- If the loss of cooling was detected through shift walk-downs, then 24 hours are (conservatively) assumed to have passed before discovery.
- It takes 16 hours to contact maintenance personnel, diagnose the cause of failure, and get new parts.
- Mean time to repair the SFP cooling system is 10 hours.
- Operating staff has received formal training and there are administrative procedures to guide them in initiating repair (NEI commitment no. 8).
- Repair crew is different than the onsite operators.

4.1.4.3 Quantification

<u>Human Error Probabilities</u>

The probability of failure to repair SFPC system is represented by the exponential repair model:

$$e^{-\lambda t}$$

where

λ = (inverse of mean time to repair)
t = available time

In the case where discovery was from the control room, probability of failure to repair SFPC

system event, HEP-COOL-REP-E, would be 0.18 based on 17 hours available to repair. In the case that the discovery was because of operator walk-down (HEP-COOL-REP-L), it is assumed that there is not enough time available to repair and restart the SFP makeup system in time to prevent bulk boiling, and the event has been assigned a value of 1.0.

4.1.4.4 Basic Event Probabilities

Basic Event	Basic Event Probability
HEP-COOL-REP-E	1.8E-1
HEP-COOL-REP-L	1.0

4.1.5 Top Event OFD – Operator Recovery Using Onsite Sources

4.1.5.1 Event Description and Timing

On the two upper branches of the event tree, the operators have recognized the loss of the SFPC system, and have tried unsuccessfully to restore the system. After 43 hours, the level of the pool has dropped below the suction of the SFP cooling system (see below), so that repair of that system will not have any effect until pool level is restored. The operating staff now has 88 hours to provide make-up to the pool using firewater (or other available onsite sources) to prevent fuel uncovery (131 hours less 43 hours). This event represents failure to provide make-up to the SFP. The operators have both an electric and a diesel-driven firewater pump available to perform this function. If both pumps were to fail, there may be time to repair one of the pumps. This event has been modeled by the fault tree LOC-OFD.

Given the operators were not successful in detecting the loss of cooling early enough to allow recovery of the normal cooling system, this event is modeled by functional fault tree LOC-OFD-L. At this stage, even though the operators have failed over several shifts to detect the need to respond, there would be several increasingly compelling cues available to the operators performing walk-downs, including a visibly lowered pool level and a hot and humid atmosphere. Since there are on the order of 88 hours before the level drops to 3 feet above the fuel, some credit has been taken for subsequent crews to recognize the loss of cooling and take corrective action.

4.1.5.2 Relevant Assumptions

- The operators have 88 hours to provide make-up.
- The operators will avoid using raw water (e.g., water not chemically controlled) if possible.
- The boil-off rate is assumed to be higher than the SFP makeup system capacity.
- The operators are aware that they must use raw water to refill the pool once the level drops to below the suction of the cooling system and the pool begins boiling, since the makeup system cannot compensate for the boiling.
- For repair of failed pumps, it is assumed that it takes 16 hours to contact maintenance personnel, identify the problem, and get new parts.

- There is a means to remotely align a makeup source to the SFP without entry to the refuel floor, so that make-up can be provided even when the environment is uninhabitable because of steam and/or high radiation (NEI commitment no.8).

- Repair crew is different than onsite operators.

- Mean time to repair the firewater pump is 10 hours.

- Operators have received formal training and there are procedures that include clear guidance on the use of the firewater system as a makeup system (NEI commitment no. 2).

- Firewater pumps are maintained and tested on a regular schedule (NEI commitment no. 10).

4.1.5.3 Quantification

Human Error Probabilities

Three human failure events are modeled in functional fault tree LOC-OFD HEP-RECG-FWSTART represents the operator's failure to recognize the need to initiate the firewater system. The conditions under which the firewater system is to be used are assumed to be explicit in a written procedure. This event was quantified using the SPAR HRA technique. The assumptions include expansive time (> 24 hours), a high level of stress, diagnostic type procedures, good ergonomic interface, 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 represents failure to start the electric or diesel firewater pump 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. This event was quantified using SPAR HRA technique. An expansive time (> 50 times the required time), high stress, highly complex task because of its non-routine nature, quality procedures available, as well as good ergonomics including equipment and tools matched to procedure, and crews that are conversant with the procedures and one another through training were assumed .

HEP-FW-REP-DEPEN represents the failure of the repair crew to repair a firewater pump. Note that the repair crew had failed to restore the SFPC system. Therefore, dependency was modeled in the failure to repair firewater system. We assume that the operator will focus his recovery efforts on only one pump. Assuming that it takes another two shifts (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] $\approx$ 1.0E-3. For HEP-FW-REP-DEPEN a low level of dependence was applied modifying the nominal failure probability of 1.0E-3 to 5.0E-2 using the THERP formulation for low dependence.

Functional fault tree LOC-OFD-L is similar except that basic event HEP-RECG-FWSTART is replaced by HEP-RECG-FWSTART-L. The probability of this event is 5E-2, representing a low

level of dependence because of the fact that a failure to detect the condition during the first few shifts may be indicative of a more serious underlying problem.

Hardware Failure Probabilities

Basic event FP-2PUMPS-FTF represents the failure of both firewater pumps. 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 (Ref. 14). Note that the relatively high unavailability assumed for the diesel driven firewater pump may be conservative if it is subject to a maintenance and testing program, and there are controls on availability. These individual pump failures result in a value of 6.7E-4 for event FP-2PUMPS-FTF.

4.1.5.4 Basic Event Probabilities

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

4.1.6 Top Event OFB – Operator Recovery Using Offsite Sources

4.1.6.1 Event Description and Timing

This event accounts for recovery of coolant make-up using offsite sources given the failure of recovery actions using onsite sources. Adequate time is available for this action, provided that the operating staff recognizes that recovery of cooling using onsite sources will not be successful, and that offsite sources are the only viable alternatives. This top event is quantified using fault tree LOC-OFB, for the upper two branches, and LOC-OFB-L for the lowest branch. Note that in this fault tree event HEP-INV-OFFSITE is ORed with the failure of the operator to recognize the need to start the firewater system (event HEP-RECG-FWSTART or HEP-RECG-FWSTART-L , described in Section 4.1.5.3). In essence, if the operators fail to recognize the need for firewater, it is assumed they will fail to recognize the need for other offsite sources of make-up.

4.1.6.2 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 make-up.
- Procedures explicitly state 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.

4.1.6.3 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 contributions from failure of both the diagnosis of the need to provide inventory from offsite sources, and of 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 on the failure to recognize the need to initiate the firewater system was assumed.

4.1.6.4 Basic Event Probability

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

4.1.7 Summary

Table 4.1 presents a summary of basic event probabilities used in the event tree quantification.

Based on the assumptions made, the frequency of fuel uncovery can be seen to be very low. A careful and thorough adherence to NEI commitments 2, 5, 8 and 10 is crucial to establishing the low frequency. In addition, however, the assumption that walk-downs 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 analysis has also assumed 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 analysis also assumed that the procedures and training are sufficiently clear in giving guidance on early preparation for using the alternate makeup sources.

Table 4.1 Basic Event Summary for the Loss of Cooling Event Tree

Basic Event Name	Description	Basic Event Probability
IE-LOC	Loss of SFP cooling initiating event	3.0E-3
HEP-DIAG-ALARM	Operators fail to respond to a signal indication in the control room	3.0E-4
HEP-WLKDWN-LSFPC	Operators fail to observe the loss of cooling in walk-downs (independent case)	1.0E-5
HEP-WLKDWN-DEPEN	Operators fail to observe the loss of cooling in walk-downs (dependent case)	5.0E-2
HEP-COOL-REP-E	Repair crew fails to repair SFPC system	1.8E-1
HEP-COOL-REP-L	Repair crew fails to repair SFPC system - Late	1.0
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 firewater system - dependent case	5.0E-2
HEP-FW-START	Operators fail to start firewater pump and provide alignment	1.0E-5
HEP-FW-REP-DEPEN	Repair crew fails to repair firewater system - dependent case	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
SPC-LVL-LOP	Local faults leading to alarm channel failure	2.0E-3

4.2 Internal Fire Event Tree

This event tree models the loss of SFP cooling caused by internal fires. Given a fire alarm, the operator will attempt to suppress the fire, and then attempt to re-start SFP cooling given that the SFP cooling system and offsite power feeder system have not been damaged by the fire. In the unlikely event that the operator fails to respond to the alarms or is unsuccessful in suppressing the fire, it is assumed that the SFPC system will be damaged to the extent where repair will not be possible. The operator then has to provide alternate cooling and inventory makeup – either using the site firewater system or by calling upon offsite resources. Figure 4.2 shows the Internal Fire event tree sequence progression.

4.2.1 Initiating Event FIR – Internal Fire

4.2.1.1 Event Description and Timing

The fire initiator includes those fires of sufficient magnitude, that if not suppressed, would cause a loss of cooling to the SFP. This loss of cooling could either result from damage to the SFPC system or the offsite power feeder system.

4.2.1.2 Relevant Assumptions

- Fire ignition frequencies from operating plants are assumed to be applicable at the SFP facility.

- Ignition sources from welding and cutting are expected to be insignificant. The facility configuration is expected to be stable, negating the need for modification and fabrication work requiring welding and cutting.

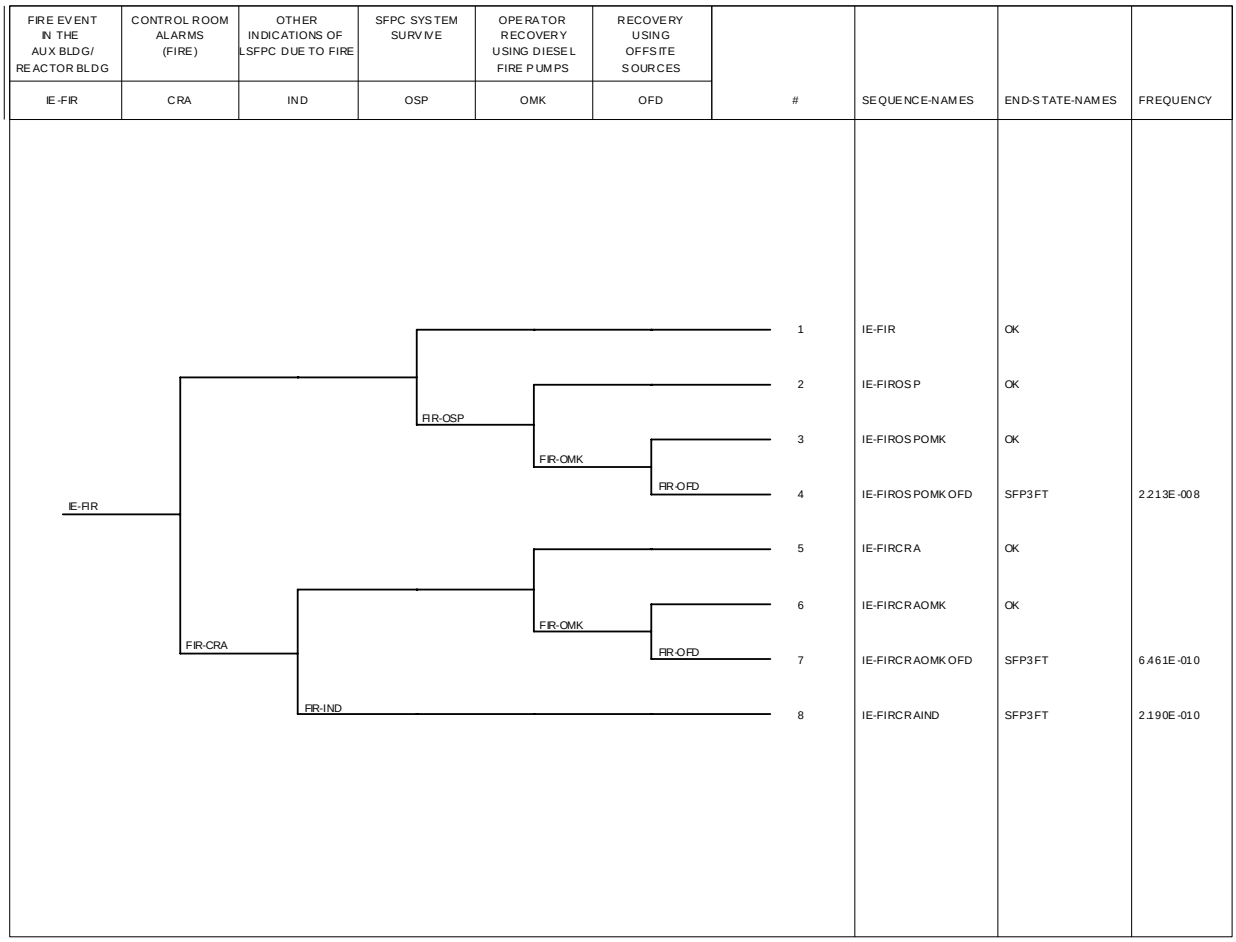
4.2.1.3 Quantification

Data compiled from historical fires at nuclear power plants is summarized in the Fire-Induced Vulnerability Evaluation (FIVE) methodology document (Ref. 15). This document identifies fire ignition sources and associated frequencies and is segregated by plant location and ignition type. Of the plant locations identified in the FIVE document, the intake structure was considered to most closely approximate the conditions and equipment associated with the SFP facilities considered in this analysis.

FIVE identifies specific frequencies associated with “electrical cabinets,” “fire pumps,” and “others” in the intake structure. In addition to these frequencies associated with specific equipment normally located in the intake structure, ignition sources from equipment (plant-wide) that may be located in the intake structure is also apportioned.

The largest ignition frequency contribution identified for intake structures is from fire pumps. In the plant configuration assumed in this study, the firewater pumps are located in an unattached structure and thus can be eliminated as ignition sources. FIVE also identifies electrical cabinets as significant ignition sources in the intake structure with an average frequency of 2.4E-3/yr. Because the number of electrical cabinets (breakers) in the spent fuel facility is expected to be less than those in the typical intake structure, a scaling factor was used to estimate the electrical cabinet contribution. Typically there are five motor-driven pumps (4 cooling pumps, 1 makeup pump) and related support equipment associated with the SPF facility. The number of electrical cabinets (breakers) was therefore estimated to be less than ten in a typical SFP facility. The number of electrical cabinets in the intake structure was estimated to be 25 (engineering judgement based on plant walk-downs). Therefore, the fire ignition frequency contribution from electrical cabinets at the SFP facility is estimated to be (10/25)(2.4E-3/yr) = 9.6E-4/yr.

Figure 4.2 Fire initiating event tree



A similar approach was used to correlate the ignition frequency for "other" to a value appropriate for the SFP facility. Intake structures typically have several pumps (e.g., circulating water, service water, screen wash, fire, etc.) as well as peripheral equipment. For this analysis, all ignition frequency associated with the "other" category was apportioned to pumps. The number of pumps in the typical intake structure was estimated to be 10 (again, engineering judgement based on plant walk-downs). Therefore, the fire ignition frequency for "other" equipment at the SFP facility is estimated to be (5/10)(3.2E-3/yr) = 1.6E-3/yr.

The contribution of ignition sources, identified as “plant-wide” sources in the FIVE document, to the ignition frequency of the SFP facility is considered to be negligible. Large ignition source contributors such as elevator motors, dryers, and MG sets do not exist in the spent fuel facility. Additionally, spontaneous cable fires are expected to be a negligible contributor because of the minimal amount of energized electrical cable. The facility configuration is expected to be stable, negating the need for modification and fabrication work requiring welding and cutting.

The fire ignition frequency for the SFP facility is therefore estimated to be 9.6E-4/yr + 1.6E-3/yr = 2.6E-3/yr. A fire frequency value of 3E-3/yr will be used in the analysis to provide additional margin and to account for any uncertainties in equipment configuration.

4.2.1.4 Basic Event Probability

Basic Event	Basic Event Probability
IE-FIRE	3.0E-3

4.2.2 Top Event CRA – Control Room Alarms

4.2.2.1 Event Description and Timing

This event represents fire detection system failure to alarm in the control room or operator failure to respond to the alarm. The proper conditions for an alarm are assumed to exist within a few minutes of fire initiation. Failure to respond could be because of operator error (failure to respond), failure of the detectors, or loss of indication because of electrical faults. Success for this event is defined as the operator recognizing the alarm and responding to the fire. Failure of this event is assumed to lead to a fire damage state where there is a loss of the SFPC system and a loss of the plant power supply system. This event is quantified by fault tree FIR-CRA and includes hardware and human failures.

4.2.2.2 Relevant Assumptions

- The SFP area is equipped with fire detectors which are alarmed in the control room. However, the area is not equipped with an automatic fire suppression system.

- Fire alarms will be activated in the control room within a few minutes of the initiation of a fire.

- Regular maintenance and testing is performed on the fire detection system and on the control room annunciators.
- Procedures are available to guide operator response to a fire, and plant operators are trained in these procedures (NEI commitment no. 2).

4.2.2.3 Quantification

<u>Human Error Probabilities</u>

One human failure event is modeled for this event (basic event HEP-DIAG-ALARM). The operator may fail to respond to a signal or indication in the control room. The source for this error rate is THERP (Table 20-23).

<u>Hardware Failure Probabilities</u>

The value used for failure of the detectors, SFP-FIRE-DETECT (5.0E-3), was taken from OREDA-92 (Ref. 14). The value used for local electrical faults leading to alarm channel failure, SFP-FIRE-LOA (2.0E-3), was estimated based on information in reference 13.

4.2.2.4 Basic Event Probabilities

Basic Event	Basic Event Probability
HEP-DIAG-ALARM	3.0E-4
SFP-FIRE-LOA	2.0E-3
SFP-FIRE-DETECT	5.0E-3

4.2.3 Top Event IND – Other Indications of Loss of Cooling

4.2.3.1 Event Description and Timing

This event models the failure of the operators to recognize the loss of SFP cooling resulting from a fire, given that either the fire alarm system failed or was not attended to. Since the assumed consequences of not attending to the alarm are a fire large enough to cause loss of power to the facility, the indications available to the operator during a walk-down include clear effects of the fire, both from visible evidence and the smell of burning, as well as the lack of power. Ultimately, if no action is taken to restore cooling, the high area temperature and humidity, and low water level from boiloff will become increasingly evident. The operators have more than 10 shifts (about 131 hours) to discover the loss of SFP cooling. Success for this event is defined as the operators recognizing the abnormal condition and understanding the need to take action within this time. This event is modeled by fault tree FIR-IND.

4.2.3.2 Relevant Assumptions

- Operators perform walk-downs once per shift (every 8 to 12 hours) and walk-downs are required to be logged.
- If the fire is discovered during the walk-down, the SFPC system is assumed to be damaged to the extent where repair will not be feasible within a few days.
- Local instrumentation and alarms are destroyed in a fire which is not extinguished within 20 minutes.
- Procedures are available to guide plant operators for off-normal conditions, and operators are trained in these procedures (NEI commitment no. 2).

4.2.3.3 Quantification

<u>Human Error Probability</u>

This event is represented by the basic event HEP-WLKDWN-LSFPC which models the operators' failure to recognize the loss of cooling during walk-downs. The failure rate was 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. Multiple opportunities for recovery were assumed.

Note that no dependency on the previous HEP was modeled. While it could be argued that, in the case where the operator has already failed to respond to control room alarms, there may be

a dependence between the event HEP-DIAG-ALARM and HEP-WLKDWN-LSFPC. However, the cues for this event are quite different. There will be obvious physical changes in the plant (e.g., loss of offsite power, a burnt out area, smoke, etc.). The only source of dependency is one where a situation would result in the operators failing to respond to control room alarms and also result in a total abandonment of plant walk-downs.

4.2.3.4 Basic Event Probability

Basic Event	Basic Event Probability
HEP-WLKDWN-LSFPC	1.0E-5

4.2.4 Top Event OSP – Fire Suppression

4.2.4.1 Event Description and Timing

This top event represents operator failure to suppress the fire before the SFP cooling system is damaged given that he responds to fire alarms. If the SFP cooling and makeup system pumps and plant power supply system are damaged to a point that they cannot be repaired in time to prevent fuel uncovery, the operator must provide cooling using available onsite (i.e., diesel fire pumps) and offsite water sources. If the fire is suppressed in time to prevent damage to SFP components, then the SFP cooling system can be restored in time to prevent fuel uncovery. The top event is represented by fault tree FIR-OSP.

4.2.4.2 Relevant Assumptions

- The automatic fire suppression system is unavailable.
- If the fire is not extinguished within 20 minutes, it is assumed that SFP cooling will be lost due either to damage of SFPC equipment, or to the plant's power supply system.
- No credit is taken for the firewater system in the suppression of the fire.
- Fire suppression extinguishers are located strategically in the SFP area, and these extinguishers are tested periodically.

4.2.4.3 Quantification

Failure of fire suppression is represented by basic event HEP-RES-FIRE. The modeling of fire growth and propagation and the determination of the effects of a fire on equipment in a room would optimally take into account the combustible loading in the room, the presence of intervening combustibles, the room size and geometry, and other characteristics such as ventilation rates and the presence of openings in the room. Because detailed inputs such as these are not applicable for a generic study such as this, fire growth and propagation was determined based on best estimate assumptions. It is assumed that the operator has 20 minutes to suppress the fire. Otherwise, it is assumed that SFP cooling will be lost (due either to damage of SFPC equipment, or to the plant's power supply system).

HEP-RES-FIRE was modeled using THERP. Because of the level of uncertainty about the size of the fire, its location, and when it is discovered, the approach taken was to model this error as a dynamic task requiring a higher level of human interaction, including keeping track of multiple functions. In addition little experience in fighting fires was assumed. Table 20-16 in THERP provides modifications of estimated HEPs for the effects of stress and experience. Using the performance shaping factors of extremely high stress (as fighting a fire would be), a dynamic task, and an operator experienced in fighting fires, this table provides an HEP of 2.5E-1.

Notes: (1) It can be argued that damage time (to disable the SFP cooling function) could be in excess of 20 minutes because typical SFP facilities are relatively large and because equipment within such facilities is usually spread out. However, in this analysis, the SFP pumps are assumed to be located in the same general vicinity with no fire barriers between them.

(2) Scenarios can be postulated where the fire damage state is less severe than that described above (e.g., fire damage to the running cooling pump, with the other pump undamaged, and with offsite power available). These scenarios can be subsumed into the "Loss of Cooling" event, and SFP cooling "recovery" in these cases would be by use of the undamaged pump train.

4.2.4.4 Basic Event Probability

Basic Event	Basic Event Probability
HEP-RES-FIRE	2.5E-1

4.2.5 Top Event OMK – Operator Recovery Using Onsite Sources

4.2.5.1 Event Description and Timing

At this point in the event tree, the SFP cooling has been lost as a result of the fire, and the operators are unable to restore the cooling system. Also, the fire has damaged the electrical system such that the motor-driven firewater pump is unavailable. If no actions are taken, SFP water level would drop to 3 ft above the top of fuel in 131 hours from the time the loss of SFP cooling occurred. This event represents failure of the operators to start the diesel-driven firewater pump and provide make-up to the SFP. If the diesel firewater pump fails, the operators have time to attempt repair. This event is modeled by fault tree FIR-OMK.

4.2.5.2 Relevant Assumptions

- There is a means to remotely align a makeup source to the SFP without entry to the refuel floor, so that make-up can be provided even when the environment is uninhabitable because of steam and/or high radiation (NEI commitment no.8).
- Inventory makeup using the firewater system is initiated by onsite operators.
- In modeling the repair of a failed firewater pump, it is assumed that it takes 16 hours to contact maintenance personnel, make a diagnosis, and get new parts.

- Mean time to repair the firewater pump is 10 hours.
- Inventory makeup using the firewater pumps is proceduralized, and the operators are trained in these procedures (NEI commitment no. 2).
- Firewater pumps are tested and maintained on a regular schedule (NEI commitment no. 10).

4.2.5.3 Quantification

Human Error Probabilities

The fault trees used to quantify this top event include three human failure events.

HEP-RECG-FWSTART represents the operators' failure to recognize the loss of SFP cooling and the need to initiate the firewater system. This event was quantified using the SPAR HRA technique. The assumptions include expansive time (> 24 hours), a high level of stress, diagnostic type procedures, good ergonomic interface, and good quality of work process. This diagnosis task provides the diagnosis for the subsequent actions taken to re-establish cooling to the pool. Although this diagnosis and subsequent actions follow a fire, no dependence between response to the fire and subsequent actions is assumed, because of the large time lag.

HEP-FW-START represents failure to start the diesel firewater pump 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 operators may have to run hoses to designated valve stations. This event HEP-FW-START was quantified using SPAR HRA technique. The following PSFs were assumed: 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. It is assume that the operators will focus their recovery efforts on only the diesel driven pump. Assuming that it takes 16 hours before technical help and parts arrive, then the operators have 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)x72] $\approx$ 1.0E-3.

Hardware Failure Probabilities

Basic event FP-DGPUMP-FTF represents the failure of the diesel-driven firewater pump. 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 1.8E-1 for failure to start and run for the diesel driven pump is used from INEL-96/0334 (Ref. 15).

4.2.5.4 Basic Event Probabilities

Basic Event	Basic Event Probability
HEP-RECG-FWSTART	2.0E-5
HEP-FW-START	1.0E-5
HEP-FW-REP-NODEP	1.0E-3
FP-DGPUMP-FTF	1.8E-1

4.2.6 Top Event OFD – Operator Recovery Using Offsite Sources

4.2.6.1 Event Description and Timing

Given the failure of recovery actions using onsite sources, this event accounts for recovery of coolant makeup using offsite sources. Adequate time is available for this action, provided that the operators recognize that recovery of cooling using onsite sources will not be successful, and that offsite sources are the only viable alternatives. This top event is quantified using fault tree FIR-OFD. This event is represented by a basic event HEP-INV-OFFSITE.

4.2.6.2 Relevant Assumptions

- The operators have 88 hours to provide make-up 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 make-up.
- Procedures explicitly state 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.

4.2.6.3 Quantification

<u>Human Error Probabilities</u>

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 the firewater pump, 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 of 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.

4.2.6.4 Basic Event Probability

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

4.2.7 Summary

Table 4.2 presents a summary of basic event probabilities used in the event tree quantification.

As in the case of the loss of cooling event, the frequency of fuel uncovery, based on the assumptions made in the analysis, is very low. The assumptions that support this low value include: careful and thorough adherence to NEI commitments 2, 5, 8 and 10; walk-downs are performed on a regular, (once per shift) (important to compensate for potential failures to the instrumentation monitoring the status of the pool); 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.

Table 4.2 Basic Event Summary for the Internal Fire Event Tree

Basic Event Name	**Description**	**Basic Event Probability**
IE-FIRE	Internal fire initiating event	3.0E-3
HEP-DIAG-ALARM	Operators fail to respond to a signal indication in the control room	3.0E-4
HEP-RES-FIRE	Operators fail to suppress fire	2.5E-1
HEP-WLKDWN-LSFPC	Operators fail to observe the loss of cooling in walk-downs (independent case)	1.0E-5
HEP-RECG-FWSTART	Operators fail to diagnoses need to start the firewater system	2.0E-5
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	1.0E-3
HEP-INV-OFFSITE	Operators fail to provide alternate sources of cooling from offsite	5.0E-2
FP-DGPUMP-FTF	Failure of firewater pump system	0.18
SFP-FIRE-LOA	Electrical faults causing loss of alarms	2.0E-3
SFP-FIRE-DETECT	Failure of fire detectors	5.0E-3

4.3 Plant-centered and Grid-related Loss of Offsite Power Event Tree

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

Figure 4.3 shows the Plant-centered and Grid-related Loss of Offsite Power (LOSP) event tree sequence progression.

4.3.1 Initiating Event LP1 – Plant-centered and Grid-related Loss of Offsite Power

4.3.1.1 Event Description

Initiating event IE-LP1 represents plant-centered and grid-related losses of offsite power. Plant-centered events typically involve hardware failures, design deficiencies, human errors (in maintenance and switching), localized weather-induced faults (e.g., lightning), or combinations of these. Grid-related events are those in which problems in the offsite power grid cause the loss of offsite power.

4.3.1.2 Quantification

For plant-centered LOSP events, NUREG/CR-5496 (Ref. 18) estimates a frequency of .04/critical year for plant centered loss of offsite power for an operating plant, and .18/unit shutdown year for a shutdown plant. For grid-related LOSP events, a frequency of 4E-3/site-yr was estimated. The frequency of grid-related losses is assumed to be directly applicable. However, neither of the plant centered frequencies is directly applicable. At a decommissioning plant there will no longer be the necessity to have the multiplicity of incoming lines typical of operating plants, which could increase the frequency of loss of offsite power from mechanical failures. On the other hand, the plant will be a normally operating facility, and it would be expected that there will be less activity and operations in the switchyard than would be expected at a shutdown plant, which would decrease the frequency of loss from human error, the dominant cause of losses for shutdown plants. For purposes of this analysis, the LOSP initiating event frequency of 0.08/yr, assumed in INEL-96/0334 (Ref. 15), is assumed for the combined losses from plant-centered and grid-related events.

4.3.2 Top Event OPR – Offsite Power Recovery

4.3.2.1 Event Description and Timing

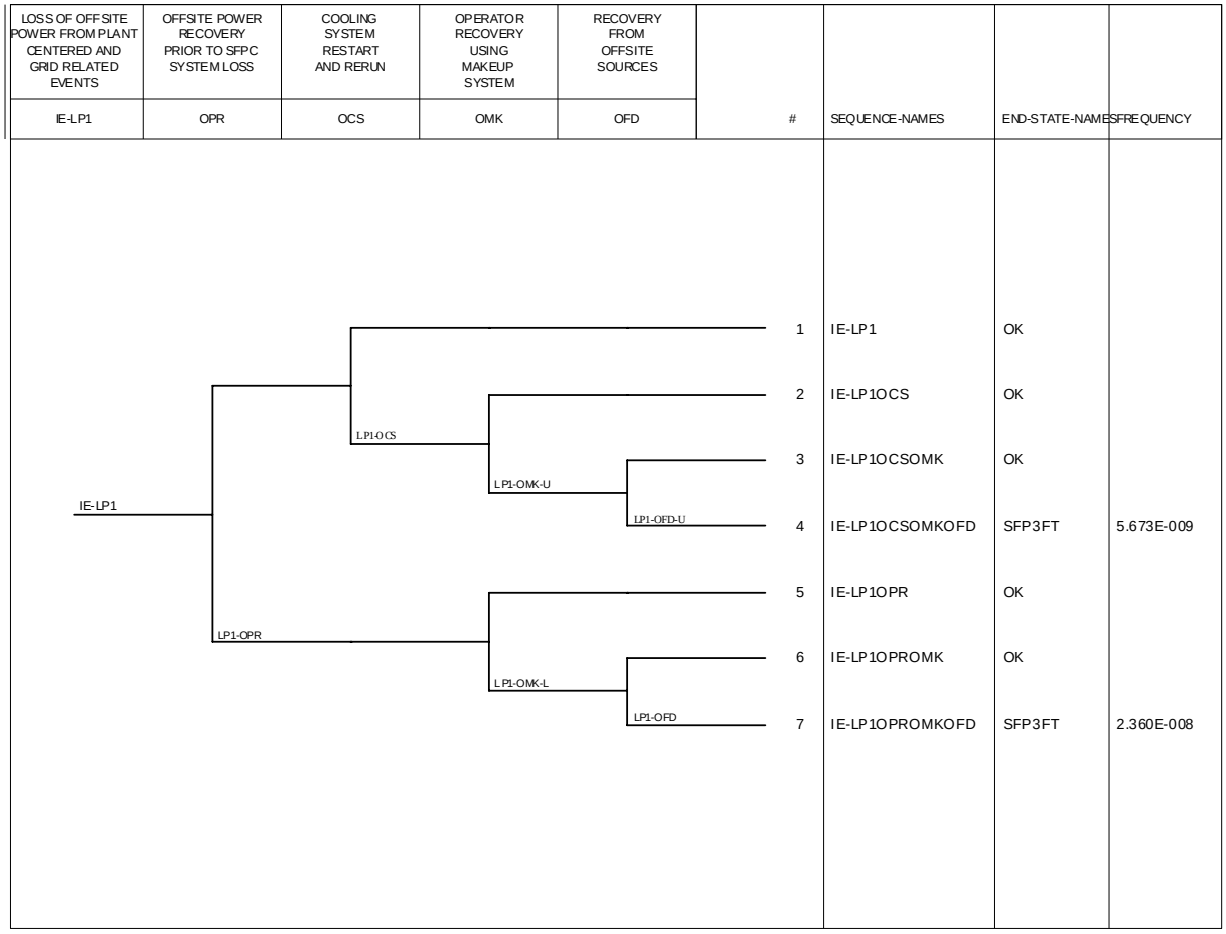
The fault tree for this top event (LP1-OPR) is a single basic event that represents the non-recovery probability of offsite power.

NUREG-1032 (Ref. 19) 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/CR-5496 (Ref. 18) divides LOSP events into three categories and estimates different values of non-recovery as functions of time.

4.3.2.2 Relevant Assumptions

- Trained electricians may not be present at the site for quick recovery from plant-centered events.

Figure 4.3 Plant centered and grid related loss of offsite power event tree



4.3.2.3 Quantification

The basic event that represents recovery of offsite power for plant-centered and grid-related LOSP 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. The majority of the plant-centered events were recovered within 7 hours, so even if there is a delay in bringing repair personnel onsite, there is a high probability of recovering offsite power within 24 hours. 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 the 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

<u>Human Error Probabilities</u>

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 because of previous failures, moderately complex task because of 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 walk-downs 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 walk-down 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 walk-down 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. 15). 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 make-up 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 SFP 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 SFP without entry to the refuel floor, so that make-up can be provided even when the environment is uninhabitable because of 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 run hoses to designated valve stations. 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 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] $\approx$ 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	5.0E-02
HEP-RECG-FWSTART	2.0E-5
HEP-FW-START	1.0E-5
HEP-FW-REP-DEPEN	5.0E-2
HEP-FW-REP-NODEP	1.0E-3
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 make-up 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 make-up, 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 commitments 2 and 4), and that preparation for this contingency is made when it is realized that it may be necessary to supplement the pool make-up.
- 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

<u>Human Error Probabilities</u>

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 includes 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 is used 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 fuel 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 walk-downs 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 the Plant-centered and Grid-related Loss of Offsite Power Event Tree

Basic Event Name	Description	Probability
IE-LP1	Loss of offsite power because of 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 because of loss of offsite power	1.0E-6
HEP-FW-REP-DEPEN	Repair crew fails to repair firewater system - dependent case	5.0E-2
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-RECG-DEPEN	Operators fail to recognize need to cool pool given prior failure	5.0E-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	1.0E-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
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
SPC-PMP-CCF	SFP cooling pumps - CCF	5.9E-4

Basic Event Name	Description	Probability
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

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 make-up.

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 an 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. 18). This includes contributions from hurricanes, snow and wind, ice, wind and salt, wind, and one tornado event, all of which occurred at a relatively small number of plants. Therefore, for the majority of sites, this frequency is conservative, whereas, for a few sites it is non-conservative. Because of their potential for severe localized damage, tornados were analyzed separately in Appendix 2E.

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. 19) 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 because of 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 because of severe weather event (REC-OSP-SW) of 2.0E-2 is used.

4.4.2.4 Basic Event Probability

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

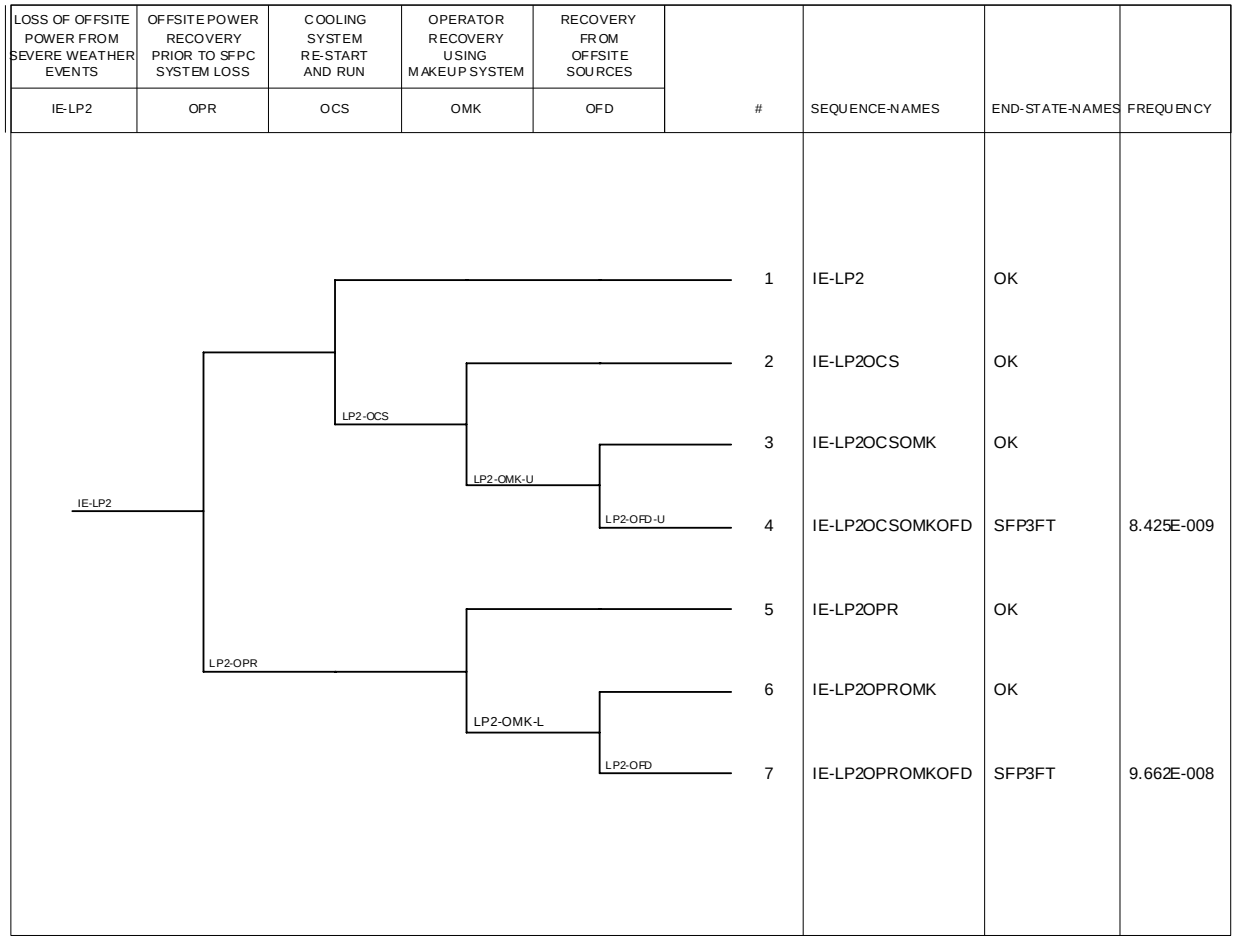
4.4.3 Top Event OCS – Cooling System Restart and Run

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 SFP 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 make-up.

Figure 4.4 Severe weather related loss of offsite power event tree



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

<u>Human Error Probabilities</u>

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 walk-downs 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 (because of potential complications from severe weather), diagnostic procedures, and good work processes. A low dependence was applied to the walk-down 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 because of severe weather, moderately complex task because of potential valve lineups and severe weather, poor ergonomics because of 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	1.0E-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 SFP 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 SFP without entry to the refuel floor, so that make-up can be provided even when the environment is uninhabitable because of 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.

- 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

<u>Human Error Probabilities</u>

The fault tree LP2-OMK-U has five operator actions, and LP2-OMK-l 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 run the fire hoses to designated valve stations. 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 because of 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)] $\approx$ 1.8E-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.

<u>Non-HEP Probabilities</u>

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.

The dependency between makeup water supply (e.g., fragility of the fire water supply tank) to events that may have caused the loss of offsite power (such as high winds) is assumed to be bounded by the dependency modeled in the HEPs.

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	1.8E-2
FP-2PUMPS-FTF	6.7E-4
FP-DGPUMP-FTF	1.8E-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 contain 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 make-up.
- 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-OFFST-SW represents failure 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 includes failures of both the diagnosis of the need to provide inventory from offsite sources, and the action itself. The contribution from the failure to diagnose is assessed by assuming a low level of dependence to account for the possible detrimental effects of the failure to complete prior tasks successfully. A relatively low contribution of 3E-02 is assumed for failure to complete the task, based on the fact that there are between five and six days for recovery of the infrastructure following a severe weather event. This results in a total HEP of 8E-02. NEI commitments 3 and 4 provide a basis for this relatively low number.

4.4.5.4 Basic Event Probability

Basic Event	**Basic Event Probability**
HEP-INV-OFFST-SW	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 fuel 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 walk-downs 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 onsite and offsite 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 Event Tree

Basic Event Name	**Description**	**Basic Event Probability**
IE-LP2	LOSP event because of severe-weather-related causes	1.1E-02

Basic Event Name	Description	Basic Event Probability
HEP-DIAG-SFPLP2	Operators fail to diagnose loss of SFP cooling because of loss of offsite power	1.0E-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	1.8E-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
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 events such as dropped loads, tornado missiles, or seismic events. The following assumption was 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 boil-off. This assumption may be significant, and potentially non-conservative for sites that do not adopt NEI commitments 6 and 7.

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. 14), 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. 14) 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 make-up 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. Although a range of leak rates is possible, the larger leak rates are postulated to be from failures in gates, seals, or from large siphoning events, and NEI commitments 6 and 7 will go a considerable way toward minimizing these events.

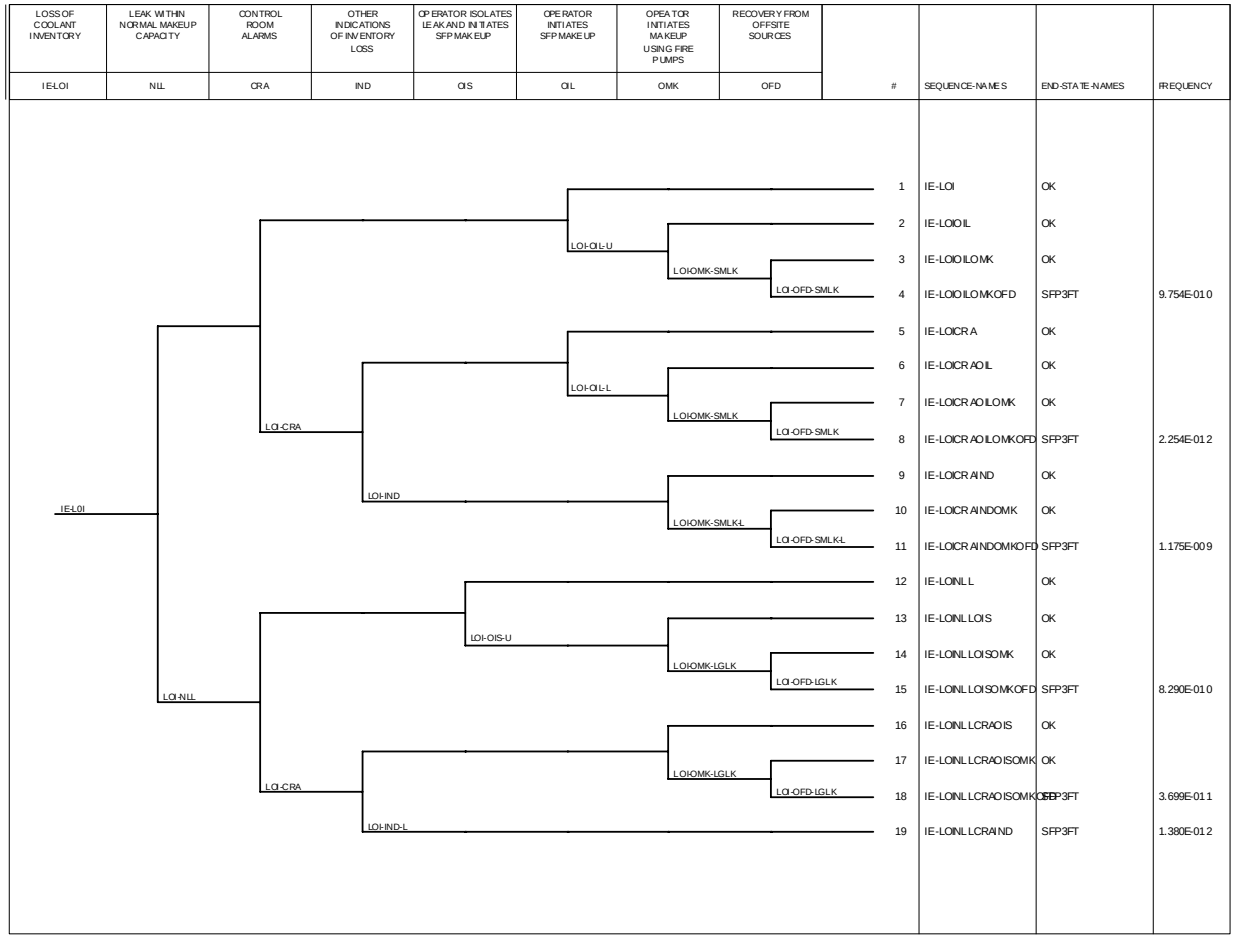
- The small leak is assumed for analysis purposes to be at the limit of the makeup system capacity, i.e., 30 gpm.

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 SFP. 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 because of 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-LOP (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-LOP	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 walk-downs over subsequent shifts. Indications available to the operators include read-outs 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 hours 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 SFP 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 because of 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 perform walk-downs 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 walk-downs 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 walk-downs, 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 Make-up

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 offsite resources.

The critical action 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 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 walk-downs, 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.

- With an assumed leak rate of 60 gpm, the operator has in excess of 4 hrs to isolate the leak and provide make-up.

- 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).

- SFP 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-U, one for failure to start the SFP makeup pump, HEP-MKUP-START-E, 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-E, it was assumed that the operator is 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 because of 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 an 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 make-up.

4.5.5.4 Basic Event Probabilities

Basic Event	Basic Event Probability
HEP-LEAK-ISO	1.3E-3
HEP-MKUP-START-E	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 SFP 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 makeup system can be achieved in less than 10 minutes.

4.5.6.3 Quantification

<u>Human Error Probabilities</u>

In the case of an early response, the operator would have more than 8 hours available to establish SFP make-up 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 make-up and is represented by the basic event HEP-MKUP-START-E (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-E, the time available is not as extensive, and is considered nominal, all other PSFs being equal.

Hardware Failure Probabilities

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

4.5.6.4 Basic Event Proababilities

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 Make-up 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 a 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 a firewater system. In all cases neither of the firewater pumps would be initially unavailable.

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 SFP without entry to the refuel floor so that make-up can be provided even when the environment is uninhabitable because of 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-driven 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 HEP-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 HEP-RECG-FWSTART is replaced by HEP-RECG-FWSTART-L. This event requires that the operators recognize that the deteriorating conditions in the SFP are because of an inventory loss. The cues will include pool heat up because of the loss of SFP 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 run hoses to designated valve stations, therefore, expansive time is assumed, with all other PSFs 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 because of 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 make-up 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, the operators have about 49 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	5.0E-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 fuel 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 walk-downs 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 walk-downs, 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 walk-downs (independent case)	1.0E-5
HEP-WLKDWN-DEPEN	Operators fail to observe the LOI event walk-downs (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 make-up	6.0E-2
HEP-MKUP-START	Operators fail to start make-up(small leak)	2.5E-6
HEP-MKUP-START-E	Operators fail to start make-up(Early Respond)	2.5E-4
HEP-MKUP-START-L	Operators fail to start make-up(Late Respond)	1.0
SFP-REGMKUP-F	Regular SFP make-up system fails	5.0E-2
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, and 5) severe weather induced losses of offsite power. The total frequency for the endstate is estimated to be 1.8E-7/year. Table 5.1 summarizes the fuel uncovery frequency for each initiator.

This frequency is to be compared with the pool performance guideline (PPG). This guideline has been established by analogy with the acceptance guidelines in RG. 1.174. In RG 1.174 it was determined that the mean value of the distribution characterizing uncertainty is the appropriate value to compare the guideline. However, it was determined that it is also necessary to investigate whether there are modeling uncertainties that could affect the decision made with respect to whether the guidelines have been met. This is the approach that has been followed here.

5.1 Characterization of Uncertainty

The frequencies are point estimates, based on the use of point estimates for the input parameters. The input parameter values were taken from a variety of sources, and in many cases were presented as point estimates with no characterization of uncertainty. In some cases, such as the initiating event frequencies derived from NUREG/CR 5496, and the HEPs derived from THERP, an uncertainty characterization was given, and the point estimates chosen corresponded to the mean values of the distributions characterizing uncertainty. For all other parameters, it was assumed that the values would be the mean values of distributions characterizing the uncertainty of the parameter value. In the case of SPAR HEPs, the authors of the SPAR HRA approach consider their estimates as mean values based on the fact that the numbers were established on the basis of considering several different sources, most of which specified mean values. Consequently, the results of this analysis are interpreted as being mean values. A propagation of parameter uncertainty through the model was not performed, nor was it considered necessary. With the exception of the SFP cooling system itself, the systems relied on are single train systems. The dominant failure contributions for the SFP cooling system are assumed to be common cause failures. Thus there are no dominant cutsets in the solutions that involved multiple repetitions of the same parameter, and under these conditions, use of mean values as input parameters produces a very close approximation to mean values of sequence frequencies. Since typical uncertainty characterization for the input parameters is a lognormal distribution with error factors of 3 or 10, the 95th percentile of the output distribution will be no more than a factor of three higher than the mean value. This is not significant to change the conclusion of the analysis.

The numerical results are a function of the assumptions made and in particular, the model used to evaluate the human error probabilities. The staff believes the models used are appropriate for the purpose of this analysis, and in particular are capable of incorporating the relevant performance shaping factors to demonstrate that low levels of risk are achievable, given an appropriate level of attention to managing the facility with a view to ensuring the health and safety of the public. Alternate HRA models could result in frequencies that are different. However, given the time scales involved, and the simplicity of the systems, we believe that the

conclusions of this study, namely that, when the NEI commitments are appropriately implemented the risks are low, are robust.

Certain assumptions may be identified as having the potential for significantly influencing the results. For example, the calculated time windows associated with the loss of inventory event tree are sensitive to the assumptions about the leak size. The SPAR HRA method is, however, not highly sensitive to the time windows assumed, primarily making a distinction between time windows that represent an inadequate time, barely adequate, nominal, extra time, and expansive time. The precise definitions of these terms can be found in Reference 9. Consequently, the assumption of the large leak rate as 60 gpm is not critical. For the loss of inventory event tree, the assumption that the leak is self-limiting after a drop in level of 15 feet, may be a more significant assumption that, on a site specific basis may be non-conservative, and requires validation. The assumption that the preparation time of several days is adequate to bring offsite sources to bear may be questioned in the case of extreme conditions. However, the very conservative assumption that this is guaranteed to fail would change the corresponding event sequences by about an order of magnitude, which would still be a very low risk contributor.

5.2 Conclusions

The analysis shows that, based on the assumptions made, the frequency of fuel 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; walk-downs 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 onsite and offsite 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 make-up to the SFP 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 for the loss of inventory events from its historical levels.

This analysis has, demonstrated to the staff that, given an appropriate implementation of the NEI commitments, the risk is indeed low, and would warrant consideration of granting exemptions. Without credit for these commitments, the risk will be more than an order of magnitude higher.

