1. Background

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

Decommissioning plants have requested exemptions to certain regulations as a result of their permanently defueled condition. Areas where regulatory relief has been requested in the past include exemptions from emergency planning (EP) requirements, Price Anderson Insurance provisions and physical security. Requests for consideration of changes in regulatory and onsite property damage requirements are appropriate since the traditional accident sequences that dominate operating reactor risk are no longer applicable. For a defueled reactor in decommissioning status, public risk is predominantly from accidents involving spent fuel. These fuel assemblies can be stored in the spent fuel pool for considerable periods of time, as remaining portions of the plant continue through decommissioning and disassembly. To date, exemptions have been requested and granted on a plant specific basis. This has resulted in some lack of consistency and uniformity in the scope of evaluations conducted and acceptance criteria applied in processing the exemption requests.

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

In support of this objective, the NRC staff has completed a draft assessment of spent fuel pool risks. This assessment utilized probabilistic risk assessment (PRA) methods (applying both quantitative and qualitative insights) and was developed from detailed analytical studies in the areas of thermal hydraulics, core physics, systems analysis, seismic and structural analysis and external hazards assessment. The focus of the risk assessment was to identify the scenarios, likelihoods and consequences that could result in loss of spent fuel pool water inventory and cooling of the spent fuel assemblies. For some period after reactor shutdown, it is possible for the fuel to heat up to the point where rapid oxidation and burning of the fuel cladding occurs leading to significant releases of radio nuclides.

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

The conclusions and findings of the study provide guidance for the design and operation of spent fuel pool cooling and inventory systems as well as practices and procedures necessary to ensure high levels of operator performance during off normal conditions. The report concludes

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that with the imposition of voluntary industry initiatives and some additional staff requirements in the areas of performance monitoring and seismic validation, the risks from spent fuel pools will be sufficiently small, to justify exemptions from selected current regulatory requirements and to form the basis for related rulemaking.

This final draft report is being issued for stakeholder review and comment.

2.0 Risk-Informed Decision Making

The regulatory framework developed for decommissioning plants is based on a risk informed process. In 1995, the NRC published its PRA policy statement **[Ref. 1]**, which stated that the use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art of the methods. Probabilistic risk assessment provides a structured analytical method to assess the various combinations of failures and events that result in undesirable consequences, for example such as core damage in an operating reactor. Related aspects of these methods can go on to assess the timing and mode of containment failure, radioactive releases to the environment and postulated health effects.

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

For decommissioning plants, the risk is due primarily to the possibility of a zirconium fire associated with the spent fuel rod cladding. Chapter 3 provides a more complete discussion of fuel pool risk scenarios. The consequences of such an event do not equate exactly to either a core damage accident or a large early release. RG 1.174 describes LERF as the frequency of unmittaged releases that have the potential for early health effects, in a time frame prior to effective evacuation of close-in population. Zirconium fires in spent fuel pools potentially have more severe consequences than a operating reactor core damage accident, because there are multiple cores involved, and because there is no containment surrounding the SFP to mitigate the consequences. On the other hand, they are somewhat different than a large early release, because the accidents progress slowly enough to allow ample warning for offsite protective actions, and because the absence of Iodine isotopes leads to ewer prompt fatalities. As a result, the criteria of RG 1.174 cannot be applied directly to the risk of a decommissioning plant without further thought.

Even though the event progresses more slowly than an operating reactor LERF and the isotopic makeup is somewhat different, the risk assessment consequence calculations, shown in Appendix 4, performed by the staff show that large inventories of radioisotopes could be released that could have significant late health effects (latent cancers) for the population at some distance from the plant, as well as the potential for a small number of early health effects (fatalities). The staff has therefore decided that the end state and consequences of a spent

fuel pool fire are sufficiently severe that the RG 1.174 LERF baseline criteria of 1E-5 per year or a change not to exceed 1E-6 per year provide appropriate frequency criteria for a decommissioning plant SFP risk, and a useful tool to assess features, systems and operator performance needs of a decommissioning pool.

2.1 Five Principles of RG 1.174

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

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

- The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change, i.e., a "specific exemption" under 10 CFR 50.12 or a "petition for rulemaking" under 10 CFR 2.802.
- The proposed change is consistent with the defense-in-depth philosophy.
- The proposed change maintains sufficient safety margins.
- When proposed changes result in an increase in core damage frequency and/or risk, the increases should be small and consistent with the intent of the Commission's Safety Goal Policy Statement
- The impact of the proposed change should be monitored using performance measurement strategies."

While the focus on Reg Guide 1.174 was decision-making regarding changes to the licensing basis of an operating plant, the same risk-informed philosophy can be applied to rulemaking for decommissioning plants or to consider potential exemptions to current requirements The intent and scope of these safety principles are discussed below. A discussion on how these principles are satisfied as demonstrated by the staff's safety assessment is provided in Chapter 4.

2.1.1 Regulations [Provide input]

2.1.2 Defense-in-Depth

The defense-in-depth philosophy applies to the operation of the spent fuel pool, whether at an operating plant or in a decommissioning plant. However, implementation is different from that applied to nuclear reactors because of the different nature of the hazards. Because the essentially quiescent (low temperature, low pressure) initial state of the spent fuel pool and the long time for taking corrective action associated with most release scenarios provide significant safety margin, a containment structure is not considered necessary as an additional barrier to provide an adequate level of protection to the public.

2.1.3 Safety Margins

The safety margins associated with fuel in the spent fuel pool are much greater than those associated with an operating reactor. The spent fuel pool is in a quiescent state, at or near ambient temperature and pressure. The decay heat levels are much lower than those of the fuel in an operating reactor. This allows much greater time for heating and boil off of the coolant water, and for heat up of the fuel itself, once uncovered. The fuel is covered with approximately 28 feet of water at near ambient temperature. The pool is designed with ample margin to criticality, using both passive (geometry) and active (poisons) means of reactivity control.

2.1.4 Impact of Proposed Changes

The impact of the proposed change should be small. As discussed above, the staff is applying the baseline and change criteria for LERF in RG 1.174 to assess the impact and acceptability of SFP risk in decommissioning plants.

