RULEMAKING ISSUE

September 26, 2001 (NEGATIVE CONSENT) SECY-01-0178

FOR: The Commissioners

FROM: William D. Travers
Executive Director for Operations

SUBJECT: MODIFIED RULEMAKING PLAN: 10 CFR Part 72 -- "GEOLOGICAL AND SEISMOLOGICAL CHARACTERISTICS FOR SITING AND DESIGN OF DRY CASK INDEPENDENT SPENT FUEL STORAGE INSTALLATIONS"

PURPOSE:

To request, by negative consent, Commission approval of the attached Modified Rulemaking Plan for amending certain sections in 10 CFR Part 72 dealing with seismic siting and design criteria for dry cask independent spent fuel storage installations (ISFSIs). The staff proposes modifications to the approved Rulemaking Plan, SECY-98-126, "Rulemaking Plan: Geological and Seismological Characteristics for the Siting and Design of Dry Cask Independent Spent Fuel Storage Installations, 10 CFR Part 72."

SUMMARY:

The Commission is amending certain sections in 10 CFR Part 72 dealing with seismic siting and design criteria for dry cask independent spent fuel storage installations (ISFSIs). The staff proposes modifications to the approved Rulemaking Plan, SECY-98-126, "Rulemaking Plan: Geological and Seismological Characteristics for the Siting and Design of Dry Cask Independent Spent Fuel Storage Installations, 10 CFR Part 72."

The Rulemaking Plan in SECY-98-126 provided three options. Option 3, recommended by the staff and approved by the Commission in its SRM to SECY-98-126, adopted the Probabilistic Seismic Hazard Analysis (PSHA) and also provided an option to use the risk-informed graded approach to seismic design for ISFSI SSCs. An additional change was recommended in SECY-98-126 to require that the design of cask storage pads and areas account for dynamic loads in addition to static loads for general licensees.

CONTACTS: Keith McDaniel, NMSS/IMNS
(301) 415-5252

Mahendra Shah, NMSS/SFPO
(301) 415-8537

State's
Exhibit 128

MODIFIED RULEMAKING PLAN
GEOLOGICAL AND SEISMOLOGICAL CHARACTERISTICS
FOR THE SITING AND DESIGN OF DRY CASK ISFSIs
10 CFR PART 72

REGULATORY PROBLEM

In 1980, the Commission added 10 CFR Part 72 to its regulations to establish licensing requirements for the storage of spent fuel in an independent spent fuel storage installation (ISFSI), (45 FR 74693). Subpart E of Part 72 contains siting evaluation factors that must be investigated and assessed with respect to the siting of an ISFSI, including a requirement for evaluation of geological and seismological characteristics. The original provision (10 CFR 72.66) (45 FR 74708) distinguished between massive water basin and air-cooled canyon types of ISFSI structures and other types of ISFSI designs. For the former, section 72.66 (now section 72.102) required seismic evaluations equivalent to those required for nuclear power plants (NPPs) when the ISFSI was located west of the Rocky Mountain Front (approximately 104° west longitude) or in areas of known potential seismic activity. At that time, ISFSIs were largely envisioned to be spent fuel pools or single, massive dry storage structures. A seismic design requirement, equivalent to the requirements for an NPP (Appendix A of 10 CFR Part 100) seemed appropriate for these types of facilities, given the potential accident scenarios. For other types of ISFSI designs, the regulation required a site-specific investigation to establish site suitability commensurate with the specific requirements of the proposed ISFSI. The Commission explained that "[f]or ISFSI's which do not involve massive structures, such as dry storage casks and canisters, the required design earthquake will be determined on a case-by-case basis until more experience is gained with the licensing of these types of units." [45 FR 74697 (1980)]. The NRC staff believed that a major seismic event at an ISFSI storing spent fuel in dry casks or canisters would most likely have minor radiological consequences compared with a major seismic event at an NPP, spent fuel pool, or single massive storage structure.

Part 72 was amended in 1988 to include the U.S. DOE Monitored Retrievable Storage Installation (MRS), (53 FR 31651). The 1988 amendment also relocated the provision governing evaluation of geological and seismological characteristics to section 72.102. It also eliminated the distinction formerly made between criteria for massive water basin and air-cooled canyon types of ISFSI structures and other types of ISFSI designs such that the criteria designed for massive structures now applied to all ISFSI and MRS facilities. Thus, section 72.102 requires that, for any site located west of the Rocky Mountain Front or in any areas of known potential seismic activity, seismicity be evaluated by the techniques of Appendix A of Part 100 and that, for sites evaluated under the Appendix A criteria, the design earthquake be equivalent to the safe shutdown earthquake (SSE) for an NPP. For sites located east of the Rocky Mountain Front and not in areas of known seismic activity, the Appendix A criteria may be used to determine a site-specific design earthquake or, alternatively, a standardized design earthquake described by an appropriate response spectrum anchored at a peak ground acceleration of 0.25 g may be used.

The procedures in Appendix A of Part 100 for determining the design basis vibratory ground motion at a site require the use of "deterministic" approaches in the development of a single set

surrounding the fuel assemblies would confine these nuclides. Therefore, the radiological risk associated with an ISFSI facility is significantly less than the risk associated with an NPP and the use of a lower design earthquake ground motion is justified.

The Commission indicated in the Statement of Considerations accompanying the initial Part 72 rulemaking that "[f]or ISFSI's which do not involve massive structures, such as dry storage casks and canisters, the required design earthquake will be determined on a case-by-case basis until more experience is gained with the licensing of these types of units." [45 FR 74697 (1980)]. With more than 10 years of experience licensing dry cask storage systems, together with analyses demonstrating their robust behavior in accident scenarios involving earthquakes, the NRC staff concludes that designing ISFSI SSCs using a single-level design earthquake with a ground motion that is commensurate with the level of risk associated with an ISFSI, is sufficient to provide reasonable assurance in demonstrating public health and safety.

The rationale for the proposed mean annual probability of exceedance of 5.0E-04 (return period of 2,000 years) for a design earthquake is based on several points:

- Use of a mean annual probability of exceedance of 5.0E-04 (return period of 2,000 years) for the design earthquake is consistent with the Commission's approval of DOE's request for an exemption from section 72.102(f)(1) for a proposed ISFSI at the INEEL to store spent fuel generated at the Three Mile Island Unit-2 nuclear power plant. Section 72.102(f)(1) requires that for sites that have been evaluated under the criteria of Appendix A of Part 100, the design earthquake must be equivalent to the SSE for an NPP. In its evaluation of the request, NRC staff considered the relative risk posed by the ISFSI. The staff concluded that considering the minor radiological consequences expected from a cask failure resulting from a seismic event, and the lack of a credible mechanism to cause such a failure, the NRC staff believes that the design earthquake using a mean annual probability of exceedance of 5.0E-04 for dry storage facilities at INEEL would be conservative.

- The total probability of exceedance for a design earthquake at an ISFSI facility with an operational period of 20 years (20 years x 5.0E-04 = 1.0E-02) is the same as the total probability of exceedance for an earthquake event at the proposed pre-closure facility at Yucca Mountain with an operational period of 100 years (100 years x 1.0E-04 = 1.0E-02).

- Because SSCs important to safety in an ISFSI are few, relative to those found in an NPP, the use of a graded approach for classifying ISFSI SSCs into one of two different categories for earthquake designs would unnecessarily increase the complexity in applications, without a commensurate improvement to safety. The SSCs important to safety in an ISFSI are associated with the storage cask, and include the canister, the canister handling systems, concrete pad supporting the cask, the transfer building supporting the handling systems, and the transfer cask. Since these SSCs are needed to be functional to prevent the dose limit of 5 rem being exceeded at the controlled area boundary, they would be required to be designed for a Category 2 design basis earthquake. Other SSCs important to safety may include the pressure monitoring system, protective cover, security lock and wire, etc. and can be designed for a lower Category 1 earthquake. However, it would be simpler to design all SSCs for a bounding Category 2 earthquake.

STATE OF UTAH's PREFACE TO TESTIMONY OF DRs STEVEN F.BARTLETT & FARHANG OSTADAN ON CONTENTION UTAH L/QQ, Lack of Design Conservatism

I. The 2,000 Year Design Earthquake is Inconsistent with Other Standards.

A. Draft U.S. Department of Energy Standard-1020-2001, which establishes design standards and guidance for nuclear facilities, including nuclear storage facilities, requires a 2,500-year return period ground motion for performance category 3 SSCs.

B. Seismic hazard maps published by U.S. Geological Survey and the National Earthquake Hazard Reduction Program adopted a 2,500-year motion for design use.

C. The International Building Code and American Association of State Highway and Transportation Officials require a 2,500-year motion for design.

D. All Utah interstate highway bridges must be designed to levels of strong ground motion equivalent to an average return period of 2,500 years.

II. Factors Affecting Selection of Design Earthquake.

A. The DE and the seismic performance of SSCs are inextricably linked, thus, any claims of conservatism are meaningless if a DE is selected without adequately determining the site specific seismic performance of SSCs and risk reduction ratios.

B. In the PFS case, in determining the appropriate DE, consideration of the seismic performance of the SSCs at PFS is even more critical because:

1. The PFS site is located in a highly seismic area where the 2,000-year DE ground acceleration are 0.711 g in the horizontal direction and 0.695 g in the vertical direction and the 84th percentile peak ground accelerations are 1.15 g in the horizontal direction and 1.17 g in the vertical direction.
2. A 2,000-year DE significantly reduced the design standard from the design basis earthquake standard posed in the regulation or the 84th percentile DE.
3. PFS's unconventional design to allow unanchored casks and storage pads to slide and rotate in a high seismic area supported by cement treated soil is unprecedented and thus, PFS cannot rely on any direct experience to support its desired DE.
4. PFS's unconservative assumptions and the lack of redundancy in the design and analysis.

C. If followed in its entirety, DOE Standard 1020 is an acceptable methodology where:

1. Documented and peer reviewed performance goals and the requisite conservatism (risk reduction ratios) are established based on the classification hazard of the SSCs.
2. Unacceptable performance is damage to the SCC beyond which hazardous material confinement and safety-related functions are impaired.
3. Fragility curves would show the expected damage or unacceptable performance of an SSC as a function of the amplitude of strong ground motion to determine whether the performance goal is met for the DE.

III. PFS Failed to Demonstrate the Seismic Performance of the SSCs are Adequate to Accommodate a 2,000 year Design Earthquake.

A. PFS failed to generated any fragility curves or other mechanism to demonstrate the seismic performance of its specific SSCs at the PFS site,

B. Numerous errors, omissions, lack of conservative assumptions in PFS's analysis result in the failure to demonstrate adequate conservatism has been applied in the seismic design of foundations for the storage pads and CTB and to the seismic stability of the pads and casks for the proposed DE.

C. PFS's cask stability analysis is inadequate. Also, the casks may tipover at the DE.

IV. Performance Goals Are Not Clearly Inherent in ISFSI and Cask SRPs.

A. Casks must be designed to NUREG-1536 and the CTB must be designed to NUREG-1567

not nuclear power plant ("NNP") standards.

B. PFS has not determined the seismic performance of its SSCs
   1. The standards in NUREG-1536 and NUREG-1567 may already be lower than standards for NNPs
   2. PFS and NRC staff claim consequences at ISFSIs are less than consequences at NNPs.
   3. Because PFS's cask may tipover at a 2,000-year DE, NUREG-1567 is not met.
   4. The unanchored casks supported by cement treated soil at PFS's ground motions are not encompassed in the NUREG/CR-6728 fragility curves.

V. Conclusion.

A. PFS has not met the requirements of DOE-STD-1020 and NUREG/CR-6728.

B. PFS Cannot assess the seismic performance for the storage pads, storage casks, and the CTB and their foundations because there are no fragility curves and because of many errors, omissions and unconservative assumptions in PFS's evaluations.

C. The selection of the proposed 2,000 year DE is not founded on a proper technical basis and is basically arbitrary.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of:	)	Docket No. 72-22-ISFSI
PRIVATE FUEL STORAGE, LLC (Independent Spent Fuel Storage Installation)	)	ASLBP No. 97-732-02-ISFSI
	)	April 1, 2002

STATE OF UTAH TESTIMONY OF DR. STEVEN BARTLETT AND R. FARHANG OSTADAN ON UNIFIED CONTENTION UTAH L/QQ, PART E (Lack of Design Conservatism)

I. Purpose of Testimony.

Q. 1: Dr. Bartlett, please state your name for the record.

A. 1: (SFB) My name is Dr. Steven F. Bartlett.

Q. 2: Dr. Ostadan, please state your name for the record.

A. 2: (FO) My name is Dr. Farhang Ostadan.

Q. 3: What is the issue that you are testifying on?

A. 3: (SFB, FO) PFS's request to the NRC to be exempted from existing regulations relating to selection of the design basis earthquake.

Q. 4: What is your understanding of the basis for PFS's request?

A. 4: (SFB, FO) PFS has requested an exemption from the seismic requirements put forth in 10 CFR § 72.102(f)(1) that requires the design basis earthquake ("DBE") be equivalent to the deterministic or maximum credible earthquake. Instead, PFS has proposed the adoption of a significantly lower DBE ground motion that has a mean annual probability of exceedance of 5×10^{-4} (*i.e.*, 2,000-year return period). Central to PFS's argument for the adoption of the 2,000-year DBE is its position that additional conservatisms are built into NRC standard review plans ("SRP") and these conservatisms justify the use of a lower DBE ground motion. However, NRC standard review plans do not address seismic design criteria applicable to PFS's unconventional design features (*e.g.*, unanchored casks undergoing

"controlled" sliding resting on a shallowly embedded foundations, buttressed by soil cement and subjected to high levels of strong ground motion).

Q. 5: What is the purpose of your testimony?

A. 5: (SFB, FO) In our testimony on dynamic analysis, filed concurrently, we show that the proposed foundation and unanchored cask designs have many unconservative assumptions, incomplete analysis, which make PFS's claim of "additional conservatisms" baseless. In this testimony we show that PFS has not demonstrated acceptable performance of the structure/foundation/soil system for the proposed 2,000-year DBE and cannot claim "additional conservatisms" exist in the seismic design.

In this testimony we will explain the basis for each of our individual professional opinions that the appropriate design earthquake must be inextricably linked to the performance of structures, systems and components important to safety ("SSCs") at the PFS facility. The key in selecting a design earthquake is to conservatively evaluate the performance of the SSC subjected to the design basis ground motion. This evaluation cannot be made absent consideration of the collective experience gained from previous design and performance of other SSCs subject to similar seismic loading. We will describe how PFS has failed to conservatively and adequately evaluate the performance of the SSCs under a 2,000-year DBE. In our opinion, a standard based on a 2,000 year DBE cannot be supported.

Q. 6: Do you consider it necessary to testify together?

A. 6: (SFB, FO) Yes. We each have different individual expertise that complements the other's expertise. As a result we each will bring a perspective from our unique engineering disciplines that we believe will aid the Licensing Board in determining this issue. Although we bring differing expertise to this issue, our individual opinions are in agreement as to the requisite factors in determining a safe design earthquake for the PFS facility. We both agree that a design earthquake cannot be designated without considering the seismic performance of specific SSCs at the PFS facility and where applicable, the appropriate risk reduction factor. Therefore, we are testifying as a panel to make our testimony more cohesive and easier to understand.

(SFB) In adequately analyzing the seismic performance and selection of a DBE, my contribution to this hearing is from the perspective of how, based on PFS's design, the capacity of the soil and foundations will withstand a 2,000-year DBE. Also, because I have applied DOE Standard 1020 ("DOE-STD-1020") and am familiar with its philosophy, I will present testimony on the concepts embedded in DOE-STD-1020.

(FO) Similarly, the expertise that I bring to this hearing relates to the loads from structures during a 2,000-year DBE that will be transferred to the foundations and soil. I

have also applied DOE-STD-1020 in seismic analysis and will also offer some testimony on the application of the DOE standard.

II. Qualifications and Background.

Q. 7: Dr. Bartlett, have you previously provided your qualifications with respect to pre-filed testimony in support of this contention?

A. 7: Yes. Please refer to my testimony on Soils Characterization and my curriculum vitae included as State's Exh. 92. In that testimony and also in my testimony on Dynamic Analysis, I discuss my involvement in assisting the State in the PFS proceeding. Especially relevant to this testimony is my professional experience at the Savannah River Site ("SRS"), in which I applied DOE-STD-1020 to seismic performance of DOE Category 3 and Category 4 nuclear facilities. While at SRS, I was part of a multi-disciplinary team responsible for the seismic qualification and upgrade of several facilities, which included: In-Tank Precipitation Facility (ITP), H-Tank Farm (High Level Waste Tank Farm), and the Defense Waste Processing Facility (DWPF) High Level Waste Vitrification Building. The goal of these qualifications was an assessment of each facility to see if it met the seismic performance goals given in DOE Standard 1020. I primarily oversaw the geotechnical assessment and calculations for the foundations of these structures.

Q. 8: Dr. Ostadan, have you previously provided your qualifications with respect to pre-filed testimony in support of this contention?

A. 8. Yes. Please refer to testimony I filed on Dynamic Analysis and my curriculum vitae included as State's Exh. 110. I have also applied DOE-STD-1020 standards and guidance to the foundations of nuclear structures, including at the Savannah River Site where I joined Dr. Bartlett on a multi-disciplinary team.

III. The 2,000-Year Design Basis Earthquake is Inconsistent with Other Design and Construction Standards.

Q. 9: Is the requested standard of a 2,000-year design basis earthquake consistent with nuclear facility design standards established by non-NRC agencies or entities?

A. 9: (SFB, FO) No. The U. S. Department of Energy ("DOE") published DOE Standard 1020, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities* ("DOE-STD-1020") in which DOE establishes design standards and guidance for nuclear facilities. In August 2001, DOE released draft DOE-STD-1020-2001, which requires a 2,500-year return period ground motion for performance category 3 ("PC3") SSCs. *See* State's Exh. 126, DOE-STD-1020-01, at C-6.

(FO) The current seismic hazard maps, such as those published by U.S. Geological Survey and the National Earthquake Hazard Reduction Program (NEHRP), have also adopted a 2,500-year motion for design use. Also, the most recent design codes such as those adopted or considered by the International Building Code and American Association of State Highway and Transportation Officials, require a 2,500-year motion for design. Based on the direction of other prominent agencies and organizations in the field of seismic design, it is my opinion that DOE will require a 2,500-year ground motion standard in the final DOE-STD-1020-01.

(SFB) The Utah Department of Transportation currently requires all interstate highway bridges to be designed to levels of strong ground motion that exceed the proposed design basis ground motion at the PFS site. The design basis ground motions are based on a uniform hazard spectrum with spectral values that have a 2 percent probability of exceedance in 50 years. This is equivalent to an average return period of 2,500-years.

(SFB, FO) In our opinion, PFS's reliance on a 2,000-year DBE is not consistent with safety and engineering standards established for DOE nuclear facilities, or even the general standard for buildings and highways.

IV. Factors Affecting Selection of Design Earthquake.

Q. 10: Does the selection of the design basis earthquake relate at all to the performance of specific SSCs?

A. 10: (SFB, FO) Yes. When selecting a design basis earthquake, one must consider the critical nature of the facility, its intended performance during the earthquake and any applicable codes and standards. For example, DOE-STD-1020 applies a graded approach where seismic performance goals are set according to the type of facility. DOE-STD-1021 gives the methods for classifying the facility into specific performance categories. There are five possible performance categories, PC 0 through PC 4, with PC 4 being the highest category and is reserved for the most critical or sensitive facilities.

For each performance category, seismic performance goals are defined in terms of a permissible annual probability of unacceptable performance P_F (*e.g.* a permissible failure frequency limit). DOE-STD-1020 requires that seismically induced unacceptable performance should have an annual probability less than or approximately equal to these goals. Thus, to meet the requirements of DOE-STD-1020, one must ultimately demonstrate that the facility can meet the seismic performance goal.

The DBE used to evaluate a structure for a given performance category is also set by DOE-STD-1020-93. The DBE is defined at specified seismic hazard exceedance probability P_H and the SSC is designed or evaluated for the prescribed DBE using adequately conservative deterministic acceptance criteria. To be adequately conservative, the

acceptance criteria must introduce an additional reduction in the risk of unacceptable performance below the annual risk of exceeding the DBE. This is known as a risk reduction ratio or risk reduction factor.

Q. 11: What is a risk reduction ratio?

A. 11: (SFB) A risk reduction ratio is a measure of the conservatism incorporated into the design of an SSC. DOE-STD-1020 requires that the risk reduction ratio must be sufficiently large to show that the target performance goals are achieved. The risk reduction ratio, R_R, in terms of probability is formally defined as:

$$R_R = \frac{P_H}{P_F}$$

where P_H is the seismic hazard exceedance probability and P_F is permissible annual probability of unacceptable performance. DOE requires minimum risk reduction ratios of 5 and 10 for PC3 and PC4 SSCs, respectively. DOE-1020-94, Table C-3 at C-5.

Q. 12: Please explain PFS's various estimations of ground motions?

A. 12: (SFB) At the time PFS requested its exemption, PFS estimated the 84th percentile peak ground accelerations at the site were 0.72 g in the horizontal direction and 0.80 g in the vertical direction. In 2001, PFS's revised 84th percentile peak ground acceleration shows 1.15 g in the horizontal direction and 1.17 g in the vertical direction. *See* Geomatrix, *Update of Deterministic Ground Motion Assessment*, Rev. 1, April 2001 at 3. The 2001 revised peak ground accelerations for a 2,000-year return period are now 0.711 g in the horizontal direction and 0.695 g in the vertical direction. SAR at 2.6-107, Rev. 22.

Q. 13: What effect does the design basis earthquake seismic exemption request have on the performance and evaluation of PFS's design?

A. 13: (SFB) Now that the NRC Staff has consented to the seismic exemption request filed by PFS, this constitutes a substantial reduction in the seismic demand used by the design standard. By using a less severe 2,000-year DBE, instead of using a deterministic DBE (maximum credible earthquake) or a 10,000-year DBE, PFS has apparently adopted the design philosophy contained in DOE-STD-1020. Inherent in demonstrating acceptable performance by this standard is the consideration of the conservatisms in the design and if the appropriate risk reduction ratio has been achieved by the design for the DBE.

Q. 14: How does the unconventional nature of PFS's design and PFS's failure to follow all applicable guidance of DOE-STD-1020 relate to the selection of a design basis earthquake?

A. 14: (SFB, FO) There is no precedent or direct experience upon which PFS can rely to support its unconventional design using cement treated soil to support the CTB or unanchored storage casks subjected to controlled sliding under high levels of seismic loading.

Furthermore, the seismic analysis becomes more critical now that the design margins or conservatism are substantially reduced. The facility is designed to the 2,000-year DBE ground motion of 0.711 g in the horizontal direction and 0.695 g in the vertical direction. These ground motions are significantly less than the 84th percentile peak ground acceleration of 1.15 g in the horizontal direction and 1.17 g in the vertical direction. Also, it is difficult to determine the seismic performance of SSCs without fragility curves because PFS's design features (*e.g.*, unanchored casks supported by cement treated soil in a high seismic area) are unique and there is no existing data on how the SSCs will perform.

Q. 15: Briefly describe how PFS supports the notion that a 2,000 year design earthquake is adequate in this case?

A. 15: (SFB) A central theory in PFS's justification for the use of 2,000-year motion is PFS's analogy to the performance goals of SSCs and risk reduction ratios in DOE-STD-1020 and NUREG/CR-6728, *Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-consistent Ground Motion Spectra Guidelines* (October 2001), and reliance on claimed conservatism built into NRC review plans. Applicant's Motion for Summary Disposition of Part B of Contention Utah L (November 2001) ("PFS SD Motion"), Declaration of Dr. C. Allin Cornell[1] at ¶¶ 20-25. PFS generally surmises that a 2,000-year DBE is warranted because "[t]ypical SSCs in nuclear facilities, such as the PFSF, that are designed to satisfy the US NRC Standard Review Plan structural and mechanical criteria have been found to have a mean component failure return period 5 to 20 times or more greater than the mean return period of the design-basis ground motion" and that the "storage-casks and safety-related structures" could withstand "the loadings resulting from an even more severe earthquake without failure." Applicant's Objections and Response to State of Utah's Eleventh Set of Discovery Requests Directed to the Applicant dated October 2, 2001 at 15. PFS's response to Utah's discovery in October 2001 is the first mention that PFS has made to its theory relating performance goals of SSCs to risk reduction ratios in DOE Standard 1020. *See* id.

---

[1] Excerpts from Declaration of Dr. C. Allin Cornell (Nov. 9, 2001) included as State's Exhibit 129.

The only other apparent justification is PFS's incorporation by reference and adoption of the bases asserted by the NRC Staff in its Safety Evaluation Report ("SER") issued September 29, 2000. Applicant's Response to Eleventh Set at 13. One of NRC Staff's five bases to justify the 2,000-year DBE is DOE-STD-1020-94 which established a 2,000-year DBE. SER (September 29, 2000) at 2-42. The NRC Staff's consent to a 2,000-year DBE in the SER did not consider the substantially greater PFS site ground motions determined in April 2001. In December 2001, the Staff issued a Supplemental SER (December 2001) in which it retained the DOE justification. *See* Supplemental SER at 2-51; Consolidated SER dated March 2002, at 2-51.

I am unaware of any other justification. In PFS's original request for an exemption from the regulations to allow a 1,000-year DBE, PFS principally relied upon the Staff's proposed rulemaking plan, SECY-98-126. PFS Exemption Request at 4-5. *See Request for Exemption to 10 CFR 72.102(f)(1) Seismic Design Requirement* dated April 2, 1999. PFS later revised its seismic analyses from considering a 1,000-year to a 2,000-year DBE, apparently in response to a request by Staff that PFS "should consider" a 2,000-year DBE. *See* PFS Commitment Resolution Letter # 14 dated August 6, 1999.

Q. 16: In this case, is DOE Standard 1020 an appropriate standard to use in the selection of a safe design earthquake?

A. 16: (SFB, FO) Yes. DOE-STD-1020 establishes design and engineering standards for nuclear facilities, including dry spent fuel storage facilities. DOE-STD-1020 and its companion documents have a carefully proscribed methodology to safely design nuclear facilities. Moreover, prior to adoption, DOE-STD-1020 was subject to extensive peer review from an array of technical experts such as seismologist, geotechnical experts, engineers, and risk experts. Thus, DOE-STD-1020 would provide appropriate guidance. The important point is: all applicable design and analysis aspects established in DOE-STD-1020 must be considered together. It is highly inappropriate to refer to a design basis earthquake without considering the probability of failure of the SSCs and the appropriate risk reduction ratio. The way in which PFS selectively relies on some aspects of DOE-STD-1020 and ignores other aspects does not constitute a rational approach.

Q. 17: Please describe how the design earthquake, and probability of failure of SSCs, and the risk reduction ratio are intertwined in the design and analysis philosophy encompassed in DOE Standard 1020.

A.17: (SFB) DOE-STD-1020 first requires that the SSC be categorized according to DOE-STD-1021, and performance goals are established based on the hazard classification. DOE-STD-1020 gives the design and evaluation criteria that control the level of conservatism introduced in the design/evaluation process. These criteria ensure that the level of conservatism and rigor in the design/ evaluation process is appropriate for the category of the facility. DOE-STD-1020 requires the selection of a target performance goal

for the SSC and sufficient evaluations that document the SSC will indeed meet the performance goal for the DBE. The performance goals used in DOE-STD-1020 are probabilistic thresholds, where the probability of unacceptable performance or failure of an SSC is expressed in terms of a mean annual probability of exceedance. Unacceptable performance is considered to be damage to the SCC beyond which hazardous material confinement and safety-related functions are impaired. Design considerations for these categories are to limit SSC damage so that hazardous materials can be controlled and confined, occupants are protected, and functioning of the SSC is not interrupted. Thus, the selection of the DBE ground motion is explicitly coupled with a thorough evaluation of the fragility of or damage to the SSC.

In selecting the performance goals for an SSC, DOE-STD-1020 adopted a graded approach for SSCs. Based on this approach, the performance goal of the SSC is selected. For PC3 SSCs, the performance goal is 10^{-4}. The key for this selection is the fragility curve for the SSCs. By evaluating the fragility curve for the SSCs and recognizing the detail design and ductility of the SSC under earthquake loading and using data from other experiences, a risk reduction factor of 4 has been adopted for PC3 SSCs. Therefore, to meet the performance goal of 10^{-4}, the DOE-STD-1020-2001 recommends a 2,500-year return earthquake for PC3 SSCs.

A probabilistic method to determine if a performance goal has been met for a particular SSC is to develop a fragility curve for each SSC. A fragility curve expresses the expected damage or unacceptable performance of an SSC as a function of the amplitude of strong ground motion. Once a fragility curve has been established for a particular SSC, the probability of unacceptable performance can be calculated for all levels of strong ground motion, even for levels beyond those incurred by the DBE.

The determination of fragility as expressed as a fragility curve allows the assessment of the conservatism of the design for multiple levels of ground motion. The calculation and application of a fragility curve are necessary to determine if an SSC has met a desired performance goal for all levels of strong ground motion. A fragility curve in combination with the seismic hazard curve yields the probability of failure of the SSC and this probability is compared with the probabilistic target performance goal for the SSC to determine if the performance is adequate.

DOE-STD-1020 also discusses the use of risk reduction ratios based on deterministic criteria to determine if the SSC performance goal has been met. Sometimes SSCs are evaluated according to deterministic methods, which are found in applicable codes and standards. When deterministic criteria are used, the basic principle embedded in DOE-STD-1020 is to ensure that the target performance goals are met when the minimum ten percent probability of failure corresponds to 1.5 times the seismic scale factor times the DBE.

Q. 18: To determine the appropriate design earthquake, what primary SSCs at the PFS facility must undergo an adequate seismic analysis?

A. 18: (SFB, FO) The SSCs of concern for seismic analysis at the proposed PFS facility, are the CTB and certain components therein, the storage pads, and the HI-STORM 100 cask system. In its request for the seismic exemption, PFS has not discussed the fragility and seismic performance of the foundation of the CTB and the foundation of the storage pads. This is a glaring omission. For example, an evaluation of whether the crane in the CTB will perform under seismic loads is pointless if the CTB foundation fails under those seismic loads.

Our individual opinion is that PFS still has not adequately addressed all necessary factors in determining the seismic performance of the SSCs.

V. PFS Fails to Demonstrate the Seismic Performance of the SSCs Are Adequate to Accommodate a 2,000-year Design Basis Earthquake.

Q. 19: In your opinion, has PFS demonstrated that the probability of failure of SSCs are appropriate for PFS's desired design basis earthquake?

