February 15, 2000

MEMORANDUM TO: Those on the Attached List

FROM: Richard F. Dudley, Jr., Senior Project Manager ***/RA/***
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Project Directorate IV & Decommissioning
Office of Nuclear Reactor Regulation

SUBJECT: TRANSMITTAL OF DRAFT FINAL TECHNICAL STUDY OF SPENT FUEL POOL ACCIDENT RISK AT DECOMMISSIONING PLANTS AND FEDERAL REGISTER NOTICE REQUESTING PUBLIC COMMENTS ON TECHNICAL STUDY

Attached is the “Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Plants.” Also provided, for your information, is a copy of the *Federal Register* notice requesting public comments on the subject report.

Project 689

Attachments: As stated (2)

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Addressees - Memorandum dated February 15, 2000

SUBJECT: TRANSMITTAL OF DRAFT FINAL TECHNICAL STUDY OF SPENT FUEL POOL ACCIDENT RISK AT DECOMMISSIONING PLANTS AND FEDERAL REGISTER NOTICE REQUESTING PUBLIC COMMENTS ON TECHNICAL STUDY

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7590-01-P

NUCLEAR REGULATORY COMMISSION

Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants

AGENCY: Nuclear Regulatory Commission.

ACTION: Request for public comments.

SUMMARY: The Nuclear Regulatory Commission (NRC) is soliciting public comment on the "Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants."

As the number of power reactors involved in the decommissioning process increases, the ability to address regulatory issues generically has become more important. After a nuclear power plant permanently shuts down and the reactor is defueled, the traditional accident sequences that dominate operating reactor risk are no longer applicable.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.

Following a Commission meeting held on March 17, 1999, the NRC staff formed a technical working group to evaluate spent fuel pool accident risk at decommissioning plants. The staff set out to develop a risk-informed technical basis that could be used to develop rulemaking and to establish a predictable method for reviewing future exemption requests and to identify the need for any research in areas of large uncertainty. The staff intends for this approach to meet the NRC outcome goals of maintaining safety, reducing unnecessary regulatory burden, increasing public confidence, and improving efficiency and effectiveness.

The NRC released a preliminary draft version of the technical basis study in June 1999. A public workshop to discuss the report was held in July 1999. Numerous comments and other input were received from the public and the nuclear industry. The NRC has revised the report to address the comments received and to incorporate needed changes based on the results of outside technical reviews.

All comments received will be considered before the final report is published as the basis for initiation of rulemaking activities.

DATES: The comment period expires April 7, 2000. Comments received after this date will be considered if practical to do so, but the staff is able to assure consideration only for comments received on or before this date.

ADDRESSES: Mail written comments to Chief, Rules and Directives Branch, Division of Administrative Services, Office of Administration, U. S. Nuclear Regulatory Commission, Washington, DC 20555-0001. Comments may also be sent electronically by completing the online comment form available on the NRC Internet web page at the address below. Comments may also be hand delivered to Room 6D59, Two White Flint North, 11555 Rockville Pike, Rockville, Maryland, between 7:30 a.m. and 4:15 p.m. on Federal workdays.

The report is available at the NRC Public Document Room, 2120 L Street NW., Washington, DC. and through the NRC Agencywide Documents Access and Management System (ADAMS). The report is also available via the Internet on the NRC web page at <http://www.nrc.gov/NRC/REACTOR/DECOMMISSIONING/SF/index.html>. Requests for single copies may be made to Richard Dudley, U. S. Nuclear Regulatory Commission, Mail Stop O-11D19, Washington, DC 20555-0001 or by telephone at 301-415-1116 or email to rfd@nrc.gov.

FOR FURTHER INFORMATION CONTACT: George Hubbard, U. S. NRC, Office of Nuclear Reactor Regulation, Mail Stop O-11A11, Washington, DC, 20555-0001; telephone 301-415-2870; email: gth@nrc.gov.

Dated at Rockville, Maryland, this 15th day of February, 2000.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Stuart A. Richards, Director
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

DRAFT FOR COMMENT
Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants

February 2000

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Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants

Table of Contents

Appendices

List of Figures

List of Tables

Executive Summary

This report documents an evaluation of spent fuel pool (SFP) accident risks at decommissioning plants. It was done to provide an interim, risk-informed technical basis for reviewing exemption requests, and to provide a regulatory framework for integrated rulemaking. The application of this report is intended to eliminate to the extent practical, unnecessary regulatory burden, while maintaining safety and improving efficiency and effectiveness of the regulatory process. By establishing a consistent, predictable process fully open to public observation and comment, the agency intends to enhance public confidence in the regulatory process for decommissioning reactors. The report was initiated by the Commission when they asked the staff to consider whether the risk from decommissioning plants was low enough to justify generic regulatory relief in the areas of emergency planning, insurance indemnification and safeguards.

The current body of NRC regulations pertaining to light-water reactors (10 CFR 50) [Ref. 1] is primarily directed towards the safety of operating units. It is generally understood that this body of regulations is conservative when a plant transitions from an operating to a decommissioning status. In the past, decommissioning plants have requested exemptions to certain regulations as a result of their permanently defueled condition. When evaluating the acceptability of exemption requests from regulations for permanently shutdown plants, the staff has assessed the susceptibility of the spent fuel to a zirconium fire accident. To date, exemptions have been granted on a plant-specific basis, resulting in different analyses and criteria being used for the basis of the exemptions. In some cases, heat up evaluations of the spent fuel cooled only by air were requested. This criterion was used because of national laboratory studies that had identified the potential concern for a significant off-site radiological release from a zirconium fire which could occur when all water is lost from the spent fuel pool. A clad temperature of 565°C, based on the onset of clad swelling, was used as a limit to ensure no radiological release.

In March 1999, the staff formed a technical working group to evaluate spent fuel pool accident risks at decommissioning plants. A two month effort was launched to review the available technical information and methods and identify areas in need of further work. A substantial effort was made to involve public and industry representatives throughout the entire effort. A series of public meetings was held with stakeholders during and following the generation of a preliminary draft study that was published in June at the request of the Nuclear Energy Institute (NEI). The partially completed DRAFT report was released to facilitate a stakeholder/NRC two day workshop that was held in July 1999. Information gained at the workshop and through other stakeholder interactions was constructive in completing this report.

Estimates of the risk from heavy load handling accidents were revised and criticality concerns were addressed in response to stakeholder feedback. A checklist was developed to establish seismic capability of SFPs, and industry commitments were documented to address the vulnerabilities that had been identified by the June 1999 draft report. Independent technical quality reviews of controversial aspects of the report were initiated to bring in outside expert opinion on the details of the report. These experts evaluated several areas of the report, including the human reliability analysis, seismic considerations, thermal-hydraulic calculations, and probabilistic risk assessment (PRA) assumptions and treatment. The PRA results were re-quantified to take into account the industry commitments to reduce risk vulnerabilities.

This report contains the results of these activities and includes three main outputs. The first is a discussion in Chapter 2 on how risk-informed decision making can be applied to decommissioning plants. The second is a summary in Chapter 3 of the risk assessment of SFPs at decommissioning plants. The third output or Chapter 4 provides the implications of SFP risks on regulatory requirements, and outlines where industry commitments in combination with additional staff assumptions may be useful in improving spent fuel pool safety at decommissioning plants. Chapter 5 is a summary of the findings of the report.

After a period of one year following permanent shutdown, the results of this report estimated the generic frequency of events leading to zirconium fires at decommissioning plants to be less than $3x10^{-6}$ per year for a plant that implements the design and operational characteristics assumed in the risk assessment performed by the staff. This frequency was estimated based on the assumptions that the characteristics of the ten industry decommissioning commitments (IDCs) proposed by NEI (See Appendix 6) and the four staff decommissioning assumptions (SDAs) identified in Chapters 3 and 4 of the report would be implemented. This estimate could be much higher for a plant that does not implement these characteristics. The most significant contributor to this risk is a seismic event which exceeds the design basis earthquake. However, the overall frequency of this event is within the staff recommended pool performance guideline (PPG) identified in this report for large radiological releases due to a zirconium fire of $1x10^{-5}$ per year. As discussed below, zirconium fires are estimated to be similar to large early release accidents postulated for operating reactors in some ways, but less severe in others.

The thermal-hydraulic analysis presented in Appendix 1 demonstrates that the decay heat necessary for a zirconium fire exists in typical spent fuel pools of decommissioning plants for a period of several years following shutdown. The analysis shows that the length of time over which the fuel is vulnerable depends on several factors, including fuel burn-up and fuel storage configuration in the SFP. In some cases analyzed in Appendix 1, the required decay time to preclude a zirconium fire is 5 years[1]. However, the exact time will be plant specific; therefore, plant-specific analysis would be needed to demonstrate shorter zirconium fire vulnerabilities.

The consequence analysis presented in Appendix 4 demonstrates that the consequences of a zirconium fire in a decommissioning plant can be very large. The integrated dose to the public is generally comparable to a large early release from an operating plant during a potential severe core damage accident and early fatalities are very sensitive to the effectiveness of evacuation. For a decommissioning plant with about one year of decay time, the onset of radiological releases from a zirconium fire is significantly delayed compared to those from the most limiting operating reactor accident scenarios. This is due to the relatively long heat up time of the fuel. For many of the sequences leading to zirconium fires, there are very large delay times due to the long time required to boil off the large spent fuel pool water inventory. Thus, while the consequences of zirconium fires are in some ways comparable to large early releases from postulated reactor accidents, the time of release occurs much later following

[1]In the area of dry storage, it is noted that currently certified casks may be loaded with spent fuel with a minimum of five years cooling. The risk of a zirconium fire in dry cask storage is largely eliminated by limiting the maximum fuel cladding temperature and minimizing the oxygen available. The temperature is explicitly modeled using bounding fuel characteristics. The maximum clad temperature occurs during vacuum drying when little oxygen is available and the fuel is in an inert environment for storage.

initiation of the accident. Therefore, this analysis indicates that for the slowly evolving SFP accident scenarios at decommissioning plants, there is a large amount of time to initiate and implement protective actions, including public evacuation in comparison to operating reactor accident sequences.

In summary, the risk assessment shows low numerical risk results in combination with satisfaction of the safety principles as described in R.G. 1.174, such as defense-in-depth, maintaining safety margins, and performance monitoring. The staff concludes that under the assumptions of this study there is a low level of public risk from SFP accidents at decommissioning plants. In addition, the study shows that, after a period of one year following final shutdown, the low likelihood that a zirconium fire would occur, in combination with the long time frames available for taking off-site protective actions, provides a basis for relaxation of emergency planning requirements.

Chapters 4 also addresses the report's implications on security and insurance provisions at decommissioning plants. For security, the risk insights can be utilized to assess what target sets are important to protect against sabotage. However, any reduction in security provisions would be constrained by an effectiveness assessment of the safeguards provisions against a design basis threat. Therefore, the staff concludes that some level of security is required as long as the fuel in the SFP is exposed to a sabotage threat. For insurance, the reports points out that no definitive criteria exists that would allow relaxation on the basis of low event probability alone while the potential for a zirconium fire exists. Finally, inconsistencies were identified in current regulations regarding vehicle-borne bomb threats and insurance indemnification requirements for ISFSIs and decommissioning nuclear power plants. These inconsistencies should be revisited during the overall integration of rules for decommissioning plants.

In summary, the report provides a comprehensive treatment of SFP risks at decommissioning plants as it relates to emergency planning, insurance, and security requirements. The report systematically examines the differences between an operating reactor spent fuel pool configuration, and the typical SFP configuration in place at one year post-shutdown for a decommissioning plant. It provides the technical basis for determining the regulatory requirements for decommissioning plants using risk-informed decision making.

1.0 Introduction

The current body of Nuclear Regulatory Commission (NRC) regulations pertaining to light-water reactors (10 CFR 50) [Ref. 1] is primarily directed towards the safety of operating units. As some reactors have reached permanent shutdown condition and entered decommissioning status, the NRC has been faced with establishing the appropriate requirements and regulatory oversight necessary to provide adequate protection to the public. For decommissioning plants, the potential public risk is due primarily to the possibility of a zirconium fire associated with the spent fuel rod cladding. Due to the decay heat generated by the spent fuel, it must be continuously cooled and remain covered by water after removal from the reactor. A postulated event could occur if the systems which provide heat removal from the fuel pool fail, causing the cooling water to boil off from the pool until the fuel is uncovered. Alternatively, a leak in the pool could occur, that if not corrected, could also result in the spent reactor fuel becoming uncovered. For either scenario, the uncovered and uncooled spent fuel could heat up causing a fire of its zirconium cladding and releasing large quantities of radionuclides.

Decommissioning plants have requested exemptions to certain regulations as a result of their permanently defueled condition. While the current Part 50 regulatory requirements (developed for operating reactors) ensure safety at the decommissioning facility, some of these requirements may be excessive and not substantially contributing to public safety. Areas where regulatory relief has been requested in the past include exemptions from off-site emergency planning (EP), insurance, and safeguards requirements. Requests for consideration of changes in regulatory requirements are appropriate since the traditional accident sequences that dominate operating reactor risk are no longer applicable. For a defueled reactor in decommissioning status, public risk is predominantly from potential accidents involving spent fuel. Spent fuel can be stored in the spent fuel pool (SFP) for considerable periods of time, as remaining portions of the plant continue through decommissioning and disassembly. To date, exemptions have been requested and granted on a plant-specific basis. This has resulted in some lack of consistency and uniformity in the scope of evaluations conducted and acceptance criteria applied in processing the exemption requests.

To improve regulatory consistency and predictability, the NRC has undertaken this effort to improve the regulatory framework applicable to decommissioning plants. This framework will utilize risk-informed approaches to identify the design and operational features necessary to ensure that risks to the public from these shutdown facilities are sufficiently small. This framework will form the foundation upon which regulatory changes will be developed, as well as the basis for requesting and approving exemption requests in the interim, until the necessary rulemaking is completed.

In support of this objective, the NRC staff has completed a draft assessment of spent fuel pool risks. This assessment utilized probabilistic risk assessment (PRA) methods and was developed from analytical studies in the areas of thermal hydraulics, core physics, systems analysis, human reliability analysis, seismic and structural analysis, external hazards assessment, and off-site radiological consequences. The focus of the risk assessment was to identify potential severe accident scenarios at decommissioning plants, and to estimate the likelihood and consequences of these scenarios. Of primary concern are events that lead to loss of spent fuel pool water inventory or loss of cooling to the spent fuel assemblies, and events that result in fuel configurations that could lead to criticality conditions. For some period after reactor shutdown and after sustained loss of inventory or cooling, it is possible for the fuel to heat up to the point where rapid oxidation and burning of the zirconium fuel cladding occurs leading to significant releases of radionuclides.

A preliminary version of this draft report was issued for public comment and technical review in June 1999. Comments received from stakeholders and other technical reviewers have been considered in preparing this assessment. Quality assessment of the staff's preliminary analysis has been aided by a small panel of human reliability analysis (HRA) experts who evaluated the human performance analysis assumptions, methods and modeling, as well as a broad quality review carried out at the Idaho National Engineering & Environmental Laboratory (INEEL).

The conclusions and findings of the study provide guidance for the design and operation of spent fuel pool cooling and inventory make-up systems as well as practices and procedures necessary to ensure high levels of operator performance during off-normal conditions. This report concludes that, with the fulfillment of industry commitments and satisfaction of a number of additional staff assumptions, the risks from spent fuel pools will be sufficiently small to justify exemptions from selected current regulatory requirements and to form the basis for related rulemaking.

This report is divided into three main parts. The first is a discussion in Chapter 2 on how risk-informed decision making can be applied to decommissioning plants. The second is a summary in Chapter 3 of the risk assessment of SFPs at decommissioning plants. The third in Chapter 4 provides the implications of SFP risk on regulatory requirements, and outlines where industry commitments in combination with additional staff assumptions may be useful in improving spent fuel pool safety**.**

2.0 Risk-Informed Decision Making

The regulatory framework proposed in this report for decommissioning plants is based on a risk-informed process. In 1995, the NRC published its PRA Policy Statement [Ref 1], which stated that the use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art of the methods. Probabilistic risk assessment provides a structured analytical method to assess the various combinations of failures and events that result in undesirable consequences, such as core damage in an operating reactor. The end points of PRAs can be extended to include public health effects by modeling the timing and mode of containment failure and radioactive releases to the environment.

Subsequent to issuance of the PRA Policy Statement, the agency published Regulatory Guide (RG) 1.174 [Ref.2] which contained general guidance for application of PRA insights to the regulation of nuclear reactors. 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 due to regulatory decisions 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 which increase LERF by $1x10^{-6}$ per year.

For decommissioning plants, the risk is primarily due to the possibility of a zirconium fire associated with the spent fuel rod cladding[2]. The consequences of such an event do not equate exactly to either a core damage accident or a large early release[3]. Zirconium fires in spent fuel pools potentially have more severe long term consequences than an operating reactor core damage accident, because there may be multiple cores involved, and because 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 very slowly (allowing time for protective actions to be taken to significantly reduce early fatalities), and the absence of short lived isotopes in the release (e.g., iodine isotopes will have decayed away though early health effects are still possible from Cesium isotopes). As a result, the criteria of RG 1.174 cannot be applied directly to the risk of a decommissioning plant.

Even though the event progresses more slowly than an operating reactor large early release event and the isotopic make-up is somewhat different, the risk assessment consequence calculations performed by the staff[4] (assuming multiple cores) show that large inventories of radioisotopes could be released that could have significant late health effects (latent cancers) for the population at some distance from the plant, as well as the potential for a small number of early fatalities. The staff has therefore decided that the end state and consequences of a spent fuel pool fire are sufficiently severe 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 frequency guideline for a decommissioning plant SFP risk, and a useful tool to be used in combination with other factors such as accident progression timing, to assess features, systems and operator performance needs of a spent fuel pool in a decommissioning plant. The staff therefore proposes $1x10^{-5}$ per year as the recommended pool performance guideline (PPG) for baseline zirconium fire frequency. In its letter of November 12, 1999 [Ref. 3], the Advisory Committee on Reactor Safeguards (ACRS) recommended that application of the LERF guideline as discussed above be utilized. The staff agrees with this recommendation.

2.1 Principles of Regulatory Guide 1.174

As discussed in RG 1.174, quantitative risk assessment is only one tool utilized in risk-informed decision making. RG 1.174 articulates the following safety principles which should be applied to the decommissioning case:

- “The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change, i.e., a “specific exemption” under 10 CFR 50.12 or a “petition for rulemaking” under 10 CFR 2.802.

- The proposed change is consistent with the defense-in-depth philosophy.

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[2]See chapter 3 for more complete discussion of fuel pool risk scenarios

[3]RG 1.174 describes LERF as the frequency of unmitigated releases that have the potential for early health effects, in a time frame prior to effective evacuation of close-in population

[4]See Appendix 4 for consequence and health impact assessment

- The proposed change maintains sufficient safety margins.

- When proposed changes result in an increase in core damage frequency and/or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement

- The impact of the proposed change should be monitored using performance measurement strategies."

While the focus of RG 1.174 was decision-making regarding changes to the licensing basis of an operating plant, the same risk-informed philosophy can be applied to rulemaking for decommissioning plants or to consider potential exemptions to current requirements. The intent and scope of these safety principles are discussed below. However, since the application of this study specifically relates to exemptions to a rule or a rule change for decommissioning plants, a discussion of the first principle regarding current regulations is not necessary nor is it provided. A discussion on how the rest of these principles are satisfied as demonstrated by the staff's safety assessment is provided in Chapter 4.

2.1.1 Defense-in-Depth

Defense-in-depth describes a multi-layered design and operational philosophy whose goal is to prevent the initiation of accidents or to prevent their progression to serious consequences. The defense-in-depth philosophy applies to the operation of the spent fuel pool, whether at an operating plant or in a decommissioning plant. In accordance with the Commission White Paper on Risk-Informed Regulation (March 11, 1999), "Defense-in-depth is an element of the NRC's Safety Philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. The defense-in-depth philosophy ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of a nuclear facility. The net effect of incorporating defense-in-depth into design, construction, maintenance and operation is that the facility or system in question tends to be more tolerant of failures and external challenges."

Therefore, application of defense-in-depth could mean in part that there is more than one source of cooling water or that pump make-up can be provided by both electric as well as direct drive diesel pumps. Additionally, defense-in-depth can mean that even if a serious outcome (such as fuel damage) occurs, there is further protection such as containment to prevent radionuclide releases to the public. However, implementation of defense-in-depth for SFPs is different from that applied to nuclear reactors because of the different nature of the hazards. The robust structural design of a fuel pool, coupled with the simple nature of the pool support systems, goes far toward preventing accidents associated with loss of water inventory or pool heat removal. Additionally, because the essentially quiescent (low temperature, low pressure) initial state of the spent fuel pool and the long time available for taking corrective action associated with most release scenarios provide significant safety margin, a containment structure is not considered necessary as an additional barrier to provide an adequate level of protection to the public. Likewise, the slow evolution of most SFP accident scenarios allows for reasonable human recovery actions to respond to system failures. Chapter 4 summarizes the specific design and operational features of the SFP, industry commitments and the additional staff assumptions that ensure that SFP defense-in-depth is maintained. This level of defense is achieved through preventative measures, appropriate mitigating systems, and an appropriate level of emergency planning.

2.1.2 Safety Margins

A safety margin can relate to the difference between the expected value of some physical parameter (e.g., temperature, pressure, stress, reactivity) and the point at which adequate performance is no longer assured. An example of this would be a containment pressure calculation which may show a peak accident pressure of 40 psig is reached for a structure which has a design capability of 60 psig and an actual ultimate capability of 110 psig. In this case there is margin from the accident calculation of 20 psig to the design limit as well as a large margin of 70 psig to the actual expected failure limit.

The safety margins associated with fuel in the spent fuel pool for many physical processes and parameters are much greater than those associated with an operating reactor. The spent fuel pool is in a quiescent state, at or near ambient temperature and pressure. The decay heat levels are much lower than those of the fuel in an operating reactor. This allows much greater time for heating and boil off of the coolant water, and for heat up of the fuel itself, once uncovered. The fuel is covered with approximately 23 feet of water at or near ambient temperature. The pool is designed with ample margin to criticality, using both passive (geometry) and active (poisons) means of reactivity control. Chapter 4 describes the provisions that ensure the SFP maintains adequate safety margins in a decommissioning plant.

2.1.3 Impact of Proposed Changes

The impact of the proposed change should be small. As discussed above, the staff is applying the pool performance guideline (PPG) of $1x10^{-5}$ per year frequency for a zirconium fire, which was developed from the treatment for LERF in RG 1.174 and a change guideline of $1X10^{-6}$ per year (assuming that the $1X10^{-5}$ per year PPG is already met). This PPG is used to assess the impact and acceptability of SFP risk in decommissioning plants. Chapters 3 and 4 discuss the design and operational characteristics of the SFP that are relied upon to produce the low baseline risk results. These are identified in the context of industry commitments as well as additional staff assumptions needed to produce the low SFP risk conclusions.

2.1.4 Implementation and Monitoring Program

RG 1.174 states that an implementation and monitoring plan should be developed to ensure that the engineering evaluation conducted to examine the impact of the proposed changes continues to reflect the actual reliability and availability of structures, systems, and components (SSCs) that have been evaluated. This will ensure that the conclusions that have been drawn will remain valid.

Therefore, with respect to all the above safety principles, implementation and monitoring of important considerations could include such actions as: comparing a check list against the spent fuel pool seismic design and construction; control of heavy load movements; development and implementation of procedures and other provisions to ensure human reliability; monitoring the capability, reliability, and availability of important equipment; and checking the effectiveness of on-site emergency response and plans for communication with off-site authorities. In many areas the implementation and monitoring may already be accomplished by utility programs such as those developed under the maintenance rule [Ref. 4].

3.0 Risk Assessment of Spent Fuel Pools at Decommissioning Plants

As discussed in Chapter 1 of this paper, the risks from a decommissioning plant are very different from an operating plant. Once fuel is permanently removed from the reactor vessel, the primary public risk in a decommissioning facility is associated with the spent fuel pool. The spent fuel assemblies are retained in the storage pool, and are submerged in water to provide cooling of the fuel's remaining decay heat as well as to provide shielding for the radioactive assemblies. The most severe accidents postulated for SFPs are associated with the loss of water (either through boil-off or draining) from the pool.

Depending on the time since reactor shutdown and fuel rack configurations, there may be sufficient heat to cause the clad to heat up over time, swell and burst in the event of loss of pool water. The breach in the clad would result in the release of radioactive gases present in the gap between the fuel and clad, called "a gap release" (See Appendix 1). If the fuel continues to heat up, the temperature of the zirconium clad will reach the point of rapid oxidation in air. This reaction of zirconium and air is exothermic. The energy released from the reaction combined with the fuel's decay energy can cause the reaction to become self-sustaining and lead to the ignition of the zirconium, or a "zirconium fire." The increase in heat from the oxidation reaction could also raise the temperature in adjacent fuel assemblies and cause the propagation of the oxidation reaction. This zirconium fire would result in a significant release of the fission products contained in the spent fuel, which would be dispersed from the reactor site due to the thermal plume from the zirconium fire. Consequence assessments (Appendix 4) have shown that such a zirconium fire could have significant latent health effects (cancers) as well as the possibility of a small number of early fatalities. Gap releases for fuel from a reactor that has been shut down more than a year release only moderately small quantities of radionuclides, in the absence of a zirconium fire, and would only be of concern for on-site effects.

Based upon the preceding insights, the staff conducted its risk evaluation to estimate the likelihood of credible accident scenarios that could result in loss of pool water and fuel heat up to the point of rapid oxidation. In addition to developing an order-of-magnitude assessment of the level of risk associated with SFPs at decommissioning plants, the objective of this risk assessment included the identification of potential vulnerabilities, and the design and operational characteristics that would minimize these vulnerabilities. Since the decay time at which air cooling alone is sufficient to prevent a zirconium fire is very plant specific, the cut off time (when a zirconium fire can no longer occur) for this risk assessment cannot be pre-determined. Rather, the insights should be considered as generally applicable to a decommissioning plant until the spent fuel decay heat level decreases to a point where rapid oxidation would not occur with complete loss of water. After a decay period that precludes fuel heat up to zirconium fire conditions, no significant risk remains from storage of the spent fuel. Preliminary calculations (see Appendix 1) show this time will vary depending on fuel burn up, SFP storage configuration and loading pattern of the assemblies, and could occur at a period as long as five years[5] from plant shutdown.

In order to support the risk evaluation, the staff conducted a thermal hydraulic assessment of the SFP for various scenarios such as loss of pool cooling and loss of inventory. These calculations provided information on heat up and boil off rates for the pool, as well as heat up rates for the uncovered fuel assemblies and timing to initiation of zirconium fire for a number of scenarios and sequences. The results of these calculations provided fundamental information on the timing of accident sequences and provided insights on the time available to recover from events and time available to initiate off-site measures, if necessary. This information was then utilized in the risk assessment to support the human reliability analysis used to assess the likelihood of recovering level or cooling before a zirconium fire occurs.

For these calculations, the end state assumed for the accident sequences was when the water level reached the top of the fuel assemblies, rather than calculating the temperature response of the fuel as the level gradually drops. This simplification was utilized because of the complex heat transfer mechanisms and chemical reactions occurring in the fuel assemblies that are slowly being uncovered. This analytical approach understates the time that is available for possible operator recovery of SFP events prior to initiation of a zirconium fire. However, since the recoverable events such as small loss of inventory or loss of power/pool cooling, are very slowly evolving events, many days are generally available for recovery whether the end point of the analysis is uncovery of the top of the fuel or complete fuel uncovery. The extra time available (estimated to be in the tens of hours) as the water level boils down the assemblies, would not impact the very high probabilities of operator recovery from these events given the industry commitments and additional staff assumptions. In its letter of November 12, 1999 [Ref. 1], the ACRS recommended that the end state of top of fuel uncovery be used for the SFP analysis along with application of the LERF criteria discussed in Chapter 2. The staff agrees with this recommendation. However, there are some exceptions noted in our response to the ACRS. The details of the staff thermal hydraulic assessment are provided in Appendix 1.

Prior to the staff's preliminary risk assessment, the most extensive work on spent fuel pool risk was in support of Generic Issue (GI) 82, "Beyond Design Basis Accidents for Spent Fuel Pools" [Ref. 2]. This report assessed the SFP risk for operating reactors and concluded that a seismic event was the dominant initiating event for the loss of inventory.

While the staff drew from the GI 82 work in its assessment, it was concluded that because of significant differences between operating and decommissioning plant spent fuel pool cooling systems, a complete assessment of SFP risk at decommissioning plants should be conducted, considering all potentially significant initiators, and reflecting the unique features found in a shutdown facility. The results of the staff assessments are discussed below. A summary of industry commitments, staff assumptions (relied upon in the risk assessment) and a discussion

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[5] In the area of dry storage, it is noted that currently certified casks may be loaded with spent fuel with a minimum of five years cooling. The risk of a zirconium fire in dry cask storage is largely eliminated by limiting the maximum fuel cladding temperature and minimizing the oxygen available. The temperature is explicitly modeled using bounding fuel characteristics. The maximum clad temperature occurs during vacuum drying when little oxygen is available and the fuel is in an inert environment for storage.

of how the decision criteria in Chapter 2 is satisfied are discussed in Chapter 4. Conclusions on how the SFP risk insights and decision criteria apply to potential changes in emergency planning, insurance, and safeguard are also discussed in Chapter 4.

3.1 Basis and Findings of SFP Risk Assessment

In order to follow the framework for the regulatory decision process described in Chapter 2, a comprehensive assessment of SFP risk was necessary. To gather information on SFP design and operational characteristics for the preliminary risk assessment done for the June 1999 draft report, the staff conducted site visits to four decommissioning plants to ascertain what would be an appropriate model for decommissioning spent fuel pools. The site visits confirmed that the as operated spent fuel pool cooling systems were different than those in operation when the plants were in power operation. The operating plant pool cooling and make-up systems have generally been removed and replaced with portable and skid-mounted pumps and heat exchangers. While in some cases there are redundant pumps, physical separation, barrier protection and emergency on-site power sources are no longer maintained. Modeling information for the PRA analysis was determined from system walk-downs as well as limited discussions with the decommissioning plant staff. Since limited information was collected for the preliminary assessment on procedural and recovery activities as well as what the minimum configuration a decommissioning plant might have, a number of assumptions and bounding conditions were assumed for the June 1999 preliminary study. These preliminary results have been refined in this draft assessment after obtaining more detailed information from industry on SFP design and operating characteristics for a decommissioning plant, as well as a number of industry commitments that contribute to achieving low risk findings from SFP incidents. These revised results also reflect improvements in the PRA model since publication of the June 1999 report.

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

In addition, a qualitative risk perspective was developed for inadvertent criticality in the SFP. The risk model, as developed by the staff and supplemented through a quality review from Idaho National Engineering & Environmental Laboratory (INEEL), is provided in Appendix 2. Appendix 2 also includes the modeling details for the heavy load drop, aircraft impacts, seismic and tornado missile assessments. Input and comments from stakeholders were also utilized in updating the June 1999 preliminary model to the present draft model.

3.2 Characteristics of SFP Design and Operations for a Decommissioning Plant

Based upon information gathered from the site visits and interactions with NEI and other stakeholders, the staff has modeled the spent fuel pool cooling system (SFPC) (see Figure 3.1) as being located in the SFP area and consisting of motor-driven pumps, a heat exchanger, an ultimate heat sink, a make-up tank, a filtration system and isolation valves.

Suction is taken from the spent fuel pool via one of the two pumps and is passed through the heat exchanger and returned back to the pool. One of the two pumps on the secondary side of the heat exchanger rejects the heat to the ultimate heat sink. A small amount of water from the suction line is diverted to the filtration process and is returned back into the discharge line. A manually operated make-up system (with a limited volumetric flow rate) supplements the small losses due to 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, if needed. There are two firewater pumps, one motor-driven (electric) and one diesel-driven, which provide firewater in the SFP area. A firewater hose station is provided in the SFP area. The firewater pumps are located in a separate structure.

Based upon information obtained during the site visits and discussions with the decommissioning plant personnel during those visits, the staff also made the following assumptions that are believed to be representative of a typical decommissioning facility:

- The make-up capacity (with respect to volumetric flow) is assumed to be as follows:

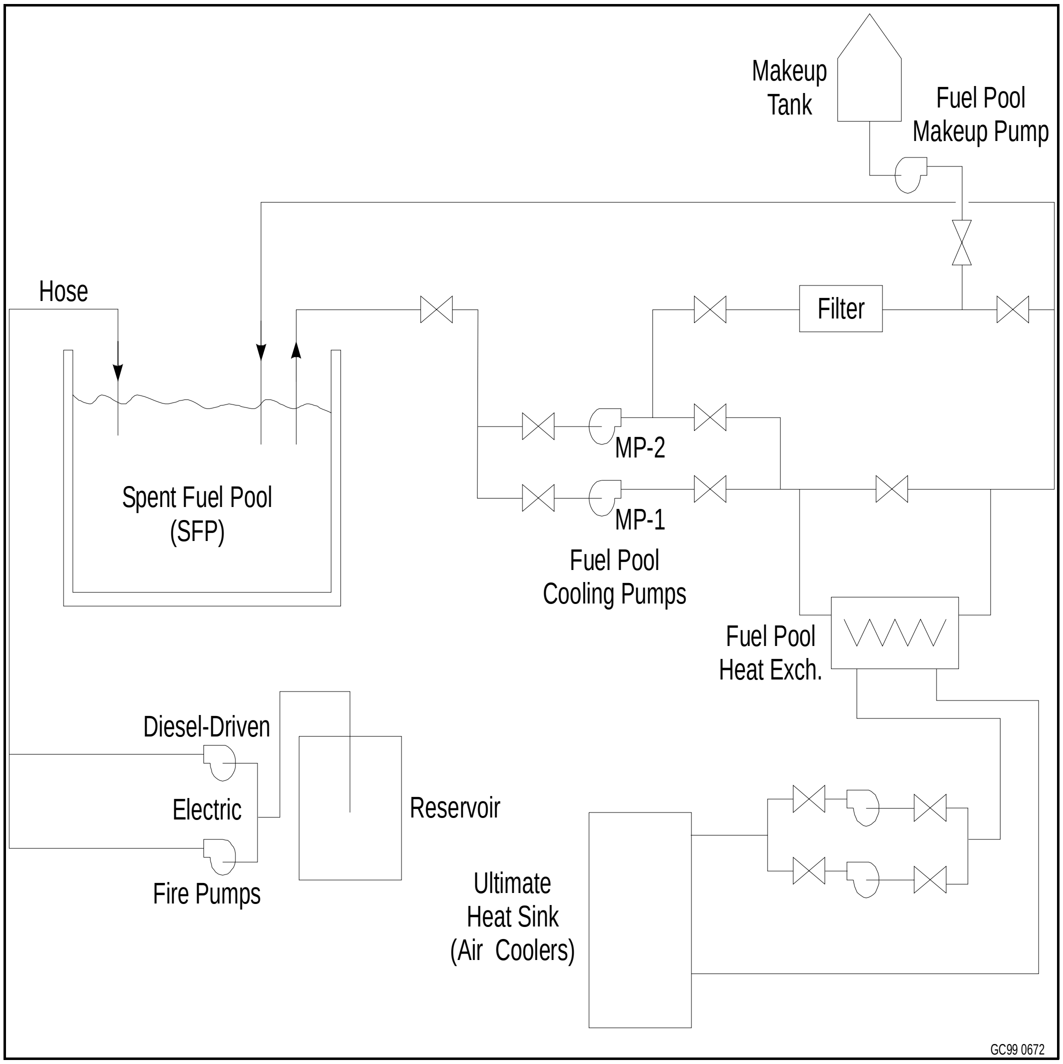
Make-up pump:	20 - 30 gpm
Firewater pump:	100 - 200 gpm
Fire engine:	100 - 250 gpm [depending on hose size: 1-½" (100 gpm) or 2-½" (250 gpm)]

  The staff also assumed that for the larger loss-of-coolant inventory accidents, water addition through the make-up pumps does not successfully mitigate the loss of inventory event unless the location of inventory loss is isolated.

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

- Plants do not have drain paths in their spent fuel pools that could lower the pool level (by draining, suction, or pumping) more that 15 feet below the normal pool operating level.

Figure 3.1 Assumed Spent Fuel Pool Cooling System



Based upon the results of the June 1999 preliminary risk analysis and its associated sensitivity cases, it became clear that many of the risk sequences were quite sensitive to the performance of the SFP operating staff in identifying and responding to off-normal conditions. This is due to the fact that the remaining systems of the SFP are relatively simple with manual rather than automatic initiation of backups or realignments. Therefore, if scenarios such as loss of cooling or inventory loss to the pool occur, operator response to diagnose the failures and bring on-site and off-site resources to bear are instrumental for ensuring that the fuel assemblies remain cooled and a zirconium fire is prevented.

As part of its technical evaluations the staff assembled a small panel of experts[6] which identified the attributes necessary to achieving very high levels of human reliability for responding to potential accident scenarios in a decommissioning plant SFP. (A discussion of these attributes and the HRA methodology used is provided in Section 3.2 of Appendix 2a.)

Upon consideration of the sensitivities identified in the staff's preliminary study and to reflect actual operating practices at many decommissioning facilities, the nuclear industry, through NEI, made important commitments (reproduced in Appendix 6) which were reflected in the staff's updated risk assessment. The revisions to the risk assessment generally reflect changes of assumptions in the areas shown below. The applicability of the specific industry decommissioning commitments (IDCs) with respect to the risk analysis results are discussed later in this chapter. How the commitments relate to specific risk conclusions and safety principles is also discussed in Chapter 4. Any future rulemaking or other regulatory activity would determine how these commitments are implemented.

Where additional operational and design considerations (beyond industry commitments) had to be assumed to ensure that the low risk estimates presented in this study are achieved, the staff identified additional staff decommissioning assumptions (SDAs) which are detailed in later sections of this report. As with the industry commitments, staff assumptions on SFP design and operational features, which were necessary to achieve the low SFP risk findings of this report, will be identified and implemented as appropriate in future regulatory activities.

Industry Decommissioning Commitments

IDC #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).

IDC #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.

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

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[6]Panel composed of Gareth Parry, U.S. NRC; Harold Blackman, INEEL; and Dennis Bley, Buttonwood Consulting

IDC #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.

IDC #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.

IDC #6 Spent fuel pool 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.

IDC #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.

IDC #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.

IDC #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.

IDC #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.

Based upon the above design and operational features, industry commitments, technical comments from stakeholders and the input from the INEEL technical review, the staff's SFP risk model was updated. The results for the initiators which were assessed quantitatively are shown in Table 3.1.

Table 3.1 Spent Fuel Pool Cooling Risk Analysis Frequency of Fuel Uncovery (per year)

INITIATING EVENT	**Frequency of Fuel Uncovery**
Loss of Pool Cooling	$1.4X10^{-08}$
Loss of Coolant Inventory	$3.1X10^{-09}$
Loss of Off-site Power - Plant centered and grid related events	$3.0X10^{-08}$
Loss of Off-site Power - Events initiated by severe weather	$1.3X10^{-07}$
Internal Fire	$4.5X10^{-08}$
Cask Drop [7]	$2.0X10^{-07}$
Seismic Event [8]	$<3.0X10^{-06}$
Aircraft Impact	$2.9X10^{-09}$
Tornado Missile	$<1.0X10^{-09}$
Total	$<3.4X10^{-06}$

This table summarizes the fuel uncovery frequency for each accident initiator. The frequencies are point estimates, based on the use of point estimates for the input parameters. For the most part these input parameter values would be used as the mean values of the probability distributions that would be used in a calculation to propagate parameter uncertainty. Because the systems are very simple with little support needs, the point estimates therefore reasonably correlate to the mean values that would be obtained from a full propagation of parameter uncertainty. Due to the large margin between the loss of cooling and inventory sequence frequencies and the pool performance guideline, this propagation was judged to be unnecessary (See Section 5 of Appendix 2a for further discussion of uncertainties).

The above results show that the estimated frequency for a zirconium fire is less than $3X10^{-6}$ per year, with the dominant contribution being from a severe seismic event. A more specific characterization of the seismic risk is discussed in Chapter 3.4.1.

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[7]For a single failure proof system without a load drop analysis. The staff assumed that facilities that chose the option in NUREG-0612 to have a non-single failure proof system performed and implemented their load drop analysis including taking mitigative actions to the extent that there would be high confidence that the risk of catastrophic failure was less than or equivalent to that of a single failure proof system.

[8]This contribution applies to SFPs that satisfy the seismic checklist and includes seismically induced catastrophic failure of the pool (which dominates the results) and a small contribution from seismically induced failure of pool support systems.

As discussed in more detail in Appendix 2, the results of the risk analysis depends on assumptions on the design and operational characteristics of the SFP facility. The inputs that have the potential to significantly influence the results are summarized below.

- The modeled system configuration is described in Chapter 3.2. The assumed availability of a diesel powered fire pump is an important element in the conclusion that fuel uncovery frequency is low for the loss of off-site power initiating events and the internal fire initiating event. The assumption of the availability of a redundant fuel pool cooling pump is not as important since the modeling of the recovery of the failed system includes repair of the failed pump, not just the startup of the redundant pump. Finally, multiple sources of make-up water are assumed for the fire pumps. This lessens the concern for possible dependencies between initiating events (e.g., severe weather events, high wind events, or seismic events) and the availability of make-up water supply (e.g., fragility of the fire water supply tank).

- Credit is taken for industry/NEI commitments as described in Chapter 3.2. Without this credit, the risk is estimated to be more than an order of magnitude higher. Specifically,

    - IDC #1 is credited for lowering the risk from cask drop accidents.

    - IDCs # 2, 3, 4, and 8 are credited for the high probability of recovery of loss of cooling (including events initiated by loss of power or fire) and loss of inventory scenarios. 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 training are explicit in giving guidance on the capability of the fuel pool make-up 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 make-up sources; and walk-downs are performed on a regular (once per shift) basis and the operators document the observations in a log. The later is important to compensate for potential failures to the instrumentation monitoring the status of the pool.

    - IDC # 5 is credited for the high probability of early identification and diagnosis (from the control room) of the loss of cooling or loss of inventory.

    - IDCs # 6, 7, and 9 are credited with lowering the initiating event frequency for the loss of inventory event from its historical levels. In addition, these commitments were used to justify the assumption that a large non-catastrophic leak rate is limited to approximately 60 gpm, and the assumption that the leak is self limiting after a drop in level of 15 feet. These assumptions may be non-conservative on a plant-specific basis depending on SFP configuration and specific commitments on configuration control.

    - IDC # 10 is credited for the equipment availabilities and reliabilities used in the analysis. In addition, if there are specific administrative procedures to control the out of service duration for the diesel fire pump, the relatively high unavailability for this pump (of 0.18) could be lowered.

- Initiating event frequencies for the loss of cooling, loss of inventory, and loss of off-site power are based on generic data. In addition, the probability of power recovery is also based on generic information. Site specific differences would proportionately affect the risk from these initiating events.

The various initiating event categories are discussed below. The staff's qualitative risk insights on the potential for SFP criticality are discussed at the end of this chapter.