Table 5.1 Summary of Results

Initiating Event	**Fuel Uncovery Frequency (per year)**
Internal Fires	2.3E-08
Loss of Cooling	1.4E-08
Loss of Inventory	3.0E-09
Loss of Offsite Power (plant centered & grid-related events)	2.9E-8
Loss of Offsite Power (severe weather events)	1.1E-7
TOTAL	1.8E-07

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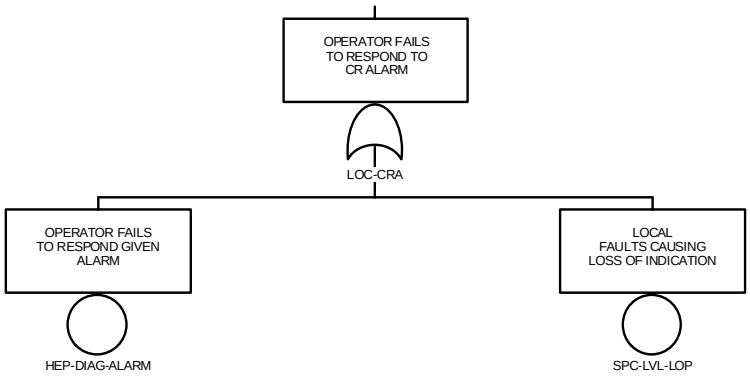
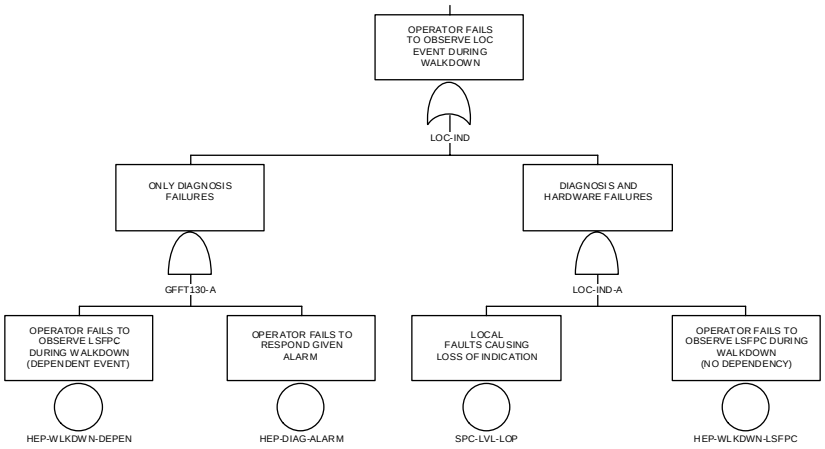
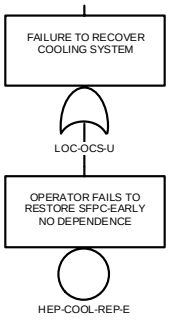
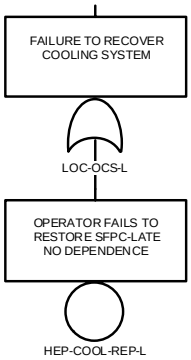
ATTACHMENT A

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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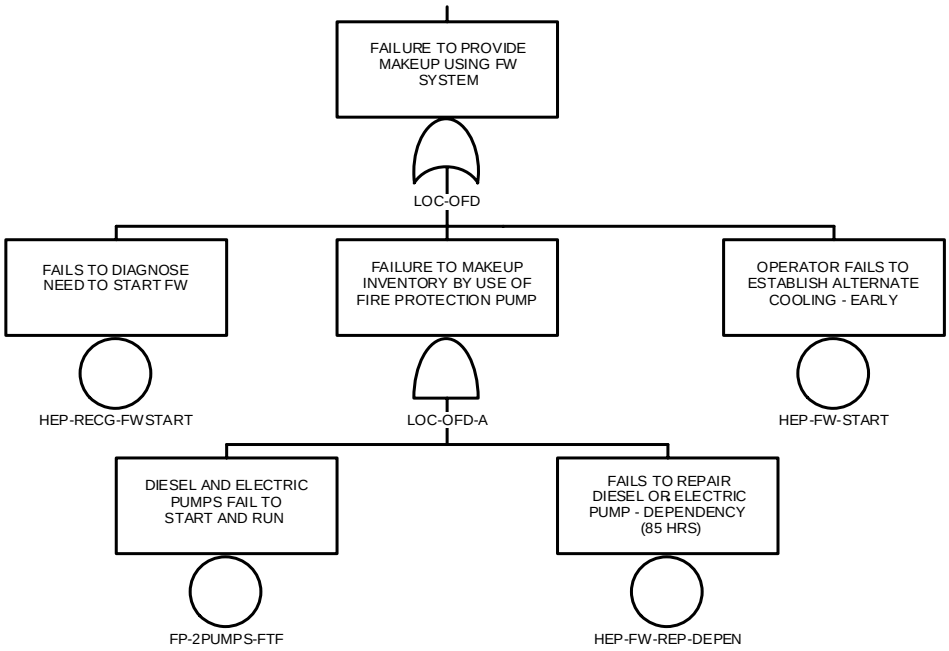
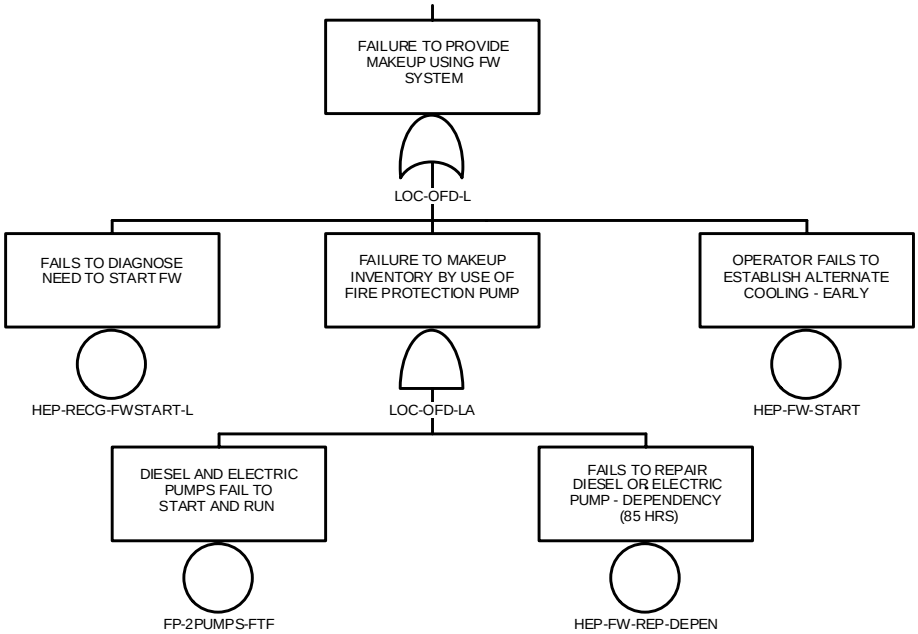
LOC-OCS-L
OPERATOR FAILS TO RESTORE SFPC-LATE NO DEPENDENCE
HEP-COOL-REP-L


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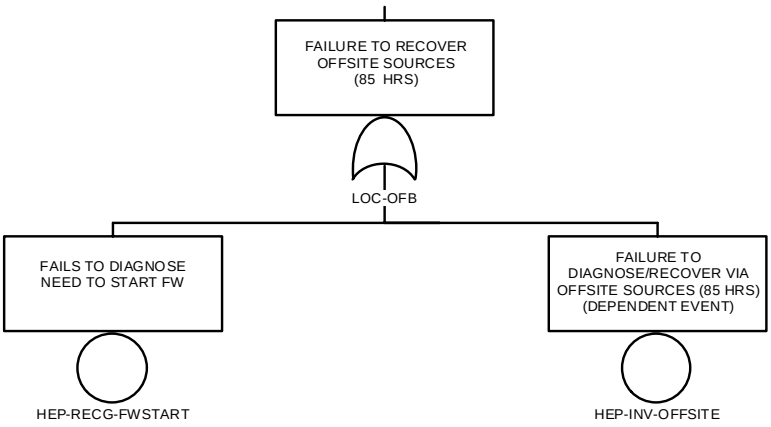
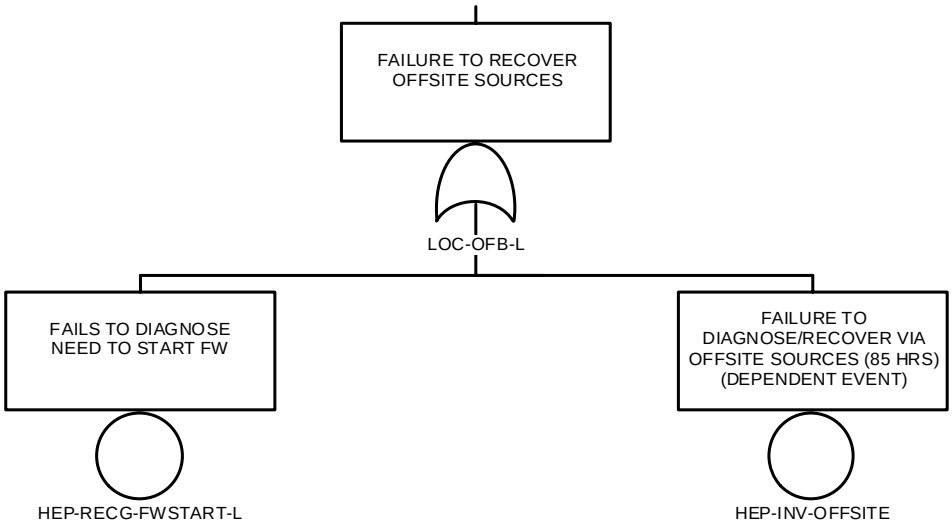
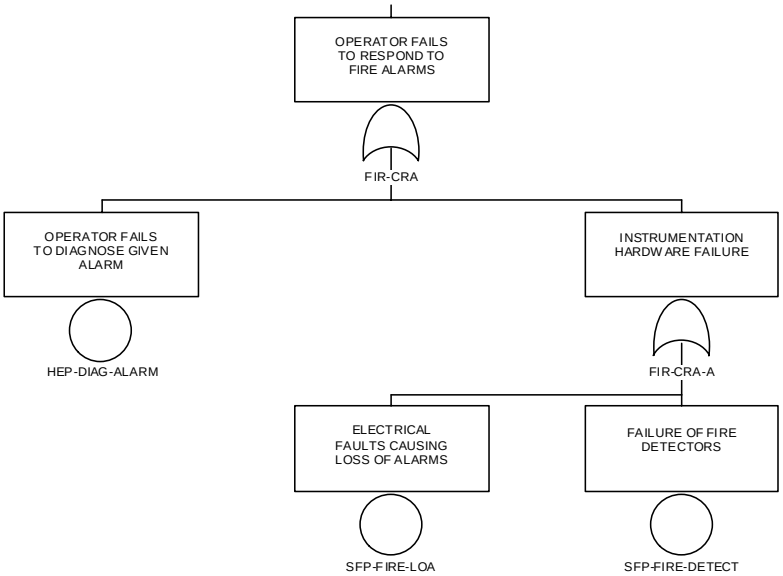
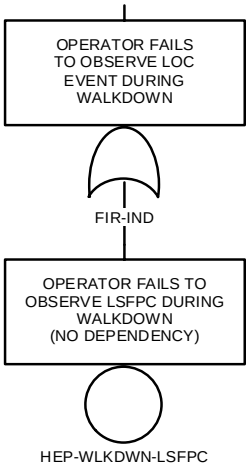
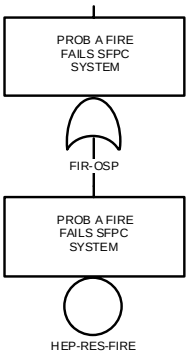
LOC-OFD-L
FAILS TO DIAGNOSE NEED TO START FW
FAILURE TO MAKEUP INVENTORY BY USE OF FIRE PROTECTION PUMP
OPERATOR FAILS TO ESTABLISH ALTERNATE COOLING - EARLY
HEP-RECG-FWSTART-L
LOC-OFD-LA
HEP-FW-START
DIESEL AND ELECTRIC PUMPS FAIL TO START AND RUN
FAILS TO REPAIR DIESEL OR ELECTRIC PUMP - DEPENDENCY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

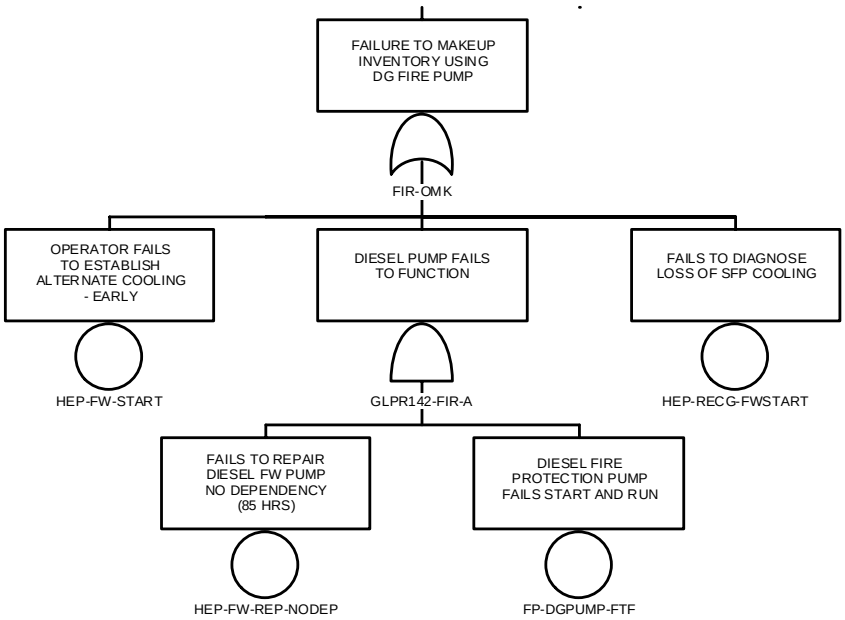

FAILURE TO RECOVER OFFSITE SOURCES (85 HRS)
LOC-OFB
FAILS TO DIAGNOSE NEED TO START FW
FAILURE TO DIAGNOSE/RECOVER VIA OFFSITE SOURCES (85 HRS) (DEPENDENT EVENT)
HEP-RECG-FWSTART
HEP-INV-OFFSITE

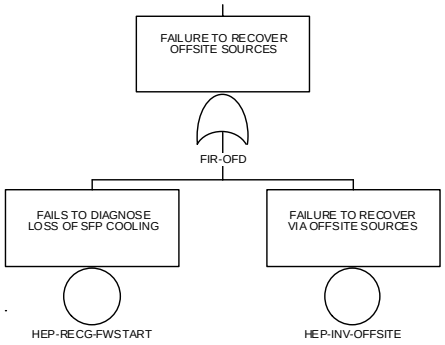

FAILURE TO RECOVER OFFSITE SOURCES
LOC-OFB-L
FAILS TO DIAGNOSE NEED TO START FW
FAILURE TO DIAGNOSE/RECOVER VIA OFFSITE SOURCES (85 HRS) (DEPENDENT EVENT)
HEP-RECG-FWSTART-L
HEP-INV-OFFSITE

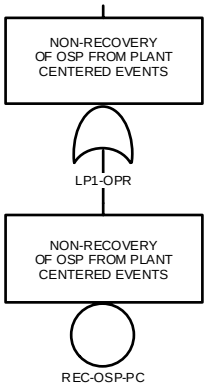

OPERATOR FAILS TO RESPOND TO FIRE ALARMS
FIR-CRA
OPERATOR FAILS TO DIAGNOSE GIVEN ALARM
INSTRUMENTATION HARDWARE FAILURE
HEP-DIAG-ALARM
FIR-CRA-A
ELECTRICAL FAULTS CAUSING LOSS OF ALARMS
FAILURE OF FIRE DETECTORS
SFP-FIRE-LOA
SFP-FIRE-DETECT

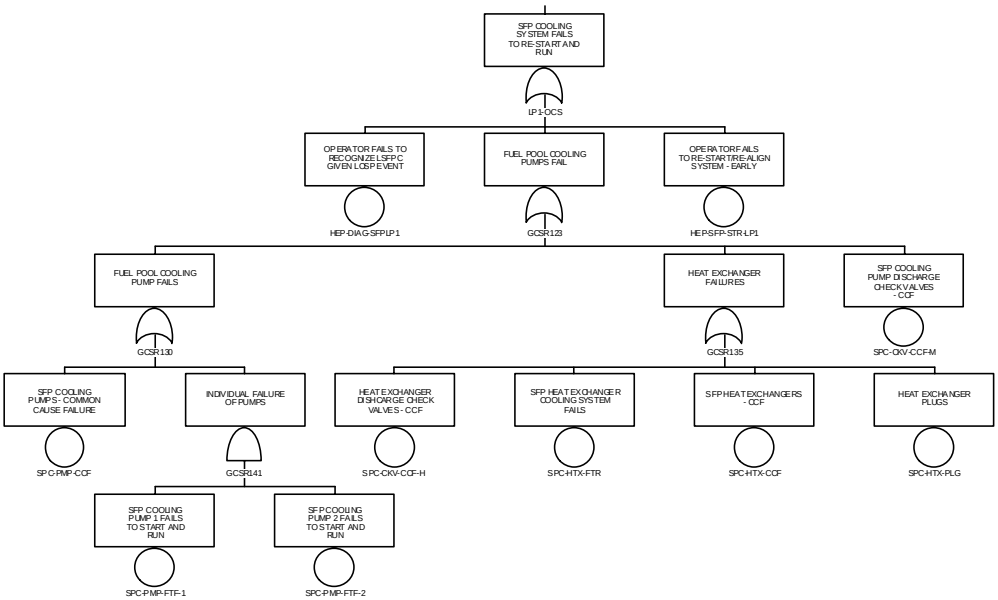

OPERATOR FAILS TO OBSERVE LOC EVENT DURING WALKDOWN
FIR-IND
OPERATOR FAILS TO OBSERVE LSFPC DURING WALKDOWN (NO DEPENDENCY)
HEP-WLKDWN-LSFPC

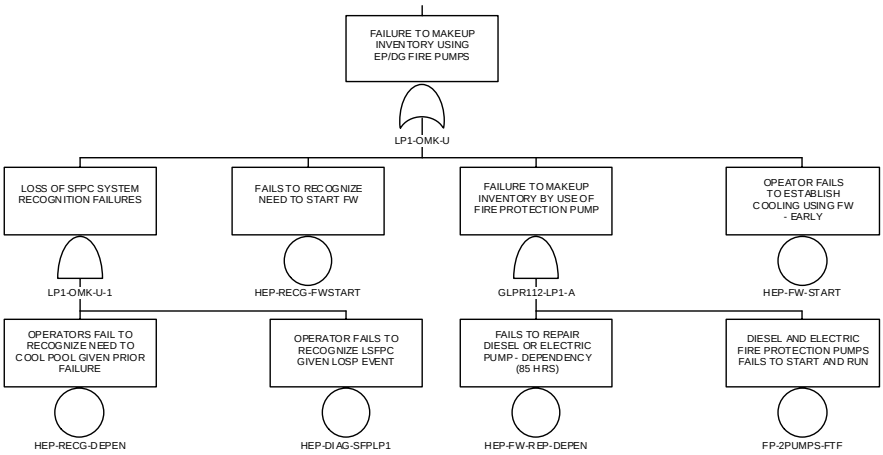

PROB A FIRE
FAILS SFPC
SYSTEM
FIR-OSP
PROB A FIRE
FAILS SFPC
SYSTEM
HEP-RES-FIRE

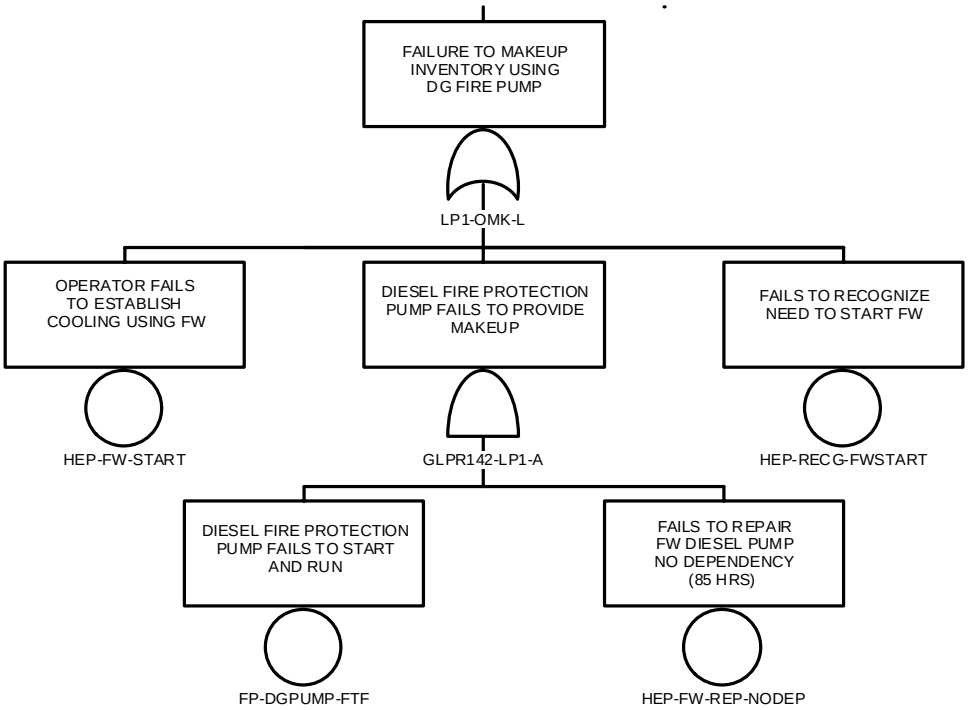

FAILURE TO MAKEUP
INVENTORY USING
DG FIRE PUMP
FIR-OMK
OPERATOR FAILS
TO ESTABLISH
ALTERNATE COOLING
- EARLY
DIESEL PUMP FAILS
TO FUNCTION
FAILS TO DIAGNOSE
LOSS OF SFP COOLING
HEP-FW-START
GLPR142-FIR-A
HEP-RECG-FWSTART
FAILS TO REPAIR
DIESEL FW PUMP
NO DEPENDENCY
(85 HRS)
DIESEL FIRE
PROTECTION PUMP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

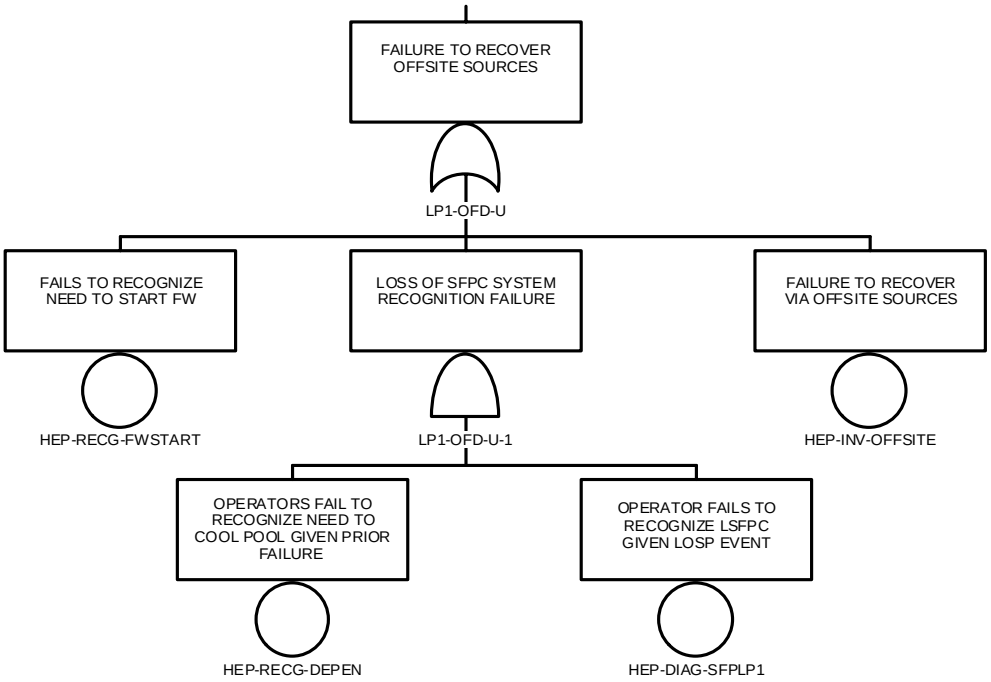

FAILURE TO RECOVER OFFSITE SOURCES
FIR-OFD
FAILS TO DIAGNOSE LOSS OF SFP COOLING
FAILURE TO RECOVER VIA OFFSITE SOURCES
HEP-RECG-FWSTART
HEP-INV-OFFSITE

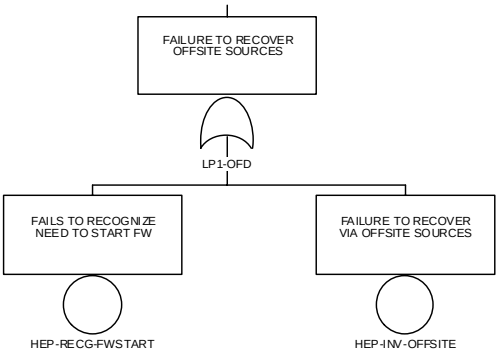

NON-RECOVERY OF OSP FROM PLANT CENTERED EVENTS
LP1-OPR
NON-RECOVERY OF OSP FROM PLANT CENTERED EVENTS
REC-OSP-PC

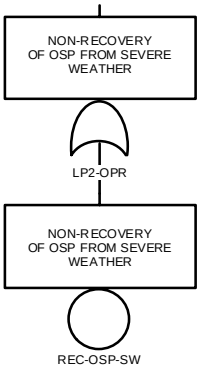

SFP COOLING SYSTEM FAILS TO RE-START AND RUN
LP1-OCS
OPERATOR FAILS TO RECOGNIZE LSFPC GIVEN LOSP EVENT
FUEL POOL COOLING PUMPS FAIL
OPERATOR FAILS TO RE-START/RE-ALIGN SYSTEM - EARLY
HEP-DIAG-SFPLP1
GCSR123
HEP-SFP-STR-LP1
FUEL POOL COOLING PUMP FAILS
HEAT EXCHANGER FAILURES
SFP COOLING PUMP DISCHARGE CHECK VALVES - CCF
GCSR130
GCSR135
SPC-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
SPC-PMP-CCF
GCSR141
SPC-CKV-CCF-H
SPC-HTX-FTR
SPC-HTX-CCF
SPC-HTX-PLG
SFP COOLING PUMP 1 FAILS TO START AND RUN
SFP COOLING PUMP 2 FAILS TO START AND RUN
SPC-PMP-FTF-1
SPC-PMP-FTF-2

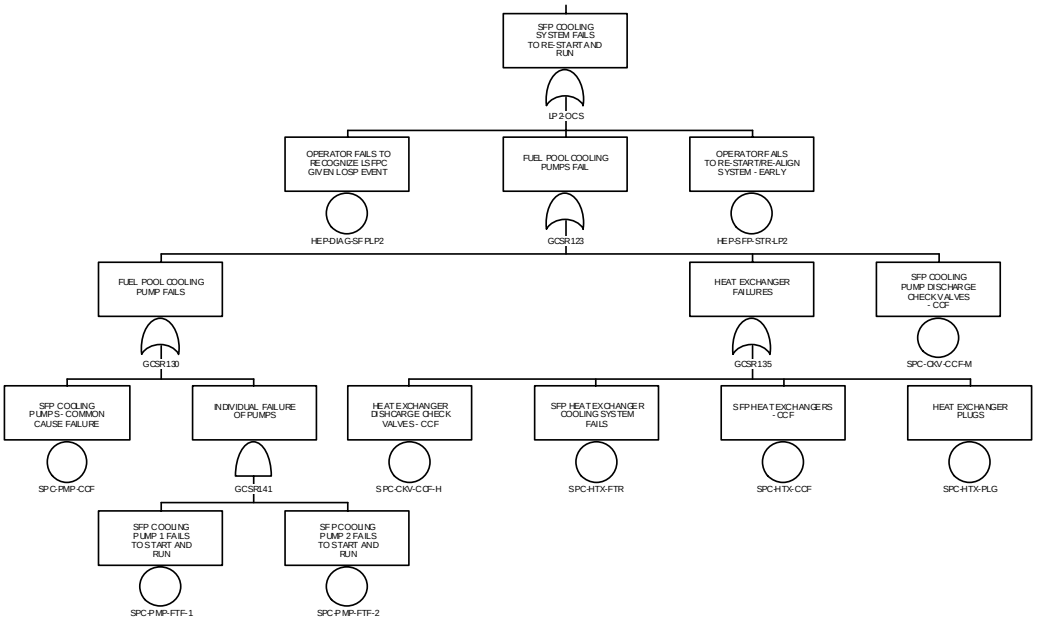

FAILURE TO MAKEUP INVENTORY USING EP/DG FIRE PUMPS
LP1-OMK-U
LOSS OF SFPC SYSTEM RECOGNITION FAILURES
FAILS TO RECOGNIZE NEED TO START FW
FAILURE TO MAKEUP INVENTORY BY USE OF FIRE PROTECTION PUMP
OPEATOR FAILS TO ESTABLISH COOLING USING FW - EARLY
LP1-OMK-U-1
HEP-RECG-FWSTART
GLPR112-LP1-A
HEP-FW-START
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

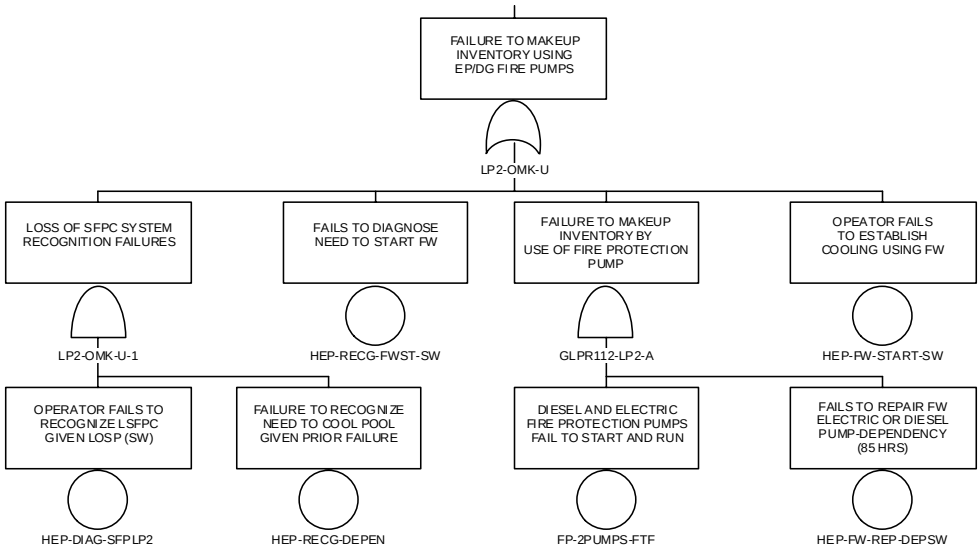

FAILURE TO MAKEUP INVENTORY USING DG FIRE PUMP
LP1-OMK-L
OPERATOR FAILS TO ESTABLISH COOLING USING FW
DIESEL FIRE PROTECTION PUMP FAILS TO PROVIDE MAKEUP
FAILS TO RECOGNIZE NEED TO START FW
HEP-FW-START
GLPR142-LP1-A
HEP-RECG-FWSTART
DIESEL FIRE PROTECTION PUMP FAILS TO START AND RUN
FAILS TO REPAIR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

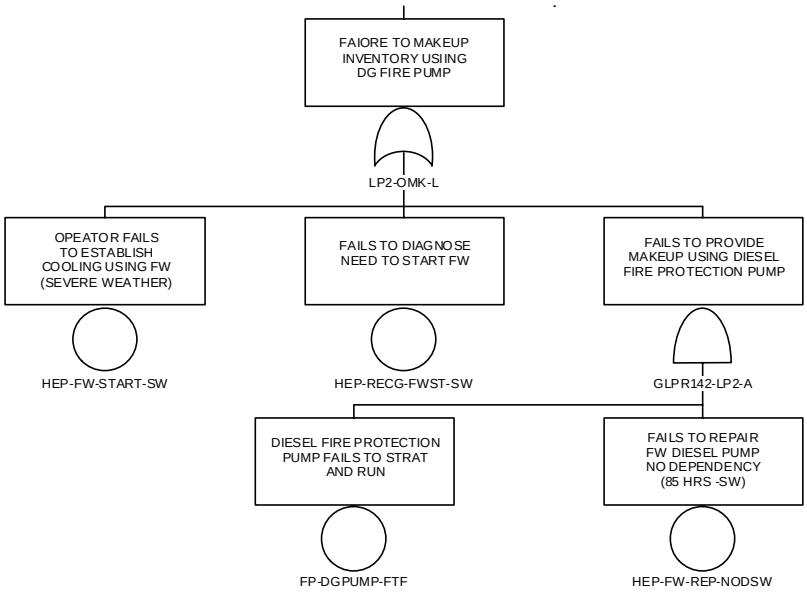

FAILURE TO RECOVER OFFSITE SOURCES
LP1-OFD-U
FAILS TO RECOGNIZE NEED TO START FW
LOSS OF SFPC SYSTEM RECOGNITION FAILURE
FAILURE TO RECOVER VIA OFFSITE SOURCES
HEP-RECG-FWSTART
LP1-OFD-U-1
HEP-INV-OFFSITE
OPERATORS FAIL TO RECOGNIZE NEED TO COOL POOL GIVEN PRIOR FAILURE
OPERATOR FAILS TO RECOGNIZE LSFPC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

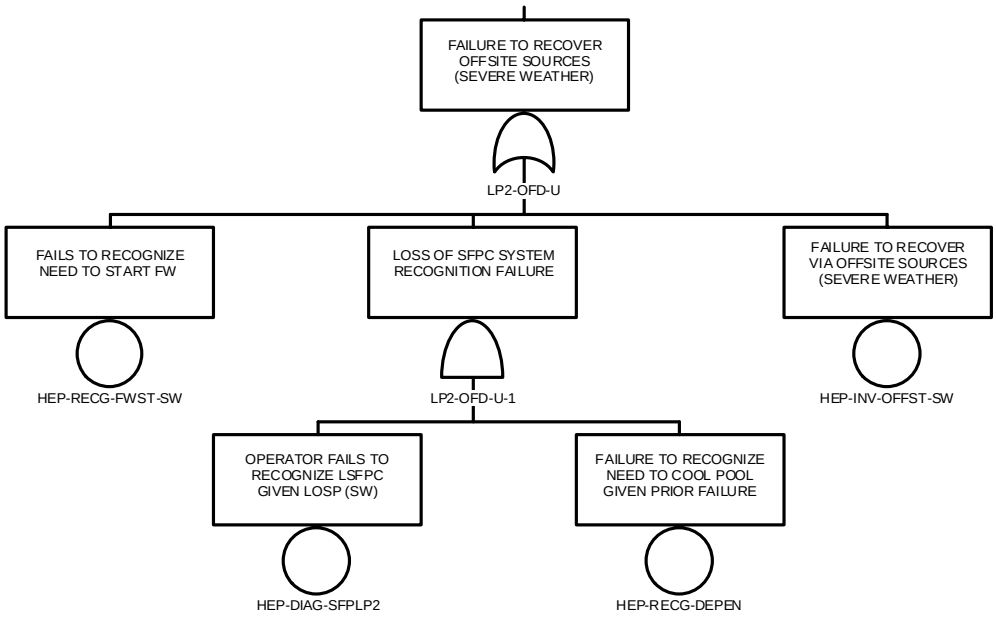

FAILURE TO RECOVER OFFSITE SOURCES
LP1-OFD
FAILS TO RECOGNIZE NEED TO START FW
FAILURE TO RECOVER VIA OFFSITE SOURCES
HEP-RECG-FWSTART
HEP-INV-OFFSITE

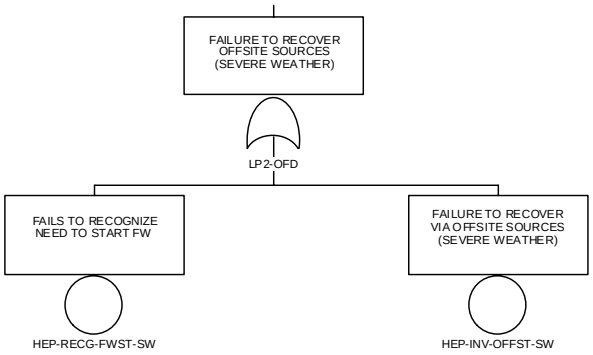

NON-RECOVERY OF OSP FROM SEVERE WEATHER
LP2-OPR
NON-RECOVERY OF OSP FROM SEVERE WEATHER
REC-OSP-SW

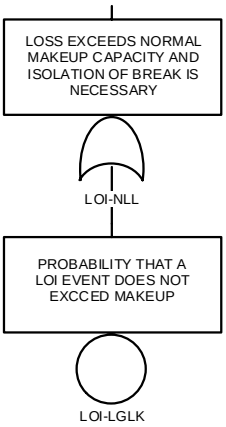

SFP COOLING SYSTEM FAILS TO RE-START AND RUN
LP2-OCS
OPERATOR FAILS TO RECOGNIZE LSFPC GIVEN LOSP EVENT
FUEL POOL COOLING PUMPS FAIL
OPERATOR FAILS TO RE-START/RE-ALIGN SYSTEM - EARLY
HEP-DIAG-SF-PLP2
GCSR123
HEP-SFP-STR-LP2
FUEL POOL COOLING PUMP FAILS
HEAT EXCHANGER FAILURES
SFP COOLING PUMP DISCHARGE CHECK VALVES - CCF
GCSR130
GCSR135
SPC-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
SPC-PMP-CCF
GCSR141
SPC-CKV-CCF-H
SPC-HTX-FTR
SPC-HTX-CCF
SPC-HTX-PLG
SFP COOLING PUMP 1 FAILS TO START AND RUN
SFP COOLING PUMP 2 FAILS TO START AND RUN
SPC-PMP-FTF-1
SPC-PMP-FTF-2

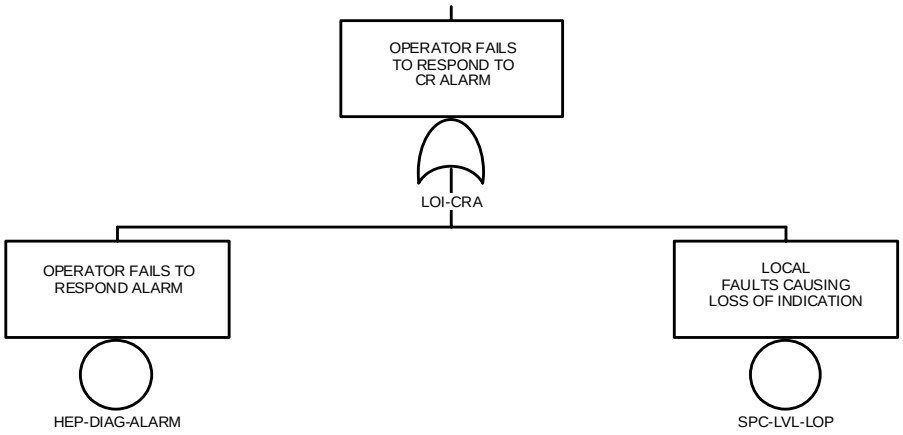

FAILURE TO MAKEUP INVENTORY USING EP/DG FIRE PUMPS
LP2-OMK-U
LOSS OF SFPC SYSTEM RECOGNITION FAILURES
FAILS TO DIAGNOSE NEED TO START FW
FAILURE TO MAKEUP INVENTORY BY USE OF FIRE PROTECTION PUMP
OPEATOR FAILS TO ESTABLISH COOLING USING FW
LP2-OMK-U-1
HEP-RECG-FWST-SW
GLPR112-LP2-A
HEP-FW-START-SW
OPERATOR 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

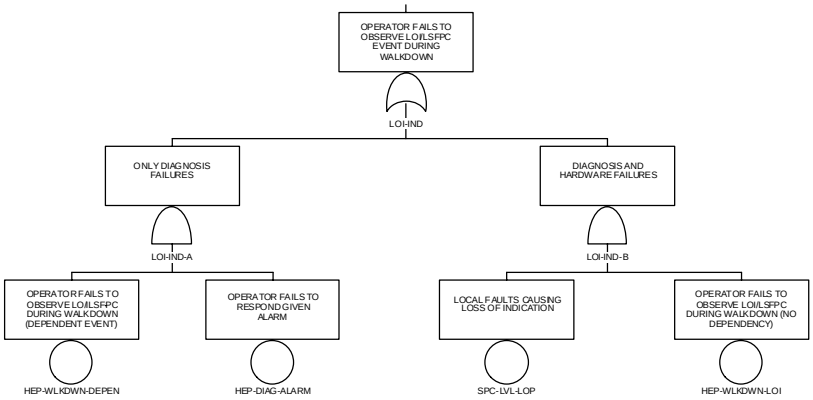

FAIORE TO MAKEUP INVENTORY USIING DG FIRE PUMP
LP2-OMK-L
OPEATOR FAILS TO ESTABLISH COOLING USING FW (SEVERE WEATHER)
FAILS TO DIAGNOSE NEED TO START FW
FAILS TO PROVIDE MAKEUP USING DIESEL FIRE PROTECTION PUMP
HEP-FW-START-SW
HEP-RECG-FWST-SW
GLPR142-LP2-A
DIESEL FIRE PROTECTION PUMP FAILS TO STRAT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

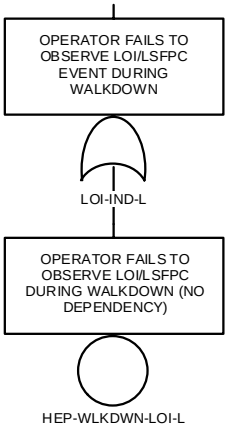

FAILURE TO RECOVER OFFSITE SOURCES (SEVERE WEATHER)
LP2-OFD-U
FAILS TO RECOGNIZE NEED TO START FW
LOSS OF SFPC SYSTEM RECOGNITION FAILURE
FAILURE TO RECOVER VIA OFFSITE SOURCES (SEVERE WEATHER)
HEP-RECG-FWST-SW
LP2-OFD-U-1
HEP-INV-OFFST-SW
OPERATOR FAILS TO RECOGNIZE LSFPC GIVEN LOSP (SW)
FAILURE TO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

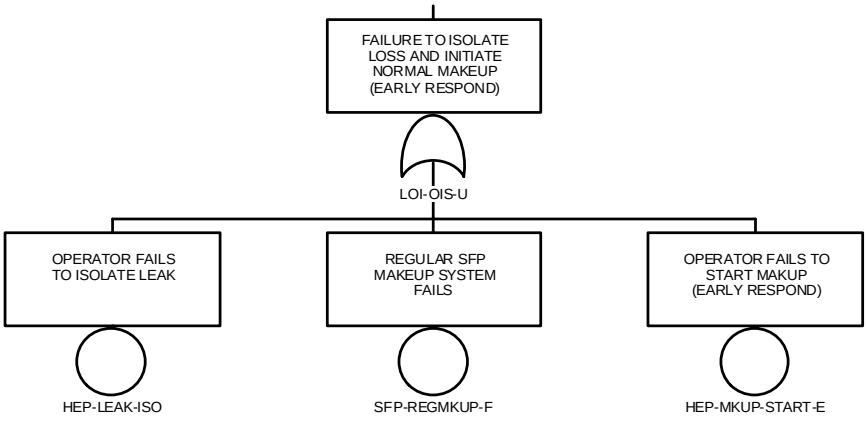

FAILURE TO RECOVER OFFSITE SOURCES (SEVERE WEATHER)
LP2-OFD
FAILS TO RECOGNIZE NEED TO START FW
FAILURE TO RECOVER VIA OFFSITE SOURCES (SEVERE WEATHER)
HEP-RECG-FWST-SW
HEP-INV-OFFST-SW

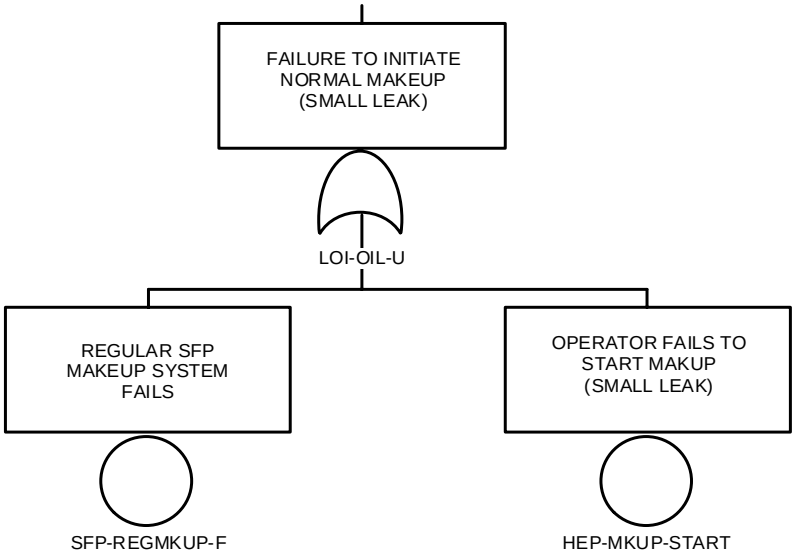

LOSS EXCEEDS NORMAL MAKEUP CAPACITY AND ISOLATION OF BREAK IS NECESSARY
LOI-NLL
PROBABILITY THAT A LOI EVENT DOES NOT EXCCED MAKEUP
LOI-LGLK

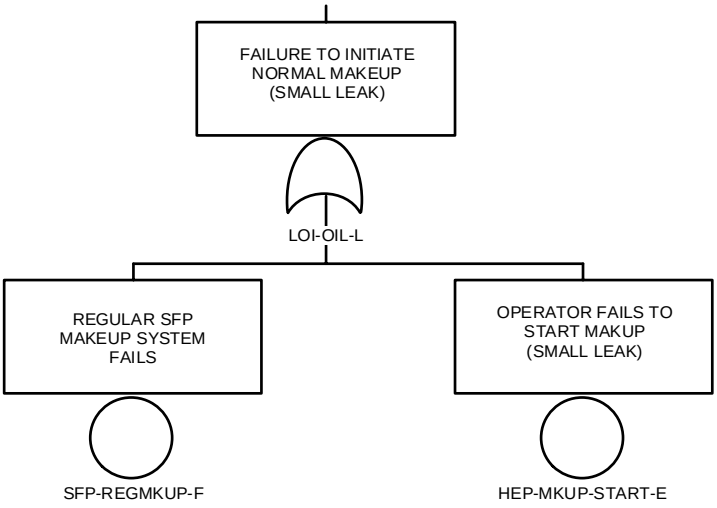

OPERATOR FAILS TO RESPOND TO CR ALARM
LOI-CRA
OPERATOR FAILS TO RESPOND ALARM
LOCAL FAULTS CAUSING LOSS OF INDICATION
HEP-DIAG-ALARM
SPC-LVL-LOP

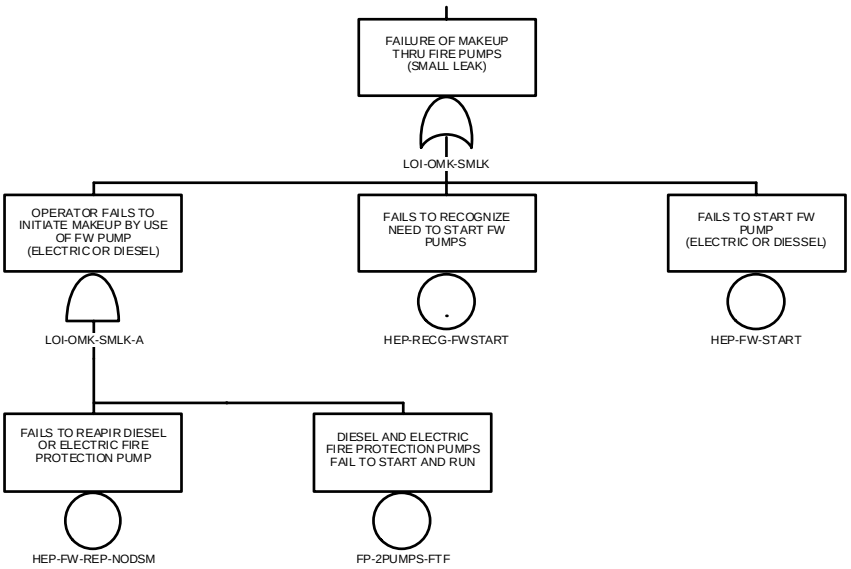

OPERATOR FAILS TO OBSERVE LOI/LSFPC EVENT DURING WALKDOWN
LOI-IND
ONLY DIAGNOSIS FAILURES
DIAGNOSIS AND HARDWARE FAILURES
LOI-IND-A
LOI-IND-B
OPERATOR FAILS TO OBSERVE LOI/LSFPC DURING WALKDOWN (DEPENDENT EVENT)
OPERATOR FAILS TO RESPOND GIVEN ALARM
LOCAL FAULTS CAUSING LOSS OF INDICATION
OPERATOR FAILS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

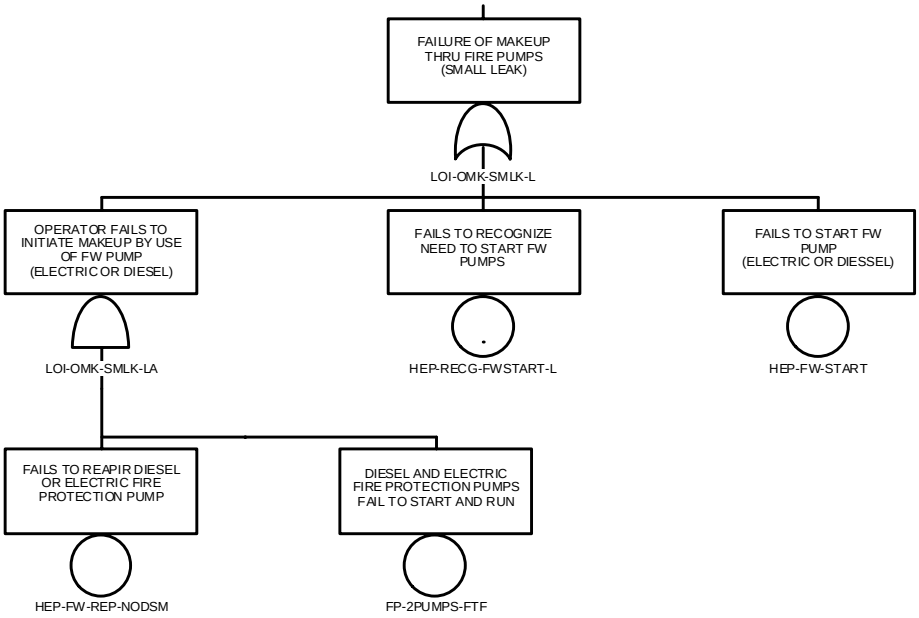

OPERATOR FAILS TO OBSERVE LOI/LSFPC EVENT DURING WALKDOWN
LOI-IND-L
OPERATOR FAILS TO OBSERVE LOI/LSFPC DURING WALKDOWN (NO DEPENDENCY)
HEP-WLKDWN-LOI-L

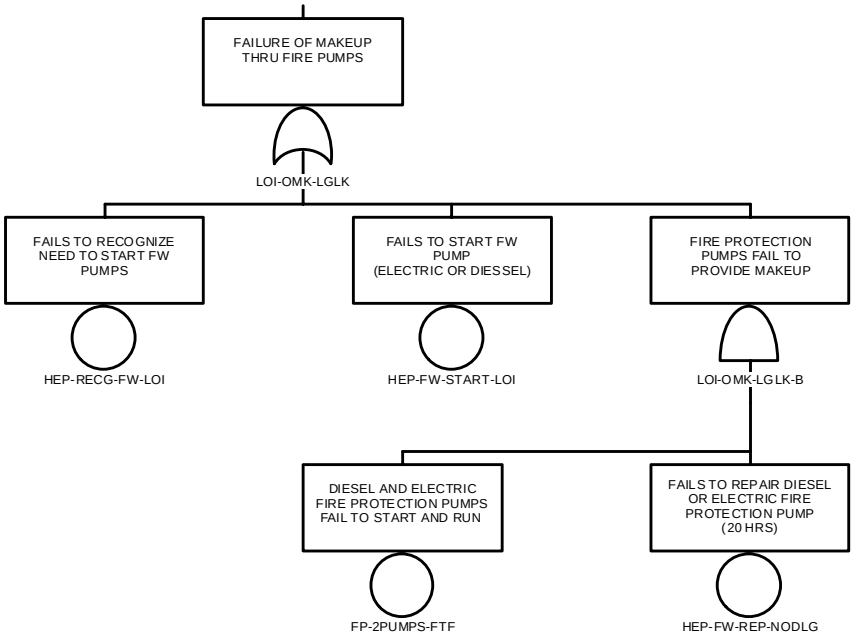

FAILURE TO ISOLATE LOSS AND INITIATE NORMAL MAKEUP (EARLY RESPOND)
LOI-OIS-U
OPERATOR FAILS TO ISOLATE LEAK
REGULAR SFP MAKEUP SYSTEM FAILS
OPERATOR FAILS TO START MAKUP (EARLY RESPOND)
HEP-LEAK-ISO
SFP-REGMKUP-F
HEP-MKUP-START-E

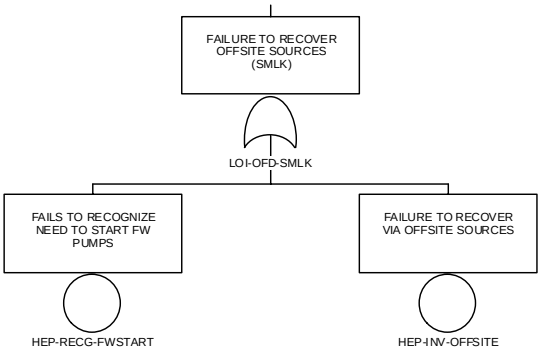

FAILURE TO INITIATE NORMAL MAKEUP (SMALL LEAK)
LOI-OIL-U
REGULAR SFP MAKEUP SYSTEM FAILS
OPERATOR FAILS TO START MAKUP (SMALL LEAK)
SFP-REGMKUP-F
HEP-MKUP-START

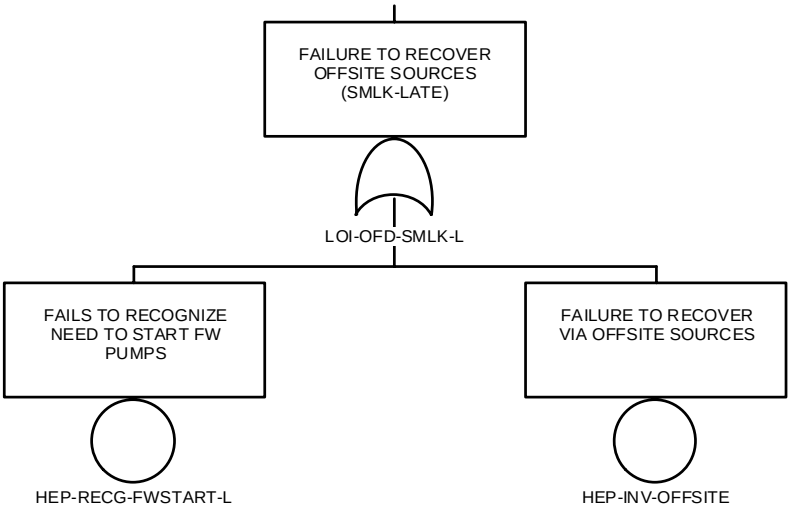

FAILURE TO INITIATE NORMAL MAKEUP (SMALL LEAK)
LOI-OIL-L
REGULAR SFP MAKEUP SYSTEM FAILS
OPERATOR FAILS TO START MAKUP (SMALL LEAK)
SFP-REGMKUP-F
HEP-MKUP-START-E

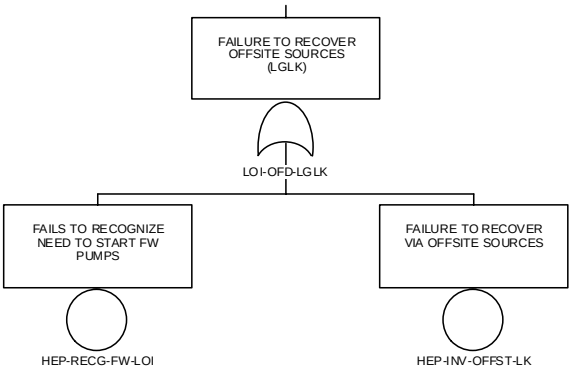

FAILURE OF MAKEUP THRU FIRE PUMPS (SMALL LEAK)
LOI-OMK-SMLK
OPERATOR FAILS TO INITIATE MAKEUP BY USE OF FW PUMP (ELECTRIC OR DIESEL)
FAILS TO RECOGNIZE NEED TO START FW PUMPS
FAILS TO START FW PUMP (ELECTRIC OR DIESSEL)
LOI-OMK-SMLK-A
HEP-RECG-FWSTART
HEP-FW-START
FAILS TO REAPIR DIESEL OR ELECTRIC FIRE 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


FAILURE TO RECOVER OFFSITE SOURCES (SMLK-LATE)
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 = 1E-2

(2) Otherwise,	**Time**	**Stress**	**Complexity**	**Experience/ Training**	**Procedures**	**Ergonomics**	**Fitness for Duty**	**Work Processes**
Diagnosis: 1E-2x___	**x___**	**x___**	**x___**	**x___**	**x___**	**x___**	**x___**	**=___**

Diagnosis

Failure Probability

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 = 1E-3

(2) Otherwise,	**Time**	**Stress**	**Complexity**	**Experience/ Training**	**Procedures**	**Ergonomics**	**Fitness for Duty**	**Work Processes**	
Action: 1E-3	**x___**	**x___**	**x___**	**x___**	**x___**	**x___**	**x___**	**x___**	**=___**

Action

Failure Probability

SPAR HRA Human Error Worksheet (Page 3 of 3)

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).

