From: "NELSON, Alan" <apn@nei.org> GTH = G. Hubbard
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Date: Fri, Aug 4, 2000 10:04 AM
Subject: SEISMIC DATA AND CORRESPONDENCE

<<Risk informed seismic EPRI response 121399.doc>> <<Risk informed seismic EPRI response 121399.doc>> <<Risk infromed seismic checklist 121399.doc>> <<Risk informed Barrett DD committ 111299.doc>> <<Risk informed comments to NRC spent fuel risk study41900.doc>> <<Risk Informed seismic charts Duke 102599>>

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4243

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 x 10^{-6}. This value is multiplied by 0.5 which results in a SFP failure probability for site 1 of about 1.0 x 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:

a. the list of high hazard 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 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.

5. 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.

6. 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:

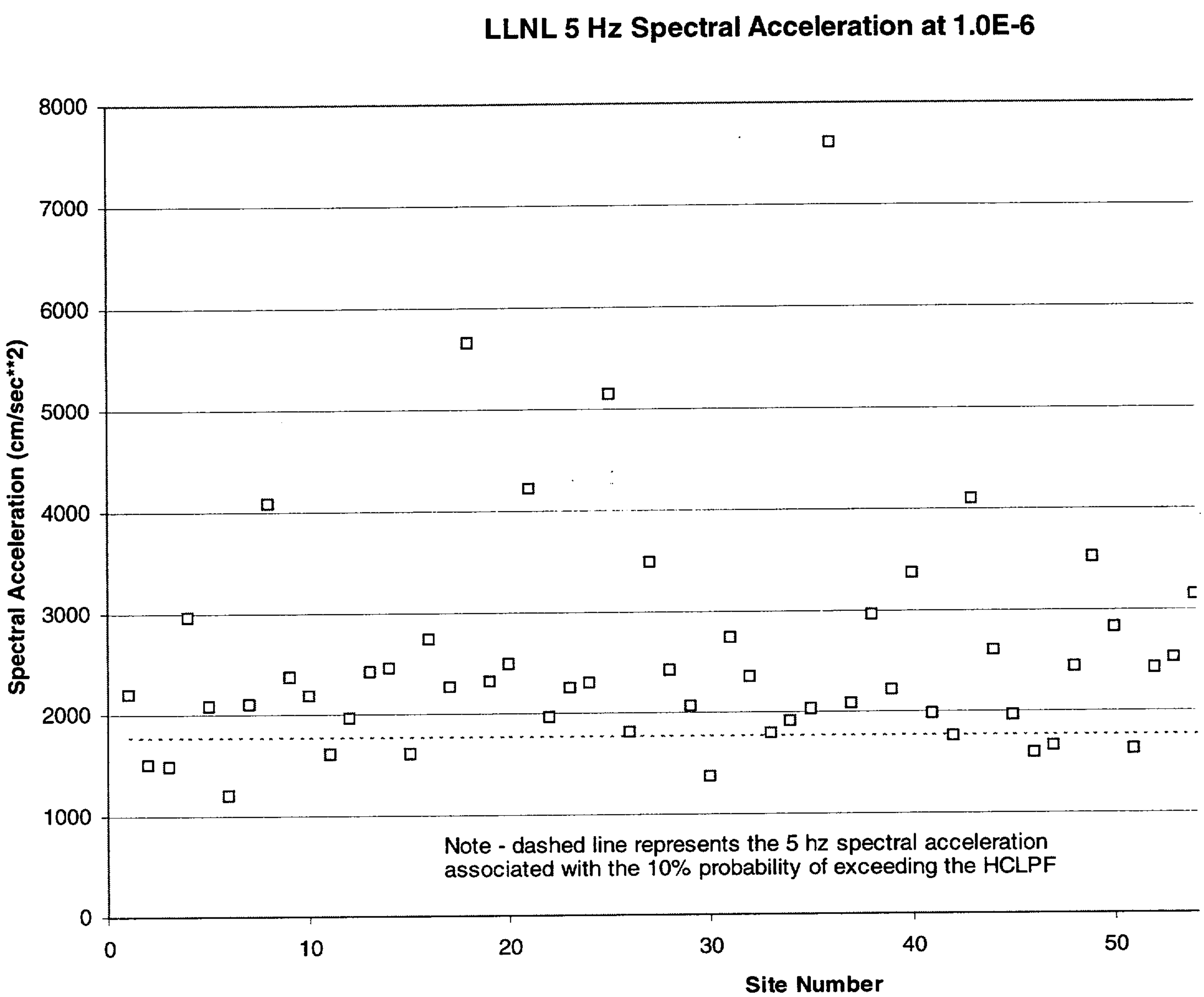
1. Memorandum, W. Hauffman to S. A. Richards, USNRC, Screening Criteria for Assessing Potential Seismic Vulnerabilities of Spent Fuel Pools at Decommissioning Plants, December 3, 1999.

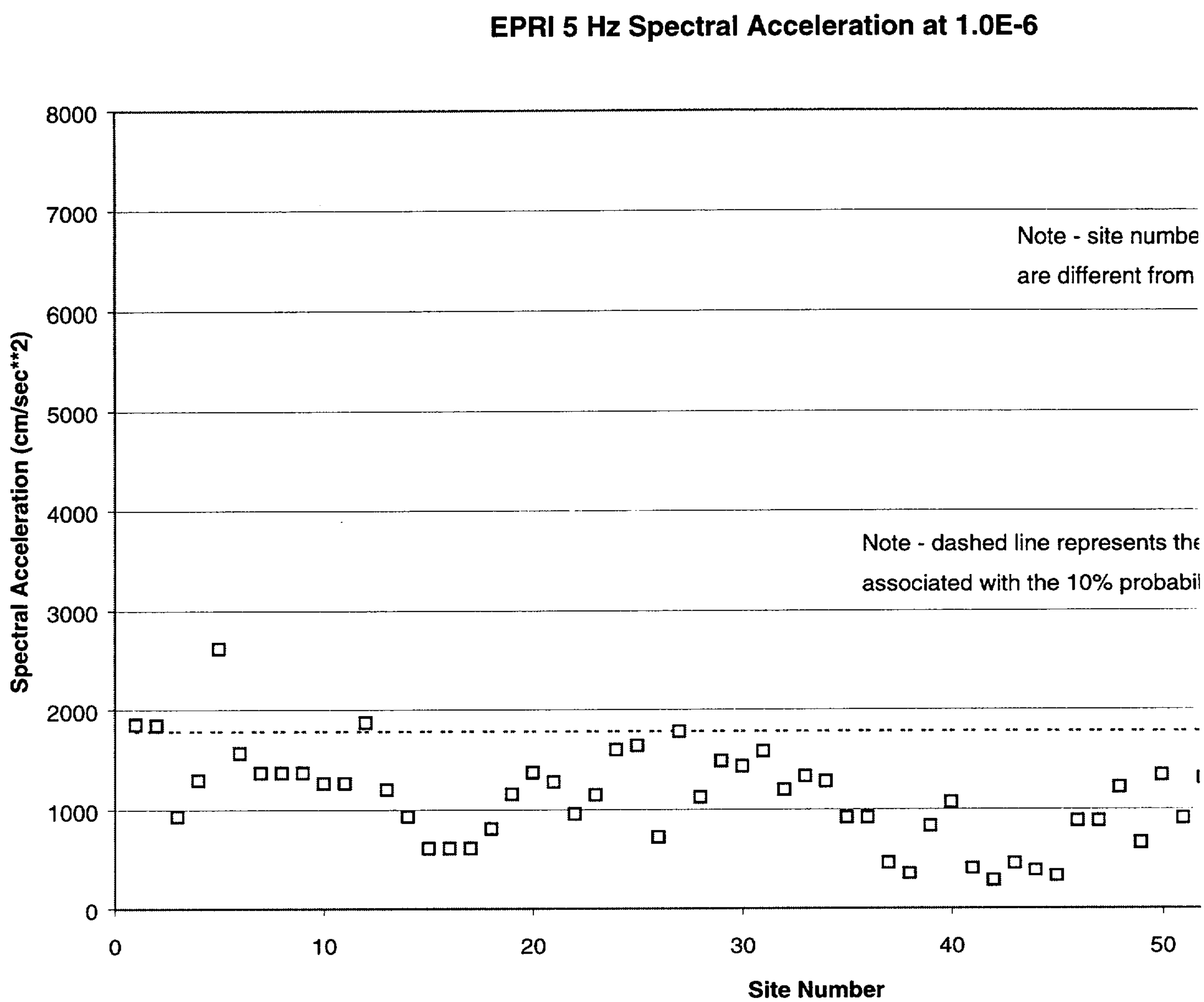
2. NUREG/CR-5176, Seismic Failure and Cask Drop Analyses of the Spent Fuel Pools at Two Representative Nuclear Power Plants, Lawrence Livermore National Laboratory, January 1989.

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.


