DRAFT FOR COMMENT

DRAFT FOR COMMENT

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

February 2000

DRAFT FOR COMMENT

Formatted Version, Rev 1 1/19/00

Technical Working Group:

Group Leaders:	Diane Jackson, DSSA/SPLB
	Vonna Ordaz, DSSA/SPLB
	George Hubbard, DSSA/SPLB
	Tanya Eaton, DSSA/SPLB

Technical Expertise:

Seismic	Goutam Bagchi, DE
Maintenance Rule	Edward Ford, DIPM/IQMB
Quality Assurance	Ken Heck, DIPM/IQMB
Consequences	Jason Schaperow, RES/DSARE/SMSAB
Safeguards	Robert Skelton, DIPM/IOLB
	Richard Rosano DIPM/IOLB
Fire Protection	Edward Connell, DSSA/SPLB
	Tanya Eaton, DSSA/SPLB
Risk Input	Mark Rubin
	Glenn Kelly, DSSA/SPSB
	Michael Cheok, DSSA/SPSB
Human Reliability	Gareth Parry, DSSA
Criticality	Larry Kopp, DSSA/SRXB
	Anthony Ulses, DSSA/SRXB
Thermal Hydraulics	Joseph Staudenmeier, DSSA/SRXB
	Christopher Boyd, RES/DSARE/SMSAB
Heavy Loads	Edward Throm, DSSA/SPSB
	Brian Thomas, DSSA/SPLB
SFP Operations	Christopher Gratton, DSSA/SPLB
EP	James O'Brien, DIPM/IOLB
	Roger Pederson, DIPM/IOLB

Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants

Table of Contents

Executive Summary 1

Introduction 3

2.0 Risk Informed Decision Making 4
- 2.1 Principles of Regulatory Guide 1.174 5
  - 2.1.1 Defense-in-Depth 6
  - 2.1.2 Safety Margins 6
  - 2.1.3 Impact of Proposed Changes 7
  - 2.1.4 Implementation and Monitoring Program 7

3.0 Risk Assessment of Spent Fuel Pools at Decommissioning Plants 7
- 3.1 Basis and Findings of SFP Risk Assessment 9
- 3.2 Characteristics of SFP Design and Operations for a Decommissioning Plant 10
- 3.3 Internal Event Scenarios Leading to Fuel Uncovery 14
  - 3.3.1 Loss of Offsite Power from Plant-Centered and Grid Related Events 14
  - 3.3.2 Loss of Offsite Power from Severe Weather Events 15
  - 3.3.3 Internal Fire 16
  - 3.3.4 Loss of Cooling 17
  - 3.3.5 Loss of Coolant Inventory 17
- 3.4 Beyond Design Basis Spent Fuel Pool Accident Scenarios (External Events) 20
  - 3.4.1 Seismic Events 20
  - 3.4.2 Aircraft 21
  - 3.4.3 Tornadoes 22
  - 3.4.4 Criticality in Spent Fuel Pool 23

4.0 Implications of Spent Fuel Pool Risk For Regulatory Requirements 25
- 4.1. Summary of the Technical Results 26
- 4.2 Risk Impact of Specific Design and Operational Characteristics 27
- 4.3. Implications for Regulatory Requirements Related to Emergency Preparedness, Security and Insurance 30
  - 4.3.1 Emergency Preparedness 31
  - 4.3.2 Security 32
  - 4.3.3 Insurance 33

5.0 References 35

6.0 Acronyms 37

Appendix 1 Thermal Hydraulics A1-1

Appendix 2 Frequency of Fuel Uncovery A2-1

Appendix 3 Criticality A3-1

Appendix 4 Consequence Assessment from Zirconium Fire A4-1

Appendix 5 Enhanced Seismic Check List and Supporting Stakeholder Documentation . . A5-1

Appendix 6 November 12, 1999 Nuclear Energy Institute Commitment Letter A6-1

Appendix 7 Spent Fuel Pool Human Reliability Analysis A7-1

Appendix 8 Stakeholder Interactions A8-1

List of Tables

Table 3.1 Spent Fuel Pool Cooling Risk Analysis Frequency of Fuel Uncovery (per year).......13

List of Figures

Figure 3.1 Assumed Spent Fuel Pool Cooling System...11

Executive Summary

This report documents an evaluation of spent fuel pool accident risk 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 reduce unnecessary regulatory burden, improve efficiency and effectiveness, and establish a consistent, predictable process that will maintain safety and enhance public confidence. The report was initiated when industry asked the NRC to consider whether the risk from decommissioning plants was low enough to justify generic regulatory relief in the areas of emergency planning, indemnification and safeguards.

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, we have requested heatup evaluations of the spent fuel cooled only by air. This criterion was used because of national laboratory studies that had identified the potential concern for a significant offsite radiological release from a zirconium fire which may 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 conservative limit to ensure no radiological release.

In March, 1999, the staff formed a technical working group to evaluate spent fuel pool accident risk 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 an industry/NRC/public 2 day workshop that was held in July, 1999. Information gained at the workshop and through other stakeholder interactions was constructive in completing the 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 PRA assumptions and treatment. The PRA results were requantified to take into account the industry commitments to reduce risk vulnerabilities.

This report contains the results of our effort. It includes three main outputs. The first is a discussion in Chapter 2 on how risk informed decision making is being applied to decommissioning plants. The second is the actual risk assessment of SFPs at decommissioning plants in Chapter 3. The third provides the implications of SFP risk on regulatory requirements in Chapter 4, and outlines where an industry initiative may be useful in improving the generic study.

As described in Chapter 2, the large early release frequency (LERF) acceptance guideline in Regulatory Guide (RG) 1.174 [Ref. 1] recognizes the need for lower frequencies in the absence of a physical means, such as a containment, of retaining the fission products. In a letter dated November 12, 1999 [Ref. 2], the ACRS suggested that the end state of uncovery of top of fuel was an appropriate PRA surrogate for zirconium fire frequency, and that comparison with LERF would be acceptable for risk-informed decision making, even though the correlation is not perfect.

The risk estimates contained in Chapter 3 demonstrate that a zirconium fire can occur during an extended period after shutdown (up to five years), depending on fuel burnup and rack configurations, if fuel uncovery were to occur. The consequences of such an event would be severe, and the zirconium fire frequencies presented in this report are comparable to the frequencies of large releases from some operating reactors. However, the requantified PRA demonstrates that if operation of the decommissioned plant is carried out in accordance with the commitments proposed by the industry and the other constraints outlined in this report are followed, such as the seismic check list, then the LERF guidelines can be met.

Chapter 4 points out that when other factors are taken into account as described in RG 1.174, such as defense in depth, maintaining safety margins, and performance monitoring, the staff has concluded that after one year following final shutdown, there is reasonable assurance that a zirconium fire will not occur such that the emergency planning requirements can be relaxed to a minimum baseline level. Any future reduction of the one year critical decay time would be contingent on improvements in the human reliability analysis. That is, any licensee wishing to gain relief from the EP requirements prior to the one year post-shutdown, would need to demonstrate a more robust reaction time than that credited in the HRA for this study. Chapter 4 also covers the need for continued indemnification requirements while the threat of a zirconium fire exists, and offers the possibility that an industry initiative to improve the thermal-hydraulic calculational methodology could result in shortening the generic 5 year window of vulnerability. And finally, Chapter 4 includes a discussion on how the risk insights contained in this report can by employed to assess the vulnerabilities to sabotage, and concludes that any reduction in security provisions would be constrained by the target set, such that some level of security is required as long as the fuel in the SFP is exposed to a sabotage threat.

In summary, this report provides the basis for determining the regulatory requirements for decommissioning plants using risk-informed decision making. It recognizes that some aspects of the regulations such as 10 CFR 20 [Ref. 3] are not amenable to this kind of analysis. However, it provides an authoritative and definitive treatment of SFP risk at decommissioning plants as it relates to emergency planning, insurance, and security requirements, and can be extrapolated to other appropriate areas of consideration such as shift staffing and fitness for duty. And finally, it points out other areas of consideration for bringing coherency to future rulemaking.

1. Introduction

The current body of NRC regulations pertaining to reactors (10 CFR 50) [Ref. 1] is primarily directed towards the safety of operating units. As reactors have reached permanent shutdown condition and entered decommissioning status, industry and the NRC have been faced with establishing the appropriate requirements and regulatory oversight necessary to provide adequate protection to the public.

Decommissioning plants have requested exemptions to certain regulations as a result of their permanently defueled condition. Areas where regulatory relief has been requested in the past include exemptions from offsite emergency planning (EP) requirements, Price Anderson Insurance provisions and physical security. 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 accidents involving spent fuel. These fuel assemblies can be stored in the spent fuel pool 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 embarked on an effort to develop a 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 (applying both quantitative and qualitative insights) and was developed from detailed analytical studies in the areas of thermal hydraulics, core physics, systems analysis, seismic and structural analysis and external hazards assessment. The focus of the risk assessment was to identify the scenarios, likelihoods and consequences that could result in loss of spent fuel pool water inventory and cooling of the spent fuel assemblies. For some period after reactor shutdown, it is possible for the fuel to heat up to the point where rapid oxidation and burning of the 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 the present assessment. Quality assessment of the staff's preliminary analysis has been aided by a blue ribbon panel of 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 systems as well as practices and procedures necessary to ensure high levels of operator performance during off normal conditions. The report concludes that with the imposition of voluntary industry initiatives and some additional staff requirements in the areas of performance monitoring and seismic validation, 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 contains is divided into three main parts. The first part is a discussion in Chapter 2 on how risk informed decision making can be applied to decommissioning plants. In Chapter 3, the staff presents the risk assessment conducted on the SFPs for decommissioning plants. In Chapter 4 of this report, the findings of SFP risk for a decommissioning plant will be assessed against each of the safety principles and objectives discussed above.

2.0 Risk Informed Decision Making

The regulatory framework developed 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, for example such as core damage in an operating reactor. Related aspects of these methods can go on to assess the timing and mode of containment failure, radioactive releases to the environment and postulated health effects.

Subsequent to issuance of the PRA Policy Statement, the agency published Regulatory Guide (RG) 1.174 [Ref.2] which contained general guidance and criteria for application of PRA to the regulation of nuclear reactors. The criteria 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 baseline frequencies and for the changes that can be allowed due to regulatory decisions. For example, if the baseline CDF for a plant is below 1E-4 per year, plant changes can be approved which increase CDF by up to 1E-5 per year. If the baseline LERF is less than 1E-5 per year, plant changes can be approved which increase LERF by 1E-6 per year.

For decommissioning plants, the risk is due primarily to the possibility of a zirconium fire associated with the spent fuel rod cladding[1]. The consequences of such an event do not equate exactly to either a core damage accident or a large early release[2]. Zirconium fires in spent fuel pools potentially have more severe consequences than an operating reactor core damage accident, because there are multiple cores involved, and because there is no

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

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

containment surrounding the SFP to mitigate the consequences. On the other hand, they are somewhat different than a large early release, because the accidents progress slowly enough to allow ample warning for offsite protective actions, and because the absence of Iodine isotopes leads to fewer prompt fatalities. As a result, the criteria of RG 1.174 cannot be applied directly to the risk of a decommissioning plant without further thought.

Even though the event progresses more slowly than an operating reactor LERF and the isotopic makeup is somewhat different, the risk assessment consequence calculations performed by the staff[3] 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 health effects (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 criteria of 1E-5 per year or a change not to exceed 1E-6 per year provide appropriate frequency criteria for a decommissioning plant SFP risk, and a useful tool to assess features, systems and operator performance needs of a decommissioning pool.

2.1 Principles of Regulatory Guide 1.174

As discussed in RG 1.174, the results of quantitative risk assessment is only one tool utilized in risk informed decision making. Due to limitations in methods and data it must be complemented by other safety principles. The RG articulates the following safety principles which should be applied to the decommissioning case, in addition to the numerical objective described above.

In RG 1.174, the NRC gave the following five principles of risk-informed regulation:

- 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.
- 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 on 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

---

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

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 these principles are satisfied as demonstrated by the staff's safety assessment is provided in Chapter 4.

2.1.1 Defense-in-Depth

The defense-in-depth philosophy applies to the operation of the spent fuel pool, whether at an operating plant or in a decommissioning plant. Traditionally defense in depth means that for various credible accident scenarios, there is more than one system or set of actions that will recover from the incident before a serious outcome occurs. This could mean that there is more than one source of cooling water or that pump makeup 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. Because the essentially quiescent (low temperature, low pressure) initial state of the spent fuel pool and the long time 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 long evolution of most SFP accident scenarios allows for reasonable human recovery actions to respond to system failures. The specific design and operational features of the SFP, industry commitments and staff requirements that ensure that SFP defense in depth is maintained, is provided in Chapter 4.

2.1.2 Safety Margins

Maintenance of sufficient safety margins is a fundamental principle of RG 1.174. A safety margin can relate to the difference between the expected value of some physical parameter (temperature, pressure, stress, reactivity) and the point at which adequate performance is no longer assured. For example a containment pressure calculation that shows 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 28 feet of water at 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 how the provisions that ensure the SFP maintains adequate 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 baseline and change criteria for LERF in RG 1.174 to assess the impact and acceptability of SFP risk in decommissioning plants. Chapters 3 and 4 discusses the design and operational characteristics of the SFP that must be relied upon to produce the low baseline risk results. These are identified in the context of industry commitments as well as staff requirements.

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 might include 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 effectiveness of onsite emergency response, and the plans for communication with offsite authorities. In many areas the implementation and monitoring may already be accomplished by utility programs such as those developed under the maintenance rule [Ref. 3]. Chapter 4 discusses the additional implementation and monitoring activities that are necessary to achieve the low SFP risk estimates of this report and support the safety principles.

3.0 Risk Assessment of Spent Fuel Pools at Decommissioning Plants

As discussed in the background section of this paper, the risks and vulnerabilities from a decommissioning plant are very different from an operating reactor. 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 both 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, swell and burst. The breach in the clad could 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 reactor combined with the 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 will result in a significant release of the fission products contained in the spent fuel, which will 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 of this age in and by themselves (without zirconium fire) release only small quantities of radionuclides and would only be of concern for onsite effects.

Based upon the preceding insights the staff conducted its risk evaluation to focus on the likelihood of scenarios that could result in loss of pool water and fuel heat up to the point of rapid oxidation. Since the decay time at which air cooling alone is sufficient to prevent 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 it reaches a point where rapid oxidation will not occur with complete loss of water. After a decay period that precludes fuel heat up to zirconium fire conditions, no significant risk remains. Preliminary calculations by the staff (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 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 offsite 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 extremely complex heat transfer mechanisms and chemical reactions occurring in the fuel assemblies. This analytical approach understates the time that is available for possible operator recovery of SFP events prior to initiation of 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 top of fuel uncovery is the end point of the analysis, or is total 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 requirements. In its letter of November 12, 1999 [Ref. 1], the Advisory Committee on Reactor Safeguards (ACRS) recommended that the end state of top of fuel uncovered 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.

Previous to the staff's preliminary risk assessment, the most extensive work to date was in support of Generic Safety Issue (GSI) 82, "Beyond Design Basis Accidents for Spent Fuel Pools" [Ref. 2]. This report assessed the 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 GSI 82 work in its assessment, it was concluded that because of significant differences between operating and decommissioning plant spent fuel pools cooling systems, a complete assessment of SFP risk 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 recommendations (relied upon in the risk assessment) and a discussion of how the decision criteria in Chapter 2 are satisfied is discussed in Chapter 4. Conclusions on how the SFP risk insights and decision criteria apply to potential changes in emergency planning, insurance, and physical security 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 1, 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 very different than those in operation when the plants were operating reactors. Modeling information was determined from both site system walkdowns as well as limited discussions with the decommissioning plant staff. Since limited information was available 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 improved information from industry on SFP design and operating characteristics for a decommissioning plant, as well as a number of 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 Offsite Power-Plant centered and grid related events
- Loss of Offsite Power- 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 re-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 include the modeling details for the cask drop, aircraft impacts, seismic and tornado missile assessments. Input and comments from stakeholders was 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 on next page) as being located in the spent fuel pool (SFP) area and consisting of motor-driven pumps, a heat exchanger, an ultimate heat sink, a makeup tank, filtration system and isolation valves.

Suction is taken via one of the two pumps on the primary side from the 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 back into the discharge line. A manually operated makeup system (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. 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 operating staff's during those visits, the staff also made the following assumptions that are believed to be representative of a typical decommissioning facility:

- The site has two operable firewater pumps, one diesel-driven and one electrically-driven from offsite power.
- We assume the makeup capacity (with respect to volumetric flow) 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)]

  We therefore assumed that for the larger loss-of-coolant inventory accidents, water addition through the makeup pumps does not successfully mitigate the loss of inventory event unless the source of inventory loss is isolated.

- The fuel handlers perform walkdowns of the SFP area once per shift (8- to 12-hour shifts). A different crew member is assumed for the next shift. We 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.