2.1.5 Implementation and Monitoring Program

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

Therefore, with respect to all the above safety principles, implementation and monitoring of important considerations might include fuel pool seismic design; control of heavy load movements; procedures and other provisions to ensure human reliability; capability, reliability and availability of safety systems; and effectiveness of onsite emergency response, and the plans for communication with offsite authorities. Examples of such features are contained in the NEI commitment letter dated November 12, 1999 [Ref. 3] and are listed in Appendix 6 for convenience. These features were considered by the staff in its risk evaluation.

In Chapter 3, the staff presents the risk assessment conducted on the SFPs for decommissioning plants. In Chapter 4 of this report, the findings of SFP risk for a decommissioning plant will be assessed against each of the safety principles and objectives discussed above.

3.0 Risk Assessment of Spent Fuel Pools at Decommissioning Plants

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

Depending on the time since reactor shutdown and fuel rack configurations, there may be

sufficient heat to cause the clad to heat up, swell and burst. The breach in the clad could result in the release of radioactive gases present in the gap between the fuel and clad, called 'a gap release" If the fuel continues to heat up, the temperature of the zirconium clad will reach the point of rapid oxidation in air. This reaction of zirconium and air is exothermic. The energy released from the reactor combined with the decay energy can cause the reaction to become self-sustaining and lead to the ignition of the zirconium, or a "zirconium fire." The increase in heat from the oxidation reaction could also raise the temperature in adjacent fuel assemblies and cause the propagation of the oxidation reaction. This zirconium fire will result in a significant release of the fission products contained in the spent fuel, which will be dispersed from the reactor site due to the thermal plume from the zirconium fire. Consequence assessments, shown in Appendix 4, show that such a zirconium fire could have significant latent health effects (cancers) as well as the possibility of a small number of early fatalities. Gap releases for fuel of this age in and by themselves (without zirconium fire) release only small quantities of radionuclides and would only be of concern for onsite effects.

Based upon the preceding insights the staff conducted its risk evaluation to focus on the likelihood of scenarios that could result in loss of pool water and fuel heat up to the point of rapid oxidation. Since the decay time at which air cooling alone is sufficient to prevent zirconium fire is very plant specific, the cut off time (when a zirconium fire can no longer occur) for this risk assessment cannot be pre-determined. Rather, the insights should be considered as generally applicable to a decommissioning plant until it reaches a point where rapid oxidation will not occur with complete loss of water. After a decay period that precludes fuel heat up to zirconium fire conditions, no significant risk remains. Preliminary calculations found in Appendix 1, show this time will vary depending on fuel burn up, SFP storage configuration and loading pattern of the assemblies, and could occur at a period as long as five years from plant shutdown.

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

For these calculations, the end state assumed for the accident sequences was when the water level reached the top of the fuel assemblies, rather than calculating the temperature response of the fuel as the level gradually drops. This simplification was utilized because of the extremely complex heat transfer mechanisms and chemical reactions occurring in the fuel assemblies. This analytical approach understates the time that is available for possible operator recovery of SFP events prior to initiation of zirconium fire. However, since the recoverable events such as small loss of inventory or loss of power/pool cooling, are very slowly evolving events, many days are generally available for recovery whether top of fuel uncovery is the end point of the analysis, or is total fuel uncovery. The extra time available (estimated to be in the tens of hours) as the water level boils down the assemblies, would not impact the very high probabilities of operator recovery from these events given the industry commitments and dated November 12, 1999, **[Ref.1]** the Advisory Committee on Reactor Safeguards (ACRS)

recommended that the end state of top of fuel uncovered be used for the SFP analysis along with application of the LERF criteria discussed in Chapter 2. The staff agrees with this recommendation. The details of the staff thermal hydraulic assessment are provided in Appendix 1.

Previous to the staff's preliminary risk assessment, the most extensive work to date was in support of Generic Safety Issue (GSI) 82 **[Ref. 2]**. This report assessed the risk for operating reactors and concluded that a seismic event was the dominant initiating event for the loss of inventory.

While the staff drew from the GSI 82 work in its assessment, it was concluded that because of significant differences between operating and decommissioning plant spent fuel pools cooling systems, a complete assessment of SFP risk should be conducted, considering all potentially significant initiators, and reflecting the unique features found in a shutdown facility. The results of the staff assessments are discussed below. A summary of industry commitments, staff recommendations (relied upon in the risk assessment), and a discussion of how the decision criteria in Chapter 2 are satisfied, is discussed in Chapter 4. Conclusions on how the SFP risk insights and decision criteria apply to potential changes in emergency planning, insurance, and physical security are also discussed in Chapter 4.

3.1 Basis and Findings of SFP Risk Assessment

In order to develop the framework for regulatory decision process described in chapter 1, a comprehensive assessment of SFP risk was desirable. To gather information on SFP design and operational characteristics for the preliminary risk assessment done for the June 1999 report, the staff conducted site visits to four decommissioning plants to ascertain what would be an appropriate model for decommissioning spent fuel pools. The site visits confirmed that the as operated spent fuel pool cooling systems were very different than those in operation when the plants were operating reactors. Modeling information was determined from both site system walkdowns as well as limited discussions with the decommissioning plant staff. Since limited information was available for the preliminary assessment on procedural and recovery activities as well as what the minimum configuration a decommissioning plant might have, a number of assumptions and bounding conditions were assumed for the June 1999 preliminary study. These preliminary results have been refined in this draft assessment after obtaining improved information from industry on SFP design and operating characteristics for a decommissioning plant, as well as a number of commitments that contribute to achieving low risk findings from SFP incidents.

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

- Loss of Offsite Power-Plant centered and grid related events
- Loss of Offsite Power- events initiated by severe weather
- Internal Fire
- Loss of Pool Cooling
- Loss of Coolant Inventory
- Seismic Event
- Cask Drop
- Aircraft Impact

- Tornado Missile

In addition, a qualitative risk perspective was developed for inadvertent re-criticality in the SFP.

The risk model as developed by the staff, and supplemented through an quality review from Idaho National Engineering & Environmental Laboratory (INEEL) is provided in Appendix 2. Subsections of this Appendix include the modeling details for the cask drop, aircraft impacts, seismic and tornado missile assessments. Input and comments from stakeholders was also utilized in updating the June 1999 preliminary model to the present draft model.

