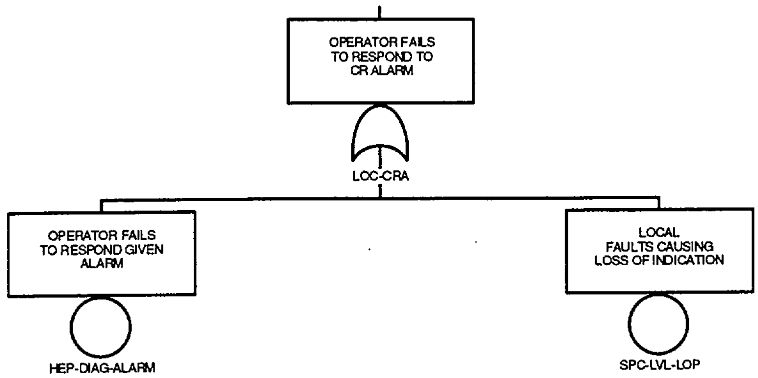
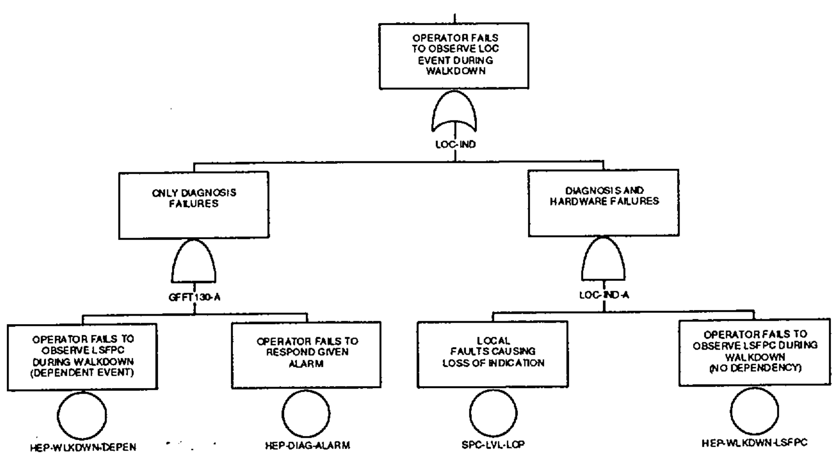
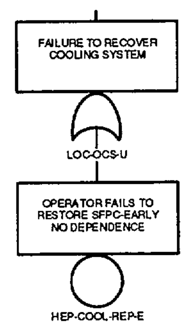
ATTACHMENT A

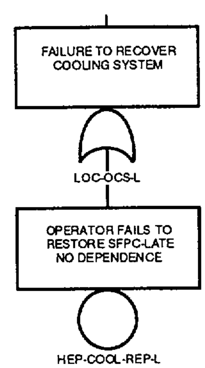
FAULT TREES USED IN THE RISK ANALYSIS

September 5, 2000 (1:54PM)

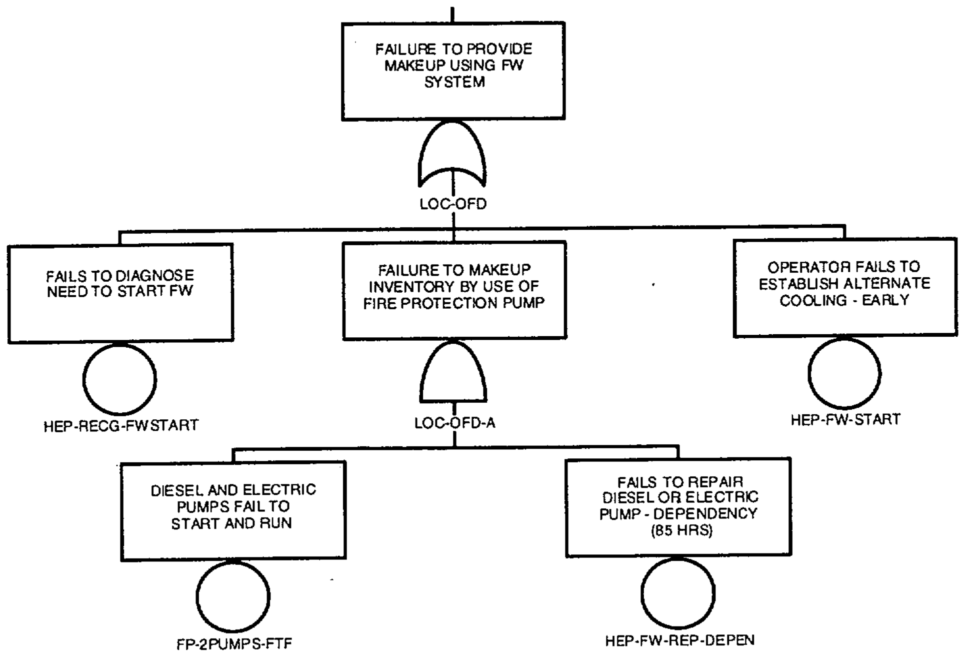


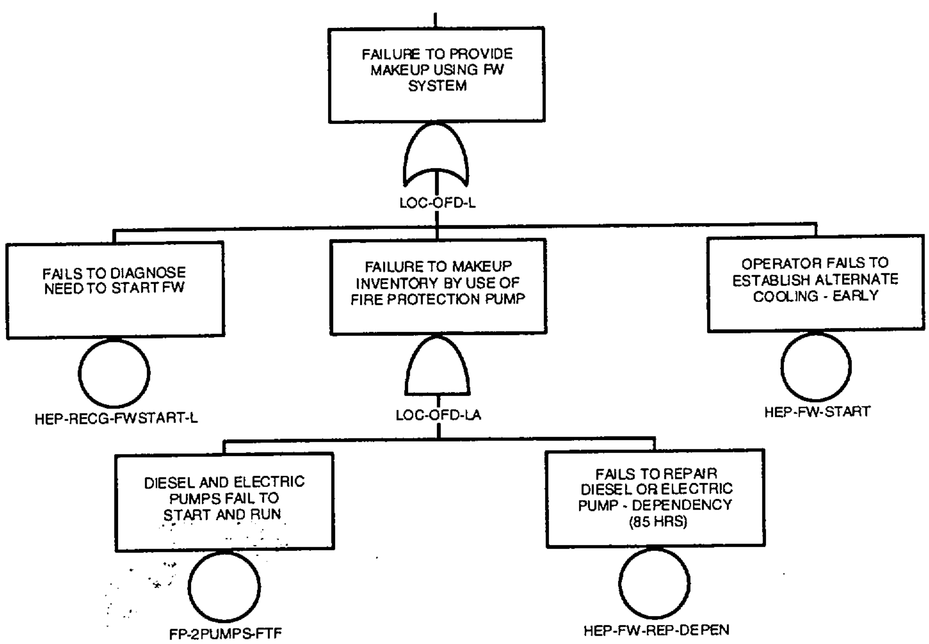


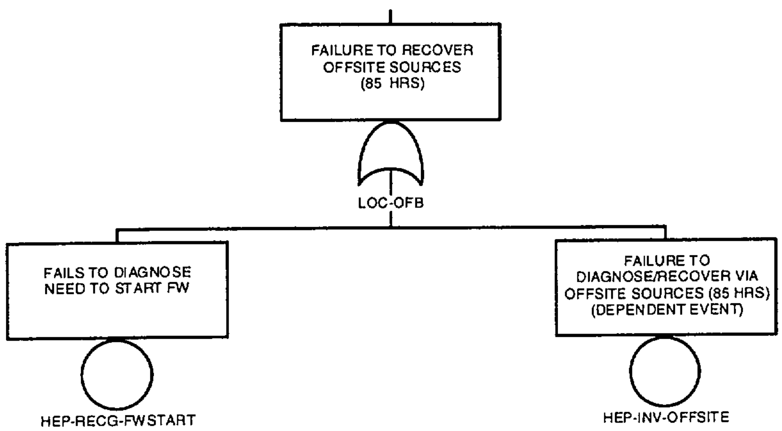

FAILURE TO RECOVER
COOLING SYSTEM
LOC-OCS-U
OPERATOR FAILS TO
RESTORE SFPC-EARLY
NO DEPENDENCE
HEP-COOL-REP-E

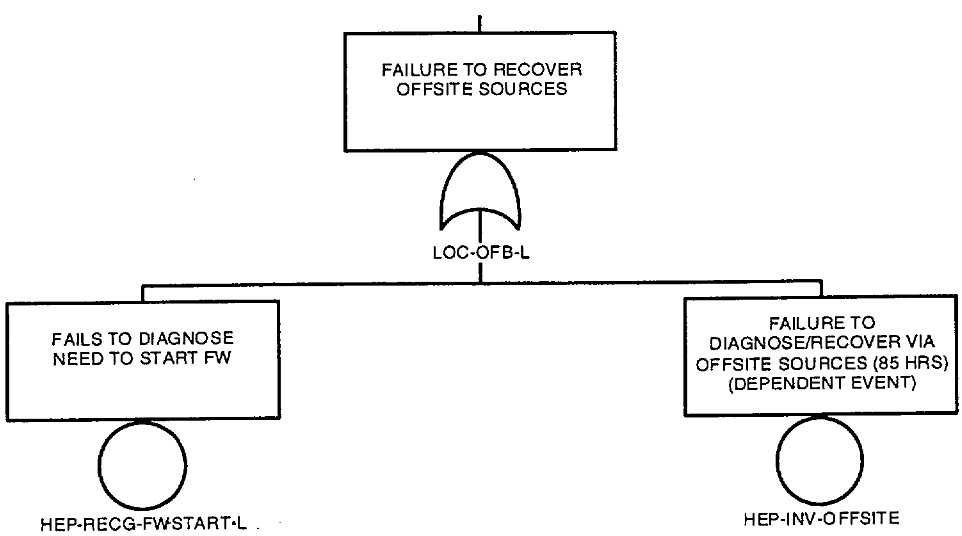

FAILURE TO RECOVER
COOLING SYSTEM
LOC-OCS-L
OPERATOR FAILS TO
RESTORE SFPC-LATE
NO DEPENDENCE
HEP-COOL-REP-L

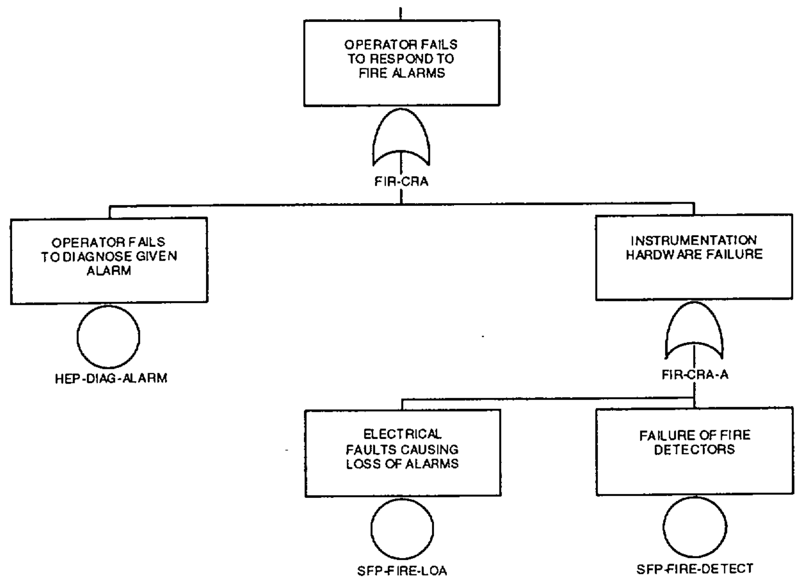
September 5, 2000 (1:54PM)

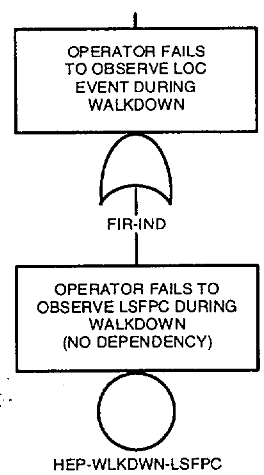




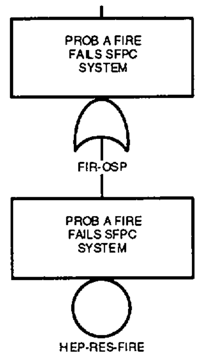

FAILURE TO RECOVER
OFFSITE SOURCES
(85 HRS)
LOC-OFB
FAILS TO DIAGNOSE
NEED TO START FW
HEP-RECG-FWSTART
FAILURE TO
DIAGNOSE/RECOVER VIA
OFFSITE SOURCES (85 HRS)
(DEPENDENT EVENT)
HEP-INV-OFFSITE

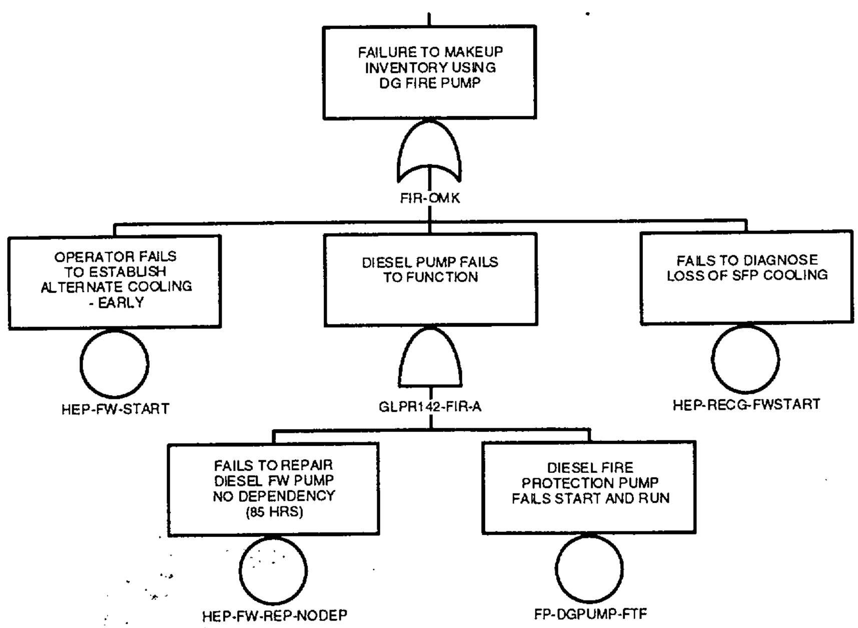

FAILURE TO RECOVER
OFFSITE SOURCES
LOC-OFB-L
FAILS TO DIAGNOSE
NEED TO START FW
HEP-RECG-FWSTART-L
FAILURE TO
DIAGNOSE/RECOVER VIA
OFFSITE SOURCES (85 HRS)
(DEPENDENT EVENT)
HEP-INV-OFFSITE


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


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

September 5, 2000 (1:54PM)





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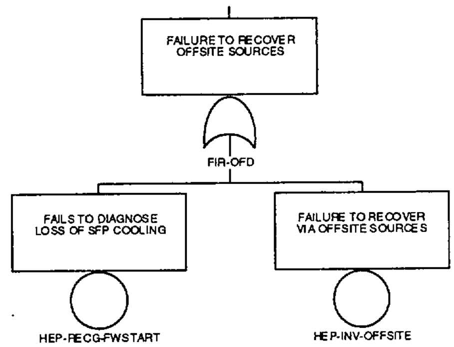

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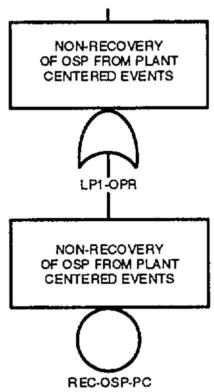

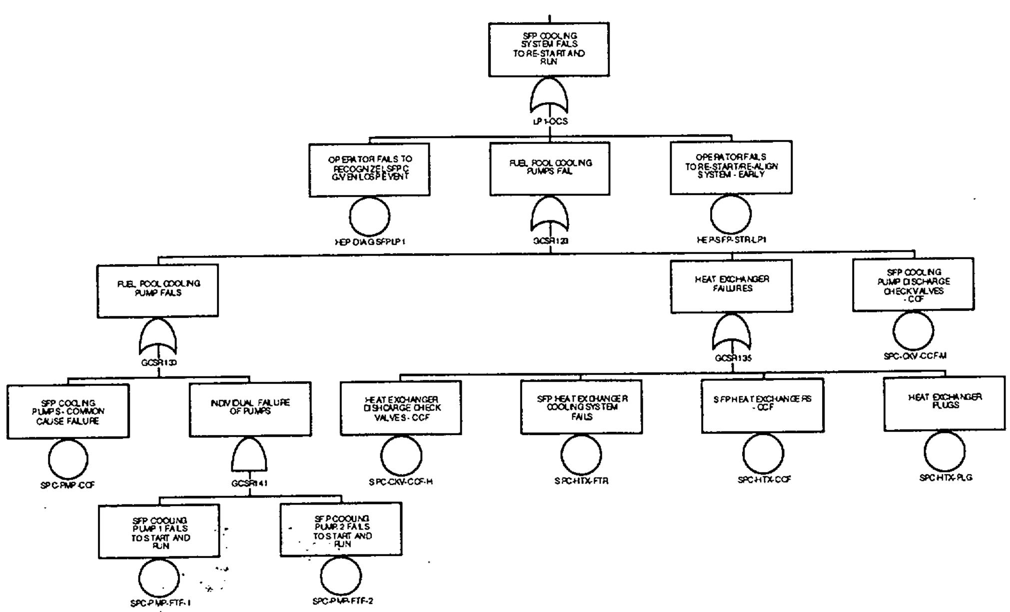

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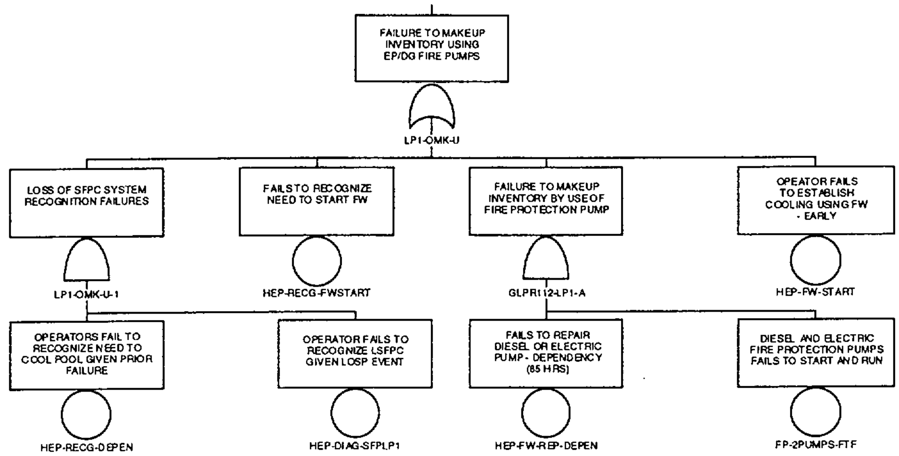
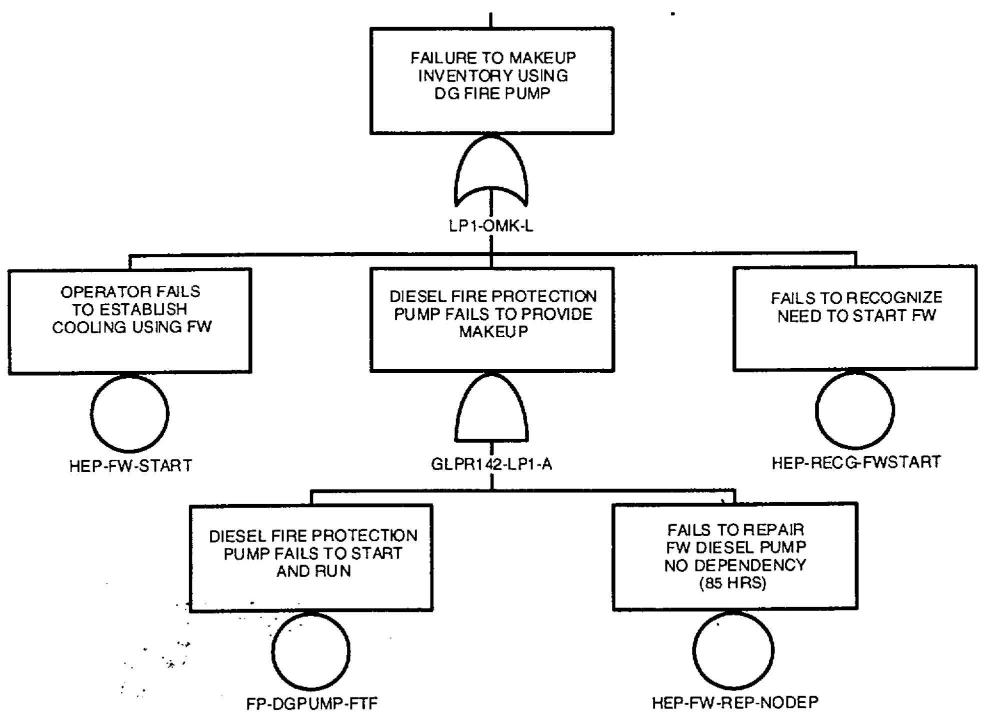
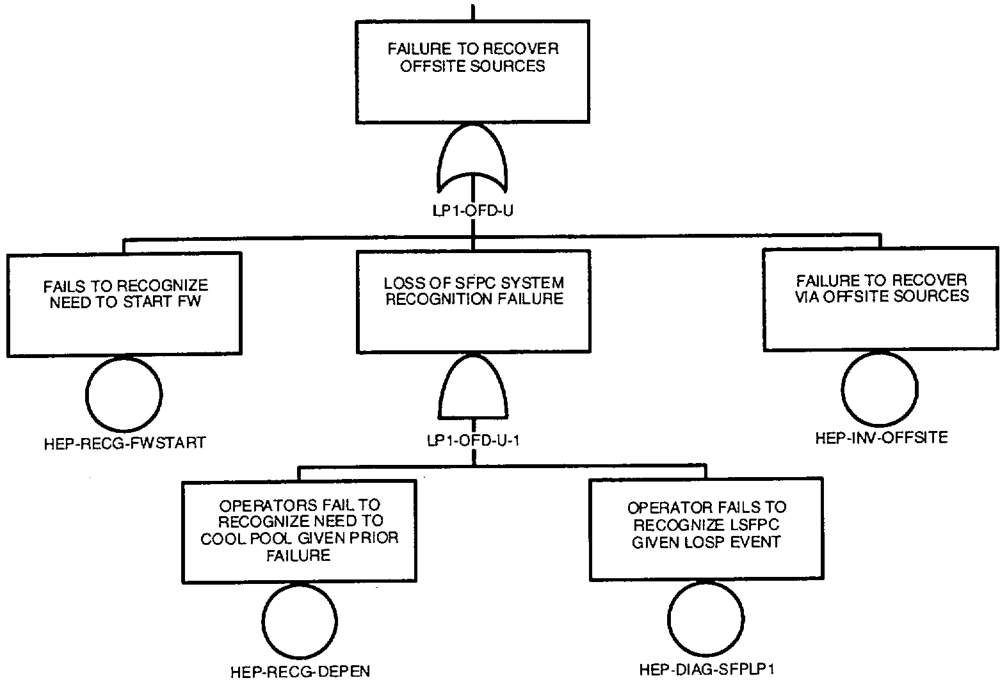
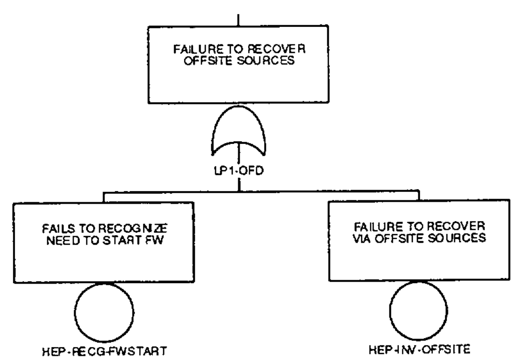
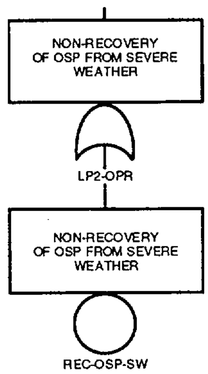
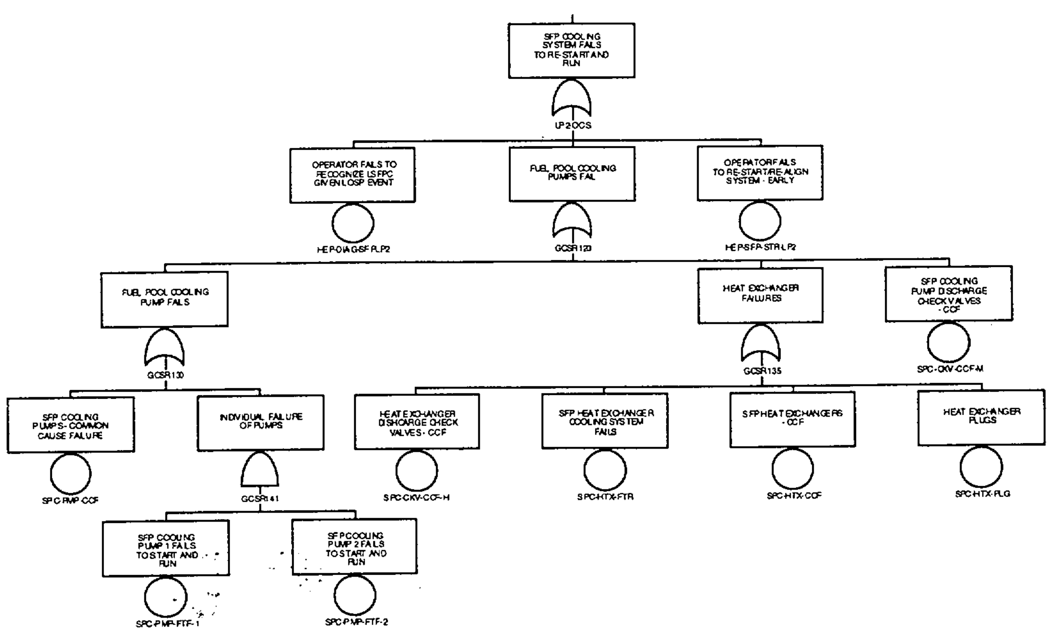


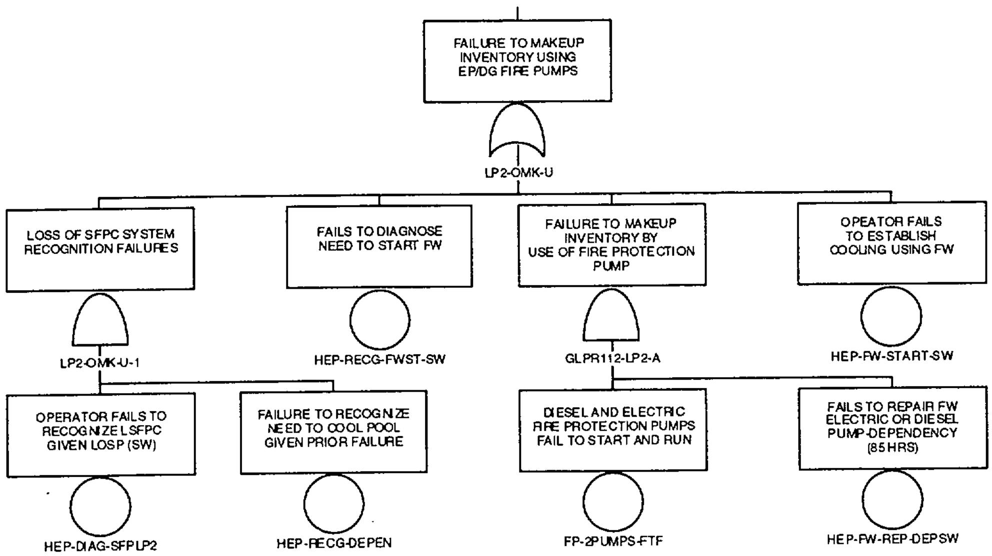


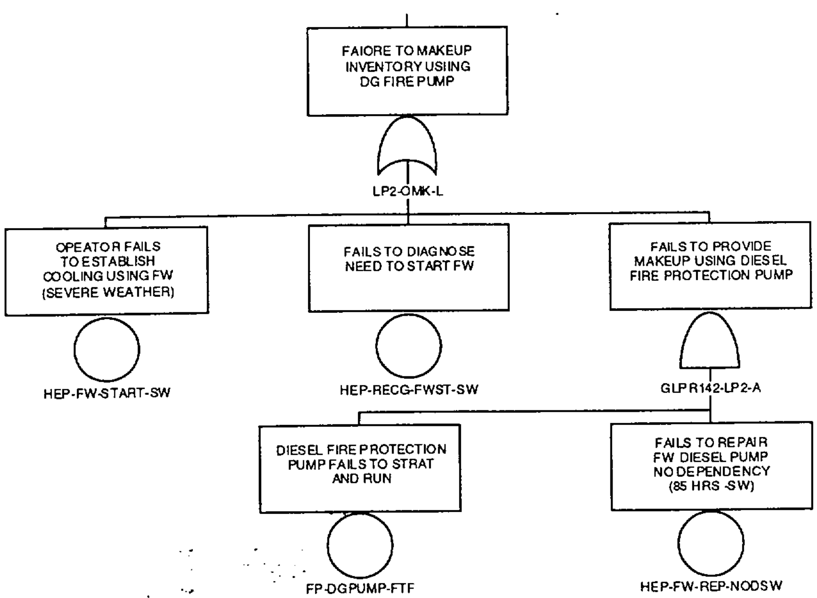

FAILURE TO RECOVER OFFSITE SOURCES
LP1-OFD-U
FAILS TO RECOGNIZE NEED TO START FW
LOSS OF SFPC SYSTEM RECOGNITION FAILURE
FAILURE TO RECOVER VIA OFFSITE SOURCES
HEP-RECG-FWSTART
LP1-OFD-U-1
HEP-INV-OFFSITE
OPERATORS FAIL TO RECOGNIZE NEED TO COOL POOL GIVEN PRIOR FAILURE
OPERATOR FAILS TO RECOGNIZE LSFPC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


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

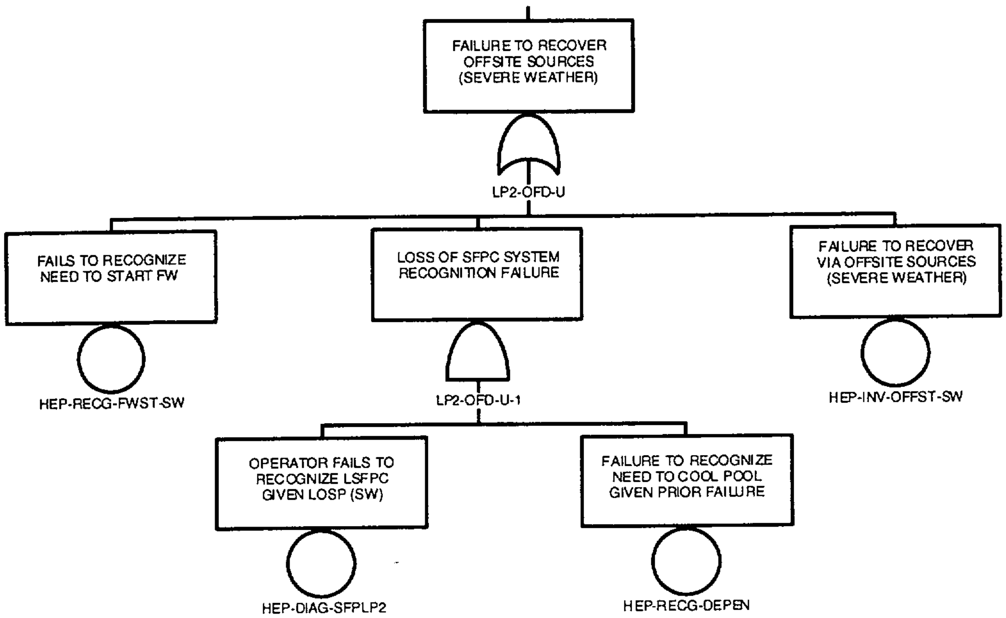

NON-RECOVERY
OF OSP FROM SEVERE
WEATHER
LP2-OPR
NON-RECOVERY
OF OSP FROM SEVERE
WEATHER
REC-OSP-SW

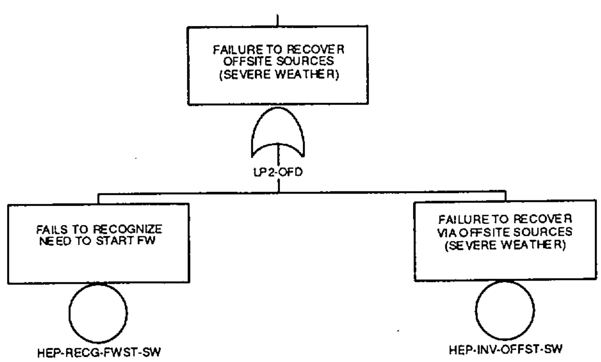

SFP COOLING
SYSTEM FAILS
TO RE-START AND
RUN
LP2OCS
OPERATOR FAILS TO
RECOGNIZE LSFPC
GIVEN LOSP EVENT
HEPOIAGSFPLP2
FUEL POOL COOLING
PUMPS FAIL
GCSR120
OPERATOR FAILS
TO RE-START/RE-ALIGN
SYSTEM - EARLY
HEPSFP-STRLP2
FUEL POOL COOLING
PUMP FAILS
GCSR130
HEAT EXCHANGER
FAILURES
GCSR135
SFP COOLING
PUMP DISCHARGE
CHECK VALVES
- CCF
SPC-CKV-CCF-M
SFP COOLING
PUMPS - COMMON
CAUSE FAILURE
SPC-PMP-CCF
INDIVIDUAL FAILURE
OF PUMPS
GCSR141
HEAT EXCHANGER
DISCHARGE CHECK
VALVES - CCF
SPC-CKV-CCF-H
SFP HEAT EXCHANGER
COOLING SYSTEM
FAILS
SPC-HTX-FTR
SFP HEAT EXCHANGERS
- CCF
SPC-HTX-CCF
HEAT EXCHANGER
PLUGS
SPC-HTX-PLG
SFP COOLING
PUMP 1 FAILS
TO START AND
RUN
SPC-PMP-FTF-1
SFP COOLING
PUMP 2 FAILS
TO START AND
RUN
SPC-PMP-FTF-2