		If all PSFs are nominal, then
Diagnosis Failure Probability: 1E-2	_____	Diagnosis Failure Probability:
Action Failure Probability: +1E-3	+_____	Action Failure Probability:
Task Failure Without Formal Dependence ($P_{w/od}$)=_____	$P_{(w/od)}$	= 1.1E-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 2B
STRUCTURAL INTEGRITY OF SPENT FUEL POOLS SUBJECT TO SEISMIC LOADS

1. INTRODUCTION

The staff's concern regarding seismic issues at spent fuel pools (SFPs) involves very large earthquake ground motions that could catastrophically fail the SFP. Under this scenario, the pool would suffer a significant breach, it would drain rapidly, and it will be incapable of being refilled. This would lead to gradual cladding heat up, possibly followed by a zirconium cladding fire. Attachment 1 to this appendix provides the checklist proposed by NEI and enhanced by the staff to identify potential weaknesses and to assure adequate seismic capacity (at least 1.2g peak spectral acceleration) at SFPs for decommissioning sites that wish to be granted exemptions to EP, safeguards, and indemnification. Attachment 2 to this appendix provides the analysis of the seismic failure probability of SFPs by the NRC's consultant, Robert Kennedy, for nuclear power plant sites based on a generic 1.2 g peak spectral acceleration high confidence, with low probability of failure (HCLPF) value for SFPs. Attachment 3 to this appendix provides a discussion of expected failure modes of SFPs because of very large ground motions and describes the expected levels of collateral damage to the area surrounding the decommissioning reactor site. The staff evaluated the frequency of large earthquake ground motions at operating reactor sites, and these results are presented in Table 3 of Attachment 2 to this appendix.

SFP structures at nuclear power plants are considered to be seismically robust. They are constructed with thick reinforced concrete walls and slabs lined with stainless steel liners 1/8 to 1/4 inch thick.[1] Pool walls are about 5 feet thick, 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 SFP structures are located outside the containment structure supported on the ground or partially embedded in the ground. The location and supporting arrangement of the pool structures influence 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 seismic demand needs. Spent fuel structures at operating nuclear power plants are able to withstand loads substantially beyond those for which they were designed.

The Commission asked the staff to determine if there were a risk-informed basis for providing exemptions from EP, safeguards, or indemnification for decommissioning plants and to provide a technical basis for potential rule making. After this, the staff began to investigate the capacity of SFPs to withstand large earthquake ground motions beyond the site's seismic design bases.

To evaluate the risk from a seismic event at an SFP, one needs to know both the likelihood of seismic ground motion at various g-levels (i.e., seismic hazard) and the conditional probability

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[1]Except for Dresden Unit 1 and Indian Point Unit 1, whose SFPs do not have any liner plates. They were permanently shutdown more than 20 years ago, and no safety significant degradation of the concrete pool structure has been reported.

that a structure, system, or component (SSC) will fail at a given acceleration level (i.e., the fragility of the SSC). These are convolved mathematically to arrive at the likelihood that the SFP will fail from a seismic event. In evaluating the effect of seismic events on SFPs, it became apparent that although information was available on the seismic hazard for nuclear power plant sites, the staff did not have fragility analyses of the pools, nor generally did licensees. The staff recognized that many of the SFPs and the buildings housing them were designed by different architect engineers. The SFPs and structures housing them were built to codes that evolved through various code editions.

The staff originally performed a simplified bounding seismic risk analysis in its June 1999 draft assessment of decommissioning plant risks to help determine if there might be a seismic concern. The analysis indicated that seismic events could not be dismissed on the basis of a simplified bounding approach. In addition after further evaluation and discussions with stakeholders, it was determined that it would not be cost effective to perform a detailed plant-specific seismic evaluation for each SFP. Working with its stakeholders, the staff developed other tools that help assure the pools are sufficiently robust.

2. RETURN PERIOD OF SFP-FAILING EARTHQUAKE GROUND MOTIONS

The staff reexamined its methods for estimating seismic risk and reexamined the results of Table 3 in Attachment 2 to this appendix, which estimates the return frequencies of large earthquake ground motions that could fail SFPs. It was decided that the HCLPF value of 1.2 g peak spectral acceleration was a good generic screening measure for seismic capacity for decommissioning plant SFPs. The staff used a simplified, but slightly conservative method (see Attachment 2) to estimate the annual probability of a zirconium fire because of seismic events (including use of site-specific seismic hazard estimates). These calculations resulted in a range of site-specific frequencies from less than $1x10^{-8}$ per year to over $1x10^{-5}$ per year, depending on the site and the seismic estimates used.

Both the EPRI and LLNL hazard estimates are judged by the NRC to be equally valid and useful for making decisions. At most sites, the LLNL hazard estimates predict a higher frequency than EPRI estimates for ground motions exceeding a given acceleration. From a regulatory stand point, it is prudent to examine the implications of the more limiting parameter inputs (e.g., shorter return period of large earthquakes) when making safety decisions. Using the LLNL hazard estimates, all central and eastern sites except H. B. Robinson have an expected seismic-induced zirconium cladding fire frequency less than the Pool Performance Guideline ($1x10^{-5}$ per year) at a 1.2g peak spectral acceleration (PSA) ground motion. Using EPRI hazard estimates, only one site east of the Rocky Mountains would have an expected frequency of a seismic-induced cladding fire in excess of $1x10^{-6}$ per year(but less than $1x10^{-5}$ per year) at the 1.2g PSA level. Neither LLNL nor EPRI developed hazard estimates for sites west of the Rocky Mountains (such as San Onofre, Diablo Canyon, and WNP2), and EPRI, unlike LLNL, did not develop hazard estimates for all operating reactor sites east of the Rockies. H.B. Robinson and Western sites with higher frequencies at the 1.2g PSA level would need to perform site-specific seismic risk analyses for their SFPs if the utilities desire to take advantage of potential exemptions or rule changes regarding EP, security, or indemnification. The plant-specific analysis would be required for the Western sites and Robinson for the following reasons: (1) the checklist helps assure an SFP has adequate capacity at the 1.2g PSA ground motion level, (2) at these sites the frequency of ground motions that exceed 1.2g PSA may be so high as to call into question whether the risk at a decommissioning plant would

exceed the NRC's Safety Goals, and (3) plant-specific seismic analyses may demonstrate that the sites actually have adequate margin to meet the staff's PPG and Safety Goals.

The staff determined the mean of all the LLNL hazard estimates for operating reactor sites east of the Rocky Mountains (i.e., $2x10^{-6}$ per year). This value is a factor of five less than PPG and the mean bounds about 70 percent of the sites east of the Rockies. Similarly, the mean of all EPRI hazard estimates is $2x10^{-7}$ per year. From a risk standpoint, the majority of potential decommissioning sites have expected frequencies of fuel uncovery far below the staff's pool performance guideline. Risk results are reported in Section 3.7 of the main study.

3. SEISMIC CHECKLIST

The staff determined that, absent specific information about SFP seismic capacities, some plant-specific evaluation of SFP capacity was warranted. During stakeholder interactions with the staff, the staff proposed the use of a seismic checklist that built on the work done to quantify seismic margins and that could provide assurance of the capacity of SFPs. In a letter dated August 18, 1999, NEI proposed a checklist that could be used to show robustness for a seismic ground motion with a peak spectral acceleration (PSA) of 1.2 g (or with peak ground acceleration (PGA), which is not as good an estimator, of approximately 0.5 g). This checklist was reviewed and enhanced by the staff (see Attachment 1). Dr. Robert Kennedy, a staff consultant, reviewed the enhanced checklist and concluded that the screening criteria are adequate for the vast majority of central and eastern U.S. sites. The seismic checklist was developed to provide a simplified method for demonstrating a HCLPF at an acceptably low value of seismic risk. The checklist includes elements to assure there are no weaknesses in the design or construction nor any service induced degradation of the pools that would make them vulnerable to failure under earthquake ground motions that exceed their design basis ground motion but are less than the HCLPF value. SFPs that satisfy the seismic checklist, as written, would have a high confidence in a low probability of failure for seismic ground motions up to 1.2 g peak spectral acceleration.

4. SEISMIC RISK - SUPPORT SYSTEM FAILURE

In its preliminary draft study published in June 1999, the staff assumed that a ground motion three times the SSE was the HCLPF of the SFP. The HCLPF capacity is such that 95 percent of the time the pool would remain intact (i.e., would not leak significantly given ground motion up to a certain value, i.e., 1.2g PSA). The staff evaluated what would happen to SFP support systems (i.e., the pool cooling and inventory make-up systems) in the event of an earthquake three times the SSE. The staff 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 study for the contribution from this scenario was $1x10^{-6}$ per year. In this study, 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 U.S. the return period of an earthquake ground motion that would damage a decommissioning plant's SFP 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 $4x10^{-4}$ that represents the failure of the fuel handlers to obtain offsite resources. The event was quantified using the SPAR HRA technique. The performance 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 because of 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 because of seismic events to be on the order of $4x10^{-8}$ per year. The frequency of fuel uncovery from support system failure because of seismic events is bounded by catastrophic failure of the SFP because of a seismic event.

5. HAZARD ESTIMATE AND FRAGILITY UNCERTAINTIES

The staff recognizes there are considerable uncertainties in both the seismic hazard estimates for nuclear power plant sites and for the fragility estimates of SFPs. The staff's evaluation used both LLNL and EPRI hazard estimates (frequency of the ground motion occurring, at a certain level) since the NRC has stated that both the EPRI and LLNL hazard estimates are reasonable and valid. For eastern U.S. sites, the hazard estimates (particularly LLNL) are relatively flat as the return period and peak spectral acceleration increase. At the return frequency (i.e., frequency of an earthquake at or exceeding a specified ground motion level) of safe shutdown earthquakes (SSEs), the LLNL and EPRI estimates are in reasonable agreement. However, as ground motion levels increase, there is little or no conclusive data, and the ground motion experts diverge on how to assign return periods to extreme seismic events. The tails of the hazard curve distributions drive the results (i.e. the mean) as would be expected of a distribution that is particularly flat (e.g., one that has large modeling uncertainties).

6. CONCLUSION

The staff recommends that those plants that are west of the Rocky Mountains and have short return periods for the occurrence of ground motions greater than 1.2 g PSA in their SFP should be required to conduct plant-specific seismic analysis beyond the confirmation of the checklist if they desire to obtain exemptions (or take advantage of rule making) from EP, indemnification, or security at decommissioning sites. In addition, because the LLNL hazard estimates indicate that the H. B. Robinson site exceeds the PPG ($1x10^{-5}$ per year) at 1.2g PSA, the utility should perform a plant-specific seismic analysis too.

To summarize the staff recommendations to provide reasonable confidence that there are no seismic vulnerabilities at decommissioning plant SFPs, (1) all sites must conduct an assessment of the SFP structures using the revised seismic check list in order to identify any structural degradation, potential for seismic interaction from superstructures and overhead cranes, and to verify that they have a seismic HCLPF value of 1.2 g PSA or higher, (2) those sites that do not pass the seismic check list may either undertake appropriate remedial action or conduct a site-specific seismic risk assessment of their decommissioning risk, and (3) sites such as H. B. Robinson, WNP2, Diablo Canyon, and San Onofre should have to pass the seismic check list to identify any structural degradation or other anomalies and then conduct a site-specific seismic risk assessment that has a fuel uncovery frequency less than the PPG if they desire an exemption from EP when their sites are in decommissioning.

Attachment 1

Seismic Check list for Commercial Nuclear Power Plants
During Decommissioning

Enhanced Seismic Checklist

Item 1:

Requirement: Identify Preexisting Concrete and Liner Plate Degradation

Basis: A detailed review of plant records concerning spent fuel pool concrete and liner plate degradation should be performed and supplemented by a detailed walkdown of the accessible portions of the spent fuel pool concrete and liner plate. The purpose of the records review and visual inspection activities is to accurately assess the material condition of the SFP concrete and liner in order to assure that these existing material conditions are properly factored into the remaining seismic screening assessments.

Design Feature: The material condition of the SFP concrete and liner, based upon the records review and the walkdown inspection, will be documented and used as an engineering input to the following seismic screening assessments.

Item 2:

Requirement: Assure Adequate Ductility of Shear Wall Structures

Basis: The expert panel involved with the development of Reference 1 concluded that, " For the Category 1 structures which comply with the requirements of either ACI 318-71 or ACI 349-76 or later building codes and are designed for an SSE of at least 0.1g pga, as long as they do not have any special problems as discussed below, the HCLPF capacity is at least 0.5g pga." This conclusion was based upon the assumption that the shear wall structure will respond in a ductile manner. The "special problems" cited deal with individual plant details which could prevent a particular plant from responding in the required ductile fashion. Examples cited in Reference 1 included an embedded structural steel frame in a common shear wall at the Zion plant (which was assumed to fail in brittle manner due to a potential shear failure of the attached shear studs) and large openings in a "crib house" roof (also at the Zion plant) which could interrupt the continuity of the structural slab.

Other examples which could impact the ductility of the spent fuel pool structure include large openings which are not adequately reinforced or reinforcing bars that are not sufficiently embedded to prevent a bond failure before the yield capacity of the steel is reached.

Design Feature: This design feature requirement will be documented based on a review of drawings and a SFP walkdown.

Item 3:

Requirement: Assure Design adequacy of Diaphragms (including roofs)

Basis: In the design of many nuclear power plants, the seismic design of roof and floor diaphragms has often not received the same level of attention as have the shear walls of the structures. Major cutouts for hatches or for pipe and electrical chases may pose special problems for diaphragms. Since more equipment tends to be anchored to the diaphragm compared to shear walls, moderate amounts of damage may be more critical for the diaphragm compared to the same amount of damage in a wall.

Based upon the guidance provided in Reference 1, diaphragms for Category I structures designed for a SSE of 0.1g or greater do not require an explicit evaluation provided that: (1) the diaphragm loads were developed using dynamic analysis methods; (2) they comply with the ductility detailing requirements of ACI 318-71 or ACI 349-76 or later editions. Diaphragms which do not comply with the above ductility detailing or which did not have loads explicitly calculated using dynamic analysis should be evaluated for a beyond-design-basis seismic event in the 0.45-0.5g pga range.

Design Feature: This design feature requirement will be documented based on a review of drawings and a SFP walkdown.

Item 4:

Requirement: Verify the Adequacy of the SFP Walls and Floor Slab to Resist Out-of-Plane Shear and Flexural Loads

Basis: For PWR pools that are fully or partially embedded, an earthquake motion that could cause a catastrophic out-of-plane shear or flexural failure is very high and is not a credible event. For BWR pools (and PWR pools that are not at least partially embedded), the seismic capacity is likely to be somewhat less and the potential for out-of-plane shear and/or flexural wall or base slab failure, at beyond-design-basis seismic loadings, is possible.

A structural assessment of the pool walls and floor slab out-of plane shear and flexural capabilities should be performed and compared to the realistic loads expected to be generated by a seismic event equal to approximately three times the site SSE. This assessment should include dead loads resulting from the masses of the pool water and racks, seismic inertial forces, sloshing effects and any significant impact forces.

Credit for out-of-plane shear or flexural ductility should not be taken unless the reinforcement associated with each failure mode can be shown to meet the ACI 318-71 or ACI 349-49 requirements.

Design Feature: Compliance with this design feature will be documented basedupon a review of drawings (in the case of embedded or partially embedded PWR pools) or based upon a review of drawings coupled with the specified beyond-design-basis shear and flexural calculations outlined above.

Item 5:

Requirement: Verify the Adequacy of Structural Steel (and Concrete) Frame Construction

Basis: At a number of older nuclear power plants, the walls and roof above the top of the spent fuel pool are constructed of structural steel. These steel frames were generally designed to resist hurricane and tornado wind loads which exceeded the anticipated design basis seismic loads. A review of these steel (or possibly concrete) framed structures should be performed to assure that they can resist the seismic forces resulting from a beyond-design-basis seismic event in the 0.45-0.5g pga range. Such a review of steel structures should concentrate on structural detailing at connections. Similarly, concrete frame reviews should concentrate on the adequacy of the reinforcement detailing and embedment.

Failure of the structural steel superstructure should be evaluated for its potential impact on the ability of the spent fuel pool to continue to successfully maintain its water inventory for cooling and shielding of the spent fuel.

Design Feature: This design feature requirement will be documented based on a review of drawings and a SFP walkdown.

Item 6:

Requirement: Verify the Adequacy of Spent Fuel Pool Penetrations

Basis: The seismic and structural adequacy of any spent fuel pool (SFP) penetrations whose failure could result in the draining or syphoning of the SFP must be evaluated for the forces and displacements resulting from a beyond-design-basis seismic event in the 0.45-0.5g pga range. Specific examples include SFP gates and gate seals and low elevation SFP penetrations, such as, the fuel transfer chute/tube and possibly piping associated with the SFP cooling system. Failures of any penetrations which could lead to draining or syphoning of the SFP should be considered.

Design Feature: This design feature requirement will be documented based on a review of drawings and a SFP walkdown.

Item 7:

Requirement: Evaluate the Potential for Impacts with Adjacent Structures

Basis: Structure-to-structure impact may become important for earthquakes significantly above the SSE, particularly for soil sites. Structures are usually conservatively designed with rattle space sufficient to preclude impact at the SSE level but there are no set standards for margins above the SSE. In most cases, impact is not a serious problem but, given the potential for impact, the consequences should be addressed. For impacts at earthquake levels below 0.5g pga, the most probable damage includes the potential for electrical equipment malfunction and for local structural damage. As cited previously, these levels of damage may be found to be acceptable or to result in the loss of SFP support equipment. The major focus of this impact review is to assure that the structure-to-structure impact does not result in the inability of the SFP to maintain its water inventory.

Design Feature: This design feature requirement will be documented based on a review of drawings and a SFP walkdown.

Item 8:

Requirement: Evaluate the Potential for Dropped Loads

Basis: A beyond-design-basis seismic event in the 0.45-0.5g pga range has the potential to cause the structural collapse of masonry walls and/or equipment supports systems. If these secondary structural failures could result in the accidental dropping of heavy loads which are always present (i.e. not loads associated with cask movements) into the SFP, then the consequences of these drops must be considered. As in previous evaluations, the focus of the drop consequence analyses should consider the possibility of draining the SFP. Additionally,

the evaluation should evaluate the consequences of any resulting damage to the spent fuel or to the spent fuel storage racks.

Design Feature: This design feature requirement will be documented based on a review of drawings and a SFP walkdown.

Item 9:

Requirement: Evaluation of Other Failure Modes

Basis: Experienced seismic engineers should review the geotechnical and structural design details for the specific site and assure that there are not any design vulnerabilities which will not be adequately addressed by the review areas listed above. Soil-related failure modes including liquefaction and slope instability should be screened by the approaches outlined in Reference 1 (Section 7 & Appendix C).

Design Feature: This design feature requirement will be documented based on a review of drawings and a SFP walkdown.

Item 10: Potential Mitigation Measures

Although beyond the scope of this seismic screening checklist, the following potential mitigation measures may be considered in the event that the requirements of the seismic screening checklist are not met at a particular plant.

a.) Delay requesting the licensing waivers (E-Plan, insurance, etc.) until the plant specific danger of a zirconium fire is no longer a credible concern.

b.) Design and install structural plant modifications to correct/address the identified areas of non-compliance with the checklist. (It must be acknowledged that this option may not be practical for significant seismic failure concerns.)

c.) Perform plant-specific seismic hazard analyses to demonstrate that the seismic risk associated with a catastrophic failure of the pool is at an acceptable level. (The exact “acceptable” risk level has not been precisely quantified but is believed to be in the range of $1x10^{-5}$ per year.)

We believe that use of the checklist and determination that the spent fuel pool HCLPF is sufficiently high will assure that the frequency of fuel uncovery from seismic events is less than or equal to $1x10^{-6}$ per year.

Comments Concerning Seismic Screening And Seismic Risk of Spent Fuel Pools for Decommissioning Plants

by
Robert P. Kennedy
October 1999

prepared for

Brookhaven National Laboratory

2. Introduction

I have been requested by Brookhaven National Laboratory, in support of the Engineering Research Applications Branch of the Nuclear Regulatory Commission, to review and comment on certain seismic related aspects of References 1 through 4. Specifically, I was requested to comment on the applicability of using seismic walkdowns and drawing reviews conducted following the guidance provided by seismic screening tables (seismic check lists) to assess that the risk of seismic-induced spent fuel pool accidents is adequately low. The desire is to use these seismic walkdowns and drawing reviews in lieu of more rigorous and much more costly seismic fragility evaluations. It is my understanding that the primary concern is with a sufficiently gross failure of the spent fuel pool so that water is rapidly drained resulting in the fuel becoming uncovered. However, there may also be a concern that the spent fuel racks maintain an acceptable geometry. It is also my understanding that any seismic walkdown assessment should be capable of providing reasonable assurance that seismic risk of a gross failure of the spent fuel pool to contain water is less than the low 10^{-6} mean annual frequency range. My review comments are based upon these understandings.

2. Background Information

The NRC Draft Technical Study of Spent Fuel Pool Accidents (Ref. 1) assumes that spent fuel pools are seismically robust. Furthermore, it is assumed that High-Confidence-Low-Probability-of Failure (HCLPF) seismic capacity of these pools is in the range of 0.4 to 0.5g peak ground acceleration (PGA). This HCLPF capacity (C_{HCLPF}) corresponds to approximately a 1% mean conditional probability of failure capacity ($C_{1\%}$), i.e.:

$$C_{HCLPF} \approx C_{1\%} \quad (1)$$

as shown in Ref. 10.

In Ref. 5, detailed seismic fragility assessments have been conducted on the gross structural failure of spent fuel pools for two plants: Vermont Yankee (BWR), and Robinson (PWR). The following HCLPF seismic capacities are obtained from the fragility information in

Ref. 5:

Vermont Yankee (BWR): $C_{HCLPF} = 0.48g$ PGA

Robinson (PWR): $C_{HCLPF} = 0.65g$ PGA (2)

These two fragility estimates provide some verification of the HCLPF capacity assumption of 0.4 to 0.5g PGA used in Ref. 1.

I am confident that a set of seismic screening tables (seismic check lists) can be developed to be used with seismic walkdowns and drawing reviews to provide reasonable assurance that the HCLPF capacity of spent fuel pools is at least in the range of 0.4 to 0.5g PGA for spent fuel pools that pass such a review. However, in order to justify a HCLPF capacity in the range of 0.4 to 0.5g PGA, these screening tables will have rather stringent criteria so that I am not so confident that the vast majority of spent fuel pools will pass the screening criteria. The screening criteria (seismic check lists) summarized in Ref. 4 provides an excellent start. The subject of screening criteria is discussed more thoroughly in Section 3.

Once the HCLPF seismic capacity (C_{HCLPF}) has been estimated, the seismic risk of failure of the spent fuel pool can be estimated by either rigorous convolution of the seismic fragility (conditional probability of failure as a function of ground motion level) and the seismic hazard (annual frequency of exceedance of various ground motion levels), or by a simplified approximate method. This subject is discussed more thoroughly in Ref. 10.

A simplified approximate method is used in Ref. 1 to estimate the annual seismic risk of failure (P_F) of the spent fuel pool given its HCLPF capacity (C_{HCLPF}). The approach used in Ref. 1 is that:

$$P_F = 0.05\ H_{HCLPF} \tag{3}$$

where H_{HCLPF} is the annual frequency of exceedance of the HCLPF capacity. Ref. 1 goes on to state that for most Central and Eastern U.S. (CEUS) plants, the mean annual frequency of exceeding 0.4 to 0.5g PGA is on the order of or less than $2x10^{-5}$ based on the Ref. 8 hazard curves. Thus, from Eqn. (3), the annual frequency of seismic-induced gross failure (P_F) of the spent fuel pool is on the order of $1x10^{-6}$ or less for most CEUS plants.

Unfortunately, the approximation of Eqn. (3) is unconservative for CEUS hazard curves that have shallow slopes. By shallow slopes, I mean that it requires more than a factor of 2 increase in ground motion to correspond to a 10-fold reduction in the annual frequency of exceedance. For most CEUS sites, Ref. 8 indicates that a factor of 2 to 3 increase in ground motion is required to reduce the hazard exceedance frequency from $1x10^{-5}$ to $1x10^{-6}$. Over this range of hazard curve slopes, Eqn. (3) is always unconservative and will be unconservative by a factor of 2 to 4. Therefore, a HCLPF capacity in the range of 0.4 to 0.5g PGA is not sufficiently high to achieve a spent fuel pool seismic risk of failure on the order of $1x10^{-6}$ or less for most CEUS plants. However, HCLPF capacities this high are sufficiently high to achieve seismic risk estimates less than $3x10^{-6}$ for most CEUS plants based upon the Ref. 8 hazard curves. This subject is further discussed in Section 4.

In lieu of using a simplified approximate method, Ref. 2 has estimated the seismic risk of spent fuel pool failure by rigorous convolution of the seismic fragility and seismic hazard estimates for the 69 CEUS sites for which seismic hazard curves are given in Ref. 8. Ref. 2 has divided the sites into 26 BWR sites and 43 PWR sites.

For the 26 BWR sites, Ref. 2 used the fragility curve defined in Ref. 5 for Vermont Yankee with the following properties:

BWR Sites		
Median Capacity	$C_{50} = 1.4$	PGA
HCLPF Capacity	$C_{HCLPF} = 0.48g$	PGA

(4)

Using the Ref. 8 seismic hazard estimates and the Eqn. (4) fragility, Ref. 2 obtained spent fuel pool mean annual failure probabilities ranging from 12.0×10^{-6} to 0.11×10^{-6} and averaging 1.6×10^{-6} for the 26 BWR sites. In my judgment, seismic screening criteria (seismic check lists) can be developed which are sufficiently stringent so as to provide reasonable assurance that the seismic capacity of spent fuel pools which pass the seismic screening roughly equals or exceeds that defined by Eqn. (4). With such a fragility estimate, based on the Ref. 8 seismic hazard estimates, for most CEUS sites, the estimated spent fuel pool seismic-induced failure probability will be less than 3×10^{-6} as further discussed in Section 4.

For the 43 PWR sites, Ref. 2 used the fragility curve defined in Ref. 5 for Robinson with the following properties:

PWR Sites		
Median Capacity	$C_{50} = 2.0$	PGA
HCLPF Capacity	$C_{HCLPF} = 0.65g$	PGA

(5)

Using the Ref. 8 seismic hazard estimates and the Eqn. (5) fragility, Ref. 2 obtained spent fuel pool mean annual failure probabilities ranging from 2.5×10^{-6} to 0.03×10^{-6} and averaging 0.48×10^{-6} for the 43 PWR sites. A fragility curve as high as that defined by Eqn. (5) is necessary to achieve an estimated spent fuel pool seismic-induced failure probability as low as 1×10^{-6} for nearly all CEUS sites. However, I don't believe realistic seismic screening criteria can be developed which are sufficiently stringent to provide reasonable assurance that the Eqn. (5) seismic fragility is achieved. In my judgment, a more rigorous seismic margin evaluation performed in accordance with the CDFM method described in Refs. 6 or 7 would be required to justify a HCLPF capacity as high as that defined by Eqn. (5).

3. Development and Use of Seismic Screening Criteria

Screening criteria are very useful to reduce the number of structure, system, and component (SSC) failure modes for which either seismic fragilities or seismic margin HCLPF capacities need to be developed. Screening criteria are presented in Ref. 6 for SSCs for which failures might lead to core damage. These screening criteria were established by an NRC sponsored "Expert Panel" based upon their review of seismic fragilities and seismic margin HCLPF capacities computed for these SSCs at more than a dozen nuclear power plants, and their review of earthquake experience data. These screening criteria were further refined in Ref. 7.

The screening criteria of Refs. 6 and 7 are defined for two seismic margin HCLPF capacity levels which will be herein called Level 1 and Level 2. Refs. 6 defines these two HCLPF capacity levels in terms of the PGA of the ground motion. However, damage to critical SSCs does not correlate very well to PGA of the ground motion. Damage correlates much better with the spectral acceleration of the ground motion over the natural frequency range of interest which is generally between 2.5 and 10 Hz for nuclear power plant SSCs. For this reason, Ref. 7 defines these same two HCLPF capacity levels in terms of the peak 5% damped spectral acceleration (PSA) of the ground motion. The two HCLPF capacity screening levels defined in Refs 6 and 7 are:

	HCLPF Screening Levels	
	Level 1	Level 2
PGA (Ref. 6)	0.3g	0.5g
PSA (Ref. 7)	0.8g	1.2g

These two definitions (PGA and PSA) are consistent with each other based upon the data upon which these screening levels are based. However, in my judgment, it is far superior to use the Ref. 7 PSA definition for the two screening levels when convolving a fragility estimate with CEUS seismic hazard estimates. For these CEUS seismic hazard estimates from Ref. 8, the ratio PSA/PGA generally lies in the range of 1.8 to 2.4 which is lower than the PSA/PGA ratio of the data from which the screening tables were developed. A more realistic and generally lower estimate of the annual probability of failure will result when the seismic fragility is defined in terms of PSA and convolved with a PSA hazard estimate in which the PSA hazard estimate is defined in the 2.5 to 10 Hz range.

In the past, a practical difficulty existed with defining the seismic fragility in terms of PSA instead of PGA. The Ref. 8 PSA hazard estimates are only carried down to 10^{-4} annual frequency of exceedance whereas the PGA hazard estimates are extended down to about 10^{-6}. Since it is necessary for the hazard estimate to be extended to at least a factor of 10 below the annual failure frequency being predicted, it has not been practical to use the PSA seismic fragility definition with the Ref. 8 hazard estimates. However, this difficulty has been overcome by Ref. 9 prepared by the Engineering Research Applications Branch of the Nuclear Regulatory

Commission which extends the PSA seismic hazard estimates also down to 10^{-6}. Ref. 9 is attached herein as Appendix A.

In order to achieve a seismic induced annual failure probability P_F in the low 10^{-6} range for nearly all of the CEUS spent fuel pools with the Ref. 8 hazard estimates, it is necessary to apply the Level 2 screening criteria of Refs. 6 or 7, i.e., screen at a HCLPF seismic capacity of 1.2g PSA (equivalent to 0.5g PGA). The seismic screening criteria presented in Ref. 4 is properly based upon screening to Level 2. Furthermore, Ref. 4 appropriately summarizes the guidance presented in Ref. 7 for screening to Level 2. In general, I support the screening criteria defined in Ref. 4. However, I do have three concerns which are discussed in the following subsections.

3.1 Out-of-Plane Flexural and Shear Failure Modes for Spent Fuel Pool Concrete Walls and Floor

The screening criteria for concrete walls and floor diaphrams were developed to provide seismic margin HCLPF capacities based upon in-plane flexural and shear failures of these walls and diaphrams. For typical auxiliary buildings, reactor buildings, diesel generator buildings, etc., it is these in-plane failure modes which are of concern. For normal building situations, seismic loads are applied predominately in the plane of the wall or floor diaphram. Out-of-plane flexure and shear are not of significant concern. As one the primary authors of the screening criteria in both Refs. 6 and 7, I am certain that these screening criteria do not address out-of-plane flexure and shear failure modes.

For an aboveground spent fuel pool in which the pool walls (and floor in some cases) are not supported by soil backfill, it is likely that either out-of-plane flexure or shear will be the expected seismic failure mode. These walls and floor slab must carry the seismic-induced hydrodynamic pressure from the water in the pool to their supports by out-of-plane flexure and shear. It is true that these walls and floor are robust (high strength), but they may not be as ductile for out-of-plane behavior as they are for in-plane behavior. For an out-of-plane shear failure to be ductile requires shear reinforcement in regions of high shear. Furthermore, if large plastic rotations are required to occur, the tensile and compression steel needs to be tied together by closely spaced stirrups. I question whether such shear reinforcement and stirrups exist at locations of high shear and flexure in the spent fuel pool walls and floor. As a result, I suspect that only limited credit for ductility can be taken.

Without taking credit for significant ductility, it is not clear to me that spent fuel pool walls and floors not supported by soil can be screened at a seismic HCLPF capacity level as high as 1.2g PSA (equivalent to 0.5g PGA). I am aware of only one seismic fragility analysis having been performed on such unsupported spent fuel pool walls. That analysis was the Vermont Yankee spent fuel pool analysis reported in Ref. 5 for which the reported seismic HCLPF capacity was 0.48g PGA. A single analysis case does not provide an adequate basis for establishing a screening level for all other cases, particularly when the computed result is right at the desired screening level. The screening criteria in Refs 6 and 7 are based upon the review of many cases at more that a dozen plants.

In my judgement, it will be necessary to have either seismic fragility or seismic margin HCLPF computations performed on at least six different aboveground spent fuel pools with walls not supported by soil before out-of-plane flexure and shear HCLPF capacity screening levels can be established for such spent fuel pools.

3.2 Spent Fuel Pool Racks

I don't know whether a gross structural failure of the spent fuel racks is of major concern. This is a topic outside of my area of expertise. However, if such a failure is of concern, no seismic HCLPF capacity screening criteria is available for such a failure. The screening criteria of Refs. 6 and 7 were never intended to be applied to spent fuel pool racks. Since I have never seen a seismic fragility or seismic margin HCLPF capacity evaluation of a spent fuel pool rack, I have no basis for deciding whether these racks can be screened at a seismic HCLPF capacity as high as 1.2g PSA (equivalent to 0.5g PGA).

3.3 Seismic Level 2 Screening Requirements

In order to screen at a seismic HCLPF capacity of 1.2g PSA (0.5g PGA), the Level 2 screening criteria for concrete walls and diaphrams requires that such walls and diaphrams essentially comply with the ductile detailing and rebar development length requirements of either ACI 318.71 or ACI 349.76 or later editions. It is not clear to me how many CEUS spent fuel pool walls and floors essentially comply with such requirements since earlier editions of these codes had less stringent requirements. Therefore, it is not clear to me how many spent fuel pool walls and floors can actually be screened at Seismic Level 2 even for in-plane flexure and shear failure mode.

4. Seismic Risk Associated With Screening Level 2

4.1 Simplified Approaches for Estimating Seismic Risk Given the HCLPF Capacity

As mentioned in Section 2, the seismic risk of failure of the spent fuel pool can be estimated by either rigorous convolution of the seismic fragility and the seismic hazard, or by a simplified approximate method. The simplified approximate method defined by Eqn. (3) was used in Ref. 1. However, as also mentioned in Section 2, this approximate method understates the seismic risk by a factor of 2 to 4 for typical CEUS hazard estimates.

Ref. 10 presents an equally simple approach for estimating the seismic risk of failure of any component given its HCLPF capacity C_{HCLPF} and a hazard estimate. This approach tends to introduce from 0% to 25% conservative bias to the computed seismic risk when compared with rigorous convolution. Given the HCLPF capacity C_{HCLPF} this approach consists of the following steps:

Step 1: Estimate the 10% conditional probability of failure capacity $C_{10\%}$ from:

$$C_{10\%} = F_\beta C_{HCLPF} \qquad (6)$$
$$F_\beta = e^{1.044\beta}$$

where β is the logarithmic standard deviation of the fragility estimate and 1.044 is the difference between the 10% non-exceedance probability (NEP) standard normal variable (-1.282) and the 1% NEP standardized normal variable (-2.326). F_β is tabulated below for various fragility logarithmic standard deviation β values.

β	Median/CDFM Capacity ($C_{50\%}/C_{CDFM}$)	$F_\beta=(C_{10\%/CHCLPF})$
0.3	2.01	1.37
0.4	2.54	1.52
0.5	3.20	1.69
0.6	4.04	1.87

For structures such as the spent fuel pool, β typically ranges from 0.3 to 0.5. Ref. 10 shows that over this range of β, the computed seismic risk is not very sensitive to β. Therefore, I recommend using a midpoint value for β of 0.4.

<u>Step 2:</u> Determine hazard exceedance frequency $H_{10\%}$, that corresponds to $C_{10\%}$ from the hazard curve.

<u>Step 3:</u> Determine seismic risk P_F from:

$$P_F = 0.5\ H_{10\%} \tag{7}$$

Table 1 presents the Peak Spectral Acceleration PSA seismic hazard estimates from Ref. 8 and 9 (LLNL93 results) for the Vermont Yankee and Robinson sites. In order to accurately estimate the seismic risk for a seismic HCLPF capacity C_{HCLPF} of:

$$C_{HCLPF} = 1.2g\ PSA = 1176\ cm/sec^2\ PSA \tag{8}$$

associated with Screening Level 2 for the Vermont Yankee site by rigorous convolution, it is necessary to extrapolate the Ref. 9 hazard estimates down to the $2x10^{-8}$ exceedance frequency. Also, intermediate values in Table 1 have been obtained by interpolation.

Table 2 compares the seismic risk of spent fuel pool failure for these two sites as estimated by the following three methods:

1. Ref. 1 simplified approach, i.e., Eqn. (3).

2. Ref. 10 simplified approach, i.e., Steps 1 through 3 above.

3. Rigorous convolution of the hazard and fragility estimates.

For all three approaches the Screening Level 2 HCLPF capacity defined by Eqn. (8) was used. In addition, for both the Ref. 10 and rigorous convolution approaches, a fragility logarithmic standard deviation β of 0.4 was used.

From Table 2, it can be seen that the Ref. 1 method (Eqn. (3)) underestimates the seismic risk by factors of 2.3 and 3.5 for Vermont Yankee and Robinson, respectively. The simplified approach recommended in Ref. 10 and described herein overestimates the seismic risk by 20% and 5% respectively for these two cases. These results are consistent with the results I have obtained for many other cases.

4.2 Estimated Seismic Risk of Spent Fuel Pools Screened at Screening Level 2 Using Mean LL93 Hazard Estimates from Ref. 8 and 9

Using the Ref. 10 simplified approach described in the previous subsection, I have estimated the spent fuel pool seismic risk of failure corresponding to Screening Level 2 for all 69 CEUS sites with LLNL93 seismic hazard estimates defined in Refs. 8 and 9. These sites are defined in terms of an NRC site number code (OCSP_) used in Ref. 9. For each site, I assumed that the HCLPF capacity C_{HCLPF} was defined by Eqn. (8). A total of 35 of the 69 sites had estimated seismic risks of spent fuel pool failure associated with Screening Level 2 of greater than 1x10-6. The estimated seismic risk of 26 of these sites exceeded 1.25×10^{-6}. These 26 sites with their estimated seismic risk corresponding to Screening Level 2 are listed in Table 3. As can be seen in Table 3, only 8 of the 69 sites had estimated seismic risks of spent fuel pool failure exceeding 3×10^{-6}. One of these sites is Shoreham at which no fuel exists.

It should be noted that the seismic risks of spent fuel pool failure tabulated in Table 3 are based on the assumption that the HCLPF capacity of the spent fuel pool exactly equals the Screening Level 2 HCLPF capacity of 1.2g PSA (equivalent to 0.5g PGA). In actuality, spent fuel pools which pass the appropriately defined screening criteria are likely to have capacities higher than the screening level capacity. Therefore these are upper bound seismic risk estimates for spent fuel pools that pass the to-be established screening criteria. Furthermore, the simplified approach used to estimate the seismic risks in Table 3 overestimates these risks by 0% to 25%.

4.3 Estimated Seismic Risk of Spent Fuel Pools Screened at Screening Level 2 Using Mean EPRI89 Hazard Estimates

Following the exact same Ref. 10 simplified approach which I followed for the LLNL93 hazard estimates, Ref. 11 provides the corresponding seismic risk of spent fuel pool failure estimates based upon EPRI89 hazard estimates for 60 of the 69 CEUS sites. Table 3 shows the corresponding seismic risk computed in Ref. 11 for the EPRI89 hazard estimates.

From Table 3, it can be seen that the EPRI89 hazard estimates produce generally much lower seismic risk estimates corresponding to Screening Level 2 than do the LLNL93 hazard estimates. Based on the EPRI89 hazard estimates, only one site has a seismic risk exceeding 1×10^{-6}. Only three other sites have seismic risks exceeding 0.5×10^{-6}. Table 3 includes all sites for which the computed seismic risk exceeds 0.5×10^{-6} based on the mean EPRI89 hazard estimates.

5. Conclusions

If based on the mean LLNL93 hazard estimates (Ref. 8 and 9) it is acceptable to have up to a mean $3x10^{-6}$ annual seismic risk of spent fuel pool failure at the screening level, then Screening Level 2 defined in Section 3 represents a practical screening level. Only 8 of the 69 sites have computed seismic risks greater than $3x10^{-6}$ at this screening level. Screening Level 2 is set at a peak 5% damped spectral acceleration (PSA) level of 1.2g (equivalent to a PGA level of 0.5g).

Based on the mean EPRI89 hazard estimates (Ref. 11), Screening Level 2 would generally result in seismic risk of spent fuel pool failure estimates less than $0.5x10^{-6}$ for spent fuel pools which passed the screening criteria. Only 4 out of 60 sites have computed seismic risks greater than $0.5x10^{-6}$ at this screening level.

The screening criteria given in Refs. 4 and 7 represent a good start on developing screening criteria for spent fuel pools at Screening Level 2. However, I have three significant concerns which are discussed in Sections 3.1 through 3.3. In my judgment, a detailed fragility review of a few spent fuel pools will be necessary in order to address my concerns. These reviews should concentrate on aboveground spent fuel pools with walls not backed by soil backfill. I believe these reviews need to be performed before a set of screening criteria can be finalized at Screening Level 2.

References

1. *Preliminary Draft Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants,* Nuclear Regulatory Commission, June 16, 1999

2. *Draft EPRI Technical Report: Evaluation of Spent Fuel Pool Seismic Failure Frequency in Support of Risk Informed Decommissioning Energy Planning,* Duke Engineering and Services

3. *A Review of Draft NRC Staff Report: Draft Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants,* NEI, August 27, 1999

4. *Seismic Screening Criteria for Assessing Potential Fuel Pool Vulnerabilities at Decommissioning Plants,* NEI, August 18, 1999

5. *Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear Power Plants,* NUREG/CR-5176, Prepared for Nuclear Regulatory Commission, January 1989

6. *An Approach to the Quantification of Seismic Margins in Nuclear Power Plants,* NUREG/CR-4334, Prepared for Nuclear Regulatory Commission, August 1985

7. *A Methodology for Assessment of Nuclear Power Plant Seismic Margin (Revision 1),* (EPRI NP-6041-SL), August 1991

8. *Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains,* NUREG-1488, Nuclear Regulatory Commission, October 1993

9. *Extension to Longer Return Periods of LLNL Spectral Acceleration Seismic Hazard Curves for 69 Sites,* provided by Engineering Research Applications Branch, Nuclear Regulatory Commission, September, 1999

10. Kennedy, R.P., *Overview of Methods for Seismic PRA and Margin Assessments Including Recent Innovations,* CSNI Seismic Risk Workshop, Tokyo, Japan, August 1999

11. Personal Communication from Tom O'Hara, Duke Engineering and Services to Robert Kennedy, October 19, 1999

Table 1
Seismic Hazard Estimates for Peak Spectral Acceleration for PSA
From Refs. 8 and 9 (LLNL 93 Results)

	Peak Spectral Acceleration PSA (cm/sec.^{2})		
Exceedance Frequency H	Vermont Yankee	Robinson	
$1x10^{-3}$	93	232	
$5x10^{-4}$	151	369	
$2x10^{-4}$	246	676	
$1x10^{-4}$	354	991	
$5x10^{-5}$	501	1349	*
$2x10^{-5}$	759	2054	*
$1x10^{-5}$	1058	2801	
$5x10^{-6}$	1396	3915	*
$2x10^{-6}$	1884	6096	*
$1x10^{-6}$	2308	8522	
$5x10^{-7}$	2661	--	**
$2x10^{-7}$	3330	--	**
$1x10^{-7}$	3802	--	**
$5x10^{-8}$	4266	--	**
$2x10^{-8}$	5248	--	**

* By Interpolation

** By Extrapolation

Table 2
Comparison of Seismic Risk Estimated by Various Approaches

C_{HCLPF} = 1.2g PSA, β = 0.4

	Computed Seismic Risk P_F (to be multiplied by 10^{-6})		
Site	Ref. 1 Method Eqn. (3)	Ref. 10 Method Steps 1 through 3	Rigorous Convolution
Vermont Yankee	0.38	1.07	0.89
Robinson	3.7	13.6	13.0

Table 3
Seismic Risk Associated With Screening Level 2

C_{HCLPF} = 1.2g Peak Spectral Acceleration

Site Number	Annual Seismic-Induced Probability of Failure P_F (to be multiplied by 10^{-6})	
	LLNL93 Hazard	EPRI89 Hazard
36	13.6	0.14
18	8.3	1.9
25	6.6	0.57
8	5.5	0.21
43	4.5	0.12
59	4.4	*
21	4.2	*
62	4.1	*
27	2.9	0.38
49	2.8	0.27
40	2.5	0.10
16	2.5	0.14
38	2.3	0.21
63	2.2	0.06
54	2.2	0.26
19	1.8	0.17
32	1.8	0.17
28	1.7	0.04
4	1.6	*
50	1.5	0.20
44	1.5	*
20	1.5	0.55
31	1.4	0.06
39	1.4	0.14
14	1.3	0.60
13	1.3	0.33

Not Available

Attachment 3

Response to Questions Concerning Spent Fuel Pool Seismic-Induced Failure Modes and Locations and the Expected Level of Collateral Damage

by
Robert P. Kennedy
September 2000

1. Introduction

This brief report responds to the following two questions from the NRC Staff:

What are the most likely spent fuel pool failure modes and locations?

I. What is the expected level of collateral damage given a seismic event necessary to fail the spent fuel pool?

The following responses are based upon my judgement without performing any calculations.

2. Most Likely Spent Fuel Pool Failure Modes and Locations

Ref. 1 presents seismic fragility estimates for the Vermont Yankee (BWR) and Robinson (PWR) spent fuel pools. These two fragility estimates are the only spent fuel pool fragility estimates that I have seen. Therefore, my judgement is heavily based on the results presented in Ref. 1.

For Vermont Yankee (BWR), Ref. 1 states that the critical failure mode for the gross structural failure of the pool is an out-of-plane shear failure of the pool floor slab. With this failure mode, the liner will be breached and a large crack will develop through the concrete floor slab within a distance equal to the floor slab thickness from the pool walls. Possibly the entire floor will drop out, but I think that such a gross failure is unlikely. However, the concrete crack will be sufficiently large that the water in the pool will quickly drain out.

Although not reported as the critical failure mode in Ref. 1, my judgement is that for BWR pools, it is at least equally likely that the critical failure mode will be an out-of-plane shear failure of one or more of the pool walls. With this failure mode, the liner will be breached and a major concrete crack will form along the length of the wall within a wall thickness distance from the top of the floor slab. Water will quickly drain out of the pool. However, as much as 4-feet of water depth will likely remain within the pool.

For Robinson (PWR), Ref. 1 states that the critical failure mode is an out-of-plane bending failure of the East wall. With this failure mode, the liner will be breached and

the concrete will become rubbled over a zone equal to the wall thickness at the base of the wall and along the two sides (ends) of the wall. The outward flow of water is likely to be somewhat slower than for a shear crack, but is still expected to be rapid. Probably less water will be retained in the pool than for the case of a shear crack through the wall, and more water will be retained than for the case of a shear crack through the pool floor.

Although not reported as the critical failure mode in Ref. 1, I believe that either of the two shear failure modes reported above for a BWR could also be the critical failure mode for some PWR pools.

Lastly, for stronger spent fuel pools with greater out-of-plane flexure and shear capacities, an in-plane shear failure mode for one or more of the pool walls could control. I suspect this will be the case for particularly some PWR pools. With this failure mode, the liner will be breached and the concrete wall will be cracked in a diagonal X pattern of cracking from near the base of the wall at the edges to near the top of the wall at the opposite edges. The pool will empty to near the base of the wall with probably some small amount of water being retained in the pool.

No matter which of these failure modes occur, drainage of the pool is expected to be fairly rapid. A small, but uncertain, amount of water is likely to remain in the pool with post-seismic-failure water depths ranging from essentially zero depth to about 4-feet of depth depending upon the critical failure mode.

3. Expected Level of Collateral Damage

The seismic capacity of spent fuel pools is high. For spent fuel pools that have successfully passed the NEI/NRC seismic walkdown procedure, I believe the spent fuel pool will have at least about the following seismic fragility capacities:

Spent Fuel Pool

$$\begin{aligned} C_{1\%} &= 0.5g \ \text{PGA} \\ C_{10\%} &= 0.75g \ \text{PGA} \\ C_{50\%} &= 1.25g \ \text{PGA} \end{aligned} \qquad (1)$$

where $C_{1\%}$, $C_{10\%}$, and $C_{50\%}$ are the 1%, 10%, and 50% non-exceedance probability (NEP) peak ground acceleration capacities.