A. 19: (SFB, FO) No. In accordance with DOE-STD-1020, the design and evaluation criteria for a critical facility, such as an ISFSI, must consider the level of conservatism or lack of conservatism introduced in the design/evaluation process by the DBE. Such an evaluation must be based on the performance of the facility under the proposed earthquake loading. For the reasons previously discussed, in our opinion, PFS's choice of a DBE cannot be segregated from the critical issues throughout the unified contention, including sections C and D. The assumptions underlying the design of the PFS facility and quantitative analyses thereof are central to whether there is conservatism in PFS's design. In our opinion, PFS's attempt to justify a 2,000-year DBE by claiming conservatism in its design cannot be judged without evaluating these claims against the unconservatism of PFS's design, such as the use of unrealistic assumptions, omissions and gross generalizations to show that certain SSCs at PFS will adequately perform given a 2,000-year DBE.

Q. 20: Is PFS's seismic design and analysis conservative?

A. 20: (SFB) No. The PFS design and analysis are not conservative. It is unprecedented to design unanchored dry storage casks for a seismically active area with such intense strong ground motions similar to those at the PFS facility. PFS's claim that the casks will only slide in a "controlled" manner atop the pads contradicts general engineering principles. The lack of conservatism in its analysis is further compounded when PFS uses its claim of "controlled" cask sliding to reduce the seismic loadings to the pad foundations.

(SFB, FO) PFS failed to demonstrate that adequate conservatism has been applied

in the seismic design of foundations for the storage pads and CTB and to the seismic stability of the pads and HI-STORM 100 storage casks for the proposed DE. As we detailed in our Dynamic Analysis testimony, there are numerous unconservative assumptions, oversights in PFS's design calculations. The lack of conservatism in the design and the inadequacy of the seismic analysis are important in determining the appropriate DBE. Rather than duplicate our opinions here, we refer the Licensing Board to our Dynamic Analysis testimony and to the Cask Stability testimony, which are being filed concurrently.

Q. 21: Please restate the purpose of a fragility curve and whether PFS has developed any?

A. 21: (SFB, FO) A fragility curve expressed the expected damage or unacceptable performance of an SSC as a function of the amplitude of strong ground motion. PFS has not produced any fragility curve for the casks, the storage pads, or the CTB foundation.

(SFB) In addition, PFS has not developed fragility curves for the HI-STORM 100 cask system relating to excessive movement and collision of the casks, tipover of the casks, excessive uplift and separation of the casks from the pad, or the consequence of such unstable cask and pad conditions. PFS's DBE witness, Dr. Cornell, had no knowledge of any fragility curves for the HI-STORM 100 cask system, the storage pad, or the CTB at the PFS facility. State's Exhibit 130,Cornell Tr. at 49. In fact, PFS's witness responsible for the seismic stability evaluations of the storage casks was unfamiliar with a fragility curve or its purpose. *See* State's Exhibit 131, Singh/Soler 2001 Tr. at 63.

Q. 22: Has PFS used SSC specific analysis other than a fragility curve to demonstrate performance goals have been satifisfied?

A. 22: (STB, FO) No. PFS has not demonstrated that the storage pad and CTB foundation meet the performance goals required in DOE-STD-1020. PFS has failed to show that the SSCs can meet a target performance goal of 1×10^{-4} for the associated 2,000-year annual return period under DOE-STD-1020-94.

Q. 23: Is it possible to select the DBE without evaluating the probability of seismic failure of each SSC at the PFS facility?

A. 23: (STB) No. As we testified, a DBE is meaningless when selected without considering the probability of seismic failure and applicable risk reduction ratios.

Q. 24: What is a "failure" of an SSC?

A. 24: (SFB) We agree with PFS's definition of a failure "as exceeding a behavior limit state that may preclude the SSC from fulfilling its intended function." State's Exhibit

129, Cornell Dec. at 14. Based on this definition, a reduction of a storage cask's ability to shield radiation, thereby causing an increase in dosage, would be a failure of the HI-STORM 100 cask. Dr. Marvin Resnikoff calculated an increase in radiation dose in the event of cask tipover. *See* Resnikoff Testimony at A. 23. In addition, Dr. Mohsin Khan and Dr. Ostadan concluded that the Holtec seismic analysis is not conservative and the results are inconclusive without analysis, test data, and other validation. *See* Khan and Ostadan Cask Stability Testimony at Answers 26-36, 38. Dr. Khan also determined that the HI-STORM 100 may in fact tipover when subject to 2,000-year DBE at the PFS site. These issues are detailed in the Joint Testimony of Dr. Mohsin Khan and Dr. Farhang Ostadan with Respect to Contention Utah L/QQ - Cask Stability. Again to eliminate duplication, we refer the Licensing Board to that testimony here which demonstrates that PFS has failed to demonstrate that the HI-STORM 100 cask will not tip over when subject to a 2,000-year DBE.

Q. 25: Does DOE-STD-1020 address acceptance performance criteria for foundations?

A. 25: (SFB) DOE-STD-1020 recognizes that specific acceptance criteria for foundations have not been developed. It states that the intent of DOE-STD-1020 must still be met for some system components for overturning or sliding of foundations. State's Exhibit 132, DOE-STD-1020-94 at 2-24. This intent is that "there should be less than 10 percent probability of unacceptable performance at input ground motion defined by a scale factor [SF] of 1.5SF times the DE." Id. PFS has not made this calculation nor demonstrated that the intent of DOE-STD-1020 has been met for the foundation systems of the storage pads and CTB.

Q. 26: When analyzing seismic performance, how do you account for nonlinear behavior?

A. 26: (SFB) For soil sites, like the PFS site, because the slope of the hazard curve can be impacted by the soil nonlinear behavior, NUREG/CR-6728 recommends to establish the slope of the hazard curve by including the nonlinear soil effects for determination of the seismic scale factor. This concept is applicable to any nonlinear behavior such as cask sliding on the pads since the response is nonlinear and is effectively based on performance design and cannot be extrapolated from the response at lower level ground motions. PFS has not considered these nonlinear effects, nor has it calculated the seismic scale factor, SF, based on considerations of the slope of the hazard curve.

VI. Performance Goals Are Not Clearly Inherent in ISFSI and Cask Standard Review Plans.

Q. 27: Do you agree with PFS that performance goals are "inherent" in the NRC Standard Review Plan design standards?

A. 27: (SFB) No. In an attempt to demonstrate that performance goals are unnecessary, PFS claims that NRC SRPs have equivalent or greater risk reduction ratios as those stated in DOE-STD-1020-94 for performance category 3 and 4 facilities. State's Exh. 129, Cornell Dec. ¶ 25. Thus, surmises PFS, risk reduction factors of approximately 5 to 20 can then be claimed for the PFS SSCs. Id.

PFS's asserted risk reduction ratios of 5 to 20 for PFS SSCs are unsubstantiated. NRC SRP requirements do not address the seismic performance requirements of unanchored casks supported by shallowly embedded pad foundations which are buttressed by cement-treated soil and subject to high levels of strong ground motion. The proposed PFS design has unique seismic interface and foundation issues and must be analyzed accordingly.

PFS itself only claims that the SRPs for nuclear power plants ("NPP") are equivalent or greater than DOE-STD-1020 design criteria. State's Exh. 129, Cornell Dec. ¶ 25. The HI-STORM 100 cask system is not designed to SRPs governing NPPs but to NUREG-1536, Standard Review Plan for Dry Cask Storage Systems. The CTB must be designed according to NUREG-1567, Standard Review Plan for Spent Fuel Dry Storage Facilities. PFS has not shown that the SRPs for dry cask storage systems and ISFSIs provide an equivalent or greater level of conservatism than that claimed for the NPP SRPs.

NRC Staff and PFS claim that the potential consequences of seismic failure of ISFSIs are much less severe than those of NPPs. *See, e.g.* State's Exh. 129, Cornell Dec. ¶ 16. PFS and the Staff further claim that ISFSI facilities are less vulnerable to earthquake-initiated accidents than NPP. *See* Id. ¶ 17. Thus, the SRPs in NUREG 1536 and 1567 may already incorporate less conservatism than NPP SRPs. Additionally, the dry cask storage system SRP design standards are based on the assumption that the design earthquake is equivalent to the safe shutdown or deterministic earthquake used for nuclear facilities, under 10 CFR Part 50. NUREG 1536 at 2-10, NUREG-1567 at 7-20, 7-54. In sum, SRPs for dry storage cask system and ISFSI may already incorporate less design conservatism than NPP SRPs. It is not good engineering practice to rely on presumed conservatism or risk reduction ratios to account for unanalyzed conditions and to assume, without any attempt to validate, that design criteria set for ISFSIs and casks will be encompassed by those standards developed for NPPs. This type of process is particularly troubling in this specific case given the substantially lower standard of a 2,000-year DBE and the unconventional plan to store unanchored casks in a highly seismic area supported by cement treated soil.

NUREG 1536 requires the applicant to demonstrate that the dry cask system will not tipover or drop as a result of a credible natural phenomenon event, such as an earthquake. NUREG 1536 at 3-6. As discussed in detail in the Joint Testimony of Dr. Mohsin Khan and Dr. Farhang Ostadan Regarding Contention Utah L/QQ, Part D - (Cask Stability), the HI-STORM 100 cask may tipover if subject to the ground accelerations for a

2,000-year earthquake. Thus, even if the SRPs for NUREG-1536 result in design criteria that are equal or more conservative than posed in DOE-STD-1020, PFS has not shown that the HI-STORM 100 cask system even meets the NUREG-1536 SRPs under the ground motions for a 2,000-year DBE at the PFS site.

Q. 28: Are you familiar with Dr. Cornell's statement supporting PFS's Motion for Summary Disposition that Chapter 7 of the recently released NUREG/CR-6728, generally supports that NRC standard review plans provide equal or greater levels of conservatism than DOE-STD-1020.

A. 28: (SFO) Yes.

Q. 29: Do the fragility curves presented in NUREG/CR-6728 include an analysis of unanchored casks in a high seismic area with equivalent or greater ground motions than the 2,000-year DBE at the PFS site?

A. 29: (SFO) No. PFS witness, Dr. Cornell, claims that NUREG/CR-6728, *Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard and Risk-consistent Ground Motion Spectra Guidelines* (October 2001), provides a "quantitative finding that the [risk reduction ratio] levels for typical systems, structures, and components designed to NRC SRPs are in the range 5 to 20 or greater" (or in the range of the DOE-STD-1020-94 risk reduction ratios). *See* State's Exh. 129, Cornell Dec. at ¶ 25. To support his claim, Dr. Cornell compares the risk reduction factors for NPP SSCs using both the NRC SRPs and DOE-STD-1020-94. *See* in general id., Attachment A, State's Exh. 129. However, in Attachment A, Dr. Cornell relies upon "numerous engineering evaluations of safety margins and 'fragility curves' of SSCs." Id. at 3. NUREG/CR-6728, Chapter 7 contains fragility curves for a variety of NPP sites. *See* NUREG/CR-6728 at 7-10 to 7-15. The fragility curves used in NUREG/CR-6728 are obtained from *Basis for Seismic Provisions of DOE-STD-1020*, R.P Kennedy and S.A. Short (1994). NUREG/CR-6728 at 7-5. It is important to note that the only site with similar peak ground accelerations to the PFS site is the California site located near Santa Maria, *i.e.*, Diablo Canyon. Id. at 7-11, 7-22. In 1994, when Kennedy and Short published the fragility curves, Diablo Canyon did not have any dry storage casks, let alone unanchored dry storage casks. *See* State's Exhibit 133, portions of the letter accompanying the Diablo Canyon Independent Spent Fuel Storage Installation License Application dated December 21, 2001. The Kennedy and Short fragility curves relied upon in NUREG/CR-6728 could not have included unanchored dry storage casks, and Dr. Cornell's attempt to correlate NUREG/CR-6728 to DOE-STD-1020-94 risk reduction ratios in his Declaration, Attachment A, fails with respect to HI-STORM 100 casks at the PFS facility.

In general, the Kennedy and Short fragility curves do not apply to SSCs such as storage casks sliding on the pads to maintain stability and control for excessive movement and tipping. The fragility curve pertains to inherent strength and ductility of the member

and the design code upon which the component was designed. The fragility curve as it pertains to controlled and stable movement of the casks on the pads has not been developed by PFS, nor any appropriate design code.

In our opinion, it is inappropriate to apply generalized risk reduction ratios deemed appropriate for NPPs to the proposed storage pad, unanchored HI-STORM 100 cask, and the CTB. The basis for selecting appropriate risk reduction factors can only adequately be conducted by evaluating a thorough uncertainty analysis of the fragility of each SSC at the PFS site, as outlined in DOE-STD-1020.

Q. 30: Please summarize your opinion.

A. 30: (SFB, FO) In summary, PFS has not met the intent and requirements of DOE-STD-1020. It is impossible to assess the fragility for the storage pads, storage casks, and the CTB and their foundations because of many errors, omissions and unconservative assumptions in PFS's evaluations. PFS has not demonstrated that the performance goal for the PFS facility has been met. Without this demonstration, the selection of the proposed 2,000-year DBE is not founded on a proper technical basis and is basically arbitrary.

November 9, 2001

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	)	
	)	
PRIVATE FUEL STORAGE L.L.C.	)	Docket No. 72-22
	)	
(Private Fuel Storage Facility)	)	ASLBP No. 97-732-02-ISFSI

DECLARATION OF C. ALLIN CORNELL

C. Allin Cornell states as follows under penalty of perjury:

I. WITNESS CREDENTIALS AND SCOPE OF TESTIMONY

1. I am currently a professor (research) at Stanford University in Stanford, California and an independent engineering consultant. In the former capacity I perform research and supervise a Senior Research Associate and several Ph.D.-level graduate students in the areas of probabilistic analysis of structural engineering and earthquake engineering. As a consultant, I assist engineering and earth sciences firms, industrial concerns, and government agencies in developing and applying methodologies and standards for probabilistic seismic hazard analysis, engineering safety assessments, natural hazards analyses, and earthquake engineering. Through my teaching, research and consulting activities (described below) I have developed an expertise in earthquake engineering, probabilistic engineering analysis of seismic loads on structures and structural responses to such loads, and the development of structural design guidelines and codes. I am providing this declaration in support of Applicant's Motion for Summary Disposition of Part B of Contention Utah L in the above captioned proceeding concerning the Private Fuel Storage Facility ("PFSF").

State's
Exhibit 129

seismic standards that explicitly use this principle include the draft International Standards Organization ("ISO") guidelines for offshore structures [Ref. 13 (Banon et al., OMAE 2001 on ISO)], Federal Emergency Management Agency ("FEMA") guidelines for building assessment [Ref. 14 (FEMA 273 pp. 2-5)], and DOE Standard 1020 [Ref. 11 (Table C-3, p. C-5)]. Further, the NRC staff has stated, with respect to the seismic design of nuclear facilities: "The use of probabilistic techniques and a risk-graded approach are compatible with the direction provided by the Commission on Direction Setting 12, 'Risk-Informed, Performance-Based Regulation.'" [Ref. 15 (SECY-98-071 pp. 3-4)].

16. Under the risk-graded approach to the seismic design, ISFSIs such as the PFSF, can be assigned a higher probability of failure than a nuclear power plant because the potential consequences of seismic failure of ISFSIs are much less severe than those for nuclear power plants. The radioactive inventory that potentially could be released to the environment from an ISFSI is less because the spent fuel has decayed significantly and because a spent fuel canister is under much lower pressures than a reactor's coolant boundary; higher pressures will disperse any released radioactivity farther from the source. The NRC has rejected the notion that licensing standards should be as high for ISFSIs as for nuclear power plants, noting that "[t]he potential ability of irradiated fuel to adversely affect the public health and safety and the environment is largely determined by the presence of a driving force behind dispersion. Therefore, it is the absence of such a driving force, due to the absence of high temperature and pressure conditions at an ISFSI (unlike a nuclear reactor operating under such conditions that could provide a driving force), that substantially eliminate the likelihood of accidents involving a major release of radioactivity from spent fuel stored in an ISFSI." [Ref. 16 (60 Fed. Reg. 20,883 (1995))].

17. Further, an ISFSI facility as a whole is inherently less vulnerable to earthquake-initiated accidents than a nuclear power plant. An ISFSI is largely passive; it does not have active cooling and safe-shutdown systems necessary for maintaining the integrity of the high-pressure reactor coolant boundary and for shutting down after a large earthquake, as does a nuclear power plant. The NRC has recognized the reduced seismic

vulnerability of an ISFSI by stating that for ISFSIs, such as dry storage casks, which do not involve massive storage structures, "the required design earthquake will be determined on a case-by-case basis until more experience is gained with licensing these types of units." [Ref. 17 (45 Fed. Reg. 74,697 (1980), as cited in Ref. 15 (SECY-98-071 p. 2).]

B. Factors Determining Failure Probability for Facilities and Structures

18. While the risk-graded approach is implemented in somewhat different ways in the various fields of seismic design, the standards of practice almost invariably utilize a DBE defined at some mean annual probability of exceedance and a set of design procedures and acceptance criteria. Both the procedures and the acceptance criteria include conservatisms that, implicitly or explicitly, are intended to implement "performance goals" (e.g., target levels of the seismic failure probability for the facility or structure), which are defined in a manner reflecting the anticipated consequences of the failure. These conservatisms are typically embedded in the various codes and standards pursuant to which the design of a structure or facility is accomplished.

19. Both the MAPE of the DBE and the level of conservatism incorporated in the design procedures and criteria affect the failure probability of seismically-designed facilities and structures. A lower (or higher) failure probability can be achieved by keeping the design procedures and criteria fixed while reducing (or increasing) the MAPE of the DBE; or, alternatively, by fixing the MAPE while making the design procedures more or less conservative; or by adjusting both elements simultaneously. Whichever choice is made among these alternatives, it is important to understand that both the MAPE and the level of conservatism in the design procedures and criteria must be considered when assessing and comparing the safety implications of various seismic design standards. One fact remains true, however: because of the conservatisms incorporated in all seismic design procedures and criteria, the probability of failure of a seismically-designed facility or structure is virtually always less than the MAPE of the DBE. In other words, virtually facilities and structures designed against a given DBE have a mean return period to failure that is longer than the mean return period of the

earthquake for which they are designed. In practical terms, this means that seismically-designed facilities and structures are able to withstand a more severe, i.e., more infrequent, earthquake than that used as the DBE.

20. The application of these principles of risk-graded seismic design is perhaps most clearly and explicitly seen in the U.S. Department of Energy's Standard 1020. The basis for DOE Standard 1020 is a set of "performance categories" (1 to 4) for seismically designed facilities and structures with increasing consequences of failure, and thus decreasing probabilities of failure as their performance goals [Ref. 11 (DOE 1020, p. 1-2 and p. C-2)]. DOE is responsible for (1) facilities such as ordinary buildings (Performance Category 1 or PC1) designed to protect occupant safety, (2) essential facilities and buildings that should continue functioning after an earthquake with minimal interruption (PC 2), (3) important facilities such as ISFSIs that contain hazardous materials (PC3), and (4) critical facilities such as those involving nuclear reactors (PC4).

21. The performance goals for DOE structures, systems and components in the four performance categories PC1 to PC4 are set as mean annual failure probabilities of 10^{-3}, $5x10^{-4}$, 10^{-4}, and 10^{-5}, respectively [Ref. 11 (DOE 1020, p. C-5)] reflecting the increasing consequences of failure. On the other hand, MAPEs for the design basis ground motions are set as $2x10^{-3}$, 10^{-3}, $5x10^{-4}$, and 10^{-4}, respectively. These values are uniformly larger than the performance goals.

22. To bridge the gap between the performance goals and the DBE MAPEs, the DOE 1020 standards call for design procedures and evaluation criteria that vary among the categories, ranging from those "corresponding closely to model building codes" for PC1 and PC2 , to those for PC4 which "approach the provisions for commercial nuclear power plants" [Ref. 11 (DOE 1020, p. 2-2, C-4 to C-5)]. The quantitative effect, in terms of reducing earthquake risk, of applying the conservatisms built into these various design procedures and criteria is reflected in the ratios between the MAPE of the design basis ground motions and the corresponding performance goal probabilities. These ratios are 2, 2, 5 and 10, respectively [Ref. 11 (DOE 1020, p. C-5)]. The ratios are called "Risk Reduction Ratios", R_R, in DOE 1020. The following table

summarizes these three parameters, the DBE MAPE, the Performance Goal, and the R_R for the four performance categories PC1 through PC4 in DOE 1020:

Table 1: DOE Std. 1020-94 Seismic Performance Goals, DBE MAPEs and R_Rs[5]

Performance Category	**Target Seismic Performance Goal (P_F)**	**DBE Exceedance Probability (MAPE)**	**Risk Reduction Ratio (R_R)**
PC1 (e.g., office building)	$1x10^{-3}$	$2x10^{-3}$	2
PC2 (e.g., essential building that should remain operational, such as hospital or police station)	$5x10^{-4}$	$1x10^{-3}$	2
PC3 (e.g., hazardous waste facilities such as ISFSIs)	$1x10^{-4}$	$5x10^{-4}$ (except $1x10^{-3}$ for Western sites near tectonic boundaries)	5 (except 10 for Western sites near tectonic boundaries)
PC4 (e.g., nuclear reactor facility)	$1x10^{-5}$	$1x10^{-4}$ (except $2x10^{-4}$ for Western sites near tectonic boundaries)	10 (except 20 for Western sites near tectonic boundaries)

[5] A revised draft version of DOE Standard 1020 was released in August of this year for comment [Ref. 18 (DOE-1020-2001)]. The primary change is that PC1 and PC2 will be based on the IBC 2000 instead of the UBC model building code. As a result, this table would differ under the proposed standard in that the MAPE of PC1 and PC2 categories would change to $4x10^{-4}$. To be consistent, the MAPE of PC3 is modified slightly to the $4x10^{-4}$ value. The performance goals remain the same in all categories. The R_R for PC3 would therefore be changed from 5 to 4, although no change would be made to the design procedures and criteria for PC3. The R_R column is left blank for PC1 and PC2, but it can be shown that the R_R is still about 2, using the information in NERHP Recommended Provisions for Seismic Regulations for New Buildings and Other structures [Ref. 19 (FEMA-303, at p. 37)] and the procedures outlined in Attachment A hereto. These proposed revisions to DOE 1020, if adopted, would not in any way alter the analyses and conclusions in this Declaration.

23. The actual value of R_R obtained from the design conservatisms for a given SSC is dependent on the shape or slope of the ground motion hazard curve. For example, the PC4 value of 10 cited in the table is representative of locations in the Central and Eastern United States. However, higher risk reduction ratios, e.g., 20 for PC4 facilities, are achieved in western US sites near tectonic boundaries, where hazard curves are steeper [Ref. 11 (DOE 1020, Table 2-1 p. 2-4)]. The higher achievable R_R values have allowed the DOE to specify that higher DBE MAPE levels can be used for PC4 facilities as well as for PC3 facilities in these regions.

24. In DOE 1020, the overall conservatism levels are controlled through acceptance criteria to achieve specific R_R levels [Ref. 11 (DOE 1020, pg. 1-5)]. The document states: "These design and evaluation criteria have been developed such that the target performance goals of the [Natural Phenomenom Hazard] Implementation Guide [set forth in Table 1 above] are achieved" [Ref. 11 (DOE 1020, p. 2-1)]. In other words, the risk reduction levels in DOE 1020 are achieved through use of the DOE design and evaluation criteria specified in Chapter 2 of DOE Standard 1020 and related appendices.[6] For PC4 facilities the risk reduction factor achieved is 10 in most regions.

25. The design guidelines provided by the NRC SRPs also contain many conservatisms that result in risk reduction factors as large as, or larger than, those for PC4 category facilities designed to DOE 1020. NRC SRP standards share with DOE's PC3 and PC4 categories many procedures leading to design conservatism [Ref. 11 (DOE 1020, pp. C-5, C-6)]. These conservatisms are introduced through prescribed analysis

[6] The State's witness has suggested that the risk reduction ratio does not measure the conservatism in a DOE PC category's design procedures and criteria, but rather that it is simply defined as the ratio of the DBE MAPE to the Performance Goal, and hence it is only the ratio required to achieve the goal. Although one might arguably draw that conclusion from the statement in DOE 1020 that the "required degree of conservatism in the deterministic acceptance criteria is a function of the specified risk reduction ratio," [Ref. 11 (DOE 1020, p. C-5)], the quote in the body of the text clearly confirms that, upon selecting the required ratio DOE then established the prerequisite design and evaluation criteria in Chapter 2 of the DOE-1020 to achieve the goals. Therefore, the ratio also becomes a measure of the conservatism provided for by the design and evaluation criteria set forth in Chapter 2 of DOE Standard 1020 and the related appendices.

methods, specification of material strengths, limits on inelastic behavior, etc. The conservatism levels in NRC seismic SRPs are not explicitly keyed to values of R_R. Nonetheless, the risk reduction factors achieved through the use of NRC guidelines for typical SSCs have been found in application to be equal to, or higher than, those called for in DOE 1020 for PC4 facilities, since they are greater than 10 in most regions. DOE 1020 acknowledges the higher R_R levels provided by the NRC SRPs by stating that the "[c]riteria for PC4 approach the provisions for commercial nuclear power plants". [Ref. 11 (DOE 1020, p. 2-2, C-4 to C5). There is recent independent technical support both for the general conclusion that NRC SRPs provide equal or greater levels of conservatism than DOE 1020, and for the quantitative finding that the R_R levels for typical systems, structures, and components designed to NRC SRPs are in the range 5 to 20 or greater [Ref. 20 (NUREG/CR-6728 at Chapter 7)]. [7]

C. Application of General Principles to the PFSF

26. At the PFSF, designing for the 2,000-year MRP DBE ground motion and using the NRC SRPs means that typical important-to-safety systems, structures and components can be expected to have seismic failure probabilities 5 to 20 or more times lower than the DBE MAPE, i.e., 2.5×10^{-5} to 1×10^{-4} or lower (i.e., seismic failure MRPs of 10,000 to 40,000 years or more). Therefore, the PFSF would easily meet the DOE performance objectives of 1×10^{-4} for PC-3 facilities under which ISFSIs, such as the PFSF, would fall. The State's expert witness, Dr. Arabasz, agreed that ISFSIs, such as the PFSF, would appropriately be classified PC-3 facilities under DOE-1020 and that the performance objective of 1×10^{-4} for the PFSF would be an appropriate standard on which to determine the acceptability of its seismic design. Arabasz Dep. at 80-81.

27. Applying a risk-graded seismic approach, a performance objective of 1×10^{-4} for ISFSIs such as the PFSF is consistent with the NRC's performance objectives for operating nuclear plants, which pose higher radiological hazard consequences than

[7]Demonstration of these conclusions requires a somewhat detailed technical discussion, which is presented in Attachment A to this Declaration.

paper)]. This is also the case with respect to the risk acceptance guidelines promulgated by the NRC where the subsidiary performance objectives are the risk metrics Core Damage Frequency (CDF) and Large Early Release Frequency (LERF). [Ref. 5 (Reg. Guide 1.174 at p. 10)] and [Ref. 22 (SECY-00-0077 at p. 6)]. The reasons for focusing on annual risks in making facility safety decisions include the fact that any facility providing a needed service will, at the end of its operating life, most likely be replaced by some other facility used for the same purposes with its own, similar risks. The spent fuel to be stored at the proposed PFSF is currently being stored in or near nuclear power plants, and after leaving the PFSF it will likely be stored at the proposed Yucca Mountain facility.

V. SUMMARY

50. In this Declaration I have explained why the use of probabilistic seismic hazard analysis to establish the design basis ground motions at the PFSF site is consistent with current NRC practice and that in other technical fields. I have showed that the 2000-year mean return period ground motions (i.e., those with mean annual probabilities of exceedance of 5×10^{-4}) together with the NRC SRPs design procedures and acceptance criteria will provide an appropriate level of public safety for the PFS ISFSI. Finally, I have addressed each of the bases asserted by the State in support of Part B of Contention Utah L and established that they do not undercut or controvert my conclusions.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on November 9, 2001.

C. Allin Cornell

ATTACHMENT A

DETERMINATION OF RISK REDUCTION FACTORS FOR SSCs AT FACILITIES DESIGNED USING NRC SEISMIC SRP STANDARDS

The objective of this Attachment is to show the analytical process used to determine quantitatively the degree of conservatism inherent in the design procedures and acceptance criteria found in both DOE Standard 1020 and the NRC SRPs. This level of conservatism is captured in the risk reduction factor or ratio R_R. By calculating the values of R_R resulting from DOE Standard 1020 and the NRC SRPs, the risk reduction factors implicit in the SRP design procedures and criteria can be compared to risk reduction factors expressly provided for in DOE 1020. The precise calculated value of R_R depends on several technical parameters (defined below) whose values may vary from site to site and from SSC to SSC. Accordingly, one can produce only a representative range of R_R values for both the SRP and DOE 1020. (As an example, Figure C-4 on page C-11 of DOE-1020 [Ref. 11] shows the range of R_R values for SSCs designed to the criteria specified for category PC4 SSCs in DOE-1020.)

The risk reduction ratio, R_R, is defined in NUREG/CR-6728 [Ref. 21 pp. 7-9] by the equation:

$$R_R = F_R^{K_H} \left(e^{x_p \beta}\right)^{K_H} e^{-\frac{1}{2}(K_H \beta)^2}$$

A different formulation of this same equation appears also in DOE-1020 at page C-9. In this equation, the variables are as follows:

- K_H, a measure of the slope of the PSHA seismic hazard curve;
- β, a measure of the degree of uncertainty in the response and capacity of SSCs;

- F_R, a measure of the margin (achieved by the procedures and criteria) between the level of the DBE and a reference SSC capacity; and
- x_p, a measure of the margin between this reference capacity and the median value of the SSC capacity.

These variables are defined in more detail in both of the references cited above (DOE 1020 at Appendix C.2 and NUREG/CR-6728 at Section 7.2).

For the purposes of this comparison, I will use for both the SRP and the DOE 1020 R_R determinations a range of values for the hazard curve slope $K_H = 2.1$ to 3.3 (NUREG/CR-6728 at pg. 7-6). These values are representative of the relevant hazard interval (10^{-4} to 10^{-5}) for nuclear power plants at CEUS sites (DOE 1020 at pg. C-8-9, and C-12)[1], and also of the relevant hazard interval (10^{-3} to 10^{-4}) for DOE PC3 (i.e., ISFSI) SSCs at the PFSF site (e.g., the K_H at the PSFF site for peak ground acceleration is 2.8, as determined from [Ref. 28 (Revised Geomatrix Appendix F at Fig. 6-11 __)]. For simplicity, I use here a typical value of $\beta = 0.4$. (The conclusions are quite insensitive to β as shown in DOE 1020 at Figure C-4 on page C-11.) These values for K_H of 2.1 to 3.3 and for β of 0.4 are common to the calculations below of the R_R for both DOE 1020 and the NRC SRP.