3.2 Characteristics of SFP Design and Operations for a Decommissioning Plant

Based upon information gathered from the site visits and interactions with NEI and other stakeholders the staff has modeled the spent fuel pool cooling system (SFPC) **(see Figure 3.0-1)** as being located in the spent fuel pool (SFP) area and consisting of motor-driven pumps, a heat exchanger, an ultimate heat sink, a makeup tank, filtration system and isolation valves. Suction is taken via one of the two pumps on the primary side from the spent fuel pool and is passed through the heat exchanger and returned back to the pool. One of the two pumps on the secondary side rejects the heat to the ultimate heat sink. A small amount of water from the suction line is diverted to the filtration process and is returned back into the discharge line. A manually operated makeup system (limited volumetric flow rate) supplements the small losses due to evaporation. In the case of prolonged loss of SFPC system or loss of inventory events, the inventory in the pool can be made up using the firewater system. There are two firewater pumps, one motor-driven (electric) and one diesel-driven, which provide firewater in the SFP area. A firewater hose station is provided in the SFP area. The firewater pumps are located in a separate structure.

Based upon information obtained during the site visits and discussions with the operating staff's during those visits the staff also made the following assumptions that are believed to be representative of a typical decommissioning facility.

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

- We assume the makeup capacity (with respect to volumetric flow) to be as follows:

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

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

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

Based upon the results of the June 1999 preliminary risk analysis and its associated sensitivity cases, it became clear that many of the risk sequences were quite sensitive to the performance of the SFP operating staff in identifying and responding to off normal conditions. This is due to the fact that the remaining systems in the SFP Island are relatively simple with manual rather than automatic initiation of backups or realignments. Therefore, if scenarios such as loss of cooling or inventory loss to the pool occurs, operator response to diagnose the failures and bring on site and off site resources to bear are instrumental for ensuring that the fuel assemblies remain cooled and a zirconium fire is prevented. As part of its technical evaluations the staff assembled a blue ribbon committee of experts which identified the attributes necessary to achieving very high levels of human reliability for responding to potential accident scenarios in a decommissioning plant SFP (See Appendix 8 HRA Study).

Upon consideration of the sensitivities identified in the staff's preliminary study and to reflect actual operating practices at many decommissioning facility, the nuclear industry, through NEI made important commitments which were reflected in the staff's updated risk assessment. These NEI commitments are shown in Appendix 6. The revisions to the risk assessment generally reflected changes of assumptions in the areas shown below. The specific industry commitments and how they relate to specific risk conclusions and safety principles are discussed in Chapter 4.

- The SFP has instrumentation/indication and alarms in the control room to alert the fuel handler to potential problems in the SFP. These include level indications, temperature indications, and radiation monitors.

- The certified fuel handlers are former reactor operators who are knowledgeable of the facility, the maintenance procedures, and the surrounding community. We assumed that these fuel handlers (through procedures and training) are aware of the available backup sources that can be used to replenish the SFP inventory (i.e., the fire protection pumps, or offsite sources such as from fire engines). We assumed that utility developed a resource plan and made arrangements in advance with fire stations or other offsite resources regarding details such as what is required from the fire department including equipment and tasks.

- Either procedures exist to limit heavy load movement over the spent fuel pool or the overhead cranes have stops to help prevent heavy loads from being moved over the spent fuel pool. Crane operators will follow safe load path procedures when moving heavy loads near or over the spent fuel pool. Phases 1 and 2 of NUREG-0612 are implemented.

- All equipment, including external sources (fire department), are available and in good working order. We assumed that alternative fuel pool makeup system components were routinely tested and administrative controls exist when equipment is out of service.

- A complete set of procedures is available to fuel handlers that is function-oriented and promote early start of actions to restore cooling. Procedures require fuel handlers to seek offsite assistance if level in the spent fuel pool drops below a prescribed point.

- A plan was developed to identify organizations or suppliers of equipment or services potentially needed in the event of problems with the spent fuel pool cooling system.

- There is an onsite restoration plan to provide repair of the SFPC system or provide access to makeup water to the spent fuel pool.

Based upon the above commitments, technical comments from stakeholders and the input from the INEEL technical review, the staff's SFP risk model was updated. The results for the initiators which were assessed quantitatively are shown in Table 3-1.

Table 3-1 Spent Fuel Pool Cooling Risk Analysis Frequency of Fuel Uncovery (per year)

INITIATING EVENT	**Base Case**
Loss of Offsite Power - Plant centered and grid related events	8.2E-08
Loss of Offsite Power - Events initiated by severe weather	1.3E-06
Internal Fire	6.7E-08
Loss of Pool Cooling	5.7E-08
Loss of Coolant Inventory	1.5E-07
Seismic Event	>1.0E-06
Cask Drop	2.2E-07
Aircraft Impact	2.9E-09
Tornado Missile	ε[1]
Total	>2.9E-06

NOTE: GARETH WILL PROVIDE A CHARACTERIZATION OF WHAT RESULTS REPRESENT< MEAN/POINT ESTIMATES..... **Gareth noted: "As far as the words on what the results mean, you can refer to them as point estimates based on mean values of the input paramteres for now. We are still working on the results, and I think that the discussion of the results in Mark's writeup will probably need to change to reflect more how the low values relate to the commitments and the assumptions. I don't see that being ready for tomorrow."**

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

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

3.3 Internal Event Scenarios Leading to Fuel Uncovery

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

[1] Frequency of less than $1x10^{-9}$ per year

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

Frequency of Fuel Uncovery

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

Scenario

Plant-centered events typically involve hardware failures, design deficiencies, human errors (in maintenance and switching), localized weather-induced faults (e.g., lightning), or combinations of these. Grid-related events are those in which problems in the offsite power grid cause the loss of offsite power. With offsite power lost (and therefore onsite power is lost too, since we assume there is no diesel generator available to pick up the necessary electrical loads), there is no effective heat removal process for the spent fuel pool (i.e., until offsite power is recovered, all electrical pumps would be unavailable, and the diesel-driven fire pump only would be available to provide makeup to the spent fuel pool.) If power were not restored quickly enough, the pool will heat up and boil off inventory until the fuel is uncovered (if there were inadequate makeup). If the diesel-driven pump fails, and if offsite power were not recovered in a timely manner, offsite recovery using fire engines is a possibility. With 1-year-old fuel (i.e., the youngest fuel in the fuel pool was shutdown in the reactor one year ago), 127 hours is available for this recovery action.

