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STATE OF UTAH

OFFICE OF THE ATTORNEY GENERAL

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March 26, 1999

Secretary
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Attn: Rulemakings and Adjudications

re: Comments on Proposed Rule to add Holtec Hi-Star 1000 Cask System to the List of Approved Spent Fuel Storage Casks.

Dear Secretary:

In response to 64 Fed. Reg. 1542, January 11, 1999, the State of Utah submits comments on the Preliminary Safety Evaluation Report and Proposed Certificate of Compliance for the Holtec HI-STAR 100 Storage Cask. These comments have been prepared with assistance from Marvin Resnikoff, Ph.D., Radioactive Waste Management Associates.

Sincerely,

Denise Chancellor
Assistant Attorney General

Enc:

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Comments from the State of Utah

Preliminary Safety Evaluation Report and Proposed Certificate of Compliance

HI-STAR 100 Storage Cask

March 29, 1998

The State of Utah, with assistance from Marvin Resnikoff, Ph.D. of Radioactive Waste Management Associates, submits these comments on the preliminary Safety Evaluation Report (SER) and proposed Certificate of Compliance (CoC) for the Holtec HI-STAR 100 irradiated fuel storage cask, NRC Docket No. 72-1008. 64 Fed. Reg. 1542 (1999). The HI-STAR 100 is an all-metal cask with an outer metal overpack that encloses a sealed helium-filled canister (MPC) containing irradiated fuel. Although the SER and CoC under review here address storage only, the HI-STAR 100 is designed for both storage and transportation of spent nuclear power plant fuel.

The State has a three-fold interest in the adequacy of the SER and CoC for the HI STAR 100 storage cask. First, its design is virtually identical to the design of the HI STAR 100 transportation cask, which Private Fuel Storage L.L.C. (PFS) proposes to use to transport spent fuel to its proposed independent spent fuel storage installation (ISFSI) in Utah. The only difference between the storage cask and transportation cask is the fact that the transportation cask uses impact limiters and must satisfy hypothetical accident conditions under 10 CFR Part 71. Second, PFS plans to use the HI-STAR 100 storage cask's internal welded canister (multi-purpose canister or MPC) to transport and store fuel. The MPC will hold the irradiated fuel during transportation to the Private Fuel Storage facility. After arrival at the PFS facility, the MPCs will be stored in the HI-STORM 100 concrete overpack at the proposed PFS facility. Third, although PFS intends to use the HI-STORM-100 cask for storage under normal conditions, in case of an accident, the all-metal cask HI-STAR 100 cask will be used as a storage backup.

These comments address the conclusions of the SER, as well as the assertions made by the cask manufacturer, Holtec International, in the Technical Safety Analysis Reports (TSARs) for the HI-STAR and HI-STORM casks. The HI-STAR 100 storage TSAR is Holtec Report HI-941184 (NRC Docket No. 72-1008). The HI-STAR 100 transportation TSAR is Holtec Report HI-951251 (NRC Docket No. 71-9261). The HI-STORM 100 storage TSAR is Holtec Report HI-951312 (NRC Docket No. 72-1014).

The State is in the process of finalizing a confidentiality agreement with Holtec that will allow the State access to the Holtec proprietary version of HI -STAR 100 SAR, Revision 9 and HI-STORM 100 TSAR, Revision 5. The State will submit additional comments, as a proprietary and confidential submittal, after it has received and reviewed Holtec proprietary documents.

General Comments

The HI-STAR 100 design should not be approved, because Holtec has not provided reasonable assurance that the cladding and cask will retain their integrity under normal, off-normal and accident conditions. Moreover, Holtec does not correctly calculate health impacts under bounding accidents. Nor has Holtec evaluated the impact of a sabotage event. Finally, the TSAR and SER do not provide assurance the cask and cladding will retain their integrity under thermal conditions that exist at an ISFSI. Rather than addressing these deficiencies, the NRC's SER has glossed them over. These issues are crucially important to protecting the public health and safety, and therefore must be addressed before the Holtec CoC can be issued.

Specific Comments

Cladding Integrity Under Impact

According to the HI-STAR 100 storage TSAR (Sec. 3.5), the HI-STAR 100 system is designed to withstand a maximum deceleration of 60 g, while a Lawrence Livermore National Laboratories report shows that the most vulnerable fuel can withstand a deceleration of 63 g in the most adverse orientation (side drop).[1] Holtec therefore asserts that fuel rod integrity will be maintained under all accident conditions. In the preliminary SER (at 11-6), the NRC Staff concurs that "there is reasonable assurance that the cladding will maintain confinement integrity during a design basis drop."

