NUREG-1738

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

U.S. Nuclear Regulatory Commission
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Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants

Manuscript Completed: January 2001
Date Published: February 2001

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ABSTRACT

This study contains the results of the NRC staff's evaluation of the potential accident risk in a spent fuel pool at decommissioning plants in the United States. This study was prepared to provide a technical basis for decommissioning rulemaking for permanently shutdown nuclear power plants. This study describes a modeling approach of a typical decommissioning plant with design assumptions and industry commitments; the thermal-hydraulic analyses performed to evaluate the behavior of spent fuel stored in the spent fuel pool at decommissioning plants; the risk assessment of spent fuel pool accidents; the consequence calculations; and the sensitivity study and implications for decommissioning regulatory requirements. Preliminary drafts of this study were issued for public comments and technical reviews in June 1999 and February 2000. Comments from interested stakeholders, the Advisory Committee on Reactor Safeguards, and other technical reviewers have been taken into account in preparing this study. A broad quality review was also carried out at the Idaho National Engineering and Environment Laboratory, and a panel of human reliability analysis experts evaluated the report's assumptions, methods, and modeling. Public comments on draft versions of this study are discussed in Appendix 6 of this NUREG.

Technical Study of Spent Fuel Pool Accidents at Decommissioning Plants

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EXECUTIVE SUMMARY

This report documents a study of spent fuel pool (SFP) accident risk at decommissioning nuclear power plants. The study was undertaken to support development of a risk-informed technical basis for reviewing exemption requests and a regulatory framework for integrated rulemaking.

The staff published a draft study in February 2000 for public comment and significant comments were received from the public and the Advisory Committee on Reactor Safeguards (ACRS). To address these comments the staff did further analyses and also added sensitivity studies on evacuation timing to assess the risk significance of relaxed offsite emergency preparedness requirements during decommissioning. The staff based its sensitivity assessment on the guidance in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis." The staff's analyses and conclusions apply to decommissioning facilities with SFPs that meet the design and operational characteristics assumed in the risk analysis. These characteristics are identified in the study as industry decommissioning commitments (IDCs) and staff decommissioning assumptions (SDAs). Provisions for confirmation of these characteristics would need to be an integral part of rulemaking.

The results of the study indicate that the risk at SFPs is low and well within the Commission's Quantitative Health Objectives (QHOs). The risk is low because of the very low likelihood of a zirconium fire even though the consequences from a zirconium fire could be serious. The results are shown in Figures ES-1 and ES-2. Because of the importance of seismic events in the analysis, and the considerable uncertainty in seismic hazard estimates, the results are presented for both the Lawrence Livermore National Laboratory (LLNL) and the Electric Power Research Institute (EPRI) seismic hazard estimates. In addition, to address a concern raised by the ACRS, the results also include a sensitivity to a large ruthenium and fuel fines release fraction. As illustrated in the figures, the risk is well below the QHOs for both the individual risk of early fatality and the individual risk of latent cancer fatality.

The study includes use of a pool performance guideline (PPG) as an indicator of low risk at decommissioning facilities. The recommended PPG value for events leading to uncovery of the spent fuel was based on similarities in the consequences from a SFP zirconium fire to the consequences from a large early release event at an operating reactor. A value equal to the large early release frequency (LERF) criterion ($1x10^{-5}$ per year) was recommended for the PPG. By maintaining the frequency of events leading to uncovery of the spent fuel at decommissioning facilities below the PPG, the risk from zirconium fires will be low and consistent with the guidance in RG 1.174 for allowing changes to the plant licensing basis that slightly increase risk. With one exception (the H.B. Robinson site) all Central and Eastern sites which implement the IDCs and SDAs would be expected to meet the PPG regardless of whether LLNL or EPRI seismic hazard estimates are assumed. The Robinson site would satisfy the PPG if the EPRI hazard estimate is applied but not if the LLNL hazard is used. Therefore, Western sites and Robinson would need to be considered on a site-specific basis because of important differences in seismically induced failure potential of the SFPs.

The appropriateness of the PPG was questioned by the ACRS in view of potential effects of the fission product ruthenium, the release of fuel fines, and the effects of revised plume parameters. The staff added sensitivity studies to its analyses to examine these issues. The consequences of a significant release of ruthenium and fuel fines were found to be notable, but not so important as to render inappropriate the staff's proposed PPG of 1×10^{-5} per year. The plume parameter sensitivities were found to be of lesser significance.

In its thermal-hydraulic analysis, documented in Appendix 1A, the staff concluded that it was not feasible, without numerous constraints, to establish a generic decay heat level (and therefore a decay time) beyond which a zirconium fire is physically impossible. Heat removal is very sensitive to these additional constraints, which involve factors such as fuel assembly geometry and SFP rack configuration. However, fuel assembly geometry and rack configuration are plant specific, and both are subject to unpredictable changes after an earthquake or cask drop that drains the pool. Therefore, since a non-negligible decay heat source lasts many years and since configurations ensuring sufficient air flow for cooling cannot be assured, the possibility of reaching the zirconium ignition temperature cannot be precluded on a generic basis.

The staff found that the event sequences important to risk at decommissioning plants are limited to large earthquakes and cask drop events. For emergency planning (EP) assessments this is an important difference relative to operating plants where typically a large number of different sequences make significant contributions to risk. Relaxation of offsite EP a few months after shutdown resulted in only a "small change" in risk, consistent with the guidance of RG 1.174. Figures ES-1 and ES-2 illustrate this finding. The change in risk due to relaxation of offsite EP is small because the overall risk is low, and because even under current EP requirements, EP was judged to have marginal impact on evacuation effectiveness in the severe earthquakes that dominate SFP risk. All other sequences including cask drops (for which emergency planning is expected to be more effective) are too low in likelihood to have a significant impact on risk. For comparison, at operating reactors additional risk-significant accidents for which EP is expected to provide dose savings are on the order of 1×10^{-5} per year, while for decommissioning facilities, the largest contributor for which EP would provide dose savings is about two orders of magnitude lower (cask drop sequence at 2×10^{-7} per year).[1] Other policy considerations beyond the scope of this technical study will need to be considered for EP requirement revisions and previous exemptions because a criteria of sufficient cooling to preclude a fire cannot be satisfied on a generic basis.

Insurance does not lend itself to a "small change in risk" analysis because insurance affects neither the probability nor the consequences of an event. As seen in figure ES-2, as long as a zirconium fire is possible, the long-term consequences of an SFP fire may be significant. These long-term consequences (and risk) decrease very slowly because cesium-137 has a half life of approximately 30 years. The thermal-hydraulic analysis indicates that when air flow has been restricted, such as might occur after a cask drop or major earthquake, the possibility of a fire lasts many years and a criterion of "sufficient cooling to preclude a fire" can not be defined on a

[1]Consistent with PRA limitations and practice, contributions to risk from safeguards events are not included in these frequency estimates. EP might also provide dose savings in such events.

generic basis. Other policy considerations beyond the scope of this technical study will therefore need to be considered for insurance requirements.

The study also discusses implications for security provisions at decommissioning plants. For security, risk insights can be used to determine what targets are important to protect against sabotage. However, any revisions in security provisions should be constrained by an effectiveness assessment of the safeguards provisions against a design-basis threat. Because the possibility of a zirconium fire leading to a large fission product release cannot be ruled out even many years after final shutdown, the safeguards provisions at decommissioning plants should undergo further review. The results of this study may have implications on previous exemptions at decommissioning sites, devitalization of spent fuel pools at operating reactors and related regulatory activities.

The staff's risk analyses were complicated by a lack of data on severe-earthquake return frequencies, source term generation in an air environment, and SFP design variability. Although the staff believes that decommissioning rulemaking can proceed on the basis of the current assessment, more research may be useful to reduce uncertainties and to provide insights on operating reactor safety. In particular, the staff believes that research may be useful on source term generation in air, which could also be important to the risk of accidents at operating reactors during shutdowns, when the reactor coolant system and the primary containment may both be open.

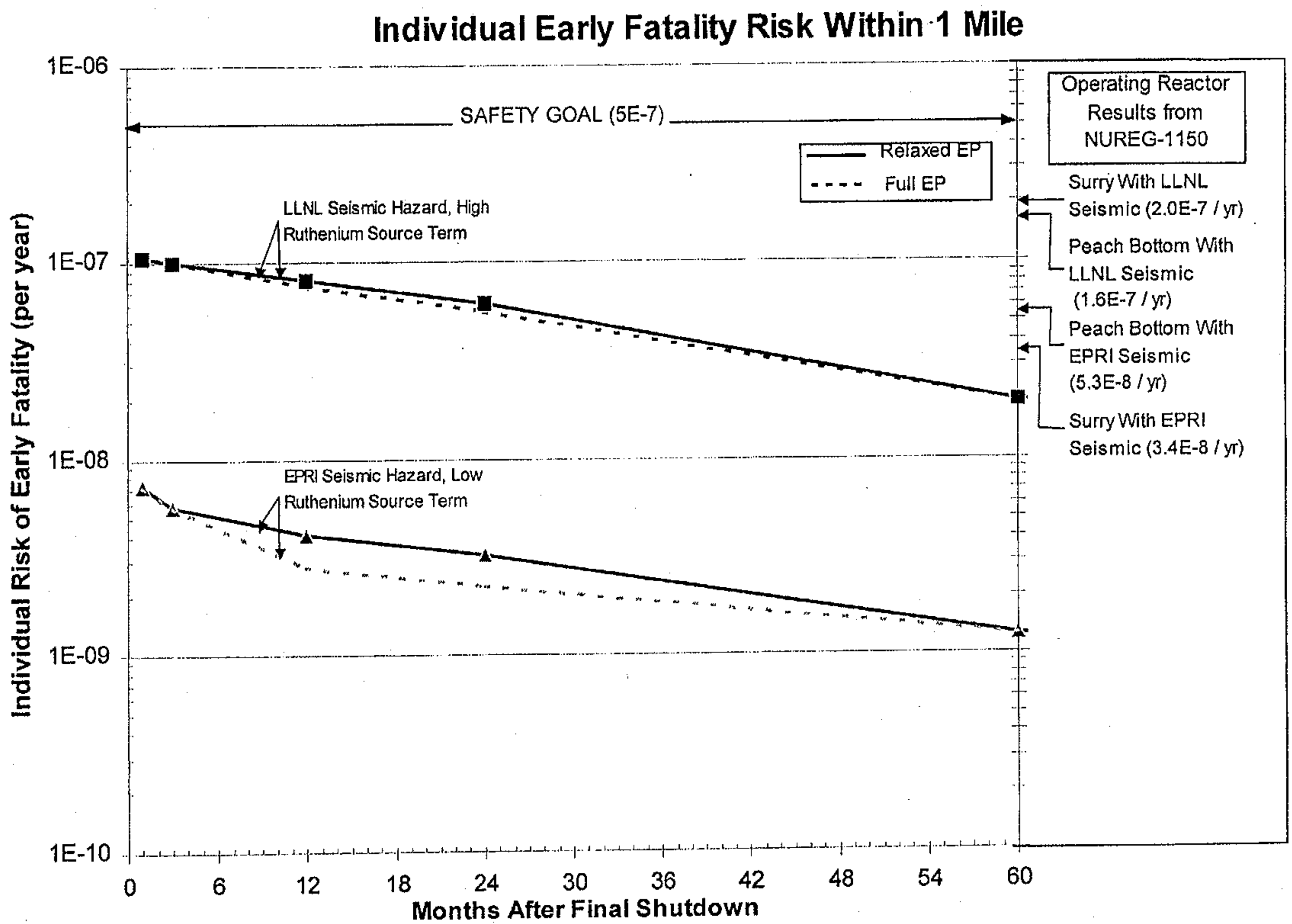
In summary, the study finds that:

1. The risk at decommissioning plants is low and well within the Commission's safety goals. The risk is low because of the very low likelihood of a zirconium fire even though the consequences from a zirconium fire could be serious.

2. The overall low risk in conjunction with important differences in dominant sequences relative to operating reactors, results in a small change in risk at decommissioning plants if offsite emergency planning is relaxed. The change is consistent with staff guidelines for small increases in risk.

3. Insurance, security, and emergency planning requirement revisions need to be considered in light of other policy considerations, because a criterion of "sufficient cooling to preclude a fire" cannot be satisfied on a generic basis.

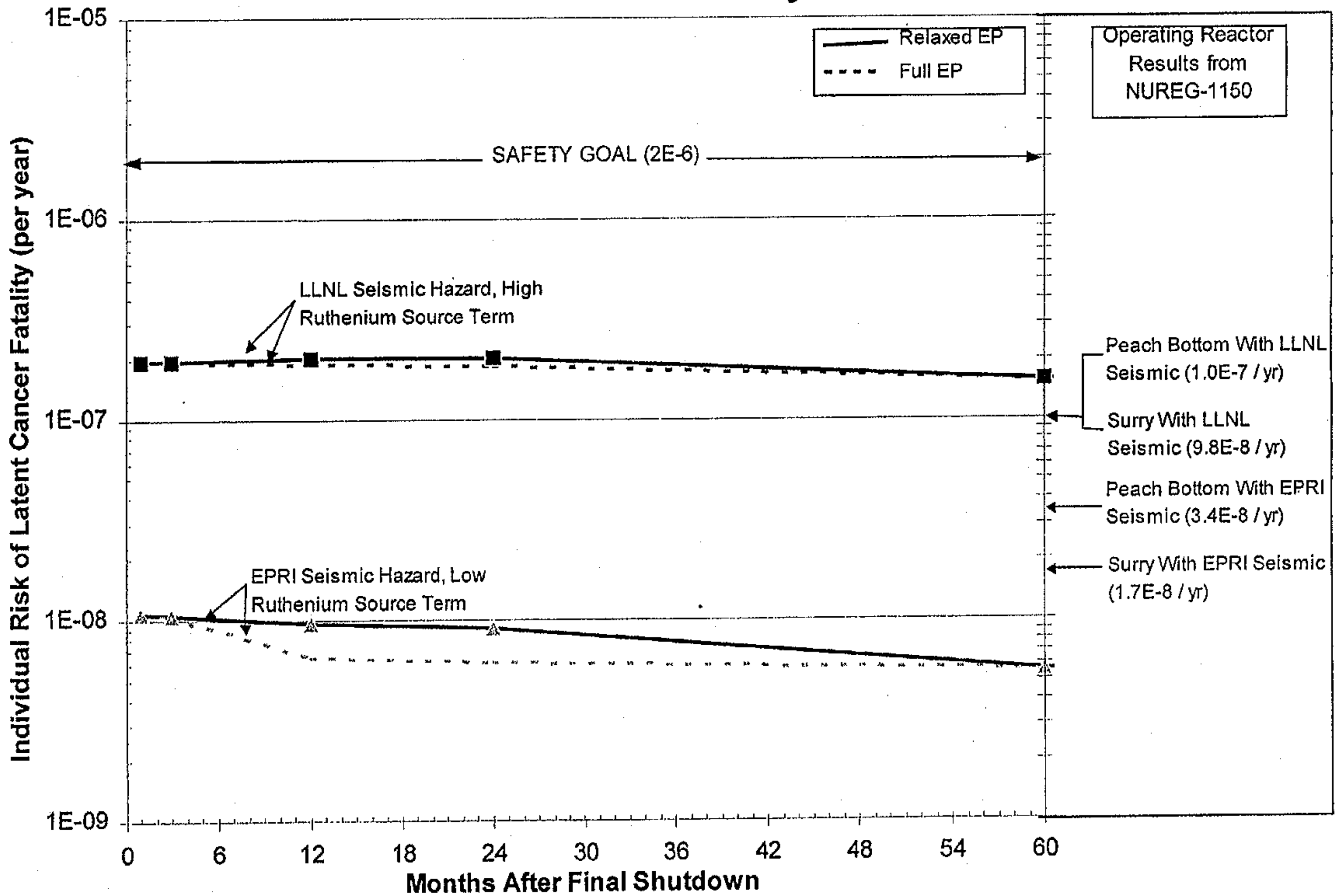
4. Research on source term generation in an air environment would be useful for reducing uncertainties.



Individual Early Fatality Risk Within 1 Mile
Individual Risk of Early Fatality (per year)
1E-06
1E-07
1E-08
1E-09
1E-10
SAFETY GOAL (5E-7)
Relaxed EP
Full EP
LLNL Seismic Hazard, High Ruthenium Source Term
EPRI Seismic Hazard, Low Ruthenium Source Term
Operating Reactor Results from NUREG-1150
Surry With LLNL Seismic (2.0E-7 / yr)
Peach Bottom With LLNL Seismic (1.6E-7 / yr)
Peach Bottom With EPRI Seismic (5.3E-8 / yr)
Surry With EPRI Seismic (3.4E-8 / yr)
0
6
12
18
24
30
36
42
48
54
60
Months After Final Shutdown

Figure ES-1

Individual Latent Cancer Fatality Risk Within 10 Miles



ES-2

ACKNOWLEDGMENTS

Major portions of the study were done by NRC staff with supporting from subcontractors and Sandia, Argonne, and Idaho Members. These people are listed below.

Glenn Kelly
Michael Cheok
Gareth Parry
Mark Rubin
Robert Palla
Goutam Bagchi
Robert Rothman
Larry Kopp
Anthony Ulses
Joseph Staudenmeier
Walt Jenson
Christopher Boyd
Jason Schaperow
Charles Tinkler
Sudhamay Basu
Jocelyn Mitchell
Edward Throm
Tanya Eaton
David Diec
Diane Jackson
John Lehning
Paula Magnanelli
United States Nuclear Regulatory Commission

Julie Gregory
Donnie Whitehead
and Randall Gauntt
Sandia National Laboratories

Hee Chung
Argonne National Laboratory

Harold Blackman
Soli Kharicha
Idaho National Engineering and Environment Laboratories

Lawrence Dickson
Atomic Energy of Canada, Limited

Robert Kennedy
RPK Structural Mechanics Consulting, Inc.

Dennis Bley
Buttonwood Consulting

1.0 INTRODUCTION

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

To improve regulatory consistency and predictability, the NRC undertook this effort to improve the regulatory framework applicable to decommissioning plants. This framework utilized 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 forms a technical foundation to be used as one input to developing regulatory changes, as well as a part of the basis for requesting and approving exemption requests until rulemaking is completed.

In support of this objective, the NRC staff has completed an assessment of SFP risks. This assessment utilized probabilistic risk assessment (PRA) methods and was developed from analytical studies in the areas of thermal hydraulics, reactivity, systems analysis, human reliability analysis, seismic and structural analysis, external hazards assessment, and offsite radiological consequences. The focus of the risk assessment was to identify potential severe accident scenarios at decommissioning plants and to estimate the likelihood and consequences of these scenarios. The staff also examined the offsite EP for decommissioning plants using an analysis strategy consistent with the principles of Regulatory Guide (RG) 1.174.

Preliminary versions of this study were issued for public comment and technical review in June 1999 and February 2000. Comments received from stakeholders and other technical reviewers have been considered in preparing this assessment. Quality assessment of the staff's preliminary analysis has been aided by a small panel of human reliability analysis (HRA) experts who evaluated the human performance analysis assumptions, methods and modeling. A broad quality review was carried out at the Idaho National Engineering and Environmental Laboratory (INEEL).

The study provides insights for the design and operation of SFP cooling and inventory makeup systems and practices and procedures necessary to ensure high levels of operator performance during off-normal conditions. The study concludes that, with the fulfillment of industry commitments and satisfaction of a number of important staff assumptions, the risks from SFPs can be sufficiently low to evaluate exemptions involving small changes to risk parameters and to contribute to the basis for related rulemaking.

As a measure of whether the risks from SFPs at decommissioning plants were sufficiently low to allow small changes to risk parameters, the concept of a pool performance guideline (PPG) was presented in the February 2000 study based on the principles of RG 1.174. In the study, the staff stated that consequences of an SFP fire are sufficiently severe that the RG 1.174 large early release frequency baseline of $1x10^{-5}$ per reactor year is an appropriate frequency guideline for a decommissioning plant SFP risk and a useful measure in combination with other factors such as accident progression timing, for assessing features, systems, and operator performance for a spent fuel pool in a decommissioning plant. Like the February 2000 study, this study uses the PPG of $1x10^{-5}$ per reactor year as the baseline frequency for a zirconium fire in the SFP.

The study is divided into three main parts. The first (Section 2) is a summary of the thermal-hydraulic analysis performed for SFPs at decommissioning plants. The second (Section 3) discusses how the principles of risk informed regulation are addressed by proposed changes. The third (Section 4) discusses the implications of the study for decommissioning regulatory requirements.

2.0 THERMAL-HYDRAULIC ANALYSES

Analyses were performed to evaluate the thermal-hydraulic characteristics of spent fuel stored in the spent fuel pools (SFPs) of decommissioning plants and determine the time available for plant operators to take actions to prevent a zirconium fire. These are discussed in Appendix 1A. The focus was the time available before fuel uncovery and the time available before the zirconium ignites after fuel uncovery. These times were utilized in performing the risk assessment discussed in Section 3.

To establish the times available before fuel uncovery, calculations were performed to determine the time to heat the SFP coolant to a point of boiling and then boil the coolant down to 3 feet above the top of the fuel. As can be seen in Table 2.1 below, the time available to take actions before any fuel uncovery is 100 hours or more for an SFP in which pressurized-water reactor (PWR) fuel has decayed at least 60 days.

Table 2.1 Time to Heatup and Boiloff SFP Inventory Down to 3 Feet Above Top of Fuel (60 GWD/MTU)

DECAY TIME	PWR	BWR
60 days	100 hours (>4 days)	145 hours (>6 days)
1 year	195 hours (>8 days)	253 hours (>10 days)
2 years	272 hours (>11 days)	337 hours (>14 days)
5 years	400 hours (>16 days)	459 hours (>19 days)
10 years	476 hours (>19 days)	532 hours (>22 days)

The analyses in Appendix 1A determined that the amount of time available (after complete fuel uncovery) before a zirconium fire depends on various factors, including decay heat rate, fuel burnup, fuel storage configuration, building ventilation rates and air flow paths, and fuel cladding oxidation rates. While the February 2000 study indicated that for the cases analyzed a required decay time of 5 years would preclude a zirconium fire, the revised analyses show that it is not feasible, without numerous constraints, to define a generic decay heat level (and therefore decay time) beyond which a zirconium fire is not physically possible. Heat removal is very sensitive to these constraints, and two of these constraints, fuel assembly geometry and spent fuel pool rack configuration, are plant specific. Both are also subject to unpredictable changes as a result of the severe seismic, cask drop, and possibly other dynamic events which could rapidly drain the pool. Therefore, since the decay heat source remains nonnegligible for many years and since configurations that ensure sufficient air flow[2] for cooling cannot be

[2]Although a reduced air flow condition could reduce the oxygen levels to a point where a fire would not be possible, there is sufficient uncertainty in the available data as to when this level would be reached and if it could be maintained. It is not possible to predict when a

assured, a zirconium fire cannot be precluded, although the likelihood may be reduced by accident management measures.

Figure 2.1 plots the heatup time air-cooled PWR and BWR fuel take to heat up from 30 °C to 900 °C versus time since reactor shutdown. The figure shows that after 4 years, PWR fuel could reach the point of fission product release in about 24 hours. Figure 2.2 shows the timing of the event by comparing the air-cooled calculations to an adiabatic heatup calculation for PWR fuel with a burnup of 60 GWD/MTU. The figure indicates an unrealistic result that until 2 years have passed the air-cooled heatup rates are faster than the adiabatic heatup rates. This is because the air-cooled case includes heat addition from oxidation while the adiabatic case does not. In the early years after shutdown, the additional heat source from oxidation at higher temperatures is high enough to offset any benefit from air cooling. This result is discussed further in Appendix 1A. The results using obstructed airflow (adiabatic heatup) show that at 5 years after shutdown, the release of fission products may occur approximately 24 hours after the accident.

In summary, 60 days after reactor shutdown for boildown type events, there is considerable time (>100 hours) to take action to preclude a fission product release or zirconium fire before uncovering the top of the fuel. However, if the fuel is uncovered, heatup to the zirconium ignition temperature during the first years after shutdown would take less than 10 hours even with unobstructed air flow. After 5 years, the heatup would take at least 24 hours even with obstructed air flow cases. Therefore, a zirconium fire would still be possible after 5 years for cases involving obstructed air flow and unsuccessful accident management measures. These results and how they affect SFP risk and decommissioning regulations are discussed in Sections 3 and 4 of this study.