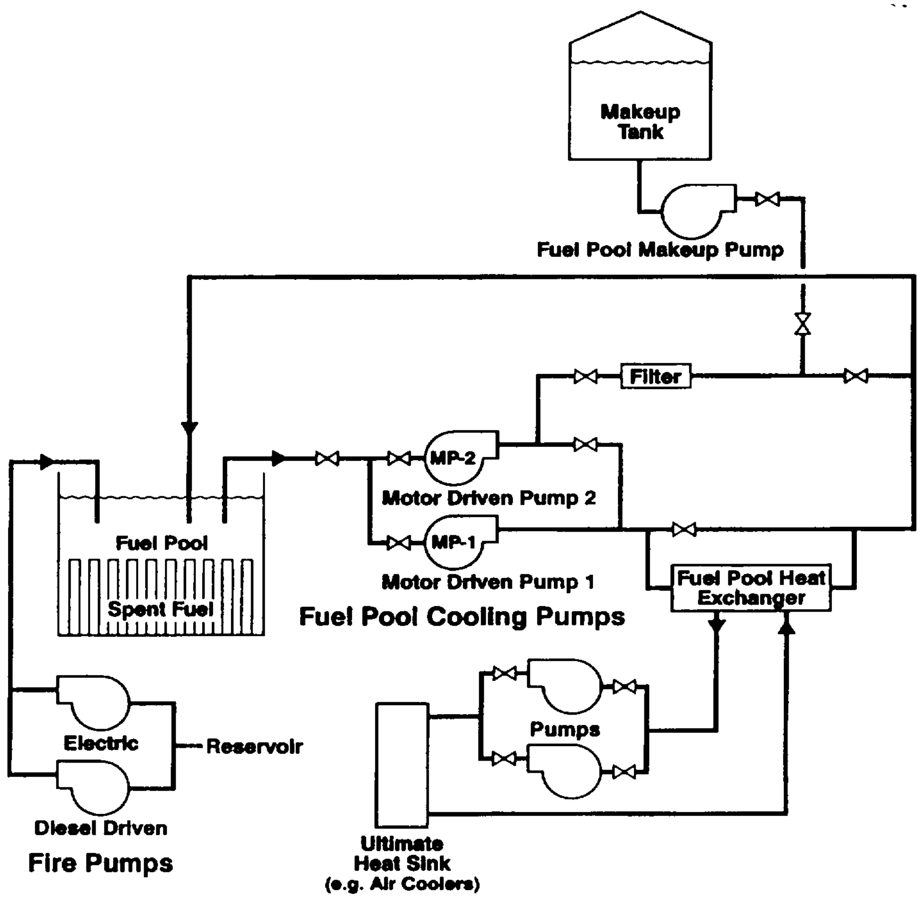


Figure 3.1 Assumed Spent Fuel Pool Cooling System

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 in the SFP Island 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 occurs, 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 blue ribbon committee of experts which identified the attributes necessary to achieving very high levels of human reliability for responding to potential accident scenarios in a decommissioning plant SFP. (See HRA Study in Appendix 2a).

Upon consideration of the sensitivities identified in the staff's preliminary study and to reflect actual operating practices at many decommissioning facility, the nuclear industry, through NEI made important commitments (located in Appendix 6) which were reflected in the staff's updated risk assessment. The revisions to the risk assessment generally reflected changes of assumptions in the areas shown below. The applicability of the specific decommissioning industry commitments (DICs) 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.

The high probability of the operators identifying and diagnosing a loss of cooling or inventory is dependent upon;

DIC #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.

DIC #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.

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

DIC #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.

DIC #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.

DIC #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.

DIC #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.

DIC #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 makeup water to the spent fuel pool. The plan will provide for remote alignment of the makeup source to the spent fuel pool without requiring entry to the refuel floor.

DIC #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.

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

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 below.

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

INITIATING EVENT	**Base Case**
Loss of Offsite Power - Plant centered and grid related events	8.2E-08
Loss of Offsite Power - Events initiated by severe weather	1.3E-06
Internal Fire	6.7E-08
Loss of Pool Cooling	5.7E-08
Loss of Coolant Inventory	1.5E-07
Seismic Event	>1.0E-06
Cask Drop	2.2E-07
Aircraft Impact	2.9E-09

Tornado Missile	ε^4
Total	>2.9E-06

NOTE: GARETH WILL PROVIDE A CHARACTERIZATION OF WHAT RESULTS REPRESENT< MEAN/POINT ESTIMATES............

The above results show that the estimated frequency for a zirconium fire is greater then approximately 3E-06 per year, with the dominant contributions being from severe seismic events and loss of offsite power initiated by severe weather.

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

3.3 Internal Event Scenarios Leading to Fuel Uncovery

The following is a description of how we modeled the cutsets[5] with the highest expected frequency of fuel uncovery for each internal event initiator: Details of the assessment are provided in Appendix 2.

3.3.1 Loss of Offsite Power from Plant-Centered and Grid Related Events

Frequency of Fuel Uncovery

Frequency of fuel uncovery = 8.2×10^{-8} per year

Scenario

Plant-centered events typically involve hardware failures, design deficiencies, human errors (in maintenance and switching), localized weather-induced faults (e.g., lightning), or combinations of these. Grid-related events are those in which problems in the offsite power grid cause the loss of offsite power. With offsite power lost (and therefore onsite 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 (i.e., until offsite power is recovered, all electrical pumps would be unavailable, and the diesel-driven fire pump only would be available to provide makeup to the spent fuel pool.) If power were not restored quickly enough, the pool will heat up and boil off inventory until the fuel is uncovered (if there were inadequate makeup). If the diesel-driven pump fails, and if offsite power were not recovered in a timely manner, offsite recovery using fire engines is a possibility. With 1-year-old fuel (i.e., the youngest fuel in the fuel pool was shutdown in the reactor one year ago), 127 hours is available for this recovery action.

Even given recovery of offsite power, the fuel handler has to restart the fuel pool cooling

[4] Frequency of less than 1×10^{-9} per year

[5]The numbered cutsets are identified and defined in Appendix 2

pumps. Failure to do this or failure of the equipment to restart will necessitate other fuel handler recovery actions. Again, considerable time is available.

Cutset

There was one important sequence minimum cutset.

Cutset for sequence 5:

> (loss of offsite power) x (fuel handlers fail to diagnose loss of SFP cooling when offsite power is lost) = 8×10^{-8} per year

PUT IN ASSUMPTIONS, COMMITMENTS RELIED UPON TO GIVE LOW RESULTS.

3.3.2 Loss of Offsite Power from Severe Weather Events

Frequency of Fuel Uncovery

Frequency of fuel uncovery = 1.3×10^{-6} per year

Scenario

This event represents the loss of SFP cooling resulting from a loss of offsite power from severe-weather-related events. Until offsite power is recovered, the electrical pumps would be unavailable and the diesel-driven fire pump would be available to only provide makeup. We assumed, given the extremely bad weather, it would be more difficult for offsite help to come and assist the fuel handlers at the site than for an ordinary loss of offsite power (LOSP) event. We assumed that given a LOSP event, the first thing the operator would do is attempt to recover power.

Cutset

There was one important minimum cutset.

Cutset for sequence 8:

> (loss of offsite power due to severe weather) x (offsite power is not recovered for more than 24 hours) x (diesel-driven firewater pump unavailable due to potential for flooding of site) x (fuel handlers fail to provide alternate sources of cooling from offsite) = 1.1×10^{-6} per year

Sensitivity

The sensitivity study showed the potential high estimated frequency of fuel uncovery if there was the lack of good communication between onsite and offsite resources, lack of formal training and lack of detailed procedures significantly increase the estimated frequency.

3.3.3 Internal Fire

Frequency of Fuel Uncovery

Frequency of fuel uncovery = 9.0×10^{-8} per year

Scenario

This event tree models the loss of SFP cooling caused by internal fires. We assumed that there is no automatic fire suppression system for the SFPC area. The fuel handler may initially attempt to recover the damaged SFP cooling system given that he responds to the alarms. If the fuel handler fails to respond the alarm, we assumed that SFPC 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 alternate makeup, either using the site firewater system or by calling upon offsite 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.

Cutset

There were three important sequence minimum cutsets.

Cutsets for sequence 4:

i) (fire starts in SFP area) x (fuel handler fails to suppress fire) x (fuel handlers fail to diagnose need to start firewater system) = 1.5×10^{-8} per year

ii) (fire starts in SFP area) x (fuel handler fails to suppress fire) x (firewater system fails to start/run) x (repair crew fails to repair firewater system) x (fuel handlers fail to provide alternate sources of water from offsite) = 6.8×10^{-9} per year

Cutset for sequence 8:

i) (fire starts in SFP area) x (fuel handler fails to respond to a signal indication from the control room that there is a fire) x (fuel handlers fail to observe loss of cooling in walkdowns [dependent case]) = 4.5×10^{-8} per year

Sensitivity

The sensitivity study again showed the potential high estimated frequency of fuel uncovery given the lack of formal training, detailed procedures, test and maintenance on important equipment, and infrequent walkdowns.

3.3.4 Loss of Cooling

Frequency of Fuel Uncovery

Frequency of fuel uncovery = 5.7×10^{-8} per year

Scenario

The initiating event frequency includes the loss of coolant system flow from the failure of pumps or valves (See Figure 3.0-1), 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.

For loss of cooling events in our decommissioning SFP case, there is a lot of time for fuel handler recovery. In the case of 1-year-old fuel (i.e., fuel that was in the reactor when it was shutdown one year previously), 127 hours is available. The result is that the risk of fuel uncovery for these events is small if industry commitments are implemented at decommissioning plants.

Based on the assumptions made, the frequency of core uncovery can be seen to be very low. A careful and thorough adherence to DICs 2, 5, 8 and 10 is crucial to establishing the low frequency. In addition, however, the assumption that walkdowns are performed on a regular, (once per shift) basis is important to compensate for potential failures to the instrumentation monitoring the status of the pool. The analysis has also assumed that the procedures and/or training are explicit in giving guidance on the capability of the fuel pool makeup system, and when it becomes essential to supplement with alternate higher volume sources. The analysis also assumed that the procedures and training are sufficiently clear in giving guidance on early preparation for using the alternate makeup sources.

The additional requirement of walkdowns being performed at least once per shift.
is identified by the staff as a decommissioning staff requirement (DSR #1)

3.3.5 Loss of Coolant Inventory

Frequency of Fuel Uncovery

Frequency of fuel failure = 1.7×10^{-7} per year

Scenario

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 (on the order of) less than one event per 100 reactor years. Most of these events are as a result of fuel handler error and are 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.

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 makeup is the spent fuel pool makeup system. For the other 94% of the cases, operation of the makeup pump is sufficient to prevent fuel uncovery.

Cutset

There was one important sequence minimum cutset.

Cutset for sequence 9:

i) (loss of inventory) x (loss exceeds normal makeup capacity) x (fuel handler fails to respond to signal indication in control room) x (fuel handler fails to notice loss of inventory - dependent case) = 1.4×10^{-7} per year

Sensitivity

The sensitivity study showed the potential for a very high estimated frequency of fuel uncovery. Due to lack of formal training, detailed procedures, test and maintenance on important equipment, and infrequent walkdowns.

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. Details of this evaluation can be found in Appendix 2. 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 onsite or offsite 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 to 12 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 bounded by other initiators.

Based on discussions with structural engineers, the staff assumed that only spent fuel shipping casks had sufficient weight to catastrophically damage the pool if dropped. We assumed there is 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 very low likelihood that it would cause catastrophic damage to the pool.

For a non-single failure proof load handling system that does not follow NUREG-0612 [Ref.4] guidelines, 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 3.4×10^{-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 a single failure proof load handling system or a plant conforming to the NUREG-0612 guidelines, is estimated to have a mean value of 9.6×10^{-6} per year, again for 100 heavy load lifts per year but using new data from U.S. Navy crane experience. Once the load is dropped, the next question is whether the drop did significant damage to the spent fuel pool.

When estimating the failure frequency of the pool floor, 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 2.1×10^{-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 2.0×10^{-7} per year for a single failure proof system or a plant conforming to all NUREG-0612 guidelines.

When estimating the failure frequency of the pool wall, the staff assumed one-in-ten heavy load drop events (0.1) will result in significant damage to the wall. For the non-single failure proof handling system, the mean value for the failure rate is 2.1×10^{-6} per year and for the single failure proof handling system the mean value for the failure rate is 2.1×10^{-8} per year. For comparison, the frequency given in NUREG/CR-4982 [Ref. 5] for wall failure was 3.7×10^{-8} per year, for 204 lifts per year. For 100 lifts, the NUREG/CR-4982 value would be 1.5×10^{-8} per year, very comparable to the estimate in this assessment.

The combined (floor and wall) expected frequency for catastrophic failure of non-single failure proof systems is 2.3×10^{-6} per year, and for single failure proof systems or a plant conforming to the NUREG-0612 guidelines is 2.2×10^{-7} per year. NEI has made a commitment (DIC #1) for the nuclear industry that future decommissioning plants will comply with phases 1 and 2 to the NUREG-0612 guidelines, which would put future decommissioning plants in the latter category.

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

The following is a description of how we modeled each of the external event initiators, 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 beginning our evaluation of the effect of seismic events on spent fuel pools, it became apparent that we do not have detailed information of how all the spent fuel pools were designed and constructed. We originally performed a simplified 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 approach. After further evaluation and discussions with stakeholders, we determined that it would not be cost effective to perform a plant-specific seismic evaluation for each spent fuel pool. Working with our stakeholders, we developed other tools that help assure the pools are sufficiently robust.

We believe spent fuel pool structures at nuclear power plants are seismically robust. They are constructed with thick reinforced concrete walls and slabs lined with thin stainless steel liners 1/8 to 1/4 inch thick.[6] 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 loads beyond their design basis. The dimensions of the pool structure are generally derived from radiation shielding considerations rather than structural needs. Spent fuel structures at operating nuclear power plants are able to withstand loads substantially beyond those for which they were designed. Consequently, they have significant seismic capacity.

Based on our work and that of an expert consultant (See Appendix 7 Kennedy report), we determined that seismic vulnerability of spent fuel pool structures is expected at levels of earthquake ground motion equal to 2.5 to 3.5 times a plant's safe shutdown earthquake (SSE). For sites east of the Rocky Mountains, ground motions three times the SSE are considered to be as high as physically possible for a site given the tectonics in the east. For the west coast sites, as the magnitude of the seismic event increases, the probability of its occurrence goes down rapidly. Thus a seismic event equal to 2.5 to 3.5 SSE at a west coast site may be considered incredible for the site. Therefore, for west coast sites a seismic event greater than two times the SSE could be considered too large to be credible.

---

[6] Except at Dresden Unit 1 and Indian Point Unit 1. 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.

Therefore, we assumed that seismic events greater than three times the SSE at a lower seismicity location (eastern US site) and two times the SSE at a higher seismicity location (west coast site) are nearly physically impossible. The seismic hazard component of the risk statement thus can be set aside if it can be demonstrated that structural capacity (i.e., the HCLPF value) is greater than or equal to 2 times the SSE at higher seismicity sites and at 3 times the SSE at lower seismicity sites. Implicit in this is the assumption that pool structures are free from pre-existing degradation or other seismic vulnerabilities. To assure there are no vulnerabilities, NEI developed a seismic checklist, which we enhanced. The enhanced checklist seeks to assure there are no weaknesses in the design or construction of the pools that might make them vulnerable to earthquake ground motions several times higher than those in the site's safe shutdown earthquake (SSE). We note that spent fuel pool configuration, layout, and structural details vary considerably from one plant to another. For sites that fail the seismic check list or have a HCLPF value lower than the ground motion goal appropriate for the area of the US the pool is situated in, the utility would need to conduct a detailed assessment of the seismically induced probability of failure of its spent fuel pool structures and components.

Our consultant's report (see Appendix 7) identifies 8 sites by site number for which seismically induced probability of failure (POF) is greater than $3X10^{-6}$ using the Lawrence Livermore National Laboratory 1993 hazard curves. For these sites it will be necessary to perform an evaluation of the POF using plant specific fragility information. For all other sites east of the Rocky Mountains, the use of the seismic check list should be adequate. The seismic checklist which the staff has developed to meet this goal is given in Appendix 5.

3.4.2 Aircraft

We 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 DOE-STD-3014-96 [Ref. 6], were used to assess the likelihood of an aircraft crash into or near a decommissioning spent fuel pool. 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 makeup sources, and may also affect recovery actions. There are two approaches that can be taken to evaluate the likelihood of an aircraft crash into a structure. The first is called the point target model which uses the area (length times width) of the target to determine the likelihood that an aircraft will strike the target. The aircraft itself does not have real dimensions when using this model. In the second approach, the DOE model modifies the point target approach to account for the wing span and the skidding of the aircraft after it hits the ground by including the additional area the aircraft could cover. Further, that model takes into account the plane's glide path by introducing the height of the structure into the equation, which effectively increase the area of the target (see Appendix 2D).

Our estimate of the frequency of catastrophic PWR spent fuel pool damage (i.e., the pool is so damaged that it rapidly drains and cannot be refilled from either onsite or offsite resources) resulting from a direct hit is based on one estimate using the point target area model for a 100 x 50 foot pool, with a conditional probability of 0.3 (large aircraft penetrating 6-ft of reinforced concrete) that the crash results in catastrophic damage. The point target model was chosen to model a direct hit on the pool. If 1-of-2 aircraft are large and 1-of-2 crashes result in significant damage, then the estimated range of catastrophic damage to the spent fuel pool is 9.6×10^{-12} to 4.3×10^{-8} per year. The mean value is estimated to be 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 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, although a crash into one of four sides of a BWR secondary containment may have a reduced likelihood of penetration due to other structures being in the way of the aircraft. 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. If instead of a direct hit, the aircraft skidded into the pool or a wing clipped the pool, catastrophic damage may not occur. We project that skidding aircraft will be negligible contributors to the frequency of fuel uncovery resulting from catastrophic failure of the pool. The estimated frequencies of air craft induced catastrophic spent fuel pool failure are bounded by other initiators.

