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Assessment of the Potential for Criticality in Decommissioned Spent Fuel Pools

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Introduction

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

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

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

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

Description Of Methods

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

Problem Definition

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

Discussion Of Results

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

PWR Spent Fuel Storage Racks

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

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

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

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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

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```
=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
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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
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  1  1  2  1  1  2  1  1  1  2  1  1  2  1  1
  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
  1  1  1  1  1  1  1  1  1  1  1  1  1  1  1
end fill
end array
```

```
read bounds all=mirror end bounds
read mixt sct=2 eps=1.e-01 end mixt
read plot
scr=yes
ttl='w15x15 in High Density Rack'
xul=-11.5 yul= 11.5 zul=0.0
xlr= 11.5 ylr=-11.5 zlr=0.0
uax=1 vdn=-1 nax=750
end plot
end data
end
```

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

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

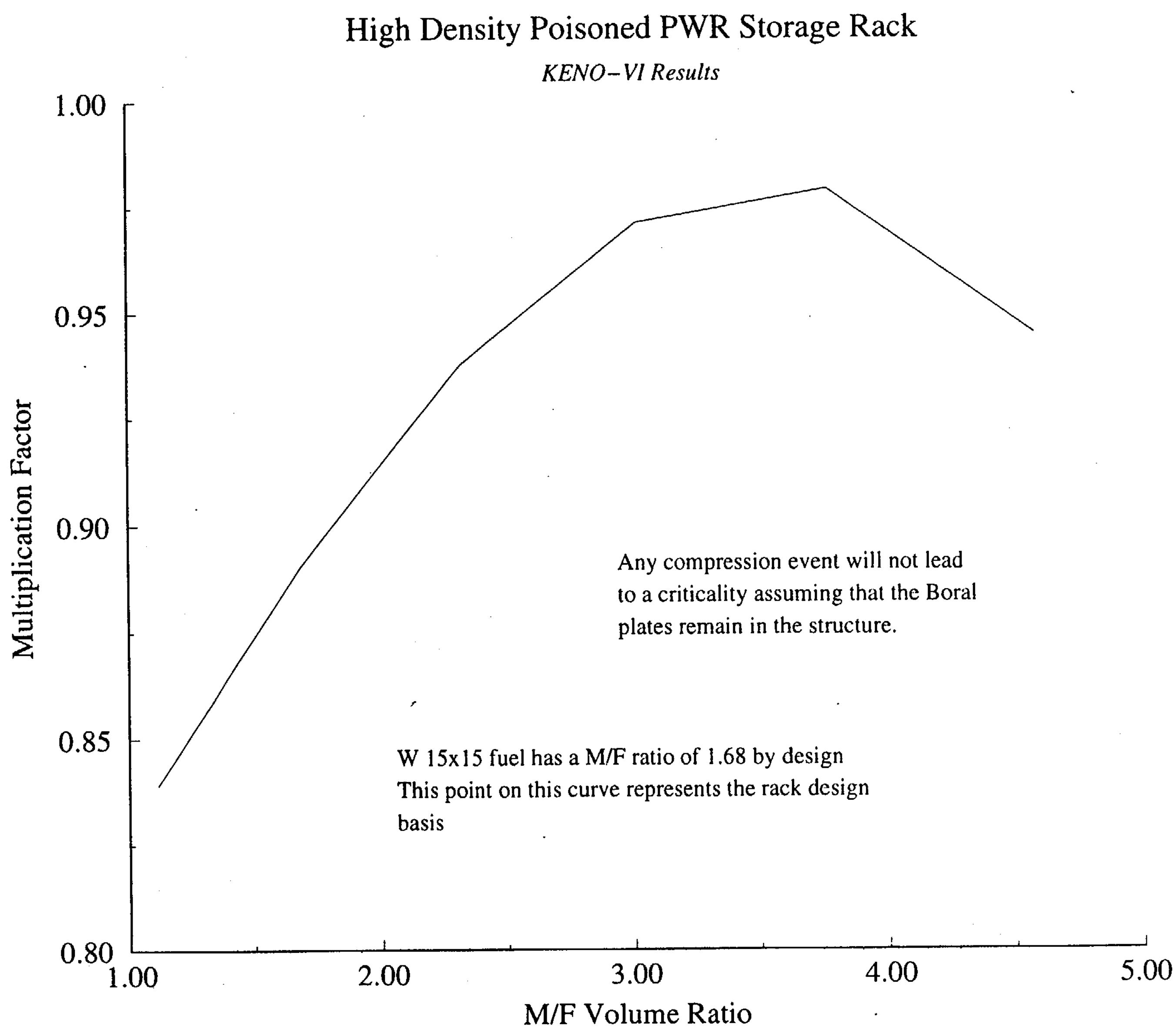

Figure 1 PWR High Density Storage Rack Eigenvalue following Compressive/Expansion Events

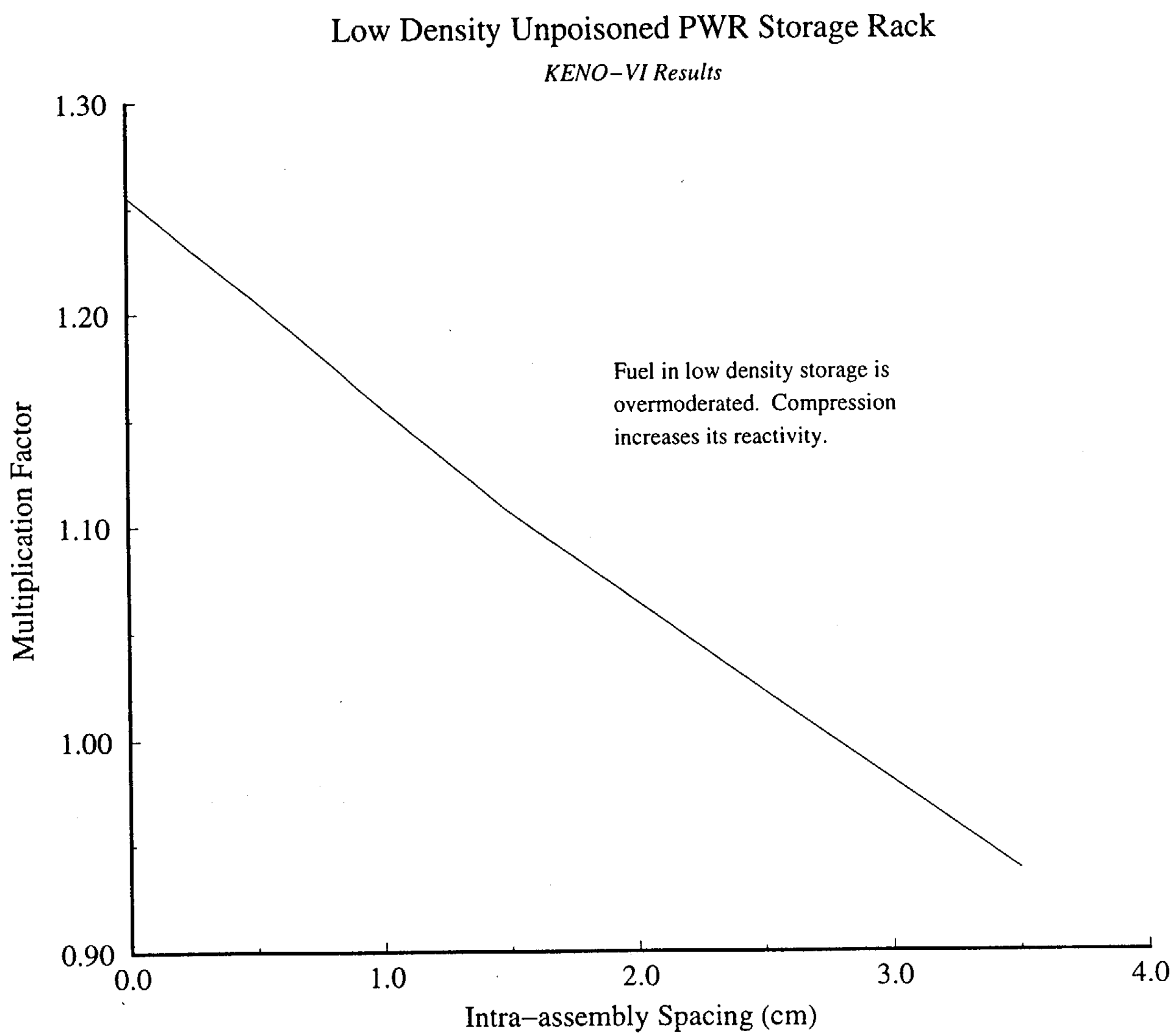

Figure 2 PWR Low Density Storage Rack Eigenvalue following Compressive/Expansion Events

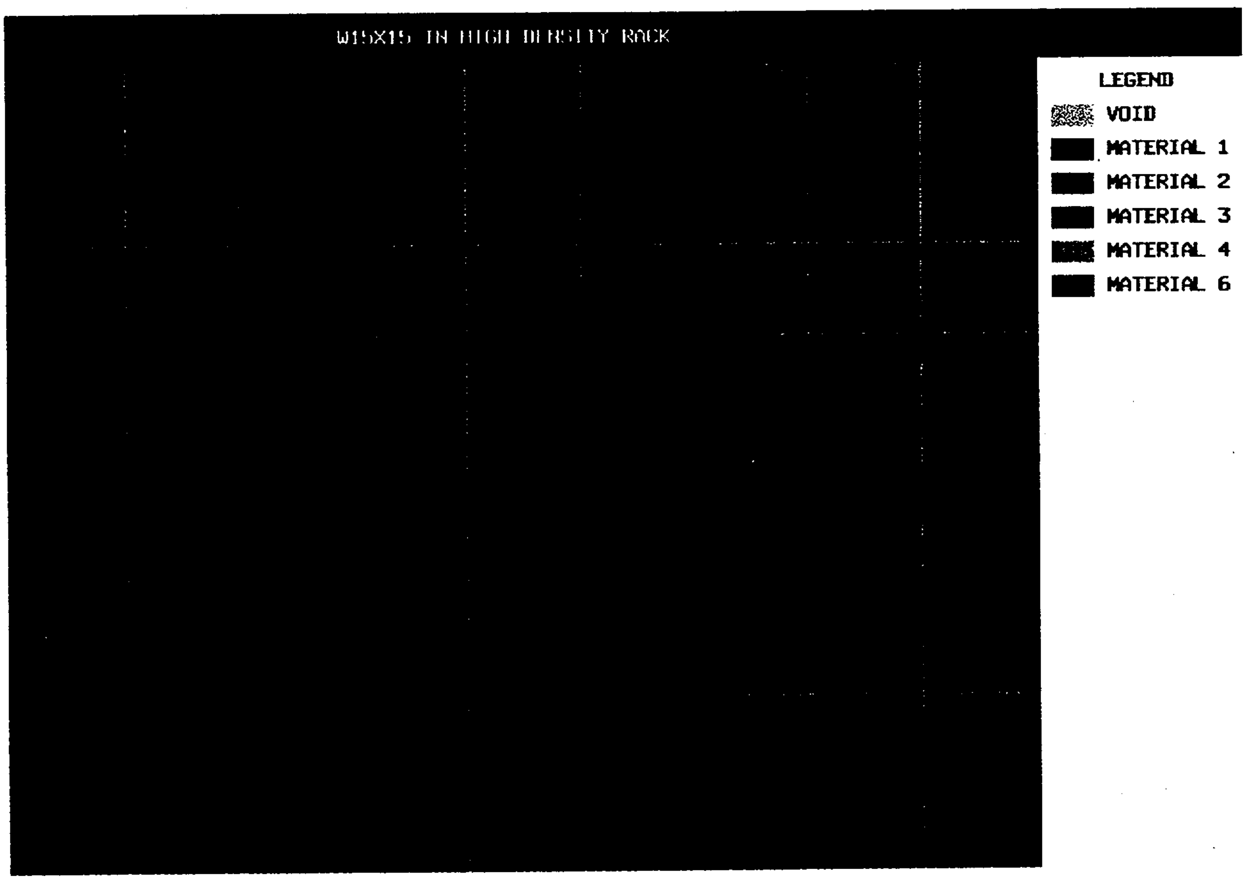

Figure 3 Sample Geometry Assuming 4 Assembly Spacing Between Most Reactive Assembly

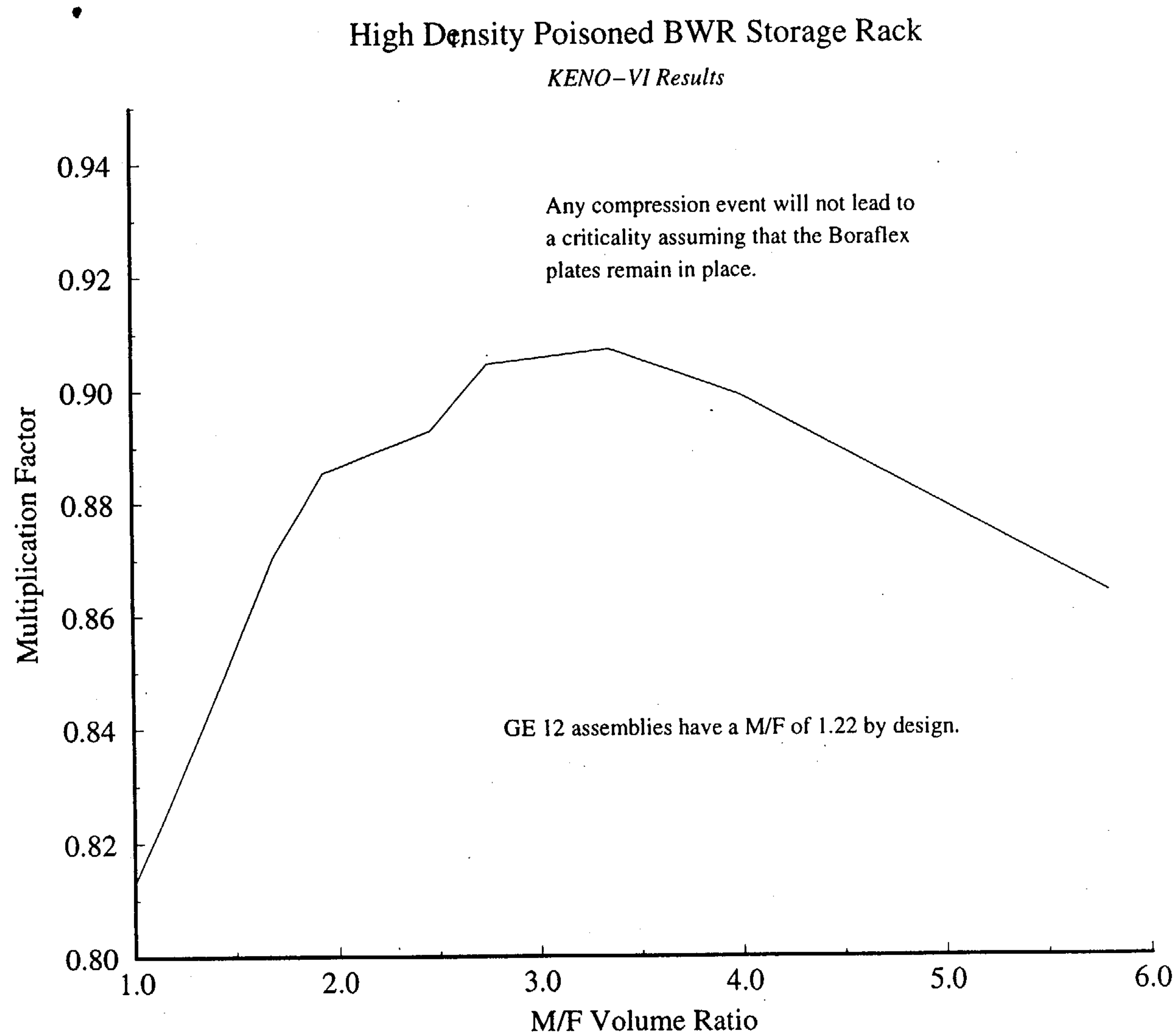

Figure 4 BWR High Density Storage Rack Eigenvalue following Compressive/Expansion Events

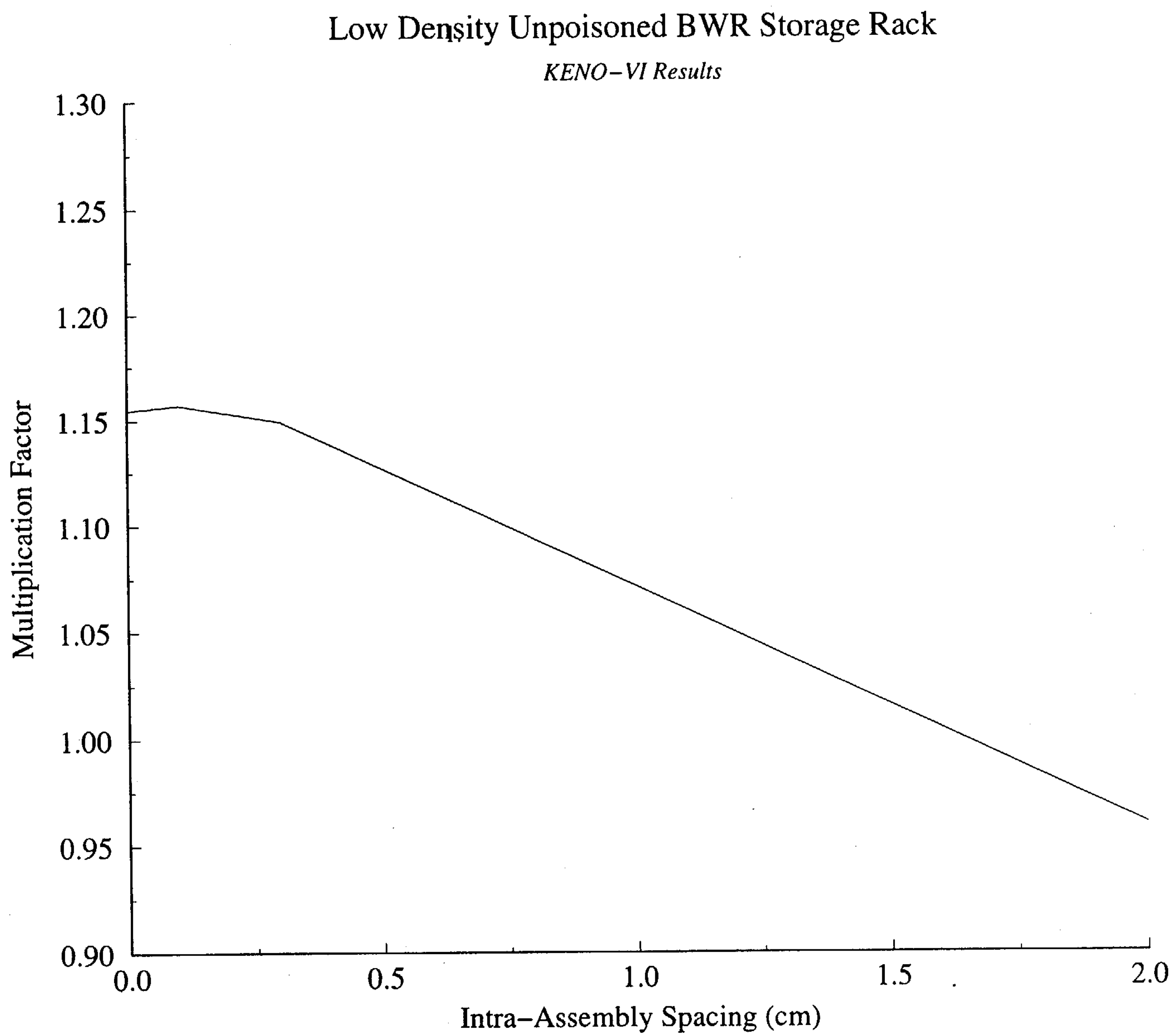

Figure 5 BWR Low Density Storage Rack Eigenvalue following Compressive/Expansion Events

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App w/ Consequence Assessment

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Appendix 4 Consequence Assessment from Zirconium Fire

Introduction

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

The purpose of this report is to document the detailed technical basis of the offsite consequence evaluation. This report documents the offsite consequence calculations we performed using the MACCS code (MELCOR Accident Consequence Code System) and includes the input files used. In addition, this report documents follow-up calculations, performed since our earlier letter, to evaluate the importance of cesium to better understand why the consequence reduction from a year of decay was not greater. These follow-up calculations showed that cesium with its long half-life (30 years) is responsible for limiting the consequence reduction. For the population within 100 miles of the site, 97 percent of the societal dose was from cesium.

Previous Consequence Assessments

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

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

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

Radionuclide Inventories

As discussed above, the current analysis was undertaken to assess the magnitude of the decrease in offsite consequences that could result from up to a year of decay in the spent fuel pool. To perform this work, it was necessary to have radionuclide inventories in the spent fuel pool for a decommissioned reactor at times up to 1 year after final shutdown. The inventories in the NUREG/CR-6451 analysis have not been retrievable, so those inventories could not be used. NUREG/CR-4982 contains spent fuel pool inventories for two operating reactors, a BWR (Millstone 1) and a PWR (Ginna). Because the current analysis may also be used as part of the probabilistic risk analysis of spent fuel pool accidents for the Susquehanna plant which is a BWR, the spent fuel inventories for Millstone 1 which is also a BWR were used for this analysis. These spent fuel pool inventories for Millstone 1 are given in Table 4.1 of NUREG/CR-4982 and are reproduced in Table A4-1 below. Two adjustments were then made to the Table A4-1 inventories. The first adjustment was to multiply the inventories by a factor of 1.7, because the thermal power of Susquehanna is 1.7 times higher than that of Millstone 1. The second adjustment, described in the next two paragraphs, was needed because NUREG/CR-4982 was for an operating reactor and this analysis is for a decommissioned reactor.

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

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

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

Radionuclide	Half-Life	Spent Fuel Pool Inventory (Ci)		
		30 days after last discharge	90 days after last discharge	1 year after last discharge
Co-58	70.9d	2.29E4	1.26E4	8.54E2
Co-60	5.3y	3.72E5	3.15E5	2.85E5
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 offsite consequences for reactor accidents. NUREG/CR-4982 contains inventories for 40 of these 60 radionuclides. Of these 40 radionuclides, 27 have half-lives from 2.4 days to a year and 13 have half-lives of a year or greater as shown in Table A4-1. The half-lives of the remaining 20 radionuclides range from 53 minutes to 1.5 days as shown in Table A4-2. Because the largest half-life of these 20 radionuclides is 1.5 days, omitting these 20 radionuclides from the initial inventories used in the MACCS analysis should not affect doses from releases occurring after a number of days of decay.

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

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

Release Fractions

NUREG/CR-4982 also provided the fission product release fractions assumed for a severe spent fuel pool accident. These fission product release fractions are shown in Table A4-3. NUREG/CR-6451 provided an updated estimate of fission product release fractions. The release fractions in NUREG/CR-6451 (also shown in Table A4-3) are the same as those in NUREG/CR-4982, with the exception of lanthanum and cerium. NUREG/CR-6451 stated that the release fraction of lanthanum and cerium should be increased from $1x10^{-6}$ in NUREG/CR-4982 to $6x10^{-6}$, because fuel fines could be released offsite from fuel with high burnup. While RES believes that it is unlikely that fuel fines would be released offsite in any substantial amount, a sensitivity was performed using a release fraction of $6x10^{-6}$ for lanthanum and cerium to determine whether such an increase could even impact offsite consequences.

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

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

Modeling of Emergency Response Actions and Other Areas

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

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

Table A4-4 Timing of Events

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

Table A4-5 Evacuation Modeling

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

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

Offsite Consequence Results

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

Table A4-6 Cases Examined Using the MACCS2 Consequence Code

Case	Population Distribution	Radionuclide Inventory	Evacuation Start Time	La/Ce Release Fraction	Evacuation Percentage
Base Case	Surry	11 batches plus rest of last core	1.4 hours after release begins	$1x10^{-6}$	99.5%
1	Surry	11 batches plus rest of last core	1.4 hours after release begins	$1x10^{-6}$	95%
2	Surry	11 batches	1.4 hours after release begins	$1x10^{-6}$	95%
3	100 people/mi^2	11 batches	1.4 hours after release begins	$1x10^{-6}$	95%
4	100 people/mi^2	11 batches plus rest of last core	1.4 hours after release begins	$1x10^{-6}$	95%
5	100 people/mi^2	11 batches plus rest of last core	3 hours before release begins	$1x10^{-6}$	95%
6	100 people/mi^2	11 batches plus rest of last core	3 hours before release begins	$6x10^{-6}$	95%
7	100 people/mi^2	11 batches plus	3 hours	$1x10^{-6}$	99.5%

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

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

Table A4-7 Mean Consequences for the Base Case

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

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

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

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

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

Table A4-8 Mean consequences for Case 2

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
30 days	0-100	.89	44,900	2,280
	0-500	.89	557,000	25,100

90 days	0-100	.78	44,500	2,250
	0-500	.78	554,000	25,000
1 year	0-100	.53	43,400	2,180
	0-500	.53	567,000	25,500

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

Table A4-9 Mean Consequences for Case 3

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

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

Table A4-10 Mean Consequences for Case 4

Decay Time in Spent Fuel Pool	Distance (miles)	Prompt Fatalities	Societal Dose (person-Sv)	Cancer Fatalities
30 days	0-100	18.3	53,500	2,610
	0-500	18.3	454,000	20,600
90 days	0-100	16.3	52,100	2,560
	0-500	16.3	465,000	21,100
1 year	0-100	12.7	50,900	2,490
	0-500	12.7	477,000	21,600

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

Table A4-11 Mean Consequences for Case 5

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

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

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

Table A4-12 Mean Consequences for Case 7

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

Comparison with Earlier Consequence Analyses

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

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

Table A4-13 Comparison of Analysis Assumptions

Parameter	NUREG/CR-4982 (BWR)	NUREG/CR-6451 (BWR)	Case 3	Case 4
population density (persons/mile^2)	100	<u>0-30 mi</u>: 1000 <u>30-50 mi</u>: 2300 (city of 10 million people, 280 outside of city) <u>50-500 mi</u>: 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 offsite consequences.

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

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

Conclusion

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

References:

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

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

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

App E Seismic History

Bldg # 67

Seismic Screening Criteria

for

Assessing Potential Fuel Pool Vulnerabilities

at

Decommissioning Plants

December 13, 1999
Revision 1

Background

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

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

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

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

Purpose of Checklist

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

DRAFT CHECKLIST

Item 1:

Requirement: Identify Preexisting Concrete and Liner Plate Degradation

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

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

Item 2:

Requirement: **Assure Adequate Ductility of Shear Wall Structures**

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

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

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

Item 3:

Requirement: **Assure Design adequacy of Diaphragms (including roofs)**

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

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

