Microfiche: Jason 42CE3 242

BROOKHAVEN NATIONAL LABORATORY

MEMORANDUM

DATE May 8, 1985

TO W. T. Pratt

FROM V. L. Sailor and K. R. Perkins

SUBJECT Study of Beyond Design Basis Accidents in Spent Fuel Pools (Generic Issue 82)

This memorandum summarizes the progress on the Work Requirements (Schedule 189, Section 3a) which provides that the work will be performed in two distinct stages. "The first stage will consist of a preliminary evaluation of the likelihood and consequences of the loss of pool integrity."

For purposes of this study "Beyond Design Basis Accidents in Spent Fuel Pools" are defined as accidents in which the water inventory of the pool is lost and, as a consequence, massive failure of the spent fuel cladding occurs resulting in the uncontrolled release of fission products from the fuel.

1. INITIATING EVENTS

1.1 Preliminary List of Events Considered

Initiating events which have been considered include:

- extreme external phenomena (earthquakes, tornadoes, hurricanes, floods, and aircraft accidents);
- prolonged loss of all cooling capability (pool boils dry);
- massive pool rupture from internal accidents (shipping cask drop, crane collapse, turbine failure missiles); and
- rapid draining of pool due to circulation system malfunction, operational error, or malicious act.
- sabotage such as the addition of reactive chemicals or deliberate damage to the pool or cooling system.

1.2 Preliminary Conclusions

Only one initiating event has been identified as having a likelihood of occurrence greater than 10^{-6} per reactor calendar year. That event is an earthquake more severe than the Design Basis Earthquake (DBE) for plants of pre-1973 vintage or the Safe Shutdown Earthquake (SSE) for plants of post-1973 vintage. (Based on the material we have examined to date, design to the SSE

E/2

criteria results in a more conservative structure than did the older DBE criteria.)

The range of probabilities of occurrence of an earthquake of severity equalling or exceeding the DBE or the SSE is discussed in Section 2 for three eastern U.S. sites.

Preliminary evaluations of other initiating events listed in Section 1.1 indicate that their probabilities are less than 10^{-6} per reactor calendar year. These tentative conclusions are based on pool design characteristics, as well as, on probability estimates for the frequency of various initiating events.[24,25] Some examples are given below:

- The pools are generally surrounded by other structures which provide partial shielding against external missiles (from tornadoes, aircraft accidents, turbine failure). In addition, the pool walls and bottom are thick, reinforced concrete that is capable of absorbing a high velocity impact without failure.

- The probabilities of missile initiating events are small[24] (tornadoes $< 10^{-5}$/yr, aircraft crash on building $< 10^{-9}$/yr, turbine failure plus energetic missile strike on pool $< 5 \times 10^{-7}$).

- Loss of water inventory resulting from loss of all sources of make-up are extremely small, because of the multiple alarm systems which monitor pool flow and levels and the relatively long time period to evaporate the pool water inventory to the level at which fuel is uncovered. The time available to provide alternate sources of make-up water is long (several days or weeks), and the quantities of water required for make-up are small (< 100 gpm).[25]

- Systems are provided to preclude inadvertent drainage of the pool (e.g., by siphoning action). The chances of such drainage going unnoticed have low probability estimated[24] at approximately 1×10^{-8}/yr.

More work is needed to refine and verify the probability estimates for events of these types, but for the present we shall assume that the published estimates are credible.

2. SEISMIC HAZARDS

2.1 Definitions and Description of Methods

The seismic hazard at a given site can be quantified in terms of the probability per year that some specified ground motion parameter will be exceeded, e.g., peak ground acceleration (PGA), peak ground velocity (PGV), spectral acceleration, spectral velocity [Ref. 1, see App. D.]. For simplicity, we shall use the PGA. A PGA seismic hazard curve consists of a plot of the probability that PGA exceeds a given value, a, i.e., P (A > a) yr^{-1} vs. the acceleration, a. (See Fig. 1.).

A comprehensive program to estimate seismic hazards in the "eastern" United States (EUS) has been underway at Lawrence Livermore National Laboratory (LLNL) for the past several years under the auspices of the U.S. Nuclear Regulatory Commission. For purposes of these studies the "eastern" U.S. is defined as the continental area to the east of the Rocky Mountains. The methodology applied to specific nuclear power plant sites is described in a five-volume series of reports.[2-6] The method uses the best-estimate of two panels of seismic experts, one for zonation and seismicity, the other for ground motion prediction. The input quantitative judgements of the experts on each panel are combined by statistical methods to obtain median values and analyses of the variances in opinions are used to estimate confidence limits. There is also a feedback loop in the solicitation of data from the panels and a self-weighting system for various classes of data input. (See Ref. 7, Fig. 2.1.)

In the analysis each expert delineates seismically active zones on a map, estimates the seismicity (frequency and magnitude distributions). These zones are then coupled to reactor sites taking account of distances, attenuation and ground motion models.

The final products for each site include seismic hazard curves (for PGA and PGV), seismic spectra and uncertainties in each. It should be noted that the seismic hazard curves derived by the above methodology are not analytical mathematical functions and therefore must be treated by numerical methods.

2.2 Availability of Site Specific Data

Table 1 summarizes the "best estimate" PGA seismic hazard data for three sites of older vintage plants which have been reviewed as part of the NRC Systematic Evaluation Program (SEP). Reference 1 also presents similar data for seven other sites where plants are/were under construction.

Data for other SEP sites* are available but in less convenient form for our initial evaluations than are the PGA seismic hazard curves cited above.[5] (Attempts will be made to obtain unpublished data. If this is unsuccessful, we will attempt to synthesize rough PGA seismic hazard curves from the published material.)

*Palisades, Big Rock Point, Dresden, Yankee Rowe, Haddam Neck, Oyster Creek and Ginna.