Our estimate of the frequency of significant damage to spent fuel pool support systems (e.g., power supply, heat exchanger, or makeup water supply) is developed for three different situations. The first case is based on the DOE model including the glide path and the wing and skid area for a 400 x 200 x 30 foot structure (i.e., the support systems are located inside a large building) with a conditional probability of 0.01 that one of these systems is hit. This model accounts for damage from the aircraft including, for example, being clipped by a wing. We assumed that critical systems occupy only 1% of the total structure area. The estimated frequency range for significant damage to the support systems is 1.0×10^{-10} to 1.0×10^{-6} per year. The mean value is estimated to be 7.0×10^{-8} per year. The second case estimates the value for the loss of a support system (power supply, heat exchanger or makeup water supply) based on the DOE model including the glide path and the wing and skid area for a 10 x 10 x 10 foot structure (i.e., the support systems are housed in a small building). The estimated frequency of support system damage ranges from 1.1×10^{-9} to 1.1×10^{-5} per year, with the mean estimated to be 7.3×10^{-7} per year. The third case uses the point model for this structure [10x 10 or 400 x 200?], and the estimated value range is 2.4×10^{-12} to 1.1×10^{-8} per year, with the mean estimated to be 7.4×10^{-10} per year. Depending on the model approach (selection of the target structure size; use of the point target model or the DOE model), 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 onsite or offsite. As an initiator to failure of a support system leading to fuel uncovery and a zirconium fire, an aircraft crash is bounded by other more probable events. Recovery of the support system will reduce the likelihood of spent fuel uncovery.

Overall, the likelihood of significant spent fuel pool damage from aircraft crashes is bounded by other more likely catastrophic spent fuel pool failure and loss of cooling modes.

3.4.3 Tornadoes

We performed a risk evaluation of tornado threats to spent fuel pools (details are in Appendix 2E). We 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. We 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 we examined the frequency and intensity of tornadoes in each of the continental United States using the methods described in NUREG/CR-2944 [Ref. 7]. The frequency of having an F4 to F5 tornado 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.

We 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 conditional 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)

We 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 makeup water supply). 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. As an initiator to failure of a support system, the tornado is bounded by other more probable events (see Table 3.1-1).

3.4.4 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. 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 requirements identified, the potential risk from SFP criticality is sufficiently small.

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

(1) A compression or buckling of the stored assemblies could result in a more optimum geometry (closer spacing) and thus create the potential for criticality (see the NRC staff report "Assessment of the Potential for Criticality in Decommissioned Spent Fuel Pools," 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. High-density racks are those that rely on both fixed neutron absorbers and geometry to control reactivity. Low-density racks rely solely upon geometry for reactivity control. In addition, all PWR pools are borated, whereas BWR pools contain no soluble absorbing material. If both PWR and BWR pools were borated, criticality would not be achievable for a compression event.

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

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

Other potential criticality events, such as loose debris of pellets or the impact of water or firefighting foam (adding neutron moderation) during personnel actions in response to accidents 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 the normal configuration. The phenomenon of a peak in reactivity due to low-density (optimum) moderation (fire-fighting foam) is not of concern in spent fuel pools since the presence of relatively weak absorber materials such as stainless steel plates or angle brackets is sufficient to preclude neutronic coupling between assemblies. Therefore, personnel actions to refill a drained spent fuel pool containing undeformed fuel assemblies would not create the potential for a criticality. Thus, the only potential scenarios described above in 1 and 2 involve crushing of fuel assemblies in low density racks or degradation of Boraflex over long periods in time.

To gain qualitative insights on the recriticality events that are credible, the staff considered the sequences of events that must occur. For scenario 1 above this would be require a heavy load drop into the a low density racked BWR pool compressing assemblies. From Appendix 2 on heavy load drop, the likelihood of a heavy load drop from a single failure proof crane has a mean value of approximately 9.6E-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 approximately 13% of the movement path length, dependant on plant specific layout specifics. The additional frequency reduction in the appendix to account for the fraction of time that the heavy load is lifted high enough to damage the pool liner is not applicable here because the fuel assemblies could be crushed without the same impact velocity being required as for the pool liner. Therefore, we observe a potential initiating frequency for crushing of approximately 1.2E-6 per year (based upon 100 lifts per year). Criticality calculations conducted for 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 1E-8 per year, which justifies its exclusion from further consideration.

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

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

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

Based upon the above conclusions and staff requirements, we believe that qualitative risk insights demonstrate conclusively that SFP recriticality poses so 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 control the risk of 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. Some of these have been proposed by the Nuclear Energy Institute in a letter to the NRC dated November 12, 1999 [See Ref. 1 or Appendix 6]. Others came to light as a result of the analysis itself. These characteristics are summarized in Chapter 4.1. The NRC intends to make these the principle aspects of the risk-informed approach to oversight of decommissioning plants.

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; namely, emergency preparedness, security and insurance. The technical results of this report might be used either to justify plant-specific exemptions from these requirements, or to determine how these areas will be treated in a risk-informed oversight process. Chapter 4.3 examines the implications of this 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 conditions necessary for a zirconium fire exist in 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 pool configuration. In some cases analyzed in Appendix 1 the required decay time is ____ years. However, the time period for any specific plant will vary. Plant-specific analysis is needed to justify the use of shorter decay periods.

The consequence analysis presented in Appendix 4 demonstrates that the consequences of a Zirconium fire in a decommissioning plant are very large. The integrated dose to the public is generally comparable to a large early release. Early fatalities, however, are low compared to those from a large early release from an operating reactor accident, and are very sensitive to the effectiveness of evacuation.

For a decommissioning plant with about one year of decay time, the timing of radiological releases from zirconium fires is significantly slower than those from the most limiting reactor accident scenarios. This is due to the slow 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 spent fuel pool water inventory. Thus, while the consequences of zirconium fires are in some ways comparable to large early releases from reactor accidents, the timing is much slower.

The annual frequency of events leading to zirconium fires at decommissioning plants is estimated to be $2x10^{-6}$ 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 embody these characteristics. The most significant contributor to this risk is a seismic event which exceeds the design basis earthquake. Other contributors are at most 10% of the seismic contribution including such scenarios as drop of heavy loads into the pool. This overall frequency is within the acceptance guidelines for large early release frequency (LERF) of $1x10^{-5}$

per year in RG 1.174. As noted above, zirconium fires are estimated to be similar to large early releases in some ways, but less severe in others.

4.2 Risk Impact of Specific Design and Operational Characteristics

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

4.2.1. 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 from a decommissioning spent fuel pool is a frequency for a zirconium fire of approximately $2x10^{-6}$ 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 the RG 1.174 criteria for baseline LERF of $1x10^{-5}$ per year should be applied. The risk assessment shows that the SFP baseline risk is well under the RG 1.174 criteria. In assessing the impact on change in risk, the staff considered a potential relief from EP requirements as the changing requirement.

Staff consequence analysis in Appendix 4 shows that the early health impacts from zirconium fire scenarios are significantly impacted by evacuation. This evacuation will greatly reduce the early fatalities near the plant site. However, this analysis also showed that for the slowly evolving SFP accident sequences, the initiation of effective evacuation can be much delayed in comparison to an operating reactor, where the accident results in high offsite does much more rapidly. Based upon this insight, the staff will require decommissioning staff requirement (DSR) #2, that a basic evacuation scheme be maintained at the plant. This scheme will include guidance on when offsite evacuation should be initiated, and ensure that current liaisons with offsite emergency organizations be maintained so that an ad hoc evacuation (as is done for transportation emergencies) can be put into place when needed. Since the slower evacuation expected from such an ad hoc effort was still shown to be effective for the SFP fire scenarios, this change from a formal offsite EP program is not expected to have any risk impact.

In addition to DSR #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 well beyond the plants 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 will require that such a checklist (DSR #3) be successfully implemented at all decommissioning facilities prior to relief from any regulatory requirements.

The accident sequences in Chapter 3 associated with loss of cooling or loss of inventory are quantified to result in low risk due to a number of elements that enhance the ability of the operators to respond successfully to the events with onsite and offsite 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 significantly increased. Some elements were also identified that reduce the likelihood of the loss of cooling or loss of inventory initiators, including both design as well as operational issues. The elements proposed by industry (Decommissioning Industry Commitments (DICs)) are identified below.

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

DIC #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.

DIC #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.

DIC #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 identifying and diagnosing a loss of cooling or inventory is dependent upon;

DIC #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.

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

DIC #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.

DIC #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.

DIC #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 makeup water to the spent fuel pool. The plan will provide for remote alignment of the makeup 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 control are not in place. The staff judges that such controls are provided by the decommissioning industry commitments (DICs).

DIC #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. 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 are implemented as assumed in the risk study. Due to the very different nature of a SFP accident versus the threat from an operating reactor, with respect to system design capability needs and event timing, the defense in depth function of reactor containment is not appropriate. However the staff has identified that the defense in depth of some form of emergency planning can be useful 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. Therefore the staff has identified the following decommissioning requirement above, which is stated:

DSR #4 Each decommissioning plant will develop and maintain a site emergency plan, that contains guidance on when a site emergency should be declared with respect to the possibility of a SFP fire. The plan will also identify off site liaisons with public emergency organizations to put in place ad hoc evacuation so as to have an effective evacuation prior to the postulated zirconium fire. The elements of this plan will be submitted to the staff for approval prior to any relief for full EP being considered.

4.2.3 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. However, the staff assessment did identify one area where additional margins are of benefit in moderating the risk from potential pool re-criticality. Due to the potential for loss of inventory events that can be recovered by use of alternate water sources, the potential exists for loss of shutdown margins with the addition of unborated water to pools that originally are borated. Additionally for pools that utilize Boraflex absorbers in high density racks, having boron on site for addition to the pool, would allow for quick restoration of shutdown margin if the rack surveillance and monitoring program did identify any significant degradation of the Boraflex. This leads to the following decommissioning staff requirement:

DSR #5 All decommissioning plants will retain on site quantities of soluble boran sufficient for maintaining pool shutdown margins in a borated pool which is assumed to have 50% of its water mass replace with unborated water. Additionally all decommissioning plants that utilize Boraflex absorbers will maintain sufficient soluble boron on site to make up shutdown reactivity margin lost due to degradation of 20% of Boraflex in the high density racks. Procedures will also be developed on the use of this boron for either scenario.

4.2.4. 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. For the SFP risk evaluation this identifies three primary areas for performance monitoring: 1) The performance and reliability of SFP cooling and associated power and inventory makeup systems, 2) Monitoring of the Boraflex condition for high density fuel racks, and 3) Monitoring crane operation and load path control for cask movements.

Monitoring of the performance and reliability of the SFP support systems, heat removal, power and inventory should be carried out under the provisions of the maintenance rule 50.65. Decommissioning plant licensees will retain the commitment to maintain a list of equipment within the scope of the maintenance rule as well as applicable performance criteria they are assessed against. Since the staff will not entertain requests for exemptions from this Rule for decommissioning plants, no additional DSR is required in this area.

With respect to monitoring of the Boraflex absorber material, the current monitoring programs required by Generic Letter 96-04 [Ref. 3] will be maintained by decommissioning plants until all fuel is removed from the SFP. This generates a decommissioning staff requirement (DSR).

DSR #6 Licensees will maintain a program to provide surveillance and monitoring of Boraflex in high density spent fuel racks until such a time as do high density racks are retained in the pool. The SFP licensees will also have procedures in place to assess degradation impact on reactivity shutdown margin and provide additional pool boration as necessary to maintain the needed margins.

With respect to monitoring and control of heavy load activities and load path control, licensee guidance in this area will be provided by DIC # 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 the 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 for are contained in 10CFR 50.47 [Ref. 4] and Appendix E [Ref. 5]. Further guidance on the basis for EP requirements is contained in NUREG-0396 [Ref. 6]. The general goal of EP requirements is to prevent early fatalities and to reduce offsite dose from accidents.

In the past, the NRC staff has granted exemptions from 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 EP requirement. The results of this technical study confirm that position for both the scenarios resulting in a potential zirconium fire as well as creditable pool recriticality 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 to allow effective offsite protective actions on an ad hoc basis, without the need for emergency planning. 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 offsite consequences of a zirconium fire are comparable to those from operating reactor 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 wnen compared with the risk guidelines from RG 1.174. Thus the risk associated with early fatalities from these scenarios is low. Based on this combination of low risk and slow evolution, the Commission might decide to reduce or eliminate EP requirements for decommissioning plants. With respect to the potential for pool recriticality, the staff's assessment discussed in Chapter 3 and Appendix 3 demonstrates that creditable scenarios for criticality are precluded by monitoring programs or are highly unlikely; and even if they do occur would not be expected to have offsite consequences. Therefore the conclusions regarding possible reductions in EP program requirements are not impacted.

One important safety principle of RG 1.174 is consistency with the defense in depth philosophy. In the rationalist approach, 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 SFP at operating reactors, defense in depth consists mainly of the mitigating effect of emergency preparedness. The Commission might consider retaining a baseline level of EP requirements for decommissioning plants as a defense in depth measure. This might be justified in view of the uncertainties associated with the risk analysis presented herein. The staff has not attempted to assess what level of emergency preparedness might be needed to provide this defense in depth. However, given the slow nature of these accidents, we believe it would be substantially lower than what is currently required for operating reactors.

The risk assessments contained in this report indicate that it would be acceptable to reduce the level of emergency preparedness to a minimum baseline level at a decommissioning reactor after a period of 1 year has elapsed. For purposes of this study, a 1 year period was considered the minimum 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. Any licensee wishing to gain relief from the EP requirements prior to the one year post-shutdown period given credit for in this report, would need to demonstrate a more robust reaction time than that credited in the human reliability analysis employed in this study. The staff would be receptive to an industry initiative or plant specific application that would attempt to advance the state of the art in this area.

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 likelihood 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, it cannot evaluate the potential consequences of a sabotage event that could directly cause off site fission product dispersion, say from a vehicle bomb that was driven into 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 the facility.

In an associated regulatory arena, 10 CFR 73.51, "Physical Protection for Spent Nuclear Fuel and High-Level Radioactive Waste," allows facilities not associated with an operating power reactor to store spent fuel at an independent spent fuel storage installation (ISFSI). This rule provides performance-based regulations specifically designed for these types of storage installations, i.e., fuel in dry cask containers or other storage formats. [redacted] 10 CFR 73.51 failed to account for the risk posed by vehicle-borne bombs at facilities where potential criticality and fuel heatup were still issues.

Ex 5

[redacted]

Ex 5

The proposed rulemaking would provide regulations specifically applicable to power reactor sites that have permanently ceased operations. The new rulemaking would codify and consolidate current regulations at a level commensurate with the reduced potential of sabotage at permanently shutdown sites. To develop this rulemaking, we will review existing regulations in 10 CFR 73.55 and determine what requirements are necessary for a permanently shutdown power reactor. After analyzing the security areas that need to be protected, we will eliminate requirements that are beyond the protection strategy needed for a permanently shutdown power reactor site and its capability to preclude a radiological release that could impact public health and safety.

[redacted]

64

The staff also noted that the applicability of 10 CFR 26 [Ref 10] has not been established 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. 11], 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 the radioactive material has been removed from the site, unless the Commission terminates the Part 50 license or otherwise modifies the financial protection requirements. The industry has asked the NRC to consider whether the likelihood of large scale radiological releases from decommissioning plants is low enough to justify modification of the financial protection requirements once the plant is permanently shutdown and prior to complete removal of all radioactive material from the site.

In the past, licensees have been granted exemptions from financial protection requirements on the basis of deterministic analyses showing that a zirconium fire could no longer occur. The analysis in this report supports continuation of this practice in the interim, and would support a revised regulatory framework for decommissioning plants that eliminates the need for insurance protection when a plant-specific thermal-hydraulic analysis demonstrates that a zirconium fire can no longer occur.

The NRC staff has considered whether the risk analysis in this report justifies relief from this requirement for decommissioning plants during the period when they are vulnerable to zirconium fires. As part of this effort, the staff determined that an analogy can be drawn between a SFP at a decommissioning plant and a wet (as opposed to dry) Independent Spent Fuel Storage Installation (ISFSI) licensed under 10 CFR 72 for which no indemnification requirement currently exists. Spent reactor fuel aged for one year can be stored in an ISFSI (wet or dry). The risk analysis in this report predicts high consequences for a zirconium fire, and identifies a generic window of vulnerability out to 5 years. The Commission has suggested in the staff requirements memorandum (SRM) for SECY-93-127 [Ref. 12] that insurance coverage is required unless a large scale radiological release is deemed incredible. Further, they instructed the staff to determine more precisely the appropriate spent fuel cooling period after plant shut down, and to determine the need for primary financial protection for ISFSIs.