In our view, this analysis is incorrect. Holtec and the NRC Staff have not demonstrated a reasonable assurance that the cladding will maintain its integrity.

Holtec's analysis does not provide reasonable assurance for the following reasons: (1) it does not take into account the possible increase in rate of oxidation of cladding of high burnup fuel; (2) Holtec relies for its analysis on a Lawrence Livermore National Laboratories (LLNL) report that fails to distinguish the effects of reactor irradiation on

[1] UCID-21246, Dynamic Impact Effects on Spent Fuel Assemblies, Chum, Witt, Schwartz (October 20, 1987) (LLNL Report).

fuel assemblies; and (3) Holtec also relies on the LLNL Report's incorrect assumption that fuel assemblies act as a static rigid rod. The first factor (increased rate of oxidation) increases the likelihood that fuel cladding may rupture, and, when the three factors are taken together, they compound the likelihood of a release of radioactive materials during a foreseeable drop accident at the proposed ISFSI.

1. Increased Rate of Oxidization.

The NRC's Information Notice IN 98-29, entitled "Predicted Increase in Fuel Rod Cladding Oxidation" (August 3, 1998), provides new information, not considered in the Holtec TSAR, that calls into question Holtec's accident analysis. IN 98-29 discusses the oxidation rate of fuel cladding for high burnup fuel based on reported experiences with Westinghouse's fuel assemblies. NRC advises recipients to "review the information for applicability to their facilities and consider action as appropriate, to avoid similar problems." Nuclear Regulatory Commission, IN 98-29, *Predicted Increase in Fuel Rod Cladding Oxidation* (August 3, 1998) at 1.

The Notice reports that in October of 1997, Westinghouse notified NRC that modification of its fuel cladding corrosion model in its fuel design code to reflect new data on Zircaloy-4 oxidation at high burnup "may create compliance issues for its Integral Fuel Burnable Absorber (IFBA) fuel with Zircaloy-4 cladding." *Id.* at 1. As noted in the Information Notice,

> The modified code may predict higher fuel temperatures and internal pressures at high burnup conditions. This, in turn, may lead to code results that do not meet the Westinghouse criterion prohibiting gap reopening and that do not meet the loss-of-coolant accident (LOCA) criterion in 10 CFR 50.46(b)(2).

Id. Although the problem was initially discovered by Westinghouse with relation to Zircaloy-4 fuel, the Information Notice notes that "the burnup related phenomena, which could result in noncompliance with the oxidation requirements of 10 CFR 50.46, may not be limited to Westinghouse IFBA fuel but might affect any Zircaloy fuel used in high burnup application." IN 98-29 at 2. Thus, the experience at Westinghouse is also germane to any high burnup fuel that may be stored in Holtec casks--not just to Westinghouse fuel.

According to IN 98-92, the increased oxidation of the cladding is a function of the fuel burnup. Oxidization may cause the cladding to become effectively thinner, decreasing its structural integrity. This thinner cladding due to oxidization also lowers the 'g' impact force at which fuel cladding will shatter. Holtec's TSAR relies on the premise that fuel cladding will not shatter for any foreseeable drop. This premise is based on the assumption that it would take a side drop of greater than 63g to damage the cladding. Our spreadsheet calculations, presented below, show that the g loading for high burnup fuel with oxidized cladding approaches 45g. The NRC Staff should not approve the Holtec application unless and until Holtec has factored the information in IN 98-29 into its calculations. The clear implication of IN 98-29 is that the lift height of the HI-STAR 100 cask must be reduced to lower the g-forces on the cladding.