3.3 Internal Event Scenarios Leading to Fuel Uncovery

The following summary is a description of the accident associated with each internal event initiator. Details of the assessment are provided in Appendix 2.

3.3.1 Loss of Cooling

The loss of cooling initiating event may be caused by the loss of coolant system flow from the failure of 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., electrical connections). While it may not be directly applicable due to design differences in a decommissioning plant, operational data from NUREG-1275, Volume 12 [Ref. 3] shows that the frequency of loss of spent fuel pool cooling events in which a temperature increase of more than 20°F occurred can be estimated to be on the order of two to three events per 1000 reactor years. The data also showed that, for the majority of events, the duration of the loss of cooling was less than one hour. Only three events exceeded 24 hours, with the maximum duration being 32 hours. There were four events where the temperature increase exceeded 20°F, with the maximum increase being 50°F.

The calculated fuel uncovery frequency for this initiating event is 1.4×10^{-8} per year. To have fuel uncovery, the plant operators would have to fail to recover the cooling system (either fails to notice the loss of cooling indications, or fails to repair or restore the cooling system). In addition, the operators would have to fail to provide make-up cooling using other on-site sources (e.g., fire pumps) or off-site sources (e.g., use of a fire brigade). For these recovery actions, there is a lot of time available. In the case of 1-year-old fuel (i.e., fuel that was in the reactor when it was shutdown one year previously), approximately 130 hours is available. Indications of a loss of pool cooling that are available to operators include: control room alarms and indicators, local temperature measurements, and eventually increasing area temperature and humidity and low pool water level from boil-off.

Based on the assumptions made, the frequency of fuel uncovery is estimated to be very low. A careful and thorough adherence to IDCs 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 of 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 make-up system, and when it becomes essential to supplement with alternative higher volume sources. The analysis also assumed that the procedures and training are sufficiently clear in giving guidance on early preparation for using the alternative make-up sources.

It should be noted that there were two recent events involving a loss of cooling at SFPs. The first, occurring in December 1998 at Browns Ferry Unit 3, involved a temperature increase of approximately 25°F over a two day period. This incident, caused by the short cycling of cooling water through a stuck-open check valve, was not detected by the control room indicators due to a design flaw in the indicators. In the second event, occurring in January 2000, the SFP temperature increased by approximately 40 to 50°F at the Duane Arnold Unit 1 plant. The incident, which was undetected for approximately two and a half days, was caused by operator failure to restore the SFP cooling system heat sink following maintenance activities. At this plant, there was no alarm for high fuel pool temperature, although temperature indicators are available in the control room. Since the conditional probability of fuel uncovery is low given a loss of cooling initiating event, the addition of these two recent events to the database will not affect the conclusion that the risk from these events is low. However, the recent events further illustrate the importance of industry commitments, particularly IDC # 5 which requires temperature instrumentation and alarms in the control room. In addition, the staff assumption that walk-downs are performed on a regular (once per shift) basis, with the operator documenting the observations in a log, is also an important element to keep the risk low, since the walk-downs compensate for potential failures of the control room instrumentation.

Even with the above referenced industry commitments, the additional need of walk-downs being performed at least once per shift had to be assumed in order to arrive at the low accident frequency calculated for this scenario. This additional assumption is identified by the staff as a staff decommissioning assumption (SDA #1). In addition, this SDA includes the assumed presence of explicit procedures and operator training which provide guidance on the capability and availability of inventory make-up sources and the time available to initiate these sources.

SDA #1 Walk-downs of SFP systems will be performed at least once per shift by the operators. Procedures will be developed for and employed by the operators to provide guidance on the capability and availability of on-site and off-site inventory make-up sources and time available to initiate these sources for various loss of cooling or inventory events.

3.3.2 Loss of Coolant Inventory

This initiator 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, Volume 12 show that the frequency of loss of inventory events in which a level decrease of more than one foot occurred can be estimated to be less than one event per 100 reactor years. Most of these events are as a result of fuel handler error and are recoverable. Many of the events are not applicable in a decommissioning facility. 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 ten feet.

Using the information from NUREG-1275, it can be estimated that 6% of the loss of inventory events will be large enough and/or occur for a duration that is long enough so that isolation of the loss is required if the only system available for make-up is the spent fuel pool make-up system. For the other 94% of the cases, operation of the make-up pump is sufficient to prevent fuel uncovery.

The calculated fuel uncovery frequency for loss of inventory events is 3.1×10^{-9} per year. Fuel uncovery occurs if plant operators fail to initiate inventory make-up either by use of on-site sources such as the fire pumps or off-site sources such as the local fire department. In the case of a large leak, isolation of the leak would also be necessary if the make-up pumps are utilized. The time available for operator action is considerable, and even in the case of a large leak, it is estimated that 40 hours will be available. Operators will be alerted to a loss of inventory condition by control room alarms and indicators, visibly decreasing water level in the pool, accumulation of water in unexpected locations and local alarms (radiation alarms, building sump high level alarms, etc.).

As in the case for the loss of pool cooling, the frequency of fuel uncovery is calculated to be very low. Again a careful and thorough adherence to IDCs 2, 5, 8 and 10 is crucial to establishing the low frequency. In addition, the assumption that walk-downs (see SDA 1 above) are performed on a regular (once per shift) basis is important to compensate for potential failures of 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 make-up system, and when it becomes essential to supplement with alternative higher volume sources. The assumption that the procedures and training are sufficiently clear in giving guidance on early preparation for using the alternative make-up sources, are crucial to establishing the low frequency. In addition, IDCs 6, 7 and 9 have been credited with lowering the initiating event frequency.

3.3.3 Loss of Off-site Power from Plant-Centered and Grid Related Events

A loss of off-site power from plant-centered events typically involves 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 off-site power grid cause the loss of off-site power. With off-site power lost (and therefore on-site power is lost too, since we assume there is no diesel generator available to pick up the necessary electrical loads), there is no effective heat removal process for the spent fuel pool. If power were not restored quickly enough, the pool would heat up and boil off inventory until the fuel is uncovered. The diesel-driven fire pump would be available to provide inventory make-up. If the diesel-driven pump fails, and if off-site power were not recovered in a timely manner, recovery using off-site fire engines is a possibility. With 1-year-old fuel (i.e., the newest fuel in the fuel pool was shutdown in the reactor one year ago), approximately 130 hours is available for this recovery action.

Even given recovery of off-site power, the plant operators have to restart the fuel pool cooling pumps. Failure to do this or failure of the equipment to restart will necessitate other operator recovery actions. Again, considerable time is available.

The calculated fuel uncovery frequency for this sequence of events is $3x10^{-8}$ per year. This frequency is very low, and similar to the cases for the loss of pool cooling and loss of inventory, is based on adherence to IDCs 2, 5, 8, and 10. In addition, the performance of regular plant walk-downs, and the availability of clear and explicit procedures and operator training is assumed as documented in SDA #1 above.

3.3.4 Loss of Off-site Power from Severe Weather Events

This event represents the loss of SFP cooling due to a loss of off-site power from severe weather-related events. This includes contributions from hurricanes, snow and wind, ice, wind and salt, wind, and one tornado event. Because of their potential for severe localized damage, tornadoes and their direct impact to the site were analyzed separately in Appendix 2e and summarized in Chapter 3.4.3 of this report.

Until off-site power is recovered, the electrical pumps would be unavailable and the diesel-driven fire pump would be available to only provide make-up. When compared to the loss of off-site power events from grid-related and plant-centered causes, recovery of off-site power in this case is assumed to be less probable. In addition, given the conditions, it would be more difficult for off-site help to assist the fuel handlers at the site than for an ordinary loss of off-site power event.

The calculated fuel uncovery frequency for this event is 1.3×10^{-7} per year. As in the previous cases, this estimate was based on IDCs 2, 5, 8, 10 and on assumptions documented in SDA #1. In addition, IDC 3, related to having procedures in place for communication between on-site and off-site organizations during severe weather, is also important in the analysis for increasing the likelihood of off-site resources being able to respond effectively.

3.3.5 Internal Fire

This event tree models the loss of SFP cooling caused by internal fires. The staff assumed that there is no automatic fire suppression system for the SFP cooling area. The fuel handler may initially attempt to manually suppress the fire given that they respond to the control room or local area alarms. If the fuel handler fails to respond to the alarm, or is unsuccessful in extinguishing the fire within the first 20 minutes, the staff assumed that the SFP cooling system will be significantly damaged and cannot be repaired within a few days. Once the inventory level drops below the SFP cooling system suction level, the fuel handlers have about 85 hours to provide some sort of alternative make-up, either using the site firewater system or by calling upon off-site resources. It was assumed that fire damages the plant power supply system such that the power to the electrical firewater pump is lost and would not be available.

The calculated fuel uncovery frequency for this event is 4.5×10^{-8} per year. As in the previous cases, this estimate was based on IDCs 2, 5, 8 and 10 and on the staff assumptions in SDA #1. In addition, IDC 3, related to having procedures in place for communication between on-site and off-site organizations during severe weather, is also important in the analysis for increasing the likelihood of off-site resources being able to respond effectively to this fire event by increasing the likelihood for recovery using off-site resources.

3.3.6 Heavy Load Drops

The staff investigated the frequency of dropping a heavy load in or near the spent fuel pool, and investigated potential damage to the pool from such a drop. The previous assessment done for resolution of Generic Issue 82 (in NUREG/CR-4982 (Ref 4)) only considered the possibility of a heavy load drop falling on the pool wall. The assessment conducted for this study identified other failure modes, such as the pool floor, as also being credible for some sites. Details of the heavy load evaluation can be found in Appendix 2c. The analysis exclusively considered drops that were severe enough to catastrophically damage the spent fuel pool such that pool inventory would be lost rapidly and it would be impossible to refill the pool using on-site or off-site resources. In essence there is no possibility for mitigation in such circumstances, only prevention. A catastrophic heavy load drop (that caused a large leakage path in the pool) would lead directly to a zirconium fire approximately 10 hours after the drop, depending on fuel age, burn up, and configuration. The dose rates in the pool area prior to any zirconium fire would be on the order of tens of thousands of rem per hour, making any potential recovery actions such as temporary large inventory addition systems very difficult. The staff concluded that non-catastrophic damage to the pool or its support systems from a load drop is captured and bound by other initiators.

Based on discussions with staff structural engineers, it was assumed that only spent fuel casks had sufficient weight to catastrophically damage the pool if dropped. The staff assumed there is a very low likelihood that other heavy loads would be moved over the spent fuel pool, and in addition, if there were a drop of one of these lighter loads over the spent fuel pool, there would be a very low likelihood that it would cause catastrophic damage to the pool.

For a non-single failure proof load handling system, the likelihood of a heavy load drop (i.e., the drop frequency) was estimated, based on NUREG-0612 information, to have a mean value of $3.4x10^{-4}$ per year. The number of heavy load lifts was based on the NEI estimate of 100 spent fuel shipping cask lifts per year, which probably is an overestimate. For single failure proof load handling system or a plant conforming to the NUREG-0612 guidelines, the plant is estimated to have a drop frequency mean value of $9.6x10^{-6}$ per year, again for 100 heavy load lifts per year but using data from U.S. Navy crane experience. Once the load is dropped, the analysis must then consider whether the drop would do significant damage to the spent fuel pool.

When estimating the failure frequency of the pool floor and pool wall, the staff assumed that heavy loads physically travel near or over the pool approximately 13% of the total path lift length (the path lift length is the distance from the lift of the load to the placement of the load on the pool floor). The staff also assumed that the critical path length (the fraction of total path the load is lifted high enough above the pool that a drop could cause damage to the structure) is approximately 16% of the time the load is near or over the pool. The staff estimated the catastrophic failure rate from heavy load drops to have a mean value of $2.1x10^{-5}$ per year for a non-single failure proof system where reliance is placed on electrical interlocks, fuel handling system reliability, and safe load path procedures. The staff estimated the catastrophic failure rate from heavy load drops to have a mean value of $2x10^{-7}$ per year for a single failure proof system. The staff assumed that licensees which chose the non-single failure proof system option in NUREG-0612 performed appropriate analyses and mitigated actions to reduce the expected frequency of catastrophic damage to the same range as that of facilities with a single failure proof system.

NEI has made a commitment (IDC #1) for the nuclear industry that future decommissioning plants will comply with Phases I and II to the NUREG-0612 guidelines. Consistent with this industry commitment, the additional assurance of a well performed and implemented load drop analysis, including mitigative actions, was assumed in order to arrive at a low accident frequency for non-single failure proof systems to be comparable to single failure proof systems.

SDA #2 Load Drop consequence analyses will be performed for facilities with non-single failure proof systems. The analyses and any mitigative actions necessary to preclude catastrophic damage to the spent fuel pool that would lead to a rapid pool draining would be sufficient to demonstrate that there is high confidence in the facilities ability to withstand a heavy load drop.

While the focus of this report is the risk associated with wet storage of spent fuel during decommissioning, the staff was alert to any implications on the storage of spent fuel during power operation. With regard to power operation, the resolution of Generic Issue (GI) 82, “Beyond Design Basis Accidents in Spent Fuel Pools,” and other studies of operating reactor spent fuel pools concluded that existing requirements for operating reactor spent fuel pools are sufficient. During this study, the staff evaluated one additional issue concerning the drop of a cask on the spent fuel pool floor. As noted above, due to the industry’s commitment to Phase II of NUREG-0612, “Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36," this is not a concern for decommissioning reactors.

Operating reactors are not required to implement Phase II of NUREG-0612. The risk for spent fuel pools at operating plants is limited by the lower expected frequency of heavy load lifts as compared to decommissioning plants. Nonetheless, this issue will be further examined as part of the Office of Nuclear Regulatory Research’s prioritization of Generic Safety Issue 186, “Potential Risk and Consequences of Heavy Load Drops in Nuclear Power Plants,” which was accepted in May 1999.

3.4 Beyond Design Basis Spent Fuel Pool Accident Scenarios (External Events)

The following is a description of how each of the external event initiators was modeled, a discussion of the frequency of fuel uncovery associated with the initiator, and a description of the most important insights regarding risk reduction strategies for each initiator.

3.4.1 Seismic Events

When performing the evaluation of the effect of seismic events on spent fuel pools, it became apparent that the staff does not have detailed information on how all the spent fuel pools were designed and constructed. Therefore, the staff originally performed a simplified bounding seismic risk analysis in our June 1999 draft risk assessment 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. After further evaluation and discussions with stakeholders, it was determined that it would not be cost effective to perform a plant-specific seismic evaluation for each spent fuel pool. Working with our stakeholders, the staff developed other tools that help assure the pools are sufficiently robust.

Spent fuel pool structures at nuclear power plants are seismically robust. They are constructed with thick reinforced concrete walls and slabs lined with stainless steel liners 1/8 to 1/4 inch thick[9]. Pool walls vary from 4.5 to 5 feet in thickness and the pool floor slabs are around 4 feet thick. The overall pool dimensions are typically about 50 feet long by 40 feet wide and 55 to 60 feet high. In boiling water reactor (BWR) plants, the pool structures are located in the reactor building at an elevation several stories above the ground. In pressurized water reactor (PWR) plants, the spent fuel pool structures are located outside the containment structure supported on the ground or partially embedded in the ground. The location and supporting arrangement of the pool structures determine their capacity to withstand seismic ground motion beyond their design basis. The dimensions of the pool structure are generally derived from radiation shielding considerations rather than structural needs. Spent fuel structures at operating nuclear power plants are able to withstand loads substantially beyond those for which they were designed. Consequently, they have significant seismic capacity.

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[9]Except at Dresden Unit 1 and Indian Point Unit 1, these two plants 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.

During stakeholder interactions with the staff, the staff proposed the use of a seismic checklist, and in a letter dated August 18, 1999 (See Appendix 5), NEI proposed a checklist that could be used to show robustness for a seismic ground motion with a peak ground acceleration (PGA) of approximately 0.5g. This checklist was reviewed and enhanced by the staff. The staff has concluded that plants that satisfy the revised seismic checklist can demonstrate with reasonable assurance a high-confidence low-probability of failure (HCLPF)[10] at a ground motion that has a very small likelihood of exceedence.

U.S. nuclear power plants, including their spent fuel pools, were designed such that they can be safely shutdown and maintained in a safe shutdown condition if subjected to ground motion from an earthquake of a specified amplitude. This design basis ground motion is referred to as the safe shutdown earthquake (SSE). The SSE was determined on a plant specific basis consistent with the seismicity of the plant's location. In general, plants located in the eastern and central parts of the US, had lower amplitude SSE ground motions established for their designs than the plants located in the western parts of the US, which had significantly higher SSEs established for them because of the higher seismicity for locations west of the Rocky Mountains. As part of this study, the staff with assistance from Dr. Kennedy (See Appendix 5), reviewed the potential for spent fuel pool failures to occur in various regions in the U.S. due to seismic events with ground motion amplitudes exceeding established SSE values.

Thus, the seismic component of risk can be limited to an acceptable level if it can be demonstrated that there is a HCLPF for seismic ground motion greater than or equal to three times SSE at CEUS sites and two times SSE at West Coast sites. As discussed in Appendix 5b, for CEUS plants that can demonstrate HCLPF at three times their SSE value and West Coast plants that can demonstrate HCLPF at two times their SSE value, the frequency of fuel uncovery is judged to be less than $3X10^{-6}$ per year.

The seismic checklist (Appendix 5d) was developed to provide a simplified method for demonstrating a high confidence of a low probability of failure and thus 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. Spent fuel pools that satisfy the seismic checklist, as written, would have a high confidence in a low probability of failure for seismic ground motions up to 0.5 g peak ground acceleration (1.2g peak spectral acceleration). Thus, sites in the central and eastern part of the U.S. that have three times SSE values less than or equal to 0.5 g PGA and pass the seismic check list would have an acceptably low level of seismic risk. Similarly, West Coast sites that have two times SSE values less than 0.5 g. and pass the seismic check list would have acceptably low values of seismic risk. From a practical point of view, a limited number of sites in the central and eastern part of the U.S. have three times SSE values greater than 0.5g; the two times SSE values exceed 0.5g for two West Coast plants. In order to demonstrate acceptably low seismic risk, those central and eastern sites for which the three times SSE values exceed 0.5g and the two West Coast sites would have to perform additional plant specific analyses to demonstrate HCLPF for their spent fuel pools at three times SSE and two times SSE values of ground acceleration, respectively. The staff notes

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

that the seismic checklist could be modified to address seismic ground motions corresponding to the range of three times and two times SSE values, making it more generally applicable. This possibility can be pursued in further discussions with external stakeholders.

3.4.2 Aircraft Crashes

The staff evaluated the likelihood of an aircraft crashing into a nuclear power plant site and seriously damaging the spent fuel pool or its support systems (details are in Appendix 2d). The generic data provided in Department of Energy (DOE) -STD-3014-96 [Ref. 6], were used to assess the likelihood of an aircraft crash into or near the spent fuel pool of a decommissioning nuclear power plant. Aircraft damage can affect the structural integrity of the spent fuel pool or affect the availability of nearby support systems, such as power supplies, heat exchangers, or water make-up sources, and may also affect recovery actions.

The estimated range of catastrophic damage to the PWR spent fuel pool, resulting in uncovery of the spent fuel, is 9.6×10^{-12} to 4.3×10^{-8} per year with a mean value estimated at 2.9×10^{-9} per year. The frequency of catastrophic BWR spent fuel pool damage resulting from a direct hit by a large aircraft is estimated to be the same as that for the PWR. Mark-I and Mark-II secondary containments generally do not appear to have any significant structures that might reduce the likelihood of aircraft penetration. Mark-III secondary containments may reduce the likelihood of penetration somewhat, as the spent fuel pool may be considered to be protected on one side by additional structures.

The mean value for an aircraft damaging a support system is in the 7×10^{-7} per year, or less, range. This is not the estimated frequency of fuel uncovery or a zirconium fire caused by damage to the support systems, since the frequency estimate does not include recovery, either from on-site or off-site sources. As an initiator to failure of a support system leading to fuel uncovery and a zirconium fire, an aircraft crash is bound by other more probable events. Recovery of the support systems will reduce the likelihood of spent fuel uncovery.

3.4.3 Tornadoes

A risk evaluation of tornado threats to spent fuel pools was performed (details are in Appendix 2e). The staff assumed that very severe tornadoes (F4 to F5 tornadoes on the Fujita scale) would be required to cause catastrophic damage to a PWR or BWR spent fuel pool. The staff then looked at the frequency of such tornadoes occurring and the conditional probability that if such a tornado hit the site, it would seriously damage the spent fuel pool or its support systems. To do this, the staff examined the frequency and intensity of tornadoes in each of the states within the continental U.S. using the methods described in NUREG/CR-2944 [Ref. 7]. The frequency of having an F4 to F5 tornado that directly impacts the site is estimated to be 5.6×10^{-7} per year for the central U.S., with a U.S. average value of 2.2×10^{-7} per year.

The staff then considered what level of damage an F4 or F5 tornado could do to a spent fuel pool or its support systems. Based on the buildings housing the spent fuel pools and the thickness of the spent fuel pools themselves, the estimated probability of catastrophic failure given a tornado missile is very low. Hence, the overall frequency of catastrophic pool failure caused by a tornado is extremely low (i.e., the calculated frequency of such an event is less than 1×10^{-9} per year)

The staff assumed that an F2 to F5 tornado would be required if significant damage were to occur to spent fuel pool support systems (e.g., power supply, cooling pumps, heat exchanger, or make-up water supply). These tornadoes have wind speeds that result in damage characterized as significant, severe or worse. The frequency of having an F2 to F5 tornado is estimated to be 1.5×10^{-5} per year for the central U.S., with a U.S. average value of 6.1×10^{-6} per year. This is not the estimated frequency of fuel uncovery or a zirconium fire caused by damage to the support systems, since the frequency estimate does not include credit for maintaining pool inventory from either on-site or off-site sources. The probability of failing to maintain inventory was estimated for the case of loss of off-site power from severe weather, where it was assumed that the principal impact of the severe weather was to hamper recovery of off-site power and also to increase the probability of failing to bring off-site sources to bear because of damage to the infrastructure. The situation with tornadoes is different, because the damage caused by a tornado is relatively localized. Therefore, while, a direct hit on the plant could also disable the diesel fire pump, it would be unlikely to also disable off-site resources to the same degree. Therefore, the probability of failing to bring in the off-site sources can be argued to be the same as for the seismic case, i.e., $1\text{X}10^{-4}$, (See Appendix 2b) under the assumption that NEI IDCs 3 and 4 are implemented. Therefore, the fuel uncovery frequency from the loss of SFP support systems from tornado events is estimated to be less than 10^{-9} per year.

Missiles generated by high winds (for example, straightwinds or hurricanes) are not as powerful as those generated by tornadoes. Therefore, high winds are estimated to have a negligible impact on catastrophic failure of the SFP resulting in fuel uncovery. With respect to impact on SFP inventory and cooling, the risk for pool uncovery from high winds is assumed to be bounded by scenarios such as those where there is a loss of off-site power from severe weather, and where off-site power is not recovered.

3.4.4 Criticality in Spent Fuel Pool

In Appendix 3 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 due to criticality in the SFP, and its conclusions that with the additional assumptions, the potential risk from SFP criticality is small.

The assessment referenced in Appendix 3 identified two scenarios as credible, which are listed below.

(1) A compression or buckling of the stored assemblies due to being impacted by a dropped heavy load (such as fuel cask) could result in a more optimum geometry (closer spacing) and thus create the potential for criticality (see the NRC staff report “Assessment of the Potential for Criticality in Decommissioned Spent Fuel Pools,” in Appendix 3). Compression is not a problem for high-density PWR or BWR racks because they have sufficient fixed neutron absorber plates to mitigate any reactivity increase, nor is it a problem for low-density PWR racks if soluble boron is credited. But compression of a low-density BWR rack could lead to a criticality since BWR racks contain no soluble or solid neutron absorbing material. This is not a surprising result since low-density BWR fuel racks use geometry and fuel spacing as the primary means of maintaining subcriticality. 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 neutron absorbing material. If BWR pools were borated, criticality would not be achievable for a low-density rack 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 in the fuel pool water would be sufficient to maintain sub-criticality. The absorber plates are generally enclosed by cover plates (stainless steel or aluminum alloy). The tolerances within a cover plate tend to prevent any appreciable fragmentation and movement of the enclosed absorber material. The total loss of the welded cover plate is not considered feasible.

Boraflex has been found to degrade in spent fuel pools due to gamma radiation and exposure to the wet pool environment. For this reason, the NRC issued Generic Letter 96-04 to all holders of operating licenses, on Boraflex degradation in spent fuel storage racks. Each addressee that uses Boraflex was requested to assess the capability of the Boraflex to maintain a 5% sub-criticality margin and to submit to the NRC proposed actions to monitor the margin or confirm that this 5% margin can be maintained for the lifetime of the storage racks. Many licensees subsequently replaced the Boraflex racks in their pools or re-analyzed 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 (adding neutron moderation) during personnel actions in response to accidents were discounted due to the basic physics and neutronic properties of the racks and fuel, which would preclude criticality conditions being reached with any credible likelihood. For example, without moderation fuel at current enrichment limits (no greater than 5 wt% U-235) cannot achieve criticality, no matter what the configuration. If it is assumed that the pool water is lost, a re-flooding of the storage racks with unborated water may occur due to personnel actions. However, both PWR and BWR storage racks are designed to remain subcritical if moderated by unborated water in the normal configuration. Thus, the only potential credible scenarios are those described above in 1 and 2 which involve crushing of fuel assemblies in low density racks or degradation of Boraflex over long periods in time. These conclusions were developed assuming present light water uranium oxide reactor fuel designs. Alternative fuel designs, such as mixed oxides (MOX) fuels would have to be reassessed to ensure that additional vulnerabilities for pool criticality did not exist.

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, a heavy load drop into a low density racked BWR pool, compressing the assemblies would be required. From the work done on heavy load drop, the likelihood of a heavy load drop from a single failure proof crane has been determined to have a mean frequency of approximately $9.6X10^{-6}$ per year, assuming 100 cask movements per year at the decommissioning facility. From the load path analysis done in that appendix it was estimated that the load could be over or near the pool approximately 13% of the movement path length, dependent on plant specific layout. The additional frequency reduction in the appendix to account for the fraction of time that the heavy load is lifted high enough to damage the pool liner is not applicable here because the fuel assemblies could be crushed without the same impact velocity being required as for the pool liner. Therefore, the staff estimated a potential initiating frequency for crushing of approximately $1.2X10^{-6}$ per year (based upon 100 lifts per year). Criticality calculations conducted in Appendix 3 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 $1X10^{-8}$ per year, which justifies its exclusion from further consideration.

Deformation of the low density BWR racks by the dropped transfer cask was shown to most likely not result in any criticality events. However, if some mode of criticality was to be induced by the dropped transfer cask it would more likely be a small return to power for a very localized region, rather than the severe response discussed in the paragraph above. This 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 above, (the gradual degradation of the Boraflex absorber material in high density storage racks), there is currently insufficient data to quantify the likelihood of criticality occurring due to its loss. However, the current programs in place at operating plants to assess the condition of the Boraflex and take remedial action if necessary provide sufficient confidence that pool reactivity requirements will be satisfied. In order to meet the RG 1.174 safety principle of maintaining sufficient safety margins, the staff judges that continuation of such programs into the decommissioning phase would be required at all plants until all high density racks are removed from the SFP. Therefore, a staff assumption is identified in Section 4.2.4 requiring continuation of this activity, which should be reflected in future regulatory activity associated with SFP requirements.

Based upon the above conclusions and staff assumption, we believe that qualitative risk insights demonstrate conclusively that SFP criticality poses no meaningful risk to the public.

4.0 Implications of Spent Fuel Pool Risk For Regulatory Requirements

An important motivation for performing the risk analysis contained in this report is to provide insight into the regulatory requirements that would be needed to limit the risk at decommissioning plants. In order to do that, Chapter 4.1 presents a brief summary of the risk results that are most pertinent to that end.

The analysis in Chapter 3 explicitly examines the risk impact of specific design and operational characteristics, taking credit for industry commitments proposed by NEI in a letter to the NRC dated November 12, 1999 [See Ref. 1 or Appendix 6]. Additional assumptions (staff decommissioning assumptions-SDAs) came to light as a result of the staff's risk assessment. These additional assumptions in SFP design and operational characteristics were found to be necessary to achieve the low risk findings in this report. One SDA is identified in Chapter 3, while the remainder are developed from the safety principles of RG 1.174 and are summarized in Chapter 4.1. Chapter 4.2 examines the design and operational elements that are important in ensuring that the risk from a SFP is sufficiently low and how these elements support the safety principles of RG 1.174 as they apply to a SFP.

In addition, the industry and other stakeholders have proposed the use of risk-informed decision-making to assess regulatory requirements in three specific areas; emergency preparedness, safeguards and insurance indemnification. The technical results of this report can be used either to justify plant-specific exemptions from these requirements, or to determine how these areas will be treated in risk-informed regulations for decommissioning sites. Since both the IDCs and SDAs are essential in achieving the levels of safety presented in this analysis, future regulatory activity would properly reflect such commitments and assumptions. Chapter 4.3 examines the implications of the technical results for those specific regulatory decisions.

4.1. Summary of the Technical Results

The thermal-hydraulic analysis presented in Appendix 1 demonstrates that the decay heat necessary for a zirconium fire exists in typical spent fuel pools of decommissioning plants for a period of several years following shutdown. The analysis shows that the length of time over which the fuel is vulnerable depends on several factors, including fuel burn up and fuel configuration. In some cases analyzed in Appendix 1, the required decay time to preclude a zirconium fire is 5 years. However, the exact time will be plant specific, and therefore plant-specific analysis is needed to justify the use of shorter decay periods. Guidelines for plant specific analyses can be found in Appendix 1.

The consequence analysis presented in Appendix 4 demonstrates that the consequences of a zirconium fire in a decommissioning plant can be very large. The integrated dose to the public is generally comparable to a large early release from an operating plant during a potential severe core damage accident. Early fatalities are very sensitive to the effectiveness of evacuation.

For a decommissioning plant with about one year of decay time, the onset of radiological releases from a zirconium fire is significantly delayed compared to those from the most limiting operating reactor accident scenarios. This is due to the relatively long heat up time of the fuel. In addition, for many of the sequences leading to zirconium fires, there are very large delay times due to the long time required to boil off the large spent fuel pool water inventory. Thus, while the consequences of zirconium fires are in some ways comparable to large early releases from postulated reactor accidents, the time of release is much longer from initiation of the accident.

The generic frequency of events leading to zirconium fires at decommissioning plants is estimated to be less than $3x10^{-6}$ per year for a plant that implements the design and operational characteristics discussed below. This estimate can be much higher for a plant that does not implement these characteristics. The most significant contributor to this risk is a seismic event which exceeds the design basis earthquake. The overall frequency of this event is within the recommended pool performance guideline (PPG) for large radionuclide releases due to zirconium fire of $1x10^{-5}$ per year. As noted above, zirconium fires are estimated to be similar to large early release accidents postulated for operating reactors in some ways, but less severe in others.

4.2 Risk Impact of Specific Design and Operational Characteristics

This section discusses the design and operational elements that are important in ensuring that the risk from a SFP is sufficiently low. The relationship of the elements to the quantitative risk findings is discussed as well as how the elements support additional safety principles of RG 1.174 as they apply to a SFP.

4.2.1. Impact of Proposed Changes

> "When proposed changes result in an increase in core damage frequency and/or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement."

The staff's risk assessment as discussed in Chapter 3 shows that the baseline risk (represented as the frequency of zirconium fire in a decommissioning spent fuel pool) is estimated to be less than 3×10^{-6} per year. As was discussed in Chapter 2, the staff has determined that such a fire results in a large radionuclide release and poses a highly undesirable end state for a spent fuel pool accident. Therefore the staff has judged that a pool performance guideline (PPG) of 1×10^{-5} per year derived from the RG 1.174 application of LERF, should be applied. The risk assessment shows that the SFP zirconium fire frequency is well under the recommended PPG. The assessments conducted for this study also show that the accident progresses much more slowly than at an operating reactor. For many scenarios, recovery and mitigation times of approximately 100 hours are available from onset of the loss of cooling initiators. Even for extremely unlikely events such as severe seismic events and heavy load drops failing the pool floor, ten hours or more time is available to initiate off-site protective actions if necessary prior to zirconium fire initiation. Therefore, the risk assessment shows that both low likelihoods and long response times are associated with SFP accidents at decommissioning plants. These conclusions are predicated on the industry commitments and staff assumptions discussed in this report being fulfilled.

The staff consequence analysis in Appendix 4 shows that the early health impacts from zirconium fire scenarios are significantly impacted by evacuation. As for operating plants, evacuation of the public is the preferred protective action to minimize exposure and early health impacts to the population surrounding the site in the event of a severe accident. Emergency planning requirements for operating plants specify that licensee's have the means for assessing the impact of an accident and have the capability of notifying off-site officials within 15 minutes of declaring an emergency. In addition, the licensee must demonstrate that there are means in place for promptly alerting and providing instructions to the public in case protective actions are needed. Furthermore, detailed off-site emergency plans are required to provide for prompt implementation of protective actions (including evacuation of the public). However, this analysis indicates that for the slowly evolving SFP accident sequences at decommissioning plants, there is a large amount of time to initiate and implement protective actions, including public evacuation, in comparison to operating reactor accident sequences.

In addition to SDA #1 and SDA #2, the low numerical risk results shown in Chapter 3 and Appendix 2 are derived from a number of design and operational elements of the SFP. As shown in those sections, the dominant risk contribution is from seismic events beyond the plant's original design basis. The baseline seismically initiated zirconium fire frequency from our risk assessment is predicated upon implementation of the seismic checklist shown in Appendix 5. The staff therefore assumed that such a checklist (SDA #3) would be successfully implemented at all decommissioning facilities.

SDA #3 Each decommissioning plant will successfully complete the seismic checklist provided in Appendix 5 to this report. If the checklist cannot be successfully completed, the decommissioning plant will perform a plant specific seismic risk assessment of the SFP and demonstrate that SFP seismically induced structural failure and rapid loss of inventory is less than the generic bounding estimates provided in this study ($<3x10^{-6}$ per year).

The quantification of accident sequences in Chapter 3 associated with loss of cooling or loss of inventory resulted in low risk due to a number of elements that enhance the ability of the operators to respond successfully to the events with on-site and off-site resources. Without these elements, the probability of the operators detecting and responding to the loss of cooling or inventory would be higher and public risk from these categories of SFP accidents could be significantly increased. Some elements were also identified that reduce the likelihood of the loss of cooling or loss of inventory initiators, including both design and operational issues. The elements proposed by industry (IDCs) are identified below.

To reduce the likelihood of loss of inventory the following was committed to by industry:

IDC #6 Spent fuel pool 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.

IDC #7 Procedures or administrative control 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) control for pump; suction and discharge points. The functionality of anti-siphon devices will be periodically verified.

IDC #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.

The high probability of the operators recovering from a loss of cooling or inventory is dependent upon the following:

IDC #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.

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

IDC #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.

IDC #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.

IDC #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.

The staff's risk evaluation also shows that the potential for pool failure due to heavy load drop to be significant if appropriate design and procedural controls are not in place.

IDC #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).

4.2.2. Defense-in-Depth

"The Proposed Change Is Consistent with the Defense-in-Depth Philosophy."

The staff's risk assessment demonstrates that the risk from a decommissioning plant SFP accident is very small if industry commitments and additional staff assumptions are implemented as assumed in the risk study. Due to the very different nature of a SFP accident versus an accident in an operating reactor, with respect to system design capability needs and event timing, the defense-in-depth function of reactor containment is not necessary. However the staff has identified that defense-in-depth in the form of accident prevention and some form of emergency planning can be useful for as long as a zirconium fire is possible, as a means of achieving consequence mitigation. The degree to which it may be required as an additional barrier is a function of the uncertainty associated with the prediction of the frequency of the more catastrophic events, such as beyond design basis earthquakes. There can be a trade off between the formality with which the elements of emergency planning (procedures, training, performance of exercises) are treated and the increasing safety margin as the fuel ages and the time for response gets longer.

4.2.3 Safety Margins

"The Proposed Change Maintains Sufficient Safety Margins."

As discussed in Chapter 2, the safety margins associated with fuel in the spent fuel pool are much greater than those associated with an operating reactor due to the low heat removal requirements and long time frames available for recovery from off normal events. Due to these larger margins the staff judges that the skid mounted and other dedicated SFP cooling and inventory systems in place do provide adequate margins. Additionally, the surveillance programs that verify Boraflex condition provide assurance of margin with respect to shutdown reactivity.

4.2.4. Implementation and Monitoring Program

"The Impact of the Proposed Change Should Be Monitored Using Performance Measurement Strategies."

RG 1.174 states that an implementation and monitoring plan should be developed to ensure that the engineering evaluation conducted to examine the impact of the proposed changes continues to reflect the actual reliability and availability of SSCs that have been evaluated. This will ensure that the conclusions that have been drawn will remain valid. Applying this guideline for the SFP risk evaluation results in identification of three primary areas for performance monitoring: 1) The performance and reliability of SFP cooling and associated power and inventory make-up systems, 2) The Boraflex condition for high density fuel racks, and 3) Crane operation and load path control for cask movements.

Performance and reliability monitoring of the SFP systems, heat removal, AC power and inventory should be carried out similar to the provisions of the maintenance rule (10 CFR 50.65).

With respect to monitoring of the Boraflex absorber material, the current monitoring programs identified in licensee's responses to Generic Letter 96-04 [Ref. 2] were assumed to be maintained by decommissioning plants until all fuel is removed from the SFP. The staff assumption is stated in SDA #4.

SDA #4 Licensees will maintain a program to provide surveillance and monitoring of Boraflex in high density spent fuel racks until such time as spent fuel is no longer stored in these high-density racks.

With respect to monitoring and control of heavy load activities and load path control, licensee guidance in this area will be provided by IDC # 1.

4.3. Implications for Regulatory Requirements Related to Emergency Preparedness, Security and Insurance

The industry and other stakeholders have expressed interest in knowing the relevance of the results of this study to decisions regarding specific regulatory requirements. These decisions could be made in response to plant-specific exemption requests, or as part of the integrated rulemaking for decommissioning plants. Such decisions can be facilitated by a risk-informed examination of both the deterministic and probabilistic aspects of decommissioning. Three examples of such regulatory decisions are presented in this section.

4.3.1 Emergency Preparedness

The requirements for emergency preparedness are contained in 10CFR 50.47 [Ref. 3] and Appendix E to 10 CFR Part 50 [Ref. 4]. Further guidance on the basis for EP requirements is contained in NUREG-0396 [Ref. 5], and NUREG-0654/FEMA-REP-1 [Ref 6]. The overall objective of EP is to provide dose savings (and in some cases immediate life saving) from accidents.

In the past, the NRC staff has typically granted exemptions from off-site emergency planning requirements for decommissioning plants that could demonstrate that they were beyond the period in which a zirconium fire could occur. The rationale for those decisions was that, in the absence of a zirconium fire, a decommissioning plant had no appreciable scenarios for which the consequences justify the imposition of an off-site EP requirement. The results of this technical study confirm that position for both the scenarios resulting in a potential zirconium fire as well as credible pool criticality events.

In some cases, emergency preparedness exemptions have also been granted to plants which were still in the window of vulnerability for zirconium fire. In these cases, the justification was that enough time had elapsed since shutdown that the evolution of a zirconium fire accident would be slow enough that the staff had confidence that mitigative measures and, if necessary, off-site protective actions could be implemented without preplanning. The staff believes that the technical analysis discussed in Chapter 3 and the decision criteria laid out in Chapter 2 have direct bearing on how such exemption requests should be viewed in the future. In addition, this information has bearing on the need for, and the extent of, emergency preparedness requirements in the integrated rulemaking.

The consequence analysis presented in Appendix 4 demonstrates that the off-site consequences of a zirconium fire are comparable to those from operating reactor postulated severe accidents. Further, the analysis demonstrates that timely evacuation can significantly reduce the number of early fatalities due to a zirconium fire. The thermal-hydraulic analysis presented in Appendix 1 confirms our earlier conclusion that zirconium fire events evolve slowly, even for initiating events that result in a catastrophic loss of fuel pool coolant. The results in Chapter 3 also show that the frequency of zirconium fires is low when compared with the risk guidelines derived from RG 1.174. Thus the risk associated with early fatalities from these scenarios is low which provides some basis to support reductions in EP requirements for decommissioning plants. With respect to the potential for pool criticality, the staff's assessment discussed in Chapter 3 and Appendix 3 demonstrates that credible scenarios for criticality are highly unlikely and are further precluded by the assumptions of Boraflex monitoring programs. Additionally, even if some criticality event was to occur, it would not be expected to have off-site consequences. Therefore, the conclusions regarding possible reductions in EP program requirements are not affected.

One important safety principle of RG 1.174 is consistency with the defense-in-depth philosophy. Defense-in-depth is included in a plant design to account for uncertainties in the analysis or operational data. The spent fuel pools at operating reactors and decommissioning facilities do not exhibit the defense-in-depth accorded to the reactor. As discussed in Chapter 1, this difference is justified in light of the considerably greater margin of safety of the SFP compared with reactors. For SFPs at operating reactors, defense-in-depth consists mainly of the mitigating effect of emergency preparedness.

The risk assessments contained in this report indicate that the safety principles of RG 1.174 can be applied to assess whether changes to emergency preparedness requirements are appropriate. The risk of a release from a spent fuel accident is very low. Notwithstanding this low risk, the safety principles in RG 1.174 dictate that defense-in-depth be considered and, as discussed previously, emergency preparedness provides defense-in-depth. However, because of the considerable time available to initiate and implement protective actions, there does not appear to be a need for formal emergency plans for rapid initiation and implementation of protective actions. The principle aspects of emergency planning which is needed for SFP events is the means for identification of the event and for notification of State and local emergency response officials. It should be noted that there will continue to be a need for on-site emergency preparedness for response to the more likely accidents which only have on-site consequences. This study indicates that a one year period provides adequate decay time necessary to reduce the pool heat load to a level that would provide sufficient human response time for anticipated transients, and minimize any potential gap release. This is also the decay time that would result in a 10-12 hour delay from fuel uncovery to zirconium fire, even for very improbable severe seismic events or heavy load drop causing total loss of pool inventory.

Any future reduction of the one year decay time would be contingent on plant specific thermal hydraulic response, scenario timing, human reliability results and system mitigation and recovery capabilities. That is, any licensee wishing to gain relief from regulatory requirements prior to the one year post-shutdown, would need to demonstrate that plant specific vulnerability to a zirconium fire satisfies the risk informed decision process, risk insights and recommended criteria described in Chapters 2 and 3.

4.3.2 Security

Currently licensees that have permanently shutdown reactor operations and have offloaded the spent fuel into the SFP are still required to meet all the security requirements for operating reactors in 10 CFR 73.55 [Ref 7]. This level of security would require a site with a permanently shutdown reactor to provide security protection at the same level as that for an operating reactor site. The industry has asked the NRC to consider whether the risk of radiological release from decommissioning plants due to sabotage is low enough to justify modification of safeguards requirements for SFPs at decommissioning plants.