Since the consequences are high, frequency of a zirconium fire occurring in a wet ISFSI or a decommissioning reactor SFP would have to be acceptably low to justify no regulatory requirement for indemnification protection. A dry ISFSI is not under consideration since the fuel is already air cooled and no threat of zirconium fire exists. The zirconium fire frequencies presented in Chapter 3 for a decommissioning reactor SFP do not fit the category of incredible. They are comparable to the frequencies of large releases from some operating reactors. The staff is not aware of any basis for concluding that the frequency of a zirconium fire occurring in a wet ISFSI would be significantly different than those presented in Chapter 3, and thus would conclude that indemnification should be required for operation of a wet ISFSI to be consistent with a decommissioning reactor SFP and provide for coherency in the regulations.

The staff knows of no frequency criterion which could be cited to justify reduction or elimination of the insurance requirement while a vulnerability to zirconium fire exists. Defining or applying such a criterion would be inconsistent with Commission direction provided in SECY-93-127. On the other hand, the possibility exists that the 5 year window of vulnerability could be reduced with more refined thermal-hydraulic calculations or other constraints on such parameters as fuel configuration. The staff would be receptive to an industry initiative designed to advance the state of the art in this area such that the period of vulnerability to zirconium fire could be reduced.

5.0 References

References for Executive Summary

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.
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. 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. 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 Safety 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. U.S. Nuclear Regulatory, "Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36," NUREG-0612, July 1980.
5. Sailor, et.al., "Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82", NUREG/CR-4982 (BNL-NUREG-52093), July 1987.
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, "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. 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."
5. U.S. Code of Federal Regulations, "Emergency Planning and Preparedness For Production and Utilization," Appendix E, Part 50, Title 10, "Energy."
6. 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.
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, "Requirements for Physical Protection of Licensed Activities," Section 50, Part 73, Title 10, "Energy."
9. 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."
10. U.S. Code of Federal Regulations, "Fitness for Duty Programs," Part 26, Title 10, "Energy."
11. U.S. Code of Federal Regulations, "Financial Protection Requirements and Indemnity Agreements," Part 140, Title 10, "Energy."
12. U.S. Nuclear Regulatory Commission, "Financial Protection Required of Licensees of Large Nuclear Power Plants During Decommissioning," SECY-93-127, dated July 13, 1993.

6.0 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
DIC	decommissioning industry commitments
DOE	Department of Energy
DSP	decommissioning status plant
DSR	decommissioning staff requirement
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
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 offsite 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
PRA	probabilistic risk assessment
PWR	pressurized water reactor
QA	quality assurance
RES	NRC Office of Research
RG	regulatory guide
SF	spent fuel
SFP	spent fuel pool
SFPC	spent fuel pool cooling system
SFPCC	spent fuel pool cooling and cleaning system
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

Formatted Version, Rev. 1 1/19/00

Appendix 1 Thermal Hydraulics

1.0 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 of 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 will 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 a 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 also not be able to maintain structural integrity because of the sustained loads at high temperature. 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.

Based on the large uncertainties in heatup calculations, the low level of sophistication and poor quality of heatup calculations submitted by licensees, and the absence of data for computer code assessment, the staff proposes an acceptance temperature of 600 °C if the radioactive iodine has decayed to the point where the gap activity is a significant contributor to offsite doses.

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 (GSI) 82. 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 GSI 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 GSI 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 include 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 air flow. 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 to within approximately one hundred degrees of oxidation, then the radiative heat from an adjacent assembly that did oxidize could raise its temperature to the oxidation level. The report also discusses small-scale experiments involving clad temperatures greater than 1000 °C. SNL hypothesized that molten zirconium material would slump or relocation 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 daysPWR, 6 kW/MTU decay power per assembly, high density rack, 10.25" pitch, 5" orifice, 1 inch from storage wall

280 daysPWR, same as above but for 1 foot from storage wall

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

unknownBWR, 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 an 5-inch orifice, a decay power below 6 kW/MTU did not result in 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. The staff has determined that the code will used as a scoping tool by the staff. 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 was it 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 on.

1.3 Heatup Calculation Uncertainties and Sensitivities

The phenomenology needed to model spent fuel heatup is dependent on the chosen cladding temperature success criteria 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 2.2.1. 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 GSI 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 air flow 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. Air flow 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 had 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 air flow 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 Office of Research (RES) using the FLUENT code and Pacific Northwest National Laboratory using the TEMPEST code indicate 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 zirconium oxidation 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.

At temperatures below the onset of self-sustaining oxidation, the heat source is dominated by the decay heat of the fuel. 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) air flow 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 GSI 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 be 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 can not 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 GSI 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 GSI 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 air flow 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.

1.4 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 an 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 PWRs and a 17x17 assembly for the BWRs. All design values are in Appendix 11. 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. 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.

1.5 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 GSI-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 counterbalance 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 3kW/MTU when in 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 3kW/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.

1.6 Fire Propagation

The staff has not performed a sufficient amount of research to 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.

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 Safety 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 regulatory issues generically 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.

Our assessment found that the frequency of spent fuel uncovery leading to a zirconium fire at decommissioning spent fuel pools is on the order of 4×10^{-6} per year when a utility follows certain industry commitments and certain of our recommendations. We also determined that if these commitments and recommendations are ignored, the estimated frequency of a zirconium fire could be significantly higher. Appendix ZZZ 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, we performed this assessment to examine the risk associated with decommissioning reactor spent fuel pools.

First, we examined whether or not it was possible from a deterministic view point for a zirconium cladding fire to occur. We chose zirconium fires as the key factor because radionuclides require an energetic source to transport them offsite if they are to have a significant health effect on local (first few miles outside the exclusion area) and more distant populations. Deterministic evaluations (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[7]. Our consequence analysis (Appendix 3) indicates that zirconium cladding fires could give offsite doses that the NRC would consider unacceptable. To assess the risk (essentially, "frequency" times "consequences") in the window from final shut down of a reactor to one year following shutdown, we initially performed a broad preliminary risk assessment, which modeled many internal and external initiating events. This assessment was the most comprehensive performed on spent fuel pool risk. The preliminary risk assessment was made publicly available early in the process (June 1999) so

---

[7] 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.

that the public and the nuclear industry could track the NRC's evaluation and provide comments. In addition, the preliminary risk assessment was subjected to a technical review and requantification by the Idaho National Engineering and Environmental Laboratory (INEEL). The NRC continued to refine its estimates, putting particular emphasis on improving the human reliability assessment (HRA), which is central to the analysis given the long periods required for lowering the water in the spent fuel pool for most initiators. We 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 our HRA effort and our ongoing reassessment of risk, the nuclear industry through NEI developed a list of commitments (See NEI letter dated November 12, 1999) 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 technical review by INEEL, and our ongoing efforts to improve the assessment. The report provides a technical basis for determining the acceptability of exemption requests and future rule making on decommissioning plant risk.

In performing the preliminary risk assessment, we chose to look at the broad aspects of the issue. We investigated a wide range of initiators (internal and external events including loss of inventory events, fires, seismic, aircraft, and tornadoes). We modeled a decommissioning plant's spent fuel pool cooling system based on the sled-mounted systems that are used at many current decommissioning plants. We chose one representative spent fuel pool configuration (See Figure 2.0-1) 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 by 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 offsite consequences, we only concerned ourselves 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 offsite in order to have potentially significant offsite consequences. We 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, we sent the assessment to INEEL for review and held public meetings and a workshop to assure that our 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 our workshop, NEI provided a list of general commitments (See November 12, 1999 letter) that proved very instrumental in refining the assumptions and models in the draft final risk assessment. Working with several PRA experts, we subsequently developed improved HRA estimates for events that lasted for extended periods. We developed a basis (see Section YYY) for helping to assure that the HRA values we used in our improved HRA analysis come true at decommissioning plants in the future.

SectionXX describes how the risk assessment was performed for beyond design bases internal event accident sequences (i.e., sequences of equipment failures or operator errors that could lead to a zirconium cladding fire and release of radionuclides offsite). We developed event trees and fault trees that model the initiating events and system or component failures that lead to fuel uncovery (these trees are provided in Appendix XXX). Table 2A.1-1 lists the internal and external initiating events[8] found to be potentially important by qualitative screening processes in the above references. The table identifies the source of each frequency estimate (they are generic and not plant-specific). Our estimates of conditional failure probabilities of mitigating systems and components (both active and passive) are given in Table 2A.1-2. Table 2A.1-3 summarizes the calculations of frequency of fuel uncovery for all initiators analyzed. Section 3.2 discusses beyond design bases external event accident sequences. Section 3.3 provides the working group's insights from this final risk evaluation.

The risk from sabotage is not normally evaluated in a PRA, in part because such acts cannot be easily analyzed analytically. We have identified to the NRC safeguards staff the structures, systems, components, industry commitments, and staff requirements that are most important in helping assure that the spent fuel pools do not represent an undue risk to the public. The safeguards staff will use this information to assist them in making decisions on the degree, location, and type of safeguards necessary to protect the public safety at decommissioning plant spent fuel pools.

2a. Detailed Assessment of Risk from Decommissioning Plant Spent Fuel Pools

[8] Internal initiating events are events that begin within the confines of the nuclear power plant and cause plant disruption. Two examples are inadvertent closure of the spent fuel pool cooling system suction valves and a pipe break in the spent fuel pool cooling system. External events are those events that begin outside the confines of the nuclear power plant. Two examples are seismic events and hurricanes. There are a few events that begin inside plants, such as internal floods and fires, that have been characterized in some PRAs as external events.

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

Introduction

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

Spent fuel pool structures at nuclear power plants are constructed with thick reinforced concrete walls and slabs lined with thin stainless steel liners 1/8 to 1/4 inch thick, except at Dresden Unit 1 and Indian Point Unit 1. 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 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 loads beyond their design basis. The dimensions of the pool structure are generally derived from radiation shielding considerations rather than structural needs. Spent fuel structures at operating nuclear power plants are inherently rugged in terms of being able to withstand loads substantially beyond those for which they were designed. Consequently, they have significant seismic capacity.

Seismic Checklist

In the preliminary draft report published in June, 1999, the staff assumed that the spent fuel pools are robust for seismic events less than three times the safe shutdown earthquake (SSE). It was assumed that the high confidence, low probability of failure (HCLPF)[9] 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 0.4 to 0.5 g (where g is the acceleration of gravity). Seismic hazard curves from the Lawrence Livermore National Laboratory (NUREG-1488) show that, for most CEUS plants, the mean frequency for PGA equal to 3 X SSE is less than 2E-5. In the June report, the working group used the approximation that the frequency of a seismic event that will challenge the spent fuel pool integrity is 5% of 2E-5, or a value of 1E-6.

---

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

Several public meetings were held from April to July, 1999 to discuss draft report of the Technical Working Group. 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, NEI proposed a checklist which would assure that any plant could show robustness for a seismic event of approximately 0.5g peak ground acceleration (PGA). A copy of this submittal is included in Appendix 5.1.

The NRC contracted with Mr. Robert P. Kennedy to perform an independent review of the seismic portion of the June TWG draft report, as well as the August 18 submittal from NEI. Mr. Kennedy's comments and recommendations were contained in an October, 1999 report entitled "Comments Concerning Seismic Screening and Seismic Risk of Spent Fuel Pools for Decommissioning Plants," which is included as Appendix 5.2 of this report. Mr. Kennedy raised three significant concerns about the completeness of the NEI checklist.

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

Exceptions

The seismic checklist is not expected to provide the solution for all sites. Some CEUS sites, and all western sites, are known to exceed the assumption on which the checklist is based; namely, that 3 X SSE is less than 0.5g PGA. For these plants, the NRC has proposed, and the industry has agreed, that a more detailed assessment of seismic fragility is needed to establish the HCLPF capacity.

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

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

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

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

Seismic risk

As noted above, the preliminary TWG report published in June, 1999 used an approximate method for estimating the risk of spent pool failure. It was assumed that the high confidence, low probability of failure (HCLPF) value for the pool integrity is 3 times SSE. For most CEUS sites, 3 X SSE is in the peak ground acceleration (PGA) range of 0.4 to 0.5 g (where g is the acceleration of gravity). Seismic hazard curves from the Lawrence Livermore National Laboratory (NUREG-1488) show that, for most CEUS plants, the mean frequency for PGA equal to 3 X SSE is less than 2E-5. In the June report, the working group used the approximation that the frequency of a seismic event that will challenge the spent fuel pool integrity is 5% of 2E-5, or a value of 1E-6.

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

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

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

Conclusions

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

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

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

Appendix 2c

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

Summary

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

Analysis

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

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

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

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

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

The handling system failure rate was estimated in NUREG-0612 to be in the range of $1.0\text{x}10^{-5}$ to $1.5\text{x}10^{-4}$ incidents per year based on the 1970s Navy crane incident data and a staff estimate of the total number of lifts per year. The staff's evaluation included a factor of two reduction for the range estimate based on improved procedures and conformance with the guidelines presented in Section 5.1.1 of NUREG-0612.

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

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

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

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

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

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

Heavy Load Drop

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

Failure of the Lifting Equipment

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

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

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

Failure to Secure the Load

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

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

Heavy Load Drop Summary

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

The estimated mean value for a heavy load drop was 2.3×10^{-6} per year for 100 lifts (FHLS) for a single-failure proof handling system, with a range of 9.5×10^{-7} to 1.0×10^{-5} per year. The contributors (mean values) included crane failure at $1.4x10^{-6}$ per year (CRANE), operator-related errors at $3.0x10^{-8}$ per year (CF1 + CF3) and improper rigging at $8.7x10^{-7}$ per year (RIGGING). For the non-single failure proof handling system, the estimated mean frequency for a heavy load drop was $1.0x10^{-3}$ per year for 100 lifts, with a range of $2.0x10^{-5}$ to $1.2x10^{-3}$ per year.

Evaluation of the Load Path

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

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

Heavy Load Drop Leading to a Loss-of-Inventory

Our heavy load drop evaluation is based on the method and fault trees developed in NUREG-0612. New 1990s Navy data was used to quantify the failure of the lifting equipment. The WIPP human error evaluation was used to quantify the failure to secure the load. We estimated the mean frequency of a loss-of-inventory from a cask drop to be $2.0x10^{-7}$ per year for 100 lifts for a single-failure proof handling system (Table A2c-2, LOI-S). The range was estimated to be between $2.1x10^{-6}$ to $2.8x10^{-8}$ per year. Table A2c-2 presents the results for a heavy load drop on or near the spent fuel pool. If the cask were dropped on the spent fuel pool floor, the likelihood of a loss-of-inventory given the drop is 1.0. If the load were dropped on the spent fuel pool wall, the likelihood of a loss-of-inventory given the drop, is 0.1. Therefore the likelihood of a loss-of-inventory from a dropped spent fuel pool cask for a single-failure proof handling system was estimated to be $2.2x10^{-7}$ per year (for 100 lifts). The range was estimated to be between $2.3x10^{-6}$ to $3.1x10^{-8}$ per year.

For a non-single failure proof handling system, we based the mean frequency of a loss-of-inventory estimate on NUREG-0612. In NUREG-0612, an alternate fault tree (Figure B-2, page B-16 of NUREG-0612) was used to estimate the frequency of exceeding the release guidelines (loss-of-inventory) for a non-single failure proof system. The mean value was estimated to be about $2.1x10^{-5}$ per year (event 2.1.1) when corrected for the new Navy data and 100 lifts per year (Table A2c-2, LOI-N). The range was estimated to be between $7.5x10^{-5}$ to $1.0x10^{-7}$ per year. Table A2c-2 presents the results for a cask drop on or near the spent fuel pool. If the cask were dropped on the spent fuel pool floor, the likelihood of a loss-of-inventory given the drop is 1.0. If the cask were dropped on the spent fuel pool wall, the likelihood of a loss-of-inventory given the drop is 0.1. Therefore we estimated the likelihood of a loss-of-inventory from a dropped spent fuel pool cask for a non-single failure proof handling system to be $2.3x10^{-5}$ per year (for 100 lifts). The range was estimated to be between $8.3x10^{-5}$ to $1.1x10^{-7}$ per year.

Comparison of results to other studies and data

Assessment of the Incident Rate

The incidents per year range was estimated to be on the order of $1.0x10^{-5}$ to $1.5x10^{-4}$ incidents per year. This range was based on Navy data and was used in the NUREG-0612 evaluation and in the current evaluation. The incident rate contains uncertainty because it is not well known how many crane operations occurred without a reportable incident. There is also some uncertainly in using the Navy data for nuclear power plant operations.

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

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

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

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

For the current experience base for LWRs, $\lambda_{95\%} = 7.1x10^{-4}$ incidents per year (assuming each cask load requires two lifts). At the 50% confidence limit, $\lambda_{50\%} = 1.6x10^{-4}$ incidents per year. If the GCR data is considered and added to the LWRs data, then $\lambda_{95\%} = 2.7x10^{-4}$ incidents per year and $\lambda_{50\%} = 6.2x10^{-5}$ incidents per year. The actual cask handling data does not call into question the incident rate range used in this assessment.