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

Cutset

There was one important sequence minimum cutset.

Cutset for sequence 5:

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

Sensitivity Study

The sensitivity study conducted for this sequence showed that significant increases could occur if not for good operator training, detailed procedures and appropriate test and maintenance on important equipment.

3.3.2 Loss of Offsite Power from Severe Weather Events

Frequency of Fuel Uncovery

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

Scenario

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

Cutset

There was one important minimum cutset

Cutset for sequence 8:

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

Sensitivity Study

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

3.3.3 Internal Fire

Frequency of Fuel Uncovery

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

Scenario

This event tree models the loss of SFP cooling caused by internal fires. We assumed that there is no automatic fire suppression system for the SFPC area. The fuel handler may initially attempt to recover the damaged SFP cooling system given that he responds to the alarms. If the fuel handler fails to respond the alarm, we assumed that SFPC system will be significantly damaged and cannot be repaired within a few days. Once the inventory level drops below the SFP cooling system suction level, the fuel handlers have about 85 hours to provide some sort of alternate makeup, either using the site firewater system or by calling upon offsite resources. It was assumed that fire damages the plant power supply system such that the power to the electrical firewater pump is lost and would not be available.

Cutset

There were three important sequence minimum cutsets.

Cutsets for sequence 4:

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

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

Cutset for sequence 8:

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

Sensitivity Study

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

3.3.4 Loss of Cooling

Frequency of Fuel Uncovery

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

Scenario

The initiating event frequency includes the loss of coolant system flow from the failure of pumps or valves (See Figure 3.0-1), from piping failures, from an ineffective heat sink (e.g., loss of heat exchangers), or from a local loss of power (e.g., electrical connections.) While it may not be directly applicable due to design differences in a decommissioning plant, operational data from NUREG-1275, Volume 12 **[Ref. 3]** shows that the frequency of loss of spent fuel pool cooling events in which a temperature increase of more than 20°F occurred can be estimated to be on the order of two to three events per 1000 reactor years. The data also showed that, for the majority of events, the duration of the loss of cooling was less than one hour. Only three events exceeded 24 hours, with the maximum duration being 32 hours. There were four events where the temperature increase exceeded 20°F, with the maximum increase being 50°F.

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

decommissioning plants. The following are the two most important minimum cutsets in the base case, Case 1, leading to fuel uncovery caused by loss of cooling:

Cutset

There are two important minimum cutsets.

Cutset for sequence 4:

i) (loss of cooling) x (repair crew fails to repair the spent fuel pool cooling system [early]) x (fuel handlers fail to diagnose need to start fire pumps) = 1.1×10^{-8} per year

Cutset for sequence 9:

i) (loss of cooling) x (failure of fuel handler to respond to control room alarms) x (fuel handlers' failure to notice pool steaming and level drop during walkdowns [dependent case]) = 4.5×10^{-8} per year

Sensitivity Study

For the sensitivity case, significant increases can be observed if the makeup pumps have poor preventive maintenance/testing programs and the fuel handlers do not have reliable emergency procedures.

3.3.5 Loss of Coolant Inventory

Frequency of Fuel Uncovery

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

Scenario

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

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

Cutset

There was one important sequence minimum cutset.

Cutset for sequence 9:

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

Sensitivity Study

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

3.3.6 Heavy Load Drops

We investigated the frequency of dropping a heavy load in or near the spent fuel pool, and investigated potential damage to the pool from such a drop. Details of this evaluation can be found in Appendix 2C. Our analysis exclusively considered drops that were severe enough to catastrophically damage the spent fuel pool such that pool inventory would be lost rapidly and it would be impossible to refill the pool using onsite or offsite resources. In essence there is no possibility for mitigation in such circumstances, only prevention. A catastrophic heavy load drop(that caused a large leakage path in the pool) would lead directly to a zirconium fire approximately 10 to 12 hours after the drop, depending on fuel age, burn up, and configuration. The dose rates in the pool area prior to any zirconium fire would be on the order of tens of thousands of rem per hour, making any potential recovery actions such as temporary large inventory addition systems very difficult. We concluded that non-catastrophic damage to the pool or its support systems from a load drop is captured and bounded by other initiators.

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

For a non-single failure proof load handling system that does not follow NUREG-0612 **[Ref. 4]** guidelines, the likelihood of a heavy load drop (i.e., the drop frequency) was estimated, based on NUREG-0612 information, to be in the range of 2.0×10^{-5} to 1.2×10^{-3} per year, with a mean value of 3.4×10^{-4} per year. The number of heavy load lifts was based on the NEI estimate of 100 spent fuel shipping cask lifts per year, which probably is an overestimate. For a single failure proof load handling system or a plant conforming to the NUREG-0612 guidelines, the frequency range for a heavy load drop is estimated to be 5.5×10^{-6} to 3.4×10^{-5} per year, with a mean value of 9.6×10^{-6} per year, again for 100 heavy load lifts per year but using new data from U.S. Navy crane experience. Once the load is dropped, the next question is whether the drop did significant damage to the spent fuel pool.

When estimating the failure frequency of the pool floor, we assumed that heavy loads physically travel near or over the pool between 5% and 25% of the total path lift length (the path lift length is the distance from the lift of the load to the placement of the load on the pool floor). We also assumed that the critical path length (the fraction of total path the load is lifted high enough above the pool that a drop could cause damage to the structure) is between 10% and 15% of the time the load is near or over the pool. We therefore estimate the catastrophic failure rate from heavy load drops to be in the range of 1.0×10^{-7} to 7.5×10^{-5} per year, with a mean value of 2.1×10^{-5} per year for the non-single failure proof system that does not follow the guidelines of NUREG-0612, and 2.8×10^{-8} to 2.1×10^{-6} per year, with a mean value of 2.0×10^{-7} per year for the single failure proof system or a plant conforming to all NUREG-0612 guidelines.