For the Central and Eastern U.S. (CEUS), I estimate the following seismic fragilities:

Loss of Offsite Power

$$\begin{aligned} C_{1\%} &= 0.10g \ \text{PGA} \\ C_{10\%} &= 0.18g \ \text{PGA} \\ C_{50\%} &= 0.35g \ \text{PGA} \end{aligned} \qquad (2)$$

Loss of Even Temporary Safe Usability of Well Designed Buildings and Bridges

$$C_{1\%} = 0.20g \text{ PGA}$$
$$C_{10\%} = 0.35g \text{ PGA} \qquad (3)$$
$$C_{50\%} = 0.75g \text{ PGA}$$

Thus, for a 0.5 PGA scenario ground motion, I would expect less than about a 1% chance of the spent fuel pool failing to hold water, about a 70 to 75% chance that offsite power to the station is lost, and about 20 to 25% of the well designed surrounding buildings (housing communication systems) and bridges being unsafe to use even temporarily. By "well designed", I mean the building or bridge has some form of lateral load carrying system, but does not have nuclear plant or California levels of seismic design. Many CEUS buildings and bridges will have lesser seismic capacity than does this "well-designed" category, and a few might be better. Therefore, over the entire population of nearby buildings and bridges, I would expect more than 20 to 25% would be unsafe for even temporary use.

For a 0.75g PGA scenario ground motion, I would expect less than about a 10% chance of the spent fuel pool failing, about a 90% chance that offsite power is lost, and more than about 50% of the CEUS buildings and bridges being unsafe for even temporary use. At this ground motion level which is within the region of ground motions that dominate the estimated seismic risk of spent fuel pool failures, sufficient power, buildings housing communication systems and emergency services, and bridges will be out-of-service that emergency responses will most likely have to be ad-hoc. Specifically, for ground motion levels that correspond to spent fuel pool failure, within at least 10 miles of the plant I would expect power to have been lost and more than about 50% of the CEUS bridges and buildings (including those housing communication systems and emergency response equipment) being unsafe for even temporary use.

4. Reference

1. *Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear Power Plants,* NUREG/CR-5176, Prepared for Nuclear Regulatory Commission, January 1989

APPENDIX 2C
STRUCTURAL INTEGRITY OF SPENT FUEL POOL STRUCTURES SUBJECT TO HEAVY LOADS DROPS

1. INTRODUCTION

A heavy load drop into the spent fuel pool (SFP) or onto the SFP wall can affect the structural integrity of the SFP. A loss-of-inventory from the SFP 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 2.0×10^{-7} per year (assuming 100 lifts per year). For a non-single failure-proof handling system where a load drop analysis has not been performed, the mean frequency of a loss-of-inventory event caused by a cask drop was estimated to be 2.1×10^{-5} per year. The staff believes that performance and implementation of a load drop analysis that has been reviewed and approved by the staff will substantially reduce the expected frequency of a loss-of-inventory event from a heavy load drop for either a single failure-proof or non-single failure-proof system.

2. ANALYSIS

The staff revisited NUREG-0612[2] (Ref. 1) to review the evaluation and the supporting data available at that time to determine its applicability to and usefulness for evaluation of heavy load drop concerns at decommissioning plants. In addition, three additional sources of information were identified by the staff and used to reassess the heavy load drop risk:

(1) U.S. Navy crane experiences (1990s Navy data) for the period 1996 through mid-1999.

(2) WIPP/WID-96-2196 (Ref. 2), "Waste Isolation Pilot Plant Trudock Crane System Analysis," October 1996 (WIPP).

(3) NEI data on actual SFP cask lifts at U.S. commercial nuclear power plants (Ref.3).

The staff's first area of evaluation was the frequency of heavy load drops. The number of occasions (incidents) where various types of faults occurred that potentially could lead to a load drop was investigated. Potential types of faults investigated included improper operation of equipment, improper rigging practices, poor procedures, and equipment failures. Navy data from the 1990s were compared to the data used in NUREG-0612. The data gave similar, but not identical, estimates of the various faults leading to heavy load drops (see Table A2c-1). The NEI cask handling experience also supported the incident data used in this evaluation, and in NUREG-0612. Once the frequency of heavy load drops was estimated (i.e., load drops per lift), the staff investigated the conditional probability that such a drop would seriously damage the SFP (either the bottom or walls of the pool) to the extent that the pool would drain very rapidly and it would not be possible to refill it using onsite or offsite resources. To do this the staff used fault trees taken from NUREG-0612 (see Figure A2c-1). By mathematically combining the frequency of load drops with the conditional probability of pool failure given a load drop, the staff was able to estimate the frequency of heavy load drops causing a zirconium fire at decommissioning facilities.

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[2]NUREG-0612 documented the results of the staff's review of the handling of heavy loads at operating nuclear power plants and included the staff's recommendations on actions that should be taken to assure safe handling of heavy loads.

3. FREQUENCY OF HEAVY LOAD DROP

The database used in this evaluation (primarily the 1990s Navy data) considered a range of values for the number of occasions where faults occurred, the frequency of heavy load drops and the availability of backup systems. The reason that there is a range of values is that while the number of equipment failures and load drops were reported, the denominator of the estimate, the actual total number of heavy load lifts, was only available based on engineering judgement. High and low estimates of the ranges were made, and it was assumed that the data had a log normal distribution with the high and low number of the range representing the 5th and 95th percentile of the distribution. From this the mean of the distribution was calculated. Data provided by NEI on actual lifts and setdowns of SFP casks at commercial U.S. nuclear power plants (light water and gas-cooled reactors) gave a similar estimated range for the incidents at the 95 percent confidence level.

Load drops were broken down into two categories: failure of lifting equipment and failure to secure the load.

Crane failures (failure of lifting equipment) were evaluated using the fault tree shown in Figure A2c-1, which comes from NUREG-0612. At the time that 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 radionuclide releases were expected if the pool were drained. It was assumed in NUREG-0612 that after this period, the fuel gap noble gas inventory had decayed and no zirconium fire would have occurred. Today, most decommissioning facilities use high-density storage racks. This analysis evaluates results at one year after reactor shutdown. Our engineering evaluations indicate that for today's fuel configurations, burnup, and enrichment, a zirconium cladding fire may occur if the pool were drained during a period as long as 5 years.

A literature search performed by the staff searching for data on failure to secure loads identified a study (WIPP report) that included a human error evaluation for improper rigging. This study was used by the staff 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 the most probable human error was associated with attaching the lifting legs to the lifting fixture. In the WIPP report, the failure to secure the load (based on a 2-out-of-3 lifting device) was estimated 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 (Ref. 4) information, the mean failure rate because of improper rigging was estimated in the WIPP report to be 8.7×10^{-7} per lift. Our requantification of the NUREG-0612 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.

These estimates provided a rate for failures per lift. Based on input from the nuclear industry at the July 1999 SFP workshop, we assumed in our analysis that there will be a maximum of 100 cask lifts per year at a decommissioning plant.

4. EVALUATION OF THE LOAD PATH

Just because a heavy load is dropped does not mean that it will drop on the SFP wall or on the pool floor. It may drop at other locations on its path. A load path analysis is plant-specific. In NUREG-0612 it was estimated that the heavy load was near or over the SFP for between

5 percent and 25 percent 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 in some circumstances from 36 feet and higher depending on the assumptions) when it is over the pool floor, and if a plant-specific load drop analysis had not been performed,[3] then damage to the pool floor would result in loss-of-inventory. In addition we looked at the probability that the load drop occurred over the pool wall from 8 to 10 inches above the edge of the pool wall. In our analysis we evaluated the chances the load was raised sufficiently high to fail the pool and evaluated the likelihood that the drop happened over a vulnerable portion of the load path. Table A2c-2 presents the results for a heavy load drop on or near the SFP. Based on NUREG-0612, if the cask were dropped on the SFP floor, the likelihood of a loss-of-inventory given the drop is 1.0. Based on the evaluation presented in NUREG/CR-5176 (Ref. 5), if the load were dropped on the SFP wall, the likelihood of a loss-of-inventory given the drop is 0.1.

5. CONCLUSION

Our heavy load drop evaluation is based on the method and fault trees developed in NUREG-0612. New 1990s Navy data were used to quantify the failure rate 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 onto the pool floor or onto the pool wall from a single failure-proof system to be 2.0×10^{-7} per year for 100 lifts per year.

However, only some of the plants that will be decommissioning plants in the future currently have single failure-proof systems. Historically, many facilities have chosen to upgrade their crane systems to become single failure-proof. However, this is not an NRC requirement. The guidance in NUREG-0612, phase 2 calls for systems to either be single failure-proof or if they are non-single failure-proof to perform a load drop analysis. The industry through NEI has indicated that it is willing to commit to follow the guidance of all phases of NUREG-0612.

For licensees that choose the non-single failure-proof handling system option in NUREG-0612, we based the mean frequency of a loss-of-inventory event on the method used in NUREG-0612. In NUREG-0612, an alternate fault tree than that used for the single failure-proof systems was used to estimate the frequency of exceeding the release guidelines (loss-of-inventory) for a non-single failure-proof system. We calculated the mean frequency of catastrophic pool failure (for drops into the pool, or on or near the edge of the pool) for non-single failure-proof systems to be about 2.1×10^{-5} per year when corrected for the 1990s Navy data and 100 lifts per year. This estimate exceeds the proposed pool performance guideline of 1×10^{-5} per year. The staff believes that a licensee which chooses the non-single failure-proof handling system option in NUREG-0612 can reduce this estimate to the same range as that for single failure-proof systems by performing a comprehensive and rigorous load drop analysis.

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[3] If a load drop analysis were performed, it means that the utility has evaluated the plant design and construction to pick out the safest path for the movement of the heavy load. In addition, it means that the path chosen has been evaluated to assure that if the cask were to drop at any location on the path, it would not catastrophically fail the pool or its support systems. If it is determined that a portion of the load path would fail if the load were dropped, the as-built plant must be modified (e.g., by addition of an impact limiter or enhancement of the structural capacity of that part of the building) to be able to take the load drop or a different safe load path must be identified.

The load drop analysis is assumed to include implementation of plant modifications or load path changes to assure the SFP would not be catastrophically damaged by a heavy load drop.

6. REFERENCES

(1) U.S. Nuclear Regulatory, “Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36,“ NUREG-0612, July 1980.

(2) Pittsburgh, Westinghouse, P.A., and Carlsbad, WID, N.M., "Waste Isolation Pilot Plant Trudock Crane System Analysis," WIPP/WID-96-2196, October 1996.

(3) Richard Dudley, NRC memorandum to Document Control Desk, “Transmittal of Information Received From the Nuclear Energy Institute (NEI) For Placement In The Public Document Room,” dated September 2, 1999.

(4) Swain, A.D., and H.E. Guttmann, “Handbook of Reliability Analysis with Emphasis on Nuclear Power Plant Applications,” NUREG/CR-1278, August 1983.

(5) P.G. Prassinos, et al., "Seismic Failure and Cask Drop Analyses of Spent Fuel Pools at Two Representative Nuclear Power Plants," NUREG/CR-5176, LLNL, January 1989.

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 report the number of lifts made and only provided information about the number of incidents. The cask loading experience at light water reactors and Ft. St. Vrain 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 SFP to result in a loss-of-inventory was estimated to be between 0.5 percent and 6.25 percent 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. No time motion study was performed to account for the fraction of time the load is over any particular location.

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.

5. Load drop analysis

   The benefit of a load drop analysis is believed to be significant, but is unquantified. A load drop analysis involves mitigation of the potential drop by methods such as changing the safe load path, installation of impact limiters, or enhancement of the structure, as necessary, to be able to withstand a heavy load drop at any location on a safe load path.

Table A2c-1 Summary of the 1996-1999 Navy Crane Data

		ID	Non-rigging Fraction	Rigging Fraction	Total Fraction
	Summary by Incident Type (fraction of events)				
	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 percent or “improper rigging” by incident cause were rigging failures during crane movement, and 70 percent of “improper rigging” by incident cause were rigging errors.

2. F1 - Load hangup resulting from operator error (assume 50 percent of “damaged load” and “load collision” lead to hangup)
   F2 - Failure of component with a backup component (assume 50 percent 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 because of 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 because of 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 because of 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 because of 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

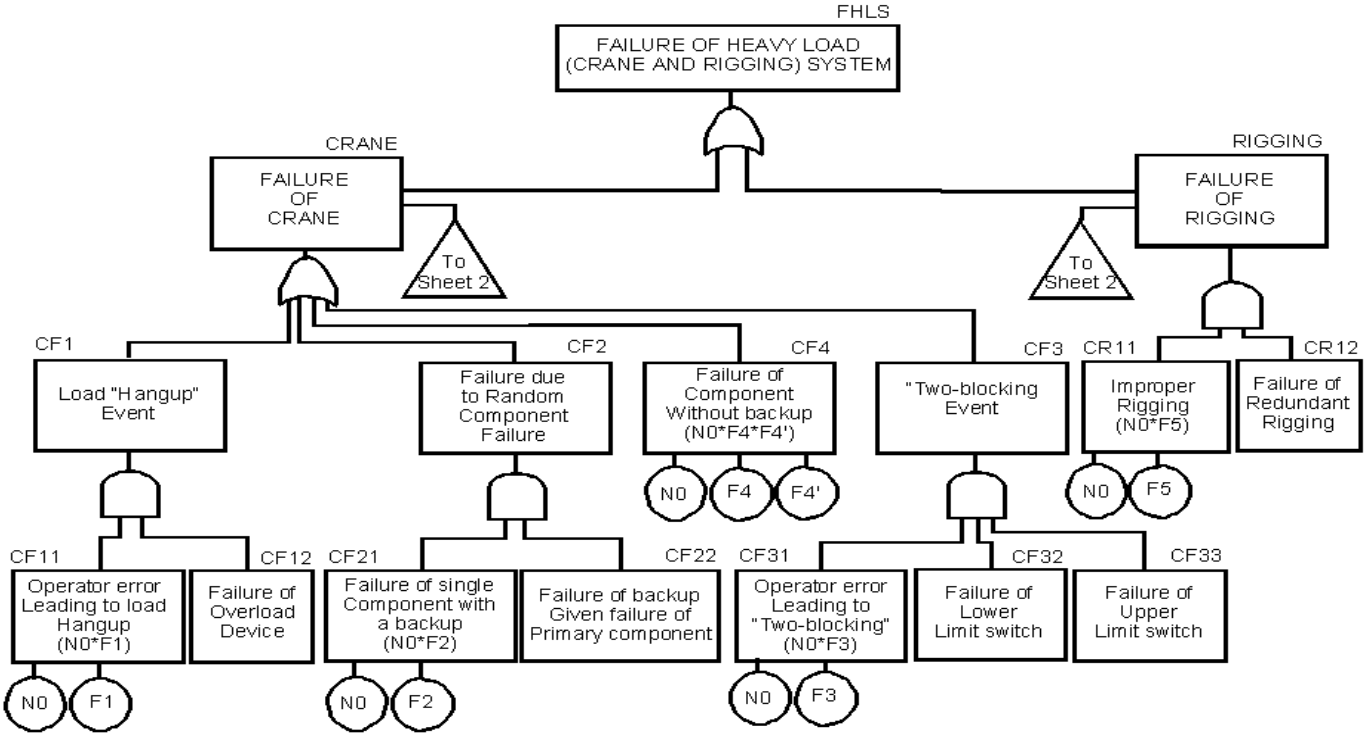
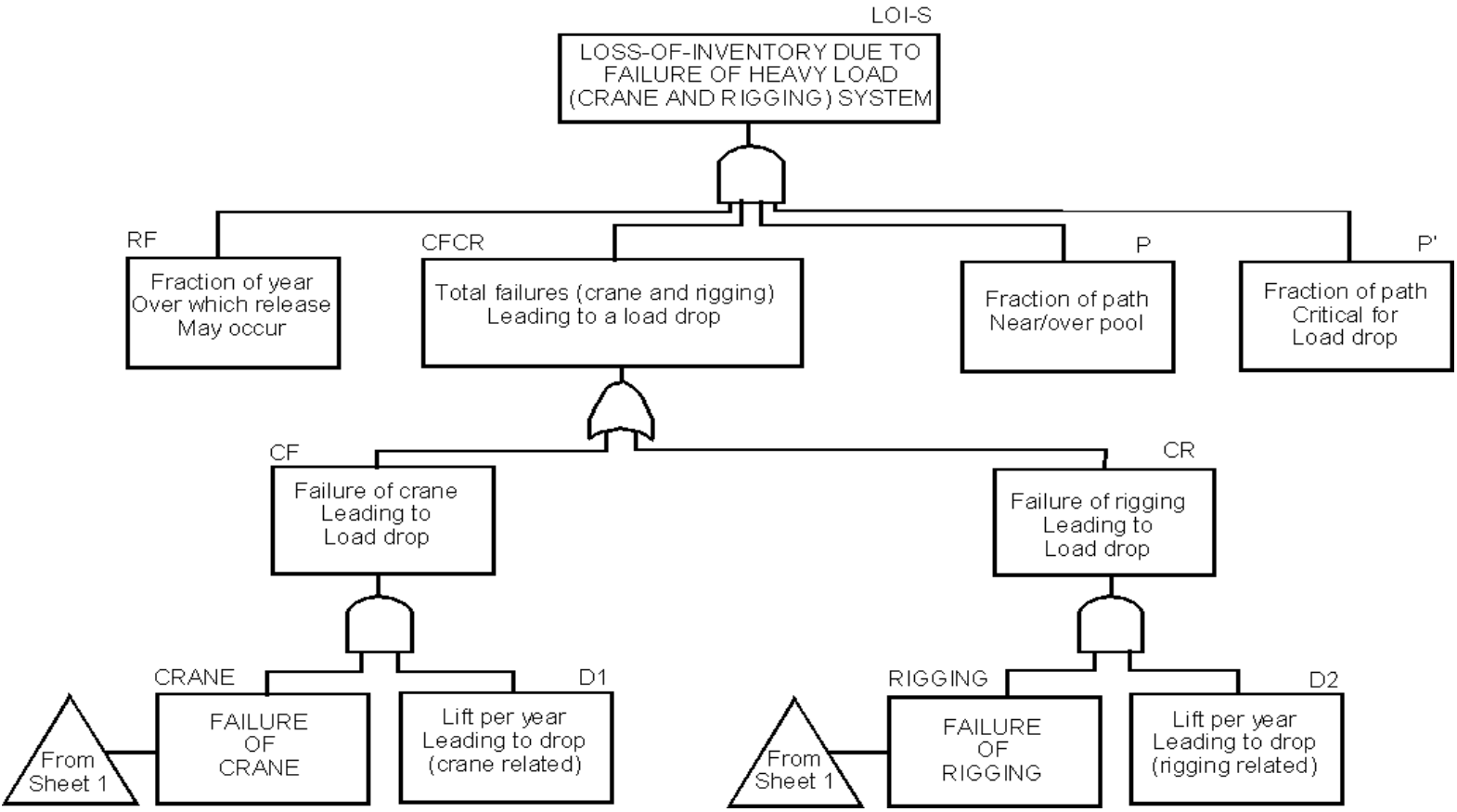

FHLS
FAILURE OF HEAVY LOAD (CRANE AND RIGGING) SYSTEM
CRANE
FAILURE OF CRANE
To Sheet 2
RIGGING
FAILURE OF RIGGING
To Sheet 2
CF1
Load "Hangup" Event
CF2
Failure due to Random Component Failure
CF4
Failure of Component Without backup (N0*F4*F4')
CF3
"Two-blocking Event
CR11
Improper Rigging (N0*F5)
CR12
Failure of Redundant Rigging
N0
F4
F4'
N0
F5
CF11
Operator error Leading to load Hangup (N0*F1)
CF12
Failure of Overload Device
CF21
Failure of single Component with a backup (N0*F2)
CF22
Failure of backup Given failure of Primary component
CF31
Operator error Leading to "Two-blocking" (N0*F3)
CF32
Failure of Lower Limit switch
CF33
Failure of Upper Limit switch
N0
F1
N0
F2
N0
F3

Figure A2c-1 (sheet 2 of 2) - Heavy Load Drop Fault Trees



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 (SFP) 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 4.1×10^{-9} per year. The estimated frequency of loss of support systems leading to SFP 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 an SFP that is contained in a PWR auxiliary building or a BWR secondary containment structure. In general, PWR SFPs are located on, or below grade, and BWR SFPs, 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 SFP. Aircraft damage can affect the structural integrity of the SFP 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:

and where:

A_{eff} = total effective target area H= height of facility

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

where:

Equation A2d-2

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

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

A_f = effective fly-in area L= length of facility
A_s = effective skid area W= width of facility
WS= wing span S= aircraft skid distance
cotΔ= mean of cotangent of aircraft impact angle 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 SFP DAMAGE

The frequency for significant PWR SFP 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 0.45 (large aircraft penetrating 5-ft of reinforced concrete) that the crash resulted in significant damage. This value (i.e., 0.45) is an interpolation from a table in NUREG/CR-0542 reproduced in Table A2d-4. If 1-of-2 aircraft are large and 1-of-2 crashes result in spent fuel uncovery, then the estimated range is 1.3×10^{-11} to 6.0×10^{-8} per year. The average frequency was estimated to be 4.1×10^{-9} per year.

The mean frequency for significant BWR SFP 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 because of other structures. Mark-III secondary containments may reduce the likelihood of penetration, since the SFP 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)	cotΔ	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)

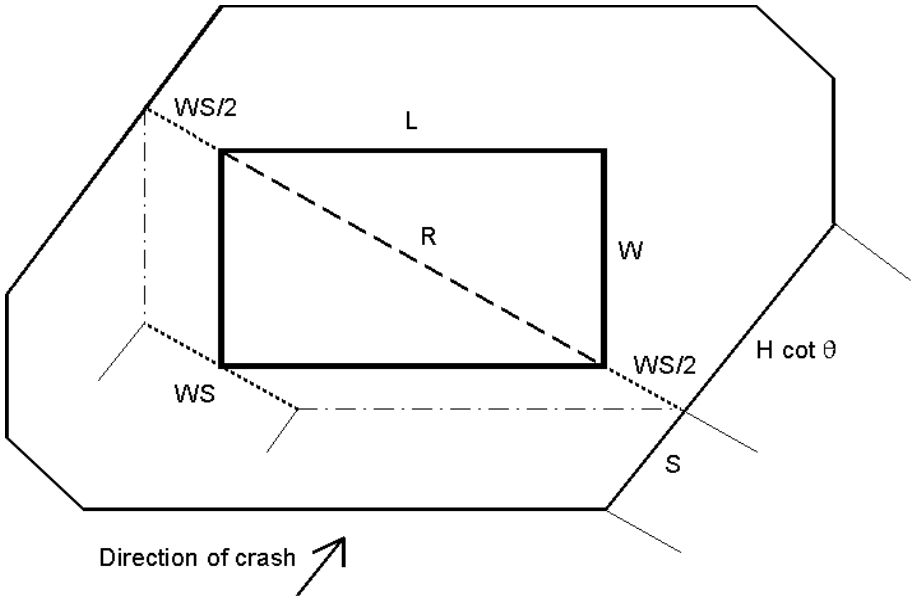
Period	Air Carriers	Air Taxes	General Aviation	Military Aviation	Total[1]
1995	1.72×10^{-7}	2.47×10^{-6}	2.45×10^{-5}	5.03×10^{-7}	2.76×10^{-5}
1993-1995	1.60×10^{-7}	2.64×10^{-6}	2.82×10^{-5}	6.47×10^{-7}	3.16×10^{-5}
1991-1995	1.57×10^{-7}	2.58×10^{-6}	2.89×19^{-5}	7.23×10^{-7}	3.23×10^{-5}
1986-1995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



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 (SFP) 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 SFP 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 onsite or offsite 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 offsite 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 offsite resources to the same degree. Therefore, the probability of failing to bring in the offsite resources can be argued to be the same as for the seismic case, i.e., $1x10^{-04}$, 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 percent of all reported tornadoes were in the F2 to F3 range and about 2.5 percent 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 SFPs

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

Δ_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 SFP 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 SFP 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 SFP, 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	7.4×10^{-05}	4.4×10^{-05}	1.5×10^{-05}	3.5×10^{-06}	5.6×10^{-07}	3.1×10^{-08}
B	5.6×10^{-05}	3.3×10^{-05}	1.1×10^{-05}	2.5×10^{-06}	3.7×10^{-07}	2.1×10^{-08}
C	2.9×10^{-05}	1.5×10^{-05}	4.1×10^{-06}	8.9×10^{-07}	1.3×10^{-07}	4.7×10^{-09}
D	3.6×10^{-06}	1.6×10^{-06}	3.9×10^{-07}	8.7×10^{-08}	1.6×10^{-08}	---
USA	3.5×10^{-05}	2.0×10^{-05}	6.1×10^{-06}	1.4×10^{-06}	2.2×10^{-07}	1.0×10^{-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

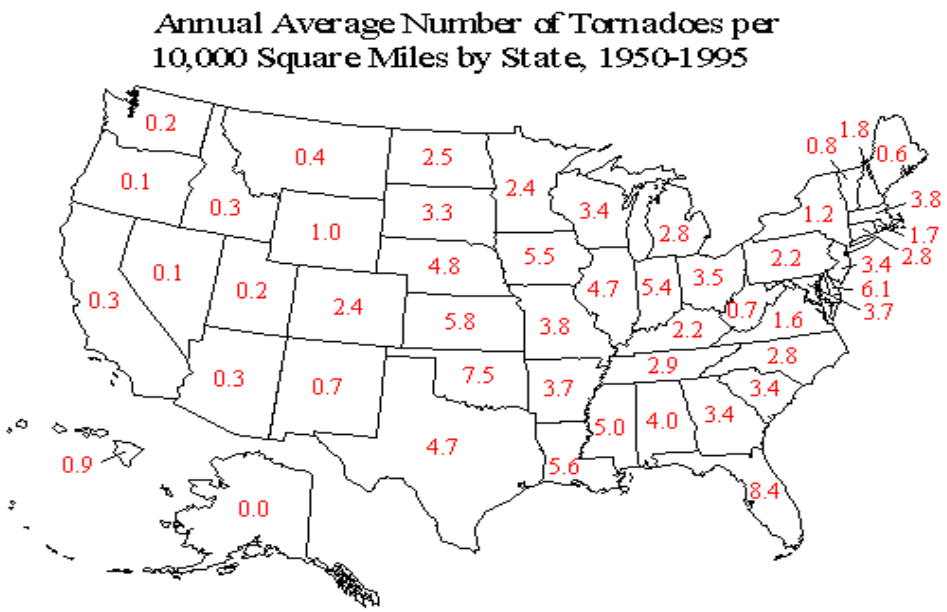


Figure A2e-2

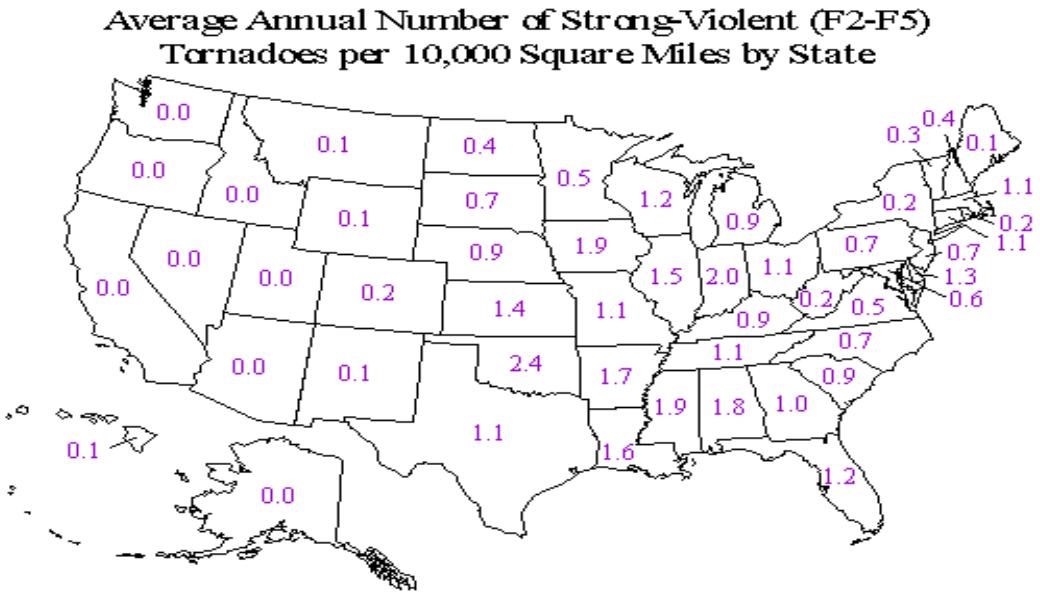


Figure A2e-3 Sketch of Hypothetical F2 Tornado Illustrating Variations

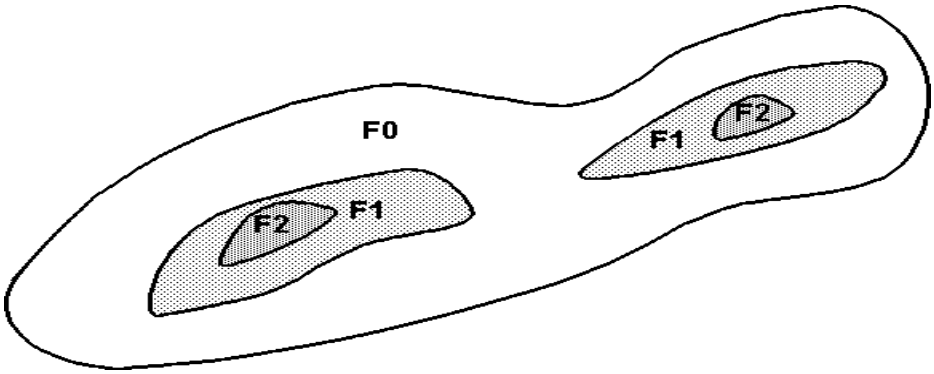


Figure A2e-4 Tornado Risk Regionalization Scheme (from NUREG/CR-2944)

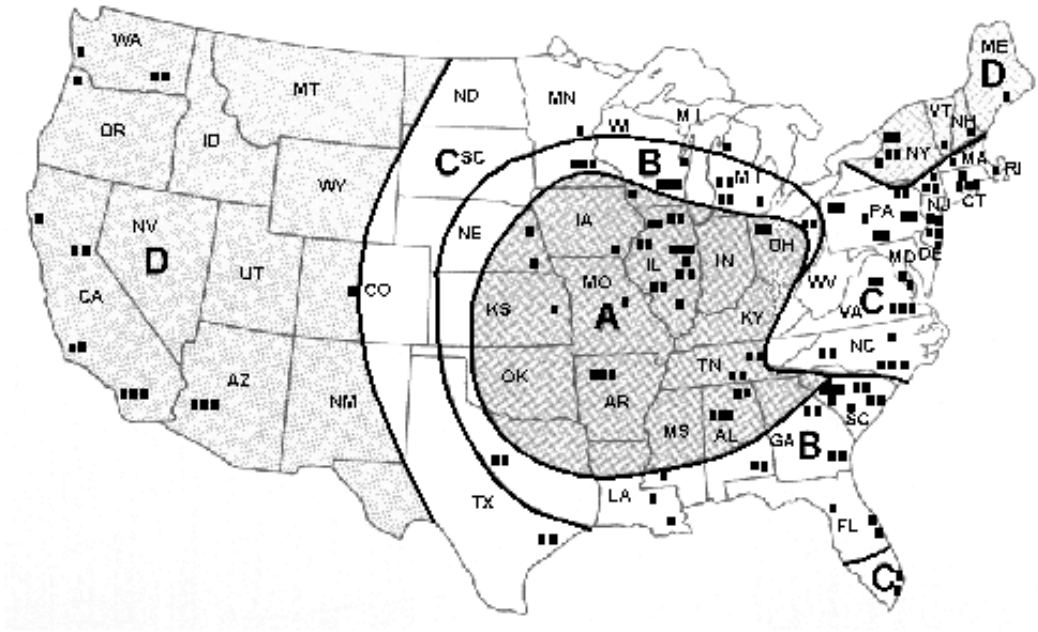
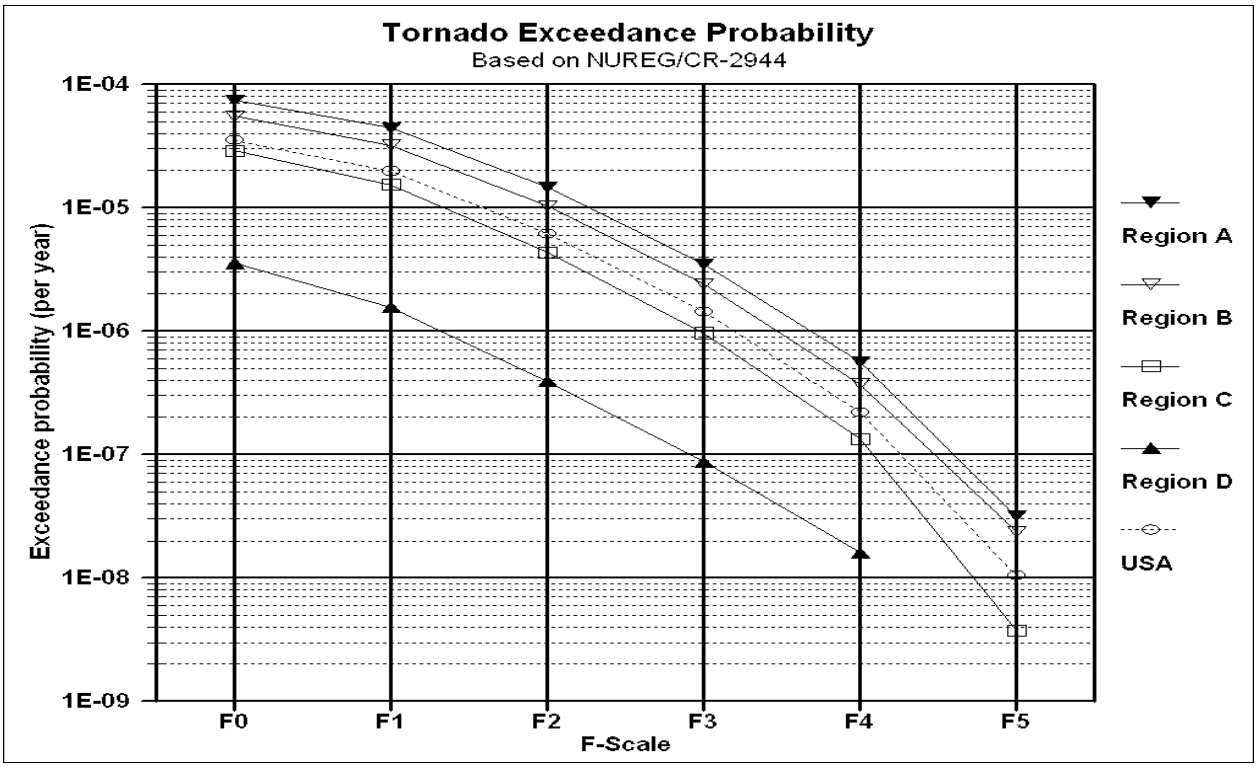


Figure A2e-5 Tornado Exceedance Probability For Each F-scale



APPENDIX 3
ASSESSMENT OF THE POTENTIAL FOR CRITICALITY IN DECOMMISSIONING SPENT FUEL POOL

3.1 INTRODUCTION

The staff criticality assessment includes both a more classical deterministic study and a qualitative risk study. The conclusion in Section 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 mitigating actions. The risk study considered whether the identified scenarios are credible and whether any of the identified compensatory measures are justified given the frequency of the initiating scenario. This appendix combines the risk study, discussed in Section 3, 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

Because of the processes involved and lack of data, it was not possible to perform a quantitative risk assessment for criticality in the spent fuel pool. Section 3.2.2 of this appendix, is a deterministic study in which the staff performed 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 because of criticality in the SFP, and its conclusions that the potential risk from SFP criticality is sufficiently small.

In section 3.2.2, the staff evaluated 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. 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 adequately 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 because of 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 percent subcriticality margin and to submit to the NRC proposed actions to monitor the margin or confirm that this 5 percent 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, were discounted because of 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 percent 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 because of 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 because of 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 percent and 25 percent 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 percent 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.

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 because of 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 should be considered at all plants until all high-density racks are removed from the SFP.

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.

Assessment of the Potential for Criticality in Decommissioned Spent Fuel Pools

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Introduction

The staff has performed a series of calculations to assess the potential for a criticality accident in the spent fuel pool of a decommissioned nuclear power plant. This work was undertaken to support the staff's efforts to develop a decommissioning rule. Unlike operating spent fuel storage pools, decommissioned pools will have to store some number of spent fuel assemblies which have not achieved full burnup potential for extended periods of time which were used in the final operating cycle of the reactor. These assemblies constitute approximately one third of the assemblies in the final operating cycle of the reactor. These assemblies are more reactive than those assemblies normally stored in the pool which have undergone full burnup. Operating reactors typically only store similarly reactive assemblies for short periods of time during refueling or maintenance outages. As we will see in this report, the loss of geometry alone could cause a criticality accident unless some mitigative measures are in place.

When spent fuel pools were originally conceived, they were intended to provide short term storage for a relatively small number of assemblies while they decayed for a period of time sufficient to allow their transport to a long term storage facility. Because a long term storage facility is not available, many reactor owners have had to change the configuration of their spent fuel pools on one or, in some cases, several occasions. This practice has led to a situation where there are many different storage configurations at U.S. plants utilizing some combination of geometry, burnup, fixed poisons, and boration, to safely store spent fuel.

The current state of spent fuel pools significantly complicates the task of generically analyzing potential spent fuel pool storage configurations. Therefore, the staff decided to take a more phenomenalogical approach to the analysis. Rather than trying to develop specific scenarios for the different types of loading configurations, we decided to analyze storage rack deformation and degradation by performing bounding analyses using typical storage racks. The results of these analyses will be used to formulate a set of generic conclusions regarding the physical controls necessary to prevent criticality. The impact of five pool storage assumptions on the conclusions in this report will be discussed throughout the text. Furthermore, for the purposes of this work, it is assumed that the postulated criticality event is unrecoverable when the water level reaches the top of the fuel. This means that events such as a loss of water leading to a low density optimal moderation condition caused by firefighting equipment will not be considered.

It is important to reinforce the point that these analyses are intended as a guide only and will be used to evaluate those controls that are either currently in place or will need to be added to maintain subcriticality. These analyses will not be used to develop specific numerical limits which must be in place to control criticality as they cannot consider all of the possible plant specific variables. We will, however, define the controls that would be effective either individually or in combination to preclude a criticality accident.

Description Of Methods

The criticality analyses were performed with three-dimensional Monte Carlo methods using ENDF/B-V based problem specific cross sections (Ref. 1). Isotopic inventories were predicted using both one- and two-dimensional transport theory based methods with point depletion. SCALE 4.3 (Ref. 2) was used to perform the Monte Carlo, one-dimensional transport, cross section processing, and depletion calculations. Specifically, the staff used KENO-VI, NITAWL-1, BONAMI, XSDRN, and ORIGEN. The two-dimensional transport theory code NEWT

(Ref. 3) was used for Boiling Water Reactor (BWR) lattice depletion studies. NEWT uses the method of characteristics to exactly represent the two-dimensional geometry of the problem. NEWT uses ORIGEN for depletion. Cross section data were tracked and used on a pin cell basis for the BWR assessments. The staff developed post processing codes to extract the information from NEWT and create an input file suitable for use with SCALE. Both the 238 and the 44 group ENDF/B-V based libraries were used in the project. Refer to Sample Input Deck at the end of Appendix 7 for a listing of one of the input decks used in this analysis. SCALE has been extensively validated for these types of assessments. (see References 4, 5, and 6)

Problem Definition

Compression (or expansion) events were analyzed in two ways. First, the assembly was assumed to crush equally in the x and y directions (horizontal plane). Analyses were performed with and without the fixed absorber panels *without* soluble boron and with fuel at the most reactive point allowed for the configuration. In these cases, the fuel pin pitch was altered to change the fuel to moderator ratio. These scenarios are intended to simulate the crushing (or expansion) of a high density configuration when little or no rack deformation is necessary to apply force to the fuel assembly. The scenarios are also applicable to low density rack deformation in which the rack structure collapses to the point at which force is applied to the assemblies. The second type of compression event involved changing the intra-assembly spacing, but leaving the basic lattice geometry unchanged. These simulations were intended to simulate compression events in which the force applied to the rack is insufficient to compress the assembly.

Discussion Of Results

Several observations are common to both Pressurized Water Reactor (PWR) and BWR rack designs. First of all, poisoned racks should remain subcritical during all compression type events assuming that the poison sheeting remains in place (in other words, that it compresses with the rack and does not have some sort of brittle failure). Secondly, criticality cannot be precluded by design following a compression event for low density, unpoisoned (referring to both soluble and fixed poisons) storage racks.

PWR Spent Fuel Storage Racks

The analyses and this discussion will differentiate between high and low density storage. High density storage is defined as racks that rely on both fixed poison sheets and geometry to control reactivity and low density storage relies solely upon geometry for reactivity control. The results of the analyses for the high density storage racks are summarized in Figure 1. When discussing Figure 1 it should be noted that the analyses supporting Figure 1 were performed without soluble boron and with fuel at the most reactive point allowed for the rack. These assumptions represent a significant conservatism of at least 20 percent delta-k. Figure 1 demonstrates that even with compression to an optimal geometric configuration, criticality is prevented by design (for these scenarios we are not trying to maintain a k_{eff} less than 0.95). The poison sheeting, boral in this case, is sufficient to keep the configuration subcritical.

The results for the low density storage rack are given in Figure 2. As can be seen, criticality cannot be entirely ruled out on the basis of geometry alone. Therefore, we examined the conservatism implicit in the methodology and assessed whether there is enough margin to not

require any additional measures for criticality control. There are two main sources of conservatism in the analyses; using fuel at the most reactive state allowed for the configuration and not crediting soluble boron. By relaxing the assumption that all of the fuel is at its peak expected reactivity, we have demonstrated by analyzing several sample storage configurations that the rack eigenvalue can be reduced to approximately 0.998 (see Table 1). The storage configurations analyzed included placing a most reactive bundle every second, fourth, sixth and eighth storage cell (see Figure 3). The assemblies used between the most reactive assembly were defined by burning the 5 w/o U_{235} enriched Westinghouse 15x15 assembly to 55 GWD/MTU which is a typical discharge burnup for an assembly of this type. This study did not examine all possible configurations so this value should be taken as an estimate only. However, the study does suggest that scattering the most reactive fuel throughout the pool would substantially reduce the risk of a criticality accident. It is difficult to entirely relax the assumption of no soluble boron in the pool, but its presence will allow time for recovery actions during an event that breaches the SFP liner and compresses the rack but does not rapidly drain the pool.

Although not all-inclusive because all fuel and rack types were not explicitly considered, the physical controls that were identified are generically applicable. The fuel used in this study is a Westinghouse 15x15 assembly enriched to 5 w/o U_{235} with no burnable absorbers. The Westinghouse 15x15 assembly has been shown by others (Ref. 7) to be the most reactive PWR fuel type when compared to a large number of different types of PWR fuel. Furthermore, the use of 5 w/o U_{235} enriched fuel will bound all available fuel types because it represents the maximum allowed enrichment for commercial nuclear fuel.

BWR Spent Fuel Storage Racks

In these analyses, we differentiated between high and low density BWR racks. The conservatism inherent in the analyses must be considered (for BWR racks, the use of the most reactive fuel allowed only) when considering the discussion of these results. The results of the analyses of high density BWR racks are given in Figure 4. As can be seen, criticality is prevented by design for the high density configurations. The poison sheets remain reasonably intact following the postulated compression event. The poison sheeting (in this case Boraflex) is sufficient to maintain subcriticality.

The results of the low density BWR rack analyses are shown in Figure 5. Here, as with the PWR low density racks, criticality cannot be prevented by design. Once again we assessed the impact of eliminating some of the conservatism in the analyses which in the case of BWR storage is only related to the reactivity of the assembly. Analyses were performed placing a most reactive assembly in every second, fourth, sixth and eighth storage cell. The assemblies placed between the most reactive assemblies were defined by burning the 4.12 w/o enriched General Electric (GE) 12 assembly to 50 GWd/MTU. These analyses demonstrate that it is possible to reduce the rack eigenvalue to approximately 1.009 (see Table 1). As previously mentioned, this study did not include all possible configurations so this value should be taken as an estimate only. Because BWR pools are not borated, there is no conservatism from the assumption of no soluble boron.

Boraflex degradation is another problem that is somewhat unique to BWR spent fuel storage racks. This is true because of the fact that BWR storage pools do not contain soluble boron that provides the negative reactivity in PWR pools to offset the positive effect of Boraflex degradation. Therefore, some compensatory measures need to be in place to provide

adequate assurance that Boraflex degradation will not contribute to a criticality event. In operating reactor spent fuel pools that use Boraflex, licensees use some sort of surveillance program to ensure that the 5 percent subcritical margin is maintained. These programs should be continued during and following decommissioning. No criticality calculations were performed for this study to assess Boraflex degradation because it is conservatively assumed that the loss of a substantial amount of Boraflex will most likely lead to a criticality accident.

These analyses are not all inclusive, but we believe that the physical controls identified are generically applicable. We examined all of the available GE designed BWR assemblies for which information was available and identified the assembly used in the study to have the largest K_{inf} in the standard cold core geometry (in other words, in the core with no control rods inserted at ambient temperature) at the time of peak reactivity. This assembly was a GE12 design (10x10 lattice) enriched to an average value of 4.12 w/o U_{235}. Only the dominant part of the lattice was analyzed and it was assumed to span the entire length of the assembly. This conservatism plus the fact that the assembly itself is highly enriched and designed for high burnup operation has led the staff to conclude that these analyses are generically applicable to BWR spent fuel storage pools.

Conclusions

One scenario that has been identified which could lead to a criticality event is a heavy load drop or some other event that compresses a low density rack filled with spent fuel at its peak expected reactivity. This event is somewhat unique to decommissioned reactors because there are more low burnup (high reactivity) assemblies stored in the spent fuel pool that were removed from the core following its last cycle of operation, than in a SFP at an operating plant.

To address the consequences of the compression of a low density rack, there are two strategies that could be used, either individually or in combination. First, the most reactive assemblies (most likely the fuel from the final cycle of operation) could be scattered throughout the pool, or placed in high density storage if available. Second, all storage pools, regardless of reactor type, could be borated.

References

1 “ENDF/B-V Nuclear Data Guidebook,” EPRI-NP 2510, July 1982.

2 "SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations," NUREG/CR-0200. Oak Ridge National Laboratory, 1995.

3 Tony Ulses, "Evaluation of NEWT for Lattice Physics Applications," Letter Report, May 1999.

4 M.D. DeHart and S.M. Bowman, "Validation of the SCALE Broad Structure 44-Group ENDF/B-V Cross Section Library for use in Criticality Safety Analysis," NUREG/CR-6102, Oak Ridge National Laboratory, 1994.

5 O.W. Hermann, et. al., "Validation of the SCALE System for PWR Spent Fuel Isotopic Composition Analyses," ORNL/TM-12667, Oak Ridge National Laboratory, March 1995.

6 W.C. Jordan, et. al., "Validation of KENOV.a Comparison with Critical Experiments," ORNL/CSD/TM-238, Oak Ridge National Laboratory, Oak Ridge National Laboratory, 1986.

7. "Licensing Report for Expanding Storage Capacity in Harris Spent Fuel Pools C and D," HI-971760, Holtec International, May 26, 1998, (Holtec International Proprietary)

Sample Input Deck Listing and
Tables and Figures

```
=csas26   parm=size=10000000
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238groupndf5  latticecell
'Data From SAS2H - Burned 5 w/o Fuel
o-16       1 0 0.4646E-01 300.00 end
kr-83      1 0 0.3694E-05 300.00 end
rh-103     1 0 0.2639E-04 300.00 end
rh-105     1 0 0.6651E-07 300.00 end
ag-109     1 0 0.4459E-05 300.00 end
xe-131     1 0 0.2215E-04 300.00 end
'xe-135    1 0 0.9315E-08 300.00 end
cs-133     1 0 0.5911E-04 300.00 end
cs-134     1 0 0.5951E-05 300.00 end
cs-135     1 0 0.2129E-04 300.00 end
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la-140     1 0 0.1485E-06 300.00 end
nd-143     1 0 0.4070E-04 300.00 end
nd-145     1 0 0.3325E-04 300.00 end
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fe         2 0 1.4838E-4   300.0 end
zr         2 0 4.2982E-2   300.0 end
'Water w/ 2000 ppm boron
h2o        3 0.99 300.0 end
'b-10       3 0 2.2061E-5   300.0 end
'SS structural material
ss304      4 0.99 300.0 end
'Boral (model as b4c-al using areal density of b-10 @ -- g/cm^2 and 0.18 atom percent b-10 in
nat. b)
'Excluded Proprietary Information
end comp
'squarepitch card excluded - Proprietary Information
more data
dab=999
end more
read param
gen=103 npg=3000 xs1=yes pki=yes gas=yes flx=yes fdn=yes far=yes nb8=999
end param
read geom
'geom cards excluded - Proprietary Information
end geom
read array
ara=1 nux=15 nuy=15 nuz=1 fill
  1   1   1   1   1   1   1   1   1   1   1   1   1   1   1
  1   1   1   1   1   1   1   1   1   1   1   1   1   1   1
  1   1   2   1   1   2   1   1   1   2   1   1   2   1   1
  1   1   1   1   1   1   1   2   1   1   1   1   1   1   1
  1   1   1   1   2   1   1   1   1   1   2   1   1   1   1
  1   1   2   1   1   1   1   1   1   1   1   1   2   1   1
  1   1   1   1   1   1   1   1   1   1   1   1   1   1   1
  1   1   1   2   1   1   1   2   1   1   1   2   1   1   1
  1   1   1   1   1   1   1   1   1   1   1   1   1   1   1
  1   1   2   1   1   1   1   1   1   1   1   1   2   1   1
  1   1   1   1   2   1   1   1   1   1   2   1   1   1   1
  1   1   1   1   1   1   1   2   1   1   1   1   1   1   1
  1   1   2   1   1   2   1   1   1   2   1   1   2   1   1
  1   1   1   1   1   1   1   1   1   1   1   1   1   1   1
  1   1   1   1   1   1   1   1   1   1   1   1   1   1   1
end fill
end array
read bounds all=mirror end bounds
read mixt sct=2 eps=1.e-01 end mixt
read plot
scr=yes
ttl='w15x15 in High Density Rack'
xul=-11.5 yul= 11.5 zul=0.0
```

```
xlr= 11.5 ylr=-11.5 zlr=0.0
uax=1 vdn=-1 nax=750
end plot
end data
end
```

Table 1 Eigenvalue (using infinite multiplication factor) reduction from skipping cells between high reactivity assemblies.