Effects of Changing Variables
Table 4 in *Dynamic Impact Effects on Spent Fuel Assemblies*

	A	B	C	D	E	F	G
Rod array	17x17	17x17	17x17	17x17	17x17	17x17	17x17
Assembly weight (lb)	1450.00	1450.00	1450.00	1450.00	1450.00	1450.00	1450.00
# of rods	264.00	264.00	264.00	264.00	264.00	264.00	264.00
Fueled length (in)	144.00	144.00	144.00	144.00	144.00	144.00	144.00
# of spacers (N)	7.00	7.00	7.00	7.00	7.00	7.00	7.00
L = (fueled length/N-	24.00	24.00	24.00	24.00	24.00	24.00	24.00
E (psi)	1.04E+07	1.04E+07	1.04E+07	1.30E+07	1.04E+07	1.04E+07	1.04E+07
oy (psi)	8.05E+04	8.05E+04	8.05E+04	8.05E+04	4.50E+04	4.50E+04	8.05E+04
t (in)	0.02	0.02	0.02	0.02	0.02	0.02	0.05
ro (in)	0.19	0.19	0.18	0.19	0.19	0.19	0.21
ri (in)	0.16	0.16	0.16	0.16	0.16	0.16	0.16
A (in2)	0.02	0.02	0.02	0.02	0.02	0.02	0.05
I = (1/4*3.14(ro^4-	3.85E-04	3.85E-04	3.09E-04	3.85E-04	3.85E-04	3.85E-04	9.37E-04
W (lb)	0.84	0.84	0.69	0.84	0.84	0.84	1.78
w (lb/in)	0.04	0.04	0.04	0.04	0.04	0.04	0.04
r (in)	0.18	0.18	0.17	0.18	0.18	0.18	0.19
pressure (lb)	2250.00	1187.80	2250.00	2250.00	2250.00	1187.80	2250.00
oa (psi)	8787.50	4639.02	10472.14	8787.50	8787.50	4639.02	4675.00
M (lb-in)	2.32	2.32	2.32	2.32	2.32	2.32	2.32
ob (psi)	1128.70	1128.70	1378.17	1128.70	1128.70	1128.70	519.50
P (lb)	68.56	68.56	55.00	85.69	68.56	68.56	166.87
ga	81.93	81.93	80.06	102.41	81.93	81.93	93.71
gy	63.54	67.21	50.81	63.54	32.08	35.76	145.96

A: Values from Westinghouse specimen (Dynamic Impact Effects...Table 4)
B: Pressure changed to a lower value (value in An Assessment of the Risk...)
C: Thickness of fuel cladding decreased due to oxidation by 17%. Column A's thickness is reduced by 17%.
D: E Modulus changed to higher value (value in An Assessment of the Risk...)
E: Yield stress lowered to half the original value
F: Yield stress lowered and pressure lowered (E and B)
G: Doubling the thickness

Note:
DIE = Dynamic Impact Effects...
AAR = An Assessment of the Risk...
AAR's E modulus was expected to be lower, as they took irradiated zircaloy into account. However, it was not, after conversion.

	AAR	DIE
E modulus	1.30E+07	1.04E+07 psi
Pressure	1187.80	2250.00 psi

2. Irradiated and Unirradiated Fuel Assemblies.

Holtec's TSAR for the HI-STAR 100 storage cask relies for its estimate of g force that will damage fuel cladding upon a 1987 report by LLNL.[2] The LLNL Report fails to take into account the increased brittleness of irradiated fuel assemblies.[3] Because the irradiated fuel assemblies may have been embrittled, they would also be less resistant to impact. During the course of a fuel assembly's life, subatomic particle bombardment, including neutron flux, significantly decreases the assembly's ductility and increases the assembly's yield stress, thereby embrittling the fuel assembly. "Cladding ductility decreases and yield stress increases with increasing neutron fluence."[4]

Furthermore, the proposed HI-STAR 100 will store only irradiated fuel assemblies; thus, the Applicant cannot rely on LLNL's analysis because the LLNL does not account for irradiation and embrittlement, which lower the impact resistance of the fuel assemblies.

These facts are significant when coupled with the increased oxidation rate reported in IN 98-29 because increased oxidation could tangentially cause an increase in cladding embrittlement.[5] Thus, IN 98-29 compounds the LLNL's error in disregarding the brittle characteristics of irradiated fuel cladding.

3. Fuel Assemblies Do Not Act as a Rigid Rod.

Holtec's calculations rely upon LLNL's erroneous assumption that the fuel within the cladding behaves as a rigid rod. Thus, Holtec merely used a static calculation instead of taking into account the dynamic loading upon impact. The LLNL Report specifically states, "It is important to emphasize that the g loadings shown in Figs. 6 and 7 are static loadings."[6] This assumption is incorrect. Instead of a homogenous, rigid rod, the fuel rod consists of fuel pellets stacked like coins within thin tubing. In any impact scenario, the fuel assembly does not act as a rigid rod; rather, it acts as a dynamic system with the fuel impacting the inside of the cladding and creating a greater likelihood of cladding rupture. Holtec has not shown that the assumption of a rigid rod is conservative. The thinner cladding due to the increased oxidation serves to compound this effect because a smaller g force would be required to rupture the assembly. The NRC staff should not approve the

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[2] LLNL Report.