LLNL 5 Hz Spectral Acceleration at 1.0E-6
Spectral Acceleration (cm/sec**2)
8000
7000
6000
5000
4000
3000
2000
1000
0
0
10
20
30
40
50
Site Number
Note - dashed line represents the 5 hz spectral acceleration associated with the 10% probability of exceeding the HCLPF


EPRI 5 Hz Spectral Acceleration at 1.0E-6
Note - site numbe
are different from
Note - dashed line represents the
associated with the 10% probabil
Spectral Acceleration (cm/sec**2)
8000
7000
6000
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10
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30
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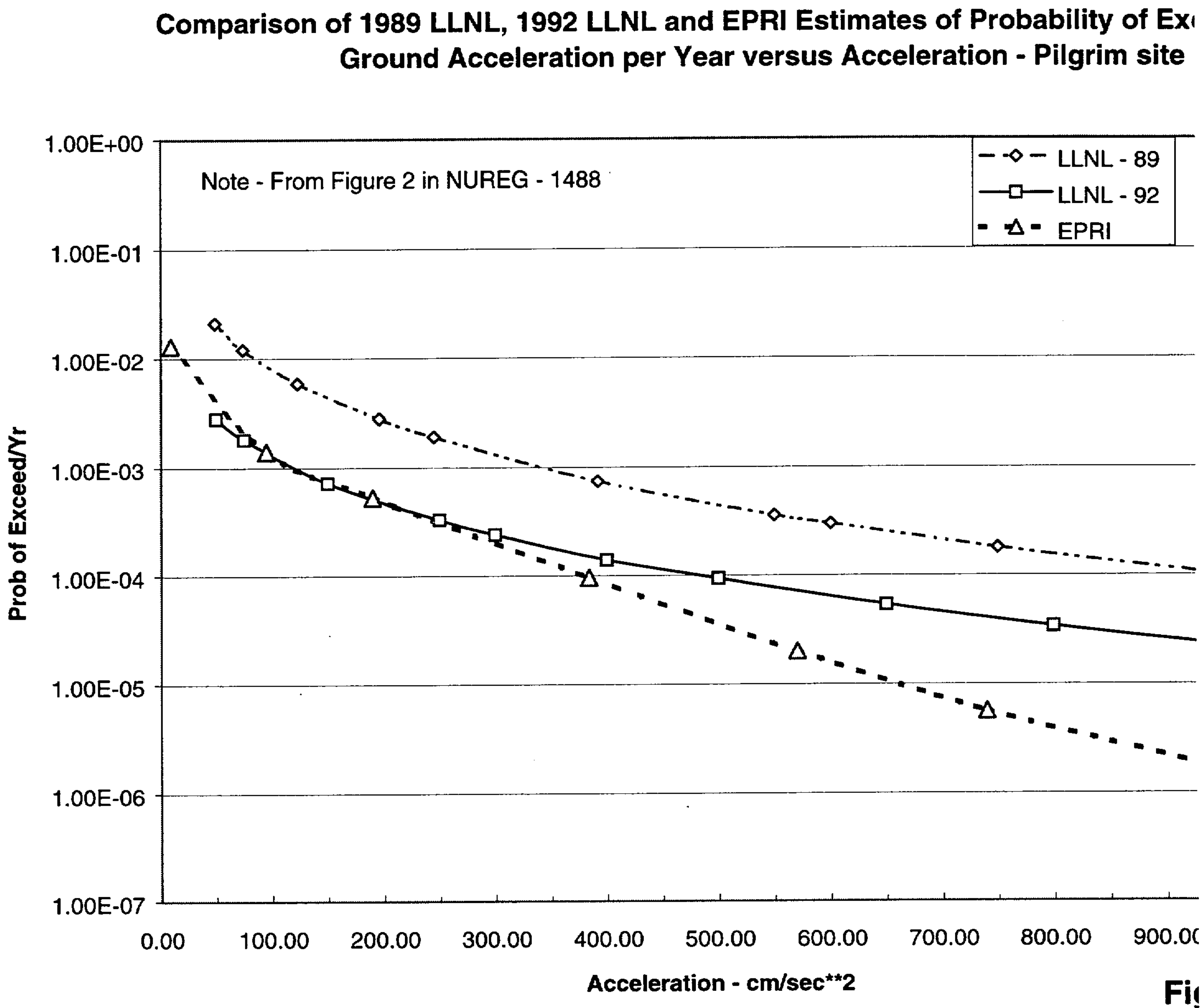
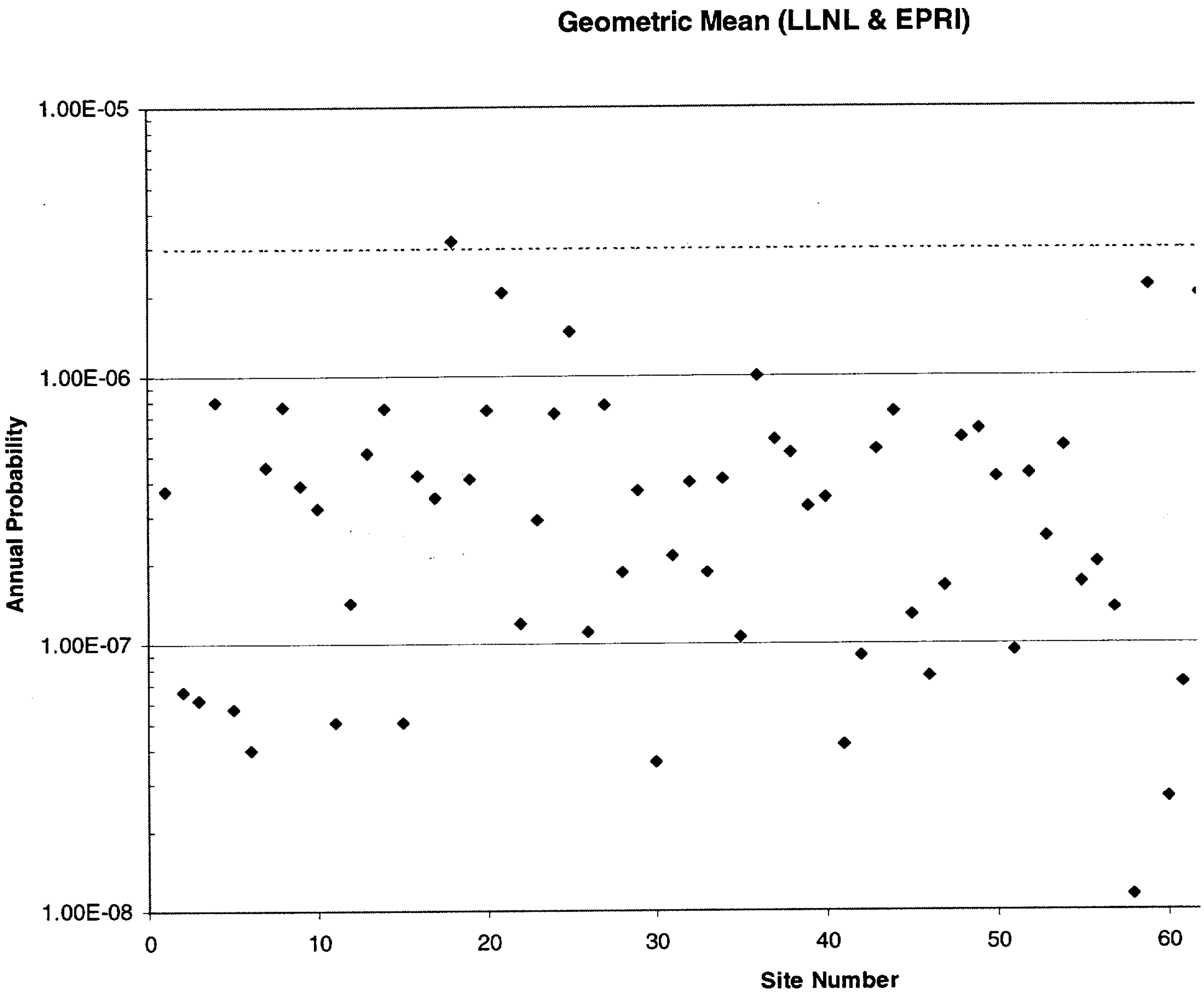
Site Number


Comparison of 1989 LLNL, 1992 LLNL and EPRI Estimates of Probability of Ex
Ground Acceleration per Year versus Acceleration - Pilgrim site
Note - From Figure 2 in NUREG - 1488
LLNL - 89
LLNL - 92
EPRI
Prob of Exceed/Yr
1.00E+00
1.00E-01
1.00E-02
1.00E-03
1.00E-04
1.00E-05
1.00E-06
1.00E-07
0.00
100.00
200.00
300.00
400.00
500.00
600.00
700.00
800.00
900.0
Acceleration - cm/sec**2
Fi



Geometric Mean (LLNL & EPRI)
1.00E-05
1.00E-06
1.00E-07
1.00E-08
Annual Probability
0
10
20
30
40
50
60
Site Number

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>

Spent Fuel Pool Seismic Vulnerability Check List

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.

Spent Fuel Pool Seismic Vulnerability Check List

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.

<u>Item 10:</u> **<u>Potential Mitigation Measures</u>**

<u>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.</u>

<u>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.</u>

<u>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.)</u>

<u>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.)</u>

Spent Fuel Pool Seismic Vulnerability Check List

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

Lynnette Hendricks
DIRECTOR
PLANT SUPPORT
NUCLEAR GENERATION DIVISION

November 12, 1999

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

Dear Mr. Barrett,

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

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

I am hereby transmitting those industry commitments as follows.

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

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

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

4. An off site resource plan will be developed which will include access to

portable pumps and emergency power to supplement on site resources. The plan would principally identify organizations or suppliers where off site resources could be obtained in a timely manner.

4. Spent fuel pool instrumentation will include readouts <u>and</u> alarms in the control room (or where personnel are stationed) for spent fuel pool temperature, water level, and area radiation levels.
5. 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.
6. 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.
7. 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.
8. 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.
9. 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 LXH@NEI.org.