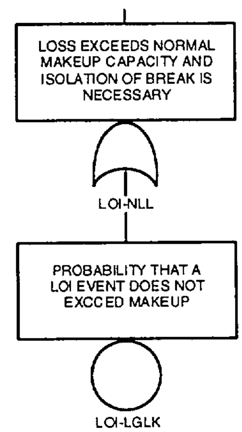

FAILURE TO MAKEUP INVENTORY USING EP/DG FIRE PUMPS
LP2-OMK-U
LOSS OF SFPC SYSTEM RECOGNITION FAILURES
FAILS TO DIAGNOSE NEED TO START FW
FAILURE TO MAKEUP INVENTORY BY USE OF FIRE PROTECTION PUMP
OPEATOR FAILS TO ESTABLISH COOLING USING FW
LP2-OMK-U-1
HEP-RECG-FWST-SW
GLPR112-LP2-A
HEP-FW-START-SW
OPERATOR FAILS TO RECOGNIZE LSFPC GIVEN LOSP (SW)
FAILURE TO RECOGNIZE NEED TO COOL POOL GIVEN PRIOR FAILURE
DIESEL AND ELECTRIC FIRE PROTECTION PUMPS FAIL TO START AND RUN
FAILS TO REPAIR FW ELECTRIC OR DIESEL PUMP-DEPENDENCY (85HRS)
HEP-DIAG-SFPLP2
HEP-RECG-DEPEN
FP-2PUMPS-FTF
HEP-FW-REP-DEPSW

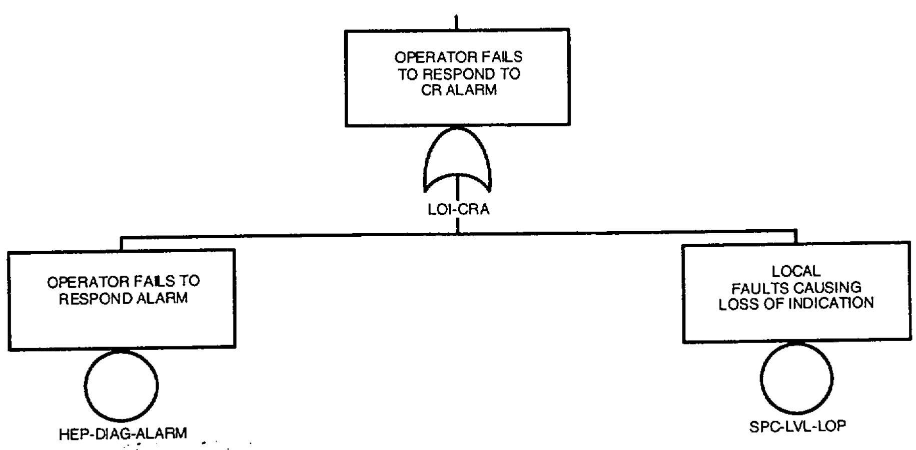

FAIORE TO MAKEUP INVENTORY USIING DG FIRE PUMP
LP2-OMK-L
OPEATOR FAILS TO ESTABLISH COOLING USING FW (SEVERE WEATHER)
FAILS TO DIAGNOSE NEED TO START FW
FAILS TO PROVIDE MAKEUP USING DIESEL FIRE PROTECTION PUMP
HEP-FW-START-SW
HEP-RECG-FWST-SW
GLPR142-LP2-A
DIESEL FIRE PROTECTION PUMP FAILS TO STRAT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

September 5, 2000 (1:54PM)

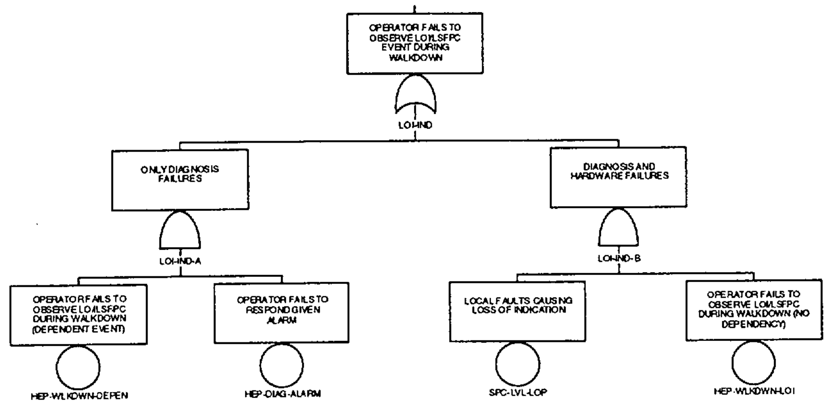


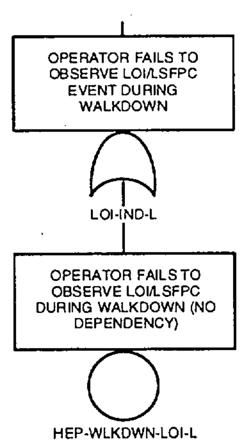


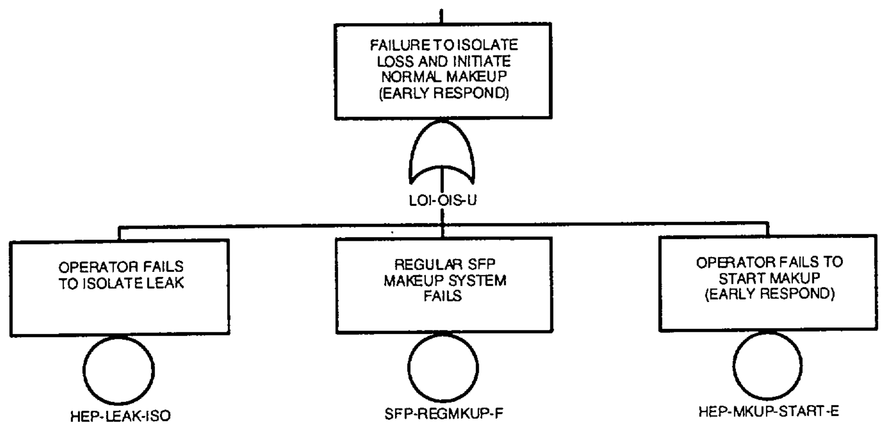

LOSS EXCEEDS NORMAL MAKEUP CAPACITY AND ISOLATION OF BREAK IS NECESSARY
LOI-NLL
PROBABILITY THAT A LOI EVENT DOES NOT EXCCED MAKEUP
LOI-LGLK

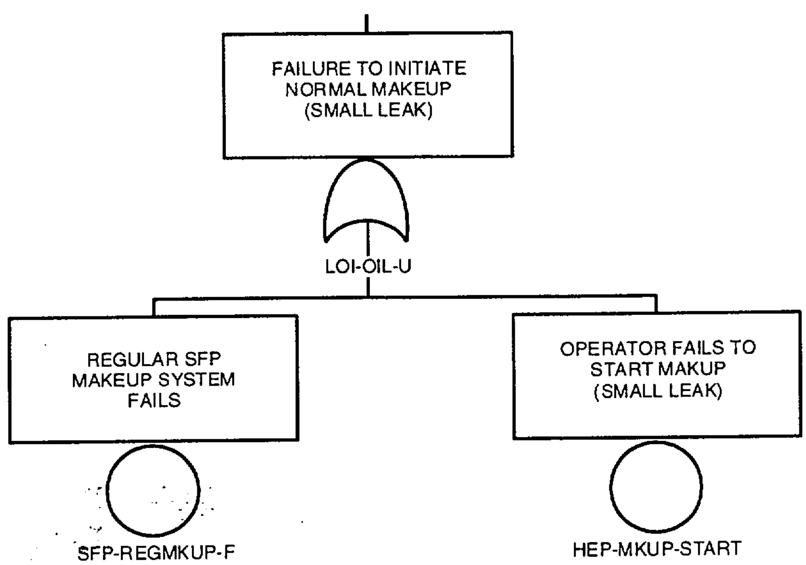

OPERATOR FAILS TO RESPOND TO CR ALARM
LOI-CRA
OPERATOR FAILS TO RESPOND ALARM
HEP-DIAG-ALARM
LOCAL FAULTS CAUSING LOSS OF INDICATION
SPC-LVL-LOP

September 5, 2000 (1:54PM)

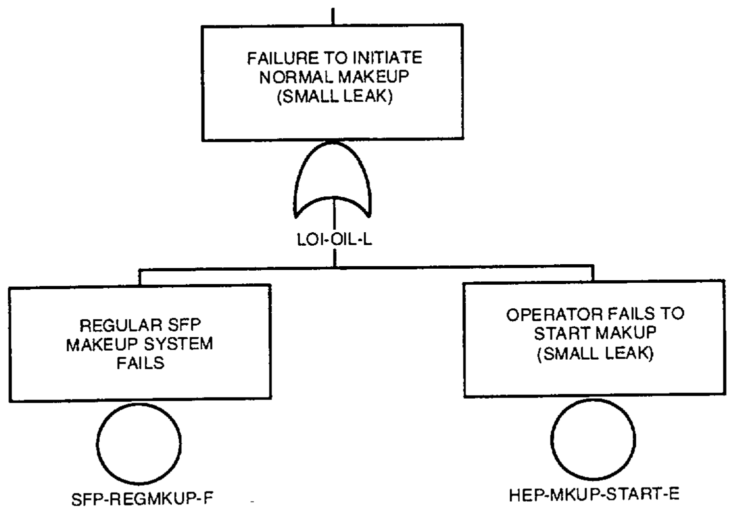


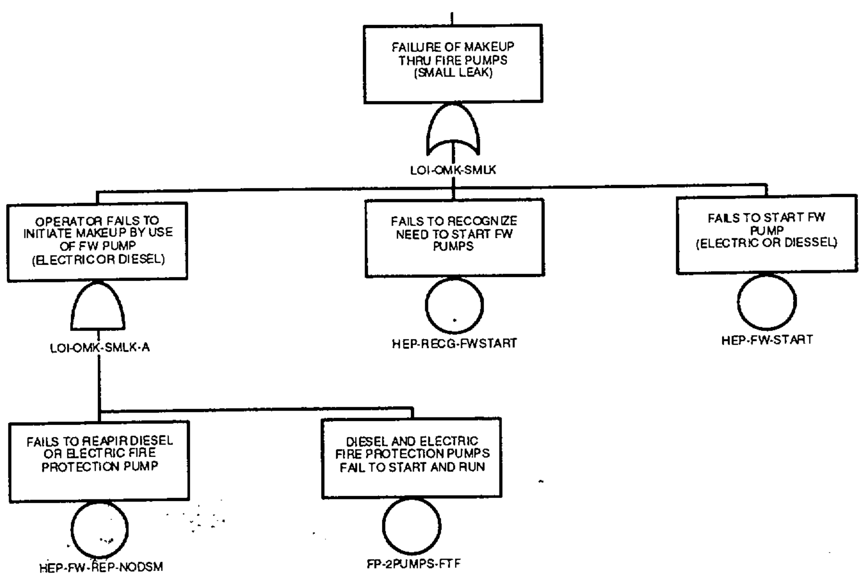



FAILURE TO ISOLATE LOSS AND INITIATE NORMAL MAKEUP (EARLY RESPOND)
LOI-OIS-U
OPERATOR FAILS TO ISOLATE LEAK
REGULAR SFP MAKEUP SYSTEM FAILS
OPERATOR FAILS TO START MAKUP (EARLY RESPOND)
HEP-LEAK-ISO
SFP-REGMKUP-F
HEP-MKUP-START-E


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

September 5, 2000 (1:54PM)





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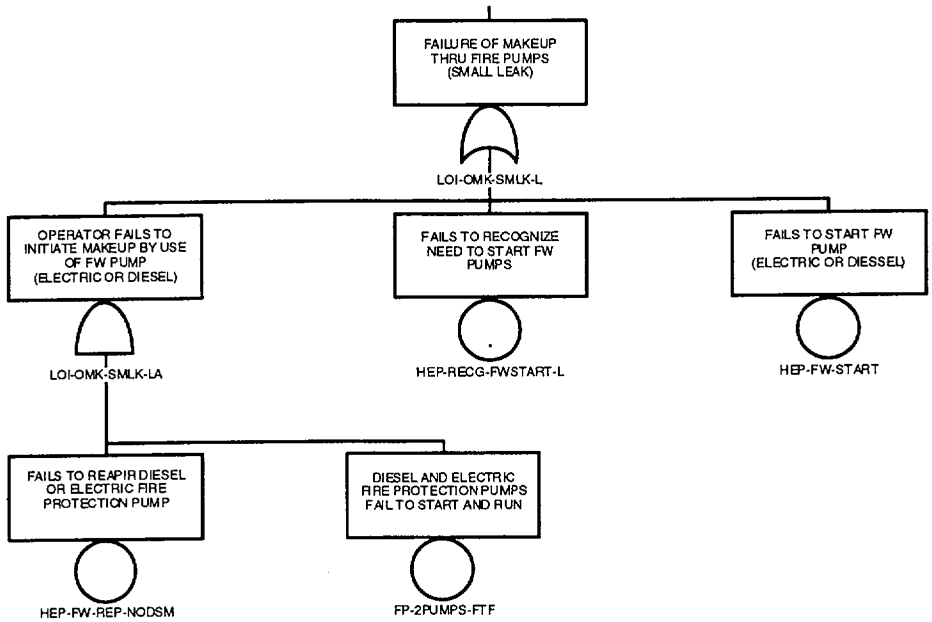
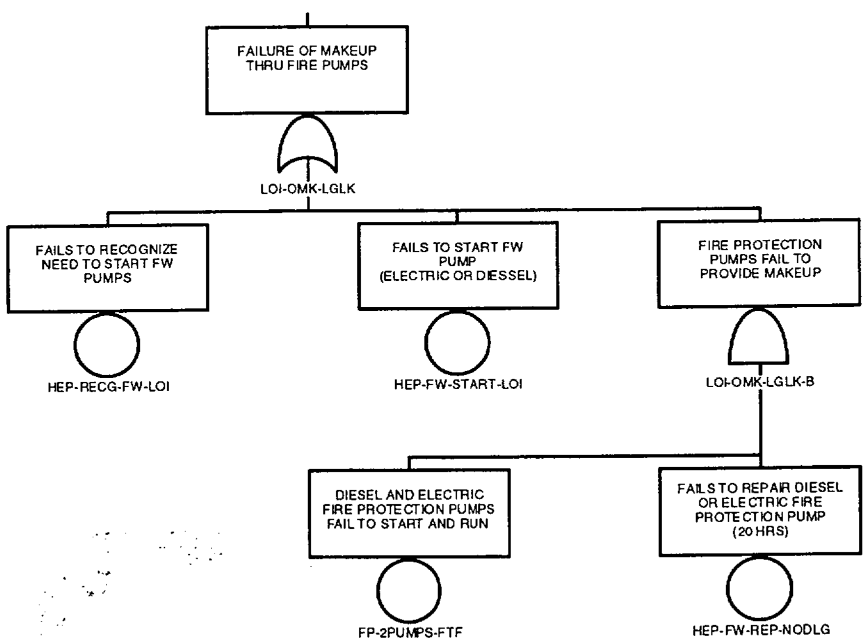
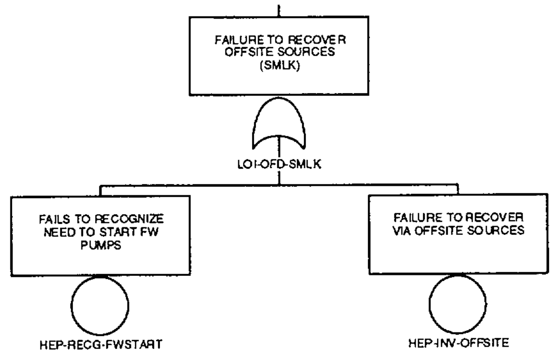
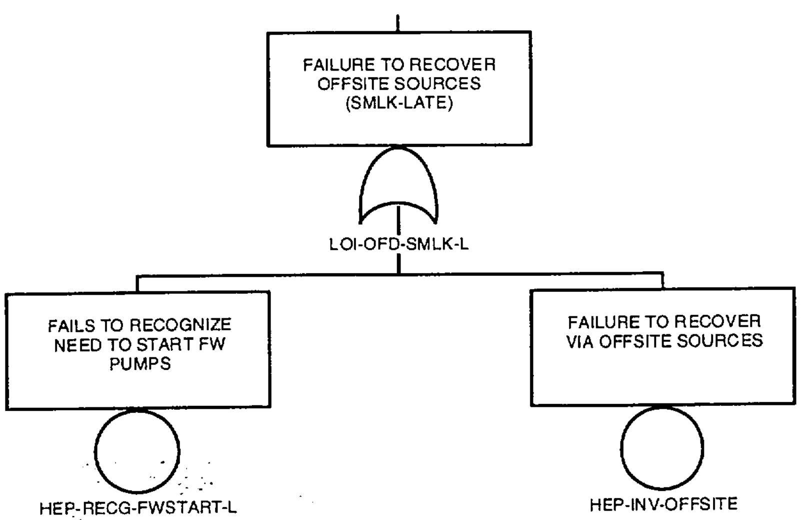


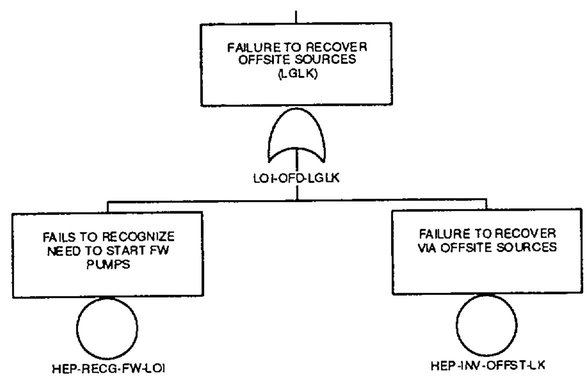


September 5, 2000 (1:54PM)





September 5, 2000 (1:54PM)



ATTACHMENT B

SPAR HRA Worksheet

SPAR HRA Human Error Worksheet (Page 1 of 3)

Plant:_____ **Initiating Event:**____ **Sequence Number:**________ **Basic Event Code:** _______________

Basic Event Context: ____

Basic Event Description:____

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

____

Part I. DIAGNOSIS

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

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

B. Calculate the Diagnosis Failure Probability

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

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

Diagnosis

Failure Probability

SPAR HRA Human Error Worksheet (Page 2 of 3)

Plant:____ Initiating Event:____ Sequence Number:_______ Basic Event Code: ______________

Basic Event Context: ____
Basic Event Description:____

Part II. ACTION

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

PSFs	PSF Levels	Multiplier for Action	If non-nominal PSF levels are selected, please note specific reasons in this column
Available Time	Inadequate time	P(failure) = 1.0	
	Time available . time required	10	
	Nominal time	1	
	Time available>50 x time required	0.01	
Stress	Extreme	5	
	High	2	
	Nominal	1	
Complexity	Highly complex	5	
	Moderately complex	2	
	Nominal	1	
Experience/Training	Low	3	
	Nominal	1	
	High	0.5	
Procedures	Not available	50	
	Available, but poor	5	
	Nominal	1	
Ergonomics	Missing/Misleading	50	
	Poor	10	
	Nominal	1	
	Good	0.5	
Fitness for Duty	Unfit	P(failure) = 1.0	
	Degraded Fitness	5	
	Nominal	1	
Work Processes	Poor	5	
	Nominal	1	
	Good	0.5	

B. Calculate the Action Failure Probability

(1) If all PSF ratings are nominal, then the Action Failure Probability = 1E-3

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

___

Action

Failure Probability

Plant:_____ Initiating Event:____ Sequence Number:________ Basic Event Code:_________

PART III. CALCULATE THE TASK FAILURE PROBABILITY WITHOUT FORMAL DEPENDENCE ($P_{W/OD}$)

Calculate the Task Failure Probability Without Formal Dependence ($P_{w/od}$) by adding the Diagnosis Failure Probability (from Part I, p.1) and the Action Failure Probability (from Part II, p. 2).

			If all PSFs are nominal, then
Diagnosis Failure Probability:	______________	Diagnosis Failure Probability:	1E-2
Action Failure Probability:	+______________	Action Failure Probability:	+1E-3
Task Failure Without Formal Dependence ($P_{w/od}$)	=______________	$P_{(w/od)}$	= 1.1E-2

Part IV. DEPENDENCY

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

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

____

Dependency Condition Table

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

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

For Complete Dependence the probability of failure is 1.

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

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

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

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

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

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

September 5, 2000 (1:54PM)

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

20. Introduction

The staff's concern regarding seismic issues at spent fuel pools (SFPs) involves very large earthquake ground motions that could catastrophically fail the SFP. Under this scenario, the pool would suffer a significant breach, it would drain rapidly, and it will be incapable of being refilled. This would lead to gradual cladding heat up, possibly followed by a zirconium cladding fire. The staff evaluated what would be the return frequency of such large earthquake ground motions. Attachment 1 to this appendix provides the checklist proposed by NEI and enhanced by the staff to assure adequate seismic capacity at SFPs for decommissioning sites that wish to be granted exemptions to EP, safeguards, and indemnification. Attachment 2 to this appendix provides the analysis of the seismic failure potential of SFPs by the NRC's consultant, Robert Kennedy, for nuclear power plant sites based on a generic 1.2 g spectral acceleration high confidence, with low probability of failure (HCLPF) value for spent fuel pools.

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

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

To evaluate the risk from a seismic event at a spent fuel pool, one needs to know both the likelihood of seismic ground motion at various g-levels (i.e., seismic hazard) and the conditional probability that a structure, system, or component (SSC) will fail at a given acceleration level (i.e., the fragility of the SSC). These are convolved mathematically to arrive at the likelihood that the spent fuel pool will fail from a seismic event. In evaluating the effect of seismic events on spent fuel pools, it became apparent that although information was available on the seismic hazard for nuclear power plant sites, the staff did not have fragility analyses of the pools, nor

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

generally did licensees. The staff recognized that many of the spent fuel pools and the buildings housing them were designed by different architect engineers. The spent fuel pools and structures housing them were built to codes and code editions.

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

2. Return Period of SFP-Failing Earthquake Ground Motions

The staff reexamined its methods for estimating seismic risk and reexamined the results of Table 3 in Attachment 2 to this appendix, which estimates the return frequencies of large earthquake ground motions that could fail spent fuel pools. It was decided that the HCLPF value of 1.2 g peak spectral acceleration was a good measure of seismic adequacy for decommissioning plant SFPs that need only be tied to the return period of the earthquake and not to the safe shutdown earthquake magnitude for the site. The staff used a simplified, but slightly conservative method (See Attachment 2) to estimate the annual probability of a zirconium fire due to seismic events (including use of site-specific seismic hazard estimates). These calculations resulted in a range of frequencies from less than $1x10^{-8}$ per year to over $1x10^{-5}$ per year, depending on the site and the seismic estimates used.

The staff finds $5X10^{-6}$ per year spent fuel pool failure annual probability to be a reasonable acceptance criterion for the ground motions, since it is a factor of 2 less than the $1x10^{-5}$ per year PPG and the estimated frequency of zirconium cladding fires from other initiators is an order of magnitude lower.[2] Such a margin is warranted due to the uncertainties of the seismic hazard and spent fuel pool fragilities at each site, and to the small margin between seismic risk results and the Quantitative Health Objectives (QHOs) of the NRC.

3. Seismic Checklist

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[2] The staff used a measure of $3x10^{-6}$ per year for the adequacy of seismic return period in its earlier versions of the report. Use of this measure meant that eight operating reactor sites east of the Rocky Mountains would need to perform additional calculations (if the LLNL hazard estimates were used) if they wanted to take advantage of EP, indemnification, or safeguards exemptions or rule making. The staff reexamined its criteria and determined that $5x10^{-6}$ per year was an acceptable measure of SFP risk given the uncertainties in the seismic hazard and fragility estimates. In addition, comments from the Advisory Committee on Reactor Safeguards and other stakeholders indicated that the proposed measure and the approach the staff was using were too conservative. Also, the proposed approach contained different assessments for the eastern and the western United States and was complicated by the fact that seismic fragility information for ground motion levels beyond 0.5 g is not readily available from a peer reviewed data base.

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

4. Seismic Risk - Support System Failure

In its preliminary draft report published in June 1999, the staff assumed that a ground motion three times the SSE was the HCLPF of the spent fuel pool. This meant that 95% of the time the pool would remain intact (i.e., would not leak significantly). The staff evaluated what would happen to spent fuel pool support systems (i.e., the pool cooling and inventory make-up systems) in the event of an earthquake three times the SSE. The staff modeled some recovery as possible (although there would be considerable damage to the area's infrastructure at such earthquake accelerations). The estimate in the preliminary report for the contribution from this scenario was 1×10^{-6} per year. In this report, this estimate has been refined based on looking at a broader range of seismic accelerations and further evaluation of the conditional probability of recovery under such circumstances. The staff estimates that for an average site in the northeast United States the return period of an earthquake ground motion that would damage a decommissioning plant's spent fuel pool cooling system equipment (assuming it had at least minimal anchoring) is about once in 4,000 years. The staff quantified a human error probability of 4×10^{-4} that represents the failure of the fuel handlers to obtain off-site resources. The event was quantified using the SPAR HRA technique. The performance shaping factors chosen were as follows: expansive time (> 50 times the required time), high stress, complex task because of the earthquake and its non-routine nature, quality procedures, poor ergonomics due to the earthquake, and finally a crew who had executed these tasks before, conversant with the procedures and one another. In combination we now estimate the risk from support failure due to seismic events to be on the order of 4×10^{-8} per year. The risk from support system failure due to seismic events is bounded by catastrophic failure of the spent fuel pool due to a seismic event.

5. Hazard Estimate and Fragility Uncertainties

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

6. Conclusion

The staff recommends that those plants that exceed $5.0X10^{-6}$ per year frequency for exceeding 1.2 g peak spectral acceleration (using the LLNL hazard estimates) in their spent fuel pool should be required to conduct plant-specific seismic analysis beyond the confirmation of the checklist if they desire to obtain exemptions (or take advantage of rule making) from EP, indemnification, or security at decommissioning sites. Using the LLNL hazard estimates, this process results in identification of four sites in the eastern U.S., only three of which are operating reactor sites - Pilgrim, H. B. Robinson, and Vogtle sites, with Maine Yankee the decommissioning site. In the western U.S., the WNP2, Diablo Canyon, and San Onofre sites are also beyond the scope of a simple screening evaluation. The use of EPRI hazard estimates similarly points these sites out as those with the highest return periods in the accelerations of interest. Based on the NRC sponsored study, "Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear Power Plants," NUREG/CR 5176, January 1989, the seismic HCLPF capacity of the H. B. Robinson spent fuel pool has been estimated to be 0.65 g peak ground motion (PGA). For the Vogtle, Pilgrim, WNP2, Diablo Canyon, and San Onofre sites, it may be necessary for the utilities to conduct a detailed site-specific seismic risk evaluation if they desire an exemption from EP when the site is in decommissioning.

To summarize the staff recommendation for seismic vulnerability of spent fuel pools, (1) all sites must conduct an assessment of the spent fuel pool structures using the revised seismic check list in order to identify any structural degradation, potential for seismic interaction from superstructures and over head cranes, and to verify that they have a seismic HCLPF value of 1.2 g PSA or higher, (2) those sites that cannot demonstrate that a seismic HCLPF value exists, may either undertake appropriate remedial action or conduct site-specific seismic risk assessment and (3) Pilgrim, H. B. Robinson, Vogtle, WNP2, Diablo Canyon, and San Onofre sites would have to use the seismic check list to identify any structural degradation or other anomalies and then conduct a site-specific seismic risk assessment if they desire an exemption from EP when their sites are in decommissioning.

September 5, 2000 (1:54PM)

Attachment 1

Seismic Check list for Commercial Nuclear Power Plants
During Decommissioning

Attachment 2

→ will be renumbered

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

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

by
Robert P. Kennedy
October 1999

prepared for

Brookhaven National Laboratory

1. Introduction

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

2. Background Information

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

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

as shown in Ref. 10.

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

Ref. 5:

$$\begin{aligned} &\text{Vermont Yankee (BWR):} && C_{HCLPF} = 0.48g \text{ PGA} \\ &\text{Robinson (PWR):} && C_{HCLPF} = 0.65g \text{ PGA} \end{aligned} \tag{2}$$

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

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

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

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

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

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

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

CEUS plants. However, HCLPF capacities this high are sufficiently high to achieve seismic risk estimates less than 3×10^{-6} for most CEUS plants based upon the Ref. 8 hazard curves. This subject is further discussed in Section 4.

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

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

BWR Sites			
Median Capacity	$C_{50} = 1.4$	PGA	
HCLPF Capacity	$C_{HCLPF} = 0.48g$	PGA	(4)

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

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

PWR Sites			
Median Capacity	$C_{50} = 2.0$	PGA	
HCLPF Capacity	$C_{HCLPF} = 0.65g$	PGA	(5)

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

3. Development and Use of Seismic Screening Criteria

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

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

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

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

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

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

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

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

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

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

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

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

3.2 Spent Fuel Pool Racks

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

3.3 Seismic Level 2 Screening Requirements

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

4. Seismic Risk Associated With Screening Level 2

4.1 Simplified Approaches for Estimating Seismic Risk Given the HCLPF Capacity

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

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

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

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

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

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

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

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

Step 3: Determine seismic risk P_F from:

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

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

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

associated with Screening Level 2 for the Vermont Yankee site by rigorous convolution, it is necessary to extrapolate the Ref. 9 hazard estimates down to the $2\text{x}10^{-8}$ exceedance frequency. Also, intermediate values in Table 1 have been obtained by interpolation.

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

1. Ref. 1 simplified approach, i.e., Eqn. (3).
2. Ref. 10 simplified approach, i.e., Steps 1 through 3 above.
3. Rigorous convolution of the hazard and fragility estimates.

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

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

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

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

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

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

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

From Table 3, it can be seen that the EPRI89 hazard estimates produce generally much lower seismic risk estimates corresponding to Screening Level 2 than do the LLNL93 hazard estimates. Based on the EPRI89 hazard estimates, only one site has a seismic risk exceeding $1x10^{-6}$. Only three other sites have seismic risks exceeding $0.5x10^{-6}$. Table 3 includes all sites for which the computed seismic risk exceeds $0.5x10^{-6}$ based on the mean EPRI89 hazard estimates.

5. Conclusions

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

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

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

References

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

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

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

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

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

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

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

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

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

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

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

Table 1

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

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

* By Interpolation

** By Extrapolation

Table 2

Comparison of Seismic Risk Estimated by Various Approaches

C_{HCLPF} = 1.2g PSA, $\beta = 0.4$

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

Table 3
Seismic Risk Associated With Screening Level 2

C_{HCLPF} = 1.2g Peak Spectral Acceleration

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

Not Available

will be renumbered

5g Enhanced Seismic Checklist

Item 1:

Requirement: Identify Preexisting Concrete and Liner Plate Degradation

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

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

Item 2:

Requirement: Assure Adequate Ductility of Shear Wall Structures

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

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

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

Item 3:

Requirement: Assure Design adequacy of Diaphragms (including roofs)

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

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

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

Item 4:

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

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

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

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

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

Item 5:

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

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

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

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

Item 6:

Requirement: Verify the Adequacy of Spent Fuel Pool Penetrations

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

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

Item 7:

Requirement: Evaluate the Potential for Impacts with Adjacent Structures

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

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

Item 8:

Requirement: Evaluate the Potential for Dropped Loads

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

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

Item 9:

Requirement: Evaluation of Other Failure Modes

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

(Section 7 & Appendix C).

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

Item 10: Potential Mitigation Measures

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

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

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

c.) Perform plant-specific seismic hazard analyses to demonstrate that the seismic risk associated with a catastrophic failure of the pool is at an acceptable level. (The exact "acceptable" risk level has not been precisely quantified but is believed to be in the range of 1.0E-06.)

We believe that use of the checklist and determination that the spent fuel pool HCLPF is sufficiently high will assure that the frequency of fuel uncovery from seismic events is less than or equal to 1×10^{-6} per year.

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Appendix 2c Structural Integrity of Spent Fuel Pool Structures Subject to Heavy Loads Drops

1. Introduction

A heavy load drop into the spent fuel pool (SFP) or onto the spent fuel pool wall can affect the structural integrity of the spent fuel pool. A loss-of-inventory from the spent fuel pool could occur as a result of a heavy load drop. For single failure proof systems where load drop analyses have not been performed at decommissioning plants, the mean frequency of a loss-of-inventory caused by a cask drop was estimated to be $2.0x10^{-7}$ per year (assuming 100 lifts per year). For a non-single failure proof handling system where a load drop analysis has not been performed, the mean frequency of a loss-of-inventory event caused by a cask drop was estimated to be $2.1x10^{-5}$ per year. The staff believes that performance and implementation of a load drop analysis that has been reviewed and approved by the staff will substantially reduce the expected frequency of a loss-of-inventory event from a heavy load drop for either a single failure proof or non-single failure proof system.

2. Analysis

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

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

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

(21) NEI data on actual spent fuel pool cask lifts at U.S. commercial nuclear power plants [Ref.3]

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

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

combining the frequency of load drops with the conditional probability of pool failure given a load drop, the staff was able to estimate the frequency of heavy load drops causing a zirconium fire at decommissioning facilities.

3. Frequency of Heavy Load Drop

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

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

Crane failures (failure of lifting equipment) were evaluated using the fault tree shown in Figure A2c-1, which comes from NUREG-0612. At the time that heavy loads were evaluated in NUREG-0612, low density storage racks were in use and after 30 to 70 days (a period of about 0.1 to 0.2 per year), no radionuclide releases were expected if the pool were drained. It was assumed in NUREG-0612 that after this period, the fuel gap noble gas inventory had decayed and no zirconium fire would have occurred. Today, most decommissioning facilities use high density storage racks. This analysis evaluates results at one year after reactor shutdown. Our engineering evaluations indicate that for today's fuel configurations, burnup, and enrichment, a zirconium cladding fire may occur if the pool were drained during a period as long as five years.

A literature search performed by the staff searching for data on failure to secure loads identified a study (WIPP report) that included a human error evaluation for improper rigging. This study was used by the staff to re-evaluate the contribution of rigging errors to the overall heavy load (cask) drop rate and to address both the common mode effect estimate and the 1990s Navy data. Failure to secure a load was evaluated in the WIPP report for the Trudock crane. The WIPP report determined that the most probable human error was associated with attaching the lifting legs to the lifting fixture. In the WIPP report, the failure to secure the load (based on a 2-out-of-3 lifting device) was estimated based on redundancy, procedures, and a checker. The report assumed that the load could be lowered without damage if no more than one of the three connections were not properly made. Using NUREG/CR-1278 [Ref. 4] information, the mean failure rate due to improper rigging was estimated in the WIPP report to be $8.7x10^{-7}$ per lift. Our requantification of the NUREG-0612 fault tree using the WIPP improper rigging failure rate is summarized in Table A2c-2. The WIPP evaluation for the human error probabilities is summarized in Table A2c-3.