In the past, decommissioning licensees have requested exemptions from specific regulations in 10 CFR 73.55, justifying their requests on the basis of a reduction in the number of target sets susceptible to sabotage attacks, and the consequent reduced hazard to public health and safety. Limited exemptions based on these assertions have been granted. The risk analysis in this report does not take exception to the reduced target set argument; however, the analysis does not support the assertion of a lesser hazard to public health and safety, given the consequences that can occur from a sabotage induced uncovery of fuel in the SFP when a zirconium fire potential exists. Further, the risk analysis in this report did not evaluate the potential consequences of a sabotage event that could directly cause off-site fission product dispersion, for example from a vehicle bomb that was driven into or otherwise significantly damaged the SFP, even if a zirconium fire was no longer possible. However, this report would support a regulatory framework that relieves licensees from selected requirements in 10 CFR 73.55 on the basis of target set reduction when all fuel has been placed in the SFP.

The risk estimates contained in this report are based on accidents initiated by random equipment failures, human errors or external events. PRA practitioners have developed and used dependable methods for estimating the frequency of such random events. By contrast, this analysis, and PRA analyses in general, do not include events due to sabotage. No established method exists for estimating the likelihood of a sabotage event. Nor is there a method for analyzing the effect of security provisions on that likelihood. Security regulations are based on a zero tolerance for sabotage involving special nuclear material - which includes spent fuel. The regulations are designed and structured to remove sabotage from design basis threats at a commercial nuclear power plant, regardless of the probability or consequences.

The technical information contained in this report shows that the consequences of a zirconium fire would be high enough to justify provisions to prevent sabotage. Moreover, the risk analysis could be used effectively to assist in determining priorities for, and details of, the security capability at a plant. However, there is no information in the analysis that bears on the level of security necessary to limit the risk from sabotage events. Those decisions will continue to be made based on a deterministic assessment of the level of threat and the difficulty of protecting a specific facility.

10 CFR 72 [Ref. 8] allows facilities not associated with an operating power reactor to store spent fuel at an independent spent fuel storage installation (ISFSI). 10 CFR 73.51 did not consider the risk posed by vehicle-borne bombs at facilities where potential criticality and fuel heat-up were still issues. The staff also noted that the applicability of 10 CFR 26 [Ref 9] has not been thoroughly evaluated for decommissioning reactors once the fuel has been removed from the reactor vessel and placed in the SFP, and specifically does not apply to ISFSIs licensed under 10 CFR 72. Given

the importance of a vehicle bomb threat to the integrity of SFP, and the significance of HRA to the conclusions reached in the SFP risk analysis, the staff recommends that for coherency in the regulations, both of these subjects be revisited during the overall integration of rules for decommissioning reactors.

4.3.3 Insurance

In accordance with 10 CFR 140 [Ref.10], each 10 CFR 50 licensee is required to maintain public liability coverage in the form of primary and secondary financial protection. This coverage is required to be in place from the time unirradiated fuel is brought onto the facility site until all of the radioactive material has been removed from the site, unless the Commission terminates the Part 50 license or otherwise modifies the financial protection requirements under Part 140. On March 17, 1999, the staff proposed to the Commission that insurance indemnity requirements for permanently shutdown reactors be developed in an integrated, risk-informed effort along with requirements for emergency preparedness and security. In the past, licensees have been granted exemptions from financial protection requirements on the basis of deterministic analyses that indicate that a zirconium fire could no longer occur. The analysis in this report supports continuation of this practice, and would support a revised regulatory framework for decommissioning plants that reduces the level of insurance protection when a generic or plant-specific thermal-hydraulic analysis demonstrates that a zirconium fire can no longer occur.

In the staff requirements memorandum (SRM) for SECY-93-127 [Ref. 11], the Commission suggested that withdrawal for secondary financial protection insurance coverage is allowed after the requisite minimum spent fuel cooling period has elapsed. Further, the Commission directed the staff to determine more precisely the appropriate spent fuel cooling period after plant shutdown, and to determine the need for primary financial protection for independent spent fuel storage installations (ISFSIs). Spent reactor fuel aged for one year can be stored in an ISFSI. The NRC staff has considered whether the risk analysis in this report justifies relief from this requirement for a decommissioning plant during the period when it is vulnerable to zirconium fires. As part of this effort, the staff recognizes the structural similarities between a SFP at a decommissioning plant and a wet (as opposed to dry) ISFSI that could be considered under 10 CFR 72; ISFSIs are generally dry. Indemnification is not required for a separately-licensed ISFSI. The risk analysis in this report indicates high consequences of a zirconium fire, identifies a generic window of vulnerability up to a period of about 5 years after shutdown, and concludes that the predicted frequency of such an accident is within the acceptance guidelines of RG 1.174 after one year, provided that certain constraints are met.

Since the postulated consequences are high, the frequency of a zirconium fire occurring in a decommissioning plant SFP would have to be low to justify a reduction in indemnification protection. The zirconium fire frequencies presented in Chapter 3 for a decommissioning plant SFP are comparable to the large early releases frequencies (LERF) from some operating reactors, and are within the LERF guidelines of RG 1.174. A zirconium fire frequency criterion to justify reduction of the insurance requirement while a vulnerability to zirconium fires exists has not been established. The potential for a zirconium fire occurring at a decommissioning plant SFP has been described in this risk study to meet the LERF guidelines in RG 1.174 after a decay time of one year, provided that certain conditions are met. On a deterministic basis, the possibility exists that the 5-year window of vulnerability could be reduced with more refined thermal-hydraulic analysis or some other constraints on other parameters such as fuel configuration.

5.0 Summary and Conclusions

The results of this report estimated the generic frequency of events leading to zirconium fires at decommissioning plants to be less than $3x10^{-6}$ per year for a plant that implements the design and operational characteristics assumed in the risk assessment performed by the staff. This frequency was estimated based on the assumptions that the characteristics of the 10 IDCs proposed by NEI (See Appendix 6) and the four SDAs identified in Chapters 3 and 4 of the report would be implemented. This estimate could be much higher for a plant that does not implement these characteristics. The most significant contributor to this risk is a seismic event which exceeds the design basis earthquake. However, the overall frequency of this event is within the staff recommended pool performance guideline (PPG) identified in this report for large radiological releases due to a zirconium fire of $1x10^{-5}$ per year. As discussed below, zirconium fires are estimated to be similar to large early release accidents postulated for operating reactors in some ways, but less severe in others.

The thermal-hydraulic analysis presented in Appendix 1 demonstrates that the decay heat necessary for a zirconium fire exists in typical spent fuel pools of decommissioning plants for a period of several years following shutdown. The analysis shows that the length of time over which the fuel is vulnerable depends on several factors, including fuel burn-up and fuel storage configuration in the SFP. In some cases analyzed in Appendix 1, the required decay time to preclude a zirconium fire is 5 years. However, the exact time will be plant specific; therefore, plant-specific analysis would be needed to demonstrate shorter zirconium fire vulnerabilities.

The consequence analysis presented in Appendix 4 demonstrates that the consequences of a zirconium fire in a decommissioning plant can be very large. The integrated dose to the public is generally comparable to a large early release from an operating plant during a potential severe core damage accident and early fatalities are very sensitive to the effectiveness of evacuation. For a decommissioning plant with about one year of decay time, the onset of radiological releases from a zirconium fire is significantly delayed compared to those from the most limiting operating reactor accident scenarios. This is due to the relatively long heat up time of the fuel. For many of the sequences leading to zirconium fires, there are very large delay times due to the long time required to boil off the large spent fuel pool water inventory. Thus, while the consequences of zirconium fires are in some ways comparable to large early releases from postulated reactor accidents, the time of release occurs much later following initiation of the accident. Therefore, this analysis indicates that for the slowly evolving SFP accident scenarios at decommissioning plants, there is a large amount of time to initiate and implement protective actions, including public evacuation, in comparison to an operating reactor accident sequences.

In summary, the risk assessment shows low numerical risk results in combination with satisfaction of the safety principles as described in R.G. 1.174, such as defense-in-depth, maintaining safety margins, and performance monitoring. The staff concludes that under the assumptions of this study there is a low level of public risk from SFP accidents at decommissioning plants. In addition, the study shows that after a period of one year following final shutdown, the low likelihood that a zirconium fire would occur, in combination with the long time frames available for taking off-site protective actions, provides a basis for relaxation of emergency planning requirements.

Chapters 4.3.2 and 4.3.3 addressed the report’s implications on security and insurance provisions at decommissioning plants. For security, the risk insights can be utilized to assess what target sets are important to protect against sabotage. However, any reduction in security provisions would be constrained by an effectiveness assessment of the safeguards provisions against a design basis threat. Therefore, the staff concludes that some level of security is required as long as the fuel in the SFP is exposed to a sabotage threat. For insurance, the reports points out that no definitive criteria exists that would allow relaxation on the basis of low event probability alone while the potential for a zirconium fire exists. Finally, inconsistencies were identified in current regulations regarding vehicle-borne bomb threats and insurance indemnification requirements for ISFSIs and decommissioning nuclear power plants. These inconsistencies should be revisited during the overall integration of rules for decommissioning plants.

6.0 References

References for Executive Summary

1. U.S. Code of Federal Regulations, “Domestic Licensing of Production and Utilization Facilities,” Part 50, Title 10, “Energy.”
1. U.S. Nuclear Regulatory Commission, “An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis,” Regulatory Guide 1.174, July 1998.
3 Dana A. Powers, Chairman of the Advisory Committee on Reactor Safeguards, U.S. Nuclear Regulatory Commission, letter to Dr. William D. Travers, U.S. Nuclear Regulatory Commission, “Spent Fuel Fires Associated With Decommissioning,” November 12, 1999.
4 U.S. Code of Federal Regulations, “Standards for Protection Against Radiation,” Part 20, Title 10, “Energy.”

References for Chapter 1.0

1. U.S. Code of Federal Regulations, “Domestic Licensing of Production and Utilization Facilities,” Part 50, Title 10, “Energy.”

References for Chapter 2.0

1. U.S. Nuclear Regulatory Commission, "Use of Probabilistic Risk Assessment Methods in Nuclear Regulatory Activities; Final Policy Statement," Federal Register, Vol. 60, No. 158, August 16, 1995/ Notices.
2. U.S. Nuclear Regulatory Commission, “An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis,” Regulatory Guide 1.174, July 1998.
3. Dana A. Powers, Chairman of the Advisory Committee on Reactor Safeguards, U.S. Nuclear Regulatory Commission, letter to Dr. William D. Travers, U.S. Nuclear Regulatory Commission, “Spent Fuel Fires Associated With Decommissioning,” November 12, 1999.
4. U.S. Code of Federal Regulations, “Requirements for Monitoring the Effectiveness of Maintenance at Nuclear Power Plants,” Section 65, Part 50, Title 10, “Energy.”

References for Chapter 3.0

1. Dana A. Powers, Chairman of the Advisory Committee on Reactor Safeguards, U.S. Nuclear Regulatory Commission, letter to Dr. William D. Travers, U.S. Nuclear Regulatory Commission, “Spent Fuel Fires Associated With Decommissioning,” November 12, 1999.
2. U.S. Nuclear Regulatory Commission, “Regulatory Analysis for the Resolution of Generic Issue 82, “Beyond design Basis Accidents in Spent Fuel Pools,” NUREG-1353, April 1989.
3. U.S. Nuclear Regulatory Commission, “ Operating Experience Feedback Report-Assessment of Spent Fuel Cooling,” NUREG-1275, Volume 12, February 1997.
4. Sailor, et.al., “Severe Accidents in Spent Fuel Pools in Support of Generic Issue 82", NUREG/CR-4982 (BNL-NUREG-52093), July 1987.
5. U.S. Nuclear Regulatory, “Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36,“ NUREG-0612, July 1980.
6. DOE-STD-3014-96, “Accident Analysis for Aircraft Crash Into Hazardous Facilities,” U.S. Department of Energy, October 1996.

7. NUREG/CR-2944, “Tornado Damage Risk Assessment,” Brookhaven National Laboratory, September 1982.

References for Chapter 4.0

1. Lynnette Hendricks, Nuclear Energy Institute, letter to Office of Nuclear Reactor Regulation, dated November 12, 1999.
2. U.S. Nuclear Regulatory Commission, "Boraflex Degradation in Spent Fuel Pool Storage Racks," Generic Letter 96-04, June 1996.
3. U.S. Code of Federal Regulations, “Emergency Plans,” Section 47, Part 50, Title 10, “Energy.”
4. U.S. Code of Federal Regulations, “Emergency Planning and Preparedness For Production and Utilization,” Appendix E, Part 50, Title 10, “Energy.”
5. U.S. Code of Federal Regulations, “Planning Basis For The Development Of State and Local Government Radiological Emergency Response Plans In Support of Light Water Nuclear Power Plants,” NUREG-0396, December 1978.
6. NUREG-0654/FEMA-REP-1, Revision 1, "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Reactors", November 1980.
7. U.S. Code of Federal Regulations, “Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Radiological Sabotage,” Section 55, Part 73, Title 10, “Energy.”
8. U.S. Code of Federal Regulations, “Licensing Requirements For the Independent Storage of Spent Nuclear Fuel And High-Level Radioactive Waste,” Part 72, Title 10, “Energy.”
9. U.S. Code of Federal Regulations, “Fitness for Duty Programs,” Part 26, Title 10, “Energy.”
10. U.S. Code of Federal Regulations, “Financial Protection Requirements and Indemnity Agreements,” Part 140, Title 10, “Energy.”
11. U.S. Nuclear Regulatory Commission, “Financial Protection Required of Licensees of Large Nuclear Power Plants During Decommissioning,” SECY-93-127, dated July 13, 1993.

7.0 Acronyms

ACRS	Advisory Committee on Reactor Safeguards
ANSI	American National Standard Institute
ANS	American Nuclear Society
ASB	NRC Auxiliary Systems Branch (Plant Systems Branch)
atm	atmosphere
BNL	Brookhaven National Laboratory
BTP	branch technical position
BWR	boiling water reactor
CFD	computational fluid dynamics
CFM	cubic feet per minute
CFR	Code of Federal Regulations
DOE	Department of Energy
DSP	decommissioning status plant
ECCS	emergency core cooling system
EP	emergency plan
EPRI	Electric Power Research Institute
ET	event tree
FFU	frequency of fuel uncovery
FT	fault tree
gpm	gallon(s) per minute
GSI	generic safety issue
GWD	gigawatt-day
HCLPF	High-Confidence/Low probability of failure
HRA	human reliability analysis
HVAC	heating, ventilation, and air conditioning
IDC	industry decommissioning commitments
INEEL	Idaho National Engineering and Environmental Laboratory
ISFSI	independent spent fuel pool installation
kW	kilowatt
LERF	large early release frequency
LLNL	Lawrence Livermore National Laboratory
LOSP	loss of off-site power
LWR	light water reactor
MR	maintenance rule
MW	megawatt
MWD	megawatt-day
MTU	megaton uranium

NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	NRC Office of Nuclear Reactor Regulation
POE	probability of exceedance
POF	probability of failure
PPG	pool performance guideline
PRA	probabilistic risk assessment
PWR	pressurized water reactor
QA	quality assurance
RES	NRC Office of Research
RG	regulatory guide
SDAs	staff decommissioning assumptions
SF	spent fuel
SFP	spent fuel pool
SFPC	spent fuel pool cooling system
SFPCC	spent fuel pool cooling and cleaning system
SHARP	Spent Fuel Heatup Analytical Response Program
SNL	Sandia National Laboratory
SRM	staff requirements memorandum
SRP	standard review plan
SSC	systems, structures, and components
SSE	safe shutdown earthquake
TS	technical specification
UKAEA	United Kingdom Atomic Energy Authority
WIPP	Waste Isolation Pilot Plant

Appendix 1 Thermal Hydraulics

1. Spent Fuel Heatup Analyses

Spent fuel heatup analyses model the decay power and configuration of the fuel to characterize the thermal hydraulic phenomena that will occur in the SFP and the building following a postulated loss of water accident. This appendix reviews the existing studies on spent fuel heatup and zirconium oxidation, the temperature criteria used in the analyses, and how it applies to decommissioned plants.

1.1 Spent Fuel Failure Criteria

Several different fuel failure criteria have been used in previously NRC-sponsored SFP accident studies. Benjamin, et. al used the onset of runaway fuel clad oxidation as the fuel failure criterion in NUREG/CR-0649 [Ref. 1]. This criterion was criticized because clad rupture can occur at a relatively low temperature causing a gap release. The consequences of gap release can be significant if the radioactive iodine has not yet decayed to insignificant amounts. SHARP calculations [Ref. 2] used the onset of clad swelling as an acceptance criterion for prevention of fuel failure. The onset of clad swelling leading to gap release occurs at approximately 565 °C, which corresponds to the temperature for 10-hour creep rupture time [Ref. 3]. A cladding temperature of 570 °C is used as a thermal limit under accident conditions for licensing of spent fuel dry storage casks.

The most severe fuel damage would be caused by rapid, runaway zirconium oxidation. This would lead to significant fission product release even after the gap activity has become insignificant. The onset of rapid oxidation may occur as low as 800 °C [Ref. 4]. Runaway oxidation can raise clad and fuel temperatures to approximately 2000 °C which corresponds to the melting temperature of zirconium. The release of fission products trapped in the fuel can occur at fuel temperatures of approximately 1400-1500 °C. Runaway oxidation starting in a high-powered channel could also propagate through radiative and convective heat transfer to lower power assemblies because of the large heat reaction in zirconium oxidation.

There are several other temperature thresholds that may be of concern in SFP accidents. The melting temperature of aluminum, which is a constituent in BORAL poison plates in some types of the spent fuel storage racks, is approximately 640 °C. No evidence was found that boron carbide would dissolve in the aluminum forming a eutectic mixture that liquefies at a temperature below the melting point of aluminum. However, if it is possible for a molten material to leak from the stainless steel spent fuel storage rack case, melting and relocation of the aluminum in the boron carbide-aluminum composite may cause flow blockages that increase hydraulic resistance. No realistic evaluation of melting and relocation of aluminum or aluminum/boron carbide eutectic has been performed.

Another concern is the structural integrity of the fuel racks at high temperatures. Several eutectic mixtures known from reactor severe accident research [Ref. 5] may be important in SFP accidents. As previously stated, the formation of an eutectic mixture allows liquification and loss of structural integrity for a mixture of materials at a lower temperature than the melting point of any of the component materials. Steel and zirconium form an eutectic mixture at approximately 935 °C. Steel and boron carbide form a eutectic mixture at approximately 1150 °C. The steel racks may not be able to maintain structural integrity because of the sustained loads at high temperatures. Loss of rack integrity may affect the propagation of a zirconium fire.

If the gap radioactivity inventory is significant, then the spent fuel cladding temperature must be kept below 565 °C. If the consequences of aluminum/boron carbide relocation are acceptable, then 800 °C is a reasonable deterministic acceptance temperature, if uncertainties are less than the margin to 800 °C, and the effects of higher temperatures on the material are modeled. Otherwise, the temperature must be lower than the aluminum melting point (640 °C) or the aluminum/boron carbide eutectic melting point.

1.2 Evaluation of Existing Spent Fuel Heatup Analyses

In the 1980's, severe accidents in operating reactor SFPs were evaluated to assess the significance of the results of some laboratory studies on the possibility of self-sustaining zirconium oxidation and fire propagation between assemblies in an air-cooled environment, and also to assess the impact of the increase in the use of high density spent fuel storage racks on severe accidents in spent fuel pools. This issue was identified as Generic Safety Issue (GI) 82. Sandia National Laboratory (SNL) and Brookhaven National Laboratory (BNL) used the SFUEL and SFUEL1W computer codes to calculate spent fuel heatup in these studies. While decommissioned plants were not addressed in the study, many of the insights gained from these studies are applicable to decommissioned plants.

More recently, BNL developed a new computer code, SHARP, that was intended to provide a simplified analysis method to model plant-specific spent fuel configurations for spent fuel heatup calculations at decommissioned plants. Some of this work was built on the assumption used by SNL and BNL in their studies in support of GI 82.

1.2.1 SFUEL Series Based Analyses

Extensive work on the phenomena of zirconium oxidation in air for a SFP configuration was performed by SNL and BNL in support of GI 82. SNL investigated the heatup of spent fuel, the potential for self-sustaining zirconium oxidation, and the propagation to adjacent assemblies [Ref. 1, 6]. SNL used SFUEL and SFUEL1W computer codes to analyze the thermal-hydraulic phenomena, assuming complete drainage of the SFP water. In NUREG/CR-4982 [Ref. 4], BNL extended the SNL studies on the phenomenology of zirconium-air oxidation and its propagation in spent fuel assemblies. The SFUEL series of codes includes all modes of heat transfer, including radiation. However, radiation heat transfer may have been underestimated due to the assumed fuel bundle arrangement.

In NUREG/CR-0649, SNL concluded that decay heat and configuration are important parameters. SNL found that key configuration variables are the baseplate hole size, downcomer width, and the availability of open spaces for airflow. They also found that building ventilation is an important configuration variable.

The draft SNL report investigated the potential for oxidation propagation to adjacent assemblies. If decay heat is sufficient to raise the clad temperature in a fuel assembly to within approximately one hundred degrees of the point of runaway oxidation, then the radiative heat from an adjacent

assembly that reached the onset of rapid oxidation could raise the temperature of the first assembly to the runaway oxidation temperature. The report also
discusses small-scale experiments involving clad temperatures greater than 1000 °C. SNL hypothesized that molten zirconium material would slump or relocate towards the bottom of the racks and consequently would not be involved in the oxidation reaction. NUREG/CR-4982 did not allow oxidation to occur at temperatures higher than 2100 °C to account for the zirconium melting and relocation. Otherwise, temperatures reached as high as 3500 °C. It was felt that not cutting off the oxidation overstated the propagation of a zirconium fire because of the fourth power temperature dependence of the radiation heat flux. The SFUEL series of codes did not model melting and relocation of materials.

In NUREG/CR-4982, BNL reviewed the SFUEL code and compared it to the SNL small-scale experiments and concluded that SFUEL was a valuable tool for assessing the likelihood of self-sustaining clad oxidation for a variety of spent fuel configurations in a drained pool. SNL reported the following critical decay times in NUREG/CR-0649 based on having no runaway oxidation. Critical decay time is defined as the length of time after shutdown when the most recently discharged fuel temperature will not exceed the chosen fuel failure criteria when cooled by air only.

700 days PWR, 6 kW/MTU decay power per assembly, high density rack, 10.25" pitch, 5" orifice, 1-inch from storage wall

280 days PWR, same as above except for 1 foot from storage wall

180 days BWR, 14 kW/MTU decay power per assembly, cylindrical baskets, 8.5" pitch, 1.5" orifice

unknown BWR, high-density rack, SFUEL1W code was limited to computation of BWR low-density racks.

High-density racks with a 5-inch orifice are the most representative of current storage practices. A critical decay time for high-density BWR racks was not provided due to code limitations. Low-density and cylindrical storage rack configurations are no longer representative of spent fuel storage. All currently operating and recently shutdown plants have some high-density racks in the pool. For an assembly in a high-density PWR rack with a 5-inch orifice, a decay power below 6 kW/MTU did not result in runaway zirconium oxidation. All of these estimates were based on perfect ventilation (i.e., unlimited, ambient-temperature air) and burnup rates of 33 GWD/MTU. Currently, some PWRs are permitted to burn up to 62 GWD/MTU and some BWRs to 60 GWD/MTU. For fuel burnup of 60 GWD/MTU, the staff estimates the decay time for a bundle to reach 6 kW/MTU will increase from 2 years to approximately 3 years. Therefore, the staff expects the difference between critical decay times for PWRs and BWRs to decrease and that the BWR critical decay time for current burnups and rack designs would now be longer than the SNL estimate for high-density PWR racks. The SNL calculations also do not appear to have included grid spacer loss coefficients, which can have a significant effect since the resistance of the grid spacers is greater than the resistance of a 5-inch orifice. There is no mixing between the rising air leaving the fuel racks, and the relatively cooler air moving down into the pool. Including the grid spacer resistance, accounting for mixing and limiting the building ventilation flow to rated conditions, will result in the critical decay power to be less than 6 kW/MTU. The SNL calculations may have understated the effective radiation heat transfer heat sink due to the assumed fuel geometry in the calculations. A more realistic fuel configuration pattern in the SFP would give a

better estimate of the radiation heat sink and raise the critical decay power needed for significant oxidation.
While the studies in support of GSI 82 provided useful insights to air-cooled spent fuel assemblies, it is the opinion of the staff that they do not provide an adequate basis for exemptions. The studies were not meant to establish exemption criteria and lack sufficient information for all the parameters that could affect the decay time. Additionally, the reports are based on burnup values at that time. Since burnup values have increased, the results may not be directly applicable to today's spent fuel.

The general conclusions and the phenomena described in the studies assist in assessing issues for decommissioned plants. However, the calculated decay time values do not represent current plant operational and storage practices.

1.2.2 SHARP Based Analyses

In NUREG/CR-6451 [Ref. 7], BNL investigated spent fuel heatup that could lead to a zirconium fire at permanently shutdown plants. BNL developed a new computer code, SHARP (Spent Fuel Heatup Analytical Response Program), to calculate critical decay times to preclude zirconium oxidation for spent fuel. The code was intended to study thermal hydraulic characteristics and to calculate spent fuel heatup up to temperatures of approximately 600 °C. SHARP is limited to low temperatures since it lacks models for radiation heat transfer, zirconium oxidation, and materials melting and relocating. SHARP also lacks modeling for grid spacer losses and neglects mixing between the rising hot air and the falling cooler air in the SFP. BNL reported the following generic critical decay times using the SHARP code.

17 months for a PWR, high density rack, 60 GWD/MTU burnup; 10.4" pitch; 5" orifice
7 months for a BWR, high density rack, 40 GWD/MTU burnup; 6.25" pitch; 4" orifice

The above decay times are based on a maximum cladding temperature of 565 °C. The parameters listed with the critical decay times are generally representative of operating practices. Current fuel burnups in some plants, however, have increased to values higher than those used by BNL and perfect ventilation was assumed, which could lead to an underestimation of the critical decay times.

The SHARP code was not significantly benchmarked, validated or verified. The critical decay times above are shorter than those calculated in NUREG/CR-0649 and NUREG/CR-4982, particularly when the lower cladding temperature used for fuel failure and the higher decay heats used in the earlier analyses are taken into account. This appears to be driven in part, by the fact that the decay heat at a given burnup in the SHARP calculations is significantly lower than what is used in the SFUEL calculations. The staff has identified several areas that require code modifications, which will increase the calculated critical decay times. It is not adequate for use as technical bases by licensees without further code modifications and verification. NUREG/CR-6541 was intended as an assessment to steer rulemaking activities. The report was neither intended nor structured to provide a basis for exemptions. The staff does not rely on this study for heatup analysis information due to the code that the decay time conclusions were based upon.

1.3 Heatup Calculation Uncertainties and Sensitivities

The phenomenology needed to model spent fuel heatup is dependent on the chosen cladding temperature success criterion and the assumed accident scenario. Many assumptions and modeling deficiencies exist in the current calculations. The staff reviewed the models to assess the impact of those modeling assumptions. Some of these uncertainties for the SFUEL series codes are further discussed in NUREG/CR-4982. For cases of flow mixing, decay heat, bundle flow resistance and other severe accident phenomena, additional information is provided here.

Calculations performed to date assume that the building, fuel, and rack geometry remain intact. This would not be a valid assumption if a seismic event or a cask drop damaged some of the fuel racks or the building. Rack integrity may not be a good assumption after the onset of significant zirconium oxidation due to fuel failure criteria issues discussed in Section 1.1. The building may also be hot enough to ignite other materials. Assuming that the racks remain intact is the most optimistic assumption that can be made about the rack geometry. Any damage to the racks or the building could significantly reduce the coolability of the fuel.

Previous SFUEL, SFUEL1W, and SHARP calculations, used in the resolution of GI 82 and decommissioning studies, used a perfect ventilation assumption. With the perfect ventilation assumption an unlimited amount of fresh, ambient-temperature air is available. This assumption would be valid if the building failed early in the event or if large portions of the walls and ceilings were open. If the building does not fail, the spent fuel building ventilation flow rate would dictate the airflow available. Mixing between the rising hot air and the descending cooler air in the spent fuel pool is not modeled in the codes.

The spent fuel building ventilation flow rate is important in determining the overall building energy balance. Airflow through the building is an important heat removal mechanism. Most of the air would recirculate in the building and the air drawn under the racks would be higher than ambient temperature and, therefore, less heat removal would occur. Airflow also provides a source of oxygen for zirconium oxidation. Sensitivity studies have shown that heatup rates increase with decreasing ventilation flow, but that very low ventilation rates limit the rate of oxidation. Other oxidation reactions (fires) that occur in the building will also deplete available oxygen in the building. Zirconium-Nitrogen reaction modeling is not included in the SFUEL code and may have an impact on zero and low ventilation cases. GSI 82 studies concluded that the perfect ventilation assumption was more conservative than no ventilation because the oxidation reaction became oxygen starved with no ventilation. These studies did not consider the failure modes of the building under high temperature scenarios. Intermediate ventilation rate results were not studied and give longer critical decay times than the perfect ventilation case.

A key fuel heat removal mechanism is buoyancy-driven natural circulation. The calculated airflow and peak temperatures are very sensitive to flow resistances in the storage racks, fuel bundles and downcomer. The downcomer flow resistance is determined by the spacing between the fuel racks and the wall of the SFP. The storage rack resistance is determined by the orifice size at the bottom entrance to the fuel bundle. Smaller inlet orifices have higher flow resistance. As shown by SFUEL and SHARP calculations, changes in the rack-wall spacing and the orifice size over the range of designs can shift critical decay times by more than a year. The fuel bundle flow resistance is determined by the rod spacing, the grid spacers, intermediate flow mixers and the upper and lower tie plates. SFUEL and SHARP calculations have neglected the losses from the grid spacers, intermediate flow mixers and the tie plates. These flow resistances will be higher than those from the rack inlet orifice in some cases. Therefore, inclusion of this additional flow resistance may significantly extend the critical decay time for some cases. NUREG/CR-4982 concluded that the largest source of uncertainty was due to the natural circulation flow rates.

The downcomer and bundle inlet air temperatures and mass flow rates are important in determining the peak cladding temperature. The extent of flow mixing will determine the air temperatures at the downcomer and bundle inlet. The SFUEL and SHARP calculations assume a well-mixed building air space. The downcomer inlet temperature is set equal to the building temperature. This assumption neglects the mixing that occurs between the hot air rising from the bundles and the cooler air descending down the SFP wall. Computational fluid dynamics calculations performed by the NRC using the FLUENT code and Pacific Northwest National Laboratory using the TEMPEST code indicates that the well-mixed building is not a good assumption. The mixing that occurs between the cool air flowing down into the pool and the hot air flowing up out of the fuel bundles can significantly increase peak cladding temperatures. Even using different turbulent mixing models can affect the peak temperatures by approximately 100 °C. The calculations indicate that fully 3-dimensional calculations may be needed to accurately predict the mixing because unrealistic flow topologies in 2-dimensional approximations may overstate the mixing. The calculations also indicate that the quasi-steady state assumptions for conditions above the fuel rack may not be appropriate. Time varying temperature fluctuations on the order of 100 °C have been observed in 3D calculations.

Radiation heat transfer is important in spent fuel pool heatup calculations. Radiation heat transfer can affect both the onset of a zirconium fire and the propagation of a fire. Both the SFP loading pattern and the geometry of the fuel racks can affect the radiation heat transfer between adjacent bundles. Simple gray body calculations show that at clad temperatures of 800 °C, a temperature difference of 100 °C between adjacent bundles would cause the radiation heat flux to exceed the critical decay power of 6 kW/MTU. Therefore, the temperature difference that could be maintained between adjacent bundles is highly constrained by the low decay heat levels. SFUEL calculations performed by SNL and BNL included radiation heat transfer, but the radiation heat transfer was underpredicted since the spent fuel placement is two-dimensional and the hottest elements are in the middle of the pool with cooler elements placed progressively toward the pool walls. Heat transfer between hotter and cooler assemblies has the potential to be significantly higher if the fuel bundles were intermixed in a realistic loading pattern.

At temperatures below 800 °C, the SFP heat source is dominated by the spent fuel decay heat. SNL and BNL found that, for high-density PWR racks, that 6 kW/MTU was the critical decay heat level for a zirconium fire to occur in configurations resembling current fuel storage practices. At the fuel burnups used in the calculations, this critical decay heat level was reached after two years. Decay heat calculations in NUREG/CR-5625 [Ref. 8] were performed to be the basis for calculating fuel assembly decay heat inputs for dry cask storage analyses. These decay heat calculations are consistent with the decay heat used in SFUEL calculations. Extrapolation of the decay heat calculations from NUREG/CR-5625 to current burnups indicate that approximately 3 years will be needed to reach a decay heat of 6 kW/MTU. The extrapolation has been confirmed to provide a reasonable decay heat approximation by performing ORIGEN calculations that extend to higher burnup. The critical decay heat may actually be as low as 3kW/MTU when in-bundle peaking effects, higher density rack configurations and rated build ventilation flows are taken into account.

Several licensees have proposed using the current Standard Review Plan (NUREG-0800) Branch Technical Position ASB 9-2 decay heat model for SFP heatup calculations. Using ASB 9-2 decay heat with a “k factor” of 0.1 produces non-conservative decay heat values in the range of 1 to 4 years after shutdown. ASB 9-2 explicitly states that it is good for times less than 10,000,000 seconds (~ 116 days). The basis of ASB 9-2 is the 1971 ANS draft decay heat standard. The standard gives “k factors” to use beyond 10,000,000 seconds. The staff has found that a “k factor of 0.2" will produce conservative decay heat values compared to ORIGEN calculations for the range of 1 to 4 years after shutdown.

1.4 Zirconium Oxidation Temperature

At temperatures below the onset of self-sustaining oxidation, decay heat of the fuel dominates the heat source. When zirconium reaches temperatures where air oxidation is significant, the heat source is dominated by oxidation. The energy of the reaction is 262 kcal per mole of zirconium. In air, the oxidation rate and the energy of the reaction is higher than zirconium-steam oxidation. Much less data exists for zirconium-air oxidation than for zirconium-steam oxidation. A large amount of data exists for zirconium-steam oxidation because of the large amount of research performed under the ECCS research program [Ref. 9]. If all of the zirconium in a full 17x17 PWR fuel bundle fully oxidizes in air over the period of an hour, the average power from the oxidation is 0.3 MW. The critical decay heat as determined with SFUEL is approximately 2.7 kW for the bundle. The oxidation power source would amount to approximately 60 MW if the whole core was burning. A 20,000 cubic feet per minute (CFM) airflow rate is needed to support that reaction rate based on 100-percent oxygen utilization. The SFUEL oxidation rate was modeled using several parabolic rate equations based on available data. SFUEL had limited verification against SNL experiments that studied the potential of zirconium fire propagation. BNL determined that although they could not find a basis for rejecting the oxidation rate model used in SFUEL, uncertainties in oxidation of zirconium in air could change the critical decay heat by up to 25-percent. It was found that the onset of runaway zirconium oxidation could occur at temperatures as low as 800 °C. Different alloys of zirconium had oxidation rates that vary by as much as a factor of four. Apparently it was found that oxidation in air was worse than oxidation in pure oxygen. This suggests that the nitrogen concentration can have a significant impact on the oxidation rate. Since the relative concentration of oxygen and nitrogen varies as oxygen is consumed this causes additional uncertainty in the oxidation rate. The oxidation was cut off at 2100 °C in the BNL calculations in support of GI 82. This was done to simulate zirconium clad relocation when the melting point of zirconium was reached. If the oxidation was not cut off, temperatures could reach as high as 3500 °C. It was felt the propagation to adjacent bundles was overpredicted if no cutoff temperature is used due to the fourth power dependence of temperature on the radiation heat fluxes.

The combustion literature cited in the June 1999 draft report shows that there is a large range in the temperature for zirconium ignition in air. Evidence cited from the literature states that bulk zirconium cannot ignite at temperatures lower than 1300-1600 °C. It is known from the extensive emergency core cooling system (ECCS) and severe accident research programs that zirconium-steam runaway oxidation occurs at temperatures below 1300 °C. Since oxidation in air occurs more rapidly than oxidation in steam, temperatures in this range are not credible for the onset of runaway oxidation in air. Correlations listed [Ref. 10] give ignition temperatures for small zirconium samples in the range of runaway oxidation computed by the SFUEL series codes when the geometry factors calculated from zirconium cladding are input into the correlations. Only one reference [Ref. 11] appears to be applicable to zirconium oxidation in sustained heating of fuel rods. In the referenced test, sections of zirconium tubing were oxidized at temperatures of 700 °C, 800 °C and 900 °C for 1 hour. The average oxidation rate tripled for each 100 °C increase in temperature. This is consistent with the change in oxidation rates predicted by the parabolic rate equations examined in NUREG/CR-4982. The zirconium combustion literature reviewed for ignition temperature did not discount or provide alternate oxidation rates that should be used in the SFUEL calculations.

As discussed earlier, current operating plants burn fuel to higher levels than used in the evaluations. The BNL and SNL studies in support of GI 82 represented operating practices of the 1980's with burnup level around 33 GWD/MTU. In NUREG/CR-6451, BNL used burnup values of 40 and 60 GWD/MTU for BWRs and PWRs, respectively. While these values are closer to current operating practices, they still underestimate peak burnup values. Additionally, the decay heat at the same burnup level used in the SHARP analyses is significantly lower than that used in the SFUEL analyses. Given that burnup is an important parameter for determining the critical decay time, this is a significant change. The increase in burnup level will increase the critical decay time needed to ensure that air-cooling is sufficient to maintain the zirconium cladding below the oxidation temperature.

The BNL and SNL studies in support of GI 82 represented storage practices of the 1980's when plants were starting to convert to high-density storage racks. The studies did not address high density BWR racks, and the high-density PWR racks in the reports were not as dense as the designs used by many plants today. The higher density racking currently used will decrease the airflow available for heat removal. Therefore, lower decay heat values are needed to ensure that air-cooling is sufficient to maintain the zirconium clad below the oxidation temperature.

1.5 Estimated Heatup Time of Uncovered Spent Fuel

The staff recognized that the decay time necessary to ensure that air cooling was adequate to remain below the temperature of self-sustaining zirconium oxidation was a conservative criteria for the reduction in emergency preparedness criteria. Using the fact that the decay heat of the fuel is reducing with time, credit could be given, if quantified, for the increasing length of time for the accident to progress after all water is lost from the SFP. The staff sought to quantify the decay time since final shutdown such that the heatup time of the fuel after uncovery was adequate for effective protective measures using local emergency response.

The heatup time of the fuel depends on the amount of decay heat in the fuel, and the amount of heat removal available for the fuel. The amount of decay heat is dependent on the burnup. The amount of heat removal is dependent on several variables, as discussed above, 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 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. 8]. The decay heat in NUREG/CR-5625 is based on ORIGEN calculations. The tables 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 staff has also considered a scenario with 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 breech occured in the liner at or below the top of active fuel. Section 5.1 of NUREG/CR-0649 analyzes the partial draindown problem. For the worst case draindown and a lower bound approximation for heat transfer to the water and the building the heatup time slightly less than the heatup time for the corresponding air cooled case. More accurate modeling could extend the heatup time to be comparable to or longer than the air cooled case.

Calculations, assuming an instant draindown of the pool and air-cooling, only show a heatup time to fission product release of 10 to 15 hours at 1 year after shutdown. The worst case partial draindown could release fission products in 5 to 10 hours at 1 year after shutdown.

1.6 Critical Decay Times to Reach Sufficient Air Cooling

Based on the above discussion, the staff concludes the following with respect to critical decay times. Calculations using the SFUEL code in support of GI-82 have determined a critical specific decay heat of 6 kW/MTU is needed for the onset of runaway zirconium oxidation. The 6 kW/MTU estimate calculated using SFUEL in a high-density storage rack configuration is reasonable and is based on the best calculations to date. However, this estimate is based on perfect ventilation conditions in the building and lower density rack configurations than exist today.

For high burnup PWR and BWR fuel, the staff estimates it will take approximately 3 years to reach the critical decay heat level cited in NUREG/CR-4982. Better modeling of flow mixing and accounting for the grid spacer and tie plate flow resistance could reduce the critical decay power level and increase the critical decay time beyond 3 years, but this may be counterbalanced by increased radiation heat transfer from realistic fuel bundle loading. Other assumptions, such as imperfect ventilation, could extend the critical decay time for the onset of a zirconium fire by 1 to 2 years. The critical decay heat may actually be as low as 3 kW/MTU when peak to average rod bundle peaking effects and higher density rack configurations are taken into account. Accounting for imperfect ventilation and higher density spent fuel storage in the racks, the staff estimates it will take approximately 4 to 5 years to reach a decay heat of 3 kW/MTU for current plant fuel burnups. Plant-specific calculations using fuel decay heat based on the actual plant operating history and spent fuel configurations could yield significantly shorter critical decay times. Calculations performed using checkerboard fuel loadings indicate that the critical decay time can be reduced by one year or more if the highest power fuel is interspersed with low powered fuel or empty rack spaces.

1.7 Fire Propagation

The staff has not performed a sufficient amount of research to fully understand and predict the propagation of zirconium fires in a spent fuel pool. Based on the limited amount of work performed to date, the propagation is probably limited to less than 2 full cores at a time of 1 year after shutdown. This estimate is based on lowering the GI 82 estimate of the 6KW/MTU fire threshold to 3KW/MTU to account for building ventilation effects. The actual propagation will probably be dependent on the actual fuel loading configuration in the spent fuel pool. A long term experimental and analytical research program would be required to reliably predict the propagation of a zirconium fire in a spent fuel pool.

1.8 Guidelines for Spent Fuel Pool Heatup Analysis

Licensees must use an appropriate evaluation model for any site specific spent fuel pool heatup calculations. An evaluation model includes one or more computer programs and other information necessary for application of the calculation framework to a specific transient or accident, such as mathematical models used, assumptions included in the programs, a procedure for treating the program input and output information, specification of those portions of the analysis not included in the computer programs, values of parameters and other information necessary to specify the calculation procedure.

The code(s) should be validated and documentation of the modeling, verification, validation and use of the computer programs should be maintained to document the adequacy of the computer program. Finally, the code should be developed and maintained under a Quality Assurance program that meets the requirements of 10 CFR Part 50, Appendix B.

Depending on the margins available, sensitivity or uncertainty analysis should be performed (and documented) to confirm that the combined code and application uncertainty is less than the design margin for the safety parameter of interest in the calculation.

Spent fuel pool heatup analyses should consider decay heat removal from both the fuel racks and the building. An accurate determination of fuel cladding temperatures in the spent fuel pool requires fluid flow and heat transfer analyses. The primary components of a heatup anlaysis are described in the paragraphs that follow.

The spent fuel pool heat source is determined by the decay heat in the spent fuel. The analysis should use methods that are appropriate for the fuel burnup and decay time. The lowest possible decay heat input can only be achieved by accurately tracking the burnup history of individual spent fuel pool bundles. The method for calulating the spent fuel pool decay heat including its uncertainty should be justified.