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zirconium fire would not occur because of a lack of oxygen. Blockage of the air flow around the fuel could be caused by collapsed structures and/or a partial draindown of the SFP coolant or by reconfiguration of the fuel assemblies during a seismic event or heavy load drop. A loss of SFP building ventilation could also preclude or inhibit effective cooling. As discussed in Appendix 1A, air flow blockage without any recovery actions could result in a near-adiabatic fuel heatup and a zirconium fire even after 5 years.

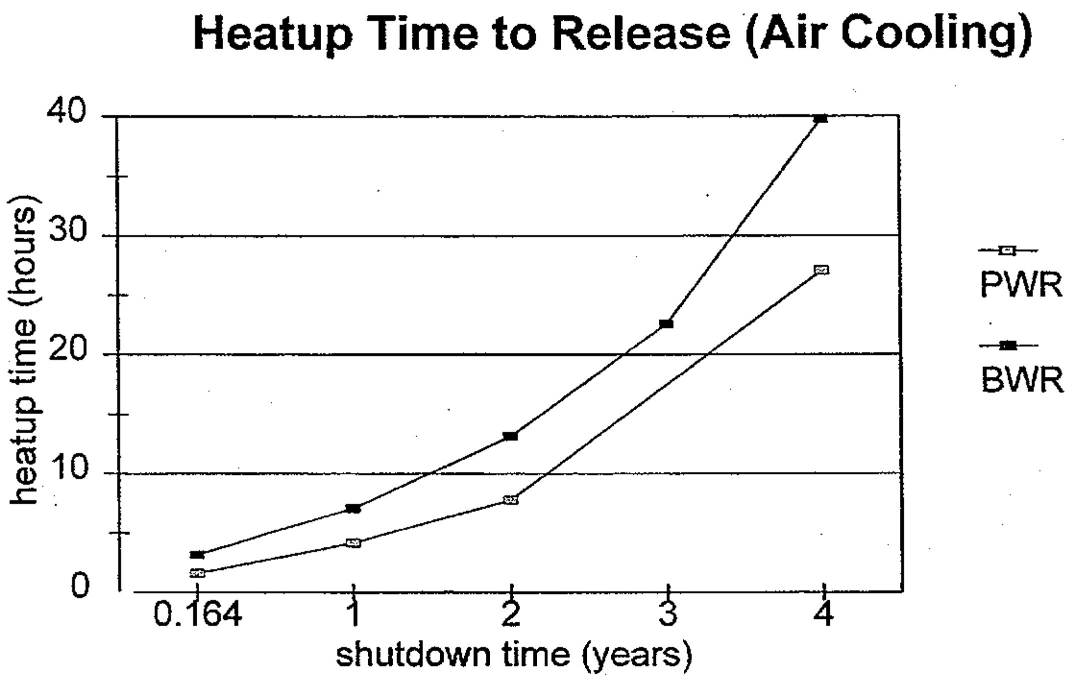

Figure 2.1 Heatup Time From 30 °C to 900 °C

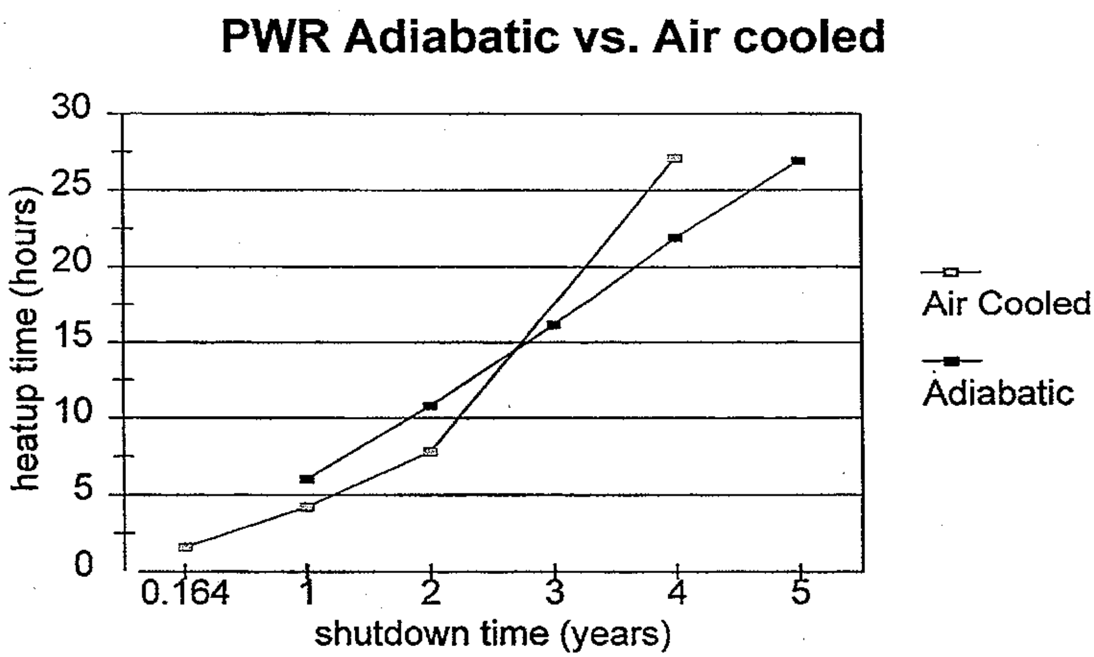

Figure 2.2 PWR Heatup Times for Air Cooling and Adiabatic Heatup

3.0 RISK ASSESSMENT OF SPENT FUEL POOLS AT DECOMMISSIONING PLANTS

The scenarios leading to significant offsite consequences at a decommissioning plant are different than at an operating plant. Once fuel is permanently removed from the reactor vessel, the primary public risk in a decommissioning facility is associated with the spent fuel pool (SFP). The spent fuel assemblies are retained in the SFP and submerged in water to cool the remaining decay heat and to shield the radioactive assemblies. The most severe accidents postulated for SFPs are associated with the loss of water from the pool.

Depending on the time since reactor shutdown, fuel burnup, and fuel rack configuration, there may be sufficient decay heat for the fuel clad to heat up, swell, and burst after a loss of pool water. The breach in the clad releases of radioactive gases present in the gap between the fuel and clad. This is called "a gap release" (see Appendix 1B). If the fuel continues to heat up, the zirconium clad will reach the point of rapid oxidation in air. This reaction of zirconium and air, or zirconium and steam is exothermic (i.e., produces heat). The energy released from the reaction, combined with the fuel's decay energy, can cause the reaction to become self-sustaining and ignite the zirconium. The increase in heat from the oxidation reaction can also raise the temperature in adjacent fuel assemblies and propagate the oxidation reaction. The zirconium fire would result in a significant release of the spent fuel fission products which would be dispersed from the reactor site in the thermal plume from the zirconium fire. Consequence assessments (Appendix 4) have shown that a zirconium fire could have significant latent health effects and resulted in a number of early fatalities. Gap releases from fuel from a reactor that has been shutdown more than a few months involve smaller quantities of radionuclides and, in the absence of a zirconium fire, would only be of concern onsite.

The staff conducted its risk evaluation to estimate the likelihood of accident scenarios that could result in loss of pool water and fuel heatup to the point of rapid oxidation. In addition to developing an assessment of the level of risk associated with SFPs at decommissioning plants, the staff's objective was to identify potential vulnerabilities and design and operational characteristics that would minimize these vulnerabilities. Finally, the staff assessed the effect of offsite emergency planning (i.e., evacuation) at selected sites using various risk metrics and the Commission's Safety Goals.

In support of the risk evaluation, the staff conducted a thermal-hydraulic assessment of the SFP for various scenarios involving loss of pool cooling and loss of inventory. These calculations provided information on heatup and boiloff rates for the pool and on heatup rates for the uncovered fuel assemblies and time to initiation of a zirconium fire (see Table 2.1 and Figures 2.1 and 2.2). The results of these calculations provided fundamental information on the timing of accident sequences, insights on the time available to recover from events, and time available to initiate offsite measures. This information was used in the risk assessment to support the human reliability analysis of the likelihood of refilling the SFP or cooling the fuel before a zirconium fire occurs.

For these calculations, the end state assumed for the accident sequences was the state at which the water level reached 3 feet from the top of the spent fuel. This simplification was used because of the lack of data and difficulty in modeling complex heat transfer mechanisms and chemical reactions in the fuel assemblies that are slowly being uncovered. As a result, the time available for fuel handler recovery from SFP events before initiation of a zirconium fire is underestimated. However, since recoverable events such as small loss of inventory or loss of power or pool cooling evolve very slowly, many days are generally available for recovery whether the end point of the analysis is uncovery of the top of the fuel or complete fuel uncovery. The extra time available (estimated to be in the tens of hours) as the water boils off would not impact the very high probabilities of fuel handler recovery from these events, given the industry decommissioning commitments (IDCs) and additional staff decommissioning assumptions (SDAs) discussed in Sections 3.2 through 4.[3] A summary of the thermal-hydraulic assessment is provided in Appendix 1A.

3.1 Basis and Findings of SFP Risk Assessment

To gather information on SFP design and operational characteristics for the preliminary risk assessment for the June 1999 draft study, the staff visited four decommissioning plants to ascertain what would be an appropriate model for decommissioning SFPs. The site visits confirmed that the as-operated SFP cooling systems were different from those in operation when the plants were in power operation. The operating plant pool cooling and makeup systems generally have been removed and replaced with portable, skid-mounted pumps and heat exchangers. In some cases there are redundant pumps. In most cases, physical separation, barrier protection, and emergency onsite power sources are no longer maintained. Modeling information for the PRA analysis was gathered from system walkdowns and discussions with the decommissioning plant staff. Since limited information was collected for the preliminary assessment on procedural and recovery activities and on the minimum configuration for a decommissioning plant, a number of assumptions and bounding conditions were in the June 1999 study. The preliminary results have been refined in this assessment, thanks to more detailed information from industry on SFP design and operating characteristics for a decommissioning plant and a number of IDCs that contribute to achieving low risk findings from SFP incidents. The revised results also reflect improvements in the PRA model since publication of the June 1999 and February 2000 studies.

The staff identified nine initiating event categories to investigate as part of the quantitative assessment on SFP risk:

1. Loss of offsite power from plant centered and grid-related events
2. Loss of offsite power from events initiated by severe weather
3. Internal fire
4. Loss of pool cooling
5. Loss of coolant inventory
6. Seismic event
7. Cask drop

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[3]The staff notes that the assumption that no recovery occurs once the water level reaches 3 feet above the fuel tends to obscure the distinction between two major types of accidents: slow boildown or draindown events and rapid draindown events. In both types of events, cooling would most likely be not by air but by water or steam. Also obscured is the effect of partial draindown events on event timing (addressed in Section 2).

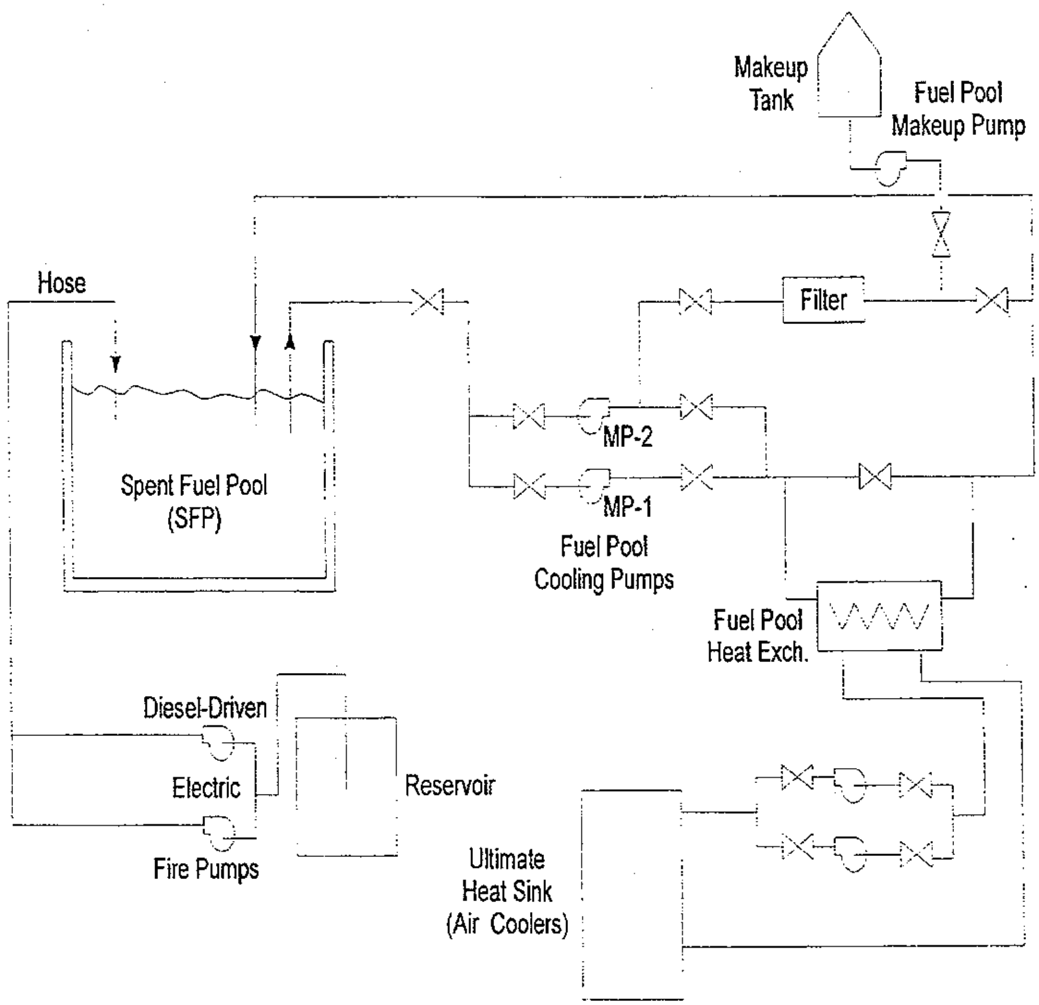
8. Aircraft impact
9. Tornado missile

In addition, a qualitative risk perspective was developed for inadvertent criticality in the SFP (see Section 3.6). The risk model, as developed by the staff and revised after a quality review by Idaho National Engineering & Environmental Laboratory (INEEL), is provided in Appendix 2A. Appendix 2A also includes the modeling details for the heavy load drop, aircraft impact, seismic, and tornado missile assessments. Input and comments from stakeholders were also utilized in updating the June 1999 and February 2000 risk models.

3.2 Characteristics of SFP Design and Operations for a Decommissioning Plant

Based on information gathered from the site visits and interactions with NEI and other stakeholders, the staff modeled the spent fuel pool cooling (SFPC) system (see Figure 3.1) as being located in the SFP area and consisting of motor-driven pumps, a heat exchanger, an ultimate heat sink, a makeup tank, a filtration system, and isolation valves. Coolant is drawn from the SFP by one of the two pumps, passed through the heat exchanger, and returned to the pool. One of the two pumps on the secondary side of the heat exchanger rejects the heat to the ultimate heat sink. A small amount of water is diverted to the filtration process and is returned into the discharge line. A manually operated makeup system (with a limited volumetric flow rate) supplements the small losses due to evaporation. During a prolonged loss of the SFPC system or a loss of inventory, inventory can be made up using the firewater system, if needed. Two firewater pumps, one motor-driven (electric) and one diesel-driven, provide firewater in the SFP area. There is a firewater hose station in the SFP area. The firewater pumps are in a separate structure.

Figure 3.1 Assumed Spent Fuel Pool Cooling System



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

- The SFP cooling design, including instrumentation, is at least as capable as that assumed in the risk assessment. Licensees have at least one motor-driven and one diesel-driven fire pump capable of delivering inventory to the SFP (SDA #1, Table 4.2-2).

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

Makeup pump:	20 – 30 gpm
Firewater pump:	100 – 200 gpm
Fire engine:	100 – 250 gpm (100 gpm, for hose: 1½-in., 250 gpm for 21/2-in. hose)

- For the larger loss-of-coolant-inventory accidents, water addition through the makeup pumps does not successfully mitigate the loss of the inventory event unless the location of inventory loss is isolated.

- The SFP fuel handlers perform walkdowns of the SFP area once per shift (8- to 12-hour shifts). A different crew member works the next shift. The SFP water is clear and the pool level is observable via a measuring stick in the pool to alert fuel handlers to level changes.

- Plants do not have drain paths in their SFPs that could lower the pool level (by draining, suction, or pumping) more that 15 feet below the normal pool operating level, and licensees must initiate recovery using offsite sources.

Based upon the results of the June 1999 preliminary risk analysis and the 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 because the remaining systems of the SFP are relatively simple, with manual rather than automatic initiation of backups or realignments. Therefore, in scenarios such as loss of cooling or inventory loss, the fuel handler's responses to diagnose the failures and bring any available resources (public or private) to bear is fundamental for ensuring that the fuel assemblies remain cooled and a zirconium fire is prevented.

As part of its technical evaluations, the staff assembled a small panel of experts[4] to identify the attributes necessary to achieving very high levels of human reliability for responding to potential accident scenarios in a decommissioning plant SFP. (These attributes and the human reliability analysis (HRA) methodology used are discussed in Section 3.2 of Appendix 2A.)

Upon considering the sensitivities identified in the staff's preliminary study and to reflect actual operating practices at decommissioning facilities, the nuclear industry, through NEI, made important commitments, which are reflected in the staff's updated risk assessment.

Industry Decommissioning Commitments (IDCs)

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

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

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

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[4]Gareth Parry, U.S. NRC; Harold Blackman, INEEL; and Dennis Bley, Buttonwood Consulting.

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

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

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

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

IDC #8 An onsite 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.

IDC #9 Procedures will be in place to control spent fuel pool operations that have the potential to rapidly decrease spent fuel pool inventory. These administrative controls may require additional operations or management review, management physical presence for designated operations or administrative limitations such as restrictions on heavy load movements.

IDC #10 Routine testing of the alternative fuel pool 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.

Additional important operational and design assumptions made by the staff in the risk estimates developed in this study are designated as SDAs and are discussed in later sections of this study.

3.3 Estimated Frequencies of Spent Fuel Uncovery and Assumptions That Influence the Results

Based upon the above design and operational features, IDCs, technical comments from stakeholders, and the input from the INEEL technical review, the staff's SFP risk model was updated. The updates have improved the estimated frequency calculations, but have not changed the need for the industry commitments or staff decommissioning assumptions. Absolute values of some sequences have decreased, but the overall insights from the risk assessment remain the same.

3.3.1 Internal and External Initiator Frequency of Spent Fuel Pool Uncovery

The results for the initiators that were assessed quantitatively are shown in Table 3.1. This table gives the fuel uncovery frequency for each accident initiator. The frequencies are point estimates because point estimates were used for the input parameters. For the most part, these input parameter values are the mean values of the probability distributions that would be used in a calculation to propagate parameter uncertainty. Because the systems are very simple and needs little support, the point estimates therefore reasonably correlate to the mean values that would be obtained from a full propagation of parameter uncertainty. Due to the large margin between the loss of cooling and inventory sequence frequencies and the pool performance guideline, this propagation was judged to be unnecessary (see Section 5.1 of Appendix 2A for further discussion of uncertainties).

Both the EPRI and LLNL hazard estimates at reactor sites were developed as best estimates and are considered valid by the NRC. Furthermore, because both sets of curves are based upon expert opinion and extrapolation in the range of interest, there is no technical basis for excluding consideration of either set of estimates. The mean frequency shown does not consider Western U.S. sites (e.g., Diablo Canyon, San Onofre, and WNP-2).

The results in Table 3.1 show the estimated frequency of a zirconium fire of fuel that has a decay time of 1 year. In characterizing the risk of seismically-induced SFP accidents for the population of sites, the staff has displayed results based on both the LLNL and EPRI hazard estimates, and has used an accident frequency corresponding to the mean value for the respective distributions, i.e., a frequency of 2×10^{-6} per year to reflect the use of LLNL hazard estimates and a frequency of 2×10^{-7} per year to reflect use of the EPRI hazard estimates. Use of the mean value facilitates comparisons with the Commission's Safety Goals and QHOs. Fire frequencies for all initiators range from about 6×10^{-7} per year to about 2×10^{-6} per year (depending on the seismic hazard estimates used), with the dominant contribution being from a severe seismic event. Plant-specific frequency estimates in some cases could be as much as an order of magnitude higher or lower because of the seismic hazard at the plant site. The mean value bounds about 70 percent of the sites for either the LLNL or the EPRI cases. A more detailed characterization of the seismic risk is given in Section 3.5.1 and Appendix 2B. The frequency of a zirconium fire is dominated by seismic events when the seismic hazard frequency is based on the LLNL estimate. Cask drop and boildown sequences become important contributors when seismic hazard frequency is based on the EPRI estimate. As a result, even though the seismic event frequency based on the EPRI estimate is an order of magnitude lower than the LLNL estimate, only a factor of four reduction in total frequency is realized with the use of the EPRI estimate since the nonseismic sequences become more important. In Section 3.4.7 the staff discusses the expected fuel uncovery frequencies for fuel that has decayed a few months, 2 years, 5 years, and 10 years.

In conjunction with the frequency of the uncovery of the spent fuel, it is important to know the time it takes the fuel to heat up once it has been uncovered fully or partially. Figures 2.1 and 2.2 in Section 2 show the time needed with and without air circulation to heat up the fuel from 30 °C to 900 °C (the temperature at which zirconium oxidation is postulated to become runaway oxidation and at which fission products are expected to be expelled from the fuel and cladding).

The staff realizes that the volumetric rate of air flow that a fuel bundle receives during a loss of cooling event significantly influences the heatup of the bundle. To achieve sufficient long-term air cooling of uncovered spent fuel, two conditions must be met: (1) an air flow path through the bundles must exist, and (2) sufficient SFP building ventilation flow must be provided. The presence of more than about 1 foot of water in the SFP, as in a seismically induced SFP failure or the late states of a boildown sequence, would effectively block the air flow path. Seismically induced collapse of the SFP building into the SFP could have a similar effect. Loss of building ventilation would tend to increase fuel heatup rates and maximum fuel temperatures, as described in Appendix 1A.

Based on engineering judgment, we have partitioned the frequency of each sequence into two parts: where the bundles in the spent fuel pool area receive two building volumes of air per hour (high air flow) and where the bundles receives little or no air flow (low air flow). Table 3.2 provides this partition.

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

INITIATING EVENT	Frequency of Fuel Uncovery (EPRI hazard)	Frequency of Fuel Uncovery (LLNL hazard)
Seismic event [5]	2×10^{-07}	2×10^{-06}
Cask drop [6]	2.0×10^{-07}	same
Loss of offsite power[7] initiated by severe weather	1.1×10^{-07}	same
Loss of offsite power from plant centered and grid-related events	2.9×10^{-08}	same
Internal fire	2.3×10^{-08}	same
Loss of pool cooling	1.4×10^{-08}	same
Loss of coolant inventory	3.0×10^{-09}	same
Aircraft impact	2.9×10^{-09}	same
Tornado Missile	$<1.0 \times 10^{-09}$	same
Total[8]	5.8×10^{-07}	2.4×10^{-06}

[5]This value is the mean of the failure probabilities for Central and Eastern SFPs that satisfy the seismic checklist and includes seismically induced catastrophic failure of the pool (which dominates the results) and a small contribution from seismically induced failure of pool support systems.

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

[7]The estimate is based upon the time available for human response when the fuel has decayed 1 year. After only a few months of decay, these estimates are not expected to increase significantly. Furthermore, for longer periods of decay, no significant change in the estimated frequency is expected because the fuel handler success rates are already so high after 1 year of decay.

[8]Consistent with PRA limitations and practice, contributions to risk from safeguards events are not included in these frequency estimates. EP might also provide dose savings in such events.