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

4. Evaluation of the Load Path

Just because a heavy load is dropped does not mean that it will drop on the spent fuel pool wall or on the pool floor. It may drop at other locations on its path. A load path analysis is plant-specific. In NUREG-0612 it was estimated that the heavy load was near or over the spent fuel pool for between 5% and 25% of the total path needed to lift, move, and set down the load. It was further estimated that if the load were dropped from 30 feet or higher (or in some circumstances from 36 feet and higher depending on the assumptions) when it is over the pool floor, and if a plant-specific load drop analysis had not been performed[2], then damage to the pool floor would result in loss-of-inventory. In addition we looked at the probability that the load drop occurred over the pool wall from eight to ten inches above the edge of the pool wall. In our analysis we evaluated the chances the load was raised sufficiently high to fail the pool and evaluated the likelihood that the drop happened over a vulnerable portion of the load path. Table A2c-2 presents the results for a heavy load drop on or near the spent fuel pool. Based on NUREG-0612, if the cask were dropped on the spent fuel pool floor, the likelihood of a loss-of-inventory given the drop is 1.0. Based on the evaluation presented in NUREG/CR-5176 [Ref. 5], if the load were dropped on the spent fuel pool wall, the likelihood of a loss-of-inventory given the drop is 0.1.

5. Conclusion

Our heavy load drop evaluation is based on the method and fault trees developed in NUREG-0612. New 1990s Navy data were used to quantify the failure rate of the lifting equipment. The WIPP human error evaluation was used to quantify the failure to secure the load. We estimated the mean frequency of a loss-of-inventory from a cask drop onto the pool floor or onto the pool wall from a single failure proof system to be 2.0×10^{-7} per year for 100 lifts per year.

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

For licensees that choose the non-single failure proof handling system option in NUREG-0612,

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

we based the mean frequency of a loss-of-inventory event on the method used in NUREG-0612. In NUREG-0612, an alternate fault tree than that used for the single failure proof systems was used to estimate the frequency of exceeding the release guidelines (loss-of-inventory) for a non-single failure proof system. We calculated the mean frequency of catastrophic pool failure (for drops into the pool, or on or near the edge of the pool) for non-single failure proof systems to be about 2.1×10^{-5} per year when corrected for the 1990s Navy data and 100 lifts per year. This estimate exceeds the proposed pool performance guideline of 1×10^{-5} per year. The staff believes that a licensee which chooses the non-single failure proof handling system option in NUREG-0612 can reduce this estimate to the same range as that for single failure proof systems by performing a comprehensive and rigorous load drop analysis. The load drop analysis is assumed to include implementation of plant modifications or load path changes to assure the spent fuel pool would not be catastrophically damaged by a heavy load drop.

References:

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

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

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

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

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

Attachment 2C-1

Uncertainties

1. Incident rate.

   The range used in this evaluation (1.0×10^{-4} to 1.5×10^{-4} incidents per year) was based on the Navy data originally assessed by the staff in NUREG-0612. The 1999 Navy data, like the 1980 data, did not report the number of lifts made and only provided information about the number of incidents. The cask loading experience at light water reactors and Ft. St. Vrain tends to support values used for the incident range.

2. Drop rate.

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

3. Load path.

   The fraction of the load path over which a load drop may cause sufficient damage to the spent fuel pool to result in a loss-of-inventory was estimated to be between 0.5% and 6.25% of the total path needed to lift, move, and set down the load. This range was developed by the staff for the NUREG-0612 evaluation. No time motion study was performed to account for the fraction of time the load is over any particular location.

4. Load handling design.

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

5. Load drop analysis

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

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

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

Notes:

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

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

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

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

	Risk reduction for a single-failure proof crane (LOI-N /LOI-S)	---	35	4	104

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

Symbol	HEP	Explanation of error	Source of HEP (NUREG/CR-1278)
A_1	3.75×10^{-3}	Improperly make a connection, including failure to test locking feature for engagement	Table 20-12 Item 13 Mean value (0.003, EF[1] = 3)
B_1	0.75	The operating repeating the actions is modeled to have a high dependency for making the same error again. It is not completely independent because the operator moves to the second lifting leg and must physically push the locking balls to insert the pins	Table 20-21 Item 4(a) High dependence for different pins. Two opportunities (the second and third pins) to repeat the error is modeled as 0.5+(1-0.5)*0.5 = 0.75
C_1	1.25×10^{-3}	Checker fails to verify proper insertion of the connector pins, and that the status affects safety when performing tasks	Table 20-22 Item 9 Mean value (0.001, EF = 3)
D_1	0.15	Checker fails to verify proper insertion of the connector pins at a later step, given the initial failure to recognize error. Sufficient separation in time and additional cues to warrant moderate rather than total or high dependency.	Table 20-21 Item 3(a) Moderate dependency for second check
F_1	5.2×10^{-7}	Failure rate if first pin improperly connected	$A_1 * B_1 * C_1 * D_1$
a_1	0.99625	Given first pin was improperly connected	
A_2	3.75×10^{-3}	Improperly make a connection, including failure to test locking feature for engagement	Table 20-12 Item 13 Mean value (0.003, EF = 3)
B_2	0.5	The operating repeating the actions is modeled to have a high dependency for making the same error again. It is not completely independent because the operator moves to the second lifting leg and must physically push the locking balls to insert the pins	Table 20-21 Item 4(a) High dependence for different pins. Only one opportunity for error (third pin)
C_2	1.25×10^{-3}	Checker fails to verify proper insertion of the connector pins, and that the status affects safety when performing tasks	Table 20-22 Item 9 Mean value (0.001, EF = 3)
D_2	0.15	Checker fails to verify proper insertion of the connector pins at a later step, given the initial failure to recognize error. Sufficient separation in time and additional cues to warrant moderate rather than total or high dependency.	Table 20-21 Item 3(a) Moderate dependency for second check
F_2	3.5×10^{-7}	Failure rate if first pin improperly connected	$a_1 * A_2 * B_2 * C_2 * D_2$
F_T	8.7×10^{-7}	Total failure due to human error	F1 + F2

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

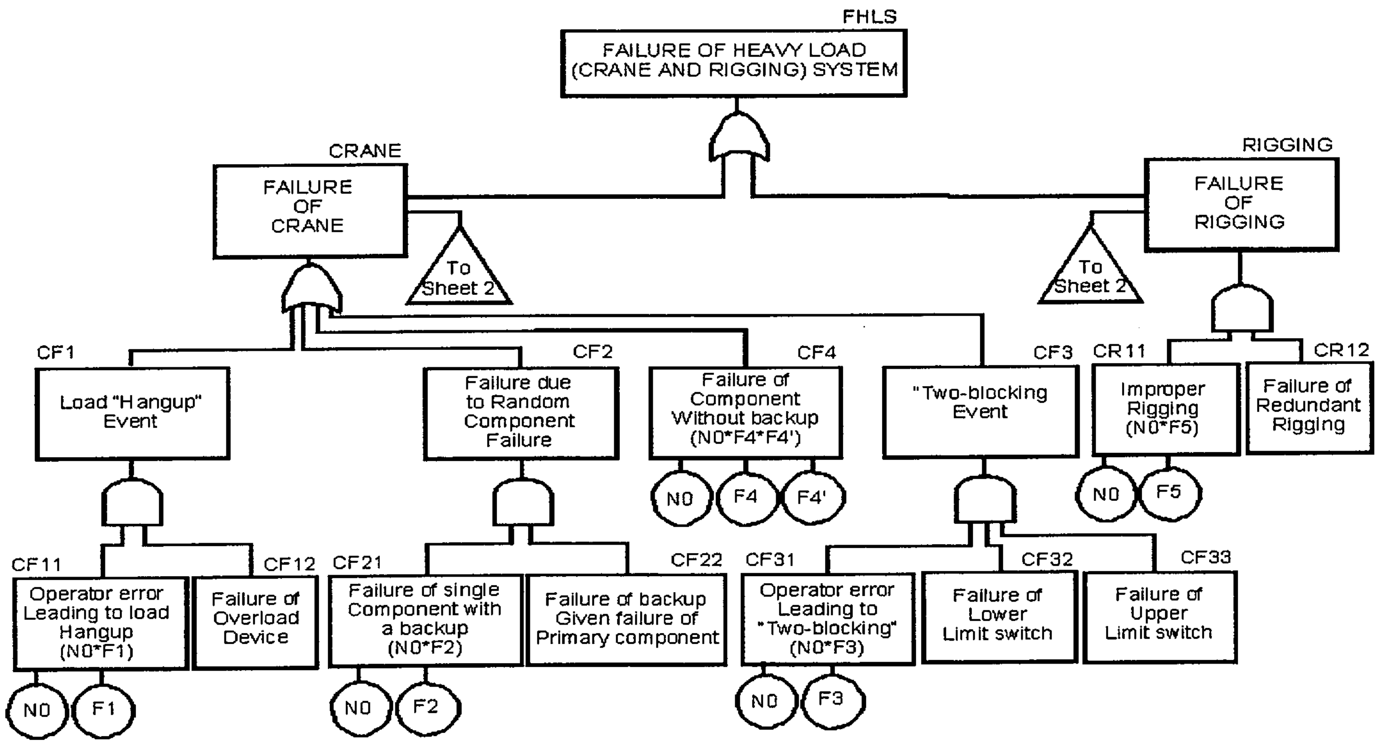
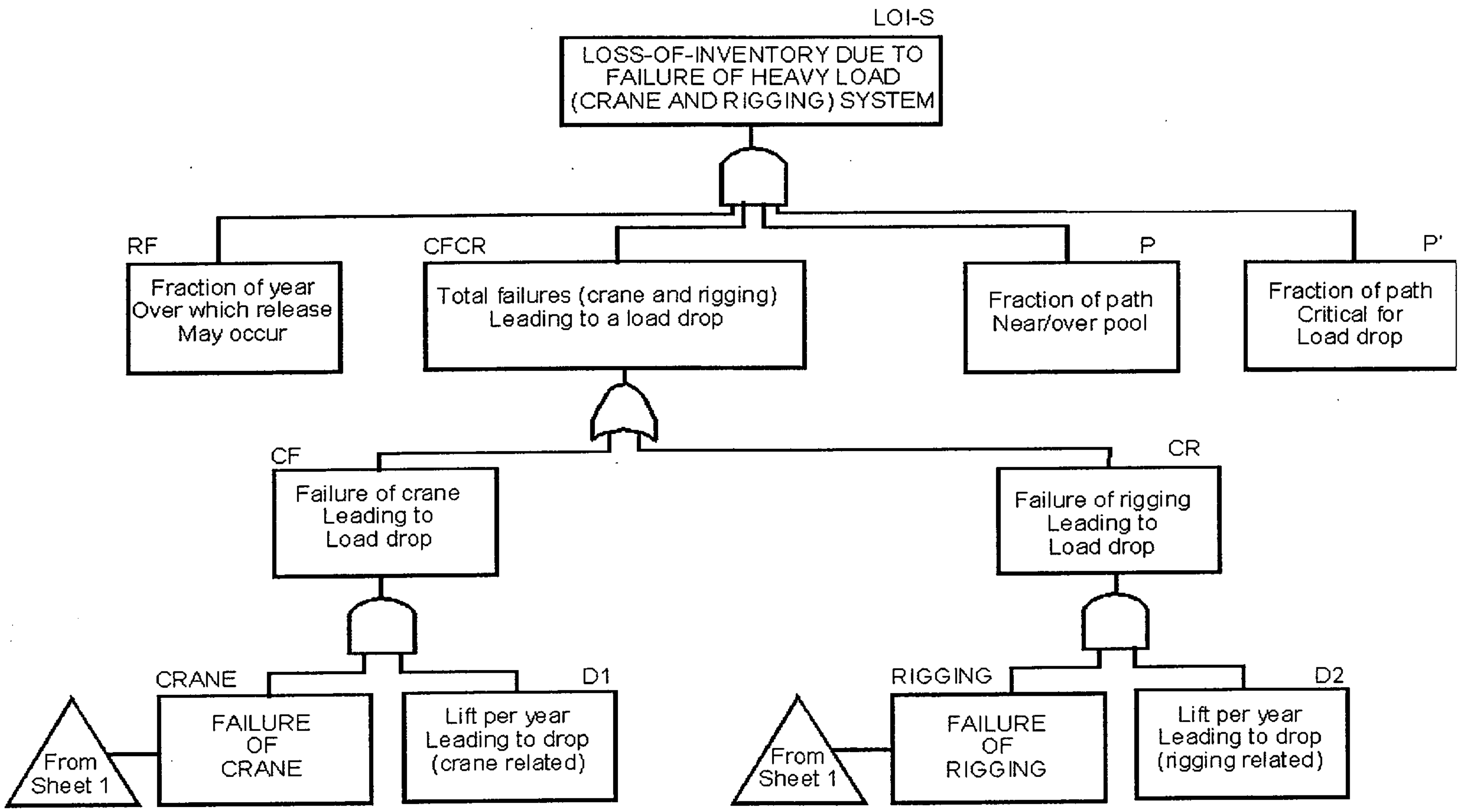


Figure A2c-1 (sheet 1 of 2) - Heavy Load Drop Fault Trees

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



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

1. Introduction

The mean frequency for significant PWR or BWR spent fuel pool damage resulting from a direct hit from an aircraft was estimated based on the point target model for a 100 x 50-foot pool to be 4.1×10^{-9} per year. The estimated frequency of loss of support systems leading to spent fuel pool uncovery is bounded by other initiators.

2. Analysis

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

The staff used the generic data provided in DOE-STD-3014-96 [Ref. 1] to assess the likelihood of an aircraft crash into or near a decommissioned spent fuel pool. Aircraft damage can affect the structural integrity of the spent fuel pool or the availability of nearby support systems, such as power supplies, heat exchangers, and make-up water sources, and may also affect recovery actions.

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

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

where:

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

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

and where:

A_{eff} = total effective target area
A_f = effective fly-in area
H= height of facility
L= length of facility

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

where:

Equation A2d-2

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

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

A_s = effective skid area
WS= wing span
cotθ= mean of cotangent of aircraft impact angle
W= width of facility
S= aircraft skid distance
R= length of facility diagonal

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

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

The results from the DOE effective aircraft target area model, using the generic data in Table A2d-1, were compared to the results of two evaluations reported in Reference 2. The first evaluation of aircraft crash hits was summarized by C.T. Kimura et al. in Reference 3. The DWTF Building 696 was assessed in the Kimura report. It was a 1-story 254-feet-long 80-feet-wide, 39-feet-high structure. The results of Kimura's study are given in Table A2d-3.

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

The second evaluation was presented in a paper by K. Jamali [Ref. 4] in which additional facility evaluations were summarized. For the Seabrook Nuclear Power Station, Jamali's application of the DOE effective aircraft target area model to the final safety analysis report (FSAR) data resulted in an impact frequency 2.4×10^{-5} per year. The Millstone Unit 3 plant area was reported as 9.5×10^{-3} square miles and the FSAR aircraft crash frequency as 1.6×10^{-6} per year. Jamali applied the DOE effective aircraft target area model to information in the Millstone Unit 3 FSAR. Jamali reported an impact frequency of 2.7×10^{-6} per year, using the areas published in the

FSAR and $2.3x10^{-5}$ per year, and using the effective area calculated the effective aircraft target area model.

When the generic DOE data in Table A2d-1 were used (for a 514 x 514 x 100-foot site), the estimated impact frequency range was $6.3x10^{-9}$ to $2.9x10^{-5}$ per year, with an average of $1.9x10^{-6}$ per year, for the point target area model. The effective aircraft target area model gave an estimated range of $3.1x10^{-8}$ to $2.4x10^{-4}$ per year, with an average of $1.6x10^{-5}$ per year.

A site-specific evaluation for Three Mile Island Units 1 and 2 was documented in NUREG/CR-5042 [Ref. 5]. The NUREG estimated the aircraft crash frequency to be $2.3x10^{-4}$ accidents per year, about the same value as would be predicted with the DOE data set for the maximum crash rate for a site area of 0.01 square miles.

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

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

3. Estimated Frequencies of Significant Spent Fuel Pool Damage

The frequency for significant PWR spent fuel pool damage resulting from a direct hit was estimated based on the point target model for a 100 x 50-foot pool with a conditional probability of 0.45 (large aircraft penetrating 5-ft of reinforced concrete) that the crash resulted in significant damage. This value (i.e., 0.45) is an interpolation from a table in NUREG/CR-0542 reproduced in Table A2d-4. If 1-of-2 aircraft are large and 1-of-2 crashes result in spent fuel uncovery, then the estimated range is $1.3x10^{-11}$ to $6.0x10^{-8}$ per year. The average frequency was estimated to be $4.1x10^{-9}$ per year.

The mean frequency for significant BWR spent fuel pool damage resulting from a direct hit was estimated to be the same as that for the PWR, $2.9x10^{-9}$ per year.

4. Support System Unavailability

The frequency for loss of a support system (e.g., power supply, heat exchanger, or make-up water supply) was estimated based on the DOE model, including wing and skid area, for a 400 x 200 x 30-foot area with a conditional probability of 0.01 that one of these systems is hit. The estimated value range was $1.0x10^{-6}$ to $1.0x10^{-10}$ per year. The average value was estimated to be $7.0x10^{-8}$ per year. This value does not credit on-site or off-site recovery actions.

As a check, we calculated the frequency for loss of a support system supply based on the DOE model, including wing and skid area, for a 10 x 10 x 10-foot structure. The estimated frequency range was $1.1x10^{-9}$ to $1.1x10^{-5}$ per year with the wing and skid area modeled, with the average estimated to be $7.3x10^{-7}$ per year. Using the point model, the estimated value range was $2.4x10^{-12}$ to $1.1x10^{-8}$ per year, with the average estimated to be $7.4x10^{-10}$ per year. This value does not credit on-site or off-site recovery actions.

5. Uncertainties

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

6. References

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

Table A2d-1 Generic Aircraft Data

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

Table A2d-2 Aircraft Hits Per Year

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

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

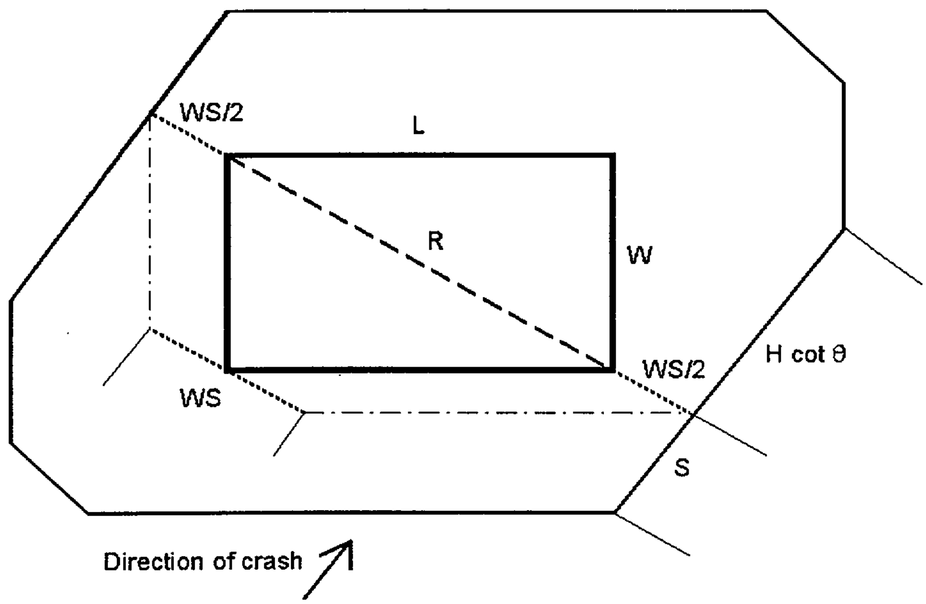
Period	Air Carriers	Air Taxes	General Aviation	Military Aviation	Total[1]
1995	1.72×10^{-7}	2.47×10^{-6}	2.45×10^{-5}	5.03×10^{-7}	2.76×10^{-5}
1993-1995	1.60×10^{-7}	2.64×10^{-6}	2.82×10^{-5}	6.47×10^{-7}	3.16×10^{-5}
1991-1995	1.57×10^{-7}	2.58×10^{-6}	2.89×19^{-5}	7.23×10^{-7}	3.23×10^{-5}
1986-1995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

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

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

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

Figure A2d-1 Rectangular Facility Effective Target Area Elements



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

1. Introduction

Tornado damage from missiles have the potential to affect the structural integrity of the spent fuel pool or the availability of nearby support systems, such as power supplies, cooling pumps, heat exchangers, and make-up water sources, and may also affect recovery actions. Department of Energy (DOE) studies indicate that the thickness of the spent fuel pool walls (greater than four feet of reinforced concrete) is more than sufficient protection from missiles that could be generated by the most powerful tornadoes ever recorded in the United States. In addition, the frequency of meeting or exceeding the wind speeds of F4 to F5 tornadoes (the most powerful tornadoes on the Fujita scale) is estimated to be on the order of $6x10^{-7}$ per year in the areas of the U.S. that are subject to the largest and most frequent tornadoes. The likelihood of meeting or exceeding the size tornado that could damage support systems is on the order of $2x10^{-5}$ per year. This is not the estimated frequency of fuel uncovery on a zirconium fire since the frequency estimate does not include credit for maintaining pool inventory from either on-site or off-site sources.

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

2. Analysis

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

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

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

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

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

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

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

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

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

where:

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

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

The tornado area, $<a>_T$, was then modified using the method described in NUREG/CR-2944 (based on Table 6b and 7b) to correct the area calculation by observations of the variations in a tornado's intensity along its path length and path width (see Figure A2e-3). Table A2e-2 gives the path-length correction data. Table A2e-3 gives the path-width correction data. The corrected effective area has a calculated $<a>_T$ of about 0.28 mi^2. The combined variation in intensity along the length and across the width of the tornado path is shown in Table A2e-4 (Table 15b from NUREG/CR-2944). For example, an F2 tornado with a path-length scale of 2 and a path-width scale of 3 has a calculated $<a>_T$ of about 0.28 mi^2. The total area is reapportioned using Table A2e-4 to assign 0.11 mi^2 to the F0 classification, 0.13 mi^2 to the F1 classification, and 0.04 mi^2 to the F2 classification.

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

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

4. Significant Pool Damage

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

5. Support System Availability

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

6. References

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

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

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

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

Table A2e-1 Tornado Characteristics

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

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

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

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

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

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

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

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

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

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

Table A2e-5 Exceedance Probability for Each F-scale

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

Table A2e-6 SPC Data Analysis Summary by State

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

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Table A2e-6 SPC Data Analysis Summary by State

State	NUREG/CR -2944 Region A	B	C	D	Years	Tornado F-scale F0	F1	F2	F3	F4	F5	Total	Point Strike Probability (per year) F0	F1	F2	F3	F4	F5	Land Area (mi²)
NV				X	34	41	8	0	0	0	0	49	2.9e-07	4.0e-08	0	0	0	0	109806
NH				X	45	24	34	15	2	0	0	75	4.7e-06	2.4e-06	4.7e-07	1.1e-08	0	0	8969
NJ			X		45	43	58	23	4	0	0	128	1.7e-05	6.6e-06	7.9e-07	7.1e-09	0	0	7419
NM			X		46	261	104	31	4	0	0	400	1.5e-06	5.2e-07	8.0e-08	1.1e-09	0	0	121365
NY				X	44	101	106	35	21	5	0	268	7.6e-06	6.1e-06	2.3e-06	8.8e-07	2.2e-07	0	47224
NC			X		46	153	321	143	44	26	0	687	1.5e-05	1.4e-05	4.9e-06	1.5e-06	2.5e-07	0	48718
ND			X		46	490	211	91	28	7	3	830	4.7e-06	3.2e-06	1.1e-06	3.6e-07	9.1e-08	1.1e-08	68994
OH	X				46	157	321	166	53	27	9	733	2.1e-05	1.8e-05	5.6e-06	1.3e-06	3.0e-07	2.8e-08	40953
OK	X				46	845	808	626	209	83	9	2580	4.1e-05	3.9e-05	1.4e-05	3.6e-06	7.0e-07	5.5e-08	68679
OR				X	45	31	15	3	0	0	0	49	2.9e-07	1.5e-07	3.1e-08	0	0	0	96003
PA			X		46	93	220	143	26	22	2	506	9.4e-06	9.0e-06	3.3e-06	9.3e-07	2.0e-07	5.4e-09	44820
RI			X		23	3	4	1	0	0	0	8	1.9e-05	1.3e-05	1.7e-06	0	0	0	1045
SC		X			46	136	234	100	31	15	0	516	1.9e-05	1.9e-05	6.8e-06	1.8e-06	3.0e-07	0	30111
SD		X	X		46	651	259	197	57	7	1	1172	9.7e-06	8.1e-06	3.0e-06	7.7e-07	1.5e-07	1.2e-08	75898
TN	X				46	107	241	139	76	29	4	596	2.2e-05	2.2e-05	8.3e-06	2.1e-06	2.0e-07	1.7e-10	41220
TX		X	X		46	2632	1837	1067	317	76	5	5934	1.6e-05	1.3e-05	4.3e-06	1.1e-06	1.8e-07	3.8e-09	261914
UT				X	43	53	19	6	1	0	0	79	5.1e-07	3.2e-07	1.0e-07	2.8e-08	0	0	82168
VT				X	41	7	14	12	0	0	0	33	3.3e-06	2.0e-06	3.4e-07	0	0	0	9249
VA			X		45	84	132	68	28	6	0	318	8.5e-06	7.0e-06	2.0e-06	4.4e-07	7.1e-08	0	39598
WA				X	41	24	17	12	3	0	0	56	4.9e-07	9.6e-08	2.3e-08	3.6e-09	0	0	66582
WV			X		45	27	36	16	8	0	0	87	2.2e-06	2.4e-06	9.7e-07	2.5e-07	0	0	24087
WI		X	X		46	204	378	276	62	24	5	949	2.6e-05	2.4e-05	7.9e-06	1.4e-06	2.5e-07	3.3e-08	54314
WY				X	46	247	145	43	8	1	0	444	2.5e-06	1.2e-06	3.1e-07	7.1e-08	1.9e-08	0	97105
Sum						13776	13251	7834	2553	924	121	38459							3536342

* DC was not included in the exceedance analysis.

Figure A2e-1

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

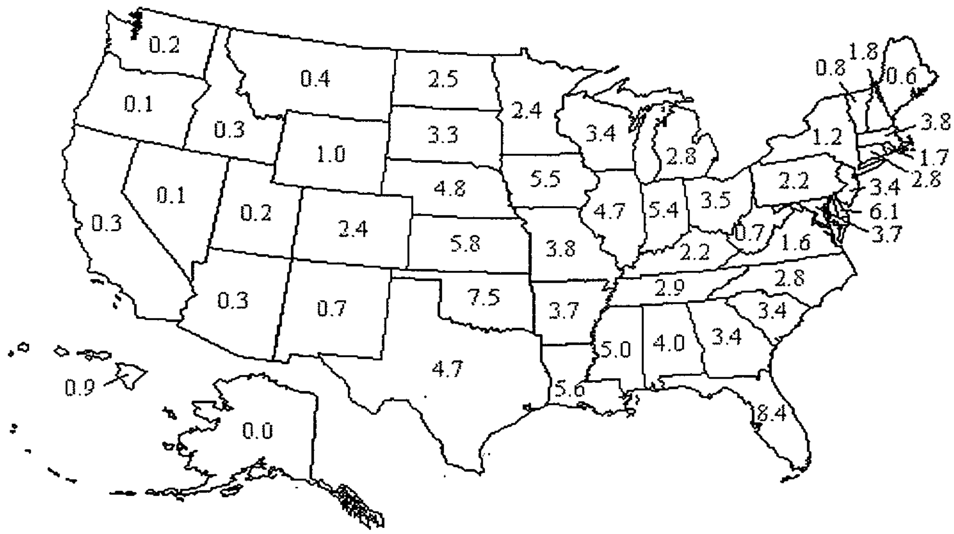


Figure A2e-2

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

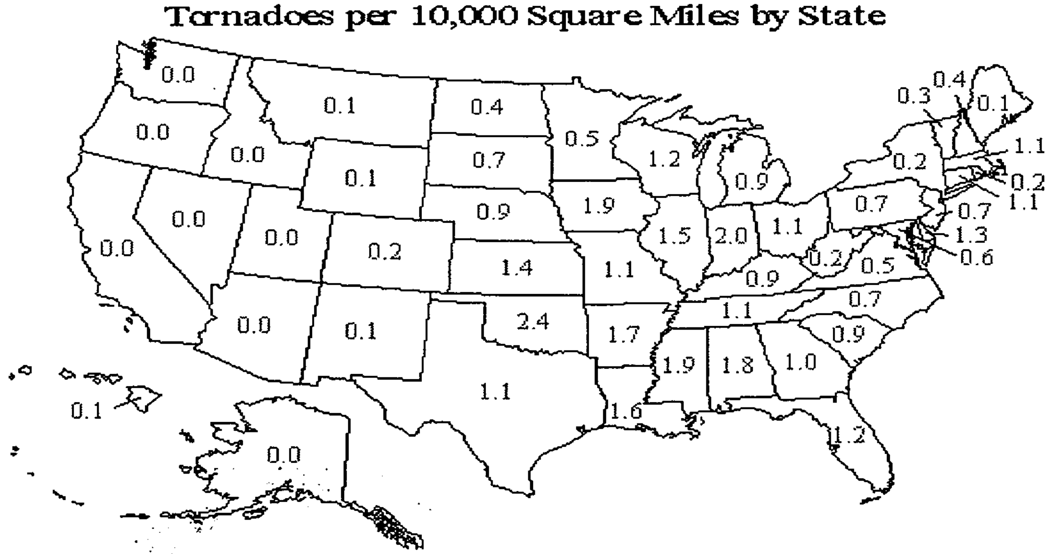


Figure A2e-3 Sketch of Hypothetical F2 Tornado Illustrating Variations

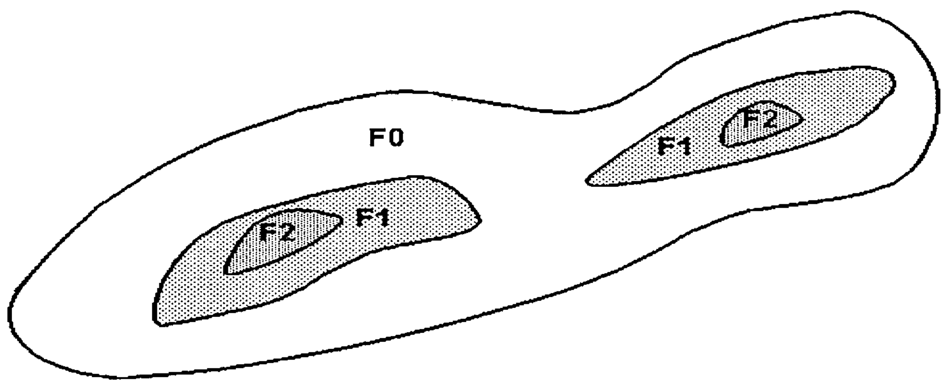


Figure A2e-4 Tornado Risk Regionalization Scheme (from NUREG/CR-2944)

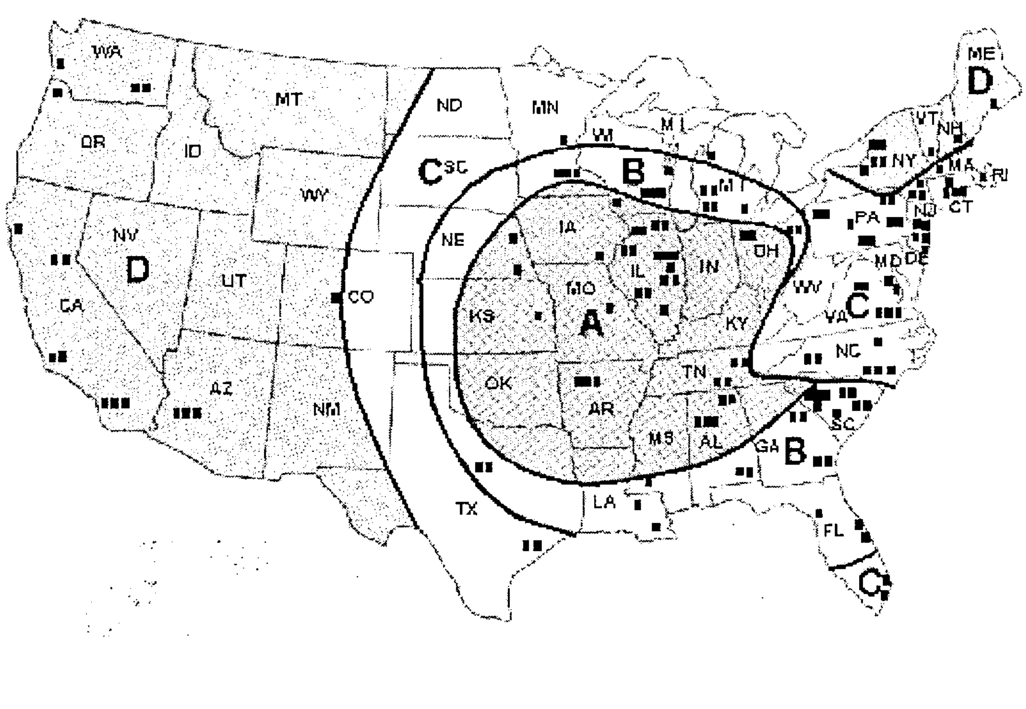
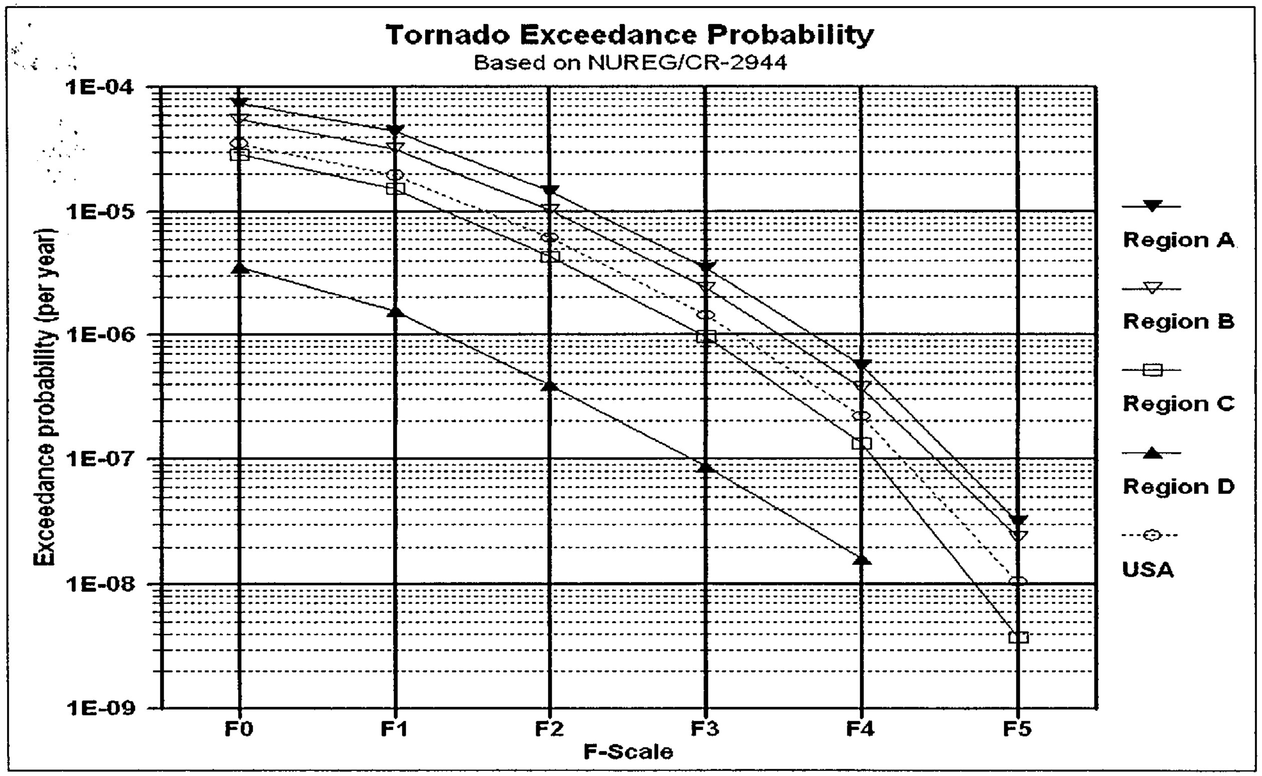


Figure A2e-5 Tornado Exceedance Probability For Each F-scale



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Appendix 3 Criticality

3.1 Introduction

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

3.2 Qualitative Risk Study

3.2.1 Criticality in Spent Fuel Pool

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

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

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

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

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

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

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

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

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

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

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

3.2.2 Deterministic Criticality Study

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

Assessment of the Potential for Criticality in Decommissioned Spent Fuel Pools

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Division of Systems Safety and Analysis

Introduction

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

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

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

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

Description Of Methods

The criticality analyses were performed with three-dimensional Monte Carlo methods using ENDF/B-V based problem specific cross sections (Ref. 1). Isotopic inventories were predicted using both one- and two-dimensional transport theory based methods with point depletion. SCALE 4.3 (Ref. 2) was used to perform the Monte Carlo, one-dimensional transport, cross section processing, and depletion calculations. Specifically, the staff used KENO-VI, NITAWL-1, BONAMI, XSDRN, and ORIGEN. The two-dimensional transport theory code NEWT (Ref. 3) was used for Boiling Water Reactor (BWR) lattice depletion studies. NEWT uses the method of characteristics to exactly represent the two-dimensional geometry of the problem. NEWT uses ORIGEN for depletion. Cross section data were tracked and used on a pin cell basis for the BWR assessments. The staff developed post processing codes to extract the information from NEWT and create an input file suitable for use with SCALE. Both the 238 and the 44 group ENDF/B-V based libraries were used in the project. Refer to Sample Input Deck at the end of Appendix 7 for a listing of one of the input decks used in this analysis. SCALE has been extensively validated for these types of assessments. (see References 4, 5, and 6)

Problem Definition

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

Discussion Of Results

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

PWR Spent Fuel Storage Racks

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

The results for the low density storage rack are given in Figure 2. As can be seen, criticality cannot be entirely ruled out on the basis of geometry alone. Therefore, we examined the conservatism implicit in the methodology and assessed whether there is enough margin to not require any additional measures for criticality control. There are two main sources of conservatism in the analyses; using fuel at the most reactive state allowed for the configuration and not crediting soluble boron. By relaxing the assumption that all of the fuel is at its peak expected reactivity, we have demonstrated by analyzing several sample storage configurations that the rack eigenvalue can be reduced to approximately 0.998 (see Table 1). The storage configurations analyzed included placing a most reactive bundle every second, fourth, sixth and eighth storage cell (see Figure 3). The assemblies used between the most reactive assembly were defined by burning the 5 w/o U_{235} enriched Westinghouse 15x15 assembly to 55 GWD/MTU which is a typical discharge burnup for an assembly of this type. This study did not examine all possible configurations so this value should be taken as an estimate only. However, the study does suggest that scattering the most reactive fuel throughout the pool would substantially reduce the risk of a criticality accident. It is difficult to entirely relax the assumption of no soluble boron in the pool, but its presence will allow time for recovery actions during an event that breaches the SFP liner and compresses the rack but does not rapidly drain the pool.