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

Item 4:

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

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

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

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

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

Item 5:

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

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

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

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

Item 6:

Requirement: **Verify the Adequacy of Spent Fuel Pool Penetrations**

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

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

Item 7:

Requirement: **Evaluate the Potential for Impacts with Adjacent Structures**

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

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

Item 8:

Requirement: **Evaluate the Potential for Dropped Loads**

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

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

Item 9:

Requirement: **Evaluation of Other Failure Modes**

Basis Experienced seismic engineers should review the geotechnical and structural design details for the specific site and assure that there are not any design vulnerabilities which will not be adequately addressed by the review areas listed above. Soil-related failure modes including liquefaction and slope instability should be screened by the approaches outlined in Reference 1 (Section 7 & Appendix C).

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

Item 10: **Potential Mitigation Measures**

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

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

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

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

Item 11: **Required Documentation**

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

References:

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

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

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

Summary of NRC Draft

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

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

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

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

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

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

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

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

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

Preliminary Industry Comments

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

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

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

Comments on Recommendation Number 4.b

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

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

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

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

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

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

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

proposed that the results from each study be geometrically averaged such that equal weight is provided the results from each study. Arithmetic averaging is considered unacceptable in light of the difficulties associated with the LLNL results. Figure 4 provides the results of geometrically averaging the LLNL and EPRI results.

Comments on Recommendation Number 4.a

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

Comments on Recommendation Number 4.c

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

Comments on Recommendation Number 5

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

References:

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

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

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

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

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

5f December 28, 1999 Kennedy Letter

Structural Mechanics Consulting, Inc.

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

December 28, 1999

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

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

Dear Dr. Hofmayer:

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

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

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

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

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

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

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

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

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

Sincerely

Robert Kennedy

cc. Mr. Goutam Bagchi
Dr. Nilesh Chokshi

5g Enhanced Seismic Checklist

Item 1:

Requirement: Identify Preexisting Concrete and Liner Plate Degradation

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

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

Item 2:

Requirement: Assure Adequate Ductility of Shear Wall Structures

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

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

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

Item 3:

Requirement: Assure Design adequacy of Diaphragms (including roofs)

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

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

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

Item 4:

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

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

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

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

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

Item 5:

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

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

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

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

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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.

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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.

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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 $1x10^{-6}$ per year.

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5h Minor Changes Made By the NRC to the NEI seismic checklist

Appendix 5h shows minor changes made by the NRC to the NEI seismic checklist, Revision 1, dated December 13, 1999.

Seismic Screening Criteria

for

Assessing Potential Fuel Pool Vulnerabilities

at

Decommissioning Plants

December 13, 1999
Revision 1 With NRC Suggested Word Changes

Background

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

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

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

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

Purpose of Checklist

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

DRAFT CHECKLIST

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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.

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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's SFP 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.

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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 with ground motion of three times the SSE with appropriate consideration of amplified motion for locations in above ground spent fuel pool structures~~ in the 0.45-0.5g pga range~~.

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

Item 4:

Requirement: **Verify the Adequacy of the SFP Walls and Floor Slab to Resist**

Out-of-Plane Shear and Flexural Loads

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

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

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

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

Item 5:

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

Basis: At a number of older nuclear power plants, the walls and roof above the top of the spent fuel pool are constructed of structural steel. These steel frames were generally designed to resist hurricane and tornado wind loads which exceeded the anticipated design basis seismic loads. A review of these steel (or possibly concrete) framed structures should be performed to assure that they can resist the seismic forces resulting from a beyond-design-basis seismic event with ground motion of three times the SSE ~~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 with ground motion of three times the SSE ~~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 with ground motion of three times the SSE ~~in the 0.45-0.5g pga range~~ has the potential to cause the structural collapse of masonry walls and/or equipment supports systems. If these secondary structural failures could result in the accidental dropping of heavy loads which are always present (i.e. not loads associated with cask movements) into the SFP, then the consequences of these drops must be considered. As in previous evaluations, the focus of the drop consequence analyses should consider the possibility of draining the SFP. Additionally, the evaluation should evaluate the consequences of any resulting damage to the spent fuel or to the spent fuel storage racks.

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

Item 9:

Requirement: **Evaluation of Other Failure Modes**

Basis Experienced seismic engineers should review the geotechnical and structural design details for the specific site and assure that there are not any design vulnerabilities which will not be adequately addressed by the review areas listed above. Soil-related failure modes including liquefaction and slope instability should be screened by the approaches outlined in Reference 1 (Section 7 & Appendix C).

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

Item 10: **Potential Mitigation Measures**

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

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

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

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

Item 11: Required Documentation

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

References:

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

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5i Other Seismic Stakeholder Interactions

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

Stakeholder Comment

"I guess I'd like to direct my questions to the seismological review for this risk-informed process. And first of all, did any of the NUREGs that you looked at take into account new information coming out of the Kobe and Northridge events? I think that what we need to be concerned with is dated information. Particularly as we are learning more about risks associated with those two particular seismological events that were never even considered when plants were sited; particularly, though I can't frame it in the seismological language, from a lay understanding, it's clear that new information was gained out of Kobe and Northridge events suggesting that you can have seismological effects of greater consequence farther afield than at the epicenter of the event."

Response

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

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

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

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

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

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