[3] See e.g, UCID-21246, Table 4, which makes no distinction between Young's modulus and yield strength of a range of fuel assemblies.

[4] "Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors," Battelle Pacific Northwest Lab, NUREG/CR-5009 (February 1988)

[5] Thin cladding acquires brittle characteristics at a faster rate than thicker cladding during fuel life. See IN 98-29 at 2 ("If this total oxidation limit were to be exceeded during an accident, the cladding could become embrittled.").

[6] LLNL Report.

Holtec application without a showing by the applicant that its calculations are conservative.

In sum, the newly discovered findings at Westinghouse, as recognized in the NRC's Notice, and the other concerns discussed above, raise significant questions about the adequacy of Holtec's accident analysis.

Health Impact of Accidents

The calculated health impacts under hypothetical accident conditions, discussed in Chapter 7 of Holtec's HI-STAR 100 TSAR, are not conservative. Three issues need to be more fully examined by NRC Staff: the design basis accident, the radiation pathways, and the dose to children.

1. Design basis accident.

Holtec's hypothetical design basis accident condition assumes 100% of the fuel rods are non-mechanically ruptured and the gases and particulates in the fuel rod gap between the cladding and fuel pellet are released to the MPC cavity and then to the external environment. Radiation doses are calculated 100 m from the cask. In the time interval between production of Rev. 4 and Rev. 6 of the TSAR, the NRC Staff requested Holtec to conduct the dose calculations in conformance with the final version of NUREG-1536.[7] The accident analysis in the final version of NUREG-1536 increased the amount of radioactivity to the MPC cavity by 5 orders of magnitude and would have placed doses at 100 m over the EPA's limit of 5 rem. In Rev. 6, Holtec responded to the NRC's request by changing the method of calculating doses to incorporate an extremely small cask leakage rate, rather than assuming 100% of the cask cavity was released to the external environment. TSAR, Rev. 6 at 11.2-15. Thus, Holtec's new analysis increased the radioactivity released to the cask cavity by 5 orders of magnitude, but the leakage rate reduced the amount released from the cask cavity to the environment by more than 5 orders of magnitude. The net effect of this sleight of hand, was to change the design basis accident, so as to reduce the doses to the thyroid and whole body at 100 m. In essence, the NRC staff has allowed the applicant to change the definition of a bounding accident to one that involves 100% fuel rod cladding rupture, with the cask lid intact, i.e., only slight leakage from the cask. The design basis accident no longer represents a loss-of-confinement-barrier accident.

Holtec's attempt to change the design basis accident for storage casks is not only inappropriate, but is unsupported. We strongly disagree that the slight cask leakage, 1.5 x

[7] Nuclear Regulatory Commission, "Standard Review Plan for Dry Cask Storage Systems," NUREG-1536, January 1997.

10^{-4} cm^3/s, constitutes a bounding accident. A scenario that could lead to a greater release rate is a welding error that allows helium to leak from the MPC if a cask is dropped. Leakage of helium will allow the maximum cladding temperature to rise and the fuel rod cladding to rupture. In this case, the percentage of fuel rods that rupture may be less, but the leakage rate from the cask cavity would be greater than assumed by Holtec.

2. Radiation pathways excluded

In Chapter 7, Holtec has calculated the radiation dose to an adult 100 m from the accident, due solely to inhalation of the passing cloud. Other relevant pathways, such as direct radiation from cesium and cobalt-60 deposited on the ground, resuspension of deposited radionuclides, ingestion of contaminated food and water and incidental soil ingestion, are not considered, in violation of 10 CFR 72.24(m).