Sincerely,

Lynnette Hendricks
LXH/lrh

Lynnette Hendricks
DIRECTOR,
PLANT SUPPORT
NUCLEAR GENERATION

April 19, 2000

Secretary
Attention: Rulemakings and Adjudications Staff
U.S. Nuclear Regulatory Commission
Washington, DC 20555-0002

Subject: Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, February 22, 2000 (65 F.R. 8752), Request for Comments

On behalf of the nuclear energy industry, NEI is pleased to provide comments on the NRC's Draft Final Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants. The Commission directed the staff to risk inform regulations for permanently shutdown nuclear power plants. A rigorous technical basis is the framework for successfully risk informing regulations. The staff's study is technically robust and is an excellent platform on which to base policy decisions. The study reflects input provided by stakeholders during the development of the study. The study includes industry commitments made, in part, in response to risk insights gleaned from the study.

We agree with the staff's conclusion on the low level of risk posed and the staff's assessment of the safety principles contained in the Commission's policy statement on risk. Specifically, we fully endorse the staff's conclusion:

> "In summary, the risk assessment shows low numerical risk results in combination with satisfaction of the safety principles as described in R.G. 1.174, such as defense-in-depth, maintaining safety margins, and performance monitoring. The staff concludes that under the assumptions of this study there is a low level of public risk from SFP accidents at decommissioning plants."

Most of our comments address the need to appropriately apply the results of the study and the staff's conclusions to fully risk inform requirements for decommissioning plants. Failure to do so will forego the bulk of the benefits to be derived from a risk informing process, i.e., applying resources to those areas that pose the highest risk and avoiding application of burdensome unnecessary regulatory requirements where the risk does not support the need for them. We believe failure to make full use of risk insights for spent fuel pools where the risk is so thoroughly and well characterized would set a negative precedent for other potentially more difficult risk informing initiatives at operating plants.

To fully accomplish the Commission's directive to risk inform decommissioning regulations NEI urges the staff charged with subsequent rulemaking to start with a clean slate. In other words, rather than determining which of the regulations applicable to operating plants apply to decommissioning, determine what controls are necessary to preserve the acceptable findings of the study, i.e., "there is a low level of public risk from SFP accidents at decommissioning plants." Namely, ensure controls are in place to preserve the assumptions in the study, e.g., the commitment to implement NUREG 0612, Control of Heavy Loads. We believe this approach will better ensure risk informed regulations and will simplify the process of amending rules where necessary for decommissioning plants. (Note, NEI will submit detailed comments on an approach and framework for amending the rules at a later date.)

Industry provides detailed comments on the seismic portion (see attachment A) of the risk study but only minor comments regarding other aspects of the methods and analysis of the risk posed by spent fuel pools. Most of our comments in attachment 1 address application of the results of the study to appropriately risk inform regulations applicable to decommissioning plants. As an example, consider the issue of off-site emergency preparedness[1]. This is one key area of regulations for operating plants where the low risk posed by decommissioning plants warrants relief. The key question is: "at what point can decommissioning licensees eliminate their off-site emergency plans?"

The risk study demonstrates very low probabilities associated with spent fuel pool accidents that could lead to the need for off-site emergency preparedness (a key element of which is evacuation). Operator recovery times for initiating events are very long and relatively insensitive to the time period after final plant shutdown.[2] Continuing the period for evacuation, as the staff has modeled it in their risk study, provides no significant benefit to public health and safety.[3] Therefore, off-site emergency preparedness is not an assumption necessary to preserve the results of the risk study.

Industry appreciates this opportunity to comment on the spent fuel pool risk study. If I can be of any assistance to you as you consider these comments please contact me (202 739-8109 or LXH@NEI.ORG.

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[1] This is an abbreviated discussion, not intended to be complete, but rather to illustrate a different thought process for the decommissioning rulemaking.

[2] According to Section 3.3 of the risk study, the time an operator has to restore makeup prior to bulk boiling is 90 hours one year after shutdown. At six months after shutdown, the time to bulk boiling is still 82 hours.

[3] See Case 1, Appendix A.

Sincerely,

Lynnette Hendricks

Attachments

Attachment 1

Industry Comments
NRC Draft Final Technical Study of
Spent Fuel Pool Accident Risk at
Decommissioning Nuclear Power Plants

Main Report Comments

Risk Assessment Report Section

Seismic:

Summary of NRC Position

The NRC study concludes that "The results of this report estimated the generic frequency of events leading to zirconium fires at decommissioning plants to be less than 3E-06 per year for a plant that implements the design and operational characteristics assumed in the risk assessment performed by the staff. ... The most significant contributor to this risk is a seismic event which exceeds the design basis earthquake."

The staff concludes that Spent Fuel Pools (SFPs) at operating Nuclear Power Plants (NPPs) are inherently rugged in terms of being able to withstand loads substantially beyond those for which they were designed. Consequently, SFPs have significant seismic capacity.

The staff also concludes that for those Central and Eastern United States (CEUS) plants where 3 X Safe Shutdown Earthquake (SSE) is less than or equal to the NEI screening criterion of 0.5g, then the seismic risk is acceptably low. A similar conclusion is reached for those Western United States (WUS) plants where 2 X SSE satisfies the screening criterion.

According to the staff, those CEUS sites (about 27) for which 3 X SSE exceeds 0.5g and the 2 WUS sites for which 2 X SSE exceeds 0.5g, would have to perform additional plant specific analyses to demonstrate a High Confidence Low Probability of Failure (HCLPF) value for their SFPs of 3 X SSE and 2X SSE, respectively, in order to demonstrate acceptably low seismic risk.

Industry Comments

NPPs that satisfy the requirements of the seismic checklist have a SFP HCLPF of 0.5g or greater. Only large Charleston-like earthquakes can generate ground motions of the amplitude, frequency content, and duration to challenge the seismic capacity of spent fuel pools that satisfy the seismic checklist. In no CEUS licensing proceeding has there been compelling data to require design to an earthquake of a magnitude which would challenge the seismic capacity of an SFP that satisfies the seismic checklist.

The basis for requiring a higher HCLPF value for plants with 3 X SSE greater than 0.5g is apparently the assumption that higher SSE levels are associated with higher seismic hazard levels - which is shown in the Appendix A to be an erroneous assumption. Furthermore, it has been previously shown, using just the LLNL results and Dr. Kennedy's methodology, that there are many sites where 3 X SSE is greater than 0.5g and the SFP failure frequency is well below those plants where 3 X SSE is less than 0.5g.

The focus of previous seismic hazard studies (LLNL and EPRI) has been at the SSE level. At high ground motion values (ground motion values that can be associated with damage to SFPs), the tail of the attenuation random uncertainty distribution allows, with some non-negligible probability, relatively small events to contribute to the probability of exceeding these high ground motion values. Deterministically, these results are not logical and therefore there is a strong basis for truncating the tail of the random uncertainty term at high ground motion values. Based on this information and information previously transmitted, use of the LLNL probabilistic estimates at low probability values may not be credible. EPRI results are also likely to be overly conservative at high ground motion values. See attachment A for a more detailed discussion of these points.

Based on the results of both probabilistic and deterministic evaluations, it is concluded that for all CEUS and some WUS Nuclear Power Plants (NPPs), regardless of SSE value, satisfaction of all the requirements of the seismic checklist provides sufficient documentation of an acceptably low level of seismic risk. This acceptably low level of seismic risk is deemed to be considerably lower than the bounding value of 3E-6 per year.

Thus, we conclude that there should be no SFP screening level distinctions based on plant SSE for the CEUS. For the WUS, it is reasonable to require that certain plants demonstrate a HCLPF of 2 X SSE.

Sabotage:

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

Nonetheless, the usual approach in granting security exemptions for decommissioning facilities has involved "shrinking" the physical and programmatic security requirements to that needed to support spent fuel safety. There is sufficient precedent, on a deterministic basis, to implement this approach in a rulemaking that avoids the need for future exemptions.

Finally, the rule on vehicle barriers is sufficiently flexible as written to allow licensees to relocate their barriers, as needed, for decommissioning.

Implications for Regulatory Requirements Report Section

1. Emergency Preparedness –

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

2. Security –

As discussed above.

3. Insurance –

The obligation for decommissioning plants to participate in the secondary financial protection (assessments for someone else's accident) should be reviewed in light of the low public risk posed by spent fuel pools for decommissioning plants. Industry does not believe that the risk justifies requiring participation, i.e., the majority of the 3 in 1 million risk of significant offsite consequences comes from an upper bound determination of the risk posed by seismic events, not on a best estimate of the seismic risk.

If it is determined that participation in the secondary financial protection will be required during the short time that decommissioning plants pose a non-zero risk, then the level of participation should be in proportion to a best estimate of the risk posed relative to the risk posed by operating plants. If any participation is required it should only be for the short period that clad surface temperatures greater than 570 degrees C (based on the spent fuel failure criteria of the thermal limit used under accident conditions for licensing of spent fuel dry storage casks) can occur in a loss of water configuration. The calculation of this temperature should be by approved methodology. However, in the absence of any calculation, the obligations should end after a period which is indicative of when there is reasonable assurance that the last core placed in a pool is incapable of attaining clad surface temperatures greater than 570 degrees C. Realistic assumptions regarding burnup histories and storage array details will lead to a time period much shorter than the 5 years proposed in the report. For example, the most recent exemption issued by the staff was issued within 18 months of shutdown.

Likewise, the capacity required for primary financial protection should be eliminated for consideration of any potential for accidents with significant off site consequences. For consideration of other events with onsite consequences, we propose that onsite coverage be reduced to $25M for the period when spent fuel remains in the pool and offsite coverage be reduced to $5-10M. (See supplemental industry comments submitted on financial protection rule for permanently shutdown plants, and NEI letter to Dave Mathews providing a basis for costs for cleanup of onsite spills.)