The fluid conditions immediately above the spent fuel racks are determined by the heat removal from the spent fuel racks to the outside of the building. This is primarily determined by the building ventilation flow rate. Heat transfer through the walls can also be important at low ventilation rates. Heat removal from the top of the fuel racks to the bulk building atmosphere is primarily determined by buoyancy driven flows. Radiation heat transfer can also be significant. A steady state solution may not exist for the problem being analyzed. Time dependent variations must be considered in the analysis if time averaging is used in order to use a steady state approximation. Spatial variations must also be considered if spatial averaging is performed to simplify the analysis. The choice of a turbulence model must be justified and its impact on the overall calculation uncertainty must be evaluated.

Heat removal from the spent fuel pool racks is governed by the fluid conditions immediately above the fuel racks and buoyancy driven natural circulation in the racks. The heat removal rates are determined by the balance between buoyancy driving forces and the flow resistance of the downflow area and the fuel racks. Downflow in low powered spent fuel bundles should be considered and accounted for. This can be very important in densely packed spent fuel pools with little downcomer area available for downflow. Calculations should use wall friction factors and additive loss coefficients (including those due to orifices and grid spacers) that are appropriate for both the flow regime and the geometry.

The staff's experience suggests conduction, convection and radiation heat transfer can all be important in spent fuel pool rack heatup calculations. Neglect of any heat transfer mode should be justified. Convective heat transfer coefficients should be appropriate for both the flow regime and the geometry.

Certain phenomena will occur as peak temperatures increase and should be considered for in the analysis. Experimental data has shown that clad ballooning will occur if cladding temperatures exceed temperatures of approximately 560 °C for longer than 10 hours. The temperature threshold will be lower for longer thermal loading times. If clad ballooning is expected additional flow losses may occur. Many spent fuel pool racks use BORAL plates for criticality control. Aluminum melts at approximately 640 °C. Heat transfer calculations within the rack should predict the temperature of any aluminum in the rack. If the temperature of any aluminum in the racks is predicted to exceed its melting temperature the consequences of the melting and relocation must be analyzed. Possible consequences of aluminum melting and relocation include flow blockages and criticality. Zirconium oxidation in air can have a significant effect on heatup calculations at temperatures above 600 °C. Zirconium oxidation must be modeled using and appropriate reaction kinetics model that is supported by experimental data.

The licensee must integrate all pieces of the analysis to determine if runaway zirconium oxidation will occur. The impact of uncertainties on the predicted temperatures must be evaluated and compared to the margin available in the calculation. The propagation of uncertainties through each part of the analysis must be properly treated.

References:

1. Benjamin, et. al., "Spent Fuel Heatup Following Loss of Water During Storage", NUREG/CR-0649, March 1979.
2. Nourbakhsh, et. al., "Analysis of Spent Fuel Heatup Following Loss of Water in a Spent Fuel Pool", NUREG/CR-6441.
3. Smith, C. W., "Calculated Fuel Perforation Temperatures, Commercial Power Reactor Fuels", NEDO-10093, September 1969.
4. Sailor, et. al., "Severe Accidents in Spent Fuel Pools in Support of Generic Issue 82", NUREG/CR-4982.
5. MELCOR 1.8.1Computer Code Manual, Volume 2: Reference Manuals and Programmers' Guides, June 1991.
6. Pisano, et. al., "The Potential for Propagation of a Self-Sustaining Zirconium Oxidation Following Loss of Water in a Spent Fuel Pool", NRC draft report, January 1984.
7. "A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants," NUREG/CR-6451, dated August 1997.
8. Hermann, et.al., "Technical Support for a Proposed Decay Heat Guide Using SAS2H/ORIGEN-S Data", NUREG/CR-5625, September 1994.
9. "Compendium of ECCS Research for Realistic LOCA Analysis", NUREG-1230, December 1988.

10. Cooper, T. D., “Review of Zirconium-Zircaloy Pyrophoricity, RHO-RE-ST-31P, Rockwell International, November 1984.
11. Kullen, et. al., “ An Assessment of Zirconium Pyrophoricity and Recommendations for Handling Wast Hulls”, ANL-77-63, Argonne National Laboratory, November 1977.

Appendix 2.0 Assessment of Spent Fuel Pool Risk at Decommissioning Plants

Introduction

As the number of decommissioning plants increases, the ability to address generically 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 shut down plants involves accidents associated with spent fuel stored in the spent fuel pool. Previously, requests for relief from regulatory requirements that are less safety significant for decommissioning plants than operating reactors were decided 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 draft final report 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 spent fuel pools is on the order of $3x10^{-6}$ per year when a utility follows certain industry commitments and certain of our recommendations. This frequency is made up of contributors from a detailed risk assessment of initiators ($4.3x10^{-7}$ per year), both internal and external, and a quasi-probabilistic contribution from seismic events ($<3x10^{-6}$ per year) 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 report 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 spent fuel pools at operating reactors that involved draining the spent fuel pool 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 spent fuel pools.

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 off-site if they are to have a significant health effect on local (first few miles outside the exclusion area) and more distant populations. Deterministic evaluations (see Appendix 1) indicate that zirconium cladding fires cannot be ruled out for loss of spent fuel pool cooling for fuel that has been shut down and removed from an operating reactor within approximately five years[1]. The consequence analysis (Appendix 4) indicates that zirconium cladding fires could give off-site doses that the NRC would consider unacceptable. To assess the risk (essentially, "frequency" times "consequences") 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. This assessment was

[1]This estimate can be significantly shorter or perhaps somewhat longer depending on fuel enrichment, fuel burnup, and configuration of the fuel in the spent fuel pool.

the most comprehensive performed on spent fuel pool risk. The preliminary risk assessment was made publicly available early in the process (June 1999) so that the public and the nuclear industry could track the NRCs 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 spent fuel pool 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 (See NEI letter dated November 12, 1999, Appendix 6) that provide boundaries within which the risk assessment's assumptions have been refined. The draft final risk 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, and the staff's ongoing efforts to improve the assessment. The report provides a technical basis for determining the acceptability of exemption requests and future rulemaking on decommissioning plant risk.

In performing the preliminary risk assessment, the staff chose to look 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). The staff modeled a decommissioning plant's spent fuel pool cooling system based on the sled-mounted systems that are used at many current decommissioning plants. One representative spent fuel pool configuration (See Appendix 2a, Figure 2.1) was chosen for the evaluation except for seismic events, where the PWR and BWR spent fuel pool 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 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 spent fuel pools, their cooling systems, and other support systems.

From the perspective of off-site consequences, the staff only concerned itself with 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 off-site in order to have potentially significant off-site consequences. The staff chose the timing of when the spent fuel pool inventory is drained to 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, and because there was uncertainty in 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) it would take five or more days to uncover the top of the fuel pool, it was considered of small numerical benefit (and significant analytical effort) if the potential additional two 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 6) that proved very 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 off-site). 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 the back of 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 this current analysis, Ref. 1 was updated based on:

- stakeholder feedback on the original analysis
- NEI commitments as documented in Ref. 2
- a revised human reliability analysis (HRA) approach
- peer review of the technical analysis by the Idaho National Engineering and Environmental Laboratory (INEEL).

This updated PRA, performed by a combination of INEEL and NRC staff, addresses the following initiating events:

- loss of spent fuel pool cooling
- fire leading to loss of spent fuel pool cooling
- loss of off-site power due to plant centered and grid related causes
- loss of off-site power due to severe weather
- non-catastrophic loss of spent fuel pool inventory

Those low frequency events such as earthquakes, aircraft crashes, heavy load drops, and tornado strikes that could lead to catastrophic pool failure are dealt with elsewhere. 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 by decommissioning project managers (NRC Staff) 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

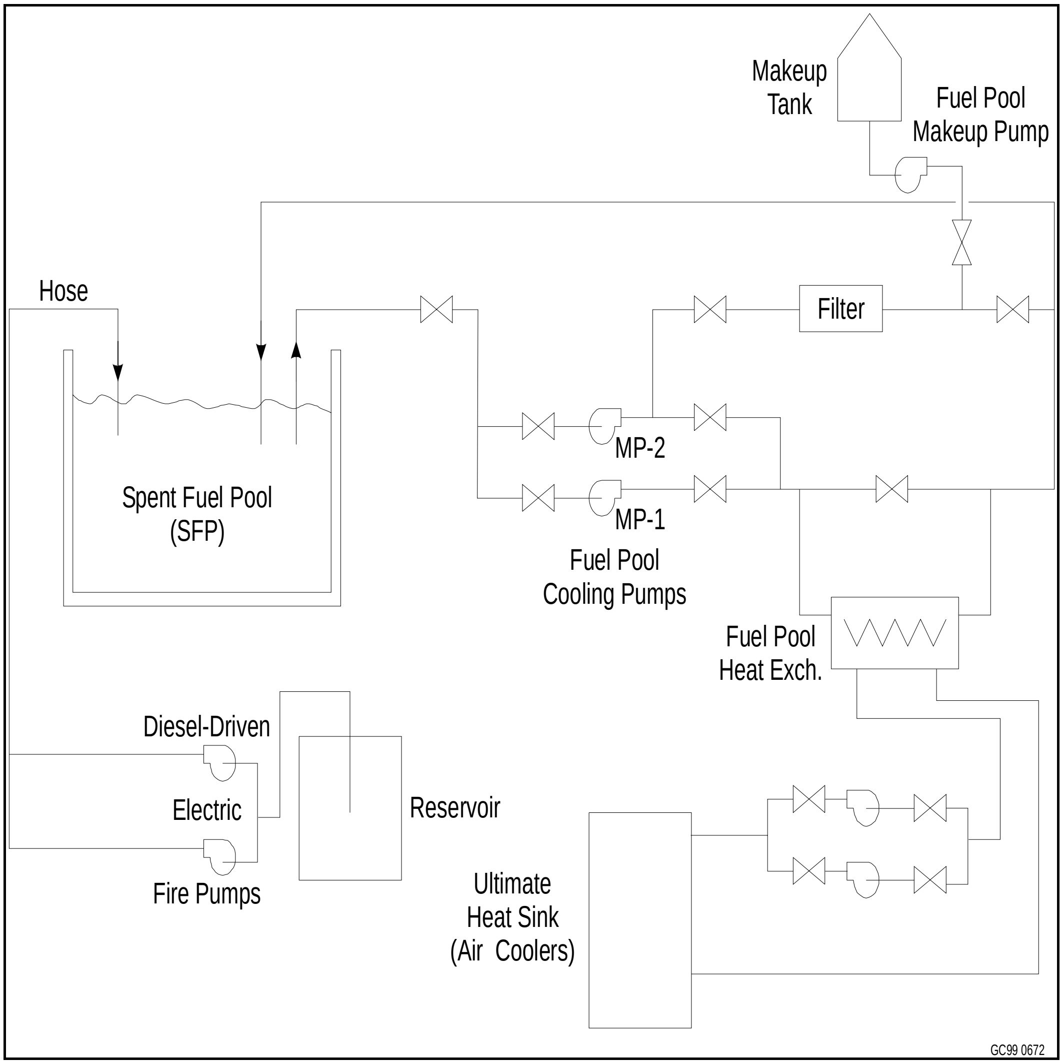


Figure 2.1 Simplified Diagram of Spent Fuel Pool Cooling and Inventory Make-up Systems

motor-driven pumps, a heat exchanger, an ultimate heat sink, a make-up tank, filtration system and isolation valves. Suction is taken via one of the two pumps on the primary side from the spent fuel pool 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 from the suction line is diverted to the filtration process and is returned to the discharge line. A regular make-up system supplements the small losses due to 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, a motor-driven (electric) and a diesel-driven, which provide firewater throughout the plant. A firewater hose station is provided in the SFP area. The firewater pumps are located in a separate structure.

3.0 Methodology

3.1 Logic Model

This section summarizes the spent fuel pool 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 detailed model documentation is provided in Attachments A and B. The event trees and fault trees presented in this report are meant to be generic enough to apply to many different configurations.

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. 3) show that less than 3 feet of water above the top of the fuel results in an environment that is rapidly lethal to anyone at the edge of the pool. For accident scenarios in which coolant inventory level has dropped to less than 3 feet above the top of fuel, recovery may require operators to approach the pool. 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 make-up 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. 4), 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 spent fuel pool during the decommissioning phase of a nuclear power plant's lifecycle is how much credit can be given to the operating staff to respond to an incident that impacts the spent fuel pool 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 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 human reliability analysis (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 spent fuel pool 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 spent fuel pool 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 50 to greater than 120 hours were estimated for heat up and boil off following loss of spent fuel pool 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 make-up systems, or even bringing in help from off site.

- 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 successful response to failures of the cooling function.

- There is expected to be little or no redundancy in the on-site mitigating capability as compared with the operating plant mode of operation. (In the staff's initial evaluation, because little redundant on-site equipment was assumed to be available, the failure to bring on off-site 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 spent fuel pool case.

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

- Due to 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 (see for example, Refs. 5, 6, and 7).

- Those factors that the industry has suggested that will help ensure adequate response (instrumentation, monitoring strategies, procedures, contingency plans) should be addressed (Ref. 2).

- 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 spent fuel pool (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:

- guidance for early action to establish contingency plans (e.g., alerting off-site agencies such as fire brigades) in parallel with a primary response such as carrying out repairs or lining up an on-site 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 quantification methods were applied, and each is briefly described below.

- The Technique for Human Error Prediction (THERP, Ref. 8). 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.

- 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 report. 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. 9) was employed for all other HEPs. This method separately evaluates the diagnosis or response planning errors and the execution errors.

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 inventory is lost in this manner due to the large amount of water in a spent fuel pool, the large specific heat of water, and the large latent heat of vaporization for water. Simple calculations for a typical-sized spent fuel pool yield the results in Table 3.1. These 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 a 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 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:

$$\text{Total Time} = 33\text{ hr} + (20\text{ ft})/(0.2\text{ ft/hr}) = 133\text{ hours}$$

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 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. 2) 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. 11) 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.

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 for 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 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.

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 off-site 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 off-site power.

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

Make-up 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 make-up 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 prior to 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, make-up pumps, firewater pumps, or any of the support systems.

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

- 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 core uncovery from spent fuel pool loss of cooling events, fire events, loss of off-site power, 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 due to 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. 12). 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 due to operator error (failure to respond), or loss of indication due to 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-LOF, 2.0E-3) was estimated based on information in reference 11. 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-LOF	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 spent fuel pool 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.

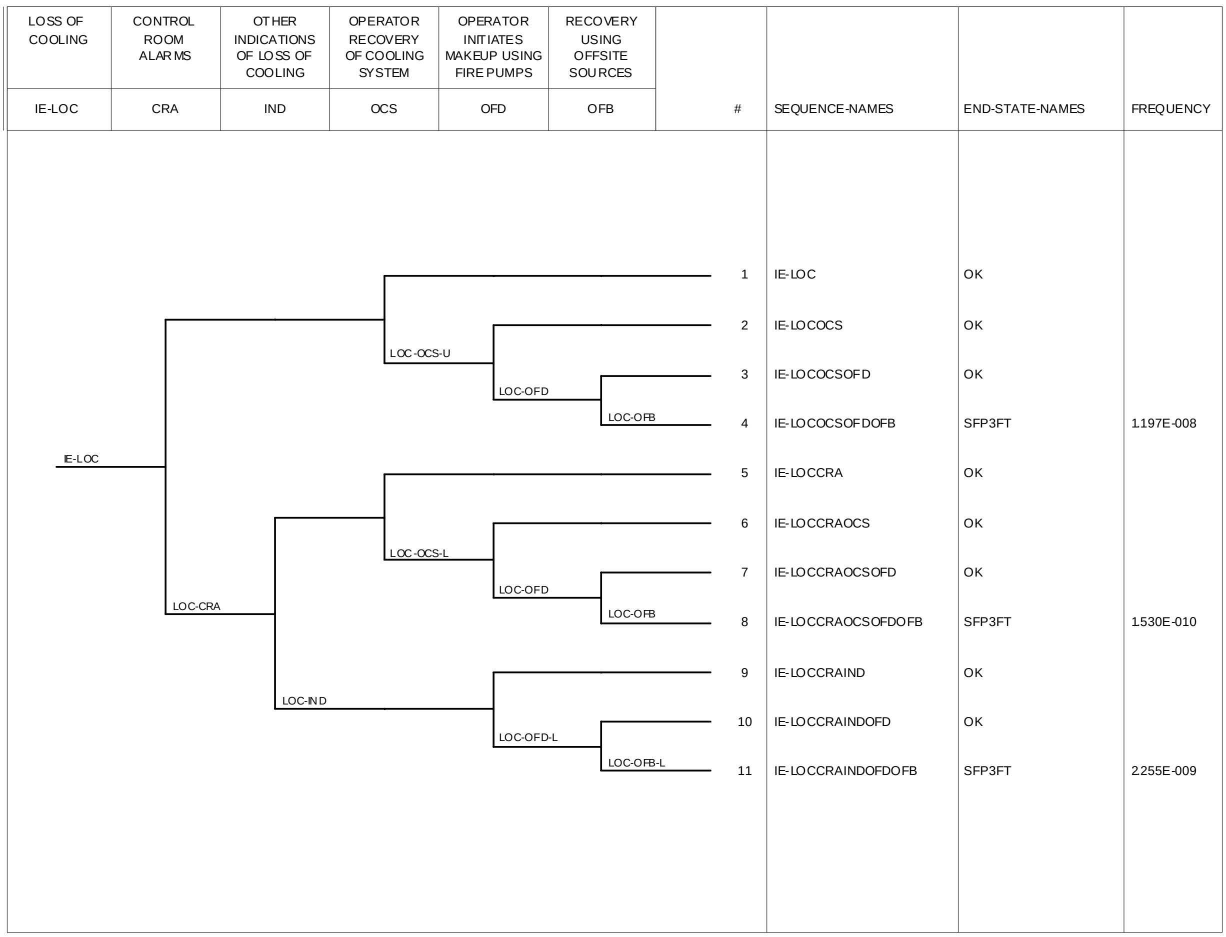
- Operators perform walk-downs and control room readouts once per shift (every 8 to 12 hours) and document observations in a log.

- 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 spent fuel pool 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 spent fuel pool cooling system becomes irrecoverable without pool make-up, it is assumed that multiple crews would have to fail. However, 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 spent fuel pool 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 make-up 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 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 due to cooling and whether the make-up 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 is not considered that this unduly biases 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 operator has 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 on-site operators.

4.1.4.3 Quantification

Human Error Probabilities

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 due to operator walk-down (HEP-COOL-REP-L), it is assumed that there is not enough time available to repair and restart the SFP make-up system in time to prevent bulk boiling, and 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 On-site 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 on-site 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 make-up 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 make-up 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 make-up source to the spent fuel pool without entry to the refuel floor, so that make-up can be provided even when the environment is uninhabitable due to steam and/or high radiation (NEI commitment no.8).

- Repair crew is different than on-site 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 make-up 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] = 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 due to 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. 12). 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 Off-site Sources

4.1.6.1 Event Description and Timing

This event accounts for recovery of coolant make-up using off-site sources given the failure of recovery actions using on-site sources. Adequate time is available for this action, provided that the operating staff recognizes that recovery of cooling using on-site sources will not be successful, and that off-site 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 off-site sources of make-up.

4.1.6.2 Relevant Assumptions

- The operators have 88 hours to provide make-up and inventory cooling.
- Procedures and training are in place that ensure that off-site 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 off-site sources.
- Operators have received formal training in the procedures.
- Off-site 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 off-site 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 off-site sources, and of the action itself. The availability of off-site 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.

4.1.6.4 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 core 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 make-up 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 make-up 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	1.0
HEP-RECG-FWSTART	Operators fail to diagnose 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-DEPEN	Repair crew fails to repair firewater system	5.0E-2
HEP-INV-OFFSITE	Operators fail to provide alternate sources of cooling from off-site	5.0E-2
FP-2PUMPS-FTF	Failure of firewater pump system	6.7E-4
SPC-LVL-LOF	Failure of control room alarm channel	1.0E-5
SPC-LVL-LOP	Electrical faults leading to alarm channel failure	2.0E-3

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 off-site 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 make-up – either using the site firewater system or by calling upon off-site 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 off-site 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. 13). 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 spent fuel pool 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 make-up 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 spent fuel pool facility is estimated to be (10/25)(2.4E-3/yr) = 9.6E-4/yr.

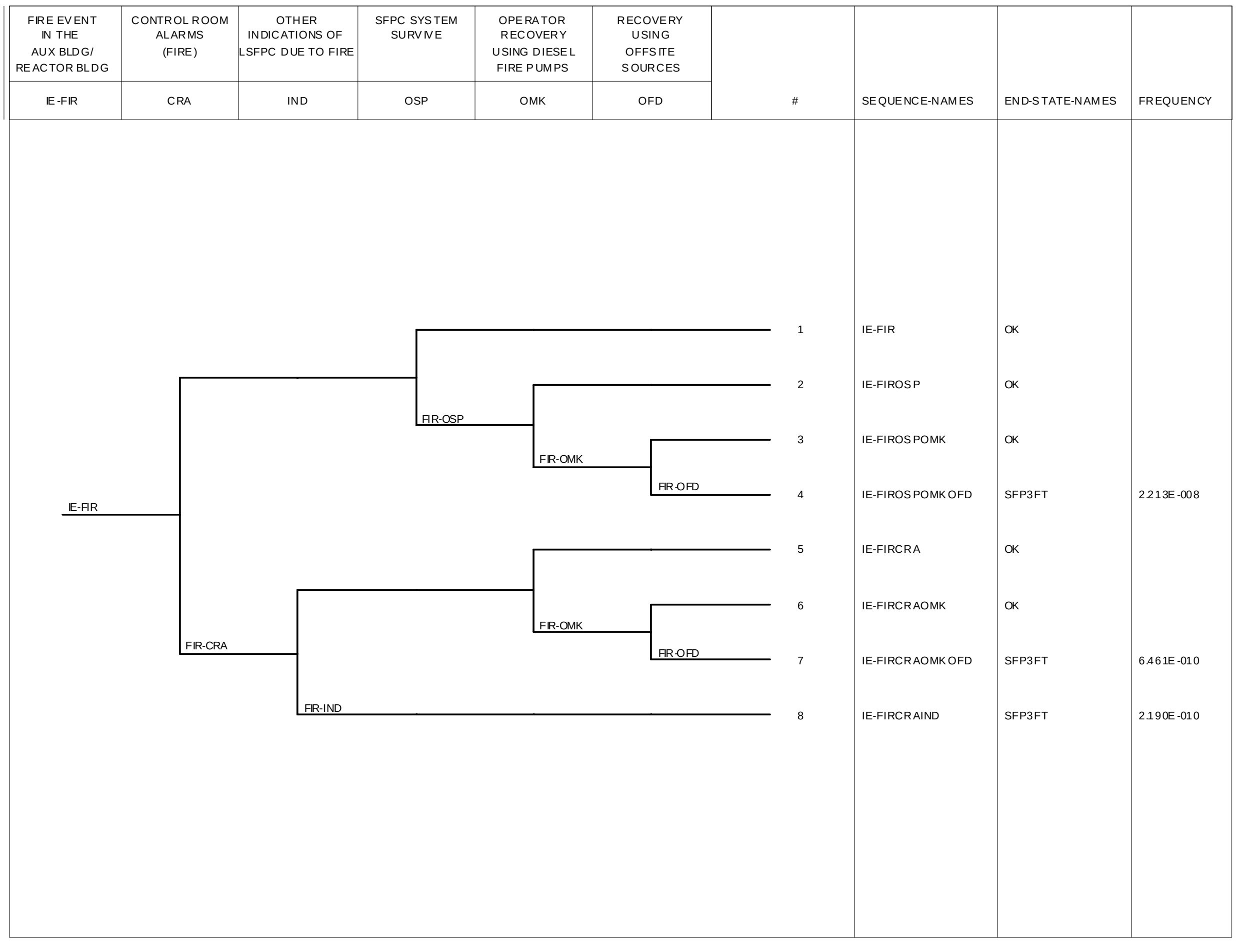


Figure 4.2 Fire initiating event tree

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 spent fuel pool 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	3E-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 due to operator error (failure to respond), failure of the detectors, or loss of indication due to 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

Human Error Probabilities

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).

Hardware Failure Probabilities

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-AOL (2.0E-3), was estimated based on information in reference 11.

4.2.2.4 Basic Event Probabilities

Basic Event	**Basic Event Probability**
HEP-DIAG-ALARM	3.0E-4
SFP-FIRE	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

Human Error Probability

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 off-site power, a burnt out area, smoke, etc.). The only source of dependency is one where a situation would result in the operator 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 make-up 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 on-site (i.e., diesel fire pumps) and off-site 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. Due to 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 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 off-site 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 On-site 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 make-up source to the spent fuel pool without entry to the refuel floor, so that make-up can be provided even when the environment is uninhabitable due to steam and/or high radiation (NEI commitment no.8).

- Inventory make-up using the firewater system is initiated by on-site 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 make-up using the firewater pumps are 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 operator may have to position a hose in the pool area. 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] = 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. 12).

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 Off-site Sources

4.2.6.1 Event Description and Timing

Given the failure of recovery actions using on-site sources, this event accounts for recovery of coolant make-up using off-site sources. Adequate time is available for this action, provided that the operators recognize that recovery of cooling using on-site sources will not be successful, and that off-site 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 off-site 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 off-site sources.
- Operators have received formal training in the procedures.
- Off-site resources are familiar with the facility.

4.2.6.3 Quantification

Human Error Probabilities

The event HEP-INV-OFFSITE represents failure to recognize that it is necessary to take the extreme measure of using off-site 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 off-site sources, and of the action itself. The availability of off-site 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 core 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 make-up 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 make-up sources.

Table 4.2 Basic Event Summary for the Internal Fire Event Tree

Basic Event Name	**Description**	**Basic Event Probability**
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-WLKDWN-DEPEN	Operators fail to observe the loss of cooling in walk-downs (dependent case)	5.0E-2
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 off-site	5.0E-2
FP-DGPUMP-FTF	Failure of firewater pump system	0.18
SFP-FIXE-LOA	Electrical faults causing loss of alarms	2E-3
SFP-FIRE-DETECT	Failure of fire detectors	5E-3

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

This event tree represents the loss of SFP cooling resulting from a loss of off-site power from plant-centered and grid-related events. Until off-site 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 Off-site Power (LOSP) event tree sequence progression.

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

4.3.1.1 Event Description

Initiating event IE-LP1 represents plant-centered and grid-related losses of off-site 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 off-site power grid cause the loss of off-site power.

4.3.1.2 Quantification

For plant-centered LOSP events, NUREG/CR-5496 (Ref. 16) estimates a frequency of .04/critical year for plant centered loss of off-site 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 off-site 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. 13), is assumed for the combined losses from plant-centered and grid-related events.

4.3.2 Top Event OPR – Off-site Power Recovery

4.3.2.1 Event Description and Timing

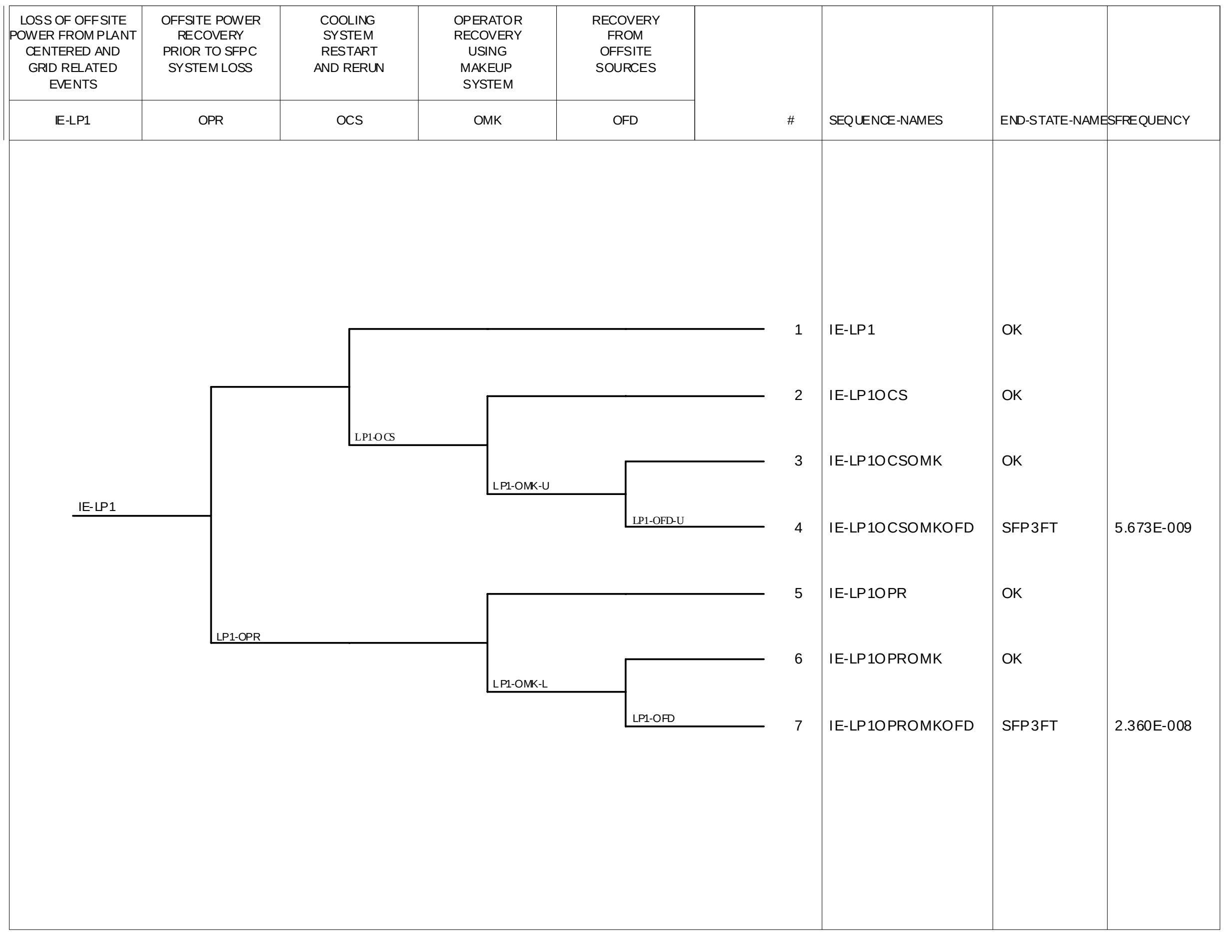
The fault tree for this top event (LP1-OPR) is a single basic event that represents the non-recovery probability of off-site power.

NUREG-1032 (Ref. 17) classified LOSP events into plant-centered, grid-related, and severe-weather-related categories, because these categories involved different mechanisms and also seemed to have different recovery times. Similarly, NUREG/CR-5496 (Ref. 16) 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 the quick recovery.
- Operators have received formal training and there are procedures to guide them (NEI commitment no. 2).

Figure 4.3 Plant centered and grid related loss of off-site power event tree



4.3.2.3 Quantification

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

4.3.2.4 Basic Event Probability

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

4.3.3 Top Event OCS – Cooling System Restart and Run

4.3.3.1 Event Description and Timing

This top event represents restarting the SFP cooling system, given that off-site 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 due to previous failures, moderately complex task due to potential valve lineups, highly trained staff, good ergonomics (well laid out and labeled matching procedures), and good work process.

A diagnosis error HEP-DIAG-SFPLP1, representing failure of the operators to recognize the loss of SFP cooling was also included. Success would most likely result from recognition that the electric pumps stop running once power is lost and require restart following recovery of power. If the operator fails to make an early diagnosis of loss of SFP cooling, then success could still be achieved during walk-downs following the loss of off-site 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. 13). It is assumed that SFPC system will be maintained since it is required to be running all the time.

4.3.3.4 Basic Event Probabilities

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

4.3.4 Top Event OMK – Operator Recovery Using Make-up Systems

4.3.4.1 Event Description and Timing

This top event represents the failure to provide make-up using the firewater pumps. If off-site 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 off-site power is not recovered then fault tree LP1-OMK-L represents this top event. In this case, the operator has only the diesel firewater pump available.

4.3.4.2 Relevant Assumptions

- It is assumed that the procedures guide the operators to wait until it is clear that spent fuel pool cooling cannot be reestablished (e.g., using cues such as the level drops to below the suction of the cooling system or the pool begins boiling) before using alternate make-up sources. Therefore, they have 88 hours to start a firewater pump.

- There is a means to remotely align a make-up source to the spent fuel pool without entry to the refuel floor, so that make-up can be provided even when the environment is uninhabitable due to steam and/or high radiation (NEI commitment no.8).
- Repair crew is different than on-site 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 make-up system, given that a loss of fuel pool cooling has been recognized. This event was quantified using the SPAR HRA technique. The assumptions included expansive time (> 24 hours), a high level of stress, diagnostic type procedures, good ergonomic interface, and good quality of work process.

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

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

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

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

Hardware Failure Probabilities

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

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

4.3.4.4 Basic Event Probabilities

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

4.3.5 Top Event OFD – Operator Recovery Using Off-site Sources

4.3.5.1 Event Description and Timing

Given the failure of recovery actions using on-site sources, this event accounts for recovery of coolant make-up using off-site 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 on-site sources will not be successful, and that off-site 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 make-up and inventory cooling.
- Procedures and training are in place that ensure that off-site 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 states that if the water level drops below a certain level (e.g., 15 ft below normal level) operator must initiate recovery using off-site sources.
- Operators have received formal training in the procedures.
- Off-site 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 off-site sources, given that even though there has been ample time up to this point to attempt recovery of both the SFP cooling system and both firewater pumps it has not been successful. This top event should include failures of both the diagnosis of the need to provide inventory from off-site sources, and the action itself. The availability of off-site 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 Off-site Power event tree.

As in the case of the loss of cooling, and fire initiating events, based on the assumptions made, the frequency of core uncovery can be seen to be very low. Again, a careful and thorough adherence to NEI commitments 2, 5, 8 and 10, the assumption that 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 make-up 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 make-up sources, are crucial to establishing the low frequency.

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

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

4.4 Severe Weather Loss of Off-site Power Event Tree

This event tree represents the loss of SFP cooling resulting from a loss of off-site power from severe-weather-related events. Until off-site 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 Off-site Power (LOSP) event tree sequence progression.

4.4.1 Initiating Event LP2 – Severe Weather Loss of Off-site Power

4.4.1.1 Event Description

Initiating event IE-LP2 represents severe-weather-related losses of off-site power. Severe weather threatens the safe operation of a SFP facility by simultaneously causing loss of off-site 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 off-site power recovery is reduced.

4.4.1.2 Quantification

The LOSP frequency from severe weather events is 1.1E-2/yr, taken from NUREG/CR-5496 (Ref. 16). This includes contributions from hurricanes, snow and wind, ice, wind and salt, wind, and one tornado event, and occurred at a relatively small number of plants. Because of their potential for severe localized damage, tornados were analyzed separately in Appendix 2e.

4.4.2 Top Event OPR – Off-site 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 off-site 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

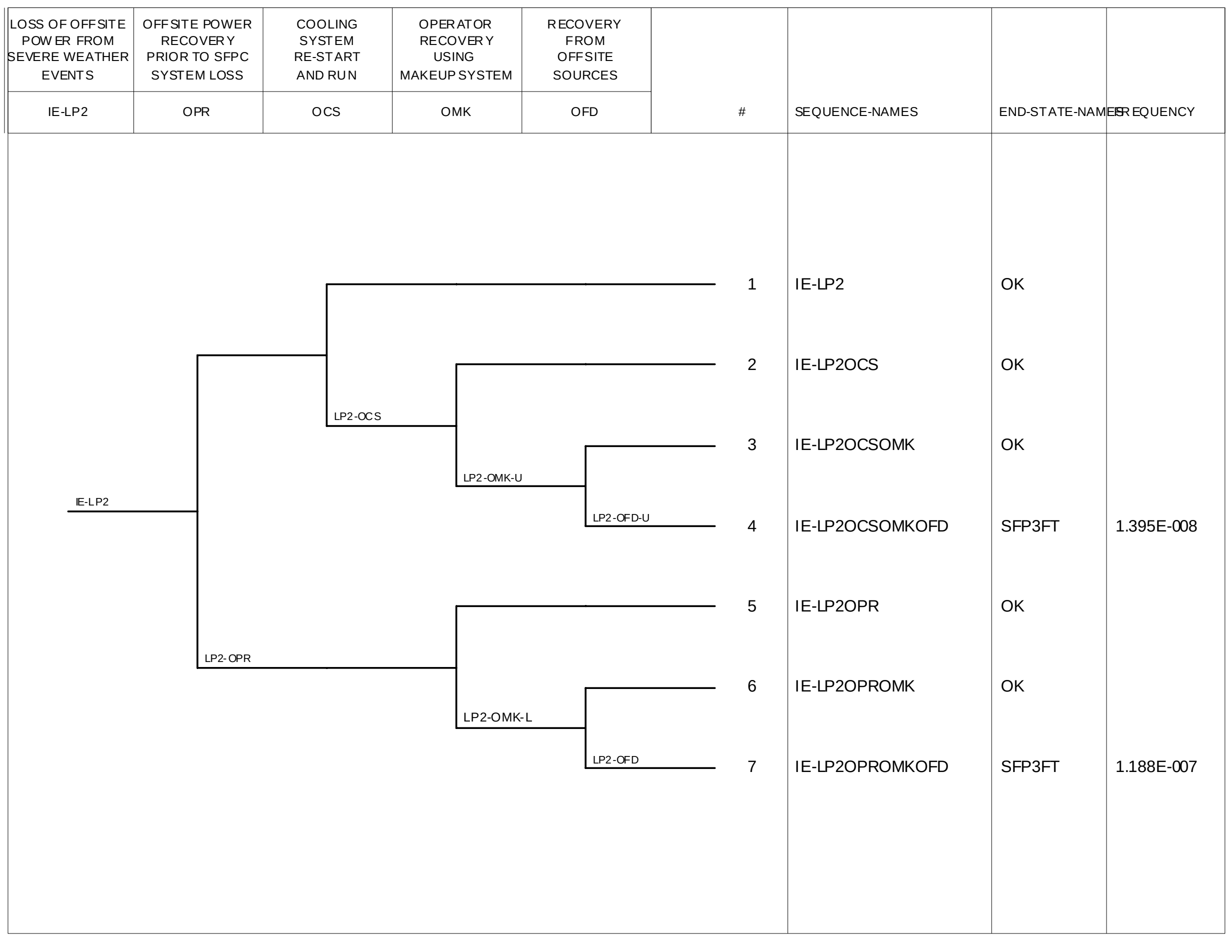
- See section 4.4.2.3 below.

4.4.2.3 Quantification

<u>Non-HEP Probability</u>

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

Figure 4.4 Severe weather related loss of off-site power event tree



4.4.2.4 Basic Event Probability

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

4.4.3 Top Event OCS – Cooling System Restart and Run

4.4.3.1 Event Description and Timing

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

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

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 off-site power. Alternatively, if power is restored, the operator will have alarms available as well. Therefore this value consists of two errors. The diagnosis error was calculated using SPAR, and the walkdown error was calculated using THERP. The relevant performance shaping factors included greater than 24 hours for diagnosis, extreme stress, moderately complex task (due to potential complications from severe weather), diagnostic procedures, and good work processes. A low dependence was applied to the 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 due to severe weather, moderately complex task due to potential valve lineups and severe weather, poor ergonomics due to severe weather, and good work process.

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

Non-HEP Probabilities

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

4.4.3.4 Basic Event Probabilities

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

4.4.4 Top Event OMK – Operator Recovery Using Make-up Systems

4.4.4.1 Event Description and Timing

This top event represents the failure probability of the firewater pumps. If off-site 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 off-site power is not recovered then fault tree LP2-OMK-L represents this top event. In this case, the operator has only the diesel firewater pump available.

4.4.4.2 Relevant Assumptions

- It is assumed that the procedures guide the operators to wait until it is clear that spent fuel pool cooling cannot be reestablished (e.g., using cues such as the level drops to below the suction of the cooling system or the pool begins boiling) before using alternate make-up 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 make-up source to the spent fuel pool without entry to the refuel floor, so that make-up can be provided even when the environment is uninhabitable due to steam and/or high radiation (NEI commitment no.8).

- Repair crew is different than on-site operators.

- Repair crew will focus his recovery efforts on only one pump

- On average, it takes 10 hours to repair a pump if it fails to start and run.

- It would take two days (48 hours) to contact maintenance personnel, make a diagnosis, and get new parts due to severe weather.

- Both firewater pumps are located in a separate structure or protected from the potential harsh environment in case of pool bulk boiling.

- Maintenance is performed per schedule on diesel and electric firewater pumps to maintain operable status.

- Operators haves received formal training on relevant procedures.

4.4.4.3 Quantification

<u>Human Error Probabilities</u>

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

HEP-FW-START-SW represents failure to start either the electric or diesel firewater pump (depending upon availability) within 88 hours after the onset of bulk boiling, given that the decision to start a firewater pump was made. No difficult valve alignment is required, but the operator may have to position a hose in the pool area. This event was quantified using the

SPAR HRA technique. The PSFs chosen were; expansive time (> 50 times the required time), high stress, highly complex task because of the multiple steps and severe weather and its non-routine nature, quality procedures, poor ergonomics due to severe weather, and finally a crew who had executed these tasks before, conversant with the procedures and one another.

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

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

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

Non-HEP Probabilities

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

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

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

The dependency between make-up water supply (e.g., fragility of the fire water supply tank) to events that may have caused the loss of off-site power (such as for 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	2.5E-2
FP-2PUMPS-FTF	6.7E-4
FP-DGPUMP-FTF	1.8E-1
FP-DGPUMP-SW	5.0E-1

4.4.5 Top Event OFD – Operator Recovery Using Off-site Sources

4.4.5.1 Event Description and Timing

Given the failure of recovery actions using on-site sources, this event accounts for recovery of coolant make-up using off-site 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 on-site sources will not be successful, and that off-site sources are the only viable alternatives. Fault tree LP2-OFD represents this top event for the lower branch (off-site power not recovered), and LP2-OFD-U for the upper branch. These fault trees contains those basic events from the fault trees LP2-OMK-U and LP2-OMK-L that relate to recognition of the need to initiate the firewater system; if OMK fails because the operator failed to recognize the need for firewater make-up, 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 make-up and inventory cooling.
- Procedures and training are in place that ensure that off-site 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 off-site sources.
- Off-site resources are familiar with the facility.

4.4.5.3 Quantification

<u>Human Error Probability</u>

The event HEP-INV-OFFSITE represents failure to take the extreme measure of using off-site 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 off-site 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-OFFSITE	8.0E-2

4.4.6 Summary

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

As in the case of the loss of off-site power from plant centered and grid related events, based on the assumptions made, the frequency of core uncovery can be seen to be very low. Again, a careful and thorough adherence to NEI commitments 2, 5, 8 and 10, the assumption that 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 make-up 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 make-up sources, are crucial to establishing the low frequency. NEI commitment 3, related to establishing communication between on site and off site organizations during severe weather, is also important, though its importance is somewhat obscured by the assumption of dependence between the events OMK and OFD. However, if no such provision were made, the availability of off-site resources could become more limiting.