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

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

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

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

3.4.1 Seismic Events

When beginning our evaluation of the effect of seismic events on spent fuel pools, it became apparent that we do not have detailed information of how all the spent fuel pools were designed and constructed. We originally performed a simplified seismic risk analysis in our June 1999 draft risk assessment to help determine if there might be a seismic concern. The analysis indicated that seismic events could not be dismissed on the basis of a simplified approach. After further evaluation and discussions with stakeholders, we determined that it would not be cost effective to perform a plant-specific seismic evaluation for each spent fuel pool. Working with our stakeholders, we developed other tools that help assure the pools are sufficiently robust.

We believe spent fuel pool structures at nuclear power plants are seismically robust. They are constructed with thick reinforced concrete walls and slabs lined with thin stainless steel liners 1/8 to 1/4 inch thick. (Dresden Unit 1 and Indian Point Unit 1 are two plants that do not have any liner plates. They were decommissioned more than 20 years ago and no safety significant degradation of the concrete pool structure has been reported). Pool walls vary from 4.5 to 5 feet in thickness and the pool floor slabs are around 4 feet thick. The overall pool dimensions

are typically about 50 feet long by 40 feet wide and 55 to 60 feet high. In boiling water reactor (BWR) plants, the pool structures are located in the reactor building at an elevation several stories above the ground. In pressurized water reactor (PWR) plants, the spent fuel pool structures are located outside the containment structure supported on the ground or partially embedded in the ground. The location and supporting arrangement of the pool structures determine their capacity to withstand loads beyond their design basis. The dimensions of the pool structure are generally derived from radiation shielding considerations rather than structural needs. Spent fuel structures at operating nuclear power plants are able to withstand loads substantially beyond those for which they were designed. Consequently, they have significant seismic capacity.

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

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

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

3.4.2 Aircraft

We evaluated the likelihood of an aircraft crashing into a nuclear power plant site and seriously damaging the spent fuel pool or its support systems (details are in Appendix 2D). The generic data provided in DOE-STD-3014-96 **[Ref. 6]**, were used to assess the likelihood of an aircraft

crash into or near a decommissioning spent fuel pool. Aircraft damage can affect the structural integrity of the spent fuel pool or affect the availability of nearby support systems, such as power supplies, heat exchangers, or water makeup sources, and may also affect recovery actions. There are two approaches that can be taken to evaluate the likelihood of an aircraft crash into a structure. The first is called the point target model which uses the area (length times width) of the target to determine the likelihood that an aircraft will strike the target. The aircraft itself does not have real dimensions when using this model. In the second approach, the DOE model modifies the point target approach to account for the wing span and the skidding of the aircraft after it hits the ground by including the additional area the aircraft could cover. Further, that model takes into account the plane's glide path by introducing the height of the structure into the equation, which effectively increase the area of the target (see Appendix 2D).

Our estimate of the frequency of catastrophic PWR spent fuel pool damage (i.e., the pool is so damaged that it rapidly drains and cannot be refilled from either onsite or offsite resources) resulting from a direct hit is based on one estimate using the point target area model for a 100 x 50 foot pool, with a conditional probability of 0.3 (large aircraft penetrating 6-ft of reinforced concrete) that the crash results in catastrophic damage. The point target model was chosen to model a direct hit on the pool. If 1-of-2 aircraft are large and 1-of-2 crashes result in significant damage, then the estimated range of catastrophic damage to the spent fuel pool is 9.6×10^{-12} to 4.3×10^{-8} per year. The mean value is estimated to be 2.9×10^{-9} per year. The frequency of catastrophic BWR spent fuel pool damage resulting from a direct hit by a large aircraft is the same as that for the PWR. Mark-I and Mark-II secondary containments generally do not appear to have any significant structures that might reduce the likelihood of aircraft penetration, although a crash into one of four sides of a BWR secondary containment may have a reduced likelihood of penetration due to other structures being in the way of the aircraft. Mark-III secondary containments may reduce the likelihood of penetration somewhat, as the spent fuel pool may be considered to be protected on one side by additional structures. If instead of a direct hit, the aircraft skidded into the pool or a wing clipped the pool, catastrophic damage may not occur. We project that skidding aircraft will be negligible contributors to the frequency of fuel uncovery resulting from catastrophic failure of the pool. The estimated frequencies of air craft induced catastrophic spent fuel pool failure are bounded by other initiators.

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

year, with the mean estimated to be 7.4×10^{-10} per year. Depending on the model approach (selection of the target structure size; use of the point target model or the DOE model), the mean value for an aircraft damaging a support system is in the 7×10^{-7} per year, or less, range. This is not the estimated frequency of fuel uncovery or a zirconium fire caused by damage to the support systems, since the frequency estimate does not include recovery, either onsite or offsite. As an initiator to failure of a support system leading to fuel uncovery and a zirconium fire, an aircraft crash is bounded by other more probable events. Recovery of the support system will reduce the likelihood of spent fuel uncovery.

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

3.4.3 Tornadoes

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

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

We assumed that an F2 to F5 tornado would be required if significant damage were to occur to spent fuel pool support systems (e.g., power supply, cooling pumps, heat exchanger, or makeup water supply). The frequency of having an F2 to F5 tornado is estimated to be 1.5×10^{-5} per year for the central U.S., with a U.S. average value of 6.1×10^{-6} per year. As an initiator to failure of a support system, the tornado is bounded by other more probable events (see Table 3.1-1).

3.4.4 Recriticality in Spent Fuel Pool

Due to the processes involved and lack of data, it was not possible to perform a quantitative risk assessment for recriticality in the spent fuel pool. In Appendix 3 the staff performed an evaluation of the potential scenarios that could lead to criticality and identified those that are credible.

In this section the staff provides its qualitative assessment of risk due to recriticality in the SFP, and its conclusions that with the additional requirements identified, the potential risk from SFP recriticality is sufficiently small.

In Appendix 3 the staff assessed the various potential scenarios that could result in inadvertent criticality. This assessment identified two scenarios as creditable, which are listed below.