First, I consider the DOE 1020 R_R standards. For these standards, the appropriate value of x_p is 1.28 and the appropriate value of F_R is 1.5 SF, both of which appear in DOE 1020 at Eq. C-6, pg. C-9. For PC4 the value of the "scale factor" SF is set at 1.25 (and for PC3 it is set at 1.0) in order to achieve the desired risk reduction ratio R_R [DOE 1020 at pg. 2-13]. Substitution of the above values for K_H, β, x_p, and F_R into the equation for R_R leads to a range of values of R_R from 8

[1] For clarity, if one uses this reference, it needs to be pointed out that the K_H range above corresponds precisely to the A_R range of 2 to 3 that will be found at this citation; A_R is an alternative hazard curve slope measure, DOE 1020, at pg. C-8).

to 17 for DOE 1020 category PC4, as can be seen on Figure C-4 on page C-11 of DOE 1020. The results of these and similar calculations were used in DOE 1020 to confirm the conclusion that the DOE 1020 design procedures and acceptance criteria set forth in Chapter 2 would achieve a value of R_R of about 10, as required to meet the PC4 performance goal. DOE 1020 at p. C-12.

Unlike DOE 1020, the NRC SRPs have not been "tuned" to give a particular R_R (or more precisely a representative value, such as 10 above, applicable to a range of sites). Accordingly, it has been necessary to depend on the numerous engineering evaluations of safety margins and "fragility curves" of SSCs designed to the SRP that have been conducted over the last 20 years in the course of research by the industry and NRC contractors, and on the seismic probabilistic risk assessments and seismic margins studies that have been undertaken at virtually all nuclear power plants in the US (via the NRC IPEEE program). These evaluations have been made by earthquake engineers familiar with nuclear power plant SSC designs prepared to the NRC SRP procedures and criteria, and with the actual behavior of such SSCs in earthquakes as observed in the field and tested in the lab. This experience is summarized in NUREG/CR-6728 at pg. 7-3 by the conclusion: "For nuclear power plant design the factor of safety has typically been 1.25 to 1.5." NUREG/CR-6728 (at pg. 7-4). This "factor of safety" is the variable F_R in the above equation. This factor is, however, coupled with a value of x_p of 2.33. NUREG/CR-6728 (at Ch. 7), which determines the definition of the reference capacity (referred to as a "HCLPF" or C_1) used in engineering evaluations of SRP conservatisms. This value of x_p is much more conservative than that used in DOE-1020.

Using this value of x_p and this range of F_R values one finds (for the same β value and range of K_H values used for the DOE 1020 calculations above) that the R_R for the SRP is in the range 8 to 32. Compared to the range of 8 to 17 calculated for DOE 1020, this result confirms that the

DOE 1020 PC4 standard does indeed only "approach" those of the NRC SRP, as stated in DOE-1020 at page C-5.

If one looks, not at the range of hazard curve slope values of 2.1 to 3.3 used for K_H in the above calculations, but rather at the specific value $K_H = 2.8$ associated with peak horizontal ground acceleration at the PFSF site, the range of NRC SRP R_R values is 12 to 21. For the subset of SSCs sensitive to 1 second spectral accelerations, the ratios range from 8 to 12 based on the reduced slope of the hazard curve for this period. Revised Geomatrix Appendix F at Fig. 6-11.

For simplicity in the body of the declaration and in the [Ref. 29] Applicants Response to the State's Int. 15, Item 9, I have summarized such detailed results in the statement that "the R_R's for typical components SSCs designed to the NRC SRP are in the range 5 to 20 or greater".

REFERENCES

Reference 1: U.S. Nuclear Regulatory Commission, NUREG-1567, *Standard Review Plan for Spent Fuel Dry Storage Facilities*, March 2000.

Reference 2: U.S. Nuclear Regulatory Commission, NUREG-0800, *Standard Review Plan*, August 1988.

Reference 3: 10 Code of Federal Regulations § 100.23.

Reference 4: U.S. Nuclear Regulatory Commission, Regulatory Guide 1.165, *Identification and Characterization of Seismic Sources and Determination of Safe Shutdown Earthquake Ground Motion*, March 1997.

Reference 5: U.S. Nuclear Regulatory Commission, Regulatory Guide 1.174, *An Approach for Using Probabilistic Risk Assessment in Risk-Informed Decisions on Plant-Specific Changes to the Licensing Basis*, July 1998.

Reference 6: U.S. Nuclear Regulatory Commission, Strategic Assessment Issue Paper, Direction Setting Issue 12, *Risk-Informed, Performance-Based Regulation Strategic Assessment*, September 16, 1996.

Reference 7: U.S. Nuclear Regulatory Commission, SECY-01-0178, *Rulemaking Plan: Geological and Seismological Characteristics for Siting and Design of Dry Cask Independent Spent Fuel Storage Installations, 10 CFR Part 72*, September 26, 2001.

Reference 8: Uniform Building Code, Vol. 2, 1997.

Reference 9: International Building Code, 2000.

Reference 10: American Petroleum Institute, API Recommended Practice 2A-WSD (RP 2A-WSD), *Recommended Practice for Planning, Designing and Constructing Fixed Offshore Platforms – Working Stress Design*, twentieth ed., July 1, 1993.

Reference 11: U.S. Department of Energy, DOE-STD-1020-94, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*, January 1996.

Reference 12: Paté-Cornell, M.E., *Quantitative Safety Goals for Risk Management of Industrial Facilities*, Structural Safety Journal 13, 1994.

Reference 13: Banon, H, Cornell, C.A., Crouse, C.B., Marshall, P.W., Nadim, F, and Younan, A. H., *ISO Seismic Design Guidelines for Offshore Platforms*, Proceedings of the 20th Offshore Mechanics and Artic Engineering Conference – OMAE 2001, Rio de Janeiro, Brazil, June 2001.

REFERENCES

Reference 14: Federal Emergency Management Agency (FEMA-273), *NEHRP Guidelines for the Seismic Rehabilitation of Buildings*, October 1997.

Reference 15: U.S. Nuclear Regulatory Commission, SECY-98-071, *Exemption to 10 CFR 72.(f)(1) Seismic Design Requirement for Three Mile Island Unit 2 Independent Spent Fuel Storage Installation*, April 8, 1998.

Reference 16: 60 Federal Register 20,883 (1995).

Reference 17: 45 Federal Register 74,697 (1980).

Reference 18: U.S. Department of Energy, DOE-STD-1020-2001, *Natural Phenomena Hazards Design and Evaluation Criteria for Department of Energy Facilities*, Rev. 1, 2001.

Reference 19: Federal Emergency Management Agency (FEMA-303), *NEHRP Recommended Provisions for Seismic Regulations for New Buildings and Other Structures*, Part 2 – Commentary, 1997 ed., February 1998.

Reference 20: U.S. Nuclear Regulatory Commission, NUREG/CR-5501, *Selection of Review Level Earthquake for Seismic Margin Studies Using Seismic PRA Results*, October 1989.

Reference 21: U.S. Nuclear Regulatory Commission, NUREG/CR-6728, *Technical Basis for Revision of Regulatory Guidance on Design Ground Motions: Hazard- and Risk-consistent Ground Motion Spectra Guidelines*, October 2001.

Reference 22: U.S. Nuclear Regulatory Commission, SECY-00-0077, *Modifications to the Reactor Safety Goal Policy Statement*, March 30, 2000.

Reference 23: PFS Memorandum and Order, CLI-01-12, June 14, 2001.

Reference 24: Private Fuel Storage, L.L.C. (Independent Spent Fuel Storage Installation) LBP-01-03, 53 NRC 84 (2001).

Reference 25: State of Utah's Request for Admission of Late-Filed Modification to Basis 2 of Contention Utah L, November 9, 2000.

Reference 26: U.S. Department of Energy, Topical Report YMP/TR-003-NP, *Preclosure Seismic Design Methodology for a Geologic Repository at Yucca Mountain*, Rev. 2, August 1997.

Reference 27: State of Utah's Objections and Responses to Applicant's Seventh Set of Formal Discovery Requests to Intervenor State of Utah, September 28, 2001.

REFERENCES

Reference 28: Geomatrix Consultants, Inc., *Fault Evaluation Study and Seismic Hazard Assessment*, Rev. 1, Final Report, Vol. 1, Private Fuel Storage Facility, Skull Valley, Utah, March 2001.

Reference 29: Applicant's Objections and Responses to the State of Utah's Eleventh Set of Discovery Requests Directed to the Applicant, October 2, 2001.

Reference 30: 51 Federal Register 28,044 (1986).

COPY OF TRANSCRIPT

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	) Docket No. 72-22
	) ASLPB No. 97-732-02-ISFSI
PRIVATE FUEL STORAGE L.L.C.	)
	) DEPOSITION OF:
(Private Fuel Storage Facility)	) **DR. C. ALLIN CORNELL**
	) **Volume II**
	) **(Utah Contention L, Part B)**

Thursday, November 1, 2001 - 8:45 a.m.

Location: PARSONS, BEHLE & LATIMER
201 S. Main, Suite 1800
Salt Lake City, UT 84111

Reporter: Vicky McDaniel
Notary Public in and for the State of Utah

State's
Exhibit 130

1 A. Do I know the number?
2 Q. Yes.
3 A. No, I don't.
4 Q. Do you know the number for the canister
5 transfer building?
6 A. I do not know the number.
7 Q. Do you know the number for the cask pad?
8 A. I do not know the number.
9 Q. Do you know if it's been developed?
10 A. The mean failure probability?
11 Q. Yes.
12 A. No, I do not know if it's been developed.
13 Q. Do you know if fragility curves were
14 developed for the canister transfer building?
15 A. No, I do not know.
16 Q. For the storage cask?
17 A. No, I do not.
18 Q. And one last one, for the concrete pad?
19 A. No, I do not.
20 Q. Have you ever estimated the mean failure
21 return period of SSC's due to exceeding the
22 design-basis ground motion? It's a general question.
23 A. Could you state it again? Have I ever
24 calculated --
25 Q. The mean -- actually, I probably should --

CONDENSED TRANSCRIPT

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	) Docket No. 72-22
	) ASLPB No. 97-732-02-ISFSI
PRIVATE FUEL STORAGE L.L.C.	)
	) TELEPHONE DEPOSITION OF:
(Private Fuel Storage Facility)	) **KRISHNA P. SINGH** and
	) **ALAN I. SOLER**
	) (Utah Contention L, Part B)

VOLUME I

Thursday, November 15, 2001 - 2:19 p.m.

Location: Office of the Attorney General
160 East 300 South, 5th Floor
Salt Lake City, Utah

Reporter: Vicky McDaniel
Notary Public in and for the State of Utah

State's
Exhibit 131

1 make you sit there while I go through my questions.
2 MR. GAUKLER: That sounds fine.
3 (Recess from 11:53 to 12:06 p.m.)
4 Q. (BY MS. NAKAHARA) Dr. Soler, are you
5 familiar with the term "fragility curves"?
6 A. (DR. SOLER) No, I'm not.
7 Q. Have you determined the probability of an
8 increase in dose as a function of levels of ground
9 motion for the HI-STORM 100 casks at the PFS facility?
10 MR. GAUKLER: Increase in what? I missed
11 one word.
12 Q. An increase in dose at the fence line as a
13 function of levels of ground motion for the HI-STORM
14 100 cask at the PFS facility.
15 A. (DR. SOLER) I have not. That's not in my
16 area of expertise.
17 Q. Do you know if anyone else has developed
18 such a probability analysis?
19 A. (DR. SOLER) I do not know.
20 Q. Okay, thank you. Have you seen any generic
21 documents such as NRC documents which determine the
22 probability of a cask that meets NRC design standards
23 to withstand certain ground motions?
24 A. (DR. SOLER) Have I seen any such documents?
25 Q. Yes, a generic document.

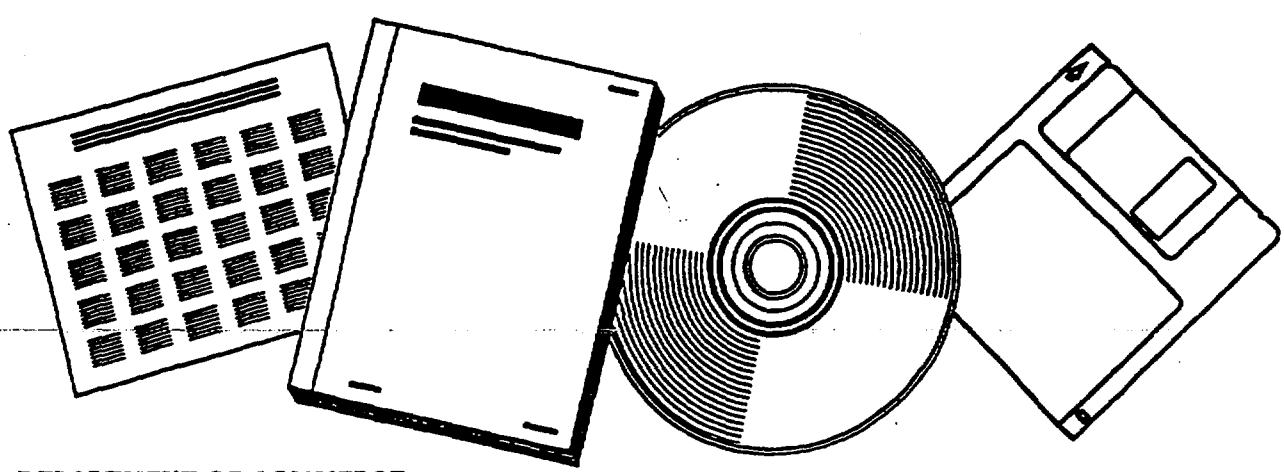
DE96006649

NTIS®
Information is our business.

NATURAL PHENOMENA HAZARDS DESIGN AND EVALUATION CRITERIA FOR DEPARTMENT OF ENERGY FACILITIES

DEPARTMENT OF ENERGY
WASHINGTON, DC

JAN 1996



U.S. DEPARTMENT OF COMMERCE
National Technical Information Service

40519

State's
Exhibit 132

factor of 1.5SF times the DBE. Equation 2-7 is useful for developing alternative evaluation and acceptance criteria which are also based on the target performance goals such as inelastic seismic response analyses. To evaluate items for which specific acceptance criteria are not yet developed, such as overturning or sliding of foundations, or some systems and components; this basic intention must be met. If a nonlinear inelastic response analysis which explicitly incorporates the hysteretic energy dissipation is performed, damping values that are no higher than Response Level 2 should be used to avoid the double counting of this hysteretic energy dissipation which would result from the use of Response Level 3 damping values.

2.5 Summary of Seismic Provisions

Table 2-5 summarizes recommended earthquake design and evaluation provisions for Performance Categories 1 through 4. Specific provisions are described in detail in Section 2.3. The basis for these provisions is described in Reference 2-1.

Table 2-5 Summary of Earthquake Evaluation Provisions

	Performance Category (PC)			
	1	2	3	4
Hazard Exceedance Probability, P_H	$2x10^{-3}$	$1x10^{-3}$	$5x10^{-4}$ $(1x10^{-3})^1$	$1x10^{-4}$ $(2x10^{-4})^1$
Response Spectra	Median amplification (no conservative bias)			
Damping for Structural Evaluation	5%		Table 2-3	
Acceptable Analysis Approaches for Structures	Static or dynamic force method normalized to code level base shear		Dynamic analysis	
Analysis approaches for systems and components	UBC Force equation for equipment and non-structural elements (or more rigorous approach)		Dynamic analysis using in-structure response spectra (Damping from Table 2-3)	
Importance Factor	I=1.0	I=1.25	Not used	
Load Factors	Code specified load factors appropriate for structural material		Load factors of unity	
Scale Factors	Not Used		SF = 1.0	SF = 1.25
Inelastic Energy Absorption Ratios	Accounted for by R_w from Table 2-2		Fμ from Table 2-4 by which elastic response is reduced to account for permissible inelastic behavior	
Material Strength	Minimum specified or 95% non-exceedance in-situ values			
Structural Capacity	Code ultimate strength or allowable behavior level		Code ultimate strength or limit-state level	
Quality Assurance Program	Required within a graded approach (i.e., with increasing rigor ranging from UBC requirements from PC-1 to nuclear power plant requirements for PC-4)			
Peer Review	Not Required	Required within a graded approach (i.e., with increasing rigor ranging from UBC requirements from PC-2 to nuclear power plant requirements for PC-4)		

[1]For sites such as LLNL, SNL-Livermore, SLAC, LBL & ETEC which are near tectonic plate boundaries

Pacific Gas and Electric Company

Lawrence F. Womack
Vice President
Nuclear Services

Diablo Canyon Power Plant
P.O. Box 56
Avila Beach, CA 93424

805.545.4600
Fax: 805.545.4234

December 21, 2001

PG&E Letter DIL-01-002

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Docket No. 72-26
Diablo Canyon Independent Spent Fuel Storage Installation
License Application for Diablo Canyon Independent Spent Fuel Storage Installation

Dear Commissioners and Staff:

In accordance with 10 CFR 72, Subpart B, Pacific Gas and Electric Company (PG&E) hereby submits an application to the Nuclear Regulatory Commission (NRC) requesting a site-specific license for an Independent Spent Fuel Storage Installation (ISFSI) at the Diablo Canyon Power Plant (DCPP).

An ISFSI at Diablo Canyon is part of PG&E's plan to provide storage capacity for spent fuel generated by DCPP through the remainder of the term of the respective NRC operating licenses (DPL 80 and 82). A permanent repository is not yet available and is not expected to be available on a schedule to meet DCPP operational needs. The ISFSI that is the subject of this 10 CFR 72 application is required beginning in 2006. This plan for handling and storing spent fuel meets PG&E's statutory obligations and will allow for continuing operation of DCPP.

PG&E is submitting: (a) calculation packages (PG&E Letters DIL-01-004, dated December 21, 2001 and DIL-01-007, dated December 21, 2001); (b) proprietary and non-proprietary Holtec drawing packages (PG&E Letter DIL-01-008, dated December 21, 2001); (c) geologic data reports (PG&E Letter DIL-01-005, dated December 21, 2001); and (d) DCPP Security Program Revisions and Exemption Requests (PG&E Letter DIL-01-003, dated December 21, 2001) as supplemental information to support NRC review of the ISFSI license application.

In connection with this submittal, PG&E previously submitted a 10 CFR 50 license amendment request (LAR) (DCL-01-096, dated September 13, 2001) seeking NRC approval to take credit for soluble boron in the spent fuel pools in order to maximally use the existing storage capacity and thus provide spent fuel storage with full core offload capability through approximately 2006. PG&E will also submit a 10 CFR 50

NMSS01
1/14

Extra Copies
To: Tim Kobetz

State's
Exhibit 133

LAR to permit cask handling activities in the DCPP fuel handling building/auxiliary building.

The Diablo Canyon ISFSI will use the Holtec dry cask storage system, which has previously been certified by the NRC. This license application and supporting materials demonstrate that the Diablo Canyon ISFSI will be built and operated in a safe manner, will have no impact on the operation of the power plant, and will have no significant environmental impacts. While additional amendments will be required for the 10 CFR 50 licenses for the power plant, as discussed below, they will involve no undue public health and safety risks.

Background

DCPP consists of two nuclear generation units located approximately 6 miles northwest of Avila Beach, California. The two units are essentially identical pressurized water reactors (PWRs), each rated at a nominal 1100 megawatts-electric (MWe). The two units share a common auxiliary building as well as certain components of auxiliary systems. The reactors, including their nuclear steam supply systems, were furnished by Westinghouse Electric Corporation. Each reactor has a dedicated fuel handling system and spent fuel storage pool. Both units and the plant site are owned and operated by PG&E.

Unit 1 began commercial operation in May 1985 and Unit 2 in March 1986. The operating licenses expire in September 2021 for Unit 1 and April 2025 for Unit 2. In general, the operating and spent fuel storage histories of DCPP Unit 1 and Unit 2 are similar to those of other PWRs. The spent fuel storage racks were initially of low-density design, capable of accommodating only one and one-third cores of spent fuel assemblies. These low-density racks were replaced in the late 1980s with the high-density racks that are currently in use.

The spent fuel pool for each unit presently has sufficient capacity for the storage of 1,324 fuel assemblies. Each reactor core contains 193 fuel assemblies, and both units are currently operating on 18 to 21-month refueling cycles. Typically, 76-96 spent fuel assemblies are permanently discharged from each unit after a refueling. Each unit has operated for 10 fuel cycles and each is presently operating in its 11th cycle. Based on the existing inventory and the expected generation of spent fuel, each spent fuel pool can accommodate the concurrent storage of a full core of irradiated fuel and the anticipated quantity of spent fuel generated from prior refueling operations until 2006. After that time, an alternative means of spent fuel storage at DCPP must be provided unless the spent fuel can be shipped offsite.

The Nuclear Waste Policy Act (NWPA) of 1982 as amended, mandated that the Department of Energy (DOE) assume responsibility for the permanent disposal of spent nuclear fuel from the nation's commercial nuclear power plants beginning in January 1998 pending the availability of a permanent DOE repository. Nuclear power plant operators such as PG&E have been given the responsibility under the NWPA to provide for the interim onsite storage of spent fuel until it is accepted by DOE. As noted above, DOE has not complied with its NWPA mandate to have a repository in operation commencing in January 1998, and no interim spent fuel storage facility has been established. Moreover, no such DOE facility is expected to become operational in a timeframe to meet DCPP's spent fuel storage needs. Thus, spent fuel generated by DCPP will need to remain at DCPP until a DOE or other facility is available. Consequently, additional spent fuel storage capacity is needed at DCPP beginning no later than 2006.

The additional capacity to accommodate discharged spent fuel, as proposed herein, will allow DCPP to continue to generate electricity. Any interruption in the availability of this capacity would almost certainly have a negative impact on the domestic sector power supply in California. Given the existing power supply situation in California and in the western United States as well as uncertainties about future power supplies, any loss of power from DCPP could have significant adverse impacts on the population, the infrastructure, and the economy. Expansion of the onsite spent fuel storage capacity at DCPP as planned by PG&E is necessary to avoid these potential significant negative impacts.

PG&E has considered several alternative means for accommodating the additional spent fuel that will be generated at DCPP through the licensed operating life of each unit. The onsite alternatives include a second reracking of the spent fuel pools to replace the existing high-density racks with racks of higher-density design and building an onsite ISFSI using dry cask storage technology. PG&E has also considered the possibility of participating in the Private Fuel Storage venture, which has an application pending before the NRC for a license to independently store spent fuel from nuclear power plants. Based on an overall assessment of operational and safety considerations, the amount of spent fuel to be generated, the transportation requirements associated with the alternatives, resources needed, and scheduling restraints, PG&E has concluded that dry cask storage of spent fuel at DCPP is the optimum alternative at this time for providing the necessary storage capacity. However, as discussed below, increasing the spent fuel pool storage capacity through a second reracking with higher density racks remains a viable option if it appears that the ISFSI cannot be licensed on a schedule that meets PG&E's storage requirements.

The expanded storage capacity provided by the use of dry casks at the ISFSI will be used to store aged spent fuel that has been stored for 5 years or longer in the DCPP spent fuel pools. The storage spaces in the respective spent fuel pools that become available following this transfer of the aged spent fuel into dry cask storage then can be used to store future discharged spent fuel from the reactor core. Storage casks will be acquired as needed to accommodate the spent fuel generated until shipment offsite occurs.

Dry Cask Storage: Licensing Considerations

The Diablo Canyon ISFSI will consist of: the ISFSI storage pad, the cask transfer facility (CTF), the onsite cask transporter, and the dry cask storage system. The dry cask storage system that has been selected by PG&E for the Diablo Canyon ISFSI is the Holtec International (Holtec) HI-STORM 100 System. The HI-STORM 100 System is comprised of a multi-purpose canister (MPC), the storage overpack, and the HI-TRAC transfer cask. The HI-STORM 100 System is certified by the NRC for use by general licensees as well as site-specific licensees, presently with a 24 PWR fuel assembly MPC and storage overpack (see NRC 10 CFR 72 Certificate of Compliance [CoC] No. 1014).

Holtec has proposed a number of changes to the certified HI-STORM 100 System in LAR 1014-1, submitted to the NRC on August 31, 2000. These proposed changes include a HI-STORM 100SA storage overpack, a higher-capacity MPC-32 design (for storage of 32 PWR spent fuel assemblies), and MPC designs with different fuel storage capabilities (e.g., high burnup fuel, certain damaged fuel). As discussed below, several of these proposed changes are desirable for the Diablo Canyon ISFSI. PG&E understands, however, that several of the proposed changes in LAR 1014-1, such as the designs to accommodate high burnup fuel, may involve extensive NRC review. As discussed below, issuance of a revised Certificate of Compliance No. 1014-1 may not necessarily be required to support the plant-specific Diablo Canyon ISFSI license.

The Diablo Canyon ISFSI is designed to hold up to 140 storage casks (138 casks plus 2 spare locations). Because of its higher capacity, the principal MPC to be used will be the MPC-32. Based on the current fuel strategy and use of the MPC-32, the ISFSI with a storage pad capacity of 140 casks will be capable of storing the spent fuel generated by DCPP Units 1 and 2 over the term of the current operating licenses (2021 and 2025, respectively). In addition, to accommodate spent fuel generated during the licensed period, as well as any damaged fuel assemblies, debris, and nonfuel hardware, PG&E may use three other MPC designs from the HI-STORM 100

STATE OF UTAH TESTIMONY OF DR. MARVIN RESNIKOFF ON UNIFIED CONTENTION UTAH L/QQ (Seismic Exemption - Dose Exposure)

I. **The Design Basis Earthquake Standard.**
   A. Reasonable assurance that the health and safety of the public and onsite workers will be protected if the casks are subjected to the peak ground accelerations from a 2,000-year mean return interval earthquake at the PFS facility.

II. **The PFS Parameters for a 2,000-Year Return Interval Earthquake Are Outside the Bounds of the HI-STORM 100 Certificate of Compliance "CoC."**
   A. CoC bounding tipover horizontal acceleration is 0.445 g with a 0.16 g vertical acceleration.
   B. CoC calculates the dose rate at controlled area boundary using 8,760 hours per year where PFS uses only 2,000 hours per year.

III. **PFS Fails to Comprehensively Estimate the Potential Dose Rate From a 2,000-Year Return Interval Earthquake.**
   A. PFS fails to use a conservative occupancy time in control boundary dose calculations.
      1. PFS used a 2,000 hour per year occupancy time.
      2. Because PFS does not control the property outside the PFS fence line, PFS should assume an 8,760 hours per year occupancy time.
      3. NRC staff required the use of 8,760 hours per year occupancy time in the HI-STORM 100 CoC.
   B. Using 8,760 hours per year occupancy, the dose at the control boundary is 25.6 mrem/year which exceeds the allowable limit.

IV. **Tipover of Cask Under a 2,000-Year Return Interval Earthquake at PFS.**
   A. Because the HI-STORM 100 cask may tipover under a 2,000-year earthquake, PFS cannot rely their non-mechanistic tipover analysis to determine the consequences.
   B. Initial angular velocity will be greater than zero if cask tips over, thus the canisters 45 g design basis impact will be exceeded.
   C. In the event of tipover, the HI-STORM 100 cask will flatten, increasing dose rate.
   D. In the event of tipover, PFS has not quantified the amount of steel stretching and concrete cracking, and whether there is an increase in dose.
   E. Less shielding on bottom of tipped over cask resulting in an increase in dose if facing towards boundary.
   F. In the event of tipover, the cask lid may be displaced.
   G. In the event of tipover, PFS must determine the temperature limit is exceeded.
      1. PFS cannot upright the casks within the 33 hours

V. **Potential Increase in Dose Even If the Cask Does Not Tipover.**
   A. The HI-STORM 100 cask may be uplifted greater than the 11 inch drop allowed under the regulations.
   B. The canister will experience greater than the allowable 45 g design limit if the cask dropped greater than 11 inches.
   C. PFS's drop calculations fail to assume the cask will drop at an angle, e.g. corner drop.

VI. **Conclusion.**
   A. PFS Seismic Consequence Analyses are Not Bounding or Conservative.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of:	)	Docket No. 72-22-ISFSI
PRIVATE FUEL STORAGE, LLC (Independent Spent Fuel Storage Installation)	)	ASLBP No. 97-732-02-ISFSI
	)	April 1, 2002

STATE OF UTAH TESTIMONY OF DR. MARVIN RESNIKOFF REGARDING UNIFIED CONTENTION UTAH L/QQ (Seismic Exemption - Dose Exposure)

Q. 1: Please state your name, affiliation, and qualifications.

A. 1: My name is Marvin Resnikoff. I am the Senior Associate of Radioactive Waste Management Associates ("RWMA"), a private technical consulting firm based in New York City. I hold a doctorate degree in high-energy theoretical physics from the University of Michigan. I have researched radioactive waste issues for the past 28 years and have extensive experience and training in the field of nuclear waste management, storage, and disposal. Our work at RWMA includes matters covered in this testimony: (i) safety issues related to the storage of irradiated fuel, and (ii) the calculation of radiation exposure. I previously prepared a declaration (January 30, 2001) for the State of Utah in response to the summary disposition motion on contention Utah L, part B. I am also testifying as a witness in the April hearing on Contention Utah K. My curriculum vitae is included as State's Exhibit 134.

During the past 28 years I have researched and evaluated technical issues related to the storage, transportation, and disposal of radioactive waste, including spent nuclear power plant fuel. I am extremely familiar with the general characteristics of spent nuclear power plant fuel, as well as the designs of spent fuel storage systems that are now in use or proposed for future use in the United States. My experience includes technical review and analysis of numerous dry cask storage designs, including proposed independent spent fuel storage installations ("ISFSIs") at the Point Beach, Palisades and Prairie Island reactors, as well as Holtec's HI-STORM and HI-STAR casks for the proposed Private Fuel Storage, LLC ("PFS") facility. I have prepared comments for the States of Utah and Nevada on the Nuclear Regulatory Commission ("NRC") Staff's preliminary Safety Evaluation Reports ("SERs") for the HI-STAR/HI-STORM systems. I have also reviewed Topical Safety Analysis Reports ("TSARs") for transportation casks, including the IF-300, NLI-1/2 and

casks used for plutonium transport.

Since 1975 I have worked on transportation issues, including cask safety, for the States of Utah, Nevada (including Churchill, Clark and White Pine Counties), Idaho, New York, New Mexico and Alaska. This work began with work for the New York Attorney General's office on the safety of transporting plutonium by plane out of John F. Kennedy International Airport. My role in the case was to determine whether the plutonium shipping container could be punctured and the amount of plutonium that could be released. I was an invited speaker at the 1976 Canadian meeting of the American Nuclear Society to discuss the risk of transporting plutonium by air. On behalf of the State of New York, I also reviewed and provided comments on NUREG-170, "Final Environmental Statement on the Transportation of Radioactive Material by Air and Other Modes." Continuing this work, I am presently preparing testimony for the Earthjustice Foundation on transportation of Pu from Rocky Flats to Lawrence Livermore lab in DT-22 casks. On behalf of the State of Nevada and Clark County, Nevada, I provided comments on the transportation cask safety studies and transportation risk assessments, such as the Modal Study and references, and more recently NUREG/CR-6672. RWMA has conducted transportation risk assessments for the State of Nevada and has employed various computer codes and formulas to estimate the amount of radioactivity released in and the health and economic consequences of a severe accident, including the computer models RADTRAN, RISKIND, RESRAD, and HOTSPOT. In addition, in hearings before state commissions and in federal court, I investigated proposed dry storage facilities at the Point Beach (WI), Prairie Island (MN) and Palisades (MI) reactors. For the Council on Economic Priorities, I have written a book on the transportation and storage of irradiated fuel.