2.3 Seismic Design Basis: Older Plants

The seismic design basis for older vintage plants was usually based on historical records of seismic events in the surrounding region. Local geology was taken into account to arrive at conservative estimates for the effect of the substructure in estimating the maximum credible ground acceleration. The OBE was usually selected to be somewhat larger than the maximum credible ground acceleration and the DBE was arbitrarily selected to be about twice the OBE. Seismic spectra from observed earthquakes were synthesized with the selected OBE and DBE for structural analyses.

Table 2 lists the OBE and DBE for several representative older vintage plants.

3. SEISMIC FRAGILITY OF EXISTING SPENT FUEL POOLS

3.1 Fragility of Plant Structures

Given an earthquake more severe than the DBE or SSE, it does not necessarily follow that a catastrophic failure of the spent fuel pool will occur.

The conditional probability of a structure or system failure as a function of seismic loading is commonly referred to as a fragility curve.[8]

To date our search of the published literature including applicant safety analysis reports (PSAR's and FSAR's) has not yielded any fragility analyses specifically for spent fuel pool structures. The reports on older vintage plants, issued by the Systematic Evaluation Program (SEP),[9-16] have generally called on the licensees to provide additional seismic analyses of plant structures.

In general, the plant structures of the older vintage plants appear to have been designed to very conservative margins of safety. Presumably, the spent fuel pools are included in those structures that are characterized by a conservative margin of strength.

Several reports refer to plant structures in general. For example, the Systematic Evaluation Program (SEP) report for the Yankee Nuclear Power Station states (Ref. 9 at pg. 4-15):

> "Most experts and the staff agree that plant design using linear elastic analysis methods and limiting calculated seismic loadings to design allowable values results in at least a factor of safety of two. Actual behavior of plants under earthquake conditions supports this conclusion."

A seismic safety analysis of the Oyster Creek plant structures, which were designed for a 0.11g DBE concludes that the median factor of safety for the reactor building is about 7.0.[17] The resulting fragility curve for this structure, taking account of the various uncertainties in the analysis, was described as following [Ref. 17, pg. 331]:

> "This fragility curve shows a very high confidence of a negligible frequency of failure for 0.2g (approximately SSE) ground acceleration. At 0.4g (double SSE) the median fragility factor is low (about 0.05) but the confidence in this estimate is poor and there is about a 5% probability that the fragility factor might be as high as 0.5. At 0.6g (three times SSE) the median fragility factor is about 0.25 but with very large uncertainty on this factor."

Also, the report notes that [Ref. 17, pg. 329]:

> "This conclusion is considered applicable to plants designed in the U.S. in the mid-1960's."

3.2 Historical Experience with Earthquakes

The historical record of earthquakes actually experienced by nuclear power plants is probably worth some investigation. (We hope to acquire a more complete record than is now available to us.) During a refueling outage on June 7, 1975, the Humboldt Bay nuclear plant experienced an earthquake that resulted in no plant damage. The operating basis earthquake (OBE) for the plant is 0.25g. The peak ground acceleration actually experienced was 0.36g.[18]

For purposes of this interim report, lacking fragility analyses for specific spent fuel pools, we shall use the structural fragility data in Ref. 17 for making our preliminary estimates of the seismic risks of catastrophic failure of a "nominal" pool structure of mid-1960 vintage design. Preliminary numerical values of the probability of structural failure as a function of acceleration, a_i, are listed in Table 3.

4. PROBABILITY OF EARTHQUAKE-INDUCED STRUCTURAL FAILURE OF SPENT FUEL POOL

The annual frequency, $F_E(a_i)$ of occurrence of an earthquake of PGA = a_i is obtained from the differential of the PGA seismic hazard curve data of the type listed in Table 1 and illustrated in Figure 1, i.e.,

$$F_E(a_i) = \Delta P\ (A > a)/\Delta a \text{ (per year)}.$$

For our preliminary evaluation, the differential has been obtained graphically. Table 3 lists $F_E(a_i)$ values for three eastern U.S. plants.

The conditional pool failure probability for a seismic event of PGA = a_i is represented by $F_S(a_i)$. The numerical values of $F_S(a_i)$, listed in Table 3, were obtained from Ref. 17 and assume that all three plants have structural strengths similar to Oyster Creek. (We hope to be able to refine these data later, on a plant specific basis and specifically for the spent fuel pool structures.)

The total probability of an earthquake induced failure of a structure, P_F, is obtained from the integral,

$$P_F = \int_0^{a_M} F_E(a_i)\, F_S(a_i)\, da,$$

where a_M is the PGA of the maximum strength earthquake that experts anticipate. The integral must be evaluated numerically. This has been done very roughly using data of the type shown in Table 3, for the "best estimate" values of both $F_E(a_i)$ and $F_S(a_i)$. Results are shown in Table 4 including the range of uncertainties based on the upper and lower confidence limits of the $F_S(a_i)$ values. It is seen in Table 4 that the estimated probability for structural failure for Site A is of the order of 1×10^{-5} with an uncertainty range from 5×10^{-5} to 2×10^{-7}, whereas for Site C the "best estimate" is 4×10^{-7} with a range from 2×10^{-6} to $< 1 \times 10^{-8}$.

The effects of uncertainties in $F_E(a_i)$ values have not yet been evaluated, but will increase the range of uncertainties shown in Table 4.

5. CONSEQUENCES OF TOTAL LOSS OF POOL WATER

5.1 Cladding Failure

We assume that total loss of pool water could, under some circumstances, lead to a cascading failure of the fuel cladding of all spent fuel assemblies stored in the pool. The progress of the cladding failure is assumed (for present purposes) to follow the scenarios described in the report of Benjamin, et al.[19] in which the exothermic oxidation of the zirconium cladding would initiate among the more recently discharged assemblies and propagate to more aged assemblies. The time scale of the cladding failures would depend on the age of the spent fuel, the density of the racking, and the configuration of batches of fuel of various decay times.