Table 4.4 Basic Event Summary for Severe Weather Loss of Off-site Power

Basic Event Name	**Description**	**Basic Event Probability**
IE-LP2	LOSP event due to severe-weather-related causes	1.1E-02
HEP-DIAG-SFPLP2	Operators fail to diagnose loss of SFP cooling due to loss of off-site power	2.0E-5
HEP-RECG-DEPEN	Failure to recognize need to cool pool given prior failure	5.0E-2
HEP-SFP-STR-LP2	Operators fail to restart and align the SFP cooling system once power is recovered	5.0E-4
HEP-RECG-FWST-SW	Operators fail to diagnose need to start the firewater system	1.0E-4
HEP-FW-START-SW	Operators fail to start firewater pump and provide alignment	1.0E-3
HEP-FW-REP-DEPSW	Repair crew fails to repair firewater system	7.0E-2
HEP-FW-REP-NODSW	Repair crew fails to repair firewater system	2.5E-2
HEP-INV-OFFST-SW	Operators fail to provide alternate sources of cooling from off-site	8.0E-2
REC-OSP-SW	Recovery of off-site 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

Basic Event Name	Description	Basic Event 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
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. 12), show that the frequency of loss of inventory events in which the level decreased more than one foot can be estimated to be less than one event per 100 reactor years. Most of these events were the result of operator error and were recoverable. NUREG-1275 shows that, except for one event that lasted for 72 hours, there were no events that lasted more than 24 hours. Eight events resulted in a level decrease of between one and five feet and another two events resulted in an inventory loss of between five and 10 feet.

4.5.1.2 Relevant Assumption

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

4.5.1.3 Quantification

The data reviewed during the development of NUREG-1275 (Ref. 12) indicated fewer than one event per 100 years in which level decreased over one foot. This would give a frequency of

1E-02. However, it is assumed that the NEI commitments 6 and 7 when implemented will reduce this frequency by an order of magnitude or more. Thus the frequency is estimated as 1E-03 per year.

4.5.2 Top Event NLL – Loss Exceeds Normal Make-up 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 make-up 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 make-up 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.

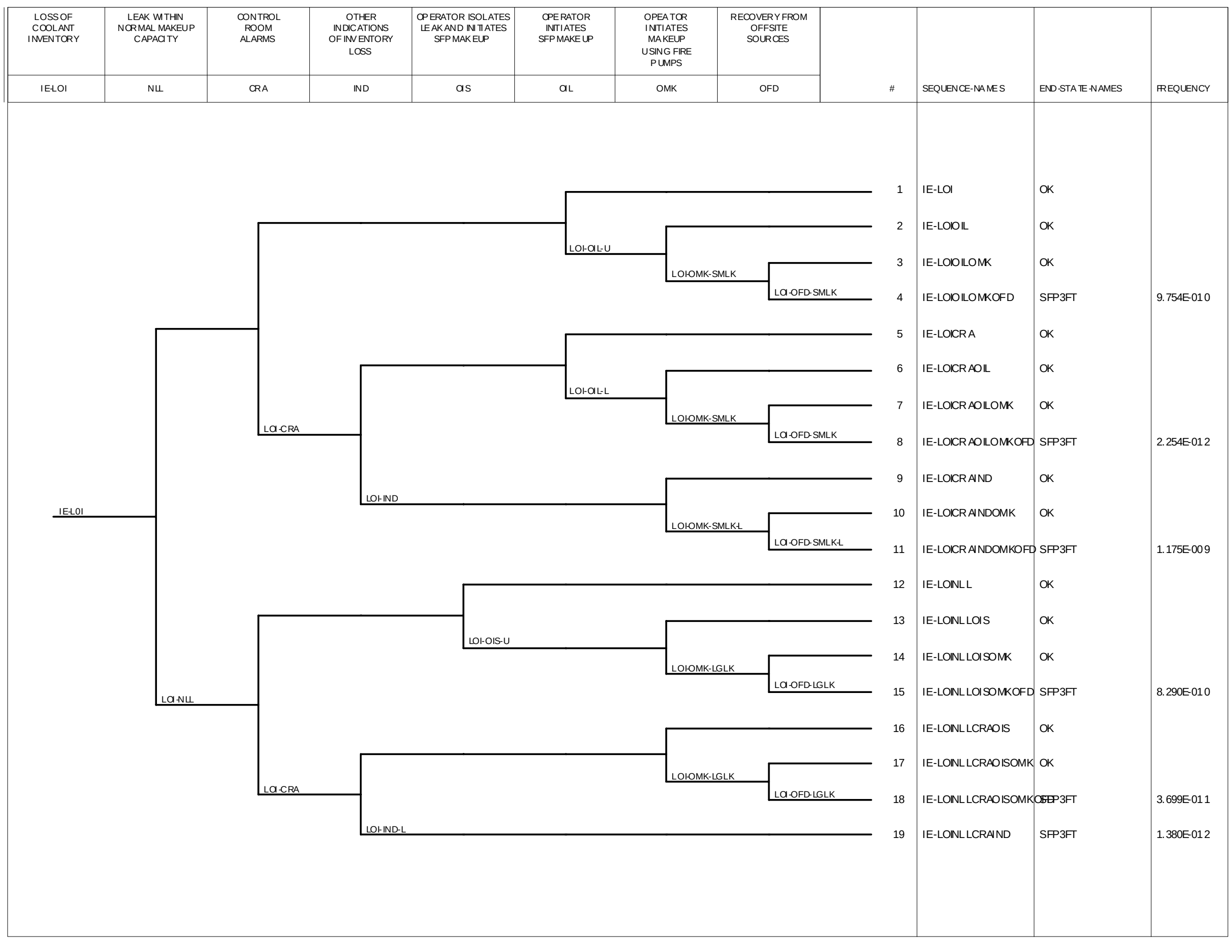
- The small leak is assumed for analysis purposes to be at the limit of the make-up system capacity, i.e., 30 gpm.

4.5.2.3 Quantification

<u>Non-HEP Probabilities</u>

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 ≈ 0.06 (6%). For the other 94% of the cases, operation of the make-up pump is sufficient to prevent fuel uncovery.

Figure 4.5 Loss of inventory event tree



4.5.3 Top Event CRA – Control Room Alarms

4.5.3.1 Event description and Timing

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

4.5.3.2 Relevant Assumptions

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

4.5.3.3 Quantification

<u>Human Error Probabilities</u>

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.

<u>Non-HEP Probabilities</u>

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

4.5.3.4 Basic Event Probabilities

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

4.5.4 Top Event IND – Other Indications of Inventory Loss

4.5.4.1 Event Description and Timing

This top event models operator failure to recognize the loss of inventory during 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 hrs before the water level would drop below the SFP cooling suction level. Therefore, for this event, only one shift is credited for recognition.

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

The critical action 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 make-up 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).

- Spent fuel pool operations that have the potential to rapidly drain the pool will be under strict administrative controls (NEI commitment no. 9). This increases the likelihood of the operators successfully terminating a leak should one occur.

4.5.5.3 Quantification

<u>Human Error Probabilities</u>

Two human failure events are included in the functional fault tree LOI-OIS, one for failure to start the SFP make-up pump, HEP-MKUP-START, and one for failure to successfully isolate the leak, HEP-LEAK-ISO.

SPAR HRA worksheets were used to quantify each of these errors. For HEP-MKUP-START, it was assumed that the operator 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 due to the fact that it is necessary to identify the source of the leak and it may be difficult to isolate, the operators are highly trained, have all the equipment available, and all components are well labeled and correspond to a procedure, and the crew has effective interactions in a quality facility.

Hardware Failure Probabilities

Unavailability of a SFP make-up system, SFP-REGMKUP-F, was assigned a value of 5.0E-2 from INEL-96/0334. It is assumed that the SFP make-up 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	2.5E-4
SFP-REGMKUP-F	5.0E-2

4.5.6 Top Event OIL – Operator Initiates SFP Make-up System

4.5.6.1 Event Description and Timing

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

4.5.6.2 Relevant Assumptions

- There are procedures to guide the operators in how to deal with loss of inventory, and the operators are trained in their use (NEI commitment no. 2).
- The manipulations required to start the make-up system can be achieved in less than 10 minutes.

4.5.6.3 Quantification

Human Error Probabilities

In the case of an early response, 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-L (see fault tree L OI-OIL-L). Success is defined as the operator starting the make-up pump and performing valve manipulation as needed.

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

<u>Hardware Failure Probabilities</u>

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

4.5.6.4 Basic Event Proabablities

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

4.5.7 Top Event OMK – Operator Initiates 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 firewater system. The case of a small leak is represented by two functional fault trees, LOI-OMK-SMLK, and LOI-OMK-SMLK-L. The difference between the two trees is that in the first, the operators are aware of the problem and are attempting to solve it, whereas in the second, the operators will need to first recognize the problem. In both small leak cases, the operator has more than 65 hrs to start firewater system. In all cases both the firewater pumps would be available.

4.5.7.2 Relevant Assumptions

- The operators have 40 to 65 hours to start a firewater pump depending on the leak size.

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

- Repair crew is different than on-site operators.

- On average, it takes 10 hours to repair a pump if it fails to start and run.
- It takes 16 hours to contact maintenance personnel, make a diagnosis, and get new parts.

- Both firewater pumps are located in a separate structure and are protected from the potential harsh environment in the case of pool bulk boiling.

- Maintenance and testing are performed on diesel and electric firewater pumps to maintain operable status (NEI commitment no. 10).

- There are procedures to guide the operators in how to deal with loss of inventory, and the operators are trained in their use. The guidance on when to begin addition of water from alternate sources is clear and related to a clearly identified condition, such as pool level or onset of boiling (NEI commitment no. 2).

4.5.7.3 Quantification

Human Error Probabilities

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

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

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

SPAR HRA worksheets were also used to quantify the HEPs.

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

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

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

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

Basic event HEP-FW-REP-NODS (see fault tree, OIL-OMK-SMLKL) represents the failure of the repair crew to repair a firewater pump for the small leak scenarios. Note that repairing the SFP regular make-up 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 50 hours (65 hours less 16 hours) to repair the pump. Therefore, assuming a 10-hour mean time to repair, the probability of failure to repair the pump would be Exp (-(1/10) * 49) = 7.5E-3 in the case of a small break scenario.

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

<u>Hardware Failure Probabilities</u>

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

4.5.7.4 Basic Event Probabilities

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

4.5.8 Top Event OFD – Recovery From Off-site Sources

4.5.8.1 Event Description and Timing

Given the failure of recovery actions using on-site sources, this event accounts for recovery of coolant make-up using off-site 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 make-up 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 off-site sources.

- Operator has received formal training and there are procedures to guide him.

- Off-site 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 make-up 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 off-site personnel, highly trained staff with good procedures, good ergonomics (equipment is available to make off-site support straightforward) and good work processes. For both cases, a low level of dependence was assumed on the failure of prior tasks.

4.5.8.4 Basic Event Probabilities

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

4.5.9 Summary

Table 4.5 presents a summary of basic events.

As in the previous cases, the frequency of core uncovery can be seen to be very low. Again, a careful and thorough adherence to NEI commitments 2, 4, 5, 8 and 10, the assumption that 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 make-up 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 make-up 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 off-site sources	5.0E-2
HEP-INV-OFFSITE	Operators fail to provide alternate sources of cooling from off-site	5.0E-2
FP-2PUMPS-FTF	Failure of firewater pump system	6.7E-4
LOI-LGLK	Loss exceeds normal make-up normal	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-LOF	Failure of control room alarm channel	1.0E-5
SPC-LVL-LOP	Electrical faults leading to alarm channel failure	2.0E-3

5.0 Summary of Results

The results of this analysis provide insight into the risks associated with storage of spent nuclear fuel in fuel pools at decommissioned nuclear power plants. The five accident initiators that were analyzed consist of: 1) internal fires, 2) Loss of cooling, 3) loss of inventory, 4) plant/grid centered losses of off-site power, and 5) severe weather induced losses of off-site power. The total frequency for the endstate is estimated to be 2.3E-7/year. Table 5.1 summarizes the fuel uncovery frequency for each initiator. The frequencies are point estimates, based on the use of point estimates for the input parameters. For the most part these input parameter values would be used as the mean values of the probability distributions that would be used in a calculation to propagate parameter uncertainty. Because the systems are essentially single train systems, the point estimates closely correlate to the mean values that would be obtained from a full propagation of parameter uncertainty.

The numerical results are a function of the assumptions made and in particular, the model used to evaluate the human error probabilities. The results represent a reasonable assessment of the levels of risk that 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 differ by an order of magnitude. However, given the time scales involved, and the simplicity of the systems, we believe that the conclusions of this study, namely that the risks are low, and that the NEI commitments play an important role in determining that low level, 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. The assumption that the preparation time of several days is adequate to bring off-site 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.

The analysis show that, based on the assumptions made, the frequency of fuel uncovery from the loss of cooling, loss of inventory, loss of off-site 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 make-up 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 make-up 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 on site and off site organizations during severe weather, is also important, though its importance is somewhat

obscured in the analysis by the assumption that there is some degree of dependence between the decision to implement supplemental make-up to the spent fuel pool from on-site sources such as fire water pumps, and that from off-site sources. However, if no such provision were made, the availability of off-site 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	**Core Uncovery Frequency (per/yr)**
Internal Fires	4.5E-08
Loss of Cooling	1.4E-08
Loss of Inventory	3.1E-09
Loss of Off-site Power (plant centered & grid-related events)	3.0E-8
Loss of Off-site Power (severe weather events)	1.3E-7
TOTAL =	2.3E-007

6.0 REFERENCES

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

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

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

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

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

6. Hollnagel, E., "Cognitive Reliability and Error Analysis Method - CREAM" Elsevier, 1998.

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

8. Swain, A. D., and Guttman, H. E., "Handbook of Human Reliability Analysis with Emphasis on Nuclear Power Plant Applications", (THERP), NUREG/CR-1278, August 1983, USNRC.

9. Byers, J.C., et al., “Revision of the 1994 ASP HRA Methodology (Draft)”, INEEL/EXT-99-00041, January 1999.

10. Sailor, et. al., "Severe Accidents in Spent Fuel Pools in Support of Generic Issue 82", NUREG/CR-4982 (BNL-NUREG-52093), July 1987.

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

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

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

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

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

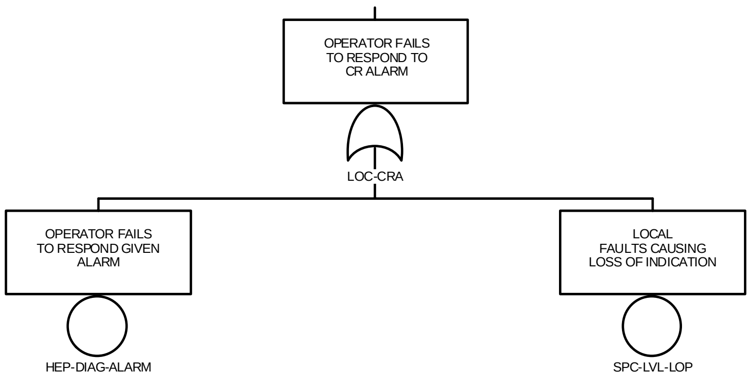
16. Atwood, et. al., "Evaluation of Loss of Offsite Power Events at Nuclear Power Plants: 1980 - 1996," NUREG/CR-5486, November 1998.

17. U.S. Nuclear Regulatory Commission, “Evaluation of Station Blackout Accidents at Nuclear Power Plants,” NUREG-1032, June 1988.

18. U.S. Nuclear Regulatory Commission, “Single-Failure-Proof Cranes for Nuclear Power Plants,” USNRC Report NUREG-0554, May 1979.

ATTACHMENT A

FAULT TREES USED IN THE RISK ANALYSIS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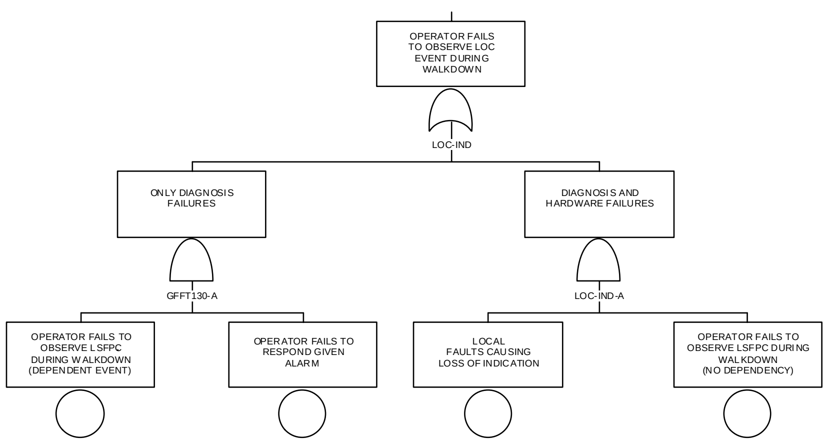
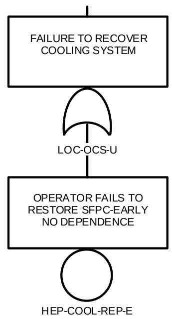
TO RESPOND TO
CR ALARM
LOC-CRA
OPERATOR FAILS
TO RESPOND GIVEN
ALARM
HEP-DIAG-ALARM
LOCAL
FAULTS CAUSING
LOSS OF INDICATION
SPC-LVL-LOP


OPERATOR FAILS
TO OBSERVE LOC
EVENT DURING
WALKDOWN
LOC-IND
ONLY DIAGNOSIS
FAILURES
GFFT130-A
DIAGNOSIS AND
HARDWARE FAILURES
LOC-IND-A
OPERATOR FAILS TO
OBSERVE LSFPC
DURING WALKDOWN
(DEPENDENT EVENT)
HEP-WLKDWN-DEPEN
OPERATOR FAILS TO
RESPOND GIVEN
ALARM
HEP-DIAG-ALARM
LOCAL
FAULTS CAUSING
LOSS OF INDICATION
SPC-LVL-LOP
OPERATOR FAILS TO
OBSERVE LSFPC DURING
WALKDOWN
(NO DEPENDENCY)
HEP-WLKDWN-LSFPC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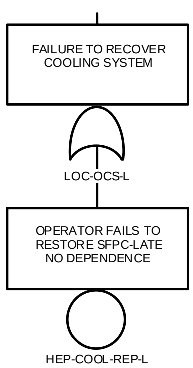
COOLING SYSTEM
LOC-OCS-U
OPERATOR FAILS TO
RESTORE SFPC-EARLY
NO DEPENDENCE
HEP-COOL-REP-E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LOC-OCS-L
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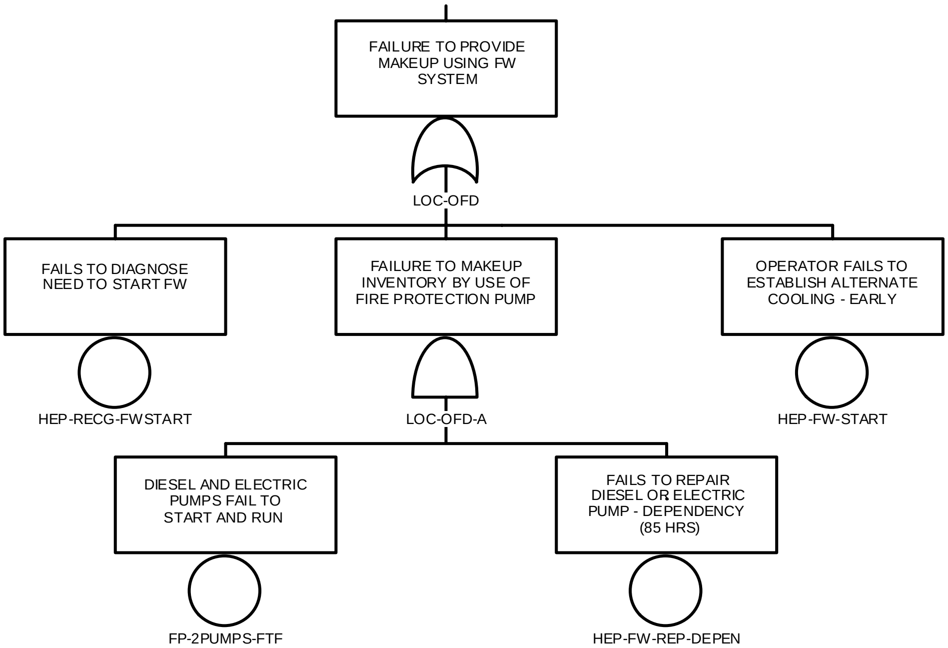
HEP-COOL-REP-L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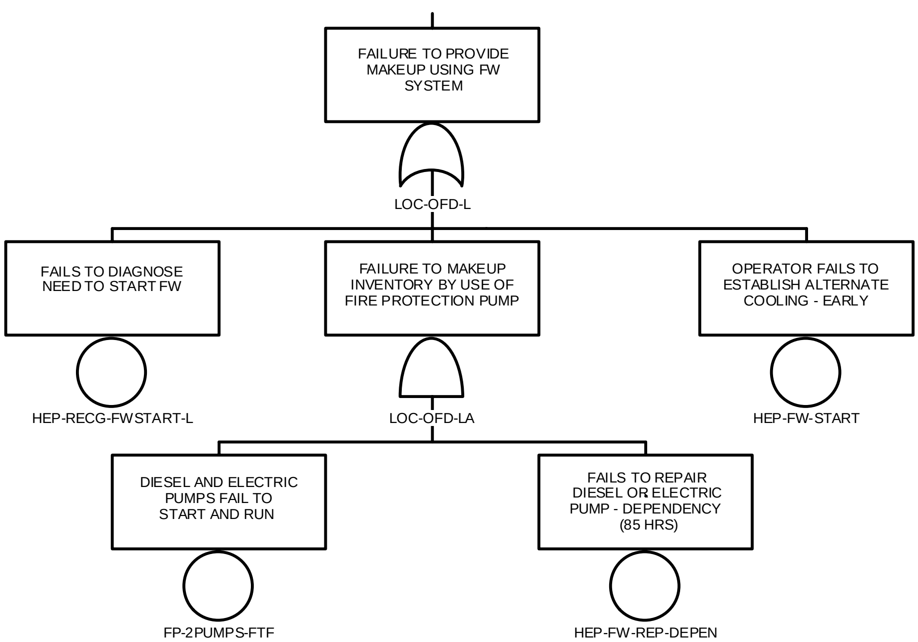
LOC-OFD
FAILS TO DIAGNOSE NEED TO START FW
FAILURE TO MAKEUP INVENTORY BY USE OF FIRE PROTECTION PUMP
OPERATOR FAILS TO ESTABLISH ALTERNATE COOLING - EARLY
HEP-RECG-FWSTART
LOC-OFD-A
HEP-FW-START
DIESEL AND ELECTRIC PUMPS FAIL TO START AND RUN
FAILS TO REPAIR DIESEL OR ELECTRIC PUMP - DEPENDENCY (85 HRS)
FP-2PUMPS-FTF
HEP-FW-REP-DEPEN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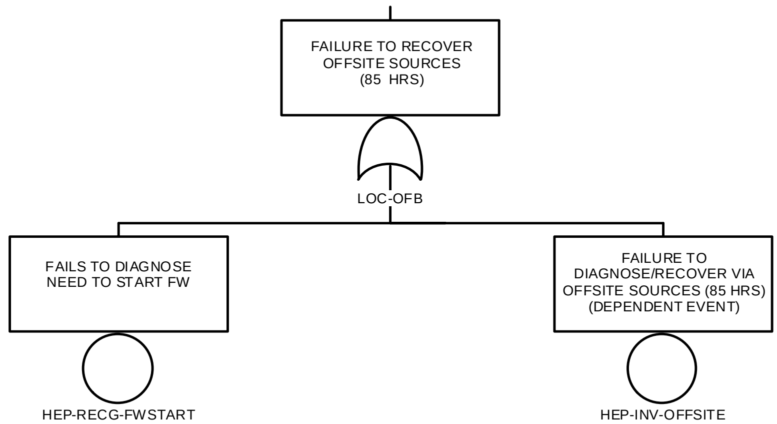
LOC-OFD-L
FAILS TO DIAGNOSE NEED TO START FW
FAILURE TO MAKEUP INVENTORY BY USE OF FIRE PROTECTION PUMP
OPERATOR FAILS TO ESTABLISH ALTERNATE COOLING - EARLY
HEP-RECG-FWSTART-L
LOC-OFD-LA
HEP-FW-START
DIESEL AND ELECTRIC PUMPS FAIL TO START AND RUN
FAILS TO REPAIR DIESEL OR ELECTRIC PUMP - DEPENDENCY (85 HRS)
FP-2PUMPS-FTF
HEP-FW-REP-DEPEN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OFFSITE SOURCES
(85 HRS)
LOC-OFB
FAILS TO DIAGNOSE
NEED TO START FW
FAILURE TO
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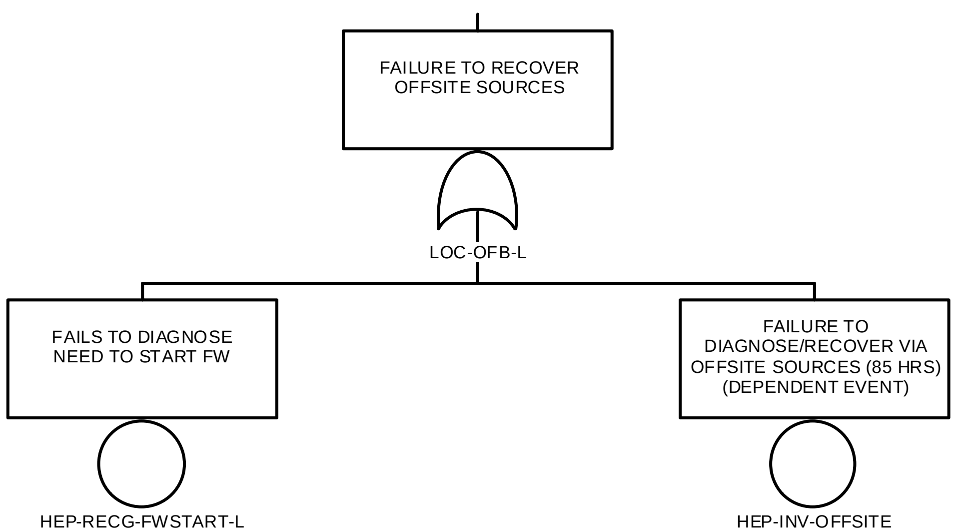
OFFSITE SOURCES (85 HRS)
(DEPENDENT EVENT)
HEP-RECG-FWSTART
HEP-INV-OFFSITE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OFFSITE SOURCES
LOC-OFB-L
FAILS TO DIAGNOSE
NEED TO START FW
FAILURE TO
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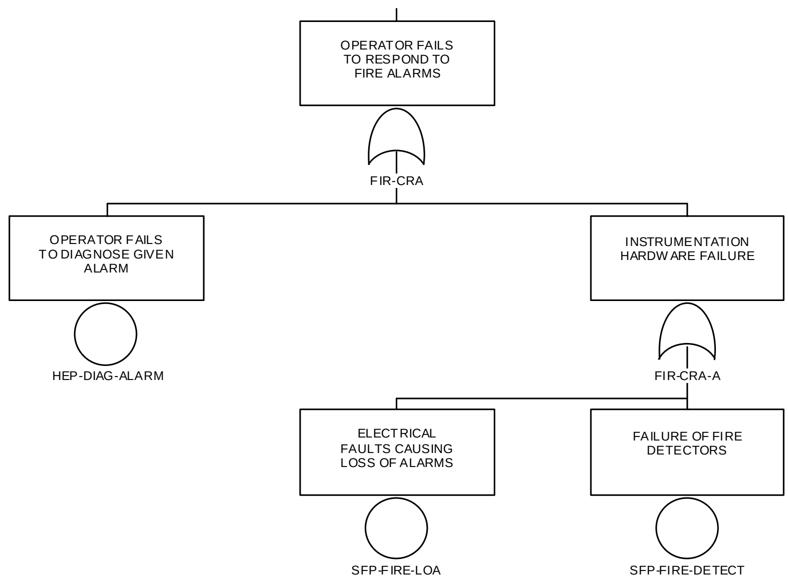
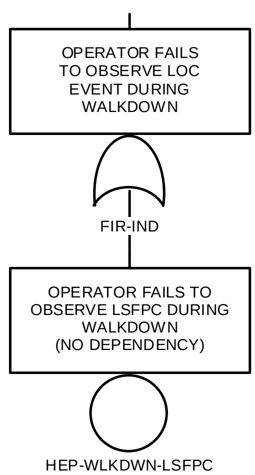
OFFSITE SOURCES (85 HRS)
(DEPENDENT EVENT)
HEP-RECG-FWSTART-L
HEP-INV-OFFSITE


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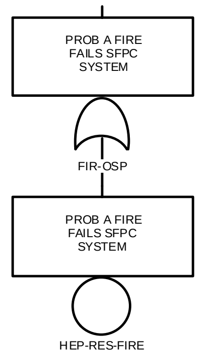
HEP-WLKDWN-LSFPC


PROB A FIRE
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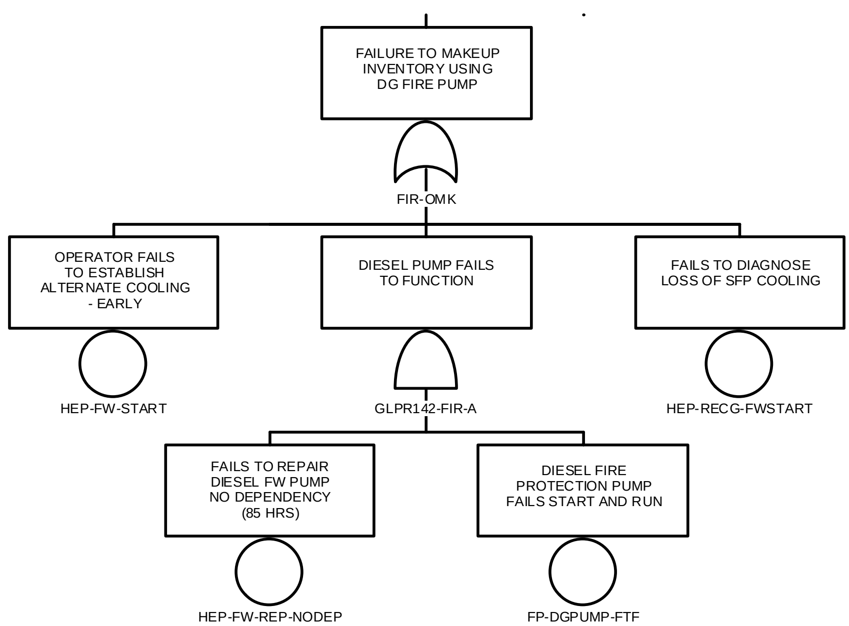
SYSTEM
FIR-OSP
PROB A FIRE
FAILS SFPC
SYSTEM
HEP-RES-FIRE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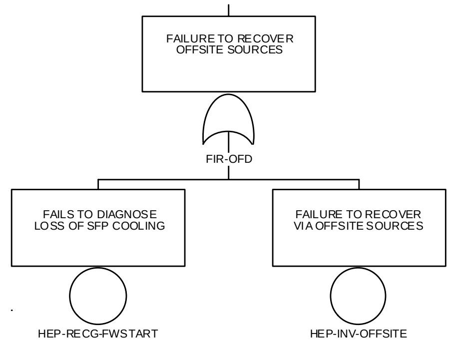
FIR-OMK
OPERATOR FAILS
TO ESTABLISH
ALTERNATE COOLING
- EARLY
DIESEL PUMP FAILS
TO FUNCTION
FAILS TO DIAGNOSE
LOSS OF SFP COOLING
HEP-FW-START
GLPR142-FIR-A
HEP-RECG-FWSTART
FAILS TO REPAIR
DIESEL FW PUMP
NO DEPENDENCY
(85 HRS)
DIESEL FIRE
PROTECTION PUMP
FAILS START AND RUN
HEP-FW-REP-NODEP
FP-DGPUMP-FTF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FAILURE TO RECOVER VIA OFFSITE SOURCES
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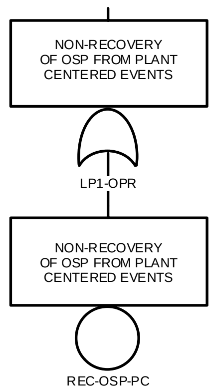
HEP-INV-OFFSITE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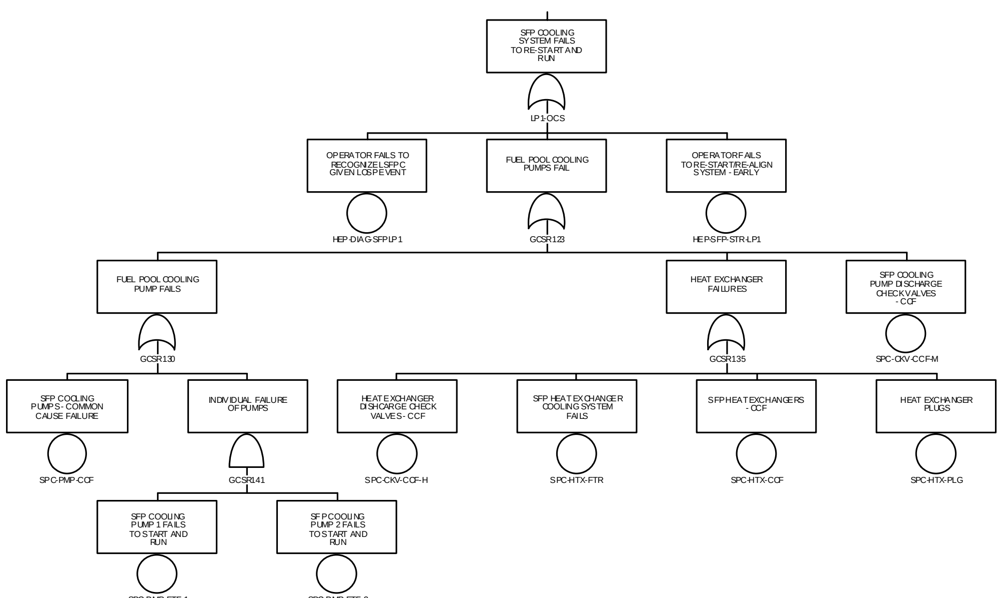
LP1-OPR
NON-RECOVERY OF OSP FROM PLANT CENTERED EVENTS
REC-OSP-PC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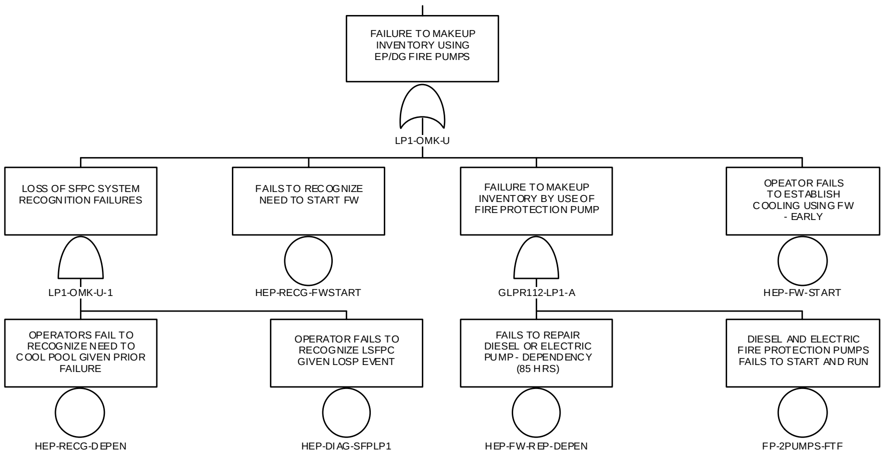
OPERATOR FAILS TO RECOGNIZE LSFPC GIVEN LOSP EVENT
HEP-DIAG-SFP LP1
FUEL POOL COOLING PUMPS FAIL
GCSR123
OPERATOR FAILS TO RE-START/RE-ALIGN SYSTEM - EARLY
HEP-SFP-STR-LP1
FUEL POOL COOLING PUMP FAILS
GCSR130
HEAT EXCHANGER FAILURES
GCSR135
SFP COOLING PUMP DISCHARGE CHECK VALVES - CCF
SPC-CKV-CCF-M
SFP COOLING PUMPS - COMMON CAUSE FAILURE
SPC-PMP-CCF
INDIVIDUAL FAILURE OF PUMPS
GCSR141
HEAT EXCHANGER DISHCARGE CHECK VALVES - CCF
SPC-CKV-CCF-H
SFP HEAT EXCHANGER COOLING SYSTEM FAILS
SPC-HTX-FTR
SFP HEAT EXCHANGERS - CCF
SPC-HTX-CCF
HEAT EXCHANGER PLUGS
SPC-HTX-PLG
SFP COOLING PUMP 1 FAILS TO START AND RUN
SPC-PMP-FTF-1
SFP COOLING PUMP 2 FAILS TO START AND RUN
SPC-PMP-FTF-2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LP1-OMK-U
LOSS OF SFPC SYSTEM RECOGNITION FAILURES
FAILS TO RECOGNIZE NEED TO START FW
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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
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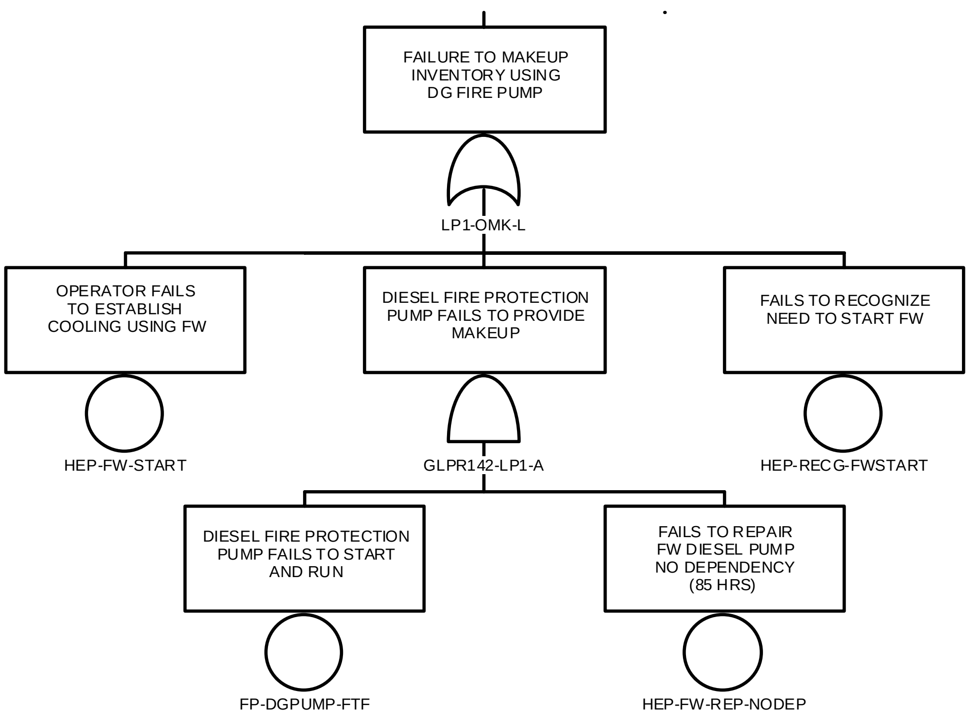
HEP-DIAG-SFPLP1
HEP-FW-REP-DEPEN
FP-2PUMPS-FTF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LP1-OMK-L
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FAILS TO RECOGNIZE NEED TO START FW
HEP-FW-START
GLPR142-LP1-A
HEP-RECG-FWSTART
DIESEL FIRE PROTECTION PUMP FAILS TO START AND RUN
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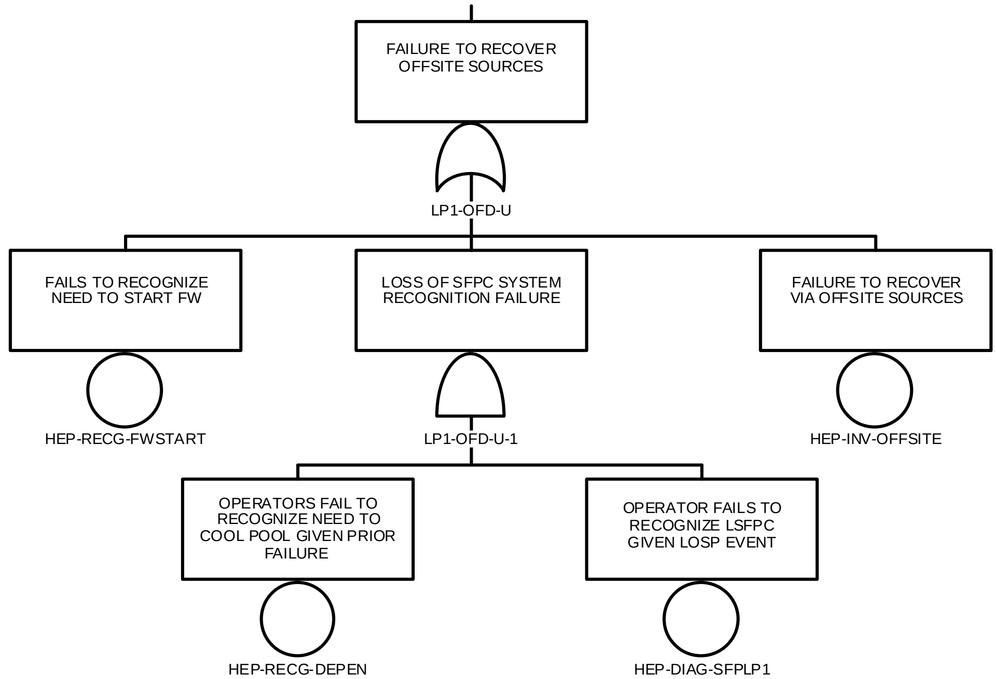
FP-DGPUMP-FTF
HEP-FW-REP-NODEP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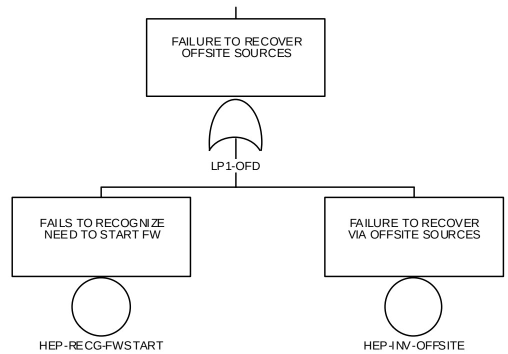
LP1-OFD-U
FAILS TO RECOGNIZE NEED TO START FW
LOSS OF SFPC SYSTEM RECOGNITION FAILURE
FAILURE TO RECOVER VIA OFFSITE SOURCES
HEP-RECG-FWSTART
LP1-OFD-U-1
HEP-INV-OFFSITE
OPERATORS FAIL TO RECOGNIZE NEED TO COOL POOL GIVEN PRIOR FAILURE
OPERATOR FAILS TO RECOGNIZE LSFPC GIVEN LOSP EVENT
HEP-RECG-DEPEN
HEP-DIAG-SFPLP1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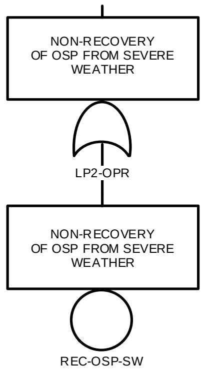
HEP-RECG-FWSTART
HEP-INV-OFFSITE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NON-RECOVERY OF OSP FROM SEVERE WEATHER
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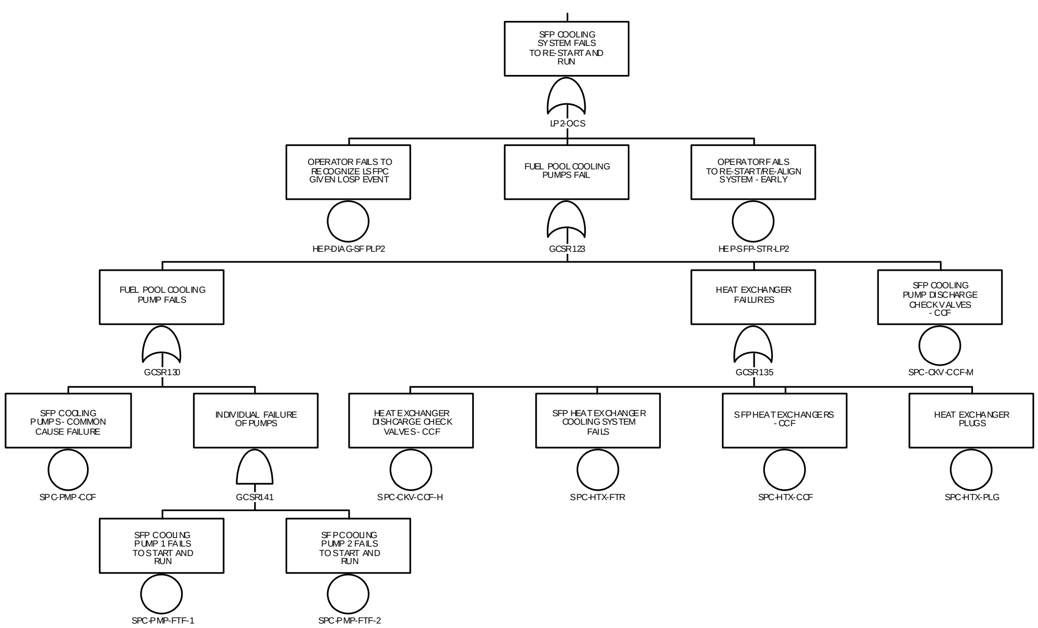
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LP2-OCS
OPERATOR FAILS TO RECOGNIZE LSFPC GIVEN LOSP EVENT
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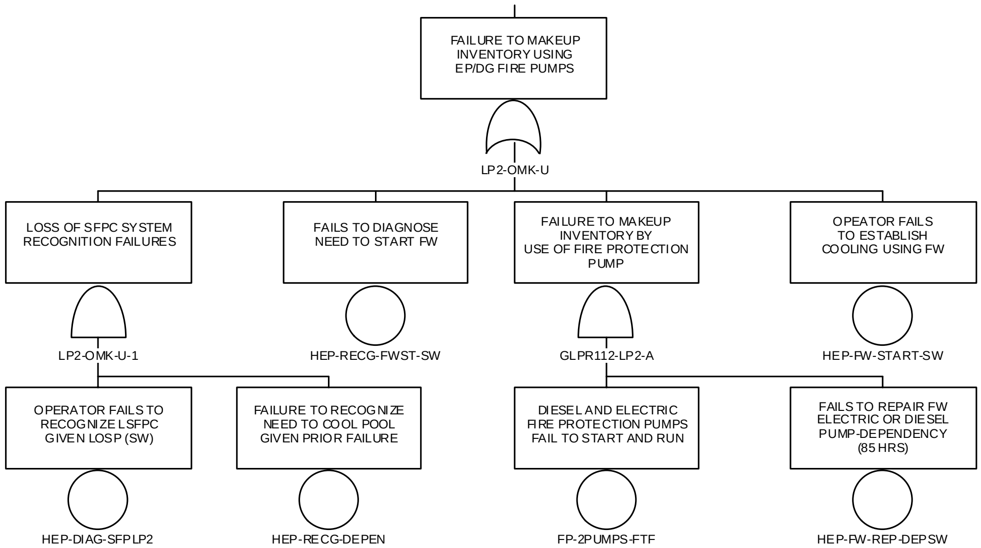
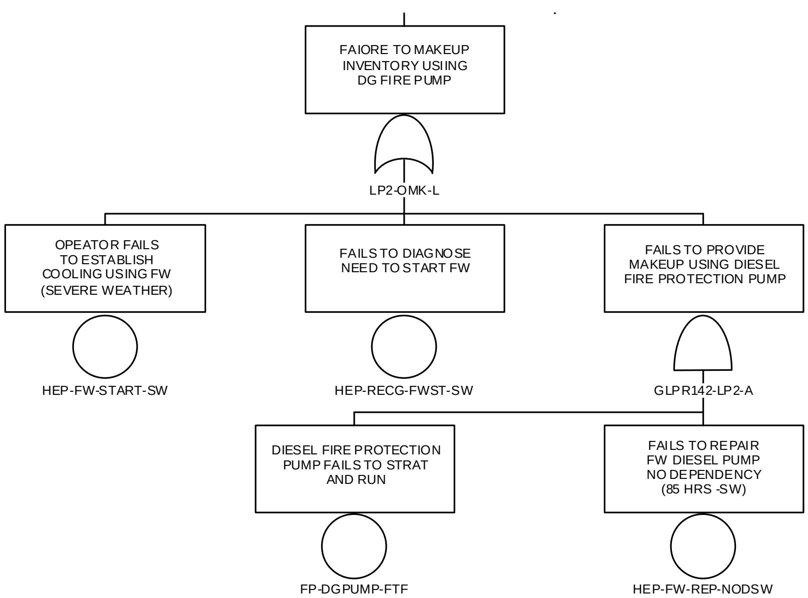
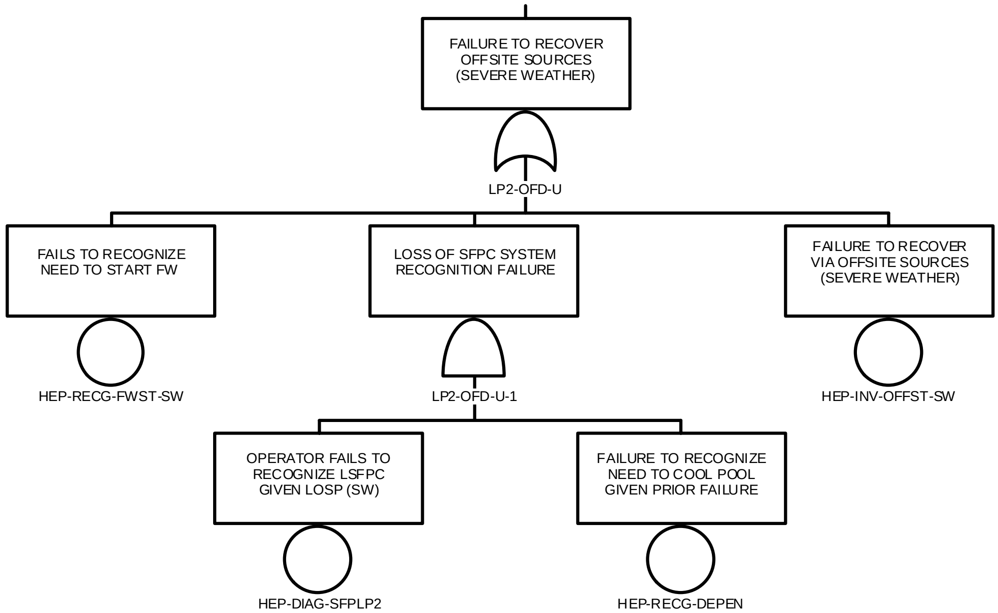
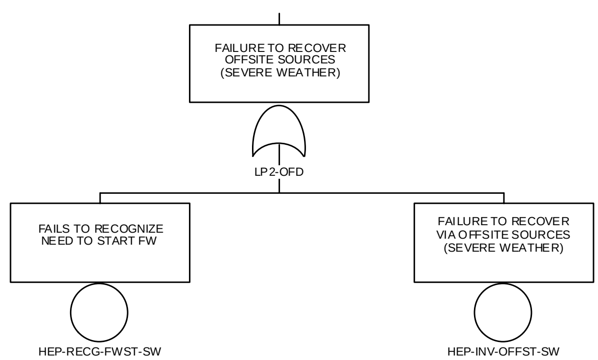
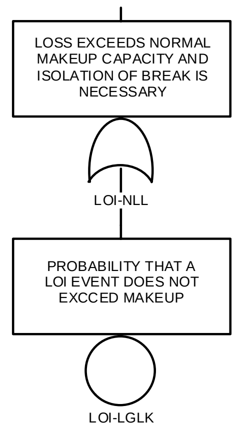
GCSR123
OPERATOR FAILS TO RE-START/RE-ALIGN SYSTEM - EARLY
HEP-SFP-STR-LP2
FUEL POOL COOLING PUMP FAILS
GCSR130
HEAT EXCHANGER FAILURES
GCSR135
SFP COOLING PUMP DISCHARGE CHECK VALVES - CCF
SPC-CKV-CCF-M
SFP COOLING PUMPS - COMMON CAUSE FAILURE
SPC-PMP-CCF
INDIVIDUAL FAILURE OF PUMPS
GCSR141
HEAT EXCHANGER DISHCARGE CHECK VALVES - CCF
SPC-CKV-CCF-H
SFP HEAT EXCHANGER COOLING SYSTEM FAILS
SPC-HTX-FTR
SFP HEAT EXCHANGERS - CCF
SPC-HTX-CCF
HEAT EXCHANGER PLUGS
SPC-HTX-PLG
SFP COOLING PUMP 1 FAILS TO START AND RUN
SPC-PMP-FTF-1
SFP COOLING PUMP 2 FAILS TO START AND RUN
SPC-PMP-FTF-2