I have considerable training and experience in the field of radiation dose exposure involving nuclear and hazardous facilities, serving as an expert witness in numerous personal injury cases in which I estimated radiation doses and the likelihood these exposures caused cancer. These cases involved uranium mining and milling, oil pipe cleaning, X-rays, thorium contamination and other issues. This work involved the use of computer codes, such as MILDOS, to estimate radiation doses and spreadsheets employing dose conversion factors. Under my oversight, the staff at RWMA and I have reviewed risk assessment studies and evaluated radiation dose exposures for proposed low-level radioactive waste facilities at Martinsville (Illinois), Boyd County (Nebraska), Wake County (North Carolina), Ward Valley (California) and Hudspeth County (Texas).

Q. 2: What has been your involvement in assisting the State with respect to PFS's seismic exemption request?

A. 2: I was designated as one of the State's testifying experts for Contention Utah L part B on September 28, 2001 – Utah L, Part B has now been unified as Utah L/QQ. My testimony relates to Section E of the unified contention. I assisted in the preparation, in

part, of State of Utah's Request for Admission of Late-Filed Modification to Basis 2 of Utah Contention L, filed on January 26, 2000 ("First Modification to Basis 2"), and have reviewed another request by the State for Admission of Late-Filed Modification to Basis 2 of Utah Contention L, filed November 9, 2000 ("Second Modification to Basis 2") and submitted Declarations in support thereof. I also participated in the preparation of discovery against the Applicant and the NRC Staff with respect to Utah L part B.

Q. 3: What is the purpose of your testimony?

A. 3: My testimony relates to whether the PFS design basis for the Holtec International Inc. ("Holtec") HI-STORM 100 cask system provides reasonable assurance that the health and safety of the public and onsite workers will be protected if the casks are subjected to the peak ground accelerations from a 2,000-year mean annual return period earthquake at the PFS site.

Q. 4: Are you familiar with the PFS license application filed in this proceeding and the proposed storage and transportation casks PFS plans to use?

A. 4: Yes, I have been assisting the State in the Private Fuel Storage, LLC ("PFS") proceeding since 1997 and have reviewed the original PFS license application and the various revisions thereto. I am familiar with PFS's Safety Analysis Report and Environmental Report, as well as the Staff's Safety Evaluation Report and Environmental Impact Statement.

PFS plans to transport spent nuclear fuel to the Skull Valley site in Holtec HI-STAR transportation casks and store the fuel on-site in Holtec HI-STORM 100 storage casks. I am familiar with Holtec's applications for the storage and transportation casks (HI-STORM and HI-STAR) PFS plans to use. I am also familiar with NRC regulations, guidance documents, and environmental studies relating to the storage and transportation of spent nuclear power plant fuel, including NUREG-0800, NUREG-1536, 10 CFR Parts 72 and 100, EPA's Protective Action Guide, and Federal Register Notice dated December 4, 1996 (61 Fed. Reg. 64257).

Q. 5: What is your familiarity, if any, with the Holtec Certificate of Compliance for the HI-STORM 100 cask?

A. 5: NRC issued a certificate of compliance ("CoC") for the HI-STORM 100 cask effective May 31, 2000. 65 Fed. Reg. 25241 (2000). By issuing a CoC, NRC determined that the "HI-STORM 100 cask system, as designed and when fabricated and used [for general licenses] in accordance with the conditions specified in its CoC, meets the requirements of 10 CFR Part 72." Id.

Q. 6: How does the CoC relate to the use of the HI-STORM 100 cask at the proposed PFS facility?

A. 6: The site-specific conditions at the PFS facility are outside the bounds of the generic CoC for the HI-STORM 100 cask system. Therefore, in order to use the HI-STORM 100 system, PFS must conduct a site-specific analysis to determine whether the performance of the casks at the PFS site are adequate to protect health and safety. There are serious shortcomings in PFS's site-specific analysis. *See* State's Testimony of Dr. Steven Bartlett and Dr. Farhang Ostadan (dynamic analysis) and State's Testimony of Dr. Ostadan and Dr. Mohsin Khan (cask stability), filed concurrently.

Q. 7: What are these serious shortcomings in PFS's site specific analysis?

A. 7: In my opinion, PFS has not shown that unanchored HI-STORM 100 casks will "reasonably maintain confinement of radioactive material" under off-normal and credible accident conditions at the proposed PFS site as required by 10 CFR § 72.236. Further, PFS and cask designer, Holtec, have not quantified the consequences of a potential 2,000-year mean annual return period, 10,000-year return period, or deterministic earthquakes that could take place at the proposed PFS site.

Q. 8: Why is the CoC unable to reflect the facts and conditions at the proposed PFS site?

A. 8: There are significant differences between the facts and conditions used to support the HI-STORM CoC and those at the PFS site; for example:

a. The calculated ground motions at the PFS facility for a 2,000-year return period earthquake are 0.711 g horizontal and 0.695 g vertical (SAR at 2.6-107, Rev. 22). As described below, the bounding ground motions in the CoC for the purpose of determining the maximum zero point acceleration that will not cause incipient tipping are bounded by a horizontal acceleration of .445 g and vertical acceleration of .16 g.

In its HI-STORM CoC[1] analysis, Holtec treated a loaded cask as a rigid body and set up the following inequality,

$G_H + \mu G_v \leq \mu$ (where μ is 0.53, G_H is the resultant horizontal acceleration, and G_V is the resultant vertical acceleration),

---

[1] Excerpts from the Holtec HI-STORM 100 Cask Certificate of Compliance for Spent Fuel Storage Casks (effective date May 31, 2000), docket number 72-1014, included as State's Exhibit 135.

stating that the maximum g loading a cask could take without tipping would occur when the horizontal force acting at the center of gravity of the cask just balances the vertical force acting at the pivot point. Any horizontal force greater than this would cause tipping in this rigid body assumption. In the above formula $\mu = r/H$. In the HI-STORM CoC Holtec reduced the value of r/H from 0.56 to 0.53, thereby giving a bounding horizontal acceleration of .445 g (with .16 g as the corresponding vertical acceleration). CoC, Appendix B at 3-8, State's Exh. 135.

As can be seen from the above, the design basis earthquake ("DBE") ground motions for the PFS site are significantly higher than those specified in the CoC for the HI-STORM 100 cask.

b. There is an inconsistency between the occupancy time at the controlled area boundary used in the Holtec CoC and that used at the PFS site. The Holtec CoC used a duration time of 8,760 hours per year whereas at the PFS site only 2,000 hours per year was used to compute dose exposure at the fence post. *See* ¶ 10 below.

c. Holtec calculated the dose consequences in a non-mechanistic single cask tip over event, whereas at PFS the entire field of casks could tip over under the accelerations caused by the DBE. *See* State's Testimony of Drs. Bartlett and Ostadan (dynamic analysis) and Testimony of Drs. Ostadan and Khan (cask stability).

Q. 9: How have these differences affected PFS's and Holtec's analyses?

A. 9: Failure to quantify the consequences of a potential 2,000-year return period, 10,000-year return period, or deterministic earthquake is fatal to PFS' and Holtec's conclusions because the calculated ground motions for a 2,000-year return period earthquake (of 0.711 g horizontal and 0.695 g vertical) at the PFS facility are so far outside the bounds of those used to support the Holtec CoC that it is fair to conclude that there is no quantification of the consequences of what will occur at ground motions of approximately 0.7g.

Q. 10: Has PFS appropriately calculated the dose rate?

A. 10: PFS calculated a 5.85 mrem/year dose for a 2,000 hour/year occupancy time at the controlled area boundary under normal operating conditions.[2] The Holtec dose calculation at the PFS controlled area boundary is inconsistent and less conservative than

[2] PFS EIS Commitment Resolution Letter # 13 (September 25, 2001), included as State's Exhibit 136.

other Holtec dose calculations which likely used an occupancy rate of 2,080 hour/year.[3] PFS has significantly underestimated the dose rate. To assure that the public is protected, PFS must calculate a radiation dose assuming a hypothetical individual is located at the site boundary the entire year or 8,760 hours/year because PFS cannot control who is at the site boundary or for what length of time. Although PFS does not control property beyond the site boundary, it calculated a dose rate at a distance of 2 miles from the site boundary.[4] In the CoC for the HI-STORM 100 System, NRC Staff agreed with my position in response to comment B.18, stating: "The NRC agrees that 8,760 hours/year should be used [for estimating the dose at the site boundary]." *See* 65 Fed. Reg. 25241, 25245 (2000). Thus, using an 8,760 hour/year assumption is consistent with the NRC Staff position in approving the HI-STORM 100 CoC.

Q. 11: What is a more appropriate calculation of the dose rate?

A. 11: I calculated the correct annual dose rate assuming a hypothetical individual remained at the site boundary for 8,760 hours. The dose rate is 5.85 mrem/year * 8,760 hours/year ÷ 2,000 hours/year = 25.6 mrem/year, which is in excess of the allowable 25 mrem/year specified in 10 CFR § 72.104(a). This is the dose rate under normal operating conditions, absent a seismic event.

Q. 12: How does your calculation of the dose rates differ from PFS's calculation?

A. 12: In addition to PFS's selection of 2,000 hour per year exposure duration being at odds with the Holtec CoC, it is also unjustified. The PFS facility is expected to have an operational life of at least 40 years. PFS SER (2002), Table 4-3, p.4-8. The site is located on the northwestern edge of the Skull Valley reservation abutting privately owned property. In my opinion it is nonconservative and unrealistic to analyze dose exposure for 40 hours per week for 50 weeks a year (*i.e.*, 2,000 hours per year). There should be an expectation that residential housing will abut the PFS site boundary. Moreover, by definition an "uncontrolled" area is an area not controlled by PFS.

Q. 13: How does PFS's and Holtec's tip over analysis impact PFS's dose rate calculation?

A. 13: Holtec and NRC Staff considered the HI-STORM tip over analysis as a non-mechanistic event. "In the absence of an identified [cask tipover] hazard" NRC allows a

---

[3] *See* Deposition Transcript of Everett Lee Redmond II (November 15, 2001) ("Tr."), excerpts included as State's Exhibit 137, at 40.

[4] PFS Consolidated Safety Evaluation Report ("PFS SER") (2002) at 7-6.

non-mechanistic cask tipover analysis.[5] *See* HI-STORM 100 Safety Evaluation Report[6], State's Exhibit 138, at § 11.2.4.1; HI-STORM 100 Topical Safety Analysis Report, State's Exhibit 139,[7] at § 11.2.3. However, a non-mechanistic tipover analysis is no longer acceptable because the HI-STORM 100 casks will likely tipover under peak ground accelerations for a 2,000-year mean annual return period earthquake. Because the dose at the controlled area boundary is already slightly greater than 25 mrem/year assuming an exposure duration of 8,760 hours/year, any further increase will put this dose that much higher than the limits allowed in 10 CFR § 72.104(a).

Q. 14: What has PFS calculated as the dose rate in the event of a tip-over accident?

A. 14: PFS acknowledges that a tip-over accident could "cause localized damage [including crushing of the concrete and associated micro-cracking] to the radial concrete shield and outer steel shell where the storage cask impacts the surface." *See* PFS Joint Dec.[8] ¶ 25. Holtec in fact states that the "overpack surface dose rate . . . could increase due to the [tipover] damage." HI-STORM 100 TSAR at 11.2-7. Contrary to the HI-STORM 100 TSAR and without any quantified analysis, PFS claims that no "noticeable increase" in radiation dose would occur at the site boundary. PFS Joint Dec. ¶ 25. PFS' radiation dose expert is unaware of any calculations that estimate the radiation consequences of concrete cracking. Redmond Tr. at 46, 47, State's Exh. 137.

Q. 15: What is your opinion of PFS's dose rate calculation in the event of a tip-over accident?

A. 15: In my opinion, there is no support for PFS's claim.

Q. 16: How is PFS's dose rate calculation insufficient?

A. 16: To determine whether fuel assemblies would be damaged in a tipover event, Holtec calculated the deceleration of the top edge of the canister as the cask struck the

---

[5] NUREG-1536, *Standard Review Plan for Dry Cask Storage Systems*, at 2-9.

[6] HI-STORM 100 Safety Evaluation Report ("HI-STORM SER"), excerpts included as State's Exhibit 138.

[7] HI-STORM 100 Topical Safety Analysis Report, HI-951312 ("TSAR") (February 4, 2000), excerpts included as State's Exhibit 139.

[8] Joint Declaration of Krishna P. Singh, Alan I. Soler, and Everett L. Redmond II (November 9, 2001) ("PFS Joint Dec."), filed with Applicant's Motion for Summary Disposition of Part B of Utah Contention L (November 9, 2001).

cement pad. *See*, e.g., HI-STORM 100 TSAR, Section 3.A. In its hypothetical tipover analysis, Holtec identified "a center of gravity over pivot point" configuration as its starting point, assuming that the initial angular velocity was zero. HI-STORM 100 TSAR, Section 3.A.6, State's Exh. 139. There are numerous problems with Holtec's analysis and the conclusion PFS draws from it.

a. PFS's witnesses conclude that during an earthquake, "the initial linear velocity of the cask centroid in the plane of precession . . . would not be significantly increased over the [hypothetical] tip-over condition already studied." PFS Joint Dec. ¶ 20. PFS again provides no supporting calculations and in my opinion, PFS's starting premise of zero initial angular velocity is unfounded.

b. If cask tip over results from earthquake accelerations, the initial angular velocity may be greater than zero. From this you can conclude that the top of the canister will decelerate at greater than 45 g, in exceedence of the 45 g design basis, thereby damaging the fuel assemblies; also the HI-STORM 100 cask will flatten more than contemplated by PFS. Claims that the "MPC has a very substantial margin built into it" are unsubstantiated; PFS has again failed to support its site specific use of the HI-STORM cask with any calculations or test data. *See* PFS Joint Dec. ¶ 20. Therefore, PFS has not substantiated whether or not the confinement boundary would be breached in a 2,000-year earthquake or a 10,000-year earthquake. *See* id.

Q. 17: What is your opinion of PFS's analysis of the potential consequences of a HI-STORM 100 cask tipover?

A. 17: Since the initial angular velocity may be greater than zero as the cask center of gravity passes the pivot point, the HI-STORM 100 cask will also flatten more than contemplated by PFS. Although PFS claims that the "MPC has a very substantial margin built into it," it again fails to support its claim with any calculations or test data. *See* PFS Joint Dec. ¶ 20. Furthermore, PFS also acknowledges that the "roundness" of the casks could be reduced following cask tipover. PFS Joint Dec. ¶ 26. However, in the event of cask tipover, PFS has not correctly quantified the amount of concrete flattening or the resultant reduction of gamma and neutron shielding. Thus, the potential consequence of a HI-STORM 100 cask tipover is another unresolved critical safety issue that must be addressed prior to determining or justifying the appropriate site specific design basis earthquake.

Q. 18: What is your opinion of PFS's assertion that in the event of cask tipover, the roundness of the cask could only be reduced in the radial area of the impact?

A. 18: If a HI-STORM 100 cask tips over, PFS further states that the roundness of

the storage cask could only be reduced in the radial area of the impact. PFS Joint Dec. ¶ 26. PFS witness, Dr. Redmond, then implies that any increase in dose from the reduction in radiation shielding caused by the flattening or localized deformation is inconsequential because the increase in dose will occur between the cask and the ground.[9] First, PFS has performed no analysis to show that the deformation will be in contact with the ground. During a seismic event, the cask could roll and the flattened end may not remain facing the ground.

Second, when the HI-STORM 100 casks are in fact uprighted, the flattened area of the cask (localized deformation) will not face the ground. PFS has failed to calculate the potential increase in dose at the site boundary or to workers from such casks.

Q. 19: What, if any, are the flaws in Holtec's analysis of the HI-STORM 100's stability in a tipover event?

A. 19: Under a HI-STORM 100 cask tipover event, Holtec has also not quantified the amount of stretching of the metal outer surface, and the amount of cracking of the cement. Cracking will lead to an increased gamma dose at the fence post and an increased neutron and gamma dose to PFS workers since gamma rays and neutrons will pass more easily through this less shielded region. The potential increase in radiation dose at the fence post must be quantified before the design basis earthquake is specified. Also, the analysis performed by Holtec in the HI-STORM TSAR does not bound cask tip-over resulting from an earthquake affecting the PFS facility because the Holtec analysis evaluates only one cask being tipped over. At a facility with up to 4,000 casks, it is highly unlikely that only one HI-STORM 100 cask will tipover as a result of peak ground accelerations from a 2,000-year mean annual return period earthquake affecting the PFS facility. *See e.g.* Utah Joint Dec. ¶ 74.

Q. 20: What would happen if the HI-STORM 100 cask were to tipover such that bottom of a row of casks faces the fence post?

A. 20: If the HI-STORM 100 casks tipover such that the bottom of a row of casks faces the fence post, the direct gamma dose at the fence post will increase. As seen in RWMA's drawing, included as State's Exhibit 140, a ring or torus of the bottom of the HI-STORM 100 cask has reduced shielding. This is not a region where the fuel is located, but indirect gamma rays and neutrons will stream through the bottom of the cask. PFS has not calculated the dose at the boundary from the bottoms of tipped over HI-STORM 100 casks. Redmond Tr. at 50, State's Exh. 137. In collaboration with my colleague, Matthew Lamb, I performed preliminary rough calculations for the reduced shielding caused by exposure from the bottom of the casks at the site boundary. I am unaware of any dose calculations performed by Holtec. *See* RWMA's dose calculations, included as State's Exhibit 141. My

[9] Redmond Tr. at 48, State's Exh. 137; PFS Joint Dec. ¶ 26.

calculations show the dose rate due to gamma rays will increase between 1.8 and 18 times that calculated by PFS at the site boundary assuming a 2,000 hour year, and between 7.7 and 77 times that calculated by PFS assuming an 8,760 hour year. Because of the likelihood that HI-STORM 100 casks will tipover during a 2,000-year mean annual return period earthquake, in order to justify that there will be no effect to health and safety from using a 2000-year DBE, in my opinion PFS must calculate a bounding radiation dose at the fence line and to workers.

Q. 21: In a tipover event, do you expect any additional damage to the cask other than flattening?

A. 21: I have further concerns about the modeling of the Holtec cask in a tipover event. HI-STORM TSAR Fig. 3.A.18[10] shows the structural details. The concrete overpack is topped with a metal lid plate about 3 ¾ inch thick, and a concrete lid bottom plate or plug that fits within the concrete cylindrical side walls of the HI-STORM cask. In a tipover event, discussed in TSAR Appendix 3.B, the cask walls at the top of the cask are expected to flatten slightly (0.11 inch, p. 3.B-5[10]) when the cask top strikes the ground. On the other hand, the cask lid plate is expected to be displaced as much as 4.9 inches in a tip over event (TSAR, p. 3.A-15[10]). This indicates to me that the 3 ¾ inch thick lid plate is going to strike the ground in a tipover event and send a strong dynamic impulse to the cask wall and canister. It does not appear that this cask detail, that may affect the canister welds, has been modeled.

Q. 22: In addition to the cracking of the concrete cask, are there any other issues that need to be addressed by PFS?

A. 22: In addition to cracking of the concrete cask, the issue of cask heat-up and loss of concrete shielding must be addressed by PFS. The HI-STORM 100 cask is designed to be cooled by a "chimney effect." Cooler air enters the bottom vent and rises and is released from the top vent. If the casks tip over, the chimney effect is reduced dramatically and this is equivalent to the intake vents being blocked. Holtec calculations show that after 33 hours of 100% air inlet blockage, the concrete temperature will exceed the short-term limit of 350°F specified in the CoC for the HI-STORM 100 cask.[11] The CoC temperature limit is established to ensure the continued effectiveness of the neutron shielding by ensuring the water does not evaporate from the concrete, reducing the amount of hydrogen available for neutron capture.[12] PFS has not analyzed the effects of an increase of neutron dose to on-site workers from the prolonged tipover of HI-STORM 100 casks.

---

[10] *See* State's Exh. 139.

[11] *See* HI-STORM 100 TSAR, p. 1.D-4, Table 1.D.1 (Rev 10), State's Exh. 139.

[12] *See* Redmond Tr. at 60-61, State's Exh. 137.

Q. 23: In the event of a cask tipover, could PFS upright all of the casks, and if not, what would be the potential dose consequences?

A. 23: At the PFS site there is the likelihood that the HI-STORM 100 casks will tip over during a 2,000-year return period DBE. Testimony of Drs. Khan and Ostadan (cask stability). In my opinion PFS could not upright all the casks within the time limits imposed by the CoC and this will result in the potential increase in neutron dose to workers.

a. The HI-STORM casks are approximately 20 feet high, 11 feet in diameter and weigh about 175 tons.[13] In restoring the casks to their original and upright position, the configuration of the casks on the pad dictates that a crane would have to work from the outside perimeter of the pads towards the center of the pads. Obtaining a crane capable of lifting 175 tons and transporting it to the Skull Valley site, maneuvering around other casks, then uprighting and re-positioning each 175 ton cask on the pad would result in only a few casks, if any, being restored to their original pad position within 33 hours. Casks remaining horizontal for extended periods of time would result in the increased temperature of the concrete overpack past the 350°F short-term temperature limit specified by the HI-STORM 100 CoC. If the temperatures resulted in the evaporation of water from the concrete, workers would then have to operate in an increased neutron dose environment.

b. The CoC temperature limit is established to ensure the continued effectiveness of the neutron shielding by ensuring the water does not evaporate from the concrete, reducing the amount of hydrogen available for neutron capture. *See* Redmond Tr. at 60-61, State's Exh. 137. In collaboration with my colleague Matthew Lamb, I performed calculations, included as State's Exhibit 143[14], that show increased neutron dose due to reduced shielding. These calculations estimate an increase in dose to workers due to neutrons of up to 57.3 times greater than the value calculated by Holtec of 1.88 mrem/hour 1 meter from the cask mid-height if all of the water evaporates from a HI-STORM cask. This would result in a worker dose of approximately 108 mrem/hour. A worker exposed to this for just over 46 hours would exceed the 5 rem/year occupational dose rate specified in 10 CFR Subpart C § 20.1201.

Q. 24: What would be the effect on the dose rate if the casks do not tipover, but slide as the Altran Report suggests that they would?

A. 24: The Altran Report, State's Exhibit 122[15] concludes that the HI-STORM 100 casks will tipover under peak ground accelerations induced by a 2,000-year earthquake at the

[13] State's Exhibit 142, PFS SAR, Table 4.2-2, Rev. 12.

[14] Calculation of Neutron Dose at Elevated Concrete Temperatures.

[15] Testimony of Drs. Khan and Ostadan (cask stability).

PFS facility. Even if the casks do not tipover, the casks may still slide approximately 370 inches in the x direction and 230 inches in the y direction and be uplifted 27 inches.[16] Contrary to PFS's claims, the casks will not move in phase with each other.[17] Under these conditions the casks will slide and collide with each other. PFS has not evaluated the damage nor calculated dose increase from colliding casks.[18] Also, the HI-STORM 100 cask will likely be lifted up to 27 inches if subjected to peak ground accelerations induced by a 2,000-year earthquake at the PFS facility.[19] The HI-STORM 100 cask was analyzed and determined capable of withstanding only a drop of 11 inches.[20] PFS has not demonstrated that its requested design basis ground motion exemption will not result in potential damage to the canister or cask. It is important to mention that a cask drop greater than 11 inches implies fuel assembly deceleration greater than 45g and therefore potential fuel damage.

Q. 25: Have you reviewed the cask drop calculations supplied by Holtec?

A. 25: Yes, I have reviewed the cask drop calculations supplied by Holtec, HI-2002572, *Evaluation of the Confinement Integrity of a Loaded Holtec MPC Under a Postulated Drop Event* (Nov. 30, 2000).

Q. 26: What is your opinion of these calculations?

A. 26: Actually there are two cask drop calculations: a 25 foot drop of the HI-TRAC transfer cask containing the fuel canister, and a 10 inch drop of the HI-STORM cask containing the inner fuel canister. Both calculations assume the cask drops vertically downward, from either a 25-foot or 10-inch height, onto a concrete base. My criticism of these calculations is that neither assumes the cask would drop at an angle. If that occurred, the shear stresses, particularly on the welds, would then be considerably more severe than in a vertical drop. The NRC Staff admits that "the SAR drop analysis does not include examination of a corner drop."[21] If the canister experiences a "corner drop," then PFS has not evaluated whether the canister welds would be impaired, exposing the canister contents to the external environment. This issue must be addressed prior to establishing the design basis earthquake.

---

[16] *See* Testimony of Drs. Khan and Ostadan(cask stability).

[17] Id..

[18] *See* PFS Joint Dec. ¶¶ 14, 17.

[19] *See* Testimony of Drs. Khan and Ostadan(cask stability).

[20] *See See* HI-STORM 100 CoC at 5.0-4, State's Exh. 135.

[21] *See* HOLTEC SER at 3-10, State's Exh. 138.

Q. 27: Overall, do you agree that the analyses performed by Holtec and PFS are conservative or bounding?

A. 27: Based on the above, I do not agree that the limited analysis performed by Holtec and PFS is conservative or bounding. In the instances discussed above, the HI-STORM cask would be operated under conditions that are outside the parameters analyzed in the SAR and SER, and would lead to doses at the fence post that exceed regulatory limits. Thus, PFS has not shown that its requested design basis ground motion will not endanger life or property or is otherwise in the public interest as required by 10 CFR § 72.7 or will not jeopardize the health and safety of on-site workers.

Q. 28: One final question, are you aware of the burn-up of fuel stored in the ISFSI at INEEL where the TMI-2 fuel is stored?

A. 28: The TMI-2 reactor went through low power testing for several months and then operated for a 3-month period before the reactor accident. As a result, the fuel burnup was 3,175 MWD/MTU, far less than the potential burnup of fuel that will be accepted at PFS, up to 45,000 MWD/MTU.

Q. 29: Does this conclude your testimony?

A. 29: Yes.

Dr. Marvin Resnikoff

Radioactive Waste Management Associates
526 West 26th Street, Room 517
New York, NY 10001
(212)620-0526 FAX (212)620-0518

241 W. 109th St, Apt. 2A
New York, NY 10025
(212) 663-7117

EXPERIENCE:

April 1989 - present **Senior Associate**, Radioactive Waste Management Associates, management of consulting firm focused on radioactive waste issues, evaluation of nuclear transportation and military and commercial radioactive waste disposal facilities.

1978 - 1981; 1983 - April 1989 **Research Director**, Radioactive Waste Campaign, directed research program for Campaign, including research for all fact sheets and the two books, *Living Without Landfills*, and *Deadly Defense*. The fact sheets dealt with low-level radioactive waste landfills, incineration of radioactive waste, transportation of high-level waste and decommissioning of nuclear reactors. Responsible for fund-raising, budget preparation and project management.

1981 - 1983 **Project Director**, Council on Economic Priorities, directed project which produced the report *The Next Nuclear Gamble*, on transportation and storage of high-level waste.

1974 - 1981 **Instructor**, Rachel Carson College, State University of New York at Buffalo, taught classes on energy and the environment, and conducted research into the economics of recycling of plutonium from irradiated fuel under a grant from the Environmental Protection Agency.

1975 - 1976 **Project Coordinator**, SUNY at Buffalo, New York Public Interest Research Group, assisted students on research projects, including project on waste from decommissioning nuclear reactor.

1973 **Fulbright Fellowship** at the Universidad de Chile, conducting research in elementary particle physics.

1967 - 1972 **Assistant Professor of Physics**, SUNY at Buffalo, conducted research in elementary particle physics and taught range of graduate and undergraduate physics courses.

1965 - 1967 **Research Associate**, Department of Physics, University of Maryland, conducted research into elementary particle physics.

EDUCATION

University of Michigan
Ann Arbor, Michigan

PhD in Physics, June 1965
M.S. in Physics, Jan 1962
B.A. in Physics/Math, June 1959

State's
Exhibit 134

Resume of Marvin Resnikoff, Ph.D.

Dr. Marvin Resnikoff is Senior Associate at Radioactive Waste Management Associates and is an international consultant on radioactive waste management issues. He is Principal Manager at Associates and is Project Director for risk assessment studies on radioactive waste facilities and transportation of radioactive materials. Dr. Resnikoff has concentrated exclusively on radioactive waste issues since 1974. He has conducted studies on the remediation and closure of the leaking Maxey Flats, Kentucky radioactive landfill for Maxey Flats Concerned Citizens, Inc. under a grant from the Environmental Protection Agency, the Wayne and Maywood, New Jersey thorium Superfund sites and on proposed low-level radioactive waste facilities at Martinsville (Illinois), Boyd County (Nebraska), Wake County (North Carolina), Ward Valley (California) and Hudspeth County (Texas). He has conducted studies on transportation accident risks and probabilities for the State of Nevada and dose reconstruction studies of oil pipe cleaners in Mississippi and Louisiana, residents of Canon City, Colorado near a former uranium mill, residents of West Chicago, Illinois near a former thorium processing plant, and residents and former workers at a thorium processing facility in Maywood, New Jersey. In West Chicago he calculated exposures and risks due to thorium contamination and served as an expert witness for plaintiffs A Muzzey, S Bryan, D Schroeder and assisted counsel for plaintiffs KL West and KA West. He is presently serving as an expert witness for a separate group of plaintiffs in West Chicago, including R Dassion. He also evaluated radiation exposures and risks in worker compensation cases involving G Boeni and M Talitsch, former workers at Maywood Chemical Works thorium processing plant.

Under a contract with the State of Utah, Dr. Resnikoff is a technical consultant to DEQ on the proposed dry cask storage facility for high-level waste at Skull Valley, Utah and proposed storage/transportation casks. He is assisting the State on licensing proceedings before the Nuclear Regulatory Commission. In addition, at hearings before state commissions and in federal court, he has investigated proposed dry storage facilities at the Point Beach (WI), Prairie Island (MN) and Palisades (MI) reactors.

In Canada, he has conducted studies on behalf of the Coalition of Environmental Groups and Northwatch for hearings before the Ontario Environmental Assessment Board on issues involving radioactive waste in the nuclear fuel cycle and Elliot Lake tailings and the Interchurch Uranium Coalition in Environmental Impact Statement hearings before a Federal panel regarding the environmental impact of uranium mining in Northern Saskatchewan. He has also worked on behalf of the Morningside Heights Consortium regarding radium-contaminated soil in Malvern and on behalf of Northwatch regarding decommissioning the Elliot Lake tailings area before a FEARO panel. More recently he completed a study for Concerned Citizens of Manitoba regarding transportation of irradiated fuel to a Canadian high-level waste repository.