The oxidation modeling in the Sandia[19] analysis has been reviewed by the Materials Technology Division at BNL.[26] While the Sandia results should be benchmarked against additional experiments, they provide a reasonable basis for estimating the consequences of spent fuel pool loss-of-integrity accidents. Although there are large uncertainties in estimating the consequences, it is likely that the total uncertainties in the risk (probability x consequences) are dominated by the inherent uncertainties in the probabilities of occurrence of extreme seismic events.

5.2 Radioactive Inventory in Spent Fuel

The inventory of radionuclides contained in spent fuel assemblies depends on the operating history and the size of the plant. In particular, during a refueling campaign, the freshly discharged fuel contains a large inventory of isotopes with short half lives in the range of approximately 1 to 30 days which subsequently decay to low levels over the course of the year until the next refueling campaign.

Older fuel, containing radionuclides with longer half-lives approaches a decay rate approximately proportional to $1/\tau$ where τ is the decay time in years. Thus, a batch of fuel aged for four years contains about 1/4 of the specific activity of fuel aged for one year.

The radionuclide inventory of "reference" spent fuel can be found in Tables 3.3.6-3.3.8 and 3.3.10 of Ref. 20. The tables cited list the specific activities of more than a hundred isotopes as a function of decay time since discharge. Separate inventories are given for activation products in fuel assembly hardware and cladding, and for fission products and actinides sealed in the fuel elements. The data were developed from a series of computer calculations using the ORIGEN code.[21] The cross sections used in the code have been calibrated so that the results are consistent with measured inventories in spent fuel samples.

For extended burnup fuel the total annual radionuclide inventory would differ only slightly from the "reference" spent fuel. There would be more fission products per tonne of discharged fuel, i.e., a higher specific radioactivity, but fewer tonnes of spent fuel per year would be discharged. The total fission product inventory per year is proportional to the total thermal energy produced per year, i.e., the total number of fissions that have occurred. Small changes in the inventory of the lighter fission products occur because, as burnup is extended, plutonium contributes a larger fraction of the fission events. Because of their larger mass, the fission product yield curves for the plutonium isotopes and are shifted slightly upward from that of uranium-235. Most of this shift occurs in the lighter fission products.

A few of the more important isotopes remaining after various periods of aging are listed in Table 5 as examples.

It is obvious that the radionuclide content in aged spent fuel differs substantially from the inventory in the fuel of an operating equilibrium reactor core. In particular, aged spent fuel contains:

- little radioiodine or other halogens, the exception being about 3.3×10^{-2} Ci/MTHM of I-129;

- no radioactive noble gases except for Kr-85; and

- much lower concentrations of alkali metals, chalcogens (O, S, Se, Te, and Po), alkali earths, lanthanides and noble metals.

On the other hand, the total inventory of longer lived isotopes in the spent fuel basin can be greater than the equilibrium core inventory. For example, if we assume that the pool contains spent fuel assemblies for ten years of operation with 1/3 of a core discharged each year, the total Cs-134, Cs-137 and Sr-90 inventories would be several times larger than those in the operating core (see Table 6). For very long-lived isotopes, e.g., half-lives > 100 years, the ten-year spent fuel basin would contain about 6.7 times the equilibrium core inventory, e.g, Tc-99 (2.14×10^5y), Pu-239 (2.41×10^4y), Pu-242 (3.76×10^5y), Am-241 (433y), Am-242m (152y).

For the case in which spent fuel has recently been discharged (for example, in the past 30 days), corrections must be made to the inventory listed in Table 5 to account for the presence of radionuclides with shorter half-lives (e.g., Xe-133, I-131, Ba-140, etc.). However, it should be noted that the probabilities of a seismic-induced event as listed in Table 4 occurring within x days of fuel discharge must be decreased by the factor x/365.25. Thus, if the probability listed in Table 4 is 1×10^{-5} per year, the probability of occurrence within ten days after discharge is reduced to

$$1 \times 10^{-5} \times (10/365.25) = 2.7 \times 10^{-7}.$$

5.3 The Radionuclide Inventory Potentially Available for Release

The radionuclide inventory available for release in any postulated accident sequence is determined by:

1) the amount (curies of each radionuclide);

2) the composition (physical and chemical form of each), and

3) the timing of the release of radioactivity to the environment.

The physical and chemical processes that would take place in a drained spent fuel basin are not well characterized at the present time. The calculations of Benjamin, et al.[19] indicate that under some conditions the zircaloy cladding could become hot enough to "burn", i.e., the exothermic zirconium oxidation rate would proceed very rapidly and the heat released would be sufficient to "ignite" adjacent assemblies, thus propagating throughout all assemblies in the basin. As noted in Ref. 22, the Sandia model does not address the relocation of the products of the reaction (molten unoxidized cladding, fuel dissolved in molten zirconium, etc.). Also, the fate of the exposed UO_2 fuel pellets has not been studied, e.g., to what degree will the pellets oxidize to U_3O_8 and in the process release less volatile fission products.

Obviously, at this stage of understanding, the source terms for "beyond design basis accidents" in spent fuel basins have not been well-defined. However, we shall make some rough estimates based on one set of assumptions which are as follows:

1) The fuel basin has in storage spent fuel from ten refuelings.

2) The last discharge was 0.5 years ago.

3) Refueling occurred at year intervals.

4) Discharged fuel averaged burnup of 33,000 MWd/MTU.

5) Annual discharge was 35 MTU.