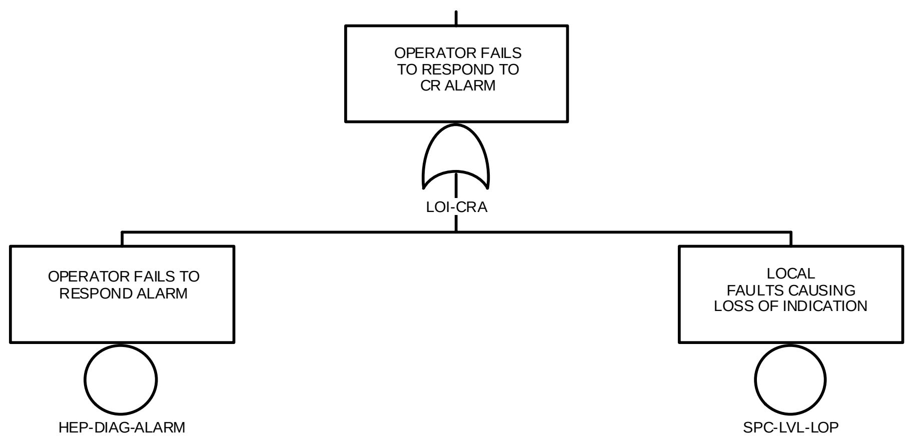

FAILURE TO MAKEUP INVENTORY USING EP/DG FIRE PUMPS
LP2-OMK-U
LOSS OF SFPC SYSTEM RECOGNITION FAILURES
FAILS TO DIAGNOSE NEED TO START FW
FAILURE TO MAKEUP INVENTORY BY USE OF FIRE PROTECTION PUMP
OPEATOR FAILS TO ESTABLISH COOLING USING FW
LP2-OMK-U-1
HEP-RECG-FWST-SW
GLPR112-LP2-A
HEP-FW-START-SW
OPERATOR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

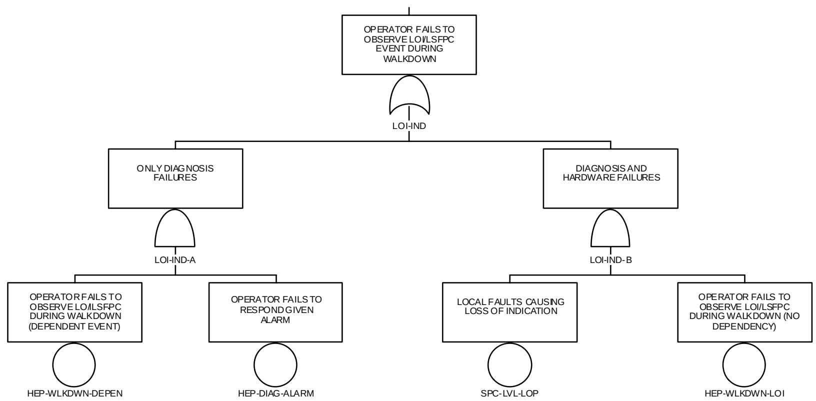

FAIORE TO MAKEUP INVENTORY USIING DG FIRE PUMP
LP2-OMK-L
OPEATOR FAILS TO ESTABLISH COOLING USING FW (SEVERE WEATHER)
FAILS TO DIAGNOSE NEED TO START FW
FAILS TO PROVIDE MAKEUP USING DIESEL FIRE PROTECTION PUMP
HEP-FW-START-SW
HEP-RECG-FWST-SW
GLPR142-LP2-A
DIESEL FIRE PROTECTION PUMP FAILS TO STRAT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

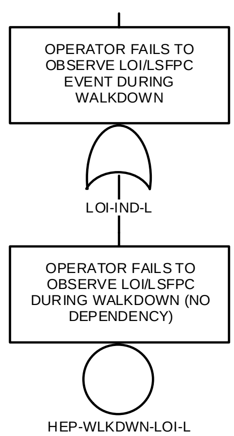

FAILURE TO RECOVER
OFFSITE SOURCES
(SEVERE WEATHER)
LP2-OFD-U
FAILS TO RECOGNIZE
NEED TO START FW
HEP-RECG-FWST-SW
LOSS OF SFPC SYSTEM
RECOGNITION FAILURE
LP2-OFD-U-1
FAILURE TO RECOVER
VIA OFFSITE SOURCES
(SEVERE WEATHER)
HEP-INV-OFFST-SW
OPERATOR FAILS TO
RECOGNIZE LSFPC
GIVEN LOSP (SW)
HEP-DIAG-SFPLP2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

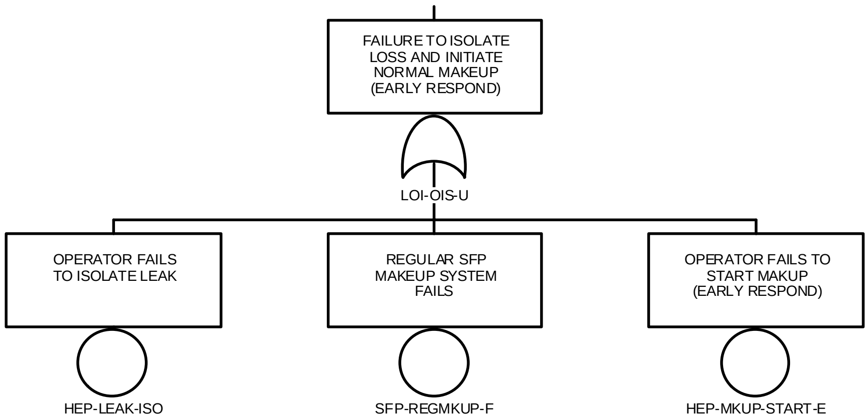

FAILURE TO RECOVER
OFFSITE SOURCES
(SEVERE WEATHER)
LP2-OFD
FAILS TO RECOGNIZE
NEED TO START FW
HEP-RECG-FWST-SW
FAILURE TO RECOVER
VIA OFFSITE SOURCES
(SEVERE WEATHER)
HEP-INV-OFFST-SW

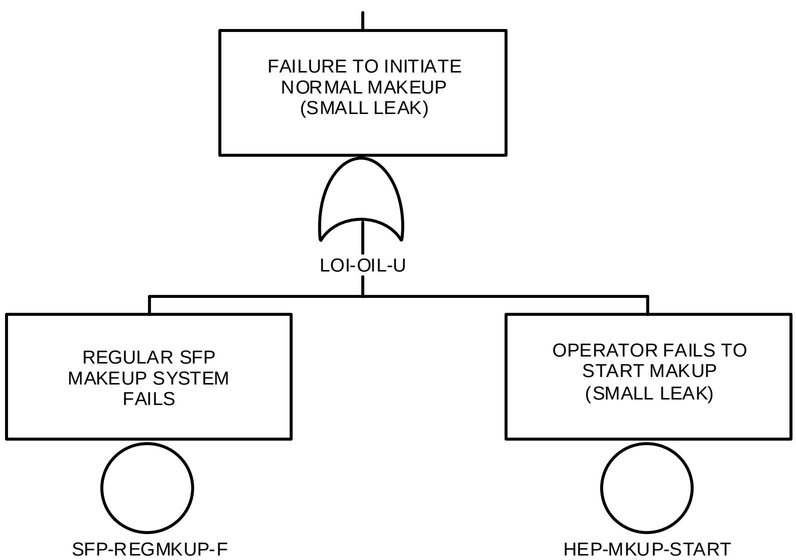

LOSS EXCEEDS NORMAL MAKEUP CAPACITY AND ISOLATION OF BREAK IS NECESSARY
LOI-NLL
PROBABILITY THAT A LOI EVENT DOES NOT EXCCED MAKEUP
LOI-LGLK

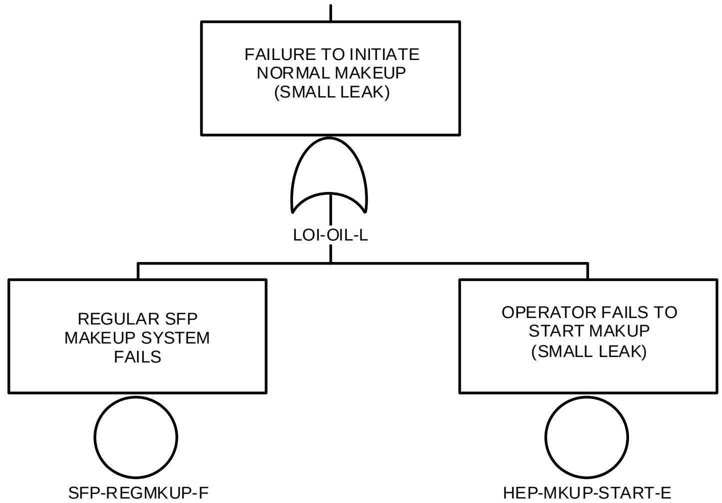

OPERATOR FAILS TO RESPOND TO CR ALARM
LOI-CRA
OPERATOR FAILS TO RESPOND ALARM
HEP-DIAG-ALARM
LOCAL FAULTS CAUSING LOSS OF INDICATION
SPC-LVL-LOP

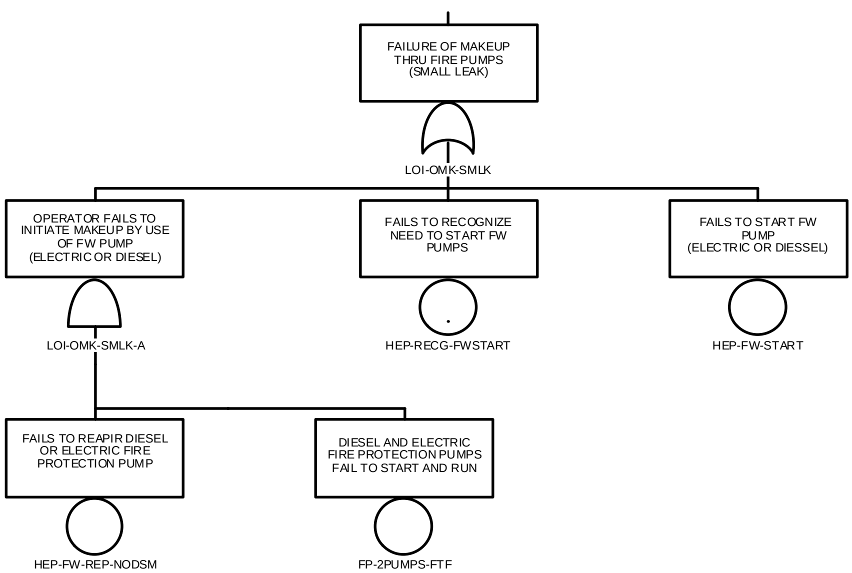

OPERATOR FAILS TO OBSERVE LOI/LSFPC EVENT DURING WALKDOWN
LOI-IND
ONLY DIAGNOSIS FAILURES
DIAGNOSIS AND HARDWARE FAILURES
LOI-IND-A
LOI-IND-B
OPERATOR FAILS TO OBSERVE LOI/LSFPC DURING WALKDOWN (DEPENDENT EVENT)
OPERATOR FAILS TO RESPOND GIVEN ALARM
LOCAL FAULTS CAUSING LOSS OF INDICATION
OPERATOR FAILS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

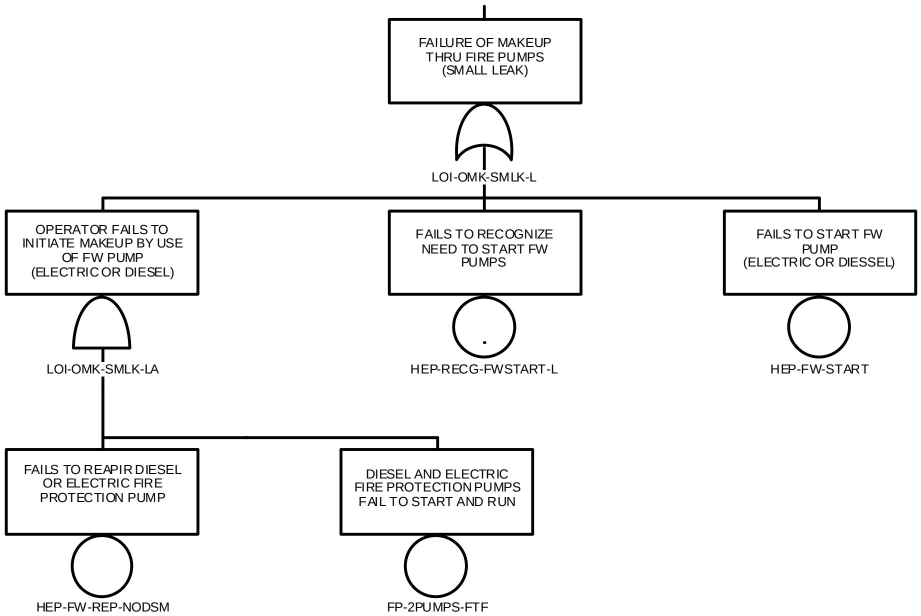

OPERATOR FAILS TO OBSERVE LOI/LSFPC EVENT DURING WALKDOWN
LOI-IND-L
OPERATOR FAILS TO OBSERVE LOI/LSFPC DURING WALKDOWN (NO DEPENDENCY)
HEP-WLKDWN-LOI-L

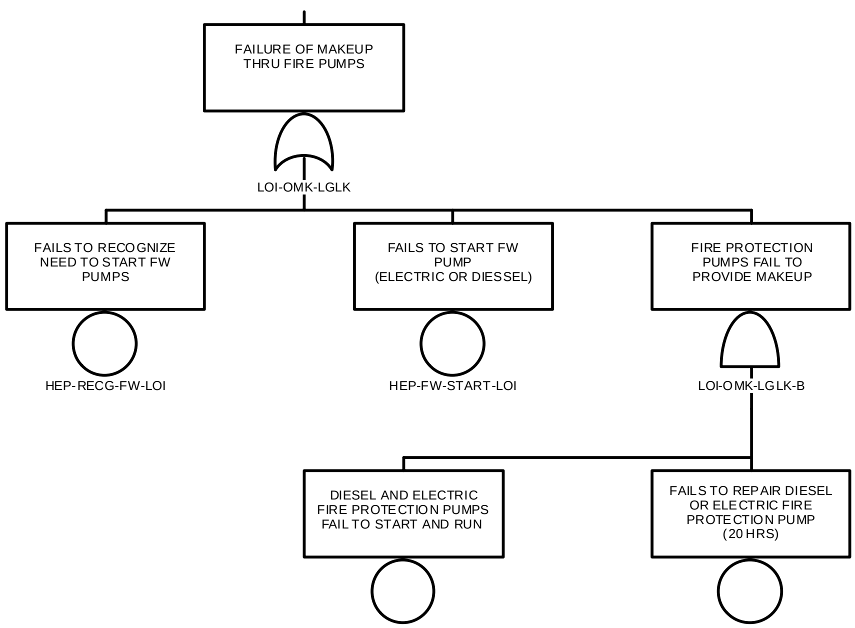

FAILURE TO ISOLATE LOSS AND INITIATE NORMAL MAKEUP (EARLY RESPOND)
LOI-OIS-U
OPERATOR FAILS TO ISOLATE LEAK
REGULAR SFP MAKEUP SYSTEM FAILS
OPERATOR FAILS TO START MAKUP (EARLY RESPOND)
HEP-LEAK-ISO
SFP-REGMKUP-F
HEP-MKUP-START-E

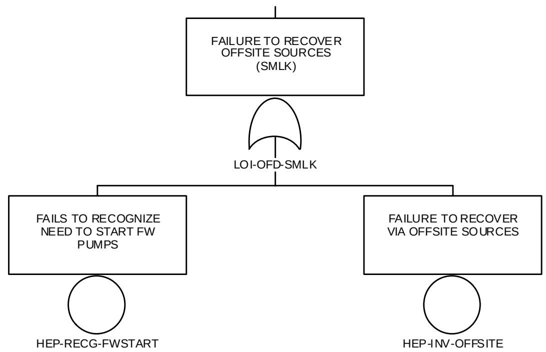

FAILURE TO INITIATE NORMAL MAKEUP (SMALL LEAK)
LOI-OIL-U
REGULAR SFP MAKEUP SYSTEM FAILS
OPERATOR FAILS TO START MAKUP (SMALL LEAK)
SFP-REGMKUP-F
HEP-MKUP-START

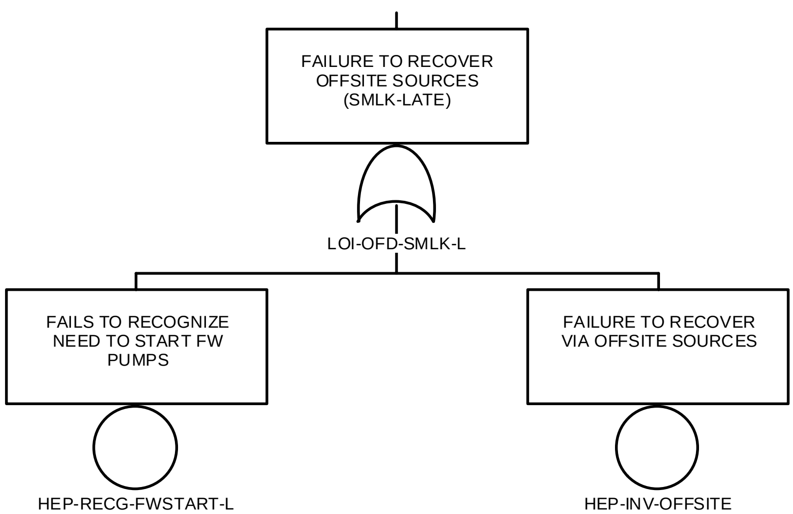

FAILURE TO INITIATE NORMAL MAKEUP (SMALL LEAK)
LOI-OIL-L
REGULAR SFP MAKEUP SYSTEM FAILS
OPERATOR FAILS TO START MAKUP (SMALL LEAK)
SFP-REGMKUP-F
HEP-MKUP-START-E

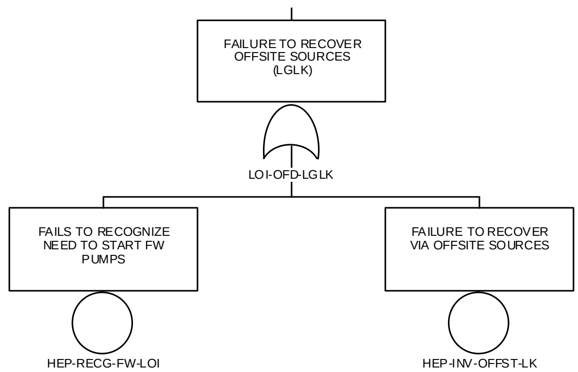

FAILURE OF MAKEUP THRU FIRE PUMPS (SMALL LEAK)
LOI-OMK-SMLK
OPERATOR FAILS TO INITIATE MAKEUP BY USE OF FW PUMP (ELECTRIC OR DIESEL)
FAILS TO RECOGNIZE NEED TO START FW PUMPS
FAILS TO START FW PUMP (ELECTRIC OR DIESSEL)
LOI-OMK-SMLK-A
HEP-RECG-FWSTART
HEP-FW-START
FAILS TO REAPIR DIESEL OR ELECTRIC FIRE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

ATTACHMENT B

SPAR HRA Worksheet

SPAR HRA Human Error Worksheet (Page 1 of 3)

Plant:_____ Initiating Event:____ Sequence Number:___________Basic Event Code: ______________

Basic Event Context: ___

Basic Event Description:___

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

Part I. DIAGNOSIS

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

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

B. Calculate the Diagnosis Failure Probability

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

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

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

Action

Failure Probability

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: _______________ Diagnosis Failure Probability:
10E-2

Action Failure Probability: +_______________ Action Failure Probability:
+10E-3

Task Failure Without
Formal Dependence ($P_{w/od}$) =_______________ $P_{(w/od)}$
= 1.1x10E-2

Part IV. DEPENDENCY

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

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

Dependency Condition Table

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

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

For Complete Dependence the probability of failure is 1.

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

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

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

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

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

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

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

1. Introduction

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

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

2. Seismic Checklist

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

These low probabilistic frequency-of-occurrence estimates are supported by deterministic considerations. The design basis earthquake ground motion, or the SSE ground motion, for

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

nuclear power plant sites were based on the assumption of the largest event geophysically ascribable to a tectonic province or a capable structure at the closest proximity of the province or fault to the site. In the case of the tectonic province in which the site is located, the event is assumed to occur at the site. For the eastern seaboard, the Charleston event is the largest magnitude earthquake and current research has established that such large events are confined to the Charleston region. The New Madrid zone is another zone in the central US where very large events have occurred. Recent research has identified the source structures of these large New Madrid earthquakes. Both of these earthquake sources are fully accounted for in the assessment of the SSE for currently licensed plants. The SSE ground motions for nuclear power plants are based on conservative estimates of the ground motion from the largest earthquake estimate to be generated under the current tectonic regime. The seismic hazards at the west coast sites are generally governed by known active fault sources, consequently, the hazard curves, which are plots of ground acceleration versus frequency of occurrence, have a much steeper slope near the higher ground motion end. In other words, as the amplitude of the seismic acceleration increases, the probability of its occurrence decreases rapidly. Therefore it is reasonable to conclude that the frequency of ground motion exceeding 3 X SSE for CEUS plants and
2 X SSE for western plants is less then 2E-5 per year.

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

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

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

3. Seismic Risk - Catastrophic Failure

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

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

For those CEUS plants where the ground motion of 3 X SSE is less than or equal to the NEI screening criterion of 0.5g PGA, the staff concludes that the risk is acceptably low. A similar conclusion can be drawn for western plants where the ground motion at 2 X SSE is within the screening criterion. For CEUS plants where 3 X SSE exceeds the screening criterion, a detailed assessment will be required to demonstrate that the pool HCLPF equals 3 X SSE. A similar conclusion can be drawn for western plants where the ground motion at 2 X SSE exceeds the screening criterion.

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

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

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

[3]These estimates are based on the Lawrence Livermore National Laboratory 1993 (LLNL 93) seismic hazard curves. Recently, the Senior Seismic Hazard Analysis Committee (SSHAC) published NUREG-CR-6372, "Recommendation for Probabilistic Seismic Hazard Analysis: Guidance On Uncertainty and Use of Experts." The report gives guidance on future application of seismic hazards. However, site specific hazard estimates have not been performed for all sites with the new method.

4. Seismic Risk - Support System Failure

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

5. Conclusion

Spent fuel pools that satisfy the seismic checklist, as written, would have a high confidence in a low probability of failure for seismic ground motions up to 0.5 g peak ground acceleration (1.2g peak spectral acceleration). Thus, sites in the central and eastern part of the U.S. that have three times SSE values less than or equal to 0.5 g PGA and pass the seismic check list would have an acceptably low level of seismic risk. Similarly, West Coast sites that have two times SSE values less than 0.5 g. and pass the seismic check list would have acceptably low values of seismic risk. From a practical point of view, a limited number of sites in the central and eastern part of the U.S. have three times SSE values greater than 0.5g; the two times SSE values exceed 0.5g for two West Coast plants. In order to demonstrate acceptably low seismic risk, those central and eastern sites for which the three times SSE values exceed 0.5g and the two West Coast sites would have to perform additional plant specific analyses to demonstrate HCLPF for their spent fuel pools at three times SSE and two times SSE values of ground acceleration, respectively. For these sites the frequency of failure is bounded by $3x10^{-6}$ per year, and other considerations indicate the frequency may be significantly lower. Plants which cannot demonstrate HCLPF values equivalent to 3 X SSE or 2 X SSE as appropriate may perform a risk assessment to demonstrate acceptably low frequency of SFP failure.

1. Introduction

A heavy load drop into the spent fuel pool (SFP) or onto the spent fuel pool wall can affect the structural integrity of the spent fuel pool. A loss-of-inventory from the spent fuel pool could occur as a result of a heavy load drop. For single failure proof systems where load drop analyses have not been performed at decommissioning plants, the mean frequency of a loss-of-inventory caused by a cask drop was estimated to be 2.0×10^{-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[1] [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 spent fuel pool 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 spent fuel pool (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

[1] 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.

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.

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 5^{th} and 95^{th} percentile of the distribution. From this the mean of the distribution was calculated. Data provided by NEI on actual lifts and setdowns of spent fuel pool 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 five 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 due to 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 spent fuel pool 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 spent fuel pool for between 5% and 25% 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[2], 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 eight to ten 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 spent fuel pool. Based on NUREG-0612, if the cask were dropped on the spent fuel pool 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 spent fuel pool 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 $2.0x10^{-7}$ 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

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[2] 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.

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

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 InThe 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.

Attachment 2C-1

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 spent fuel pool to result in a loss-of-inventory was estimated to be between 0.5% and 6.25% of the total path needed to lift, move, and set down the load. This range was developed by the staff for the NUREG-0612 evaluation. 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% or "improper rigging" by incident cause were rigging failures during crane movement, and 70% of "improper rigging" by incident cause were rigging errors.

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

Table A2c-2 Summary of NUREG-0612 Heavy Loads Evaluation (for cask drop) with New 1990s Navy Crane Data Values and WIPP Rigging HEP Method

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

Table A2c-3 WIPP Evaluation for Failure to Secure Load (improper rigging estimate**)**

Symbol	HEP	Explanation of error	Source of HEP (NUREG/CR-1278)
A_1	$3.75x10^{-3}$	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	$1.25x10^{-3}$	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	$5.2x10^{-7}$	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	$3.75x10^{-3}$	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	$1.25x10^{-3}$	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	$3.5x10^{-7}$	Failure rate if first pin improperly connected	$a_1 * A_2 * B_2 * C_2 * D_2$
F_T	$8.7x10^{-7}$	Total failure due to human error	F1 + F2

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

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

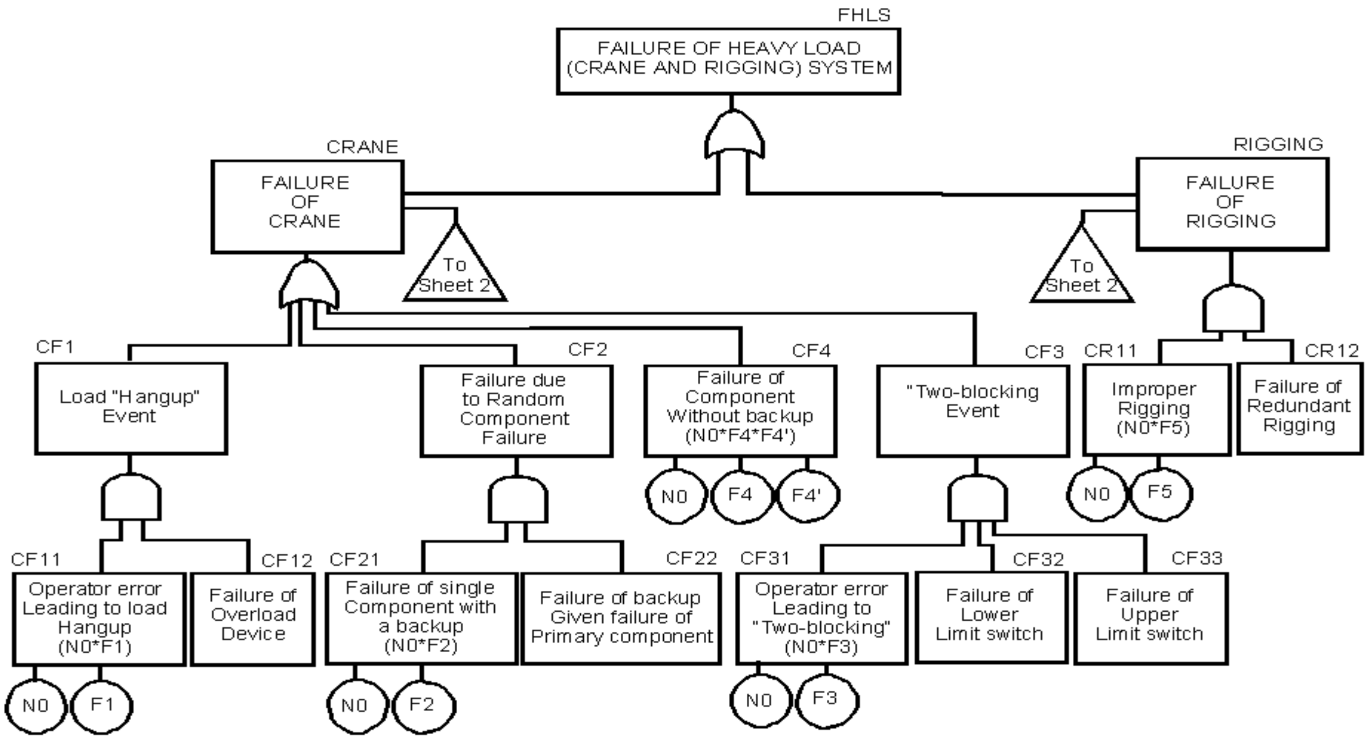
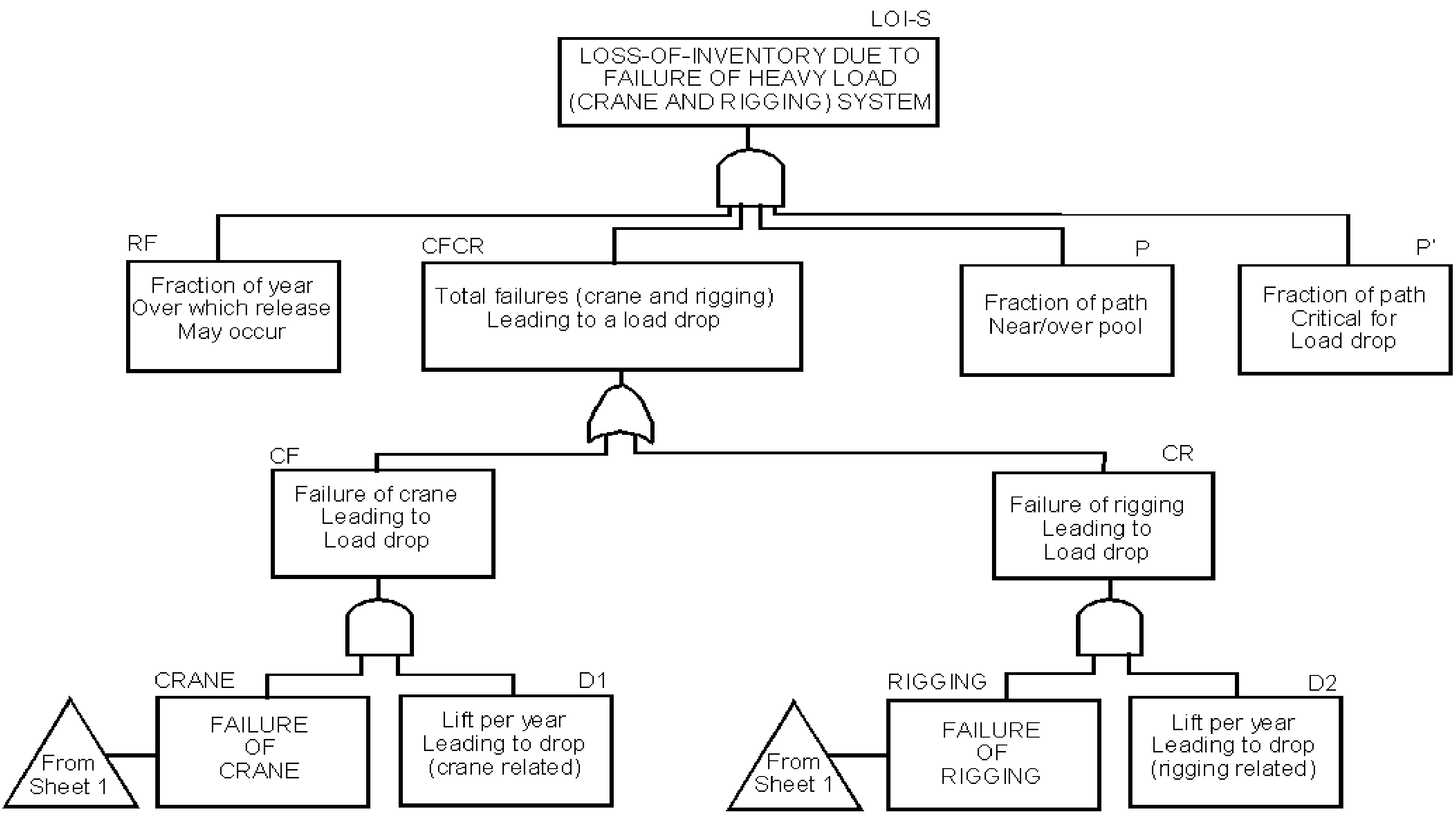


Figure A2c-1 (sheet 2 of 2) - Heavy Load Drop Fault Trees



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

2. Analysis

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

The staff used the generic data provided in DOE-STD-3014-96 [Ref. 1] to assess the likelihood of an aircraft crash into or near a decommissioned spent fuel pool. Aircraft damage can affect the structural integrity of the spent fuel pool or the availability of nearby support systems, such as power supplies, heat exchangers, and make-up 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
A_f = effective fly-in area
A_s = effective skid area
WS= wing span
H= height of facility
L= length of facility
W= width of facility
S= aircraft skid distance

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

where:

Equation A2d-2

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

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

cotθ= 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 Spent Fuel Pool Damage

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

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

4. Support System Unavailability

The frequency for loss of a support system (e.g., power supply, heat exchanger, or make-up 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 on-site or off-site 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 on-site or off-site recovery actions.