When fuel has been removed offsite or placed in an onsite ISFSI, we recommend onsite coverage be reduced to $25M while the site still contains significant sources of radioactive material (more than 1000 gallons of contaminated liquids). Onsite coverage could be reduced to zero when there are no sources exceeding 1000 gallons of liquid. Offsite coverage should be reduced to $5-10M for plants with fuel off site or in an onsite ISFSI.

If some consideration is required for the negligible potential for events with significant offsite consequences, the primary coverage required should be reduced in proportion to the reduced risk, i.e., in the same manner discussed above for proportional reduction in participation in secondary financial protection, and for the same time period.

Appendices - Section by Section Technical Comments:

1. Thermal Hydraulics

The range of outcomes, which depend on specific fuel burnup histories and storage array details, suggests that standard methods will need to be developed for a consistent application in applying the regulations.

2. Risk Assessment
    a. Methodology: *No Comments*
    b. Structural Integrity - Seismic Loads: (see Attachment A).
    c. Structural Integrity - Heavy loads

In Section 3.3.6 and footnote 7, the staff mischaracterizes the risk of heavy load drops for licensees choosing to do load drop analyses. A successful load drop analysis, by definition, demonstrates that off-site dose consequences are acceptable. Therefore, the risk associated with a heavy load drop that has been analyzed is negligible– i.e., it is not considered for events resulting in consequences that propagate to either a complete loss of inventory (and potential zircalloy fire), or, in license basis terms, fuel pin damage resulting in consequences in excess of Part 100.

Therefore, for purposes of a risk study, the only heavy loads component of risk is that contributed by a single failure proof crane approach.

d. Structural Integrity - Aircraft Crashes: *No Comments*
e. Structural Integrity – Tornadoes: *No Comments*

3. Criticality

No Comments

4. Consequences Assessment from Zirconium Fire

The Consequence Assessment for Zirconium Fires in the NRC draft final study provides the misleading conclusion that there is "about a factor-of-two reduction in prompt fatalities if the accident occurs after 1 year instead of thirty days." What the study does not note is that the absolute value of fatalities is a couple of orders of magnitude below the numbers for an operating plant. This is not surprising since it is the short-lived nuclides that drive this result. In addition the study does not highlight the fact that the most significant reduction in early fatalities occurs within the first thirty days. Although there is an additional factor of two reduction over the next 11 months, the more significant reduction is in the first month, again since the short-lived nuclides have largely decayed off in this period.

By failing to emphasize the above, the staff's risk study lends misleading support to the idea that a one year waiting period is justified prior to reducing emergency planning requirements. In fact, the risk study does not support this conclusion.

The consequence analyses contained in Appendix A also seem to contradict the staff's conclusion that one year is an appropriate waiting time for emergency planning. Presumably, the primary benefit of off-site emergency preparedness is to reduce prompt fatalities through evacuation. Yet, Case 1 in Appendix 4 which apparently was intended to support that assumption, contradicts this assumption. While there is not sufficient information in Appendix 4 to clearly understand the consequence analyses, Case 1 appears to indicate that evacuation provides no benefit in reducing prompt fatalities.

Finally, the staff's study seems to establish the one year delay time based on providing sufficient time for operator response to upset conditions. For instance, in Section 4.3.1, page 34, the staff notes: "This study indicates that a one-year period provides adequate 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." A true, but again, misleading statement.

Actually, a much shorter delay period supports the same conclusion. For instance, referring to Table 3.1 and subsequent text in Appendix 2, we see that one year after shutdown, the total time available for operator action (time to bulk boiling plus time to boil down) is 133 hours. Performing the same calculation for a six month delay period (which the staff does not do in the report) reveals 118 hours available for operator action. This is a substantial period of time, which allows the same conclusion that, i.e., this study indicates that a six-month period provides adequate decay time necessary to reduce the pool heat load to a level that would provide sufficient human response time.

Thus the risk informed conclusion that should be drawn from the Consequence Analysis is that the prompt fatalities are very small in comparison to operating reactor accidents, and are sufficiently reduced in the first month after shutdown to support eliminating off site emergency preparedness. Furthermore, even after a relatively short delay time, there is substantial time for operator action to respond to upset conditions.

On the other hand, there are restrictions on reducing off-site emergency preparedness that are part of the pre-existing license basis of the facility, that have little to do with decommissioning or the risk study, but nonetheless must be satisfied by a licensee in transitioning from operations to decommissioning. Most significant is the one rem off-site dose consequence (the so-called EPA protective action guideline) that distinguishes between off-site and on-site response. Below one rem, no off-site response is called for.

Independent of spent fuel pool events, there are accidents within a plant's license basis that can generate off-site doses during decommissioning. The dominant event is a fuel handling accident (e.g., dropping a fuel bundle that breeches the integrity of some fuel rods, thereby releases radioactivity). Examination of this event shows that the vast majority of off-site dose is due to iodine which fairly rapidly decays following fuel offload. In fact, it is straightforward to reanalyze a fuel handling accident to determine the point following shutdown at which the accident offsite dose drops below one rem, thereby establishing the point at which off-site emergency response capability can be eliminated.

5. Seismic Checklist

As a result of stakeholder interactions with NRC in 1999, it was 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. The industry developed a seismic screening checklist to identify and evaluate specific seismic characteristics. The checklist has been incorporated into the bases for the NRC evaluation. Successful application of the revised seismic checklist provides a high degree of assurance that the Spent Fuel Pool (SFP), High Confidence Low Probability of Failure (HCLPF) is 0.5g or greater. In no Central or Eastern United States licensing proceeding has there been compelling data to require design to an earthquake of a magnitude which would challenge the seismic capacity of an SFP that satisfies the seismic checklist. The industry is committed to completion of the requirements of the checklist, including a thorough spent fuel pool walkdown.

6. NEI Commitment Letter:

NEI reiterates that the industry will perform decommissioning with the same high level of commitment to safety as during operation of the plants. To that end, industry has made several commitments for procedures and equipment which would reduce the probability and consequence of spent fuel pool events during decommissioning. These commitments have been incorporated into the bases for the NRC evaluation and the industry stands ready to fulfill them.

Attachment A

Comments on Appendix 2.b. "Structural Integrity Seismic Loads"

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. Reference 1 provides the technical basis for determining the regulatory requirements for decommissioning plants using risk-informed decision making. Table 3.1 (Reference 1) provides a summary of the annual frequency of fuel uncovery associated with internal and external initiating events. Based on Table 3.1 it is estimated that the frequency of a zirconium fire is less than 3×10^{-6}, with the dominant contribution coming from seismic events. The seismic contribution is estimated to be less than 3×10^{-6}, while the contribution from all other initiating events is estimated to be 4×10^{-7}. <u>As described by the staff, other considerations indicate that the seismic contribution may be considerably lower.</u> Assumption of the generic frequency of events leading to a zirconium fire at decommissioning plants to be less than 3×10^{-6} per year is based on a plant satisfying the design and operational characteristics assumed in the risk assessment performed by the staff.

Comments on Appendix 2b Structural Integrity of Spent Fuel Pools Subject to Seismic Loads (Reference 1)

1. Introduction

No significant comments on this section other than to concur that spent fuel pools (SFPs) at operating nuclear power plants <u>and at decommissioning NPPs</u> are inherently rugged in terms of being able to withstand loads substantially beyond those for which they were designed. Consequently, SFPs have significant seismic capacity.

2. <u>Seismic Checklist</u>

It is not clearly noted in this section, but the important point is that successful application of the revised seismic checklist provides a high degree of assurance that the SFP HCLPF is 0.5g or greater. The comments on the conservatisms (in paragraph 2) associated with the design basis earthquake at licensed NPPs should be moved to a separate section. Furthermore, the deterministic method should be contrasted with the probabilistic method. This contrast is important because the deterministic method provides a powerful counter to the veracity of the probabilistic results at low probability levels.

Deterministic Methods vs Probabilistic Methods

Deterministic Methods

The design basis earthquake ground motion, or the SSE ground motion, for NPPs were based on the assumption of the largest event geophysically ascribable to a tectonic province or to a capable structure at the closest proximity of the province or fault to the site. In the case of the tectonic province in which the site is located, the event is assumed to occur at the site. For the Eastern seaboard, the Charleston event is the largest magnitude earthquake and current research has established that such large events are confined to the Charleston region. The New Madrid zone is another zone in the Central US where very large events have occurred. Recent research has identified the source structures of these large New Madrid earthquakes. Both of these earthquake sources are fully accounted for in the assessment of the SSE for currently licensed NPPs. The SSE ground motions for NPPs are based on conservative estimates of the ground motion from the largest earthquake estimate to be generated from the current tectonic regime. In deterministic analyses used in the licensing of existing NPPs, one standard deviation is considered sufficient to incorporate all the conservatism in the final ground motion estimate. For CEUS sites the typical NPP is designed for about a magnitude 5.3 to 5.5 (about 0.15g). The largest design basis earthquake for a CEUS site, based on detailed seismological, geological, and geophysical investigations, is magnitude 6.0 (about 0.25g). In no EUS licensing proceeding has there been compelling data to require design to an earthquake of a magnitude which would challenge the seismic capacity of an SFP that satisfies the seismic checklist. For WUS sites the design basis ground motion is generally governed by known active faults at known distances. Based on fault length and other deterministic factors the maximum earthquake potential can be estimated.