Table 3.2 Spent Fuel Pool Cooling Risk Analysis — Frequency Partition for Air Flow

SEQUENCES	% HIGH AIR FLOW	% LOW AIR FLOW (ADIABATIC)
Seismic	30%	70%
Heavy load drop	50%	50%
Loss of offsite power, severe weather	90%	10%

In Table 3.2 for seismic sequences, we have assumed that 30 percent of the time the building will turn over two building volumes of air per hour (high air flow) and 70 percent of the time the individual fuel bundle of concern will receive little or no air cooling. These percentages are based on discussions with staff structural engineers, who believe that, at accelerations in excess of 1.2 g spectral acceleration (which is greater than three times the safe shutdown earthquake (SSE) for many reactor sites east of the Rocky Mountains), there is a high likelihood of building damage that blocks air flow. For heavy load drop sequences, the staff assumed a 50 percent partition for the high air flow case. This is based on considering both damage to fuel bundles due to a heavy load drop that renders bundles uncoolable and the alternative possibility that the drop damages the building structure in a way that blocks some spent fuel bundles. For loss of offsite power events caused by severe weather, the staff assumed a 90 percent partition for the high airflow case. This is based on a staff assumption that openings in the SFP building (e.g., doors and roof hatches) are large enough that, if forced circulation is lost, natural circulation cooling will provide at least two building volume of air per hour to the SFP. This assumption may need to be confirmed on a plant-specific basis. The staff did not partition the rest of the sequences in Table 3.2, since their absolute value and contribution to the overall zirconium fire frequency are so low.

The frequency partitioning shows that a large portion of the seismic and heavy load drop sequences have low air flow. This partitioning did not consider the possibility that the air flow path is blocked by residual water in the SFP. When the potential for flow blockage by residual water is considered, an even greater portion of the events would result in no (or low) air flow. Fuel heatup calculations in Section 2 show that for the first several years after shutdown, the fuel heatup time (e.g., time to reach 900 °C) for the adiabatic and air-cooled cases is comparable. Thus, the effects of partitioning are negligible for this period. Because SFP and SFP building fragilities and failure modes are plant-specific, and the heatup time for the adiabatic and air-cooled cases differ only slightly, the staff did not consider the partitioning in estimating the frequency of SFP fires. Whether or not a spent fuel bundle receives high air flow or low air flow fuel uncovery does not change our insights into the risk associated with operation of SFPs.

3.3.2 Important Assumptions

As discussed in more detail in Appendix 2A, the results of the risk analysis depend on assumptions about the design and operational characteristics of the SFP facility. The following inputs can significantly influence the results:

- The modeled system configuration is described in Section 3.2. The assumed availability of a diesel-driven fire pump is an important factor in the conclusion that fuel uncovery frequency is low for the loss of offsite power initiating events and the internal fire initiating event. The assumption of the availability of a redundant fuel pool cooling pump is not as important since the modeling of the recovery of the failed system includes repair of the failed pump as well as the startup of the redundant pump. Finally, multiple sources of makeup water are assumed for the fire pumps. This lessens the possible dependencies between initiating events (e.g., severe weather, high winds, or earthquakes) and the availability of makeup water supply (e.g., the fragility of the fire water supply tank).

- Plants have no drain paths in their SFPs that could lower the pool level (by draining, suction, or pumping) more that 15 feet below the normal pool operating level, and licensees must initiate recovery using offsite sources.

- Openings in the SFP building (e.g., doors and roof hatches) are large enough that, if forced circulation is lost, natural circulation cooling will provide at least two building volumes of air per hour to the SFP. Procedures exist to implement natural circulation.

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

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

  - IDCs #2, 3, 4, and 8 are credited for the high probability of recovery from loss of cooling scenarios (including events initiated by loss of power or fire) and loss of inventory scenarios. To take full credit for these commitments, additional assumptions have been made about how these commitments will be implemented. Procedures and training are assume to give explicit guidance on the capability of the fuel pool makeup system and on when it becomes essential to supplement with alternate higher volume sources. Procedures and training are assumed to give sufficiently clear guidance on early preparation for using the alternate makeup sources. Walkdowns are assumed once per shift and the fuel handlers are assumed to document their observations in a log. The last assumption compensates for potential failures of the instrumentation monitoring the status of the pool.

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

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

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

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

The various initiating event categories are discussed below. The staff's qualitative risk insights on the potential for SFP criticality are discussed in Section 3.6.

3.4 Internal Event Scenarios Leading to Fuel Uncovery

This section describes the events associated with internal event initiators. More details are given in Appendix 2A.

3.4.1 Loss of Cooling

The loss of cooling initiating event may be caused by the failure of pumps or valves, by piping failures, by an ineffective heat sink (e.g., loss of heat exchangers), or by a local loss of power (e.g., electrical connections). Although it may not be directly applicable because of design differences in decommissioning plants, operational data from NUREG-1275, Volume 12 (Ref. 3), shows that the frequency of loss of SFP cooling events in which temperature increases more than 20 °F is on the order of two to three events per 1000 reactor years. The data also shows that was the loss of cooling lasted less than 1 hour. Only three events exceeded 24 hours: the longest was 32 hours. In four events the temperature increase exceeded 20 °F, the largest increase being 50 °F.

The calculated fuel uncovery frequency for this initiating event is 1.4×10^{-8} per year. Indications of a loss of pool cooling available to fuel handlers include control room alarms and indicators, local temperature measurements, increasing area temperature and humidity, and low pool water level from boiloff. For a fuel uncovery, the plant fuel handlers must fail to recover the cooling system (either fail to notice the loss of cooling indications or fail to repair or restore the cooling system). In addition, the fuel handlers must fail to provide makeup cooling using other onsite sources (e.g., fire pumps) or offsite sources (e.g., a fire brigade). A long time is available for these recovery actions. In the case of 1-year-old fuel (i.e., fuel that was in the reactor when it was shutdown 1 year ago), approximately 195 hours is available for a PWR and 253 hours for a BWR before the water level drops to within 3 feet of the spent fuel. If the fuel most recently offloaded is only 2 months out of the reactor, the time available is still long (100–150 hours), and the likelihood of fuel handler success is still very high. These heatup and boiloff times are about double those reported by the staff before a correction in the staff's heat load assumptions.

Because the uncovery frequency is already very low (on the order of 1 in 1 million per year) both absolutely and relative to other initiators, and because the quantification of human reliability analysis values for such extended periods of recovery is beyond the state-of-the-art, the staff did not attempt to recalculate the expected uncovery frequency. For 2-year-old, 5-year-old, and 10-year-old fuel, much longer periods are available than at 1 year (see Table 2.1).

A careful and thorough adherence to IDCs #2, #5, #8, and #10 is crucial to establishing and maintaining the low frequency. In addition, however, the assumption that walkdowns are performed on a regular basis (once per shift) is important to compensate for potential failures of the instrumentation monitoring the status of the pool. The analysis has also assumed that the procedures and/or training give explicit guidance on the capability of the fuel pool makeup system and on when it becomes essential to supplement with alternative higher volume sources. The analysis also assumes that the procedures and training give sufficiently clear guidance on early preparation for using the alternative makeup sources.

There have been two recent events involving a loss of cooling at SFPs. The first, at Browns Ferry Unit 3 occurring in December 1998, involved a temperature increase of approximately 25 °F over a 2-day period. This event, caused by the short cycling of cooling water through a stuck-open check valve, was not detected by the control room indicators because of a design flaw in the indicators. In the second event, at the Duane Arnold Unit 1 in January 2000, the SFP temperature increased by 40 to 50 °F. The incident, which was undetected for approximately 2½ days, was caused by operator failure to restore the SFP cooling system heat sink after maintenance activities. The plant had no alarm for high fuel pool temperature, although there are temperature indicators in the control room. Since the conditional probability of fuel uncovery is low given a loss of cooling initiating event, the addition of these two recent events to the database will not affect the conclusion that the risk from these events is low. However, the recent events illustrate the importance of industry commitments, particularly IDC #5 which requires temperature instrumentation and alarms in the control room. In addition, the staff assumptions that walkdowns are performed on a regular basis (once per shift), with the fuel handler documenting the observations in a log, and the assumption that control room instrumentation that monitors SFP temperature and water level directly measures the parameters involved are important for keeping the risk low, since the walkdowns compensate for potential failures of the control room instrumentation and direct measurement precludes failures such as occurred at Browns Ferry.

Even with the above referenced industry commitments, the additional need for walkdowns to be performed at least once per shift and the specific need for direct indication of level and temperature had to be assumed in order to arrive at the low accident frequency calculated for this scenario. These additional assumptions are identified by the staff as staff decommissioning assumptions (SDAs) #2 and #3. SDA #2 assumes the existence of explicit procedures and fuel handler training to provide guidance on the capability and availability of inventory makeup sources and the time available to utilize these sources.

SDA #2 Walkdowns of SFP systems are performed at least once per shift by the fuel handlers. Procedures are in place to give the fuel handlers guidance on the capability and availability of onsite and offsite inventory makeup sources and on the time available to utilize these sources for various loss of cooling or inventory events.

SDA #3 Control room instrumentation that monitors SFP temperature and water level directly measures the parameters involved. Level instrumentation provides alarms for calling in offsite resources and for declaring a general emergency.

3.4.2 Loss of Coolant Inventory

This initiator includes loss of coolant inventory resulting from configuration control errors, siphoning, piping failures, and gate and seal failures. Operational data in NUREG-1275, Volume 12, shows that the frequency of loss of inventory events in which a level decrease of more than 1 foot occurred is less than one event per 100 reactor years. Most of these events are as a result of fuel handler errors and are recoverable. Many of the events are not applicable in a decommissioning facility.

NUREG-1275 shows that, except for one event that lasted 72 hours, no events lasted more than 24 hours. Eight events resulted in a level decrease of between 1 and 5 feet, and another two events resulted in an inventory loss of between 5 and 10 feet.

Using the information from NUREG-1275, it can be estimated that 6 percent of the loss of inventory events will be large enough and/or long enough to require that isolating the loss if the only system available for makeup is the SFP makeup system. For the other 94 percent of the cases, operation of the makeup pump is sufficient to prevent fuel uncovery.

The calculated fuel uncovery frequency for loss of inventory events is 3.0×10^{-9} per year. The uncovery frequency is low primarily due to the assumption that loss of inventory can drain the pool only so far. Once that level is reached, additional inventory loss must come from pool heatup and boiloff. Fuel uncovery occurs if plant fuel handlers fail to initiate inventory makeup either by use of onsite sources such as the fire pumps or offsite sources such as the local fire department. In the case of a large leak, isolation of the leak would also be necessary if the makeup pumps are used. The time available for fuel handler action is considerable, and even in the case of a large leak, it is estimated that 40 hours will be available. Fuel handlers are alerted to a loss of inventory condition by control room alarms and indicators, by the visibly dropping water level in the pool, by the accumulation of water in unexpected locations, and by local alarms (radiation alarms, building sump high level alarms, etc.).

As with the loss of pool cooling, the frequency of fuel uncovery is calculated to be very low. Again, a careful and thorough adherence to IDCs #2, #5, #8, and #10 is crucial to establishing the low frequency. In addition, the assumptions that walkdowns (see SDA #2 above) are performed on a regular basis (once per shift) and that instrumentation directly measures temperature and level are important to compensate for potential failures of the instrumentation monitoring the status of the pool. The assumption that the procedures and/or training give explicit guidance on the capability of the fuel pool makeup system lowers the expected probability of fuel handler human errors, and the assumption that fuel handlers will supplement SFP makeup at appropriate times from alternative higher volume sources lowers the estimated frequency of failure of the fuel handler to mitigate the loss of coolant inventory. IDCs #6, #7, and #9 are also credited with lowering the initiating event frequency.

Even with these industry commitments, the staff had to assume the drop in pool inventory due to loss of inventory events is limited in order to arrive at the low accident frequency calculated for this scenario. This additional assumption is identified by the staff as SDA #4.

SDA #4 The licensee has determined that the SFP has no drain paths that could lower the pool level (by draining, suction, or pumping) more than 15 feet below the normal pool operating level and that the licensee initiates recovery using offsite sources.

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

A loss of offsite power from plant-centered events typically involves hardware failures, design deficiencies, human errors (in maintenance and switching), localized weather-induced faults (e.g., lightning), or combinations. Grid-related offsite power events are caused by problems in the offsite power grid. With the loss of offsite power (onsite power is lost too, since the staff assumes no diesel generator is available to pick up the necessary electrical loads), there is no effective way of removing heat from the SFP. If power is not restored in time, the pool will heatup and boiloff inventory until the fuel is uncovered. The diesel-driven fire pump is available to provide inventory makeup. If the diesel-driven pump fails, and if offsite power is not recovered promply, recovery using offsite fire engines is a possibility. Recovery times are the same as for loss of cooling (discussed in Section 3.4.1).

Even after recovering offsite power, the fuel handlers have to restart the fuel pool cooling pumps. Failure to do this or failure of the equipment to restart will necessitate other fuel handler recovery actions. Again, considerable time is available.

The calculated fuel uncovery frequency for this sequence of events is 2.9×10^{-8} per year. This frequency is very low and, as with loss of pool cooling and loss of inventory, is based on adherence to IDCs #2, #5, #8, and #10. In addition, regular plant walkdowns, clear and explicit procedures, fuel handler training (SDA #2), and the direct measurement of level and temperature in the SFP (SDA #3) are assumed as documented.

3.4.4 Loss of Offsite Power from Severe Weather Events

This event represents the loss of SFP cooling because of a loss of offsite power from severe weather-related events (hurricanes, snow and wind, ice, wind and salt, wind, and one tornado event). Because of the potential for severe localized damage, tornadoes are analyzed separately in Appendix 2E. The analysis is summarized in Section 3.5.3 of this study.

Until offsite power is recovered, the electrical pumps are unavailable and the diesel-driven fire pump is available only for makeup. Recovery of offsite power after severe weather events is assumed to be less probable than after grid-related and plant-centered events. In addition, it is more difficult for offsite help to reach the site.

The calculated fuel uncovery frequency for this event is 1.1×10^{-7} per year. As in the previous cases, this estimate was based on IDCs #2, #5, #8, #10 and on assumptions documented in SDA #2 and SDA #3. In addition, IDC #3, the commitment to have procedures in place for communications between onsite and offsite organizations during severe weather, is also important in the analysis for increasing the likelihood that offsite organization can respond effectively.

3.4.5 Internal Fire

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

The calculated fuel uncovery frequency for this event is 2.3×10^{-8} per year. As in the previous cases, this estimate was based on IDCs #2, #5, #8, and #10 and on SDA #2 and SDA #3. In addition, IDC #3, the commitment to have procedures in place for communications between onsite and offsite organizations during severe weather, is also important in the analysis for increasing the likelihood that offsite organizations can respond effectively to a fire event because the availability of offsite resources increases the likelihood of recovery.

3.4.6 Heavy Load Drops

The staff investigated the frequency of a heavy load drop in or near the SFP and the potential damage to the pool from such a drop. The previous assessment done for resolution of Generic Issue 82 (in NUREG/CR-4982 (Ref. 4)) only considered the possibility of a heavy load drop on the pool wall. The assessment conducted for this study identifies other failure modes, such as the collapse of the pool floor, as also credible for some sites. Details of the heavy load assessment are given in Appendix 2C. The analysis exclusively considered drops severe enough to catastrophically damage the SFP so that pool inventory would be lost rapidly and it would be impossible to refill the pool using onsite or offsite resources. There is no possibility of mitigating the damage, only preventing it. The staff has not attempted to partition the initiator into events where there is full rapid draindown and events where there is rapid, but partial draindown. The staff assumes a catastrophic heavy load drop (creating a large leakage path in the pool) would lead directly to a zirconium fire. The time from the load drop until a fire varies depending on fuel age, burn up, and configuration. The dose rates in the pool area before any zirconium fire are tens of thousands of rem per hour, making any recovery actions (such as temporary large inventory addition) very difficult.

Based on discussions with staff structural engineers, it is assumed that only spent fuel casks are heavy enough to catastrophically damage the pool if dropped. The staff assumes a very low likelihood that other heavy loads will be moved over the SFP and that if one of these lighter loads over the SFP is dropped, it is unlikely to cause catastrophic damage to the pool.

For a non-single-failure-proof load handling system, the drop frequency of a heavy load drop is 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 plants with a single-failure-proof load handling system or plants conforming to the NUREG-0612 guidelines, the drop frequency is estimated to have a mean value of 9.6×10^{-6} per year, again for 100 heavy load lifts per year but using data from U.S. Navy crane experience. Once the load is dropped, the analysis must then consider whether the drop significantly damages the SFP.

When estimating the failure frequency of the pool floor and pool wall, the staff assumes that heavy loads travel near or over the pool approximately 13 percent of the total path lift length (the path lift length is the distance from where the load is lifted to where it is placed on the pool floor). The staff also assumes that the critical path (the fraction of total path the load is lifted high enough above the pool to damage the structure in a drop) is approximately 16 percent. The staff estimates 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 relying on electrical interlocks, fuel handling system reliability, and safe load path procedures. The staff estimates the catastrophic failure rate from heavy load drops to have a mean value of 2×10^{-7} per year for a single-failure-proof system. The staff assumes that licensees that chose the non-single-failure-proof system option in NUREG-0612 performed appropriate analyses and took mitigative actions to reduce the expected frequency of catastrophic damage to the same range as for facilities with a single-failure-proof system.

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

SDA #5 Load Drop consequence analyses will be performed for facilities with non-single-failure-proof systems. The analyses and any mitigative actions necessary to preclude catastrophic damage to the SFP that would lead to a rapid pool draining should be performed with sufficient rigor to demonstrate that there is high confidence in the facility's ability to withstand a heavy load drop.

Although this study focuses on the risk associated with wet storage of spent fuel during decommissioning, the staff has been alert to any implications for the storage of spent fuel during power operation. With regard to power operation, the resolution of Generic Issue (GI) 82, "Beyond Design Basis Accidents in Spent Fuel Pools," and other studies of operating reactor SFPs concluded that existing requirements for operating reactor SFPs are sufficient. In developing the risk assessment for decommissioning plants, the staff evaluated the additional issue of a drop of a cask on the SFP floor rather than just on a SFP wall. As noted above, because the industry has committed to Phase II of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36," this is not a concern for decommissioning reactors.

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

3.4.7 Spent Fuel Pool Uncovery Frequency at Times Other Than 1 Year After Shutdown

The staff has considered how changes in recovery time available to fuel handlers at 2 months, 2 years, 5 years, and 10 years after shutdown (see Table 2.1) change the insights or bottom-line numerical results from the risk assessment . The different recovery times primarily affect the human reliability analysis (HRA) results and insights. Even at 2 months after shutdown, the HRA failure estimates are small and are dominated by institutional factors (e.g., training, quality of procedures, staffing). It is therefore expected that the total fuel uncovery frequency at 2 months will continue to be dominated by the seismic contribution. At periods beyond 1 year, the increased recovery time (from a very long time to an even longer time) lowers the uncertainty that these HRA estimates really are very small, but the increased time has not translated into significant changes in the bottom-line numerical estimates because quantification of the effect of such extensions on organizational problems is beyond the state-of-the-art.

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

In the following sections, the staff explains how each of the external event initiators was modeled, discusses the frequency of fuel uncovery associated with the initiator, and describes the most important insights regarding risk reduction strategies for each initiator.

3.5.1 Seismic Events

The staff performed a simplified seismic risk analysis in its June 1999 preliminary draft risk assessment to gain initial insights on seismic contribution to SFP risk. The analysis indicated that seismic events could not be dismissed on the basis of a simplified bounding approach. The additional efforts by the staff to evaluate the seismic risk to SFPs are addressed here and in Appendix 2B.

SFP structures at nuclear power plants should be seismically robust. They are constructed of thick, reinforced concrete walls and slabs lined with stainless steel liners 1/8 to 1/4 inch thick.[9] Pool walls are about 5 feet thick and the pool floor slabs are around 4 feet thick. The overall pool dimensions are typically about 50 feet long by 40 feet wide and 55 to 60 feet high. In boiling-water reactor (BWR) plants, the pool structures are located in the reactor building at an elevation several stories above the ground. In pressurized-water reactor (PWR) plants, the SFP structures are outside the containment structure and supported on the ground or partially

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[9]Except at Dresden Unit 1 and Indian Point Unit 1, have no liner plates. The plants were permanently shut down more than 20 years ago and no safety significant degradation of the concrete pool structure has been reported.

embedded in the ground. The location and supporting arrangement of the pool structures affect their capacity to withstand seismic ground motion beyond their design basis. The dimensions of the pool structure are generally derived from radiation shielding considerations rather than seismic demand needs. Spent fuel structures at nuclear power plants are able to withstand loads substantially beyond those for which they were designed.

To evaluate the risk from a seismic event at an SFP, one needs to know both the likelihood of seismic ground motion at various acceleration levels (i.e., seismic hazard) and the conditional probability that a structure, system, or component (SSC) will fail at a given acceleration level (i.e., the fragility of the SSC). These can be convolved mathematically to arrive at the likelihood that the SFP will fail from a seismic event. In evaluating the effect of seismic events on SFPs, it became apparent that although information was available on seismic hazard for nuclear power plant sites, the staff did not have fragility analyses of the pools, nor generally did licensees. The staff recognized that many of the SFPs and the buildings housing them were designed by different architect-engineers. Additionally, the pools were built to different standards as the rules changed over the years.

To compensate for the lack of knowledge of the capacity of the SFPs, the staff proposed the use of a seismic checklist during stakeholder interactions, and in a letter dated August 18, 1999, NEI proposed a checklist that could be used to show an SFP would retain its structural integrity at a peak spectral acceleration of about 1.2 g. This value was chosen, in part, because existing databases that could be used in conjunction with the checklist only go up to 1.2 g peak spectral acceleration. The checklist was reviewed and enhanced by the staff (see Appendix 2B). The checklist includes elements to assure there are no weaknesses in the design or construction or any service-induced degradation of the pools that would make them vulnerable to failure during earthquake ground motions that exceed their design-basis ground motion but are less than the 1.2 g peak spectral acceleration. The staff used a simplified, but slightly conservative method to estimate the annual probability of a zirconium fire due to seismic events and site-specific seismic hazard estimates (see Appendix 2B, Attachment 2). These calculations resulted in a range of frequencies from less than 1×10^{-8} per year to over 1×10^{-5} per year, depending on the site and the seismic estimates used.

Figures 3.2 and 3.3 show the estimated annual probabilities of a zirconium fire from a seismic event in ascending order. Figure 3.2 shows the results of convolving the site-specific LLNL seismic hazard estimates (from NUREG-1488, "Revised Livermore Seismic Hazard Estimates for 69 Nuclear Power Plant Sites East of the Rocky Mountains," P. Sobel, October 1993) with the generic SFP fragility analysis, and Figure 3.3 shows the results of convolving the EPRI site-specific seismic hazard estimates (Ref. 10) in a similar manner.[10] Note that the order of the sites differs somewhat In the EPRI and LLNL estimates. These figures show that for the zirconium fire frequencies using the LLNL estimates, the annual probabilities for most site clusters just above 1×10^{-6} per year. The mean failure probability for the sites analyzed by LLNL is about 2×10^{-6} per year. This value bounds 70 percent of the sites using the LLNL curves. For the EPRI

[10] At higher accelerations, especially for plant sites east of the Rocky Mountains, there is great modeling uncertainty about the ground motions, return periods, and the possibility of cutoff. There is virtually no data at these acceleration levels.

curve, the mean value of the pool failure frequency is about 2×10^{-7} per year. In considering these two different sets of hazard estimates, the NRC has found that both sets are reasonable and equally valid.

By passing the checklist, the SFP will be assured a high confidence with low probability of failure (HCLPF)[11] of at least 1.2 g peak spectral acceleration. The performance of the seismic checklist is identified by the staff as SDA #6.

SDA #6 Each decommissioning plant will successfully complete the seismic checklist provided in Appendix 2B to this study. If the checklist cannot be successfully completed, the decommissioning plant will perform a plant specific seismic risk assessment of the SFP and demonstrate that SFP seismically induced structural failure and rapid loss of inventory is less than the generic bounding estimates provided in this study ($<1 \times 10^{-5}$ per year including non-seismic events).

For many sites (particularly PWRs because their SFPs are closer to ground level or embedded and the motion is therefore less amplified), the plant-specific risk may be considerably lower. There are only two plant-specific SFP fragility analyses of which the staff is aware, and these were used in the analyses performed to help confirm the generic seismic capacities assumed for SFPs.