5. Uncertainties

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

6. References

1. DOE-STD-3014-96, “Accident Analysis for Aircraft Crash Into Hazardous Facilities,” U.S. Department of Energy (DOE), October 1996
2. A. Mosleh and R.A. Bari (eds),“Probabilistic Safety Assessment and Management,” *Proceedings of the 4th International Conference on Probabilistic Safety Assessment and Management*, PSAM 4, Volume 3, 13-18 September 1998, New York City.
3. C.T. Kimura et al.,“Aircraft Crash Hit Analysis of the Decontamination and Waste Treatment Facility (DWTF), Lawrence Livermore National Laboratory.
4. K. Jamali, et al., “Application of Aircraft Crash Hazard Assessment Methods to Various Facilities in the Nuclear Industry.”
5. NUREG/CR-5042, “Evaluation of External Hazards to Nuclear Power Plants in the United States,” Lawrence Livermore National Laboratory, December 1987.

Table A2d-1 Generic Aircraft Data

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

Table A2d-2 Aircraft Hits Per Year

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

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

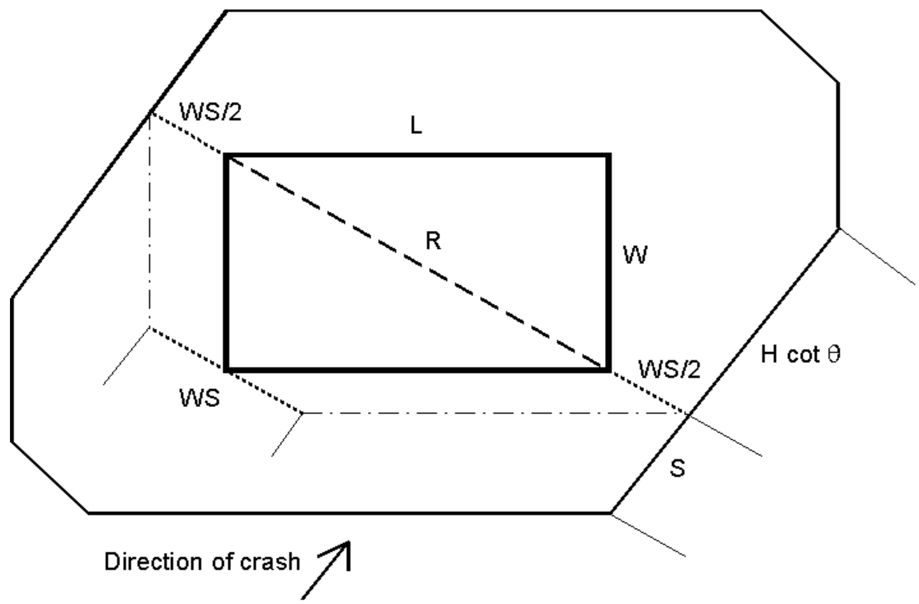
Period	Air Carriers	Air Taxes	General Aviation	Military Aviation	Total[1]
1995	1.72×10^{-7}	2.47×10^{-6}	2.45×10^{-5}	5.03×10^{-7}	2.76×10^{-5}
1993-1995	1.60×10^{-7}	2.64×10^{-6}	2.82×10^{-5}	6.47×10^{-7}	3.16×10^{-5}
1991-1995	1.57×10^{-7}	2.58×10^{-6}	2.89×19^{-5}	7.23×10^{-7}	3.23×10^{-5}
1986-1995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

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

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

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

Figure A2d-1 Rectangular Facility Effective Target Area Elements



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

1. Introduction

Tornado damage from missiles have the potential to affect the structural integrity of the spent fuel pool or the availability of nearby support systems, such as power supplies, cooling pumps, heat exchangers, and make-up water sources, and may also affect recovery actions. Department of Energy (DOE) studies indicate that the thickness of the spent fuel pool walls (greater than four feet of reinforced concrete) is more than sufficient protection from missiles that could be generated by the most powerful tornadoes ever recorded in the United States. In addition, the frequency of meeting or exceeding the wind speeds of F4 to F5 tornadoes (the most powerful tornadoes on the Fujita scale) is estimated to be on the order of 6×10^{-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 2×10^{-5} per year. This is not the estimated frequency of fuel uncovery on a zirconium fire since the frequency estimate does not include credit for maintaining pool inventory from either on-site or off-site sources.

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

2. Analysis

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

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

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

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

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

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

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

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

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

where:

P_{fs} = strike probability for F-scale (fs)
$<a>_T$ = tornado area, mi^2
A_{ob} = area of observation, mi^2 (state land area)
Y_{int} = interval over which observations were made, years
Σ_N = sum of reported tornados in the area of observation

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

The tornado area, $\langle a \rangle_T$, 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 $\langle a \rangle_T$ 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 $\langle a \rangle_T$ of about 0.28 mi^2. The total area is reapportioned using Table A2e-4 to assign 0.11 mi^2 to the F0 classification, 0.13 mi^2 to the F1 classification, and 0.04 mi^2 to the F2 classification.

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

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

4. Significant Pool Damage

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

5. Support System Availability

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

6. References

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

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

3 NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the

United States," Lawrence Livermore National Laboratory, December 1987.
4 DOE-STD-1020-94, "Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities," January 1996, Department of Energy

Table A2e-1 Tornado Characteristics

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

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

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

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

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

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

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

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

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

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

Table A2e-5 Exceedance Probability for Each F-scale

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

Table A2e-6 SPC Data Analysis Summary by State

	NUREG/CR -2944 Region					Tornado F-scale							Point Strike Probability (per year)						Land Area
State	A	B	C	D	Years	F0	F1	F2	F3	F4	F5	Total	F0	F1	F2	F3	F4	F5	(mi^2)
AL	X	X			46	165	364	323	129	36	14	1031	2.9e-05	3.2e-05	1.3e-05	3.7e-06	6.9e-07	4.3e-08	50750
AZ				X	44	90	57	11	2	0	0	160	6.7e-07	2.9e-07	3.6e-08	1.8e-09	0	0	113642
AR	X				46	198	298	331	149	31	0	1007	3.2e-05	3.5e-05	1.3e-05	2.4e-06	1.9e-07	0	52075
CA				X	45	142	58	21	2	0	0	223	5.1e-07	2.7e-07	6.0e-08	2.7e-09	0	0	155973
CO			X	X	46	616	441	99	15	1	0	1172	4.4e-06	2.0e-06	4.2e-07	3.9e-08	3.3e-11	0	103730
CT			X		46	9	29	20	5	2	0	65	1.1e-05	1.1e-05	3.6e-06	8.5e-07	2.2e-07	0	4845
DE			X		42	20	23	11	1	0	0	55	2.6e-05	1.5e-05	1.5e-06	6.4e-09	0	0	1955
DC*					1	1	0	0	0	0	0	1	1.3e-04	0	0	0	0	0	61
FL		X	X		46	1156	665	293	30	4	0	2148	1.5e-05	8.6e-06	2.2e-06	2.8e-07	2.0e-08	0	53997
GA		X			46	147	537	266	65	17	0	1032	2.9e-05	3.0e-05	1.2e-05	3.4e-06	4.3e-07	0	57919
ID				X	42	63	53	8	0	0	0	124	4.7e-07	1.9e-07	1.4e-08	0	0	0	82751
IN	X				46	246	336	263	108	77	8	1038	3.3e-05	3.5e-05	1.5e-05	5.2e-06	1.2e-06	6.7e-08	35870
IA	X				46	478	506	421	119	74	9	1607	3.7e-05	3.7e-05	1.4e-05	3.1e-06	6.1e-07	2.5e-08	55875
IL	X				46	431	440	316	113	39	3	1342	3.0e-05	2.7e-05	9.8e-06	2.5e-06	3.3e-07	2.1e-08	55875
KS	X	X			46	1111	610	404	168	54	16	2363	3.5e-05	3.0e-05	1.1e-05	3.0e-06	5.8e-07	1.1e-07	81823
KY	X				46	79	168	133	65	35	3	483	1.6e-05	1.7e-05	6.9e-06	1.8e-06	3.1e-07	1.4e-08	39732
LA		X			46	225	620	268	123	16	2	1254	2.4e-05	2.2e-05	6.9e-06	1.4e-06	1.2e-07	1.9e-08	43566
ME				X	42	21	44	17	0	0	0	82	1.8e-06	1.1e-06	1.7e-07	0	0	0	30865
MD			X		46	49	92	26	5	0	0	172	1.5e-05	9.2e-06	9.4e-07	8.2e-09	0	0	9775
MA			X		45	24	72	31	8	3	0	138	1.2e-05	1.1e-05	4.3e-06	1.6e-06	3.7e-07	0.0e+00	7838
MI		X	X		45	195	308	210	57	30	7	807	1.4e-05	1.4e-05	5.2e-06	1.4e-06	2.8e-07	1.4e-08	56809
MN		X	X		46	372	336	158	53	28	6	953	1.4e-05	1.2e-05	3.5e-06	7.2e-07	1.3e-07	6.6e-09	79617
MS	X	X			46	226	468	369	136	59	10	1268	4.4e-05	4.4e-05	1.7e-05	5.0e-06	1.0e-06	1.3e-08	46914
MO	X				46	298	577	334	109	48	1	1367	1.8e-05	1.6e-05	5.3e-06	1.3e-06	2.3e-07	2.6e-11	68898
MT				X	44	174	42	33	4	0	0	253	1.0e-06	7.0e-07	2.3e-07	2.2e-08	0	0	145556
NE		X	X		46	827	585	255	105	42	4	1818	2.9e-05	2.9e-05	1.2e-05	3.5e-06	3.5e-07	1.6e-08	76878
NV				X	34	41	8	0	0	0	0	49	2.9e-07	4.0e-08	0	0	0	0	109806

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)
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

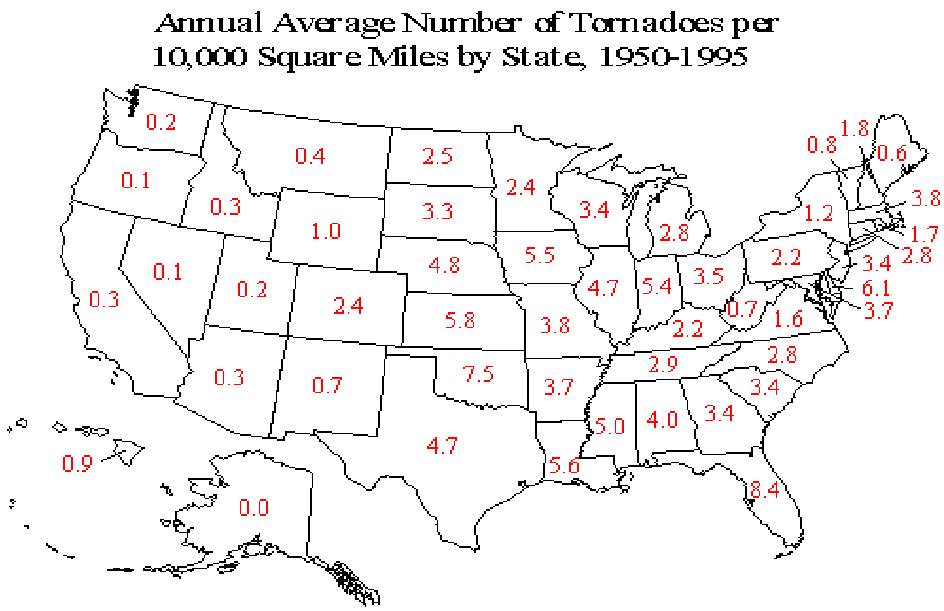


Figure A2e-2

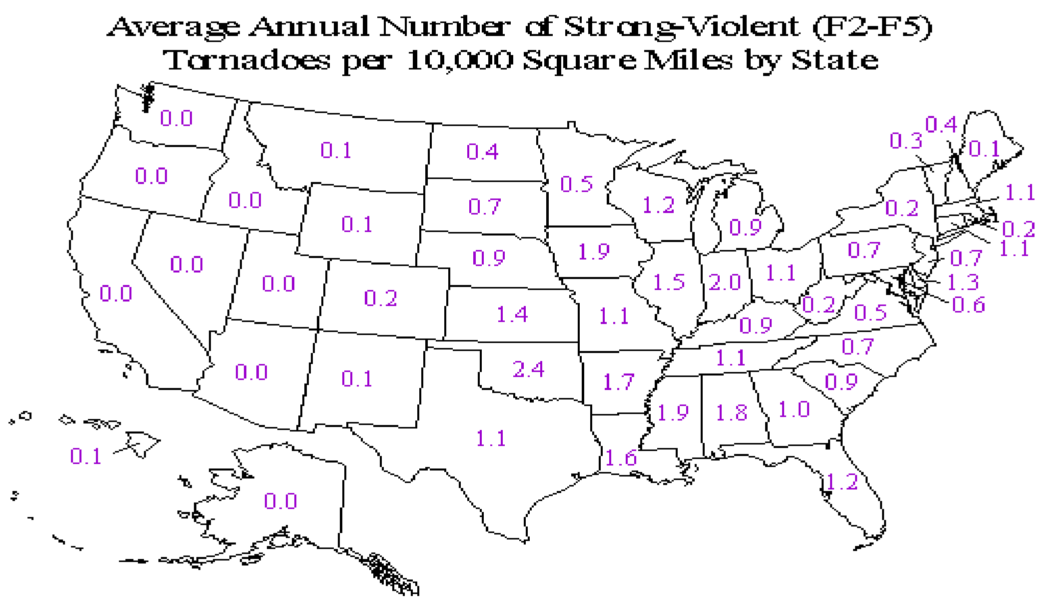


Figure A2e-3 Sketch of Hypothetical F2 Tornado Illustrating Variations

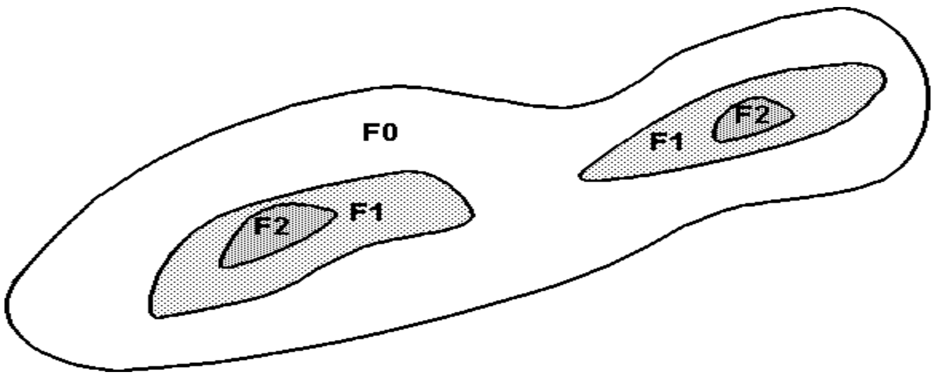


Figure A2e-4 Tornado Risk Regionalization Scheme (from NUREG/CR-2944)

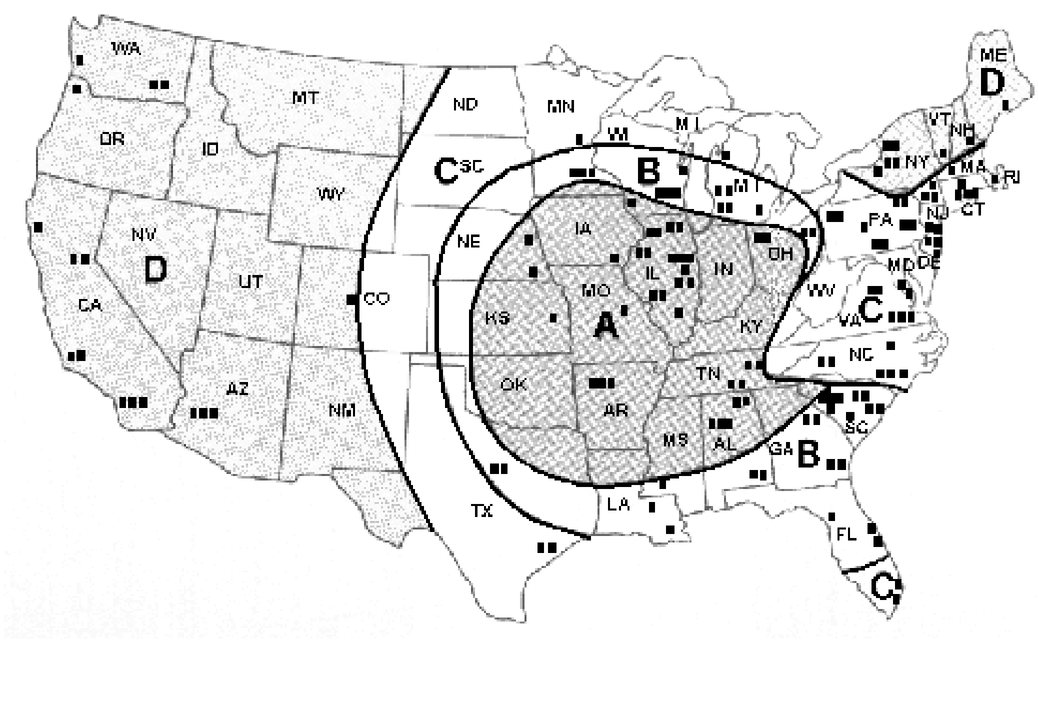
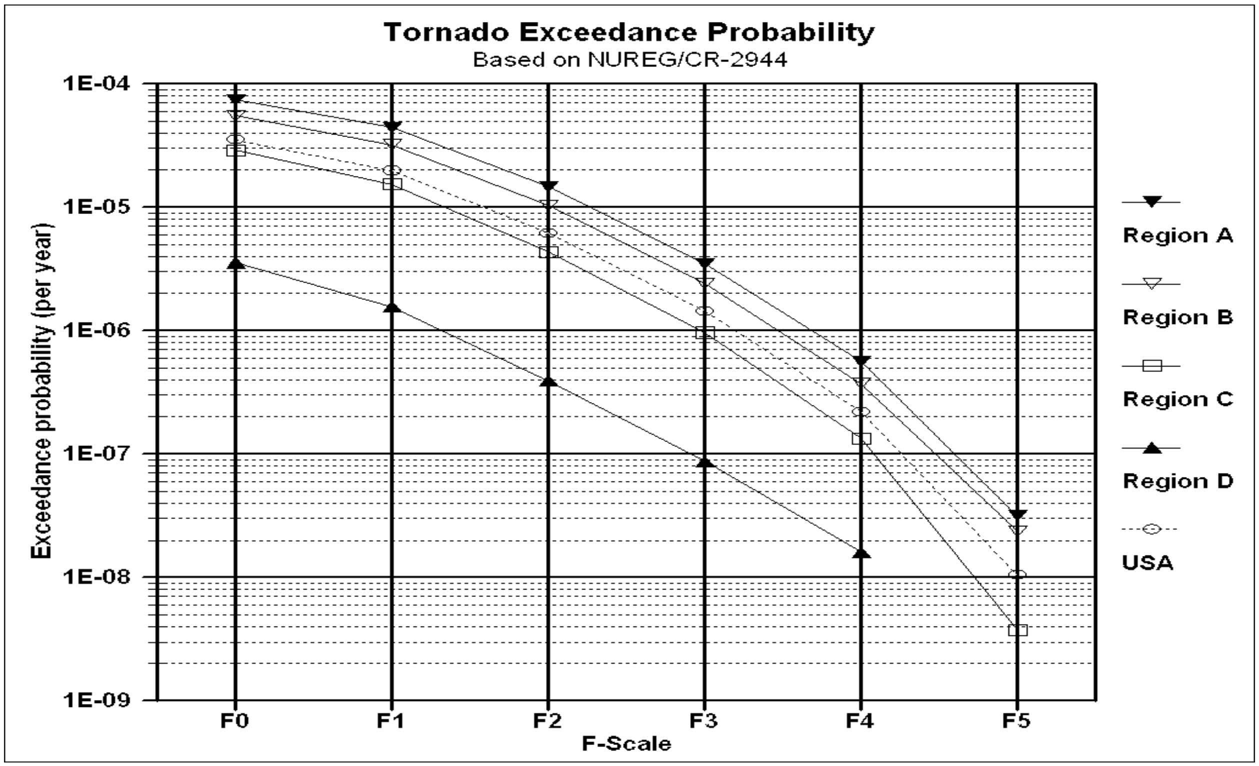


Figure A2e-5 Tornado Exceedance Probability For Each F-scale



Appendix 3 Criticality

3.1 Introduction

The staff criticality assessment includes both a more classical deterministic study and a qualitative risk study. The conclusion in chapter 3 of this report that criticality is not a risk significant event, is based upon consideration of both of these studies. The deterministic study was used to define the possible precursor scenarios and any 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 chapter 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

Due to the processes involved and lack of data, it was not possible to perform a quantitative risk assessment for criticality in the spent fuel pool. 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 due to 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 due to gamma radiation and exposure to the wet pool environment. For this reason, the NRC issued Generic Letter 96-04 to all holders of operating licenses, on Boraflex degradation in spent fuel storage racks. Each addressee that uses Boraflex was requested to assess the capability of the Boraflex to maintain a 5% subcriticality margin and to submit to the NRC proposed actions to monitor the margin or confirm that this 5% margin can be maintained for the lifetime of the storage racks. Many licensees subsequently replaced the Boraflex racks in their pools or reanalyzed the criticality aspects of their pools, assuming no reactivity credit for Boraflex.

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

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

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

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

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

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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Division of Systems Safety and Analysis

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
KENO-VI Input for Storage Cell Calc. High Density Poisoned Rack
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
ba-140     1 0 0.1097E-05 300.00 end
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
pm-147     1 0 0.8045E-05 300.00 end
pm-148     1 0 0.4711E-07 300.00 end
pm-148     1 0 0.6040E-07 300.00 end
pm-149     1 0 0.6407E-07 300.00 end
sm-147     1 0 0.3349E-05 300.00 end
sm-149     1 0 0.1276E-06 300.00 end
sm-150     1 0 0.1409E-04 300.00 end
sm-151     1 0 0.7151E-06 300.00 end
sm-152     1 0 0.5350E-05 300.00 end
eu-153     1 0 0.4698E-05 300.00 end
eu-154     1 0 0.1710E-05 300.00 end
eu-155     1 0 0.6732E-06 300.00 end
gd-154     1 0 0.1215E-06 300.00 end
gd-155     1 0 0.5101E-08 300.00 end
gd-156     1 0 0.2252E-05 300.00 end
gd-157     1 0 0.3928E-08 300.00 end
gd-158     1 0 0.6153E-06 300.00 end
gd-160     1 0 0.3549E-07 300.00 end
u-234      1 0 0.6189E-07 300.00 end
u-235      1 0 0.3502E-03 300.00 end
u-236      1 0 0.1428E-03 300.00 end
u-238      1 0 0.2146E-01 300.00 end
np-237     1 0 0.1383E-04 300.00 end
pu-238     1 0 0.4534E-05 300.00 end
pu-239     1 0 0.1373E-03 300.00 end
pu-240     1 0 0.5351E-04 300.00 end
pu-241     1 0 0.3208E-04 300.00 end
pu-242     1 0 0.1127E-04 300.00 end
am-241     1 0 0.9976E-06 300.00 end
am-242     1 0 0.2071E-07 300.00 end
am-243     1 0 0.2359E-05 300.00 end
cm-242     1 0 0.3017E-06 300.00 end
cm-244     1 0 0.6846E-06 300.00 end
i-135      1 0 0.2543E-07 300.00 end
'Zirc
cr         2 0 7.5891E-5   300.0 end
```

```
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

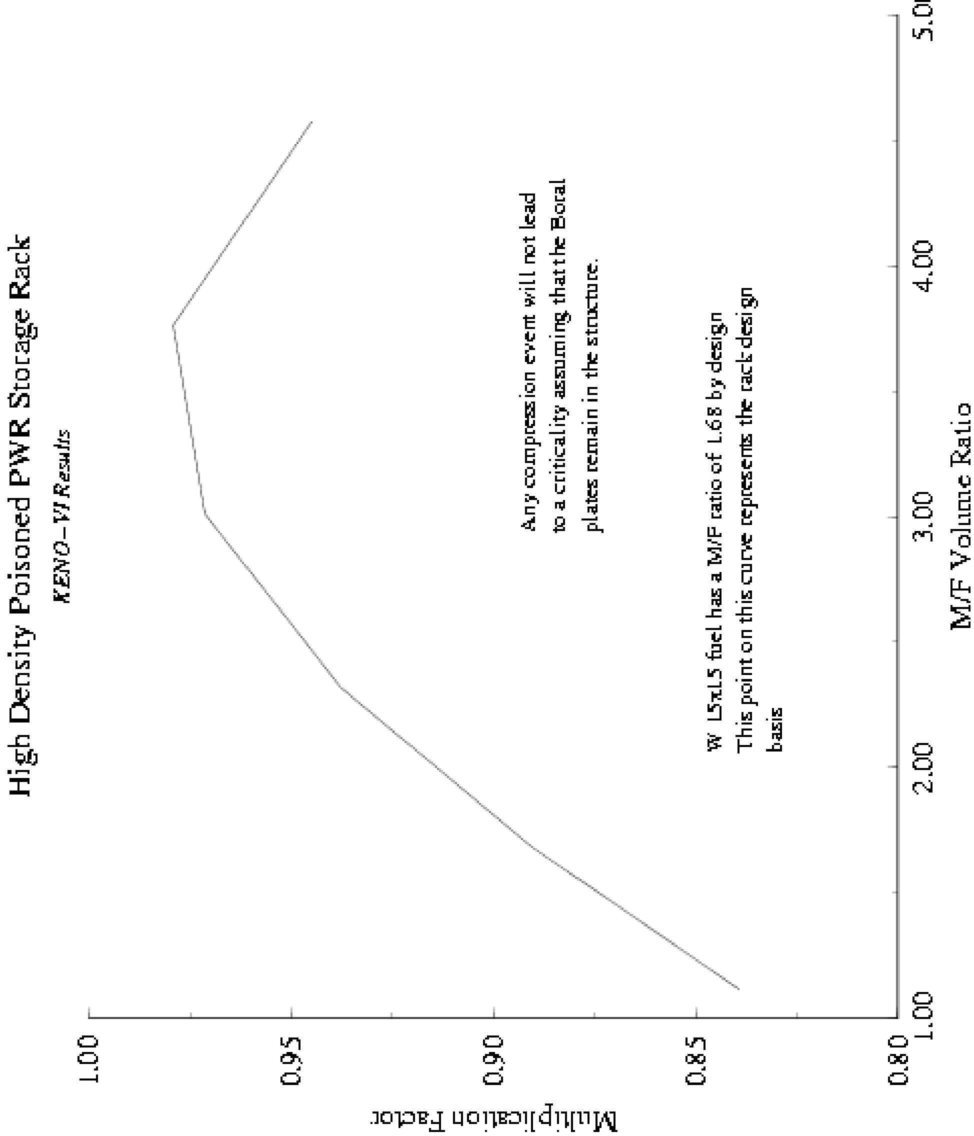
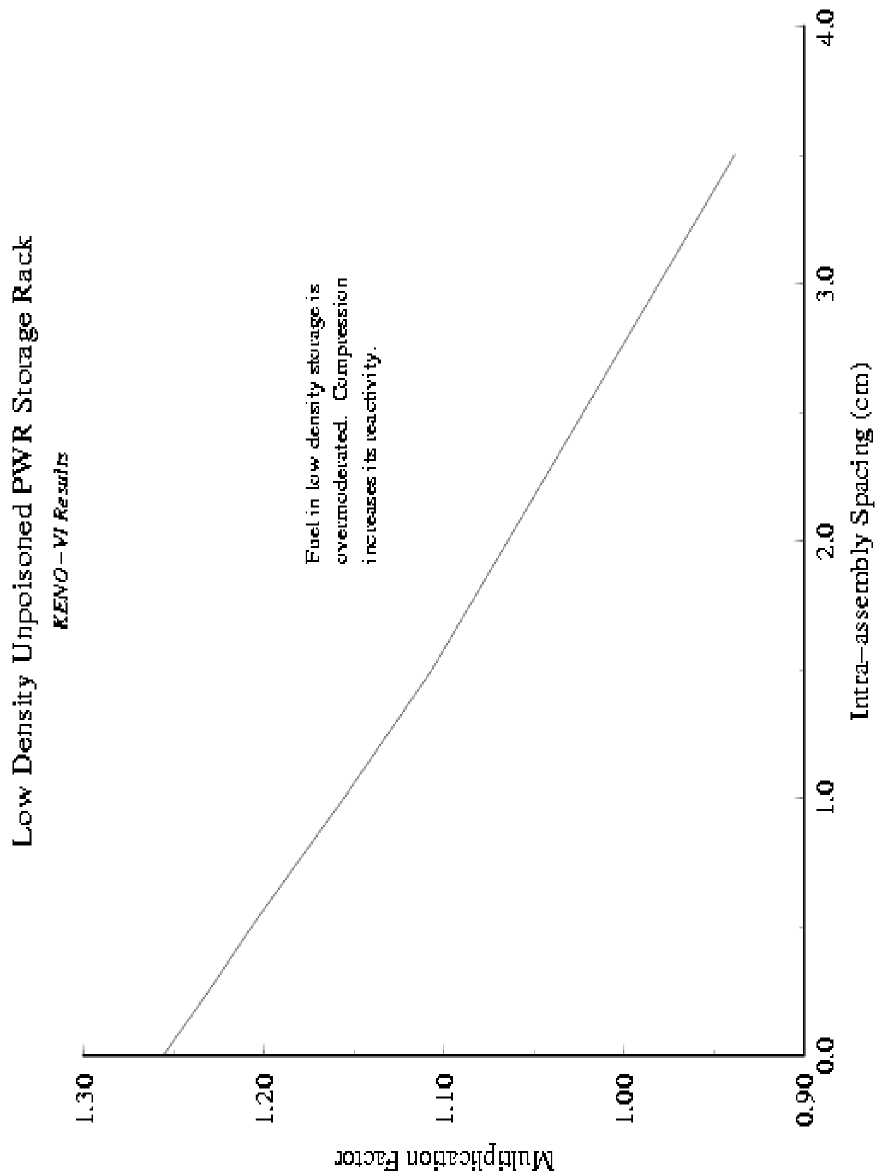


Figure 2 PWR Low Density Storage Rack Eigenvalue Following Compressive/Expansion Events



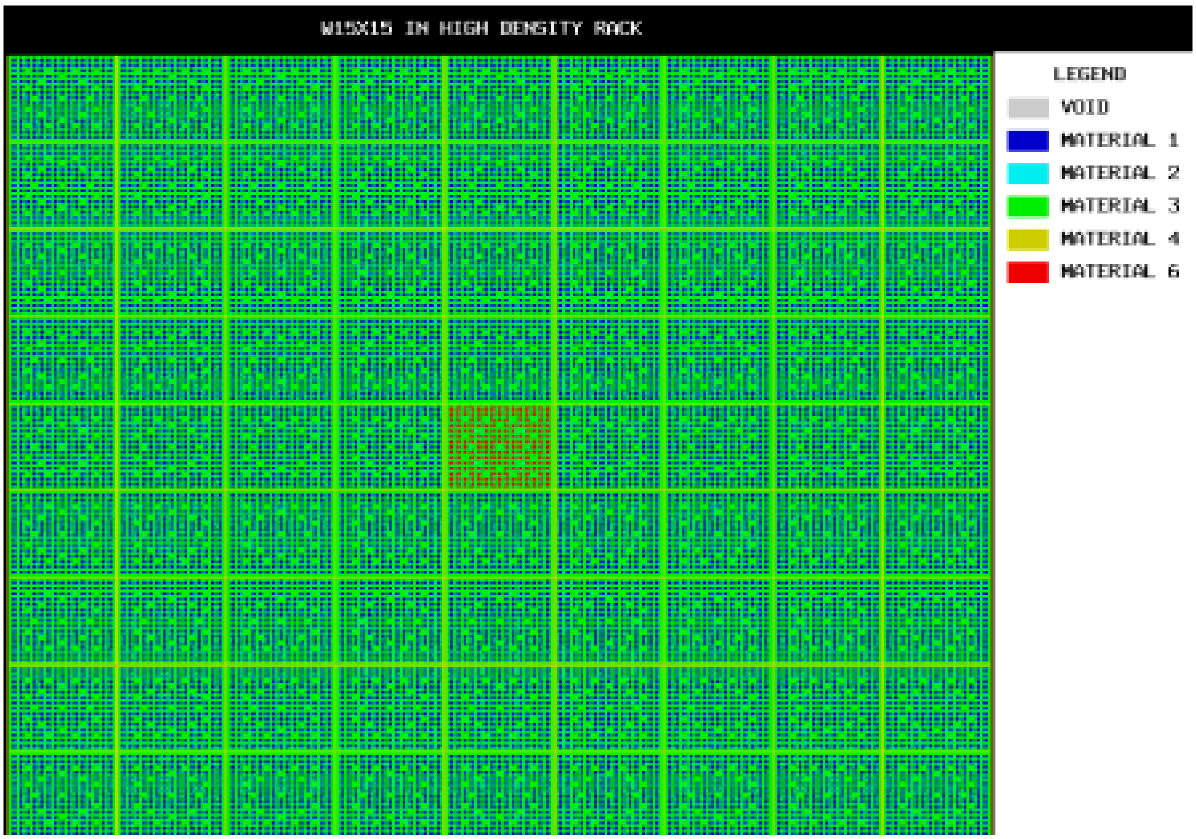


Figure 3 Sample Geometry Assuming 4 Assembly Spacing Between Most Reactive Assembly

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

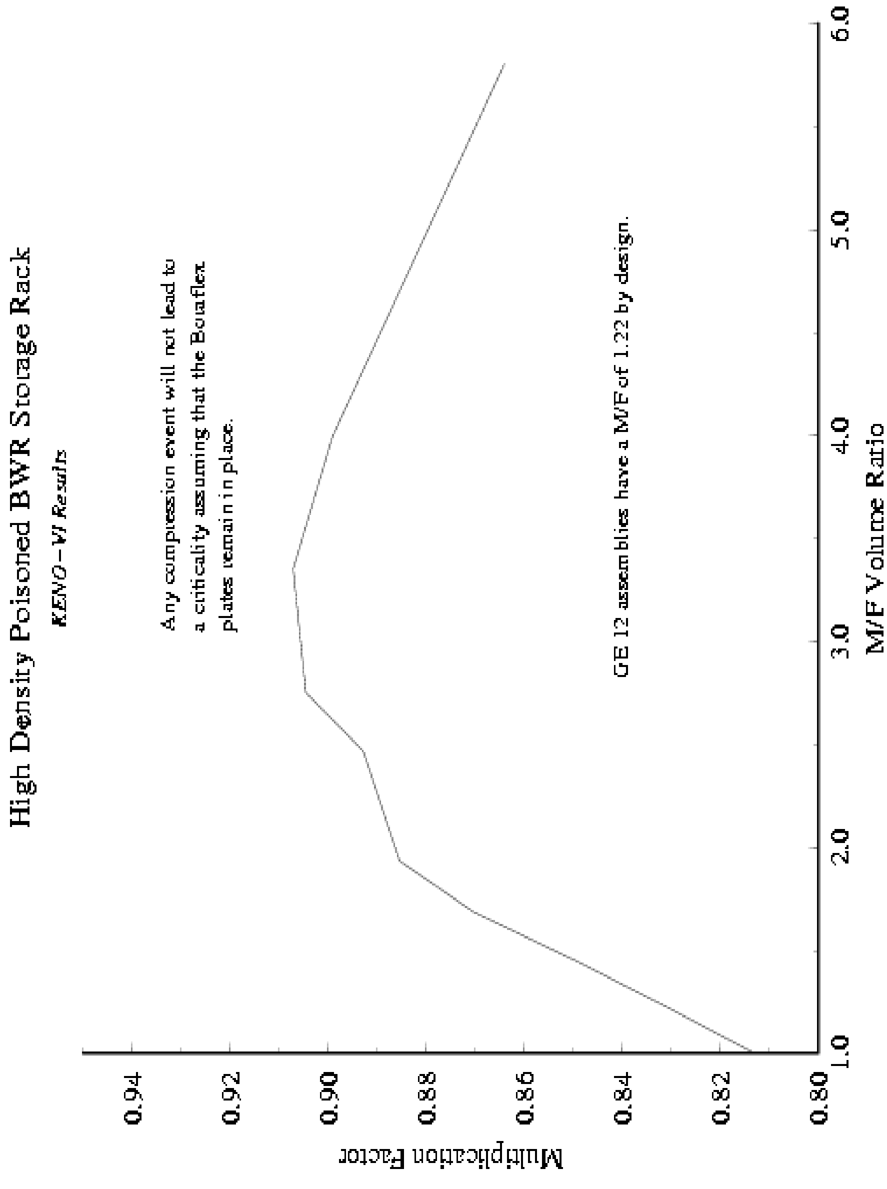
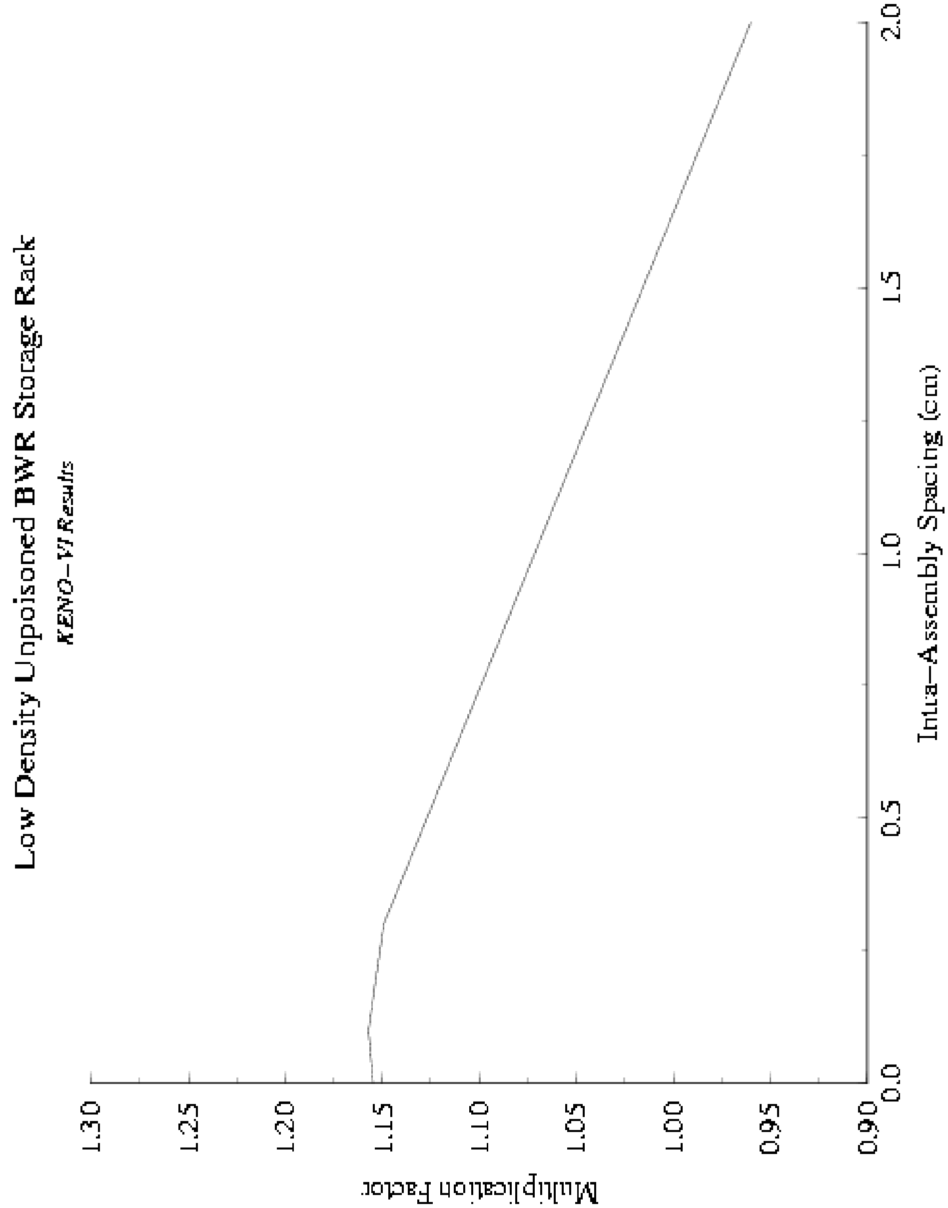


Figure 5 BWR Low Density Storage Rack Eigenvalue Following Compressive/Expansion Events



Appendix 4 Consequence Assessment from Zirconium Fire

Spent fuel pool 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 a spent fuel pool 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 spent fuel pool to heat up to the point of releasing radionuclides than in some reactor accidents.

Earlier analyses in NUREG/CR-4982[1] and NUREG/CR-6451[2] have assessed the frequency and consequences of spent fuel pool accidents. These analyses included a limited evaluation of off-site consequences of a severe spent fuel pool 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 spent fuel pool. 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 off-site consequences for a severe spent fuel pool 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 a spent fuel pool, and the resulting radioactive decay, on off-site 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[3] (version 2) to estimate off-site consequences for a severe spent fuel pool 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 spent fuel pool accident is discussed below.

Radionuclide Inventories

As discussed above, the current analysis was undertaken to assess the magnitude of the decrease in off-site consequences that could result from up to a year of decay in the spent fuel pool. To perform this work, it was necessary to have radionuclide inventories in the spent fuel pool 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 spent fuel pool 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 spent fuel pool 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 spent fuel pool risk for an operating reactor, the Millstone 1 spent fuel pool 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 spent fuel pool.

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 off-site 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 spent fuel pool 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 off-site from fuel with high burnup. While the staff believes that it is unlikely that fuel fines would be released off-site 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 off-site 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 off-site release ends. Only people within 10 miles of the spent fuel pool 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 off-site emergency response officials	0	0
start time of off-site release	2400	.7
end time of off-site 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.

Off-site 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 off-site consequences resulting from extended decay prior to 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 off-site consequences for a severe spent fuel pool 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 spent fuel pool 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 off-site consequences for a severe spent fuel pool 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 off-site 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 due to 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 spent fuel pool 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 spent fuel pool 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 a spent fuel pool 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 off-site, there would be no change in off-site 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 spent fuel pool 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 spent fuel pool 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 spent fuel pool 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 spent fuel pool 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 due to 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 due to 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 off-site 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 a spent fuel pool, and the resulting radioactive decay, on off-site consequences of a severe spent fuel pool accident at a decommissioned reactor. This evaluation was performed in support of the generic evaluation of spent fuel pool 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 5 Enhanced Seismic Checklist and Supporting Stakeholder Documentation

Appendix 5 contains the following sub-sections:

Alan Nelson
SENIOR PROJECT MANAGER
PLANT SUPPORT
NUCLEAR GENERATION DIVISION

Mr. Richard Dudley
Project Manager
U.S. Nuclear Regulatory Commission
Mail Stop 11 D19
Washington, DC 20555-0001

Dear Mr. Dudley:

On July 15-16, 1999, the NRC held a workshop on spent fuel accidents at decommissioning plants. During the course of the workshop, presentations by the NRC and the industry concluded that spent fuel pools possess substantial capability beyond their design basis to with stand seismic events but that variations in seismic capacity existed due to plant specific designs and locations.