Skipped Cells	PWR	BWR
2	1.03533	1.02628
4	1.01192	1.01503
6	1.00363	1.01218
8	0.99786	1.01059

Figure 1 PWR High Density Storage Rack Eigenvalue Following Compressive/Expansion Events

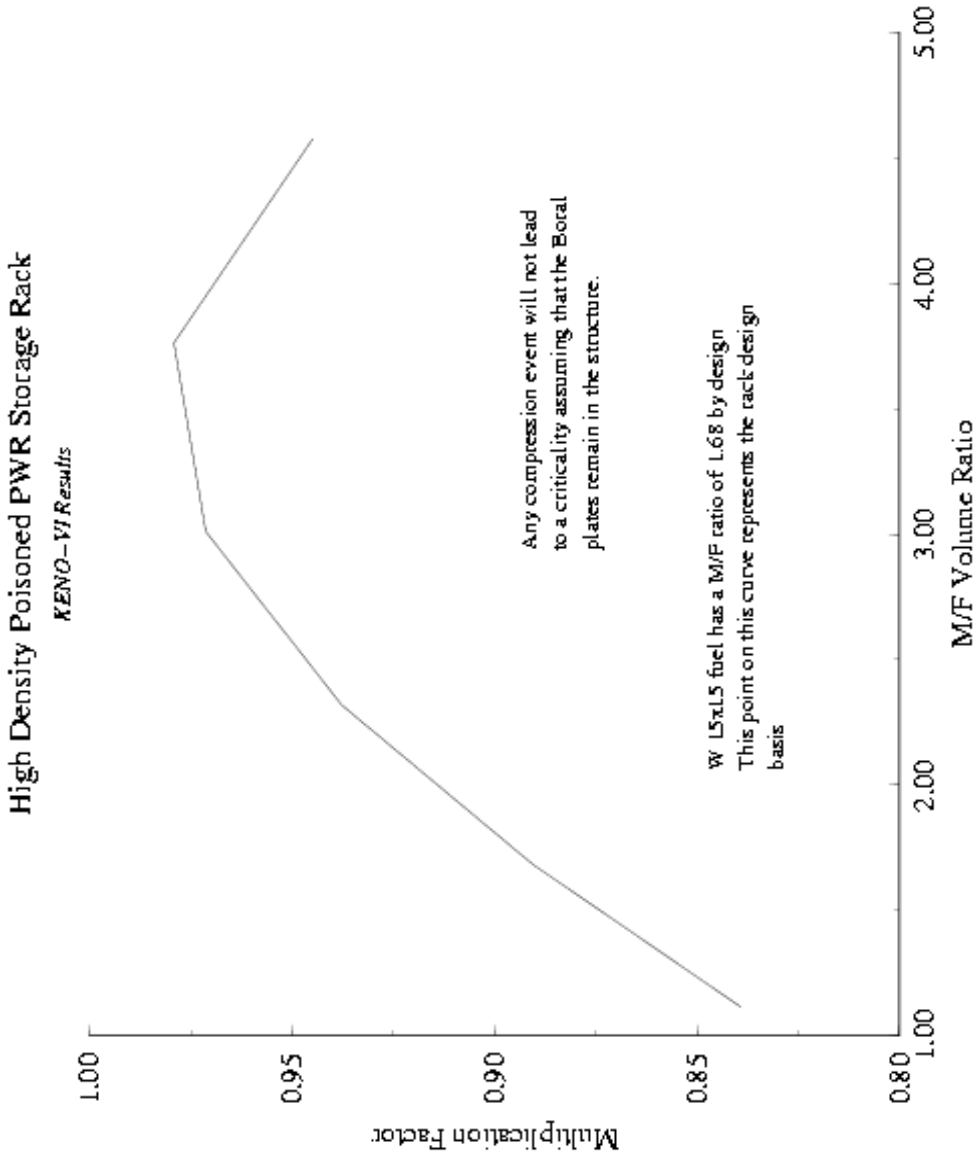
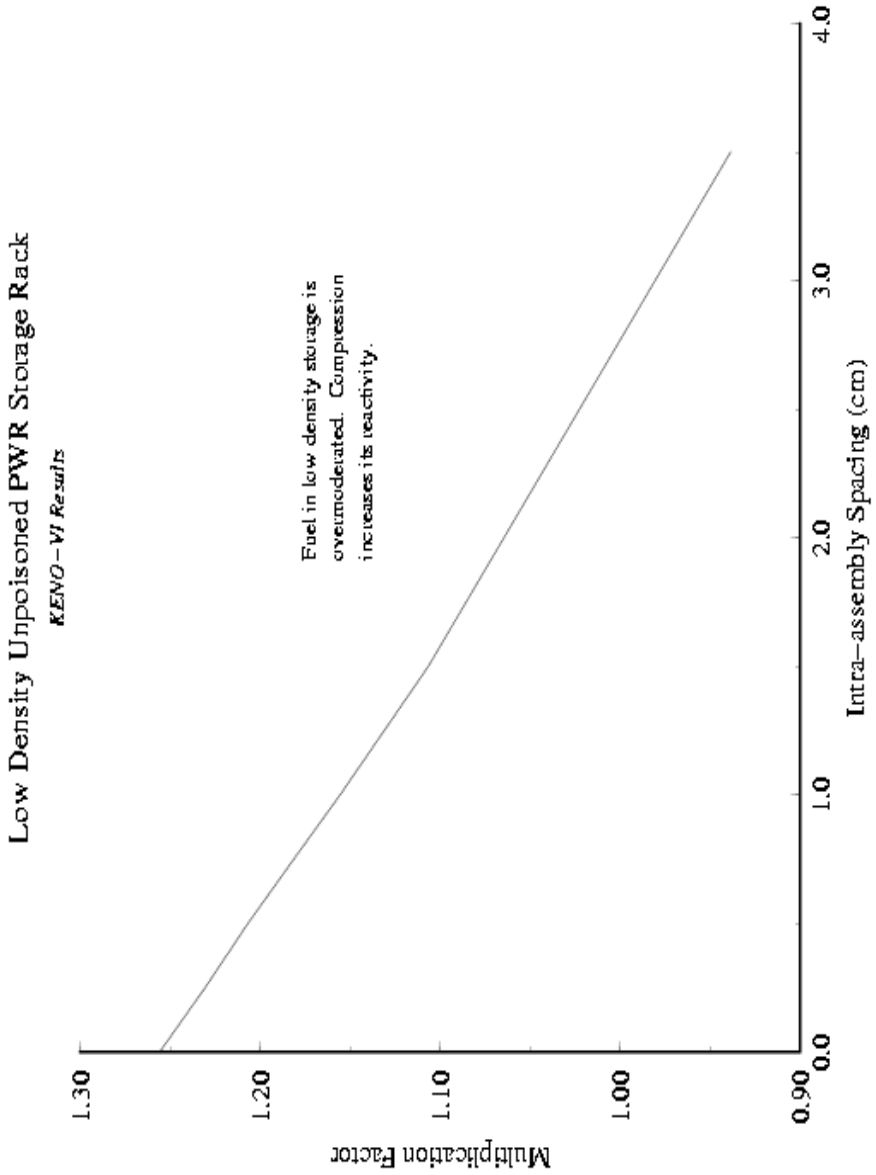


Figure 2 PWR Low Density Storage Rack Eigenvalue Following Compressive/Expansion Events



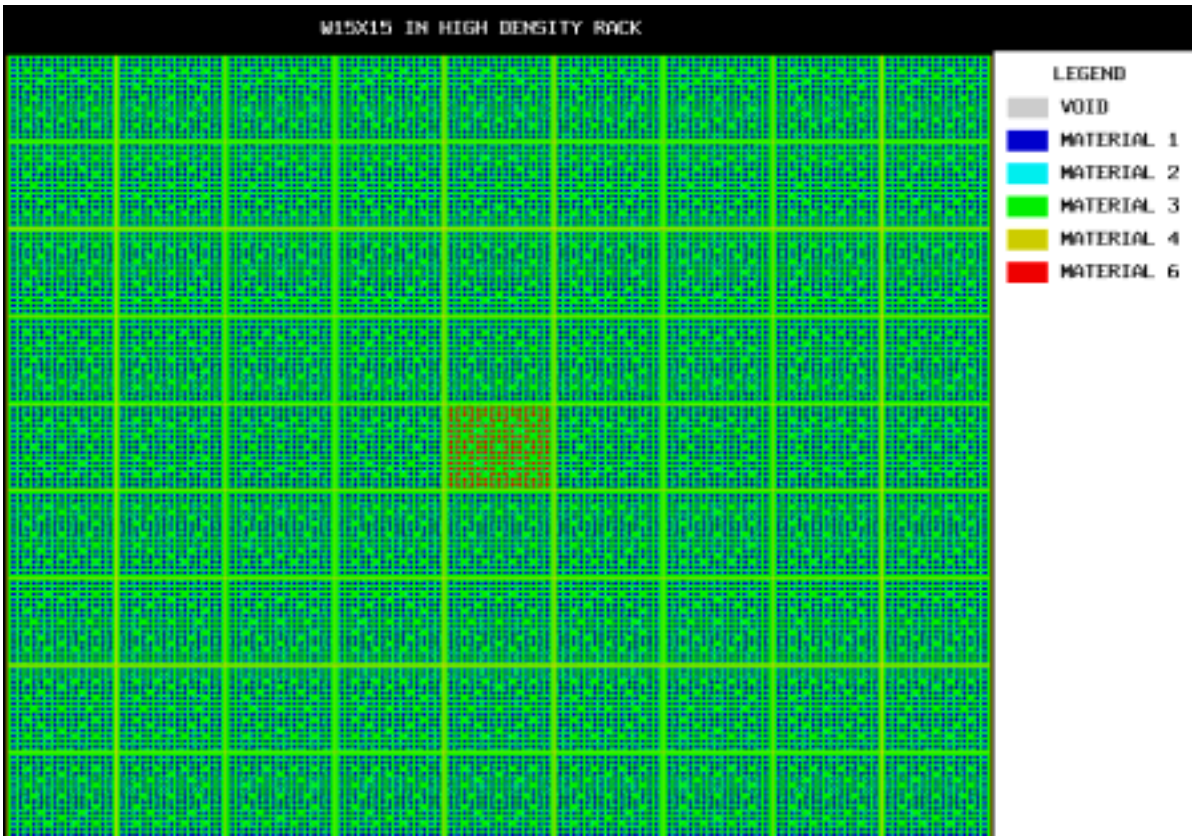


Figure 3 Sample Geometry Assuming 4 Assembly Spacing Between Most Reactive Assembly

Figure 4 BWR High Density Storage Rack Eigenvalue Following Compressive/Expansion Events

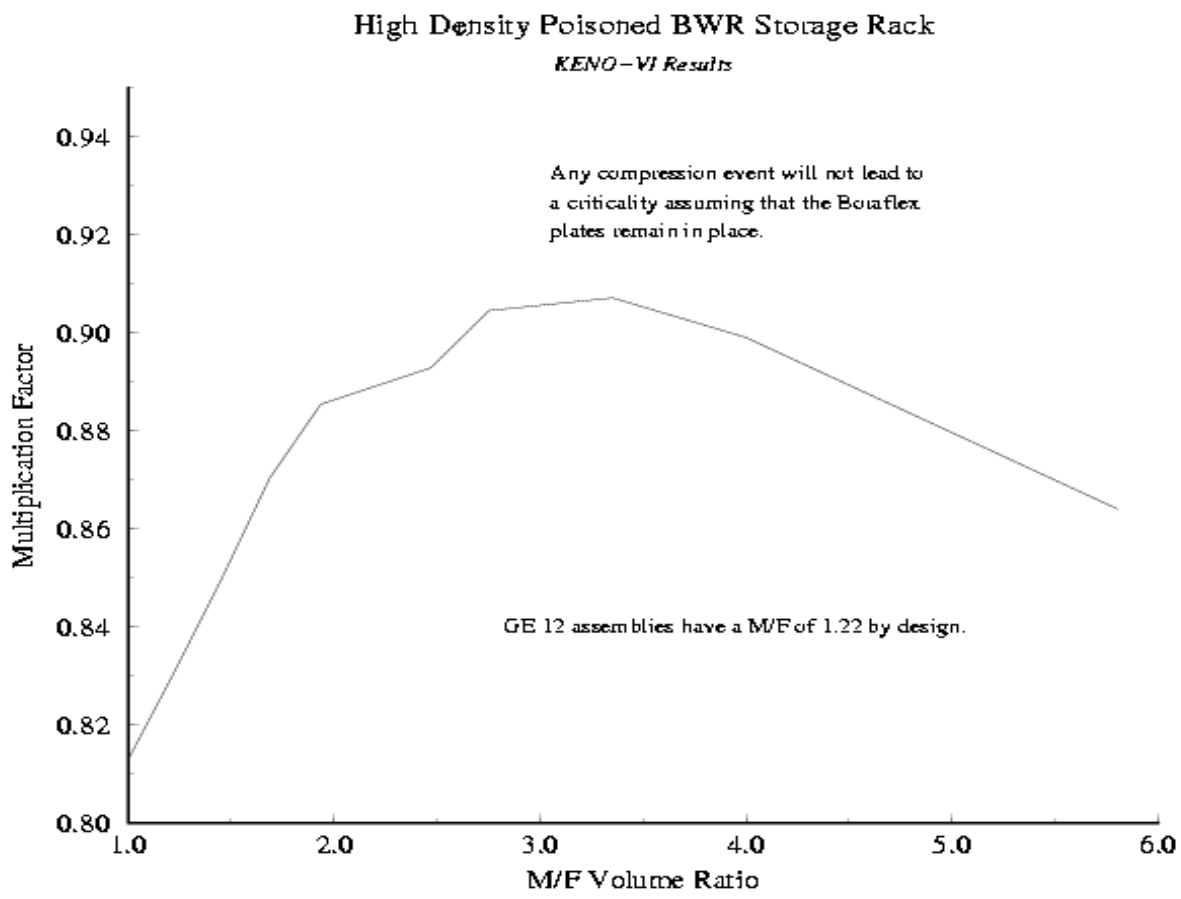
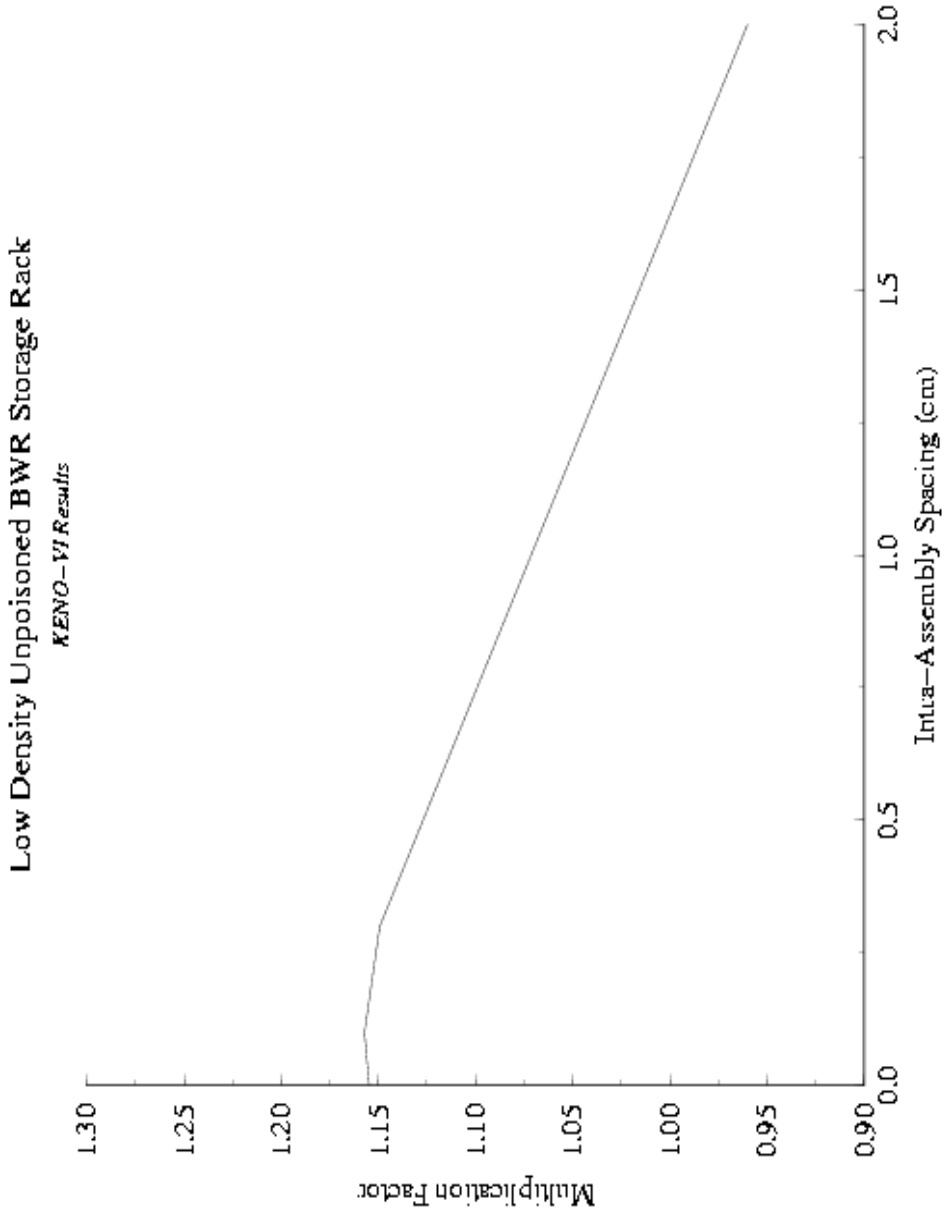

High Density Poisoned BWR Storage Rack
KENO-VI Results
Any compression event will not lead to a criticality assuming that the Boraflex plates remain in place.
GE 12 assemblies have a M/F of 1.22 by design.
Multiplication Factor
0.94
0.92
0.90
0.88
0.86
0.84
0.82
0.80
1.0
2.0
3.0
4.0
5.0
6.0
M/F Volume Ratio

Figure 5 BWR Low Density Storage Rack Eigenvalue Following Compressive/Expansion Events



APPENDIX 4
CONSEQUENCE ASSESSMENT FROM ZIRCONIUM FIRE

Spent fuel pool (SFP) accidents involving a sustained loss of coolant have the potential for leading to significant fuel heat up and resultant release of fission products to the environment. Such an accident would involve decay heat raising the fuel temperature to the point of exothermic cladding oxidation, which would cause additional temperature escalation to the point of fission product release. However, because fuel in an SFP has a lower decay power than fuel in the reactor vessel of an operating reactor, it will take much longer for the fuel in the SFP to heat up to the point of releasing radionuclides than in some reactor accidents.

Earlier analyses in NUREG/CR-4982 (Ref. 1) and NUREG/CR-6451 (Ref. 2) have assessed the frequency and consequences of SFP accidents. These analyses included a limited evaluation of offsite consequences of a severe SFP accident. NUREG/CR-4982 results included consequence estimates for the societal dose for accidents occurring 30 days and 90 days after the last discharge of spent fuel into the SFP. NUREG/CR-6451 results included consequence estimates for societal dose, prompt fatalities, and cancer fatalities for accidents occurring 12 days after the last discharge of spent fuel. The work described in this appendix extends the earlier analyses by calculating offsite consequences for a severe SFP accident occurring up to one year after discharge of the last load of spent fuel, and supplements that earlier analysis with additional sensitivity studies, including varying evacuation assumptions as well as other modeling assumptions. The primary objective of this analysis was to assess the effect of extended storage in an SFP, and the resulting radioactive decay, on offsite consequences. However, as part of this work, the sensitivity to a variety of other parameters was also evaluated.

The current analysis used the MACCS code (Ref. 3)(version 2) to estimate offsite consequences for a severe SFP accident. Major input parameters for MACCS include radionuclide inventories, radionuclide release fractions, evacuation and relocation criteria, and population density. The specification of values for these input parameters for a severe SFP accident is discussed below.

Radionuclide Inventories

As discussed above, the current analysis was undertaken to assess the magnitude of the decrease in offsite consequences that could result from up to a year of decay in the SFP. To perform this work, it was necessary to have radionuclide inventories in the SFP for a decommissioned reactor at times up to 1 year after final shutdown. The inventories in the NUREG/CR-6451 analysis have not been retrievable, so those inventories could not be used. NUREG/CR-4982 contains SFP inventories for two operating reactors, a BWR (Millstone 1) and a PWR (Ginna). Since the staff had radionuclide inventory data for a small BWR (Millstone 1), the staff adjusted the radionuclide inventory of Millstone 1 to represent a large BWR with a thermal power of 3441 megawatts. These SFP inventories for Millstone 1 are given in Table 4.1 of NUREG/CR-4982 and are reproduced in Table A4-1 below. Two adjustments were then made to the Table A4-1 inventories. The first adjustment was to multiply the inventories by a factor of 1.7, because the thermal power of the large BWR is 1.7 times higher than that of Millstone 1. The second adjustment, described in the next two

paragraphs, was needed because NUREG/CR-4982 was for an operating reactor and this analysis is for a decommissioned reactor.

Because NUREG/CR-4982 was a study of SFP risk for an operating reactor, the Millstone 1 SFP inventories shown in Table A4-1 were for the fuel that was discharged during the 11th refueling outage (about 1/3 of the core) and the previous 10 refueling outages. The inventories shown in Table A4-1 did not include the fuel which remained in the vessel (about 2/3 of the core) that was used further when the reactor was restarted after the outage. Because the current study is for a decommissioned reactor, the inventories shown in Table A4-1 were adjusted by adding the inventories in the remaining 2/3 of the core. This remaining 2/3 of the core is expected to contain a significant amount of short half-life radionuclides in comparison with the 11 batches of spent fuel in the SFP.

The radionuclide inventories in the remaining 2/3 of the core were derived from the data in Tables A.5 and A.6 in NUREG/CR-4982. Tables A.5 and A.6 give inventory data for the 11th refueling outage. Table A.5 gives the inventories for the entire core at the time of reactor shutdown. Table A.6 gives the inventories (at 30 days after shutdown) for the batch of fuel discharged during the outage. First, the inventories for the entire core at the time of shutdown were reduced by radioactive decay to give the inventories for the entire core at 30 days after shutdown. Then, the inventories (at 30 days after shutdown) for the batch of fuel discharged were subtracted to give the inventories for the remaining 2/3 of the core at 30 days after shutdown. Inventories for the remaining 2/3 of the core at 90 days and 1 year after shutdown were subsequently calculated by reducing the 30-day inventories by radioactive decay.

Table A4-1 Radionuclide Inventories in the Millstone 1 Spent Fuel Pool

Radionuclide	Half-Life	Spent Fuel Pool Inventory (Ci)		
		30 days after last discharge	90 days after last discharge	1 year after last discharge
Co-58	70.9d	2.29E4	1.26E4	8.54E2
Co-60	5.3y	3.72E5	3.15E5	2.85E5
Kr-85	10.8y	1.41E6	1.39E6	1.33E6
Rb-86	18.7d	1.01E4	1.05E3	3.84E-2
Sr-89	50.5d	8.39E6	3.63E6	8.33E4
Sr-90	28.8y	1.42E7	1.42E7	1.39E7
Y-90	28.8y	1.43E7	1.42E7	1.39E7
Y-91	58.5d	1.18E7	5.75E6	2.21E5
Zr-95	64.0d	1.94E7	1.00E7	5.10E5
Nb-95	64.0d	2.54E7	1.70E7	1.11E6
Mo-99	2.7d	1.49E4	3.12E-3	0
Tc-99m	2.7d	1.43E4	3.01E-3	0
Ru-103	37.3d	1.53E7	5.21E6	4.07E4
Ru-106	1.0y	1.72E7	1.53E7	9.13E6
Sb-127	3.8d	8.21E3	1.39E-1	0
Te-127	109d	2.21E5	1.45E5	2.52E4
Te-127m	109d	2.18E5	1.48E5	2.57E4
Te-129	33.6d	2.74E5	7.79E4	2.68E2
Te-129m	33.6d	4.21E5	1.20E5	4.12E2
Te-132	3.2d	3.74E4	8.64E-2	0
I-131	8.0d	1.22E6	6.35E3	0
I-132	3.2d	3.85E4	8.90E-2	0
Xe-133	5.2d	7.29E5	2.30E2	0
Cs-134	2.1y	7.90E6	7.47E6	5.80E6

Cs-136	13.2d	2.05E5	8.13E3	3.91E-3
Cs-137	30.0y	2.02E7	2.01E7	1.97E7
Ba-140	12.8d	5.19E6	1.90E5	6.41E-2
La-140	12.8d	5.97E6	2.19E5	7.37E-2
Ce-141	32.5d	1.32E7	3.61E6	1.03E4
Ce-144	284.6d	2.64E7	2.27E7	1.16E7
Pr-143	13.6d	5.44E6	2.41E5	1.90E-1
Nd-147	11.0d	1.54E6	3.36E4	1.10E-3
Np-239	2.4d	5.59E4	2.88E3	2.88E3
Pu-238	87.7y	4.51E5	4.53E5	4.54E5
Pu-239	24100y	8.89E4	8.89E4	8.89E4
Pu-240	6560y	1.30E5	1.30E5	1.30E5
Pu-241	14.4y	2.29E7	2.27E7	2.19E7
Am-241	432.7y	2.88E5	2.94E5	3.21E5
Cm-242	162.8d	1.45E6	1.12E6	3.50E5
Cm-244	18.1y	2.27E5	2.25E5	2.19E5

MACCS has a default list of 60 radionuclides that are important for offsite consequences for reactor accidents. NUREG/CR-4982 contains inventories for 40 of these 60 radionuclides. Of these 40 radionuclides, 27 have half-lives from 2.4 days to a year and 13 have half-lives of a year or greater as shown in Table A4-1. The half-lives of the remaining 20 radionuclides range from 53 minutes to 1.5 days as shown in Table A4-2. Because the largest half-life of these 20 radionuclides is 1.5 days, omitting these 20 radionuclides from the initial inventories used in the MACCS analysis should not affect doses from releases occurring after a number of days of decay.

Table A4-2 Half-lives of MACCS Radionuclides Whose Inventories Were Not in NUREG/CR-4982

Radionuclide	Half-Life (days)
Kr-85m	.19
Kr-87	.05
Kr-88	.12
Sr-91	.40
Sr-92	.11
Y-92	.15
Y-93	.42
Zr-97	.70
Ru-105	.19
Rh-105	1.48
Sb-129	.18
Te-131m	1.25
I-133	.87
I-134	.04
I-135	.27
Xe-135	.38
Ba-139	.06
La-141	.16
La-142	.07
Ce-143	1.38

Release Fractions

NUREG/CR-4982 also provided the fission product release fractions assumed for a severe SFP accident. These fission product release fractions are shown in Table A4-3. NUREG/CR-6451 provided an updated estimate of fission product release fractions. The release fractions in NUREG/CR-6451 (also shown in Table A4-3) are the same as those in NUREG/CR-4982, with the exception of lanthanum and cerium. NUREG/CR-6451 stated that the release fraction

of lanthanum and cerium should be increased from $1x10^{-6}$ in NUREG/CR-4982 to $6x10^{-6}$, because fuel fines could be released offsite from fuel with high burnup. While the staff believes that it is unlikely that fuel fines would be released offsite in any substantial amount, a sensitivity was performed using a release fraction of $6x10^{-6}$ for lanthanum and cerium to determine whether such an increase could even impact offsite consequences.

Table A4-3 Release Fractions for a Severe Spent Fuel Pool Accident

Radionuclide Group	Release Fractions	
	NUREG/CR-4982	NUREG/CR-6451
noble gases	1	1
iodine	1	1
cesium	1	1
tellurium	$2x10^{-2}$	$2x10^{-2}$
strontium	$2x10^{-3}$	$2x10^{-3}$
ruthenium	$2x10^{-5}$	$2x10^{-5}$
lanthanum	$1x10^{-6}$	$6x10^{-6}$
cerium	$1x10^{-6}$	$6x10^{-6}$
barium	$2x10^{-3}$	$2x10^{-3}$

Modeling of Emergency Response Actions and Other Areas

Modeling of emergency response actions was essentially the same as that used for Surry in NUREG-1150. The timing of events is given in Table A4-4. Evacuation begins exactly two hours after emergency response officials receive notification to take protective measures. This results in the evacuation beginning approximately .8 hours after the offsite release ends. Only people within 10 miles of the SFP evacuate, and, of those people, .5% do not evacuate. Details of the evacuation modeling are given in Table A4-5.

People outside of 10 miles are relocated to uncontaminated areas after a specified period of time depending on the dose they are projected to receive in the first week. There are two relocation criteria. The first criterion is that, if the dose to an individual is projected to be greater that 50 rem in one week, then the individual is relocated outside of the affected area after 12 hours. The second criterion is that, if the dose to an individual is projected to be greater that 25 rem in one week, then the individual is relocated outside of the affected area after 24 hours.

Table A4-4 Timing of Events

Event	Time (sec)	Time (hour)
notification given to offsite emergency response officials	0	0
start time of offsite release	2400	.7
end time of offsite release	4200	1.2
evacuation begins	7200	2.0

Table A4-5 Evacuation Modeling

Parameter	Value
size of evacuation zone	10 miles
sheltering in evacuation zone	no sheltering
evacuation direction	radially outward
evacuation speed	4 miles/hr
other	after evacuee reaches 20 miles from fuel pool, no further exposure is calculated

After the first week, the pre-accident population in each sector (including the evacuation zone) is assumed to be present unless the dose to an individual in a sector will be greater than 4 rem over a period of 5 years. If the dose to an individual in a sector is greater than 4 rem over a period of 5 years, then the population in that sector is relocated. Dose and cost criteria are used to determine when the relocated population returns to a sector. The dose criterion is that the relocated population is returned at a time when it is estimated that an individual's dose will not exceed 4 rem over the next 5 years. The actual population dose is calculated for exposure for the next 300 years following the population's return.

Offsite Consequence Results

MACCS calculations for a decommissioned reactor for accidents occurring 30 days, 90 days, and 1 year after final shutdown were performed to assess the magnitude of the decrease in the offsite consequences resulting from extended decay before the release. These calculations were performed for a Base Case along with a number of sensitivity cases to evaluate the impact of alternative modeling. These cases are summarized in Table A4-6. The results of these calculations are discussed below.

Table A4-6 Cases Examined Using the MACCS2 Consequence Code

Case	Population Distribution	Radionuclide Inventory	Evacuation Start Time	La/Ce Release Fraction	Evacuation Percentage
Base Case	Surry	11 batches plus rest of last core	1.4 hours after release begins	$1x10^{-6}$	99.5%
1	Surry	11 batches plus rest of last core	1.4 hours after release begins	$1x10^{-6}$	95%
2	Surry	11 batches	1.4 hours after release begins	$1x10^{-6}$	95%
3	100 people/mi^2	11 batches	1.4 hours after release begins	$1x10^{-6}$	95%
4	100 people/mi^2	11 batches plus rest of last core	1.4 hours after release begins	$1x10^{-6}$	95%
5	100 people/mi^2	11 batches plus rest of last core	3 hours before release begins	$1x10^{-6}$	95%
6	100 people/mi^2	11 batches plus rest of last core	3 hours before release begins	$6x10^{-6}$	95%
7	100 people/mi^2	11 batches plus rest of last core	3 hours before release begins	$1x10^{-6}$	99.5%

The Base Case was intended to model the offsite consequences for a severe SFP accident for a decommissioned reactor. To accomplish this, the Base Case used the Millstone 1 inventories from NUREG/CR-4982 adjusted for reactor power and the rest of the last core as discussed above. Accordingly, the Base Case used the Millstone 1 radionuclide inventories for the fuel from the first 11 refueling outages (1649 assemblies) together with the rest of the last core (413 assemblies). Because the Millstone 1 core design has 580 assemblies, the amount of fuel assumed to be in the SFP is equivalent to about 3.5 cores.

Other modeling in the Base Case, such as the population distribution, the evacuation percentage of 99.5% of the population, and the meteorology, are from the NUREG-1150 consequence assessment model for Surry. The results of the Base Case are shown in Table A4-7.

Table A4-7 Mean Consequences for the Base Case

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
30 days	0-100	1.75	47,700	2,460
	0-500	1.75	571,000	25,800
90 days	0-100	1.49	46,300	2,390
	0-500	1.49	586,000	26,400
1 year	0-100	1.01	45,400	2,320
	0-500	1.01	595,000	26,800

Table A4-7 shows the offsite consequences for a severe SFP accident at 30 days, 90 days, and 1 year following final reactor shutdown. The decay times for fuel transferred to the pool during the 11th refueling outage were 30 days, 90 days, and 1 year, respectively. The decay times for spent fuel in the pool from earlier refueling outages were much longer and were accounted for in the inventories used in this analysis.

These results in Table A4-7 show virtually no change in long-term offsite consequences (i.e., societal dose and cancer fatalities) as a function of decay time, because they are controlled by inventories of radionuclides with long half-lives and relocation assumptions. However, these results also show about a factor-of-two reduction in the short-term consequences (i.e., prompt fatalities) from 30 days to 1 year of decay. (All of the prompt fatalities occur within 10 miles of the site.) As a rough check on the prompt fatality results, the change in decay power was evaluated for an operating reactor shut down for 30 days and for 1 year. The decay power decreased by about a factor of three. This is consistent with a factor-of-two decrease in prompt fatalities. The factor-of-three decrease in decay power by radioactive decay will also increase the time it takes to heat up the spent fuel, which provides additional time to take action to mitigate the accident.

The results of Case 1, which used a lower evacuation percentage than the Base Case, are identical to the results of the Base Case shown in Table A4-7. Case 1 used an evacuation percentage of 95%, while the Base Case used an evacuation percentage of 99.5%. Although it might be expected to see an increase in prompt fatalities from reducing the evacuation percentage, no such increase was observed. This is because of the assumption that the release ends at 1.2 hours, while the evacuation does not begin until 2 hours.

Case 2, shown in Table A4-8, used a radionuclide inventory that consisted of 11 batches of spent fuel, but did not include the remaining two-thirds of the core in the vessel. This was done to facilitate comparison of the consequence results with the results of the analyses in

NUREG/CR-4982 and NUREG/CR-6451. This also allowed examination of the relative contribution of the short-lived radionuclides to consequences. Because the length of time between refueling outages is on the order of a year, short-lived radionuclides in the SFP will decay away between refueling outages. As a result, all of the short-lived radionuclides are in the core at the start of the 11th refueling outage for Millstone 1. When Millstone 1 discharged one-third of its core at the beginning of the 11th refueling outage, two-thirds of its short-lived isotopes remained in the vessel. Therefore, use of 11 batches of fuel in Case 2 without the remaining two-thirds of the core represents about a factor-of-three reduction in short-lived radionuclides in the SFP from what was modeled in Case 1. As shown in Table A4-8, use of 11 batches of spent fuel without the remaining two-thirds of the core resulted in a factor-of-two reduction in the prompt fatalities and no change in the societal dose and cancer fatalities. This factor-of-two reduction in prompt fatalities is consistent with the factor-of-three reduction in the inventories of the short-lived radionuclides when the remaining two-thirds of the core in the vessel is not included in the consequence calculation.

Table A4-8 Mean consequences for Case 2

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
30 days	0-100	.89	44,900	2,280
	0-500	.89	557,000	25,100
90 days	0-100	.78	44,500	2,250
	0-500	.78	554,000	25,000
1 year	0-100	.53	43,400	2,180
	0-500	.53	567,000	25,500

The results of the next case, Case 3, are shown in Table A4-9. This case used a generic population distribution of 100 persons/mile^2 (uniform). This was done to facilitate comparison of the consequence results with the results of the analyses in NUREG/CR-4982 and NUREG/CR-6451. Use of a uniform population density of 100 persons/mile^2 results in an order-of-magnitude increase in prompt fatalities and relatively small changes in the societal dose and cancer fatalities.

Table A4-9 Mean Consequences for Case 3

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
30 days	0-100	11.7	50,100	2,440
	0-500	11.7	449,000	20,300
90 days	0-100	10.6	50,300	2,460
	0-500	10.6	447,000	20,200
1 year	0-100	8.19	49,000	2,380
	0-500	8.19	453,000	20,500

The results of the next case, Case 4, are shown in Table A4-10. This case includes the remaining two-thirds of the core in the vessel. This was done to facilitate comparison of the consequence results with the results of the analysis in NUREG/CR-6451. As discussed above in the comparison of Case 1 with Case 2, this increases the prompt fatalities by about a factor of two with no change in the societal dose or cancer fatalities.

Table A4-10 Mean Consequences for Case 4

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
30 days	0-100	18.3	53,500	2,610
	0-500	18.3	454,000	20,600
90 days	0-100	16.3	52,100	2,560
	0-500	16.3	465,000	21,100
1 year	0-100	12.7	50,900	2,490
	0-500	12.7	477,000	21,600

Heat up of fuel in an SFP following a complete loss of coolant takes much longer than in some reactor accidents. Therefore, it may be possible to begin evacuating before the release begins. Case 5, which uses an evacuation start time of three hours before the release begins, was performed to assess the impact of early evacuation. As shown in Table A4-11, prompt fatalities were significantly reduced and societal dose and cancer fatalities remained unchanged.

Table A4-11 Mean Consequences for Case 5

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
30 days	0-100	.96	48,300	2,260
	0-500	.96	449,000	20,200
90 days	0-100	.83	47,500	2,220
	0-500	.83	460,000	20,700
1 year	0-100	.67	46,700	2,180
	0-500	.67	473,000	21,300

As noted above, NUREG/CR-6451 estimated the release of lanthanum and cerium to be a factor of six higher than that originally estimated in NUREG/CR-4982. Case 6 was performed to assess the potential impact of that higher release. The Case 6 consequence results were identical to those of Case 5 shown in Table A4-11. Therefore, even it were possible for fuel fines to be released offsite, there would be no change in offsite consequences as a result.

The final case, Case 7 was performed to examine the impact of a 99.5% evacuation for a case with evacuation before the release begins. This sensitivity (see Table A4-12) showed an order of magnitude decrease in the prompt fatalities. Again, as expected, no change in the societal dose or cancer fatalities was observed.

Table A4-12 Mean Consequences for Case 7

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
30 days	0-100	.096	48,100	2,250
	0-500	.096	449,000	20,200
90 days	0-100	.083	47,400	2,210
	0-500	.083	460,000	20,700
1 year	0-100	.067	46,600	2,170
	0-500	.067	473,000	21,300

Comparison with Earlier Consequence Analyses

As a check on the above calculations and to provide additional insight into the consequence analysis for severe SFP accidents, the above calculations were compared to the consequence results reported in NUREG/CR-4982 and NUREG/CR-6451. Table A4-13 shows the analysis assumptions used for BWRs in these earlier reports together with those of Cases 3 and 4 of

the current analysis.

NUREG/CR-4982 results included consequence estimates for societal dose for an operating reactor for severe SFP accidents occurring 30 days and 90 days after the last discharge of spent fuel into the pool. The Case 3 results were compared against the NUREG/CR-4982 results, because they use the same population density (100 persons/mile^2) and 11 batches of spent fuel in the pool. However, one difference is that Case 3 uses a radionuclide inventory that is a factor of 1.7 higher than NUREG/CR-4982 to reflect the relative power levels of a large BWR and Millstone 1. Therefore, Case 3 was rerun with the radionuclide inventory of NUREG/CR-4982. As shown in Table A4-14, the Case 3 rerun results generally compared well with the NUREG/CR-4982 results.

Table A4-13 Comparison of Analysis Assumptions

Parameter	NUREG/CR-4982 (BWR)	NUREG/CR-6451 (BWR)	Case 3	Case 4
population density (persons/ mile^2)	100	0-30 mi: 1000 30-50 mi: 2300 (city of 10 million people, 280 outside of city) 50-500 mi: 200	100	100
meteorology	uniform wind rose, average weather conditions	representative for continental U.S.	Surry	Surry
radionuclide inventory	11 batches of spent fuel	full fuel pool after decommissioning (3300 assemblies)	11 batches of spent fuel, increased by x1.7	11 batches of spent fuel plus last of rest core, increased by x1.7
exclusion area	not reported	.4 mi	none	none
emergency response	relocation at one day if projected doses exceed 25 rem	relocation at one day if projected doses exceed 25 rem	NUREG-1150 Surry analysis (see above)	NUREG-1150 Surry analysis (see above)

Table A4-14 Comparison with NUREG/CR-4982 Results

Decay Time in Spent Fuel Pool	Distance (miles)	Societal Dose (person-Sv)		
		NUREG/CR-4982	Case 3	Case 3 Rerun
30 days	0-50	26,000	20,900	16,700
	0-500	710,000	449,000	379,000
90 days	0-50	26,000	20,400	16,500

The NUREG/CR-6451 results included consequence estimates for societal dose, cancer fatalities, and prompt fatalities for a decommissioned reactor for a severe SFP accident occurring 12 days after the final shutdown. The Case 4 results for 30 days after final shutdown were compared against the NUREG/CR-6451 results, because (1) they included the entire last core in the SFP and (2) Case 4 had a uniform population density which could be easily adjusted to approximate that in NUREG/CR-6451. Differences between Case 4 and

NUREG/CR-6451 included the population density, the amount of spent fuel in the pool, and the exclusion area size. To provide a more consistent basis to compare the NUREG/CR-6451 results with the Case 4 results, Case 4 was rerun using population densities, an amount of spent fuel, and an exclusion area size similar to NUREG/CR-6451.

The average population densities in the NUREG/CR-6451 analysis were about 1800 persons/mile^2 within 50 miles and 215 persons/mile^2 within 500 miles. Also, NUREG/CR-6451 used an inventory with substantially higher quantities of long-lived radionuclides than the 11 batches of spent fuel in NUREG/CR-4982. NUREG/CR-6451 stated that it used an inventory of Cs-137 (30 year half-life) that was three times greater than that used in NUREG/CR-4982. To provide a more consistent basis to compare with NUREG/CR-6451 long-term consequences, Case 4 was rerun using uniform population densities of 1800 persons/mile^2 within 50 miles and 215 persons/mile^2 outside of 50 miles and a power correction factor of 3 instead of 1.7. As shown in Table A4-15, Case 4 rerun is in generally good agreement with NUREG/CR-6451. These calculations indicate a very strong dependence of long-term consequences on population density. Remaining differences in long-term consequences may be because of remaining differences in population density and inventories as well as differences in meteorology and emergency response.

Table A4-15 Comparison with NUREG/CR-6451 Results (long-term consequences)

Dist. (miles)	Societal Dose (person-Sv)			Cancer Fatalities		
	NUREG/ CR-6451	Case 4	Case 4 Rerun	NUREG/ CR-6451	Case 4	Case 4 Rerun
0-50	750,000	23,600	389,000	31,900	1,260	20,800
0-500	3,270,000	454,000	1,330,000	138,000	20,600	44,900

To provide a more consistent basis to compare with NUREG/CR-6451 short-term consequences, Case 4 was again rerun, this time using a uniform population density of 1000 persons/mile^2 and an exclusion area of .32 miles. As shown in Table A4-16, Case 4 rerun is in generally good agreement with NUREG/CR-6451. Overall, these calculations indicate a very strong dependence of short-term consequences on population density and a small dependence (about 10% change in prompt fatality results) on exclusion area size. Remaining differences in short-term consequences may be because of remaining differences in population density and inventories as well as differences in meteorology and emergency response.

Table A4-16 Comparison with NUREG/CR-6451 Results (short-term consequences)

Dist. (miles)	Prompt Fatalities		
	NUREG/CR-6451	Case 4	Case 4 Rerun
0-50	74	18.3	168
0-500	101	18.3	168

Effect of Cesium

Cesium is volatile under severe accident conditions and was previously estimated to be completely released from fuel under these conditions. Also, the half-lives of the cesium isotopes are 2 years for cesium-134, 13 days for cesium-136, and 30 years for cesium-137. Therefore, we performed additional sensitivity calculations on the Base Case to evaluate the importance of cesium to better understand why the consequence reduction from a year of decay was not greater. The results of our calculations are shown in Table A4-17. As shown in this table, we found that the cesium isotopes with their relatively long half-lives were responsible for limiting the reduction in offsite consequences.

Table A4-17 Mean Consequences for the Base Case with and Without Cesium

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
1 year	0-100	1.01	45,400	2,320
1 year (without cesium)	0-100	0.00	1,460	42

Conclusion

The primary objective of this evaluation was to assess the effect of extended storage in an SFP, and the resulting radioactive decay, on offsite consequences of a severe SFP accident at a decommissioned reactor. This evaluation was performed in support of the generic evaluation of SFP risk that is being performed to support related risk-informed requirements for decommissioned reactors. This evaluation showed about a factor-of-two reduction in prompt fatalities if the accident occurs after 1 year instead of after 30 days. Sensitivity studies showed that cesium with its long half-life (30 years) is responsible for limiting the consequence reduction. For the population within 100 miles of the site, 97 percent of the societal dose was from cesium. Also, this evaluation showed that beginning evacuation three hours before the release begins reduces prompt fatalities by more than an order of magnitude.

References:

1. NUREG/CR-4982, Severe Accidents in Spent Fuel Pools in Support of Generic Issue 82, July 1987.

2. NUREG/CR-6451, A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants, August 1997.

3. NUREG/CR-6613, Code Manual for MACCS2, May 1998.

APPENDIX 4A

Memo to Gary M. Holahan from Farouk Eltawila dated August 25, 2000, re:
RISK-INFORMED REQUIREMENTS FOR DECOMMISSIONING

August 25, 2000

MEMORANDUM TO: Gary M. Holahan, Director
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

FROM: Farouk Eltawila, Acting Director
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

SUBJECT: RISK-INFORMED REQUIREMENTS FOR DECOMMISSIONING

As part of its effort to develop generic, risk-informed requirements for decommissioning, NRR requested (Reference 1) an evaluation of the offsite radiological consequences of beyond-design-basis spent fuel pool accidents. In response to that user need, we completed an in-house analysis (Reference 2) that concluded the following:

- The short-term consequences (i.e., early fatalities) decreased by a factor of two when the fission product inventory decreased from that for 30 days to that for one year after final shutdown.

- At one year after final shutdown, the short-term consequences decreased by up to a factor of 100 as a result of early evacuation. Early evacuation is likely after one year, because of the decreased decay heat level and the number of hours required for the fuel with the highest decay power to heat up to the point of releasing fission products.

- The long-term consequences (i.e., cancer fatalities and societal dose) were unaffected by the additional decay and early evacuation.

Although the reductions in the short-term consequences were significant, emergency planning requirements could not be relaxed solely on the basis of these reductions. NRR also used our consequence evaluation in the Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, February 2000, as an absolute measure of spent fuel pool accident consequences and concluded that the consequences were generally comparable to those of reactor accidents.

Subsequently, the ACRS raised issues with the source term and plume modeling associated with spent fuel pool accidents. In particular, the ACRS believed that the ruthenium and fuel fines releases and plume spreading were too low. To address these issues, we completed a series of sensitivity studies and concluded:

- With the exception of the ruthenium release fraction, the parameters varied did not sufficiently impact the results, nor change the conclusion that the consequences were generally comparable to those of reactor accidents.

- Increasing the ruthenium release fraction from that for a non-volatile ($2x10^{-5}$) to that for a volatile (.75) resulted in a large increase in both short-term and long-term consequences due to ruthenium's high dose per curie inhaled. However, consequence increases from ruthenium were demonstrated to be largely offset by early evacuation.

- Although using updated values for plume-spreading model parameters resulted in up to a 60% increase in long-term consequences, similar increases are expected when these updated values are used to calculate reactor accident consequences. Using updated values also resulted in up to a factor-of-15 decrease in short-term consequences.

The results of these sensitivity studies are described in Attachment 1, which was written, at NRR request, to be incorporated into the final technical study as an appendix. The range of consequences for a beyond-design-basis spent fuel pool accident occurring one year after final shutdown is shown below for early evacuation. This range reflects the uncertainty in the ruthenium and fuel fines release fractions. NRR also requested our assistance in responding to the public comments on the Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants. Our responses to these comments in the areas of offsite radiological consequences and emergency response are provided in Attachment 2.

End of Range	Consequences within 100 Miles (Surry population density)		
	Early Fatalities	Societal Dose (rem)	Cancer Fatalities
Lower	.005	$4x10^6$	2,000
Upper	.5	$8x10^6$	7,000

Recently, NRR requested additional consequence calculations using fission product inventories at 30 and 90 days and two, five, and ten years after final shutdown to provide additional insight into the effect of reductions in inventory available for release. We are currently performing these calculations and expect to provide the results shortly.

References:
1. Memorandum from G. Holahan to T. King dated March 26, 1999
2. Memorandum from A. Thadani to S. Collins dated November 12, 1999

Attachments:
1. Effect of Source Term and Plume-Related Parameters on Consequences
2. Response to Public Comments on the Consequence Assessment

cc: T. Collins
R. Barrett
J. Hannon
J. Wermiel

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Appendix 4A Effect of Source Term and Plume-Related Parameters on Consequences

Introduction

Appendix 4 documents the staff's evaluation of the offsite consequences of a spent fuel pool accident involving a sustained loss of coolant, leading to a significant fuel heatup and resultant release of fission products to the environment. The objectives of the consequence evaluation were (1) to assess the effect of one year of decay and (2) to assess the effect of early versus late evacuation because spent fuel pool accidents are slowly evolving accidents. The staff's evaluation was an extension of an earlier study performed by Brookhaven National Laboratory (BNL) for spent fuel pools at operating reactors, which assessed consequences using inventories for 30 days after shutdown.[1]

To perform the evaluation documented in Appendix 4, the staff used the MACCS code (MELCOR Accident Consequence Code System)[2] with fission product inventories for 30 days and 1 year after final shutdown. The evaluation showed that short-term consequences (early fatalities) decreased by a factor of two when the fission product inventory was changed from that for 30 days after final shutdown to that for one year after final shutdown. It also showed that, at one year after final shutdown, early evacuation decreased early fatalities by up to a factor of 100. Long-term consequences (cancer fatalities and societal dose) were unaffected by the additional decay and early evacuation. Representative results for the Surry population density are shown in Table 1.

Table 1 Representative Results
(99.5% evacuation, Surry Population Density)

Decay Time Prior to Accident	Mean Consequences (within 100 miles)		
	Early Fatalities	Societal Dose (person-rem)	Cancer Fatalities
30 days	1.75	4.77×10^6	2,460
1 year	1.01	4.54×10^6	2,320
1 year[a]	.0048	4.18×10^6	1,990

[a] Based on evacuation before release.

As noted above, the staff's consequence evaluation was an extension of an earlier consequence evaluation to gain insight into the effect of one year of decay and of early evacuation. Subsequent reviews of the staff's consequence evaluation identified issues with the earlier evaluation performed by BNL in the areas of fractional release from the fuel of each fission product (i.e., fission product source term) and plume-related parameters. To address these issues, the staff performed additional MACCS sensitivity calculations which are documented below.

Fission Product Source Term

The Appendix 4 consequence assessment was based on the release fractions shown in Table 2, which are from the BNL study.[1] It also was based on releasing fission products from a number of fuel assemblies equivalent to 3.5 reactor cores. These release fractions include relatively small release fractions for the low-volatile and non-volatile fission products.

Table 2 Fission Product Release Fractions from the BNL Study

xenon, krypto n	iodine	cesium	telluriu m	strontium	barium	ruthenium	lantha-num	cerium

1	1	1	$2x10^{-2}$	$2x10^{-3}$	$2x10^{-3}$	$2x10^{-5}$	$1x10^{-6}$	$1x10^{-6}$

A subsequent review of the staff's spent fuel pool risk assessment indicated that significant air ingression, influencing fission product release, will occur in accidents involving quick drain-down, and the staff's consequence assessment should accommodate any reasonable uncertainty in the progression of the accident with the possible exception of an increase in the ruthenium release. The ruthenium release fraction used in the staff's consequence assessment was $2x10^{-5}$. Small-scale Canadian experiments show that, in an air environment, significant ruthenium releases begin after the oxidation of 75% to 100% of the cladding, and that the ruthenium release fraction can be as high as the release fraction of the volatile fission products. However, in a spent fuel pool accident, rubbling of the fuel may limit the ruthenium release fraction to a smaller value than that of the volatile fission products.

With regard to the number of fuel assemblies releasing fission products, the thermal-hydraulic evaluation in the BNL study indicated that, as a result of radioactive decay, assemblies other than those from the final core may not reach temperatures high enough to release fission products. The number of assemblies assumed to release fission products in the Appendix 4 consequence assessment is equivalent to 3.5 cores. With regard to the release fractions of the low-volatile and non-volatile fission products, higher release fractions than those in the BNL study may be possible as a result of the release of fuel fines due to fuel pellet decrepitation associated with high fuel burnup.

Ruthenium:

To assess the sensitivity of the consequences to the ruthenium release fraction, the staff performed consequence calculations with and without significant ruthenium releases. The starting point for this assessment was the Base Case calculation from Appendix 4. Then, sensitivity cases were run with a ruthenium release fraction of one and a uniform population density of 100 people/mile^2. The results of these cases (i.e., Base Case, Cases 11, 21, 22) are given in Table 3. For these cases, the effect of ruthenium is to increase the number of prompt fatalities by a factor of ten to 90. The effect on societal dose and cancer fatalities is a more modest increase, with the largest effect being a factor-of-four increase in cancer fatalities for the Surry population density.

Table 3 Results of Ruthenium Release Sensitivities (99.5% evacuation)

Case	Population Density[b]	Ruthenium release fraction	Mean Consequences (within 100 miles)		
			Prompt Fatalities	Societal Dose (person-rem)	Cancer Fatalities
Base Case	Surry	$2x10^{-5}$	1.01	$4.54x10^6$	2,320
11	Surry	1	95.3	$9.53x10^6$	9,150
21	uniform	$2x10^{-5}$	9.33	$5.05x10^6$	2,490
22	uniform	1	134	$9.46x10^6$	6,490
13[a]	Surry	$2x10^{-5}$	.0048	$4.18x10^6$	1,990
14[a]	Surry	1	.132	$6.75x10^6$	6,300
15[a]	uniform	$2x10^{-5}$	.045	$4.65x10^6$	2,170
16[a]	uniform	1	.277	$6.38x10^6$	4,940

[a]Based on evacuation before release.
[b]The uniform population density site has a population density of 100 people/mile^2 with an Exclusion Area Boundary of .75 miles.

The Base Case calculation assumed that evacuation begins about an hour after the fission product release begins. However, Appendix 1 states that, after a year of decay, it will take a number of hours

for the fuel with the highest decay power density to heat up to the point of releasing fission products in the fastest progressing accident scenarios. As a result, it is more likely to have evacuation before the release begins. Therefore, the Base Case calculation then was modified to begin the evacuation three hours before the fission product release begins. This modified Base Case is called Case 13. Starting with Case 13, sensitivity cases were run with a ruthenium release fraction of one and a uniform population density of 100 people/mile^2. The results of these cases (i.e., Cases 13, 14, 15, 16) are given in Table 3. For these cases, the effect of ruthenium is to increase the number of prompt fatalities by a factor of six to 30. The effect on societal dose and cancer fatalities is a more modest increase, with the largest effect being a factor-of-three increase in cancer fatalities for the Surry population density.

For the cases in Table 3, the total number of prompt fatalities increases by a larger factor for Surry than for the uniform population density when a significant ruthenium release is included. Therefore, as part of the ruthenium sensitivity assessment, the staff further examined the effect of population density on prompt fatalities. For the cases with late evacuation (i.e., Base Case, Cases 11, 21, 22), Table 4 gives the MACCS results for the individual risk of a prompt fatality in each radial ring which is composed of 16 sectors. The individual risk of a prompt fatality is a function of the dose to an individual and is independent of the population density. The total number of prompt fatalities is calculated in MACCS by multiplying, in each sector, the individual risk of a prompt fatality by the total number of people in that sector. Table 5, which is the result of multiplying the individual risk of a prompt fatality in each ring from Table 4 by the population in each ring, indicates that Surry's higher increase in prompt fatalities is caused by the jump in the Surry population density at 8.1 km shown in Table 4.