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

Frequency of Spent Fuel Pool Seismically Induced Failure Based on LLNL Estimates and HCLPF of 1.2 Peak Spectral Acceleration

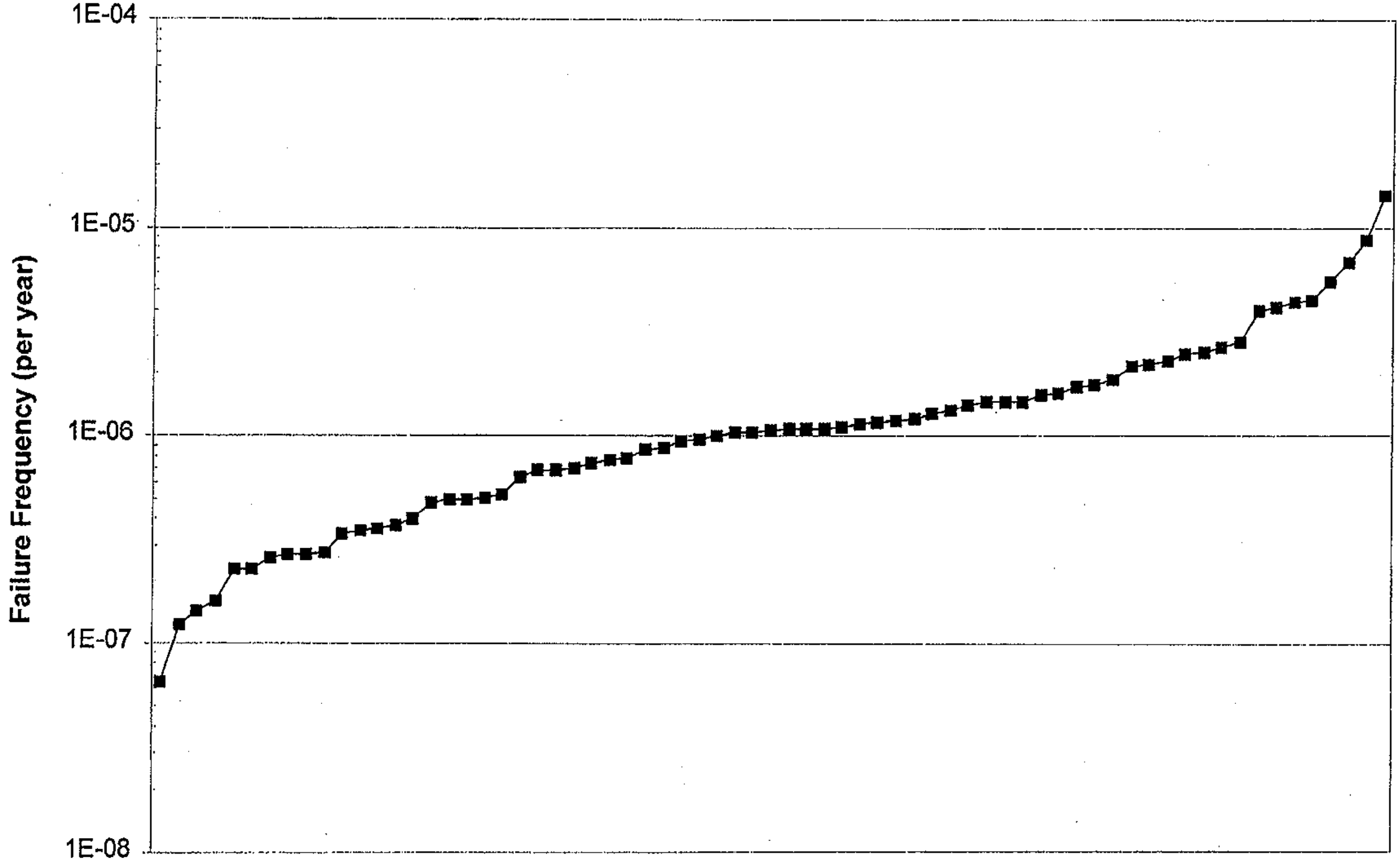


Figure 3.2

Frequency of Spent Fuel Pool Seismically Induced Failure Based on EPRI Estimates and HCLPF of 1.2 Peak Spectral Acceleration

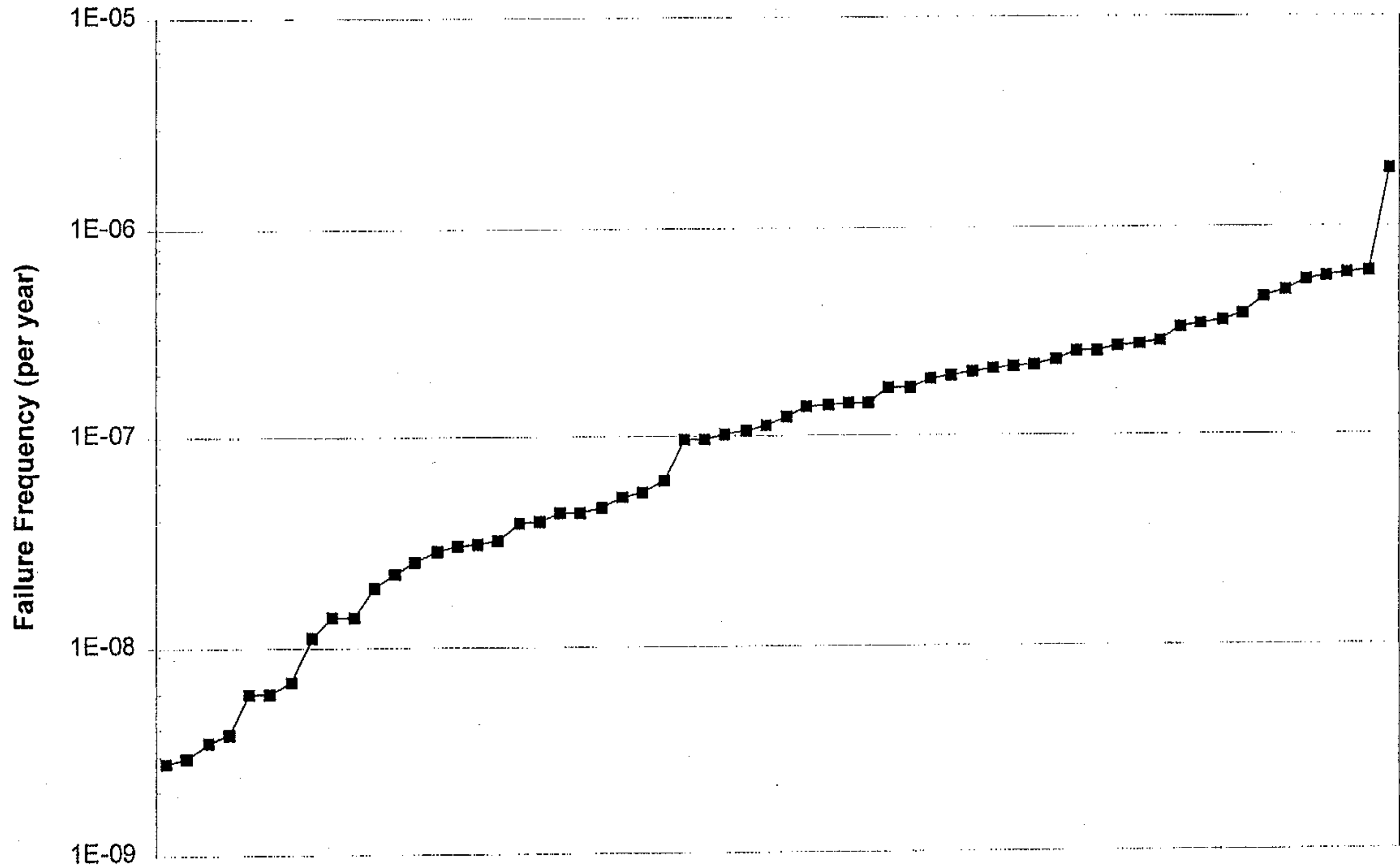


Figure 3.3

3.5.2 Aircraft Crashes

The staff evaluated the likelihood that an aircraft crashing into a nuclear power plant site would seriously damage the spent fuel pool or its support systems (details are in Appendix 2D). The generic data provided in DOE-STD-3014-96 (Ref. 6) was 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 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 to evaluating the likelihood of an aircraft crash into a structure. The first is 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 in 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. The DOE model also takes into account the plane's glide path by introducing the height of the structure into the equation, which effectively increases the area of the target.

In estimating the frequency of catastrophic PWR spent fuel pool damage from an aircraft crash (i.e., the pool is so damaged that it rapidly drains and cannot be refilled from either onsite or offsite resources), the staff uses the point target area model and assumes a direct hit on a 100 x 50 foot spent fuel pool. Based on studies in NUREG/CR-5042, "Evaluation of External Hazards to Nuclear Power Plants in the United States," it is estimated that 1 of 2 aircrafts are large enough to penetrate a 5-foot-thick reinforced concrete wall. The conditional probability that a large aircraft crash will penetrate a 5-foot-thick reinforced concrete wall is taken as 0.45 (interpolated from NUREG/CR-5042). It is further estimated that 1 of 2 crashes damage the spent fuel pool enough to uncover the stored fuel (for example, 50 percent of the time the location of the damage is above the height of the stored fuel). The estimated range of catastrophic damage to the spent fuel pool resulting in uncovery of the spent fuel is 1.3×10^{-11} to 6.0×10^{-8} per year. The mean value is estimated to be 4.1×10^{-9} per year. The frequency of catastrophic BWR spent fuel pool damage resulting from a direct hit by a large aircraft is estimated to be the same as for a 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 1 of 4 sides of a BWR secondary containment may be less likely to penetrate because other structures are in the way of the aircraft. Mark-III secondary containments may reduce the likelihood of penetration somewhat, since the spent fuel pool may be protected on one side by additional structures. If instead of a direct hit, the aircraft skids into the pool or a wing clips the pool, catastrophic damage may not occur. The staff estimates that skidding aircraft are negligible contributors to the frequency of fuel uncovery resulting from catastrophic damage to the pool because skidding decreases the impact velocity. The estimated frequencies of aircraft-induced catastrophic spent fuel pool failure are bounded by other initiators.

The staff estimated the frequency of significant damage to spent fuel pool support systems (e.g., power supply, heat exchanger, makeup water supply) for three different situations. The first case is based on the DOE model including the glide path and the wing and skid area and assumes a structure 400 x 200 x 30 feet (i.e., the large building housing the support systems) with a conditional probability of 0.01 that one of these systems is hit (the critical system occupies a 30 x 30 x 30 foot cube within the large building). This model accounts for damage from the aircraft (including, for example, being clipped by a wing). 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 this case assumes 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 10x10 structure, 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 used and the target structure size, the mean value for an aircraft damaging a support system is 7×10^{-7} per year or less. 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 on site or off site. As an initiator of 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.

3.5.3 Tornadoes and High Winds

The staff performed a risk evaluation of tornado threats to spent fuel pools (details are in Appendix 2E). The staff assumed that very severe tornadoes (F4 to F5 tornadoes on the Fujita scale) would be required to cause catastrophic damage to a PWR or BWR spent fuel pool. These tornados have wind speeds that result in damage characterized as "devastating" or "incredible." The staff then looked at the frequency of such tornadoes and the conditional probability that if such a tornado hit the site, it would seriously damage the spent fuel pool. To do this the staff examined the frequency and intensity of tornadoes the continental United States, using the methods described in NUREG/CR-2944 (Ref. 7). The frequency of an F4 to F5 tornado is estimated to be 5.6×10^{-7} per year for the Central United States, with a U.S. average value of 2.2×10^{-7} per year.

The staff then considered what level of damage an F4 or F5 tornado could do to a spent fuel pool. 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).

It was assumed that an F2 to F5 tornado would be required to significantly damage SFP support systems (e.g., power supply, cooling pumps, heat exchanger, or makeup water supply). These tornados have wind speeds that result in damage characterized as "significant," "severe," or "worse." The frequency of an F2 to F5 tornado is estimated to be 1.5×10^{-5} per year for the Central United States, with a U.S. average value of 6.1×10^{-6} per year. This is not the estimated frequency of fuel uncovery or a zirconium fire caused by damage to the support systems, since the frequency estimate does not include recovery, either on site or off site. As an initiator of failure of a support system leading to fuel uncovery and a zirconium fire, a tornado is bounded by other more probable events. Recovery of the support system(s) will reduce the likelihood of spent fuel uncovery.

Missiles generated by high winds (for example, straight winds or hurricanes) are not as powerful as those generated by tornados. Therefore high winds are estimated to have a negligible impact on the frequency of catastrophic failure of the SFP resulting in fuel uncovery. Long-term loss of offsite power due to straight winds is evaluated in Section 3.4.4.

The staff estimated the frequency of significant damage to SFP support systems from straight-line winds to be very low. Damage was assumed to be caused by building collapse. Based on the construction requirements for secondary containments, the staff believes that the buildings containing BWR spent fuel pools are sufficiently robust that straight line winds will not challenge the integrity of the building. The staff assumes buildings covering PWR spent fuel pools have a concrete foundation that extends part way up the side of the building. The exterior of the rest of the building has a steel frame covered by corrugated steel siding. The PWR spent fuel buildings are assumed to be constructed to American National Standards Institute (ANSI) or American Society of Civil Engineers (ASCE) standards. Based on these assumptions, the staff believes that straight-line winds will cause buildings housing PWR spent fuel pools to fail at a frequency of 1×10^{-3} per year or less. This failure rate for support systems is subsumed in the initiating event frequency for loss of offsite power from severe weather events. The event tree for this initiator takes into account the time available for recovery of spent fuel pool cooling (approximately 195 hours for 1-year old PWR fuel and 253 hours for 1-year-old BWR fuel).

3.6 Criticality in Spent Fuel Pool

In Appendix 3, the staff performed an evaluation of the potential scenarios that could lead to criticality and identified those that are credible.

In this section the staff gives its qualitative assessment of risk due to criticality in the SFP, concluding that, with the additional assumptions, the potential risk from SFP criticality is small.

Appendix 3 references the NRC staff report "Assessment of the Potential for Criticality in Decommissioned Spent Fuel Pools." The assessment identified two credible scenarios listed below:

(1) A compression or buckling of the stored assemblies from the impact of a dropped heavy load (such as a fuel cask) could result in a more optimum geometry (closer spacing) and thus create the potential for criticality. Compression is not a problem for high-density PWR or BWR racks because they have sufficient fixed neutron absorber plates to mitigate any reactivity increase, nor is it a problem for low-density PWR racks if soluble boron is credited. But compression of a low-density BWR rack could lead to a criticality since BWR racks contain no soluble or solid neutron-absorbing material. This is not a surprising result since low-density BWR fuel racks use geometry and fuel spacing as the primary means of maintaining subcriticality. High-density racks rely on both fixed neutron absorbers and geometry to control reactivity. Low-density racks rely solely upon geometry for reactivity control. In addition, all PWR pools are borated, whereas BWR pools contain no soluble

neutron-absorbing material. If BWR pools were borated, criticality would not be possible during a low-density rack compression event.

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

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

Other potential criticality events, such as events involving loose pellets or the impact of water (adding neutron moderation) during personnel actions in response to accidents, were discounted because the basic physics and neutronic properties of the racks and fuel would prevent criticality conditions from being reached with any credible likelihood. For example, without moderation fuel at current enrichment limits (no greater than 5 wt% U-235) cannot achieve criticality, no matter what the configuration. If it is assumed that the pool water is lost, a reflooding of the storage racks with unborated water may occur during personnel actions. However, both PWR and BWR storage racks are designed to remain subcritical if moderated by unborated water in the normal configuration. Thus, the only potential credible scenarios are the two scenarios described above, which involve crushing of fuel assemblies in low-density racks or degradation of Boraflex over long periods in time. These conclusions assume present light-water uranium oxide reactor fuel designs. Alternative fuel designs, such as mixed oxide (MOX) fuels will have to be reassessed to ensure that additional vulnerabilities for pool criticality do not exist.

To gain qualitative insights on credible criticality events, the staff considered the sequences of events that must occur. For scenario 1, a heavy load drop into a low-density racked BWR pool, compressing the assemblies would be required. From its analysis of the heavy load drop documented in Appendix 2C, the staff has determined the likelihood of a heavy load drop from a single failure-proof crane to have a mean frequency of approximately $9.6X10^{-6}$ per year, assuming 100 cask movements per year at the decommissioning facility. From the load path analysis done in Appendix 2C, the staff estimates that the load is over or near the pool approximately 13 percent of the movement path length, depending on the plant's layout. The additional frequency reduction in the appendix to account for the fraction of time that the heavy load is lifted high enough to damage the pool liner is not applicable here because the fuel assemblies can be crushed by a smaller impact velocity than required to need to crush the pool liner. Therefore, the staff estimates that the potential initiating frequency for crushing is approximately $1.2X10^{-6}$ per year (based upon 100 lifts per year). The criticality calculations in Appendix 3 show that even if the low-density BWR assemblies were crushed by a transfer cask, it is "highly unlikely" that a configuration would be produced that would result in a severe reactivity event, such as a steam explosion that 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 if the transfer cask drops directly onto the assemblies. This would put the significant criticality likelihood well below $1X10^{-8}$ per year.

Deformation of the low-density BWR racks by the dropped transfer cask was shown to most likely not result in any criticality events. However, if some mode of criticality was to be induced by the dropped transfer cask, it would likely be a small return to power for a very localized region, rather than the severe response discussed in the paragraph above. This type of event would have essentially no offsite (or onsite) consequences since the heat of the reaction would be removed by localized boiling in the pool, and water would shield 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 while the fuel remains stored in the pool.

With respect to scenario 2 (the gradual degradation of the Boraflex absorber material in high-density storage racks), there is currently insufficient data to quantify the likelihood of criticality due to the degradation. However, the current programs in place at operating plants to assess the condition of the Boraflex and take remedial action if necessary provide sufficient confidence that pool reactivity requirements will be satisfied. In order to meet the RG 1.174 safety principle of maintaining sufficient safety margins, the staff judges that continuation of such programs into the decommissioning phase should be considered at all plants until all high-density racks are removed from the SFP. As such, SDA #7 should be considered in future regulatory activities associated with SFP requirements. This additional assumption is identified as SDA #7.

SDA #7 Licensees will maintain a program to provide surveillance and monitoring of Boraflex in high-density spent fuel racks until such time as spent fuel is no longer stored in these high-density racks.

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

3.7 Consequences and Risks of SFP Accidents

This section assesses the consequences and risks associated with SFP accidents. The consequences are assessed in Section 3.7.1. Results are provided for both early evacuation and late evacuation cases[12] to address the impact of evacuation on consequences, and for two

[12] Early evacuation is initiated and completed before the SFP release. Late evacuation is not completed before release.

different source terms to show the impact of source term uncertainties on results. In Section 3.7.2, the severe accident consequences for either the early or late evacuation cases are assigned to each of the major types of SFP accidents, as appropriate, and then combined with the respective event frequencies to provide a scoping estimate of SFP risks. The risks of SFP accidents are shown to meet the Commission's safety goals. The impact of changes in EP regulations on these risk measures is discussed later in Section 4.

3.7.1 Consequences of SFP Accidents

Earlier analyses in NUREG/CR-4982, "Severe Accidents in Spent Fuel Pools in Support of Generic Issue 82," and NUREG/CR-6451, "A Safety and Regulatory Assessment of Generic BWR and PWR Permanently Shutdown Nuclear Power Plants," included a limited analysis of the offsite consequences of a severe SFP accident occurring up to 90 days after the last discharge of spent fuel into the SFP. The analysis showed that the consequences of an SFP accident could be comparable to those for a severe reactor accident. As part of its effort to develop generic, risk-informed requirements for decommissioning, the staff performed a further analysis of the offsite radiological consequences of beyond-design-basis SFP accidents. Varying the evaluation and other modeling assumptions, the staff performed an initial set of calculations to extend the earlier analyses to SFP accidents occurring 1 year after plant shutdown and to supplement the earlier analyses with additional sensitivity studies. The results of these calculations were documented in the February 2000 study, and are provided in Appendix 4.

Subsequently, the ACRS raised issues with the source term and plume modeling for SFP accidents. In particular, the ACRS believed that the ruthenium and fuel fines releases were too low and the plume was too narrow. To address these issues, the staff performed additional sensitivity studies, as documented in Appendix 4A of this study.

To provide insight into the impact on results of decay times shorter or longer than 1 year, additional consequence calculations were performed using fission product inventories at 30 and 90 days and 2, 5, and 10 years after final shutdown. The results of these consequence calculations were used as the basis for assessing the risk from SFP accidents. These results are summarized in Tables 3.7-1 and 3.7-2 for several key consequence measures, and are described in more detail in Appendix 4B. These consequences are conditional upon the occurrence of an accident that results in an SFP fire, i.e., the consequences are on a "per event" rather than a "per year," basis and do not account for the probability of the event.

These calculations were based on the Surry site, although the SFP accident consequences could be greater at higher population sites, the quantitative health objectives used in comparisons to the Commission's Safety Goals (see Section 3.7.3) represent risk to the average individual within 1 mile and 10 miles of the plant, and should be relatively insensitive to the site specific population.

Table 3.7-1 Consequences of an SFP Accident With a High Ruthenium Source Term (per event)

Time After Shutdown	Mean Consequences for High Ruthenium Source Term (Surry population, 95% evacuation)			
	Early Fatalities	Societal Dose (p-rem within 50 miles)	Individual Risk* of Early Fatality (within 1mile)	Individual Risk* of Latent Cancer Fatality (within 10 miles)
Late Evacuation				
30 days	192	$2.37x10^7$	$4.43x10^{-2}$	$8.24x10^{-2}$
90 days	162	$2.25x10^7$	$4.19x10^{-2}$	$8.20x10^{-2}$
1 year	77	$1.93x10^7$	$3.46x10^{-2}$	$8.49x10^{-2}$
2 years	19	$1.69x10^7$	$2.57x10^{-2}$	$8.42x10^{-2}$
5 years	1	$1.45x10^7$	$8.96x10^{-2}$	$7.08x10^{-2}$
10 years	-	$1.34x10^7$	$4.68x10^{-2}$	$6.39x10^{-2}$
Early Evacuation				
30 days	7	$1.35x10^7$	$2.01x10^{-3}$	$4.79x10^{-3}$
90 days	4	$1.29x10^7$	$1.87x10^{-3}$	$4.77x10^{-3}$
1 year	1	$1.12x10^7$	$1.50x10^{-3}$	$4.33x10^{-3}$
2 years	-	$9.93x10^6$	$1.12x10^{-3}$	$3.70x10^{-3}$
5 years	-	$8.69x10^6$	$3.99x10^{-4}$	$2.93x10^{-3}$
10 years	-	$8.13x10^6$	$2.05x10^{-4}$	$2.64x10^{-3}$

* Conditional on event - Total frequency for all events is shown in Table 3.1 as less than $3x10^{-6}$ per year.