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

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

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

2. During the July workshop, members of the public raised concerns about the hazard of the fuel transfer tube interacting with the pool structure during a large earthquake. There was also another concern about the effect of aging on the spent fuel pool liner plate and the reinforced concrete pool structure.

   Transfer tubes are generally used in PWR plants where the fuel assembly exits the containment structure through the tube and enters the pool. These transfer tubes are generally located inside a concrete structure that is buried under the ground and attached to the pool structure through a seismic gap and seal arrangement. These layouts and arrangements can vary from one PWR plant to another, and the seismic hazard caused by transfer tubes should be examined on a case-by-case basis. This is included in the seismic checklist.

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

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

With aging, concrete gains compressive strength of about 20% in an asymptotic manner and spent fuel pool structures are expected to have this increased strength at the time of their decommissioning. Degradation of concrete structures can be divided into two parts, long term and short term. The long-term degradation can occur due to freezing and thawing effects when concrete is exposed to outside air. This is the predominant long-term failure mode of concrete; observed on bridge decks, pavements, and structures exposed to weather. Degradation of concrete can also occur when chemical contaminants attack concrete. These types of degradation have not been observed in spent fuel pools in any of the operating reactors. Additionally, inspection and maintenance of spent fuel pool structures are within the scope of the maintenance rule, 10 CFR 50.65, and corrective actions are required if any degradation is observed. An inspection of the spent fuel pool structure to identify cracks, spalling of concrete, etc., is also recommended as a part of the seismic checklist. Significant degradation of reinforced concrete structures would take more than 5 years or so, the time necessary to lose decay heat in the spent fuel. Substantial loss of structural strength requires long-term corrosion of reinforcing steel bars and substantial cracking of concrete. This is not likely to happen because of inspection and maintenance requirements.

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

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

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

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

NUCLEAR ENERGY INSTITUTE

Alan Nelson
SENIOR PROJECT MANAGER,
PLANT SUPPORT
NUCLEAR GENERATION DIVISION

December 13, 1999

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

Dear Mr. Huffman:

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

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

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

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

Sincerely,

Alan Nelson

Alan Nelson
APN/dc
Enclosure

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.

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2. During the July workshop, members of the public raised concerns about the hazard of the fuel transfer tube interacting with the pool structure during a large earthquake. There was also another concern about the effect of aging on the spent fuel pool liner plate and the reinforced concrete pool structure.

   Transfer tubes are generally used in PWR plants where the fuel assembly exits the containment structure through the tube and enters the pool. These transfer tubes are generally located inside a concrete structure that is buried under the ground and attached to the pool structure through a seismic gap and seal arrangement. These layouts and arrangements can vary from one PWR plant to another, and the seismic hazard caused by transfer tubes needs to be examined on a case-by-case basis. This is a good candidate for a seismic checklist. This has been changed in the on 1/20/00 4PM

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

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

   With aging, concrete gains compressive strength of about 20% in an asymptotic manner and spent fuel pool structures are expected to have this increased strength at the time of their decommissioning. Degradation of concrete structures can be divided into two parts, a long term and short term. The long-term degradation can occur due to freezing and thawing effects when concrete is exposed to outside air. This is the predominant long-term failure mode of concrete, observed on bridge decks, pavements, and structures exposed to weather. Degradation of concrete can also occur when chemical contaminants attack concrete. These types of degradation have not been observed in spent fuel pools in any of the operating reactors. Additionally, inspection and maintenance of spent fuel pool structures are within the scope of the maintenance rule, 10 CFR 50.65, and corrective actions are required if any degradation is observed. An inspection of the spent fuel pool structure to identify cracks, spalling of concrete, etc., is also recommended as a part of the seismic checklist. Significant degradation of reinforced concrete structures would take more than 5 years or so, the time necessary to lose decay heat in the spent fuel. Substantial loss of structural strength requires long-term corrosion of reinforcing steel bars and substantial cracking of concrete. This is not likely to happen because of inspection and maintenance requirements.

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

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

Appendix 5

Appendix 5 contains the following sub-sections:

Appendix 5a Original NEI Screening Criteria, August 18, 1999

Alan Nelson
SENIOR PROJECT MANAGER
PLANT SUPPORT
NUCLEAR GENERATION DIVISION

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

Dear Mr. Dudley:

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

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

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

Sincerely,

Alan Nelson

APN:tnb

Enclosure

Seismic Screening Criteria

For

Assessing Potential Fuel Pool Vulnerabilities

At

Decommissioning Plants

August 18, 1999

Background

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

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

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

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

Purpose of Checklist

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

DRAFT CHECKLIST

Item 1:

Requirement: **Assure Adequate Ductility of Shear Wall Structures**

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

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

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

Item 2:

Requirement: **Assure Design adequacy of Diaphragms (including roofs)**

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

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

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

Item 3:

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

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

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

Item 4:

Requirement: **Verify the Adequacy of Spent Fuel Pool Penetrations**

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

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

Item 5:

Requirement: **Evaluate the Potential for Impacts with Adjacent Structures**

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

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

Item 6:

Requirement: **Evaluate the Potential for Dropped Loads**

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

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

Item 7:

Requirement: **Evaluation of Other Failure Modes**

Basis: Experienced seismic engineers should review the geotechnical and structural design details for the specific site and assure that there are not any design vulnerabilities which will not be adequately addressed by the review areas listed above. Soil-related failure modes including liquefaction and slope instability should be screened by the approaches outlined in Reference 1 (Section 7 & Appendix C).

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

Item 7: **Required Documentation**

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

References:

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

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

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

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

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

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

by
Robert P. Kennedy
October 1999

prepared for

Brookhaven National Laboratory

1. Introduction

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

2. Background Information

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

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

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:

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

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:

$$P_F = 0.05 \text{ HHCLPF} \tag{3}$$

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 $3x10^{-6}$ for most CEUS plants based upon the Ref. 8 hazard curves. This subject is further discussed in Section 4.

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

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

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 $12.0x10^{-6}$ to $0.11x10^{-6}$ and averaging $1.6x10^{-6}$ for the 26 BWR sites. In my judgment, seismic screening criteria (seismic check lists) can be developed which are sufficiently stringent so as to provide reasonable assurance that the seismic capacity of spent fuel pools which pass the seismic screening roughly equals or exceeds that defined by Eqn. (4). With such a fragility estimate, based on the Ref. 8 seismic hazard estimates, for most CEUS sites, the estimated spent fuel pool seismic-induced failure probability will be less than $3x10^{-6}$ as further discussed in Section 4.

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

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 $2.5x10^{-6}$ to $0.03x10^{-6}$ and averaging $0.48x10^{-6}$ for the 43 PWR sites. A fragility curve as high as that defined by Eqn. (5) is necessary to achieve an estimated spent fuel pool seismic-induced failure probability as low as $1x10^{-6}$ for nearly all CEUS sites. However, I don't believe realistic seismic screening criteria can be developed which are sufficiently stringent to provide reasonable assurance that the Eqn. (5) seismic fragility is achieved. In my judgment, a more rigorous seismic margin evaluation performed in accordance with the CDFM method described in Refs. 6 or 7 would be required to justify a HCLPF capacity as high as that defined by Eqn. (5).

3. Development and Use of Seismic Screening Criteria

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

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

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

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

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

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

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

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

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

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

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

3.2 Spent Fuel Pool Racks

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