Table 7 presents the calculated radioactive inventory stored in the pool, based on the above assumptions. Included are activation products contained in the zircaloy cladding and fuel assembly hardware. Many isotopes with low specific activity have been omitted. The calculations were based on data presented in Ref. 20.

We shall assume that catastrophic oxidation of all zircaloy cladding propagates throughout the pool. 3 1/3 acres

Rough estimates of the fractional release of various isotopes have been presented in a handwritten attachment to Ref. 22, including noble gases (100%), halogens (100%), alkali metals (100%), tellurium (2 to 100%), barium (2%), strontium (0.2%) and ruthenium (0.002%). These estimates are listed in column 4, "Release Fraction" of Table 7.

We have estimated release fractions of the other isotopes in Table 7, based on engineering judgement. Comments on these estimates follow:

Tellurium: The releases shown assume the lower limit of Ref. 22 based on the tellurium release model recently proposed by Lorenz, et al.[28] The low release value assumes that a fraction of the zircaloy cladding relocates (melts and flows downward) before oxidation is complete.[23]

most of the release for an accident involving the core is tied up in noble gases and halogens, whereas the spent fuel pools release is principally alkali metals (cesium).

The dose equivalence of the release estimates depends on many factors including the location of the fuel pool and equipment operability. Sandia[26] has estimated the offsite dose for the WASH-1400 spent fuel pool releases assuming the pool is located in the auxiliary building. The result is shown in Figure 2 with and without air filtering. Note that cesium is expected to be released as an aerosol and the filters will provide an effective removal mechanism. (It should be noted that some older vintage plants do not have filters in the ventilation systems in the spent fuel pool buildings.) However, if the building develops cracks or the fans fail to function due to the seismic event, the release will be substantially above the Protective Action Guidelines given in NUREG-0654.

6. CONCLUSIONS

Section 4 indicates the likelihood of spent fuel pool failure and drainage due to seismic events ranging from 10^{-5} to 10^{-7}/yr depending on location and seismic qualification for several of the oldest plants. Thus, for some sites, the seismic hazard is well above the frequency estimate given in WASH-1400 for a typical plant (10^{-6}/yr).

The release estimates given in Section 5 are somewhat different than those estimated in WASH-1400 due to the different assumptions (1 unit with 10 year accumulation instead of the WASH-1400 analysis for two units with shipping after 150 days) but the total release estimates are fairly consistent and both the WASH-1400 and the present release estimates tend to be principally cesium.

For the case in which the pool fails and the air filter systems fail or are ineffective due to building leakage, the consequences (Figure 2) are significant and appear to be comparable to core melt sequences.

Since the principal hazard for spent fuel pools is expected to be due beyond design basis seismic events, there appear to be no obvious retrofits which might readily reduce or eliminate the risk in a cost effective manner (any mitigation system such as sprays, ventilation/filtering or inerting would have to be designed to a stricter seismic standard than the rest of the plant). However, this situation deserves further investigation. It is noted that the susceptibility of spent fuel pools to seismic events is far from uniform and appears to be limited to several plants that were designed and built prior to the adoption of more stringent seismic regulations in 1973. A logical next step would be to do plant specific fragility analysis of the structures and safety systems for the few spent fuel pools which are judged to be most vulnerable. Depending on the outcome, it may be necessary to investigate what steps could be taken to reduce the seismic risks for some plants.

Transition Elements: We assume that 50% of the activation products contained in the cladding are levitated as aerosols of the oxides (smoke). Note that the small release fraction of Zr-95 (0.01) takes into account the large inventory of fission product Z-95 trapped in the fuel pellets. We assume that none of this escapes.

We assume that only 10% of the activation products in the assembly hardware escapes (see, Fe-55, Co-58 and Co-60). The Co-60 fraction is corrected for the small content in the cladding.

Antimony: We assume Sb-125 to be very mobile and that 100% is roasted out of the fuel pellets.

Lathanide and Actinides: We have assumed negligible release of the oxides of the lathanides and actinides because of their chemical stability, low vapor pressures and ceramic characteristics.

The resultant releases of radioactivity from the fuel and cladding are given in the last column of Table 7. These estimates should be considered as very preliminary.

Reference 19 assumes that the building surrounding the pool will begin to leak at an internal gage pressure of 2.0 psi during clad oxidation. We shall provisionally adopt this assumption.

At the present stage of our analysis, we have not attempted to estimate what fractions of the isotopes released from the pool might deposit on surfaces inside the building. For the present, we shall make the conservative assumption that there is no deposition.

Based on scenarios presented in Ref. 23, the onset of cladding failure could begin within 1 to 8 hours following complete sudden drainage of the pool, depending on the age of the most recently discharged fuel, and on details of storage rack design, rack density, arrangements of fuel batches of differing ages, and building ventilation available at the time. The duration of the "burn" could be as long as 8 hours (see, e.g., Fig. 2.17 of Ref. 23).

5.4 Comparison of Spent Fuel Pool Release Estimates with Other Severe Accidents

The radionuclide release for the spent fuel pool has been grouped for comparison to release estimates for fuel pool accidents and core accidents given in WASH-1400. This comparison is given in Table 8. Note that even with the shipping assumption*, the two unit plant analyzed in WASH-1400 will have significantly more activity due to the more frequent discharge. Note that

*It was assumed that all fuel would be shipped out for storage or reprocessing after 150 days.

7. RECOMMENDATIONS

Based on our preliminary estimates of the likelihood and consequences of beyond design basis accidents (Phase I), we believe that the seismic hazard is the principal contributor to risk for spent fuel pools. Our estimate of the consequences due to seismic events indicates that for some plants the risk is significant compared to core melt accidents. However, the uncertainty in this risk estimate is large due to the uncertainty in the structural behavior of spent fuel pools, and in the seismic hazard itself.