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

BWR Spent Fuel Storage Racks

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

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

Boraflex degradation is another problem that is somewhat unique to BWR spent fuel storage racks. This is true because of the fact that BWR storage pools do not contain soluble boron that provides the negative reactivity in PWR pools to offset the positive effect of Boraflex degradation. Therefore, some compensatory measures need to be in place to provide adequate assurance that Boraflex degradation will not contribute to a criticality event. In operating reactor spent fuel pools that use Boraflex, licensees use some sort of surveillance program to ensure that the 5 percent subcritical margin is maintained. These programs should be continued during and following decommissioning. No criticality calculations were performed for this study to assess Boraflex degradation because it is conservatively assumed that the loss of a substantial amount of Boraflex will most likely lead to a criticality accident.

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

Conclusions

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

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

References

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

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

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

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

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

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

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

Sample Input Deck Listing and
Tables and Figures

```
=csas26   parm=size=10000000
KENO-VI Input for Storage Cell Calc. High Density Poisoned Rack
238groupndf5  latticecell
'Data From SAS2H - Burned 5 w/o Fuel
o-16      1 0 0.4646E-01 300.00 end
kr-83     1 0 0.3694E-05 300.00 end
rh-103    1 0 0.2639E-04 300.00 end
rh-105    1 0 0.6651E-07 300.00 end
ag-109    1 0 0.4459E-05 300.00 end
xe-131    1 0 0.2215E-04 300.00 end
'xe-135   1 0 0.9315E-08 300.00 end
cs-133    1 0 0.5911E-04 300.00 end
cs-134    1 0 0.5951E-05 300.00 end
cs-135    1 0 0.2129E-04 300.00 end
ba-140    1 0 0.1097E-05 300.00 end
la-140    1 0 0.1485E-06 300.00 end
nd-143    1 0 0.4070E-04 300.00 end
nd-145    1 0 0.3325E-04 300.00 end
pm-147    1 0 0.8045E-05 300.00 end
pm-148    1 0 0.4711E-07 300.00 end
pm-148    1 0 0.6040E-07 300.00 end
pm-149    1 0 0.6407E-07 300.00 end
sm-147    1 0 0.3349E-05 300.00 end
sm-149    1 0 0.1276E-06 300.00 end
sm-150    1 0 0.1409E-04 300.00 end
sm-151    1 0 0.7151E-06 300.00 end
sm-152    1 0 0.5350E-05 300.00 end
eu-153    1 0 0.4698E-05 300.00 end
eu-154    1 0 0.1710E-05 300.00 end
eu-155    1 0 0.6732E-06 300.00 end
gd-154    1 0 0.1215E-06 300.00 end
gd-155    1 0 0.5101E-08 300.00 end
gd-156    1 0 0.2252E-05 300.00 end
gd-157    1 0 0.3928E-08 300.00 end
gd-158    1 0 0.6153E-06 300.00 end
gd-160    1 0 0.3549E-07 300.00 end
u-234     1 0 0.6189E-07 300.00 end
u-235     1 0 0.3502E-03 300.00 end
u-236     1 0 0.1428E-03 300.00 end
u-238     1 0 0.2146E-01 300.00 end
np-237    1 0 0.1383E-04 300.00 end
pu-238    1 0 0.4534E-05 300.00 end
pu-239    1 0 0.1373E-03 300.00 end
pu-240    1 0 0.5351E-04 300.00 end
pu-241    1 0 0.3208E-04 300.00 end
pu-242    1 0 0.1127E-04 300.00 end
am-241    1 0 0.9976E-06 300.00 end
am-242    1 0 0.2071E-07 300.00 end
am-243    1 0 0.2359E-05 300.00 end
cm-242    1 0 0.3017E-06 300.00 end
```

```
cm-244     1 0 0.6846E-06 300.00 end
i-135      1 0 0.2543E-07 300.00 end
'Zirc
cr         2 0 7.5891E-5   300.0 end
fe         2 0 1.4838E-4   300.0 end
zr         2 0 4.2982E-2   300.0 end
'Water w/ 2000 ppm boron
h2o        3 0.99 300.0 end
'b-10      3 0 2.2061E-5   300.0 end
'SS structural material
ss304      4 0.99 300.0 end
'Boral (model as b4c-al using areal density of b-10 @ -- g/cm^2 and 0.18 atom percent b-10 in
nat. b)
'Excluded Proprietary Information
end comp
'squarepitch card excluded - Proprietary Information
more data
dab=999
end more
read param
gen=103 npg=3000 xs1=yes pki=yes gas=yes flx=yes fdn=yes far=yes nb8=999
end param
read geom
'geom cards excluded - Proprietary Information
end geom
read array
ara=1 nux=15 nuy=15 nuz=1 fill
  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  1  1  2  1  1  2  1  1  1  2  1  1  2  1  1
  1  1  1  1  1  1  1  2  1  1  1  1  1  1  1
  1  1  1  1  2  1  1  1  1  1  2  1  1  1  1
  1  1  2  1  1  1  1  1  1  1  1  1  2  1  1
  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
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  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
end fill
end array
read bounds all=mirror end bounds
read mixt sct=2 eps=1.e-01 end mixt
read plot
scr=yes
ttl='w15x15 in High Density Rack'
xul=-11.5 yul= 11.5 zul=0.0
```

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

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

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

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

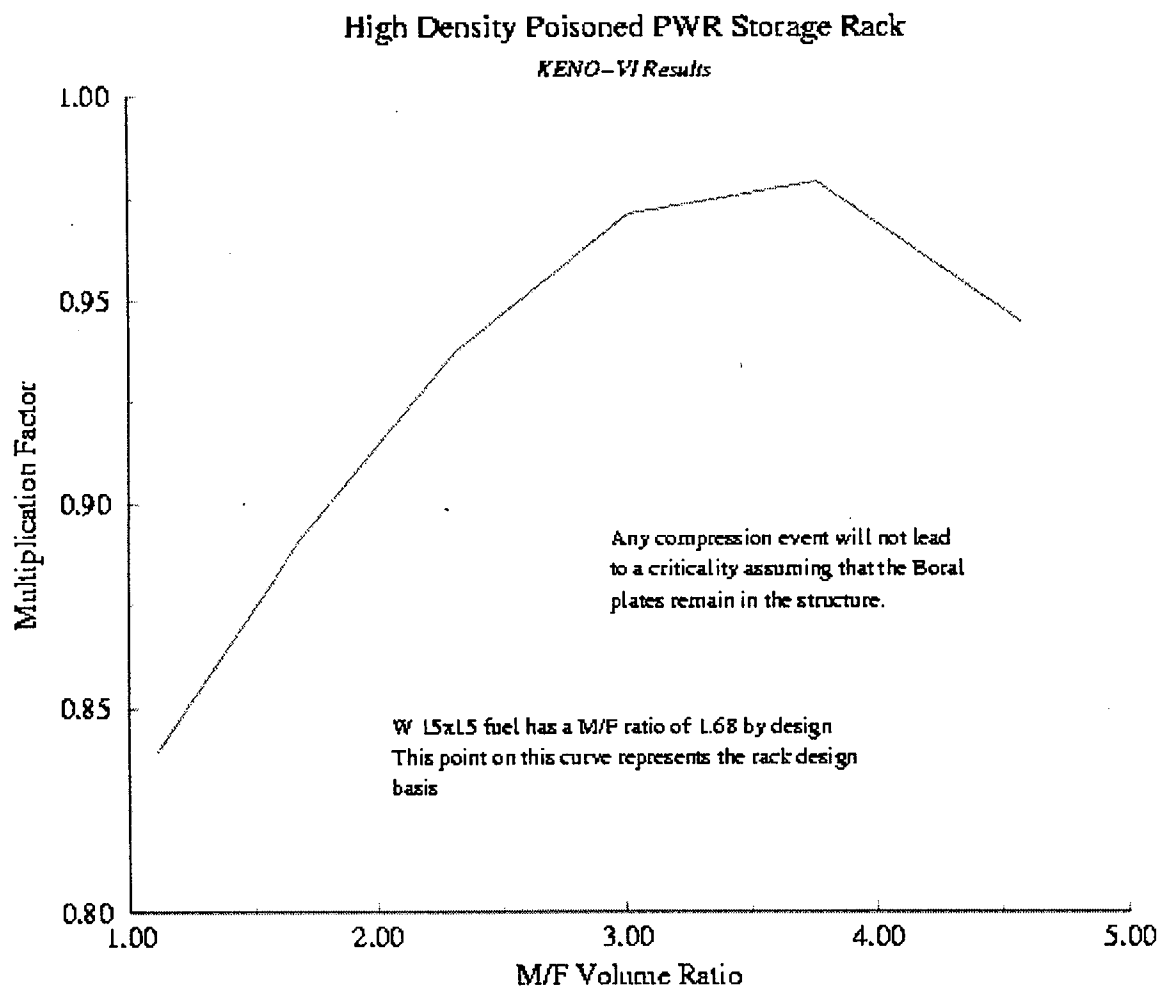
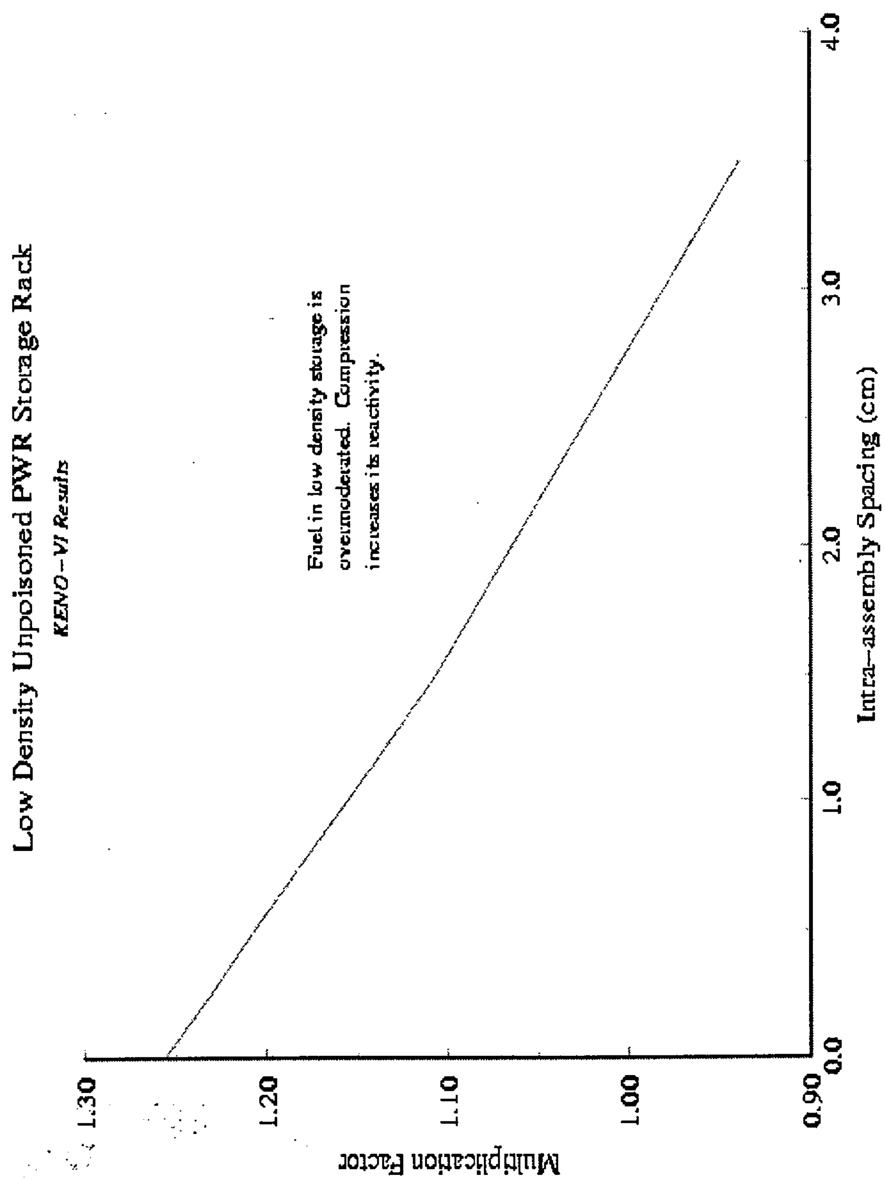


Figure 2 PWR Low Density Storage Rack Eigenvalue Following Compressive/Expansion Events



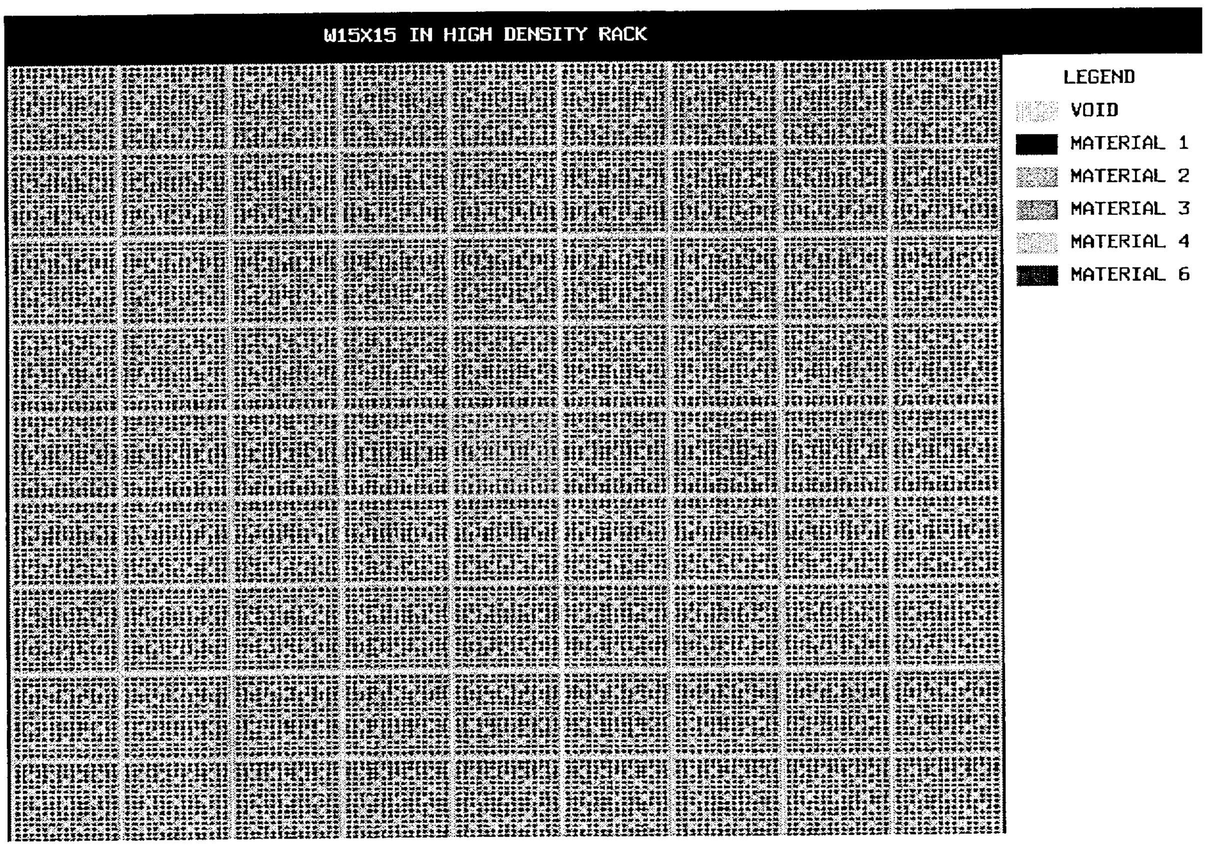


Figure 3 Sample Geometry Assuming 4 Assembly Spacing Between Most Reactive Assembly

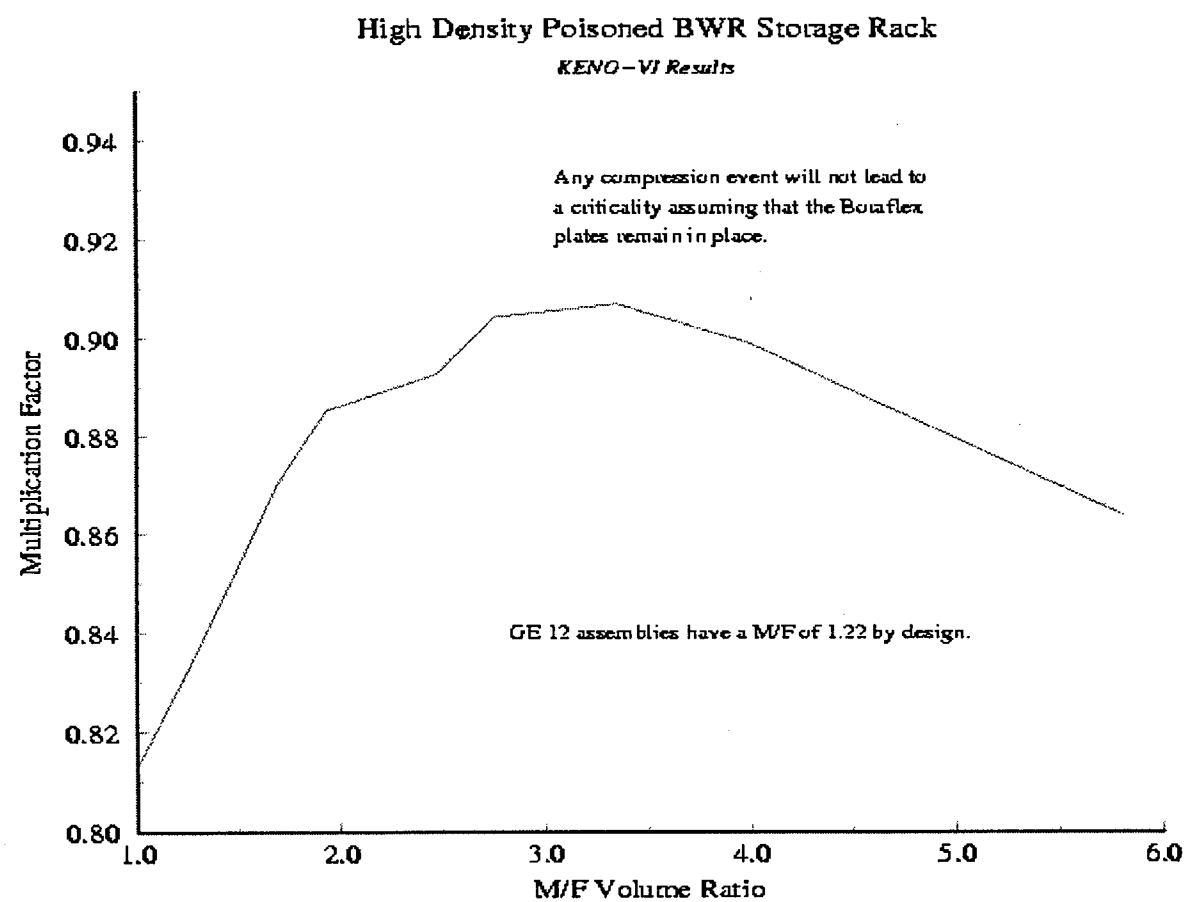
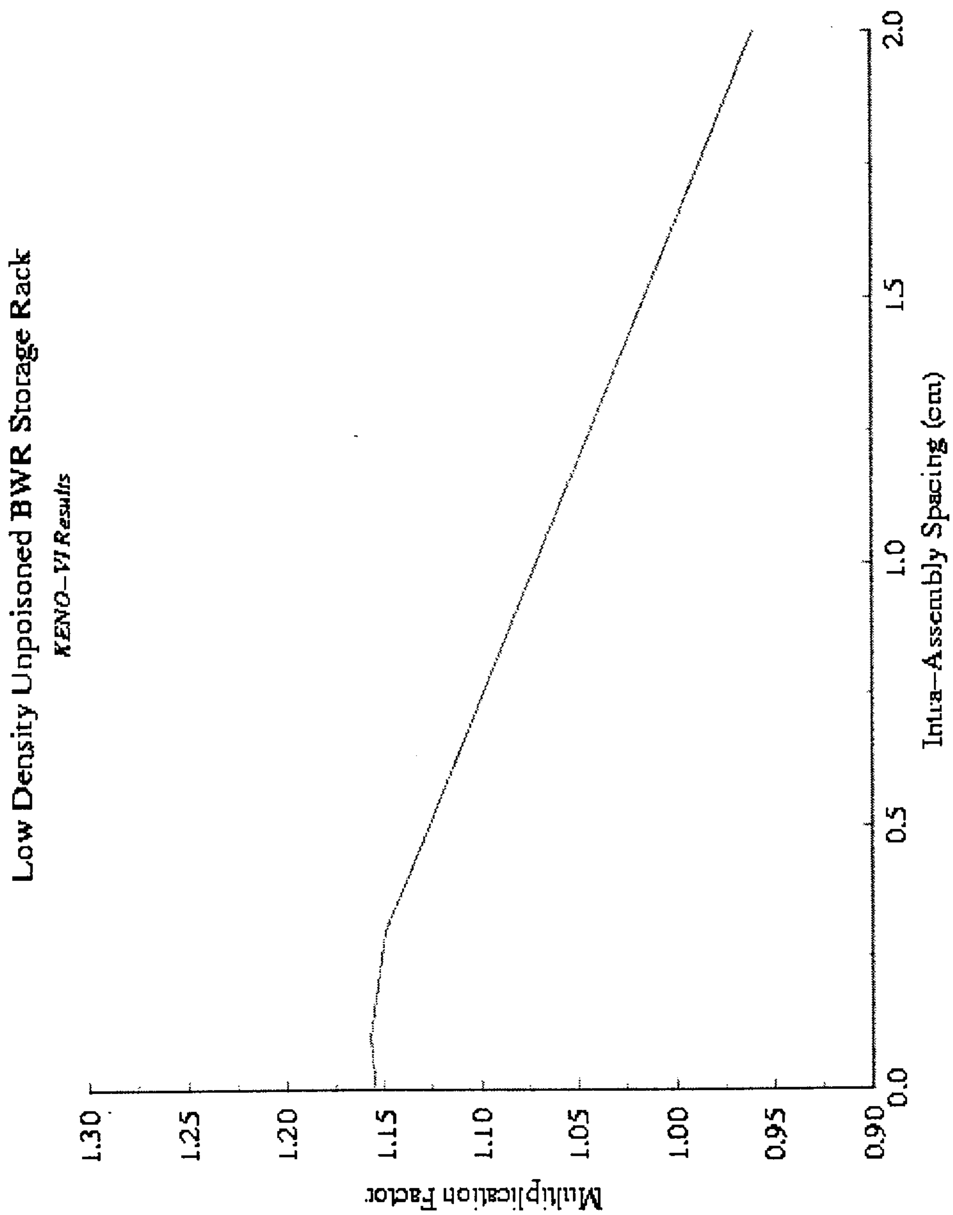


Figure 4 BWR High Density Storage Rack Eigenvalue Following Compressive/Expansion Events

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



Appendix 4 Consequence Assessment from Zirconium Fire

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

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

The current analysis used the MACCS code[3] (version 2) to estimate off-site consequences for a severe spent fuel pool accident. Major input parameters for MACCS include radionuclide inventories, radionuclide release fractions, evacuation and relocation criteria, and population density. The specification of values for these input parameters for a severe spent fuel pool accident is discussed below.

Radionuclide Inventories

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

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

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

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

Radionuclide	Half-Life	Spent Fuel Pool Inventory (Ci)		
		30 days after last discharge	90 days after last discharge	1 year after last discharge
Co-58	70.9d	2.29E4	1.26E4	8.54E2
Co-60	5.3y	3.72E5	3.15E5	2.85E5
Kr-85	10.8y	1.41E6	1.39E6	1.33E6
Rb-86	18.7d	1.01E4	1.05E3	3.84E-2
Sr-89	50.5d	8.39E6	3.63E6	8.33E4
Sr-90	28.8y	1.42E7	1.42E7	1.39E7
Y-90	28.8y	1.43E7	1.42E7	1.39E7
Y-91	58.5d	1.18E7	5.75E6	2.21E5
Zr-95	64.0d	1.94E7	1.00E7	5.10E5
Nb-95	64.0d	2.54E7	1.70E7	1.11E6
Mo-99	2.7d	1.49E4	3.12E-3	0
Tc-99m	2.7d	1.43E4	3.01E-3	0
Ru-103	37.3d	1.53E7	5.21E6	4.07E4

Ru-106	1.0y	1.72E7	1.53E7	9.13E6
Sb-127	3.8d	8.21E3	1.39E-1	0
Te-127	109d	2.21E5	1.45E5	2.52E4
Te-127m	109d	2.18E5	1.48E5	2.57E4
Te-129	33.6d	2.74E5	7.79E4	2.68E2
Te-129m	33.6d	4.21E5	1.20E5	4.12E2
Te-132	3.2d	3.74E4	8.64E-2	0
I-131	8.0d	1.22E6	6.35E3	0
I-132	3.2d	3.85E4	8.90E-2	0
Xe-133	5.2d	7.29E5	2.30E2	0
Cs-134	2.1y	7.90E6	7.47E6	5.80E6
Cs-136	13.2d	2.05E5	8.13E3	3.91E-3
Cs-137	30.0y	2.02E7	2.01E7	1.97E7
Ba-140	12.8d	5.19E6	1.90E5	6.41E-2
La-140	12.8d	5.97E6	2.19E5	7.37E-2
Ce-141	32.5d	1.32E7	3.61E6	1.03E4
Ce-144	284.6d	2.64E7	2.27E7	1.16E7
Pr-143	13.6d	5.44E6	2.41E5	1.90E-1
Nd-147	11.0d	1.54E6	3.36E4	1.10E-3
Np-239	2.4d	5.59E4	2.88E3	2.88E3
Pu-238	87.7y	4.51E5	4.53E5	4.54E5
Pu-239	24100y	8.89E4	8.89E4	8.89E4
Pu-240	6560y	1.30E5	1.30E5	1.30E5
Pu-241	14.4y	2.29E7	2.27E7	2.19E7
Am-241	432.7y	2.88E5	2.94E5	3.21E5
Cm-242	162.8d	1.45E6	1.12E6	3.50E5
Cm-244	18.1y	2.27E5	2.25E5	2.19E5

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

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

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

Release Fractions

NUREG/CR-4982 also provided the fission product release fractions assumed for a severe spent fuel pool accident. These fission product release fractions are shown in Table A4-3. NUREG/CR-6451 provided an updated estimate of fission product release fractions. The release fractions in NUREG/CR-6451 (also shown in Table A4-3) are the same as those in NUREG/CR-4982, with the exception of lanthanum and cerium. NUREG/CR-6451 stated that the release fraction of lanthanum and cerium should be increased from 1×10^{-6} in NUREG/CR-4982 to 6×10^{-6}, because fuel fines could be released off-site from fuel with high burnup. While the staff believes that it is unlikely that fuel fines would be released off-site in any substantial amount, a sensitivity was performed using a release fraction of 6×10^{-6} for lanthanum and cerium to determine whether such an increase could even impact off-site consequences.

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

Radionuclide Group	Release Fractions	
	NUREG/CR-4982	NUREG/CR-6451
noble gases	1	1
iodine	1	1
cesium	1	1
tellurium	2×10^{-2}	2×10^{-2}
strontium	2×10^{-3}	2×10^{-3}
ruthenium	2×10^{-5}	2×10^{-5}
lanthanum	1×10^{-6}	6×10^{-6}
cerium	1×10^{-6}	6×10^{-6}
barium	2×10^{-3}	2×10^{-3}

Modeling of Emergency Response Actions and Other Areas

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

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

Table A4-4 Timing of Events

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

Table A4-5 Evacuation Modeling

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

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

Off-site Consequence Results

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

Table A4-6 Cases Examined Using the MACCS2 Consequence Code

Case	Population Distribution	Radionuclide Inventory	Evacuation Start Time	La/Ce Release Fraction	Evacuation Percentage
Base Case	Surry	11 batches plus rest of last core	1.4 hours after release begins	1×10^{-6}	99.5%
1	Surry	11 batches plus rest of last core	1.4 hours after release begins	1×10^{-6}	95%
2	Surry	11 batches	1.4 hours after release begins	1×10^{-6}	95%
3	100 people/mi^2	11 batches	1.4 hours after release begins	1×10^{-6}	95%
4	100 people/mi^2	11 batches plus rest of last core	1.4 hours after release begins	1×10^{-6}	95%
5	100 people/mi^2	11 batches plus rest of last core	3 hours before release begins	1×10^{-6}	95%
6	100 people/mi^2	11 batches plus rest of last core	3 hours before release begins	6×10^{-6}	95%
7	100 people/mi^2	11 batches plus rest of last core	3 hours before release begins	1×10^{-6}	99.5%

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

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

Table A4-7 Mean Consequences for the Base Case

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
30 days	0-100	1.75	47,700	2,460
	0-500	1.75	571,000	25,800

90 days	0-100	1.49	46,300	2,390
	0-500	1.49	586,000	26,400
1 year	0-100	1.01	45,400	2,320
	0-500	1.01	595,000	26,800

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

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

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

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

Table A4-8 Mean consequences for Case 2

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

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

Table A4-9 Mean Consequences for Case 3

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

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

Table A4-10 Mean Consequences for Case 4

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities

30 days	0-100	18.3	53,500	2,610
	0-500	18.3	454,000	20,600
90 days	0-100	16.3	52,100	2,560
	0-500	16.3	465,000	21,100
1 year	0-100	12.7	50,900	2,490
	0-500	12.7	477,000	21,600

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

Table A4-11 Mean Consequences for Case 5

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

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

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

Table A4-12 Mean Consequences for Case 7

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

Comparison with Earlier Consequence Analyses

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

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

Table A4-13 Comparison of Analysis Assumptions

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

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

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

The NUREG/CR-6451 results included consequence estimates for societal dose, cancer fatalities, and prompt fatalities for a decommissioned reactor for a severe spent fuel pool accident occurring 12 days after the final shutdown. The Case 4 results for 30 days after final shutdown were compared against the NUREG/CR-6451 results, because (1) they included the entire last core in the spent fuel pool and (2) Case 4 had a uniform population density which could be easily adjusted to approximate that in NUREG/CR-6451. Differences between Case 4 and NUREG/CR-6451 included the population density, the amount of spent fuel in the pool, and the exclusion area size. To provide a more consistent basis to compare the NUREG/CR-6451 results with the Case 4 results, Case 4 was rerun using population densities, an amount of spent fuel, and an exclusion area size similar to NUREG/CR-6451.