Probabilistic Methods

References 2 and 3 describe the Lawrence Livermore National Labs (LLNL) and Electric Power Research Institute (EPRI) seismic hazard methodologies. A seismic hazard analysis (SHA) estimates the seismic hazard at a site due to the potential occurrence of earthquakes in the region surrounding the site. Importantly, the historic seismic data is insufficient, at least for the CEUS, to use as the sole source of information for estimating the various parameters of the overall probability model. Therefore, it is necessary to rely on "expert opinion" to supplement the data. One fundamental expert opinion input to the SHA is the upper bound magnitude distribution for each earthquake source. Figure 1 contrasts the distribution of upper bound magnitude estimates assessed by the experts in the LLNL study for the host zones containing a New England NPP with the SSE determined by the 10CFR Part 100 Appendix A process. This distribution of upper bound magnitude may be plausible, but not necessarily a possible outcome. In other words, it is not based on any known structure in each host zone description that could cause earthquakes this large. Within this context, the assessed seismic hazard will generally be higher – because less is known and the distribution has more probability associated with extreme outcomes, **or,**

outcomes that in fact cannot occur. The effect of including these extreme outcomes is to predict incredible ground motions at credible probability levels. Expert opinion on the distribution of upper bound magnitude is but one of the many opinions rendered in the LLNL and EPRI studies that have profound effects on the perceived seismic hazard at low (10^{-6}) probability levels.

The LLNL methodology was initially developed in 1979 to determine SSE values for older NPPs in the Systematic Evaluation Program. The methodology was further developed to address the Charleston Issue (SECY-91-135, Reference 4), i.e., to evaluate the contribution to the seismic hazard from large earthquakes along the eastern seaboard outside the Charleston region. It should be noted that the focus of these studies was on the relative contribution of large earthquakes to the overall seismic hazard, not on the absolute effect. Also, comparisons between the LLNL and EPRI results was typically made at the SSE level (0.15g to 0.25g - annual probability of 10^{-3} to 10^{-4}), not at the ground motion level associated with a HCLPF of 0.5g. It is noted that given a HCLPF of 0.5g the median capacity (A_m) of an SFP is about 1.0g (A_m = HCLPF/$e^{-1.65(Bc)}$) – far from typical SSE values. Realistically, only large Charleston like earthquakes can generate ground motions of the amplitude, frequency content, and duration to challenge the seismic capacity of spent fuel pools. However, at high ground motion values (1000 cm/sec^2), the tail of the attenuation random uncertainty distribution (sigma) allows, with some non-negligible probability, relatively small events to contribute to the probability of exceeding a ground motion of 1000 cm/sec^2. Figure 2 shows the effect of changing sigma for a point source at a given distance. These results were analytically determined. As can be seen, at low ground motions (125 cm/sec^2), changes in sigma have a small effect on the probability of exceedance. However, at high accelerations (1000 cm/sec^2) the effect of changes in sigma is profound. The high probability of exceeding 1000 cm/sec^2 based on use of a sigma of 0.6g in Figure 2, is driven by the tail of the attenuation random uncertainty term. For example, 1000 cm/sec^2 is about 3 standard deviations above the expected ground motion from a magnitude 6.5 earthquake at 100 km. Clearly there must be a physical limit on the strength of ground motion that a given earthquake can generate. These results don't make sense and provide a basis for truncating the tail of the random uncertainty term at high ground motion values. As described previously, in deterministic analyses one standard deviation is considered sufficient to incorporate all the conservatism in the final ground motion estimate. Use of a smaller sigma value is a form of truncation. As can be seen on Figure 2, the probability of exceeding 1000 cm/sec^2 is reduced by about a factor 600 by simply changing sigma from 0.6 to 0.4. EPRI results are based on use of a sigma of 0.5. Based on this information and information previously described in Reference 5, use of the LLNL probabilistic estimates at high ground motion values may not be credible. EPRI results are also likely to be overly conservative at high ground motion values.

3. <u>Seismic Risk - Catastrophic Failure</u>

The staff concludes that for those CEUS plants where 3 X SSE is less than or equal to the NEI screening criterion of 0.5g, then the seismic risk is acceptable low. A similar conclusion is reached for those WUS plants where 2 X SSE satisfies the screening criterion. For CEUS plants that exceed the 3 X SSE screening criterion, a detailed SFP assessment will be required to demonstrate the SFP HCLPF equals 3 X SSE. A similar conclusion is reached for those WUS plants where 2 X SSE exceeds the screening criterion. This requirement that some plants with higher SSE values perform detailed HCLPF assessments of their SFPs is not be warranted. The assumption of this requirement is that the SSE is correlated with seismic hazard, in other words, the higher the SSE the higher the seismic hazard. Previous studies have shown that the SSE is poorly correlated with the seismic hazard (see Figure 3). In particular, there are many 0.2g to 0.25g SSE sites with lower seismic hazard estimates than 0.1g to 0.2g SSE sites. SSE tends to be more correlated with plant vintage than seismic hazard. **Based on this information, we conclude that there should be no SFP screening level distinctions based on plant SSE for the CEUS.** For the WUS, it is reasonable to require that certain plants demonstrate a HCLPF of 2 X SSE.

4. Seismic Risk - Support System Failure

No comments.

5. Conclusion

The staff concludes that for SFPs in the CEUS with HCLPF values of 3 X SSE or 0.5g whichever is greater and for WUS SFPs with HCLPF values of 2 X SSE or 0.5g, whichever is greater, the SFP failure frequency due to seismic is bounded by 3×10^{-6} per year. As stated by the staff, "other considerations indicate that the frequency may be significantly lower."

For CEUS plants that satisfy the seismic checklist and 3 X SSE is less than 0.5g, the seismic risk is considered by the staff to be acceptably low and no additional work is required. According to the staff, those CEUS sites (about 27) for which 3 X SSE exceeds 0.5g and 2 WUS sites for which 2 X SSE exceeds 0.5g would have to perform additional plant specific analyses to demonstrate a HCLPF value for their SFPs of 3 X SSE and 2X SSE respectively in order to demonstrate acceptably low seismic risk.

The conclusion that the SFP failure frequency is bounded by 3×10^{-6} per year can be found in previous submittals. In particular, it was shown that the assumption of a 0.5g HCLPF and applying Dr. Kennedy's conservative methodology to estimate SFP failure frequency at all CEUS sites using both the LLNL and EPRI seismic hazard results, the SFP failure frequency is bounded by 3×10^{-6} per year. It is noted that no distinction was made in the previous analysis concerning cases where 3 X SSE was greater than 0.5g. The basis for requiring a higher HCLPF value for plants with 3 X SSE greater than 0.5g is neither clear nor compelling. If the basis for requiring a higher HCLPF value for plants with high SSEs is that the SSE is assumed to be correlated with hazard it can readily be shown that seismic hazard and SSE are poorly correlated (Figure 3). Furthermore, it can be also be shown, using just the LLNL results and Dr. Kennedy's methodology, that there are many sites where 3 X SSE is greater than 0.5g **AND** the

SFP failure frequency is well below those sites where 3 X SSE is less than 0.5g.

Successful application of the revised seismic checklist provides a high degree of assurance that the SFP HCLPF is 0.5g or greater. It is noted that given a HCLPF of 0.5g the median capacity of an SFP is about 1.0g. Realistically, only large Charleston like earthquakes can generate ground motions of the amplitude, frequency content, and duration to challenge the seismic capacity of spent fuel pools. In no EUS licensing proceeding has there been compelling data to require design to an earthquake of a magnitude which would challenge the seismic capacity of an SFP that satisfies the seismic checklist. The focus of previous seismic hazard studies (LLNL and EPRI) has been at the SSE level. At high ground motion values (ground motion values that can be associated with damage to SFPs), the tail of the attenuation random uncertainty distribution (sigma) allows, with some non-negligible probability, relatively small events to contribute to the probability of exceeding these high ground motion values. These results don't make sense and provide a basis for truncating the tail of the random uncertainty term at high ground motion values. In deterministic analyses used in the licensing of existing NPPs, one standard deviation is considered sufficient to incorporate all the conservatism in the final ground motion estimate. Based on this information and information previously described in Reference 5, use of the LLNL probabilistic estimates at low probability values may not be credible. EPRI results are also likely to be overly conservative at high ground motion values.

Based on the results of both probabilistic and deterministic evaluations, it is concluded that for all CEUS and some WUS NPPs, regardless of SSE value, satisfaction of all the requirements of the seismic checklist provides sufficient documentation of an acceptably low level of seismic risk. For the 2 WUS plants at known high seismic hazard locations, a HCLPF value of 2 X SSE should be demonstrated. This acceptably low level of seismic risk is deemed to be considerably lower than the bounding value of 3E-6 per year.

References:

1. USNRC, Draft for Comment Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, February, 2000.

2. NUREG/CR-5250, Seismic Hazard Characterization of 69 Nuclear Plant Sites East of the Rocky Mountains, Lawrence Livermore Nation Laboratory, January, 1989.

3. EPRI NP-6396-D, Probabilistic Seismic Hazard Evaluations at Nuclear Plant Sites in the Central and Eastern United States: Resolution of the Charleston Issue, Electric Power Research Institute, April 1989.

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. Appendix 5e, USNRC, Draft for Comment Technical Study of Spent Fuel Pool Accident Risk at Decommissioning Nuclear Power Plants, February 2000.