While our review[27] also indicates substantial uncertainty in the cladding oxidation and propagation analyses,[19] we believe that there is sufficient basis to conclude that given a loss-of-pool-integrity accident, a substantial fraction of the pool inventory will reach the clad oxidation limits. If such an accident is plausible, the ultimate fate of the fuel rubble would need investigation.

We recommend that:

1. Phase II of the program be initiated with an emphasis on identifying the specific plants which have a significant seismic hazard.

2. The status of fragility analyses of existing pool structures needs further investigation for plants perceived to have the highest risks. This investigation should consist of: (1) a critical review of existing structural analyses, if such analyses can be obtained from the vendors or A/E; and/or (2) independent dynamic stress analyses of one or more selected pool structures.

3. The release estimates need to be refined and developed for the actual situations (i.e., the anticipated stored inventory) for one or more specific plants. Included should be: (1) analyses of the fuel degradation, and (2) the final coolable state of the fuel rubble.

4. The risk profile should be refined. Included would be a refinement of: (1) the non-seismic risks discussed in Section 1.2, and (2) a better definition of the seismic risks, including the interaction with the plant systems, if approved by the project manager.

VLS/KRP:tr

Table 1 "Best-Estimate" Seismic Hazard Data for Three Reactor Sites in the "Eastern" United States [source: Ref. 1]. The first column lists the effective peak ground acceleration, PGA, in units of the acceleration of gravity, g. The other three columns list the annual probability of ground motion acceleration exceeding PGA, i.e., P (A > a), for the respective sites. Uncertainties in the probabilities, which are large, are not shown.

Peak Ground Acceleration, a	Probability of Exceeding PGA (per year), P (A > a)		
	Site A	Site B	Site C
0.1 g	8.0×10^{-3}	5.4×10^{-3}	5.5×10^{-4}
0.2	1.7×10^{-3}	1.0×10^{-3}	8.1×10^{-5}
0.3	5.8×10^{-4}	4.6×10^{-4}	2.6×10^{-5}
0.4	2.7×10^{-4}	1.8×10^{-4}	1.1×10^{-5}
0.5	1.2×10^{-4}	9.0×10^{-5}	3.3×10^{-6}
0.6	8.0×10^{-5}	5.0×10^{-5}	2.2×10^{-6}
0.7	4.8×10^{-5}	3.1×10^{-5}	1.2×10^{-6}
0.8	3.0×10^{-5}	2.0×10^{-5}	7.0×10^{-7}
0.9	2.0×10^{-5}	1.2×10^{-5}	3.0×10^{-7}
1.0	1.5×10^{-5}	9×10^{-6}	1.8×10^{-7}

Table 2 Operating Basis Earthquakes (OBE) and Design Basis Earthquakes (DBE) for Several Older Vintage Plants

Plant	OBE	DBE
	(peak ground acceleration in units of g)	
Big Rock Point	None	0.05 (static only)
Browns Ferry-1	0.10	0.20
Dresden-1	0.10	0.20[a]
Ginna	0.08	0.20[a]
Haddam Neck	None	0.17
Humboldt Bay	0.25	0.50
La Crosse	0.06	0.12
Maine Yankee	0.05	0.10
Millstone-1	0.09	0.17
Oyster Creek	0.11	0.22[a]
Palisades	0.10	0.20
San Onofre-1	0.12[b]	0.25[c]
Trojan	0.15	0.25[a]
Yankee Rowe	None	None

(a) In the SAR this was called the "safe shutdown earthquake" but it is not equivalent to the SSE as presently defined.
(b) "100-year" earthquake.
(c) "600-year" earthquake.

Table 3 Preliminary Numerical Data Used for Estimating the Probability (per year) of Earthquake-Induced Structural Failure, P_F. The estimates are subject to large uncertainties which are not shown in the table. The fragility values, $F_S(a_i)$, were obtained from Ref. 17 and are not plant specific. The PGA frequencies, $F_E(a_i)$ were derived graphically from data in Ref. 1, and are site specific.

PGA = a_i (in units of g)	Fragility $F_S(a_i)$	Frequency of PGA = a_i, $F_E(a_i)$ per year: Site A	Site B	Site C
0.2	0	20.0×10^{-4}	10.3×10^{-4}	12.0×10^{-5}
0.3	0.04	5.0×10^{-4}	3.8×10^{-4}	3.5×10^{-5}
0.4	0.11	1.8×10^{-4}	1.8×10^{-4}	0.9×10^{-5}
0.5	0.22	1.0×10^{-4}	0.6×10^{-4}	0.6×10^{-5}
0.6	0.33	0.4×10^{-4}	0.3×10^{-4}	0.13×10^{-5}
0.7	0.43	0.23×10^{-4}	0.13×10^{-4}	0.06×10^{-5}
0.8	0.53	0.13×10^{-4}	0.08×10^{-4}	-
0.9	0.62	0.08×10^{-4}	0.05×10^{-4}	-

Table 4 Rough Estimates of Probabilities, P_F, of Earthquake-Induced Structural Failure at Three Eastern U.S. Sites. The seismic hazard contribution is based on site specific data (Ref. 6). The structural fragility is based on generic data (Ref. 17).

	Estimated P_f (per year)		
	Site A	Site B	Site C
"Best Estimate"	1×10^{-5}	7×10^{-6}	4×10^{-7}
Upper Limit (90 percentile)*	5×10^{-5}	3×10^{-5}	2×10^{-6}
Lower Limit (10 percentile)*	2×10^{-7}	1×10^{-7}	$<1 \times 10^{-8}$

*Based only on uncertainties in fragility data. Effects of uncertainties in seismic hazard have not yet been evaluated.