3.3 Seismic Level 2 Screening Requirements

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

4. Seismic Risk Associated With Screening Level 2

4.1 Simplified Approaches for Estimating Seismic Risk Given the HCLPF Capacity

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

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

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

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

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

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

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

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:

$$P_F = 0.5\, H_{10\%} \qquad (7)$$

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

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

associated with Screening Level 2 for the Vermont Yankee site by rigorous convolution, it is necessary to extrapolate the Ref. 9 hazard estimates down to the $2x10^{-8}$ exceedance frequency. Also, intermediate values in Table 1 have been obtained by interpolation.

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

1. Ref. 1 simplified approach, i.e., Eqn. (3).
2. Ref. 10 simplified approach, i.e., Steps 1 through 3 above.

3. Rigorous convolution of the hazard and fragility estimates.

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

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

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

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

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

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

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

From Table 3, it can be seen that the EPRI89 hazard estimates produce generally much lower seismic risk estimates corresponding to Screening Level 2 than do the LLNL93 hazard estimates. Based on the EPRI89 hazard estimates, only one site has a seismic risk exceeding $1x10^{-6}$. 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)

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

* By Interpolation

** By Extrapolation

Table 2
Comparison of Seismic Risk Estimated by Various Approaches

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

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

Table 3
Seismic Risk Associated With Screening Level 2

C_{HCLPF} = 1.2g Peak Spectral Acceleration

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

Not Available

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

December 3, 1999

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

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

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

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

The Attachment to this memorandum contains additional details on the deficiencies the staff has found with use of the current seismic checklist. Copies of this memorandum with the attached information will be provided to NEI and all other interested stakeholders in an effort to further the dialogue relating to the seismic checklist and support the development of additional modifications that will resolve the deficiencies currently identified.

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

Attachment: As stated

cc w/att: See next page

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

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

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Structural Failure Modes

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

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

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

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

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

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

- Inspect the pool structure and its vicinity and note:
  - physical condition such as cracking and spalling of concrete, signs of leakage or leaching and separation of pool walls from the grade surface, potential for piping connections, either buried underground or above ground, to fail due to a large seismic excitation or interaction with adjacent equipment, and cause drainage of the pool below the safety level of the pool water,
  - arrangement and layout of supporting columns and shear walls, assessment of other loads from tributary load areas carried by the supporting structure of the pool, as-built dimensions and mapping of any existing structural cracks,
  - adjacent structures that can impact the pool structure both above and below the grade surface, supporting arrangement for superstructure and crane and potential for failure of the superstructure and the crane, potential impact from heavy objects that can drop in the pool structure and the corresponding drop heights.

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

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

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

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

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

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

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

Other Considerations

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

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

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

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

Using the data from NUREG-1488 (new Lawrence Livermore National Laboratory (LLNL) data) for currently operating plants in the eastern and central United States, the mean probability of exceedance (POE) of the peak ground acceleration values for the SSE were examined. The plant grouping approach, Reduced Scope, Focused Scope, Full Scope, etc., used in NUREG-1407, "Procedural and Submittal Guidance for the Individual Plant Examination of External Events (IPEEE) for Severe Accident Vulnerabilities" Final Report was also reviewed. The objective of plant grouping for IPEEE was to put plants into groups with similar seismic vulnerability; consequently, it was useful to look at these plant groups. However, the evaluation in this draft study is driven by the 1993 LLNL seismic hazard results, and it was determined that, except for a small number of plants, the POEs for SSE are lower than $1X10^{-4}$ per reactor year and for three times the SSE, the POEs are below $1X10^{-5}$. For these plants, the likelihood of a catastrophic pool structure failure at a HCLPF value of three times the SSE should be less than $5X10^{-7}$. This makes the simplifying assumption that the conditional probability of failure (POF) or reaching the end state of a structure is $5X10^{-2}$. In this approach there is confidence that the seismic hazard is low (at three times the SSE) and there is also a plant specific structural assessment of the HCLPF value which is more than or equal to three times the SSE. For spent fuel pools located at sites that meet the HCLPF value of three times the SSE, a catastrophic structural failure from an earthquake much larger than the design basis SSE is not credible. However, this approach may not be feasible at sites where the likelihood of the spent fuel pool structure failure due to beyond design basis earthquake is higher. For such sites in the eastern United States, a more detailed examination of the probability of the earthquake, a realistic assessment of the ground motion caused by the event at the site and the structural capacity of the spent fuel pool structure may be necessary.

NEI Draft Seismic Checklist

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

Kennedy Report

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

Recommendation

The following actions are recommended:

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

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

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

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

5. Proposed treatment of sites West of the Rocky Mountains

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

NUCLEAR ENERGY INSTITUTE

Alan Nelson
SENIOR PROJECT MANAGER,
PLANT SUPPORT
NUCLEAR GENERATION DIVISION

December 13, 1999

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

Dear Mr. Huffman:

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

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

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

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

Sincerely,

Alan Nelson

Alan Nelson
APN/dc
Enclosure

3 7/150

Appendix 6
NEI Commitment
Ltr

Appendix 6
Nuclear Energy Institute Commitment letter
dated November 12, 1999
to the
Nuclear Regulatory Commission

NEI

NUCLEAR ENERGY INSTITUTE

Lynnette Hendricks
DIRECTOR
PLANT SUPPORT
NUCLEAR GENERATION DIVISION

November 12, 1999

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

Dear Mr. Barrett,

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

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

I am hereby transmitting those industry commitments as follows.

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

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

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

4. An off site resource plan will be developed which will include access to portable pumps and emergency power to supplement on site resources. The plan would principally identify organizations or suppliers where off

site resources could be obtained in a timely manner.

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

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

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

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

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

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

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

Sincerely,

Lynnette Hendricks
LXH/1rh

6 #9/17

App 7 Stakeholders

Attached w/ changes 1/27 1:32 pm

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Appendix 7 Stakeholder Interactions

1. Introduction

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

Public meetings on the Technical Working Group Study

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

2. Probabilistic Risk Assessment (PRA)

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

3. Human Reliability Analysis

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

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

4. Heavy Loads

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

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

5. Seismic Assessment

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

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

6. Other Seismic Stakeholders Interactions

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

7. Criticality

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

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

8. Thermal-Hydraulic Assessment

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

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

9. Partial Draindown and Exothermic Reaction of SFP

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

10. Impact of Decommissioning on Operating Units

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

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

11. Safeguards

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

Appendix 7 Stakeholder Interactions

1. Introduction

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

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November 19, 1999	Stakeholder meeting with NRC staff in Rockville, MD

2. Probabilistic Risk Assessment (PRA)

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

3. Human Reliability Analysis

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

The staff enlisted the support of HRA experts to refine the analysis in the June 1999 draft report. The HRA results were also subjected to an independent technical review. This topic is discussed in Appendix 2.