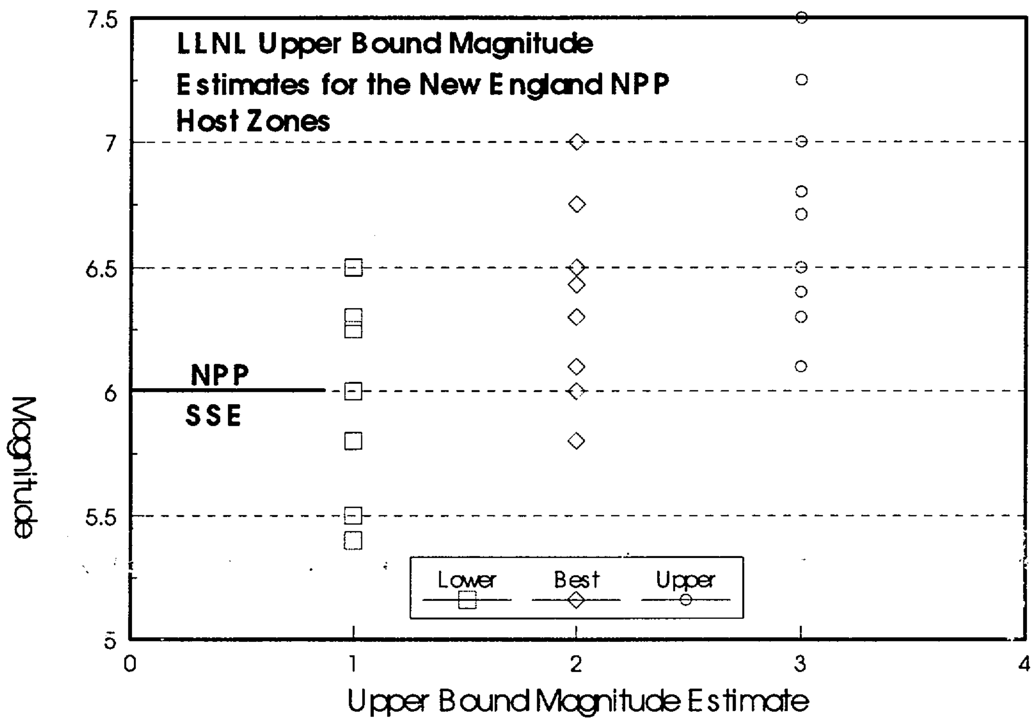


Figure 1 – Distribution of Upper Bound Magnitude Estimates from Reference 2 for a New England site.

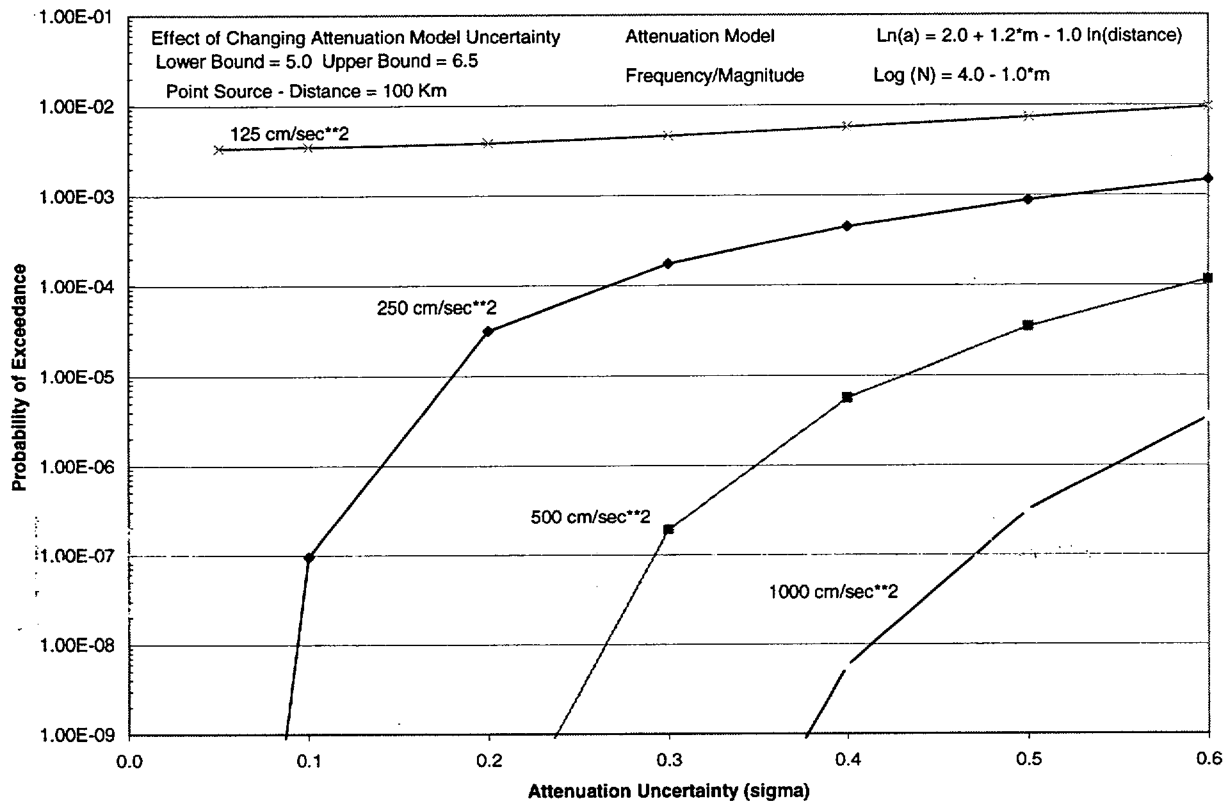


Figure 2 Effect of Attenuation Random Uncertainty on Probability of Exceedance from a Point Source