Table 4 Individual Risk of a Prompt Fatality for Cases with Late Evacuation

Distance (km)	Individual risk of a prompt fatality		Ratio	Surry population density* (persons/ km^2)
	Base Case and Case 21, Ru release fraction of 2x10^{-5}	Cases 11 and 22, Ru release fraction of 1		
0 - .2	.146	.169	1.16	0
.2 - .5	.0302	.0657	2.18	0
.5 - 1.2	.0138	.0374	2.71	1.33
1.2 - 1.6	.00828	.0301	3.64	1.13
1.6 - 2.1	.00575	.0266	4.63	1.80
2.1 - 3.2	.00326	.0216	6.63	1.58
3.2 - 4.0	.00151	.0146	9.67	7.15
4.0 - 4.8	.00167	.0132	7.90	7.77
4.8 - 5.6	.00171	.0110	6.43	7.84
5.6 - 8.1	.0000672	.0131	194.94	8.07
8.1 - 11.3	.000000254	.00301	11850.39	117.80
11.3 - 16.1	0	.0000225	NA	118.36
16.1 - 20.9	0	0	NA	83.75

*This data is from the MACCS input file SURSIT.INP.

Table 5 Number of Prompt Fatalities in Each Radial Ring for Cases with Late Evacuation

Distance (km)	Number of early fatalities with Surry population density		Number of early fatalities with uniform population density	
	Base Case, Ru release fraction of $2x10^{-5}$	Case 11, Ru release fraction of 1	Case 21, Ru release fraction of $2x10^{-5}$	Case 22, Ru release fraction of 1
0 - .2	0	0	0	0
.2 - .5	0	0	0	0
.5 - 1.2	.0690	.1870	0	0
1.2 - 1.6	.0331	.1204	1.1329	4.1184
1.6 - 2.1	.0633	.2926	1.3564	6.2750
2.1 - 3.2	.0945	.6264	2.3060	15.2788
3.2 - 4.0	.1963	1.8980	1.0609	10.2574
4.0 - 4.8	.2923	2.3100	1.4521	11.4777
4.8 - 5.6	.3523	2.2660	1.7357	11.1653
5.6 - 8.1	.0564	10.9909	.2699	52.6050
8.1 - 11.3	.0058	69.2661	.0019	22.7135
11.3 - 16.1	0	1.1027	0	.3599
16.1 - 20.9	0	0	0	0
Total	1.16	89.06	9.32	134.25

The staff also performed sensitivity calculations to determine which isotope in the ruthenium group is responsible for the increase in consequences when a significant ruthenium release is included in the consequence calculations. Sensitivity calculations were performed with different ruthenium-group isotopes included in the consequence calculations. The ruthenium-group isotopes remaining after a year of radioactive decay are Co-58, Co-60, Ru-103, and Ru-106. These cases were run starting with the Base Case. The results of these calculations are shown in Table 6. These results show that the dominant isotope in the ruthenium group is Ru-106.

Table 6 Cases with Different Ruthenium-Group Isotopes Included

Case	Ruthenium Release Fraction	Isotopes Included	Mean Consequences (within 100 miles)		
			Prompt Fatalities	Societal Dose (person-rem)	Cancer Fatalities
Base Case	$2x10^{-5}$	Co-58,Co-60,Ru-103,Ru-106	1.01	$4.54x10^{6}$	2,320
11	1	Co-58,Co-60,Ru-103,Ru-106	95.3	$9.53x10^{6}$	9,150
11a	1	Ru-103,Ru-106	94.4	$9.51x10^{6}$	9,120
11b	1	Ru-106	94.3	$9.51x10^{6}$	9,120
11c	1	Ru-103	1.02	$4.54x10^{6}$	2,320

The amounts of the dominant cesium isotope, Cs-137, and the dominant ruthenium isotope, Ru-106, in a spent fuel pool at one year after final shutdown are about the same. After one year, the inventories of Cs-137 and Ru-106 are $8.38x10^{17}$ Bq and $5.77x10^{17}$ Bq, respectively. This would suggest a modest increase in the individual risk of a prompt fatality ruthenium is included in the consequence calculation. However, Table 4 shows large increases in the individual risk of a prompt fatality. A comparison of the dose conversion factors for Cs-137 and Ru-106 is given in Table 7. These dose conversion factors were taken from the MACCS input file DOSDATA.INP. An examination of these dose conversion factors indicates that the large Ru-106 inhalation dose conversion factor in MACCS used to calculate acute doses is partly responsible for the increase in individual risk of a prompt fatality beyond what would be expected as a result of the additional amount of Ru-106.

Table 7 Dose Conversion Factors for Ru-106 and Cs-137

	organ	cloud-shine (Sv sec/ Bq m^3)	ground-shine (Sv sec/ Bq m^2)	inhalation/ acute (Sv/Bq)	inhalation/ chronic (Sv/Bq)	ingestion (Sv/Bq)
Ru-106	lungs	7.99E-15	1.58E-16	2.09E-08	1.04E-06	1.48E-09
	red marrow	8.05E-15	1.61E-16	8.74E-11	1.77E-09	1.48E-09
Cs-137	lungs	2.88E-14	4.35E-16	8.29E-10	8.80E-09	1.27E-08
	red marrow	2.22E-14	4.41E-16	5.63E-10	8.30E-09	1.32E-08
Ratio of Ru-106 to Cs-137	lungs	.4	.4	**25**	118	.1
	red marrow	.4	.4	.2	.2	.1

Fuel Fines:

The staff performed MACCS calculations with different fuel fines release fractions to assess the sensitivity of the consequences. The results of these calculations are shown in Table 8. Case 11, which used a ruthenium release fraction of one, is the shown in the second row of Table 8 and was the starting point for these calculations. Then, Case 96 was run with the large fuel fines release fraction of .01. As a result of increasing the fuel fines release fraction from $1x10^{-6}$ to .01, a small increase in the offsite consequences was seen.

Table 8 Results of Release Fraction Sensitivities (99.5% evacuation, Surry Population Density)

Case	Release Fraction							Mean Consequences (within 100 miles)		
	I,Cs	Ru	Te	Ba	Sr	Ce	La	Early Fatalities	Societal Dose (person-rem)	Cancer Fatalities
Base	1	$2x10^{-5}$	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	1.01	$4.54x10^{6}$	2,320
11	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	95.3	$9.53x10^{6}$	9,150
96	1	1	.02	.01	.01	.01	.01	106	$1.33x10^{7}$	11,700
95	.75	.75	.02	.01	.01	.01	.01	57.0	$1.17x10^{7}$	10,400
94	.75	.75	.02	.002	.002	.001	.001	50.2	$8.35x10^{6}$	7,850
14[a]	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	.132	$6.75x10^{6}$	6,300
97[a]	1	1	.02	.01	.01	.01	.01	.154	$8.74x10^{6}$	7,990

[a]Based on evacuation before release.

The evaluation documented in Appendix 4 used a conservative release fraction of one for the volatile fission products. NUREG-1465, *Accident Source Terms for Light-Water Nuclear Power Plants*, February 1995, specifies a more realistic release fraction of .75 for volatile fission products. As part of the sensitivity of the effect of fuel fines release fraction, this more realistic release fraction was used. In Case 95, the consequences decreased as a result of decreasing the volatile fission product release fraction from 1 to .75. In this case, a factor-of-two decrease in the early fatalities and a small decrease in the long-term consequences were seen.

Finally, Case 94 was run to investigate the sensitivity of the consequences to a fuel fines release fraction intermediate between $1x10^{-6}$ and .01. This case used a fuel fines release fraction of .001. As a result of decreasing the fuel fines release fraction from .01 to .001, a small decrease in the consequences was seen.

In Case 11, evacuation begins about an hour after the fission product release begins. However, Appendix 1 states that, after a year of decay, it will take a number of hours for the fuel with the highest decay power density to heat up to the point of releasing fission products in the fastest progressing accident scenarios. As a result, it is more likely to have evacuation before the release begins. Therefore, a sensitivity calculation on fuel fines release fraction also was run using Case 14 as the starting point; Case 14 includes evacuation three hours before the release begins. Case 97 was run with a fuel fines release fraction of .01. As a result of increasing the fuel fines release fraction from $1x10^{-6}$ to .01, a small increase in the offsite consequences was seen.

The above sensitivity calculations for fuel fines release fractions were performed with 99.5% of the population evacuating. This translates into one person in 200 not evacuating. It has been suggested that the percentage of the population evacuating may be smaller. Therefore, the staff performed additional calculations with 95% of the population evacuating. This translates into one person in 20 not evacuating. The results of these calculations are shown in Table 9. Case 45, which used a ruthenium release fraction of one, is the shown in the second row of Table 9 and was the starting point for these calculations. Then, Case 45a was run with a fuel fines release fraction of .01, and Case 45b was run with a volatile fission product release fraction of .75. The same trends were seen as in the 99.5% evacuation cases, Cases 11, 96, and 95.

Table 9 Results of Release Fraction Sensitivities (95% evacuation, Surry Population Density)

Case	Release Fraction							Mean Consequences (within 100 miles)		
	I,Cs	Ru	Te	Ba	Sr	Ce	La	Early Fatalities	Societal Dose (person-rem)	Cancer Fatalities
1	1	$2x10^{-5}$	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	1.01	$4.54x10^{6}$	2,320
45	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	92.2	$9.50x10^{6}$	9,150
45a	1	1	.02	.01	.01	.01	.01	103	$1.33x10^{7}$	11,700
45b	.75	.75	.02	.01	.01	.01	.01	54.9	$1.17x10^{7}$	10,300
46[a]	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	1.32	$6.84x10^{6}$	6,430
46a[a]	1	1	.02	.01	.01	.01	.01	1.54	$8.89x10^{6}$	8,160
46b[a]	.75	.75	.02	.01	.01	.01	.01	.543	$7.94x10^{6}$	6,880
46c[a]	.75	.75	.75	.01	.01	.01	.01	.544	$7.94x10^{6}$	6,880
46d[a]	.75	.75	.75	.75	.01	.01	.01	.544	$7.94x10^{6}$	6,880
46e[a]	.75	.75	.75	.75	.75	.01	.01	.644	$1.01x10^{7}$	8,350

[a]Based on evacuation before release.

In addition, the staff performed calculations with 95% of the population evacuating with the evacuation beginning three hours before the release begins. The results of these calculations are shown in Table 9. The starting point for these calculations was Case 46, which includes evacuation beginning three hours before the release begins. Then, Case 46a was run with a fuel fines release fraction of .01. The same trends were seen as in the 99.5% evacuation cases, Cases 14 and 97.

The main difference between the results for 99.5% and 95% evacuation is in the area of early fatalities for cases with evacuation before release. In comparing Cases 14 and 97 with Cases 46 and 46a, a factor-of-ten increase in early fatalities is seen, because of the factor-of-ten increase in persons not evacuating. Cases 14 and 97 use one out of 200 people not evacuating, while Cases 46 and 46a use ten out of 200 people not evacuating.

The staff also performed sensitivity calculations for tellurium, barium, and strontium by increasing their release fractions to that of the volatile fission products, that is, .75. In Case 46c, the release fraction for tellurium was increased from .02 to .75. In Case 46d, the release fraction for barium was increased from .01 to .75. No change in consequences were seen in these two cases, because of the small inventories of these isotopes after a year of decay. In Case 46e, the release fraction for strontium was increased from .01 to .75. A small increase in the consequences was seen in this case.

The results in Table 9 are the total number of early fatalities, societal dose, and cancer fatalities for the population within 100 miles of the facility. However, the NRC's quantitative health objectives are given in terms of individual risk of an early fatality within one mile and individual risk of a cancer fatality within

ten miles. The MACCS results in terms of these two consequence measures are given in Table 10.

Table 10 Results of Release Fraction Sensitivities (95% evacuation, Surry Population Density)

Case	Release Fraction							Mean Consequences	
	I,Cs	Ru	Te	Ba	Sr	Ce	La	Individual Risk of an Early Fatality (within one mile)	Individual Risk of a Cancer Fatality (within ten miles)
45a	1	1	.02	.01	.01	.01	.01	3.66×10^{-2}	5.16×10^{-2}
45b	.75	.75	.02	.01	.01	.01	.01	3.23×10^{-2}	4.98×10^{-2}
46a[a]	1	1	.02	.01	.01	.01	.01	1.61×10^{-3}	2.83×10^{-3}
46b[a]	.75	.75	.02	.01	.01	.01	.01	1.40×10^{-3}	2.55×10^{-3}

[a]Based on evacuation before release.

Amount of Fuel Releasing Fission Products:

To assess the sensitivity to the fission product inventory released, the staff performed calculations with all of the spent fuel (i.e., 3.5 cores) and the final core offload releasing fission products. These calculations were run for cases with evacuation beginning after the release begins. The inventories used in the MACCS calculations for one core are the Table A.5 inventories in the BNL study reduced by one year of radioactive decay. The results of the MACCS calculations are given in Table 11.

Table 11 Sensitivities on Amount of Fuel Assemblies Releasing Fission Products (99.5% evacuation)

Case	Population Density	Ruthenium Release Fraction	# of cores	Mean Consequences (within 100 miles)		
				Prompt Fatalities	Societal Dose (person-rem)	Cancer Fatalities
Base Case	Surry	2×10^{-5}	3.5	1.01	4.54×10^{6}	2,320
31	Surry	2×10^{-5}	1	.014	3.23×10^{6}	1,530
11	Surry	1	3.5	95.3	9.53×10^{6}	9,150
32	Surry	1	1	50.5	7.25×10^{6}	7,360
21	uniform	2×10^{-5}	3.5	9.33	5.05×10^{6}	2,490
33	uniform	2×10^{-5}	1	.177	3.10×10^{6}	1,480
22	uniform	1	3.5	134	9.46×10^{6}	6,490
34	uniform	1	1	103	6.59×10^{6}	4,960

For the cases with a ruthenium release fraction of 2×10^{-5}, the reduction in prompt fatalities is caused by the reduction in the Cs-137 inventory which decreases from 8.38×10^{17} Bq to 2.11×10^{17} Bq in going from 3.5 cores to one core. This was confirmed by rerunning Case 33 with a Cs-137 inventory of 8.38×10^{17} Bq. The reductions in prompt fatalities for uniform and Surry population densities are factors of 52 and 72, respectively. These reductions are more than proportional to the factor-of-four reduction in Cs-137

inventory, because of the combined effects of individual risk of early fatality and non-uniform population density as discussed in the above analysis of the effect of ruthenium on offsite consequences.

For the cases with a ruthenium release fraction of one, the reduction in prompt fatalities is caused by the reduction in the Ru-106 inventory which decreases from 5.77×10^{17} Bq to 4.59×10^{17} Bq in going from 3.5 cores to 1 core. This was confirmed by rerunning Case 34 with a Ru-106 inventory of 5.77×10^{17} Bq. The reductions in prompt fatalities for uniform and Surry population densities are factors of 1.30 and 1.89, respectively. These reductions are nearly proportional to the factor of 1.26 reduction in the Ru-106 inventory. Again, deviations from being proportional are due to the combined effects of individual risk of early fatality and non-uniform population density. Overall, the effect of reducing the number of assemblies on prompt fatalities is less pronounced for the cases with a ruthenium release fraction of one, in part, because the additional 2.5 cores has a small amount of Ru-106 (one year half-life) in comparison with Cs-137 (30 year half-life). Finally, in all of the cases, the effect of reducing the amount of fuel releasing fission products from 3.5 cores to one core is a modest decrease (20 to 40%) in societal dose and cancer fatalities.

Plume-Related Parameters

The evaluation documented in Appendix 4 used the plume heat content associated with a large early release for a reactor accident. The plume heat content for a spent fuel pool accident may be higher, because (1) a spent fuel pool does not have a containment as a heat sink and (2) the heat of reaction for zirconium oxidation is 85% higher in air than in steam. Also, the evaluation documented in Appendix 4 used the default values for the plume-spreading model in MACCS version 2.[2] NUREG/CR-6244, *Probabilistic Accident Consequence Uncertainty Analysis*, January 1995, provides improved values for these parameters.

Plume Heat Content:

The staff estimated that the complete oxidation in air (in a half hour) of the amount of zircalloy cladding in a large BWR core would generate 256 MW. Subsequently, Sandia National Laboratories (SNL) performed a more detailed assessment of the plume heat content for a spent fuel pool accident.[3] SNL calculated that oxidation of 36% of the zircalloy cladding and fuel channels by the oxygen in the air flow would heat up the accompanying nitrogen and the spent fuel to 2500 K. Once the spent fuel reaches 2500 K, it will degrade into a geometry in which continued exposure to air and, therefore, oxidation, will be precluded. For a spent fuel pool accident involving the amount of fuel in a large BWR core, SNL estimated the heat content of the nitrogen plume to be 43 MW. The SNL estimate was made by subtracting (a) the energy absorbed by the spent fuel in heating up to 2500 K from (b) the energy released by the oxidation of 36% of the zircalloy cladding and fuel channels.

The staff performed calculations with different plume heat contents to assess the sensitivity of the consequences. The results of these calculations are shown in Table 12. Case 45, which used a ruthenium release fraction of one, is shown in the second row of Table 12 and was the starting point for these calculations. Case 45 used a plume heat content of 3.7 MW, which is associated with a large early release for a reactor accident. Then, Cases 47 and 49 were run with plume heat contents of 83.0 MW and 256 MW, respectively. Increasing the plume heat content from 3.7 MW to 83.0 MW resulted in a factor-of-two decrease in the early fatalities and no change in the long-term consequences. Increasing the plume heat content from 83.0 MW to 256 MW resulted in a factor-of-three decrease in the early fatalities and a small decrease in the long-term consequences.

Table 12 Results of Plume Heat Content Sensitivities (95% evacuation, Surry Population Density)

Case	Release Fraction							Plume Heat Content (MW)	Mean Consequences (within 100 miles)		
	I,Cs	Ru	Te	Ba	Sr	Ce	La		Early Fatalities	Societal Dose (person-rem)	Cancer Fatalities
1	1	$2x10^{-5}$	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	3.7	1.01	$4.54x10^{6}$	2,320
45	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	3.7	92.2	$9.50x10^{6}$	9,150
47	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	83.0	57.3	$9.24x10^{6}$	9,280
49	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	256.0	18.3	$8.24x10^{6}$	8,380
46[a]	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	3.7	1.32	$6.84x10^{6}$	6,430
48[a]	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	83.0	.00509	$7.28x10^{6}$	7,060
50[a]	1	1	.02	.002	.002	$1x10^{-6}$	$1x10^{-6}$	256.0	.00357	$6.96x10^{6}$	6,650

[a]Based on evacuation before release.

Cases 45, 47, and 49 were based on evacuation about an hour after the release began. The staff also performed calculations based on evacuation beginning three hours before the release begins. Case 46, which used a ruthenium release fraction of one and evacuation beginning three hours before the release begins, is shown in the fourth row of Table 12 and was the starting point for these calculations. Then, Cases 48 and 50 were run with plume heat contents of 83.0 MW and 256 MW, respectively. Increasing the plume heat content from 3.7 MW to 83.0 MW resulted in a factor-of-300 decrease in the early fatalities and a small increase in the long-term consequences. Increasing the plume heat content from 83.0 MW to 256 MW resulted in a small decrease in the early fatalities and a small decrease in the long-term consequences.

Plume Spreading:

MACCS uses a Gaussian plume model with the amount of spreading determined by the parameters σ_y and σ_z, where y is the cross-wind direction and z is the vertical direction. In NUREG/CR-6244, phenomenological experts provided updated values for σ_y and σ_z. However, the experts did not provide single values of these parameters. Instead, they provided probability distributions. To assess the sensitivity of spent fuel pool accident consequences to the updated values for σ_y and σ_z, Sandia National Laboratories performed MACCS calculations using values for σ_y and σ_z randomly selected from the experts distributions.[4] These MACCS calculations were based on Cases 11 and 14 (see Table 3), which use the Surry population density and a ruthenium release fraction of one. Case 11 has evacuation beginning about an hour after the release begins, while Case 14 has evacuation beginning three hours before the release begins. A total of 300 MACCS runs were performed to generate distributions of early fatalities, population dose, and cancer fatalities. The results of these MACCS runs are shown in Tables 13 and 14. For the late evacuation case, Case 11, the 50th percentile and mean results using NUREG/CR-6244 plume spreading are lower for early fatalities and higher for societal dose and cancer fatalities. The same trend is seen for the early evacuation case, Case 14. Overall, the effect of the plume spreading model on offsite consequences is not large.

Table 13 Results of Plume-Spreading Model Sensitivity for Case 11 (99.5% evacuation, Surry Population Density)

Plume-Spreading Model	Point in Distribution	Early Fatalities	Societal Dose (rem)	Cancer Fatalities
default	not applicable	95.3	9.53×10^6	9,150
NUREG/CR-6244	10th percentile	.527	9.04×10^6	8,343
	50th percentile	8.89	1.26×10^7	10,100
	mean	54.1	1.28×10^7	10,100
	90th percentile	171	1.66×10^7	11,900

Table 14 Results of Plume-Spreading Model Sensitivity for Case 14 (99.5% evacuation, Surry Population Density)

Plume-Spreading Model	Point in Distribution	Early Fatalities	Societal Dose (rem)	Cancer Fatalities
default	not applicable	.132	6.75×10^6	6,300
NUREG/CR-6244	10th percentile	.00197	7.00×10^6	6,010
	50th percentile	.00855	1.03×10^7	7,730
	mean	.118	1.07×10^7	7,810
	90th percentile	.0637	1.46×10^7	9,590

Conclusion

Appendix 4 documents the staff's evaluation of the offsite consequences of a spent fuel pool accident involving a sustained loss of coolant, leading to a significant fuel heatup and resultant release of fission products to the environment. The objectives of the staff's evaluation were (1) to assess the effect of one year of decay and (2) to assess the effect of early versus late evacuation because spent fuel pool accidents are slowly evolving accidents. The staff's evaluation was an extension of an earlier study performed by BNL for spent fuel pools at operating reactors, which assessed consequences using inventories for 30 days after shutdown. Subsequent reviews of the staff's consequence evaluation identified issues with the earlier evaluation performed by BNL in the areas of fission product source term and plume-related parameters. To address these issues, the staff performed additional MACCS sensitivity calculations which are documented in the current appendix.

With regard to the fission product source term, sensitivity calculations were performed using different release fractions for the nine fission product groups. These calculations also included variations in population density, evacuation start time, percentage of the population evacuating, and number of fuel assemblies releasing fission products. With regard to plume-related parameters, sensitivity calculations were performed using different plume heat contents and updated values for the plume-spreading parameters.

With the exception of ruthenium, increasing the release fraction of each fission product group resulted in a negligible to modest (less than 40%) increase in consequences. Increasing the ruthenium release fraction resulted in a larger increase in consequences. However, these consequence increases were demonstrated to be largely offset by beginning the evacuation before the release begins. Such an early evacuation is likely, because after a year of decay, it will take a number of hours for the fuel with the highest decay power to heat up to the point of releasing fission products.

Other sensitivity calculations involved examining the effect of (1) decreasing the amount of fuel

releasing fission products from the entire spent fuel pool inventory to the final core offload and (2) decreasing the percentage of the population evacuating from 99.5% and 95%. For cases with a small ruthenium release, the main effect of decreasing the amount of fuel releasing fission products was a large reduction in prompt fatalities. However, for cases with a large ruthenium release, the prompt fatalities did not change as much, because most of the ruthenium is in the final core offload due to its one-year half-life. With regard to the percentage of the population evacuating, the main difference between 99.5% and 95% evacuation is in the area of early fatalities for cases with evacuation before release. In these cases, the number of early fatalities increases by a factor of ten, because a change from 99.5% to 95% is a factor-of-ten increase in the number of persons not evacuating.

The sensitivity calculations also showed that increasing the plume heat content resulted in reductions in early fatalities and no change in societal dose or cancer fatalities. In addition, updating the values of the plume-spreading parameters to those in the NUREG/CR-6244 expert elicitation results in a decrease in early fatalities and up to a 60% increase in societal dose and cancer fatalities, because of the additional plume spreading associated with the updated plume-spreading parameter values.

References

1. *Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82*, NUREG/CR-4982, July 1987

2. *Code Manual for MACCS2*, NUREG/CR-6613, May 1998

3. *Analysis of Plume Energy from Air Oxidation in Spent Fuel Storage Pool*, Sandia National Laboratories, August 7, 2000

4. *Task 7 Letter Report: Investigation of Plume Spreading Uncertainties on the Radiological Consequences Associated with a Spent Fuel Pool Accident*, Sandia National Laboratories, June 2000

APPENDIX 4B

Memo to Gary M. Holahan from Farouk Eltawila dated 10/26/00 re:
RADIOLOGICAL CONSEQUENCES OF SPENT FUEL POOL ACCIDENT OCCURRING UP TO 10 YEARS AFTER FINAL REACTOR SHUTDOWN

October 26, 2000

MEMORANDUM TO: Gary M. Holahan, Director
Division of Systems Safety and Analysis
Office of Nuclear Reactor Regulation

FROM: Farouk Eltawila, Acting Director **(Original signed by)**
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

SUBJECT: RADIOLOGICAL CONSEQUENCES OF SPENT FUEL POOL ACCIDENTS OCCURRING UP TO 10 YEARS AFTER FINAL REACTOR SHUTDOWN

As part of its effort to develop generic, risk-informed requirements for decommissioning, NRR requested (Reference 1) that RES evaluate the offsite radiological consequences of beyond-design-basis spent fuel pool accidents. In response to that user need, RES completed an in-house analysis (References 2 and 3) using the MACCS code (Reference 4). The focus of that work was estimation of consequences of accidents occurring between 30 days and 1 year after final reactor shutdown. Recently, NRR requested (References 5 and 6) that RES extend the consequence evaluation to accidents occurring up to 10 years after final shutdown.

RES performed the requested calculations using the release fractions in Table 1 and the fission product inventories at 30 and 90 days and 1, 2, 5, and 10 years after final shutdown. The release fractions in the first row of Table 1 are the sum of the in-vessel and ex-vessel release fractions in NUREG-1465, “Accident Source Terms for Light-Water Nuclear Power Plants,” February 1995 (Reference 7). NUREG-1465 has received significant peer review and is representative of a low pressure core-melt accident. The release fractions in the second row of Table 1, other than those for ruthenium and fuel fines, also are from NUREG-1465. In this case, the ruthenium release fraction is that for a volatile fission product, and the fuel fines release fraction is that from the Chernobyl accident (Reference 8). Results of the RES calculations for distances of 1, 10, and 50 miles are given in Tables 2 and 3.

Table 1 Fission Product Release Fractions

Source Term	Release Fractions								
	Xe,Kr	I	Cs	Te	Sr	Ba	Ru	La	Ce
NUREG-1465	1	.75	.75	.31	.12	.12	.005	.0052	.0055
NUREG-1465 (modified)	1	.75	.75	.31	.12	.12	.75	.035	.035

Table 2 Results based on NUREG-1465 Source Term

Case	Decay Time	Mean Consequences[a] (Surry population, 95% evacuation)			
		Individual Risk of Early Fatality (within 1mile)	Individual Risk of Cancer Fatality (within 10 miles)	Societal Dose (rem) (within 50 miles)	Early Fatalities (within 10 miles)
77a	30 days	1.27×10^{-2}	1.88×10^{-2}	5.58×10^{6}	2.21
77b	90 days	9.86×10^{-3}	1.82×10^{-2}	5.43×10^{6}	1.37
77c	1 year	7.13×10^{-3}	1.68×10^{-2}	5.28×10^{6}	.736
77d	2 years	5.64×10^{-3}	1.58×10^{-2}	5.12×10^{6}	.481
77e	5 years	3.18×10^{-3}	1.43×10^{-2}	4.90×10^{6}	.192
77f	10 years	1.63×10^{-3}	1.29×10^{-2}	4.72×10^{6}	.0778
78a[b]	30 days	8.36×10^{-4}	9.92×10^{-4}	4.12×10^{6}	.0720
78b[b]	90 days	6.83×10^{-4}	9.62×10^{-4}	4.02×10^{6}	.0461
78c[b]	1 year	5.44×10^{-4}	9.09×10^{-4}	3.95×10^{6}	.0301
78d[b]	2 years	4.41×10^{-4}	8.71×10^{-4}	3.87×10^{6}	.0208
78e[b]	5 years	2.54×10^{-4}	8.14×10^{-4}	3.77×10^{6}	.00882
78f[b]	10 years	1.47×10^{-4}	7.70×10^{-4}	3.69×10^{6}	.00400

[a]Accident frequencies approximately 10^{-6}/year or less.
[b]Based on early evacuation.

Table 3 Results based on NUREG-1465 (modified) Source Term

Case	Decay Time	Mean Consequences[a] (Surry population, 95% evacuation)			
		Individual Risk of Early Fatality (within 1 mile)	Individual Risk of Cancer Fatality (within 10 miles)	Societal Dose (rem) (within 50 miles)	Early Fatalities (within 10 miles)
79a	30 days	4.43×10^{-2}	8.24×10^{-2}	2.37×10^{7}	191
79b	90 days	4.19×10^{-2}	8.20×10^{-2}	2.25×10^{7}	162
79c	1 year	3.46×10^{-2}	8.49×10^{-2}	1.93×10^{7}	76.9
79d	2 years	2.57×10^{-2}	8.42×10^{-2}	1.69×10^{7}	19.2
79e	5 years	8.96×10^{-3}	7.08×10^{-2}	1.45×10^{7}	1.34
79f	10 years	4.68×10^{-3}	6.39×10^{-2}	1.34×10^{7}	.360
80a[b]	30 days	2.01×10^{-3}	4.79×10^{-3}	1.35×10^{7}	5.38
80b[b]	90 days	1.87×10^{-3}	4.77×10^{-3}	1.29×10^{7}	3.61
80c[b]	1 year	1.50×10^{-3}	4.33×10^{-3}	1.12×10^{7}	.951
80d[b]	2 years	1.12×10^{-3}	3.70×10^{-3}	9.93×10^{6}	.149
80e[b]	5 years	3.99×10^{-4}	2.93×10^{-3}	8.69×10^{6}	.0162
80f[b]	10 years	2.05×10^{-4}	2.64×10^{-3}	8.13×10^{6}	.00601

[a]Accident frequencies approximately 10^{-6}/year or less.
[b]Based on early evacuation.

References:
1. Memorandum from G. Holahan to T. King dated March 26, 1999
2. Memorandum from A. Thadani to S. Collins dated November 12, 1999
3. Memorandum from F. Eltawila to G. Holahan dated August 25, 2000
4. Code Manual for MACCS2, NUREG/CR-6613, May 1998
5. Memorandum from R. Barrett to J. Flack dated August 25, 2000
6. Memorandum from S. Collins to A. Thadani dated September 11, 2000
7. Accident Source Terms for Light-Water Nuclear Power Plants, NUREG-1465, February 1995
8. Chernobyl Ten Years On, Radiological and Health Impact, An Appraisal by the NEA Committee on Radiation Protection and Public Health, November 1995

cc: T. Collins
R. Barrett
J. Hannon
J. Wermiel
G. Hubbard

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APPENDIX 4C
POOL PERFORMANCE GUIDELINE

1. INTRODUCTION

The Pool Performance Guideline (PPG) provides a threshold for controlling the risk from a decommissioning plant spent fuel pool (SFP). By maintaining the frequency of events leading to uncovery of the spent fuel at a value less than the recommended PPG value of $1x10^{-5}$ per year, zirconium fires will remain highly unlikely, the risk will continue to meet the Commission's Quantitative Health Objectives [1], and changes to the plant licensing basis that result in very small increases in LERF may be permitted consistent with the logic in Regulatory Guide 1.174 [2]. The purpose of this appendix is to present the rationale for the PPG, and to illustrate how conformance with the recommended PPG will assure that SFP risk in decommissioning plants will continue to meet the Commission's quantitative health objectives (QHOs).

Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," contains general guidance for application of PRA insights to the regulation of nuclear reactors. The same concepts can also be applied in the regulation of SFPs. The guidelines in RG 1.174 pertain to the frequency of core damage accidents (CDF) and large early releases (LERF). For both CDF and LERF, RG 1.174 contains guidance on acceptable values for the changes that can be allowed as a function of the baseline frequencies. For example, if the baseline CDF for a plant is below $1x10^{-4}$ per year, plant changes can be approved that increase CDF by up to $1x10^{-5}$ per year. If the baseline LERF is less than $1x10^{-5}$ per year, plant changes can be approved that increase LERF by up to $1x10^{-6}$ per year.

For decommissioning plants, the risk is primarily because of the possibility of a zirconium fire associated with the spent fuel cladding. The consequences of such an event do not equate directly to either a core damage accident or a large early release as modeled for an operating reactor. Zirconium fires in SFPs have the potential for significant long-term consequences because: there may be multiple cores involved; the relevant clad/fuel degradation mechanisms could lead to increased releases of certain isotopes (e.g., short-lived isotopes such as iodine will have decayed, but the release of longer-lived isotopes such as ruthenium could be increased because of air-fuel reactions); and there is no containment surrounding the SFP to mitigate the consequences. On the other hand, they are different from a large early release because the postulated accidents progress more slowly, allowing time for protective actions to be taken to significantly reduce early fatalities (and to a lesser extent latent fatalities). In effect, an SFP fire would result in a "large" release, but this release would not generally be considered "early" because of the significant time delay before fission products are released.

Even though the event progresses more slowly than an operating reactor large early release event and the isotopic make-up is somewhat different, the consequence calculations performed by the staff (reported in Appendix 4 and 4A) show that SFP fires could have significant health effects on par with those for a severe reactor accident. These calculations considered the effects of different source terms, evacuation assumptions, and plume-related parameters on offsite consequences. Since an SFP fire scenario would involve a direct release to the environment with significant consequences, the staff has decided that the RG 1.174 LERF baseline guideline of $1x10^{-5}$ per year (the value of baseline risk above which the staff will only consider very small increases in risk) provides an appropriate threshold for controlling the risk from a decommissioning plant SFP, and has established $1x10^{-5}$ per year as the recommended PPG for this purpose. Maintaining the frequency of events leading to uncovery of the spent fuel at a value less than the PPG, will assure that zirconium fires remain highly unlikely and that the risk in a decommissioning plant will continue to meet the Commission's QHOs, as discussed below. Conformance with the PPG is also essential if the staff is to permit changes to the licensing basis that result in small increases in risk, such as

relaxations in Emergency Preparedness requirements.

Our conclusion in the draft final study was that, even though there are some differences in source term and timing, scenarios involving an SFP zirconium fire may result in population doses that are generally comparable to those expected from accident scenarios involving a large early release at operating reactors, and therefore a PPG of $1x10^{-5}$ per year was appropriate. The staff has reassessed these conclusions following the performance of additional consequence calculations in Appendix 4A and 4B that took into account the possibility of significant ruthenium release fractions. This assessment was undertaken to address concerns raised during review of the draft final study that large ruthenium releases from a spent fuel fire could substantially increase both early and latent health effects, as well as shift the controlling decision criteria from early fatalities to latent health effects because of the combined effect of longer times for evacuation and longer ruthenium half life.

In reassessing the appropriateness of the $1x10^{-5}$ per year PPG as discussed below, the staff contrasted the SFP risk for a licensee maintaining its facility at the PPG with the Commission's Safety Goal Policy Statement. The Policy Statement expressed the Commission's policy regarding the acceptable level of radiological risk from nuclear power plant operation as follows:

- Individual members of the public should be provided a level of protection from the consequences of nuclear power plant operation such that individuals bear no significant additional risk to life and health

- Societal risks to life and health from nuclear power plant operation should be comparable to or less than the risks of generating electricity by viable competing technologies and should not be a significant addition to other societal risks.

The following quantitative health objectives (QHOs) are used in determining achievement of the safety goals:

- The risk to an average individual in the vicinity of a nuclear power plant of prompt fatalities that might result from reactor accidents should not exceed one-tenth of 1 percent (0.1 percent) of the sum of prompt fatality risks resulting from other accidents to which members of the U.S. population are generally exposed.

- The risk to the population in the area near a nuclear power plant of cancer fatalities that might result from nuclear power plant operation should not exceed one-tenth of 1 percent (0.1 percent) of the sum of cancer fatality risks resulting from all other causes.

These QHOs have been translated into two numerical objectives as follows:

- The individual risk of a prompt fatality from all "other accidents to which members of the U.S. population are generally exposed," such as fatal automobile accidents, is about $5x10^{-4}$ per year. One-tenth of one percent of this figure implies that the individual risk of prompt fatality from a reactor accident should be less than $5x10^{-7}$ per reactor year.

- "The sum of cancer fatality risks resulting from all other causes" for an individual is taken to be the cancer fatality rate in the U.S. which is about 1 in 500 or $2x10^{-3}$ per year. One-tenth of 1 percent of this implies that the risk of cancer to the population in the area near a nuclear power plant because of its operation should be limited to $2x10^{-6}$ per reactor year.

Although the Policy Statement and related numerical objectives were developed to address the risk associated with power operation, is it reasonable to require that these objectives continue to be met for as long as nuclear materials remain on the plant site. Accordingly, the staff has compared the risks to an individual with the QHOs, assuming the licensee maintains the facility

at the recommended PPG of $1x10^{-5}$ per year. The relevant risk measures are the early fatality risk to an average individual within 1 mile of the plant, and the latent cancer fatality risk to an average individual within 10 miles of the plant. These measures would not be significantly impacted by population density since they are determined on the basis of the risk to the average individual.

Appendix 4B of this study provides the results of offsite consequence calculations for an SFP fire occurring at various times following final shutdown at a hypothetical 3441 MWth BWR SFP located at the Surry site. Additional calculations provided in Appendix 4A address the sensitivity of early and latent health effects to source terms, time of evacuation, percent of population participating in the evacuation, population distribution, number of cores participating in the SFP fire, and plume-related parameters. The risk measures corresponding to the above numerical objectives were calculated using the MACCS2 computer code for each of the cases reported in Appendix 4B (i.e., low and high ruthenium source term with both early and late evacuation), and for the worst case SFP accident source term reported in Appendix 4A. The latter case, identified as Case 45a, corresponds to a complete release of the volatiles (cesium and iodine) and ruthenium, a 0.01 release of fuel fines, and late evacuation of 95 percent of the population. The results are reported in Table 1. For comparison with the numerical objectives, the staff assumed that the licensee maintains the facility at the recommended PPG of $1x10^{-5}$ per year.

The risk results indicate that at a PPG of $1x10^{-5}$ per year, the QHOs would continue to be met for even the most severe cases considered in Appendix 4A and 4B. The margins to both QHOs are substantial (about two orders of magnitude) for the case with early evacuation even with the large ruthenium release. The margins are considerably reduced in the late evacuation cases, but sufficient to conclude that the QHOs would be met given the bounding nature of the source terms and fission product inventories used in these calculations. Although the comparisons in Table 1 are at a time 1 year after shutdown, the staff also evaluated the risk at 30 days after shutdown and found that it continues to meet the QHOs.

The margin to the QHO is smallest (i.e., the percent of QHO is the largest) for early fatality risk. Thus, similar to severe accidents in operating reactors, acceptable levels of risk for an SFP accident would be controlled by the early fatality risk measure. The margins to the QHO observed in these calculations suggest that the recommended PPG of $1x10^{-5}$ per year provides an appropriate level of safety.

The role of the PPG in plant-specific implementation of regulatory changes for decommissioning plants will be established as part of the integrated rulemaking. In one possible approach shown in Figure 1, a licensee that fully complies with all IDCs and SDAs (including the seismic checklist) might be permitted to implement changes under the revised rule without a plant-specific analysis and detailed staff review. However, if the licensee/site does not comply with all of the IDCs and SDAs, a plant-specific analysis of SFP risks would be required in order to support relaxations to existing regulations. The PPG could be used to establish an acceptable level of risk in the review of such submittals.

2. CONCLUSIONS

The frequency of events leading to uncovery of the spent fuel must be less than $1x10^{-5}$ per year in order to consider risk-informed changes that could result in the equivalent of a $1x10^{-6}$ per year increase in LERF. Based upon the above comparisons, the staff believes that the LERF-based pool performance criteria of $1x10^{-5}$ per year is reasonable and appropriate. This is supported by the comparisons that show that the conditional health effects for SFP fires may be in the range of health effects considered for severe accidents in operating reactors, and that the Commission's QHOs continue to be met for SFP fires even if the ruthenium release fraction is substantially increased. Given these observations, there does not appear to be sufficient justification to revise the proposed pool performance guideline of $1x10^{-5}$ per year which was developed from the RG 1.174 LERF considerations.

Although the above comparisons focus on the Surry site, the results are expected to be generally applicable to other sites as well. The QHOs represent risk to the average individual within 1 mile and 10 miles of the plant, and should be relatively insensitive to the site-specific population.

It should also be acknowledged that long term health impacts are sensitive to public policy decisions such as land interdiction criteria for returning populations. The long term protective assumption used in both the NUREG-1150 and SFP studies was to interdict land which could give a projected dose to an individual via the groundshine and resuspension inhalation pathways of more than 4 rem in 5 years (2 rem in the first year and 0.5 rem per year for the next 4 years, for an average of 800 mrem per year). Comparisons of consequence results at various distances for each of the NUREG-1150 reference plants are provided in NUREG/CR-6349, and clearly show that the increase in population dose with distance is because of a large number of people receiving very small doses, below the assumed long-term interdiction limit of 4 rem in 5 years, since the offsite consequences because of land condemnation, etc., remain essentially the same over the range of distances. The effect of varying long-term interdiction dose limits on population doses, latent fatalities, and offsite costs was estimated in NUREG/CR-6349 by recalculating the consequences for each of the NUREG-1150 plants for various lower limits. The results show that as the interdiction limit is reduced, the population dose and latent cancers decrease and the offsite costs progressively increase. For a reduction in the interdiction limit from 800 mrem per year to 300 mrem per year the risk measures decreased by typically 20 to 30 percent, and offsite costs increased by about a factor of two. Thus, changes in risk results on this order can be expected as a result of public policy decisions.

3. REFERENCES

1. Safety Goals for the Operations of Nuclear Power; Policy Statement, 51 Federal Register 28044, August 4, 1986.

2. Regulatory Guide 1.174, “An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” July 1998.

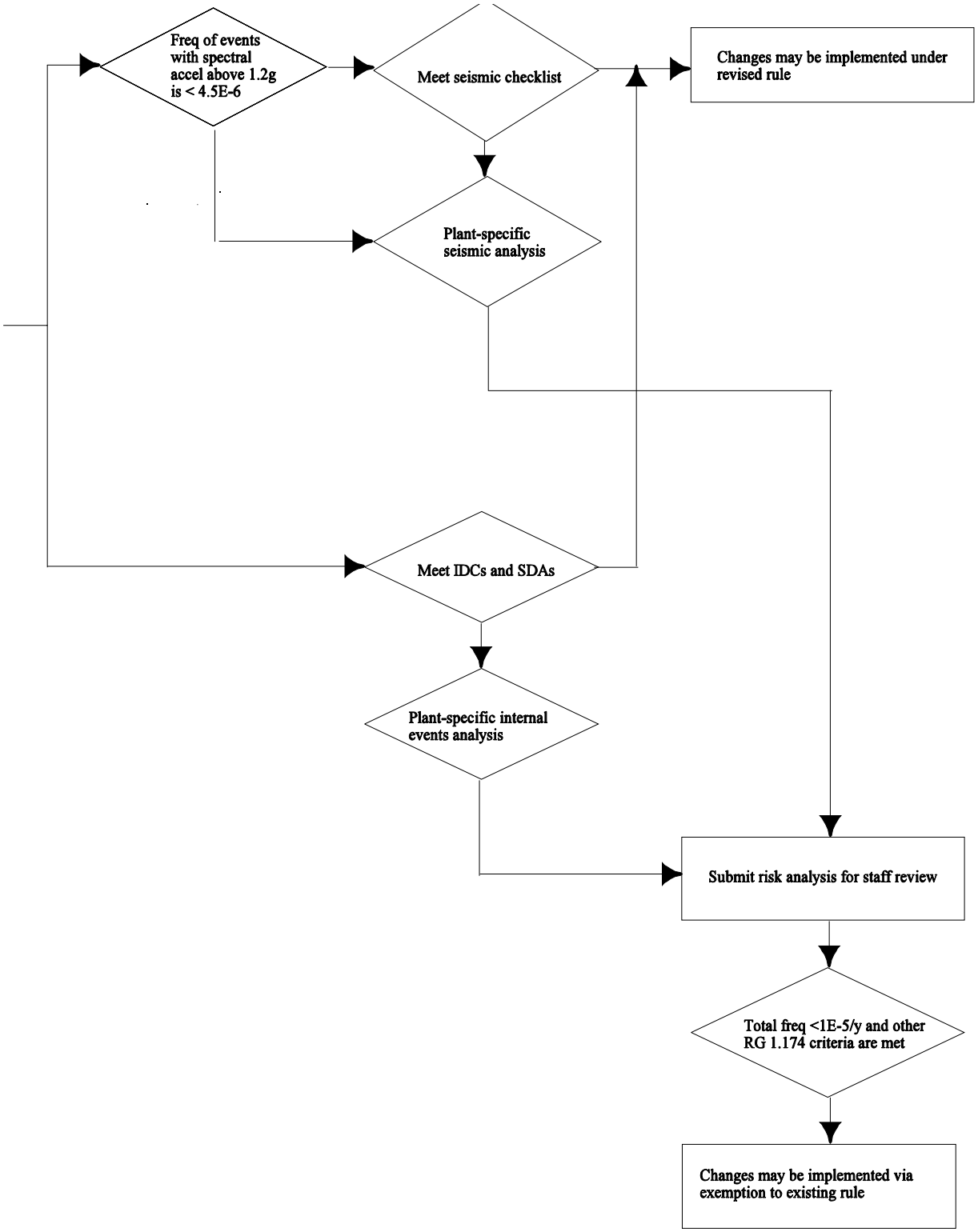
3. U.S. Nuclear Regulatory Commission, *Evaluation of Severe Accident Risks: Surry Unit 1*, NUREG/CR-4551, Vol. 3, Rev. 1, Part 1, Sandia National Laboratory, October 1990.

4. U.S. Nuclear Regulatory Commission, *MACCS Version 1.5.11.1: A Maintenance Release of the Code*, NUREG/CR-6059, Sandia National Laboratory, October 1993.

Table 1 - Comparison of Spent Fuel Pool Accident Risk One Year After Shutdown with Quantitative Health Objectives (QHOs)

Case	QHO for Individual Risk of Prompt Fatality					QHO for Individual Risk of Latent Cancer Fatality				
	Ind. Early Fatality Risk (per event)	PPG (events per year)	Prob of Early Fatality (per year)	QHO (per year)	% of QHO	Ind. Latent C. Fatality Risk (per event)	PPG (events per year)	Prob of Latent C. Fatality (per year)	QHO (per year)	% of QHO
Low Ruthenium Source Term, Early Evacuation	5.44×10^{-4}	1×10^{-5}	5.44×10^{-9}	5×10^{-7}	1	9.09×10^{-4}	1×10^{-5}	9.09×10^{-9}	2×10^{-6}	<1
Low Ruthenium Source Term, Late Evacuation	7.13×10^{-3}	1×10^{-5}	7.13×10^{-8}	5×10^{-7}	14	1.68×10^{-2}	1×10^{-5}	1.68×10^{-7}	2×10^{-6}	8
High Ruthenium Source Term, Early Evacuation	1.50×10^{-3}	1×10^{-5}	1.50×10^{-8}	5×10^{-7}	3	4.33×10^{-3}	1×10^{-5}	4.33×10^{-8}	2×10^{-6}	2
High Ruthenium Source Term, Late Evacuation	3.46×10^{-2}	1×10^{-5}	3.46×10^{-7}	5×10^{-7}	69	8.49×10^{-2}	1×10^{-5}	8.49×10^{-7}	2×10^{-6}	42
Worst Source Term in App. 4A, Late Evacuation	3.66×10^{-2}	1×10^{-5}	3.66×10^{-7}	5×10^{-7}	73	5.16×10^{-2}	1×10^{-5}	5.16×10^{-7}	2×10^{-6}	26

Figure 1 - Use of the PPG in Review of SFP Risk Submittals



APPENDIX 4D
CHANGE IN RISK ASSOCIATED WITH EP RELAXATIONS

Regulatory Guide (RG)1.174 provides guidance on the allowable increase in the frequency of large early release associated with a proposed change to the licensing basis. In accordance with RG 1.174, if the baseline LERF is less than $1x10^{-5}$ per year, plant changes can be approved that increase LERF by up to $1x10^{-6}$ per year. Relaxations in emergency planning (EP) requirements do not impact the frequency of events involving a large early release (i.e., SFP fire frequency) but instead could increase the consequences associated with the large release. Hence, in applying the ΔLERF concept to plant changes that impact consequences it is necessary to translate the allowable increase in LERF into an allowable increase in risk.

The risk increase associated with a ΔLERF of $1x10^{-6}$ per year can be bounded by considering the consequences for a worst case large early release sequence, in conjunction with the maximum allowable frequency increase (i.e., $1x10^{-6}$ per year). This approach provides an upper limit on the increase in risk that might be approved in accordance with RG 1.174 principle of permitting only small increases in risk. The allowable increase in risk will be plant specific since the allowable increase in LERF of $1x10^{-6}$ per year applies to all sites irrespective of such factors as population and meteorology. However, risk-significant differences between sites will tend to similarly impact both the SFP and reactor accident consequences. Hence, the comparisons of SFP risks to the allowable risk increases derived for Surry should be generally applicable to other sites.

The consequences associated with the source term that produced the greatest number of early fatalities in the NUREG-1150 study for Surry are provided in Table 1 below. The consequences are reported separately for internal events and seismic events. The risk measures reported for seismic events are based on the LLNL hazard curve and are about an order or magnitude more severe than those based on the EPRI hazard curve. The maximum allowable level of risk increase is the product of the consequences (in this case, the consequences for the worst seismic event since it is bounding) and the allowable frequency increase of $1x10^{-6}$ per year. This risk increase is provided in the last column of Table 1.

It should be noted that the Commission's Quantitative Health Objectives (QHOs) correspond to an individual early fatality risk of $5x10^{-7}$ per year and an individual latent cancer fatality risk of $2x10^{-6}$ per year. Thus, the risk increase values inferred from RG 1.174 for individual early fatality risk ($8.7x10^{-8}$ per year) and individual latent cancer fatality risk ($6.9x10^{-8}$ per year) represent about 17 percent and 4 percent of these QHOs, respectively. This margin reflects the strategy taken in establishing the acceptance guidelines for risk increase in RG 1.174. Specifically, in RG 1.174 the NRC adopted more restrictive acceptance guidelines than might be derived directly from the Commission's Safety Goal Policy Statement. This policy was adopted to account for uncertainties and for the fact that safety issues continue to emerge regarding design, construction, and operational matters.

Table 2 summarizes the bases for evacuation modeling for each of the major contributors to SFP fires. The effectiveness of EP was characterized is such a way to maximize the value of formal EP in the "full EP" case and minimize the value of ad hoc EP in the "relaxed radiological preplanning" case. As such, the resulting estimates of the risk increase associated with EP relaxations represent an upper bound on the potential risk increase.

The consequences associated with each of the events leading to SFP fires are provided in Table 3 for the "full EP" case and "relaxed radiological preplanning" case. The consequences are based on results of calculations reported in Appendix 4A. In several cases where MACCS2 runs were not available, the results for the closest corresponding calculation were used as an approximation. The risk increase associated with the EP relaxation is the product

of the event frequency and the change in consequences, summed over all contributors.

The sensitivity of the risk increase estimates is strongly dependent on the assumptions regarding the effectiveness of emergency evacuation in seismic events, since these events dominate the SFP fire frequency. In NUREG-1150, evacuation in seismic events was treated either of two ways depending on the peak ground acceleration (PGA) of the earthquake:

- for low PGA earthquakes (<0.6g), the population was assumed to evacuate however the evacuation was assumed to start later and proceed more slowly than evacuation for internally-initiated events. A delay time of 1.5 times the normal delay time and an evacuation speed of 0.5 times the normal evacuation speed was assumed for this case.

- for high PGA earthquakes (>0.6g), it was assumed that there would be no effective evacuation and that many structures would be uninhabitable. The population in the emergency response zone was modeled as being outdoors for the first 24 hours, and then relocating at 24 hours.