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

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

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

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

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

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

Effect of Cesium

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

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

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

Conclusion

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

References:

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

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

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

August 25, 2000

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

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

SUBJECT: RISK-INFORMED REQUIREMENTS FOR DECOMMISSIONING

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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Lower	.005	$4x10^6$	2,000
Upper	.5	$8x10^6$	7,000

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

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

Introduction

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

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

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

Decay Time Prior to Accident	Mean Consequences (within 100 miles)		
	Early Fatalities	Societal Dose (person-rem)	Cancer Fatalities
30 days	1.75	$4.77x10^6$	2,460
1 year	1.01	$4.54x10^6$	2,320
1 year[a]	.0048	$4.18x10^6$	1,990

[a] Based on evacuation before release.

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

Fission Product Source Term

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

Table 2 Fission Product Release Fractions from the BNL Study

xenon, krypton	iodine	cesium	tellurium	strontium	barium	ruthenium	lanthanum	cerium
1	1	1	2×10^{-2}	2×10^{-3}	2×10^{-3}	2×10^{-5}	1×10^{-6}	1×10^{-6}

A subsequent review of the staff's spent fuel pool risk assessment indicated that significant air ingression, influencing fission product release, will occur in accidents involving quick drain-down, and the staff's consequence assessment should accommodate any reasonable uncertainty in the progression of the accident with the possible exception of an increase in the ruthenium release. The ruthenium release fraction used in the staff's consequence assessment was 2×10^{-5}. Small-scale Canadian experiments show that, in an air environment, significant ruthenium releases begin after the oxidation of 75% to 100% of the cladding, and that the ruthenium release fraction can be as high as the release fraction of the volatile fission products. However, in a spent fuel pool accident, rubbling of the fuel may limit the ruthenium release fraction to a smaller value than that of the volatile fission products.

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

Ruthenium:

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

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

Case	Population Density[b]	Ruthenium release fraction	Mean Consequences (within 100 miles)		
			Prompt Fatalities	Societal Dose (person-rem)	Cancer Fatalities
Base Case	Surry	2×10^{-5}	1.01	4.54×10^{6}	2,320
11	Surry	1	95.3	9.53×10^{6}	9,150
21	uniform	2×10^{-5}	9.33	5.05×10^{6}	2,490
22	uniform	1	134	9.46×10^{6}	6,490
13[a]	Surry	2×10^{-5}	.0048	4.18×10^{6}	1,990
14[a]	Surry	1	.132	6.75×10^{6}	6,300
15[a]	uniform	2×10^{-5}	.045	4.65×10^{6}	2,170
16[a]	uniform	1	.277	6.38×10^{6}	4,940

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

The Base Case calculation assumed that evacuation begins about an hour after the fission product release begins. However, Appendix 1 states that, after a year of decay, it will take a number of hours for the fuel with the highest decay power density to heat up to the point of releasing fission products in the fastest progressing accident scenarios. As a result, it is more likely to have evacuation before the release begins. Therefore, the Base Case calculation then was modified to begin the evacuation three hours before the fission product release begins. This modified Base Case is called Case 13. Starting with Case 13, sensitivity cases were run with a ruthenium release fraction of one and a uniform population density of 100 people/mile^2. The results of these cases (i.e., Cases 13, 14, 15, 16) are given in Table 3. For these cases, the effect of ruthenium is to increase the number of prompt fatalities by a factor of six to 30. The effect on societal dose and cancer fatalities is a more modest increase, with the largest effect being a factor-of-three increase in cancer fatalities for the Surry population density.

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

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

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

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

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

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

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

Table 6 Cases with Different Ruthenium-Group Isotopes Included

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

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

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

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

Fuel Fines:

The staff performed MACCS calculations with different fuel fines release fractions to assess the sensitivity of the consequences. The results of these calculations are shown in Table 8. Case

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

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

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

[a]Based on evacuation before release.

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

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

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

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

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

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

[a]Based on evacuation before release.

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

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

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

The results in Table 9 are the total number of early fatalities, societal dose, and cancer fatalities for the population within 100 miles of the facility. However, the NRC's quantitative health objectives are given in terms of individual risk of an early fatality within one mile and individual risk of a cancer fatality within ten miles. The MACCS results in terms of these two consequence measures are given in Table 10.

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

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

[a]Based on evacuation before release.

Amount of Fuel Releasing Fission Products:

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

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

Case	Population Density	Ruthenium Release Fraction	# of cores	Mean Consequences (within 100 miles)		
				Prompt Fatalities	Societal Dose (person-rem)	Cancer Fatalities
Base Case	Surry	$2x10^{-5}$	3.5	1.01	$4.54x10^{6}$	2,320
31	Surry	$2x10^{-5}$	1	.014	$3.23x10^{6}$	1,530
11	Surry	1	3.5	95.3	$9.53x10^{6}$	9,150
32	Surry	1	1	50.5	$7.25x10^{6}$	7,360
21	uniform	$2x10^{-5}$	3.5	9.33	$5.05x10^{6}$	2,490
33	uniform	$2x10^{-5}$	1	.177	$3.10x10^{6}$	1,480
22	uniform	1	3.5	134	$9.46x10^{6}$	6,490
34	uniform	1	1	103	$6.59x10^{6}$	4,960

For the cases with a ruthenium release fraction of $2x10^{-5}$, the reduction in prompt fatalities is caused by the reduction in the Cs-137 inventory which decreases from $8.38x10^{17}$ Bq to $2.11x10^{17}$ Bq in going from 3.5 cores to one core. This was confirmed by rerunning Case 33 with a Cs-137 inventory of $8.38x10^{17}$ Bq. The reductions in prompt fatalities for uniform and Surry population densities are factors of 52 and 72, respectively. These reductions are more than proportional to the factor-of-four reduction in Cs-137 inventory, because of the combined effects of individual risk of early fatality and non-uniform population density as discussed in the above analysis of the effect of ruthenium on offsite consequences.

For the cases with a ruthenium release fraction of one, the reduction in prompt fatalities is caused by the reduction in the Ru-106 inventory which decreases from $5.77x10^{17}$ Bq to $4.59x10^{17}$ Bq in going from 3.5 cores to 1 core. This was confirmed by rerunning Case 34 with a Ru-106 inventory of $5.77x10^{17}$ Bq. The reductions in prompt fatalities for uniform and Surry population densities are factors of 1.30 and 1.89, respectively. These reductions are nearly proportional to the factor of 1.26 reduction in the Ru-106 inventory. Again, deviations from being proportional are due to the combined effects of individual risk of early fatality and non-uniform population density. Overall, the effect of reducing the number of assemblies on prompt fatalities is less pronounced for the cases with a ruthenium release fraction of one, in part, because the additional 2.5 cores has a small amount of Ru-106 (one year half-life) in comparison with Cs-137 (30 year half-life). Finally, in all of the cases, the effect of reducing the amount of fuel releasing fission products from 3.5 cores to one core is a modest decrease (20 to 40%) in societal dose and cancer fatalities.

Plume-Related Parameters

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

Plume Heat Content:

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

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

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

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

[a]Based on evacuation before release.

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

Plume Spreading:

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

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

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

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

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

Conclusion

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

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

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

Other sensitivity calculations involved examining the effect of (1) decreasing the amount of fuel releasing fission products from the entire spent fuel pool inventory to the final core offload and (2) decreasing the percentage of the population evacuating from 99.5% and 95%. For cases with a small ruthenium release, the main effect of decreasing the amount of fuel releasing fission products was a large reduction in prompt fatalities. However, for cases with a large ruthenium release, the prompt fatalities did not change as much, because most of the ruthenium is in the final core offload due to its one-year half-life. With regard to the percentage of the population evacuating, the main difference between 99.5% and 95% evacuation is in the area of early fatalities for cases with evacuation before release. In these cases, the number of early fatalities increases by a factor of ten, because a change from 99.5% to 95% is a factor-of-ten increase in the number of persons not evacuating.

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

References

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

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

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

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

Attachment 2

Response to Public Comments on the Consequence Assessment

Public Comment #1:

Page 2, ACRS: The staff made additional MACCS calculations which assumed 100% release of the ruthenium inventory. For a 1 year decay time with no evacuation, the prompt fatalities increase by 2 orders of magnitude over those in the draft report which did not include ruthenium release. The societal dose doubled, and the cancer fatalities increased four-fold. [Ref. 11]

The staff has included, in Appendix 4A, the additional MACCS calculations with a large ruthenium release fraction. These calculations show an increase in consequences over the cases with the small ruthenium release fraction characteristic of fission product releases under steam conditions. However, the increased consequences resulting from a large ruthenium release are demonstrated to be largely offset by a consequence reduction due to early evacuation which is likely given the long time it takes for a spent fuel pool to heat up.

Public Comment #2:

Page 2, ACRS: The ACRS is concerned about the appropriateness of the source term used in the study. The staff did consider the possibility that "fuel fines" could be released from fuel with ruptured cladding (as a result of decrepitation). It did not, believe these fuel fines could escape from the plant site. Evidence suggests that fuel fines could be entrained in the vigorous natural convection flows produced in a SFP accident. Nevertheless, the staff considered the effect of 6×10^{-6} release fraction of fines. This minuscule release fraction did not affect the calculated findings. There is no reason to think that such a low release fraction would be encountered with decrepitating fuel. [Ref. 11]

The staff has included, in Appendix 4A, additional MACCS calculations with a fuel fines release fractions of .001 and .01. These calculations show a negligible to modest (less than 40%) increase in consequences.

Public Comment #3:

Page 3, ACRS: The uncertainties associated with many of the critical features of the MACCS code do not seem to have been considered in the analyses of the SFP accident. [Ref. 11]

> -One of the uncertainties is that the spread of the radioactive plume from a power plant site is much larger than what is taken as the default spread in the MACCS calculations.
> - The initial plume energy assumed in the MACCS calculations, which determines the extent of plume rise, was taken to be the same as that of a reactor accident rather than one appropriate for a zirconium fire.
> -The consequences found by the staff tend to overestimate prompt fatalities and underestimate latent fatalities just because of the narrow plume used in the MACCS calculations and the assumed default plume energy.

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

The staff has included, in Appendix 4A, additional MACCS calculations using different plume heat contents and updated values for the plume-spreading model parameters. The sensitivity calculations showed that increasing the plume heat content resulted in reductions in early fatalities and no change in societal dose or cancer fatalities. In addition, updating the values of the plume-spreading model parameters to those in NUREG/CR-6244 results in a decrease in early fatalities and up to a 60% increase in societal dose and cancer fatalities, because of the additional plume spreading associated with the updated values.

Public Comment #4:

Page 3, ACRS: The staff needs to review the air oxidation fission products release data from Oak Ridge National Laboratory and from Canada that found large releases of cesium, tellurium, and ruthenium at temperatures lower than 1000°C. Based on these release values for ruthenium, and incorporating uncertainties in the MACCS plume dispersal models, the consequence analysis should be redone. [Ref. 11]

The release values for ruthenium and the uncertainties in the MACCS plume dispersal models are discussed in the responses to Public Comment #1 and Public Comment #3, respectively. The consequence evaluation documented in Appendix 4 uses a cesium release fraction of one and a tellurium release fraction of .02. Also, the staff has included, in Appendix 4A, additional MACCS calculations using a tellurium release fraction of .75. No change in consequences were seen, because of the small inventories of the tellurium isotopes after one year of decay.

Public Comment #5:

Page 3, Mats Sjöberg/ Ferenc Müller on report, [Ref. 9]: Is a gap release considered to give moderate off-site consequences at the time when Zr-fire is no longer a threat?

NUREG/CR-4982, *Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82*, July 1987, provides societal doses for spent fuel pool accidents involving a fuel melt release and a gap release. These societal doses, which are for the population within 50 miles, are $3x10^6$ rem and 4 rem for a fuel melt release and a gap release, respectively. The NUREG/CR-4982 gap release includes releases of noble gases and iodine, but does not include releases of the less-volatile fission products. The fission product inventory used for the gap release case is for one year after final reactor shutdown. These societal dose results indicate that a gap release is expected to give negligible off-site radiological consequences at the time when rapid zirconium oxidation is no longer a threat.

In the Appendix 4A consequence assessment, a one-year decay time was used. However, the decay time for when rapid zirconium oxidation is no longer a threat is expected to be about five years. After five years of decay, the time available for mitigation, evacuation, and relocation will be much greater. An adiabatic heat-up calculation shows that, after five years of decay, fuel with a burn-up of 60 Gwd/t will take over a day to reach 600°C, the temperature at which it takes cladding 10 hours to rupture. Also, after five years of decay, the fission product inventory available for release will be much smaller. Finally, given the low decay power after five years, there may not be sufficient heat to carry released fission products out of the spent fuel pool and off-site. Based on these considerations, a gap release is expected to give negligible off-site radiological consequences at the time when rapid zirconium oxidation is no longer a threat.

Public Comment #6:

Orange County comment: Draft study does not address where people who have been relocated from uninhabitable land will reside while the land recover from radioactive contamination. Furthermore, the study does not explain the regulatory basis for using 4 rem over 5 years as the threshold dose for relocation **(RES to address)**. Finally, the study fails to address the social and economic implications of losing the use of thousands of square kilometers of land for several generations. [Ref. 8]

EPA 400-R-92-001, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*, May 1992, states that, after the early phase of a nuclear incident, protective actions should be taken to limit the dose received by an individual to 2 rem in the first year, .5 rem/year after the first year, and 5 rem over 50 years. These Protective Action Guides are implemented in the MACCS code by limiting the dose to 4 rem over 5 years, that is, 2 rem in the first year plus .5 rem for each of the second through fifth years.

Appendix 4B Pool Performance Guideline

Introduction

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

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

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

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

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

In reassessing the appropriateness of the 1E-5 per year PPG as discussed below, the staff contrasts the range of SFP accident consequences (early and latent health effects) reported in Appendices 4 and 4A with the consequences of the most risk-significant accidents evaluated in the NUREG-1150 study for Surry. The staff also compares the SFP risk for a licensee maintaining its facility at the PPG with the level of risk associated with reactor operation at the Surry site, and with the Commission's QHOs.

Comparison of Health Consequences

For at-power reactor accidents, the sequences that dominate early fatalities also tend to dominate latent cancer fatalities and population dose. These sequences generally involve early containment failure or containment bypass. Based on a survey of consequence results for the NUREG-1150 plants, early containment failure and containment bypass accident progression bins account for 80 to 100 percent of early fatalities and 60 to 80 percent of the latent cancer fatalities and population dose.

Using NUREG-1150 results for Surry (documented in NUREG/CR-4551 [3]) as a basis for comparison, early fatalities are dominated by interfacing system LOCA ("V") sequences. Steam generator tube rupture (SGTR) sequences with a stuck open secondary safety relief valve also lead to large releases but these releases occur after evacuation is complete and cause relatively few early fatalities. Consequence measures that depend on the total amount of radioactivity released (latent cancer fatalities and population dose) are dominated by V and SGTR sequences with a stuck open secondary safety relief valve.

Mean source terms for the frequency-dominant accident progression bins for each plant damage state are reported in Section 3.3 of NUREG/CR-4551. The source terms for the most probable wet and dry V sequence and SGTR sequence with a stuck open secondary safety relief valve are also identified. The "wet" V sequence represents sequences in which the break location is low enough in the auxiliary building that water escaping through the break would form a pool that would cover the break and scrub a significant portion of the release. The "dry" V sequence represents sequences in which this pool will not occur. These source terms were compared to the source terms resulting from the binning/partitioning process (Table 3.4-4 of NUREG/CR-4551) to identify the closest match. (This was done since consequence results are only reported in NUREG/CR-4551 for the source terms produced through the partitioning process.) The source terms for the most probable wet and dry V sequence and SGTR sequence with a stuck open secondary safety relief valve correspond closely with source terms SUR-03-3, SUR-05-3, and SUR-14-1, respectively, in NUREG/CR-4551. The mean consequence results for these source terms are provided in Table 1. Also provided in Table 1 are the reported consequences for the source terms that produced the greatest early fatalities and latent health effects in the internal events analysis (identified as source terms SUR-10-3 and SUR-10-1, respectively), and the source term that produced the greatest health effects in the seismic analysis (SRH-10-3). The NUREG-1150 study assumed that 99.5% of the population would be evacuated. However, for large earthquakes (greater than 0.5g) it was assumed that there would be no effective evacuation until 24 hours, at which time the population in the emergency response zone would be relocated.

It should be noted that the latent cancer fatality results reported in NUREG-1150 and NUREG/CR-4551 are based on an earlier cancer risk model than used in the SFP consequence calculations. The model used in the SFP calculations, described in NUREG/CR-6059 [4], results in about a factor of three increase in latent cancer fatalities relative to the earlier model. The other risk measures (early fatalities, and population dose) are also slightly higher. More recent calculations based on the later version of the MACCS code are reported in NUREG/CR-6349 for most of the NUREG-1150 reference plant source terms (for internal events). The results from these later calculations are cited where available. Otherwise, the latent cancer fatality results from NUREG-1150 were increased by a factor of three to provide a more meaningful comparison.

Briefly stated, the conditional number of early fatalities considered in NUREG-1150 study for the Surry plant varied from essentially zero to approximately 250, the population dose within 50 miles ranged from 1E6 to 1.1E7 person-rem, and the number of latent cancer fatalities ranged from about 2400 to 22000. Radiological consequences of seismic events are substantially greater than for internal events due largely to the ineffectiveness of emergency response in high acceleration earthquakes.

Appendices 4 and 4A of this report provide the results of offsite consequence calculations for a SFP fire occurring one year following reactor shutdown at a hypothetical 3441 MWth BWR spent fuel pool located at the Surry site. The calculations address the sensitivity of early and latent health effects to source terms, time of evacuation, percent of population participating in the evacuation, population distribution, number of cores participating in the SFP fire, and plume-related parameters.

Given the long delays to the onset of fission product release in SFP accidents, combined with the Industry Decommissioning Commitments (IDCs) and Staff Decommissioning Assumptions (SDAs) related to SFP instrumentation and offsite communication, the staff considers the consequence cases with early evacuation to be most representative for internally-initiated SFP accidents. Although 99.5% of the population was assumed to evacuate in NUREG-1150, this value may be somewhat optimistic, especially if existing EP requirements are relaxed, such as the requirement for notification systems. Accordingly, cases assuming reduced participation (i.e., 95% of the population) are considered more representative of an evacuation carried out on an ad hoc basis without the benefit of current radiological preplanning. For the large seismic events that dominate the frequency of SFP fires, it is expected that there would be extensive damage to the infrastructure needed for effective emergency response. As a result, evacuation would be ineffective regardless of radiological emergency planning, and the case with late evacuation would be more representative for these events.

The baseline calculation reported in Appendix 4 assumes the release fractions from NUREG/CR-4982 (including a ruthenium release fraction of 2E-5), the release of no additional "fuel fines", and the participation of essentially 3.5 cores. The baseline calculation assumed late evacuation (i.e., an evacuation start time of 1.4 hours after the beginning of the release), however, additional cases assuming earlier evacuation are also provided (i.e., an evacuation start time of 3 hours before the beginning of the release). The consequences for the baseline calculation with early and late evacuation of 99.5% of the population are provided in Table 1. The consequences for the baseline source term are well within the range of consequences predicted for large releases in an operating reactor accident for either evacuation time.

The consequence calculations presented in Appendix 4A show that when the ruthenium release fraction is increased from the original value of 2E-5 to a level equivalent to that for volatile fission products (cesium and iodine), the early and latent health effects increase considerably. Sensitivity cases with a 0.75 release of cesium, iodine and ruthenium and a 0.01 release of fuel fines were used for comparison. A release fraction of 0.75 is considered realistic for volatile isotopes and reflects the expectation that the combined effect of rubbling of the fuel, incomplete fission product release from parts of the assemblies, and fission product deposition would limit the release fraction of volatile fission products to less than 1.0. Rubbling of the fuel may limit the ruthenium to much less than 1.0. Thus, the 0.75 release of ruthenium is judged to be conservative.

The consequences for the large ruthenium release case with early and late evacuation of 95% of the population are provided in Table 1. These are identified as cases 46b and 45b respectively in Appendix 4A. The number of early fatalities increases by approximately two orders of magnitude, population dose increases by a factor of 2, and latent cancer fatalities increase by about a factor of 4 relative to the corresponding baseline calculations. For the case with early evacuation, early fatalities and population dose within 50 miles remain within the range considered in NUREG-1150, but latent cancer fatalities exceed the maximum values considered in NUREG-1150 by about 30%. For the case with late evacuation, the early fatalities and population dose within 50 miles are comparable to those for the worst seismic event considered in NUREG-1150. Long term risk measures are about a factor of 2 higher than the maximum values considered in NUREG-1150.

Consequences for the worst case SFP accident reported in Appendix 4A are also included in Table 1. This case, identified as case 45a, corresponds to a 1.0 release of the volatiles and ruthenium, a 0.01 release of fuel fines, and late evacuation of 95% of the population. Even with these high release fractions the early fatalities and population dose are comparable to the maximum values considered in NUREG-1150, and long term risk measures are about a factor of 2 higher than the maximum values considered in NUREG-1150.

Although the latent cancer fatality values mentioned above may appear large, they must be considered in perspective. The calculated latent fatalities from a nuclear accident occur throughout the entire region around the plant (1000 miles) and over several decades. The population within 1000 miles of the plant is about 160 million. Given the cancer fatality rate in the U.S. of about 1 in 500 per year, there would be about 300,000 deaths per year and 6 million deaths over a 2 decade period within the region from all other cancers. When spread over two or three decades, even tens of thousands of additional latent cancer fatalities are statistically indistinguishable from the background morbidity due to cancer fatalities from other causes.

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

Finally, in comparing the SFP consequences with those for a reactor accident at Surry it should be kept in mind that the NUREG-1150 results for Surry are for a power level of 2441 MWth, and that the SFP consequences will be overstated slightly due to the different power levels. Results for one case with a SFP decay heat level corresponding to a reactor power of 2440 MWth (values in brackets in Table 2) indicate that the latent health consequences would be about 30 percent lower than those based on 3441 MWth.

Comparison of Risk

The previous discussion provides a comparison of reactor and SFP accident consequences but does not address the relative frequency of these events. The quantitative assessment of risk involves combining severe accident sequence frequency data with corresponding offsite consequence effects. To provide insights into the relative levels of risk for reactor accidents versus SFP accidents, the staff compared the level of risk associated with reactor operation at Surry with the level of risk associated with a SFP fire in the hypothetical BWR spent fuel pool located at the Surry site. The contribution to reactor risk from both internal and seismic events were considered since these contributors were important in the SFP study. The aforementioned caveats regarding the differences in power level apply here as well.

The mean risk associated with power operation of the Surry plant, as estimated in the NUREG-1150 study, is reported in Table 2. These risk results reflect a frequency-weighted sum of the consequences of all releases -- severe as well as benign. Also included in Table 2 are estimates of the risk of a SFP fire. The SFP estimates were developed by assuming that the licensee maintains its facility consistent with the assumptions in the SFP study (i.e., the frequency of events leading to uncovery of the spent fuel is 3.4E-6 per year), and that the SFP fire results in one of the previously discussed release cases. Three different releases cases were considered, corresponding to: (1) the baseline releases with early evacuation, (2) a 0.75 release of cesium, iodine and ruthenium, 0.01 release of fuel fines, and early evacuation, and (3) a 1.0 release of cesium, iodine and ruthenium, 0.01 release of fuel fines, and late evacuation.

For the baseline release from a SFP accident, the early fatality risk results are about two orders of magnitude lower than for an internally-initiated reactor accident, due primarily to lower inventories of cesium and iodine in the SFP source term. Population dose is a factor of 2 higher for the SFP accident but latent cancer fatalities are comparable.

For the case with 0.75 release of cesium, iodine and ruthenium, 0.01 release of fuel fines, and early evacuation, the early fatality risk results are comparable to those for an internally-initiated reactor accident. Population dose and latent cancer fatalities for the SFP accident are about a factor of 4 higher than for internally-initiated events, due primarily to the larger quantities of long-lived radionuclides released, but are comparable to the results for seismic events which assume no evacuation.

For the case with 1.0 release of cesium, iodine and ruthenium, 0.01 release of fuel fines, and late evacuation, early fatalities, population doses, and latent fatalities are generally comparable to those for the worst seismically-initiated reactor accident. Although the source term for the SFP accident is larger than the reactor accident, this effect is partly offset by the late evacuation in the SFP case.

Even though the risk associated with a fire in the hypothetical SFP at Surry could be an order of magnitude greater than the risk of power operation at Surry, the individual health effect risks for a SFP accident would not exceed the Commission's QHOs. Comparisons of individual health effect risks with the QHOs are presented below.

<u>Comparison with Quantitative Health Objectives</u>

The Safety Goal Policy Statement expressed the Commission's policy regarding the acceptable level of radiological risk from nuclear power plant operation as follows:

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

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

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

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

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

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

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

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

Although the Policy Statement and related numerical objectives were developed to address the risk associated with power operation, is it reasonable to require that these objectives continue to be met for as long as nuclear materials remain on the plant site. Accordingly, the staff has compared the risks to an individual with the QHOs, assuming the licensee maintains the facility at the recommended PPG of 1E-5 per year.

The risk measures corresponding to the above numerical objectives were calculated by MACCS2 for each of the cases reported in Appendix 4 and 4A. The relevant risk measures are the early fatality risk to an average individual within 1 mile of the plant, and the latent cancer fatality risk to an average individual within 10 miles of the plant. These measures would not be significantly impacted by population density since they are determined on the basis of the risk to the average individual. The risk results are reported in Table 3 for the previously mentioned cases involving a 0.75 release of cesium, iodine and ruthenium and a 0.01 release of fuel fines (with early and late evacuation), and a 1.0 release of cesium, iodine and ruthenium and a 0.01 release of fuel fines with late evacuation (i.e., the worst case reported in Appendix 4A). For comparison with the numerical objectives, the staff assumed that the licensee maintains the facility at the recommended PPG of 1E-5 per year.

The risk results indicate that at a PPG of 1E-5 per year, the QHOs would continue to be met for even the worst case considered in Appendix 4A. The margins to both QHOs are substantial (about two orders of magnitude) for the case with early evacuation even with the large ruthenium release. The margins are considerably reduced in the late evacuation cases, but sufficient to conclude that the QHOs would be met given the bounding nature of these calculations.

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

Conclusions

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

In the above comparisons the SFP accident is assumed to occur one year following shutdown. The consequences of the accident would be markedly lower if it were to occur at a later time due to fission product decay. Specifically, after about 5 years the contribution from ruthenium would be virtually eliminated, and consequences would be dominated by cesium. Accordingly, the results reported for the baseline source term would be most representative for events occurring 5 years or beyond.

Although the above comparisons focus on the Surry site, the results are expected to be generally applicable to other sites as well. At higher population sites the SFP accident consequences would be higher, but the risk associated with reactor accidents would be proportionally higher as well. Thus, the results of the relative comparisons should remain valid. Similarly, the QHOs represent risk to the average individual within 1 mile and 10 miles of the plant, and should be relatively insensitive to the site-specific population.

References

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

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

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

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

Table 1 - Comparison of Health Consequences for Reactor and Spent Fuel Pool Accidents [1]

Consequence Measure	Consequences for Operating Reactor Accident[2] (Surry, NUREG-1150)						Consequences for SFP Accident One Year After Shutdown				
	Internal Events					Seismic Events	Baseline Source Term		Release of 0.75 Ru and 0.01 Fuel Fines		Worst Case
	SGTR[3] (SUR-14-1)	"V" - Wet[3] (SUR-03-3)	"V" - Dry[3] (SUR-05-3)	Worst EF[3] (SUR-10-3)	Worst LCF[4] (SUR-10-1)	Worst EF[4] and LCF (SRH-10-3)	Early Evac of 99.5% (Case 13)	Late Evac of 99.5% (Base)	Early Evac of 95% (Case 46b)	Late Evac of 95% (Case 45b)	Late Evac of 95% (Case 45a)
Early fatalities (EF)	0.017	0.23	2.7	15	0.84	249	0.005	1.0	0.54 [0.17]	55	103
Population dose within 50 miles (person-rem)	2.1E6	1.3E6	2.9E6	3.6E6	4.8E6	1.1E7	2.8E6	3.2E6	6.3E6 [5.1E6]	1.0E7	1.1E7
Latent cancer fatalities (LCF)	7850	2460	7930	11300	14300	21700	**1990**	**2320**	**6880 [4420]**	**10300**	**11700**

1 - Except where noted in brackets, consequence results for spent fuel pool accidents are based on a reactor power of 3441 MWth. Values in brackets are for a 2440 MWth reactor, equivalent to Surry.

2 - NUREG-1150 study assumed that 99.5% of the population would be evacuated. However, for large seismic events it was assumed that there would be no effective evacuation until 24 hours, at which time the population in the emergency response zone would be relocated.

3 - Based on results reported in NUREG/CR-6349.

4 - Based on results reported in NUREG/CR-4551. Values shown for latent cancer fatalities include a factor of three adjustment to account for differences in the cancer risk model used for NUREG-1150 and SFP accident calculations

Table 2 - Comparison of Risk Results for Reactor and Spent Fuel Pool Accidents [1]

Risk Measure	Risk for Operating Reactor Accident (Surry, NUREG-1150)			Risk for SFP Accident One Year After Shutdown (conditional on SFP source term and 5E-6 per year fire frequency)		
	Internal Events	Seismic Events [3]	Internal and Seismic	Baseline Release, Early Evac of 99.5% (Case 13)	Release of 0.75 Ru and 0.01 Fuel Fines, Early Evac of 95% (Case 46b)	Release of 1.0 Ru and 0.01 Fuel Fines, Late Evac of 95% (Case 45a)
Early fatalities (per year)	2.0E-6	9.3E-5	9.5E-5	2.4E-8	2.7E-6 [8.5E-7]	5.2E-4
Population dose within 50 miles (person-rem per year)	5.8	45	51	14	31 [25]	57
Latent cancer fatalities (per year) [2]	0.016	0.12	0.13	**0.010**	**0.034 [0.022]**	**0.059**

1 - Except where noted in brackets, consequence results for spent fuel pool accidents are based on a reactor power of 3441 MWth. Values in brackets are for a 2440 MWth reactor, equivalent to Surry.

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

3 - Based on Lawrence Livermore National Laboratory (LLNL) seismic hazard distributions

Table 3 - Comparison of Spent Fuel Pool Accident Risk One Year After Shutdown with Quantitative Health Objectives

Case	QHO for Individual Risk of Prompt Fatalities					QHO for Societal Risk of Latent Cancer Fatalities				
	Ind. Early Fatality Risk (per event)	PPG (events per year)	Prob of Early Fatality (per year)	QHO (per year)	% of QHO	Ind. Latent C. Fatality Risk (per event)	PPG (events per year)	Prob of Latent C. Fatality (per year)	QHO (per year)	% of QHO
0.75 Ru w/ fuel fines, early evac of 95% (Case 46b)	1.40E-3	1E-5	1.40E-8	5E-7	3	2.55E-3	1E-5	2.55E-8	2E-6	1
0.75 Ru w/ fuel fines, late evac of 95% (Case 45b)	3.23E-2	1E-5	3.23E-7	5E-7	65	4.98E-2	1E-5	4.98E-7	2E-6	25
1.0 Ru w/ fuel fines, late evac of 95% (Case 45a)	3.66E-2	1E-5	3.66E-7	5E-7	73	5.16E-2	1E-5	5.16E-7	2E-6	26

Appendix 5 November 12, 1999 Nuclear Energy Institute Commitment Letter

NEI

NUCLEAR ENERGY INSTITUTE

Lynnette Hendricks
DIRECTOR
PLANT SUPPORT
NUCLEAR GENERATION DIVISION

November 12, 1999

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

Dear Mr. Barrett,

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

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

I am hereby transmitting those industry commitments as follows.

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

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

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

4. An off-site resource plan will be developed which will include access to portable pumps and emergency power to supplement on site resources.

The plan would principally identify organizations or suppliers where off site resources could be obtained in a timely manner.

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

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

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

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

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

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

If you have any questions regarding industry's commitments, please contact me at 202 739-8109 or LXH@NEI.org.

Sincerely,

Lynnette Hendricks
LXH/1rh

Appendix 6 Stakeholder Concerns Raised During the Public Comment Period

On February 15, 2000, the Nuclear Regulatory Commission (NRC) released the "Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Plants," for public comment. The NRC encouraged stakeholders to review the draft report and to formally submit comments for review. Appendix 7 of that report included a list of public meetings and how the staff addressed stakeholder comments received on the draft report, issued June 1999, in various technical areas. After review of the February draft final report, several public groups commented that it appeared that the NRC did not address some of the public's comments. While all stakeholder comments were considered and many resulted in changes in the study, the staff did not include a discussion for some of the comments in Appendix 7. In order to ensure that adequate consideration had been given to public comments, the staff reviewed comments which had been received prior to February 15, 2000, as well as comments received as a result of a review of the draft final report. Comments received prior to February 15 were identified by reviewing transcripts of publically attended meetings, letters from the public, and other available documentation related to the staff's efforts in completing the draft final report.

This appendix provides the NRC's responses to the comments and concerns received as described above. In most cases, responses are documented in this appendix. However, in other cases, comments or concerns identified in this appendix are referred to other parts of the report where the identified issues are addressed. For cases where similar comments were received by more than one commenters, the comments were combined for one response. The comments are grouped in the following technical categories: Criticality, Consequences, Probability and Human Reliability, Seismic, Security/Safety Culture/EP, Thermal hydraulics, Insurance, and Rulemaking/ NRC Process Concerns.

CRITICALITY

DLPM Public Comment #5g: A member of the public stated that SECY 99-168 doesn't cover all decommissioning issues. The commenter stated that potential criticality should be addressed.

Response: The issue of nuclear criticality is addressed in **Section 3.4.4** in the body of the report.