4. Heavy Loads

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

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

5. Seismic Assessment

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

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

6. Other Seismic Stakeholders Interactions

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

7. Criticality

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

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

8. Thermal-Hydraulic Assessment

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

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

9. Partial Draindown and Exothermic Reaction of SFP

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

10. Impact of Decommissioning on Operating Units

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

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

11. Safeguards

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

Tech Editor Comments

#1.12

Note:
Generic Safety Issue changed to Generic Issue throughout the report.
JC

Changes made 1/31 @ 1:30 p.m. JC

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11. U.S. Code of Federal Regulations, "Financial Protection Requirements and Indemnity Agreements," Part 140, Title 10, "Energy."
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6.0 Acronyms

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

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

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

February 2000

I left the following words as one word, rather than two.

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Technical Study of Spent Fuel Pool Accidents for Decommissioning Plants

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Executive Summary

The evaluation

This report documents an evaluation of spent fuel pool (SFP) accident risk at decommissioning plants. ~~It~~ was done to provide an interim, risk-informed technical basis for reviewing exemption requests, and to provide a regulatory framework for integrated rulemaking. The application of this report is intended to reduce unnecessary regulatory burden, improve efficiency and effectiveness, and establish a consistent, predictable process that will maintain safety and enhance public confidence. The report was initiated when industry asked the NRC to consider whether the risk from decommissioning plants was low enough to justify generic regulatory relief in the areas of emergency planning, indemnification and safeguards.

w/o NRC

In the past, decommissioning plants have requested exemptions to certain regulations as a result of their permanently defueled condition. When evaluating the acceptability of exemption requests from regulations for permanently shutdown plants, the staff has assessed the susceptibility of the spent fuel to a zirconium fire accident. To date, exemptions have been granted on a plant-specific basis, resulting in different analyses and criteria being used for the basis of the exemptions. In some cases, we have requested heatup evaluations of the spent fuel cooled only by air. This criterion was used because of national laboratory studies that had identified the potential concern for a significant offsite radiological release from a zirconium fire

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SFP zirconium

which may occur when all water is lost from the ~~spent fuel pool~~. A clad temperature of 565 *C, based on the onset of clad swelling, was used as a conservative limit to ensure no radiological release.

SFP

In March, 1999, the staff formed a technical working group to evaluate ~~spent fuel pool~~ accident risk at decommissioning plants. A ~~two~~ 3 month effort was launched to review the available technical information and methods and identify areas in need of further work. A substantial effort was made to involve public and industry representatives throughout the entire effort. A series of public meetings was held with stakeholders during and following the generation of a preliminary draft study that was published in June at the request of the Nuclear Energy Institute (NEI). The partially completed DRAFT report was released to facilitate an industry/NRC/public 2-day workshop that was held in July, 1999. Information ~~gained at~~ acquired the workshop and through other stakeholder interactions was constructive in completing ~~the~~ report.

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

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

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

w/o ACRS

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

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

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

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1. Introduction

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

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

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

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

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

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The conclusions and findings of the study provide guidance for the design and operation of spent fuel pool cooling and inventory systems as well as practices and procedures necessary to ensure high levels of operator performance during off normal conditions. The report concludes that with the imposition of voluntary industry initiatives and some additional staff requirements in the areas of performance monitoring and seismic validation, the risks from spent fuel pools will be sufficiently small, ~~to~~ justify exemptions from selected current regulatory requirements and to form the basis for related rulemaking.

This report ~~contains~~ is divided into three main parts. The first part is a discussion in Chapter 2 on how risk informed decision making can be applied to decommissioning plants. ~~In~~ Chapter 3, the staff presents the risk assessment conducted on the SFPs for decommissioning plants. ~~In~~ Chapter 4 of this report, the findings of SFP risk for a decommissioning plant will be assessed against each of the safety principles and objectives discussed above.

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2.0 Risk Informed Decision Making

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

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

For decommissioning plants, the risk is due primarily to the possibility of a zirconium fire

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

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

associated with the spent fuel rod cladding[1]. The consequences of such an event do not equate exactly to either a core damage accident or a large early release[2]. Zirconium fires in spent fuel pools potentially have more severe consequences than an operating reactor core damage accident, because there are multiple cores involved, and because there is no containment surrounding the SFP to mitigate the consequences. On the other hand, they are somewhat different than a large early release, because the accidents progress slowly enough to allow ample warning for offsite protective actions, and because the absence of Iodine isotopes leads to fewer prompt fatalities. As a result, the criteria of RG 1.174 cannot be applied directly to the risk of a decommissioning plant without further thought.

Even though the event progresses more slowly than an operating reactor LERF and the isotopic makeup is somewhat different, the risk assessment consequence calculations performed by the staff[3] show that large inventories of radioisotopes could be released that could have significant late health effects (latent cancers) for the population at some distance from the plant, as well as the potential for a small number of early health effects (fatalities). The staff has therefore decided that the end state and consequences of a spent fuel pool fire are sufficiently severe that the RG 1.174 LERF baseline criteria of 1E-5 per year or a change not to exceed 1E-6 per year provide appropriate frequency criteria for a decommissioning plant SFP risk, and a useful tool to assess features, systems and operator performance needs of a decommissioning pool.

2.1 Principles of Regulatory Guide 1.174

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

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

- The proposed change meets the current regulations unless it is explicitly related to a requested exemption or rule change, i.e., a "specific exemption" under 10 CFR 50.12 or a "petition for rulemaking" under 10 CFR 2.802.
- The proposed change is consistent with the defense-in-depth philosophy.
- The proposed change maintains sufficient safety margins.

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

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

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

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

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

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

2.1.1 Defense-in-Depth

The defense-in-depth philosophy applies to the operation of the spent fuel pool, whether at an operating plant or in a decommissioning plant. Traditionally defense in depth means that for various credible accident scenarios, there is more than one system or set of actions that will recover from the incident before a serious outcome occurs. This could mean that there is more than one source of cooling water or that pump makeup can be provided by both electric as well as direct drive diesel pumps. Additionally, defense in depth can mean that even if a serious outcome (such as fuel damage) occurs, there is further protection such as containment to prevent radionuclide releases to the public. However, implementation of defense in depth for SFPs is different from that applied to nuclear reactors because of the different nature of the hazards. Because the essentially quiescent (low temperature, low pressure) initial state of the spent fuel pool and the long time for taking corrective action associated with most release scenarios provide significant safety margin, a containment structure is not considered necessary as an additional barrier to provide an adequate level of protection to the public. Likewise, the long evolution of most SFP accident scenarios allows for reasonable human recovery actions to respond to system failures. The specific design and operational features of the SFP, industry commitments and staff requirements that ensure that SFP defense in depth is maintained, is provided in Chapter 4.

2.1.2 Safety Margins

Maintenance of sufficient safety margins is a fundamental principle of RG 1.174. A safety margin can relate to the difference between the expected value of some physical parameter (temperature, pressure, stress, reactivity) and the point at which adequate performance is no longer assured. For example a containment pressure calculation that shows a peak accident pressure of 40 psig is reached for a structure which has a design capability of 60 psig and an actual ultimate capability of 110 psig. In this case there is margin from the accident calculation of 20 psig to the design limit as well as a large margin of 70 psig to the actual expected failure

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limit.

The safety margins associated with fuel in the spent fuel pool for many physical processes and parameters are much greater than those associated with an operating reactor. The spent fuel pool is in a quiescent state, at or near ambient temperature and pressure. The decay heat levels are much lower than those of the fuel in an operating reactor. This allows much greater time for heating and boil off of the coolant water, and for heat up of the fuel itself, once uncovered. The fuel is covered with approximately 28 feet of water at near ambient temperature. The pool is designed with ample margin to criticality, using both passive (geometry) and active (poisons) means of reactivity control. Chapter 4 describes how the provisions that ensure the SFP maintains adequate margins in a decommissioning plant.