(1) A compression or buckling of the stored assemblies could result in a more optimum geometry (closer spacing) and thus create the potential for criticality (see the NRC staff report "Assessment of the Potential for Criticality in Decommissioned Spent Fuel Pools," in Appendix 7). Compression is not a problem for high-density PWR or BWR racks because they have sufficient fixed neutron absorber plates to mitigate any reactivity increase, nor is it a problem for low-density PWR racks if soluble boron is credited. But compression of a low-density BWR rack could lead to a criticality since BWR racks contain no soluble or solid neutron absorbing material. High-density racks are those that rely on both fixed neutron absorbers and geometry to control reactivity. Low-density racks rely solely upon geometry for reactivity control. In addition, all PWR pools are borated, whereas BWR pools contain no soluble absorbing material. If both PWR and BWR pools were borated, criticality would not be achievable for a compression event.

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

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

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

potential scenarios described above in 1 and 2 involve crushing of fuel assemblies in low density racks or degradation of Boraflex over long periods in time.

To gain qualitative insights on the recriticality events that are credible, the staff considered the sequences of events that must occur. For scenario 1 above this would be require a heavy load drop into the a low density racked BWR pool compressing assemblies. From appendix 2 on heavy load drop, the likelihood of a heavy load drop from a single failure proof crane is approximately 2E-6 per year, assuming 100 cask movements per year at the decommissioning facility. From the load path analysis done for that appendix it was estimated that the load could be over or near the pool between 25% and 5% of the movement path length, dependant on plant specific layout specifics. The additional frequency reduction in the appendix to account for the fraction of time that the heavy load is lifted high enough to damage the pool liner is not applicable here because the fuel assemblies could be crushed without the same impact velocity being required as for the pool liner. Therefore, if we assume 10% load path vulnerability, we observe a potential initiating frequency for crushing of approximately 2E-7 per year (based upon 100 lifts per year). Criticality calculations conducted for appendix 3 show that even if the low density BWR assemblies were crushed by a transfer cask, it is "highly unlikely" that a configuration would be reached that would result in a severe reactivity event, such as a steam explosion which could damage and drain the spent fuel pool. The staff judges the chances of such a criticality event to be well below 1 chance in 100 even given that the transfer cask drops directly onto the assemblies. This would put the significant criticality likelihood well below 1E-8 per year, which justifies its exclusion from further consideration.

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

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

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

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

4.0 Implications of Spent Fuel Pool Risk For Regulatory Requirements

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

The analysis in chapter 3 explicitly examines the risk impact of specific design and operational characteristics. Some of these have been proposed by the Nuclear Energy Institute in a letter to the NRC dated November 12, 1999 [Ref. 1]. Others came to light as a result of the analysis itself. These characteristics are summarized in section 4.1. The NRC intends to make these the principle aspects of the risk-informed approach to oversight of decommissioning plants.

In addition, the industry and other stakeholders have proposed the use of risk-informed decision-making to assess regulatory requirements in three specific areas; namely, emergency preparedness, security and insurance. The technical results of this report might be used either to justify plant-specific exemptions from these requirements, or to determine how these areas will be treated in a risk-informed oversight process. Section 4.3 examines the implications of this technical results for those specific regulatory decisions.

4.1 Summary of the Technical Results

The thermal-hydraulic analysis presented in Appendix 1 demonstrates that the conditions necessary for a zirconium fire exist in spent fuel pools of decommissioning plants for a period of several years following shutdown. The analysis shows that the length of time over which the fuel is vulnerable depends on several factors, including fuel burnup and pool configuration. In some cases analyzed in Appendix ----, the required decay time is ____ years. However, the time period for any specific plant will vary. Plant-specific analysis is needed to justify the use of shorter decay periods.

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

For a decommissioning plant with about one year of decay time, the timing of radiological releases from zirconium fires is significantly slower than those from the most limiting reactor accident scenarios. This is due to the slow heatup time of the fuel. In addition, for many of the sequences leading to zirconium fires, there are very large delay times due to the long time required to boil off the spent fuel pool water inventory. Thus, while the consequences of zirconium fires are in some ways comparable to large early releases from reactor accidents, the timing is much slower.

The annual frequency of events leading to zirconium fires at decommissioning plants is estimated to be _____, for a plant that implements the design and operational characteristics discussed below. This estimate can be much higher for a plant that does not embody these characteristics. The most significant contributor to this risk is a seismic event which exceeds

the design basis earthquake. Other contributors are _____. This overall frequency is within the acceptance guidelines for large early release frequency (LERF) in Regulatory Guide (RG) 1.174 [Ref. 2]. As noted above, zirconium fires are estimated to be similar to large early releases in some ways, but less severe in others.

4.2 Risk Impact of Specific Design and Operational Characteristics

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

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

The staff's risk assessment as discussed in Chapter 3 shows that the baseline risk from a decommissioning spent fuel pool is a frequency for a zirconium fire of approximately $2x10^{-6}$ per year. As was discussed in Chapter 2, the staff has determined that such a fire results in a large radionuclide release and poses a highly undesirable end state for a spent fuel pool accident. Therefore the staff has judged that the RG 1.174 criteria for baseline LERF of $1x10^{-5}$ per year should be applied. The risk assessment shows that the SFP baseline risk is well under the RG 1.174 criteria. In assessing the impact on change in risk, the staff considered a potential relief from EP requirements as the changing requirement.

The staff consequence analysis in Appendix 4 shows that the early health impacts from zirconium fire scenarios are significantly impacted by evacuation. This evacuation will greatly reduce the early fatalities near the plant site. However, this analysis also showed that for the slowly evolving SFP accident sequences, the initiation of effective evacuation can be much delayed in comparison to an operating reactor, where the accident results in high offsite does much more rapidly. Based upon this insight, the staff will require (decommissioning staff requirement (DSR #1)) that a basic evacuation scheme be maintained at the plant. This scheme will include guidance on when offsite evacuation should be initiated, and ensure that current liaisons with offsite emergency organizations be maintained so that an ad hoc evacuation (as is done for transportation emergencies) can be put into place when needed. Since the slower evacuation expected from such an ad hoc effort was still shown to be effective for the SFP fire scenarios, this change from a formal EP program is not expected to have any risk impact.

In addition to DSR #1, the low numerical risk results shown in Chapter 3 and Appendix 2 are derived from a number of design and operational elements of the SFP. As shown in those sections, the dominant risk contribution is from seismic events well beyond the plants original design basis. The baseline seismically initiated zirconium fire frequency from our risk assessment is predicated upon implementation of the seismic checklist shown in Appendix 5. The staff will require that such a checklist (DSR #2) be successfully implemented at all decommissioning facilities prior to relief from any regulatory requirements.