SRXB Public Comments #1, 3, and 4: A public commenter raised several concerns related to SFP criticality. (1) Can a criticality occur due to chemical stripping of primary piping? (2) During primary system decontamination, can contaminated solution go "overboard" and into public waters? (3) During primary system decontamination at decommissioning reactors, is it possible to misalign the valves and send corrosive chemicals into the SFP? Could these chemicals precipitate boron from the SFP water? Is there a potential for criticality? Is there a potential for fuel damage?

Response: The precipitation of boron out of the pool water, due to chemicals or any other means, will not increase in criticality risk because soluble boron is not credited to maintain spent fuel pool subcriticality (k-eff < 1.0). Consideration of such things as chemical intrusion into the spent fuel pool or offsite discharge pathways will be considered when the staff reviews the plant specific decommissioning plans. Main plant buildings have a drain system in the event of any liquid spillage to prevent contamination of public waters or land.

SRXB Public Comment #6: The NRC should identify the scenario where a steam explosion is possible because of a severe criticality event and the basis upon which the probability was determined to be "highly unlikely."

Response: The discussion in the paper was intended to mean that a steam explosion from a super-prompt critical event is highly unlikely not because of the low probability of the scenario, but because of the fact that inherent negative feedback in the fuel would prevent a super-prompt critical event in all load drop scenarios, which are themselves of a low probability. Super prompt critical events are those in which the reactor power changes very rapidly. When we speak of inherent negative feedback we are referring to feedback caused by fuel temperature increases which is always negative. In other words, for every increase in reactivity (which manifests itself as power) the increase in fuel temperature will attempt to shut the reaction down.

SRXB Public Comment #7: The NRC should identify all radioactivity in the SFP and that capable of being dispersed in an accident (beyond that on p A3-11 to A3-13).

Response: The information supplied in pages A.3-11 to A.3-13 does not relate to the generation of the source term. These nuclides were selected because they contribute to the reactivity of the spent fuel. The nuclides listed there represent well over 90 percent of the reactivity contribution in spent fuel. Therefore, it is not necessary to expand the list because such an expansion will not significantly alter the predicted reactivity of the spent fuel in the storage racks. The source term is addressed in detail in Appendix 4.

SRXB Public Comment #8: The criticality accident analysis does not consider the risk of a criticality accident that arises from placement of low-burnup fuel assemblies in a pool where the licensee relies on burnup credit to prevent criticality.

Response: The double contingency principle discussed in ANS 8.1, "Nuclear Criticality Safety in Operations with Fissionable Material Outside Reactors" which has been endorsed by the staff, requires that only the worst highly unlikely single failure or event needs to considered in a criticality evaluation. The staff considers fuel misloading events to be highly unlikely and has demonstrated via analysis (affidavit of A. Ulses in hearing before the Atomic Safety and Licensing Board, ASLBP No. 99-762-02-LA, January 4, 2000) that the worst possible misloading scenario will not lead to a criticality event. Therefore, further consideration is not needed.

CONSEQUENCES

RES Public Comment #5:
Is a gap release considered to give moderate off-site consequences at the time when Zr-fire is no longer a threat?

Response: NUREG/CR-4982, *Severe Accidents in Spent Fuel Pools in Support of Generic Safety Issue 82*, July 1987, provides societal doses for spent fuel pool accidents involving a fuel melt release and a gap release. These societal doses, which are for the population within 50 miles, are $3x10^6$ rem and 4 rem for a fuel melt release and a gap release, respectively. The NUREG/CR-4982 gap release includes releases of noble gases and iodine, but does not include releases of the less-volatile fission products. The fission product inventory used for the gap release case is for one year after final reactor shutdown. These societal dose results indicate that a gap release is expected to give negligible off-site radiological consequences at the time when rapid zirconium oxidation is no longer a threat.

In the Appendix 4A consequence assessment, a one-year decay time was used. However, the decay time for when rapid zirconium oxidation is no longer a threat is expected to be about five years. After five years of decay, the time available for mitigation, evacuation, and relocation will be much greater. An adiabatic heat-up calculation shows that, after five years of decay, fuel with a burn-up of 60 Gwd/t will take over a day to reach 600°C, the temperature at which it takes cladding 10 hours to rupture. Also, after five years of decay, the fission product inventory available for release will be much smaller. Finally, given the low decay power after five years, there may not be sufficient heat to carry released fission products out of the spent fuel pool and off-site. Based on these considerations, a gap release is expected to give negligible off-site radiological consequences at the time when rapid zirconium oxidation is no longer a threat.

RES Public Comment #6:
The draft study does not address where people who have been relocated from uninhabitable land will reside while the land recover from radioactive contamination. Furthermore, the study does not explain the regulatory basis for using 4 rem over 5 years as the threshold dose for relocation **(RES to address)**. Finally, the study fails to address the social and economic implications of losing the use of thousands of square kilometers of land for several generations.

Response: EPA 400-R-92-001, *Manual of Protective Action Guides and Protective Actions for Nuclear Incidents*, May 1992, states that, after the early phase of a nuclear incident, protective actions should be taken to limit the dose received by an individual to 2 rem in the first year, .5 rem/year after the first year, and 5 rem over 50 years. These Protective Action Guides are implemented in the MACCS code by limiting the dose to 4 rem over 5 years, that is, 2 rem in the first year plus .5 rem for each of the second through fifth years.

PROBABILITY AND HUMAN RELIABILITY ASSESSMENT

SPSB Public comment # 1: Experience at nuclear power plants demonstrates that safety problems are not caused by workers making mistakes or by not following procedures. Problems are caused by bad management.

Response: The staff agrees that utility safety culture and utility oversight/expectations in the day-to-day operations of a facility are important contributors to either a well run plant or a poorly run one. The staff is proposing that utilities with decommissioning sites develop a process that will help insure that proper attention, regardless of management, is given to spent fuel pool status, procedures are developed that guide fuel handlers in the event of a spent fuel pool accident, communications are established between onsite and offsite organizations, and cask drop analyses are performed or a single failure proof crane is used for handling very heavy loads. These prescriptions and commitments are discussed in Sections 3.2, 3.3.1, 3.3.6, 4.2.1, 4.2.4, and Appendix 6 of the Draft Final Technical Study. Additionally, while it is not Commission policy to assess plant management, actions taken in the past illustrate the willingness of the NRC to evaluate plant management as necessary. As appropriate, the NRC will evaluate plant management at decommissioning facilities.

SPSB Public comment #2: Experience at nuclear power plants shows that multiple shifts can make the same error and not recognize it for a long time. With watching the pool being their major responsibility, a fuel handler's life would be very tedious and boredom would set in. This should result in a poorer response by the fuel handler in the event of an accident.

Response: The Commission, through the "Policy on Factors Causing Fatigue of Operating Personnel at Nuclear Reactors" provides guidelines on working hours that were consistent with the objective of ensuring that the mental alertness and decision-making abilities of plant staff were not significantly degraded by fatigue. The staff shares the commenter's concern that operator boredom and their ability to maintain alertness while standing watch may contribute to fatigue-induced impairment of personnel and thereby increase the likelihood of personnel errors. For this study, our modeling and quantification of spent fuel pool risk includes consideration of multiple shift turnovers and the chance that shift after shift makes the same mistake. However, for almost all postulated SFP accidents, there is a very long time available to the fuel handlers to discover and recover from the existence of a problem in the spent fuel pool or its support systems. The staff believes that the commitments made by the industry and the NRC's staff decommissioning assumptions provide a basis for reducing the chances of multiple shift errors to the point where they do not contribute significantly to the overall risk of spent fuel pool operation (See Sections 3.2, 3.3.1, 3.3.6, 4.2.1, 4.2.4, and Appendix 6 of the Draft Final Technical Study). The rest of the accidents (i.e., seismic and heavy load drop), which progress rapidly, proceed independent of operator intervention once the accident has occurred because the SFP is drained so rapidly.

SPSB Public comment #3: Over time, tedious tasks will cause workers to make mistakes. The NRC needs to address this in a conservative manner.

Response: The staff agrees that tedious tasks can increase the chances of a fuel handler making a careless mistake. The Commission, through the "Policy on Factors Causing Fatigue of Operating Personnel at Nuclear Reactors" provides guidelines on working hours that were consistent with the objective of ensuring that the mental alertness and decision-making abilities of plant staff were not significantly degraded by fatigue. However, we do not agree that fuel handler errors need be handled in a conservative manner when performing a probabilistic risk assessment. It is the NRC's policy to make its risk assessments as realistic as possible. The staff performed the analyses for this report consistent with the agency's policy.

SPSB Public comment #4: How is common mode failure accounted for in the staff's risk analysis? How confident are you of your ability to model and quantify common mode failures?

Response: The staff's risk analysis accounts for dependencies among the initiating events, the equipment needed to mitigate the events, and also the operator actions needed for accident mitigation. Initiating events that have the potential of simultaneously degrading mitigating equipment or impeding operator actions are modeled in the construction of the event trees and in the estimation of equipment failure rates and human failure probabilities. For example, for an event where a fire is not extinguished within 20 minutes, it was assumed that the SFP cooling system and the electric-driven firewater pumps are failed (either due to fire damage or due to loss of the electrical supply to the plant). Therefore, no credit is taken for this equipment. In addition, the estimation of the human error probability (for starting backup diesel pumps or for offsite recovery) took into account a high level of operator stress, which increases the failure probability.

Equipment hardware failure dependencies, usually referred to as common cause failures, have also been modeled in the risk analysis. Since these failures have the potential for disabling multiple trains of equipment at the same time, they can be large contributors to the risk. In the staff's analysis, the only multiple train system modeled is the spent fuel pool cooling system. In the fault tree model for this system, common cause failures are modeled for the cooling pumps, the heat exchangers, and the discharge check valves. The modeling of dependent failures, including common-cause hardware failures, in the staff's risk analysis is consistent with NRC and industry guidelines. Based on the above, the staff has confidence that its modeling and quantification of common mode failures is adequate.

SPSB Public comment # 5: NRC should set guidelines on how often fuel handlers make their rounds at decommissioning facilities. This would help assure operator attentiveness.

Response: The staff agrees that, if fuel handlers make the rounds of the SFP and its equipment on a frequent basis, the probability of the handlers detecting problems early is greatly enhanced. To this end SDA #1 states in part that walk-downs of the SFP systems will be performed at least once per shift by the fuel handlers. This is documented in **Section 3.3.1** of the report. The staff expects that these assumptions will be translated into requirements or industry guidance during the rulemaking process.

SPSB Public comment # 6: NRC should assure that the probability of failure of systems required to mitigate the consequences of design bases and beyond design bases spent fuel pool events are minimized.

Response: The need to have highly reliable systems to prevent or mitigate an accident is partly a function of how rapidly the accident progresses and how serious its consequences are. If an accident would result in serious consequences unless a rapid response were achieved, then highly reliable systems and components are needed to prevent and/or mitigate the event. If the accident is very slow in progressing or has benign consequences, the equipment designed to prevent or mitigate it need not be as reliable. For spent fuel pools at decommissioning plants, the large volume of water above the spent fuel provides an inherent delay time before fuel can be uncovered, except for two potential beyond design basis accidents which are discussed later. This delay time (measured in days) allows for repair or replacement of equipment. If it were impossible to repair or replace the equipment, inventory could be added to the pool to match the boil-off rate. The industry has committed in IDC #4 **(Section 3.2)** to implement an off-site resource plan to include access to portable pumps and emergency power. IDC #7 and IDC #9 commit the industry to implement procedures or administrative controls to reduce the likelihood of rapid draindown events. The SDA #1 (**Section 3.3.1)** calls for procedures to be developed that will provide guidance on the availability of on-site and off-site inventory make-up sources and time available to initiate these sources. In addition, the industry has committed in IDC #10 to perform routine testing of the alternative spent fuel pool make-up system components and to have procedural controls on equipment out of service to increase confidence that components will be available. The two accidents that could lead to very rapid draining of the SFP are extremely large seismic events and heavy load drops. IDC #1 and SDA #2 (**Section 3.3.6**) address heavy load drop concerns. SDA #3 (**Section 4.2.1**) calls for each decommissioning plant to successfully complete the seismic checklist provided in Appendix 2 to this report. Implementation of these commitments and assumptions will help assure the frequency of a zirconium fire remains below the pool performance guideline of $1X10^{-5}$ per year.

SPSB Public comment #7 and SPLB #2: Is station blackout at a decommissioning site acceptable to the staff?

Response: The staff does not find having station blackouts to be an acceptable practice. At the same time, as with an operating reactor, the staff recognizes that there is some small annual probability that a station blackout will occur at a decommissioning site. Unlike an operating reactor, decommissioning spent fuel pools (at one year or greater after the last fuel was irradiated in the reactor) can go without electrical power for almost a week and not suffer serious consequences. This is due to the inherent margin provided by the large volume of water sitting above the spent fuel in the pool. It takes a long time to heat this water up to boiling and then to continue to boil it off until fuel is uncovered. IDC #2 commits the industry to develop procedures and train personnel to ensure that on-site and off-site resources can be brought to bear during an event. IDC #3 calls for communication systems to be set up between the SFP site and off-site resources that can survive severe weather and seismic events, which can cause a station blackout. **Section 3.2** of this report discusses this issue.

SPSB Public comment #8: The risk assessment should take into account changes in local aircraft traffic when evaluating the probability and consequences from aircraft crashing into SFPs.

Response: The risk from aircraft crashes is small, and even large increases in traffic should not make aircraft crashes a dominant contributor to risk. A decommissioning plant will continue to be governed by 10 CFR Part 50 for the evaluation of hazards as discussed in Standard Review Plan 2.2.3, "Evaluation of Potential Accidents," including accidents involving nearby industrial, military, and transportation facilities. Changes in local aircraft traffic would continue to be assessed on a deterministic basis at a decommissioning plant and a reassessment of risk would be performed, as needed.

The frequency of an aircraft crash leading to an accident in a spent fuel pool was estimated in the report to be in the range of 4.3×10^{-8} to 9.6×10^{-12} per year where damage to the pool was significant enough that it resulted in a rapid loss of water from the pool **(See Section 3.4.2 and Appendix 2b)**. The mean value was estimated to be 2.9×10^{-9} per year. These values are a small fraction of the overall risk of uncovering the spent fuel in the pool at a decommissioned plant, which was estimated to be less than 5.0×10^{-6} per year. An aircraft crash could also result in damage to a spent fuel pool support system. The estimated range of striking a support system was estimated to be in the range of 1.0×10^{-5} to 1.0×10^{-9} per year, with a mean value of 7.0×10^{-7} per year, without consideration of recovery actions. These values are also a small fraction of the estimated frequencies for the loss of cooling initiator (3.0×10^{-3} per year), the internal fire initiator (3.0×10^{-3} per year), or the loss of inventory initiator (1.0×10^{-3} per year).

Aircraft traffic and accident data were reviewed by the staff (Ref: "Data Development Technical Support Document for the Aircraft Crash Risk Analysis Methodology (ACRAM) Standard," C.Y. Kimura, et al., UCRL-ID-124837, Lawrence Livermore National Laboratory, August 1, 1996). The number of U.S. air carrier operations increased from about 5.5 million departures per year in the 1970s to about 8.7 million departures per year in the mid-1990s. The average miles traveled per departure increased from about 500 to 650. For the period from 1986 to 1993 general aviation operations remained relatively constant, with a decrease in activities reported in 1992 and 1993. Military aircraft data, which are a small fraction of the total risk **(see Table A2d-1**, "Generic Aircraft Data"), was not reviewed.

While it is very unlikely that changes to aircraft traffic near a decommissioning plant will significantly increase the estimated risk of uncovering the spent fuel in the pool, changes in aircraft traffic would continue to be assessed at a decommissioning plant.

SPSB Public comment #9: What is the generic frequency of events leading to zirconium fires at decommissioning plants before the implementation of industry commitments and staff assumptions?

Response: The staff visited four decommissioning sites as part of the preparation for developing the risk assessment of decommissioning spent fuel pools. The insights from those visits include that the facilities appeared to have been staffed by well trained and knowledgeable individuals with significant nuclear power plant experience. Procedures were in place for dealing with routine losses of inventory. Fuel handlers appeared to know whom to contact off-site if difficulties arose with the SFP. The staff recognized that these attributes were not required by any NRC regulations nor suggested in any NRC guidance for decommissioning sites. The industry's IDCs and the staff's SDAs are an attempt to increase the assurance that fuel handlers will continue to be knowledgeable of offsite resources and have good procedures available to them. The staff believes that the initiating event frequencies at the visited decommissioning sites are very similar to those estimated in the staff's decommissioning SFP risk assessment. The response of the fuel handlers at the visited sites would probably be as good as estimated in the report. If somehow it were possible for a zirconium fire to begin at one of these pools, the staff believes that the frequency of this fire would be on the same order of magnitude as that estimated in the report.

<u>SPSB Public comment #10</u>: What will the NRC staff do to protect plant workers and the public from spent fuel pool risks at permanently closed plants and operating plants before the industry commitments and staff assumptions are implemented?

<u>Response</u>: Regarding protection of the public, for plants that are currently in a decommissioning status, the staff has no reason to believe that these sites have characteristics significantly worse than those discovered by the staff during its visits to four decommissioning sites. The as-found conditions at these sites were the basis for the modeling of the spent fuel pool cooling system and operator actions in the report. In addition, most decommissioning sites have even lower decay heat levels than assumed in the report, and the likelihood of a zirconium cladding fire should be even lower at these sites than estimated in the report since these sites have longer periods within which to recover spent fuel pool cooling or inventory. **The staff intends to review the heavy load operations at current decommissioning sites to assure that there are no vulnerabilities.** Future decommissioning plants will either implement the industry commitments and staff assumptions or will have to continue with full emergency preparedness, security, and insurance. Operating reactors are fully staffed, have multiple backup systems, and have full emergency preparedness, security, and insurance. The staff believes that the risks from operating reactor spent fuel pools are less than those of decommissioning plants and are within the NRC's Safety Goals.

The dominant health concern for decommissioning site workers caused by beyond design bases accidents is the potential for very high exposures should the spent fuel become uncovered (the field at the edge of the pool would be in the range of tens of thousands of rem per hour.) However, since the expected frequency of spent fuel uncovery is so low and workers already are aware that uncovering the fuel could subject them to high doses, the staff believes that no additional warnings to the fuel handlers are deemed necessary at this time regarding the potential dose rates at the edge of the spent fuel pool associated with fuel uncovery. Decommissioning plant workers continue to have radiation dose limits set by the NRC and their utility, just as workers do at operating nuclear power plants.

<u>SPSB Public comment #11</u>: There are several places in the draft report where the staff refers to "uncovering the core" rather than "uncovering the fuel."

<u>Response</u>: The phrase "uncovering the core" has been replaced by "uncovering the fuel."

SPSB Pubic comment #12: Recalculating the frequencies for event trees produced numerical results for some sequences that were off by one or two orders of magnitude.

Response: In the staff's risk analysis, the accident scenario frequencies in the event trees were calculated such that dependencies among the failure events (in the event tree branches) were taken into account. Therefore, if an event resulted in functional failure in more than one branch in the event tree, this dependency was taken into account, and the resultant scenario frequency is therefore larger (in some cases, by as much as two orders of magnitude) than if the events were assumed to be independent.

SPSB Public comment #13: The initiating frequencies, human error rates, and equipment failure rates should more accurately take into account the occurrence of actual events such as Chernobyl and Three Mile Island.

Response: The decommissioning SFP risk assessment takes into account actual events that are applicable to spent fuel pools and their support systems. The staff used initiating event frequencies from staff studies from actual events at spent fuel pools, from actual crane lift data, from site-specific seismic hazard curves, from studies on aircraft crashes and tornadoes, and from large databases developed to provide estimates for initiating events and equipment failure rates. Human error rates were developed by the staff in conjunction with experts at Idaho National Engineering and Environmental Laboratory. The staff believes that the values used in the report provide a reasonable picture of the risks associated with operation of decommissioning spent fuel pools under the assumptions and commitments documented in the study.

SPSB Public comment #14: The NRC should determine which failure rates used in the report are reliable and which are not, and the results should be included in the study.

Response: The staff uses the most reliable information on failure rates that is available. Because of the long time it takes for water above the spent fuel to heat up and boil off, the failure rates of specific equipment that support a spent fuel pool are not important contributors to spent fuel pool risk for long term sequences (i.e., the results are not particularly sensitive to the assumed failure rate of equipment.) However, very large seismic events or heavy load drops could rapidly drain the spent fuel pool. For seismic events, the robustness of the spent fuel pool is assured by implementation of a seismic checklist (**See Appendix 2**). For heavy load drops, IDC #1 calls for performance of cask drop analyses or use of a single-failure-proof crane when moving heavy loads over or near the spent fuel pool (**See Section 3.2**), which should help assure that the risk from heavy load drops is extremely low.

SPSB Public comment #15: Mitigating systems at decommissioning spent fuel pools are not automatic. The NRC should assure that fuel handlers are available in the event of an accident.

Response: The staff has presented to the Commission a rulemaking plan related to decommissioning that includes operator staffing requirements and safeguards arrangements for facilities undergoing decommissioning. Staffing at present day decommissioning sites is controlled by Technical Specifications on a plant-specific basis. In addition, SDA #1 calls for walkdowns of the spent fuel pool area by fuel handlers every shift **(See Section 3.2.)**

SPSB Public comment #16: What measures have been taken to assure that fuel handlers remain attentive?

Response: The Commission, through the "Policy on Factors Causing Fatigue of Operating Personnel at Nuclear Reactors" provides guidelines on working hours that were consistent with the objective of ensuring that the mental alertness and decision-making abilities of plant staff were not significantly degraded by fatigue. For this study, the staff incorporated several measures into the risk assessment to help assure fuel handler attentiveness. First, SDA #1 calls for walkdowns of the spent fuel pool area by fuel handlers every shift. Second, IDC #4 states that SFP instrumentation will be in place providing readouts and alarms in the control room or where the fuel handlers are stationed. Additionally, discussions with the industry indicate that it is a general practice for sites to log instrument readings from the decommissioning spent fuel pools at least once per shift. Such practices help maintain fuel handler alertness and keep them abreast of the status of the pool and its support systems. **Sections 3.2 and 3.3.1** of this report discuss this issue.

SPSB Public comment #17: What measures have been taken to help minimize fuel handler error in postulated SFP accident scenarios?

Response: Having procedures in place helps reduce that chance of human errors, especially under stressful conditions such as during a severe accident. The industry has committed to providing procedures or administrative controls to reduce the likelihood of rapid drain down events. IDC #2 is credited for ensuring that procedures and training of personnel are to be in place to ensure that on-site and off-site resources can be brought to bear during an accident. IDC #3 is credited to have procedures for establishing communication between on-site and off-site organizations during severe weather and seismic events. IDC #4 is credited to ensure that an off-site resource plan will be developed that will include access to portable pumps and emergency power. In addition, IDC #5 is credited to ensure that fuel handlers will have available to them spent fuel pool instrumentation that monitors spent fuel pool temperature, water level, and area radiation levels. **Section 3.2** of this report discusses this issue.

SPSB Public comment #18: The NRC should review the need to place a containment around spent fuel pools.

Response: The staff evaluated the risk from spent fuel pool operation and from zirconium fires at operating plants in Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools." NUREG-1353 determined that the risks of spent fuel pool operation and the cost of alterations did not justify performing any generic backfits at operating plants, including installation of improved containment structures. Risk estimates from the decommissioning spent fuel pool risk assessment are similar to risk numbers (same order of magnitude) found in NUREG-1353, and decommissioning sites have a shorter period of vulnerability to zirconium fires than do operating reactors. The staff believes that an additional containment structure is not warranted for decommissioning spent fuel pools.

SPSB Public comment #19: To the extent possible, experimental validation of risk-informed results should be addressed.

Response: The staff does not plan on performing any proto-typical tests of SFP configurations. However, the predictive models used for estimating the risk from spent fuel pools are based on a wealth of experimentation. Many experiments have been performed in the areas of human reliability analysis, seismic fragility of equipment, fires, and thermal hydraulics (where billions of dollars have been spent to better understand the phenomenology of reactor accidents.) The results of the decommissioning spent fuel pool risk assessment come from a systematic analytical modeling of the spent fuel pool and its support systems at a "typical" decommissioning site. The model of the spent fuel pool and its support systems was based on plant-specific visits made by the staff. The staff used failure rates of support system equipment based on existing large databases of equipment failure rates. Human error rates were developed by the staff with help from experts at Idaho National Engineering and Environmental Laboratory. Heavy load drops were based on modeling performed for NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants, Resolution of Generic Technical Activity A-36" with additional sources of data from U.S. Navy crane experiences, Waste Isolation Plant Trudock Crane System experience, and data supplied by NEI (See Appendix 2c). The effects of aircraft crashes were analyzed using Department of Energy models (See Appendix 2d) and generic aircraft crash data.

SPSB Public comment #25: The staff's report is misleading when it states that there is about a factor-of-two reduction in prompt fatalities if the accident occurs after one year instead of thirty days. The real insight should be that compared to operating plants, the absolute value of prompt fatalities from zirconium fires at SFPs is a couple of orders of magnitude lower. In fact, the report does not justify a one-year delay in eliminating off-site emergency preparedness. Prompt fatalities are sufficiently reduced one month after reactor shutdown to support eliminating off-site emergency preparedness.

Response: The report does not focus on comparing the results of an accident at thirty days versus one year. The staff evaluated the risk to the public from spent fuel pool operation at decommissioning plants at one year and longer after final reactor shutdown. The basis for our recommendations on delaying reduction or elimination of off-site emergency preparedness is based on a number of factors, two of which are the estimated frequency of spent fuel pool zirconium cladding fires and the estimated consequences of such a fire.

SPSB Public comment #28: The human error probabilities (HEPs) used for the operator action "Operator Recovery Using Off-Site Sources" are too conservative.

Response: The HEPs for recovery using off-site sources were quantified with the assumption that the fuel handlers/plant operators will initially attempt to mitigate the upset condition using in-house resources, and having failed this, attempt recovery using off-site sources. This was based on input obtained from licensees during public meetings on this subject, and on the assumption that fuel handlers will initially avoid using raw water (i.e., water not chemically controlled) when possible. It was however assumed that licensee procedures and training are in place to ensure that off-site resources can be brought to bear (IDC # 2 and 4), and that these procedures explicitly state that if the water level drops below a certain level (e.g., 15 feet below normal level), the fuel handler must initiate recovery using off-site sources. The probability of this event was quantified under the assumption that there is a low dependence with preceding fuel handler failures. Given that the event is always coupled with other fuel handler failures, it would, in the staff's opinion, be inappropriate to argue for zero dependence. When looked at in the context of the complete cutsets, it can be seen that the likelihood of failure to respond to any of the initiating events (excluding seismic and heavy load drops) where meaningful responses are possible is indeed very low, as is evident from the very low sequence frequencies.

<u>SPSB Public comment #29</u>: Is it realistic to assume "good communication" with off-site emergency organizations once the plant is shutdown and "forgotten"?

<u>Response</u>: As the time after shutdown increases, the decay heat loads decrease and more time is needed to heat up the pool water and boil off if heat removal were lost. After one year, the decay heat levels are such that there is at least a week of delay between loss of cooling and spent fuel uncovery. Even following a seismic or severe weather event, the staff expects that a utility will be aware of the resources that are available in the area to provide pool cooling or inventory make up and that the utility will have assured the availability of the resources. In addition, the utility should have a plan for communicating with suppliers and government officials during such emergencies by means that would not be disrupted by such events (e.g., by portable radio). IDC #2 and #3 provide assurance that good communication will be maintained.

<u>SPSB Public comment #30</u>: Will commitments lead to practices better than current? If not, use historic data.

<u>Response</u>: It is the staff's expectation that the commitments will in general provide guidance that assures that the good practices found at decommissioning sites visited by the staff will be implemented at future decommissioning sites. Some industry commitments and staff assumptions, such as IDC #1 (**See Section 3.2**) and SDA #2 (**See Section 3.3**) and SDA #3 (**See Section 4.2.1**), may enhance of the capabilities currently practiced by existing decommissioning plants. Where possible (e.g., for some initiating event frequencies), the staff has used actual data from spent fuel pool events. The commitments provide a basis for the staff to conclude that the low human error probabilities associated with the loss of SFP cooling and loss of inventory events are justified. In addition, the commitments provide a bound on the risk associated with the two events that could rapidly drain the spent fuel pool (i.e., seismic and heavy load drop events.)

<u>SPSB Public comment #31</u>: The staff noted a recent event (January 2000) that occurred during shutdown, when SFP monitoring should have been a priority. This event should have raised the initiating event frequencies, not lowered them.

Response: Including the two recent loss-of-cooling events mentioned in **Section 3.3.1** of the draft report would increase the initiating event frequency for loss of cooling accidents. However, since the fuel uncovery frequency from this event is very low (approximately 10^{-8} per year), the conclusion in the report that the loss of cooling events are not a major risk contributors is not affected. However, these recent events illustrate the importance of industry commitments, particularly IDC #5, which requires temperature instrumentation and alarms in the control room.

SPSB Public Comment #32: The discussion in Section 3.3.2 states that many of the events listed in NUREG-1275, Volume 12, do not apply to a decommissioning facility. Therefore, adherence to IDCs #2, 5, 8, and 10 are not really important to establishing a low frequency of fuel uncovery.

Response: The commenter correctly noted that many of the initiating events from operating reactor spent fuel pool incidents that are discussed in NUREG-1275 do not apply to decommissioning facilities. The staff likewise did not include these events when estimating the frequency of events at decommissioning plants. To help assure that the frequency of these events does not end up being much higher than assumed by the staff in its risk assessment, the industry committed to various IDC's and SDA's regarding procedures and planning for contingencies to limit, prevent, or mitigate loss of inventory and loss of cooling events.

SPSB Public comment #33: How did the staff come up with the factor of 100 reduction in the failure rate for heavy load drops for single-failure-proof systems?

Response: For a non-single-failure-proof handling system, the mean probability of a loss-of-inventory event was estimated based on NUREG-0612. In NUREG-0612, an alternate fault tree (Figure B-2, page B-16) was used to estimate the probability of exceeding the release guidelines (loss-of-inventory) for a non-single failure proof system. The mean value was estimated to be about 2.1×10^{-5} per year when corrected for the new Navy data and 100 lifts per year. A comparison of this mean value to the 2.0×10^{-7} per year mean value for the single-failure-proof crane shows a factor of 100 reduction.

SPLB Public Comment #1: Were heavy objects, such as crane rail or masonry wall, falling into the SFP or taking out electricity during decommissioning activities addressed in the study?

Response: The loss of electricity and the control of heavy loads were considered in the study. The loss of electricity would result in a loss of the spent fuel pool cooling system. IDC # 1 and SDA #2 deal with controlling heavy loads over the spent fuel pool. With regards to a masonry wall, any design feature specific to an individual plant would be dealt with on a case-by-case basis.

SPLB Public Comment #4: Since the National Severe Storm Center is predicting more frequent and more intense severe weather phenomena, shouldn't the size and velocity of wind-driven missiles and maximum height of storm surges be reassessed?

Response:
If more severe storms were occurring and a plant (or plants) did not function as expected, we would evaluate the need to update plants' storm-related analyses. Also, if a licensee requests a change to its licensing basis dealing with storms, such as tornados, or storm-generated missiles, then they would look at more recent data collected since the licensing of the plant. If an individual or organization believes that a rule should be changed, a rulemaking petition can be filed in accordance with 10 CFR 2.802.

SPLB Public Comment #7: All pools leak, dry storage is the only way for long term safety.
SPLB Public Comment #8: The NRC should identify all SFP's that leak. Degradation of the lines and concrete should be investigated. The leaks should be sealed.

Response to #7 & #8: The statement that all pools leak implies leakage to the environment, which is mostly certainly not true. Most pools have a leak detection system between the steel liner and the concrete wall to identify and quantify if leakage from the liner occurs. This is not leakage to the environment. This water is collected by the system in the plant. This system allows licensees to monitor a situation and evaluate if there is a safety concern. Two plants do have leaking spent fuel pools. The licensees are closely monitoring the leak to ensure that there is no public hazard.

Dry storage casks are a viable option for spent fuel storage for licensees. Dry storage casks are currently approved for fuel that have been removed from the reactor for at least five years. Many licensees are choosing to use dry cask storage in addition to the spent fuel pool. The NRC has approved the use of spent fuel pool for the life the plant.

SPLB Public Comment #11: What happened to the commitment verbally agreed up on through a public stakeholder to install a single failure proof crane system using safety grade electrical equipment?

Response:
NEI verbally committed decommissioning plants to implement Phase II of NUREG-0612 (Control of Heavy Loads), which prescribed the use of single failure proof cranes or to implement a load drop analysis. NEI provided this commitment in writing on November 12, 1999. The commitment was included in the analysis and documented in the report as IDC #1.

DLPM Public Comment #4: The staff's spent fuel pool risk study only considered accidents scenarios that could lead to a spent fuel zirconium fire. A member of the public questioned what other design basis accidents are considered for decommissioning nuclear power plants beyond those addressed in the study?
DLPM Public Comment #5b: A member of the public stated that SECY 99-168 doesn't cover all decommissioning issues. The commenter asked what design basis accidents do we need to consider?

Response: There are typically no new or unique conditions associated with decommissioning that result in the creation or possibility of a different type of accident not previously bounded by the design basis accidents considered for the plant while it was operating. When a licensee updates its Final Safety Analysis Report for decommissioning, a suite of accidents are considered that have a reasonable potential to adversely impact public health and safety. The offsite consequences of these accidents are very small and should not require offsite emergency response. Examples of the types of accidents that are considered by the licensees include:

- Materials handling event (non-fuel)
- Radioactive liquid waste releases
- Accidents from handling spent resin
- Fire
- Explosions
- External events
- Transportation accidents
- Fuel handling accident

In addition to plant specific assessments of the postulated accidents, the staff has performed some generic evaluations. Consideration of environmental impacts of such events has been provided in NUREG-0586, "Final Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities."

DLPM Public Comment #15: A public stakeholder stated that Industry Decommissioning Commitment #5 should be revised to require direct measurement of SFP temperature and water level.

Response: The staff agrees and has incorporated this clarification in its sample regulatory language for emergency preparedness in the integrated decommissioning rulemaking plan, SECY-00-0145, issued on June 28, 2000. **Confirm this is addressed in the final report!!!**

DLPM Public Comment #18: Dr. Hanauer was quoted in a 1975 memo to say, "you can make probabilistic numbers prove anything, by which I mean that probabilistic numbers prove nothing." If a respected technical advisor has expressed doubts about the NRC's use of probabilistic numbers, how is the NRC going to use probabilities convincingly to protect health and safety? A member of the public stated that, "this is an invalid way of measuring safety, and should not be used. Each day these reactors stay opened you are poisoning the environment. This is unacceptable."