2.1.3 Impact of Proposed Changes

The impact of the proposed change should be small. As discussed above, the staff is applying the baseline and change criteria for LERF in RG 1.174 to assess the impact and acceptability of SFP risk in decommissioning plants. Chapters 3 and 4 discusses the design and operational characteristics of the SFP that must be relied upon to produce the low baseline risk results. These are identified in the context of industry commitments as well as staff requirements.

2.1.4 Implementation and Monitoring Program

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

Therefore, with respect to all the above safety principles, implementation and monitoring of important considerations might include comparing a check list against the spent fuel pool seismic design and construction, control of heavy load movements, development and implementation of procedures and other provisions to ensure human reliability, monitoring the capability, reliability, and availability of important equipment, and checking effectiveness of onsite emergency response, and the plans for communication with offsite authorities. In many areas the implementation and monitoring may already be accomplished by utility programs such as those developed under the maintenance rule [Ref. 3]. Chapter 4 discusses the additional implementation and monitoring activities that are necessary to achieve the low SFP risk estimates of this report and support the safety principles.

3.0 Risk Assessment of Spent Fuel Pools at Decommissioning Plants

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

Depending on the time since reactor shutdown and fuel rack configurations, there may be sufficient heat to cause the clad to heat up, swell and burst. The breach in the clad could result in the release of radioactive gases present in the gap between the fuel and clad, called "a gap release" (See Appendix 1). If the fuel continues to heat up, the temperature of the zirconium clad will reach the point of rapid oxidation in air. This reaction of zirconium and air is exothermic. The energy released from the reactor combined with the decay energy can cause the reaction to become self-sustaining and lead to the ignition of the zirconium, or a "zirconium fire." The increase in heat from the oxidation reaction could also raise the temperature in adjacent fuel assemblies and cause the propagation of the oxidation reaction. This zirconium fire will result in a significant release of the fission products contained in the spent fuel, which will be dispersed from the reactor site due to the thermal plume from the zirconium fire. Consequence assessments (Appendix 4) have shown that such a zirconium fire could have significant latent health effects (cancers) as well as the possibility of a small number of early fatalities. Gap releases for fuel of this age in and by themselves (without zirconium fire) release only small quantities of radionuclides and would only be of concern for onsite effects.

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

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

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

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

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

3.1 Basis and Findings of SFP Risk Assessment

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

SFP incidents. These revised results also reflect improvements in the PRA model since publication of the June 1999 report.

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

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

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

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

3.2 Characteristics of SFP Design and Operations for a Decommissioning Plant

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

Suction is taken via one of the two pumps on the primary side from the spent fuel pool and is passed through the heat exchanger and returned back to the pool. One of the two pumps on the secondary side rejects the heat to the ultimate heat sink. A small amount of water from the suction line is diverted to the filtration process and is returned back into the discharge line. A manually operated makeup system (limited volumetric flow rate) supplements the small losses due to evaporation. In the case of prolonged loss of SFPC system or loss of inventory events, the inventory in the pool can be made up using the firewater system. There are two firewater pumps, one motor-driven (electric) and one diesel-driven, which provide firewater in the SFP area. A firewater hose station is provided in the SFP area. The firewater pumps are located in a separate structure.

Based upon information obtained during the site visits and discussions with the operating staff's during those visits, the staff also made the following assumptions that are believed to be

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representative of a typical decommissioning facility:

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- The site has two operable firewater pumps, one diesel-driven and one electrically-driven from offsite power. line up w/ left indent
- We assume the makeup capacity (with respect to volumetric flow) to be as follows:

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

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

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

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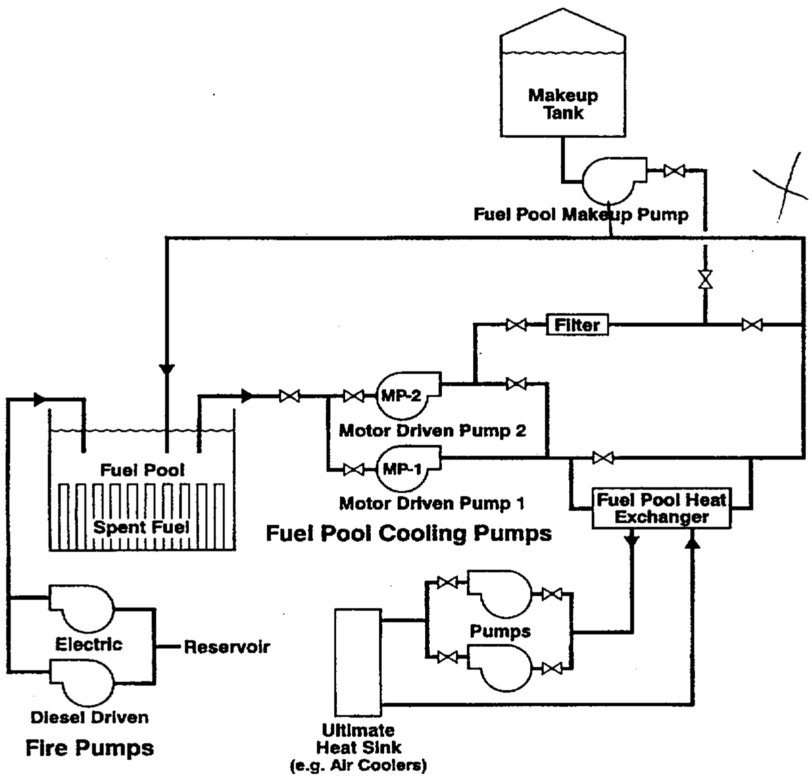

Figure 3.1 Assumed Spent Fuel Pool Cooling System

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

As part of its technical evaluations the staff assembled a blue ribbon committee of experts which identified the attributes necessary to achieving very high levels of human reliability for responding to potential accident scenarios in a decommissioning plant SFP. (See HRA Study in Appendix 2a).

Upon consideration of the sensitivities identified in the staff's preliminary study and to reflect actual operating practices at many decommissioning facility, the nuclear industry, through NEI made important commitments (located in Appendix 6) which were reflected in the staff's updated risk assessment. The revisions to the risk assessment generally reflected changes of assumptions in the areas shown below. The applicability of the specific decommissioning industry commitments (DICs) with respect to the risk analysis results are discussed later in this chapter. How the commitments relate to specific risk conclusions and safety principles is also discussed in Chapter 4.

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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

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Tornado Missile	ϵ[4]
Total	>2.9E-06

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

The above results show that the estimated frequency for a zirconium fire is greater ~~then~~ than approximately 3E-06 per year, with the dominant contributions being from severe seismic events and loss of offsite power initiated by severe weather. already Δ'd no longer there

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

3.3 Internal Event Scenarios Leading to Fuel Uncovery

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

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

Frequency of Fuel Uncovery

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

Scenario

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

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

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

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

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

Cutset

There was one important sequence minimum cutset.

Cutset for sequence 5:

(loss of offsite power) x (fuel handlers fail to diagnose loss of SFP cooling when offsite power is lost) = $8x10^{-8}$ per year

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

3.3.2 Loss of Offsite Power from Severe Weather Events

Frequency of Fuel Uncovery

Frequency of fuel uncovery = $1.3x10^{-6}$ per year

Scenario

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

Cutset

There was one important minimum cutset.

Cutset for sequence 8:

(loss of offsite power due to severe weather) x (offsite power is not recovered for more than 24 hours) x (diesel-driven firewater pump unavailable due to potential for flooding of site) x (fuel handlers fail to provide alternate sources of cooling from offsite) = $1.1x10^{-6}$ per year

Sensitivity

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

3.3.3 Internal Fire

Frequency of Fuel Uncovery

Frequency of fuel uncovery = $9.0x10^{-8}$ per year

Scenario

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

Cutset

There were three important sequence minimum cutsets.

Cutsets for sequence 4:

i) (fire starts in SFP area) x (fuel handler fails to suppress fire) x (fuel handlers fail to diagnose need to start firewater system) = $1.5x10^{-8}$ per year

ii) (fire starts in SFP area) x (fuel handler fails to suppress fire) x (firewater system fails to start/run) x (repair crew fails to repair firewater system) x (fuel handlers fail to provide alternate sources of water from offsite) = $6.8x10^{-9}$ per year

Cutset for sequence 8:

i) (fire starts in SFP area) x (fuel handler fails to respond to a signal indication from the control room that there is a fire) x (fuel handlers fail to observe loss of cooling in walkdowns [dependent case]) = $4.5x10^{-8}$ per year

Sensitivity

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

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3.3.4 Loss of Cooling

Frequency of Fuel Uncovery

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

Scenario

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

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

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

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

3.3.5 Loss of Coolant Inventory

Frequency of Fuel Uncovery

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

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Scenario

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

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

Cutset

There was one important sequence minimum cutset.

Cutset for sequence 9:

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

Sensitivity

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

3.3.6 Heavy Load Drops

The staff investigated the frequency of dropping a heavy load in or near the spent fuel pool, and investigated potential damage to the pool from such a drop. Details of this evaluation can be found in Appendix 2. The analysis exclusively considered drops that were severe enough to catastrophically damage the spent fuel pool such that pool inventory would be lost rapidly and it would be impossible to refill the pool using onsite or offsite resources. In essence there is no possibility for mitigation in such circumstances, only prevention. A catastrophic heavy load drop(that caused a large leakage path in the pool) would lead directly to a zirconium fire approximately 10 to 12 hours after the drop, depending on fuel age, burn up, and configuration.

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The dose rates in the pool area prior to any zirconium fire would be on the order of tens of thousands of rem per hour, making any potential recovery actions such as temporary large inventory addition systems very difficult. The staff concluded that non-catastrophic damage to the pool or its support systems from a load drop is captured and bounded by other initiators.