Table 3.7-2 Consequences of an SFP Accident With a Low Ruthenium Source Term (per event)

Time After Shutdown	Mean Consequences for Low Ruthenium Source Term (Surry population, 95% evacuation)			
	Early Fatalities	Societal Dose (p-rem within 50 miles)	Individual Risk* of Early Fatality (within 1mile)	Individual Risk* of Latent Cancer Fatality (within 10 miles)
Late Evacuation				
30 days	2	5.58×10^{6}	1.27×10^{-2}	1.88×10^{-2}
90 days	1	5.43×10^{6}	9.86×10^{-3}	1.82×10^{-2}
1 year	1	5.28×10^{6}	7.13×10^{-3}	1.68×10^{-2}
2 years	-	5.12×10^{6}	5.64×10^{-3}	1.58×10^{-2}
5 years	-	4.90×10^{6}	3.18×10^{-3}	1.43×10^{-2}
10 years	-	4.72×10^{6}	1.63×10^{-3}	1.29×10^{-2}
Early Evacuation				
30 days	-	4.12×10^{6}	8.36×10^{-4}	9.92×10^{-4}
90 days	-	4.02×10^{6}	6.83×10^{-4}	9.62×10^{-4}
1 year	-	3.95×10^{6}	5.44×10^{-4}	9.09×10^{-4}
2 years	-	3.87×10^{6}	4.41×10^{-4}	8.71×10^{-4}
5 years	-	3.77×10^{6}	2.54×10^{-4}	8.14×10^{-4}
10 years	-	3.69×10^{6}	1.47×10^{-4}	7.70×10^{-4}

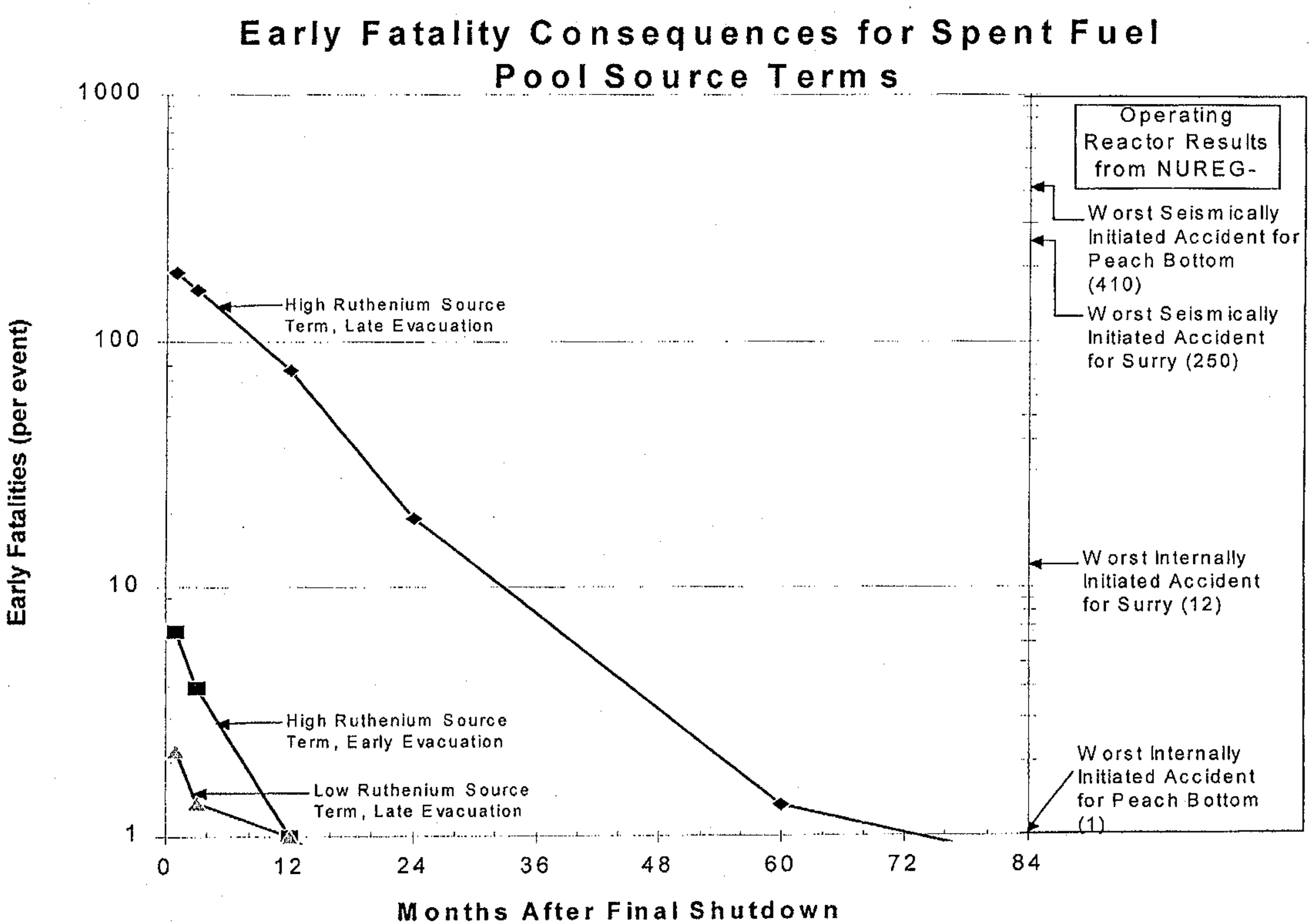
* Conditional on event - Total frequency for all events is shown in Table 3.1 as less than 3×10^{-5} per year.

The consequences in Table 3.7-1 are based on the upper bound source term described in Appendix 4B. With the exception of ruthenium and fuel fines, the release fractions are from NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants" (Ref. 1), and include the ex-vessel and late in-vessel phase releases. The ruthenium release fraction is for a volatile fission product in an oxidic (rather than metallic) form. This is consistent with the experimental data reported in Reference 8. The source term is considered to be bounding for several reasons. First, rubbling of the spent fuel after heatup to about 2500 °K is expected to limit the potential for ruthenium release to a value less than that for volatile fission products. Second, following the Chernobyl accident, ruthenium in the environment was found to be in the metallic form (Ref. 2). Metallic ruthenium (Ru-106) has about a factor of 50 lower dose conversion factor (rem per Curie inhaled) than the oxidic ruthenium assumed in the Melcor Accident Consequence Code System (MACCS) calculations. Finally, the fuel fines release fraction is that from the Chernobyl accident (Ref. 3). This is considered to be bounding because the Chernobyl accident involved more extreme conditions (i.e., two explosions followed by a prolonged graphite fire) than an SFP accident. In subsequent discussions, this source term is referred to as the high ruthenium source term.

The consequences obtained using the source term in NUREG-1465 (which treats ruthenium as a less volatile fission product) in conjunction with SFP fission product inventories are provided in Table 3.7-2 for comparison. In subsequent discussions, this source term is referred to as the low ruthenium source term.

The consequence calculations for both the high and low ruthenium source terms assume that all of the fuel assemblies discharged in the final core off-load and the previous 10 refueling outages participate in the SFP fire. These assemblies are equivalent to about 3.5 reactor cores. Approximately 85 percent of all the ruthenium in the pool is in the last core off-loaded since the ruthenium-106 half-life is about 1 year. For cesium-137, with a 30-year half-life, the inventory decays very slowly and is abundant in all of the batches considered. The staff assumed that the number of fuel assemblies participating in the SFP fire remains constant and did not consider the possibility that fewer assemblies might be involved in an SFP fire in later years because of substantially lower decay heat in the older assemblies. Based on the limited analyses performed to date, fire propagation is expected to be limited to less than two full cores 1 year after shutdown (see Appendix 1A). Thus, the assumption that 3.5 cores participate adds some conservatism to the calculation of long-terms effects associated with cesium, but is not important with regard to the effects of ruthenium.

The results for early fatality and societal dose (person-rem) consequences for an SFP accident are graphed in Figures 3.7-1 and 3.7-2. The early fatality plots are truncated at a value of one early fatality since fractions of a fatality are not meaningful. Since no early fatalities were predicted for the low ruthenium source term with early evacuation, a curve is not shown for that case in Figure 3.7-1. Because latent cancer fatalities are directly proportional to societal dose through a dose-to-cancer-risk conversion factor within the MACCS2 consequence code (Ref. 9), results for latent cancer fatalities are not displayed separately.



Early Fatality Consequences for Spent Fuel Pool Source Terms
Early Fatalities (per event)
1000
100
10
1
0
12
24
36
48
60
72
84
Months After Final Shutdown
High Ruthenium Source Term, Late Evacuation
High Ruthenium Source Term, Early Evacuation
Low Ruthenium Source Term, Late Evacuation
Operating Reactor Results from NUREG-
Worst Seismically Initiated Accident for Peach Bottom (410)
Worst Seismically Initiated Accident for Surry (250)
Worst Internally Initiated Accident for Surry (12)
Worst Internally Initiated Accident for Peach Bottom (1)

Figure 3.7-1

Societal (Person-rem) Consequences for Spent Fuel Pool Source Terms

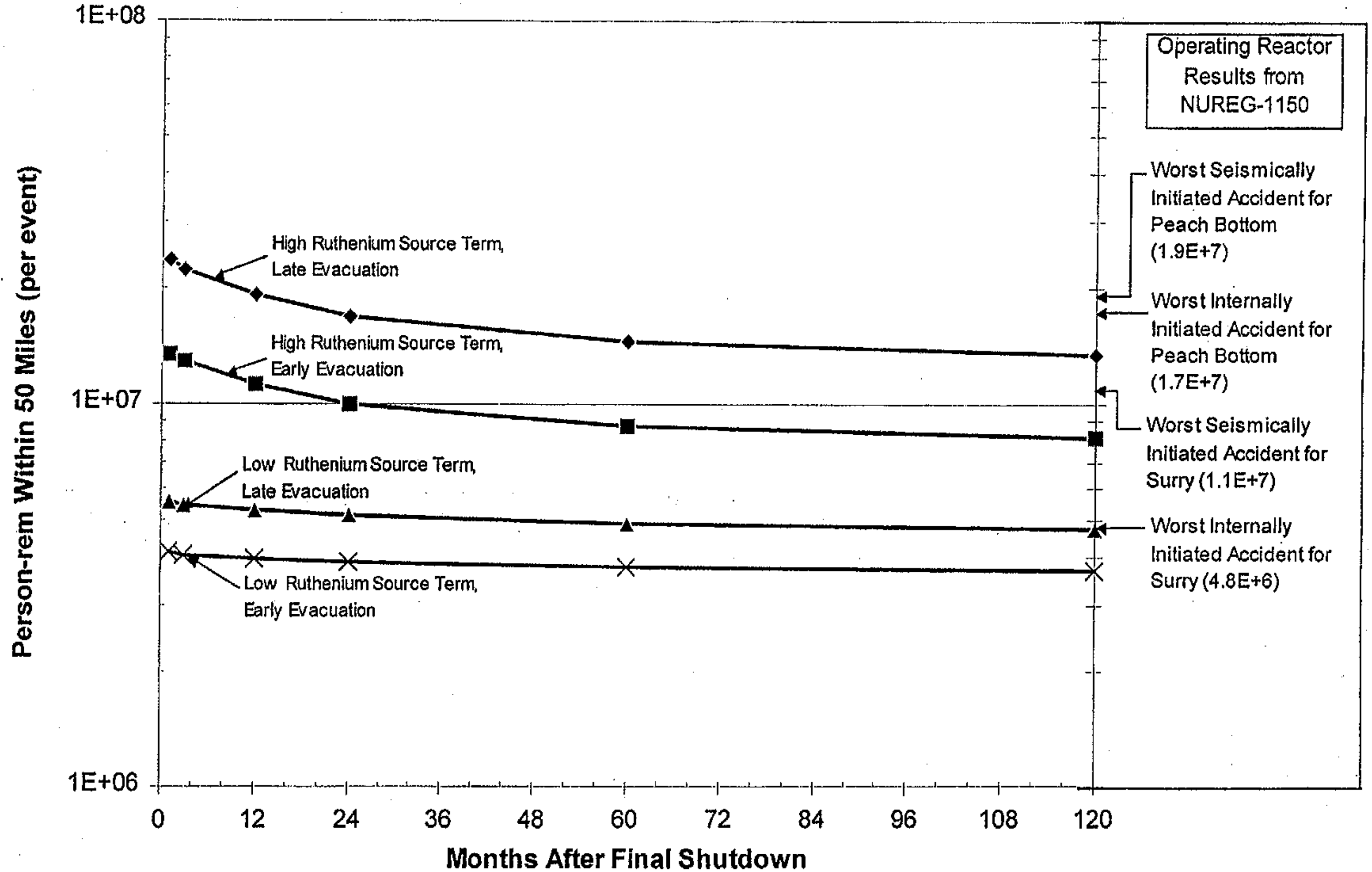


Figure 3.7-2

Consequence estimates are also included on Figures 3.7-1 and 3.7-2 for the two operating reactors for which risk results for both internal and seismic events are available in NUREG-1150, "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants," and the supporting NUREG/CR-4551 reports, "Evaluation of Severe Accident Risks: Surry Unit 1" and "Evaluation of Severe Accident Risks: Peach Bottom, Unit 2." The values shown are for the reactor accident source terms that produced the greatest number of early fatalities (Figure 3.7-1) or the greatest societal dose and latent cancer fatalities (Figure 3.7-2). Results are displayed separately for internally and seismically initiated accidents and indicate that for these plants, reactor accident consequences for seismically initiated events are substantially higher than those for internally initiated events. Although the consequences for the high ruthenium source term diminish more quickly than for the low ruthenium source term, these curves do not converge because of the long half-lives of the fuel fines in the high ruthenium source term.

An examination of Figure 3.7-1 indicates the following:

- Early fatality consequences for spent fuel pool accidents can be as large as for a severe reactor accident even if the fuel has decayed several years. This is attributable to the significant health effect of ruthenium, and the ruthenium-106 half-life of about 1 year. There is also an important but lesser contribution from cesium.

- A large ruthenium release fraction is important to consequences, but not more important than the consequences of a reactor accident large early release.

- The effect of early evacuation (if possible) is to offset the effect of a large ruthenium release fraction. This effect is comparable to that for reactor accidents.

- For the low ruthenium source term, no early fatality is expected after 1 year decay even with late evacuation.

For the longer term consequences Figure 3.7-2 indicates:

- Long-term consequences remain significant as long as a fire is possible. These consequences are due primarily to the effect of cesium-137, which remains abundant even in significantly older fuel because of its long (30-year) half-life. Ruthenium and evacuation have notable long-term consequences but do not change the conclusion.

3.7.2 Risk Modeling for SFP Accidents

The quantitative assessment of risk involves combining the estimated frequencies of severe accident sequences with their corresponding offsite consequences. In this section, severe accident consequences reported in Tables 3.7-1 and 3.7-2 are assigned to each of the major types of events that lead to uncovery of the spent fuel, and then combined with the respective event frequencies to provide a scoping estimate of SFP risks.

The SFP accidents discussed in Section 3 can be broadly classified as either boildown or rapid draindown sequences. Rapid draindown sequences are further divided into seismically- and non-seismically-initiated events. In assigning consequences to each of these events, the staff considered whether protective measures to evacuate the population around the site could be effectively implemented before fission product release. This included consideration of the effectiveness of offsite notification, the delay between event initiation and fission product release (dependent on time after shutdown), the time required to initiate and complete an evacuation, and the impact that a relaxation in current emergency planning requirements might have on these factors. As a result of this assessment, consequences were assigned based on either the early evacuation case or late evacuation case.

The frequency and consequence modeling is briefly described below for each type of SFP accident. The resulting risk estimates for each sequence (in terms of early fatalities and societal dose per year) are presented in Figures 3.7-3 through 3.7-6 and discussed in Section 3.7.3.

Boil Down Sequences

Boil down sequences (including loss of inventory events) and their associated frequencies are listed in Table 3.7-3. These sequences involve heatup of the pool to boiling followed by gradual reduction in pool level until the spent fuel is eventually uncovered. This process would take over 100 hours at 60 days, and substantially longer at later times as shown in Table 2.1. The long delay provides sufficient time for licensee staff to effectively intervene in the large majority of these events, and results in very low frequencies of fuel uncovery. For those events that proceed to fuel uncovery, fuel heatup will continue until either steady-state conditions are achieved or cladding oxidation occurs. All boil down sequences that uncover spent fuel were assumed to result in an SFP fire. Loss of inventory events are classified as boil down events since the time to uncover the fuel will be in excess of 24 hours (as described in Section 4.5.4.1 of Appendix 2A) and will provide ample time for licensee to take corrective measures.

Table 3.7-3 Frequency of Boil Down Events Leading to Spent Fuel Uncovery (for times greater than 60 days after shutdown)

Initiating Event	Frequency (per year)
Loss of offsite power—severe weather	1.1×10^{-7}
Loss of offsite power—plant-centered and grid-related events	2.9×10^{-8}
Internal fire	2.3×10^{-8}
Loss of pool cooling	1.4×10^{-8}
Loss of coolant inventory	3.0×10^{-9}
Total	1.8×10^{-7}

The failure paths leading to a zirconium fire involve failure to acquire offsite resources to makeup pool inventory, despite the large amount of time available for recovery in the boildown event. For sequences involving loss of offsite power due to severe weather, the weather is assumed to drain regional resources or limit access to the facility. The staff reasoned that if it is difficult for offsite resources to reach the facility or if regional resources are engaged in other efforts, then it would also be unlikely that the population in the area would be effectively notified and/or evacuated under these conditions. For sequences other than loss of off-site power due to severe weather, the dominant reason that recovery is not provided in the failure paths is a general breakdown in the overall facility organization. The failure to acquire offsite resources also implies a failure to contact regional authorities and declare an emergency when the SFP level drops below the proceduralized limit in these sequences. Accordingly, the consequences for boildown sequences are based on results for the late evacuation case (Tables 3.7-1 and 3.7-2). This same reasoning is applied for cases with and without EP relaxations and for all times after shutdown. The net effect is that EP, as well as relaxations in EP, do not impact the risk associated with those boildown sequences that proceed to spent fuel uncovery.

Rapid Draindown Due to Seismic Events

Given the robust structural design of SFPs, it is expected that a seismic event with peak spectral acceleration several times larger than the safe shutdown earthquake (SSE) would be required to produce catastrophic failure of the structure. The estimated frequency of events of this magnitude differs greatly among experts and is driven by modeling uncertainties. The estimated frequency of seismic events sufficiently large to result in structural failure of the SFP is given in Table 3.7-4 and is based on the LLNL and EPRI seismic hazard estimates.

Both the LLNL and EPRI hazard estimates were developed as best estimates and are considered valid by the NRC. Furthermore, because both sets of curves are based upon data extrapolation and expert opinion, there is no technical basis for excluding consideration of either set.

Using the LLNL hazard estimates, a return frequency equivalent to the pool performance guideline ($1x10^{-5}$ per year) for a 1.2g peak spectral acceleration (PSA) ground motion bounds all but four sites (one Central and Eastern and three Western U.S. sites). The frequency for the remaining sites falls in the range of less than $7x10^{-8}$ per year to $9x10^{-6}$ per year. The majority (45 sites) have hazard estimates (for a 1.2 PSA ground motion) near $1x10^{-6}$ per year and 20 sites fall below $6x10^{-7}$ per year. The mean value for the population of plants is approximately $2x10^{-6}$ per year.

If EPRI hazard estimates were used, only one site would have an estimate that exceeds $1x10^{-6}$ per year (excluding Western sites).[13] Ten sites are near $5x10^{-7}$ per year, and the remaining 49 sites analyzed by EPRI have estimates less than $3x10^{-7}$ per year, with half of these sites (25 sites) estimated at less than $7x10^{-8}$ per year. The mean value for the population of plants is approximately $2x10^{-7}$ per year.

In characterizing the risk of seismically induced SFP accidents for the population of sites, the staff has displayed results based on both the LLNL and the EPRI hazard estimates, and has used an accident frequency corresponding to the mean value for the respective distributions,

[13]EPRI seismic estimates were not developed for all sites east of the Rocky Mountains. Six sites have LLNL but no EPRI hazard estimates.

i.e., a frequency of 2×10^{-6} per year to reflect the use of the LLNL hazard estimates and a frequency of 2×10^{-7} per year to reflect use of the EPRI hazard estimates. Use of the mean value facilitates comparisons with the Commission's quantitative safety goals and quantitative health objectives (QHOs). About 70 percent of the sites are bounded using the mean value.

Table 3.7-4 Mean Frequency of Rapid Draindown Due to Seismic Events

Source of Hazard Estimate	Frequency (per year)
LLNL	2×10^{-6}
EPRI	2×10^{-7}

Likely SFP failure modes and locations are discussed in Attachment 2 to Appendix 2B. The conclusion is that drainage of the pool would be fairly rapid and a small amount of water is likely to remain in the pool, with post-seismic-failure, water depths ranging from about zero to about 4 feet depending upon the critical failure mode. For purposes of consequence assessment, all seismically initiated sequences were assumed to result in a rapid draindown followed by an SFP fire, regardless of the SFP failure mode and location, which are plant-specific.

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

- For low PGA earthquakes, the population was assumed to evacuate; however, the evacuation was assumed to start later and proceed more slowly than evacuation for internally initiated events.

- For high PGA earthquakes, it was reasoned that there would be no effective evacuation and that many structures would be uninhabitable.

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

The consequences for seismic sequences are therefore based on results for the late evacuation cases in Tables 3.7-1 and 3.7-2. The same reasoning is applied for cases with and without EP relaxations and for all times after shutdown. The net effect is that EP, as well as relaxations in EP, do not impact the risk associated with seismic events that result in SFP failure. A sensitivity study was also done to explore the impact on risk if the seismic event only partially degrades the emergency response (see Section 4.2.1).

Rapid Draindown Due to Non-Seismic Events

Non-seismically-initiated events leading to rapid draindown are listed in Table 3.7-5. These events are dominated by cask drop accidents, with the next highest contributor nearly two orders of magnitude lower.

Table 3.7-5 Frequency of Rapid Draindown Spent Fuel Uncovery Due to Nonseismic Events

Initiating Event	Frequency (per year)
Cask drop	2.0×10^{-7}
Aircraft impact	2.9×10^{-9}
Tornado missile	$<1.0 \times 10^{-9}$
Total	2.0×10^{-7}

Cask drop accidents that lead to catastrophic failure of the SFP include accidents in which the load is dropped either on the pool floor or on or near the pool wall. Load drops on the pool floor are more likely to result in complete draindown of the pool and create an air flow path through the fuel assemblies. Load drops on the pool wall would likely result in residual water in the pool, which would obstruct air flow. For purposes of consequence assessment, all cask drop accidents leading to fuel uncovery were assumed to result in a rapid draindown followed by an SFP fire.

Depending upon the pool failure mode and location, the fuel could be air cooled, or heatup could be close to adiabatic as a result of air flow blockage. As discussed in Appendix 1A for either adiabatic or air flow conditions (at 60 GWD/MTU burnup), the time of fission product release would be about 4 hours for a PWR and 8 hours for a BWR for accidents initiated 1 year following shutdown. For cases with air cooling, close to 1 day is available after 3 years decay. Even with adiabatic heatup, 1 day is available after 5 years of decay. At 60 days after shutdown, fission product release could begin as early as 2 hours after fuel uncovery. The actual time would depend on reactor type, fuel burnup, fuel rack structure, and other plant-specific parameters, as discussed in Appendix 1A. The fuel handlers would be immediately aware of a cask drop accident. It is expected that with procedures that specify the SFP water level at which an emergency is to be declared, the proper offsite authorities would be promptly informed.

For the case in which current EP requirements are retained, it was assumed that cask drop accidents occurring 1 or more years following shutdown would afford sufficient time to implement protective measures before fission products were released. This is consistent with the evacuation time estimates in the NUREG-1150 study for Surry, which assumed a 1.5 hour delay time and a 4 mile per hour evacuation speed. Thus the consequences at less than 1 year following shutdown are based on late evacuation, and the consequences at 1 year and beyond are based on early evacuation when full EP requirements are retained.

Relaxations in EP requirements are expected to result in additional delays in initiation and implementation of protective measures relative to the case in which current EP requirements are retained. If offsite preplanning requirements were to be relaxed, as many as 10 to 15 hours may be required at some sites to initiate an evacuation. Based on either air-cooled or adiabatic heatup rates for the reference pool, the minimum time to fission product release following a load drop that catastrophically damages the pool is about 8–9 hours for PWR pools and about 15 hours for BWR pools 2 years following shutdown (see Appendix 1A). These release times increase significantly by 5 years following shutdown (i.e., greater than 24 hours even with adiabatic heatup rates). For the case in which current EP requirements are relaxed, the consequences within the first 2 years following shutdown are based on late evacuation, and the consequences at 5 years and after are based on the early evacuation results reported in Tables 3.7-1 and 3.7-2.

3.7.3 Risk Results

The frequency and consequences for each SFP accident were combined to provide a scoping estimate of the risk of SFP accidents. The frequency of each event was based on the estimated value at 1 year following shutdown as described above, and was assumed to remain constant over time. In reality, the frequency would vary with time, and could be higher or lower than the 1-year estimate, as a result of plant configuration changes described in Section 3.1 (e.g., replacement of operating plant pool cooling and makeup systems with skid-mounted systems) and reductions in decay heat levels (which would impact human reliability estimates). However, as described in Section 3.4.7, these impacts are not expected to change the insights from the risk assessment for decay times greater than 60 days.

Figures 3.7-3 and 3.7-4 show the total early fatality risk and societal risk as a function of time after final shutdown. Companion curves are provided based on both the LLNL and the EPRI seismic hazard studies since both studies are considered equally valid. The SFP risk results are shown in these figures for both the high ruthenium source term and a fuel burnup of 60 GWD/MTU. Also shown are the corresponding mean risk measures for two operating plants, Surry and Peach Bottom,[14] for which risk results for both internal and seismic events are given in NUREG-1150.

Figures 3.7-5 and 3.7-6 show the risk contribution from cask drop events, which are the only events modeled that are significantly impacted by EP. For the case in which current EP requirements are retained, the consequences at 1 year and beyond are based on early evacuation (the lower, solid curve). For the case in which current EP requirements are relaxed, the consequences within the first 2 years following shutdown are based on late evacuation (the upper, solid curve), and the consequences at 5 years and beyond are based on early evacuation, as discussed in Section 3.7.2.