Since the SFP fire frequency is driven by seismic events with PGA several times larger than the SSE, the reasoning that there would be no effective evacuation was adopted in developing the baseline estimate of the risk. This is consistent with the expert opinion provided in Attachment 2 to Appendix 2B regarding the expected level of collateral damage within the Emergency Planning Zone given a seismic event large enough to fail the SFP. Specifically, for ground motion levels that correspond to SFP failure in the Central and Eastern U.S., it is expected that electrical power would be lost and more than half of the bridges and buildings (including those housing communication systems and emergency response equipment) would be unsafe even for temporary use within at least 10 miles of the plant. This reasoning is also consistent with previous Commission rulings on San Onofre and Diablo Canyon in which the Commission found that for those risk-dominant earthquakes that cause very severe damage to both the plant and the offsite area, emergency response would have marginal benefit because of its impairment by offsite damage.

This same reasoning is applied to the full EP and the relaxed EP cases. The net effect is that EP, as well as relaxations in EP, do not impact the risk associated with seismic events that result in SFP failure. A sensitivity case was also performed to explore the impact on risk increase if the seismic event only partially degrades the emergency response, as discussed below.

In the sensitivity case, it was assumed that evacuation would be carried out consistent with the NUREG-1150 model for low g earthquakes if current EP requirements are maintained, i.e., the population evacuates, but the evacuation delay time is increased by 50 percent and the time to complete the evacuation is doubled. This is extremely optimistic given the damage to communication and notification systems, buildings and structures, and roads that would accompany any seismic event severe enough to fail the SFP. With no preplanning for radiological accidents, the evacuation delay time was further increased to three times the normal delay time.

For purposes of assigning consequences in the seismic sensitivity case, the “full EP” case was represented by the results from the early evacuation case (i.e., evacuation is started and completed before the release) and the “relaxed preplanning for radiological accidents” case was represented by the results from the late evacuation case (i.e., evacuation is not started until after the release has occurred). This maximizes the effectiveness of evacuation in the full EP case and minimizes its effectiveness in the relaxed preplanning case, thereby tending to maximize the risk increase associated with EP relaxations.

The estimated risk increases associated with the EP relaxation are summarized in Table 4.

The results indicate that relaxation of the requirements for radiological preplanning would result in an increase of about 1.5×10^{-5} early fatalities per year and 2 person-rem per year, which is about a factor of 15 and five below the allowable increase inferred from the RG 1.174 LERF criteria. The other risk measures are also substantially lower than the allowables from RG 1.174. Since the SFP fire frequency assumed in these comparisons (2.4×10^{-6} per year) is about a factor of four lower than the PPG of 1×10^{-5} per year, a plant operating nominally at the PPG would have a smaller margin to the allowable risk limits for the reference plant but would still be at or below the limits under the above assumptions.

The results of the sensitivity studies indicate that even under unrealistically optimistic assumptions regarding the value of EP in seismic events, the change in risk associated with relaxation of the requirements for radiological preplanning is still relatively small. The increases in early fatalities and individual early fatality risk remain below the maximum allowable for each risk measure. Population dose and individual latent cancer fatality risk are about a factor of two higher than the allowable value inferred from RG 1.174. However, this increase in individual latent cancer risk represents only about 9 percent of the QHO, and considerable margin to the QHO would still remain.

It must be kept in mind that the evacuation effectiveness assumed for “Full EP” in the sensitivity case is unrealistic for high g earthquakes, and that the risk increase associated with the EP relaxations would be closer to the baseline value. Also, the risk reduction estimates are based on the LLNL seismic hazard frequencies and the high ruthenium source term, and would be substantially lower if either the EPRI seismic hazard frequencies or the low ruthenium source term were used. Finally, the above comparisons are based on the risk levels 1 year after shutdown.

The impact of the above factors on the maximum risk increase for the EP relaxations is shown in Figures 1 and 2 for early fatalities and population dose (person-rem). Use of either the EPRI seismic hazard frequencies or the low ruthenium source term would reduce each of the risk measures by about a factor of 10, to values which are well below the RG 1.174 allowables. The risk impact will decrease in later years because of reduced consequences as fission products decay further.

Table 1 - Guideline Level of Risk Increase In Accordance With RG 1.174 ΔLERF Criterion (Based on Surry)

Risk Measure	Consequences -- conditional upon source term that produces greatest early fatalities (per event)		Guideline frequency increase in accordance with RG 1.174 (events per year)	Guideline risk increase (per year)
	Internal Events	Seismic Events		
Early fatalities	15	250	$1x10^{-6}$	$2.5x10^{-4}$
Population dose (p-rem within 50 miles)	$3.6x10^{6}$	$1.1x10^{7}$	$1x10^{-6}$	11
Individual early fatality risk at 1 mile	$2.9x10^{-2}$	$8.7x10^{-2}$	$1x10^{-6}$	$8.7x10^{-8}$
Individual latent cancer fatality risk at 10 mile [1]	$5.5x10^{-3}$	$6.9x10^{-2}$	$1x10^{-6}$	$6.9x10^{-8}$

1 - Values shown include a factor of three adjustment to account for differences in the cancer risk model used for NUREG-1150 and SFP accident calculations

Table 2 - Evacuation Modeling for Major Contributors to SFP Fires

Event Type	Major Contributor	Freq (per year)	Minimum Time to Release at One Year (h)	Timely Notification of Off-Site Authorities?	Intact Infrastructure for Emergency Response?	Evacuation Model	
						Full EP	Relaxed Preplanning for Radiological Accidents
Boildown	LOOP (severe weather)	1.8×10^{-7}	>200	No	Yes	Late	Late
Rapid Draindown	Cask Drop	2.0×10^{-7}	~10	Yes	Yes	Early	Late
	Seismic [1]	2.0×10^{-6}	~10	Yes	No	No evacuation Relocation at 24 h	No evacuation Relocation at 24 h
	Seismic Sensitivity [2]					1.5x normal delay 0.5x normal speed (Model as Early)	3x normal delay 0.5x normal speed (Model as Late)

1 - Evacuation model for full EP case is consistent with NUREG-1150 assumptions for high acceleration earthquakes
2 - Evacuation model for full EP case is consistent with NUREG-1150 assumptions for low acceleration earthquakes

Table 3 - Estimated Risk Increase Associated With Relaxing EP Requirements at SFP Facility (at one year)

Event	Freq (per year)	Consequences Per Event with Full EP				Consequences Per Event with Relaxed Preplanning for Radiological Accidents				ΔRisk Per Year from EP Relaxation			
		EF	p-rem	Ind Risk of EF	Ind Risk of LCF	EF	p-rem	Ind Risk of EF	Ind Risk of LCF	ΔEF	Δp-rem	ΔInd Risk of EF	ΔInd Risk of LCF
Boildown[1]	$1.8x10^{-7}$	See Note 1				See Note 1				0	0	0	0
Cask Drop	$2.0x10^{-7}$	0.95	$1.1x10^{7}$	$1.50x10^{-3}$	$4.33x10^{-3}$	77	$1.9x10^{7}$	$3.46x10^{-2}$	$8.49x10^{-2}$	$1.5x10^{-5}$	1.6	$6.6x10^{-9}$	$1.6x10^{-8}$
Seismic[2]	$2.0x10^{-6}$	See Note 2				See Note 2				0	0	0	0
Total	$2.4x10^{-6}$									$1.5x10^{-5}$	1.6	$6.6x10^{-9}$	$1.6x10^{-8}$
Seismic Sensitivity	$2.0x10^{-6}$	0.95	1.1x107	$1.50x10^{-3}$	$4.33x10^{-3}$	77	$1.9x10^{7}$	$3.46x10^{-2}$	$8.49x10^{-2}$	$1.5x10^{-4}$	16	$6.6x10^{-8}$	$1.6x10^{-7}$

1 - Risk results with and without EP would be comparable for boildown sequences since the failure paths in these sequences involve failures to notify offsite authorities and would not be impacted by EP

2 - Risk results with and without EP would be comparable for large seismic events since emergency response would have marginal benefit because of its impairment by offsite damage

Table 4 - Comparison of Risk Increase with RG 1.174 Guideline (at one year)

Risk Measure	Risk Increase Because of EP Relaxation (per year)		RG 1.174 Guideline Risk Increase (per year)
	Baseline [1]	Seismic Sensitivity [2]	
Early Fatalities	1.5×10^{-5}	1.6×10^{-4}	2.5×10^{-4}
Population Dose	1.6	17.6	11
Individual Early Fatality Risk	6.6×10^{-9}	7.3×10^{-8}	8.7×10^{-8}
Individual Latent Cancer Fatality Risk	1.6×10^{-8}	1.8×10^{-7}	6.9×10^{-8}

1 - Assumes no effective evacuation in seismic events, regardless of pre-planning
2 - Assumes maximum effectiveness of emergency planning (i.e., early evacuation) when EP requirements are maintained, and minimum effectiveness (i.e., late evacuation) when EP requirements are relaxed

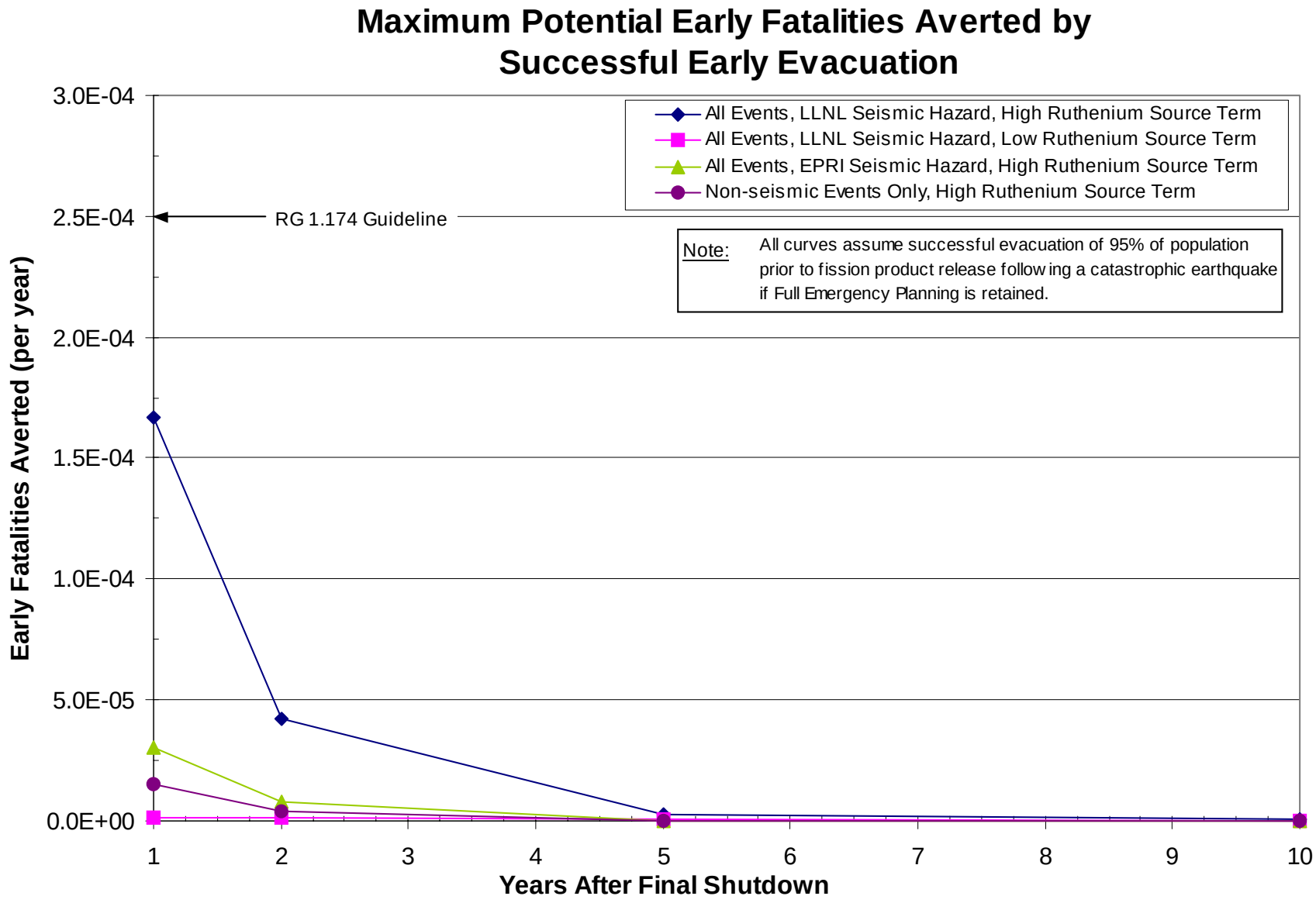

Maximum Potential Early Fatalities Averted by Successful Early Evacuation
Early Fatalities Averted (per year)
3.0E-04
2.5E-04
2.0E-04
1.5E-04
1.0E-04
5.0E-05
0.0E+00
1
2
3
4
5
6
7
8
9
10
Years After Final Shutdown
All Events, LLNL Seismic Hazard, High Ruthenium Source Term
All Events, LLNL Seismic Hazard, Low Ruthenium Source Term
All Events, EPRI Seismic Hazard, High Ruthenium Source Term
Non-seismic Events Only, High Ruthenium Source Term
RG 1.174 Guideline
Note: All curves assume successful evacuation of 95% of population prior to fission product release following a catastrophic earthquake if Full Emergency Planning is retained.

Figure 4D-1

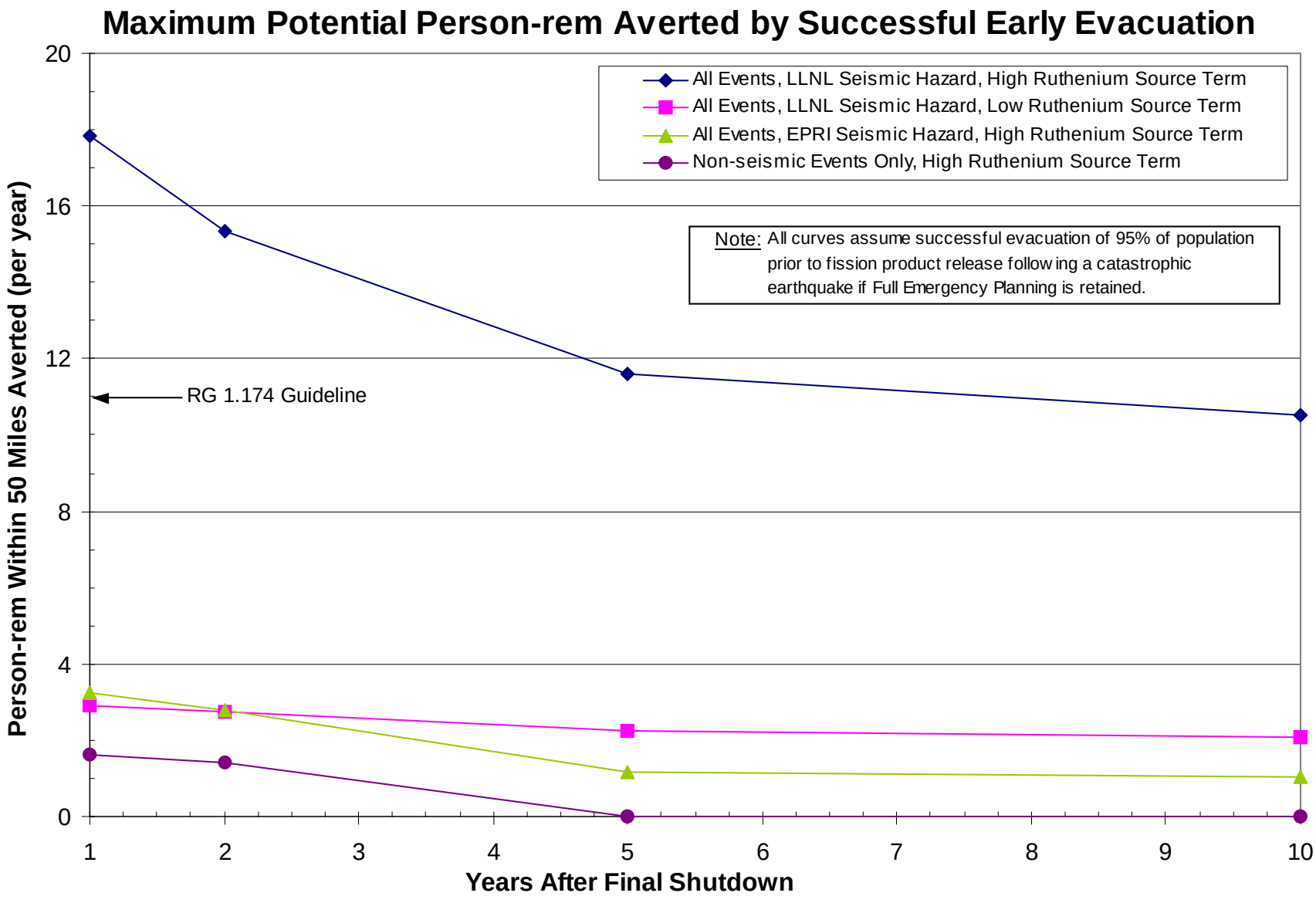

Maximum Potential Person-rem Averted by Successful Early Evacuation
Person-rem Within 50 Miles Averted (per year)
Years After Final Shutdown
20
16
12
8
4
0
1
2
3
4
5
6
7
8
9
10
All Events, LLNL Seismic Hazard, High Ruthenium Source Term
All Events, LLNL Seismic Hazard, Low Ruthenium Source Term
All Events, EPRI Seismic Hazard, High Ruthenium Source Term
Non-seismic Events Only, High Ruthenium Source Term
Note: All curves assume successful evacuation of 95% of population prior to fission product release following a catastrophic earthquake if Full Emergency Planning is retained.
RG 1.174 Guideline

Figure 4D-2

APPENDIX 5
NOVEMBER 12, 1999 NUCLEAR ENERGY INSTITUTE COMMITMENT LETTER

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NUCLEAR ENERGY INSTITUTE

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November 12, 1999

Richard J. Barrett
Chief, Probabilistic Safety Assessment Branch
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Barrett,

Industry is committed to performing decommissioning with the same high level of commitment to safety for its workers and the public that was present during operation of the plants. To that end, industry is making several commitments for procedures and equipment which would reduce the probability of spent fuel pool events during decommissioning and would mitigate the consequences of those events while fuel remains in the spent fuel pool. Most of these commitments are already in place in the emergency plans, FSAR requirements, technical specifications or regulatory guidance that decommissioning plants must follow.

These commitments were initially presented at the NRC public workshop on decommissioning, July 15-16, in Gaithersburg, Maryland. They were further discussed in detailed industry comments prepared by Erin Engineering. At a recent public meeting with NRC management it was determined that a letter clearly delineating these commitments could be useful to NRC as it considers input to its technical analyses.

I am hereby transmitting those industry commitments as follows.

1. Cask drop analyses will be performed or single failure proof cranes will be in use for handling of heavy loads (i.e., phase II of NUREG 0612 will be implemented).

2. Procedures and training of personnel will be in place to ensure that on site and off-site resources can be brought to bear during an event. \c)o(

3. Procedures will be in place to establish communication between on site and off-site organizations during severe weather and seismic events.

4. An off-site resource plan will be developed which will include access to portable pumps and emergency power to supplement on site resources. The plan would principally identify organizations or suppliers where off site resources could be obtained in a timely manner.

5. Spent fuel pool instrumentation will include readouts and alarms in the control room (or where personnel are stationed) for spent fuel pool temperature, water level, and area radiation levels.

6. Spent fuel pool boundary seals that could cause leakage leading to fuel uncovery in the event of seal failure shall be self limiting to leakage or otherwise engineered so that drainage cannot occur.

7. Procedures or administrative controls to reduce the likelihood of rapid drain down events will include (1) prohibitions on the use of pumps that lack adequate siphon protection or (2) controls for pump suction and discharge points. The functionality of anti-siphon devices will be periodically verified.

8. An on site restoration plan will be in place to provide repair of the spent fuel pool cooling systems or to provide access for make-up water to the spent fuel pool. The plan will provide for remote alignment of the make-up source to the spent fuel pool without requiring entry to the refuel floor.

9. Procedures will be in place to control spent fuel pool operations that have the potential to rapidly decrease spent fuel pool inventory. These administrative controls may require additional operations or management review, management physical presence for designated operations or administrative limitations such as restrictions on heavy load movements.

10. Routine testing of the alternative fuel pool make-up system components will be performed and administrative controls for equipment out of service will be implemented to provide added assurance that the components would be available, if needed.

If you have any questions regarding industry's commitments, please contact me at 202 739-8109 or LXII@NEI.org.

Sincerely,

Lynnette Hendricks
LXH/1rh

APPENDIX 6
STAKEHOLDER CONCERNS RAISED DURING THE PUBLIC COMMENT PERIOD

On February 15, 2000, the Nuclear Regulatory Commission (NRC) released the "Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Plants," for public comment. The NRC encouraged stakeholders to review the draft study and to formally submit comments for review. Appendix 7 of that report included a list of public meetings and how the staff addressed stakeholder comments received on the draft report, issued June 1999, in various technical areas. After review of the February 2000 study, several public groups commented that it appeared that the NRC did not address some of the public's comments. While all stakeholder comments were considered and many resulted in changes in the study, the staff did not include a discussion for some of the comments in Appendix 7. In order to ensure that adequate consideration had been given to public comments, the staff reviewed comments which had been received prior to February 15, 2000, as well as comments received as a result of a review of the draft final report. Comments received prior to February 15 were identified by reviewing transcripts of publically attended meetings, letters from the public, and other available documentation related to the staff's efforts in completing the draft final report.

This appendix provides the NRC's responses to the comments and concerns received as described above. In most cases, responses are documented in this appendix. However, in other cases, comments or concerns identified in this appendix are referred to other parts of the report where the identified issues are addressed. For cases where similar comments were received by more than one commenters, the comments were combined for one response. The comments are grouped in the following technical categories: Criticality, Consequences, Probability and Human Reliability, Seismic, Security/Safety Culture/EP, Thermal hydraulics, Insurance, and Rulemaking/ NRC Process Concerns.

CRITICALITY

Comment #1: A commenter stated that the potential criticality should be addressed.

Response: The staff agrees. The issue of nuclear criticality is addressed in Section 3.6 and Appendix 3.

Comments #2 and 3: A commenter raised several concerns related to SFP criticality. (a) Can a criticality occur due to chemical stripping of primary piping? (b) During primary system decontamination at decommissioning reactors, is it possible to misalign the valves and send corrosive chemicals into the SFP? Could these chemicals precipitate boron from the SFP water? Is there a potential for criticality? Is there a potential for fuel damage?

Response: The precipitation of boron out of the pool water, due to chemicals or any other means, will not result in criticality because soluble boron is not credited to maintain spent fuel pool subcriticality (k-eff < 1.0). The main connection between the spent fuel pool and primary system is the transfer tube used to transfer fuel for refueling. After a plant ceases to operate, this tube is sealed on both ends with flanges. As a result, there is no communication between the primary system and SFP from this connection. Support systems connected to the SFP vary from one plant to another. At most decommissioning plants, there would be no communication between the SFP and the primary reactor systems, while others may use a primary support water system to add water to the pool. In any event, even if fuel damage did occur, the shielding provided by the large volume of water above the fuel (usually 23 feet of water) would preclude any significant radiation release. In addition, decommissioning activities are performed according to procedures, which reduces the possibility of operator error. For

example, 10 CFR 50.120 requires training and qualification in nine categories of personnel involved with spent fuel pool maintenance and support.

Comment #4: During primary system decontamination, can contaminated solution go "overboard" and into public waters?

Response: Main plant buildings have a drain system in the event of liquid spills to prevent possible contamination of public waters or land. Additionally, the licensee performs decommissioning activities using procedures and training, which reduces the likelihood and consequences of such occurrences.

Comment #5: The NRC should identify the scenario where a steam explosion is possible because of a severe criticality event and the basis upon which the probability was determined to be "highly unlikely."

Response: The staff did not construct a sequence of events that would lead to a steam explosion. The discussion in the paper (Appendix 3) indicates that the likelihood of any criticality event is low. The likelihood of a super-prompt critical event is even lower, particularly in view of the inherent negative feedback provided by fuel temperature increases.

Comment #6: The NRC should identify all radioactivity in the SFP and is capable of being dispersed in an accident (beyond that on p A3-11 to A3-13).

Response: In Appendix 3, the staff addresses nuclides that contribute to the reactivity of the spent fuel and does not relate to the generation of the source term. The nuclides listed there represent well over 90 percent of the reactivity contribution in spent fuel. Therefore, it is not necessary to expand the list because such an expansion will not significantly alter the predicted reactivity of the spent fuel in the storage racks. Similarly, nuclides used for the source term, which is addressed in Appendix 4, represent the dominant contributors to public and worker dose. The results would not vary significantly if all of the radionuclides were included.

Comment #7: The criticality accident analysis does not consider the risk of a criticality accident that arises from placement of low-burnup fuel assemblies in a pool where the licensee relies on burnup credit to prevent criticality.

Response: The scenario suggested by the commenter would only occur as a misloading event in a pressurized water reactor spent fuel pool. In order to meet General Design Criterion 62, licensees analyze a misloading event in the spent fuel pool to demonstrate that the fuel remains subcritical even for the misloading of a fuel assembly of highest reactivity worth (i.e., a fresh, unirradiated fuel assembly) into a region designed to accommodate lower worth assemblies (i.e., highly irradiated fuel). Further, the staff demonstrated via analysis (affidavit of A. Ulses in hearing before the Atomic Safety and Licensing Board, ASLBP No. 99-762-02-LA, January 4, 2000) that there was sufficient soluble boron in a pressurized water reactor spent fuel pool such that even an inadvertent misloading of a complete rack of fresh assemblies would not lead to a criticality event. Therefore, the staff did not give further consideration to this scenario.

CONSEQUENCES

Comment #8: Is a gap release considered to give moderate off-site consequences at the time when a zirconium fire is no longer a threat?

Response: As time increases since permanent shutdown, the fission product inventory available for release gets smaller and the decay power decreases. As a result, there may not be sufficient energy to carry released fission products out of the spent fuel pool and offsite. NUREG/CR-4982, *Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82*, July 1987, provides societal doses for SFP accidents involving a fuel melt release and a gap release at one year after final shutdown. These societal doses, which are for the population within 50 miles, are $3x10^6$ rem and 4 rem for a fuel melt release and a gap release. A gap release is expected to give negligible off-site radiological consequences at this time. This study did not calculate the consequences of a gap release.

Comment #9: The draft study does not explain the regulatory basis for using 4 rem over 5 years as the threshold dose for relocation.

Response: The Environmental Protection Agency (EPA) report 400-R-92-001, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*, May 1992, states that, after the early phase of a nuclear incident, protective actions should be taken to limit the dose received by an individual to 2 rem in the first year, 0.5 rem/year after the first year, and 5 rem over 50 years. These Protective Action Guides are implemented in the MACCS code for relocation criteria by limiting the dose to 4 rem over 5 years, that is, 2 rem in the first year plus 0.5 rem for each of the second through fifth years.

PROBABILITY AND HUMAN RELIABILITY ASSESSMENT

Comment #10: Experience at nuclear power plants demonstrates that safety problems are not caused by workers making mistakes or by not following procedures. Problems are caused by bad management.

Response: The staff agrees that utility safety culture and utility oversight/expectations in the day-to-day operations of a facility are important contributors to either a well run plant or a poorly run one. The staff decommissioning assumptions and industry commitments will help insure that proper attention is given to spent fuel pool status, procedures are developed that guide fuel handlers in the event of a spent fuel pool accident, communications are established between on-site and off-site organizations, and cask drop analyses are performed or a single failure proof crane is used for handling very heavy loads. These staff assumptions and industry commitments are discussed in Sections 3 and 4.

Comment #11: Experience at nuclear power plants shows that multiple shifts can make the same error and not recognize it for a long time. With watching the pool being their major responsibility, a fuel handler's life would be very tedious and boredom would set in. This should result in a poorer response by the fuel handler in the event of an accident. An example of this is the recent Browns Ferry event.

Response: The NRC, through the "Policy on Factors Causing Fatigue of Operating Personnel at Nuclear Reactors" provides guidelines on working hours that were consistent with the objective of ensuring that the mental alertness and decision-making abilities of plant staff were not significantly degraded by fatigue. The staff shares the commenter's concern that operator boredom and their ability to maintain alertness while standing watch may contribute to fatigue-induced impairment of personnel and thereby increase the likelihood of personnel errors. For this study, our modeling and quantification of SFP risk includes consideration of multiple shift turnovers and the chance that shift after shift makes the same mistake. However, for almost all postulated SFP accidents, there is a very long time available to the fuel handlers to discover and recover from the existence of a problem in the spent fuel pool or its support systems. The staff believes that the commitments made by the industry and the NRC's staff decommissioning assumptions provide a basis for reducing the chances of multiple shift errors to the point where they do not contribute significantly to the overall risk of spent fuel pool operation (See Sections 3 and 4). Other accidents (i.e., seismic and heavy load drop), which progress rapidly, are assumed to proceed independent of operator intervention once the accident has occurred because the SFP is assumed to drain very rapidly.

Comment #12: Over time, tedious tasks will cause workers to make mistakes. The NRC needs to address this in a conservative manner.

Response: The staff agrees that tedious tasks can increase the chances of a fuel handler making a careless mistake. The NRC, through the "Policy on Factors Causing Fatigue of Operating Personnel at Nuclear Reactors" provides guidelines on working hours that were consistent with the objective of ensuring that the mental alertness and decision-making abilities of plant staff were not significantly degraded by fatigue. However, we do not agree that fuel handler errors need be modeled in a conservative manner when performing a probabilistic risk assessment. It is the NRC's policy to make its risk assessments as realistic as possible. The staff performed the analysis for this report consistent with the agency's policy.

Comment #13: How is common mode failure accounted for in the staff's risk analysis? How confident are you of your ability to model and quantify common mode failures?

Response: The staff's risk analysis accounts for dependencies among the initiating events, the equipment needed to mitigate the events, and also the operator actions needed for accident mitigation. Initiating events that have the potential of simultaneously degrading mitigating equipment or impeding operator actions are modeled in the construction of the event trees and in the estimation of equipment failure rates and human failure probabilities. For example, for an event where a fire is not extinguished within 20 minutes, it was assumed that the SFP cooling system and the electric-driven firewater pumps are failed (either due to fire damage or due to loss of the electrical supply to the plant). Therefore, no credit is taken for this equipment. In addition, the estimation of the human error probability (for starting backup diesel pumps or for off-site recovery) took into account a high level of operator stress, which increases the failure probability.

Equipment hardware failure dependencies, usually referred to as common cause failures, have also been modeled in the risk analysis. Since these failures have the potential for disabling multiple trains of equipment at the same time, they can be large contributors to the risk. In the staff's analysis, the only multiple train system modeled is the spent fuel pool cooling system. In the fault tree model for this system, common cause failures are modeled for the cooling pumps, the heat exchangers, and the discharge check valves. The modeling of dependent failures, including common-cause hardware failures, in the staff's risk analysis is consistent with NRC and industry guidelines. Based on the above, the staff has confidence that its modeling and quantification of common mode failures is adequate.

Comment #14: NRC should set guidelines on how often fuel handlers make their rounds at decommissioning facilities. This would help assure operator attentiveness.

Response: The staff agrees that, if fuel handlers make the rounds of the SFP and its equipment on a frequent basis, the probability of the handlers detecting problems early is greatly enhanced. To this end, SDA #2 states in part that walk-downs of the SFP systems will be performed at least once per shift by the fuel handlers. The staff expects that these assumptions will be translated into requirements or industry guidance during the rulemaking process.

Comment #15: NRC should assure that the probability of failure of systems required to mitigate the consequences of design bases and beyond design bases spent fuel pool events are minimized.

Response: The need to have highly reliable systems to prevent or mitigate an accident is partly a function of how rapidly the accident progresses and how serious its consequences are. If an accident would result in serious consequences unless a rapid response were achieved, then highly reliable systems and components are needed to prevent and/or mitigate the event. If the accident is very slow in progressing or has benign consequences, the equipment designed to prevent or mitigate it need not be as reliable. For SFPs at decommissioning plants, the large volume of water above the spent fuel provides an inherent delay time before fuel can be uncovered, except for two potential beyond design basis accidents which are discussed later. This delay time (measured in days) allows for repair or replacement of equipment. If it were impossible to repair or replace the equipment, inventory could be added to the pool to match the boil-off rate. The industry has committed in IDC #4 to implement an off-site resource plan to include access to portable pumps and emergency power. IDC #7 and IDC #9 commit the industry to implement procedures or administrative controls to reduce the likelihood of rapid

draindown events. SDA #2 calls for procedures to be developed that will provide guidance on the availability of on-site and off-site inventory make-up sources and time available to initiate these sources. In addition, the industry has committed in IDC #10 to perform routine testing of the alternative spent fuel pool make-up system components and to have procedural controls on equipment out of service to increase confidence that components will be available. The two accidents that could lead to very rapid draining of the SFP are extremely large seismic events and heavy load drops. IDC #1 and SDA #5 address heavy load drop concerns. SDA #6 calls for each decommissioning plant to successfully complete the seismic checklist provided in Appendix 2b to this report. Implementation of these commitments and assumptions will help assure the frequency of a zirconium fire remains within the assumptions of the analysis.

Comment #16: Is station blackout at a decommissioning site acceptable to the staff?

Response: As with an operating reactor, the staff recognizes that there is some small annual probability that a station blackout will occur at a decommissioning site. For both an operating and decommissioning plant, there is a need to recover from a station blackout. However, unlike an operating reactor, decommissioning SFPs can go without electrical power for a long period of time (days to weeks) and not suffer safety consequences. This is due to the inherent margin provided by the large volume of water sitting above the spent fuel in the pool. It takes a long time to heat this water up to boiling and then to continue to boil it off until fuel is uncovered (almost a week at one year after the last fuel was irradiated in the reactor). IDC #2 commits the industry to develop procedures and train personnel to ensure that on-site and off-site resources can be brought to bear during an event. IDC #3 calls for communication systems to be set up between the SFP site and off-site resources that can survive severe weather and seismic events, which can cause a station blackout.

Comment #17: The risk assessment should take into account changes in local aircraft traffic when evaluating the probability and consequences from aircraft crashing into SFPs.

Response: The risk from aircraft crashes is small, and even large increases in traffic should not make aircraft crashes a dominant contributor to risk. A decommissioning plant will continue to be governed by 10 CFR Part 50 for the evaluation of hazards as discussed in Standard Review Plan 2.2.3, “Evaluation of Potential Accidents,” including accidents involving nearby industrial, military, and transportation facilities.

The frequency of an aircraft crash leading to an accident in a spent fuel pool was estimated in the report to be in the range of 4.3×10^{-8} to 9.6×10^{-12} per year where damage to the pool was significant enough that it resulted in a rapid loss of water from the pool (See Section 3 and Appendix 2d). The mean value was estimated to be 2.9×10^{-9} per year. These values are a small fraction of the overall risk of uncovering the spent fuel in the pool at a decommissioned plant. An aircraft crash could also result in damage to a SFP support system. The estimated range of striking a support system was estimated to be in the range of 1.0×10^{-5} to 1.0×10^{-9} per year, with a mean value of 7.0×10^{-7} per year, without consideration of recovery actions. These values are also a small fraction of the estimated frequencies for the loss of cooling initiator (3.0×10^{-3} per year), the internal fire initiator (3.0×10^{-3} per year), or the loss of inventory initiator (1.0×10^{-3} per year).

Aircraft traffic and accident data were reviewed by the staff (Ref: “Data Development Technical Support Document for the Aircraft Crash Risk Analysis Methodology (ACRAM) Standard,” C.Y. Kimura, et al., UCRL-ID-124837, Lawrence Livermore National Laboratory, August 1, 1996). The number of U.S. air carrier operations increased from about 5.5 million departures per year in the 1970s to about 8.7 million departures per year in the mid-1990s. The average

miles traveled per departure increased from about 500 to 650. For the period from 1986 to 1993 general aviation operations remained relatively constant, with a decrease in activities reported in 1992 and 1993. Military aircraft data, which are a small fraction of the total risk (see Appendix 2D, Table A2d-1, "Generic Aircraft Data"), was not reviewed.

Comments #18 and 19: (A) What is the generic frequency of events leading to zirconium fires at decommissioning plants before the implementation of industry commitments and staff assumptions? (B) This question is relevant to operating plants.

Response: Risk assessments are performed as realistic as possible. As such, the analysis for this study reflects practices already in place. The staff visited four decommissioning sites as part of the preparation for developing the risk assessment of decommissioning spent fuel pools. The insights from those visits include that the facilities appeared to have been staffed by well trained and knowledgeable individuals with significant nuclear power plant experience. Procedures were in place for dealing with routine losses of inventory. Fuel handlers appeared to know whom to contact off-site if difficulties arose with the SFP. The staff recognized that these attributes were not required by NRC regulations nor suggested in NRC guidance for decommissioning sites. The IDCs and SDAs are an attempt to increase the assurance that plant personnel will continue to be knowledgeable of off-site resources and have good procedures available to them.

This study does not reflect the risk at operating plants. As with the practices discussed above, this study reflects the support systems and staffing generally found at decommissioning plants, which are different than at operating plants. For example, the spent fuel pool cooling and makeup systems at decommissioning plants are generally replacing smaller capacity systems to match the reduced decay heat level of the spent fuel. The staff believes that a direct comparison of this risk study on decommissioning plants can not be made to operating plants. However, the staff is sensitive to possible implications to operating plants.

Comment #20: There are several places in the draft report where the staff refers to "uncovering the core" rather than "uncovering the fuel."

Response: The phrase "uncovering the core" has been replaced by "uncovering the fuel."

Comment #21: Recalculating the frequencies for event trees produced numerical results for some sequences that were off by one or two orders of magnitude.

Response: In the staff's risk analysis, the accident scenario frequencies in the event trees were calculated such that dependencies among the failure events (in the event tree branches) were taken into account. Therefore, if an event resulted in functional failure in more than one branch in the event tree, this dependency was taken into account, and the resultant scenario frequency is therefore larger (in some cases, by as much as two orders of magnitude) than if the events were assumed to be independent.

Comment #22: The initiating frequencies, human error rates, and equipment failure rates should more accurately take into account the occurrence of actual events such as Chernobyl and Three Mile Island.

Response: The decommissioning SFP risk assessment takes into account actual events that are applicable to spent fuel pools and their support systems. The staff used initiating event

frequencies from staff studies from actual events at spent fuel pools, actual crane lift data, site-specific seismic hazard curves, studies on aircraft crashes and tornadoes, and large databases developed to provide estimates for initiating events and equipment failure rates. Human error rates were developed by the staff in conjunction with experts at Idaho National Engineering and Environmental Laboratory. The staff believes that the values used in the report provide a reasonable picture of the risks associated with operation of decommissioning spent fuel pools under the assumptions and commitments documented in the study.

Comment #23: The NRC should determine which failure rates used in the report are reliable and which are not, and the results should be included in the study.

Response: The staff uses the most reliable information on failure rates that is available. Because of the long time it takes for water above the spent fuel to heat up and boil off, the failure rates of specific equipment that support a spent fuel pool are not important contributors to spent fuel pool risk for long term sequences (i.e., the results are not particularly sensitive to the assumed failure rate of equipment.) However, very large seismic events or heavy load drops could rapidly drain the spent fuel pool. For seismic events, the robustness of the spent fuel pool is assured by implementation of a seismic checklist (See Appendix 2b). For heavy load drops, IDC #1 and SDA #5 calls for performance of cask drop analyses/load drop consequence analysis or use of a single-failure-proof crane when moving heavy loads over or near the spent fuel pool, which should help assure that the risk from heavy load drops is extremely low.

Comment #24: Mitigating systems at decommissioning spent fuel pools are not automatic. The NRC should assure that fuel handlers are available in the event of an accident.

Response: The decommissioning rulemaking plan will address operator staffing requirements and safeguards staffing for facilities undergoing decommissioning. Staffing at present day decommissioning sites is controlled by Technical Specifications on a plant-specific basis. In addition, SDA #2 calls for walkdowns of the spent fuel pool area by fuel handlers every shift.

Comment #25: What measures have been taken to assure that fuel handlers remain attentive?

Response: The Commission, through the “Policy on Factors Causing Fatigue of Operating Personnel at Nuclear Reactors” provides guidelines on working hours that were consistent with the objective of ensuring that the mental alertness and decision-making abilities of plant staff were not significantly degraded by fatigue. For this study, the staff incorporated several measures into the risk assessment to help assure fuel handler attentiveness. First, SDA #2 calls for walkdowns of the spent fuel pool area by fuel handlers every shift. Second, IDC #5 states that SFP instrumentation will be in place providing readouts and alarms in the control room or where the fuel handlers are stationed. Additionally, discussions with the industry indicate that it is a general practice for sites to log instrument readings from the decommissioning spent fuel pools at least once per shift. Such practices help maintain fuel handler alertness and keep them abreast of the status of the pool and its support systems.

Comment #26: What measures have been taken to help minimize fuel handler error in postulated SFP accident scenarios?

Response: Having procedures in place helps reduce that chance of human errors, especially under stressful conditions such as during a severe accident. The industry has committed to

providing procedures or administrative controls to reduce the likelihood of rapid drain down events. IDC #2 is credited for ensuring that procedures and training of personnel are to be in place to ensure that on-site and off-site resources can be brought to bear during an accident. IDC #3 is credited to have procedures for establishing communication between onsite and offsite organizations during severe weather and seismic events. IDC #4 is credited to ensure that an off-site resource plan will be developed that will include access to portable pumps and emergency power. IDC #5 is credited to ensure that fuel handlers will have available to them spent fuel pool instrumentation that monitors spent fuel pool temperature, water level, and area radiation levels. In addition, SDA #2 calls for procedures and guidance for plant personnel on onsite and offsite makeup capability. SDA #3 calls for the direct measurement of water level and temperature. These staff assumptions and industry commitments are discussed in this report.

Comment #27: The NRC should review the need to place a containment around spent fuel pools.

Response: The staff evaluated the risk from spent fuel pool operation and from zirconium fires at operating plants in Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools." NUREG-1353 determined that the risks of spent fuel pool operation and the cost of alterations did not justify performing any generic backfits at operating plants, including installation of containment structures. The staff believes that containment structure is not warranted for decommissioning spent fuel pools. This issue is discussed in Section 4.1.2 as part of evaluation of defense in depth.

Comment #28: To the extent possible, experimental validation of risk-informed results should be addressed.

Response: The predictive models used for estimating the risk from spent fuel pools are based on a wealth of experimentation. Many experiments have been performed in the areas of human reliability analysis, seismic fragility of equipment, fires, and thermal hydraulics. The results of the decommissioning SFP risk assessment come from a systematic analytical modeling of the SFP and its support systems at a "typical" decommissioning site. The model of the SFP and its support systems was based on plant-specific visits made by the staff. The staff used failure rates of support system equipment based on existing large databases of equipment failure rates. Human error rates were developed by the staff with help from experts at Idaho National Engineering and Environmental Laboratory. Heavy load drops were based on modeling performed for NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36" with additional sources of data from U.S. Navy crane experiences, Waste Isolation Plant Trudock Crane System experience, and data supplied by NEI. The effects of aircraft crashes were analyzed using Department of Energy models and generic aircraft crash data. (See Appendix 2d)

Comment #29: The staff's report is misleading when it states that there is about a factor-of-two reduction in prompt fatalities if the accident occurs after one year instead of thirty days. The real insight should be that compared to operating plants, the absolute value of prompt fatalities from zirconium fires at SFPs is a couple of orders of magnitude lower. In fact, the report does not justify a one-year delay in eliminating off-site emergency preparedness. Prompt fatalities are sufficiently reduced one month after reactor shutdown to support eliminating off-site emergency preparedness.

Response: This report provides the staff's re-evaluation of the risk of spent fuel pool

accidents, as well as the effectiveness of emergency preparedness at decommissioning plants. The staff has evaluated the consequences at several times after shutdown at times less than one year.

Comment #30: The human error probabilities (HEPs) used for the operator action "Operator Recovery Using Off-Site Sources" are too conservative.

Response: The HEPs for recovery using off-site sources were quantified with the assumption that the fuel handlers/plant operators will initially attempt to mitigate the upset condition using in-house resources, and having failed this, attempt recovery using off-site sources. This was based on input obtained from licensees during public meetings on this subject, and on the assumption that fuel handlers will initially avoid using raw water (i.e., water not chemically controlled) when possible. It was however assumed that licensee procedures and training are in place to ensure that off-site resources can be brought to bear (IDCs # 2 and 4), and that these procedures explicitly state that if the water level drops below a certain level (e.g., 15 feet below normal level), licensee personnel must initiate recovery using off-site sources (SDA # 4) The probability of this event was quantified under the assumption that there is a low dependence with preceding fuel handler failures. Given that the event is always coupled with other fuel handler failures, it would, in the staff's opinion, be inappropriate to argue for zero dependence. When looked at in the context of the complete cutsets, it can be seen that the likelihood of failure to respond to any of the initiating events (excluding seismic and heavy load drops) where meaningful responses are possible is indeed low, as is evident from the low sequence frequencies.

Comment #31: Is it realistic to assume "good communication" with off-site emergency organizations once the plant is shutdown and "forgotten"?

Response: As the time after shutdown increases, the decay heat loads decrease and more time is needed to heat up the pool water and boil off if heat removal were lost. After one year, the decay heat levels are such that there is at least a week of delay between loss of cooling and spent fuel uncovery. Even following a seismic or severe weather event, the staff expects that a utility will be aware of the resources that are available in the area to provide pool cooling or inventory make up and that the utility will have assured the availability of the resources. In addition, the utility should have a plan for communicating with suppliers and government officials during such emergencies by means that would not be disrupted by such events (e.g., by portable radio). IDCs #2 and #3 provide assurance that good communication will be maintained.

Comment #32: Will commitments lead to practices better than current? If not, use historic data.

Response: It is the staff's expectation that the commitments will in general provide guidance that assures that the good practices found at decommissioning sites visited by the staff will be implemented at future decommissioning sites. Some industry commitments and staff assumptions, such as IDC #1 and SDAs #2, #3, and #4, may enhance the capabilities currently practiced by existing decommissioning plants. Where possible (e.g., for some initiating event frequencies), the staff has used actual data from spent fuel pool events. The industry commitments and staff assumptions provide a basis for the staff's conclusion that the low human error probabilities associated with the loss of SFP cooling and loss of inventory events are justified. In addition, the commitments provide a bound on the risk associated with the two events that could rapidly drain the spent fuel pool (i.e., seismic and heavy load drop

events.)

Comment #33: The staff noted a recent event (January 2000) that occurred during shutdown, when SFP monitoring should have been a priority. This event should have raised the initiating event frequencies, not lowered them.

Response: The staff agrees that including the two recent loss-of-cooling events mentioned the draft report would increase the initiating event frequency for loss of cooling accidents. However, since the fuel uncovery frequency from this event is very low (approximately 10^{-8} per year), the conclusion in the report that the loss of cooling events are not a major risk contributors is not affected. However, these recent events illustrate the importance of industry commitments, particularly IDC #5, which requires temperature instrumentation and alarms in the control room. Additionally, SDA #3 calls for direct measurement of water temperature and level in the spent fuel pool.

Comment #34: The discussion in Section 3.3.2 [of the draft report] states that many of the events listed in NUREG-1275, Volume 12, do not apply to a decommissioning facility. Therefore, adherence to IDCs #2, 5, 8, and 10 are not really important to establishing a low frequency of fuel uncovery.

Response: The commenter correctly noted that many of the initiating events from operating reactor spent fuel pool incidents that are discussed in NUREG-1275 do not apply to decommissioning facilities. The staff likewise did not include these events when estimating the frequency of events at decommissioning plants. To help assure that the frequency of these events does not end up being much higher than assumed by the staff in its risk assessment, the industry committed to IDCs and SDAs including those mentioned regarding procedures and planning for contingencies to limit, prevent, or mitigate loss of inventory and loss of cooling events.

Comment #35: How did the staff come up with the factor of 100 reduction in the failure rate for heavy load drops for single-failure-proof systems?

Response: For a non-single-failure-proof handling system, the mean probability of a loss-of-inventory event was estimated based on NUREG-0612. In NUREG-0612, an alternate fault tree (Figure B-2, page B-16) was used to estimate the probability 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 when corrected for the new Navy data and 100 lifts per year. A comparison of this mean value to the 2.0×10^{-7} per year mean value for the single-failure-proof crane shows a factor of 100 reduction.

Comment #36: Were heavy objects, such as crane rail or masonry wall, falling into the SFP or taking out electricity during decommissioning activities addressed in the study?

Response: The loss of electricity and the control of heavy loads were considered in the study. The loss of electricity would result in a loss of the spent fuel pool cooling system. IDC # 1 and SDA #5 deal with controlling heavy loads over the spent fuel pool. SDA #6 requires that licensees complete a seismic checklist that evaluates that seismic robustness of the pool and building structure. If a plant cannot successfully complete the generic seismic checklist, then a site-specific assessment would be performed.

Comment #37: Since the National Severe Storm Center is predicting more frequent and more intense severe weather phenomena, shouldn't the size and velocity of wind-driven missiles and maximum height of storm surges be reassessed?

Response: The licensing basis for severe weather phenomena is conservative, such that increases in storm intensity should still be bounded by the current licensing basis. When severe weather is expected, such as hurricanes, the NRC monitors the licensee's readiness and performance closely. If more severe storms were occurring or a plant (or plants) did not function as expected, we would evaluate the need to update plants' storm-related analyses and communicate with industry, such as using information notices and bulletins, to ensure that licensees were aware of events that occur at other plants and our expectations of their performance. Also, if a licensee requests a change to its licensing basis dealing with storms, such as tornados, or storm-generated missiles, then they would look at more recent data collected since the licensing of the plant.

Comments #38 and 39: (A) All pools leak, dry storage is the only way for long term safety. (B) The NRC should identify all SFP's that leak. Degradation of the liner and concrete should be investigated. The leaks should be sealed.

Response: The staff has determined that both wet storage (in a SFP) and dry storage (in casks) are safe methods to hold spent fuel. Most pools have a leak detection system between the steel liner and the concrete wall to identify and quantify if leakage from the liner occurs. This is not leakage to the environment. This water is collected by the system in the plant. This system allows licensees to monitor a situation and evaluate if there is a safety concern. Two plants do have leaking spent fuel pools. The licensees are closely monitoring the leak to ensure that there is no public hazard. Dry storage casks are a viable option for spent fuel storage for licensees. Dry storage casks are currently approved for fuel that have been removed from the reactor for at least five years. Many licensees are choosing to use dry cask storage in addition to the spent fuel pool.

Comment #40: What happened to the commitment verbally agreed upon through a public stakeholder to install a single failure proof crane system using safety grade electrical equipment?

Response: The staff reviewed the transcript of the public meeting. NEI verbally committed decommissioning plants to implement Phase II of NUREG-0612 (Control of Heavy Loads), which prescribed the use of single failure proof cranes or to implement a load drop analysis. NEI provided this commitment in writing on November 12, 1999 (See Appendix 5). The commitment was included in the analysis and documented in the report as IDC #1.

Comments #41 and 42: (A) The staff's spent fuel pool risk study only considered accident scenarios that could lead to a spent fuel zirconium fire and asked what other design basis accidents are considered for decommissioning nuclear power plants beyond those addressed in the study. (B) What design basis accidents do we need to consider?

Response: There are typically no new or unique conditions associated with decommissioning that result in the creation or possibility of a different type of accident not previously bounded by the design basis accidents considered for the plant while it was operating. When a licensee updates its Final Safety Analysis Report for decommissioning, a suite of accidents are considered that have a reasonable potential to adversely impact public health and safety. The

off-site consequences of these accidents are generally very small and should not require off-site emergency response. Examples of the types of accidents that are considered by the licensees include:

- Materials handling event (non-fuel)
- Radioactive liquid waste releases
- Accidents from handling spent resin
- Fire
- Explosions
- External events
- Transportation accidents
- Fuel handling accident

In addition to plant specific assessments of the postulated accidents, the staff has performed some generic evaluations. Consideration of environmental impacts of such events has been provided in NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities."

Comment #43: A commenter stated that Industry Decommissioning Commitment #5 should be revised to require direct measurement of SFP temperature and water level.

Response: The staff agrees; SDA #3 calls for direct measurement of SFP temperature and water level.

Comment #44: Dr. Hanauer was quoted in a 1975 memo to say, "you can make probabilistic numbers prove anything, by which I mean that probabilistic numbers prove nothing." If a respected technical advisor has expressed doubts about the NRC's use of probabilistic numbers, how is the NRC going to use probabilities convincingly to protect health and safety? A commenter stated that, "this is an invalid way of measuring safety, and should not be used. Each day these reactors stay opened you are poisoning the environment. This is unacceptable."

Response: In the two and a half decades since this statement, there have been significant advances in risk assessment methodologies. In that time frame, the NRC has also gained a great deal of experience in applying these methodologies to the regulatory arena, which has led to improved safety. The NRC has determined that PRA is an acceptable technology and uses it in a manner that complements a deterministic approach and supports the traditional defense-in-depth philosophy.

Comment #45: Has the NRC considered the events with the "second" worst off-site consequences at decommissioning plants? For example, in another country which has nuclear power plants, a fire in the bitumen storage (waste handling area) was found to have the second worst, although limited, off-site consequences.