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

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

When estimating the failure frequency of the pool floor, the staff assumed that heavy loads physically travel near or over the pool approximately 13% of the total path lift length (the path lift length is the distance from the lift of the load to the placement of the load on the pool floor). The staff also assumed that the critical path length (the fraction of total path the load is lifted high enough above the pool that a drop could cause damage to the structure) is approximately 16% of the time the load is near or over the pool. The staff estimated the catastrophic failure rate from heavy load drops to have a mean value of 2.1×10^{-5} per year for a non-single failure proof system where reliance is placed on electrical interlocks, fuel handling system reliability, and safe load path procedures. The staff estimated the catastrophic failure rate from heavy load drops to have a mean value of 2.0×10^{-7} per year for a single failure proof system or a plant conforming to all NUREG-0612 guidelines.

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

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

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NUREG-0612 guidelines, which would put future decommissioning plants in the latter category.

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

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

3.4.1 Seismic Events

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

We believe spent fuel pool structures at nuclear power plants are seismically robust. They are constructed with thick reinforced concrete walls and slabs lined with thin stainless steel liners 1/8 to 1/4 inch thick.[6] Pool walls vary from 4.5 to 5 feet in thickness and the pool floor slabs are around 4 feet thick. The overall pool dimensions are typically about 50 feet long by 40 feet wide and 55 to 60 feet high. In boiling water reactor (BWR) plants, the pool structures are located in the reactor building at an elevation several stories above the ground. In pressurized water reactor (PWR) plants, the spent fuel pool structures are located outside the containment structure supported on the ground or partially embedded in the ground. The location and supporting arrangement of the pool structures determine their capacity to withstand loads beyond their design basis. The dimensions of the pool structure are generally derived from radiation shielding considerations rather than structural needs. Spent fuel structures at operating nuclear power plants are able to withstand loads substantially beyond those for which they were designed. Consequently, they have significant seismic capacity.

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

[6] Except at Dresden Unit 1 and Indian Point Unit 1. These two plants do not have any liner plates. They were decommissioned more than 20 years ago and no safety significant degradation of the concrete pool structure has been reported.

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down rapidly. Thus a seismic event equal to 2.5 to 3.5 SSE at a west coast site may be considered incredible for the site. Therefore, for west coast sites a seismic event greater than two times the SSE could be considered too large to be credible.

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

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

3.4.2 Aircraft

We evaluated the likelihood of an aircraft crashing into a nuclear power plant site and seriously damaging the spent fuel pool or its support systems (details are in Appendix 2D). The generic data provided in DOE-STD-3014-96 [Ref. 6], were used to assess the likelihood of an aircraft crash into or near a decommissioning spent fuel pool. Aircraft damage can affect the structural integrity of the spent fuel pool or affect the availability of nearby support systems, such as power supplies, heat exchangers, or water makeup sources, and may also affect recovery actions. There are two approaches that can be taken to evaluate the likelihood of an aircraft crash into a structure. The first is called the point target model which uses the area (length times width) of the target to determine the likelihood that an aircraft will strike the target. The aircraft itself does not have real dimensions when using this model. In the second approach, the DOE model modifies the point target approach to account for the wing span and the skidding of the aircraft after it hits the ground by including the additional area the aircraft could cover. Further, that model takes into account the plane's glide path by introducing the height of the structure into the equation, which effectively increase the area of the target (see Appendix 2D).

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

Our estimate of the frequency of significant damage to spent fuel pool support systems (e.g., power supply, heat exchanger, or makeup water supply) is developed for three different situations. The first case is based on the DOE model including the glide path and the wing and skid area for a 400 x 200 x 30 foot structure (i.e., the support systems are located inside a large building) with a conditional probability of 0.01 that one of these systems is hit. This model accounts for damage from the aircraft including, for example, being clipped by a wing. We assumed that critical systems occupy only 1% of the total structure area. The estimated frequency range for significant damage to the support systems is 1.0×10^{-10} to 1.0×10^{-6} per year. The mean value is estimated to be 7.0×10^{-8} per year. The second case estimates the value for the loss of a support system (power supply, heat exchanger or makeup water supply) based on the DOE model including the glide path and the wing and skid area for a 10 x 10 x 10 foot structure (i.e., the support systems are housed in a small building). The estimated frequency of support system damage ranges from 1.1×10^{-9} to 1.1×10^{-5} per year, with the mean estimated to be 7.3×10^{-7} per year. The third case uses the point model for this structure [10x 10 or 400 x 200?], and the estimated value range is 2.4×10^{-12} to 1.1×10^{-8} per year, with the mean estimated to be 7.4×10^{-10} per year. Depending on the model approach (selection of the target structure size; use of the point target model or the DOE model), the mean value for an aircraft damaging a support system is in the 7×10^{-7} per year, or less, range. This is not the estimated frequency of fuel uncovery or a zirconium fire caused by damage to the support systems, since the frequency estimate does not include recovery, either onsite or offsite. As an initiator to failure of a support system leading to fuel uncovery and a zirconium fire, an aircraft crash is ~~bounded~~ by other more probable events. Recovery of the support system will reduce the likelihood of spent fuel uncovery.

Overall, the likelihood of significant spent fuel pool damage from aircraft crashes is ~~bounded by~~

other more likely catastrophic spent fuel pool failure and loss of cooling modes.

3.4.3 Tornadoes

We performed a risk evaluation of tornado threats to spent fuel pools (details are in Appendix 2E). We assumed that very severe tornadoes (F4 to F5 tornadoes on the Fujita scale) would be required to cause catastrophic damage to a PWR or BWR spent fuel pool. We then looked at the frequency of such tornadoes occurring and the conditional probability that if such a tornado hit the site, it would seriously damage the spent fuel pool or its support systems. To do this we examined the frequency and intensity of tornadoes in each of the continental United States using the methods described in NUREG/CR-2944 [Ref. 7]. The frequency of having an F4 to F5 tornado is estimated to be $5.6x10^{-7}$ per year for the central U.S., with a U.S. average value of $2.2x10^{-7}$ per year.

We then considered what level of damage an F4 or F5 tornado could do to a spent fuel pool or its support systems. Based on the buildings housing the spent fuel pools and the thickness of the spent fuel pools themselves, the conditional probability of catastrophic failure given a tornado missile is very low. Hence, the overall frequency of catastrophic pool failure caused by a tornado is extremely low (i.e., the calculated frequency of such an event is less than $1x10^{-9}$ per year)

We assumed that an F2 to F5 tornado would be required if significant damage were to occur to spent fuel pool support systems (e.g., power supply, cooling pumps, heat exchanger, or makeup water supply). The frequency of having an F2 to F5 tornado is estimated to be $1.5x10^{-5}$ per year for the central U.S., with a U.S. average value of $6.1x10^{-6}$ per year. As an initiator to failure of a support system, the tornado is bounded by other more probable events (see Table 3.1-1).

3.4.4 Criticality in Spent Fuel Pool

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

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

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

(1) A compression or buckling of the stored assemblies could result in a more optimum geometry (closer spacing) and thus create the potential for criticality (see the NRC staff report "Assessment of the Potential for Criticality in Decommissioned Spent Fuel Pools," in Appendix 3). Compression is not a problem for high-density PWR or BWR racks because they have sufficient fixed neutron absorber plates to mitigate any reactivity

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increase, nor is it a problem for low-density PWR racks if soluble boron is credited. But compression of a low-density BWR rack could lead to a criticality since BWR racks contain no soluble or solid neutron absorbing material. High-density racks are those that rely on both fixed neutron absorbers and geometry to control reactivity. Low-density racks rely solely upon geometry for reactivity control. In addition, all PWR pools are borated, whereas BWR pools contain no soluble absorbing material. If both PWR and BWR pools were borated, criticality would not be achievable for a compression event.

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

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

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

To gain qualitative insights on the recriticality events that are credible, the staff considered the sequences of events that must occur. For scenario 1 above, ~~this would be require~~ a heavy load drop into the a low density racked BWR pool compressing assemblies would be required. From Appendix 2 on

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heavy load drop, the likelihood of a heavy load drop from a single failure proof crane has a mean value of approximately 9.6E-6 per year, assuming 100 cask movements per year at the decommissioning facility. From the load path analysis done for that appendix it was estimated that the load could be over or near the pool approximately 13% of the movement path length, dependant on plant specific layout specifics. The additional frequency reduction in the appendix to account for the fraction of time that the heavy load is lifted high enough to damage the pool liner is not applicable here because the fuel assemblies could be crushed without the same impact velocity being required as for the pool liner. Therefore, we observe a potential initiating frequency for crushing of approximately 1.2E-6 per year (based upon 100 lifts per year). Criticality calculations conducted for Appendix 3 show that even if the low density BWR assemblies were crushed by a transfer cask, it is "highly unlikely" that a configuration would be reached that would result in a severe reactivity event, such as a steam explosion which could damage and drain the spent fuel pool. The staff judges the chances of such a criticality event to be well below 1 chance in 100 even given that the transfer cask drops directly onto the assemblies. This would put the significant criticality likelihood well below 1E-8 per year, which justifies its exclusion from further consideration.