He was formerly Research Director of the Radioactive Waste Campaign, a public interest organization conducting research and public education on the radioactive waste issue. His duties with the Campaign included directing the research program on low-level commercial and military waste and irradiated nuclear fuel transportation, writing articles, fact sheets and reports, formulating policy and networking with numerous environmental and public interest organizations and the media. He is author of the Campaign's book on "low-level" waste, *Living Without Landfills*, and co-author of the Campaign's book, *Deadly Defense, A Citizen Guide to Military Landfills*.

Between 1981 and 1983, Dr. Resnikoff was a Project Director at the Council on Economic Priorities, a New York-based non-profit research organization, where he authored the 390-page study, *The Next Nuclear Gamble, Transportation and Storage of Nuclear Waste*. The CEP study details the hazard of transporting irradiated nuclear fuel and outlines safer options.

In February 1976, assisted by four engineering students at State University of New York at Buffalo, Dr. Resnikoff authored a paper that changed the direction of power reactor decommissioning in the United States. His paper showed that power reactors could not be entombed for long enough periods to allow the radioactivity to decay to safe enough levels for unrestricted release. The presence of long-lived radionuclides meant that large volumes of dismantled reactors would still have to go to low-level waste disposal facilities. He has assisted public interest groups NECNP and CAN on the decommissioning of the Yankee-Rowe reactor.

Dr. Resnikoff is an international expert in nuclear waste management, and has testified often before State Legislatures and the U.S. Congress. He has extensively investigated the safety of the West Valley, New York and Barnwell, South Carolina nuclear fuel reprocessing facilities. His paper on reprocessing economics (Environment, July/August, 1975) was the first to show the marginal economics of recycling plutonium. He completed a more detailed study on the same subject for the Environmental Protection Agency, "Cost/Benefits of U/Pu Recycle," in 1983. His paper on decommissioning nuclear reactors (Environment, December, 1976) was the first to show that reactors would remain radioactive for hundreds of thousands of years.

Dr. Resnikoff has prepared reports on incineration of radioactive materials, transportation of irradiated fuel and plutonium, reprocessing, and management of low-level radioactive waste. He has served as an expert witness in state and federal court cases and agency proceedings. He has served as a consultant to the State of Kansas on low-level waste management, to the Town of Wayne, New Jersey, in reviewing the cleanup of a local thorium waste dump, to WARD on disposal of radium wastes in Vernon, New Jersey, to the Southwest Research and Information Center and New Mexico Attorney General on shipments of plutonium-contaminated waste to the WIPP facility in

New Mexico and the State of Utah on nuclear fuel transport. He has served as a consultant to the New York Attorney General on air shipments of plutonium through New York's Kennedy Airport, and transport of irradiated fuel through New York City, and to the Illinois Attorney General on the expansion of the spent fuel pools at the Morris Operation and the Zion reactor, to the Idaho Attorney General on the transportation of irradiated submarine fuel to the INEL facility in Idaho and to the Alaska Attorney General on shipments of plutonium through Alaska. He was an invited speaker at the 1976 Canadian meeting of the American Nuclear Society to discuss the risk of transporting plutonium by air. As part of an international team of experts for the State of Lower Saxony, the Gorleben International Review, he reviewed the plans of the nuclear industry to locate a reprocessing and waste disposal operation at Gorleben, West Germany. He presented evidence at the Sizewell B Inquiry on behalf of the Town and Country Planning Association (England) on transporting nuclear fuel through London. In July and August 1989, he was an invited guest of Japanese public interest groups, Fishermen's Cooperatives and the Japanese Congress Against A- and H- Bombs (Gensuikin).

Between 1974 and 1981, he was a lecturer at Rachel Carson College, an undergraduate environmental studies division of the State University of New York at Buffalo, where he taught energy and environmental courses. The years 1975-1977 he also worked for the New York Public Interest Group (NYPIRG).

In 1973, Dr. Resnikoff was a Fulbright lecturer in particle physics at the Universidad de Chile in Santiago, Chile. From 1967 to 1973, he was an Assistant Professor of Physics at the State University of New York at Buffalo. He has written numerous papers in particle physics, under grants from the National Science Foundation. He is a 1965 graduate of the University of Michigan with a Doctor of Philosophy in Theoretical Physics, specializing in group theory and particle physics.

Marvin Resnikoff, PhD: Court Proceedings

Case	*Official Title, Case #, Court*	*Plaintiff Attorney*	*Nature of case*
Adams v Dupont	Anita Adams et al v EI DuPont DeNemours et al, US District Court, San Antonio, SA-98-CA-816(EP)	John Emerson Whittington, vonSternberg, Emerson 2600 S Gessner, Ste 600 Houston, TX 77063 713.789.8850	Personal injury, U mining and milling
Boeni v Stepan Chemical	George L. Boeni v. Stepan Chemical Co.; heard before Div. Of Worker's Compensation, Hackensack, NJ	David Tykulsker Ball, Livingston & Tykulsker 108 Washington St. Newark, NJ 07102 (201) 622-4545	Worker's comp; thorium processing plant
Carey v Kerr-McGee	Jesse Carey et al v Kerr-McGee Corp, US District Ct, Northern Dist of Illinois, No. 96-C-8583	Paul Weiss Hagens & Berman 1301 Fifth Ave, Ste 2900 Seattle, WA 98101 206.623.7292	Personal injury, thorium processing plant
Case v Chevron	James Edward Case et al. v. Chevron USA, Inc. et al.; USDC; Southern Miss.; Jackson District; Case #J92-0269(W)(N)	Stuart Smith Sacks & Smith 1615 Poydras St. Suite 860 New Orleans, LA 70112 (504) 593-9600	Dose reconstruction: oil pipe scale
Davis v Transelco		Stuart Smith Sacks & Smith 1615 Poydras St. Suite 860 New Orleans, LA 70112 (504) 593-9600	Thorium lens polishing compound; occupational exposure
Dodge v Comm Edison	Joseph Dodge et al. v. The Cotter Corporation and Commonwealth Edison Co.; USDC; District of Colorado; Civil Action #91 Z 1861	Rebecca Lorenz Melat, Pressman, Ezell & Higbie 711 South Tejon St. Colorado Springs, CO, 80903-4059 (719) 475-0304	Dose reconstruction; Cotter U mill
Ferguson v Ashland	Victor Ferguson v Ashland Inc, et al Johnson Circuit Court, KY, No. 95-CI-124	Pete Petroski, Esq Fleming, Hovenkamp & Grayson 1330 Post Oak Blvd Houston, TX 77056-3019	NORM contamination

Updated March 15, 2001

Case	*Official Title, Case #, Court*	*Plaintiff Attorney*	*Nature of case*
Garza v Wyoming Minerals	A Garza et al v Westinghouse Electric Corporation, et al; UDC in Corpus Christi, TX; Civ No. C-95-505	Andrew Schirrmeister, 2603 Augusta, Ste 1200, Houston, TX 77057; (713) 781-0771	Personal injury case re. birth defects; parents involved in U mining
Kenny v Shore Regional	Robt Kenny v Shore Regional Bd of Ed, *et al* Superior Court of NJ, Monmouth Cty, MON-L-6617-93	Norm Hobbie, Esq, Giordano, Halleran, 125 Half Mile Rd, Middletown, NJ 07748, (732)219-5484	Personal injury, industrial accident
Idaho AG	Public Service Co. of Colorado v. Cecil D Andrus, individually and as the Governor of the State of Idaho; UDC; District of Idaho; Civil # 91-0035-S-HLR	Larry Echohawk, Atty General State of Idaho; Office of the Attorney General; Boise, ID 83720-1000 (208) 234-2400	Federal Court proceedings re. impact of transportation of sub fuel to Idaho
Kennedy v Southern California Edison	Joe Kennedy and Ellen Marie Kennedy v Southern California Edison, et al US District Ct, Southern District of California Case No. 95-3769 J (RBB)	Katy Jacobs, Esq Howarth & Smith 700 So Flower St, Suite 2900 Los Angeles, CA 90017 213/955-9400	Personal injury at nuclear reactor
Longoria v URI	Manuel T Longoria et al. v. Uranium Resources, Inc. et al.; in the District Court of Duval County, 229th Judicial District, Texas; Cause # 16264	Ricardo DeAnda DeAnda Law Firm Plaza de San Augustin 212 Flores Ave. Laredo, TX 78040 (210) 726-0038	Property damage due to uranium solution mining
Marina Owners v MM	Euchee Marina & Campground, et al. v. Martin Marietta Energy System, Inc. et al.; USDC; East Dist; North Div; Docket: CIV-3-91-0510	Wm Vines; Butler, Vines & Babb; Suite 810, First American Center; Knoxville, TN 37901-2649; (423) 637-3531	Federal Court proceedings, potential risk from Oak Ridge releases
Miller v Tetra Pak, et al	Miller v Tetra Pak, et al, District Court, Denton County, Texas	Greg Marks, Esq 1333 Corporate Dr, Suite 209, Irving, TX 75038, (972) 790-4400	Personal injury, electron beam machine
Radioactive Waste Campaign		Louie Roselle; Waite, Schneider, Bayless & Chesley; 1513 Central Trust Tower, Cincinnati, OH 45202; (513) 621-0267	Fernald U exposures and radiation doses to nearby residents
Rock v Southern California Edison	Joshua Rock v Southern California Edison, et al, Case No. 95-3821 J (RBB)	Katy Jacobs, Esq Howarth and Smith 700 S Flower St. Suite 2900 Los Angeles, CA 90017-4216	Personal injury, 'hot' particles
Talitsch v Stepan Chemical	Mathis Talitsch v. Stepan Chemical Co.; Division of Worker's Compensation, Hackensack, NJ; C.P. #91-001800	David Tykulsker Ball, Livingston & Tykulsker 108 Washington St. Newark, NJ 07102 (201) 622-4545	Worker's comp; thorium processing plant

Case	*Official Title, Case #, Court*	*Plaintiff Attorney*	*Nature of case*
Vercher v Intracoastal		Stuart Smith; Sacks & Smith; 1615 Poydras St. Suite 860; New Orleans, LA 70112; (504) 593-9600	Oil pipe scale
West v Kerr-McGee	Kristy Lee West et al. v. Kerr-McGee Chemical Corp.; in USDC; Northern District of Illinois; Eastern Div.; Case # 92 C 4211	Thomas Trinley Patrick J Kenneally Ltd 2 North LaSalle St. Suite 1950 Chicago. ILL 60602 (312) 236-2522	Personal injury cases re. KMCC thorium
Muzzey/Bryan v Kerr-McGee	Gary Muzzey et al. v. Kerr-McGee Chemical Corp.; in USDC; Northern District of Illinois; Eastern Div.; Case # 93 C 3623	Thomas Trinley Patrick J Kenneally Ltd 2 North LaSalle St. Suite 1950 Chicago. ILL 60602 (312) 236-2522	Personal injury cases re. KMCC thorium

Publications of Marvin Resnikoff, PhD, 1985/2000

January 1985 "U.S. Radioactive Landfill Experience," paper, presented to the Annual Institute of British Geographers in Leeds, England. Incorporated into Nuclear Power in Crisis: Politics and Planning for the Nuclear State, edited by Blowers, A., and Pepper, D., Nichols Publishing Co. (1987).

February 1985 "Comments on the transportation sections of the draft Environmental Assessment for a high level waste repository in Utah," prepared for the State of Utah.

March 1985 Testimony before the House Committee on Interior and Insular Affairs on the long-lived hazard of "low-level" radioactive waste.

May 1985 "Radioactive Waste Incineration in Bladen County, What's Coming Out of the Stack?," Campaign report on the environmental impact of incinerating radioactive waste in Bladen County, North Carolina, 33 pages.

August 1985 Paper submitted to the House Energy and Commerce Committee, on the hazard of long-lived low-level waste.

September 1985 "Radioactive Waste Incineration in Parks Township, Pennsylvania, What's Coming Out of the Stack?," Campaign report on the environmental impact of incinerating radioactive waste in Parks Township, Pennsylvania, 32 pages.

September 2, 1985 "Critique of Submission by Dr. John Till to the House Committee on Interior and Insular Affairs."

February 1986 "Alternatives to Radioactive Landfills, An Environmental Perspective," paper presented at the International Symposium on Alternatives to Radioactive Landfills, Chicago, Ill. Paper incorporated in the Proceedings, published by the Illinois Dept of Nuclear Safety and the Central Midwest Compact Commission.

June 1986 "Feed Materials Production Center, Uranium Contamination of Off-site Wells," Campaign report prepared with Dana Coyle on the health impact of uranium contamination of off-site wells, 35 pages.

June 1986 Testimony on behalf of the Northwest Inland Waters Coalition, a public interest organization, for the Federal District Court, State of Washington, on the need for an Environmental Impact Statement to evaluate the import of irradiated nuclear fuel from Taiwan through the Port of Seattle.

July 1986 Paper on the Kerr-McGee uranium conversion facility near Salisaw, Oklahoma presented to conference organized by Native Americans for a Clean Environment.

September 1986 Affidavit in Federal Court in New York City on behalf of a Warwick, New York public interest group (WARD), and in New Jersey State Courts on behalf of the New York-New Jersey Trails Conference, opposing plans by the New Jersey Department of Environmental Protection to move radium residues from Montclair to Vernon, New Jersey.

August, 1986 Supplement to June 1986 Campaign report on the Feed Materials Production Center discussing contamination of public water supplies, 20 pages.

September 1986 "Disposal of high-level waste in Canada," paper presented at high-level waste conference, Winnipeg, Manitoba. Workshop on the transportation of irradiated fuel in Canada.

Incorporated into Challenges to Nuclear Waste, Proceedings of Nuclear Waste Issues Conference, Sept 12-14, 1986, edited by Weiser, A., Concerned Citizens of Manitoba, Winnipeg, Manitoba, 1987.

November 1986 Associates report to the State of Kansas on draft Request for Proposal for contractor to the Central States Compact, 10 pages.

November 1986 "Transportation of irradiated fuel," paper presented to a subcommittee of the National Association of Attorneys General, Las Vegas, Nevada.

December 1986 Associates affidavit prepared for the Coalition on West Valley Nuclear Wastes and the Radioactive Waste Campaign in a successful U.S. District Court action on the need for a federal Environmental Impact Statement before disposing of low-level waste at West Valley.

February 1987 "Off-site radioactive contamination at DOE's Oak Ridge, Tennessee facility," Campaign report prepared with Dana Coyle on radioactive leakage from the Oak Ridge Reservation, 65 pages.

May 1987 "At-reactor storage of irradiated fuel," paper presented at conference sponsored by Blue Ridge Environmental Defense League and other citizen organizations at Maryville, Tennessee.

June 1987 Associates affidavit prepared for the Sierra Club Legal Defense Fund on the need for an Environmental Impact Statement before incinerating plutonium-contaminated waste at the Rocky Flats Plant.

September 1987 *Living Without Landfills*, Campaign book on the hazard of radioactive landfills, and safer alternatives, 119 pages.

September 1987 Associates affidavit prepared on behalf of the Alaska Attorney General in a U.S. District Court action on the need for a federal Environmental Impact Statement for air shipments of plutonium in Alaska.

November 1987 "Low-level waste in Michigan," talk before a joint session of the Michigan Legislature, East Lansing, Michigan.

February 1988 Testimony before the Vermont House Committee on Natural Resources and the Environment, Montpelier, Vermont.

May 1988 Talks at Chadron State College (Chadron, Neb), Alliance, and Scottsbluff, on "low-level" waste in the Central States.

June 1988 Co-authored the Radioactive Waste Campaign's *Deadly Defense*, 170 page book on radioactive waste at nuclear weapons facilities. Released at a national press conference in Washington, D.C.

June 1988 Wayne and Clark counties, Illinois; public meetings near proposed llw dump sites; jointly-sponsored with local groups (Individuals for a Clean Environment)

July 1988 Briefing before Congressional Legislative Assistants on the findings of *Deadly Defense*, jointly conducted with the Sierra Club and sponsored by Representative Don Bonker.

September 1988 Reno, NV; talk before Northern Colorado Gaming Executives re. transportation of irradiated fuel to a proposed high-level waste repository; jointly sponsored by Citizens

Alert and State of Nevada

September 1988 "Rebuttal of NRC Critique of *Living Without Landfills*, 12 pages.

October 1988 Boulder, CO; talk, participation in conference and chapter of book, Environmental Impacts of Warfare; sponsored by the Sierra Club.

November 1988 Nucla, CO; prepared testimony before Colorado Department of Health re. suitability of proposed "low-level" waste disposal site in Uravan, Colorado on behalf of Western Colorado Congress.

November 1988 Augusta, Maine; participation in debate sponsored by the Maine Low-Level Radioactive Waste Authority

December 1988 Preparation of court affidavit re. proposed irradiated fuel shipments from Taiwan through Portsmouth, Virginia, before the United States District Court, District Of Columbia, on behalf of the Sierra Club Legal Defense Fund.

February 1989 "Uranium Releases at Fernald, Radiation Doses to Nearby Residents," report released by the Radioactive Waste Campaign at Cincinnati, Ohio press conference.

April 1989 "Risks of Low-Level Radioactive Waste Transportation," 8-page fact sheet, prepared for the Radioactive Waste Campaign.

May 12, 1989, "Preliminary Report on RI/FS Study," prepared on behalf of Maxey Flats Concerned Citizens, Flemingsburg, Kentucky.

August 30, 1989, "Analysis of RADTRAN Computer Model," paper presented at meeting of the American Nuclear Society Meeting, Las Vegas, Nevada.

October 1989 "Report on Maxey Flats Remediation Program," 75-page report, prepared for Maxey Flats Concerned Citizens, Inc.

November 1989 "RADTRAN Analysis," 60-page report on the probability and consequences of accidents in transporting high-level waste to the proposed Yucca Mountain repository, prepared for the University of Nevada, Las Vegas.

February 1990 "Radioactive Waste Mismanagement at Nine Mile Point 1."

April 9, 1990 "Comments on the Final Supplemental Environmental Impact Statement, Waste Isolation Pilot Plant," on behalf of Concerned Citizens for Nuclear Safety, Santa Fe, New Mexico.

April 25, 1990, talk before the Hazardous Materials/Nuclear Symposium on nuclear transportation issues, Ely, Nevada.

April 26, 1990, Statement before the Nevada Commission on Nuclear Projects on nuclear transportation issues, Las Vegas, Nevada.

July 19, 1990, "Report on Feasibility Study, Risk Assessment, App. D, iodine hazard," prepared on behalf of Maxey Flats Concerned Citizens, Flemingsburg, Kentucky.

August 1, 1990, "Report on the State of Kentucky, Maxey Flats Closure Plan," prepared on behalf

of Maxey Flats Concerned Citizens, Flemingsburg, Kentucky.

August 1990 Preparation of second court affidavit re. proposed irradiated fuel shipments from Taiwan through Portsmouth, Virginia, before the United States District Court, District Of Columbia, on behalf of the Sierra Club Legal Defense Fund.

September 1990 "The Generation Time-Bomb: Radioactive and Chemical Defense Wastes," in *Hidden Dangers, Environmental Consequences of Preparing for War*, edited by AH Ehrlich and JW Birks, Sierra Club Books, San Francisco

October 22, 1990, "Review of Environmental Report for the Central Interstate Compact Low-level Radioactive Waste Facility," on behalf of Heartland Operation to Protect the Environment, Auburn, Nebraska.

December 1990 Declaration re. the constitutionality of the Low-Level Radioactive Waste Policy Act before the U.S. District Court, District of Nebraska on behalf of Concerned Citizens of Nebraska.

December 1990 Preparation of third court affidavit re. proposed irradiated fuel shipments from Taiwan through Portsmouth, Virginia, before the United States District Court, District Of Columbia, on behalf of the Sierra Club Legal Defense Fund.

February 8, 1991, "Review of 'Risk Assessment and Safety Analysis, University of Michigan Waste Handling Facility,'" on behalf of No. Campus Residents Council, Ann Arbor, Michigan.

April 1, 1991, "Health and Safety Impact of NMI," on behalf of Citizens Concerned About NMI, Concord, Massachusetts.

May 6, 1991, "Comments on Final Environmental Impact Statement, Prairie Island Independent Spent Fuel Storage Installation," on behalf of the Sioux Tribal Council, Red Wing, Minnesota.

May 16, 1991, "Managing Low-Level Radioactive Waste," talk at Future Options Symposium, International Institute for Low Level Radioactive Waste, East Lansing, Michigan.

May 23, 1991, "Radiac Accident Analysis," prepared on behalf of the Radioactive Waste Campaign, Brooklyn, New York.

May 30, 1991, "Nuclear Power in the United States," talk sponsored by the Green Party, Rikstag, Green Party Group Room, Stockholm, Sweden.

June 20, 1991, "Comments on the Department of Energy Environmental Assessment on Off-Site Fuels Policy," prepared on behalf of the Sierra Club Legal Defense Fund, Washington, D.C.

July 1, 1991, "Comments on the Final Environmental Impact Statement for the Proposed Ward Valley Low-level Waste Landfill," submitted to the California Department of Health Services, on behalf of Don't Waste California.

July 12, 1991, "Comments on EPA Proposed Plan," prepared on behalf of Maxey Flats Concerned Citizens, Flemingsburg, Kentucky.

September 8, 1991 Preparation of 4th court affidavit re. proposed irradiated fuel shipments from Taiwan through Portsmouth, Virginia, before the United States District Court, District Of Columbia, on behalf of the Sierra Club Legal Defense Fund.

September 20, 1991 "Consequences of a Severe HEU Ship Accident," memo to Greenpeace

September 30, 1991 "Prairie Island Independent Spent Fuel Storage Facility, Cost and Radiation Analysis," before the Minnesota Public Utility Commission on behalf of the Prairie Island Mdewakanton Sioux Indian Community.

October 23, 1991 "Health and Safety Impacts of NMI, 2nd Report," prepared on behalf of Concerned Citizens about NMI, Concord, Mass.

October 31, 1991 Preparation of 5th court affidavit re. proposed irradiated fuel shipments from Taiwan through Portsmouth, Virginia, before the United States District Court, District Of Columbia, on behalf of the Sierra Club Legal Defense Fund.

November 4, 1991 Statement before the City of Albuquerque Common Council regarding disposal of radioactive waste into the city sewer system.

November 9, 1991 Affidavit re. shipments of Pu-contaminated waste to the proposed WIPP facility, before the US District Court, District of Columbia, on behalf of the New Mexico Attorney General.

November 1991 "Prairie Island Independent Spent Fuel Storage Facility, Prefiled Reply Testimony," before the Minnesota Public Utility Commission on behalf of the Prairie Island Mdewakanton Sioux Indian Community.

RWMA, White Paper #1, *Sources of Low-Level Waste in Connecticut*, prepared on behalf of the Towns of East Windsor, Ellington and South Windsor, September 30, 1991.

RWMA, White Paper #2, *Low-Level Waste Transportation in Connecticut*, prepared on behalf of the Towns of East Windsor, Ellington and South Windsor, October 2, 1991.

RWMA, White Paper #3, *Statement by Dr. Marvin Resnikoff on Chem-Nuclear*, prepared on behalf of the Towns of East Windsor, Ellington and South Windsor, October 29, 1991.

RWMA, White Paper #4, *Leakage From Existing 'Low-Level' Waste Disposal Facilities*, prepared on behalf of the Towns of East Windsor, Ellington and South Windsor, January 6, 1992.

Marvin Resnikoff and Anne Vanrenterghem, *Preliminary Review of US Ecology Safety Analysis Report, Proposed Boyd County, Nebraska Low-Level Waste Facility,* prepared on behalf of the Boyd County Local Monitoring Committee, February 2, 1992.

Marvin Resnikoff, *Radon Releases from Uranium Tailings and Projected Health Effects*, prepared on behalf of Northwatch Coalition, February 17, 1992.

RWMA, White Paper #5, *Storage of Low-Level Radioactive Waste,* prepared on behalf of the Towns of East Windsor, Ellington and South Windsor, February 19, 1992.

Marvin Resnikoff, *Scope: McArthur River and Cigar Lake Projects*, Memo to Inter Church Uranium Council, February 27, 1992.

Richard Leigh, Marvin Resnikoff and Anne Vanrenterghem, *Environmental Impacts of Elliot Lake Mill Tailings*, prepared on behalf of Northwatch Coalition, March 30, 1992.

Marvin Resnikoff, *Canadian High-Level Waste Repository Costs*, Memo to David Poch and David Argue, Coalition of Environmental Groups, April 2, 1992.

Minard Hamilton, *Low Level Waste Facilities in Canada and the U.S.*, prepared on behalf of Northwatch Coalition, April 22, 1992.

Marvin Resnikoff, *Comment on Midwest Joint Venture EIS*, Memo to Inter Church Uranium Council, April 23, 1992.

Benjamin A. Goldman, *Review of Environmental Report Social and Economic Impact Assessments: Proposed Low-Level Radioactive Waste Disposal Facility*, prepared on behalf of Northwatch Coalition, June 25, 1992.

Lee DiTullio and Marvin Resnikoff, *Review of Safety Analysis Report Part 1: Geology, Hydrology Proposed Low-Level Waste Facility Butte, Nebraska*, prepared on behalf of the Boyd County Local Monitoring Committee, June 29, 1992.

Marvin Resnikoff, Mythbuster#8, *"Low-Level" Radioactive Waste*, for Safe Energy Communications Council, Summer 1992.

Marvin Resnikoff, *Comments on Final Guidelines for the Preparation of an Environmental Impact Statement on the Nuclear Fuel Waste Management and Disposal Concept*, July 22, 1992.

Marvin Resnikoff, *NMI's Proposed Hydromet Project*, Memo to Judy Scotnicki, Concerned Citizens of Concord, July 29, 1992.

Marvin Resnikoff and Lee DiTullio, *Review of Safety Analysis Report Part 2: Risk Assessment Proposed Low-Level Waste Facility Butte, Nebraska*, prepared on behalf of the Boyd County Local Monitoring Committee, August 7, 1992.

RWMA, *Comments on McClean Lake Project EIS*, prepared on behalf of the Inter-Uranium Coalition, June 30, 1992.

Lee DiTullio and Karen Levine, *Comments on Cluff Lake EIS*, prepared on behalf of the Inter-Church Uranium Coalition, July 20, 1992.

Marvin Resnikoff, *Plutonium Ship Akatsuki Maru Consequences of Fire at the Pearl Harbor Naval Shipyard*, prepared on behalf of Greenpeace, August 24, 1992.

Marvin Resnikoff, *Waste Impacts of the Nuclear Fuel Cycle*, prepared on behalf of Coalition of Environmental Groups, November 1992.

Marvin Resnikoff, Declarations on the safety of shipping naval fuel from shipyards to Idaho before the Federal District Court, prepared on behalf of the Idaho Attorney General, March 1993.

Marvin Resnikoff, Declaration on the safety of the VSC-24 storage cask before the Federal District Court on behalf of the Lake Michigan Federation, May 1993.

Marvin Resnikoff, Talk at a Town Meeting in Grand Rapids, Michigan, June 22, regarding the safety of the VSC-24 storage container at the Palisades reactor.

Marvin Resnikoff, Reports to two environmental assessment panels reviewing the environmental impact of proposed mining operations in Northern Saskatchewan, prepared on behalf of the Interchurch Uranium Coalition, May 12 and June 14, 1993.

Marvin Resnikoff, Presentation before the Ohio Governor's Blue Ribbon Committee on siting a low-level waste facility in Ohio for the Midwest Compact, July 1993.

Marvin Resnikoff, Report on the safety of processing and storing radium-contaminated wastes in the Tapscott district of Scarborough, Toronto, prepared on behalf of the Coalition Against Radioactive Tapscott, November 1, 1993.

Marvin Resnikoff, Remarks before the Department of Energy meeting on the Multi-Purpose Canister, Washington, D.C., November 16, 1993.

Marvin Resnikoff, Report on the scoping guidelines for production of an Environmental Impact Statement (EIS) for decommissioning of the Elliot Lake uranium tailings and report on the draft EIS by Rio Algom for the decommissioning of Quirk and Panel tailings, Elliot Lake, prepared on behalf of Algoma-Manitoulin Nuclear Awareness, December 15, 1993.

Resnikoff, M and Haaker, R, "Estimated Radiation Dose received by James E Case, et al, during Pipe De-scaling Operations at Brookhaven, Mississippi," report prepared in the case Case v. Chevron, January 23, 1994.

Radioactive Waste Management Associates, "Soil Separation: What It Means For Wayne," report prepared for the Town of Wayne, New Jersey, May 24, 1994.

Resnikoff, M and Fuchsman, P, "Comments on the Department of Energy's Baseline Risk Assessment for the Wayne Site, Wayne, New Jersey, January 1994," May 31, 1994.

Resnikoff, M, "Radiation Dose Exposures Received by William Davis During Lens Polishing Operation," report prepared for the case Davis v Transelco *et al*, July 1, 1994.

Leigh, RL and Resnikoff, M, "Estimated Exposure to Radiation and Metals Received by Lincoln Park Residents from Cotter Mill Operations," report prepared for the case J Dodge *et al* v. Commonwealth Edison, July 1, 1994.

Resnikoff, M, Affidavit prepared on behalf of plaintiffs in the United States District Court for the Eastern District Of Tennessee at Knoxville, Euchee Marina & Campground, Inc. *et al,* plaintiffs, v Union Carbide Corporation, *et al*, defendants, July 15, 1994.

Resnikoff, M and Knowlton, K, "Preliminary Critique of the Safety Analysis Report, Wake/Chatham Proposed Low-Level Waste Facility," report prepared for the Chatham County Preferred Site Local Advisory Committee, July 19, 1994.

Resnikoff, M, Leigh, RL and Fuchsman, P, "Comments on the Department of Energy's Baseline Risk Assessment for the Maywood Site, Maywood, New Jersey, April 1993, July 27, 1994.

Resnikoff, M, "Prefiled Testimony Of Marvin Resnikoff, Ph.D. On Behalf of Lake Michigan Federation, before the Public Service Commission of Wisconsin, in the case of Application of Wisconsin Electric Power Company for Authority to Construct and Place in Operation an Independent Spent Fuel Storage Facility Utilizing Dry Cask Storage Technology at the Point Beach Nuclear Plant Located in the Town of Two Creeks, Manitowoc County, September 11,

1994. Also Rebuttal Testimony, dated September 27, 1994 and Supplemental Testimony, dated October 3, 1994.

Resnikoff, M, affidavit prepared on behalf of plaintiffs in the United States District Court for the District of Massachusetts, Citizens Awareness Network, Inc., plaintiff, v. United States Nuclear Regulatory Commission, defendant, October 4, 1994.