The accident sequences in Chapter 3 associated with loss of cooling or loss of inventory are quantified to result in low risk due to a number of elements that enhance the ability of the

operators to respond successfully to the events with onsite and offsite resources. Without these elements, the probability of the operators detecting and responding to the loss of cooling or inventory would be higher and public risk from these categories of SFP accidents could significantly increased. Some elements were also identified that reduce the likelihood of the loss of cooling or loss of inventory initiators, including both design as well as operational issues. The elements proposed by industry (Decommissioning Industry Commitments (DICs)) are identified below.

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

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

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

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

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

DIC #2 Procedures and training of personnel will be in place to ensure that on site and off site resources can be brought to bear during an event.

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

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

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

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

The staff's risk evaluation also shows that the potential for pool failure due to heavy load drop to be significant if appropriate design and procedural control are not in place. The staff judges that such controls are provided by the decommissioning industry commitments (DICs)

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

4.2.2 The Proposed Change Is Consistent with the Defense-in-depth Philosophy.

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

DSR #2 Each decommissioning plant will develop and maintain a site emergency plan, that contains guidance on when a site emergency should be declared with respect to the possibility of a SFP fire. The plan will also identify off site liaisons with public emergency organizations to put in place ad hoc evacuation so as to have an effective evacuation prior to the postulated zirconium fire. The elements of this plan will be submitted to the staff for approval prior to any relief for full EP being considered.

4.2.3 The Proposed Change Maintains Sufficient Safety Margins.

As discussed in Chapter 2 the safety margins associated with fuel in the spent fuel pool are much greater than those associated with an operating reactor due to the low heat removal requirements and long time frames available for recovery from off normal events. Due to these larger margins the staff judges that the skid mounted and other dedicated SFP cooling and inventory systems in place do provide adequate margins. However, the staff assessment did identify one area where additional margins are of benefit in moderating the risk from potential pool re-criticality. Due to the potential for loss of inventory events that can be recovered by use of alternate water sources, the potential exists for loss of shutdown margins with the addition of unborated water to pools that originally are borated. Additionally for pools that utilize Boraflex absorbers in high density racks, having boron on site for addition to the pool, would allow for quick restoration of shutdown margin if the rack surveillance and monitoring program did identify any significant degradation of the Boraflex. This leads to the following requirement:

DSR #3 All decommissioning plants will retain on site quantities of soluble boran sufficient for maintaining pool shutdown margins in a borated pool which is assumed to have 50% of its water mass replace with unborated water.

Additionally all decommissioning plants that utilize Boraflex absorbers will maintain sufficient soluble boron on site to make up shutdown reactivity margin lost due to degradation of 20% of Boraflex in the high density racks. Procedures will also be developed on the use of this boron for either scenario.

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

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

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

With respect to monitoring of the Boraflex absorber material, the current monitoring programs required by **(STATE REFERENCE FOR REQUIREMENT)** will be maintained by decommissioning plants until all fuel is removed from the SFP. This generates a decommissioning staff requirement (DSR)

DSR #4 Licensees will maintain a program to provide surveillance and monitoring of Boraflex in high density spent fuel racks until such a time as do high density racks are retained in the pool. The SFP licensees will also have procedures in place to assess degradation impact on reactivity shutdown margin and provide additional pool boration as necessary to maintain the needed margins.

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

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

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

4.3.1 Emergency Preparedness

The requirements for emergency preparedness for are contained in 10CFR 50.47 **[Ref. 3]** and Appendix E **[Ref. 4]**. Further guidance on the basis for EP requirements is contained in NUREG-0396 **[Ref. 5]**. The general goal of EP requirements is to prevent early fatalities and to reduce offsite dose from accidents.

In the past, the NRC staff has granted exemptions from emergency planning requirements for decommissioning plants that could demonstrate that they were beyond the period in which a zirconium fire could occur. The rationale for those decisions was that, in the absence of a zirconium fire, a decommissioning plant had no appreciable scenarios for which the consequences justify the imposition of an EP requirement. The results of this technical study confirm that position (**but what about criticality?**).

In some cases, emergency preparedness exemptions have also been granted to plants which were still in the window of vulnerability for zirconium fire. In these cases, the justification was that enough time had elapsed since shutdown that the evolution of a zirconium fire accident would be slow enough to allow effective offsite protective actions on an ad hoc basis, without the need for emergency planning. The staff believes that the technical analysis discussed in chapter 3 and the decision criteria laid out in chapter 1 have direct bearing on how such exemption requests should be viewed in the future. In addition, this information has bearing on the need for, and the extent of, emergency preparedness requirements in the integrated rulemaking.

The consequence analysis presented in appendix __ demonstrates that the offsite consequences of a zirconium are comparable to those from operating reactor severe accidents. Further, the analysis demonstrates that timely evacuation can significantly reduce the number of early fatalities due to a zirconium fire. The thermal-hydraulic analysis presented in appendix __ confirms our earlier conclusion that zirconium fire events evolve slowly, even for initiating events that result in a catastrophic loss of fuel pool coolant. The results in chapter 3 also show that the frequency of zirconium fires is low when compared with the risk guidelines from RG 1.174. Thus the risk associated with early fatalities from these scenarios is low. Based on this combination of low risk and slow evolution, the Commission might decide to reduce or eliminate EP requirements for decommissioning plants.

One important safety principle of RG 1.174 is consistency with the defense in depth philosophy. In the rationalist approach, defense in depth is included in a plant design to account for uncertainties in the analysis or operational data. The spent fuel pools at operating reactors and decommissioning facilities do not exhibit the defense in depth accorded to the reactor. As discussed in chapter 1, this difference is justified in light of the considerably greater margin of safety of the SFP compared with reactors. For SFP at operating reactors, defense in depth consists mainly of the mitigating effect of emergency preparedness. The Commission might consider retaining a baseline level of EP requirements for decommissioning plants as a defense in depth measure. This might be justified in view of the uncertainties associated with the risk analysis presented herein. The staff has not attempted to assess what level of emergency preparedness might be needed to provide this defense in depth. However, given the slow nature of these accidents, we believe it would be substantially lower than what is currently required for operating reactors.