Table 5 Examples of Specific Activities of Radionuclides Contained in "Reference" Spent Fuel Assemblies (Source: Ref. 20)

ISOTOPE	- - DECAY TIME AFTER DISCHARGE (in years) - - 0.5 yr	1.5 yr	3.5 yr	10 yr
	(specific activity in Ci/MTHM[a])			
A. Fuel Ass[illegible] Hardware, Activation Products				
Fe-55	6.0×10^3	4.0×10^3	2.5×10^3	4.0×10^2
Co-58	2.0×10^3	2.0×10^0	negl.[b]	negl.
Co-60	5.0×10^3	4.0×10^3	3.0×10^3	1.0×10^3
B. Zircaloy Cladding, Activation Products				
Zr-95	4.0×10^3	8.0×10^1	negl.	negl.
Nb-93m	9.0×10^3	negl.	negl.	negl.
Nb-95	7.0×10^3	2.0×10^2	negl.	negl.
C. Fission Products				
Kr-85	9.5×10^3	8.9×10^3	7.9×10^3	5.0×10^3
Sr-90	6.7×10^4	6.5×10^4	6.3×10^4	5.2×10^4
Y-90[c]	6.7×10^4	6.5×10^4	6.3×10^4	5.2×10^4
Zr-95	1.7×10^5	3.4×10^3	1.4×10^0	negl.
Ru-106	3.4×10^5	1.7×10^5	4.2×10^4	3.4×10^2
Rh-106[d]	3.4×10^5	1.7×10^5	4.2×10^4	3.4×10^2
Sb-125	6.8×10^3	5.3×10^3	3.1×10^3	5.3×10^2
Te-125m[e]	2.8×10^3	2.2×10^3	1.3×10^3	2.2×10^2
Cs-134	1.7×10^5	1.2×10^5	6.0×10^4	5.7×10^3
Cs-137	9.4×10^4	9.2×10^4	8.8×10^4	7.5×10^4
Pr-144	6.1×10^5	2.4×10^5	4.2×10^4	8.2×10^1
Eu-154	5.8×10^3	5.5×10^3	5.0×10^3	3.7×10^3

(continued on next page)

a) Ci/MTHM = curies per metric ton of heavy metal.
b) negl. = less than 1 curie.
c) Y-90 is daughter product of Sr-90.
d) Rh-106 is daughter product of Ru-106.
e) Te-125 is daughter product of Sb-125.

Table 5 (Cont.)

	0.5 yr	1.5 yr	3.5 yr	10 yr
D. Actinides				
Np-239[f]	1.4×10^{1}	1.4×10^{1}	1.4×10^{1}	1.4×10^{1}
Pu-238	2.1×10^{3}	2.1×10^{3}	2.1×10^{3}	2.0×10^{3}
Pu-239	2.9×10^{2}	2.9×10^{2}	2.9×10^{2}	2.9×10^{2}
Pu-240	4.5×10^{2}	4.5×10^{2}	4.5×10^{2}	4.5×10^{2}
Pu-241	1.1×10^{5}	1.1×10^{5}	9.2×10^{4}	6.9×10^{4}
Am-241[g]	2.0×10^{2}	3.7×10^{2}	6.9×10^{2}	1.6×10^{3}
Cm-242	1.7×10^{4}	3.6×10^{3}	1.7×10^{2}	8.5×10^{0}
Cm-244	1.3×10^{3}	1.3×10^{3}	1.2×10^{3}	9.0×10^{2}

f) Np-239 is "granddaughter" of Pu-241 alpha decay branch.
g) Am-241 is daughter of Pu-241 beta decay branch.

Table 6 Comparison of Selected Radionuclide Inventory in Equilibrium Core with Ten Years of Stored Spent Fuel. It is assumed that each year one-third of the equilibrium core is discharged, that the equilibrium core has an average burnup of 16,500 MWd/MTU* and that the spent fuel at time of discharge has a burnup of 33,000 MWd/MTU.

	INVENTORY (10^6 Ci)		
	Cs-134	Cs-137	Sr-90
Equilibrium Core	5.26	5.26	4.04
10-years stored spent fuel	8.48	30.96	23.65
Ratio: spent fuel to core	1.61	5.89	5.85

*Burnup is expressed as Megawatt Days (MWd) per Metric Ton of Uranium (MTU).

Table 7 Example of Estimate of Radionuclides Released During a Spent Fuel Pool Accident Resulting in Complete Destruction of Cladding. Conditions assumed were: (1) annual discharge of 35 MTHM each year for 10 refueling cycles; (2) 0.5 years since last discharge (oldest fuel aged 9.5 years).

Chemical Family	Isotope	Pool Inventory (Ci)	Release Fraction	Radioactivity Released (Ci)
Noble gases	Kr-85	2.5×10^6	1.00	2.5×10^6
Halogens	I-131	5.0×10^2	1.00	5.0×10^2
Alkali Metals	Cs-134	2.0×10^7	1.00	2.0×10^7
	Cs-137	3.0×10^7	1.00	3.0×10^7
Chalcogens	Te-125m	4.0×10^5	0.02	8.0×10^3
	Te-127	3.3×10^5	0.02	6.6×10^3
Alkali Earths	Sr-90	2.1×10^7	2×10^{-3}	4.2×10^4
Transition Elements	Fe-55	8.6×10^5	0.10	8.6×10^4
	Co-58	7.2×10^4	0.10	7.2×10^4
	Co-60	1.0×10^6	0.12	1.2×10^5
	Nb-93m	2.2×10^6	0.5	1.1×10^6
	Nb-95	2.5×10^5	0.5	1.3×10^5
	Zr-95	6.1×10^7	0.01	6.1×10^5
Miscellaneous	Ru-106	2.4×10^7	2×10^{-5}	4.8×10^2
	Sb-125	1.0×10^6	1.00	1.0×10^6
	Pr-144	3.6×10^7	1×10^{-6}	3.6×10^1
	Eu-154	1.5×10^6	1×10^{-6}	1.5×10^0
Transuranics	Pu-238	7.4×10^5	1×10^{-6}	7.4×10^{-1}
	Pu-239	1.0×10^5	1×10^{-6}	1.0×10^{-1}
	Pu-240	1.6×10^5	1×10^{-6}	1.0×10^{-1}
	Pu-241	3.1×10^7	1×10^{-6}	3.1×10^1
	Cm-242	7.6×10^5	1×10^{-6}	7.6×10^{-1}
	Cm-244	3.9×10^5	1×10^{-6}	3.9×10^{-1}