The consensus was that the risk was low enough that precise quantification was not necessary to support exemption requests but that this needed to be confirmed on a plant specific basis with deterministic criteria. It was recommended that a simple spent fuel pool (SFP) vulnerability check list be developed to provide additional assurance that no beyond-design-basis seismic structural vulnerabilities exist at decommissioning plants. Enclosed for your review is the “Seismic Screening Criteria For Assessing Potential Pool Vulnerabilities At Decommissioning Plants.”

Please contact me at (202) 739-8110 or by e-mail (apn@.nei.org) if you have any questions or if a meeting should be scheduled to discuss the enclosed seismic checklist.

Sincerely,

Alan Nelson

APN:tnb

Enclosure

Seismic Screening Criteria

For

Assessing Potential Fuel Pool Vulnerabilities

At

Decommissioning Plants

August 18, 1999

Background

To increase the efficiency and effectiveness of decommissioning regulations, the NRC staff has engaged in rulemaking activities that would reduce the need to routinely process exemptions once a plant is permanently shut down. With this goal in mind, members of the NRC staff, industry representatives and other stakeholders held a two-day workshop on risk related spent fuel pool accidents at decommissioning plants.

At this workshop, based upon presentations by the NRC staff (Goutam Bagchi et al.) and the nuclear industry (T. O'Hara - DE&S), it was concluded that a large seismic event (in the range of three times the design level earthquake) would represent a risk of exceeding the structural capacity of the spent fuel pool and thus potentially result in draining the pool.

Although the methodologies presented by the NRC staff and the industry differed somewhat, they both concluded that, in general, spent fuel pools possess substantial capacity beyond their design basis but that variations in seismic capacity existed due to plant specific details (i.e. "Differences in seismic capacity due to spent fuel location and other details.").

The consensus was that the risk was low enough that precise quantification was not necessary to support exemption requests but that this needed to be confirmed on a plant specific basis with deterministic criteria. It was recommended that a simple spent fuel pool (SFP) vulnerability check list be developed to provide additional assurance that no beyond-design-basis seismic structural vulnerabilities exist at decommissioning plants. The following pages provide the proposed structural vulnerability check list/screening criteria.

Purpose of Checklist

As discussed briefly in the "Background" section, the purpose of this checklist is to identify and evaluate specific seismic characteristics which might result in a specific spent fuel pool from not being capable of withstanding, without catastrophic failure, a beyond-design-basis seismic event equal in magnitude to approximately three times its design basis. Completion of the requirements will be performed by a qualified seismic engineer. This effort will include a thorough SFP walkdown and a review of appropriate SFP design drawings.

DRAFT CHECKLIST

Item 1:

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 a 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 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 2:

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 3:

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 could 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 4:

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 siphoning 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 5:

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 6:

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 7:

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 7: Required Documentation

A simple report describing the results of the seismic engineer's walkdown and drawing review findings is judged to provide sufficient documentation to rule out a beyond-design-basis seismic event as a significant risk contributor to a decommissioned nuclear power plant.

References:

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

2. "Seismic Discussion Session," Workshop on Risk Related to Spent Fuel Pool Accidents at Decommissioning Plants, Stuart Richards, Goutam Bagchi and Gareth Parry, July 16, 1999

3. "Draft Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants," by USNRC Technical Working Group - Vonna Ordaz et al., dated June 1999

4. "Risk Informed Decommissioning Emergency Planning," EPRI/NEI Project by Tom O'Hara (presented July 16, 1999)

5b Craig Memo to Holahan Forwarding Kennedy Report, November 19, 1999.

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

1. 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.:

$$C_{HCLPF} » C_{1\%} \quad (1)$$

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:

$$\text{Vermont Yankee (BWR):} \qquad C_{HCLPF} = 0.48g \ \text{PGA}$$
$$\text{Robinson (PWR):} \qquad C_{HCLPF} = 0.65g \ \text{PGA} \qquad (2)$$

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:

$$P_F = 0.05 \text{ HHCLPF} \qquad (3)$$

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:

<u>BWR Sites</u>			
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 $12.0x10^{-6}$ to $0.11x10^{-6}$ and averaging $1.6x10^{-6}$ 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 $3x10^{-6}$ 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:

<u>PWR Sites</u>			
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 $2.5x10^{-6}$ to $0.03x10^{-6}$ and averaging $0.48x10^{-6}$ 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 $1x10^{-6}$ 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_β=($C_{10\%/CHCLPF}$)
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 $1.25\text{x}10^{-6}$. 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 $3\text{x}10^{-6}$. 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 $1\text{x}10^{-6}$. Only three other sites have seismic risks exceeding $0.5\text{x}10^{-6}$. Table 3 includes all sites for which the computed seismic risk exceeds $0.5\text{x}10^{-6}$ 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, $\beta = 0.4$

	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

5c Huffman Memo to Richards with Staff Evaluation of Screening Criteria, December 3, 1999

December 3, 1999

MEMORANDUM TO: Stuart A. Richards, Director
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

FROM: William C. Huffman, Project Manager/S/ P. RAY FOR
Decommissioning Section
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

SUBJECT: SCREENING CRITERIA FOR ASSESSING POTENTIAL SEISMIC VULNERABILITIES OF SPENT FUEL POOLS AT DECOMMISSIONING PLANTS

The staff is in the process of preparing a final draft of its technical study on spent fuel pool accident risks at decommissioning plants. This final draft will be issued for public comment in early January 2000. Included in this report will be a discussion on risks from a large seismic event that exceeds the structural capacity of the spent fuel pool to the extent that a catastrophic failure occurs. Such a failure would result in rapid draining of the spent fuel pool with no capability of retaining water even if reflooded. The staff has previously acknowledged that spent fuel pools are inherently robust and can withstand loads substantially beyond those for which they were designed. Consequently, they have a significant seismic capacity. To take credit for the seismic design margins existent in spent fuel pools, the staff sought an appropriate method to identify potential structural vulnerabilities without having to perform a detailed fragility review. At a public workshop conducted on July 15-16, 1999, development of a simple spent fuel pool seismic screening checklist was proposed as way of assessing the seismic vulnerabilities of spent fuel pools without performing quantifying analyses. In a letter to the staff dated August 18, 1999, the Nuclear Energy Institute (NEI) proposed a “seismic checklist” for screening potential spent fuel pool structural vulnerabilities on a plant-specific basis. Based on the staff’s recent input to the final draft report, the use of a checklist is considered to be an excellent approach to plant-specific seismic assessments; however, some deficiencies have been identified in the checklist proposed by NEI. The nature of the deficiencies with the current version of the checklist was generally discussed in a public meeting with NEI and other stakeholders on November 19, 1999. NEI indicated that it needed additional details on the staff’s findings relative to the checklist in order to propose effective improvements.

The Attachment to this memorandum contains additional details on the deficiencies the staff has found with use of the current seismic checklist. Copies of this memorandum with the attached information will be provided to NEI and all other interested stakeholders in an effort to

further the dialogue relating to the seismic checklist and support the development of additional modifications that will resolve the deficiencies currently identified.

For comments to be considered for the draft report that will be issued in January 2000 for public comment, written comments must be received by the staff no later than December 13, 1999. Comments received after December 13, 1999, will be addressed in the final report that will be issued in early April 2000. The NRC staff contact for public comments is Mr. William Huffman. Mr. Huffman can be reached at (301) 415-1141.

Attachment: As stated

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Structural Failure Modes

Amongst the various ways a pool structure can fail, the only failure modes that are of concern are those that involve pool floor slab failure, failure of side walls at the bottom of the pool or at the bottom corners. It is important to ensure that the structural integrity assessment is based on realistic failure modes for catastrophic loss of structural integrity. This should take into account physical interactions with adjacent structures and equipment.

For PWR spent fuel pools, the pool floor slab is not likely to fail except through the effect of local concrete spalling due to foundation uplift and impact with the subgrade or adjacent structures. Failure of walls in partially embedded pools is not likely. Bending moment capacity of the pool walls is very much dependent on reinforcing patterns and the walls are generally reinforced in an orthotropic pattern, such that the resistance in the horizontal and vertical directions are unequal. The resistance is also unequal between one wall and another wall. This requires a case by case assessment of the bending capacity of walls.

For BWR spent fuel pools, the floor slab, walls and supporting columns and shear walls need scrutiny to determine the critical failure mode. As in the case of PWR spent fuel pools, the effect of adjacent structures and equipment on structural failure needs to be evaluated.

The stainless steel liner plate is used to assure leak-tightness; cracks in the welded seams are not likely to lead to catastrophic loss of water inventory unless there is a simultaneous massive failure of the concrete structure.

The emphasis here is that spent fuel pool structures not only vary in layout and elevation between PWRs and BWRs, they can also vary within each group. The process of realistic assessment of structural capacity of pool structures begins with a methodical consideration of likely failure modes associated with a catastrophic loss of integrity.

The efforts involved in the assessment of seismic capacity of pool structures typically consist of the following:

- Inspect the pool structure and its vicinity and note:

    - physical condition such as cracking and spalling of concrete, signs of leakage or leaching and separation of pool walls from the grade surface, potential for piping connections, either buried underground or above ground, to fail due to a large seismic excitation or interaction with adjacent equipment, and cause drainage of the pool below the safety level of the pool water,

    5. arrangement and layout of supporting columns and shear walls, assessment of other loads from tributary load areas carried by the supporting structure of the pool, as-built dimensions and mapping of any existing structural cracks,

    6. adjacent structures that can impact the pool structure both above and below the grade surface, supporting arrangement for superstructure and crane and potential for failure of the superstructure and the crane, potential impact from heavy objects that can drop in the pool structure and the corresponding drop heights.

- Seismic capacity assessments of the pool structure typically consist of the following:

- review existing layout drawings and structural dimensions and reconcile the differences, if any, between the as-built and as designed information and consider the effects of structural degradation as appropriate,

- from design calculations determine the margin to failure and assess the extrapolated multiple of SSE level that the pool structure could survive, determine whether or not design dynamic response analysis including soil-structure interaction effects are still applicable at the capacity level seismic event; if not, conduct a new analysis using properties of soil at higher strain levels and reduced stiffness of cracked reinforced concrete,

- determine the loads from pool structure foundation uplift and from impact of pool structure with adjacent structures during the capacity level seismic event, determine loads from the impact of a spent fuel rack on the pool floor and the side walls and determine the loads from dropping of heavy objects from the collapse of a superstructure or the overhead crane,

- determine a list of plausible failure modes; failure of side walls due to the worst loading from the capacity level earthquake in combination with fluid hydrostatic and sloshing head and dynamic earth pressure as appropriate, failure of the pool floor slab in flexure and bending due to loads from the masses of water and the spent fuel and racks, local failure by punching shear due to impact between structures and the spent fuel racks or dropping of heavy objects,

- the assessments to determine the lowest structural capacity can be based on ultimate strength of reinforced concrete structures due to flexure, shear and punching shear. When conducting a yield line analysis, differences in flexural yield capacities for the negative and positive bending moments in two orthogonal directions influence the crack patterns, and several sets of yield lines may have to be investigated to obtain the lowest capacity. For heterogeneous materials, the traditional yield line analysis provides upper bound solutions; consequently, considerable skill is needed to determine the structural capacity based on the yield lines that approximate the lower bound capacity.

Although the inspection of the pool structure is an essential part of establishing that the structure is in sound condition, some of the other attributes of a detailed capacity evaluation, as discussed above, may only be undertaken for plants that do not pass simple examination using a seismic checklist. Such an effort may be necessary for plants in high seismic hazard areas.

Other Considerations

NRC sponsored studies have treated the assessment of seismic capacity of spent fuel pools relying on the seismic margins method to determine the high confidence of low probability (less than 5% failure) of failure (HCLPF). The HCLPF value for a structural failure may well be unrealistic and unnecessarily conservative in terms of an instantaneous loss of water inventory.

This point needs to be emphasized because the shear and moment capacity of the walls and slabs are determined by using upper limits of allowable stresses. In the study which resulted in NUREG/CR 4982, the seismic capacities were based on the Oyster Creek reactor building and a shear wall from the Zion auxiliary building. For elevated pool structures, the Oyster Creek estimate may be an acceptable approximation, but the Zion shear wall may be too highly simplified to substitute for the catastrophic failure of the spent fuel pool structure. However, it is important to emphasize that out of plane loading on the pool walls from the hydrostatic head of the pool water can lead to flexure and shear-induced failures. Relatively low margin on allowable out-of-plane shear strength combined with the uncertainty of the extent to which reinforcement details ensure ductile behaviors make it imperative to ensure that seismic capacities of the pool walls and slab elements are adequate. The stainless steel pool liner was not designed to resist any structural load; nevertheless, it can provide substantial water- retaining capacity near the bottom half of the pool where structural deformations are likely to be low from seismic loading (this is due to the aspect ratio of the pool walls which are thick and form a deep box shape) except in a highly unlikely failure mode, such as puncturing the pool slab or the wall near the bottom of the pool.

For PWR pools that are fully or partially embedded, an earthquake motion that could cause a catastrophic failure is very high and is not a credible event. However, interaction with adjacent structures and equipment may have to be evaluated to determine the structural capacity on a case-by-case basis.

For BWR pools, the seismic capacity is likely to be somewhat less than that of a PWR pool and can vary significantly from one plant to another. This is because for most BWR pools that are at higher elevation there is amplification of seismic motion, and the pool floor may not be supported on the subgrade. Shear failure of the pool floor can occur at a relatively lower level of seismic input for BWR pools. More important, a combination of the hazard and the spent fuel pool structural capacity can bring down the likelihood of a catastrophic structural failure to a negligible risk. On the other hand, plant-specific hazard and seismic fragility of spent fuel pools can combine to produce a risk that needs to be examined on a case-by-case basis.

Using the data from NUREG-1488 (new Lawrence Livermore National Laboratory (LLNL) data) for currently operating plants in the eastern and central United States, the mean probability of exceedance (POE) of the peak ground acceleration values for the SSE were examined. The plant grouping approach, Reduced Scope, Focused Scope, Full Scope, etc., used in NUREG-1407, “Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities” Final Report was also reviewed. The objective of plant grouping for IPEEE was to put plants into groups with similar seismic vulnerability; consequently, it was useful to look at these plant groups. However, the evaluation in this draft study is driven by the 1993 LLNL seismic hazard results, and it was determined that, except for a small number of plants, the POEs for SSE are lower than $1X10^{-4}$ per reactor year and for three times the SSE, the POEs are below $1X10^{-5}$. For these plants, the likelihood of a catastrophic pool structure failure at a HCLPF value of three times the SSE should be less than $5X10^{-7}$. This makes the simplifying assumption that the conditional probability of failure (POF) or reaching the end state of a structure is $5X10^{-2}$. In this approach there is confidence that the seismic hazard is low (at three times the SSE) and there is also a plant specific structural assessment of the HCLPF value which is more than or equal to three times the SSE.

For spent fuel pools located at sites that meet the HCLPF value of three times the SSE, a catastrophic structural failure from an earthquake much larger than the design basis SSE is not credible. However, this approach may not be feasible at sites where the likelihood of the spent fuel pool structure failure due to beyond design basis earthquake is higher. For such sites in the eastern United States, a more detailed examination of the probability of the earthquake, a realistic assessment of the ground motion caused by the event at the site and the structural capacity of the spent fuel pool structure may be necessary.

NEI Draft Seismic Checklist

The draft checklist provided in an NEI letter to the staff postmarked August 18, 1999, includes seven elements that identify areas of potential weaknesses. The use of such a checklist would ensure that potential vulnerabilities are either rectified or mitigation measures are put in place. The checklist is quite comprehensive. But it can be improved by taking into account out-of-plane shear capacity of shear walls such as those that form the pool when they are not backed up by backfill. Other considerations might include pre-existing degradation of concrete and the liner plate. With minor modifications the checklist can be finalized.

Kennedy Report

As a part of an independent technical review, Dr. Robert P. Kennedy was requested to conduct this review. This review activity was supported by the Office of Nuclear Regulatory Research, Division of Engineering Technology. Dr. Kennedy attended the public workshop on July 16, 1999. The report does endorse the feasibility of the use of the seismic screening concept and identifies eight sites by site numbers for which seismically induced probability of failure (POF) is greater than $3x10^{-6}$ using the LLNL 93 Hazard. It is important to recognize that sites where POF is greater than $3x10^{-6}$, in addition to the use of the seismic checklist, an evaluation of the POF using plant-specific fragility information will be necessary. For all other sites, the use of the seismic checklist should be adequate. Appropriate excerpts of the Kennedy Report are contained in the Enclosure.

Recommendation

The following actions are recommended:

1. The seismic checklist should consider out of plane shear and flexure.

2. Identification of preexisting concrete and liner plate degradation be added to the checklist.

3. The checklist should be augmented to discuss potential mitigation measures for vulnerabilities that may be identified.

4. Higher seismic hazard sites in the Eastern U.S., should be further evaluated by the industry to determine (a) a list of such sites, (b) a credible ground motion description at which the seismic hazard frequency is low enough at these sites, and (c) plant specific seismic capacity evaluation using credible ground motion description at the site.

5. Proposed treatment of sites West of the Rocky Mountains

NOTE: Additional supplemental information from the Kennedy report is included in the following pages.

5d Nelson Letter to Huffman with Revised Criteria, December 13, 1999

NUCLEAR ENERGY INSTITUTE

Alan Nelson
SENIOR PROJECT
MANAGER,
PLANT SUPPORT
NUCLEAR
GENERATION
DIVISION

December 13, 1999

Mr. William C. Huffman
Project Manager
Decommissioning Section
Projects Directorate IV & Decommissioning
U.S. Nuclear Regulatory Commission
Mail Stop 11 D19
Washington, DC 20555-0001

Dear Mr. Huffman:

On July 15-16, 1999, the NRC held a workshop on spent fuel accidents at decommissioning plants. During the course of the workshop, presentations by the NRC and the industry concluded that spent fuel pools possess substantial capability beyond their design basis to withstand seismic events but that variations in seismic capacity existed due to plant specific designs and locations.

NEI forwarded "Seismic Screening Criteria for Assessing Potential Pool Vulnerabilities at Decommissioning Plants, to the NRC " August 18, 1999 for review and comment. Based on NRC review, the staff proposed additional details to the submitted checklist. Detailed NRC comments were made available on December 3, 1999 "Screening Criteria for Assessing Potential Seismic Vulnerabilities of Spent Fuel Pools at Decommissioning Plants."

Enclosed is the revised screening criteria addressing the December 3, 1999 NRC memorandum. We believe the revision addresses the deficiencies identified. We request that the revised checklist be considered as the NRC prepares its draft report to be issued in January 2000.

Please contact me at (202) 739-8110 or by e-mail (apn@nei.org) if you have any questions or if you would like to schedule a meeting to discuss industry's response to the staff's recommendations. .

Sincerely,

Alan Nelson
APN/dc
Enclosure

Seismic Screening Criteria

for

Assessing Potential Fuel Pool Vulnerabilities

at

Decommissioning Plants

December 13, 1999
Revision 1

Background

To increase the efficiency and effectiveness of decommissioning regulations, the NRC staff has engaged in rulemaking activities that would reduce the need to routinely process exemptions once a plant is permanently shut down. With this goal in mind, members of the NRC staff, industry representatives and other stakeholders held a two-day workshop on risk related spent fuel pool accidents at decommissioning plants.

At this workshop, based upon presentations by the NRC staff (Goutam Bagchi et al.) and the nuclear industry (T. O'Hara - DE&S), it was concluded that a large seismic event (in the range of three times the design level earthquake) would represent a risk of exceeding the structural capacity of the spent fuel pool and thus potentially result in draining the pool.

Although the methodologies presented by the NRC staff and the industry differed somewhat, they both concluded that, in general, spent fuel pools possess substantial capacity beyond their design basis but that variations in seismic capacity existed due to plant specific details (i.e. "Differences in seismic capacity due to spent fuel location and other details.").

The consensus was that the risk was low enough that precise quantification was not necessary to support exemption requests but that this needed to be confirmed on a plant specific basis with deterministic criteria. It was recommended that a simple spent fuel pool (SFP) vulnerability check list be developed to provide additional assurance that no beyond-design-basis seismic structural vulnerabilities exist at decommissioning plants. A draft seismic screening checklist was provided to the Staff by NEI in August 1999. Comments on this draft were discussed during a conference call held on December 7, 1999 and the following draft screening checklist has been revised to address the issues raised..

Purpose of Checklist

As discussed briefly in the "Background" section, the purpose of this checklist is to identify and evaluate specific seismic characteristics which might result in a specific spent fuel pool from not being capable of withstanding, without catastrophic failure, a beyond-design-basis seismic event equal in magnitude to approximately three times its design basis. Completion of the requirements will be performed by a qualified seismic engineer. This effort will include a thorough SFP walkdown and a review of appropriate SFP design drawings.

DRAFT 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 our-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 based upon 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 "zirc-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 1.0E-06.)

Item 11: Required Documentation

A simple report describing the results of the seismic engineer's walkdown and drawing review findings is judged to provide sufficient documentation to rule out a beyond-design-basis seismic event as a significant risk contributor to a decommissioned nuclear power plant.

References:

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

5e The "Industry Comments" Referred to in the December 28, 1999 Kennedy letter

Comments on NRC Draft Screening Criteria for Assessing Potential Seismic Vulnerabilities of Spent Fuel Pools at Decommissioning Plants – December 3, 1999 NRC Memorandum

Summary of NRC Draft

To increase the efficiency and effectiveness of decommissioning regulations, the NRC staff has engaged in rulemaking activities that would reduce the need to routinely process exemptions once a plant is permanently shut down. The December 3, 1999 memorandum from W. Huffman to S. Richards (Reference 1) provides a summary of the staff's current concerns regarding a screening criteria for assessing potential seismic vulnerabilities to spent fuel pools (SFP) at decommissioning plants. Attachments to this memorandum contain suggested enhancements to the proposed seismic checklist and also excerpts from an independent technical review by Dr. Robert Kennedy. The report by Kennedy endorsed the feasibility of the use of a seismic screening concept. The Kennedy report identified eight sites for which the seismically induced probability of SFP failure is greater than 3.0×10^{-6} using the LLNL 93 hazard data.

The seismic risk of failure of the spent fuel pool can be estimated by rigorously convolving a family of fragility curves with a family of seismic hazard curves (Reference 2), or by simplified approximation methods. Two simplified methods are described in the attachments to the December 3, 1999 memorandum (Reference 1).

The first simplified method was presented by the Staff in their preliminary draft of June 16, 1999 (Reference 3). This method is based on use of the SFP high confidence low probability of failure (HCLPF) value and the simplifying assumption that the conditional probability of SFP failure is about a factor of 20 less than the annual probability of exceeding the SFP HCLPF value. Given that the SFP HCLPF value is more than or equal to three times the SSE (and less than 10^{-5}) then the SFP failure frequency should be less than 5×10^{-7}. This simplified method is based on use of peak ground acceleration (PGA) curves.

The second simplified method was suggested by Kennedy and is based on use of spectral acceleration (S_a) rather than PGA. Kennedy states that damage to structures, systems, and components (SSCs) does not correlate well to PGA ground motions but correlates much better with spectral accelerations between 2.5 and 10 Hz at nuclear power plants. Based on previous studies Kennedy proposes to screen SFPs based on use of the peak spectral acceleration (PSA) HCLPF seismic capacity of 1.2g. This value is equivalent to 0.5g PGA. This simplified approach is based on calculating the 10% conditional probability of failure capacity ($C_{10\%}$) given the PSA value of 1.2g. Using Equation 6 in the Reference 1 attachment results in a $C_{10\%}$ S_a value of 1.82g. The annual probability of exceeding this value at 10, 5 and 2.5 Hz is then calculated using the LLNL hazard results. These value are then multiplied by 0.5 and the highest of the 10, 5, and 2.5 Hz results is used as the SFP failure probability. For example, the $C_{10\%}$ at 5 Hz is 1.82g or about 56.8 cm/sec spectral velocity. For LLNL site 1, the annual probability of exceeding 56.8 cm/sec is about 2.0×10^{-6}. This value is multiplied by 0.5 which results in a SFP failure probability for site 1 of about 1.0×10^{-6}. This same calculation is performed at 10 and 2.5 Hz.

Based on comparisons made by Kennedy he concludes that simplified method 1 (Reference 3) underestimates the seismic risk by factors of 2.3 and 3.5 for Vermont Yankee and Robinson respectively. Using simplified method 2 the seismic risk is overestimated by 20% and 5% respectively for these two cases.

Kennedy noted that in his judgement it will be necessary to have seismic fragility HCLPF computations performed on at least six different aboveground SFPs with walls not supported by soil before HCLPF screening levels can be established for these SFPs.

Recommendation Number 4 of the December 3, 1999 memorandum requested that industry provide input concerning:

f. the list of high hazard sites,
g. a credible ground motion description at which the seismic hazard frequency is low enough at these sites, and
h. plant specific seismic capacity evaluations using credible ground motion descriptions at these sites.

Recommendation Number 5 requests that industry propose treatment of sites West of the Rocky Mountains.

Preliminary Industry Comments

Industry concurs that use of a seismic screening checklist is an excellent approach to plant-specific seismic assessments. In addition, we will incorporate into our earlier seismic checklist those suggestions presented in Recommendation numbers 1, 2, and 3 to the December 3, 1999 memorandum.

With respect to the simplified methods to estimate seismic failure frequency of SFP failure the method proposed by Kennedy appears to be reasonable.

In the recommendations section of the 12/3/99 memorandum (Reference 1) some actions by industry are proposed. Recommendation Number 4.b requests that industry recommend a credible ground motion description at which the seismic hazard frequency is low enough at these "high" hazard sites. These "high" hazard sites were identified based on use of the Kennedy simplified SFP failure methodology and the LLNL 1993 hazard results. The response to Recommendation Numbers 4.a and 4.c are dependent on the resolution of 4.b.

Comments on Recommendation Number 4.b

1. Using the Kennedy simplified SFP failure methodology $C_{10\%}$ values are determined at 10, 5, and 2.5 Hz. At 5 Hz the spectral acceleration value is 1.82g or about 56.8 cm/sec.

2. The PSA values associated with these $C_{10\%}$ values are consistent with spectral values which describe the San Onofre and Diablo Canyon SSEs, i.e., large magnitude, near field earthquakes.

3. The issue of large earthquakes occurring near EUS NPPs was resolved by the Charleston Issue (SECY-91-135, Reference 4). As stated in SECY-91-135, "Large 1886 Charleston-size earthquakes, greater than or equal to magnitude 6.5, are not significant contributors to the seismic hazard for nuclear facilities along the eastern seaboard outside the Charleston region. This result is consistent with the results emerging from the ongoing studies of earthquake-induced liquefaction features along the eastern seaboard. These studies have found no evidence of large prehistoric earthquakes originating outside the South Carolina region. Thus the issue of the Charleston earthquake occurring elsewhere in the eastern seaboard is considered to be closed."

4. Credible, versus not credible in terms of annual probability, is typically associated with greater than about 10^{-6} (credible) and 10^{-6} or less (not credible). Within the context of the Kennedy simplified SFP failure methodology, if the annual probability of exceeding the screening level value (for example 56.8 cm/sec at 5 Hz) times 0.5 is less than 10^{-6}, then only the seismic checklist must be satisfied. Implicit in this approach is that the probabilistic estimates at the $C_{10\%}$ level are credible.

5. For a site to be screened out the $C_{10\%}$ value should be on the order of 10^{-6}. Figure 1 (attached) shows the 5 Hz spectral acceleration values associated with the 10^{-6} LLNL results at each of the 69 sites. As can be seen, for site number 36 (which in Table 3 of the Kennedy report is the site with the highest SFP failure frequency) the 10^{-6} spectral acceleration is about 7,700 cm/sec^2 or about 245 cm/sec. As stated previously, 57 cm/sec is consistent with 5 Hz spectral velocities associated with a magnitude 6.6 earthquake 8 km from the site (San Onofre SSE), therefore these predicted groundmotions must be associated with a very large earthquake, greater than magnitude 6.5, very near to the site – which is counter to the conclusions of SECY-91-135. Other values at other sites are equally incredible. Based on these results, it is concluded that the LLNL results, at the probability/ground motion levels of interest, are deterministically incredible and therefore their use in screening is questionable. Figure 2 (attached) shows the 5 Hz spectral acceleration values associated with the 10^{-6} EPRI results. As can be seen, the EPRI results, at the probability/ground motion levels of interest, are credible, and consistent with SECY-91-135.

6. Figure 3 (Figure 2 from NUREG-1488, Reference 5) illustrates the problems associated with the LLNL results at high ground motions/low annual probabilities. As can be seen from Figure 3, at high probabilities there is reasonable agreement between LLNL and EPRI. However, the slope of the LLNL results at high ground motions is too shallow. The effect of this shallow slope is to predict incredible ground motions at credible probability levels.

7. Based on this review, industry contends that it would be appropriate to only use EPRI results in the SFP seismic screening analysis. We believe this to be reasonable in light of the difficulties associated with the LLNL results at low probabilities. The effect of using only the EPRI results is shown in column 3 of Table 3 in the Kennedy report (Reference 1). As can be seen, only 1 plant would be required to perform further analyses. However, because both LLNL and EPRI are considered to provide valid results, it is proposed that the results from each study be geometrically averaged such that equal weight is provided the results from each study. Arithmetic averaging is

considered unacceptable in light of the difficulties associated with the LLNL results. Figure 4 provides the results of geometrically averaging the LLNL and EPRI results.

Comments on Recommendation Number 4.a

Based on Figure 4 about 6 sites would be preliminarily screened in due to exceeding the 10^{-6} criterion. One of the 6 sites is Shoreham. If these screened in SFPs are above ground then further analyses will be required.

Comments on Recommendation Number 4.c

It is industry's understanding of Section 4.2 of the Kennedy report that given that a plant satisfies the seismic screening checklist then the SFP is likely to have a seismic capacity higher than the screening level capacity. If plant-specific information is conveniently available, additional seismic capacity values will be developed in a manner similar to that described in NUREG/CR-5176.

Comments on Recommendation Number 5

A response to the NRC Recommendation Number 5 requesting industry to provide "Proposed treatment of sites West of the Rocky Mountains" will be provided later. However, as a result of detailed deterministic investigations at and around each site, a better understanding of the sources and causes of earthquakes is developed in the licensing of Western U.S. (WUS) plants. Therefore, it would be reasonable to describe the credible ground motion for WUS sites deterministically.

References:

1. Memorandum, W. Hauffman to S. A. Richards, USNRC, Screening Criteria for Assessing Potential Seismic Vulnerabilities of Spent Fuel Pools at Decommissioning Plants, December 3, 1999.

2. NUREG/CR-5176, Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear Power Plants, Lawrence Livermore National Laboratory, January 1989.

3. USNRC, Preliminary Draft Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants, June 16, 1999.

4. SECY-91-135, Conclusions of the Probabilistic Seismic Hazard Studies Conducted for Nuclear Power Plants in the Eastern United States, May 14, 1991.

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

5f December 28, 1999 Kennedy Letter

Structural Mechanics Consulting, Inc.

Robert P. Kennedy 18971 Villa Terrace, Yorba Linda, CA 92686 (714) 777-2163

December 28, 1999

Dr. Charles Hofmayer
Environmental & Systems Engineering Division
Brookhaven National Lab
Building 130, 32 Lewis Road Upton, NY 11973-5000

Subject: Additional Documents Concerning Seismic Screening and Seismic Risk of Spent Fuel Pools For Decommissioning Plants

Dear Dr. Hofmayer:

I have reviewed the December 3,1999 memorandum from W. Huffman to S. Richards entitled *Screening Criteria for Assessing Potential Seismic Vulnerabilities of Spent Fuel Pools at.Decommissioning Plants*. I have also reviewed the "Industry Comments" on the material presented in this memorandum. Lastly, I reviewed Revision I of the *Industiy Seismic Screening Criteria* dated December 13, 1999.

I concur with the adequacy of the *Industry Seismic Screening Criteria* presented in Revision I for the vast majority of Central and Eastern US (CEUS) sites. So long as Screening Items I through 9 are satisfied, the seismic risk of spent fuel pool failure to contain water for these sites should be so low as to not warrant further assessment. The addition of Screening Item 4 in Revision I removes my concern about the previous draft. For spent fuel pool walls and floor slab not supported by soil, Screening Item 4 requires a structural assessment of the pool walls and floor slab out-of-plane shear and flexural capabilities be performed and compared to the realistic demands expected to be generated by seismic input equal to approximately three times the site SSE input. In order to demonstrate a HCLPF capacity in excess of approximately 3 SSE, this assessment should be performed with the degree of conservatism defined for the Conservative Deterministic Failure Margin (CDFM) method in EPRI 6041.

Spent fuel pools at a few higher seismic hazard sites in the CEUS and all Western US sites should be further evaluated beyond this screening criteria. I concur with the approach presented on page 4 of the "Industry Comments" for defining these few higher seismic hazard CEUS sites. Based on Figure 4 of the "Industry Comments", it appears that no more than 4 CEUS sites (excluding Shoreham) would fall into this higher seismic hazard category.

Either Seismic Margin or Seismic Fragility HCLPF capacity estimates should be made for spent fuel pools at decommissioning plants in each of the following cases:

1. Out-of-plane flexural and shear capacity of aboveground spent fuel pool walls and floors not supported by soil.

2. Spent fuel pools which do not pass the Revision I *Industry Seismic Screening Criteria*.

3. A few higher seismic hazard CEUS sites and all Western sites.

For the above situations where HCLPF capacity assessments should be made, I understand that Goutam Bagehi and Bob Rothman of the NRC have recommended that a plant coming in for decommissioning which can show that their spent fuel pool structural resistance has a HCLPF value of 3*SSE for CEUS sites and 2*SSE for West Coast sites has demonstrated an adequately low seismic risk for their spent fuel pool. This recommended approach represents a reasonable engineering approach with which I concur.

I believe the approach outlined above is a practical approach for demonstrating the seismic risk of spent fuel pools at decommissioning plants is very low. Please contact me if you desire further discussion.

Sincerely

Robert Kennedy

cc. Mr. Goutam Bagchi
Dr. Nilesh Chokshi

5g 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.

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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.

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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 our-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 1.0E-06.)

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 1×10^{-6} per year.

5h Other Seismic Stakeholder Interactions

1. A member of the public raised a concern about the potential effects of Kobe and Northridge earthquakes related to risk-informed considerations for decommissioning during the Reactor Decommissioning Public Meeting on Tuesday, April 13, 1999, in Rockville, MD.

Stakeholder Comment

> "I guess I'd like to direct my questions to the seismological review for this risk-informed process. And first of all, did any of the NUREGs that you looked at take into account new information coming out of the Kobe and Northridge events? I think that what we need to be concerned with is dated information. 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 two NUREGs mentioned by a member of the public 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 for central and eastern U.S. nuclear power plants for the NRC. The results of these newer studies indicated lower seismic hazards for the plants than the earlier studies estimated. If the probabilistic hazard studies were to be performed again, hazard estimates for most sites would probably be reduced further than the LLNL 1993 study due to: new methods of eliciting information, newer methods of sampling hazard parameters' uncertainties, better information on ground motion attenuation in the U.S. and a more certain understanding of the seismicity of the central and eastern U.S.

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 safe shutdown earthquake (SSE). This ground motion was used in the design and analysis of the plant.

The determination of the SSEs had to follow the criteria and procedures required by NRC regulations and apply 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 well beyond their design basis and far 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 individual 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 do 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 do not cause anomalous amplification, therefore, there is no “new information gained from the Kobe and Northridge events,” which raises 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.

2. 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. There was also another concern about the effect of aging on the spent fuel pool liner plate and the reinforced concrete pool structure.

   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. These 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. This is included in the seismic checklist.

3. During the July workshop, members of the public raised concerns about the effect of aging on the spent fuel pool liner plate and the reinforced concrete pool structure.

   Irradiation-induced degradation of steel requires high neutron fluency, which is not present in the spent fuel pools. Operating experience has not indicated any 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 spent fuel pool structures are expected to have this increased strength at the time of their decommissioning. 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; 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 spent fuel pool structure to identify cracks, spalling of concrete, etc., is also recommended as a part of the seismic checklist. Significant degradation of reinforced concrete structures would take more than 5 years or so, the time necessary to lose decay heat in the spent fuel. 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.

The short-term period of concern for the beyond-design-basis seismic event can be considered to last no more than several days. Any seepage of water during this time will not degrade the capacity of concrete. Degradation of concrete strength would require loss of cross-section of reinforcing bars due to corrosion, and a period of several days is too short to cause such a loss.

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, it can be assumed that the spent fuel pool structure will be at its full strength at the initiation of a postulated beyond-design-basis event.

Appendix 6 November 12, 1999 Nuclear Energy Institute Commitment Letter

NEI

NUCLEAR ENERGY INSTITUTE

Lynnette Hendricks
DIRECTOR
PLANT SUPPORT
NUCLEAR GENERATION DIVISION

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 7 Stakeholder Interactions

1. Introduction

The technical staff reviewed and evaluated available technical information and methods to use as the risk-informed technical basis for reviewing decommissioning exemption requests and rulemaking related to emergency preparedness, safeguards, indemnification, and other areas. When the draft report was released for public comment in June,1999, stakeholders identified concerns, which were addressed for inclusion in the final report. The early stakeholder input has improved the overall quality of the report. Meetings held with the stakeholders are provided below. Afterward, stakeholder comments in various technical areas and how the staff addressed them are discussed.

Public meetings on the Technical Working Group Study

March 17, 1999	Commission meeting in Rockville, MD
April 13, 1999	Stakeholder meeting with NRC staff in Rockville, MD
May 5, 1999	Stakeholder meeting with NRC staff in Rockville, MD
June 7, 1999	Stakeholder meeting with NRC staff in Rockville, MD
June 8, 1999	Stakeholder meeting with Sam Collins in Rockville, MD
June 21, 1999	Pre-workshop stakeholder meeting with NRC staff in Rockville, MD
July 15-16, 1999	Workshop on decommissioning plant spent fuel pool accident risk in Gaithersburg, MD
November 3, 1999	Stakeholder meeting with Sam Collins in Rockville, MD
November 5, 1999	ACRS meeting in Rockville, MD
November 8, 1999	Commission meeting in Rockville, MD
November 19, 1999	Stakeholder meeting with NRC staff in Rockville, MD

2. Probabilistic Risk Assessment (PRA)

An industry stakeholder raised the concern that the PRA was too conservative and that some of the assumptions were unrealistic. The staff refined the PRA analysis, incorporating industry commitments, and subjected the results to an independent technical review. The results are summarized in Chapter 3. A more detailed description of the risk analysis is presented in Appendix 2.

3. Human Reliability Analysis

Industry stakeholders raised a concern that the June 1999 draft report did not give sufficient credit for operator actions in the area of human reliability analysis (HRA). Specifically, industry stated that the NRC draft report did not reflect the potential for actions such as self-checking, longer reaction times available, management oversight, design simplicity, second crew member check, additional shift attention in recovery, or additional cues causing increased attention.

The staff enlisted the support of HRA experts, who independently reviewed the analysis in the June 1999 draft report and provided refinements to the HRA analysis. The HRA results were also subjected to an independent technical review. This topic is discussed in Appendix 2.

4. Heavy Loads

Industry stakeholders raised a concern that the heavy load risk assessment in the draft report did not give sufficient credit for NUREG-0612 actions and used the conservative upper bound values.

To address these concerns, the staff employed more recent Navy data to requantify the fault tree, included the mean value estimate for compatibility with Regulatory Guide 1.174 and addressed industry voluntary commitment to Phase II of NUREG-0612. The results and conclusions are discussed in Chapter 3.3.6 and Appendix 2 (section 2c).

5. Seismic Assessment

To take credit for the seismic design margins existent in spent fuel pools, the staff sought an appropriate method to identify potential structural vulnerabilities without having to perform a detailed fragility review. At a July 15-16, 1999 public workshop, industry proposed development of a simple spent fuel pool seismic checklist as a way of assessing seismic vulnerabilities without performing quantifying analyses.

In a letter dated August 18, 1999, NEI submitted a “seismic checklist” for screening. The staff considered it an acceptable alternative to plant specific fragility reviews; provided that some deficiencies in the checklist proposed by NEI were corrected. After these concerns were identified to NEI, a revised checklist was submitted in a letter dated December 13, 1999. Details of the seismic checklist and other seismic issues are provided in Chapter 3.4.1 and Appendices 2 (section 2b) and 5.

1.06 Other Seismic Stakeholders Interactions

Members of the public raised other seismic concerns at the Reactor Decommissioning Public Meeting on Tuesday, April 13, 1999 and during the July workshop. The concerns raised related to: the potential effects of the Kobe and Northridge earthquakes on risk-informed considerations for decommissioning; the hazard of the fuel transfer tube interacting with the pool structure during an earthquake; and the effect of aging on the spent fuel pool liner and the reinforced concrete pool structure. These concerns are addressed in Appendix 5.h.

7. Criticality

A public stakeholder concluded that the June 1999 draft report did not address the potential for a criticality accident in the SFP of a decommissioned plant. The subject was also raised by a member of the public during the November 8, 1999 Commission meeting.

The staff examined the mechanisms by which a criticality accident could occur to assess the potential for criticality, the consequences, and the likelihood of a criticality event. The results were subjected to an independent contractor review where additional mechanisms were proposed and examined. The results are presented in Appendix 3.

8. Thermal-Hydraulic Assessment

Industry stakeholders raised a concern that the thermal-hydraulic assessment in the June 1999 draft report used overly conservative adiabatic heat-up calculations and a maximum clad temperature that was too conservative for the zirconium ignition temperature.

We refined the thermal-hydraulic analysis presented in the draft report. The results of the analysis are included in Appendix 1.

9. Partial Draindown and Exothermic Reaction of SFP

An industry stakeholder stated that we did not consider the implications of a partial draindown as being as serious as, or worse than, a complete draindown. The stakeholder also stated that the draft report did not address the potential for a hydrogen explosion resulting from an exothermic reaction between steam and zirconium. A discussion of these topics are found in Appendix 1.

10. Impact of Decommissioning on Operating Units

A public stakeholder stated that we did not consider the impacts on operating units of removing the water from the SFP at a decommissioning site, such as Millstone and San Onofre.

It is recognized that the loss of water in a decommissioning SFP (note: this concern relates only to reduced quantities of water in the SFP and not with zirconium fires) has the potential to have an impact on adjacent operating units at the same site. For a site where there are no shared systems, components, or structures between plants, the major concern would be a harsh radiation environment which would cause increased radiation doses to operators in the plant. For plants where systems, components, or structures are shared between plants, the concern would be a harsh environment (e.g. radiation or temperature) which could cause concerns for operators and/or equipment which might be unable to perform its safety function due to the harsh environment being greater than its design basis. While these concerns are recognized, the staff believes that with the low probability of the uncovery of spent fuel, as discussed in Chapter 3 and Appendix 2 of this report, the risks associated with this event are acceptable.

11. Safeguards

A public stakeholder stated that the draft report did not address the potential or threat for vehicle-borne bombs. This issue is addressed in Chapter 4.3.2.