Resnikoff, M, affidavit in opposition to motion of Westinghouse for summary judgment, prepared on behalf of plaintiffs in the United States District Court for the Western District Of Washington at Yakima, in re Hanford Nuclear Reservation, October 15, 1994.

RWMA, "Comments on proposed rule change: radiation standards for low-level waste facilities," January 9, 1995.*

Resnikoff, M, "Nuclear waste transportation and the role of the public, Las Vegas, Nevada: unresolved safety issues," February 1, 1995.

Resnikoff, M, "Ohio low-level waste legislation," Lobby Day, Ohio Environmental Council Border Opposes Nuclear Dump, February 22, 1995.

Fuchsman, P, Hamilton, M, Knowlton, K, Levine, K and Resnikoff, M, "Wayne Health Survey," prepared for the Town of Wayne, April, 1995.

Knowlton, K and Resnikiff, M, "A review of the phase II field investigation and financial resources of NMI," report prepared for CREW, May 22, 1995.

Resnikoff, M, Knowlton, K and Fuchsman, P, "Low-level waste transportation in Texas," prepared for Alert Citizens for Environmental Safety, June 29, 1995.

Resnikoff, M and Knowlton, K, "Comments on 'Engineering evaluation/cost analysis for the cleanup of residential and municipal vicinity properties at the Maywood site, Bergen County, New Jersey,'" on behalf of Concerned Citizens of Maywood, August 11, 1995.

Resnikoff, M, Fuchsman, P and Knowlton, K, "Low-level waste transportation in North Carolina," prepared for the Chatham County Preferred Site Local Advisory Committee, August 13, 1995.

Edelman, G and Fuchsman, P, "Business survey on the Socioeconomic impact of the low-level radioactive waste facility proposed for North Carolina," prepared for Chatham County, August, 16, 1995

Resnikoff, M, Knowlton, K and Fuchsman, P, "Comments on environmental impact statements for the Cigar Lake and Midwest Joint Ventures proposals," prepared for the Saskatchewan Uranium Coalition, October, 27, 1995.

Knowlton, K, Resnikoff, M and Fuchsman, P, "Review of the license application for the Proposed LLRW facility near Sierra Blanca, Texas," prepared for Alert Citizens for Environmental Safety, November 30, 1995.

Resnikoff, M, "Scoping Comments for the Environmental Impact Statement for a Geologic Repository for the Disposal of Spent Nuclear Fuel and High-Level Radioactive Waste at Yucca

* All publications in the year 1995 have appeared since my Rule 26 statement in the Muzzey/Bryan case.

Mountain, Nye, County, Nevada," December 1, 1995.

Resnikoff, M, "Calculation of Radiation Exposures Received by Donald A. Schroeder, West Chicago, Illinois." Prepared for case Schroeder v Kerr-McGee, February 21,1996.

Resnikoff, M, K Knowlton, and K Island, "Comments on Environmental Impact Statement for the McArthur River Proposal," prepared for Saskatchewan Uranium Coalition, March 1, 1996.

Resnikoff, M, K Knowlton, P Fuchsman, and K Island, "Site Suitability and Impact of Proposed Radioactive Waste Facility, Wake/Chatham Counties, North Carolina," prepared for Chatham County Preferred Site Local Advisory Committee, March 20, 1996.

Resnikoff, M, "Mississippi Oil and Gas Board Proposed Rule 69: Control of Oilfield NORM," March 25, 1996.

Resnikoff, M, "Before the Illinois LLRW Task Group: Comments on Revised Siting Criteria," May 15, 1996.

Resnikoff, M, "Decommissioning of Big Rock Point," prepared for Don't Waste Michigan - Northern Chapter, July 1996.

Resnikoff, M, "Preliminary Report: Occupational Exposures for Plaintiffs Garza and Depain from Uranium Solution Mining Activities, Bruni, Texas" July 31,1996.

RWMA, "Comments on the IRP Remedial Design Work Plan, Maxey Flats, Kentucky," prepared for Maxey Flats Concerned Citizens, August 2, 1996.

Knowlton K, and M Resnikoff, "Review of the License Application for the Proposed LLRW Facility Near Sierra Blanca, Texas," prepared for Sierra Blanca Legal Defense Fund, updated August 14, 1996, (original November 30,1995).

Resnikoff, M, "Radiation Dose Exposure Received by Milt Vercher During Oil Pipe Cleaning Operations," September 27, 1996.

Resnikoff, M, and K Knowlton, "Comments on the Department of Energy's Draft Feasibility Study and Draft Record of Decision for the Wayne, New Jersey Site," October 15, 1996.

Knowlton, K, M Medina, and M Resnikoff, "Comments on Fall 1996 *Addendums* to JEB Pit Tailings Disposal Plans," prepared on the behalf of the Saskatchewan Uranium Coalition, December 2,1996.

Resnikoff, M., "Critique of Ontario Hydro Irradiated Fuel Transportation Assessment," prepared on behalf of Concerned Citizens of Manitoba, January 1997.

Resnikoff, M., and Waligora, Jr., S.J., "Estimated Radiation Doses Received by Victor Ferguson, Martha, Kentucky," report prepared in the case, Victor Ferguson v. Ashland Oil Inc, et al, January 16, 1997.

Resnikoff, M., "Preliminary Report, Environmental and Health-Related Impacts of the Mobil Mining and Minerals Phosphoric Acid Plant," February 1997.

Resnikoff, M., Affidavit, before the Court of Appeals, State of Minnesota, re. appeal of decision by the Minnesota Dept of Environmental Quality, Prairie Island dry storage facility, on behalf of the Prairie Island Indian Community, February 1997.

Resnikoff, M., "Comments on Environmental Management Advisory Board Report 'Report of a Stakeholder Process to Develop Guiding Principles for the Formerly Utilized Sites Remedial Action Project, Background Information Document, " prepared on behalf of Thorium Advisory Committee, Town of Wayne, New Jersey, March 7, 1997.

Resnikoff, M., "Comments On 'Report of a Stakeholder Process to Develop Guiding Principles for the Formerly Utilized Sites Remedial Action Project,' Background Information Document, " prepared on behalf of Concerned Citizens of Maywood, Maywood, New Jersey, March 12, 1997.

Resnikoff, M., "Wastes Generated in Decontaminating and Decommissioning a Nuclear Power Plant," presented at the conference, "Impacts of Nuclear Power and Nuclear Waste on Indigenous and Local Communities, Prairie Island Indian Community," March 25-26, 1997.

Resnikoff, M, "Radiation Risks on Reclaimed Phosphate Mined Lands," April 30, 1997.

Resnikoff, M., andChampion, A., "Transportation Accident of Ship Carrying Vitrified High-Level Radioactive Waste: Part 1. Impact on the Federated States of Micronesia," prepared on behalf of Greenpeace Pacific, July 31, 1997.

Resnikoff, M., Prepared statement before the Joint Federal-Provincial Panel on U Mining on Cigar Lake and McArthur River, on behalf of the Saskatchewan Uranium Coalition, August 27, 1997.

Resnikoff, M., "Comments on the Department of Energy's Draft Feasibility Study and Draft Record of Decision for the Wayne Site," prepared on behalf of Thorium Advisory Committee, Town of Wayne, New Jersey, September 15, 1997.

Resnikoff, M., "Presentation Before Panel on the Waste Management System, Nuclear Waste Technical Review Board," November 19, 1997.

Fairlie, I., and Resnikoff, M., *No Dose Too Low*, article prepared for The Bulletin of The Atomic Scientists, November/December 1997.

Resnikoff, M., "Proposed Remediation Work at Maxey Flats," prepared on behalf of Maxey Flats Concerned Citizens, January 1998.

Resnikoff, M., "Comments on the draft Safety Evaluation Report (DSER), Proposed LLRW Facility at Boyd County, Nebraska by the State of Nebraska," prepared on behalf of the Boyd County Monitoring Committee, February 1998.

Resnikoff, M., "Comments on Draft PEIS for the Long-Term Management and Use of Depleted Uranium Hexafluoride," March 1998.

Resnikoff, M., Affidavit prepared for Giordano, Halleran, & Ciesla, in the case *Robert Kenny v. Shore Regional High School, et al.*, March 1998.

Resnikoff, M., "Comments on FGR No. 13, Part I " Interim Version, Health Risks from Low-Level Environmental Exposure to Radionuclides," June 30, 1998.

Resnikoff, M., "Radioactive Waste Trends," prepared on behalf of the Chatham County PSLAC,

June 1998.

Resnikoff, M., and Tsui, C., "Health Impacts of Oil Production in Brookhaven, Mississippi," prepared on behalf of Sacks & Smith, July 30, 1998.

Resnikoff, M., "Comments on Preliminary Safety Evaluation Report and Proposed Certificate of Compliance HI-Star 100 Storage Cask," prepared on behalf of The State of Utah, Department of Environmental Quality, October 26, 1998.

Resnikoff, M., "Calculation of Radiation Exposures Received by Rebekah Dassion," prepared on behalf of Hagens & Berman as part of a federal personal injury case, November 23, 1998.

Resnikoff, M., Review of Risk-Based Remedial Action Levels and Holding Basin Feasibility Study, Starmet Corp, prepared on behalf of CREW, April 1999.

Resnikoff, M., Comments on the Feasibility Study and Proposed Plan for the Wayne Interim Storage Site, prepared on behalf of the Township of Wayne, August 9, 1999.

Dela Merced, M. and Resnikoff, M., Occupational Exposure to Radioactive Scale During Oil Pipe Cleaning Operations, prepared on behalf of Sacks & Smith, November 1999.

Dela Merced, M., Fairlie, I., and Resnikoff, M., Radiation Exposures Due to Mining, Milling and Waste Disposal Operations in South Texas (with 1st set of plaintiffs Adams), prepared on behalf of Whittington, von Sternberg, Emerson & Wilsher, LLP, March 2000.

- 2nd set of plaintiffs Cantu, prepared April 2000.
- 3rd set of plaintiffs Aguirre, prepared May 2000.

Resnikoff, M., Presentation by Marvin Resnikoff, Radioactive Waste Management Associates, to the Scientific Advisory Panel of the California Advisory Group on Management of Low-Level Radioactive Waste, March 17, 2000.

Resnikoff, M., Radiation Dose Exposures Received by Milt Vercher During Oil Pipe Cleaning Operations, prepared on behalf of Sacks & Smith, March 30, 2000.

Dela Merced, M. and Resnikoff, M., Occupational Exposure to Radioactive Scale During Oil Pipe Cleaning Operations (update of November 1999 document), prepared on behalf of Sacks & Smith, March 31, 2000.

Resnikoff, M., Rebuttal Affidavit in the matter of Anita Adams, et al, Plaintiffs vs. E.I. Dupont Denemours & Company, et al, Defendants, prepared on behalf of Whittington, von Sternberg, Emerson & Wilsher, LLP, April 2000.

RWMA, Uranium Movement at the NMI/Starmet Site, Concord, MA, prepared on behalf of CREW, September 2000.

Resnikoff, M., Radiation Exposures Received by Thomas Miller, prepared on behalf of J. Gregory Marks, October 2000.

Dela Merced, M., and Resnikoff, M., Comments on the Response by NMI/Starmet to the Proposed Addition of the Starmet Site to the Superfund List, prepared on behalf of CREW, November 2000.

CERTIFICATE OF COMPLIANCE FOR SPENT FUEL STORAGE CASKS

The U.S. Nuclear Regulatory Commission is issuing this Certificate of Compliance pursuant to Title 10 of the Code of Federal Regulation Part 72, "Licensing Requirements for Independent Storage of Spent Nuclear Fuel and High-Level Radioactive Waste" (10 CFR Part 72). This certificate is issued in accordance with 10 CFR 72.238, certifying that the storage design and contents described below meet the applicable safety standards set forth in 10 CFR Part 72, Subpart L, and on the basis of the Final Safety Analysis Report (FSAR) of the ca design. This certificate is conditional upon fulfilling the requirements of 10 CFR Part 72, as applicable, and the conditions specified belov

Certificate No.	Effective Date	Expiration Date	Docket Number	Amendment No.	Amendment Date	Package Identification No.
1014	05/31/00	06/01/20	72-1014	0		USA/72-1014

Issued To: (Name/Address)

Holtec International
Holtec Center
555 Lincoln Drive West
Marlton, NJ 08053

Safety Analysis Report Title

Holtec International Inc., Final Safety Analysis Report for the HI-STORM 100 Cask System
Docket No. 72-1014

CONDITIONS

This certificate is conditioned upon fulfilling the requirements of 10 CFR Part 72, as applicable, the attached Appendix A (Technical Specifications) and Appendix B (Approved Contents and Design Features), and the conditions specified below:

1. CASK

   a. Model No.: HI-STORM 100 Cask System

   The HI-STORM 100 Cask System (the cask) consists of the following components: (1) interchangeable multi-purpose canisters (MPCs), which contain the fuel; (2) a storage overpack (HI-STORM 100), which contains the MPC during storage; and (3) a transfer cask (HI-TRAC), which contains the MPC during loading, unloading and transfer operations. The cask stores up to 24 pressurized water reactor (PWR) fuel assemblies or 68 boiling water reactor (BWR) fuel assemblies.

   b. Description

   The HI-STORM 100 Cask System is certified as described in the Topical Safety Analysis Report (SAR) and in NRC's Safety Evaluation Report (SER) accompanying the Certificate of Compliance. The cask comprises three discrete components: the MPCs, the HI-TRAC transfer cask, and the HI-STORM 100 storage overpack.

5.5 Cask Transport Evaluation Program (continued)

b. For site-specific transport conditions which are not bounded by the surface characteristics in Section 3.4.6 of Appendix B to Certificate of Compliance No. 1014, the program may evaluate the site-specific conditions to ensure that the impact loading due to design basis drop events does not exceed 45 g. This alternative analysis shall be commensurate with the drop analyses described in the Topical Safety Analysis Report for the HI-STORM 100 Cask System. The program shall ensure that these alternative analyses are documented and controlled.

c. The TRANSFER CASK and MPC, when loaded with spent fuel, may be lifted to those heights necessary to perform cask handling operations, including MPC transfer, provided the lifts are made with structures and components designed in accordance with the criteria specified in Section 3.5 of Appendix B to Certificate of Compliance No. 1014, as applicable.

Table 5-1

TRANSFER CASK and OVERPACK Lifting Requirements

ITEM	ORIENTATION	LIFTING HEIGHT LIMIT (in.)
TRANSFER CASK	Horizontal	42 (Note 1)
TRANSFER CASK	Vertical	None Established (Note 2)
OVERPACK	Horizontal	Not Permitted
OVERPACK	Vertical	11

Notes: 1. To be measured from the lowest point on the TRANSFER CASK (i.e., the bottom edge of the transfer lid).

1.01 See Technical Specification 5.5c.

DESIGN FEATURES (continued)

3.4 Site-Specific Parameters and Analyses

Site-specific parameters and analyses that will require verification by the system user are, as a minimum, as follows:

1.01 The temperature of 80° F is the maximum average yearly temperature.

2. The allowed temperature extremes, averaged over a 3-day period, shall be greater than -40° F and less than 125° F.

3. The resultant horizontal acceleration (vectorial sum of two horizontal ZPA's at a three-dimensional seismic site), G_H, and vertical acceleration, G_V, expressed as fractions of 'g', shall satisfy the following inequality:

$$G_H + \mu G_V \leq \mu$$

where μ is the Coulomb friction coefficient for the HI-STORM 100/ISFSI pad interface. Unless demonstrated by appropriate testing that a higher value of μ is appropriate for a specific ISFSI, the value of μ used shall be 0.53. Representative values of G_H and G_V combinations for μ = 0.53 are provided in Table 3-2.

Table 3-2

Representative DBE Acceleration Values to Prevent HI-STORM 100 Sliding (μ = 0.53)

Equivalent Vectorial Sum of Two Horizontal ZPA's (G_H in g's)	**Corresponding Vertical ZPA (G_V in g's)**
0.445	0.160
0.424	0.200
0.397	0.250

(continued)

Private Fuel Storage, L.L.C.

7677 East Berry Ave., Englewood, CO 80111-2137
Phone 303-741-7009 Fax: 303-741-7806
John L. Donnell, P.E., Project Director

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

September 25, 2001

EIS COMMITMENT RESOLUTION LETTER #13
DOCKET NO. 72-22 / TAC NO. L22462
PRIVATE FUEL STORAGE FACILITY
PRIVATE FUEL STORAGE L.L.C.

Reference: September 24, 2001 telephone call between the NRC and Stone and Webster (S&W)

During the above referenced telephone call, Mr. Mike Waters of the NRC requested that Private Fuel Storage L.L.C. (PFS) include a change to Section 4.2.9.1.1 of the Private Fuel Storage Facility (PFSF) Environmental Report (ER) in the next amendment to the PFSF License Application. ER Section 4.2.9.1.1 states the following:

> "As described in Section 7.6 of the PFSF SAR, a maximum dose rate of 2.10 mrem/yr (assuming a 2,000 hour annual occupancy) was calculated at the OCA boundary fence 600 meters from the RA fence at its closest points of approach. This dose rate is comprised of direct and scattered gamma and neutron radiation assumed to emanate from 4,000 HI-STORM storage casks and is based on the assumption that all 4,000 casks contain typical fuel expected to be received at the PFSF with 35-GWd/MTU burnup and 20-year cooling time."

The PFSF storage cask array dose assessment evaluated dose rates at the north and west owner controlled area (OCA) boundaries, both of which are a minimum distance of 600 meters out from the restricted area fence. The dose assessment determined that the maximum dose rate occurs at the north OCA boundary. Section 7.3.3.5 of the PFSF SAR presents dose rates from the PFSF cask array at the OCA boundary, based on the full complement of 4,000 HI-STORM storage casks, assuming that all the casks contain 1) relatively hot fuel having 40 GWd/MTU burnup and 10-year cooling time, and 2) fuel having 35 GWd/MTU burnup and 20-year cooling time, which is considered to be representative of typical fuel expected to be received at the PFSF as explained in SAR Section 7.3.3.5. The dose rates at the OCA boundary for the relatively hot fuel were calculated to be 5.85 mrem/yr at a point on the boundary 600 meters north of the RA

State's
Exhibit 136

fence, and 4.35 mrem/yr at a point on the boundary 600 meters west of the RA fence, assuming a hypothetical individual spends 2,000 hours per year at the OCA boundary. Dose rates will be lower at points along the south and east sides of the OCA boundary, since these points are further from the storage casks than the north and west OCA boundaries. The annual dose at the north OCA boundary (maximum) for typical fuel expected to be received at the PFSF (having 35 GWd/MTU burnup and 20-year cooling time) was calculated to be 2.10 mrem, again assuming that a hypothetical individual spends 2,000 hours per year at the boundary. Section 4.2.9.1.1 of the ER identifies the 2.10 mrem annual dose at the OCA boundary, based on the stated assumption that all 4,000 casks contain typical fuel expected to be received at the PFSF with 35-GWd/MTU burnup and 20-year cooling time. The NRC requested that PFS revise this section of the ER to include the higher calculated annual dose of 5.85 mrem, based on the assumption that all 4,000 HI-STORM storage casks contain the relatively hot fuel assumed to have 40 GWd/MTU burnup and 10-year cooling time.

PFS will revise ER Section 4.2.9.1.1 to incorporate the 5.85 mrem calculated annual dose at the OCA boundary that is based on the assumption that all 4,000 storage casks contain the relatively hot fuel, and PFS will make a similar revision to SAR Section 7.6, which contains essentially the same information as ER Section 4.2.9.1.1. These revisions will be included in the next amendment to the PFSF License Application.

If you have any questions regarding this response, please contact me at 303-741-7009.

Sincerely,

John L. Donnell
Project Director
Private Fuel Storage L.L.C.

Copy to (with enclosure):
Mark Delligatti
Scott Flanders
John Parkyn
Jay Silberg
Sherwin Turk
Greg Zimmerman
Scott Northard
Denise Chancellor
Richard E. Condit
John Paul Kennedy
Joro Walker

CONDENSED TRANSCRIPT

UNITED STATES OF AMERICA

NUCLEAR REGULATORY COMMISSION

Before the Atomic Safety and Licensing Board

In the Matter of	) Docket No. 72-22
	) ASLPB No. 97-732-02-ISFSI
PRIVATE FUEL STORAGE L.L.C.	)
	) TELEPHONE DEPOSITION OF:
(Private Fuel Storage Facility)	) **EVERETT LEE REDMOND II**
	) (Utah Contention L, Part B)

Thursday, November 15, 2001 - 8:15 a.m.

Location: Office of the Attorney General
160 East 300 South, 5th Floor
Salt Lake City, Utah

Reporter: Vicky McDaniel
Notary Public in and for the State of Utah

State's Exhibit 137

PAGE 37

37

Q. Okay. And no involvement or contribution to paragraph -- to items 2, 3, and 4 in paragraph 8?
A. No.
Q. Thank you. I'm going to go through a list of some technical documents to see if you have reviewed any of these documents or had any contribution to them.
A. Okay.
Q. The seismic analysis of the pads as done by Stone and Webster?
A. No, I have not reviewed that.
Q. A Holtec report entitled Evaluation and Confinement Integrity of a Loaded Holtec MPC under a Postulated Drop Event?
A. No, I have not reviewed it.
Q. Dynamic Response of Freestanding HI-STORM 100 Excited by 10,000-year Return Period Earthquake at PFS. That's another Holtec report.
A. No, I have not reviewed that.
Q. I believe this is your report, the Radiation Shielding Analysis, right, of PFS?
A. That I have reviewed.
Q. Right. And finally, another Holtec report, Multi Cask Response at the PFS ISFSI from a 2,000-year Seismic Event?
A. No, I have not reviewed that.

PAGE 38

38

Q. Paragraph 10 of your declaration deals with the geometry of the pad. Does this take into account the new spacing that the PFS -- let me start that again. PFS amended its license application this year to show new spacing on the pad. Does the geometry that you refer to take into account this new spacing?
A. Yes, it does.
Q. Did that have any effect on the dose analysis, the new spacing?
A. Yes, it did. The dose analysis was redone to account for the spacing.
Q. Did that end up -- if you know, did that end up with a higher or lower dose analysis with the new spacing as opposed to the old spacing?
A. I don't remember if it was higher or lower. It was extremely close to the previous analysis is what I remember.
Q. Okay. If you would turn to paragraph 22 of the declaration. Mr. Gaukler may have told me this, but I can't remember. There's no attribution to paragraph 22.
A. Okay.
Q. Do you know who authored paragraph 22?
A. No, I do not.
MS. CHANCELLOR: If Mr. Gaukler knows, if

PAGE 39

39

you could tell me again.
MR. GAUKLER: I told you that Soler was responsible for that.
MS. CHANCELLOR: I knew you did. I couldn't remember.
Q. (BY MS. CHANCELLOR) If you would turn to paragraph 23 of your declaration. Towards the end of -- two thirds of the way down paragraph 23 you state that there's a table in the PFS SAR that shows a maximum value of 5.85 millirems per year was calculated for a 2,000-hour year occupancy time.
A. Yes.
Q. At the controlled area boundary. Can you explain to me why you used a 2,000-hour year?
A. Two thousand hours was used because it's roughly equivalent to a 40-hour work week and the area outside the controlled area of boundary is unoccupied, and that was a conservative estimate for the amount of time an individual would be spending there.
Q. You used the term "at the controlled area boundary." I believe the regulations used "beyond the controlled area boundary." Do you have an opinion whether there's any difference between at the controlled area boundary as opposed to beyond the controlled area boundary?

PAGE 40

40

A. I don't have the regulations in front of me to review them to know the exact wording, but to me they would be the same.
Q. Have you done any dose calculations for an 8,760-hour year, which would be somebody at the boundary 24 hours a day, 365 days for the entire year?
A. Not at the controlled area boundary. The dose rate was estimated at the nearest occupant, nearest resident for a full 8,760-hour occupancy time.
Q. What is the distance to the closest resident?
A. I believe it's a mile and a half, or maybe it's two and a half miles. I don't remember exactly.
Q. In your work with Trojan and Diablo, for example, when you did dose calculations there, do you know whether you used 2,000 hours per year at the controlled area boundary or an 8,760-hour year?
A. Both of those analyses was done using not 2,000 hours but I believe 2,080 hours.
Q. So they didn't get any time off for Christmas. Is that what you're saying?
A. Basically.
Q. Paragraph 25 of the joint declaration is attributable both to you and to Dr. Soler.
A. Yes.

PAGE 45

45

1 Q. (BY MR. CHANCELLOR) Mr. Redmond, you stated
2 Hatch, Dresden, and then I didn't hear the rest of your
3 answer. Something about another utility.
4 **A. Yes. We've manufactured casks for**
5 **Fitzpatrick. We have not manufactured casks for Diablo**
6 **Canyon or Trojan.**
7 Q. Do you intend to, if you know?
8 **A. We intend to manufacture casks for Diablo**
9 **Canyon. For Trojan, they have existing casks already.**
10 **Trojan is a unique situation. We will not be**
11 **manufacturing casks for them.**
12 Q. Okay, thank you. You state that it's highly
13 unlikely that any localized crushing and associated
14 microcracking would create, quote, "uninterrupted
15 radiation streaming path." Does this statement imply
16 that the concrete has nowhere to go and that it
17 completely fills the steel walls?
18 **A. Yes.**
19 Q. Does your statement imply that the steel
20 cannot stretch under torsional movement?
21 MR. GAUKLER: Please restate the question.
22 I didn't hear that.
23 MS. CHANCELLOR: Does Mr. Everett's
24 statement about uninterrupted radiation streaming path,
25 does his statement imply that the steel cannot stretch

PAGE 46

46

1 under torsional movement.
2 **A. It implies that it's my belief that the**
3 **steel will not stretch. It's my understanding that the**
4 **steel will not stretch.**
5 Q. Have you done any calculations or do you
6 know of any calculations that have been done to support
7 that belief?
8 **A. I have not done any calculations, and I'm**
9 **not familiar with the calculations that have been done.**
10 Q. Have you done any calculations or do you
11 know of any calculations that have been done of the
12 radiation consequences of concrete cracking?
13 **A. No.**
14 Q. Have you considered concrete cracking such
15 that it would not provide radiation shielding?
16 MR. GAUKLER: Objection, vague and ambiguous
17 question.
18 Q. (BY MS. CHANCELLOR) If you understand the
19 question, you may answer it.
20 **A. I don't actually understand the question.**
21 **Please rephrase.**
22 Q. Have you looked at any concrete cracking
23 scenario where there would be a loss of radiation
24 shielding?
25 **A. No, I have not analyzed any concrete**

PAGE 47

47

1 **cracking.**
2 Q. If you would turn to paragraph 26 of your
3 declaration.
4 MR. GAUKLER: How much longer do you have,
5 Denise? We've been going about an hour and a half.
6 May be time for a break.
7 MS. CHANCELLOR: I think, Paul, I can wrap
8 it up -- I mean, if the witness is tired and would like
9 a break, that's just fine. But I really don't have a
10 whole lot more.
11 MR. GAUKLER: Do you want to take a break,
12 Everett?
13 THE WITNESS: If I could take about two
14 minutes, that would be great.
15 MS. CHANCELLOR: Yeah, that would be fine.
16 Why don't we call you back in five minutes, then.
17 (Recess from 9:32 to 9:49 a.m.)
18 MS. CHANCELLOR: Do you still have the same
19 people in the room?
20 MR. GAUKLER: Yeah.
21 Q. (BY MS. CHANCELLOR) Okay, where were we?
22 Welcome back, Mr. Redmond. If you would turn to
23 paragraph 26 of your declaration.
24 **A. Okay.**
25 MR. TURK: We're not on the record yet, are

PAGE 48

48

1 we?
2 MS. CHANCELLOR: Yes, we are.
3 MR. TURK: Could we go off one second?
4 MS. CHANCELLOR: Okay.
5 (Discussion off the record.)
6 Q. (BY MS. CHANCELLOR) Mr. Redmond, if you'd
7 look at paragraph 26 of your declaration.
8 **A. Yes.**
9 Q. Towards the end of the last sentence in
10 paragraph 26, the "therefore" clause, you talk about
11 the roundness of the storage cask could only be reduced
12 in the radial area of impact. Could you explain that
13 statement?
14 **A. The analysis for the tipover, as I**
15 **understand it, indicates that any localized**
16 **deformations that might occur would occur between the**
17 **cask and the ground. And so what I'm talking about**
18 **here is if there's a localized deformation, the**
19 **roundness of the storage cask would be affected,**
20 **obviously, and that that only occurs between the cask**
21 **and the ground in that localized area. I'm defining**
22 **the area of impact, if you will.**
23 Q. Now, I didn't -- my understanding was that
24 you hadn't reviewed the tipover analysis.
25 **A. That's true, I haven't reviewed the tipover**

1 **analysis; but I have had discussions, like I've said**
2 **before, with Dr. Soler.**
3 Q. Okay, and it's based on those discussions
4 that -- is this your contribution, the roundness of the
5 storage cask could only be reduced in the radial area
6 of impact?
7 **A. Yes, it is. Well, yes. I wrote it, but**
8 **obviously Dr. Soler is on the paragraph as well.**
9 Q. Just give me a moment. I've sort of lost my
10 train of thought. I'll only be a second.
11 On paragraph 28 you claim that if the casks
12 were lying on their sides, the side dose from a cask
13 would decrease and more than compensate the increased
14 dose from the top and bottom of the casks. Have you
15 done any calculations that support this claim?
16 **A. I've done some very simplified calculations**
17 **to look at side dose from a tipped-over concrete**
18 **cylinder. I have not analyzed the HI-STORM 100 cask.**
19 **I have done some very simplified calculations, but I**
20 **have not looked at the top, any radiation coming off**
21 **the top or the bottom of the cask.**
22 MS. CHANCELLOR: Mr. Gaukler, if we don't
23 have a copy of those calculations, could we get a copy?
24 MR. GAUKLER: Okay, I'll take it under
25 advisement.