[14] The LLNL seismic risk results reported in NUREG-1150 are based on a 1989 version of the LLNL hazard estimates. An update of these estimates performed in 1993 resulted in a factor of 10 reduction in the LLNL mean hazard for Peach Bottom and a smaller reduction for Surry. To provide a more meaningful comparison, the LLNL seismic risk results for Peach Bottom reported in NUREG-1150 have been reduced by a factor of 10. The results for Surry and the EPRI seismic risk results are not affected by this adjustment.

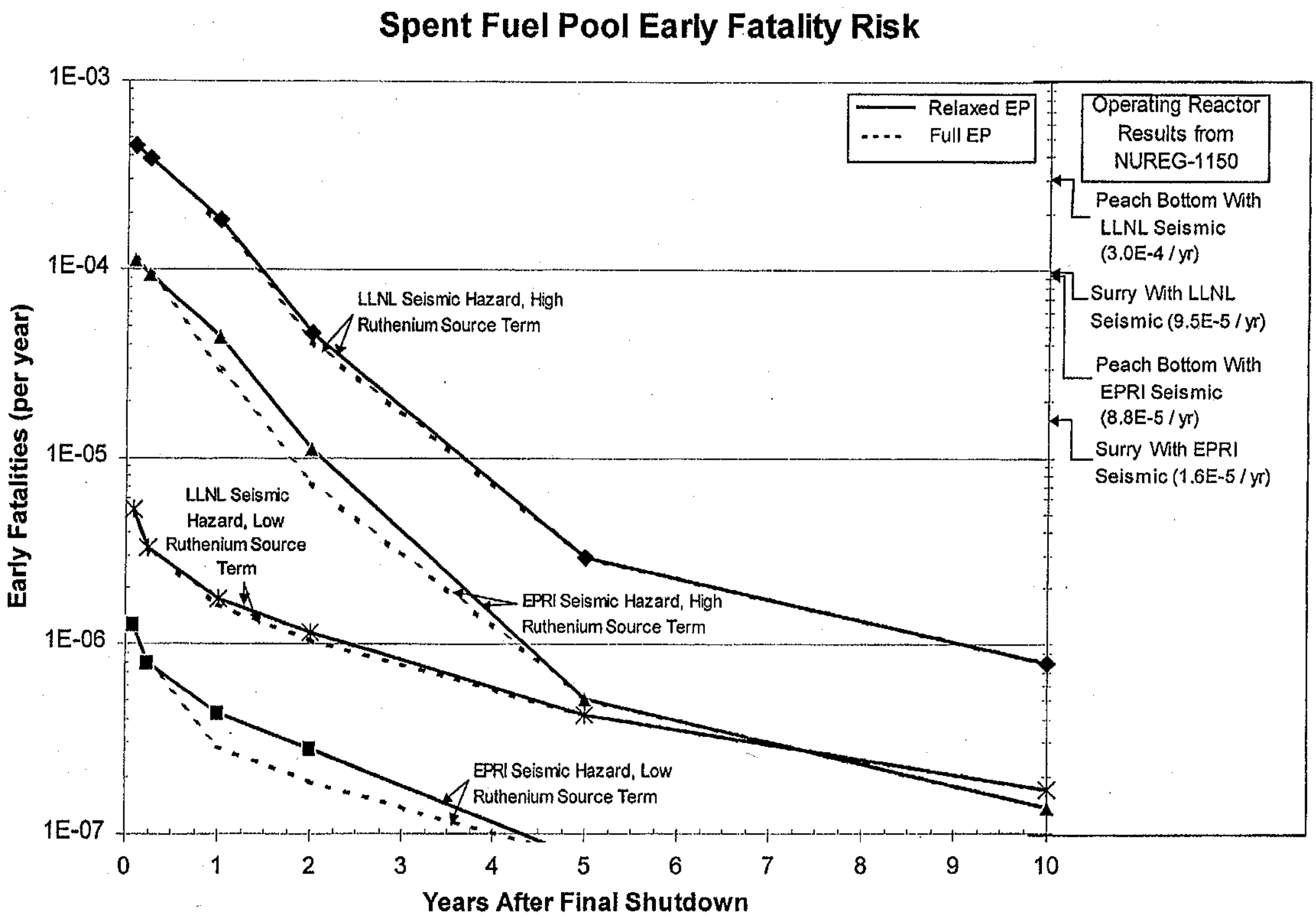
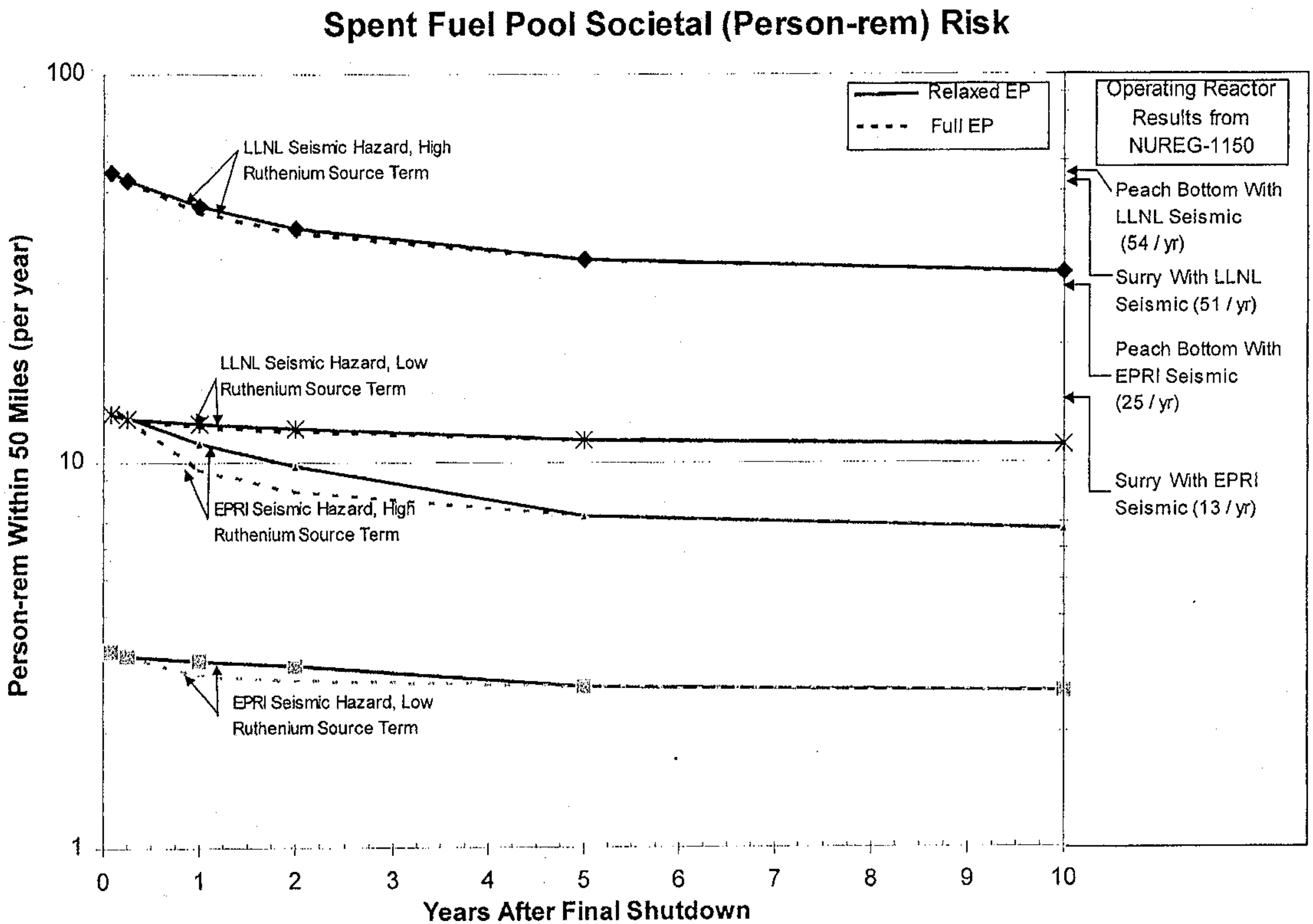

Spent Fuel Pool Early Fatality Risk
Early Fatalities (per year)
1E-03
1E-04
1E-05
1E-06
1E-07
0
1
2
3
4
5
6
7
8
9
10
Years After Final Shutdown
Relaxed EP
Full EP
Operating Reactor Results from NUREG-1150
Peach Bottom With LLNL Seismic (3.0E-4 / yr)
Surry With LLNL Seismic (9.5E-5 / yr)
Peach Bottom With EPRI Seismic (8.8E-5 / yr)
Surry With EPRI Seismic (1.6E-5 / yr)
LLNL Seismic Hazard, High Ruthenium Source Term
LLNL Seismic Hazard, Low Ruthenium Source Term
EPRI Seismic Hazard, High Ruthenium Source Term
EPRI Seismic Hazard, Low Ruthenium Source Term

Figure 3.7-3



Spent Fuel Pool Societal (Person-rem) Risk
Person-rem Within 50 Miles (per year)
Years After Final Shutdown
Relaxed EP
Full EP
Operating Reactor Results from NUREG-1150
LLNL Seismic Hazard, High Ruthenium Source Term
LLNL Seismic Hazard, Low Ruthenium Source Term
EPRI Seismic Hazard, High Ruthenium Source Term
EPRI Seismic Hazard, Low Ruthenium Source Term
Peach Bottom With LLNL Seismic (54 / yr)
Surry With LLNL Seismic (51 / yr)
Peach Bottom With EPRI Seismic (25 / yr)
Surry With EPRI Seismic (13 / yr)
100
10
1
0
1
2
3
4
5
6
7
8
9
10

Figure 3.7-4

Sensitivity of Early Fatality Risk to Emergency Planning -- Cask Drop Event

(Conditional upon: High Ruthenium Source Term)

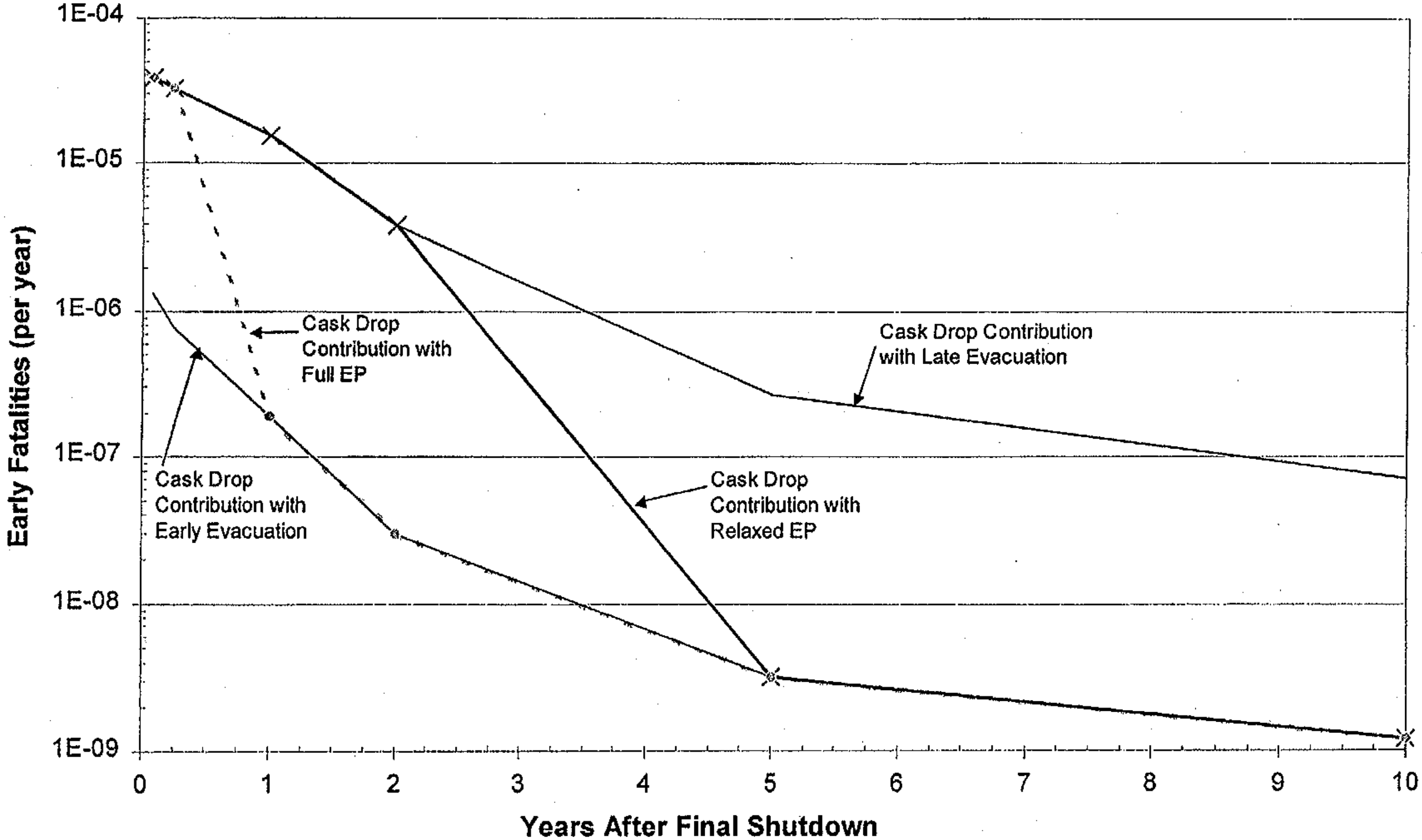


Figure 3.7-5

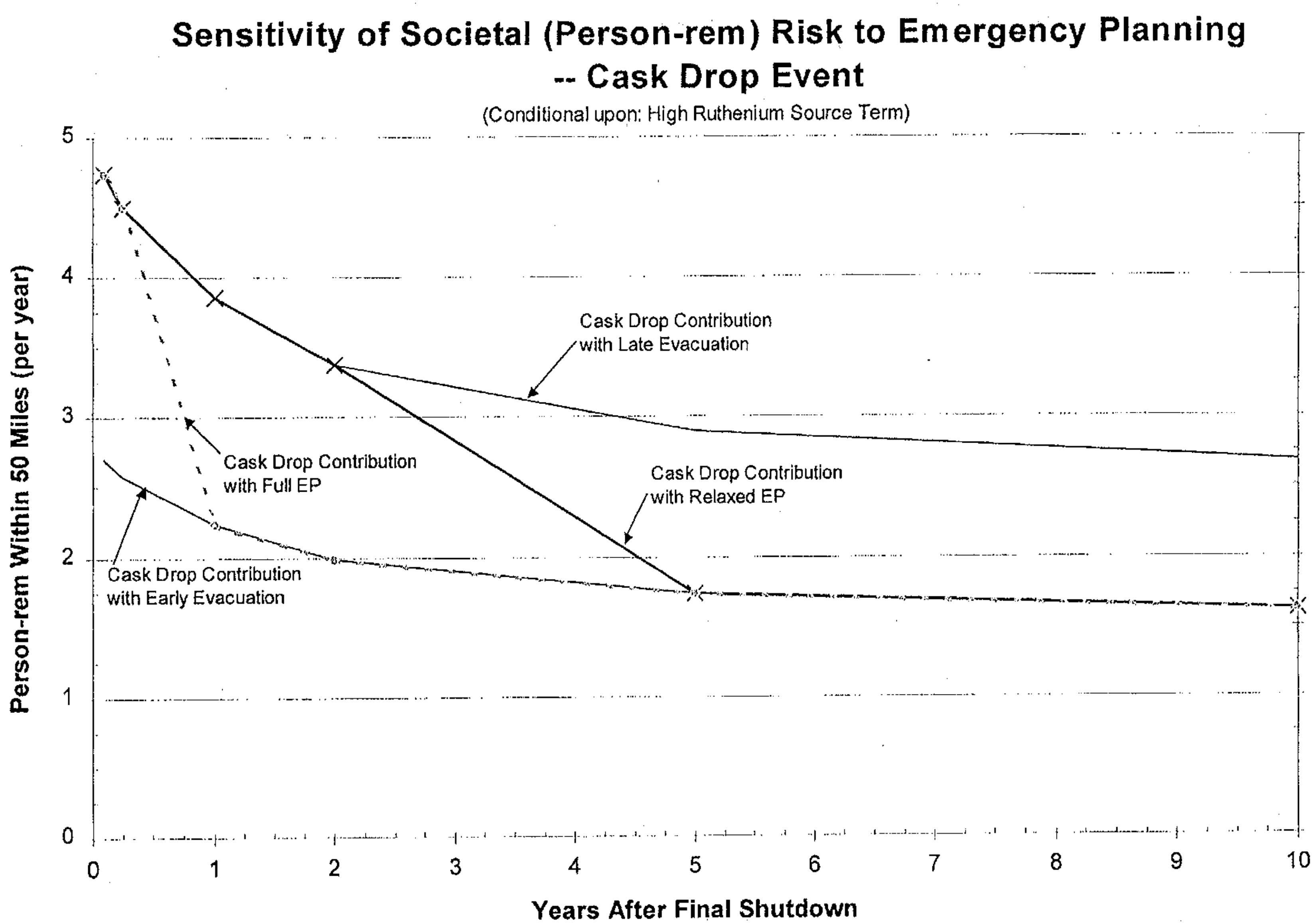

Sensitivity of Societal (Person-rem) Risk to Emergency Planning
-- Cask Drop Event
(Conditional upon: High Ruthenium Source Term)
Person-rem Within 50 Miles (per year)
Years After Final Shutdown
Cask Drop Contribution with Late Evacuation
Cask Drop Contribution with Full EP
Cask Drop Contribution with Relaxed EP
Cask Drop Contribution with Early Evacuation
0
1
2
3
4
5
6
7
8
9
10

Figure 3.7-6

On high ruthenium source term, The staff concludes:

- For the first 1 to 2 years following shutdown, the early fatality risk for an SFP fire is low, but may be comparable to that for a severe accident in an operating reactor (based on the two operating reactors considered). At 5 years following shutdown, the early fatality risk for SFP accidents is approximately two orders of magnitude lower than at shutdown. This is attributable to the effect of ruthenium, which decays to negligible amounts at 5 years.

- Societal risk for an SFP fire may be comparable to that for a severe accident in an operating reactor, but does not exhibit a substantial reduction with time because of the slower decay of fission products and the interdiction modeling assumptions that drive long-term doses.

- Of the SFP accidents assessed, only the cask drop accident is affected by changes to EP requirements. However, these changes do not substantially impact the total risk because the frequency of cask drop accidents is very low. As discussed previously, changes to EP requirements affect only the risk from cask drop accidents in the time period between 1 and 5 years.

- These observations are valid regardless of whether seismic event frequencies are based on the LLNL or the EPRI seismic hazard study.

About the low ruthenium source term the staff concludes:

- Use of the low ruthenium source term reduces early fatality risk by about a factor of 100 (relative to the high ruthenium source term) within the first 1 to 2 years and by about a factor of 10 at 5 years and after.

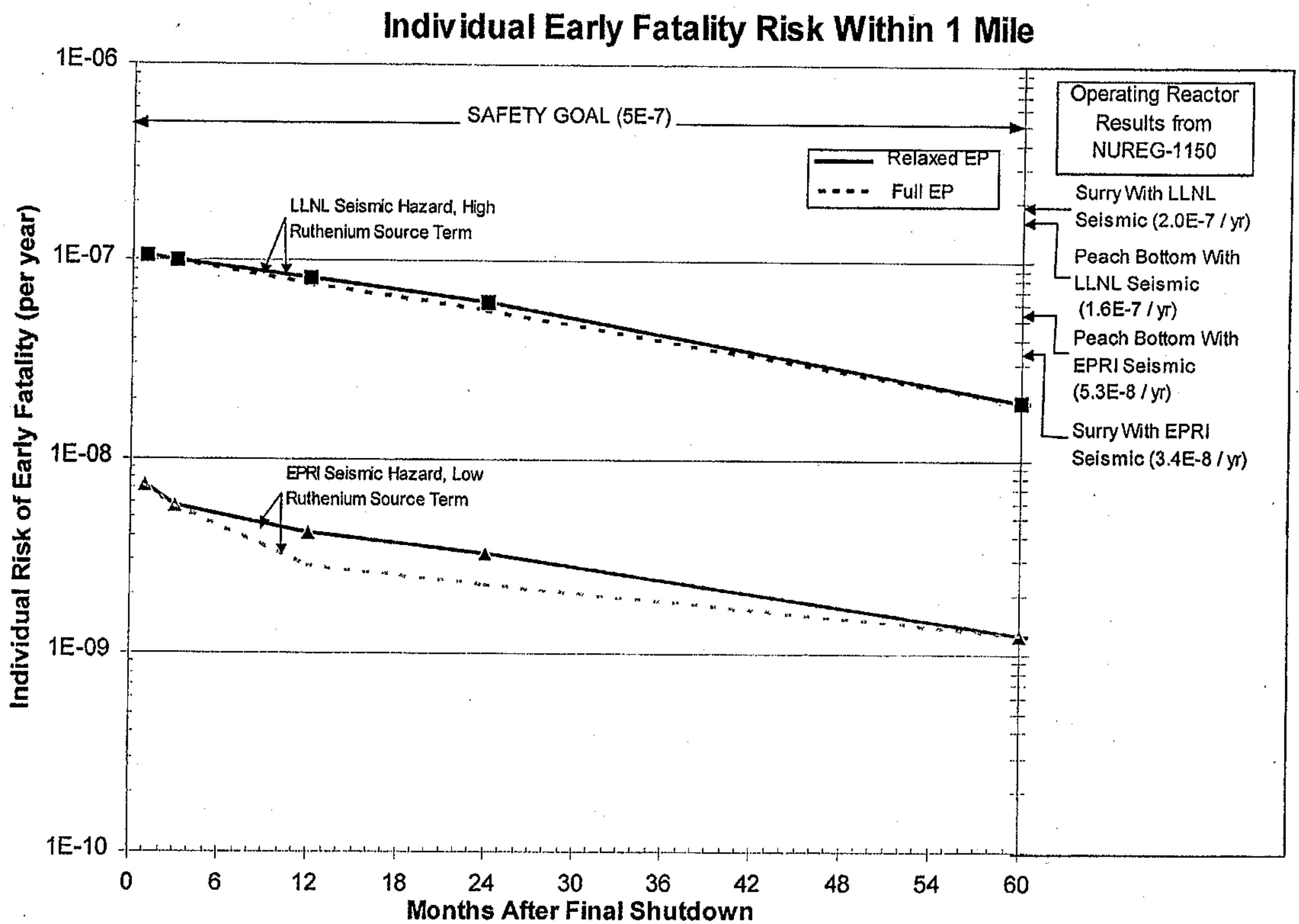
- With the low ruthenium source term, the early fatality risk for SFP accidents is about an order of magnitude lower than the corresponding values for a reactor accident shortly following shutdown and about two orders of magnitude lower at 2 years following shutdown. (In making these comparisons it is important to compare the SFP risks based on a particular seismic hazard estimate, e.g., EPRI, with reactor accident risks based on the same hazard estimate.)

- With the low ruthenium source term, the societal risk for SFP accidents is also about an order of magnitude lower than the corresponding values for a reactor accident shortly following shutdown, but does not exhibit a substantial reduction with time because of the slower decay of fission products and the interdiction modeling assumptions (discussed in Appendix 4) that drive long-term doses. Substantial reductions would only occur after about 5 years, when sufficient time appears to be available to initiate unplanned accident management recovery actions.

- As with the high ruthenium source term, changes to EP requirements affect the cask drop accident, and do not substantially impact the total risk due to the low frequency of cask drop accidents.