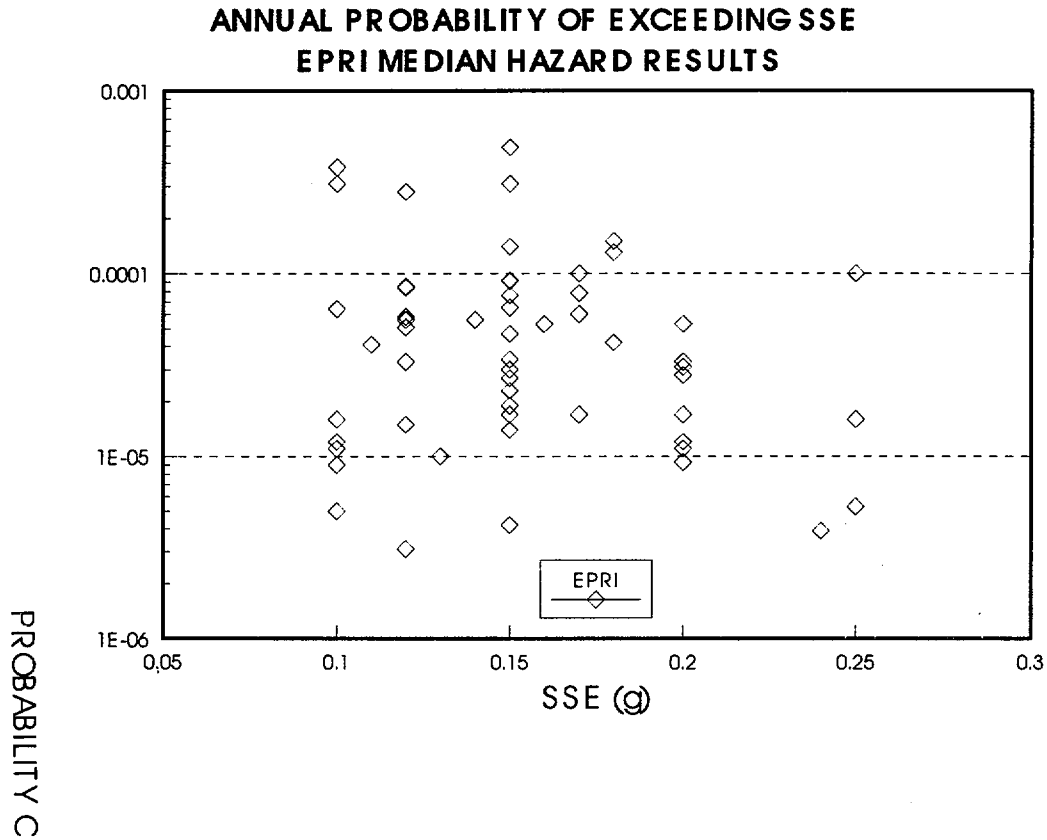
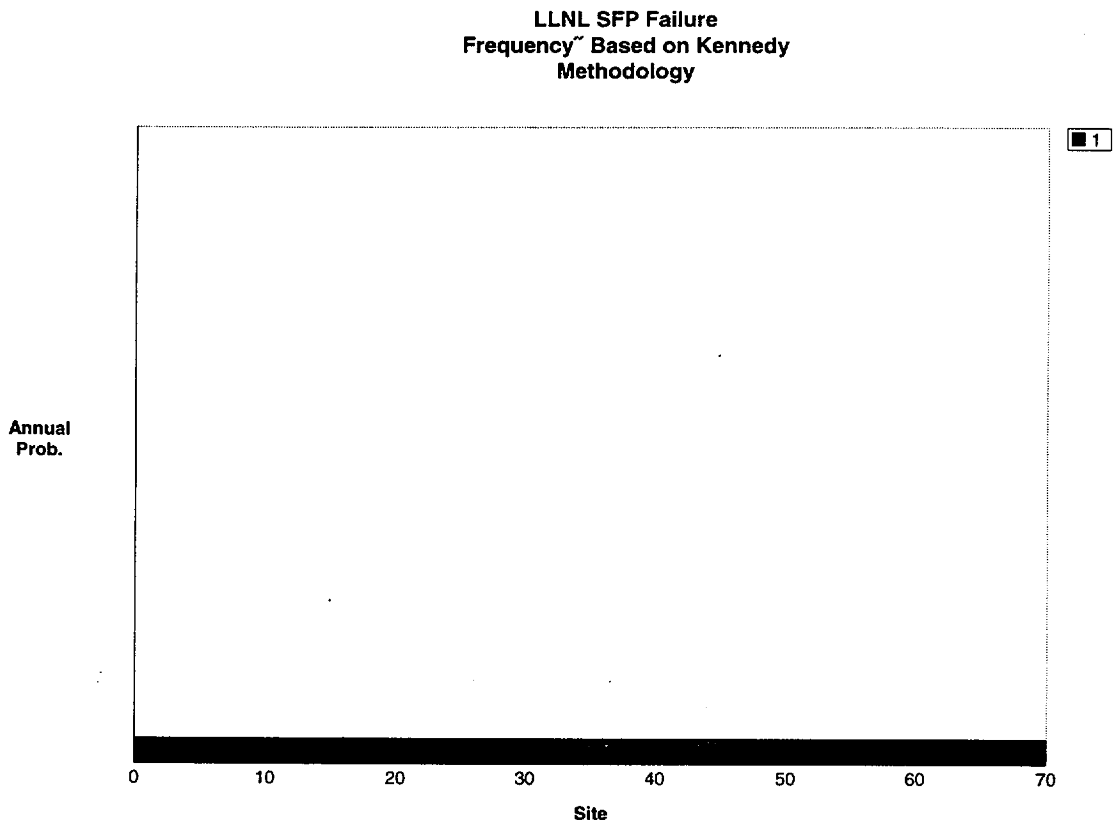
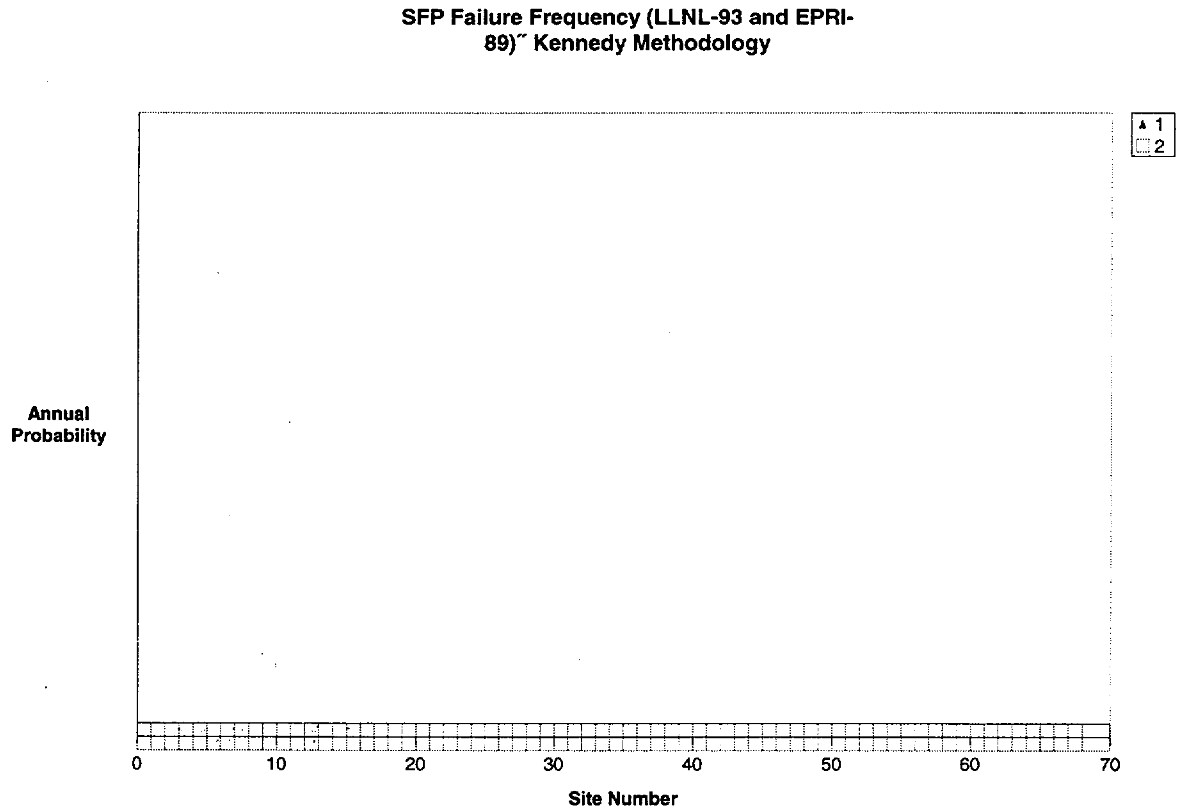
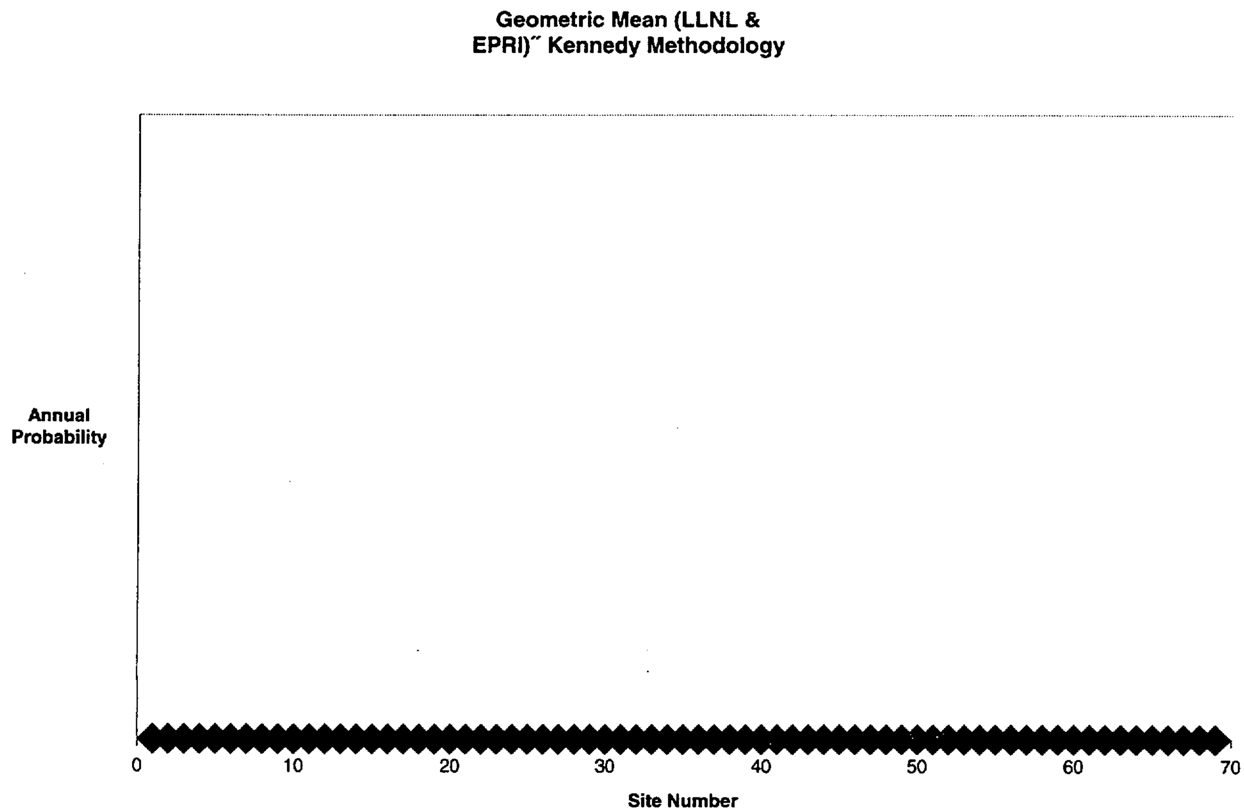


Figure 3 – Annual Probability of Exceeding the SSE at CEUS sites based on EPRI (Reference 3)


LLNL SFP Failure
Frequency˜ Based on Kennedy
Methodology
Annual
Prob.
1
0
10
20
30
40
50
60
70
Site


SFP Failure Frequency (LLNL-93 and EPRI-89)" Kennedy Methodology
1
2
Annual Probability
0
10
20
30
40
50
60
70
Site Number


Geometric Mean (LLNL & EPRI)˝ Kennedy Methodology
Annual Probability
0
10
20
30
40
50
60
70
Site Number