Response: Dr. Hanauer was a respected NRC technical advisor in the 1970's. However, in the two and a half decades since his statement was quoted ("you can make probabilistic numbers prove anything, by which I mean, that probabilistic numbers prove nothing"), there have been significant advances in risk assessment methodologies. In that time frame, the NRC has also gained a great deal of experience in applying these methodologies to the regulatory arena, which has led to improved safety. The NRC has determined that PRA is an acceptable technology and uses it in a manner that complements a deterministic approach and supports the traditional defense-in-depth philosophy.

DLPM Public Comment #21: Has the NRC considered the events with the "second" worst offsite consequences at decommissioning plants? For example, in another country which has nuclear power plants, a fire in the bitumen storage (waste handling area) was found to have the second worst, although limited, offsite consequences.

Response: The draft NRC study evaluated a spectrum of potentially severe spent fuel pool accidents that could lead to uncovery of the fuel. Separate from the draft report, the NRC considered other, less severe accidents with offsite consequences. The rulemaking plan established for the first group of rule changes (i.e. the integrated rulemaking), recommends that licensees perform reviews at their facilities to ensure that there are no other possible accidents that could result in offsite consequences exceeding EPA Protective Action Guidelines before reductions may be made in emergency preparedness and insurance requirements.

SEISMIC

DE Public Comment #1: The staff should look at stresses on the transfer tunnel
DE Public Comment #4: Seismic vulnerabilities of the SFP transfer tube should be assessed to properly determine the risk of SFP draining.
DE Comment #11: (formerly Appendix 5h #2)
During the July workshop, members of the public raised concerns about the hazard of the fuel transfer tube interacting with the pool structure during a large earthquake.

Response: Transfer tubes are generally used in PWR plants where the fuel assembly exits the containment structure through the tube and enters the pool. These transfer tubes are generally located inside a concrete structure that is buried under the ground and attached to the pool structure through a seismic gap and seal arrangement. In most spent fuel pools, the transfer tube is not connected directly to the area of the pool that contains the spent fuel. The transfer tube is usually in a separate portion of the pool that has a weir wall separating the area from the main section that holds the spent fuel. The weir wall is higher than the top of the spent fuel. As such, even if water was drained through the transfer tube, the fuel would not be uncovered. Additionally, following the final off-load of the fuel into the spent fuel pool, the transfer tube is permanently capped at both ends. However, the layouts and arrangements can vary from one PWR plant to another and the seismic hazard caused by transfer tubes should be examined on a case-by-case basis. As such, as part of the seismic checklist each licensee must verify the adequacy of spent fuel pool penetrations whose failure could lead to drainage or siphoning (See Appendix 2).

DE/**SPLB** Public Comment #2: The staff should address aging effects on the qualification of equipment.

Response: "Equipment Qualification" is required for components that are exposed to harsh environments, such as high radiation or high temperature. Systems around the spent fuel pool are not exposed to harsh environments and therefore do not needed special consideration. To address the effect of normal aging on spent fuel pool support systems, the maintenance rule (10 CFR 50.65) requires that the licensee monitor systems or components associated with the storage, control, and maintenance of the spent fuel in a safe condition. Additionally, the probability of equipment failure was included in this study as part of the accident sequences. The staff believes that aging of equipment at decommissioning plants has been addressed through existing programs and in this study.

DE Public Comment #3: The staff should address aging effects on the spent fuel pool, in particular, the strengthening or hardening of the concrete and the strength of the liner over time.
DE Public Comment #5: The NRC should perform a rigorous engineering analysis of the effects of aging[1] upon the spent fuel pool and its associated structures and equipment. Most SFPs were never designed to be quasi-permanent fuel storage facilities. Because there is, as

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[1] Aging could include degradation, failure, etc. of structures & equipment.

of yet, no permanent place to store used fuel, SFPs have had to accept more fuel than they were originally designed to hold. To allow SFPs to continue to store spent fuel for, as of yet, an undetermined period of time requires, I suggest a comprehensive look at aging.
DE Comment #12: (formerly Appendix 5h #3) Members of the public raised concerns about the effect of aging on the spent fuel pool liner plate and the reinforced concrete pool structure.

Response:
Irradiation-induced degradation of steel requires high neutron fluency, which is not present in the spent fuel pools. Operating experience has not indicated any degradation of liner plates or the concrete that can be attributed to radiation effects.

With aging, concrete gains compressive strength of about 20% in an asymptotic manner and the strength of reinforcing bars does not change with age, provided that rebars are not degraded by corrosion. Spent fuel pool structures are expected to have this increased strength at the time of their decommissioning. In general, degradation of concrete structures can be divided into two parts, long term and short term. The long-term degradation can occur due to freezing and thawing effects when concrete is exposed to outside air. This is the predominant long-term failure mode of concrete; observed on bridge decks, pavements, and structures exposed to weather. Degradation of concrete can also occur when chemical contaminants attack concrete. These types of degradation have not been observed in spent fuel pools in any of the operating reactors. Additionally, inspection and maintenance of spent fuel pool structures are within the scope of the maintenance rule, 10 CFR 50.65, and corrective actions are required if any degradation is observed. An inspection of the spent fuel pool structure to identify cracks, spalling of concrete, etc., is also recommended as a part of the seismic checklist. Substantial loss of structural strength requires long-term corrosion of reinforcing steel bars and substantial cracking of concrete. This is not likely to happen because of inspection and maintenance requirements. Through the use of the proposed seismic checklist, any degradation such as spalling of concrete or cracks and indications of rust and stains, etc., will be detected and appropriate corrective actions taken.

Degradation of the liner plate can occur due to cracks that can develop at the welded joints. Seepage of water through minute cracks at welded seams has been minimal and has not been observed at existing plants to cause structural degradation of concrete. Nevertheless, preexisting cracks would require a surveillance program to ensure that structural degradation is not progressing.

Based on the discussion above, any potential aging of the spent fuel pool structure is managed during decommissioning. The structural strength will be verified using the seismic checklist in the early stages of decommissioning, which may include site-specific analysis. While its structural strength is not expected degrade during decommissioning, it is managed under the maintenance rule. As a result, the staff does not believe that detailed generic analysis is needed.

DE Public Comment #6: To my knowledge, not every spent fuel pool was designed to the seismic criteria in use today. The use of works like "robust" does not necessarily address seismic qualifications. The NRC should identify all spent fuel pools that were not initially designed to seismic criteria and explain their level of qualification, including the SF racks.

Response: All spent fuel pools have undergone seismic and structural reevaluation, at least once, during a licensing review when the licensee requests to expand the spent fuel storage capacity. Spent fuel pool structures, as well as the spent fuel racks, undergo detailed analysis and staff review. All currently operating nuclear power plants have expanded their spent fuel storage capacity and met their safe shutdown earthquake criteria.

DE Public Comment #7: Not all PWR buildings housing spent fuel are seismically qualified. The NRC should perform a worst case analysis of the result of a seismic event which collapses the spent fuel pool building, and/or drains the pool and/or damages the spent fuel. Both criticality and zirconium fires are of concern. The nine initiating events listed on p. 11 which could occur concurrently with the earthquake should also be considered if the events contribute to the worst case scenario.

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

Loss of Off-site Power from plant centered and grid related events
Loss of Off-site Power from events initiated by severe weather
Internal Fire
Loss of Pool Cooling
Loss of Coolant Inventory
Seismic Event
Cask Drop
Aircraft Impact
Tornado Missile

The initiating events indicated above are independent. However, the event sequences that emanate from each event are carefully modeled in the event tree and could include some of the same circumstances. This means that a seismic event tree would include the consideration of off-site and on-site power loss. In a PRA assessment no risk insight can be gained by considering worst case combination of truly random and independent events such as a seismic event and a tornado missile. However, the frequency of a combined seismic and tornado missile is much less than $1x10^{-8}$. Also, with respect to other structures, such as crane girders and super-structures, they are covered in the seismic check list for the spent fuel pool structure.

DE Public Comment #8: The NEI seismic checklist requires a seismic engineer to review drawings in addition to conducting a walkdown of the SFP. It has been my experience that many electrical drawings of NAP's do not reflect the existing plant electrical installation. How is the seismic engineer going to verify drawings to the existing SFP building and pool if much of the pool is inaccessible? For instance, how does he verify concrete degradation under the steel liner? The NRC should require that specific areas be inspected and that these areas be accessible. If these areas are not accessible, then the checklist is not complete and susceptibility to seismic activity remains a concern.

Response: The staff considers the review of construction drawings to be very important. Minimum reinforcing areas are dictated by codes. Thick walls and slabs forming spent fuel pool structure are in many cases governed by minimum reinforcing requirements. Should there be any additional shear or flexural steel requirements, engineering calculations would indicate where they are need and how much is needed. Therefore, a review of drawings and design calculations would present a more complete picture. With respect to accessibility, cracks, spalling of concrete and stains and efflorescence are indications of a degradation in progress in inaccessible areas. In order to determine the root cause of the external signs, it is necessary to use more invasive procedures, such as chipping and breaking concrete, etc. This is not unique to spent fuel pool structures, and there are several examples of this type of inspection in the operating experience of several plants.

<u>DE Public Comment #9</u>: The NRC should specify why it is not cost effective to perform a plant-specific seismic evaluation for each spent fuel pool and what impact this has on safety. Because there are so many differently designed spent fuel pools, it is difficult to perceive how a generic approach could be acceptable without assembling a list of similar and/or identical designs and performing a seismic evaluation of the various groups which are assembled. Specific seismic evaluations for each plant or groups of similar/identical plants should be considered.

<u>Response</u>: A significant body of work exists characterizing the strength and capacity of shear walls based on tests and analyses. The use of a generic parameter, with the underpinning of data, solely for the purpose of screening is very appropriate and reliable. Using the seismic checklist, a structure is not acceptable unless all the conditions in the checklist are met. At sites where the prescribed seismic demand is greater than the 0.5g peak ground acceleration value or the 1.2g spectral acceleration value, a plant specific evaluation is to be conducted. The use of a screening parameter is a reliable way to determine the need for further evaluation. This concept was developed without any consideration of cost.

<u>DE Public Comment #10 (formerly Appendix 5h #1)</u>: A member of the public raised a concern about the potential effects of Kobe and Northridge earthquakes related to risk-informed considerations for decommissioning

Did any of the NUREGs that you looked at take into account new information coming out of the Kobe and Northridge events? Particularly as we are learning more about risks associated with those two particular seismological events that were never even considered when plants were sited; particularly, though I can't frame it in the seismological language, from a lay understanding, it's clear that new information was gained out of Kobe and Northridge events suggesting that you can have seismological effects of greater consequence farther afield than at the epicenter of the event."

<u>Response</u>: The **two NUREGs** mentioned by a member of the public were written in the middle and late 1980s and used probabilistic seismic hazard analyses performed for the NRC by Lawrence Livermore National Laboratory (LLNL) for nuclear power plants in the central and eastern U.S. Since then, LLNL has performed additional probabilistic hazard studies for central and eastern U.S. nuclear power plants for the NRC. The results of these newer studies indicated lower seismic hazards for the plants than the earlier studies estimated. If the probabilistic hazard studies were to be performed again, hazard estimates for most sites would probably be reduced further than the LLNL 1993 study due to: new methods of eliciting information, newer methods of sampling hazard parameters' uncertainties, better information on

ground motion attenuation in the U.S. and a more certain understanding of the seismicity of the central and eastern U.S.

The design basis for each nuclear power plant took into account the effects of earthquake ground motion. The seismic design basis, called the safe shutdown earthquake (SSE), defines the maximum ground motion for which certain structures, systems, and components necessary for safe shutdown were designed to remain functional. The licensees were required to obtain the geologic and seismic information necessary to determine site suitability and provide reasonable assurance that a nuclear power plant could be constructed and operated at a site without undue risk to the health and safety of the public.

The information collected in the investigations was used to determine the earthquake ground motion at the site, assuming that the epicenters of the earthquakes are situated at the point on the tectonic structures or in the tectonic provinces nearest to the site. The earthquake which could cause the maximum vibratory ground motion at the site was designated the safe shutdown earthquake (SSE). This ground motion was used in the design and analysis of the plant.

The determination of the SSEs followed the criteria and procedures required by NRC regulations and applied a multiple hypothesis approach. In this approach, several different methods were applied to determine each parameter, and sensitivity studies were performed to account for the uncertainties in the geophysical phenomena. In addition, nuclear power plants have design margins (capability) well beyond the demands of the SSE. The ability of a nuclear power plant to resist the forces generated by the ground motion during an earthquake is thoroughly incorporated in the design and construction. As a result, nuclear power plants are able to resist earthquake ground motions well beyond their design basis and far above the ground motion that would result in severe damage to residential and commercial buildings designed and built to standard building codes.

Following large damaging earthquakes such as the Kobe and Northridge events, the staff reviewed the seismological and engineering information obtained from these events to determine if the new information challenged previous design and licensing decisions. The Kobe and Northridge earthquakes were tectonic plate boundary events occurring in regions of very active tectonics. The operating U.S. nuclear power plants (except for San Onofre and Diablo Canyon) are located in the stable interior portion of the North American tectonic plate. This is a region of relatively low seismicity and seismic hazard. Earthquakes with the characteristics of the Kobe and Northridge events will not occur near central and eastern U.S. nuclear power plant sites.

The ground motion from an earthquake at a particular site is a function of the earthquake source characteristics, the magnitude and the focal mechanism. It is also a function of the distance of the facility to the fault, the geology along the travel path of the seismic waves, and the geology immediately under the facility site. Two U.S. operating nuclear power plant sites can be considered as having the potential to be subjected to the near field ground motion of moderate to large earthquakes. These are the San Onofre Nuclear Generating Station (SONGS) near San Clemente and the Diablo Canyon Power Plant (DCPP) near San Luis Obispo. The seismic design of SONGS Units 2 and 3 is based on the assumed occurrence of a Magnitude 7 earthquake on the Offshore Zone of Deformation, a fault zone approximately 8 kilometers from the site. The design of DCPP has been analyzed for the postulated occurrence of a Magnitude 7.5 earthquake on the Hosgri Fault Zone, approximately 4 kilometers from the site. The response spectra, used for both the SONGS and the DCPP, was evaluated against the actual spectra of near field ground motions of a suite of earthquakes gathered on a worldwide basis.

The commenter stated, "... it's clear that new information was gained out of Kobe and Northridge events suggesting that you can have seismological effects of greater consequence farther afield than at the epicenter of the event." A review of the strong motion data and the damage resulting from these events do not bear out the validity of this concern at SONGS and DCPP.

The staff assumes that the individual alluded to the fact that the amplitudes of the ground motion from the 1994 Northridge earthquake were larger in Santa Monica than those at similar and lesser distances from the earthquake source. The cause of the larger ground motions in the Santa Monica area is believed to be the subsurface geology along the travel path of the waves. One theory (Gao et al, 1996) is that the anomalous ground motion in Santa Monica is explained by focusing due to a deep convex structure (several kilometers beneath the surface) that focuses the ground motion in mid-Santa Monica. Another theory (Graves and Pitarka, 1998) is that the large amplitudes of the ground motions in Santa Monica from the Northridge earthquake are caused by the shallow basin-edge structure (1 kilometer deep) at the northern edge of the Los Angles Basin. This theory suggests that the large amplification results from constructive interference of direct waves with the basin-edge generated surface waves. Earthquake recordings at San Onofre and Diablo Canyon do not indicate anomalous amplification of ground motion. In addition, there have been numerous seismic reflection and refraction studies of the site areas for the site evaluations, and for petroleum exploration and geophysical research. They, along with other well-proven methods, were used to determine the nature of the geologic structure in the site vicinity, the location of any faults, and the nature of the faults. None of these studies have indicated anomalous conditions, like those postulated for Santa Monica, at either SONGS or DCPP. In addition, the empirical ground motion database used to develop the ground motion attenuation relationships contains events recorded at sites with anomalous, as well as typical ground motion amplitudes. The design basis ground motion for both SONGS and DCPP were compared to 84th percentile level of ground motion obtained using the attenuation relationships and the appropriate earthquake magnitude, distance and geology for each site. The geology of the SONGS and DCPP sites do not cause anomalous amplification, therefore, there is no "new information gained from the Kobe and Northridge events," which raises safety concerns for U.S. nuclear power plants.

In summary, earthquakes of the type that occurred in Kobe and Northridge are different from those that can occur near nuclear power plants in the central and eastern U.S. The higher ground motions recorded in the Santa Monica area from the Northridge earthquake were due to the specific geology through which the waves traveled. Improvements in our understanding of central and eastern U.S. geology, seismic wave attenuation, seismicity, and seismic hazard calculation methodology result in less uncertainty and lower hazard estimates today than have previous studies.

SPSB Public comment #26: The use of Lawrence Livermore National Laboratory (LLNL) hazard curves at high ground motion values may not be credible. Even EPRI results are likely to be overly conservative at high ground motions. The requirement that some plants with higher SSE values perform detailed HCLPF assessments of their SFPs is not warranted. In conclusion, there should be no SFP screening level distinctions based on plant SSEs for the central and eastern U.S. All that is needed is that the sites pass the screening criteria (**Appendix 2b).** For a few western sites, it is reasonable to require that the plants demonstrate a HCLPF of 2 X SSE.

Response: While it is possible that there is some conservatism in the EPRI and LLNL hazard curves at higher ground motions, the staff finds this prudent since the geologic record east of the Rocky Mountains is sparse and does not provide many examples of very large ground motions. The EPRI and LLNL hazard curves were made by different experts who gave their best judgement as to how to reflect the risks from seismic events at various nuclear power plant sites. They provided expert advice for high and low ground motions.

SPLB Public Comment #5: How can there be no spent fuel pool degradation issues if type 304 stainless steel employed in fuel racks and assemblies is known to exhibit stress-corrosion cracking in oxygenated or stagnant borated water?

Response:
Type 304 stainless steel material is susceptible to stress corrosion cracking in oxygenated water environment at relatively high temperature conditions. At the temperature levels that exist in the spent fuel pools, stress corrosion cracking of the spent fuel racks made of stainless steel is not a concern, and there has been no report of any actual incidence of stress corrosion cracking of spent fuel racks. The stagnant, borated condition of the spent fuel pool water is not a significant factor in inducing stress corrosion cracking of the racks. Most spent fuel assemblies are clad with zirconium and are not known to be susceptible to stress corrosion cracking.

SPSB Public Comment #20: A significant seismic event which damages and drains the SFP is also likely to wreak havoc upon the local infrastructure. How has NRC considered the availability of local resources as identified by IDC #2, #3, and #4 should the local infrastructure be destroyed?

Response: Seismic capacity of spent fuel structures against catastrophic failures, such that a very rapid loss of water can be assumed, is substantially above the safe shutdown earthquake levels of the spent fuel pools. Consequently, high ground motion levels are necessary to initiate failures. The response by local, state, or national authorities needed at the spent fuel pool site will depend on the actual or potential damage to the spent fuel pool. The most likely damage to the spent fuel pool and support systems would be to the support systems that provide cooling to the pool. The large inventory of water above the spent fuel should provide adequate time (it would take about a week without pool cooling before boiling would occur) for repairing or bringing in replacement pumps and heat exchangers. If the local infrastructure was damaged by a seismic event such that the prearranged off-site response could not occur, the industry commitments provide a good foundation for an ad hoc response.

SPSB Public comment #24: For all central and eastern U.S. nuclear power plant sites and for some western U.S. nuclear power plant sites, all that is necessary to have an adequately safe spent fuel pool with respect to seismic-induced risk is for the pool to meet the requirements of the seismic checklist. Several western U.S. sites may need to demonstrate a high confidence with low probability of failure (HCLPF) of 2 X SSE.

Response: The staff agrees that, for most sites throughout the U.S., meeting the enhanced seismic checklist (**Appendix 2**) is sufficient to demonstrate acceptable seismic risk for decommissioning spent fuel pools. However, four sites east and two sites west of the Rocky Mountains are beyond the scope of a simple screening evaluation; these sites must perform a plant-specific seismic risk evaluation of their spent fuel pools if relaxation of EP, indemnification, or safeguards is desired.

SPSB Public comment #27: The value of three times the SSE for the SFP HCLPF should not be a hard and fast acceptance criteria, since this is only a screening criteria.

Response: The staff agrees that this value is only a screening criterion. In **Appendix 2 ???,** the staff discusses potential mitigation measures that can be taken by a plant that does not pass the seismic checklist. Options offered include delay in requesting an exemption, correction of the identified areas on non-compliance with the checklist, or performance of a plant-specific seismic risk analysis to demonstrate that the risk associated with a catastrophic failure of the pool is at an acceptable level.

SPLB Public Comment #9: The NRC should determine the qualifications and degradation of spent fuel racks.

Response: Spent fuel rack designs are reviewed and approved by the NRC. Additionally, when a licensee changes its technical specification for the amount of fuel allowed to be stored in the pool even using approved spent fuel racks, an NRC review and approval is required. The staff technical reviewers are provided guidelines in NUREG-0800, Standard Review Plan (SRP). The SRP incorporates the regulations specified in the Code of Federal Regulations, Appendix A, General Design Criteria, which require safe handling and storage under normal and accident conditions. A specific question on degradation of the spent fuel racks is also addressed in this section.

THERMAL HYDRAULICS

SRXB Public Comment #9: The draft study is deficient in that it ignores the phenomenon associated with partial draindown of SFP that will suppress convective heat transfer by presence of residual water at the base of fuel assemblies.

Response: The partial drain down scenario may extend the critical decay time well beyond 5 years. Current calculations indicate that decay times in excess of 20 years may be needed to preclude fuel damage from a partial drain down.

SRXB Public Comment #10: The draft study is deficient in that partial draindown will lead to a steam-zirconium reaction producing hydrogen gas which could reach explosive concentrations in the atmosphere of the spent fuel building, potentially leading to a breach of that building.

Response: Steam oxidation will release hydrogen. The hydrogen concentrations or the consequences of any subsequent hydrogen burn or explosion have not been calculated.

SRXB Public Comment #11: The energy of reaction for air oxidation in the draft report is incorrect.

Response: The draft report is correct. The author of the comment has made a fundamental error. There are 92 grams of Zirconium in a mole. The authors calculation is based on 92 kg in a mole.

SRXB Public Comment #18: Depending on fuel burnup/storage array details, the development of standard methods is needed for consistent application of regulations.

Response: There is no current technical basis to support a standard methodology for thermal hydraulic analysis.

SRXB Public Comment #19: Gap release temperature too conservative for success criteria.

Response: The gap release temperature is the temperature at which the metal rod, called cladding, can blister and allow gases trapped between the fuel pellets and the cladding to escape. The criteria for gap release may also be the threshold for releasing fuel fines and ruthenium. Ruthenium trapped in the fuel could provide a source term that significantly exceeds the classical gap release. However, there may not be sufficient energy in a gap release to create an off-site hazard. These considerations and others were included in determining the success criterion for a thermal hydraulic analysis.

SRXB Public Comment #20: Fire propagation to low powered fuel unlikely.

Response: Sufficient research has not been performed to rule out propagation to even the lowest powered assemblies and past studies (i.e., GSI 82) did not evaluate potentially significant effects such as the impact of rubble from failed assemblies on fire propagation. In any event, the uncertainty in the source term is probably exceeded by the uncertainty in the PRA.

SPLB Public Comment #3: Could foreign materials with lower ignition temperatures enter a drained SFP and catch fire, thus raising the temperature of SF to the point of rapid zirconium oxidation?

Response: Licensees have programs to keep any unintended objects (called foreign objects) from entering the spent fuel pool. Retrievable foreign objects that fall into the pool are moved to designated storage areas within the pool. The staff does not have any evidence to show that the current foreign object exclusion programs are unacceptable. The staff determined that additional analysis is not merited at this time.

SECURITY/EP/RESIN FIRE/ SAFETY CULTURE

IOLB Public Comment #1: Section 4.3.2, "Security" of the draft report casts a shadow on the entire 10 CFR 73.51 rulemaking and needs to clarify the scope of the safety issues. The last paragraph in Section 4.3.2 should be clear and completely identify the scope and basis of the ISFSI safety concerns from the radiological sabotage and theft identified in 10 CFR 73.1. Finally, the last paragraph appears to contradict the May 15, 1998, NRC rulemaking on Physical Protection for Spent Nuclear Fuel and High-Level Radioactive Waste, Federal Register Vol. 63, No. 94 Pages 26955 - 26963.

(I think the response to this comment should go into the body of the report(4.3.2, "Security) instead of in public comments. Because if you read the report, we still leave in the sentence in that is confusing..Talk To Tanya).

Response: The NRC staff agrees that Section 4.3.2, Security, as written, appears to be inconsistent with the changes to Part 73 as described in FRN 26955 dated May 15,1998. The description of risk associated with potential criticality and fuel heat up is for spent fuel recently discharged from the reactor vessel and not spent fuel stored at an ISFSI.

The staff believes that, as written, 10 CFR 73.51 provides proper physical protection for the storage of all spent nuclear fuel (wet or dry storage) at an ISFSI. The design basis threat for radiological sabotage of power reactors under 10 CFR 73.1 is not considered appropriate for the types of facilities subject to 73.51, and therefore, a separate protection goal is defined for these facilities. The protection goal states that "The physical protection system must be designed to protect against loss of control of the facility that could be sufficient to cause radiation exposure exceeding the dose as described in 10 CFR 72.106 and referenced by 73.51(b)(3)."

With regard to protection against malevolent use of land-based vehicles, NRC continues to believe that there is no compelling justification for requiring a vehicle barrier as perimeter protection at this time. The staff will however, continue to review the requirements to ensure that proper level of security is provided for new cask designs and other changing technologies.

IOLB Public comment #2: With new personnel and decommissioning personnel, what methods are available to instill or ensure the same "safety culture" as during operation?

Response: There are several methods of instilling/ensuring "safety culture" in new personnel at both operating and decommissioning facilities. Methods include management policies and procedures, training, and qualification. OSHA requires employers to provide employees with safety training and education. Section 1926.21(b)(2) of Title 29 of the CFR requires training in the recognition and avoidance of unsafe conditions, 29 CFR 1926.21(b)(3) requires training in the safe handling and use of poisons, caustics, and other harmful substances, 29 CFR 1926.21(b)(5) requires training in the safe handling and use of flammable liquids, gases, or toxic materials, and 29 CFR 1926.21(b)(6) requires confined or enclosed space training. In addition, 10 CFR 50.120 requires training and qualification of nine categories of personnel involved with spent fuel pool maintenance and support. The training programs for the nine categories of personnel should include occupational safety and radiation protection training. While NRC and OSHA require training, it is incumbent upon the licensee to provide the training and instill/ensure upon the workers the proper "safety culture."

IOLB Public comment #3: The report concludes that there is no methodology currently available to access probabilities of terrorist activity or behaviors which might culminate in attempted sabotage of spent fuel. We disagree. For instance, Sandia National Laboratories, a key contractor employed by the NRC on security matters, has applied a probabilistic approach to security in decommissioning on the Maine Yankee docket. We encourage the staff to review this report.

Response: The staff disagrees with this comment and again states there is no methodology available to access the probability of terrorist activity. The report in question, its identity verified through NEI, is "A Vulnerability Analysis of a Proposed Security Plan for the Maine Yankee Power Plant," dated January 9, 1998. The purpose of this report was twofold: first, it presents the results of an analysis of the effectiveness of the proposed physical security system in preventing or mitigating an attempt by the design basis threat adversaries attempting radiological sabotage, and second, it presents the results of a study to determine the need for a vehicle barrier systems. This report does not predict the probability of terrorist activities or behaviors. The staff has read this report, and conducted an on-site inspection (June 8, 1999) of its technical findings and found them to be deficient. It is recommended that the commenter read the June 8 inspection report (Inspection Report #:50-289/99-06) for further information.

IOLB Public comment #4: The decommissioning rule should specify that the licensee is excused from 10 CFR 50.47 off-site EP requirements after the short-lived nuclides important to dose have undergone substantial decay resulting in offsite dose consequences due to license basis accidents of less than 1 rem (the EPA protective action guideline).

Response: The staff has considered the decay time of short lived nuclides and the offsite dose consequences along with the risks of both design basis accidents and beyond design basis events in efforts to determine an appropriate point at which requirements for offsite EP could be relaxed. The staff also considered the effects of the substantial decay heat and longer lived nuclides available in stored spent fuel which could result in offsite dose consequences. In consideration of these effects and the associated risks, the staff has proposed the one year decay time before considering relaxation of offsite emergency planning requirements.

IOLB Public comment #5: What does "reducing unnecessary regulatory burden" mean in practice, when it comes to emergency planning? What kind of reductions are foreseen for the following: manpower onsite/offsite, emergency equipment, communication means, alarm means, notification of personnel/public, EP, plans, KI [potassium iodide], EPZ [emergency planning zone] radius?

Response: The specific reductions in the areas mentioned is a subject that is beyond the intent of this study. Generally speaking, it is anticipated that onsite manpower could be reduced early in the decommissioning process provided adequate personnel are available to provide emergency response duties. Offsite manpower needs, equipment, communication, alarms, notifications, plans, and planning areas, would be relaxed consistent with the relaxation of requirements for offsite emergency planning. The consideration of the use of KI would not be necessary when iodine releases are no longer a concern.

IOLB Public comment #6: It's conspicuously absent from your review of risk in this overall subject, that we (the staff) haven't looked at the issue of sabotage and terrorism. (comment from a member of the public)
SPLB Public Comment #12: The draft report omitted acts of sabotage and vandalism. Emergency evacuation plans should be prepared with this consideration of terrorism.
SPLB Public Comment #13: Atherton comment: It is suggested that NRC "err on the side of safety" since terrorist acts can not be specifically addressed. [Ref. 7]

Response: The commenters are correct that security is identified, but not highlighted, in the report. The report is a technical study to quantify the risks as it relates to the draining of a decommissioned spent fuel pool and the issue of a zirconium fire. It was not intended to address security in any detail. The integrated rulemaking, which is an outgrowth of the technical study, addresses safeguards as one of the major components of the decommissioning integrated rulemaking. An entire section is devoted to security with none of the requirements less than those currently required in 10CFR 73.51. A rulemaking package is before the Commission which details the schedule for the rulemaking. As with any rulemaking, there will be opportunities for the public to comment on the security requirements the staff is recommending.

IOLB Public comment #7: A commenter requested that the consequences of an offsite radiological release from an onsite fire involving radioactive material from a resin container fire; fire in a waste storage building; and fire in a container vehicle with waste stored in it that could trigger emergency response mechanisms, be re-evaluated.

Response: This evaluation is beyond the scope of this study which is focused on spent fuel pool accident risk.

IOLB Public comment #8: Discuss protection of plant workers, particularly for less severe accidents such as pool uncovery without a zirconium fire.

Response: Existing regulatory requirements address the need for emergency plans to consider protective actions and a means for controlling exposures in an emergency for emergency workers as well as the public.

IOLB Public comment #9: Asked about calculations for radiation dose experienced by members of the fire brigade responding to resin fires.

Response: IOLB Comments #8 and #9 are very similar in nature, the comments ask about the protection of emergency responders onsite. In accordance with existing emergency planning requirements, each site has established procedures for the protection of works responding to emergency situations. Generally, these procedures include the consideration of radiological conditions when responding to events.

SPLB Public Comment #6: The draft report should be revised to include credible hazards to plant workers at permanently closed plants.

Response: This report is a technical study to quantify the frequencies and risks as they relate to accidents draining a spent fuel pool and the issue of a zirconium fire at decommissioning plants. While the staff is concerned about the worker safety at decommissioning plants, existing regulatory requirements address the need for emergency plans to consider protective actions and a means for controlling exposures in an emergency for emergency workers. OSHA and NRC regulations require safety training and education, including safe handling and use of poisons, caustics, flammable liquids, gases and toxic materials; radiation protection; and occupational safety.

SPLB Public Comment #10: The NRC should determine the proper methods of extinguishing a possible zirconium fire.

Response: At the present time, the state-of-art for zirconium fire experiments has not advanced to researching the various methods for extinguishing. Additional research would need to be performed to investigate acceptable methods, required quantities of fire-fighting materials, conditions of use, and guidelines. Due to the low probability of the event, this research is not recommended at this time.

DLPM Public Comment #8: A member of the public stated that since more radioactive materials are being handled [during decommissioning] than at an operating plant, and under conditions more likely to lead to inadvertent exposures, why are licensees left without the supervision of resident inspectors, or at least radiation protection personnel?

Response: During operation of a reactor, radioactive material is produced by neutron absorption by various materials. These radioactive materials are handled in many ways, including liquids contained in pipes and tanks and radioactive solids contained in plastic bags or specialized containers. After the reactor is shut down, no additional radioactive material is produced and the radioactive material decay process reduces the total amount of radioactive material over time. The handling of radioactive material after shutdown is controlled in the same manner as before shutdown. Supervision of radioactive material handling is performed by the licensee before and after reactor shutdown with the oversight of licensee radiation protection personnel. Region-based NRC inspectors provide a periodic verification that the licensee is handling radioactive materials within the bounds of the current regulations. NRC experience over the last few years with the current region-based reactor decommissioning inspection process has shown that the oversight process is working well to ensure both public health and safety and protection of plant workers.

DLPM Public Comment #10: A member of the public stated that little of what operators or reactor inspectors have learned is applicable to decommissioning. NRC needs personnel specifically trained in and dedicated to decommissioning.

Response: Significant changes take place during the transition from an operating plant to a decommissioning plant. However, many decommissioning activities are similar to activities conducted during plant operation. For example, the complete removal of components and systems, radiological waste shipments, fuel handling operations, and spent fuel pool system operations and maintenance which occur during decommissioning are very similar to activities that occurred during plant operation and refueling outages. Objectives during decommissioning, such as, protecting the spent fuel from sabotage and maintaining the spent fuel pool operational, were also accomplished during plant operation. The training received by operators and inspectors associated with radiological fundamentals, system operations, etc., still applies during decommissioning.

Although there is not an NRC inspector on-site during all of decommissioning, as there is during plant operation, there is a group of inspectors in each region who are specifically assigned to oversee plants undergoing decommissioning, and who make routine visits to the site (commensurate with the quantity and significance of the ongoing work). Each plant in decommissioning is also assigned to a project manager located at NRC Headquarters. These project managers are assigned to a section that is responsible only for decommissioned power reactors.

RULEMAKING & NRC PROCESS CONCERNS

Rulemaking Public Comment #1: For EP, the integrated decommissioning rule should specify that the licensee is excused from 10 CFR 50.47 requirements after a period of one-year from final shutdown. The basis for this recommendation is drawn directly from the technical material presented, and little can be gained by closer analysis.

Response: The staff has recommended in its rulemaking plan that at least 1 year of spent fuel decay has elapsed before offsite EP be discontinued as supported by the conclusions of the staff's technical risk study.