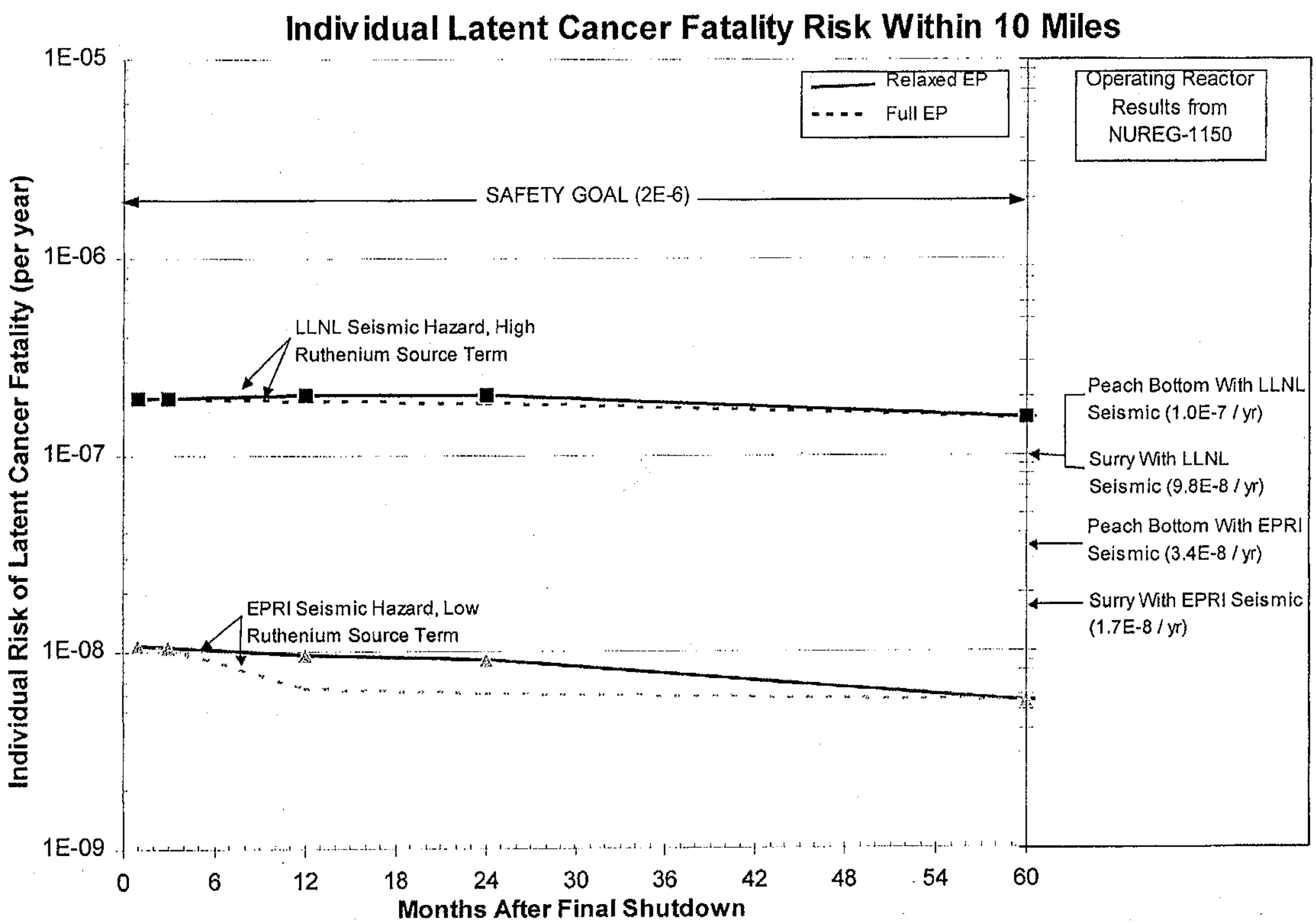
- These observations are valid regardless of whether seismic event frequencies are based on the LLNL or the EPRI seismic hazard estimates.

Figures 3.7-7 and 3.7-8 show the risk measures relevant to the Commission's safety goal policy statement, specifically, the individual risk of early fatality (to an individual within 1 mile of the site) and the individual risk of latent cancer fatality (to an individual within 10 miles of the site). The upper curves are based on the LLNL seismic hazard curves and the high ruthenium source term, and the lower curves are based on the EPRI hazard curves and the low ruthenium source term. Accordingly, these results may be viewed as a representative range of risk results for spent fuel pools uncovery given the conservative assumption that all SFP accidents result in a fire.



Individual Early Fatality Risk Within 1 Mile
1E-06
1E-07
1E-08
1E-09
1E-10
Individual Risk of Early Fatality (per year)
SAFETY GOAL (5E-7)
Relaxed EP
Full EP
LLNL Seismic Hazard, High Ruthenium Source Term
EPRI Seismic Hazard, Low Ruthenium Source Term
Operating Reactor Results from NUREG-1150
Surry With LLNL Seismic (2.0E-7 / yr)
Peach Bottom With LLNL Seismic (1.6E-7 / yr)
Peach Bottom With EPRI Seismic (5.3E-8 / yr)
Surry With EPRI Seismic (3.4E-8 / yr)
0
6
12
18
24
30
36
42
48
54
60
Months After Final Shutdown

Figure 3.7-7



Individual Latent Cancer Fatality Risk Within 10 Miles
Relaxed EP
Full EP
Operating Reactor Results from NUREG-1150
SAFETY GOAL (2E-6)
LLNL Seismic Hazard, High Ruthenium Source Term
EPRI Seismic Hazard, Low Ruthenium Source Term
Peach Bottom With LLNL Seismic (1.0E-7 / yr)
Surry With LLNL Seismic (9.8E-8 / yr)
Peach Bottom With EPRI Seismic (3.4E-8 / yr)
Surry With EPRI Seismic (1.7E-8 / yr)
1E-05
1E-06
1E-07
1E-08
1E-09
Individual Risk of Latent Cancer Fatality (per year)
0
6
12
18
24
30
36
42
48
54
60
Months After Final Shutdown

Figure 3.7-8

The individual early fatality risk for an SFP accident is about one to two orders of magnitude lower than the Commission's safety goal, depending on assumptions about the SFP accident source term and seismic hazard. For the upper curve (corresponding to use of the mean of the LLNL seismic hazard estimates and the high ruthenium source term), the risks are about a decade lower than the safety goal. For the lower curve (corresponding to use of the mean of the EPRI seismic hazard estimates and the low ruthenium source term) the risks are about 2 decades lower than the safety goal. The individual early fatality risk for an SFP accident decreases with time and is about a factor of 5 lower at 5 years following shutdown (relative to the value at 30 days).

Similarly, the individual latent cancer fatality risk for an SFP accident is about one to two orders of magnitude lower than the Commission's safety goal, depending on assumptions about the SFP accident source term and seismic hazard. For the upper curve (corresponding to use of the mean of the LLNL seismic hazard estimates and the high ruthenium source term), the risks are about a decade lower than the safety goal. For the lower curve (corresponding to use of the mean of the EPRI seismic hazard estimates and the low ruthenium source term), the risks are about 2 decades lower than the safety goal. The individual latent cancer fatality risk for an SFP accident are not substantially reduced with time because of the slower decay of fission products and the interdiction modeling assumptions that drive long-term doses.

Changes to EP requirements, as modeled, do not substantially impact the margin between SFP risk and the safety goals because of the low frequency of events for which EP would be effective.

4.0 IMPLICATIONS OF SPENT FUEL POOL (SFP) RISK FOR REGULATORY REQUIREMENTS

The primary purpose of this study is to provide risk insights to support possible revisions to regulatory requirements for decommissioning plants. Section 4.1 below describes the safety principles of Regulatory Guide (RG) 1.174 as they apply to an SFP, and examines the design, operational, and regulatory elements that are important in ensuring that the risk from an SFP continues to meet these principles. This technical assessment explores possible implications for EP requirements, but the same technical information also provides risk insights to inform regulatory decisions on changes in insurance, safeguards, staffing and training, backfit, and other requirements for decommissioning plants. Section 4.2 examines the implications of the technical results for these regulatory decisions, and how future regulatory activity might reflect commitments and assumptions. The implications of safeguards events are not included in this evaluation.

4.1 Risk-Informed Decision Making

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. Subsequent to issuance of the PRA policy statement, the agency published RG 1.174, which contained general guidance for application of PRA insights to the regulation of nuclear reactors. The regulatory framework proposed in this study for decommissioning plants is based on the risk-informed decision-making process described in RG 1.174 (Ref. 2). Although the focus of RG 1.174 is decision making regarding changes to the licensing basis of an operating plant, the same risk-informed philosophy can be applied generically as part of the evaluation of potential exemptions or changes to current regulatory requirements for decommissioning plants.

RG 1.174 articulates the following safety principles, which can be applied in evaluating regulatory changes for decommissioning plants:

- 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.

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

- The proposed change maintains sufficient safety margins.

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

A discussion of each of these safety principles and how they would continue to be satisfied at a decommissioning plant is provided in the sections that follow. 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.

4.1.1 Increases in Risk

RG 1.174 states that 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 has evaluated the risks associated with SFP accidents and the impacts of potential changes to regulatory requirements for decommissioning plants relative to applicable regulatory guidance. Guidance on acceptable levels of (total) risk to the public from nuclear power plant operation is provided in the Commission's safety goal policy statement (Ref. 3). Additional guidance on the acceptable levels of risk increase from a change to the plant licensing basis is provided in RG 1.174. The guidance contained in these documents is summarized below and used in this study to evaluate the risks associated with SFP accidents and the impacts of potential changes to regulatory requirements for decommissioning plants.

SFP Risk Relative to the Safety Goal Policy Statement

The "Policy Statement on Safety Goals for the Operation of Nuclear Power Plants," issued in 1986, establishes goals that broadly define an acceptable level of radiological risk to the public as a result of nuclear power plant operation. These goals are used generically to assess the adequacy of current requirements and potential changes to the requirements. The Commission established two qualitative safety goals that are supported by two quantitative objectives for use in the regulatory decision-making process. The qualitative safety goals stipulate the following:

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

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

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

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

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

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

- The individual risk of a prompt fatality from all "other accidents to which members of the U.S. population are generally exposed," such as fatal automobile accidents, is about 5×10^{-4} per year. One-tenth of 1 percent of this figure implies that the individual risk of prompt fatality from a reactor accident should be less than 5×10^{-7} per reactor year.

- "The sum of cancer fatality risks resulting from all other causes" for an individual is taken to be the cancer fatality rate in the U.S. which is about 1 in 500 or 2×10^{-3} per year. One-tenth of 1 percent of this risk means that the risk of cancer to the population in the area near a nuclear power plant due to its operation should be limited to 2×10^{-6} per reactor year.

Although the policy statement and related numerical objectives were developed to address the risk associated with power operation, the QHOs provide a convenient benchmark for SFP evaluations. Accordingly, the staff has compared the estimated risks associated with SFP accidents to the QHOs.

The risks associated with SFP accidents compare favorably with the QHOs. The comparisons, presented in Section 3.7.3, show that a typical site that conforms with the IDCs and SDAs would meet the QHOs by one to two orders of magnitude a few months following shutdown and by greater margins later. The risk comparisons provided in Appendix 4C show that SFP facilities maintained at or below the recommended pool performance guideline (PPG) of 1×10^{-5} per year, would continue to meet the QHO even with a severe SFP source term. With the exception of H.B. Robinson (using the LLNL seismic hazard estimates and generic fragilities), all Central and Eastern U.S. plants which satisfy the IDCs and SDAs (and pass the seismic checklist) will meet the PPG. Western plants (including San Onofre, Diablo Canyon, and WNP-2) were not included in the LLNL or EPRI seismic hazard studies and need to demonstrate compliance with the PPG on a plant-specific basis.

Risk Increases Relative to Regulatory Guide 1.174

The guidelines in RG 1.174 pertain to the core damage frequency (CDF) and large early release frequency (LERF). For both CDF and LERF, RG 1.174 contains guidance on acceptable values for the changes that can be allowed due to regulatory decisions as a function of the baseline frequencies. For example, if the baseline CDF for a plant is below 1×10^{-4} per year, plant changes can be approved that increase CDF by up to 1×10^{-5} per year. If the baseline LERF is less than 1×10^{-5} per year, plant changes can be approved which increase LERF by up to 1×10^{-6} per year.

For decommissioning plants, the risk is primarily due to the possibility of a zirconium fire involving the spent fuel cladding. The consequences of such an event do not equate directly to either a core damage accident or a large early release as modeled for an operating reactor. Zirconium fires in SFPs have the potential for significant long-term consequences because multiple cores may be involved; the relevant cladding and fuel degradation mechanisms could lead to increased releases of certain isotopes (e.g., short-lived isotopes such as iodine will have decayed, but the release of long-lived isotopes such as ruthenium could be increased due to air-fuel reactions); and there is no containment surrounding the SFP to mitigate the consequences. On the other hand, after about 2 years, the consequences are different than from a large early release because the postulated accidents progress more slowly, allowing time for protective actions to be taken to significantly reduce early fatalities (and to a lesser extent latent fatalities). In effect, an SFP fire would result in a "large" release, but this release would not generally be considered "early" due to the significant time delay before fission products are released.

In spite of the differences relative to an operating reactor large early release event and the differences in isotopic makeup, the consequence calculations performed by the staff and discussed in Section 3.7 show that SFP fires could have health effects comparable to those of a severe reactor accident. These calculations considered the effects of different source terms and evacuation assumptions on offsite consequences. Since an SFP fire scenario would involve a direct release to the environment with significant consequences, the staff has decided that the RG 1.174 guidance concerning LERF can be applied to the issue of SFP risks for decommissioning plants.

The LERF guidance is applied in two ways in this study:

(1) Because the changes in EP requirements affect not the frequency of events involving a large early release (i.e., the SFP fire frequency) but the consequences of these releases, the allowable increase in LERF in RG 1.174 is translated into an allowable increase in key risk measures. The estimated risk increases associated with changes in EP requirements are then compared to the allowable increases inferred from RG 1.174. These comparisons are presented in Appendix 4D.

(2) The RG 1.174 guidance is used to establish a PPG. The PPG provides a threshold for controlling the risk from a decommissioning SFP. By maintaining the frequency of events leading to uncovery of the spent fuel at a value less than the recommended PPG value of 1×10^{-5} per year, zirconium fires will remain highly unlikely, the risk will continue to meet the Commission's QHOs, and changes to the plant SFP licensing basis that result in very small increase in risk may be permitted consistent with the logic in RG 1.174. A licensee would need to assure that the frequency of events leading to uncovery of the spent fuel would be less than the PPG in order to consider the risk-informed changes in a rule for decommissioning plants. With the exception of those plants mentioned above, this assurance could be provided by conforming with the IDCs and SDAs listed in Tables 4.2-1 and -2. The use of the LERF guidance (1×10^{-5} per year frequency of fuel uncovery) was questioned by the ACRS because of concerns related to SFP source terms and accident consequences. The rationale for the PPG is presented in Appendix 4C.

The risk increases associated with relaxations in EP requirements compare favorably with the guidance contained in RG 1.174 (see Table 4 of Appendix 4D). Relaxation of EP requirements would result in an increase of about 1.5×10^{-5} early fatalities per year and 2 person-rem per year for the Surry analysis, the first is about a factor of 15 and the second a factor of 5 below the allowable increase inferred from the RG 1.174 LERF criteria. The increase in the QHO risk measures is also substantially lower than that allowed in RG 1.174. Since the SFP fire frequency assumed in these comparisons is about a factor of 4 lower than the PPG of $1x10^{-5}$ per year, an SFP facility operating nominally at the PPG would have a smaller margin to the allowable risk limits for the reference plant but would still be at or below these limits.

As discussed in Section 3.7, the basis for these results is that EP is of marginal benefit in large earthquakes because of offsite damage. However, as described in Appendix 4D, even with unrealistically optimistic assumptions about the effectiveness of EP in seismic events (i.e., assuming full and relaxed EP results in early and late evacuation, respectively, and using the LLNL seismic hazard frequency and the high ruthenium source term), the change in risk is small and the QHOs continue to be met with adequate margin.

Measures to Assure Risk Increases Remain Small

The analysis in Section 3 explicitly examines the risk impact of specific design and operational characteristics. This analysis credits the industry decommissioning commitments (IDCs) proposed by NEI in a letter to the NRC dated November 12, 1999 (Ref. 4) and several additional staff decommissioning assumptions (SDAs) identified through the staff's risk assessment and the staff's evaluation of the RG 1.174 safety principles for decommissioning plants. The IDCs and SDAs are summarized in Tables 4.2-1 and 4.2-2.

The low numerical risk results shown in Section 3 and Appendix 2 are predicated on the IDCs and SDAs being fulfilled. Specifically,

- IDC #5 and SDAs #2 and #3 provide assurance of timely operator response for a broad range of operational events.
- The low likelihood of pool failure due to heavy load drop is dependent on design and procedural controls for handling of heavy loads (IDC #1 and #9 and SDA #5).
- The low baseline frequency for a seismically initiated zirconium fire is predicated upon implementation of the seismic checklist shown in Appendix 5 (SDA #6).
- The low likelihood of loss of cooling is dependent upon procedures and training (IDC #2) and instrumentation (IDC #5 and SDA #3).
- The low likelihood of loss of inventory is dependent upon design provisions (IDC #6) and procedures and controls (IDC #7) to limit leakage.
- The high probability that the operators will identify and recover from a loss of cooling or a loss of inventory event is dependent upon procedures and training for effective use of onsite and offsite resources (IDCs #2 through #4, IDC #8, and SDA #3) and SFP instrumentation (IDC #5 and SDA #3).
- The low likelihood criticality issues is dependent on continuation of programs to assess the condition of Boraflex absorber material (SDA #7).

- Applicability of the staff's generic risk assessment to a specific facility is assured by SDA #1.

With regard to SFP risks and risk increases associated with EP relaxations, the staff concludes:

- An SFP facility that conforms with the IDCs and SDAs would meet the QHOs by one to two orders of magnitude shortly after shutdown and with greater margins at later times.

- The risk increase associated with relaxations in EP requirements is very small, even under assumptions that maximize the effectiveness of emergency preparedness in seismic events (i.e., assigning consequences for the "full EP" case based on early evacuation and consequences for the "relaxed EP" case based on late evacuation), and the QHOs continue to be met with adequate margin.

- Continued conformance with IDCs and SDAs provides reasonable assurance that the SFP risk and risk increases associated with regulatory changes would remain small.

4.1.2 Defense-in-Depth

RG 1.174 states that the proposed change should be consistent with the defense-in-depth philosophy.

In accordance with the Commission white paper on risk-informed regulation (March 11, 1999), "Defense-in-depth is an element of the NRC's Safety Philosophy that employs successive compensatory measures to prevent accidents or mitigate damage if a malfunction, accident, or naturally caused event occurs at a nuclear facility. The defense-in-depth philosophy ensures that safety will not be wholly dependent on any single element of the design, construction, maintenance, or operation of a nuclear facility. The net effect of incorporating defense-in-depth into design, construction, maintenance and operation is that the facility or system in question tends to be more tolerant of failures and external challenges." Therefore, application of defense-in-depth could mean in part that there is more than one source of cooling water or that pump make-up can be provided by both electric as well as direct-drive diesel pumps. Additionally, defense-in-depth can mean that even if a serious outcome (such as fuel damage) occurs, there is further protection, such as containment for operating plants to prevent radionuclide releases to the environment and emergency response measures to provide dose savings to the public.

The defense-in-depth philosophy applies to the operation of the SFP in a decommissioning plant. The philosophy also applies to the potential regulatory changes contemplated for decommissioning plants. Implementation of defense-in-depth for SFPs is different than for nuclear reactors because the hazards are different. The robust structural design of a fuel pool, coupled with the simple nature of the pool support systems, goes far toward preventing accidents associated with loss of water inventory or pool heat removal. Additionally, because the essentially quiescent (low-temperature, low-pressure) initial state of the SFP and the long time available for taking corrective action associated with most release scenarios provide significant safety margin, a containment structure is not considered necessary as an additional barrier to provide an adequate level of protection to the public. Likewise, the slow evolution of most SFP accident scenarios allows for reasonable human recovery actions to respond to system failures, and provides sufficient time to allow for the implementation of protective actions.

The staff's risk assessment demonstrates that the risk from a decommissioning plant SFP accident is small if IDCs and additional SDAs are implemented as assumed in the risk study. Due to the different nature of an SFP accident versus an accident in an operating reactor with respect to system design capability and event timing, the defense-in-depth function of reactor containment is not required. However, the staff has found that defense-in-depth in the form of accident prevention measures and an appropriate level of emergency planning can limit risk and provide dose savings for as long as a zirconium fire is possible.

Defense-in-depth for accident prevention and mitigation is provided by licensee conformance with the IDCs and SDAs, as discussed previously. Defense-in-depth for consequence mitigation should continue to be provided by retaining requirements for an appropriate level of EP in consideration of the amount of time available before fission product release in specific events.

For the purpose of the analysis in this study, when referring to relaxation of offsite EP, the study assumed conditions that would be similar to those at sites in decommissioning that have already received exemptions from some EP requirements. For instance, licensees may no longer be required: to have a formalized emergency planning zone (EPZ); to have an emergency operations facility (EOF), technical support center (TSC), or operations support center (OSC); to promptly notify the public using a siren system, tone alert radios, or National Weather Service radios; or to conduct biennial full-participation exercises. The analyses in the study were simplified to focus on conditions which assumed evacuation occurred either early or late.

It is understood that EP involves more than just evacuation considerations. In the analysis of the study, it was assumed that the decommissioning licensee would still be required to notify offsite authorities, characterize the releases, and make protective action recommendations; have a means of notifying offsite organizations and providing information to the public; and hold onsite biennial exercises and semiannual drills.

The assessments conducted for this study show that, 60 days after final shutdown, recovery and mitigation times of more than 100 hours are available before release occurs, except for the most severe events. These times appear to be sufficient to permit offsite protective actions to be implemented on an ad hoc basis, if necessary, without the full compliment of regulatory requirements associated with operating reactors. The staff notes that potential relaxation of EP requirements for decommissioning plants could be phased in such that the relaxation would not result in an immediate lapse of all offsite emergency response capabilities following final shutdown, but would more likely result in early elimination of some capabilities (e.g., sirens) and more gradual relaxation of certain other capabilities (e.g., pre-planning of evacuations and communications), with a transition towards longer ad hoc response times over several years due to such factors as attrition of experienced personnel. Shortly after final shutdown, when SFP heatup rates and risks are greatest, response capabilities are expected to be largely intact and comparable to those for full EP. These capabilities could be expected to diminish over time, resulting in longer ad hoc response times. However, continued fission product decay in the spent fuel will result in longer times to release, providing additional time during which emergency response measures could be implemented.

Only during the first several years and in the most severe events, such as severe seismic events, heavy load drops, and other dynamic events, that cause the pool to fail, would the accident progress so rapidly that emergency response measures might not be implemented in a timely manner. The staff's risk study indicates that the frequency of such events is dominated by earthquakes with a magnitude several times the safe shutdown earthquake (SSE). As discussed in Section 3.7.2, for ground motion levels that correspond to SFP failure, emergency planning would have marginal benefit because of extensive collateral damage to infrastructure (e.g., power, communications, buildings, roads, and bridges). Emergency response action would likely require substantial ad hoc action regardless of pre-planned actions in these events.

The next largest contribution is from cask drop sequences. The frequency of such events is low in the staff's risk study ($2x10^{-7}$ per year) due to implementation of IDCs and SDAs concerning movement of heavy loads. Relaxations in EP requirements could result in some increase in the risk associated with these events for a limited time following shutdown (1 to 5 years in the staff's analysis). However, the increase is a small fraction of the total risk from SFPs, as shown in Section 3.7. For the remaining SFP accidents that were analyzed and lead to SFP fires (e.g., boildown sequences due to organizational failures), current emergency planning was assumed to be ineffective or the frequencies of accidents, (e.g., aircraft impact) would be at least an order of magnitude lower than for the cask drop accident. Thus, mitigation of these events would not be risk significant.

With regard to defense-in-depth, the staff concludes:

- Defense-in-depth for accident prevention and mitigation is provided by the robust design of the SFP, the simple nature of pool support systems, and the long time available for taking corrective action in response to system failures.
- The substantial amount of time available for ad hoc offsite emergency response should provide some level of defense-in-depth for consequence mitigation in SFP accidents.
- In the large seismic events that dominate SFP risk, pre-planning for radiological accidents would have marginal benefit due to extensive collateral damage offsite. Accordingly, relaxations in EP requirements are not expected to substantially alter the outcome from such a large seismic event.
- There can be a tradeoff 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 available to respond gets longer.

4.1.3 Safety Margins

RG 1.174 states that the proposed change should maintain sufficient safety margins.

As discussed in Section 2, the safety margins associated with fuel in the SFP 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 provide adequate margins for accident prevention. Additionally, the presence of soluble boron or Boraflex provides additional assurance of margin with respect to shutdown reactivity.