Response: This study evaluated a spectrum of potentially severe spent fuel pool accidents. However, before offsite EP at a decommissioning plant could be eliminated, a licensee would need to perform reviews of their facilities to ensure that there are no other possible accidents that could result in off-site consequences exceeding the EPA protective action guidelines per existing requirements under 10 CFR 30.32(i) and 10 CFR 30.72.

SEISMIC

Comments #46, 47 and 48: (A) The staff should look at stresses on the transfer tunnel; (B) Seismic vulnerabilities of the SFP transfer tube should be assessed to properly determine the risk of SFP draining; (C) During the July workshop, members of the public raised concerns about the hazard of the fuel transfer tube interacting with the pool structure during a large earthquake.

Response: Transfer tubes are generally used in PWR plants where the fuel assembly exits the containment structure through the tube and enters the pool. These transfer tubes are generally located inside a concrete structure that is buried under the ground and attached to the pool structure through a seismic gap and seal arrangement. In most spent fuel pools, the transfer tube is not connected directly to the area of the pool that contains the spent fuel. The transfer tube is usually in a separate portion of the pool that has a weir wall separating the area from the main section that holds the spent fuel. The weir wall is higher than the top of the spent fuel. As such, even if water was drained through the transfer tube, the fuel would not be uncovered. Additionally, following the final off-load of the fuel into the spent fuel pool, the transfer tube is permanently capped at both ends. However, the layouts and arrangements can vary from one PWR plant to another and the seismic hazard caused by transfer tubes should be examined on a case-by-case basis. As such, as part of the seismic checklist each licensee must verify the adequacy of spent fuel pool penetrations whose failure could lead to drainage or siphoning. (See Appendix 2B)

Comment #49: The staff should address aging effects on the qualification of equipment.

Response: The "equipment qualification" program is required for components that are or have the potential to be exposed to harsh environments, such as high radiation or high temperature. Systems around the spent fuel pool are not exposed to harsh environments and therefore do not needed special consideration. To address the effect of normal aging on spent fuel pool support systems, the maintenance rule (10 CFR 50.65) requires that the licensee monitor systems or components associated with the storage, control, and maintenance of the spent fuel in a safe condition. Additionally, the probability of equipment failure was included in this study as part of the accident sequences. The staff believes that aging of equipment at decommissioning plants is adequately addressed through existing programs and characterized in this study.

Comment #50, 51 and 52: (A) The staff should address aging effects on the spent fuel pool, in particular, the strengthening or hardening of the concrete and the strength of the liner over time. (B) The NRC should perform a rigorous engineering analysis of the effects of aging[4] upon the spent fuel pool and its associated structures and equipment. Most SFPs were never designed to be quasi-permanent fuel storage facilities. Because there is, as of yet, no permanent place to store used fuel, SFPs have had to accept more fuel than they were originally designed to hold. To allow SFPs to continue to store spent fuel for, as of yet, an undetermined period of time requires, I suggest a comprehensive look at aging.
(C) A commenter raised concerns about the effect of aging on the spent fuel pool liner plate and the reinforced concrete pool structure.

Response: Irradiation-induced degradation of steel requires high neutron fluency, which is not present in the spent fuel pools. Over 30 years of operating experience has not indicated any

[4] Aging could include degradation, failure, etc. of structures & equipment.

degradation of liner plates or the concrete that can be attributed to radiation effects.

With aging, concrete gains compressive strength of about 20% in an asymptotic manner and the strength of reinforcing bars does not change with age, provided that rebars are not degraded by corrosion. In general, degradation of concrete structures can be divided into two parts, long term and short term. The long-term degradation can occur due to freezing and thawing effects when concrete is exposed to outside air. This is the predominant long-term failure mode of concrete, which is observed on bridge decks, pavements, and structures exposed to weather. Degradation of concrete can also occur when chemical contaminants attack concrete. These types of degradation have not been observed in spent fuel pools in any of the operating reactors. Additionally, inspection and maintenance of spent fuel pool structures are within the scope of the maintenance rule, 10 CFR 50.65, and corrective actions are required if any degradation is observed. An inspection of the SFP structure to identify cracks, spalling of concrete, etc., is also part of the seismic checklist. Substantial loss of structural strength requires long-term corrosion of reinforcing steel bars and substantial cracking of concrete. This is not likely to happen because of inspection and maintenance requirements. Through the use of the seismic checklist, any degradation such as spalling of concrete or cracks and indications of rust and stains, etc., will be detected and appropriate corrective actions taken. (See Appendix 2b)

Degradation of the liner plate can occur due to cracks that can develop at the welded joints. Seepage of water through minute cracks at welded seams has been minimal and has not been observed at existing plants to cause structural degradation of concrete. Nevertheless, preexisting cracks would require a surveillance program to ensure that structural degradation is not progressing.

Based on the discussion above, any potential aging of the spent fuel pool structure is managed during decommissioning. The structural strength will be verified using the seismic checklist in the early stages of decommissioning, which may include site-specific analysis. While its structural strength is not expected to degrade during decommissioning, it is managed under the maintenance rule. As a result, the staff does not believe that detailed generic analysis is needed.

Comment #53: To my knowledge, not every spent fuel pool was designed to the seismic criteria in use today. The use of works like “robust” does not necessarily address seismic qualifications. The NRC should identify all spent fuel pools that were not initially designed to seismic criteria and explain their level of qualification, including the SF racks.

Response: When the licensee requests to expand the spent fuel storage capacity, spent fuel pools undergo seismic and structural reevaluation during a licensing review. Spent fuel pool structures, as well as the spent fuel racks, undergo detailed analysis and staff review. All currently operating nuclear power plants have expanded their spent fuel storage capacity and met their safe shutdown earthquake criteria.

Comment #54: Not all PWR buildings housing spent fuel are seismically qualified. The NRC should perform a worst case analysis of the result of a seismic event which collapses the spent fuel pool building, and/or drains the pool and/or damages the spent fuel. Both criticality and zirconium fires are of concern. The nine initiating events listed on p. 11 which could occur concurrently with the earthquake should also be considered if the events contribute to the worst case scenario.

Response: Risk assessments are performed as realistic as possible, not worst case. The

staff identified the following nine initiating event categories to investigate as part of the quantitative risk assessment on SFP risk:

Loss of Off-site Power from plant centered and grid related events
Loss of Off-site Power from events initiated by severe weather
Internal Fire
Loss of Pool Cooling
Loss of Coolant Inventory
Seismic Event
Cask Drop
Aircraft Impact
Tornado Missile

The initiating events indicated above are independent. However, the event sequences that emanate from each event are carefully modeled in the event tree and could include some of the same circumstances. This means that a seismic event tree would include the consideration of off-site and on-site power loss. In a PRA assessment no risk insight can be gained by considering worst case combination of truly random and independent events such as a seismic event and a tornado missile. However, the frequency of a combined seismic and tornado missile is much less than $1x10^{-8}$. Also, with respect to other structures, such as crane girders and super-structures, they are covered in the seismic check list for the spent fuel pool structure.

Comment #55: The NEI seismic checklist requires a seismic engineer to review drawings in addition to conducting a walkdown of the SFP. It has been my experience that many electrical drawings of NPP's do not reflect the existing plant electrical installation. How is the seismic engineer going to verify drawings to the existing SFP building and pool if much of the pool is inaccessible? For instance, how does he verify concrete degradation under the steel liner? The NRC should require that specific areas be inspected and that these areas be accessible. If these areas are not accessible, then the checklist is not complete and susceptibility to seismic activity remains a concern.

Response: Plant walkdowns should provide an opportunity to verify and correct plant drawings to the as-built conditions, and design calculation would verify the compliance to Codes and Standards. The staff considers the review of construction drawings to be very important. Minimum reinforcing areas are dictated by codes. Thick walls and slabs forming spent fuel pool structure are in many cases governed by minimum reinforcing requirements. Should there be any additional shear or flexural steel requirements, engineering calculations would indicate where they are needed and how much is needed. Therefore, a review of drawings and design calculations would present a more complete picture. With respect to inaccessible areas, cracks, spalling of concrete and stains, and efflorescence are of a degradation process. In order to determine the root cause of the external signs, it is necessary to use more invasive procedures, such as chipping and breaking concrete, etc. This is not unique to spent fuel pool structures, and there are several examples of this type of inspection in the operating experience of several plants.

Comment #56: The NRC should specify why it is not cost effective to perform a plant-specific seismic evaluation for each spent fuel pool and what impact this has on safety. Because there are so many differently designed spent fuel pools, it is difficult to perceive how a generic approach could be acceptable without assembling a list of similar and/or identical designs and performing a seismic evaluation of the various groups which are assembled. Specific seismic evaluations for each plant or groups of similar/identical plants should be considered.

Response: A significant body of work exists characterizing the strength and capacity of shear walls based on tests and analyses. The use of a generic parameter, with the underpinning of data, solely for the purpose of screening is very appropriate and reliable. Using the seismic checklist, a structure is not acceptable unless all the conditions in the checklist are met. At sites where the seismic checklist is not met, a plant specific evaluation can be conducted. The use of a screening parameter is a reliable way to determine the need for further evaluation.

Comment #57: Did the NUREGs that you looked at take into account new information coming out of the Kobe and Northridge events? Particularly as we are learning more about risks associated with those two particular seismological events that were never even considered when plants were sited; particularly, though I can't frame it in the seismological language, from a lay understanding, it's clear that new information was gained out of Kobe and Northridge events suggesting that you can have seismological effects of greater consequence farther afield than at the epicenter of the event.

Response: The staff believes that the NUREG reports mentioned by the commenter were NUREG 1488, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains" and NUREG/CR-5250, "Seismic Hazard Characterization of 69 Nuclear Plant Sites East of The Rocky Mountains," which were written in the middle and late 1980s and used probabilistic seismic hazard analyses performed for the NRC by Lawrence Livermore National Laboratory (LLNL) for nuclear power plants in the central and eastern U.S. Since then, LLNL has performed additional probabilistic hazard studies (1993) for central and eastern U.S. nuclear power plants for the NRC. The results of the more recent study indicated lower seismic hazards for the plants than the earlier study estimated. EPRI has also performed probabilistic hazard studies. The LLNL hazard curves generally predict higher frequency estimates than those generated by EPRI. This is a result of different expert judgements. However, both are valid methodologies.

The design basis for each nuclear power plant took into account the effects of earthquake ground motion. The seismic design basis, called the safe shutdown earthquake (SSE), defines the maximum ground motion for which certain structures, systems, and components necessary for safe shutdown were designed to remain functional. The licensees were required to obtain the geologic and seismic information necessary to determine site suitability and provide reasonable assurance that a nuclear power plant could be constructed and operated at a site without undue risk to the health and safety of the public.

The information collected in the investigations was used to determine the earthquake ground motion at the site, assuming that the epicenters of the earthquakes are situated at the point on the tectonic structures or in the tectonic provinces nearest to the site. The earthquake which could cause the maximum vibratory ground motion at the site was designated the SSE. This ground motion was used in the design and analysis of the plant.

The determination of the SSEs followed the criteria and procedures required by NRC regulations and applied a multiple hypothesis approach. In this approach, several different methods were applied to determine each parameter, and sensitivity studies were performed to account for the uncertainties in the geophysical phenomena. In addition, nuclear power plants have design margins (capability) well beyond the demands of the SSE. The ability of a nuclear power plant to resist the forces generated by the ground motion during an earthquake is thoroughly incorporated in the design and construction. As a result, nuclear power plants are able to resist earthquake ground motions beyond their design basis and above the ground motion that would result in severe damage to residential and commercial buildings designed and built to standard building codes.

Following large damaging earthquakes such as the Kobe and Northridge events, the staff reviewed the seismological and engineering information obtained from these events to determine if the new information challenged previous design and licensing decisions. The Kobe and Northridge earthquakes were tectonic plate boundary events occurring in regions of very active tectonics. The operating U.S. nuclear power plants (except for San Onofre and Diablo Canyon) are located in the stable interior portion of the North American tectonic plate. This is a region of relatively low seismicity and seismic hazard. Earthquakes with the characteristics of the Kobe and Northridge events will not occur near central and eastern U.S. nuclear power plant sites.

The ground motion from an earthquake at a particular site is a function of the earthquake source characteristics, the magnitude and the focal mechanism. It is also a function of the distance of the facility to the fault, the geology along the travel path of the seismic waves, and the geology immediately under the facility site. Two U.S. operating nuclear power plant sites can be considered as having the potential to be subjected to the near field ground motion of moderate to large earthquakes. These are the San Onofre Nuclear Generating Station (SONGS) near San Clemente and the Diablo Canyon Power Plant (DCPP) near San Luis Obispo. The seismic design of SONGS Units 2 and 3 is based on the assumed occurrence of a Magnitude 7 earthquake on the Offshore Zone of Deformation, a fault zone approximately 8 kilometers from the site. The design of DCPP has been analyzed for the postulated occurrence of a Magnitude 7.5 earthquake on the Hosgri Fault Zone, approximately 4 kilometers from the site. The response spectra, used for both the SONGS and the DCPP, was evaluated against the actual spectra of near field ground motions of a suite of earthquakes gathered on a worldwide basis.

The commenter stated, "... it's clear that new information was gained out of Kobe and Northridge events suggesting that you can have seismological effects of greater consequence farther afield than at the epicenter of the event." A review of the strong motion data and the damage resulting from these events does not bear out the validity of this concern at SONGS and DCPP.

The staff assumes that the individual alluded to the fact that the amplitudes of the ground motion from the 1994 Northridge earthquake were larger in Santa Monica than those at similar and lesser distances from the earthquake source. The cause of the larger ground motions in the Santa Monica area is believed to be the subsurface geology along the travel path of the waves. One theory (Gao et al, 1996) is that the anomalous ground motion in Santa Monica is explained by focusing due to a deep convex structure (several kilometers beneath the surface) that focuses the ground motion in mid-Santa Monica. Another theory (Graves and Pitarka, 1998) is that the large amplitudes of the ground motions in Santa Monica from the Northridge earthquake are caused by the shallow basin-edge structure (1 kilometer deep) at the northern edge of the Los Angles Basin. This theory suggests that the large amplification results from constructive interference of direct waves with the basin-edge generated surface waves. Earthquake recordings at San Onofre and Diablo Canyon do not indicate anomalous amplification of ground motion. In addition, there have been numerous seismic reflection and refraction studies of the site areas for the site evaluations, and for petroleum exploration and geophysical research. They, along with other well-proven methods, were used to determine the nature of the geologic structure in the site vicinity, the location of any faults, and the nature of the faults. None of these studies have indicated anomalous conditions, like those postulated for Santa Monica, at either SONGS or DCPP. In addition, the empirical ground motion database used to develop the ground motion attenuation relationships contains events recorded at sites with anomalous, as well as typical ground motion amplitudes. The design basis ground motion for both SONGS and DCPP were compared to 84th percentile level of ground motion obtained using the attenuation relationships and the appropriate earthquake magnitude, distance and geology for each site. The geology of the SONGS and DCPP sites

does not cause anomalous amplification, therefore, the information gained from the Kobe and Northridge events does not raise safety concerns for U.S. nuclear power plants.

In summary, earthquakes of the type that occurred in Kobe and Northridge are different from those that can occur near nuclear power plants in the central and eastern U.S. The higher ground motions recorded in the Santa Monica area from the Northridge earthquake were due to the specific geology through which the waves traveled. Improvements in our understanding of central and eastern U.S. geology, seismic wave attenuation, seismicity, and seismic hazard calculation methodology result in less uncertainty and lower hazard estimates today than have previous studies.

Comment #58: The use of Lawrence Livermore National Laboratory (LLNL) hazard curves at high ground motion values may not be credible. Even EPRI results are likely to be overly conservative at high ground motions. The requirement that some plants with higher SSE values perform detailed HCLPF assessments of their SFPs is not warranted. In conclusion, there should be no SFP screening level distinctions based on plant SSEs for the central and eastern U.S. All that is needed is that the sites pass the screening criteria (Appendix 2b). For a few western sites, it is reasonable to require that the plants demonstrate a HCLPF of 2 X SSE?

Response: The staff is no longer using the 2 or 3 times the SSE as a criterion. The staff agrees that there is considerable uncertainty in the EPRI and LLNL hazard curves at higher ground motions, since the geologic record east of the Rocky Mountains is sparse and does not provide many examples of very large ground motions. The EPRI and LLNL hazard curves were developed as best estimates and were made by different experts who gave their best judgement as to how to reflect the risks from seismic events at various nuclear power plant sites. They provided expert advice for high and low ground motions. The staff's re-evaluation is discussed in this report.

Comments #59 and 60: (A) The NRC should determine the qualifications and degradation of spent fuel racks. (B) How can there be no spent fuel pool degradation issues if type 304 stainless steel employed in fuel racks and assemblies is known to exhibit stress-corrosion cracking in oxygenated or stagnant borated water?

Response: Spent fuel rack designs do have qualifications. The designs are reviewed and approved by the NRC. Additionally, when a licensee changes its technical specification for the amount of fuel allowed to be stored in the pool even using approved spent fuel racks, an NRC review and approval is required. The staff technical reviewers use the guidelines in NUREG-0800, Standard Review Plan (SRP), which incorporates the regulations specified in the Code of Federal Regulations, Appendix A, General Design Criteria, which require safe handling and storage under normal and accident conditions.

Regarding degradation, type 304 stainless steel material, which is used for the spent fuel racks, is susceptible to stress corrosion cracking in oxygenated water environment at relatively high temperature conditions. At the temperature levels that exist in the spent fuel pools, stress corrosion cracking of the spent fuel racks made of stainless steel is not a concern, and there has been no report of any actual incidence of stress corrosion cracking of spent fuel racks. The stagnant, borated condition of the spent fuel pool water is not a significant factor in inducing stress corrosion cracking of the racks. Most spent fuel assemblies are clad with zirconium and are not known to be susceptible to stress corrosion cracking.

Comment #61: A significant seismic event which damages and drains the SFP is also likely to wreak havoc upon the local infrastructure. How has NRC considered the availability of local resources as identified by IDC #2, #3, and #4 should the local infrastructure be destroyed?

Response: Seismic capacity of spent fuel structures against catastrophic failures, such that a very rapid loss of water can be assumed, is substantially above the safe shutdown earthquake levels of the spent fuel pools. Consequently, high ground motion levels are necessary to initiate failures. The response by local, state, or national authorities needed at the spent fuel pool site will depend on the actual or potential damage to the spent fuel pool. The most likely damage to the spent fuel pool and support systems would be to the support systems that provide cooling to the pool. The large inventory of water above the spent fuel should provide adequate time (it would take about a week without pool cooling before boiling would occur) for repairing or bringing in replacement pumps and heat exchangers. If the local infrastructure was damaged by a seismic event such that the prearranged off-site response could not occur, the industry commitments provide a good foundation for an ad hoc response.

Comment #62: For all central and eastern U.S. nuclear power plant sites and for some western U.S. nuclear power plant sites, all that is necessary to have an adequately safe spent fuel pool with respect to seismic-induced risk is for the pool to meet the requirements of the seismic checklist. Several western U.S. sites may need to demonstrate a high confidence with low probability of failure (HCLPF) of 2 X SSE.

Response: The staff agrees that, for most sites throughout the U.S., meeting the enhanced seismic checklist (Appendix 2D) is sufficient to demonstrate acceptable seismic risk for decommissioning spent fuel pools. However, if a site does not pass the seismic checklist, a plant-specific seismic risk evaluation of the spent fuel pool could be performed. The staff is no longer recommending using 2 or 3 times the SSE as a criterion. The staff's re-evaluation is included in this report.

Comment #63: The value of three times the SSE for the SFP HCLPF should not be a hard and fast acceptance criteria, since this is only a screening criteria.

Response: The staff is no longer recommending the use of 2 or 3 times the SSE as a criterion. This report provides the staff's re-evaluation of the SFP accident risk at decommissioning plants and the criteria necessary to demonstrate low risk.

THERMAL HYDRAULICS

Comment #64: The draft study is deficient in that it ignores the phenomenon associated with partial draindown of SFP that will suppress convective heat transfer by presence of residual water at the base of fuel assemblies.

Response: The staff agrees that the partial drain down scenario should be considered and has included the scenario in the thermal hydraulic analyses in this study.

Comment #65: The draft study is deficient in that partial draindown will lead to a steam-zirconium reaction producing hydrogen gas which could reach explosive concentrations in the atmosphere of the spent fuel building, potentially leading to a breach of that building.

Response: The staff agrees that, in a partial draindown scenario, hydrogen would be produced. The hydrogen concentrations or the consequences of any subsequent hydrogen burn or explosion have not been calculated for this study. However, the staff believes that the consequences are bounded by the zirconium fire consequences because no credit is given in the staff's analysis for building integrity to retain fission products. In effect, a complete building breach is assumed in the staff's consequence analysis.

Comment #66: Depending on fuel burnup/storage array details, the development of standard methods is needed for consistent application of regulations.

Response: The staff agrees that a standard methodology or guidelines for thermal hydraulic analysis would assist in the consistent application of regulations. After a rulemaking plan is approved by the Commission, the staff will evaluate the best methods to develop the necessary guidance.

Comment #67: The gap release temperature is too conservative for a success criterion.

Response: The gap release temperature is the temperature at which the cladding can blister and allow gases trapped between the fuel pellets and the cladding to escape. The temperature criterion for gap release may also be the threshold for releasing fuel fines and ruthenium. These considerations and others were included in determining the temperature criterion for the thermal hydraulic analysis. This is discussed in Appendix 1b.

Comment #68: Fire propagation to low powered fuel is unlikely.

Response: As the fuel decays, the involvement in a fire becomes less likely. However, sufficient research has not been performed to define clear limits of propagation. The staff therefore assumed the involvement of 3.5 cores in its analysis based upon previous analyses (NUREG/CR-0649 and NUREG/CR-4982). This assumption is discussed in Section 3 and Appendix 4 of this report.

Comment #69: Could foreign materials with lower ignition temperatures enter a drained SFP and catch fire, thus raising the temperature of the spent fuel to the point of rapid zirconium oxidation?

Response: Foreign objects that would fall between or into the fuel assemblies to be close to

the fuel would be of a small size and would have an insignificant effect on the heat input in comparison to the spent fuel. As part of this study, the staff performed a thermal hydraulic sensitivity study assuming adiabatic conditions (no heat removal), which would produce a faster heatup of the fuel than would actually be expected. Additionally, licensees have programs to keep foreign objects from entering the spent fuel pool. Retrievable foreign objects that fall into the pool are moved to designated storage areas within the pool.

Comment #70: The energy of reaction for air oxidation in the draft report is incorrect.

Response: The staff confirmed that the draft report is correct. It appears that the commenter based the calculation on 92 kg of zirconium in a mole, when there are 92 grams of zirconium in a mole.

ISSUES OUTSIDE OF THE STUDY

EMERGENCY PREPAREDNESS, SECURITY, INSURANCE, FIRE, AND SAFETY CULTURE

Comment #71: For EP, the integrated decommissioning rule should specify that the licensee is excused from 10 CFR 50.47 requirements after a period of one-year from final shutdown. The basis for this recommendation is drawn directly from the technical material presented, and little can be gained by closer analysis.

Response: The rulemaking will be based on the conclusions drawn from this technical study. The staff has used all of the information gathered during the technical study to perform its risk assessment; evaluating the frequencies of events, potential consequences, thermal hydraulic analysis and the effectiveness of EP.

Comment #72: The decommissioning rule should specify that the licensee is excused from 10 CFR 50.47 off-site EP requirements after the short-lived nuclides important to dose have undergone substantial decay resulting in off-site dose consequences due to license basis accidents of less than 1 rem (the EPA protective action guideline).

Response: The staff has considered the decay time of short-lived nuclides and the off-site dose consequences of the short-lived nuclides. However, to assess the whole risk, the staff also considered the consequences of longer-lived nuclides, the risks of both design basis accidents and beyond design basis events, and the effectiveness of EP in efforts to determine an appropriate point at which requirements for off-site EP could be relaxed.

Comment #73: Section 4.3.2, “Security” of the draft report (February 2000) casts a shadow on the entire 10 CFR 73.51 rulemaking and needs to clarify the scope of the safety issues. The last paragraph in Section 4.3.2 should be clear and completely identify the scope and basis of the ISFSI safety concerns from the radiological sabotage and theft identified in 10 CFR 73.1. Finally, the last paragraph appears to contradict the May 15, 1998, NRC rulemaking on Physical Protection for Spent Nuclear Fuel and High-Level Radioactive Waste, Federal Register Vol. 63, No. 94, Pages 26955 - 26963.

Response: The NRC staff agrees that Section 4.3.2, Security, as written in the February 2000 report, appears to be inconsistent with the changes to Part 73 as described in FRN 26955 dated May 15,1998. The description of risk associated with potential criticality and fuel heat up is for spent fuel recently discharged from the reactor vessel and not spent fuel stored at an ISFSI. The discussion is no longer in the report, however, the staff provides additional information to clarify the apparent inconsistency.

The staff believes that, as written, 10 CFR 73.51 provides proper physical protection for the storage of spent nuclear fuel licensed under Part 72 at an independent spent fuel storage installation (ISFSI). The design-basis threat for radiological sabotage of power reactors under 10 CFR 73.1 is not considered appropriate for the types of facilities subject to Section 73.51 and, therefore, a separate protection goal is defined for these facilities. The protection goal states: "The physical protection system must be designed to protect against loss of control of the facility that could be sufficient to cause radiation exposure exceeding the dose as described in 10 CFR 72.106 and referenced by 73.51(b)(3)."

With regard to protection against malevolent use of land-based vehicle, the staff continues to believe there is no compelling justification for requiring a vehicle barrier as perimeter protection

at this time for ISFSIs. The staff will, however, continue to review the requirements to ensure that a proper level of security is provided for existing casks, new cask designs, and other changing technologies.

Comment #74: For Security, the integrated decommissioning rule should allow licensees to be excused from 10 CFR 73.55 requirements upon a showing that the consequences of sabotage can not exceed a defined dose to the public at the site boundary.

Response: The staff believes that 10 CFR 73.55 should be modified to a level commensurate with the risk associated with safeguarding permanently shutdown plants, but not to a level less than that provided for an ISFSI as described in 10 CFR 73.51.

Comment #75: The report concludes that there is no methodology currently available to access probabilities of terrorist activity or behaviors which might culminate in attempted sabotage of spent fuel. We disagree. For instance, Sandia National Laboratories, a key contractor employed by the NRC on security matters, has applied a probabilistic approach to security in decommissioning on the Maine Yankee docket. We encourage the staff to review this report.

Response: The staff reviewed the information provided by the commenter. After review, the staff maintains that there is currently not an acceptable methodology available to access the probability of terrorist activity. The report in question, its identity verified through NEI, is "A Vulnerability Analysis of a Proposed Security Plan for the Maine Yankee Power Plant," dated January 9, 1998. The purpose of this report was twofold: first, it presents the results of an analysis of the effectiveness of the proposed physical security system in preventing or mitigating an attempt by the design basis threat adversaries attempting radiological sabotage, and second, it presents the results of a study to determine the need for a vehicle barrier systems. This report does not predict the probability of terrorist activities or behaviors. The staff has read this report, and conducted an on-site inspection (report dated June 8, 1999) of its technical findings and found them to be deficient. Further information on the inspection can be found in the June 8 inspection report, Inspection Report #:50-289/99-06.

Comments #76, 77 and 78: (A) It's conspicuously absent from your review of risk in this overall subject, that the staff hasn't looked at the issue of sabotage and terrorism. (B) The draft report omitted acts of sabotage and vandalism. Emergency evacuation plans should be prepared with this consideration of terrorism. (C) It is suggested that NRC "err on the side of safety" since terrorist acts can not be specifically addressed.

Response: The commenters are correct that security is identified, but not highlighted, in the report. The report is a technical study to quantify the risks as it relates to the draining of a decommissioning plant's spent fuel pool and the issue of a zirconium fire. It was not intended to address security in any detail. The staff will be addressing specific security requirements for decommissioning plants in its integrated rulemaking, which is an outgrowth of the technical study. As with any rulemaking, there will be opportunities for the public to comment on the security requirements the staff is recommending.

Comment #79: For insurance, the obligation for secondary financial protection should end at such time that a determination can be made that clad surface temperatures greater than 570 °C can not occur in a dry configuration. The calculation of this temperature should be by approved methodology. However as supported in the technical report, in the absence of any

calculation, the obligation should end after a period which is less than five years. The capacity required of primary financial protection should be reduced after the period of time determined as above for secondary financial protection.

Response: The liability insurance requirements of our regulations are meant to ensure that the public is protected in the event of a low probability, high consequences event. The accident sequences that could lead to a zirconium fire are low probability but could result in high consequences in terms of property damage and land contamination. In this study, a spectrum of event initiators and sequences of events were analyzed to determine if the events were creditable. The staff will evaluate the need for analytical calculations and regulatory guidance during the rulemaking process.

Comment #80: The obligation for decommissioning plants to participate in the secondary financial protection should be reviewed in light of the low public risk posed for SFPs for decommissioned plants. Industry does not believe that the risk justifies requiring participation. (The majority of the $3x10^{-6}$ risk of significant off-site consequences comes from an upper bound determination of the risk posed by seismic events, not on a best estimate of the seismic risk).

If it is determined that participation [in secondary financial protection] will be required during the short time that decommissioning plants pose a non-zero risk, then the level of participation should be in proportion to a best estimate of the risk posed relative to the risk posed by operating plants. If any participation is required, it should be only for the short period that clad surface temperatures greater than 570°C can occur in a loss of water configuration. The calculation of this temperature should be by an approved methodology.

The commenter also stated that the capacity required for primary financial protection should be eliminated for consideration of any potential for accidents with significant off-site consequences. For other events with off-site consequences, on-site coverage should be reduced to $25 million dollars (M) for the period when the spent fuel remains in the pool and off-site coverage should be reduced to $5-10M. When the fuel has been removed off-site or placed in an off-site ISFSI, on-site coverage should be reduced to $25M while the site still contains significant sources of radioactive material. On-site coverage could be reduced to zero when there are no sources exceeding 1000 gallons of fluid. Off-site coverage should be reduced to $5-10M for plants with fuel off-site or in an on-site ISFSI.

Response: The underlying purpose of 10 CFR 50.54(w) is to provide sufficient property damage insurance coverage to ensure funding for on-site post-accident recovery stabilization and decontamination costs in the unlikely event of a nuclear accident. Section 140.11 of Title 10 of the CFR also serves to provide sufficient liability insurance to ensure funding for claims resulting from a nuclear incident or precautionary evacuation. The property and liability insurance requirements of our regulations are meant to ensure that the public is protected in the event of a low probability, high consequence event.

In SECY-93-127, “Financial Protection Required of Licensees of Large Nuclear Power Plants During Decommissioning,” the staff explains that insurance coverage is necessary for reactor licensees as determined by “reasonably conceivable” accidents. Reasonably conceivable accidents may exceed design basis accidents but are less severe than remotely possible hypothetical accidents that are often termed “incredible.” Also, the consequences of such accidents need to be considered. The staff has previously stated that while it is correct that the frequency of events that could lead to a zirconium fire is small, the consequences of such a fire could be significant. The SRM for SECY-93-127 approved the staff’s recommendation that after an appropriate cooling period for the spent fuel had elapsed that primary level

coverage could be reduced and licensees would be allowed to withdraw from participation in the secondary financial protection layer. The actual amount of coverage and acceptable methods to demonstrate approval will be determined during rulemaking.

Comment #81: With new personnel and decommissioning personnel, what methods are available to instill or ensure the same "safety culture" as during operation?

Response: There are several methods of instilling/ensuring "safety culture" in new personnel at both operating and decommissioning facilities. Methods include management policies and procedures, training, and qualification. OSHA requires employers to provide employees with safety training and education. Section 1926.21(b)(2) of Title 29 of the CFR requires training in the recognition and avoidance of unsafe conditions, 29 CFR 1926.21(b)(3) requires training in the safe handling and use of poisons, caustics, and other harmful substances, 29 CFR 1926.21(b)(5) requires training in the safe handling and use of flammable liquids, gases, or toxic materials, and 29 CFR 1926.21(b)(6) requires confined or enclosed space training. In addition, 10 CFR 50.120 requires training and qualification of nine categories of personnel involved with spent fuel pool maintenance and support. The training programs for the nine categories of personnel should include occupational safety and radiation protection training. While NRC and OSHA require training, it is incumbent upon the licensee to provide the training and instill/ensure upon the workers the proper "safety culture."

Comment #82: A commenter asked about calculations for radiation dose experienced by members of the fire brigade responding to resin fires.

Response: Existing regulatory requirements address the need for onsite worker radiation protection and emergency plans to consider protective actions and a means for controlling exposures in an emergency for emergency workers as well as the public. For example, the regulatory requirements for emergency worker protective actions and exposure control are found in 10 CFR 50.47(b)(10) and 10 CFR 50.47(b)(11). Each site has established procedures and training for the protection of workers responding to emergency situations. Generally, these procedures include the consideration of radiological conditions when responding to events. Calculations for occupational exposure for emergency workers would be consistent with the EPA Emergency Worker and Lifesaving Activity Protective Action Guidelines.

Comments #83 and 84: (A) Discuss protection of plant workers, particularly for less severe accidents such as pool uncovery without a zirconium fire. (B) The draft report should be revised to include credible hazards to plant workers at permanently closed plants.

Response: This technical study was limited to accidents involving the draining a decommissioning plant spent fuel pool. For on-site hazards, the staff believes that existing regulatory requirements adequately address the need for emergency plans to consider protective actions and a means for controlling exposures in an emergency for emergency workers. For example, 10 CFR Part 20 establishes standards for radiation protection for onsite workers and the public, and 10 CFR 50.47 (b)(10) and (11) establish protective actions and exposure control regulations for emergency workers. Nuclear power plant licensees are also subject to regulations for byproduct material under 10 CFR Part 30. Emergency plans under Section 30.32 require identification of accidents and means for mitigation, including the protection of onsite workers. Additionally, OSHA and NRC regulations require safety training and education, including safe handling and use of poisons, caustics, flammable liquids, gases and toxic materials; radiation protection; and occupational safety.

Although this study does not directly assess accidents or hazards that could occur to plant personnel, measures for worker safety were included. For example, IDC #8 calls for remote alignment of the water makeup source to the SFP without requiring entry to the refueling floor, which prevents workers and other accident responders from entering a potential radiation area.

Comment #85: What will the NRC staff do to protect plant workers and the public from spent fuel pool risks at permanently closed plants and operating plants before the industry commitments and staff assumptions are implemented?

Response: The analysis for this study reflects practices already in place. The staff visited four decommissioning sites as part of the preparation for developing the risk assessment of decommissioning spent fuel pools. The insights from those visits include that the facilities appeared to have been staffed by well trained and knowledgeable individuals with significant nuclear power plant experience. Procedures were in place for dealing with routine losses of inventory. Fuel handlers appeared to know whom to contact off-site if difficulties arose with the SFP. The staff recognized that these attributes were not required by any NRC regulations nor suggested in any NRC guidance for decommissioning sites. The IDCs and SDAs are an attempt to increase the assurance that plant personnel will continue to be knowledgeable of off-site resources and have good procedures available to them.

The staff believes that current worker safety regulations adequately protect workers. The regulations for the protection of workers are the same at decommissioning plants as at operating plants, such as 10 CFR 20 for standards for protection against radiation. Several other comments in this appendix also address worker safety regulations.

Comment #86: The NRC should determine the proper methods of extinguishing a possible zirconium fire.

Response: At the present time, the state-of-art for zirconium fire experiments has not advanced to determined the various methods for extinguishing. Additional research would need to be performed to investigate acceptable methods, required quantities of fire-fighting materials, conditions of use, and guidelines. Due to the low probability of the event, this research is not recommended at this time.

Comment #87: The consequences should be re-evaluated to include an off-site radiological release from an on-site fire involving radioactive material from a resin container fire; fire in a waste storage building; and fire in a container vehicle with waste stored in it that could trigger emergency response mechanisms.

Response: This evaluation is beyond the scope of this study which is focused on spent fuel pool accident risk. However, before offsite EP at decommissioning plants could be eliminated, a licensee would need to perform reviews at their facilities to ensure that there are no other possible accidents that could result in off-site consequences exceeding EPA protective action guidelines per existing requirements under 10 CFR 30.32 (i) and 10 CFR 30.72.

Comment #88: Decommissioning nuclear power plants should be evaluated for fires in the low level waste storage (LLW) area. This stakeholder states that large amounts of LLW could be stored in on-site LLW storage areas if off-site waste disposal sites are lost by a licensee "mid-stream" during the decommissioning process.

Response: As part of the staff's broad-scope decommissioning regulatory improvement effort, the staff will ensure that regulations are in place that would reasonably preclude threats to the public health and safety from accidents that are significantly less severe than a spent fuel pool zirconium fire but perhaps more probable, such as the LLW fire described above. To address the specific concern of the public stakeholder, 10 CFR 50.48 requires decommissioning nuclear power plant licensees to maintain a fire protection program to address fires which could cause the release or spread of radioactive materials which could result in a radiological hazard. In addition, nuclear power plants are also subject to the Commission's regulations for byproduct materials under 10 CFR Part 30. Specifically, 10 CFR 30.32(i) would require a licensee to maintain an appropriate EP program for radioactive materials stored on-site in quantities in excess of those specified in 10 CFR 30.72, "Schedule C - Quantities of Radioactive Material Requiring Consideration of the Need for an Emergency Plan for Responding to a Release."

RULEMAKING & NRC PROCESS CONCERNS

Comment #89: Since more radioactive materials are being handled [during decommissioning] than at an operating plant, and under conditions more likely to lead to inadvertent exposures, why are licensees left without the supervision of resident inspectors, or at least radiation protection personnel?

Response: During operation of a reactor, radioactive material is produced by neutron absorption by various materials. These radioactive materials are handled in many ways, including liquids contained in pipes and tanks, and radioactive solids contained in plastic bags or specialized containers. After the reactor is shut down, no additional radioactive material is produced and the radioactive material decay process reduces the total amount of radioactive material over time. The handling of radioactive material after shutdown is controlled in the same manner as before shutdown. Supervision of radioactive material handling is performed by the licensee before and after reactor shutdown with the oversight of licensee radiation protection personnel. Region-based NRC inspectors provide periodic verification that the licensee is handling radioactive materials within the bounds of the current regulations. NRC experience over the last few years with the current region-based reactor decommissioning inspection process has shown that the oversight process is effective in ensuring both public health and safety and protection of plant workers.

Comment #90: A commenter stated that little of what operators or reactor inspectors have learned is applicable to decommissioning. NRC needs personnel specifically trained in and dedicated to decommissioning.

Response: Significant changes take place during the transition from an operating plant to a decommissioning plant. However, many decommissioning activities are similar to activities conducted during plant operation. For example, the complete removal of components and systems, radiological waste shipments, fuel handling operations, and spent fuel pool system operations and maintenance which occur during decommissioning are very similar to activities that occurred during plant operation and refueling outages. Objectives during decommissioning, such as, protecting the spent fuel from sabotage and maintaining the spent fuel pool operational, were also accomplished during plant operation. The training received by operators and inspectors associated with radiological fundamentals, system operations, etc., still applies during decommissioning.

Although there is not an NRC inspector on-site during all of decommissioning, as there is during plant operation, there is a group of inspectors in each region who are specifically assigned to oversee plants undergoing decommissioning, and who make routine visits to the site (commensurate with the quantity and significance of the ongoing work). Each plant in decommissioning is also assigned to a project manager located at NRC Headquarters. These project managers are assigned to a section that is responsible only for permanently shutdown power reactors.

Comment #91: What does “reducing unnecessary regulatory burden” mean in practice, when it comes to emergency planning? What kind of reductions are foreseen for the following: manpower on-site/off-site, emergency equipment, communication means, alarm means, notification of personnel/public, EP, plans, KI [potassium iodide], EPZ [emergency planning zone] radius?

Response: The specific reductions in the areas mentioned is a subject that is beyond the intent of this study and will be determined during the rulemaking process. Generally speaking,

it is anticipated that on-site manpower could be reduced early in the decommissioning process provided adequate personnel are available to provide emergency response duties. Off-site manpower needs, equipment, communication, alarms, notifications, plans, and planning areas, would be relaxed consistent with the relaxation of requirements for off-site emergency planning. The consideration of the use of KI would not be necessary when iodine releases are no longer a concern.

Comments #92 and 93: (A) It is difficult to figure out how this effort fits into the overall big picture of what the NRC is doing on decommissioning. (B) A commenter asked the staff to "look at all of the activities that happen during decommissioning when developing regulations, not just a narrow view of the spent fuel pool."

Response: The focus of the decommissioning spent fuel pool risk study was intentionally limited to address potential severe accidents associated only with spent fuel. An additional rulemaking effort, termed the regulatory improvement initiative, is planned by the NRC and will include a comprehensive look at all decommissioning regulations to determine if any additional changes are required. An overall assessment of decommissioning issues and other activities that take place at decommissioning sites will be addressed during this subsequent effort.

Comments #94 and 95: (A) A commenter stated that he was confused on the way Part 50 is being applied in places where Part 72 might be more applicable. (B) Why does the NRC apply Part 50 (reactor) regulations to decommissioning reactors when the rules in Part 72 for storage of high-level waste are more clearly outlined? Part 50 regulations are not appropriate for long-term storage of high-level waste.

Response: Although 10 CFR Part 50 was developed with the operating power reactors in mind, many of the requirements still apply to decommissioning power reactors. The NRC believes that the 10 CFR Part 50 regulations applicable to decommissioning reactors are sufficient to assure public health and safety. The Part 50 license allows for safe storage of spent fuel in a spent fuel pool during operation and the staff believes that license remains adequate for spent fuel pool storage during decommissioning. The staff does not require a Part 50 licensee to obtain a Part 72 license for spent fuel storage in a spent fuel pool. All reactor decommissioning activities will remain under the Part 50 license until the decommissioning is completed and the license is formally terminated. When a licensee chooses to store spent fuel in an independent spent fuel storage installation, then the requirements of Part 72 license and regulations will be applicable.

In SECY-99-168, dated June 30, 1999, the NRC staff proposed to the Commission that all NRC regulations under Title 10 be reviewed and modified as necessary to ensure proper applicability to decommissioning. At the direction of the Commission, the staff is currently performing a comprehensive review of all applicable NRC regulations that may need modification to more effectively address decommissioning reactors.

Comment #96: Although NRC and EPA disagree on site remediation criteria, the commenter stated that either level would provide reasonable assurance to the public of undue risk.

Response: Resolution of the disagreement between NRC and EPA on release criteria is not within the scope of the technical study.

Comment #97: What is the applicability of 10 CFR Part 26 fitness-for-duty regulations to

decommissioning reactors?

Response: Fitness-for-duty at decommissioning facilities is one of the issues that will be evaluated by the decommissioning regulatory improvement initiative.

Comment #98: Quality assurance, emergency planning, fire protection, and application of codes and standards differs from site to site. Right now the decommissioning industry is being regulated by exemption to Part 50.

Response: The staff is planning to propose new emergency planning rules for decommissioning reactors to eliminate the need for addressing the issue on a plant-specific basis by processing exemptions. A final regulatory guide on decommissioning reactor fire protection programs is expected to be issued in 2001. The remaining issues will be addressed by the decommissioning regulatory improvement initiative. The planned rulemaking and guidance should assist in making regulations predictable and consistent.

Comment #99: The issue of on-site disposal of clean waste (rubblization) needs clarification.

Response: The development of NRC policy on rubblization is now ongoing in the Office of Nuclear Materials Safety and Safeguards.

Comment #100: A commenter requested an adjudicatory hearing and a prior NRC review/approval step at the onset of the decommissioning process.

Response: This issue of a hearing and NRC review and approval prior to decommissioning has been previously considered by the Commission. The Commission addressed the issue in the statements of consideration for the rulemaking for decommissioning published July 29, 1996, in the *Federal Register* (61 FR39278) by stating: "...initial decommissioning activities (dismantlement) are not significantly different from routine operational activities such as replacement or refurbishment. Because of the framework of regulatory provisions embodied in the licensing basis for the facility, these activities do not present significant safety issues for which an NRC decision would be warranted." Therefore, an NRC review and approval process that allows a public hearing before decommissioning begins is not necessary. However, in the 1996 rulemaking the Commission decided to offer a public hearing opportunity later in the decommissioning process at the license termination stage when issues such as to the adequacy of site cleanup could be raised.

Comment #101: A commenter felt that the NRC should hire a contractor to determine why/how 10 CFR Part 50 was contorted to fit decommissioning reactors with the duct tape of 10 CFR 50.82 to avoid adjudicatory processes with regulatory handles.

Response: When the NRC issued decommissioning regulations in 1988, it was assumed that decommissioning would normally take place after the facility's operating license expired. The licensee was obligated to submit a preliminary decommissioning plan 5 years before the license expired. The preliminary decommissioning plan contained a cost estimate for decommissioning and an up-to-date technical assessment of the factors that could affect planning for decommissioning. This included (1) the decommissioning alternative selected, (2) the major technical actions necessary to carry out decommissioning safely, (3) the current situation with regard to disposal of high-level and low-level radioactive waste, (4) the residual radioactivity criteria, and (5) other site-specific factors that could affect decommissioning

planning and cost.

The 1988 rule also required that no later than 1 year before expiration of the license (or within 2 years of permanent cessation of operations for plants closing before their license expires), a licensee had to submit an application for authority to decommission the facility. The application was to be accompanied by or preceded by a proposed decommissioning plan. The proposed decommissioning plan was to include (1) the alternative selected for decommissioning with a description of the activities involved, (2) a description of controls and limits on procedures and equipment to protect occupational and public health and safety, (3) a description of the planned final radiation survey, (4) an updated cost estimate for the chosen alternative and a plan for ensuring the availability of adequate funding, and (5) a description of the technical specifications, quality assurance provisions, and physical security plan provisions in place during decommissioning. A supplemental environmental report that described any substantive environmental impacts that were anticipated but not already covered in other environmental impact documents was also required.

The NRC would review the decommissioning plan and would approve it by issuing an order if the plan demonstrated that the decommissioning would be performed in accordance with regulations and there were no security, health, or safety issues. The NRC would also require that notice be given to interested persons. However, the NRC could add other conditions and limits to the plan that it deemed appropriate. The license would then be terminated if the NRC determined that the decommissioning had been performed in accordance with the approved decommissioning plan and the order authorizing decommissioning, and if the final radiation survey and associated documentation demonstrated that the facility and site were suitable for release for unrestricted use.

In August 1996 the regulations were revised for several reasons. First, the experience gained in the early decommissioning activities associated with several facilities did not reveal any activities that required NRC review and approval of a decommissioning plan. Second, environmental impacts associated with decommissioning those early facilities resulted in impacts consistent with those evaluated in the "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," NUREG-0586. And finally, experience gained from reviewing numerous decommissioning oversight activities at a number of these facilities also indicated that the decommissioning activities were in general no more complicated than activities normally undertaken at operating reactors without prior and specific NRC approval. The revised rule redefined the decommissioning process and required licensees to provide the NRC with early notification of planned decommissioning activities at their facilities went into effect. The rule made the decommissioning process more efficient and uniform. It provided for greater public awareness and clarified the opportunity for participation in the decommissioning process. It also gave plant personnel a clearer understanding of the process for changing from an operating organization to a decommissioning organization.

Comment #102: Untrained NRC public representatives frequently misinform the public, particularly about the opportunities for a hearing on reactor decommissioning.

Response: The NRC endeavors to train all NRC employees for their specific work assignments. In the event that misinformation is inadvertently communicated by an individual staff member, the NRC staff upon identifying the misinformation provides the correct information in the most expedient manner.

Comment #103: A commenter cited several specific examples of interactions with NRC staff that he felt demonstrated improper or inaccurate information provided by NRC staff members.

Response: In the course of oral communication with the public in an open and unrestrained fashion, errors, misspoken words, and misunderstandings may occur by the individuals from the public and the NRC staff. The NRC endeavors to minimize these miscommunications from our staff, but should they occur, NRC staff will act to correct them by the most expedient means available.

Comment #104: The time delays experienced by licensees who must submit individual heatup analyses and applications for exemption from NRC regulations could be mitigated by preparation of such documentation well in advance of decommissioning.

Response: It is true that decommissioning licensees who have planned reactor shutdown schedules far in advance would be able to submit exemption requests and conduct supporting thermal-hydraulic analyses in advance of reactor shutdown so that lengthy regulatory delays could be minimized. However, plants that shut down unexpectedly would not be able to submit such analyses in advance. The NRC believes that it should promulgate new decommissioning regulations that ensure public health and safety, reduce unnecessary regulatory burden and increase the efficiency and effectiveness of operations for both licensees and the NRC.

Comment #105: In a letter to the NRC, a commenter stated that the NRC staff owes its stakeholders the courtesy of addressing their concerns, particularly when comments are solicited by the NRC staff. Otherwise, the NRC staff must stop actively soliciting public comment when it has no intention of considering.

Response: At the July 15-16, 1999 public workshop on decommissioning spent fuel pool risk, the public stakeholder raised a concern that the NRC evaluate potential hazards that decommissioning accidents could impose upon plant workers. When the NRC issued its final draft report, the stakeholder's issue was not specifically addressed in the comment evaluation section. However, the NRC had received an industry decommissioning commitment that licensees would provide a remote method of adding water to spent fuel pools that would reduce potential risk to plant workers and which resulted from the issue the stakeholder had raised. The NRC seriously considers public comments received on all issues within its jurisdiction. In this case, the staff regrets the appearance that a public comment had been ignored. In order to ensure that proper consideration was given to all stakeholder comments, the NRC staff reviewed written comments received and examined transcripts of public meetings to ensure that all issues had been addressed.

Comment #106: A commenter requested on April 10, 2000, that the comment period on the spent fuel pool risk report be extended by 3 months.

Response: The original 45 day comment period ended on April 7, 2000. In a public meeting on May 9, 2000, NRC managers told the stakeholder that the comment period would be extended until June 9, 2000.

Comment #107: The NRC should identify and address possible conflicts of interests, and differing professional opinions as to the use of PRA (probabilistic risk assessment). For instance, Dr. Hanauer was quoted in a memo to say, "you can make probabilistic numbers prove anything, by which I mean that probabilistic numbers mean prove nothing."

Response: It is the policy of the Commission to maintain a working environment that

encourages the employees to make known their best professional judgements even though they may differ from a prevailing staff view. An objective of this policy is to ensure full consideration and prompt disposition of differing opinions and views by affording an independent, impartial review by qualified personnel. The content of the quote is responded to in this appendix as a separate comment.

Comment #108: The NRC should make references used in the spent fuel pool risk study available at no cost.

Response: The NRC policy is that all pertinent regulatory information is made available to the public via the Public Document Room and/or through the Agency Document and Management System (ADAMS) where this information is available for inspection at no charge. However, during the period of this study, the NRC took additional actions to provide the stakeholder with free copies of all routine correspondence and of numerous studies and reports that he specifically requested. Additionally, the NRC provided free copies of the draft June spent fuel pool risk study to all interested persons who attended the July 1999 public workshop and to all other members of the public who requested it.

Comment #109: Changes to decommissioning regulations should be made on an interim basis, to be reviewed again at some future date.

Response: The NRC does not plan to issue interim regulations for decommissioning. Rulemaking is a methodical and deliberately lengthy procedure to ensure that a rule is not issued without due process. Provisions for public comment as well as independent review committees afford ample opportunity to examine a rulemaking prior to issuing a new rule. Any person who believes an NRC regulation is no longer applicable may petition the Commission to issue rescind, or amend that regulation in accordance with 10 CFR 2.802.

Comment #110: The Draft Study completely sidesteps the question of where all the people who are relocated will be able to go for the decades that must pass while the land where they live recovers from radioactive contamination. This issue is graphically illustrated by the consequences of the Chernobyl accident, which rendered huge land areas uninhabitable and unsuitable for agriculture for an extended period of time. Finally, the Draft Study fails entirely to address the social and economic implications of losing the use of thousands of square kilometers of land for several generations.

Response: The staff agrees with the commenter that the study did not address the topics of relocation and societal impacts, such as land interdiction. As part of its original licensing review, every operating plant had an environmental impact statement that addressed land use for the area surrounding that plant. When a plant enters decommissioning, an environmental assessment is performed to determine whether activities will remain bounded by that environmental impact statement.

The calculations in support of this risk study were performed following the principles and approach of Regulatory Guide (RG) 1.174, “An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis,” which does not include environmental considerations. While overall societal risk is not considered directly in RG 1.174, a large early release is used to gauge the severity of the event outside of the plant boundary. Similarly, in this study, early and latent effects were directly calculated and reported.

The Commission recently considered whether an additional agency safety goal or objective was needed to directly address land contamination and overall societal risk. It was decided by the Commission that the current policy would not change. For further discussion, read SECY-00-0077, dated March 30, 2000, and staff requirements memorandum dated June 27, 2000. Consistent with Commission guidance, the staff does not plan to include this issue in the study.