Summary of Other Heavy Load Drop Studies

Heavy load drops were evaluated as part of Generic Safety Issue 82. In NUREG/CR-4982 ("Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82) the total human error rate associated with cask movement was estimated to be $6.0x10^{-4}$ incidents per lift. It was further assumed that only 1-in-100 human errors would result in a cask drop. It was also estimated that the cask was above the pool edge (wall) about 25% of the lift time. Based on two shipment per week with two lifts per shipment (208 lifts), the estimate for a load drop on the spent fuel pool wall was $3.1x10^{-4}$ per year. Damage to the pool wall sufficient to cause a loss-of-inventory was further estimated to have a conditional probability of 0.1, based on the evaluation presented in NUREG/CR-5176, "Seismic Failure and Cask Drop Analyses of Spent

Fuel Pools at Two Representative Nuclear Power Plants," LLNL, P.G. Prassinos, et al., January 1989. The analysis assumes the height of the load above the pool wall is only about 8 to 10 inches. The estimate of a loss-of-inventory from a heavy load drop on the spent fuel pool wall was $3.1x10^{-5}$ per year (for a non-single failure proof handling system.) Damage resulting from a load drop onto the spent fuel pool floor was not addressed as part of Generic Safety Issue 82. We believe that if the load were dropped from a high enough elevation, e.g., 30 to 40 feet above the spent fuel pool floor, it is likely that significant damage would occur, resulting in a loss of inventory. Based on 100 lifts per year, the NUREG/CR-4982 evaluation would estimate the loss-of-inventory from a heavy load drop on the spent fuel pool wall to be about $1.5x10^{-5}$ per year (for a non-single-failure proof handling system).

In NUREG-1353, it was decided based on engineering judgement that conformance with NUREG-0612 guidelines would reduce the probability of a load drop as presented in NUREG/CR-4982 by a factor of 1,000. Based on Table A2c-2, the fault tree method indicates that the expected reduction was in the 10 to 100 range. For 100 lifts per year, the NUREG/CR-4982 evaluation would estimate the loss-of-inventory from a heavy load drop on the pool wall to be $1.5x10^{-8}$ per year. As a comparison to this current evaluation, for a load drop on the pool floor, this value should be increased by a factor of 10 to $1.5x10^{-7}$ per year to account for a load drop from 30 to 40 feet above the spent fuel pool floor (a drop onto the pool from this height likely will cause a loss of inventory.) Based on the fault tree quantification (Table A2c-2), the mean probability for the loss-of-inventory from a heavy load drop was estimated to be $2.0x10^{-7}$ per year for 100 lifts (for a single-failure proof handling system) for a drop on the spent fuel pool floor and $2.0x10^{-8}$ per year for a drop on the spent fuel pool wall.

Conclusion

This generic assessment of a heavy load (cask) drop that may result in significant damage to the spent fuel pool indicates that the likelihood of a loss-of-inventory from the spent fuel pool is in the range of $3.1x10^{-8}$ to $2.3x10^{-6}$ per year for 100 lifts with a mean value of $2.2x10^{-7}$ per year for a single-failure proof handling system. These values include the contribution from a heavy load drop on the spent fuel pool floor and a heavy load drop on the spent fuel pool wall.

Uncertainties

1. Incident rate.

   The range used in this evaluation ($1.0x10^{-4}$ to $1.5x10^{-4}$ incidents per year) was based on the Navy data originally assessed by the staff in NUREG-0612. The 1999 Navy data, like the 1980 data, did not include the number of lifts made and only provided information about the number of incidents. The cask loading experience at LWRs and the GCR tends to support values used for the incident range.

2. Drop rate.

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

3. Load path.

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

4. Load handling design.

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

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

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

Notes:

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

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

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

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

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

Symbol	HEP	Explanation of error	Source of HEP (NUREG/CR-1278)
A_1	$3.75x10^{-3}$	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 95[th] percentile/50[th] percentile (median). For an EF of 3, the mean-to-median multiplier is 0.8.

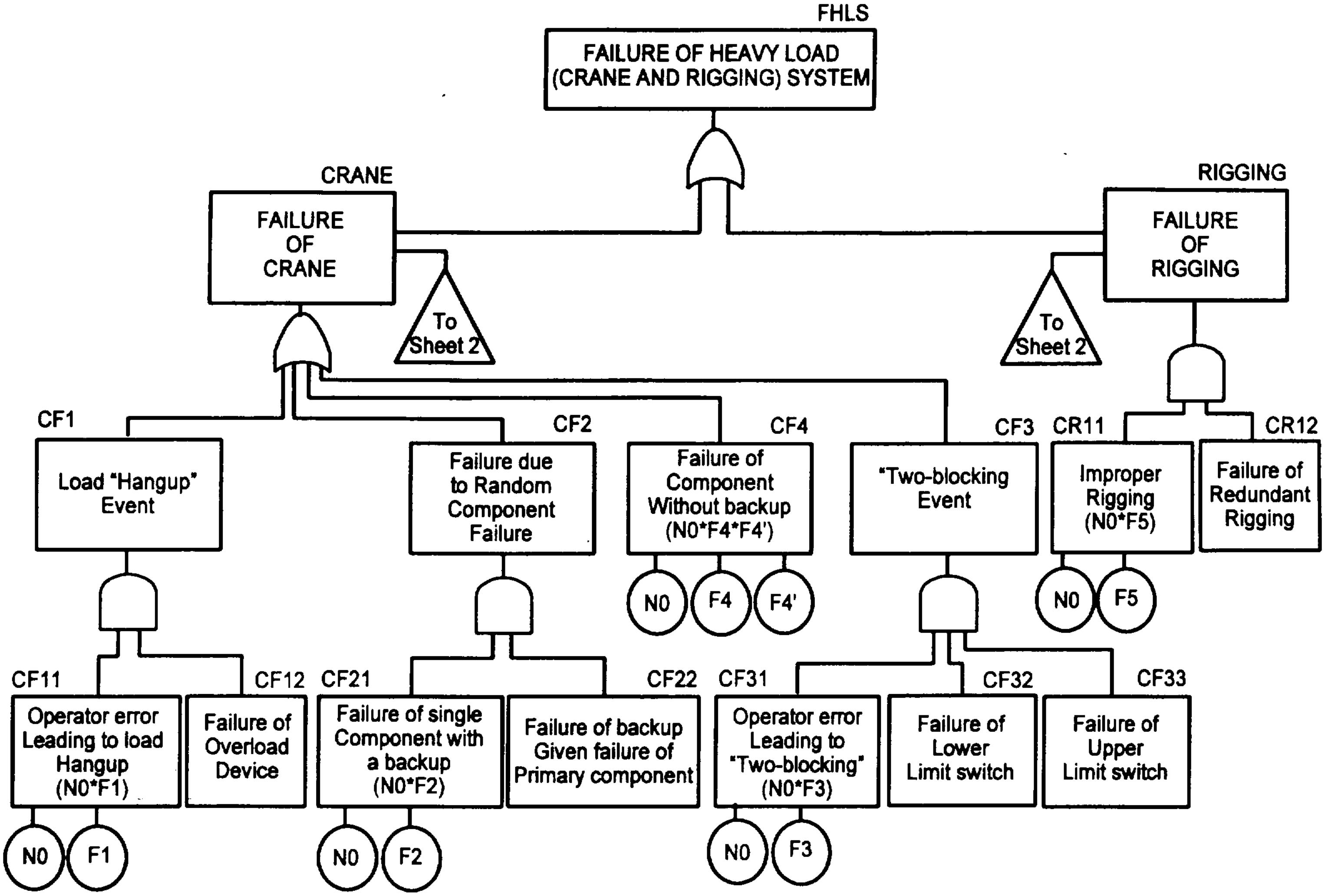


Figure A2c-1 (sheet 1 of 2) - Heavy load drop fault trees

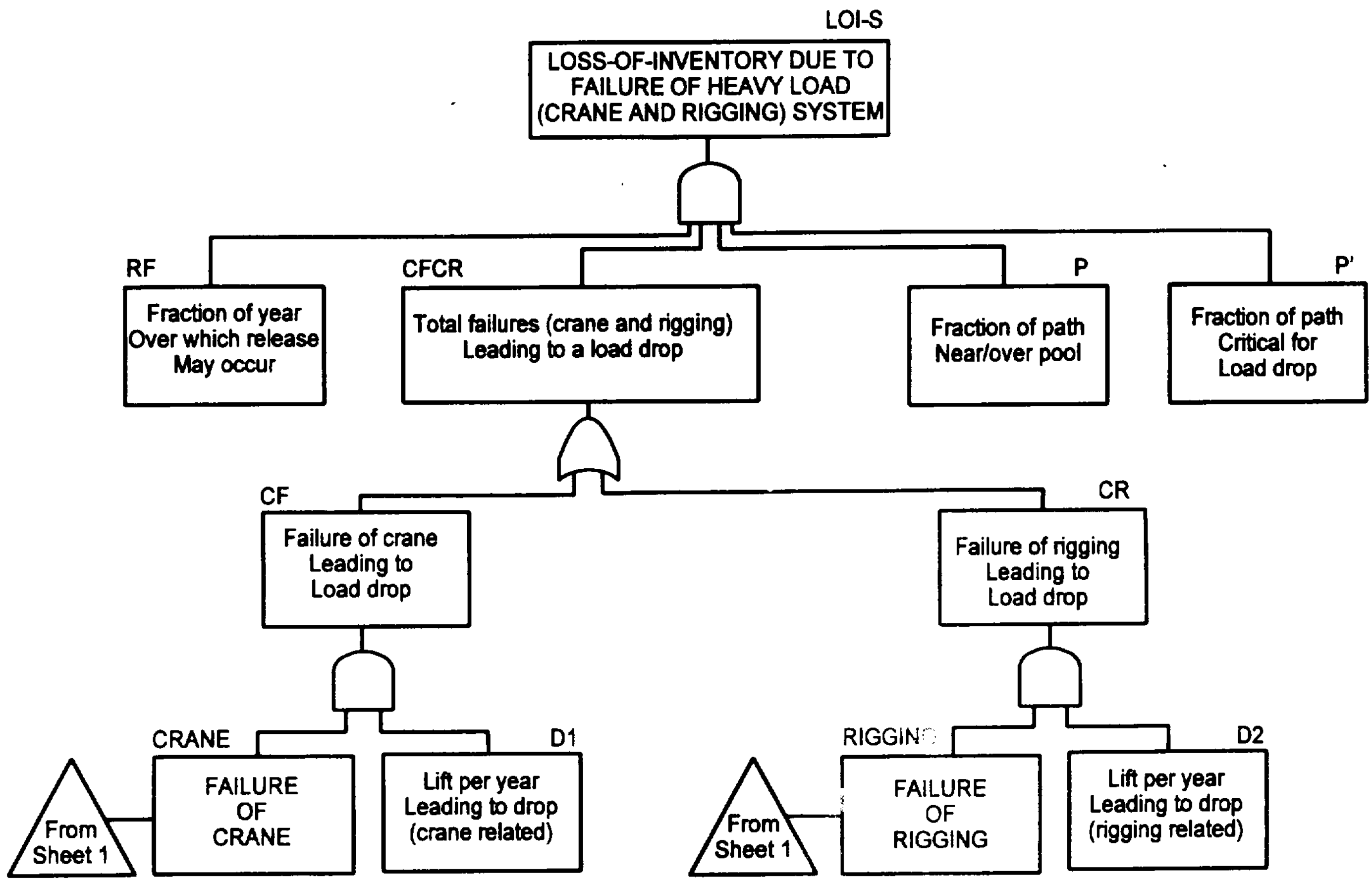


Figure A2c-1 (sheet 2 of 2) - Heavy load drop fault trees

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

Summary

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 100x50 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.

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 exchanges and makeup water supplies) requires a knowledge of the size and location of these systems at decommissioning plants, information not readily available. However, we believe this analysis is adequately broad to provide a reasonable approximation of decommissioning plant vulnerability to aircraft crashes.

The generic data provided in DOE-STD-3014-96, "Accident Analysis for Aircraft Crash Into Hazardous Facilities," U.S. Department of Energy (DOE), October 1996, were used 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 affect the availability of nearby support systems, such as power supplies, heat exchangers, and water makeup sources, and may also affect recovery actions.

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

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

where:

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

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

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

where:

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

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

Equation A2d-2

and where:

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

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

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

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

Applying the DOE generic data to the DWTF resulted in a frequency range of $6.5x10^{-9}$ hits per year to $6.6x10^{-5}$ hits per year, with an average value of $4.4x10^{-6}$ per year, for the effective aircraft target area model. For the point target area model, the range was $4.4x10^{-10}$ to $2.2x10^{-6}$ per year, with an average value of $1.5x10^{-7}$ per year.

The second evaluation was presented in a paper by K. Jamali, et al., "Application of Aircraft Crash Hazard Assessment Methods to Various Facilities in the Nuclear Industry," In which additional facility evaluations were summarized. For the Seabrook Nuclear Power Station, Jamali's application of the DOE effective aircraft target area model to the Final Safety Analysis Report (FSAR) data resulted in an impact frequency $2.4x10^{-5}$ per year. The Millstone 3 plant area was reported as $9.5x10^{-3}$ square miles and the FSAR aircraft crash frequency was reported to be $1.6x10^{-6}$ per year. Jamali applied the DOE effective aircraft target area model to information found in the Millstone 3 FSAR. Jamali reported an impact frequency of $2.7x10^{-6}$ per year using the areas published in the FSAR and $2.3x10^{-5}$ per year using the effective area calculated the effective aircraft target area model.

When the generic DOE data in Table A2d-1 were used (for a 514x514x100 foot site), the estimated impact frequency range was $6.3x10^{-9}$ to $2.9x10^{-5}$ per year, with an average of $1.9x10^{-6}$ per year, for the point target area model. The effective aircraft target area model resulted in estimated range between $3.1x10^{-8}$ to $2.4x10^{-4}$ per year, with an average of $1.6x10^{-5}$ per year.

A site-specific evaluation for Three-Mile Island Units 1 and 2 was documented in NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the United States," Lawrence Livermore National Laboratory, December 1987. The NUREG estimated the aircraft crash frequency to be $2.3x10^{-4}$ accidents per year, about the same value as would be predicted with the DOE data set for the maximum crash rate for a site area of 0.01 square miles.

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

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

Estimated Frequencies of Significant Spent Fuel Pool Damage

The frequency for significant PWR spent fuel pool damage resulting from a direct hit was estimated based on the point target model for a 100x50 foot pool with a conditional probability of 0.32 (large aircraft penetrating 6-ft of reinforced concrete) that the crash resulted in significant damage. If 1-of-2 aircraft are large and 1-of-2 crashes result in spent fuel uncovery, then the estimated range is $9.6x10^{-12}$ to $4.3x10^{-8}$ per year. The average frequency was estimated to be $2.9x10^{-9}$ 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.

Support System Unavailability

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

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

Uncertainties

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

Table A2d-1 Generic Aircraft Data

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

Table A2d-2 Aircraft Hits Per Year

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

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

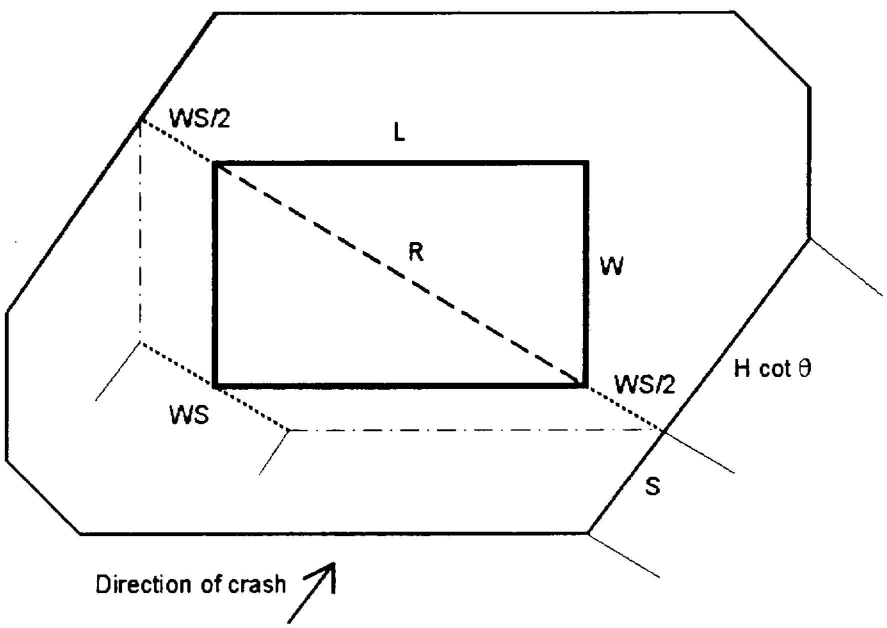
Period	Air Carriers	Air Taxes	General Aviation	Military Aviation	Total[1]
1995	$1.72x10^{-7}$	$2.47x10^{-6}$	$2.45x10^{-5}$	$5.03x10^{-7}$	$2.76x10^{-5}$
1993-1995	$1.60x10^{-7}$	$2.64x10^{-6}$	$2.82x10^{-5}$	$6.47x10^{-7}$	$3.16x10^{-5}$
1991-1995	$1.57x10^{-7}$	$2.58x10^{-6}$	$2.89x19^{-5}$	$7.23x10^{-7}$	$3.23x10^{-5}$
1986-1995	$1.52x10^{-7}$	$2.41x10^{-6}$	$2.89x10^{-5}$	$8.96x10^{-7}$	$3.23x10^{-5}$

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 and High Winds

1 Summary

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

2 Analysis

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

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

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

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

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

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

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

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

Recommended Values for Risk-informed Assessment of Spent Fuel Pool

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

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

where:

P_{fs} = strike probability for F-scale (fs)

$<a>_T$ = tornado area, mi^2

A_{ob} = area of observation, mi^2 (state land area)

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

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

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

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

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

Significant Pool Damage

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

Support System Availability

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

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

References:

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

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

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

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

Table A2e-1 Tornado and High Wind Data Summary

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

Table A2e-2a Tornado Characteristics

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

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

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

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

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

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

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

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

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

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

Table A2e-3 Exceedance Probability for Each F-scale

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

Table A2e-4 SPC Data Analysis Summary by State

	NUREG/CR -2944 Region					Tornado F-scale							Point strike probability (per year)						Land Area
State	A	B	C	D	Years	F0	F1	F2	F3	F4	F5	Total	F0	F1	F2	F3	F4	F5	(mi²)
AL	X	X			46	165	364	323	129	36	14	1031	2.9e-05	3.2e-05	1.3e-05	3.7e-06	6.9e-07	4.3e-08	50750
AZ				X	44	90	57	11	2	0	0	160	6.7e-07	2.9e-07	3.6e-08	1.8e-09	0	0	113642
AR	X				46	198	298	331	149	31	0	1007	3.2e-05	3.5e-05	1.3e-05	2.4e-06	1.9e-07	0	52075
CA				X	45	142	58	21	2	0	0	223	5.1e-07	2.7e-07	6.0e-08	2.7e-09	0	0	155973
CO			X	X	46	616	441	99	15	1	0	1172	4.4e-06	2.0e-06	4.2e-07	3.9e-08	3.3e-11	0	103730
CT			X		46	9	29	20	5	2	0	65	1.1e-05	1.1e-05	3.6e-06	8.5e-07	2.2e-07	0	4845
DE			X		42	20	23	11	1	0	0	55	2.6e-05	1.5e-05	1.5e-06	6.4e-09	0	0	1955
DC*					1	1	0	0	0	0	0	1	1.3e-04	0	0	0	0	0	61
FL		X	X		46	1156	665	293	30	4	0	2148	1.5e-05	8.6e-06	2.2e-06	2.8e-07	2.0e-08	0	53997
GA		X			46	147	537	266	65	17	0	1032	2.9e-05	3.0e-05	1.2e-05	3.4e-06	4.3e-07	0	57919
ID				X	42	63	53	8	0	0	0	124	4.7e-07	1.9e-07	1.4e-08	0	0	0	82751
IN	X				46	246	336	263	108	77	8	1038	3.3e-05	3.5e-05	1.5e-05	5.2e-06	1.2e-06	6.7e-08	35870
IA	X				46	478	506	421	119	74	9	1607	3.7e-05	3.7e-05	1.4e-05	3.1e-06	6.1e-07	2.5e-08	55875
IL	X				46	431	440	316	113	39	3	1342	3.0e-05	2.7e-05	9.8e-06	2.5e-06	3.3e-07	2.1e-08	55875
KS	X	X			46	1111	610	404	168	54	16	2363	3.5e-05	3.0e-05	1.1e-05	3.0e-06	5.8e-07	1.1e-07	81823
KY	X				46	79	168	133	65	35	3	483	1.6e-05	1.7e-05	6.9e-06	1.8e-06	3.1e-07	1.4e-08	39732
LA		X			46	225	620	268	123	16	2	1254	2.4e-05	2.2e-05	6.9e-06	1.4e-06	1.2e-07	1.9e-08	43566
ME				X	42	21	44	17	0	0	0	82	1.8e-06	1.1e-06	1.7e-07	0	0	0	30865
MD			X		46	49	92	26	5	0	0	172	1.5e-05	9.2e-06	9.4e-07	8.2e-09	0	0	9775
MA			X		45	24	72	31	8	3	0	138	1.2e-05	1.1e-05	4.3e-06	1.6e-06	3.7e-07	0.0e+00	7838
MI		X	X		45	195	308	210	57	30	7	807	1.4e-05	1.4e-05	5.2e-06	1.4e-06	2.8e-07	1.4e-08	56809
MN		X	X		46	372	336	158	53	28	6	953	1.4e-05	1.2e-05	3.5e-06	7.2e-07	1.3e-07	6.6e-09	79617
MS	X	X			46	226	468	369	136	59	10	1268	4.4e-05	4.4e-05	1.7e-05	5.0e-06	1.0e-06	1.3e-08	46914

Table A2e-4 SPC Data Analysis Summary by State

	NUREG/CR -2944 Region					Tornado F-scale							Point strike probability (per year)						Land Area
State	A	B	C	D	Years	F0	F1	F2	F3	F4	F5	Total	F0	F1	F2	F3	F4	F5	(mi²)
MO	X				46	298	577	334	109	48	1	1367	1.8e-05	1.6e-05	5.3e-06	1.3e-06	2.3e-07	2.6e-11	68898
MT				X	44	174	42	33	4	0	0	253	1.0e-06	7.0e-07	2.3e-07	2.2e-08	0	0	145556
NE		X	X		46	827	585	255	105	42	4	1818	2.9e-05	2.9e-05	1.2e-05	3.5e-06	3.5e-07	1.6e-08	76878
NV				X	34	41	8	0	0	0	0	49	2.9e-07	4.0e-08	0	0	0	0	109806
NH				X	45	24	34	15	2	0	0	75	4.7e-06	2.4e-06	4.7e-07	1.1e-08	0	0	8969
NJ			X		45	43	58	23	4	0	0	128	1.7e-05	6.6e-06	7.9e-07	7.1e-09	0	0	7419
NM			X		46	261	104	31	4	0	0	400	1.5e-06	5.2e-07	8.0e-08	1.1e-09	0	0	121365
NY				X	44	101	106	35	21	5	0	268	7.6e-06	6.1e-06	2.3e-06	8.8e-07	2.2e-07	0	47224
NC			X		46	153	321	143	44	26	0	687	1.5e-05	1.4e-05	4.9e-06	1.5e-06	2.5e-07	0	48718
ND			X		46	490	211	91	28	7	3	830	4.7e-06	3.2e-06	1.1e-06	3.6e-07	9.1e-08	1.1e-08	68994
OH	X				46	157	321	166	53	27	9	733	2.1e-05	1.8e-05	5.6e-06	1.3e-06	3.0e-07	2.8e-08	40953
OK	X				46	845	808	626	209	83	9	2580	4.1e-05	3.9e-05	1.4e-05	3.6e-06	7.0e-07	5.5e-08	68679
OR				X	45	31	15	3	0	0	0	49	2.9e-07	1.5e-07	3.1e-08	0	0	0	96003
PA			X		46	93	220	143	26	22	2	506	9.4e-06	9.0e-06	3.3e-06	9.3e-07	2.0e-07	5.4e-09	44820
RI			X		23	3	4	1	0	0	0	8	1.9e-05	1.3e-05	1.7e-06	0	0	0	1045
SC		X			46	136	234	100	31	15	0	516	1.9e-05	1.9e-05	6.8e-06	1.8e-06	3.0e-07	0	30111
SD		X	X		46	651	259	197	57	7	1	1172	9.7e-06	8.1e-06	3.0e-06	7.7e-07	1.5e-07	1.2e-08	75898
TN	X				46	107	241	139	76	29	4	596	2.2e-05	2.2e-05	8.3e-06	2.1e-06	2.0e-07	1.7e-10	41220
TX		X	X		46	2632	1837	1067	317	76	5	5934	1.6e-05	1.3e-05	4.3e-06	1.1e-06	1.8e-07	3.8e-09	261914
UT				X	43	53	19	6	1	0	0	79	5.1e-07	3.2e-07	1.0e-07	2.8e-08	0	0	82168
VT				X	41	7	14	12	0	0	0	33	3.3e-06	2.0e-06	3.4e-07	0	0	0	9249
VA			X		45	84	132	68	28	6	0	318	8.5e-06	7.0e-06	2.0e-06	4.4e-07	7.1e-08	0	39598
WA				X	41	24	17	12	3	0	0	56	4.9e-07	9.6e-08	2.3e-08	3.6e-09	0	0	66582
WV			X		45	27	36	16	8	0	0	87	2.2e-06	2.4e-06	9.7e-07	2.5e-07	0	0	24087
WI		X	X		46	204	378	276	62	24	5	949	2.6e-05	2.4e-05	7.9e-06	1.4e-06	2.5e-07	3.3e-08	54314

Table A2e-4 SPC Data Analysis Summary by State																			
	NUREG/CR -2944 Region					Tornado F-scale							Point strike probability (per year)						Land Area
State	A	B	C	D	Years	F0	F1	F2	F3	F4	F5	Total	F0	F1	F2	F3	F4	F5	(mi^2)
WY				X	46	247	145	43	8	1	0	444	2.5e-06	1.2e-06	3.1e-07	7.1e-08	1.9e-08	0	97105
Sum						13776	13251	7834	2553	924	121	38459							3536342

* - DC was not included in the exceedance analysis.

Figure **A2e-1**

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

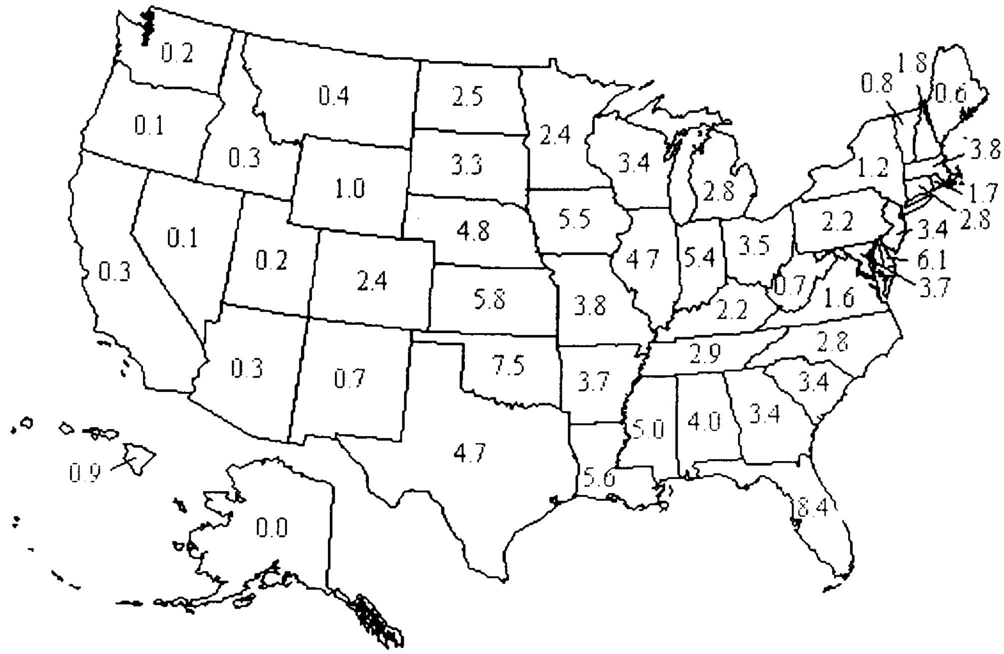


Figure A2 **e-2**

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

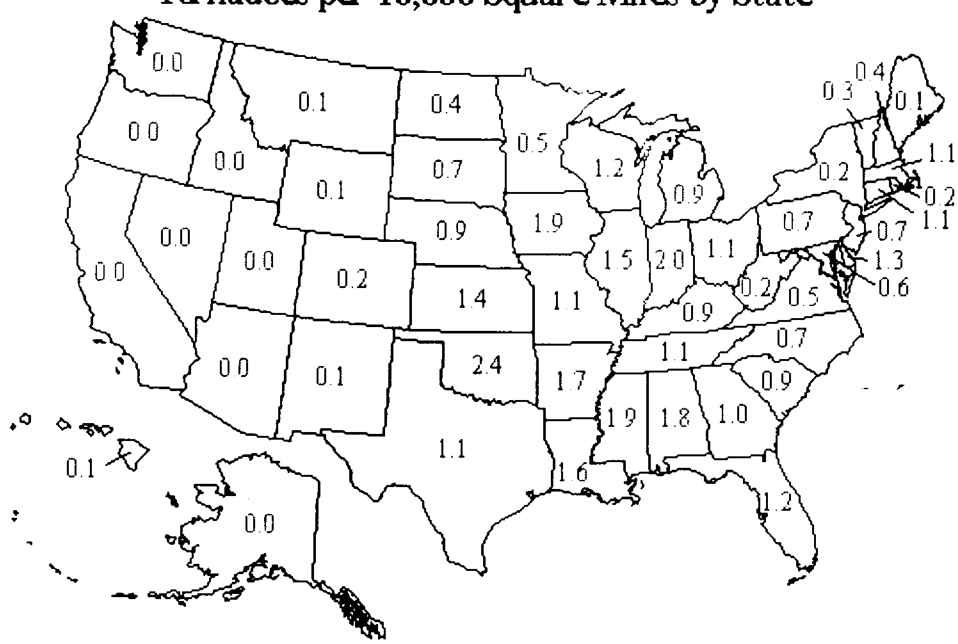
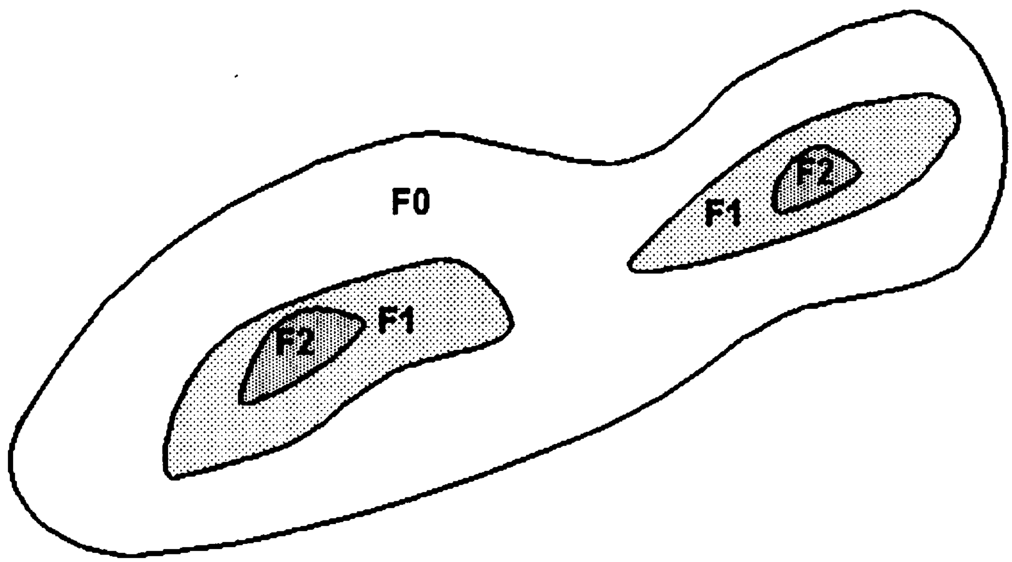


