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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	)	ASLBP No. 97-732-02-ISFSI
(Independent Spent Fuel	)	
Storage Installation)	)	~~June 21, 2001 April 1, 2002~~

AMENDED 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

Template = SECY-055

SECY-02

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. ~~My amended testimony makes corrections to dose calculations that I provided previously. The new dose calculations are found in Exhibit 141a, which is attached herewith. Corrections to my April 1, 2002, testimony can be found in responses to Questions 11, 13, and 20.~~

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),

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

[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.

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 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

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

[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.

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 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?

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[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).

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 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.

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

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 141a. My calculations show the dose rate due to neutrons and gamma rays will increase by 5 ~~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, but the dose may also be ½ the PFS calculated dose. 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

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

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.

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

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

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 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).

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[14] Calculation of Neutron Dose at Elevated Concrete Temperatures.
[15] Testimony of Drs. Khan and Ostadan (cask stability).
[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.

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.

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.

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

<u>CERTIFICATE OF SERVICE</u>

I hereby certify that on June 21, 2002, electronic copies of AMENDED STATE OF UTAH TESTIMONY OF DR. MARVIN RESNIKOFF REGARDING UNIFIED CONTENTION UTAH L/QQ (Seismic Exemption – Dose Exposure) and Exhibit 141a were served electronically on the the persons listed below (unless otherwise noted) with conforming copies to follow by hand on June 24, 2002, or by United States mail first class:

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Diane Curran

Exhibit 141a
Amended Rough Calculations: Dose Emanating from Bottom of Tipped-Over Cask

In this supplementary exhibit to Exhibit 141, I corrected several assumptions in prefiled Exhibit 141 and recalculated the exposures at the fence post. The changed assumptions are the following.

a) In exhibit 141, HI-TRAC was assumed to have a 1" lead thickness. The correct value is 1.5". This increases the original estimate of internal dose rate.
b) In exhibit 141, the HI-STORM was assumed to have a 3" base plate, whereas the correct value is 2". This increases the original estimate of external dose rate at 1 meter.
c) The closest distance from the overturned casks to the fence post was assumed to be 555m in exhibit 141, whereas the correct value is 600m. This decreases the dose.
d) The inverse relation between distance and dose rate was taken in exhibit 141 to be $I = S^*\theta/h$, where h is the distance from the line of overturned casks to the fence post, S is the linear dose rate and θ is the angle to each side of the row of casks, approximately 45°. In fact, the relation is $I = S^*\theta/4\pi h$. This decreases the dose by a factor of 4π.

All gamma dose calculations are identical except for the above changes and correction of minor errors in the previous calculation. The results for gamma dose rates are shown in Tables 11-14 for both the pre-filed testimony and the recalculated dose rates. The final gamma dose results are given for hourly, 2000 hr/y and 8760 hr/y doses, the latter being for a full-time resident.

Following the gamma dose rates I calculated the neutron dose rate from the bottom of the cask that was inadvertently neglected in exhibit 141. The bottom of the HI-TRAC cask and the annulus of the HI-STORM cask have relatively thin layers of lead and steel, neither of which is effective as a neutron absorber. Hydrogen in the form of water or cement is far more effective as a neutron absorber. For neutrons I go through the same calculations as done previously for gamma doses. I limit the area to a ring annulus that effectively reduces the bottom dose rate by 13.45%. I consider the neutron source as a line source and evaluate it at the distance 600m to the fence post. Note we do not take into account the humidity that contains water. We also do not take into account the production of gamma rays due to the interaction between neutrons and steel or lead. The result for neutron dose rate is given in Table 15. The combined neutron plus gamma dose rate is given in Table 16.

Gamma Dose Rate Calculation

Table 1 below, materials parameters, is reproduced from Exhibit 141. Tables 11 through 16 below consecutively follow the numbering of tables from Exhibit 141.