Table 8 Estimated Radionuclide Releases for Spent Fuel Pool Accidents Compared to Typical (PWR 3) Release Estimates for Core Melt Accidents

Radionuclide Group	Spent Fuel Pool WASH-1400 Av Loading) (Curies x 10^{-5})	Spent Fuel Pool[1] (10/yr Inventory) (Curies x 10^{-5})	PWR 3 Release (Curies x 10^{-5})
Noble Gases	5.8	8.3	2740
Halogens	1.8	.002	5005
Alkali Metals	217	167	30
Alkali Earths	18.8	.14	74
Te, Sb	.44	.05	528
Rare Earths	0	0	48
Noble Metals	0	0	166

(1) Assuming 1/3 of inventory involved with release fractions given in Table 7.

(2) Assuming full core melt for 1000 MWe equilibrium core with PWR 3 release rates from WASH-1400.

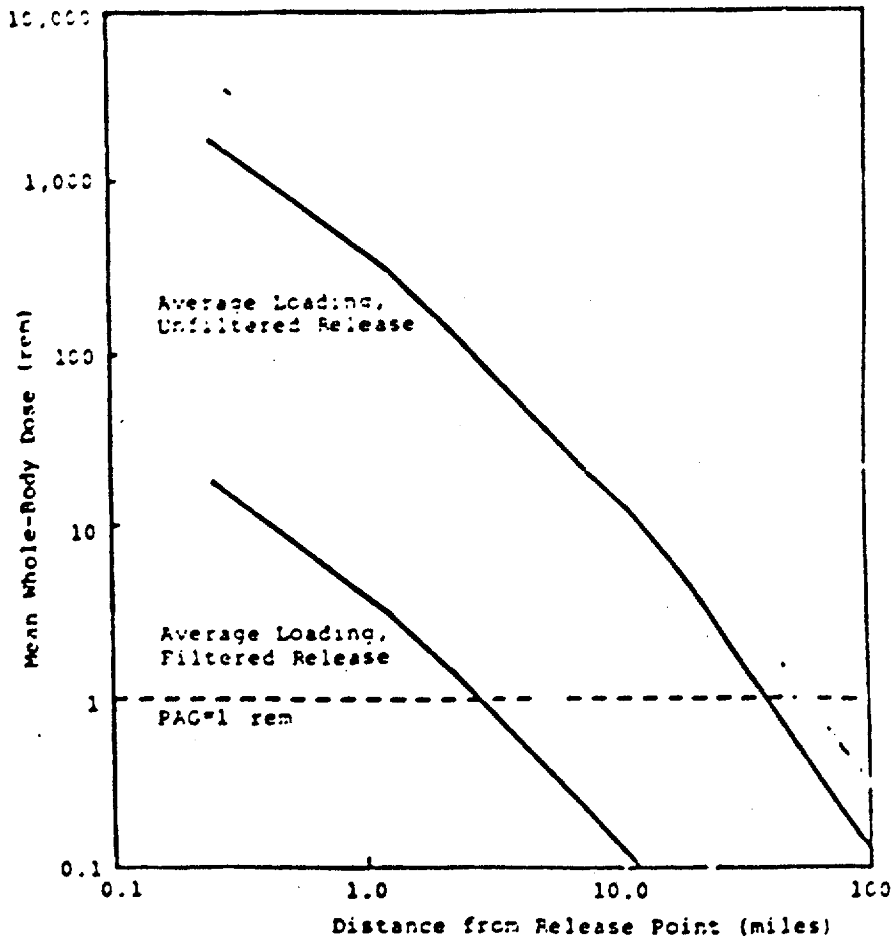

Figure 2 Spent Fuel Storage Pool Fuel Melt Accidents Mean Whole-Body Dose vs. Distance CRAC2 - ≈100 Weather Sequences (Source: Reference 26. Note that these calculations assume 48-hour exposures, which is very conservative compared with 4-hour exposures normally assumed in consequence analyses. However, the extreme exposure time might be consistent with a beyond design basis seismic event in which failure of roads and bridges disrupt evacuation.)

References

1. D. L. Bernreuter, J. B. Savy, R. W. Mensing, and D. H. Chung, Seismic Hazard Characterization of the Eastern United States: Methodology and Interim Results for Ten Sites, prepared for the U.S. Nuclear Regulatory Commission by Lawrence Livermore National Laboratory, NUREG/CR-3756 (UCRL-53527), (April 1984).

2. D. L. Bernreuter, C. Minichino Seismic Hazard Analysis: Overview and Executive Summary, prepared for the U.S. Nuclear Regulatory Commission by Lawrence Livermore National Laboratory REG/CR-1582, Vol ' (UCRL-53030, Vol. 1), (April 1983).

3. TERA Corporation, Seismic Hazard Analysis: A Methodology for the Eastern United States, prepared for the U.S. Nuclear Regulatory Commission, NUREG/CR-1582, Vol. 2, (August 1980).