Figure A2e-3 Sketch of Hypothetical F2 Tornado Illustrating Variations



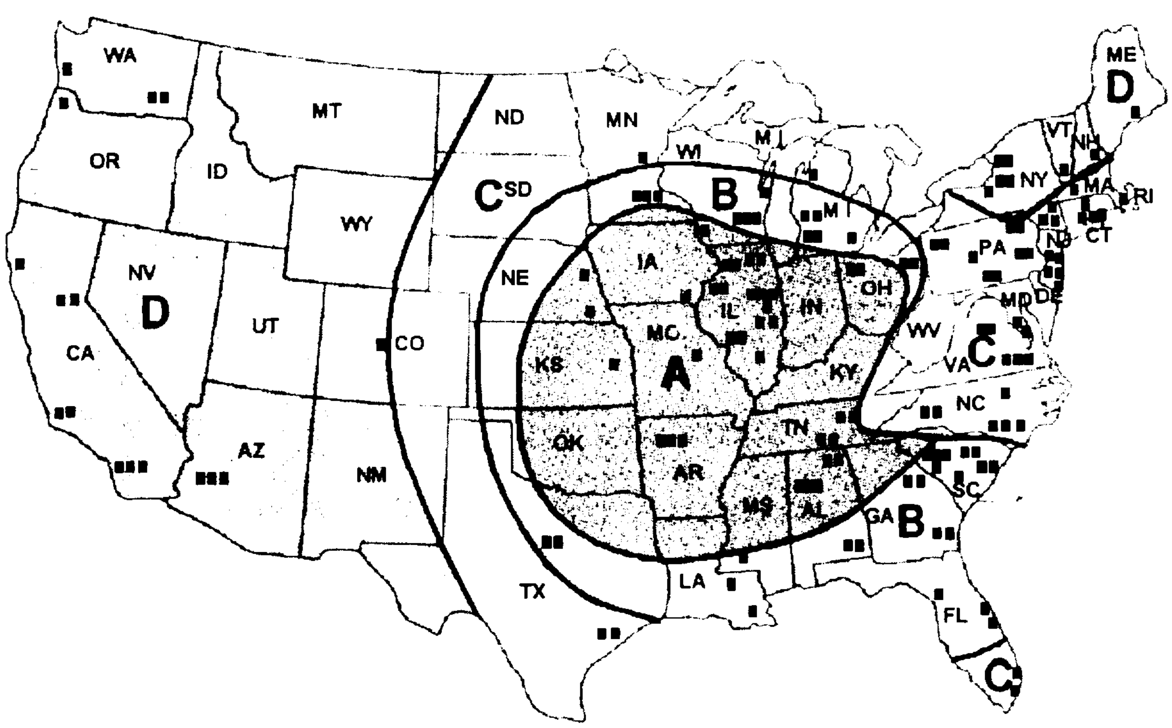
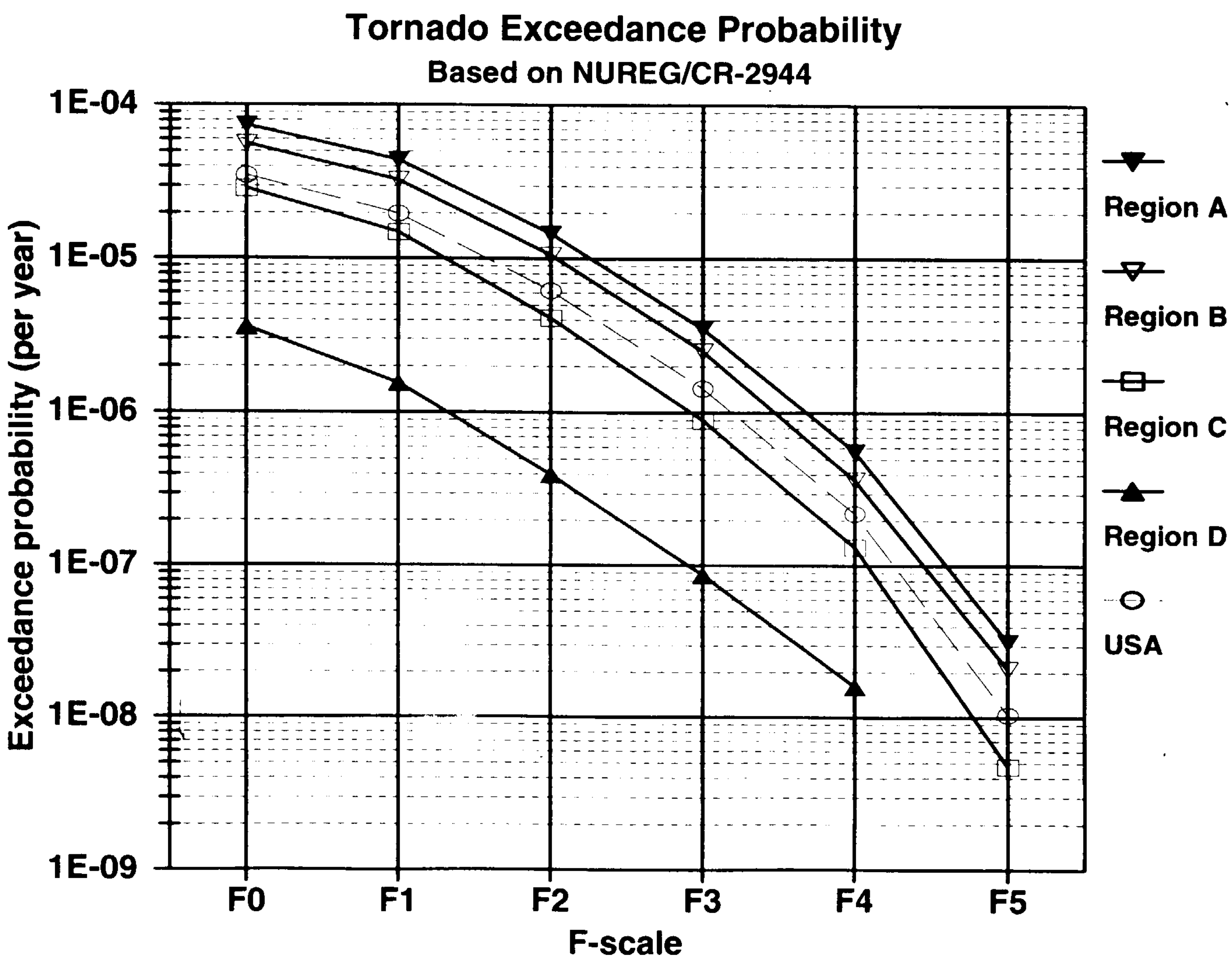


Figure A2e-4 Tornado Risk Regionalization Scheme (from NUREG/CR-2944)

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



Appendix 3 Criticality

3.1 Introduction

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

3.2 Qualitative Risk Study

3.2.1 Criticality in Spent Fuel Pool

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

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

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

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

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

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

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

To gain qualitative insights on the criticality events that are credible, the staff considered the sequences of events that must occur. For scenario 1, above this would require a heavy load drop into the a low density racked BWR pool compressing assemblies. From appendix 2 on heavy load drop, the likelihood of a heavy load drop from a single failure proof crane is approximately 2E-6 per year, assuming 100 cask movements per year at the decommissioning facility. From the load path analysis done for that appendix it was estimated that the load could be over or near the pool between 25% and 5% of the movement path length, dependent on plant specific layout specifics. The additional frequency reduction in the appendix to account for the fraction of time that the heavy load is lifted high enough to damage the pool liner is not applicable here because the fuel assemblies could be crushed without the same impact velocity being required as for the pool liner. Therefore, if we assume 10% load path vulnerability, we observe a potential initiating frequency for crushing of approximately 2E-7 per year (based upon 100 lifts per year). Criticality calculations show that even if the low density BWR assemblies were crushed by a transfer cask, it is "highly unlikely" that a configuration would be reached that would result in a severe reactivity event, such as a steam explosion which could damage and drain the spent fuel pool. The staff judges the chances of such a criticality event to be well below 1 chance in 100 even given that the transfer cask drops directly onto the assemblies. This would put the significant criticality likelihood well below 1E-8 per year, which justifies its exclusion from further consideration.

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

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

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

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

3.2.2 Deterministic Criticality Study

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

Assessment of the Potential for Criticality in Decommissioned Spent Fuel Pools

Tony P. Ulses
Reactor Systems Branch
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. Operating reactors typically only store highly reactive assemblies for short periods of time. 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 is 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 should 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