The risk results provided in Section 3.6.3 show that a typical site that conforms with the IDCs and SDAs would meet the Commission's QHOs by one to two orders of magnitude, depending on assumptions about the SFP source term and seismic hazard frequency. The risk comparisons provided in Appendix 4C show that SFP facilities maintained at or below the recommended PPG of $1x10^{-5}$ per year would continue to meet the QHOs for even the most severe source term.

The estimated risk increases associated with the EP relaxations are also well below the allowable increases developed from the RG 1.174 LERF criteria. As discussed in Section 4.2.1 and Appendix 4D, the increases in risk from the EP relaxation would be about a factor of 10 below the maximum allowable increases developed from RG 1.174. Since the SFP fire frequency assumed in the RG 1.174 comparisons is about a factor of 4 lower than the PPG of $1x10^{-5}$ per year, a plant operating nominally at the PPG would have a smaller margin to the allowable risk limits for the reference plant but would still be at or below the limits.

The results of a sensitivity case in Appendix 4D indicate that even with assumptions that maximize the effectiveness of EP in seismic events, the change in risk associated with relaxation of the requirements for radiological preplanning is still relatively small. The increases in early fatalities and individual early fatality risk remain below the maximum allowable for each risk measure. Population dose and individual latent cancer fatality risk are about a factor of 2 higher than the allowable value inferred from RG 1.174. This increase in individual latent cancer risk represents about 9 percent of the QHO; thus, considerable margin to the QHO would still remain.

The evacuation effectiveness assumed for "full EP" in the sensitivity case is unrealistic for high ground motion earthquakes, and the risk increase associated with the EP relaxations is expected to be closer to the baseline value. Also, the risk reduction estimates are based on the LLNL seismic hazard frequencies and the high ruthenium source term, and would be substantially lower if either the EPRI seismic hazard frequencies or the low ruthenium source term were used. The above comparisons are based on the risk levels 1 year after shutdown but would also be valid several months following shutdown. Use of either the EPRI seismic hazard frequencies or the low ruthenium source term would reduce each of the risk measures by about a factor of 10, to values well below the RG 1.174 guidelines and the QHOs. The risk impact will decrease even further in later years due to reduced consequences as fission products decay.

The study concludes that relaxation of certain EP requirements can be considered for decommissioning plants in which conformance with the IDCs and SDAs provides reasonable assurance that sufficient margins to the safety goals will be maintained.

4.1.4 Implementation and Monitoring Program

RG 1.174 states that the impact of the proposed change should be monitored using performance measurement strategies. RG 1.174 further states that an implementation and monitoring plan should be developed to ensure that the engineering evaluation conducted to examine the impact of the proposed changes continues to reflect the actual reliability and availability of SSCs that have been evaluated. This will ensure that the conclusions that have been drawn will remain valid.

Applying this guideline for the SFP risk evaluation results in identification of four primary areas for performance monitoring: (1) the performance and reliability of SFP cooling and associated power and inventory makeup systems, (2) the Boraflex condition for high-density fuel racks, (3) crane operation and load path control for cask movements, and (4) onsite emergency response capabilities. The following monitoring should continue after decommissioning in order to assure SFP risk remains low:

- Performance and reliability monitoring of the SFP systems, heat removal, AC power, and inventory should comply with the provisions of the Maintenance Rule (10 CFR 50.65).

- The current monitoring programs identified in licensee's responses to Generic Letter 96-04 (Ref. 2) with respect to monitoring of the Boraflex absorber material should be maintained by decommissioning plants until all fuel is removed from the SFP. This staff assumption is stated in SDA #7 (see Table 4.1-2).

- Heavy load activities and load paths should be monitored and controlled by the licensee in accordance with IDC #1 (see Table 4.1-1).

- Licensees should continue to provide a level of onsite capabilities to assure prompt notification of offsite authorities, characterization of potential releases, development of protective action recommendations, and communication with the public. These capabilities should be monitored by holding periodic onsite exercises and drills.

The staff concludes that continued compliance with the Maintenance Rule, the IDCs, and the SDAs, together with some level of EP, provides a reasonable level of monitoring of SFP safety.

Table 4.1-1 Industry Decommissioning Commitments (IDCs)

IDC No.	Industry commitments
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 onsite and offsite resources can be brought to bear during an event.
3	Procedures will be in place to establish communication between onsite and offsite organizations during severe weather and seismic events.
4	An offsite resource plan will be developed which will include access to portable pumps and emergency power to supplement onsite resources. The plan would principally identify organizations or suppliers where offsite resources could be obtained in a timely manner.
5	SFP instrumentation will include readouts and alarms in the control room (or where personnel are stationed) for SFP temperature, water level, and area radiation levels.
6	SFP seals that could cause leakage leading to fuel uncovery in the event of seal failure shall be self limiting to leakage or otherwise engineered so that drainage cannot occur.
7	Procedures or administrative controls to reduce the likelihood of rapid draindown events will include (1) prohibitions on the use of pumps that lack adequate siphon protection or (2) controls for pump suction and discharge points. The functionality of anti-siphon devices will be periodically verified.
8	An onsite restoration plan will be in place to provide repair of the SFP cooling systems or to provide access for makeup water to the SFP. The plan will provide for remote alignment of the makeup source to the SFP without requiring entry to the refuel floor.
9	Procedures will be in place to control SFP operations that have the potential to rapidly decrease SFP inventory. These administrative controls may require additional operations or 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.

Table 4.1-2 Staff Decommissioning Assumptions (SDAs)

SDA No.	Staff Assumptions
1	Licensee's SFP cooling design will be at least as capable as that assumed in the risk assessment, including instrumentation. Licensees will have at least one motor-driven and one diesel-driven fire pump capable of delivering inventory to the SFP.
2	Walk-downs of SFP systems will be performed at least once per shift by the operators. Procedures will be developed for and employed by the operators to provide guidance on the capability and availability of onsite and offsite inventory makeup sources and time available to initiate these sources for various loss of cooling or inventory events.
3	Control room instrumentation that monitors SFP temperature and water level will directly measure the parameters involved. Level instrumentation will provide alarms at levels associated with calling in offsite resources and with declaring a general emergency.
4	Licensee determines that there are no drain paths in the SFP that could lower the pool level (by draining, suction, or pumping) more than 15 feet below the normal pool operating level and that licensee must initiate recovery using offsite sources.
5	Load Drop consequence analyses will be performed for facilities with non-single failure-proof systems. The analyses and any mitigative actions necessary to preclude catastrophic damage to the SFP that would lead to a rapid pool draining would be sufficient to demonstrate that there is high confidence in the facilities ability to withstand a heavy load drop.
6	Each decommissioning plant will successfully complete the seismic checklist provided in Appendix2B to this study. If the checklist cannot be successfully completed, the decommissioning plant will perform a plant specific seismic risk assessment of the SFP and demonstrate that SFP seismically induced structural failure and rapid loss of inventory is less than the generic bounding estimates provided in this study ($<1x10^{-5}$ per year including non-seismic events).
7	Licensees will maintain a program to provide surveillance and monitoring of Boraflex in high-density spent fuel racks until such time as spent fuel is no longer stored in these high-density racks.

4.2 Implications for Regulatory Requirements for 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 on specific regulatory requirements. These decisions could be made in response to plant-specific exemption requests or as part of the integrated rulemaking for decommissioning plants. Such decisions can be facilitated by a risk-informed examination of both the deterministic and probabilistic aspects of decommissioning. Three examples of such regulatory decisions are presented in this section: regulatory requirements for emergency preparedness, security, and insurance.

4.2.1 Emergency Preparedness

The requirements for emergency preparedness are contained in 10 CFR 50.47 (Ref. 5) and Appendix E to 10 CFR Part 50 (Ref. 6). Further guidance on the basis for EP requirements is contained in NUREG-0396 (Ref. 7) and NUREG-0654/FEMA-REP-1 (Ref. 8). The task force of NRC and Environmental Protection Agency (EPA) representatives formed to address the planning bases for emergency preparedness concluded that the overall objective of EP is to provide dose savings (and in some cases immediate life saving) for a spectrum of accidents that could produce offsite doses in excess of predetermined protective action guides (PAGs).

In the past, the NRC staff has typically granted exemptions from offsite 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, there were no decommissioning plant scenarios for which the consequences justify the imposition of an offsite EP requirement. The results of this technical study confirm that the frequency of events leading to SFP fires is very low (ranging from about $4x10^{-7}$ at sites where seismic events are a minimal contributor to less than $1x10^{-5}$ per year at sites where seismic events dominate SFP risk and no plant-specific seismic analyses need to be performed), and that the subset of events in which EP can produce significant dose savings is even smaller (about $2x10^{-7}$ per year). However, the staff concludes that the possibility an SFP accident will lead to a large fission product release cannot be ruled out even many years after final shutdown, since several SFP accidents could involve either blockage of the air cooling path (e.g., due to partial draining of the SFP) or inadequate air circulation within the SFP building, resulting in near-adiabatic heatup of the spent fuel. The impact of this new information on previously granted exemptions is being evaluated by the staff. Large seismic events that fail the SFP are the dominant contributor to these failure modes. Emergency planning would be of marginal benefit in reducing the risk of such events due to its impairment by offsite damage. The next largest contributor, cask drop accidents, is about an order of magnitude lower in frequency. In the first few years following final shutdown (when time to fission product release is less than about 10 hours), EP could provide some dose savings, but does not substantially impact risk due to the low frequency of these events. Finally, although large releases from the SFP would remain possible for these failure modes, the time available before release would be in excess of 24 hours 5 years after final shutdown and sufficient to support implementation of protective measures on an ad hoc basis.

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 so that the evolution of a zirconium fire accident would evolve slowly enough to allow mitigative measures and, if necessary, offsite protective actions to be implemented without preplanning. The staff believes that the technical analysis discussed in Section 3 and the decision criteria laid out in Section 4.1 provides information on how such exemption requests could be viewed in the future. In addition, this information bears on the need for, and the extent of, emergency preparedness requirements in the integrated rulemaking. In consideration of the study's conclusion that air cooling may not always be available for some event sequences, the basis for some previous exemptions may need to be reconsidered.

The consequence analysis presented in Appendix 4 indicates that the offsite consequences of a zirconium fire may be comparable to those from operating reactor postulated severe accidents. Further, the analysis indicates that timely evacuation, implemented through either pre-planned or ad hoc measures, can significantly reduce the number of early fatalities due to a zirconium fire. The results in Section 3.7.3 indicate that early fatality and societal risk for an SFP fire may be comparable to that for an operating reactor, and that the risk is one to two orders of magnitude lower than the Commission's safety goal. The results in Appendix 4D show that even with the most optimistic assumptions about the effectiveness of EP in large seismic events, the increase in risk associated with relaxations in EP requirements is small and the QHOs continue to be met. Thus, the risk assessment provides some basis for reductions in EP requirements for decommissioning plants. With respect to the potential for pool criticality, the staff's assessment discussed in Section 3 and Appendix 3 indicates that credible scenarios for criticality are highly unlikely and are further precluded by the assumption of Boraflex monitoring programs. Additionally, even if a criticality event did occur, it would not have offsite consequences. Therefore, the conclusions regarding possible reductions in EP program requirements are not affected.

In Section 4.1, the safety principles of RG 1.174 are applied to assess whether changes to emergency preparedness requirements are appropriate. Notwithstanding the low risk associated with SFP accidents, the safety principles in RG 1.174 dictate that defense-in-depth be considered. As discussed previously, emergency preparedness provides defense-in-depth. However, because of the considerable time available to initiate protective actions, in most SFP accidents (and the low frequency of events in which sufficient time is not available to implement protective actions on an ad hoc basis), the level of formal emergency plans needed for rapid initiation and implementation of offsite protective actions can be evaluated. The principle emergency planning measures needed for SFP events is the means for identifying the event and notifying of State and local emergency response officials.

4.2.2 Security

Currently licensees that have permanently shut down reactor operations and have off-loaded the spent fuel into the SFP are still required to meet all the security requirements for operating reactors in 10 CFR 73.55 (Ref. 9). This level of security requires a site with a permanently shutdown reactor to provide security protection at the same level as for an operating reactor site. The industry has asked the NRC to consider whether the risk of radiological release from decommissioning plants due to sabotage is low enough to justify modification of safeguards requirements for SFPs at decommissioning plants.

In the past, decommissioning licensees have requested exemptions from specific regulations in 10 CFR 73.55 on the basis of the reduced number of target sets susceptible to sabotage attacks and the consequent reduced hazard to public health and safety. Limited exemptions have been granted on this basis. The risk analysis in this study does not refute the reduced target set argument; however, the analysis does not support the assertion of a lesser hazard to public health and safety, given the possible consequences of sabotage-induced uncovery of fuel in the SFP when a zirconium fire potential exists. Further, the risk analysis in this study did not evaluate the potential consequences of a sabotage event that could directly cause offsite fission product dispersion, for example, a vehicle bomb driven into or otherwise significantly damaging the SFP, even after a zirconium fire was no longer possible. However, this study supports 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.

As a result of the conclusions from this study, the bases for previous exemptions for defueled facilities, the devitalization of the spent fuel pool at operating reactors, and certain concerns at ISFSIs may need to be reconsidered. This is due to differences in the findings relative to the specific periods of time historically used for the devitalization of spent fuel pools at operating reactors and certain operational concerns and potential vulnerabilities at decommissioning sites.

The risk estimates contained in this study 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 designed and structured to prevent sabotage on the assumption that the design-basis threat could occur at commercial nuclear power plants without assessing the actual probability or consequences.

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

4.2.3 Insurance

In accordance with 10 CFR Part 140 (Ref. 10), each 10 CFR Part 50 licensee is required to maintain public liability coverage in the form of primary and secondary financial protection. This coverage is required to be in place from the time unirradiated fuel is brought onto the facility site until all of the radioactive material has been removed form the site, unless the Commission terminates the Part 50 license or otherwise modifies the financial protection requirements under Part 140. On March 17, 1999, the staff proposed to the Commission that insurance indemnity requirements for permanently shutdown reactors be developed in an integrated, risk-informed effort along with emergency preparedness and security requirements. In the past, licensees have been granted exemptions from financial protection requirements on the basis of deterministic analyses that indicate that a zirconium fire could no longer occur.

In the staff requirements memorandum (SRM) for SECY-93-127 (Ref. 11), the Commission suggested that withdrawal of secondary financial protection insurance coverage be allowed after the requisite minimum spent fuel cooling period had elapsed. Further, the Commission directed the staff to determine more precisely the appropriate spent fuel cooling period after plant shutdown. While insurance does not lend itself to a "small change in risk" analysis because insurance affects neither the probability nor the consequences of an event, the NRC staff has considered whether the risk analysis in this study justifies relief from this requirement for a decommissioning plant while it is vulnerable to zirconium fires. The risk analysis in the February 2000 study identified a generic window of vulnerability for an SFP fire until about 5 years after shutdown. The analysis in this study, however, indicates that a zirconium fire cannot be precluded on a generic basis even after 5 years decay. This is because a spent fuel configuration necessary to assure air cooling cannot be assured following a severe earthquake or cask drop event that drains the pool. Since a criteria of "sufficient cooling to preclude a fire" cannot be met and the long-term consequences could be significant (e.g., the long-term consequences (and risk) decrease very slowly because cesium-137 has a half life of approximately 30 years), the staff will need to consider alternative criteria if changes to insurance requirements are to be pursued.

5.0 SUMMARY AND CONCLUSIONS

This study documents an evaluation of spent fuel pool (SFP) accident risk at decommissioning plants. The study was undertaken to develop a risk-informed technical basis for reviewing exemption requests and a regulatory framework for integrated rulemaking. The staff tried to actively involve the public and industry representatives throughout the study. The staff held a series of public meetings with stakeholders during and after the preparation of a preliminary study (published in June 1999 at the request of the Nuclear Energy Institute (NEI)).

The staff published a draft study in February 2000 for public comment and significant comments were received from the public and the Advisory Committee on Reactor Safeguards (ACRS). To address these comments the staff did further analyses and also added sensitivity studies on evacuation timing to assess the risk significance of relaxed offsite emergency preparedness requirements during decommissioning. The staff based its sensitivity assessment on the guidance in Regulatory Guide (RG) 1.174, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis." The staff's analyses and conclusions apply to decommissioning facilities with SFPs that meet the design and operational characteristics assumed in the risk analysis. These characteristics are identified in the study as industry decommissioning commitments (IDCs) and staff decommissioning assumptions (SDAs). Provisions for confirmation of these characteristics would need to be an integral part of rulemaking.

The results of the study indicate that the risk at SFPs is low and well within the Commission's Quantitative Health Objectives (QHOs). The risk is low because of the very low likelihood of a zirconium fire even though the consequences from a zirconium fire could be serious. Because of the importance of seismic events in the analysis, and the considerable uncertainty in seismic hazard estimates, the results are presented for both the Lawrence Livermore National Laboratory (LLNL) and the Electric Power Research Institute (EPRI) seismic hazard estimates. In addition, to address a concern raised by the ACRS, the results also include a sensitivity to a large ruthenium and fuel fines release fraction. The results indicate that the risk is well below the QHOs for both the individual risk of early fatality and the individual risk of latent cancer fatality.

The study includes use of a pool performance guideline (PPG) as an indicator of low risk at decommissioning facilities. The recommended PPG value for events leading to uncovery of the spent fuel was based on similarities in the consequences from a SFP zirconium fire to the consequences from a large early release event at an operating reactor. A value equal to the large early release frequency (LERF) criterion (1×10^{-5} per year) was recommended for the PPG. By maintaining the frequency of events leading to uncovery of the spent fuel at decommissioning facilities below the PPG, the risk from zirconium fires will be low and consistent with the guidance in RG 1.174 for allowing changes to the plant licensing basis that slightly increase risk. With one exception (the H.B. Robinson site) all Central and Eastern sites which implement the IDCs and SDAs would be expected to meet the PPG regardless of whether LLNL or EPRI seismic hazard estimates are assumed. The Robinson site would satisfy the PPG if the EPRI hazard estimate is applied but not if the LLNL hazard is used. Therefore, Western sites and Robinson would need to be considered on a site-specific basis because of important differences in seismically induced failure potential of the SFPs.

The appropriateness of the PPG was questioned by the ACRS in view of potential effects of the fission product ruthenium, the release of fuel fines, and the effects of revised plume parameters. The staff added sensitivity studies to its analyses to examine these issues. The consequences of a significant release of ruthenium and fuel fines were found to be notable, but not so important as to render inappropriate the staff's proposed PPG of $1x10^{-5}$ per year. The plume parameter sensitivities were found to be of lesser significance.

In its thermal-hydraulic analysis the staff concluded that it was not feasible, without numerous constraints, to establish a generic decay heat level (and therefore a decay time) beyond which a zirconium fire is physically impossible. Heat removal is very sensitive to these additional constraints, which involve factors such as fuel assembly geometry and SFP rack configuration. However, fuel assembly geometry and rack configuration are plant specific, and both are subject to unpredictable changes after an earthquake or cask drop that drains the pool. Therefore, since a non-negligible decay heat source lasts many years and since configurations ensuring sufficient air flow for cooling cannot be assured, the possibility of reaching the zirconium ignition temperature cannot be precluded on a generic basis.

The staff found that the event sequences important to risk at decommissioning plants are limited to large earthquakes and cask drop events. For emergency planning (EP) assessments this is an important difference relative to operating plants where typically a large number of different sequences make significant contributions to risk. Relaxation of offsite EP a few months after shutdown resulted in only a "small change" in risk, consistent with the guidance of RG 1.174. The change in risk due to relaxation of offsite EP is small because the overall risk is low, and because even under current EP requirements, EP was judged to have marginal impact on evacuation effectiveness in the severe earthquakes that dominate SFP risk. All other sequences including cask drops (for which emergency planning is expected to be more effective) are too low in likelihood to have a significant impact on risk. For comparison, at operating reactors additional risk-significant accidents for which EP is expected to provide dose savings are on the order of $1x10^{-5}$ per year, while for decommissioning facilities, the largest contributor for which EP would provide dose savings is about two orders of magnitude lower (cask drop sequence at $2x10^{-7}$ per year). Other policy considerations beyond the scope of this technical study will need to be considered for EP requirement revisions and previous exemptions because a criteria of sufficient cooling to preclude a fire cannot be satisfied on a generic basis.

Insurance does not lend itself to a "small change in risk" analysis because insurance affects neither the probability nor the consequences of an event. The study found that as long as a zirconium fire is possible, the long-term consequences of an SFP fire may be significant. These long-term consequences (and risk) decrease very slowly because cesium-137 has a half life of approximately 30 years. The thermal-hydraulic analysis indicates that when air flow has been restricted, such as might occur after a cask drop or major earthquake, the possibility of a fire lasts many years and a criterion of "sufficient cooling to preclude a fire" can not be defined on a generic basis. Other policy considerations beyond the scope of this technical study will therefore need to be considered for insurance requirements.

The study also discusses implications for security provisions at decommissioning plants. For security, risk insights can be used to determine what targets are important to protect against sabotage. However, any revisions in security provisions should be constrained by an effectiveness assessment of the safeguards provisions against a design-basis threat. Because the possibility of a zirconium fire leading to a large fission product release cannot be ruled out even many years after final shutdown, the safeguards provisions at decommissioning plants should undergo further review. The results of this study may have implications on previous exemptions at decommissioning sites, devitalization of spent fuel pools at operating reactors and related regulatory activities.

The staff's risk analyses were complicated by a lack of data on severe-earthquake return frequencies, source term generation in an air environment, and SFP design variability. Although the staff believes that decommissioning rulemaking can proceed on the basis of the current assessment, more research may be useful to reduce uncertainties and to provide insights on operating reactor safety. In particular, the staff believes that research may be useful on source term generation in air, which could also be important to the risk of accidents at operating reactors during shutdowns, when the reactor coolant system and the primary containment may both be open.

In summary, the study finds that:

1. The risk at decommissioning plants is low and well within the Commission's safety goals. The risk is low because of the very low likelihood of a zirconium fire even though the consequences from a zirconium fire could be serious.

2. The overall low risk in conjunction with important differences in dominant sequences relative to operating reactors, results in a small change in risk at decommissioning plants if offsite emergency planning is relaxed. The change is consistent with staff guidelines for small increases in risk.

3. Insurance, security, and EP requirement revisions need to be considered in light of other policy considerations because a criterion of "sufficient cooling to preclude a fire" cannot be satisfied on a generic basis.

4. Research on source term generation in an air environment would be useful for reducing uncertainties.

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7.0 ACRONYMS

ACRS	Advisory Committee on Reactor Safeguards
ANSI	American National Standard Institute
ANS	American Nuclear Society
ASB	NRC Auxiliary Systems Branch (Plant Systems Branch)
atm	atmosphere
BNL	Brookhaven National Laboratory
BTP	branch technical position
BWR	boiling-water reactor
CFD	computational fluid dynamics
CFM	cubic feet per minute
CFR	*U.S. Code of Federal Regulations*
DOE	Department of Energy
DSP	decommissioning status plant
ECCS	emergency core cooling system
EOF	emergency operations facility
EP	emergency plan
EPRI	Electric Power Research Institute
EPZ	emergency planning zone
ET	event tree
FFU	frequency of fuel uncovery
FT	fault tree
gpm	gallon(s) per minute
GI	generic issue
GWD	gigawatt-day
HCLPF	high confidence in low probability of failure
HRA	human reliability analysis
HVAC	heating, ventilation, and air conditioning
IDC	industry decommissioning commitment
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	metric ton uranium
NEI	Nuclear Energy Institute
NRC	Nuclear Regulatory Commission
NRR	NRC Office of Nuclear Reactor Regulation
OSC	operations support center
POE	probability of exceedance
POF	probability of failure
PPG	pool performance guideline
PRA	probabilistic risk assessment
PWR	pressurized-water reactor
QA	quality assurance
QHO	quantitative health objective
RES	NRC Office of Regulatory Research
RG	regulatory guide
SDA	staff decommissioning assumption
SF	spent fuel
SFP	spent fuel pool
SFPC	spent fuel pool cooling system
SFPCC	spent fuel pool cooling and cleaning system
SHARP	Spent Fuel Heatup Analytical Response Program
SNL	Sandia National Laboratory
SRM	staff requirements memorandum
SRP	standard review plan
SSC	systems, structures, and components
SSE	safe shutdown earthquake
TS	technical specification
TSC	technical support center
UKAEA	United Kingdom Atomic Energy Authority
WIPP	waste isolation pilot plant