3. Dose to children not considered

Contrary to the standards in 10 C.F.R. Parts 72 and 20, Holtec has not calculated the dose to children. These standards prescribe dose limits for "an individual outside the controlled area" (10 C.F.R. § 72.24(m)), and "individual members of the public" (10 C.F.R. §§ 20.1301, 20.1302). For purposes of the Part 20 dose standards, the regulations define "individual" as "***any*** human being," and "member of the public" as ***any*** individual except when that individual is receiving an occupational dose." (Emphasis added). The concept of "any individual" clearly includes people other than adult men, *i.e.*, children. Nor does the Atomic Energy Act limit its protection against undue risk to adult males. In fact, NRC regulations already make special exception for the dose to a minor (10 CFR § 20.1207) and the dose to an embryo/fetus (10 CFR § 20.1208) within restricted areas. Further, Regulatory Guide 3.51, "Calculational Models for Estimating Radiation Doses to Man from Airborne Radioactive Materials Resulting from Uranium Milling Operations," also calculates the dose to children and infants by adjusting the organ size, breathing rate and dose conversion factors.

Children are more vulnerable to radiation than adults because of their higher surface-area-to-volume of organs ratio.[8] Other contributing factors include the fact that children have higher soil ingestion rates than adults.[9] Children also have reduced ingestion and inhalation rates compared to adults;[10] nevertheless, the dose to children under a design basis accident is likely to be significantly higher than the dose to an adult. Thus, in order to satisfy the regulations and the Atomic Energy Act, it is necessary to determine whether

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[8] International Commission on Radiological Protection, "1990 Recommendations of the International Commission on Radiological Protection," ICRP 60, 1991, Pergamon Press.

[9] EPA, "Risk Assessment Guidance for Superfund: Volume I - Human Health Evaluation Manual (Part B, Development of Risk-based Preliminary Remediation Goals," EPA/540/R-92/003, December 1991.

[10] Eckerman, KF et al, "Health Risks from Low-Level Environmental Exposure to Radionuclides," Federal Guidance Report No. 13, Part I - Interim Version, prepared for the EPA, 1998.

the dose limits are satisfied for children. In addition, children are at a higher risk than adults of developing cancer because children live longer than adults and their cells grow more rapidly than adults' cells.

Note that it is not the regulation of 25 mr/yr or 100 mr/y or the EPA accident dose limits of 5 rems that are deficient. Rather, it is the NRC Staff's methodology in calculating exposures to children.

Sabotage Event

We disagree that an accident involving 100% fuel rod cladding rupture with slight lid leakage is a bounding accident. *See supra,* discussion on design basis accident. We urge NRC staff to consider the effect of a sabotage event with an anti-tank missile. The NRC already considers the impact of a tornado missile and explosion, but a tornado missile, like an 8" diameter steel rod striking the cask at 126 mph,[11] does not have the impact of an anti-tank missile. Similarly, an explosion with an external pressure of 300 psig does not have the impact of an anti-tank missile. The lack of a comprehensive assessment of the risks of sabotage and terrorism against nuclear waste facilities and shipments is well established. Terrorists have shown that they are capable of exploiting the weak interfaces associated with transportation as well as causing tremendous damage to static structures such as the World Trade Center and the Oklahoma City government building. As NRC Staff is aware, German regulatory authorities have imposed an additional condition on casks, namely, that they be able to withstand the impact of a 1-ton missile impacting a cask at the speed of sound. By this condition, German casks are able to withstand the impact of a jet engine striking a cask. NRC staff could impose additional conditions on dry storage casks and ISFSIs, *e.g.*, the CoC could require that an ISFSI be designed with an earthen berm to remove the line-of-sight.

Since the early 1980s, the NRC has relied on and has poorly interpreted an outdated set of experiments carried out by Sandia and Battelle Columbus Laboratories that measured the release of radioactive materials as a result of cask sabotage. In one of the Sandia experiments, a GE IF-200 truck cask containing one unirradiated fuel assembly was attacked with an M3A1, a military "shaped charge." Although the results "demonstrated that casks could indeed be breached by military explosives and that a considerable fraction of spent fuel could be released by such an attack,"[12] the NRC responded to Sandia's findings by concluding that since only 2/1,000,000 of the total fuel weight was released in inhalable form, the "average radiological consequences of a release in a heavily populated urban area such as New York City would be no early fatalities and less

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[11] Holtec HI-STAR 100 Storage TSAR, Table 2.2.5.