	LLNL	LLNL	LLNL	LLNL	LLNL		L+E	L+E	
Site #	2.5hz	5hz	10hz	Max (2.5, 5, 10)	Max*0.5	EPRI	Avg.	Geometric	
1	1.56E-06	2.00E-06	1.04E-06	2.00E-06	1.00E-06	2.27E-07	6.14E-07	3.73E-07	1.50E-06
2	4.23E-07	5.18E-07	2.44E-07	5.18E-07	2.59E-07	3.03E-08	1.45E-07	6.62E-08	1.50E-06
3	4.04E-07	5.45E-07	2.69E-07	5.45E-07	2.72E-07	2.55E-08	1.49E-07	6.16E-08	1.50E-06
4	2.33E-06	3.22E-06	1.96E-06	3.22E-06	1.61E-06		8.06E-07	8.06E-07	1.50E-06
5	1.94E-06	1.40E-06	4.35E-07	1.94E-06	9.68E-07	6.80E-09	4.88E-07	5.76E-08	1.50E-06
6	2.02E-07	2.44E-07	1.16E-07	2.44E-07	1.22E-07	2.22E-08	7.20E-08	4.00E-08	1.50E-06
7	1.36E-06	1.76E-06	9.54E-07	1.76E-06	8.79E-07	3.42E-07	6.11E-07	4.57E-07	1.50E-06
8	1.11E-05	9.02E-06	2.43E-06	1.11E-05	5.54E-06	2.08E-07	2.87E-06	7.73E-07	1.50E-06
9	1.71E-06	2.41E-06	1.34E-06	2.41E-06	1.21E-06	2.16E-07	7.11E-07	3.92E-07	1.50E-06
10	1.39E-06	1.90E-06	1.10E-06	1.90E-06	9.50E-07	1.85E-07	5.68E-07	3.24E-07	1.50E-06
11	5.24E-07	7.07E-07	3.63E-07	7.07E-07	3.53E-07	1.40E-08	1.84E-07	5.07E-08	1.50E-06
12	1.01E-06	1.39E-06	7.70E-07	1.39E-06	6.95E-07	5.41E-08	3.75E-07	1.42E-07	1.50E-06
13	1.84E-06	2.57E-06	1.42E-06	2.57E-06	1.29E-06	3.27E-07	8.07E-07	5.14E-07	1.50E-06
14	1.88E-06	2.63E-06	1.41E-06	2.63E-06	1.32E-06	5.99E-07	9.58E-07	7.58E-07	1.50E-06
15	5.28E-07	7.14E-07	3.68E-07	7.14E-07	3.57E-07	1.40E-08	1.86E-07	5.10E-08	1.50E-06
16	5.02E-06	3.41E-06	1.00E-06	5.02E-06	2.51E-06	1.37E-07	1.32E-06	4.26E-07	1.50E-06
17	1.65E-06	2.15E-06	1.09E-06	2.15E-06	1.08E-06	1.94E-07	6.35E-07	3.51E-07	1.50E-06
18	9.05E-06	1.75E-05	1.08E-05	1.75E-05	8.74E-06	1.89E-06	5.32E-06	3.17E-06	1.50E-06
19	3.75E-06	2.23E-06	1.05E-06	3.75E-06	1.88E-06	1.68E-07	1.02E-06	4.14E-07	1.50E-06
20	2.28E-06	2.94E-06	1.67E-06	2.94E-06	1.47E-06	5.50E-07	1.01E-06	7.46E-07	1.50E-06
21	6.36E-06	8.23E-06	1.87E-06	8.23E-06	4.11E-06		2.06E-06	2.06E-06	1.50E-06
22	1.13E-06	1.36E-06	7.00E-07	1.36E-06	6.82E-07	3.89E-08	3.61E-07	1.18E-07	1.50E-06
23	1.63E-06	2.14E-06	1.10E-06	2.14E-06	1.07E-06	1.42E-07	6.07E-07	2.94E-07	1.50E-06
24	1.59E-06	2.19E-06	1.25E-06	2.19E-06	1.10E-06	6.11E-07	8.54E-07	7.22E-07	1.50E-06
25	1.21E-05	1.37E-05	4.41E-06	1.37E-05	6.83E-06	5.71E-07	3.70E-06	1.45E-06	1.50E-06
26	8.32E-07	1.05E-06	5.11E-07	1.05E-06	5.24E-07	4.29E-08	2.84E-07	1.10E-07	1.50E-06
27	2.30E-06	5.33E-06	5.70E-06	5.70E-06	2.85E-06	3.78E-07	1.61E-06	7.81E-07	1.50E-06
28	3.46E-06	2.43E-06	8.62E-07	3.46E-06	1.73E-06	3.86E-08	8.83E-07	1.85E-07	1.50E-06
29	1.17E-06	1.56E-06	9.36E-07	1.56E-06	7.78E-07	2.65E-07	5.21E-07	3.72E-07	1.50E-06
30	3.20E-07	4.59E-07	2.61E-07	4.59E-07	2.30E-07	1.11E-08	1.20E-07	3.66E-08	1.50E-06
31	1.79E-06	2.89E-06	5.35E-07	2.89E-06	1.45E-06	6.15E-08	7.53E-07	2.15E-07	1.50E-06
32	3.54E-06	2.31E-06	1.06E-06	3.54E-06	1.77E-06	1.68E-07	9.70E-07	4.04E-07	1.50E-06
33	8.50E-07	1.02E-06	5.15E-07	1.02E-06	5.10E-07	1.11E-07	3.11E-07	1.86E-07	1.50E-06
34	1.02E-06	1.27E-06	6.24E-07	1.27E-06	6.33E-07	3.52E-07	4.92E-07	4.16E-07	1.50E-06
35	1.13E-06	1.54E-06	8.11E-07	1.54E-06	7.69E-07	2.86E-08	3.99E-07	1.07E-07	1.50E-06
36	2.32E-05	2.81E-05	4.67E-06	2.81E-05	1.41E-05	1.42E-07	7.10E-06	1.00E-06	1.50E-06
37	1.33E-06	1.71E-06	9.13E-07	1.71E-06	8.56E-07	4.95E-07	6.75E-07	5.78E-07	1.50E-06
38	3.42E-06	4.59E-06	2.35E-06	4.59E-06	2.29E-06	2.13E-07	1.25E-06	5.17E-07	1.50E-06
39	2.79E-06	1.84E-06	9.73E-07	2.79E-06	1.40E-06	1.40E-07	7.68E-07	3.28E-07	1.50E-06
40	1.25E-06	5.01E-06	2.83E-06	5.01E-06	2.50E-06	9.71E-08	1.30E-06	3.55E-07	1.50E-06
41	1.02E-06	1.32E-06	2.07E-06	2.07E-06	1.03E-06	3.45E-09	5.19E-07	4.23E-08	1.50E-06
42	7.60E-07	9.45E-07	4.85E-07	9.45E-07	4.73E-07	3.20E-08	2.52E-07	8.98E-08	1.50E-06
43	5.49E-06	8.90E-06	1.87E-06	8.90E-06	4.45E-06	1.23E-07	2.29E-06	5.30E-07	1.50E-06
44	2.65E-06	2.79E-06	2.93E-06	2.93E-06	1.46E-06		7.32E-07	7.32E-07	1.50E-06
45	1.08E-06	1.37E-06	6.85E-07	1.37E-06	6.85E-07	4.53E-08	3.65E-07	1.29E-07	1.50E-06
46	5.07E-07	6.74E-07	3.29E-07	6.74E-07	3.37E-07	3.08E-08	1.84E-07	7.53E-08	1.50E-06
47	6.17E-07	8.00E-07	4.18E-07	8.00E-07	4.00E-07	1.06E-07	2.53E-07	1.64E-07	1.50E-06
48	6.36E-07	2.13E-06	1.16E-06	2.13E-06	1.06E-06	4.61E-07	7.62E-07	5.93E-07	1.50E-06
49	4.55E-06	5.39E-06	2.91E-06	5.39E-06	2.70E-06	2.71E-07	1.48E-06	6.34E-07	1.50E-06
50	1.84E-06	3.17E-06	1.25E-06	3.17E-06	1.59E-06	2.01E-07	8.94E-07	4.24E-07	1.50E-06
51	5.71E-07	7.41E-07	3.88E-07	7.41E-07	3.71E-07	4.26E-08	2.07E-07	9.38E-08	1.50E-06
52	6.09E-07	2.06E-06	1.16E-06	2.06E-06	1.03E-06	2.83E-07	6.57E-07	4.31E-07	1.50E-06
53	1.14E-06	2.29E-06	1.12E-06	2.29E-06	1.14E-06	1.01E-07	6.22E-07	2.51E-07	1.50E-06
54	2.37E-06	4.32E-06	2.64E-06	4.32E-06	2.16E-06	2.51E-07	1.21E-06	5.50E-07	1.50E-06
55	1.69E-06	2.15E-06	9.83E-07	2.15E-06	1.08E-06	5.13E-08	5.64E-07	1.70E-07	1.50E-06
56	1.18E-06	1.49E-06	7.93E-07	1.49E-06	7.43E-07	9.62E-08	4.20E-07	2.01E-07	1.50E-06
57	4.80E-07	5.48E-07	2.49E-07	5.48E-07	2.74E-07		1.37E-07	1.37E-07	1.50E-06
58	9.12E-08	1.30E-07	7.10E-08	1.30E-07	6.51E-08	3.80E-09	3.45E-08	1.14E-08	1.50E-06
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61	2.22E-07	2.86E-07	1.61E-07	2.86E-07	1.43E-07		7.15E-08	7.15E-08	1.50E-06
62	4.25E-06	7.43E-06	8.03E-06	8.03E-06	4.02E-06		2.01E-06	2.01E-06	1.50E-06
63	4.41E-06	2.75E-06	4.70E-07	4.41E-06	2.20E-06	5.91E-09	1.10E-06	8.08E-08	1.50E-06
64	6.10E-07	2.33E-06	1.75E-06	2.33E-06	1.16E-06	2.50E-07	7.07E-07	4.20E-07	1.50E-06
65	3.96E-07	5.41E-07	2.41E-07	5.41E-07	2.70E-07	1.91E-08	1.45E-07	5.26E-08	1.50E-06
66	9.98E-07	7.16E-07	1.93E-07	9.98E-07	4.99E-07	2.90E-09	2.51E-07	2.70E-08	1.50E-06
67	9.94E-07	9.40E-07	1.68E-07	9.94E-07	4.97E-07		2.49E-07	2.49E-07	1.50E-06
68	2.34E-07	3.19E-07	1.81E-07	3.19E-07	1.59E-07		7.97E-08	7.97E-08	1.50E-06
69	2.38E-06	1.37E-06	3.73E-07	2.38E-06	1.19E-06	2.75E-09	5.96E-07	4.05E-08	1.50E-06

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From: "NELSON, Alan" <apn@nei.org>
To: "'GTH@NRC.GOV'" <GTH@nrc.gov>, "'TEC@NRC.GOV'" <TEC@nrc.gov>
Subject: SEISMIC DATA AND CORRESPONDENCE
Date: Fri, 4 Aug 2000 10:04:27 -0400
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