4. TERA Corporation, Seismic Hazard Analysis: Solicitation of Expert Opinion, prepared for the U.S. Nuclear Regulatory Commission, NUREG/CR-1582, Vol. 3, (August 1980).

5. D. L. Bernreuter, Seismic Hazard Analysis: Application of Methodology, Results, and Sensitivity Studies, prepared for the U.S. Nuclear Regulatory Commission by Lawrence Livermore National Laboratory, NUREG/CR-1582, Vol. 4 (UCRL-53030, Vol. 4), (October 1981).

6. D. L. Bernreuter, Seismic Hazard Analysis: Review Panel, Ground Motion Panel and Feedback Results, prepared for the U.S. Nuclear Regulatory Commission by Lawrence Livermore National Laboratory, NUREG/CR-1582, Vol. 5 (UCRL-53030, Vol. 5), (1981).

7. J. B. Savy, D. L. Bernreuter, and R. W. Mensing, "Uncertainties in Seismic Hazard Analysis Using Expert Opinion for the Eastern United States," Lawrence Livermore National Laboratory, UCRL-92190 (preprint), (February 1985).

8. R. P. Kennedy, R. D. Campbell, G. Hardy, and H. Banon, Subsystem Fragility: Seismic Safety Margins Research Program (Phase I), Structural Mechanics Associates, prepared for the U.S. Nuclear Regulatory Commission, NUREG/CR-2405, UCRL-15407, (February 1982).

9. Integrated Plant Safety Assessment, Systematic Evaluation Program, Yankee Nuclear Power Station, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Final Report, NUREG-0825, (June 1983).

10. Integrated Plant Safety Assessment, Systematic Evaluation Program, R. E. Ginna Nuclear Power Plant, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Final Report, NUREG-0821, (December 1982).

11. Integrated Plant Safety Assessment, Systematic Evaluation Program, Oyster Creek Nuclear Generating Station, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Final Report, NUREG-0822, (January 1983).

12. Integrated Plant Safety Assessment, Systematic Evaluation Program, Dresden Nuclear Power Station, Unit 2, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Final Report, NUREG-0823, (February 1983).

13. Integrated Plant Safety Assessment, Systematic Evaluation Program, Millstone Nuclear Power Station, Unit 1, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Final Report, NUREG-0824, (February 1983).

14. Integrated Plant Safety Assessment, Systematic Evaluation Program, Haddam Neck Plant, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Final Report, NUREG-0826, (March 1983).

15. Integrated Plant Safety Assessment, Systematic Evaluation Program, La Crosse Boiling Water Reactor, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Final Report, NUREG-0827, (June 1983).

16. Integrated Plant Safety Assessment, Systematic Evaluation Program, Big Rock Point Plant, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, Final Report, NUREG-0828, (May 1984).

17. R. P. Kennedy, C. A. Cornell, R. D. Campbell, S. Kaplan and H. F. Perla, "Probabilistic Seismic Safety Study of an Existing Nuclear Power Plant," Nuclear Engineering and Design, Vol. 59, pp. 315-318, (1980).

18. J. W. Minarick and C. A. Kukielka, Precursors to Potential Severe Core Damage Accidents: 1969-1979, Science Applications, Inc., prepared for the U.S. Nuclear Regulatory Commission, NUREG/CR-2497, Vol. 2, Appendix B, p. B-252, (June 1982).

19. A. S. Benjamin, D. J. McCloskey, D. A. Powers, and S. A. Dupree, Spent Fuel Heatup Following Loss of Water During Storage, prepared for the U.S. Nuclear Regulatory Commission by Sandia Laboratories, NUREG/CR-0649 (SAND77-1371), Rev. 3, (March 1979).

20. U.S. Department of Energy, Technology for Commercial Radioactive Waste Management, Volume 1, Section 3.3, DOE/ET-0028, (May 1979).

21. M. J. Bell, ORIGEN - The ORNL Isotope Generation and Population Code, ORNL-4628, (May 1963).

22. Memorandum of J. T. Han to M. Silberberg, "Response to a NRR Request to Review SNL Studies Regarding Spent Fuel Heatup and Burning Following Loss of Water in Storage Pool," U.S. Nuclear Regulatory Commission, (May 21, 1984).

23. N. A. Pisano, F. Best, A. S. Benjamin and K. T. Stalker, The Potential for Propagation of a Self-Sustaining Zirconium Oxidation Following Loss of Water in a Spent Storage Pool, prepared for the U.S. Nuclear Regulatory Commission, draft report, (January 1984).

24. WASH-1400 (NUREG-75/014), Reactor Safety Study, An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants, U.S. Nuclear Regulatory Commission, (October 1975), Appendix I.

25. R. Emrit, W. Minners, H. VanderMolen, R. Colmar, D. Thatcher, J. Pittman, W. Milstead, R. Riggs, G. Sege, P. Matthews, L. Riani, A Prioritization of Generic Safety Issues, Division of Safety Technology, Office of Nuclear Reactor Regulation, U.S. Nuclear Regulatory Commission, NUREG-0933, (December 1983).

26. R. P. Burke, C. D. Heising, and D. C. Aldrich, "In-Plant Considerations for Optimal Off-Site Response to Reactor Accidents," NUREG/CR-2925, (November 1982).

27. L. J. Tentonico, "Study of Beyond Design Basis Accidents in Spent Fuel Pools", BNL Letter Report to K. Perkins, (March 27, 1985).

28. R. A. Lorenz, E. C. Beahm, and R. P. Wichner, "Review of Tellurium Release Rates from LWR Fuel Elements Under Accident Conditions," Proceedings of the International Meeting on Light Water Reactor Severe Accident Evaluation, August 28-September 1, 1983, pg. 4.4-1, American Nuclear Society Order 700085, ISBN 0-89448-1112-6.