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

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

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

Based upon the above conclusions and staff requirements, we believe that qualitative risk insights demonstrate conclusively that SFP recriticality poses ~~so~~ no meaningful risk to the public.

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4.0 Implications of Spent Fuel Pool Risk For Regulatory Requirements

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

The analysis in Chapter 3 explicitly examines the risk impact of specific design and operational characteristics. Some of these have been proposed by the Nuclear Energy Institute in a letter to the NRC dated November 12, 1999 [See Ref. 1 or Appendix 6]. Others came to light as a result of the analysis itself. These characteristics are summarized in Chapter 4.1. The NRC intends to make these the principle aspects of the risk-informed approach to oversight of decommissioning plants.

Chapter 4.2 examines the design and operational elements that are important in ensuring that the risk from a SFP is sufficiently low and how these elements support the safety principles of RG 1.174 as they apply to a SFP.

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

4.1. Summary of the Technical Results

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

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

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

The annual frequency of events leading to zirconium fires at decommissioning plants is estimated to be $2x10^{-6}$ per year for a plant that implements the design and operational characteristics discussed below. This estimate can be much higher for a plant that does not embody these characteristics. The most significant contributor to this risk is a seismic event which exceeds the design basis earthquake. Other contributors are at most 10% of the seismic contribution including such scenarios as drop of heavy loads into the pool. This overall frequency is within the acceptance guidelines for large early release frequency (LERF) of $1x10^{-5}$ per year in RG 1.174. As noted above, zirconium fires are estimated to be similar to large early releases in some ways, but less severe in others.

4.2 Risk Impact of Specific Design and Operational Characteristics

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

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

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

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

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

The accident sequences in Chapter 3 associated with loss of cooling or loss of inventory are quantified to result in low risk due to a number of elements that enhance the ability of the operators to respond successfully to the events with onsite and offsite resources. Without these elements, the probability of the operators detecting and responding to the loss of cooling or inventory would be higher and public risk from these categories of SFP accidents could significantly increased. Some elements were also identified that reduce the likelihood of the loss of cooling or loss of inventory initiators, including both design ~~as well as~~ and operational issues. The elements proposed by industry (Decommissioning Industry Commitments (DICs)) are identified below.

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

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

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

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

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

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

DIC #3 Procedures will be in place to establish communication between on site and off site

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organizations during severe weather and seismic events.

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

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

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

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

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

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

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

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

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4.2.3 The Proposed Change Maintains Sufficient Safety Margins

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

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

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

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

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

With respect to monitoring of the Boraflex absorber material, the current monitoring programs required by Generic Letter 96-04 [Ref. 3] will be maintained by decommissioning plants until all

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fuel is removed from the SFP. This generates a decommissioning staff requirement (DSR).

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

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

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

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

4.3.1 Emergency Preparedness

The requirements for emergency preparedness ~~for~~ are contained in 10CFR 50.47 [Ref. 4] and Appendix E [Ref. 5]. Further guidance on the basis for EP requirements is contained in NUREG-0396 [Ref. 6]. The general goal of EP requirements is to prevent early fatalities and to reduce offsite dose from accidents.

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

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

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The consequence analysis presented in Appendix 4 demonstrates that the offsite consequences of a zirconium fire are comparable to those from operating reactor severe accidents. Further, the analysis demonstrates that timely evacuation can significantly reduce the number of early fatalities due to a zirconium fire. The thermal-hydraulic analysis presented in appendix 1 confirms our earlier conclusion that zirconium fire events evolve slowly, even for initiating events that result in a catastrophic loss of fuel pool coolant. The results in Chapter 3 also show that the frequency of zirconium fires is low when compared with the risk guidelines from RG 1.174. Thus the risk associated with early fatalities from these scenarios is low. Based on this combination of low risk and slow evolution, the Commission might decide to reduce or eliminate EP requirements for decommissioning plants. With respect to the potential for pool recriticality, the staff's assessment discussed in Chapter 3 and Appendix 3 demonstrates that creditable scenarios for criticality are precluded by monitoring programs or are highly unlikely; and even if they do occur would not be expected to have offsite consequences. Therefore the conclusions regarding possible reductions in EP program requirements are not impacted.

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

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

4.3.2 Security

Currently licensees that have permanently shutdown reactor operations and have offloaded the spent fuel into the SFP are still required to meet all the security requirements for operating

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reactors in 10 CFR 73.55 [Ref 7]. This level of security would require a site with a permanently shutdown reactor to provide security protection at the same level as that for an operating reactor site. The industry has asked the NRC to consider whether the likelihood of radiological release from decommissioning plants due to sabotage is low enough to justify modification of safeguards requirements for SFPs at decommissioning plants.

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

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

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

In an associated regulatory arena, 10 CFR 73.51, "Physical Protection for Spent Nuclear Fuel and High-Level Radioactive Waste," ~~allows~~ facilities not associated with an operating power reactor are allowed to store spent fuel at an independent spent fuel storage installation (ISFSI). This rule provides performance-based regulations specifically designed for these types of storage installations, i.e., fuel in dry cask containers or other storage formats. The objective of the 10 CFR 73.51 rule was to reduce the regulatory burden regarding security requirements without reducing protection levels to public health and safety for spent fuel storage not associated with an operating reactor. When drafted, 10 CFR 73.51 included permanently shutdown reactors, but these facilities were removed from the scope of the rule when NRR technical staff identified a potential safety issue addressed herein. 10 CFR 73.51 failed to account for the risk posed by vehicle-borne bombs at facilities where potential criticality and fuel heatup were still issues.

The risk analysis in this study indicates the need to prepare a performance-based regulation similar to 10 CFR 73.51 that will not only reduce the regulatory burden and be appropriate for spent fuel storage at power reactor sites but also will account for the threat of vehicle-borne bombs. In addition security officers will be armed, but the bullet-resisting alarm station will not necessarily be in the protected area.

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

As noted above, this new regulation will be very similar to 10 CFR 73.51 except for the use of armed security officers, the off-site bullet-resisting alarm station, and the retention of the vehicle barrier system. The following additional open or unresolved issues will be resolved during the formal rulemaking process: (1) the impact of this technical study as it relates to timing of the downgrading of requirements, (2) grandfathering sites that defueled before the vehicle barrier system rule, and (3) the use of vital and protected areas, as currently defined in the regulations.

The staff also noted that the applicability of 10 CFR 26 [Ref 10] has not been established for decommissioning reactors once the fuel has been removed from the reactor vessel and placed in the SFP, and specifically does not apply to ISFSIs licensed under 10 CFR 72. Given the importance of a vehicle bomb threat to the integrity of SFP, and the significance of HRA to the conclusions reached in the SFP risk analysis, the staff recommends that for coherency in the regulations, both of these subjects be revisited during the overall integration of rules for decommissioning reactors.

4.3.3 Insurance

In accordance with 10 CFR 140 [Ref. 11], each 10 CFR 50 licensee is required to maintain public liability coverage in the form of primary and secondary financial protection. This coverage is required to be in place from the time unirradiated fuel is brought onto the facility site until all the radioactive material has been removed from the site, unless the Commission terminates the Part 50 license or otherwise modifies the financial protection requirements. The industry has asked the NRC to consider whether the likelihood of large scale radiological releases from decommissioning plants is low enough to justify modification of the financial protection requirements once the plant is permanently shutdown and prior to complete removal of all radioactive material from the site.

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

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

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

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