1 Q. (BY MS. CHANCELLOR) Have you estimated what
2 the dose on the bottom of the HI-STORM cylinder would
3 be in a tipover event, in a tipover event this side
4 would potentially be exposed and contribute to a dose
5 to someone at the boundary?
6 **A. No, I have not.**
7 Q. And in paragraph 28, the dose rate of the
8 controlled area of boundary, is that also based on the
9 2,000-hour year as it was in the early part in the
10 declaration?
11 **A. Yes, it would be.**
12 Q. In paragraphs 29 and 30 you basically look
13 at all 4,000 casks tipping over. Can you explain how
14 the tops and bottoms of the casks would be facing each
15 other, or, in other words, are they lined up on the
16 ground top to bottom?
17 **A. I can try to explain it. This is my, again,**
18 **my opinion here. The Private Fuel Storage, this is**
19 **arranged in various rows of casks. Each pad has a --**
20 **well, as I discuss here in paragraph, well, 30, there 2**
21 **by 40 arrays of casks. So they're in basically rows of**
22 **80 casks, and then there's a pathway 35 feet wide**
23 **between them. The distance between the casks in that 2**
24 **by 40 array is 15 or 16 feet, depending on the**
25 **direction.**

1 **And so the cask is better than 11 feet in**
2 **diameter, and so in order for the casks to fall over,**
3 **all 80 casks on a pad, they're essentially going to**
4 **have to fall away from each other. And if they fall**
5 **away from each other, you know, the tops of the casks**
6 **would then fall away from each other, and when they**
7 **land the bottoms would be facing each other.**
8 Q. Why do you say that they would have to fall
9 away from each other?
10 **A. Well, it's again my opinion that if you**
11 **wanted to have all 80 casks fall down and to be laying**
12 **on the ground flat, that's about the only way I can**
13 **think of that that would happen.**
14 Q. Oh, so you're looking at the end result,
15 you're not looking at whether some would topple on top
16 of other casks?
17 **A. Right. In fact, in paragraph 29 I talk**
18 **about that and say that, you know, in all likelihood**
19 **you wouldn't have all of them fall down. But paragraph**
20 **30 is talking about all 4,000 casks laying flat on the**
21 **ground.**
22 Q. So is it your opinion that this would be the
23 worst case scenario if the casks were lying on the
24 ground top to bottom as opposed to at random some on
25 top of each other, some on their sides?

1 **A. I wouldn't use the word "worst case**
2 **scenario." As I say in here, my opinion is that the**
3 **end result of all 4,000 casks tipped over is**
4 **essentially no change in the dose rate.**
5 Q. Provided that they're lying --
6 **A. Well, and also --**
7 Q. -- top to bottom?
8 **A. -- the same would be true for casks being**
9 **randomly oriented and partially tipped over and some**
10 **upright.**
11 Q. So your opinion is the same whether they
12 are -- whether they are lying top to bottom or whether
13 they are randomly distributed; is that correct? Is
14 that what you just said?
15 **A. In the sense that if they're randomly**
16 **distributed, they're not all 4,000 lying down.**
17 Q. Well, let's just say the 4,000 aren't all
18 upright.
19 **A. Right. Well, if they're randomly**
20 **distributed, some of the casks would be upright, some**
21 **would be leaning against others, and others would be**
22 **laying down.**
23 Q. So you wouldn't get the shielding from the
24 side of the cask like you would in the top to bottom
25 scenario if they were randomly distributed. Is that

1 the transportation license will not permit you to
2 transport the same contents that will be permitted for
3 storage in HI-STORM. In other words, the HI-STORM will
4 be permitted to store high burnup fuel. The
5 transportation license will be amended to store high
6 burnup fuel or to be able to transport high burnup
7 fuel. But the cooling time that will be permitted in
8 the transportation will be significantly longer than
9 will be permitted in the HI-STORM storage cask.
10 Q. So it will be higher burnup fuel with a
11 longer cooling time, but I'm not sure longer than what.
12 A. Well, longer than what will be permitted in
13 the HI-STORM storage cask.
14 Q. Okay, all right. Thank you. That clarifies
15 it.
16 I assume in paragraph 32 the radiation dose
17 at the site boundary is for the 2,000-hour years?
18 A. Just a second, please.
19 Q. Certainly.
20 A. Yes.
21 Q. Paragraph 33 of your declaration, you give
22 your opinion that a fully engulfing fire creating a
23 temperature of 1,475 degrees Fahrenheit for 3.6 minutes
24 at the surface of the storage cask would provide a more
25 severe thermal transient on the cask system with regard

1 to shielding. Can you explain the basis of this
2 assumption?
3 A. I have to preface it by saying I'm not a
4 thermal expert, and again, this is my opinion. The
5 fire that was analyzed is 1,475, and I say here the
6 localized temperature excursion affects only 4 percent
7 of the total thickness, and therefore it's my opinion
8 that if you tip over the cask that the temperatures
9 that the concrete would see, I don't think they could
10 exceed the temperatures that have been calculated here
11 for the fire. Again, that's my opinion. I haven't
12 done any thermal analysis and I don't know what that
13 would show.
14 Q. Now, you say you're not a thermal expert,
15 but what are you basing your opinion on?
16 A. It's based on my feelings, actually, that in
17 a tipover situation the MPC would now be resting on the
18 inner portion of the overpack so there would be some
19 additional heat transfer mechanisms that are not
20 currently available in the HI-STORM overpack.
21 In addition, the heat load capability, or
22 the heat load that will be in the MPC at Private Fuel
23 Storage is far less than the heat load that the
24 HI-STORM is generically able to take, and that's
25 because of the transportation limitations we discussed

1 earlier.
2 Based on those and the fact that the fire
3 also affects basically the entire cask, whereas in a
4 tipover condition if there's any effect on the concrete
5 it would be localized again, probably below the MPC,
6 not the whole body of the overpack. Based on those
7 viewpoints, that's what my opinion is based on.
8 Q. And you state that a 1,475 degree fire
9 condition, the 3.6 minutes would bound the thermal
10 events of cask tipover even if the casks remained in a
11 horizontal position for extended periods of time. Can
12 you describe the duration of extended periods of time?
13 A. The extended periods of time would be -- I
14 cannot give you a number for it, but if it takes a long
15 period of time to upright the cask, that would be an
16 extended period of time. I don't have a number for
17 you.
18 Q. Not even ballpark? Are we talking about
19 hours or days?
20 A. To me it would be probably tens of hours,
21 but I don't have a feel for it.
22 Q. Do you have an opinion on whether the
23 HI-STORM 100S provides better radiation shielding than
24 the HI-STORM 100?
25 MR. GAUKLER: Objection, lack of relevance.

1 You can answer if you can.
2 A. The HI-STORM 100S has the same thickness,
3 same shielding in the radial direction as the HI-STORM
4 100. It is shorter, and therefore the dose rates out
5 the top of the overpack are somewhat higher on the
6 HI-STORM 100S than they are on the HI-STORM 100.
7 MS. CHANCELLOR: If you'd just hold on for a
8 moment, I think I'm about done. Let's go off the
9 record for a moment.
10 (Discussion off the record.)
11 Q. (BY MS. CHANCELLOR) I just have one more
12 question. Mr. Redmond, do you have an opinion of what
13 happens to concrete if the temperature is greater than
14 the design basis limits, both short-term limits and
15 long-term limits?
16 A. What would happen to concrete if --
17 Q. What would happen to the shielding
18 properties.
19 A. Right. If temperature is elevated for a
20 duration of time above -- I don't know the limits, the
21 temperature limits of concrete. I'm not a concrete
22 expert. But what would happen is the water that is in
23 the concrete would be driven out, and water has
24 hydrogen in it, so what you're doing is you're
25 decreasing neutron shielding when that happens, when

PAGE 61

61

1 the water's removed.
2 The gamma, the shielding properties of
3 concrete for gamma radiation are predominantly
4 unaffected because that's provided by the other
5 constituents in the concrete. So it's my opinion that
6 what would happen is that you would get an increase in
7 the neutron dose rate but really no significant change
8 in the gamma dose rate. And the gamma dose is the
9 dominant, by far the dominant portion of the radiation
10 dose coming off of the overpack, and therefore there
11 would be no significant change in the performance of
12 the overpack.
13 In addition, the localized areas, the
14 temperature increase would be localized areas. The
15 entire body of the overpack would not -- would not be
16 at the same temperature. There's still more gradients
17 through the overpack. So any increase in temperature
18 would be localized, which would further reduce the
19 shielding impact.
20 Q. Well, Mr. Redmond, thank you very much. I
21 have no further questions.
22 A. There's one thing that I was thinking I
23 should add in regards to the high burnup issue.
24 Q. Have you had discussion with counsel on this
25 issue?

PAGE 62

62

1 A. Only to the extent of --
2 MR. GAUKLER: Only to the extent we thought
3 clarification was appropriate.
4 THE WITNESS: And it was my idea.
5 Q. Okay, go ahead.
6 A. We had talked about the HI-STORM cask being
7 able to, in the amendment, take high burnup fuel. It
8 should be understood that as a general rule, the higher
9 burnup, longer cooling time fuel gives a lower dose
10 rate than shorter cooling time, lower burnup fuel. So
11 that means that even though the cask will be permitted
12 to store higher burnup fuel, that doesn't mean that,
13 necessarily mean that the dose rates would be higher
14 when you put that fuel in, because the permitted
15 cooling times for that fuel would also be longer.
16 That's the only point I wanted to clarify or add.
17 Q. Was there anything else you wanted to add or
18 clarify to your deposition?
19 A. No, that was it. That was something I
20 thought I left out. Thank you.
21 MS. CHANCELLOR: Okay, I have no further
22 questions.
23 MR. TURK: Paul, I would like to ask a
24 couple of questions. I don't know if this is something
25 that you're going to go into or not.

PAGE 63

63

1 MR. GAUKLER: I just have one question I'm
2 going to ask, so probably not.
3 EXAMINATION
4 BY MR. TURK:
5 Q. Okay. My name is Sherwin Turk. I'm an
6 attorney with the NRC staff. Mr. Redmond, I just want
7 to ask you a couple of questions. Is it Mister or
8 Doctor?
9 A. It's Doctor, actually.
10 Q. I apologize. I don't have your resume in
11 front of me.
12 A. No, that's quite all right.
13 Q. You were talking at some point about whether
14 Holtec is manufacturing the casks being used at other
15 sites.
16 A. Yes.
17 Q. When you used the phrase "manufacture," were
18 you including -- let me just ask you a direct question.
19 Is Holtec involved in the pouring of the concrete for
20 casks at any site? And if so, which site?
21 A. It's my understanding -- again, I'm not the
22 manufacturing person here -- our casks are -- the steel
23 portion of the cask is manufactured at U.S. Tool and
24 Dye. The casks are shipped then to the sites where the
25 concrete is poured. The concrete is poured, as I

PAGE 64

64

1 understand it, under supervision or observation, at
2 least, at a minimum by Holtec personnel using Holtec
3 approved procedures. Again, I'm not the manufacturing
4 person, so the final word on that would come from
5 somebody else. But that's my understanding. And I
6 believe that's been done at Dresden, Hatch and
7 Fitzpatrick. Those are the only sites we've delivered
8 casks to, HI-STORM casks to.
9 Q. Is that different in any way from your
10 understanding of what will be done at PFS?
11 MS. CHANCELLOR: Objection.
12 A. I have no reason to doubt it would be done
13 differently, but I don't know.
14 Q. Also you were asked some questions about the
15 hypothetical incident in which many casks or all casks
16 have been tipped over and are lying either on their
17 sides or some on their sides and some at a 45-degree
18 angle. Looking at that in a hypothetical situation
19 where there is a large amount of tipover casks, if a
20 cask is lying on the ground facing north-south and in
21 this orientation it's facing another cask that's lying
22 on its side facing east-west, that's oriented
23 east-west, would the north-south cask radiation be
24 shielded by that other cask that it's pointed at which
25 is lying in an east-west direction?

HOLTEC INTERNATIONAL

HI-STORM 100 CASK SYSTEM

SAFETY EVALUATION REPORT

ML003711779

State's
Exhibit 138

and the overpack in the tipover event for the maximum centrifugal acceleration (when the overpack is essentially horizontal and the MPC is only restrained longitudinally by steel on stainless steel friction). Due to the finite distance of MPC travel necessary for it to impact the overpack lid, any such impact would not be concurrent with maximum deceleration due to ground impact. Therefore, the tensile load to the lid restraining studs would not occur at the time of maximum shear in those studs.

As a result of the above considerations, the staff concludes that the scope of the SAR tipover analysis is acceptable.

Accidental Drop

The SAR structural analysis of accidental drops of an overpack with a full MPC is in Appendix 3.M. This analysis determines factors of safety for the overpack structural components in the load path associated with an 11 inch vertical drop onto the reference pad. Appendix 3.A provides the determination of the maximum height (11 inches) that the overpack and MPC within the overpack may be dropped with overpack longitudinal axis vertical without imposing more than 45g deceleration on the MPC. This drop height is used as a limiting condition of use for the height that the overpack with MPC may be above a receiving surface. Summary minimum factors of safety are shown in SAR Tables 3.4.5 and 3.4.9.

The analyses of the overpack structural elements in the SAR determine that the factors of safety of the most critically loaded elements would be above 1.0, and that any deformation would not impose loads on the MPC or impair ready retrievability of the stored materials.

The staff review determined that the analytical approaches, computations, results, and acceptance criteria are acceptable. The assumptions relating to the receiving surface (used for both drop and tipover) are acceptable. The pad is the reference pad used for the HI-STAR 100 Cask System SAR. This pad is identical to that described in UCRL-ID-126295. The factors of safety determined are considered to include the lowest factors of safety associated with tipover of the overpack.

The SAR drop analysis does not include examination of a corner drop or drop with the overpack longitudinal axis horizontal. The analyses also do not include the stresses in all of the welds or all of the component members of the overpack body or lid weldments.

A corner drop with the center of gravity over the point of impact is considered to be most likely to cause local permanent distortion of the overpack. A drop from the maximum design drop height of 11 inches would, however, result in greater penetration of the receiving surface, reducing the maximum decelerations experienced by the MPC. The effects of subsequent overturning would be within the effects determined for the non-mechanistic tipover event. Any simultaneous deformation of the overpack would further reduce the deceleration. Any significant permanent deformation of the overpack at the point of impact would be readily observable following the event. The restraint of 11 inches vertical height for overpack handling should preclude a situation in which the full overpack were raised sufficiently to permit a corner drop with c.g. vertically over a tangent to the base plate edge.

adequate for certification and licensing at those sites where it is shown that handling of the overpack will not be greater than 11 inches and that the receiving surface hardness does not exceed that analyzed in the SAR.

Under this postulated accident, all stresses remain within allowable values, thereby assuring that the confinement boundary remains in tact.

11.2.4 HI-STORM 100 Storage Overpack Tipover

11.2.4.1 Cause of Tipover

Although analyses have shown that the overpack will not tip over as a result of severe natural phenomena, such as earthquakes and tornadoes, a tipover analysis is required as a bounding design event to demonstrate the defense-in-depth of the design.

11.2.4.2 Consequences of Tipover Accidents

The tipover is described in Section 3.4.10 of the SAR. Analyses included a structural analysis of the tipover event, the determination of maximum accelerations that may be experienced, an analysis of the integrity of the overpack lid during the event, and an analysis of the studs securing the lid to the overpack. Staff review of the structural analyses are in Section 3 of this SER. The maximum acceleration of the MPC inside the overpack was shown to be 43.2 g as a result of the tipover event. This acceleration is bounded by the 45 g acceleration for which the MPC has been designed and analyzed. The structural analyses of tipover in the SAR concluded that the overpack would maintain safety, that the factors of safety of the most critically loaded elements would be above 1.0, and that deformations of the overpack would not impose loads on the MPC or impair retrievability following a tipover event. The staff review determined that the analytical approaches, computations, results, and acceptance criteria are acceptable.

11.2.5 Burial Under Debris

11.2.5.1 Cause of Burial

Natural phenomena that could lead to burial of the cask under man-made or earthen material.

11.2.5.2 Consequences of Burial

The applicant analyzed the effects of a postulated accident in which the cask is buried under debris which would act as an additional thermal resistance to heat removal from the cask surface as well as 100% blockage of all air inlets. This scenario satisfies the requirement of NUREG-1536 to perform an adiabatic heatup calculation. The thermal effect of debris was modeled as adiabatic insulation on the overpack along with complete air inlet blockage. The results of this analysis show that the short-term cladding temperature limit would not be reached until more than 100 hours. As in the case of the 100% air inlet blockage accident, the concrete short-term limit of 350°F would be expected to be reached at approximately 33 hours. This accident analysis demonstrates that the 24 hour surveillance interval for the cask air inlets is

HOLTEC INTERNATIONAL

Holtec Center, 555 Lincoln Drive West, Marlton, NJ 08053

Telephone (856) 797-0900
Fax (856) 797-0909

BY OVERNIGHT MAIL

February 4, 2000

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, DC 20555-0001

Subject: USNRC Docket No. 72-1014; TAC No. L22221
HI-STORM 100 Storage Application
HI-STORM TSAR, Revision 10

References:
1. Holtec Project 5014
2. Holtec Letter, B. Gutherman to NRC, dated February 1, 2000

Dear Sir:

In accordance with our recent verbal commitments and the Reference 2 letter, Holtec International is pleased to forward replacement pages comprising Revision 10 of the HI-STORM 100 System Topical Safety Analysis Report (TSAR). This revision includes changes to reflect the resolution of comments received by the NRC during the rulemaking process. In addition, several editorial corrections have been made. Changes in the affected chapters are described in the enclosed document entitled "Summary of Changes in Revision 10."

Thank you for your continued support in the HI-STORM 100 review process. We look forward to receiving the final Certificate of Compliance and Safety Evaluation Report by July 31, 2000.

If you have any questions or require additional information, please contact us.

Sincerely,

Brian Gutherman, P.E.
Licensing Manager

Approval:

K.P. Singh, Ph.D, P.E.
President and CEO

cc: Ms. Marissa Bailey, USNRC (w/14 copies of TSAR Rev. 10, including instructions)
Mr. E. William Brach, USNRC (w/o encl.)

Document ID: 5014363

State's
Exhibit 139

HOLTEC INTERNATIONAL

REPORT NO. HI-951312

TOPICAL SAFETY ANALYSIS REPORT FOR

THE HI-STORM 100 CASK SYSTEM

VOLUME I OF II

Table 1.D.1: Requirements on Plain Concrete

ITEM	APPLICABLE LIMIT OR REFERENCE
Density (Minimum)	146 (lb/cubic feet)
Specified Compressive Strength	4,000 psi (min.)
Compressive and Bearing Stress Limit	Per ACI 318-95
Cement Type and Mill Test Report	Type II; Section 3.2 (ASTM C 150 or ASTM C595)
Aggregate Type	Section 3.3 (including ASTM C33(Note 2))
Nominal Maximum Aggregate Size	3/4 (inch)
Water Quality	Per Section 3.4
Material Testing	Per Section 3.1
Admixtures	Per Section 3.6
Air Content	6%[1] (Table 4.5.1)
Maximum Water to Cement Ratio	0.5 (Table 4.5.2)
Maximum Water Soluble Chloride Ion Cl in Concrete	1.00 percent by weight of cement (Table 4.5.4)
Concrete Quality	Per Chapter 4 of ACI 349
Mixing and Placing	Per Chapter 5 of ACI 349
Consolidation	Per ACI 309-87
Quality Assurance	Per Holtec Quality Assurance Manual, 10 CFR Part 72, Appendix G commitments
Maximum Local Temperature Limit Under Normal and Off-normal Conditions	200°F (See Note 3)
Maximum Local Temperature Limit Under Accident Conditions	350°F (Appendix A, Subsection A.4.2)
Aggregate Maximum Value[2] of Coefficient of Thermal Expansion (tangent in the range of 70°F to 100°F)	6E-06 inch/inch/°F (NUREG-1536, 3.V.2.b.i.(2)(c)2.b)

Notes:

1. All section and table references are to ACI 349 (85).
2. The coarse aggregate shall meet the requirements of ASTM C33 for class designation 1S from Table 3. However, if the requirements of ASTM C33 cannot be met, concrete that has been shown by special tests or actual service to produce concrete of adequate strength and durability meeting the requirements of Tables 1.D.1 and 1.D.2 is acceptable in accordance with ACI 349 Section 3.3.2.
3. The 200 °F long term temperature limit is specified in accordance with Paragraph A.4.3 of ACI 349 for normal conditions. The 200 °F long term temperature limit is based on (1) the use of Type II cement, specified aggregate criteria, and the specified compressive stress in Table 1.D.1, (2) the relatively small increase in long term temperature limit over the 150°F specified in Paragraph A.4.1, and (3) the very low maximum stresses calculated for normal and off-normal conditions in Section 3.4 of this TSAR.

---

1 This limit is specified to accommodate severe exposure to freezing and thawing (Table 4.5.1).

2 The following aggregate types are a priori acceptable: limestone, dolomite, marble, basalt, granite, gabbro, or rhyolite. The thermal expansion coefficient limit does not apply when these aggregates are used. Careful consideration shall be given to the potential of long-term degradation of concrete due to chemical reactions between the aggregate and cement selected for HI-STORM 100 overpack concrete.

The total weight used in the analysis is approximately 2,000 lb. lighter than the HI-STORM 100 containing the lightest weight MPC.

Analysis of a single mass impacting a spring with a given initial velocity shows that both the maximum deceleration "a_M" of the mass and the time duration of contact with the spring "t_c" are related to the dropped weight "w" and drop height "h" as follows:

$$a_M \sim \frac{\sqrt{h}}{\sqrt{w}}; t_c \sim \sqrt{w}$$

Therefore, the most conservatism is introduced into the results by using the minimum weight. It is emphasized that the finite element model described in the foregoing is identical in its approach to the "Holtec model" described in the benchmark report [3.A.4]. Gaps between the MPC and the overpack are included in the model.

3.A.6 Impact Velocity

a. Linear Velocity: Vertical Drops

For the vertical drop event, the impact velocity, v, is readily calculated from the Newtonian formula:

$$v = \sqrt{(2gh)}$$

where

g = acceleration due to gravity
h = free-fall height

b. Angular Velocity: Tip-Over

The tipover event is an artificial construct wherein the HI-STORM 100 overpack is assumed to be perched on its edge with its C.G. directly over the pivot point A (Figure 3.A.16). In this orientation, the overpack begins its downward rotation with zero initial velocity. Towards the end of the tip-over, the overpack is horizontal with its downward velocity ranging from zero at the pivot point (point A) to a maximum at the farthest point of impact (point E in Figure 3.A.17). The angular velocity at the instant of impact defines the downward velocity distribution along the contact line.

In the following, an explicit expression for calculating the angular velocity of the cask at the instant when it impacts on the ISFSI pad is derived. Referring to Figure 3.A.16, let r be the length AC where C is the cask centroid. Therefore,

$$r = \left(\frac{d^2}{4} + h^2 \right)^{1/2}$$

The mass moment of inertia of the HI-STORM 100 System, considered as a rigid body, can be written about an axis through point A, as

$$I_A = I_c + \frac{W}{g} r^2$$

where I_c is the mass moment of inertia about a parallel axis through the cask centroid C and W is the weight of the cask (W = Mg).

Let $\theta_1(t)$ be the rotation angle between a vertical line and the line AC. The equation of motion for rotation of the cask around point A, during the time interval prior to contact with the ISFSI pad, is

$$I_A \frac{d^2\theta_1}{dt^2} = Mgr \sin\theta_1$$

This equation can be rewritten in the form

$$\frac{I_A}{2} \frac{d(\dot{\theta}_1)^2}{d\theta_1} = Mgr \sin\theta_1$$

which can be integrated over the limits $\theta_1 = 0$ to $\theta_1 = \theta_{2f}$ (See Figure 3.A.17).

The final angular velocity $\dot{\theta}_1$ at the time instant just prior to contact with the ISFSI pad is given by the expression

$$\dot{\theta}_1(t_B) = \sqrt{\frac{2\,Mgr}{I_A}(1 - \cos\theta_{2f})}$$

where, from Figure 3.A.17

$$\theta_{2f} = \cos^{-1}\left(\frac{d}{2r_1}\right)$$

This equation establishes the initial conditions for the final phase of the tip-over analysis; namely, the portion of the motion when the cask is decelerated by the resistive force at the ISFSI pad interface.

Using the data germane to HI-STORM 100 (Table 3.A.3), and the above equations, the angular velocity of impact is calculated as 1.49 rad/sec.

3.A.7 Results

It has been previously demonstrated in the benchmark report[3.A.4] that bounding rigid body decelerations are achieved if the cask is assumed to be rigid with only the target (ISFSI pad) considered as an energy absorbing media. Therefore, for the determination of the bounding decelerations reported in this appendix, the HI-STORM storage overpack was conservatively made rigid except for the radial channels that position the MPC inside of the overpack. The MPC material behavior was characterized in the identical manner used in the Livermore Laboratory analysis as was the target ISFSI pad and underlying soil. The LS-DYNA3D time-history results are processed using the Butterworth filter (in conformance with the LLNL methodology) to establish the rigid body motion time-history of the cask. The material points on the cask where the acceleration displacement and velocity are computed for each of the drop scenarios are shown in Figure 3.A.18.

Node 82533 (Channel A1), which is located at the center of the outer surface of the baseplate, serves as the reference point for end-drop scenarios.

Node 84392 (Channel A2), which is located at the center of the cask top lid outer surface, serves as the reference point for the tipover scenario with the pivot point indicated as Point 0 in Figure 3.A.18.

The final results are shown in Table 3.A.4.

Table 3.A.4: Results

Drop Event	Max. Displ (in)	Impact Velocity (in/sec)	Max. Acc. (g's)	Acc. Pulse Duration (msec.)
End-11"	0.696	92.20	44.13	2.96
Tipover Cask Top[1]	4.903	341.3	48.41	9.76
Tipover (Basket Top)	4.368	304.03	43.12	--
Tipover (with Increased Initial Clearance) Cask Top[1]	4.998	341.3	48.52	10.0
Tipover (with Increased Initial Clearance) (Basket Top)	4.452	304.03	43.22	--

1 The distance of the top of the fuel basket is 206" from the pivot point. The distance of the top of the cask is 231.25" from the pivot point. Therefore, all displacements, velocities, and accelerations at the top of the fuel basket are 89.08% of those at the cask top (206"/231.25").

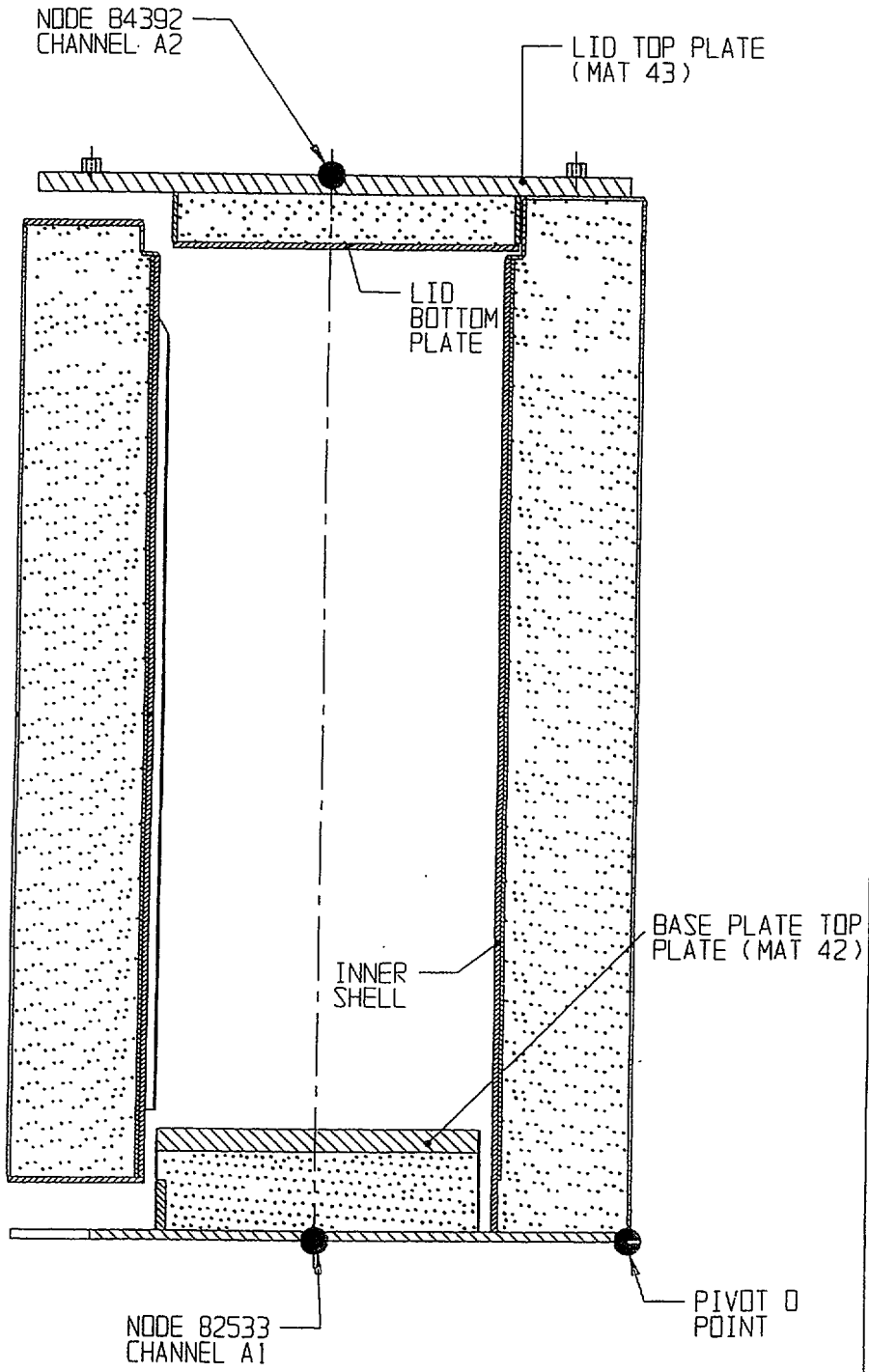


FIGURE 3.A.18; MEASUREMENT POINTS AND CORRESPONDING FINITE-ELEMENT MODEL NODES

The results from the case 1 analysis in the attachment are appropriate to examine the non-mechanistic tipover analysis. It is determined there that the diametrical change under the load is 0.11". This demonstrates that there is no restraint to remove the MPC after the tipover since a positive clearance is still maintained. (0.1875" is the radial gap prior to the accident).
The other cases considered in the attachment are for potential use elsewhere.

3.B.7 Conclusion

Classical ring solutions have been presented for use in examination of the ovalization of the storage overpack. The solutions have been derived using the weight of the storage overpack without an MPC. The weight has been amplified by 45 to represent a bounding accident condition. It is shown by analysis that ready retrievability of the fuel is maintained after a tipover accident.

11.2.2.4 HI-STORM Overpack Handling Accident Corrective Action

Following a handling accident, the ISFSI operator shall first perform a radiological and visual inspection to determine the extent of the damage to the overpack. Special handling procedures, as required, will be developed and approved by the ISFSI operator.

If upon inspection of the MPC, structural damage of the MPC is observed, the MPC is to be returned to the facility for fuel unloading in accordance with Chapter 8. After unloading, the structural damage of the MPC shall be assessed and a determination shall be made if repairs will enable the MPC to return to service. Likewise, the HI-STORM overpack shall be thoroughly inspected and a determination shall be made if repairs will enable the HI-STORM overpack to return to service. Subsequent to the repairs, the equipment shall be inspected and appropriate tests shall be performed to certify the HI-STORM 100 System for service. If the equipment cannot be repaired and returned to service, the equipment shall be disposed of in accordance with the appropriate regulations.