Rulemaking Public Comment #2: For Security, the integrated decommissioning rule should allow licensees to be excused from 10 CFR 73.55 requirements upon a showing that the consequences of sabotage can not exceed a defined dose to the public at the site boundary.

Response: The staff agrees that 10 CFR 73.55 should be modified to a level commensurate with the risk associated with safeguarding permanently shutdown plants, but not to a level less than that provided for an ISFSI as described in 10 CFR 73.51.

While the new regulation does not require that the spent fuel pool be a vital area, it will correct the existing problem in the 10 CFR 73.55 regarding the implementation of protected areas and isolations zones. The new rule will have a protected area and limited use of isolation zones.

Rulemaking Public Comment #3: For Insurance, the obligation for secondary financial protection should end at such time that a determination can be made that clad surface temperatures greater than 570C can not occur in a dry configuration. The calculation of this temperature should be by approved methodology. However as supported in the technical report, in the absence of any calculation, the obligation should end after a period which is less than five years. The capacity required of primary financial protection should be reduced after the period of time determined as above for secondary financial protection.

Response: Since the zirconium fire scenario would be possible for up to several years following shutdown, and since the consequences of such a fire are severe in terms of property damage and land contamination, the staff position is that full onsite liability coverage must be retained for five years or until analysis has indicated that a zirconium fire is no longer possible.

For those licensees who choose to analytically demonstrate the non-viability of a zirconium fire, the staff is now analyzing comments provided by the Advisory Committee for Reactor Safeguards to determine the threshold temperature for rapid oxidation. The staff will also evaluate the need for preparing regulatory guidance for such analytical calculations during the rulemaking process.

The NRC believes that the amount of primary financial protection required should be determined by the consequences and not the probability of the worst "reasonably conceivable" accident. The low probability of such an accident is considered by insurers who may reduce the premiums for the required coverage to account for the reduced risk at decommissioning plants.

DLPM Public Comment #1: At the decommissioning spent fuel pool risk public workshop held on July 15-16, 1999, a public stakeholder, stated, "It is difficult to figure out how this effort fits into the overall big picture of what the NRC is doing on decommissioning."

Response: The focus of the decommissioning spent fuel pool risk study was intentionally limited to address potential severe accidents associated only with spent fuel. An additional rulemaking effort, termed the regulatory improvement initiative, is planned by the NRC and will include a comprehensive look at all decommissioning regulations to determine if any additional changes are required. An overall assessment of decommissioning issues will be addressed during this subsequent effort.

DLPM Public Comment #2: At the decommissioning spent fuel pool risk public workshop held on July 15-16, 1999, a member of the public stated, "Look at all of the activities that happen during decommissioning when developing regulations, not just a narrow view of the spent fuel pool."

Response: The focus of the decommissioning spent fuel pool risk study was intentionally limited to address potential severe accidents associated only with spent fuel. An additional rulemaking effort, termed the regulatory improvement initiative, is planned by the NRC and will include a comprehensive look at the decommissioning regulations to determine if any additional changes are required. Other activities that take place at decommissioning sites will be considered during this subsequent effort.

DLPM Public Comment #3: At the decommissioning spent fuel pool risk public workshop held on July 15-16, 1999, a member of the public stated that he was confused on the way Part 50 is being applied in places where Part 72 might be more applicable.

Response: Although 10 CFR Part 50 was developed with the operating power reactors in mind, many of the requirements still apply to decommissioning power reactors. Decommissioning nuclear power plant licensees remain subject to their Part 50 license after they have permanently shut down and have offloaded all fuel from the reactor to the spent fuel pool. The Part 50 license allows for safe storage of spent fuel in a spent fuel pool during operation and the staff believes that license remains adequate for spent fuel pool storage during decommissioning. The staff does not require a Part 50 licensee to obtain a Part 72 license for spent fuel storage in a spent fuel pool. When a licensee chooses to store spent fuel in an independent spent fuel storage installation, then the appropriate requirements of Part 72 will be applicable. All reactor decommissioning activities will remain under the Part 50 license until the decommissioning is completed and the license is formally terminated.

In SECY-99-168, dated June 30, 1999, the NRC staff proposed to the Commission that all NRC regulations under Title 10 be reviewed and modified as necessary to ensure proper applicability to decommissioning. At the direction of the Commission, the staff is currently assessing the regulations that may need modification to more effectively address decommissioning reactors.

DLPM Public Comment # 5a:

A member of the public stated that SECY 99-168 doesn't cover all decommissioning issues. The commenter stated that although NRC and EPA disagree on site remediation criteria, the public stakeholder stated that either level would provide reasonable assurance to the public of undue risk.

Response: Resolution of the disagreement between NRC and EPA on release criteria is not within the scope of the current rulemaking effort.

DLPM Public Comment # 5c: A member of the public stated that SECY 99-168 doesn't cover all decommissioning issues. The commenter asked why does the NRC apply Part 50 (reactor) regulations to decommissioning reactors when the rules in Part 72 for storage of high-level waste are more clearly outlined? Part 50 regulations are not appropriate for long-term storage of high-level waste.

Response: The NRC believes that the 10 CFR Part 50 regulations applicable to decommissioning reactors are sufficient to assure public health and safety. Further assurance of the adequacy of these regulations will be provided in the near future as part of the decommissioning regulatory improvement effort in which a comprehensive review of all applicable NRC regulations will be undertaken. This issue is also addressed in the response to Comment 3 above.

DLPM Public Comment # 5d: A member of the public stated that SECY 99-168 doesn't cover all decommissioning issues. The commenter asked what is the applicability of 10 CFR Part 26 fitness-for-duty regulations to decommissioning reactors?

Response: Fitness-for-duty at decommissioning facilities is one of the issues that will be evaluated by the decommissioning regulatory improvement initiative.

DLPM Public Comment # 5e: A member of the public stated that SECY 99-168 doesn't cover all decommissioning issues. The commenter stated that quality assurance, emergency planning, fire protection, and application of codes and standards differs from site to site. Right now the decommissioning industry is being regulated by exemption to Part 50.

Response: The NRC is planning to propose new emergency planning rules for decommissioning reactors to eliminate the need for addressing the issue on a plant-specific basis by processing exemptions. A final regulatory guide on decommissioning reactor fire protection programs is expected to be issued in a few months. The remaining issues will be addressed by the decommissioning regulatory improvement initiative.

DLPM Public Comment # 5f: A member of the public stated that SECY 99-168 doesn't cover all decommissioning issues. The commenter stated that the issue of onsite disposal of clean waste (rubblization) needs clarification.

Response: Although outside the scope of the spent fuel pool risk study, development of NRC policy on rubblization is now ongoing in the Office of Nuclear Materials Safety and Safeguards.

DLPM Public Comment #6: A member of the public felt that decommissioning nuclear power plants should be evaluated for fires in the low level waste storage (LLW) area. This stakeholder states that large amounts of LLW could be stored in onsite LLW storage areas if offsite waste disposal sites are lost by a licensee "mid-stream" during the decommissioning process.

Response: As part of the staff's broad-scope decommissioning regulatory improvement effort, the staff will ensure that regulations are in place that would reasonably preclude threats to the public health and safety from accidents that are significantly less severe than a spent fuel pool zirconium fire but perhaps more probable, such as the LLW fire described above. To address the specific concern of the public stakeholder, 10 CFR 50.48 requires decommissioning nuclear power plant licensees to maintain a fire protection program to address fires which could cause the release or spread of radioactive materials which could result in a radiological hazard. In addition, nuclear power plants are also subject to the Commission's regulations for byproduct materials under 10 CFR Part 30. Specifically, 10 CFR 30.32(i) would require a licensee to maintain an appropriate EP program for radioactive materials stored onsite in quantities in excess of those specified in 10 CFR 30.72, "Schedule C - Quantities of Radioactive Material Requiring Consideration of the Need for an Emergency Plan for Responding to a Release." As part of the staff's recent effort on the integrated decommissioning rulemaking plan, the staff considered other less severe accidents with offsite consequences. The rulemaking plan recommends requiring licensees to perform reviews at their facilities to ensure that there are no other possible accidents that could result in offsite consequences exceeding EPA Protective Action Guidelines before reductions may be made in emergency preparedness and insurance requirements.

DLPM Public Comment #7: A member of the public stated the desire for an adjudicatory hearing and a prior NRC review/approval step at the onset of the decommissioning process.

Response: This issue of a hearing and NRC review and approval prior to decommissioning has been previously considered by the Commission. The Commission addressed the issue in the statements of consideration for the rulemaking for decommissioning published July 29, 1996, in the *Federal Register* (61 FR39278) by stating: "...initial decommissioning activities (dismantlement) are not significantly different from routine operational activities such as replacement or refurbishment. Because of the framework of regulatory provisions embodied in the licensing basis for the facility, these activities do not present significant safety issues for which an NRC decision would be warranted." Therefore, an NRC review and approval process that allows a public hearing before decommissioning begins is not necessary. Instead, in the 1996 rulemaking the Commission decided to offer a public hearing opportunity later in the decommissioning process at the license termination stage when issues such as to the adequacy of site cleanup could be raised.

DLPM Public Comment #9: A member of the public felt that the NRC should hire a contractor to determine why/how 10 CFR Part 50 was contorted to fit decommissioning reactors with the duct tape of 10 CFR 50.82 to avoid adjudicatory processes with regulatory handles.

Response: When the NRC issued decommissioning regulations in 1988, it was assumed that decommissioning would normally take place after the facility's operating license expired. The licensee was obligated to submit a preliminary decommissioning plan 5 years before the license expired. The preliminary decommissioning plan contained a cost estimate for decommissioning and an up-to-date technical assessment of the factors that could affect planning for decommissioning. This included (1) the choice of decommissioning alternative selected, (2) the major technical actions necessary to carry out decommissioning safely, (3) the current situation with regard to disposal of high-level and low-level radioactive waste, (4) the residual radioactivity criteria, and (5) other site-specific factors that could affect decommissioning planning and cost.

The 1988 rule also required that no later than 1 year before expiration of the license (or within 2 years of permanent cessation of operations for plants closing before their license expires), a licensee had to submit an application for authority to decommission the facility. The application was to be accompanied by or preceded by a proposed decommissioning plan. The proposed decommissioning plan was to include (1) the choice of the alternative for decommissioning with a description of the activities involved, (2) a description of controls and limits on procedures and equipment to protect occupational and public health and safety, (3) a description of the planned final radiation survey, (4) an updated cost estimate for the chosen alternative and a plan for ensuring the availability of adequate funding, and (5) a description of the technical specifications, quality assurance provisions, and physical security plan provisions in place during decommissioning. A supplemental environmental report that described any substantive environmental impacts that were anticipated but not already covered in other environmental impact documents was also required.

The NRC would review the decommissioning plan and would approve it by issuing an order if the plan demonstrated that the decommissioning would be performed in accordance with regulations and there were no security, health, or safety issues. The NRC would also require that notice be given to interested persons. However, the NRC could add other conditions and limits to the plan that it deemed appropriate. The license would then be terminated if the NRC determined that the decommissioning had been performed in accordance with the approved decommissioning plan and the order authorizing decommissioning, and if the final radiation survey and associated documentation demonstrated that the facility and site were suitable for release for unrestricted use.

In August 1996 the regulations were revised for several reasons. First, the experience gained in the early decommissioning activities associated with several facilities did not reveal any activities that required NRC review and approval of a decommissioning plan. Second, environmental impacts associated with decommissioning those early facilities resulted in impacts consistent with those evaluated in the "Generic Environmental Impact Statement on Decommissioning of Nuclear Facilities," NUREG-0586. And finally, experience gained from reviewing numerous decommissioning oversight activities at a number of these facilities also indicated that the decommissioning activities were in general no more complicated than activities normally undertaken at operating reactors without prior and specific NRC approval. The revised rule redefined the decommissioning process and required licensees to provide the NRC with early notification of planned decommissioning activities at their facilities went into effect. The rule made the decommissioning process more efficient and uniform. It provided for greater public awareness and clarified the opportunity for participation in the decommissioning process. It also gave plant personnel a clearer understanding of the process for changing from an operating organization to a decommissioning organization.

DLPM Public Comment #11: Untrained NRC public representatives frequently misinform the public, particularly about the opportunities for a hearing on reactor decommissioning.

Response: The NRC endeavors to train all NRC employees for their specific work assignments. In the event that misinformation is inadvertently communicated by an individual staff member, the NRC staff upon identifying the misinformation provides the correct information in the most expedient manner.

DLPM Public Comment #12: A member of the public cited several specific examples of interactions with NRC staff that he felt demonstrated improper or inaccurate information provided by NRC staff members.

Response: In the course of oral communication with the public in an open and unrestrained fashion, errors, miss-spoken words, and misunderstandings will occur by the individuals from the public and the NRC staff. The NRC endeavors to minimize these miss communications from our staff, but should they occur, NRC staff will act to correct them by the most expedient means available.

DLPM Public Comment #13: At the November 8, 1999, Commission meeting, a public stakeholder said that the time delays experienced by licensees who must submit individual heatup analyses and applications for exemption from NRC regulations could be mitigated by preparation of such documentation well in advance of decommissioning.

Response: It is true that decommissioning licensees who have planned reactor shutdown schedules far in advance would be able to submit exemption requests and conduct supporting thermal-hydraulic analyses in advance of reactor shutdown so that lengthy regulatory delays could be minimized. However, plants that shut down unexpectedly would not be able to submit such analyses in advance. The NRC believes that it should promulgate new decommissioning regulations that ensure public health and safety, reduce unnecessary regulatory burden and increase the efficiency and effectiveness of operations for both licensees and the NRC.

DLPM Public Comment #14: In a March 15, 2000, letter to the NRC, a public stakeholder, said that the NRC staff owes its stakeholders the courtesy of addressing their concerns, particularly when comments are solicited by the NRC staff. Otherwise, the NRC staff must stop actively soliciting public comment when it has no intention of considering.

Response: At the July 15-16, 1999 public workshop on decommissioning spent fuel pool risk, the public stakeholder raised a concern that the NRC evaluate potential hazards that decommissioning accidents could impose upon plant workers. When the NRC issued its final draft report, the stakeholder's issue was not specifically addressed in the comment evaluation section. However, the NRC had received an industry decommissioning commitment that licensees would provide a remote method of adding water to spent fuel pools that would reduce potential risk to plant workers and which resulted from the issue the stakeholder had raised. The NRC seriously considers public comments received on all issues within its jurisdiction. In this case, the staff regrets the appearance that a public comment had been ignored. In order to ensure that proper consideration was given to all stakeholder comments, the NRC staff reviewed all written comments received and examined transcripts of public meetings to ensure that all issues had been addressed. An evaluation of the stakeholder's initial concern on potential impacts to plant workers expressed at the July 1999 public workshop is included in the **IOLB Section of the REPORT** ?????**TANYA TO PROVIDE REFERENCE**?????????.

DLPM Public Comment #16: A member of the public requested on April 10, 2000, that the comment period on the spent fuel pool risk report be extended by 3 months.

Response: The original 45 day comment period ended on April 7, 2000. In a public meeting on May 9, 2000, NRC managers told the stakeholder that the comment period would be extended until June 9, 2000.

DLPM Public Comment #17: The NRC should identify and address possible conflicts of interests, and differing professional opinions as to the use of PRA (probabilistic risk assessment). For instance, Dr. Hanauer was quoted in a memo to say, "you can make probabilistic numbers prove anything, by which I mean that probabilistic numbers mean prove nothing."

Response: It is the policy of the Commission to maintain a working environment that encourages the employees to make known their best professional judgements even though they may differ from a prevailing staff view. An objective of this policy is to ensure full consideration and prompt disposition of differing opinions and views by affording an independent, impartial review by qualified personnel. The content of the quote is responded to in **DPLM public comment #18**

DLPM Public Comment #19: A stakeholder stated that the NRC should make references used in the spent fuel pool risk study available at no cost.

Response: The NRC policy is that all pertinent regulatory information is made available to the public via the Public Document Room and/or through the Agency Document and Management System (ADAMS) where this information is available for inspection at no charge. However, during the period of this study, the NRC took additional actions to provide the stakeholder with free copies of all routine correspondence and of numerous studies and reports that he specifically requested. Additionally, the NRC provided free copies of the draft June spent fuel pool risk study to all interested persons who attended the July 1999 public workshop and to all other members of the public who requested it.

DLPM Public Comment #20: A member of the public commented that changes to decommissioning regulations should be made on an interim basis, to be reviewed again at some future date.

Response: The NRC does not plan to issue interim regulations for decommissioning. Rulemaking is a methodical and deliberately lengthy procedure to ensure that a rule is not issued without due process. Provisions for public comment as well as independent review committees afford ample opportunity to examine a rulemaking prior to issuing a new rule. Any person who believes an NRC regulation is no longer applicable may petition the Commission to issue rescind, or amend that regulation in accordance with 10 CFR 2.802.

SPLB Public Comment #14:
"The Draft Study completely sidesteps the question of where all the people who are relocated will be able to go for the decades that must pass while the land where they live recovers from radioactive contamination. This issue is graphically illustrated by the consequences of the Chernobyl accident, which rendered huge land areas uninhabitable and unsuitable for agriculture for an extended period of time."

"Finally, the Draft Study fails entirely to address the social and economic implications of losing the use of thousands of square kilometers of land for several generations."

Response:
The staff agrees with the commenter that the study did not address the topics of relocation and societal impacts, such as land interdiction. The calculations in support of this risk study were performed following the principles and approach of Regulatory Guide (RG) 1.174, "An Approach For Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis," which does not include environmental considerations. While overall societal risk is not considered directly in RG 1.174, a large early release fraction (LERF) is used to gage the severity of the event outside of the plant boundary. Because RG 1.174 is applied to full power operation, the definition of LERF is not applicable to spent fuel pool accidents. Therefore, in this study, early fatalities were directly calculated and reported.

The Commission recently considered whether an additional agency safety goal or objective was needed to directly address land contamination and overall societal risk. It was decided by the Commission that the current policy would not change. For further discussion, read SECY-00-0077, dated March 30, 2000, and staff requirements memorandum dated June 27, 2000. Consistent with Commission guidance, the staff does not plan to include this issue in the study.

As part of its original licensing review, every operating plant had an environmental impact statement that addressed land use for the area surrounding that plant. When a plant enters decommissioning, an environmental assessment is performed to determine whether activities will remain bounded by that environmental impact statement.

INSURANCE

RGEB Public Comment #1: The obligation for decommissioning plants to participate in the secondary financial protection should be reviewed in light of the low public risk posed for SFPs for decommissioned plants. Industry does not believe that the risk justifies requiring participation. (The majority of the $3x10^{-6}$ risk of significant offsite consequences comes from an upper bound determination of the risk posed by seismic events, not on a best estimate of the seismic risk).

If it is determined that participation will be required during the short time that decommissioning plants pose a non-zero risk, then the level of participation should be in proportion to a best estimate of the risk posed relative to the risk posed by operating plants. If any participation is required, it should be only for the short period that clad surface temperatures greater than 570°C can occur in a loss of water configuration. The calculation of this temperature should be by an approved methodology.

The commenter also stated that the capacity required for primary financial protection should be eliminated for consideration of any potential for accidents with significant offsite consequences. For other events with offsite consequences, onsite coverage should be reduced to $25M for the period when the spent fuel remains in the pool and offsite coverage should be reduced to $5-10M. When the fuel has been removed offsite or placed in an offsite ISFSI, onsite coverage should be reduced to $25M while the site still contains significant sources of radioactive material. Onsite coverage could be reduced to zero when there are no sources exceeding 1000 gallons of fluid. Offsite coverage should be reduced to $5-10M for plants with fuel offsite or in an onsite ISFSI.

Response: The staff has previously stated that, while it is correct that the risk of a zirconium fire is not significant, the property and liability insurance requirements of our regulations are meant to ensure that the public is protected in the event of a low probability, high consequence event. The underlying purpose of 10 CFR 50.54(w) is to provide sufficient property damage insurance coverage to ensure funding for onsite post-accident recovery stabilization and decontamination costs in the unlikely event of a nuclear accident. Section 140.11 of Title 10 of the CFR also serves to provide sufficient liability insurance to ensure funding for claims resulting from a nuclear incident or precautionary evacuation.

In SECY-93-127, the Commission established that the amount of insurance coverage necessary for reactor licensees should be determined by the worst "reasonably conceivable" accident possible. Reasonably conceivable accidents may exceed design basis accidents but are less severe than remotely possible hypothetical accidents that are often termed "incredible." The TWG risk study concluded that the probability of a zirconium fire at a permanently shutdown plant is low but did not conclude that its probability is low enough to be considered "incredible." Also, the consequences of such a fire, which are severe in terms of property damage and land contamination, need to be considered. Thus, adequate insurance coverage is necessary for such an event.

Appendix 6c ACRS comments

SRXB Public Comment #13: The ACRS has difficulties with the time at which the risk of zirconium fires becomes negligible. Issues related with the formation of zirconium-hydride precipitates in the fuel cladding are spontaneously combustible in air. Spontaneous combustion of zirconium-hydrides would render moot the issue of "ignition" temperature which is the focus of the staff analysis of air interactions with exposed cladding. The staff neglected the issue of hydrides and suggested that uncertainties in the critical decay heat times and the critical temperatures can be found by sensitivity analysis. Sensitivity analysis with models lacking essential physics and chemistry would be of little use in determining the real uncertainties.

Response: Fuel cladding can contain high concentrations of zirconium hydride at the oxide-cladding interface in high burnup fuel. The effect of zirconium hydride on cladding oxidation rates is unknown at this time. If the oxide layer stays intact, the reaction rates should be similar to cladding oxidation rates without zirconium hydride since the rate is determined by the diffusion of oxygen through the zirconium oxide layer. The effect of the hydrogen reaction product on the oxide film and oxidation rate is unknown. It is possible that cladding rupture at a temperature near 700 °C may lead to autoignition of the cladding due to the reaction of oxygen with zirconium hydride. Air oxidation experiments with high burnup cladding are needed to resolve the reaction rate and autoignition issues.

SRXB Public Comment #14: The staff analysis of the interaction of air with cladding has relied heavily on geriatric work. New findings through a cooperative international program PHEBUS FP provide information relating to the well-known tendency for zirconium to undergo breakaway oxidation in air whereas no tendency is encountered in steam or in pure oxygen. Other findings relate to how nitrogen from air depleted of oxygen will interact exothermically with zircaloy [zirconium alloy] cladding. The ACRS does not accept the staff's claim that it has performed "bounding" calculations of the heatup of Zircaloy clad fuel even when it neglects heat losses.

Response: Breakaway oxidation can have a significant impact. Breakaway oxidation has been observed to occur in experiments Ref [6,7] measuring oxidation rates of zirconium and Zircaloy-4 in air. Breakaway oxidation has not been observed in pure oxygen. The lower temperature limit for breakaway oxidation in Zircaloy-2, Zircaloy-4 or any advanced zirconium alloy is unknown. An experimental program would be required to quantify the effect of this potentially important physical phenomenon. The experiments should examine the effect of fuel burnup on this phenomenon. The limited data available indicates that the lower temperature limit for breakaway oxidation in Zircaloy-4 is lower than the lower limit observed in pure zirconium but the lower limit has not been determined. The mechanisms that induce breakaway oxidation are unknown at the present time. Therefore data should be taken under conditions that are as prototypical as can be achieved.

SRXB Public Comment #15: Since the staff has neglected any reaction with nitrogen and did not consider breakaway oxidation, it had not made an appropriate analysis to find this "ignition temperature". (from the ACRS)

Response: It has been shown that the presence of nitrogen increases the rate of oxidation of zirconium. The oxidation rate is a weekly increasing function of nitrogen fraction over a wide range of relative nitrogen fractions. [Ref 6] The reaction rate of nitrogen with zirconium is approximately 20 times lower than the oxidation rate. The energy of reaction of zirconium with nitrogen is also less than the energy of reaction with oxygen. Therefore, the heat input from the nitrogen reaction should be a small perturbation to the oxidation heat input except for very low oxygen concentrations and in that case the fuel has already reached its failure point and a large release is underway.

SRXB Public Comment #16: The search for ignition temperature may be the wrong criterion for the analysis. The staff should be looking at the point at which cladding ruptures and fission products can be released. One arrives at a lower temperature criteria for concern over the release of radionuclides. (From the ACRS)

Response: Cladding rupture can release gap gases. Additionally the interaction of the fuel with air can cause the release of fuel fines and fission products such as ruthenium trapped in the fuel that will provide a source term that significantly exceeds the classical gap release.

SRXB Public Comment #17: The staff focuses on eutectic formations when intermetallic reactions are more germane to the issues at hand.

Response: **RES has not provided the information needed to evaluate this.**

SPSB Public comment # 21: The ruthenium inventory in spent fuel is substantial. Ruthenium has a biological effectiveness equivalent to that of Iodine-131 and has a relatively long half-life. If there were significant releases of ruthenium in a zirconium fire, the Regulatory Guide 1.174 large early release frequency (LERF) value might not be an appropriate surrogate for the prompt fatality quantitative health objective. The controlling consequence may become latent cancer deaths.

Response: The staff's conclusion in the draft final report was that, even though there are some differences in source term and timing, scenarios involving a spent fuel pool zirconium fire would result in population doses that are generally comparable to those expected from accident scenarios at operating reactors. Since a zirconium fire in the SFP would involve a direct release to the environment, the LERF guideline was applied. The staff reassessed these conclusions following the performance of additional consequence calculations that took into account the possibility of significant ruthenium release fractions.

The staff's reassessment showed that, when the ruthenium release fraction was increased to 100% from the originally assumed fraction of $2x10^{-5}$, the number of early fatalities increased by approximately two orders of magnitude. However, the resulting early fatality consequences are still relatively low when compared to those predicted for operating reactor accidents. For example, for the various source terms considered in the NUREG-1150 assessment of Surry, the conditional number of early fatalities varied from essentially zero to approximately 11. The reassessment for SFP zirconium fire consequences (assuming 100% ruthenium release fraction, and a population distribution like Surry) indicated conditional prompt fatalities of 0.13 for the scenarios where evacuation was initiated before onset of a zirconium fire.

When considering latent cancer fatalities, the staff analysis also provided a sensitivity study for total latent cancer deaths up to 500 miles away, with and without the increased ruthenium release fraction. For the situation where evacuation is initiated prior to zirconium fire, latent cancer fatalities increased by approximately 17%, indicating that latent effects were only slightly sensitive to the ruthenium release fraction. It should also be acknowledged that these long term health impacts are sensitive to public policy decisions such as land interdiction criteria for returning populations. Appendix 4 of this report discusses this issue.

SPSB Public comment #22: (moved to seismic section and then to here) The seismic risk was treated in a conservative manner. Risk-informed decision making regarding spent fuel pool zirconium fire issues should use realistic analysis, including uncertainty assessment.

Response: The assessments of the frequency of fuel uncovery from seismic events were performed using the Lawrence Livermore National Laboratory (LLNL) seismic hazard curves. The LLNL hazard curves are generally conservative with respect to those generated by EPRI. This is a result of different expert judgements. An assumed HCLPF (high confidence of low probability of failure) value of 0.5g was used in the seismic analysis. The HCLPF value was chosen on the basis that it was the value that was felt to be attainable by a plant that met the seismic checklist (see Appendix 5). It was recognized by the staff that the HCLPF value at a plant could be greater than 0.5g (i.e., the plant might actually have a higher capacity than the minimum predicted if the checklist were met.) However, in the absence of plant-specific assessments of fuel pool capacities, this is a good approximation, which is bounding. The draft report also states that the approach used to evaluate the frequency gives a slightly conservative estimate of the mean value that would be calculated from a convolution of the hazard curve and the fragility curve. Since the treatment of uncertainties is an inherent part of the development of the hazard curves and the fragility curves, this mean value does indeed address uncertainties. While it can be concluded that the frequency of fuel uncovery from seismic events is potentially conservative, it is not considered by the staff that this will impact the quality of the decisions that will be made on a generic basis using this information.

SPSB Public comment #23: Because the accident analysis is dominated by sequences involving human errors and seismic events that involve large uncertainties, the absence of an uncertainty analysis of frequencies of accidents is unacceptable. Absent knowledge of the uncertainties, the decision making process is flawed.

Response: The staff intends to use the decommissioning spent fuel pool risk assessment results and insights in decision making based on the principles used in Regulatory Guide (RG) 1.174. In this approach, when acceptance (in this case performance) guidelines are established, it is understood that the appropriate measure with which to make the comparison is the mean value of a distribution characterizing the quantified uncertainty. Uncertainties that cannot be incorporated into this quantification and that are usually associated with modeling issues or the adoption of specific assumptions are to be addressed in the decision making process. The uncertainties in the decision making process are addressed by demonstrating that the adoption of alternate, plausible modeling assumptions would not lead to a change in the conclusion that the guidelines have (or have not) been met.

Seismic analysis and the assessment of the human performance in response to losses of heat removal and fuel pool inventory were pointed out as having large uncertainties. With respect to the accident sequences developed using a detailed logic model for losses of heat removal and pool inventory, the frequencies generated for those sequences are point estimates, based on the use of point estimates for the input parameters. The input parameter values were taken from a variety of sources, and in many cases were presented as point estimates with no characterization of uncertainty. In some cases, such as the initiating event frequencies derived from NUREG/CR 5496 and the human error probabilities (HEPs) derived from THERP (Technique for Human Error Rate Prediction), an uncertainty characterization was given, and the point estimates chosen corresponded to the mean values of the distributions characterizing uncertainty. For all other parameters, it was assumed that the values would be the mean values of distributions characterizing the uncertainty on the parameter value. In the case of the Simplified Plant Risk (SPAR) HEPs, the authors of the SPAR human reliability analysis approach consider their estimates to be mean values since the numbers were established on the basis of considering several different sources, most of which specified mean values. Consequently, the results of this analysis are interpreted as being mean values.

A propagation of parameter uncertainty through the model was not performed, nor was it considered necessary. With the exception of the spent fuel pool cooling system itself, the systems relied on are single train systems. The dominant failure contributions for the spent fuel pool cooling system are assumed to be common cause failures. Thus, there are no dominant cutsets in the solutions that involve multiple repetitions of the same parameter and under these conditions, use of mean values as input parameters produces a very close approximation to mean values of sequence frequencies. Since typical uncertainty characterization for the input parameters is a lognormal distribution with error factors of 3 or 10, the 95th percentile of the output distribution will be no more than a factor of three higher than the mean value. This is not significant enough to change the conclusion of the analysis.

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

Certain assumptions may be identified as having the potential for significantly influencing the results. For example, the calculated time windows associated with the loss of inventory event tree are sensitive to the assumptions about the leak rate. The SPAR HRA method is, however, not highly sensitive to the time windows within the ranges determined to be plausible for the scenarios modeled. Consequently, the assumption of the large leak rate as 60 gpm to represent those leaks that require isolation is not critical. For the loss of inventory event tree, the assumption that the leak is self-limiting after a drop in level of 15 feet may be a more significant assumption that, on a site-specific basis, may be non-conservative and requires validation. The assumption that the preparation time of several days is adequate to bring off-site sources to bear may be questioned in the case of extreme conditions. However, the very conservative assumption that offsite recovery is guaranteed to fail would increase the corresponding event sequences by about an order of magnitude, which would still be a very low risk contributor. In conclusion, the staff considers that, by determining that the estimates for the sequence frequencies are equivalent to mean values, and in identifying those assumptions that could affect the numerical results, and in understanding the effects of these assumptions on the numerical results, the uncertainty analysis performed is sufficient to support the decision making process.

RES Public Comment #1:
The staff made additional MACCS calculations which assumed 100% release of the ruthenium inventory. For a 1 year decay time with no evacuation, the prompt fatalities increase by 2 orders of magnitude over those in the draft report which did not include ruthenium release. The societal dose doubled, and the cancer fatalities increased four-fold. [Ref. 11]

Response: The staff has included, in Appendix 4A, the additional MACCS calculations with a large ruthenium release fraction. These calculations show an increase in consequences over the cases with the small ruthenium release fraction characteristic of fission product releases under steam conditions. However, the increased consequences resulting from a large ruthenium release are demonstrated to be largely offset by a consequence reduction due to early evacuation which is likely given the long time it takes for a spent fuel pool to heat up.

RES Public Comment #2:
The ACRS is concerned about the appropriateness of the source term used in the study. The staff did consider the possibility that "fuel fines" could be released from fuel with ruptured cladding (as a result of decrepitation). It did not, believe these fuel fines could escape from the plant site. Evidence suggests that fuel fines could be entrained in the vigorous natural convection flows produced in a SFP accident. Nevertheless, the staff considered the effect of 6×10^{-6} release fraction of fines. This minuscule release fraction did not affect the calculated findings. There is no reason to think that such a low release fraction would be encountered with decrepitating fuel.

Response: The staff has included, in Appendix 4A, additional MACCS calculations with a fuel fines release fractions of .001 and .01. These calculations show a negligible to modest (less than 40%) increase in consequences.

RES Public Comment #3:
The uncertainties associated with many of the critical features of the MACCS code do not seem to have been considered in the analyses of the SFP accident.

-One of the uncertainties is that the spread of the radioactive plume from a power plant site is much larger than what is taken as the default spread in the MACCS calculations.

- The initial plume energy assumed in the MACCS calculations, which determines the extent of plume rise, was taken to be the same as that of a reactor accident rather than one appropriate for a zirconium fire.
-The consequences found by the staff tend to overestimate prompt fatalities and underestimate latent fatalities just because of the narrow plume used in the MACCS calculations and the assumed default plume energy.

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

The staff has included, in Appendix 4A, additional MACCS calculations using different plume heat contents and updated values for the plume-spreading model parameters. The sensitivity calculations showed that increasing the plume heat content resulted in reductions in early fatalities and no change in societal dose or cancer fatalities. In addition, updating the values of the plume-spreading model parameters to those in NUREG/CR-6244 results in a decrease in early fatalities and up to a 60% increase in societal dose and cancer fatalities, because of the additional plume spreading associated with the updated values.

RES Public Comment #4:
The staff needs to review the air oxidation fission products release data from Oak Ridge National Laboratory and from Canada that found large releases of cesium, tellurium, and ruthenium at temperatures lower than 1000°C. Based on these release values for ruthenium, and incorporating uncertainties in the MACCS plume dispersal models, the consequence analysis should be redone.

Response: The release values for ruthenium and the uncertainties in the MACCS plume dispersal models are discussed in the responses to Public Comment #1 and Public Comment #3, respectively. The consequence evaluation documented in Appendix 4 uses a cesium release fraction of one and a tellurium release fraction of .02. Also, the staff has included, in Appendix 4A, additional MACCS calculations using a tellurium release fraction of .75. No change in consequences were seen, because of the small inventories of the tellurium isotopes after one year of decay.