4.3.2 Security

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

In the past, decommissioning licensees have requested exemptions from specific regulations in 10 CFR 73.55, justifying their requests on the basis of a reduction in the number of target sets susceptible to sabotage attacks, and the consequent reduced hazard to public health and safety. Limited exemptions based on these assertions have been granted. The risk analysis in this report does not take exception to the reduced target set argument; however, the analysis does not support the assertion of a lesser hazard to public health and safety, given the consequences that can occur from a sabotage induced draining of the SFP and subsequent zirconium fire. Further, it does not evaluate the potential for a sabotage event that would directly cause off site fission product dispersion, say from a vehicle bomb that was driven into the SFP. However, this report would support a regulatory framework that relieves licensees from selected requirements in 10 CFR 73.55 on the basis of target set reduction when all fuel has been placed in the SFP.

The risk estimates contained in this report are based on accidents initiated by random equipment failures, human errors or external events. PRA practitioners have developed and used dependable methods for estimating the frequency of such random events. By contrast, this analysis, and PRA analyses in general, do not include events due to sabotage. No 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. As it happens, security regulations are based on a zero tolerance for sabotage; in other words, the regulations are designed and structured to prevent any successful sabotage event at a commercial nuclear power plant, regardless of the probability or consequences.

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

The staff noted that 10 CFR 73.51 **[Ref]**, which specifies the security requirements for an ISFSI licensed under 10 CFR 72 **[Ref],** is silent on the vehicle bomb threat. The staff also noted that 10 CFR 26 **[Ref]** - Fitness for Duty Programs, is not being applied to decommissioning reactors once the fuel has been removed from the reactor vessel and placed in the SFP, and specifically does not apply to ISFSIs licensed under 10 CFR 72. Given the importance of a vehicle bomb threat to the integrity of SFP, and the significance of HRA to the conclusions reached in the

SFP risk analysis, the staff recommends that for coherency in the regulations, these subjects be revisited during the overall integration of rules for decommissioning reactors.

4.3.3 Insurance

In accordance with 10 CFR 140 **[Ref. 6]**, each 10 CFR 50 licensee is required to maintain public liability coverage in the form of primary and secondary financial protection. This coverage is required to be in place from the time unirradiated fuel is brought onto the facility site until all the radioactive material has been removed from the site, unless the Commission terminates the Part 50 license or otherwise modifies the financial protection requirements. The industry has asked the NRC to consider whether the likelihood of large scale radiological releases from decommissioning plants is low enough to justify modification of the financial protection requirements once the plant is permanently shutdown and prior to complete removal of all radioactive material from the site.

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

The NRC staff has considered whether the risk analysis in this report justifies relief from this requirement for decommissioning plants during the period when they are vulnerable to zirconium fires. As part of this effort, the staff determined that an analogy can be drawn between a SFP at a decommissioning plant and a wet (as opposed to dry) independent spent fuel storage installation (ISFSI) licensed under 10 CFR 72 **[Ref. 7]** for which no indemnification requirement currently exists. Spent reactor fuel aged for one year can be stored in a wet or dry ISFSI. The risk analysis in this report predicts high consequences for a zirconium fire, and identifies a generic window of vulnerability out to 5 years. In the staff requirements memorandum (SRM) for SECY-93-127 [Ref. 8] the Commission suggested that insurance coverage is required unless a large scale radiological release is deemed incredible. Further, they instructed the staff to determine more precisely the appropriate spent fuel cooling period after plant shut down, and to determine the need for primary financial protection for ISFSIs.

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

The staff knows of no frequency criterion which could be cited to justify reduction or elimination of the insurance requirement while a vulnerability to zirconium fire exists. Defining or applying such a criterion would be inconsistent with Commission direction provided in SECY-93-127.

5.0 References

References for Chapter 1.0
1. 10 CFR Part 50
2.

References for Chapter 2.0
1.01 In 1995, the NRC published its PRA policy statement
1.02 Regulatory Guide 1.174, "An approach for using probabilistic risk assessment in risk-informed decisions on plant-specific changes to the licensing basis,
1.03 Lynnette Hendricks, Nuclear Energy Institute, letter to Office of Nuclear Reactor Regulation, dated November 12, 1999.

References for Chapter 3.0
1.01 November 12, 1999, the Advisory Committee on Reactor Safeguards (ACRS)
1.02 Generic Safety Issue (GSI) 82 [Ref. 2], "Beyond Design Basis Accident for ?Spent Fuel Pools."
1.03 NUREG-1275, Volume 12
1.04 NUREG-0612
1.05 NUREG/CR-4982
1.06 DOE-STD-3014-96, "Accident Analysis for Aircraft Crash Into Hazardous Facilities," U.S. Department of Energy, October 1996
1.07 NUREG/CR-2944, "Tornado Damage Risk Assessment," Brookhaven National Laboratory, September 1982.

References for Chapter 4.0

1.01 Lynnette Hendricks, Nuclear Energy Institute, letter to Office of Nuclear Reactor Regulation, dated November 12, 1999.
1.02 U.S. Nuclear Regulatory Commission, "An Approach for Using Probabilistic Risk Assessment In Risk-Informed Decisions On Plant-Specific Changes to the Licensing Basis," Regulatory Guide 1.174, July 1998.
1.03 *U.S. Code of Federal Regulations*, "Emergency Plans," Section 47, Part 50, Title 10, "Energy."
1.04 *U.S. Code of Federal Regulations*, "Emergency Planning and Preparedness For Production and Utilization," Appendix E, Part 50, Title 10, "Energy."
1.05 NUREG-0396
(6) *U.S. Code of Federal Regulations*, "Financial Protection Requirements and Indemnity Agreements," Part 140, Title 10, "Energy."
I. *U.S. Code of Federal Regulations*, "Licensing Requirements for the Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste," Part 72, Title 10, "Energy."
II. U.S. Nuclear Regulatory Commission, "Financial Protection Required of Licensees of Large Nuclear Power Plants During Decommissioning," SECY-93-127, dated July 13, 1993.

6.0 Acronyms

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

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