Table 1: Tenth Value Layers of Radionuclides

Radionuclide	Gamma Energy (MeV)	Tenth-Value Layer Concrete (cm)	Tenth-Value Layer Steel (cm)	Tenth-Value Layer Lead (cm)	Shielding Coefficient Concrete (1/cm)	Shielding Coefficient Steel (1/cm)	Shielding Coefficient Lead (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	0.1118	0.3337	0.5756

Table 11: Tipped-Over Cask Gamma Dose Rate

Radionuclide	HI-TRAC Dose Rate Outside (mrem/hr)	Pre-filed Dose Rate Inside (mrem/hr)	Recalculated Dose Rate Inside (mrem/hr)	Prefiled Bottom Dose Rate unshielded by concrete 13.45% (mrem/hr)	Prefiled Bottom Dose Rate unshielded by concrete 1.35% (mrem/hr)	Recalculated Bottom Dose Rate unshielded by concrete 13.45% (mrem/hr)	Recalculated Bottom Dose Rate unshielded by concrete 1.35% (mrem/hr)
Cs-137	238.28	80266.80	323070.89	394.02	39.40	4781.12	478.11
Co-60	3058.38	135772.02	282039.73	1436.10	143.61	6963.07	696.31

Table 12: Line Source Term and Dose Rate at Boundary

Radionuclide	Prefiled Single Cask Dose Rate @ 1 meter 13.45% (mrem/h)	Prefiled Single Cask Dose Rate @ 1 meter 1.35% (mrem/h)	Recalculated Single Cask Dose Rate @ 1 meter 13.45% (mrem/h)	Recalculated Single Cask Dose Rate @ 1 meter 1.35% (mrem/h)	Prefiled Linear Source Term 13.45% (mrem/m-hr)	Prefiled Linear Source Term 1.35% (mrem/m-hr)	Recalculated Linear Source Term 13.45% (mrem/m-hr)	Recalculated Linear Source Term 1.35% (mrem/m-hr)
Cs-137	394.02	39.40	4781.12	478.11	68.08	6.81	826.13	82.61
Co-60	1436.10	143.61	6963.07	696.31	248.14	24.81	1203.14	120.31

Table 13: Dose Rate at Controlled Area Boundary; No Attenuation in Air

	Prefiled Dose Rate @ 555 meters (13.45%)			Recalculated Dose Rate @ 600 meters (13.45%)		
Radionuclide	mrem/hr	2000 hr/yr mrem/yr	8760 hr/yr mrem/yr	mrem/hr	2000 hr/yr mrem/yr	8760 hr/yr mrem/yr
Cs-137	0.0970	194.02	849.80	0.0807	161.48	707.29
Co-60	0.3536	707.14	3097.27	0.1176	235.18	1030.07

Table 14: Dose Rate at Controlled Area Boundary with Attenuation in Air

	Prefiled Dose Rate @ 555 meters (13.45%)			Recalculated Dose Rate @ 600 meters (13.45%)		
Radionuclide	mrem/hr	2000 hr/yr mrem/yr	8760 hr/yr mrem/yr	mrem/hr	2000 hr/yr mrem/yr	8760 hr/yr mrem/yr
Cs-137	0.0045	9.06	39.68	0.0024	4.88	21.37
Co-60	0.0455	91.08	398.93	0.0110	21.99	96.30
		Total =	438.60		Total =	117.67

Neutron Dose Rate Calculation

Table 15: Tipped-Over Cask Neutron Dose Rate

	HI-TRAC Dose Rate Outside (mrem/hr)	Single Cask Dose Rate @ 1 meter 13.45% (mrem/h)	Linear Source Term (mrem/m-hr)	Neutron Dose Rate @ 600 meters (13.45%) mrem/hr	2000 hr/yr mrem/yr	8760 hr/yr mrem/yr
Neutron	554.51	74.58	12.89	0.0013	2.52	11.03

As seen, the neutron dose contributes an additional 10% to the dose rate at the fence post. The combined gamma and neutron dose is shown in Table 16 below. As seen, the total yearly dose for a person continually residing at the fence post is 128.71 mrem/year, and the 50-year dose is then 6.4 rems, above the EPA's Protective Action Guide of 5 rems.

Table 16: Combined Neutron and Gamma Dose Rate

	Dose Rate @ 600 meters (13.45%)		
	mrem/hr	2000 hr/yr mrem/yr	8760 hr/yr mrem/yr
Neutron	0.0013	2.52	11.03
Gamma	0.0134	26.87	117.67
Total	0.0147	29.39	128.71