11.2.3 Tip-Over

11.2.3.1 Cause of Tip-Over

The analysis of the HI-STORM 100 System has shown that the overpack does not tip over as a result of the accidents (i.e., tornado missiles, flood water velocity, and seismic activity) analyzed in this section. It is highly unlikely that the overpack will tip-over during on-site movement because of the low handling height limit. The tip-over accident is stipulated as a non-mechanistic accident.

11.2.3.2 Tip-Over Analysis

The tip-over accident analysis evaluates the effects of the loaded overpack tipping-over onto a reinforced concrete pad. The tip-over analysis is provided in Section 3.4. The structural analysis provided in Appendix 3.A demonstrates that the resultant deceleration loading on the MPC as a result of the tip-over accident is less than the design basis 45g's. The analysis shows that the HI-STORM 100 System meets all structural requirements and there is no adverse effect on the structural, confinement, thermal, or subcriticality performance of the MPC. However, the side impact will cause some localized damage to the concrete and outer shell of the overpack in the radial area of impact.

Structural

The structural evaluation of the MPC presented in Section 3.4 demonstrates that under a 45g loading the stresses are well within the allowable values. Analysis presented in Chapter 3 shows that the concrete shields attached to the underside and top of the overpack lid remains attached. As a result of the tip-over accident there will be localized crushing of the concrete in the area of impact.

Thermal

The thermal analysis of the overpack and MPC is based on vertical storage. The thermal consequences of this accident while the overpack is in the horizontal orientation are bounded by the burial under debris accident evaluated in Subsection 11.2.14. Damage to the overpack will be limited as discussed above. As the structural analysis demonstrates that there is no significant change in the MPC or overpack, once the overpack and MPC are returned to their vertical orientation there is no effect on the thermal performance of the system.

Shielding

The effect on the shielding performance of the system as a result of this event is limited to a localized decrease in the shielding thickness of the concrete.

Criticality

There is no effect on the criticality control features of the system as a result of this event.

Confinement

There is no effect on the confinement function of the MPC as a result of this event. As discussed in the structural evaluation above, all stresses remain within allowable values, assuring confinement boundary integrity.

Radiation Protection

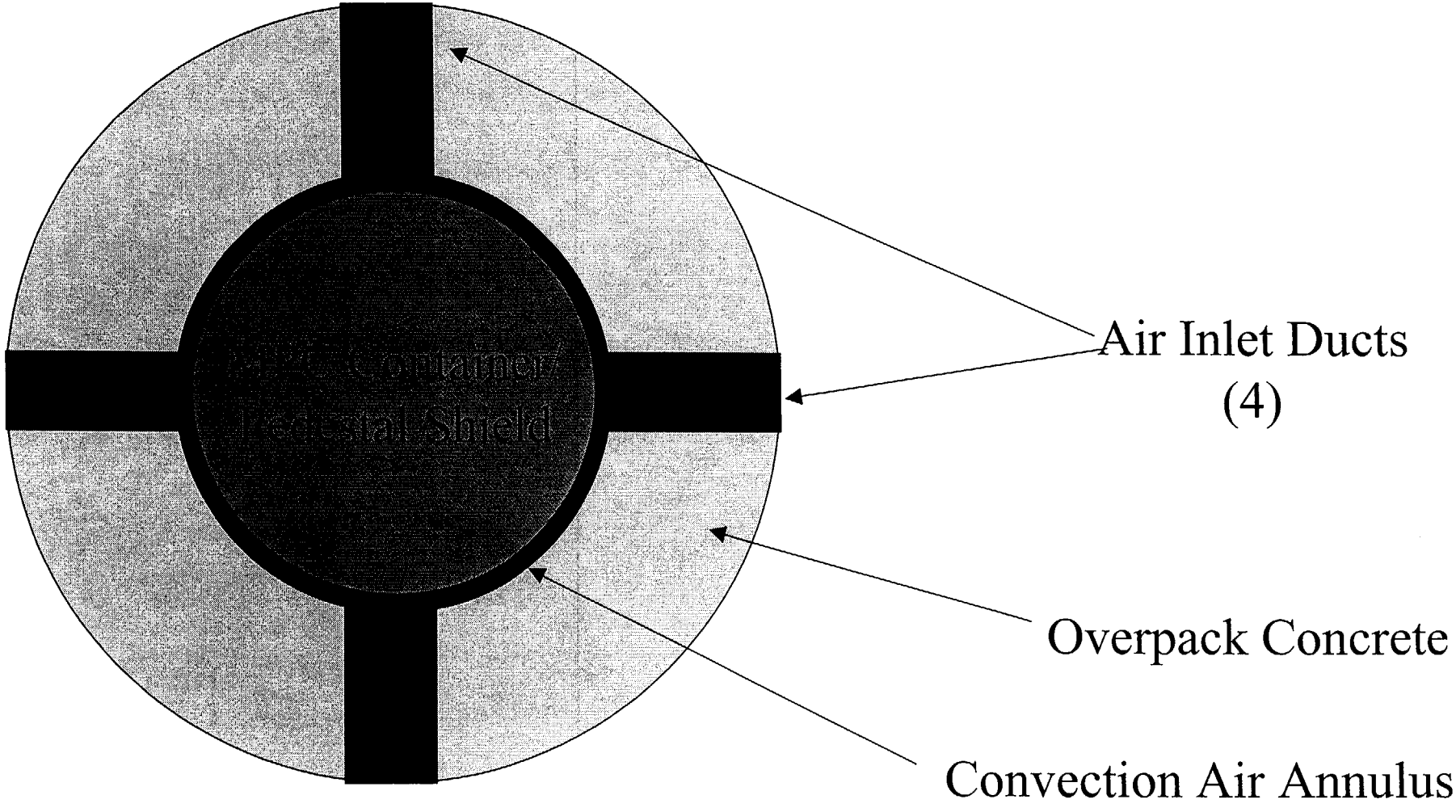
Since there is a very localized reduction in shielding and no effect on the confinement capabilities as discussed above, there is no effect on occupational or public exposures as a result of this accident event.

Based on this evaluation, it is concluded that the accident pressure does not affect the safe operation of the HI-STORM 100 System.

11.2.3.3 Tip-Over Dose Calculations

The tip-over accident could cause localized damage to the radial concrete shield and outer steel shell where the overpack impacts the surface. The overpack surface dose rate in the affected area could increase due to the damage. However, there should be no noticeable increase in the ISFSI site or boundary dose rate, because the affected areas will be small and localized. The analysis of the tip-over accident has shown that the MPC confinement barrier will not be compromised and, therefore, there will be no release of radioactivity or increase in site-boundary dose rates.

Schematic Cross Section of HI-STORM 100 Cask Bottom



Note: Bottom of cask contains approximately 3 inches of steel, not pictured in this drawing

State's
Exhibit 140

Rough Calculations: Dose Emanating from Bottom of Tipped-Over Cask

This calculation attempts to estimate the gamma dose emanating from the bottom of a single tipped-over HI-STORM 100 cask. To do this, I will attempt to calculate the dose rate at the bottom of an unshielded MPC container, given the dose rate at the bottom of a HI-TRAC transfer cask. Once this has been estimated, I will estimate the dose rate on the outside of the bottom of a HI-STORM cask which has been tipped over. I will first calculate the dose rate due to Co-60, then for fuel gammas (assuming all the dose is from Cs-137).

According to the SAR for the HI-STORM 100 cask, the dose rate adjacent to the bottom of a HI-TRAC transfer cask filled with design-basis fuel is 3058.38 mrem/hour. The "fuel gamma" dose rate at the same point is 238.28 mrem/hour. From drawings of the HI-TRAC contained in the HI-STORM SAR, this dose rate is after passing through shielding (2.75 inches of steel and approximately 1 inch of lead, in addition to the meter of air). To calculate the dose before attenuation by the shielding, the following equation is used, solving for D_0.

$$D(x) = D_0 \times e^{-\lambda_s x}$$

The shielding coefficient λ_s can be calculated with the tenth-value layer as follows:

λ_s (1/cm) = ln10/tenth-value layer

Table 1: Tenth Value Layers of Radionuclides

Radionuclide	Gamma Energy	Tenth-value layer			Shielding coefficient		
		Concrete	Steel	Lead	Concrete	Steel	Lead
	(MeV)	(cm)	(cm)	(cm)	(1/cm)	(1/cm)	(1/cm)
Cs-137	0.66	15.7	5.3	2.1	0.1467	0.4345	1.0965
Co-60	1.17, 1.33	20.6	6.9	4.0	0.1118	0.3337	0.5756

1). Co-60

A). Inside Dose Calculation

As was previously mentioned, the dose rate at the bottom of a HI-TRAC cask due to CO-60 was given as 3058.38 mrem/hour adjacent to the cask. Using the above shielding coefficients for steel and lead (ignoring any attenuation in air), we obtain the following value for the dose rate inside the shielding.

State's
Exhibit 141

$$D_{inside} = \frac{D_{outside}}{e^{-(\lambda_L T_L + \lambda_{st} T_{st})}} \;;$$

where the thickness of steel and lead were previously given (2.75 inches and 1 inch, respectively).

Using these values, we obtain a dose rate inside the shielding of **135,748.5 mrem/hour**.

B). Annulus Area Calculation

A central assumption in this calculation is that the majority of the gamma ray dose emanating from the bottom of the cask will travel through the annulus which, under normal conditions, allows for convective heat transfer. The annulus creates a ring in which it is possible for streams of radiation to pass through without being shielded by any concrete, with the only shielding being provided by the steel baseplate, estimated to be 3 inches thick.

The formula for calculating the area of an annulus is as follows:

$A = \pi(r_o^2 - r_i^2)$;

In this case, r_o = 36.75 inches and r_i = 34.1875 inches, based on drawings in the HI-STORM SAR.

Using these numbers, the area of the annulus is calculated to be 571 in^2.

Next, we need to estimate what percentage of the source term is able to stream through this annulus. As a bounding case, we take the percentage of area occupied by the annulus when compared to the area of the bottom of the HI-STORM that is considered part of the source term. This is bounding because we assume that the entire radiation dose is contained in the area bounded by the outer diameter of the annulus, thus arriving at the maximum fractional area for the annulus.

The area bounded by the outer radius (36.75 inches) is 4243 in^2. Therefore the annulus occupies 13.45% of this area.

C). Computation of Dose outside HI-STORM Annulus

We need to make and assumption regarding the fraction of radiation that can emanate directly through the annulus. As a first approximation, we make the assumption that 13.45% of the radiation emanating from the MPC container travels directly through the annulus, unencumbered by any concrete. This is an overestimate, since the MPC container stops at the inner radius of the annulus, meaning that there is no direct path through it. Therefore, as a second approximation, we reduce the area by a factor of 10.

Next, we recomputed the dose rate, this time on the outside of the HI-STORM cask, after passing through approximately 3 inches of steel. From the previous table, the shielding coefficient for Co-60 through steel is .3337 cm^{-1}. The shielding coefficient for Co-60 in air is 7.12e-5 cm^{-1}, which will be applied to obtain a dose rate at 1 meter from the bottom of the cask. The results are provided below for both cases (13.45% and 1.345% of the radiation emanating through the annulus, respectively.)

Table 2: Tipped-Over Cask Dose Rate from Co-60

Case	% of source term unshielded by concrete	Dose Rate inside shielding (mrem/hr)	Dose rate at 1 meter outside HI-STORM bottom (mrem/hour)
1	13.45	18,258	1426
2	1.345	1,825.8	142.6

D). Estimate of dose at site boundary assuming a line of casks overturning

If a line of casks overturn, we can estimate the dose at the boundary assuming a line source. The largest "line source" would consist of 80 casks overturning, creating a line 1,520 feet (463.3 meters). If we assume all 80 tip over so their bottoms are perpendicular to and facing the site boundary, we can estimate a linear source term for the two cases analyzed above as follows:

Table 3: Development of Line Source Term for Co-60

Case	Single Cask Dose rate @ 1 meter (mrem/hour)	Linear Source term (mrem/meter-hour)
1	1426	246.2
2	142.6	24.62

The dose rate from a linear source term decreases linearly with distance by the following formula:

$I_2 = I_1\theta/h$;

Where theta is the angle in the following diagram. In this case, theta =.79 rad, or approximately 45 degrees and h is equal to 555 meters.

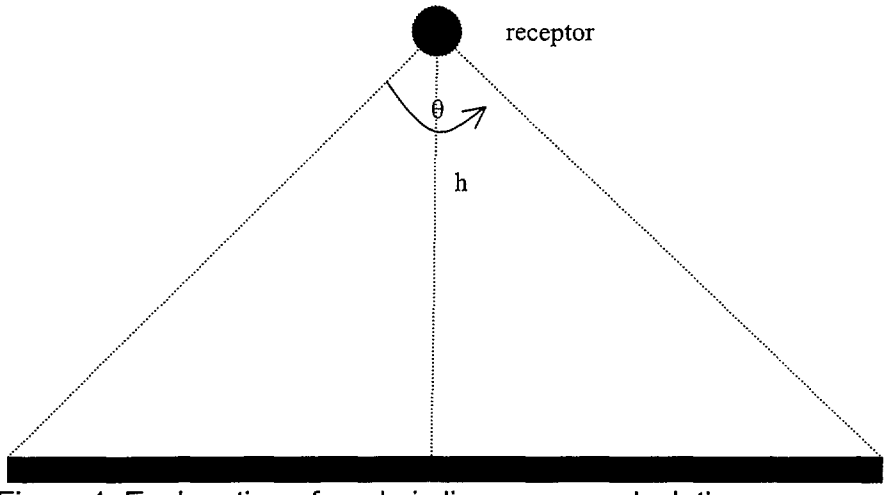


Figure 1: Explanation of angle in line source calculation

Using this formula, we obtain the following dose rates at the site boundary for the two cases.

Table 4: Dose Rate at Controlled Area Boundary from Co-60; No Attenuation in Air

Case	Dose Rate @ 555 meters		
	mrem/hour	mrem/year, 2000 hours/year	mrem/year, 8760 hours/year
1	.35	701	3070
2	.035	70.1	307

However, we did not take into account the shielding by the steel of the casks in this calculation. Therefore, we will discount value using Beer's Law, correcting for the broad-beam geometry.

$I/I_0 = Be^{-ut}$

In this case, μ=7.12e-5 cm^{-1} for Co-60 through air and the buildup factor is approximately 6.7 (based on values from table 6.5.1 of *Handbook of Health Physics and Radiological Health*. Using this results in a reduction by a factor of .133. Thus, the final dose rate due to the postulated line source is given as:

Table 5: Dose Rate at Controlled Area Boundary from Co-60

Case	Dose Rate @ 555 meters, assuming attenuation in air		
	mrem/hour	mrem/year, 2000 hours/year	mrem/year, 8760 hours/year
1	.047	93.6	410
2	.0047	9.36	41.0

2). Cs-137

A). Inside Dose Calculation

The fuel gamma dose rate at the bottom of a HI-TRAC cask was given in the HI-STORM SAR as 238.28 mrem/hour adjacent to the cask. Using the above shielding coefficients for steel and lead (ignoring any attenuation in air), we obtain the following value for the dose rate inside the shielding.

$$D_{inside} = \frac{D_{outside}}{e^{-(\lambda_L T_L + \lambda_{st} T_{st})}} \;;$$

where the thickness of steel and lead were previously given (2.75 inches and 1 inch, respectively).

Using these values, we obtain a dose rate inside the shielding of **80,301.1 mrem/hour**.

B). Annulus Area Calculation

A central assumption in this calculation is that the majority of the gamma ray dose emanating from the bottom of the cask will travel through the annulus which, under normal conditions, allows for convective heat transfer. The annulus creates a ring in which it is possible for streams of radiation to pass through without being shielded by any concrete, with the only shielding being provided by the steel baseplate, estimated to be 3 inches thick.

The formula for calculating the area of an annulus is as follows:

$A = \pi(r_o^2 - r_i^2)$;

In this case, r_o = 36.75 inches and r_i = 34.1875 inches, based on drawings in the HI-STORM SAR.

Using these numbers, the area of the annulus is calculated to be 571 in^2.

Next, we need to estimate what percentage of the source term is able to stream through this annulus. As a bounding case, we take the percentage of area occupied by the annulus when compared to the area of the bottom of the HI-STORM that is considered part of the source term. This is bounding because we assume that the entire radiation dose is contained in the area bounded by the outer diameter of the annulus, thus arriving at the maximum fractional area for the annulus.

The area bounded by the outer radius (36.75 inches) is 4243 in^2. Therefore the annulus occupies 13.45% of this area.

C). Computation of Dose outside HI-STORM Annulus

We need to make and assumption regarding the fraction of radiation that can emanate directly through the annulus. As a first approximation, we make the assumption that 13.45% of the radiation emanating from the MPC container travels directly through the annulus, unencumbered by any concrete. This is an overestimate, since the MPC container stops at the inner radius of the annulus, meaning that there is no direct path through it. Therefore, as a second approximation, we reduce the area by a factor of 10.

Next, we recomputed the dose rate, this time on the outside of the HI-STORM cask, after passing through approximately 3 inches of steel. From the previous table, the shielding coefficient for Cs-137 through steel is .4345 cm^{-1}. The shielding coefficient for Cs-137 in air is 9.31e-5 cm^{-1}, which will be applied to obtain a dose rate at 1 meter from the bottom of the cask. The results are provided below for both cases (13.45% and 1.345% of the radiation emanating through the annulus, respectively.)

Table 6: Tipped-Over Cask Dose Rate from Cs-137

Case	% of source term unshielded by concrete	Dose Rate inside shielding (mrem/hr)	Dose rate at 1 meter outside HI-STORM bottom (mrem/hour)
1	13.45	10,801	391
2	1.345	1,825.8	39.1

3). Estimate of dose at site boundary assuming a line of casks overturning

If a line of casks overturn, we can estimate the dose at the boundary assuming a line source. The largest "line source" would consist of 80 casks overturning, creating a line 1,520 feet (463.3 meters). If we assume all 80 tip over so their bottoms are perpendicular to and facing the site boundary, we can estimate a linear source term for the two cases analyzed above as follows:

Table 7: Development of Line Source Term for Cs-137

Case	Single Cask Dose rate @ 1 meter (mrem/hour)	Linear Source term (mrem/meter-hour)
1	394	67.5
2	39.4	6.75

The dose rate from a linear source term decreases linearly with distance by the following formula:

$I_2 = I_1\theta/h$;

Using this formula, we obtain the following dose rates due to Cs-137 at the site boundary for the two cases.

Table 8: Dose Rate at Controlled Area Boundary from Cs-137; No Attenuation in Air

Case	Dose Rate @ 555 meters		
	mrem/hour	mrem/year, 2000 hours/year	mrem/year, 8760 hours/year
1	.097	194	849
2	.0097	19.4	84.9

However, we did not take into account the shielding by the steel of the casks in this calculation. Therefore, we will discount value using Beer's Law, correcting for the broad-beam geometry.

$I/I_o = Be^{-ut}$

In this case, μ=9.67e-5 cm^{-1} for Cs-137 through air and the buildup factor is approximately 10 (based on values from table 6.5.1 of *Handbook of Health Physics and Radiological Health*. Using this results in a reduction by a factor of .049. Thus, the final Cs-137 dose rate due to the postulated line source is given as:

Table 9: Dose Rate at Controlled Area Boundary from Cs-137

Case	Dose Rate @ 555 meters, assuming attenuation in air		
	mrem/hour	mrem/year, 2000 hours/year	mrem/year, 8760 hours/year
1	.0048	9.52	41.7
2	.00048	0.952	4.17

A

dding up the doses from Co-60 and Cs-137, we obtain the following dose rates at the controlled area boundary"

Table 10: Estimated Dose Rate at Controlled Area Boundary, Multiple Cask Tip-Over

Case	Dose Rate @ 555 meters, assuming attenuation in air		
	mrem/hour	mrem/year, 2000 hours/year	mrem/year, 8760 hours/year
1	.05	103	451
2	.005	10.3	45.1

TABLE 4.2-2

PHYSICAL CHARACTERISTICS OF THE HI-STORM STORAGE CASK

PARAMETER	VALUE
Height	239.5 inches
Outside Diameter	132.5 inches
Capacity	1 loaded canister
Max. Radiation Dose Rate[1] 1 meter from surface: Side Top On contact with surface: Side Top Top vents Bottom vents	 17 mrem/hr 2 mrem/hr 35 mrem/hr 5 mrem/hr 9 mrem/hr 15 mrem/hr
Material of Construction	Concrete (core and lid) Steel (liner and shell)
Weight, maximum	268,334 lb (empty) 348,321 lb (with loaded MPC-24) 355,575 lb (with loaded MPC-68)
Service Life	>100 years

1. Dose rate is based on HI-STORM design basis zircaloy clad fuel for normal conditions.

State's
Exhibit 142

Calculation of Neutron Dose at Elevated Concrete Temperatures

The following calculations are based on information from:

Kaplan, M.F, 1989. *Concrete Radiation Shielding: Nuclear Physics, Concrete Properties, Design and Construction.* Avon, Great Britain: Longman Scientific and Technical.

In order to calculate the loss of neutron shielding associated with evaporation of water in the concrete, it is necessary to discuss briefly the interaction mechanisms at play. Neutrons are grouped into categories based on their energy levels: thermal neutrons have low energies and can be absorbed or captured, while fast neutrons have higher energy and need to be slowed before capture. Table 1, below, shows the energy distribution of fast neutrons used as the design basis fuel in the Holtec HI-STORM SAR.

Table 1: Neutron Energy Distribution of Design Basis Fuel: From HI-STORM SAR

Lower Energy (MeV)	Upper Energy (MeV)	35,000 MWD/MTU, 5-year cooled		45,000 MWD/MTU, 5-year cooled		45,000 MWD/MTU, 9-year cooled	
		Neutrons/s	% of total	Neutrons/s	% of total	Neutrons/s	% of total
0.1	0.4	7.19E+06	3.8%	1.63E+07	3.8%	1.40E+07	3.8%
0.4	0.9	3.68E+07	19.3%	8.33E+07	19.4%	7.15E+07	19.4%
0.9	1.4	3.37E+07	17.7%	7.63E+07	17.8%	6.55E+07	17.7%
1.4	1.85	2.49E+07	13.1%	5.62E+07	13.1%	4.84E+07	13.1%
1.85	3	4.42E+07	23.2%	9.92E+07	23.1%	8.56E+07	23.2%
3	6.43	3.99E+07	21.0%	9.01E+07	21.0%	7.75E+07	21.0%
6.43	20	3.52E+06	1.9%	7.98E+06	1.9%	6.85E+06	1.9%
Totals		1.90E+08	100.0%	4.29E+08	100.0%	3.69E+08	100.0%

For fast neutrons, the *effective removal cross section* (Σ_R) describes the removal of neutrons by a shielding mechanism. It is used in the following equation:

$I=I_0e^{-\Sigma_r T}$; where T is the thickness of the shielding.

Neutron cross-sections are greatly affected by the neutron energy and the atomic weight of the various chemical elements in the shielding medium. However, according to Kaplan's text, the effective removal cross section is considered to be approximately constant for neutron energies between 2 and 12 MeV (Kaplan, pp 235) which accounts for approximately 50% of the neutron distribution given above. For comparison, we will assume the cross sections are constant throughout the range of energies listed above.

For materials (such as concrete) which are comprised of a variety of elements, the total effective removal cross section can be calculated as the sum of the weighted averages of the individual effective removal cross sections. Kaplan has listed various effective removal cross sections for components commonly found in concrete. In Table 2, below, we list the elemental makeup of "ordinary concrete" used in the Kaplan text and the concrete to be used for the HI-STORM 100 overpack. The HI-STORM concrete data is taken from Table 5.3.2 of the HI-STORM SAR

Table 2: Elemental Makeup of Concrete and Neutron Removal Cross Sections

	Ordinary Concrete			**Holtec HI-STORM 100 Overpack Concrete**		
Element	g element/cm^3 concrete	Σ_R/ρ (cm^2/g)	Σ_R (cm^{-1})	G element/cm^3 concrete	Σ_R/ρ (cm^2/g)	Σ_R (cm^{-1})
H	0.015	0.598	0.0090	0.0141	0.598	0.0084
O	1.057	0.0346	0.0366	1.175	0.0346	0.0407
Na	0.041	0.0341	0.0014	0.03995	0.0341	0.001362295

Mg	0.085	0.0333	0.0028	--		
Al	0.137	0.0292	0.0040	0.1128	0.0292	0.00329376
Si	0.487	0.0295	0.0144	0.74025	0.0295	0.021837375
P	0.002	0.0283	0.0001	--		
S	0.002	0.0275	0.0001	--		
K	0.015	0.0247	0.0004	0.04465	0.0247	0.001102855
Ca	0.295	0.0243	0.0072	0.19505	0.0243	0.004739715
Ti	0.011	0.022	0.0002	--		
Mn	0.003	0.0202	0.0001	--		
Fe	0.178	0.0214	0.0038	0.0282	0.0214	0.00060348
Concrete Properties	**2.328**		**0.0799**	**2.35**		**0.0820**

The HI-STORM 100 cask contains a concrete layer 67.95 cm thick. Therefore, we can estimate the attenuation of neutrons by this shielding by solving the exponential absorption equation:

$I/I_o = e^{-\Sigma rT}$

For the "ordinary concrete" used in the Kaplan text, $I/I_o = 0.0044$, while for the HI-STORM overpack concrete $I/I_o = 0.0038$. Further, the HI-STORM 100 SAR has specified a neutron dose rate adjacent to the mid-height of the HI-STORM overpack as 1.88 mrem/hour assuming 45,000 MWD/MTU, 5-year cooled MPC-24 fuel. Using this value as I in the above equation we can solve for I_o to estimate the neutron dose rate assuming no shielding.

$I_o = 1.88$ mrem/hour x $\exp^{.0820 \times 67.95}$ = **495 mrem/hour assuming no neutron shielding by concrete**

Temperature Effects of Neutron Shielding Ability of Concrete

Increased temperature of concrete results in a decrease in the amount of water, which results in an increase in the neutron flux density transmitted through a concrete shield of given thickness. The Kaplan text presents the results of experiments in which the effective removal cross section of concrete were estimated (and experimentally measured) at temperature values of room temperature, 100°C, 200°C, and 300°C. Below, I use these results to estimate the effect on the HI-STORM concrete, assuming that an equivalent loss of hydrogen (by weight %) in the HI-STORM overpack concrete as in the experimental concrete. The same calculations were performed as appear in Table 2 of this report: the relative proportions of the various elements are not included in Table 3 for brevity.

Table 3: Effect of Temperature Increase on Neutron Shielding

		"Ordinary Concrete" used in Kaplan text	HI-STORM 100 Overpack Concrete
Unheated (as cured)	ρ, g/cm^3	2.328	2.35
	H density, g H/cm^3 concrete	0.015	0.0141
	Σ_R (calculated), cm^{-1}	0.0801	0.0820
	Σ_R (measured), cm^{-1}	0.0780	--
100 °C	ρ, g/cm^3	2.258	2.283
	H density, g H/cm^3 concrete	0.007	0.00658
	Σ_R (calculated), cm^{-1}	0.0731	0.0755
	Σ_R (measured), cm^{-1}	0.0735	--
200°C	ρ, g/cm^3	2.238	2.266
	H density, g H/cm^3 concrete	0.005	0.0047

	Σ_R (calculated), cm^{-1}	0.0713	0.0738
	Σ_R (measured), cm^{-1}	0.0724	--
300°C	ρ, g/cm^3	2.227	2.258
	H density, g H/cm^3 concrete	0.004	0.00376
	Σ_R (calculated), cm^{-1}	0.0704	0.0730
	Σ_R (measured), cm^{-1}	0.0702	--
All Water Evaporates	ρ, g/cm^3	2.194	2.242
	H density, g H/cm^3 concrete	0	0
	Σ_R (calculated), cm^{-1}	.0668	.0703
	Σ_R (measured), cm^{-1}	--	--

Previously, we have estimated the unshielded dose rate to be 495 mrem/hour. To estimate what the shielded dose rate is as a function of temperature, we simply repeat the calculation: $I=I_0e^{-\Sigma rT}$ for the varying temperatures. This is done for the HI-STORM 100 cask below.

Table 4: Estimated Dose Rates Due To Neutrons as a Function of Concrete Temperature

Temperature	**Dose Rate Adjacent to Cask Mid-Height (mrem/hour)**
Unheated (as cured)	1.88
100°C	2.94
200°C	3.28
300°C	3.47
All Water Evaporates	4.16

However, the above assumes that thermal neutrons will be attenuated once they are reduced in energy. According to the Kaplan Text, "the concept of an effective removal cross-section is dependent on the presence of hydrogen." (70) If there are insufficient hydrogen atoms to thermalize and consequently contribute to the absorption of a neutron after it has been slowed, then the equations used above may underpredict the amount of radiation emanating from a hydrogen-free shielding material. Usually, concrete contains sufficient hydrogen and is dominated by the collision reactions. When hydrogen is not present, the thermal interactions may dominate from a shielding perspective. If this is the case, the neutron dose rate computed above is likely to be somewhat higher.

For example, the Kaplan text provides values of the thermal diffusion length of a certain type of concrete at two different temperatures: unheated (as cured) and at 100°C). Assuming a shield thickness equal to 67.95 cm, the following I/I_0 values for thermal neutrons are presented:

Table 5: Thermal Diffusion Length as a function of Concrete Temperature

Concrete	Temperature	Density (g/cm^3)	L (cm)	I/I_0
0-HW1	Unheated (as cured)	2.33	6.98	5.92×10^{-5}
0-HW2	100°C	2.26	8.97	5.12×10^{-4}

The density of this concrete is very similar to that of the HI-STORM overpack concrete. For our purposes, we assume they are identical in terms of shielding. If we assume a similar loss in hydrogen content (from .015 to .007 g/cm^3 concrete) as a result of heating to 100°C that was witnessed in the "ordinary concrete" discussed in the Kaplan text, we note that a decrease in hydrogen content by approximately 50% leads to a decrease in thermal neutron shielding by an order of magnitude. If we assume a linear relationship between thermal diffusion length and hydrogen content, we can make the following estimates of loss of thermal shielding as a function of temperature (and consequently, hydrogen content).

Table 6: Estimated Thermal Diffusion Length as a Function of Hydrogen Content

Temperature	Density (g/cm^3)	H content (assumed)	L (cm)	I/I_o
Unheated (as cured)	2.33	.015	6.98	5.92×10^{-5}
100°C	2.26	.007	8.97	5.12×10^{-4}
200°C	2.238	.005	9.66	8.80×10^{-4}
300°C	2.227	.004	10.04	1.15×10^{-3}
No Hydrogen Left	2.194	0	11.95	3.39×10^{-3}

Thus, if the concrete in a HI-STORM 100 cask were to lose all of its water, it is estimated that the amount of thermal neutron radiation passing through would be approximately 57.3 times greater than that calculated in the HI-STORM SAR. In terms of radiation dose, assuming proportionality, this would increase the neutron dose to workers to **1.88 mrem/hour x 57.3, or 108 mrem/hour.**