Formatted Version, Rev. 1 1/19/00

```
=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

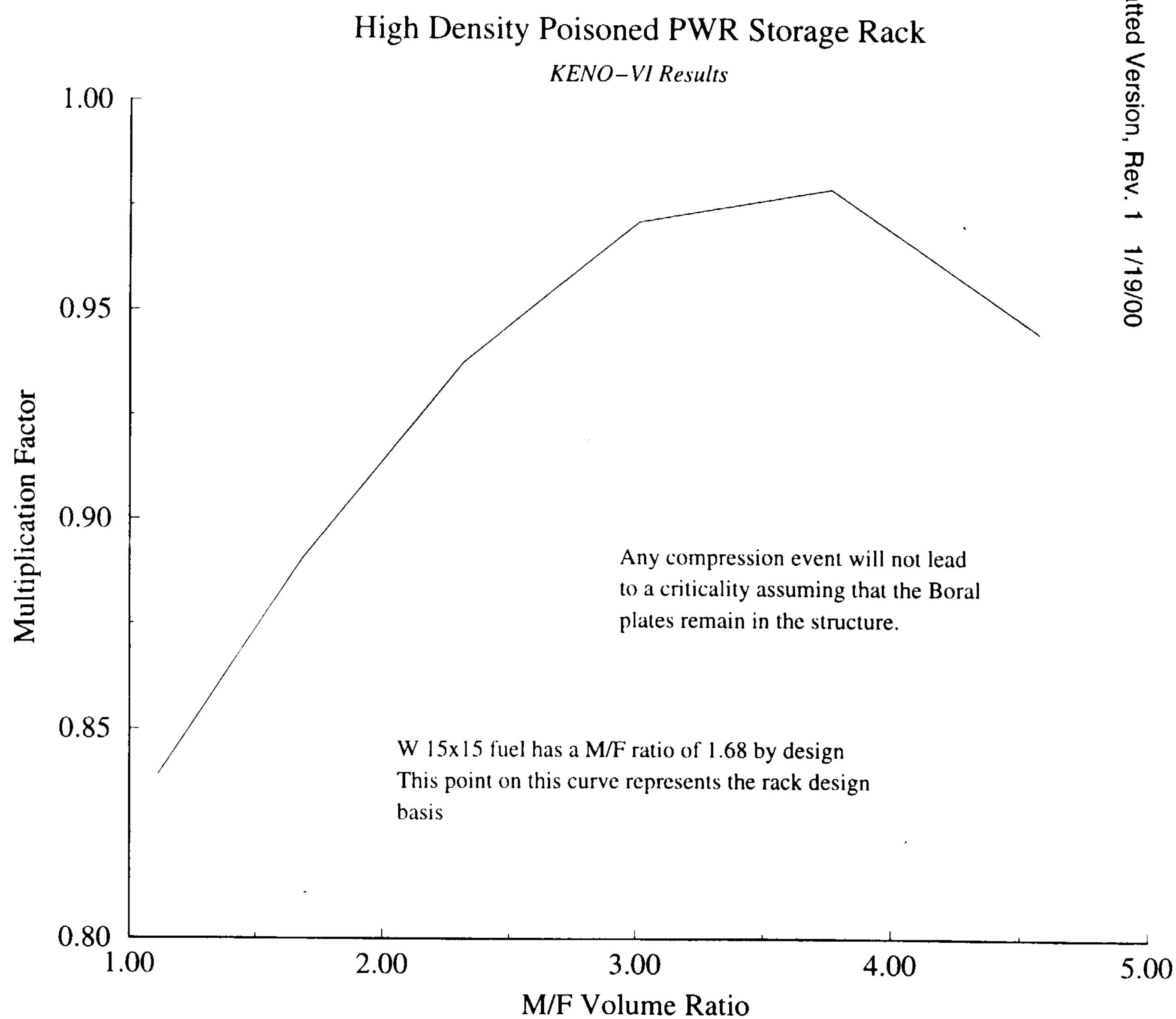


Figure 1 PWR High Density Storage Rack Eigenvalue following Compressive/Expansion Events

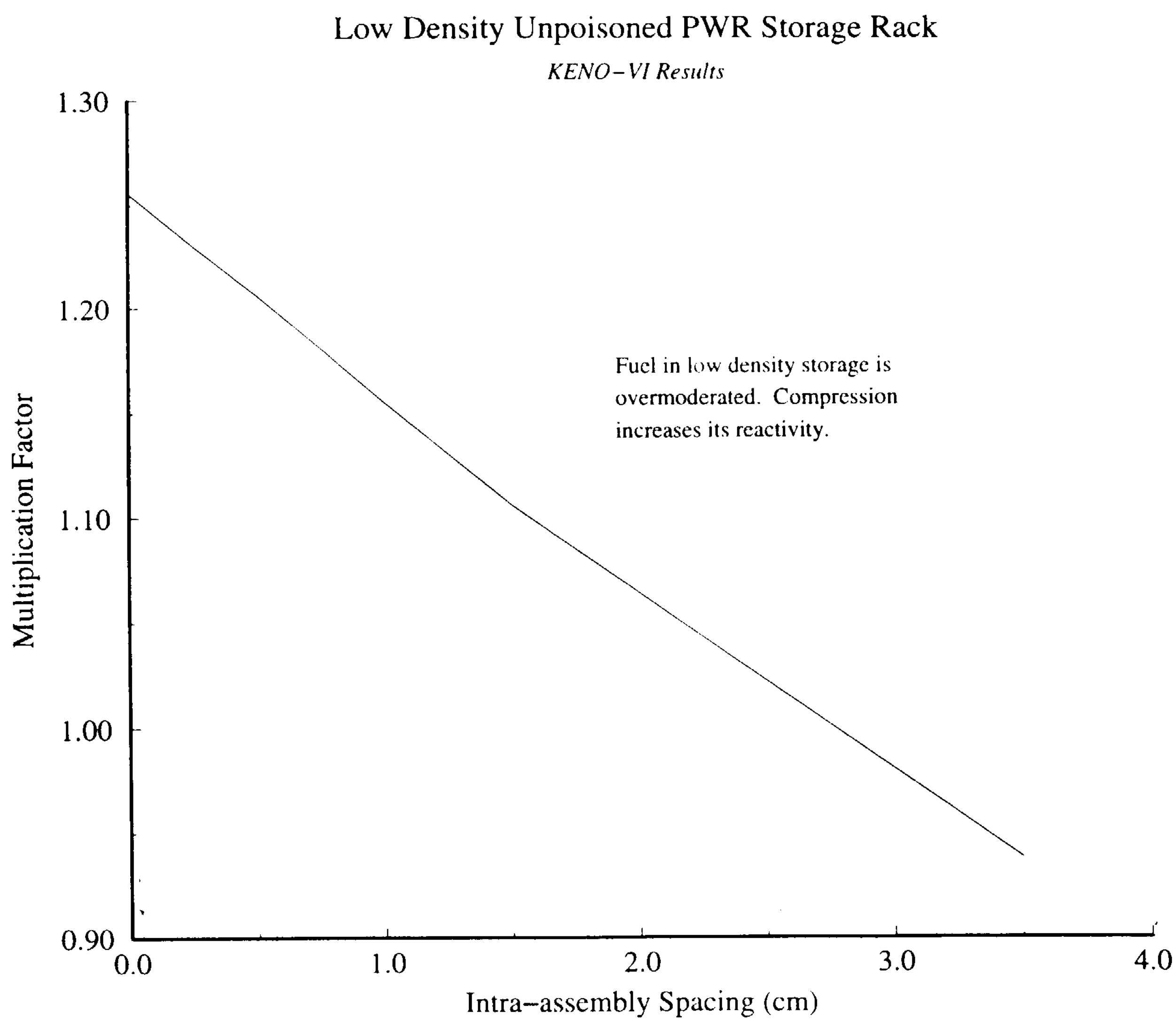


Figure 2 PWR Low Density Storage Rack Eigenvalue following Compressive/Expansion Events

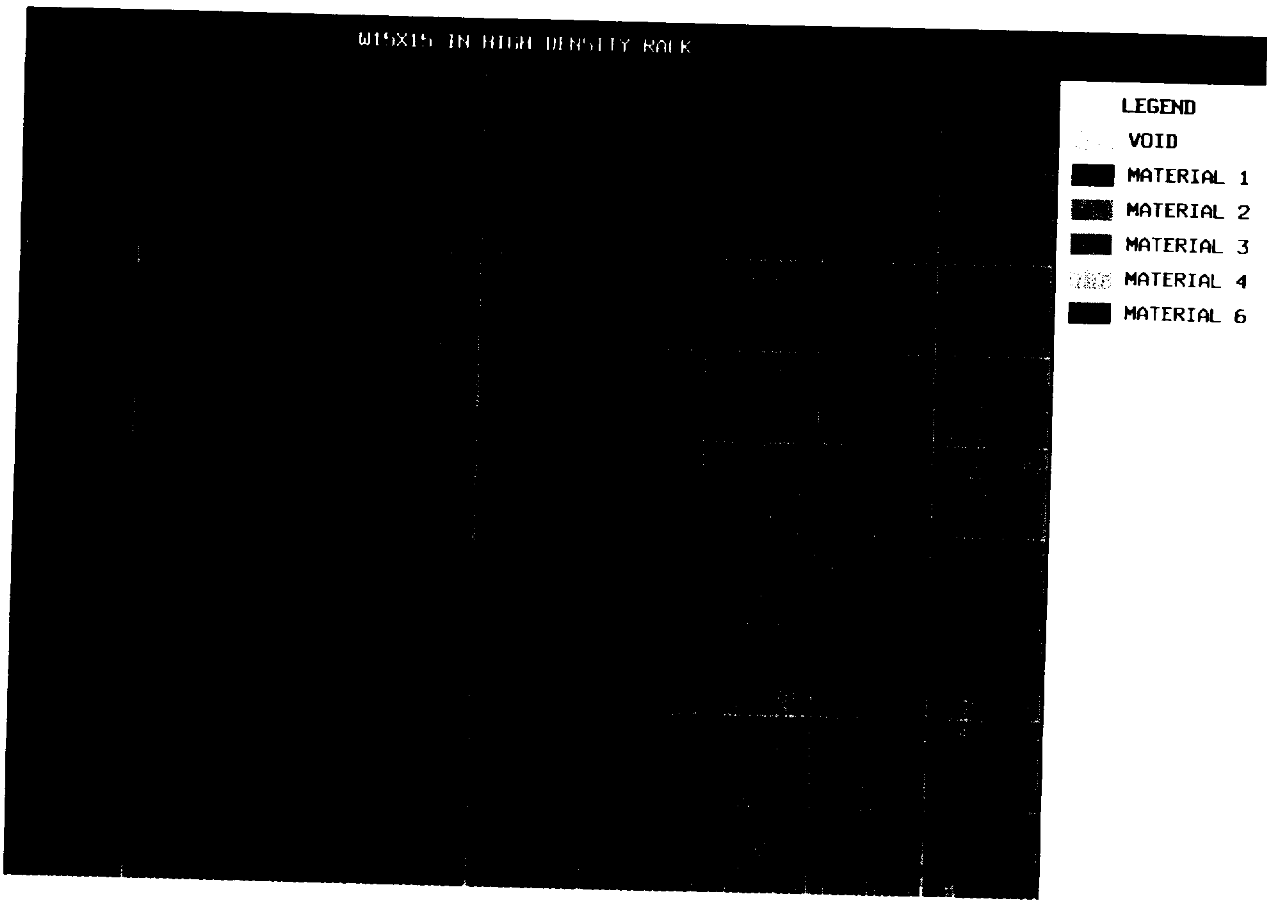


Figure 3 Sample Geometry Assuming 4 Assembly Spacing Between Most Reactive Assembly

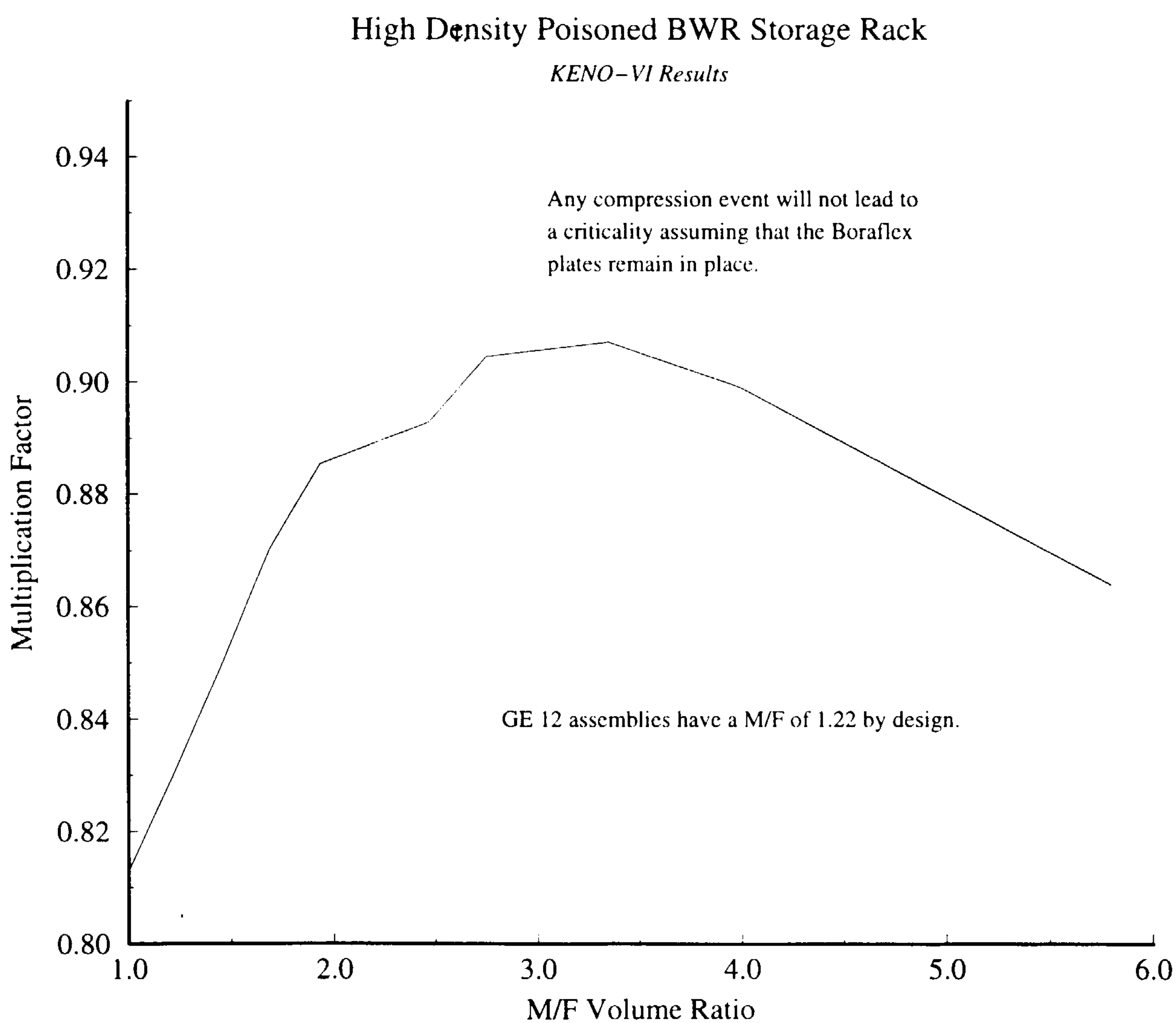


Figure 4 BWR High Density Storage Rack Eigenvalue following Compressive/Expansion Events

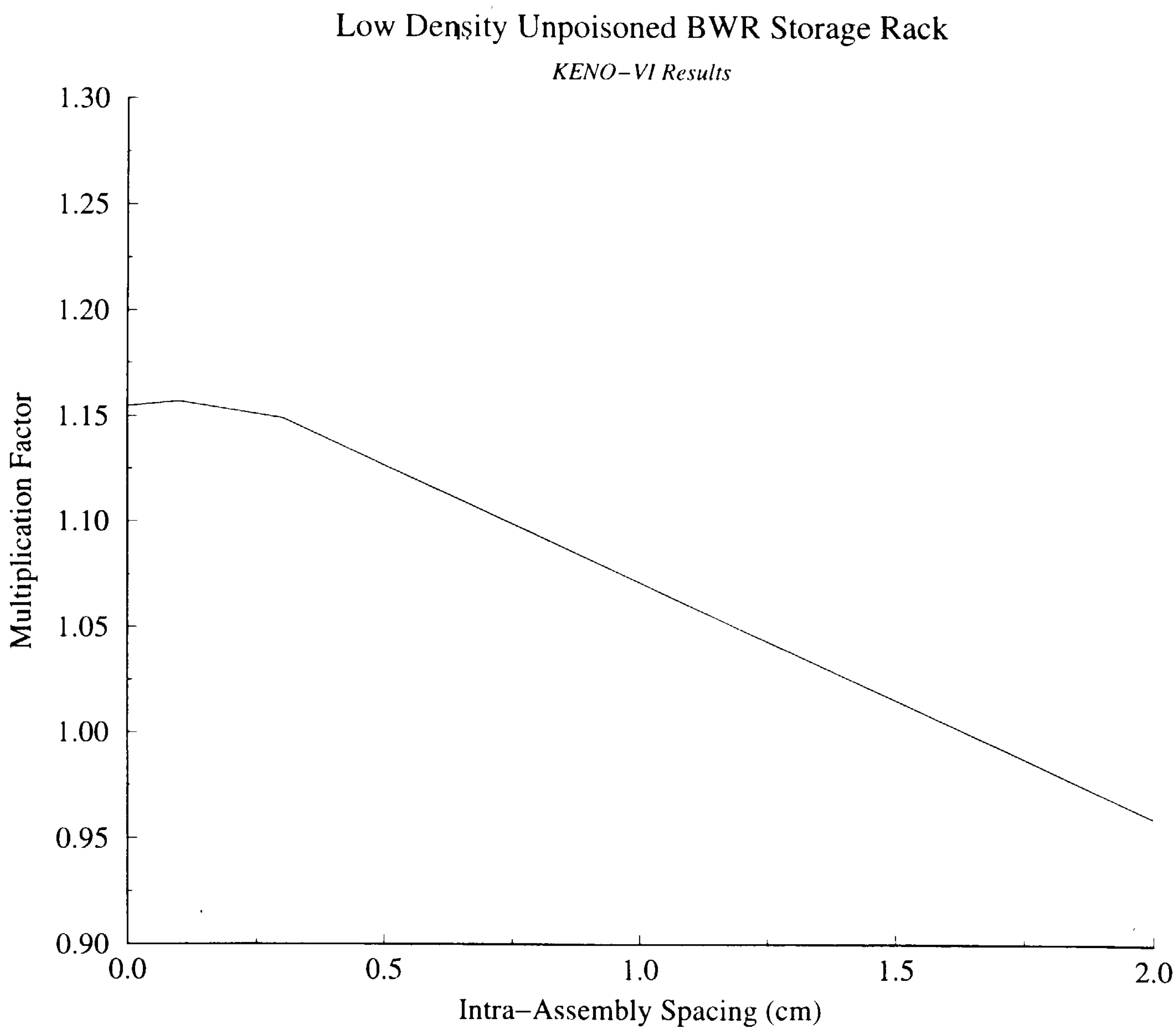


Figure 5 BWR Low Density Storage Rack Eigenvalue following Compressive/Expansion Events

Appendix 4 Consequence Assessment from Zirconium Fire

SMSAB-99-02

United States
Nuclear Regulatory Commission

Assessment of Offsite Consequences for a Severe Spent Fuel Pool Accident

Prepared by
Jason H. Schaperow

Safety Margins and Systems Analysis Branch
Division of Systems Analysis and Regulatory Effectiveness
Office of Nuclear Regulatory Research

November 1999

Introduction

As part of its generic study of spent fuel pool accidents, undertaken to develop generic, risk-informed regulatory requirements for plants that are being decommissioned, the Office of Nuclear Reactor Regulation (NRR) had requested the Office of Nuclear Regulatory Research (RES) to perform an evaluation of the offsite radiological consequences of a severe spent fuel pool accident. Accordingly, RES completed an in-house analysis of offsite radiological consequences, which included sensitivity and uncertainty analysis to assess the effect of critical parameters and assumptions. On May 25, 1999, RES forwarded to NRR a summary of the evaluation. A primary objective of the evaluation was to assess the effect of extended storage in a spent fuel pool, and the resulting radioactive decay, on offsite consequences. The evaluation showed about a factor-of-two reduction in prompt fatalities if the accident occurs after 1 year instead of after 30 days. The evaluation also showed that beginning evacuation three hours before the release begins reduces prompt fatalities by more than an order of magnitude.

The purpose of this report is to document the detailed technical basis of the offsite consequence evaluation. This report documents the offsite consequence calculations we performed using the MACCS code (MELCOR Accident Consequence Code System) and includes the input files used. In addition, this report documents follow-up calculations, performed since our earlier letter, to evaluate the importance of cesium to better understand why the consequence reduction from a year of decay was not greater. These follow-up calculations 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.

Previous Consequence Assessments

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 offsite 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 current report extends the earlier analyses by calculating offsite 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 offsite consequences. However, as part of this work, the sensitivity to a variety of other parameters was also evaluated.

The current analysis used the MACCS code[3] (version 2) to estimate offsite 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 offsite 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). Because the current analysis may also be used as part of the probabilistic risk analysis of spent fuel pool accidents for the Susquehanna plant which is a BWR, the spent fuel inventories for Millstone 1 which is also a BWR were used for this analysis. 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 Susquehanna 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	2.35E5
Kr-85	10.8y	1.41E6	1.39E6	1.33E6
Rb-86	18.7d	1.01E4	1.05E3	3.84E-2
Sr-89	50.5d	8.39E6	3.63E6	8.33E4
Sr-90	28.8y	1.42E7	1.42E7	1.39E7
Y-90	28.8y	1.43E7	1.42E7	1.39E7
Y-91	58.5d	1.18E7	5.75E6	2.21E5
Zr-95	64.0d	1.94E7	1.00E7	5.10E5
Nb-95	64.0d	2.54E7	1.70E7	1.11E6
Mo-99	2.7d	1.49E4	3.12E-3	0
Tc-99m	2.7d	1.43E4	3.01E-3	0
Ru-103	37.3d	1.53E7	5.21E6	4.07E4
Ru-106	1.0y	1.72E7	1.53E7	9.13E6
Sb-127	3.8d	8.21E3	1.39E-1	0
Te-127	109d	2.21E5	1.45E5	2.52E4
Te-127m	109d	2.18E5	1.48E5	2.57E4
Te-129	33.6d	2.74E5	7.79E4	2.68E2
Te-129m	33.6d	4.21E5	1.20E5	4.12E2

Te-132	3.2d	3.74E4	8.64E-2	0
I-131	8.0d	1.22E6	6.35E3	0
I-132	3.2d	3.85E4	8.90E-2	0
Xe-133	5.2d	7.29E5	2.30E2	0
Cs-134	2.1y	7.90E6	7.47E6	5.80E6
Cs-136	13.2d	2.05E5	8.13E3	3.91E-3
Cs-137	30.0y	2.02E7	2.01E7	1.97E7
Ba-140	12.8d	5.19E6	1.90E5	6.41E-2
La-140	12.8d	5.97E6	2.19E5	7.37E-2
Ce-141	32.5d	1.32E7	3.61E6	1.03E4
Ce-144	284.6d	2.64E7	2.27E7	1.16E7
Pr-143	13.6d	5.44E6	2.41E5	1.90E-1
Nd-147	11.0d	1.54E6	3.36E4	1.10E-3
Np-239	2.4d	5.59E4	2.88E3	2.88E3
Pu-238	87.7y	4.51E5	4.53E5	4.54E5
Pu-239	24100y	8.89E4	8.89E4	8.89E4
Pu-240	6560y	1.30E5	1.30E5	1.30E5
Pu-241	14.4y	2.29E7	2.27E7	2.19E7
Am-241	432.7y	2.88E5	2.94E5	3.21E5
Cm-242	162.8d	1.45E6	1.12E6	3.50E5
Cm-244	18.1y	2.27E5	2.25E5	2.19E5

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

Formatted Version, Rev. 1 1/19/00

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

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

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

Modeling of Emergency Response Actions and Other Areas

Modeling of emergency response actions was essentially the same as that used for Surry in

NUREG-1150. The timing of events is given in Table A4-4. Evacuation begins exactly two hours after emergency response officials receive notification to take protective measures. This results in the evacuation beginning approximately .8 hours after the offsite release ends. Only people within 10 miles of the 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 offsite emergency response officials	0	0
start time of offsite release	2400	.7
end time of offsite release	4200	1.2
evacuation begins	7200	2.0

Table A4-5 Evacuation Modeling

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

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

Offsite Consequence Results

MACCS calculations for a decommissioned reactor for accidents occurring 30 days, 90 days, and 1 year after final shutdown were performed to assess the magnitude of the decrease in the offsite consequences resulting from extended decay 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	3 hours	$1x10^{-6}$	99.5%

The Base Case was intended to model the offsite 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 input files for the Base Case are given in Appendix A. The results of the Base Case are shown in Table A4-7.

Table A4-7 Mean Consequences for the Base Case

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

Table A4-7 shows the offsite consequences for a severe 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 offsite consequences (i.e., societal dose and cancer fatalities) as a function of decay time, because they are controlled by inventories of radionuclides with long half-lives and relocation assumptions. However, these results also show about a factor-of-two reduction in the short-term consequences (i.e., prompt fatalities) from 30 days to 1 year of decay. (All of the prompt fatalities occur within 10 miles of the site.) As a rough check on the prompt fatality results, the change in decay power was evaluated for an operating reactor shut down for 30 days and for 1 year. The decay power decreased by about a factor of three. This is consistent with a factor-of-two decrease in prompt fatalities. The factor-of-three decrease in decay power by radioactive decay will also increase the time it takes to heat up the spent fuel, which provides additional time to take action to mitigate the accident.

The results of Case 1, which used a lower evacuation percentage than the Base Case, are identical to the results of the Base Case shown in Table A4-7. Case 1 used an evacuation percentage of 95%, while the Base Case used an evacuation percentage of 99.5%. Although it might be expected to see an increase in prompt fatalities from reducing the evacuation percentage, no such increase was observed. This is 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 offsite, there would be no change in offsite consequences as a result.

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

Table A4-12 Mean Consequences for Case 7

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

Comparison with Earlier Consequence Analyses

As a check on the above calculations and to provide additional insight into the consequence analysis for severe 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 Susquehanna 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 'ong-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 offsite consequences.

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

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

Conclusion

The primary objective of this evaluation was to assess the effect of extended storage in a spent fuel pool, and the resulting radioactive decay, on offsite consequences of a severe spent fuel pool accident at a decommissioned reactor. This evaluation was performed in support of the NRR 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 Safety 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 Check List and Supporting Stakeholder Documentation

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 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.

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 makeup water to the spent fuel pool. The plan will provide for remote alignment of the makeup 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 makeup system components will be performed and administrative controls for equipment out of service will be implemented to provide added assurance that the components would be available, if needed.

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

Formatted Version, Rev. 1 1/19/00

Appendix 7 Stakeholder Interactions

3. 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

4. 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.

5. 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 to refine the analysis in the June 1999 draft report. The HRA results were also subjected to an independent technical review. This topic is discussed in Appendices 2.

6. 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, 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.

6. 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 heatup 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.

BC's: John Hannon (JH), Rich Barrett (RB), Jared Wermiel (JW)

The purpose of this table is to show that you agree/disagree with the chapter version show in this folder, prior to tech. editor comments and your formal concurrence. Please initial the column which applies to you and return to Tanya Eaton ASAP. Thanks.

Chapter/Appendix	Completed (Yes/No)	Concur (use initials)	Concur w/comments (See comments written in chapter)	Do not concur (initial)
Executive Summary				
Ch 1.0 Background				
Ch 2.0 Risk-Informed Decision Making				
Chapter 3.0 Risk Assessment of Spent Fuel Pools at Decom. Plants				
Chapter 4.0 Implications of SFP Risk for Regulatory Requirements				
Appendix 1 Thermal Hydraulics				
Appendix 2 Frequency of Fuel Uncovery				
Appendix 2a INEL				
Appendix 2b Seismic				
Appendix 2c Heavy Loads				
Appendix 2d Aircraft				
Appendix 2e Tornado/Weather	Yes 1/14/00			
Appendix 3 Criticality				
Appendix 4 Consequence	Yes 1/14/00			
Appendix 5 Seismic Checklist	Yes 1/14/00			
Appendix 6 NEI Commitments				
Appendix 7 Kennedy's Report	Yes 1/14/00			
Appendix 8 Gareth's HRA Study	Yes 1/14/00			
Appendix 9 Stakeholder Concerns				