[12] Halstead, Robert J, and James David Ballard, "Nuclear Waste Transportation Security and Safety Issues; The Risk of Terrorism and Sabotage Against Repository Shipments," prepared for the Nevada Agency for Nuclear Projects, Carson City, Nevada, October, 1997, p.25.

that one (0.4) latent cancer fatality."[13] But this analysis is highly deficient, making a large number of questionable assumptions regarding evacuation and failing to include several significant radiation pathways, such as direct gamma exposures from deposited radionuclides. A more recent analysis[14] of a transportation accident in a rural area for a cask holding 14 (not 24) PWR fuel assemblies shows a cost of $620 million and a recovery time of 460 days for the cleanup operation. This cleanup is only to a level that would reduce doses to 500 mrem/year. Based on this accident scenario in a rural setting, a properly conducted, realistic accident scenario in an urban area can be expected to show billions of dollars in cleanup costs and lost revenues. An urban accident would also cause a large number of adverse health effects. Deficient as the analysis of a sabotage event during transportation is, the NRC Staff has never estimated the economic and safety implications of a sabotage event at a fixed storage facility.

The NRC has never explained why it considered the Sandia experiment indicative of what could occur in any type of terrorist attack, no matter the circumstances. Following the publication of these Sandia results, the NRC proposed elimination of many of the safety requirements for shipments of spent fuel aged more than 150 days, such as, no armed guards for the shipments in highly populated areas, no advance notice to the NRC or local law enforcement officials, and no periodic communication between escorts and a communications center.[15] "At least 32 parties submitted more than 100 pages of comments in response to the notice," to which the NRC never publicly responded. *See supra*, Halstead and Ballard, note 12.

In the intervening years since the Sandia experiments, anti-tank weapons with much greater accuracy and penetrating power have been manufactured and widely distributed. These devices could release much more radioactive material. The NRC suspended action on the rule-making, but it inappropriately continues to use the unrevised conclusions in the proposed rule as a basis for its policies on terrorism and sabotage of nuclear shipments.

The numerous terrorist attacks of the last several years have graphically demonstrated that the NRC continues to ignore the risks of sabotage at significant peril to the public. The NRC should adopt the specific recommendations of Halstead and Ballard for creating a realistic, up-to-date terrorism risk assessment. Some of the reference parameters Halstead and Ballard suggest are as follows:

- The reference weapon should be portable anti-tank missiles for their ability to permeate the strong cask materials, their range and availability.

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13 *Id.*, at 26.

14 Sandquist, GM et al, "Exposures and Health Effects from Spent Fuel Transportation," Rogers & Associates, RAE-8339/12-1, November 29, 1985.

15 Halstead and Ballard at 27.

- A 10-year-cooled, medium burn-up, Westinghouse PWR assembly should be the reference spent fuel. A Holtec HI-STAR 100 storage cask loaded with 24 PWR assemblies of the reference fuel would represent a total radioactivity of about 5.5 million Curies. A terrorist incident resulting in a one-percent release would have radiological consequences far greater than those assumed in the HI-STAR TSAR.

- The following two scenarios, at a minimum should be considered: "an attack in which the cask is captured, penetrated by one or more explosive devices, and releases a significant amount (at least one percent) of its radioactive contents; and an attack in which the cask is perforated by one or more armor-piercing rockets or missiles and releases a significant amount (at least one percent) of its radioactive contents."[16]

Note that Halstead and Ballard recommend a 1% release because that is the percentage of unirradiated fuel released in the Sandia sabotage tests.[17] We maintain that a design basis accident should not be the release of 2×10^{-5} or less of the cesium inventory, but 1%, based on the Sandia sabotage tests. Further, it is not simply inhalable-sized particulates that are important. Larger-sized particulates will be released and deposited downwind, giving rise to a direct gamma dose.

Thermal Requirements

The proposed CoC temperature conditions for the Holtec HI-STAR 100 storage cask are not sufficient to guarantee that cladding and neutron shield degradation will be minimized. To reduce high temperatures, NRC staff must incorporate an additional condition into the CoC, a minimum pitch or center-to-center distance between casks. While Holtec has suggested a pitch of 12' or a 4' spacing between casks, this analysis is likely not based on rigorous calculations. Until the State receives the proprietary calculations from Holtec, it cannot comment with specificity on them. However, based on review of similar proprietary calculations for the HI-STORM 100 casks we have reviewed, we are skeptical that the proprietary calculations for the HI-STAR 100 cask are rigorous and sufficient.

Under the present regulatory framework, NRC staff and contractors must show that **individual** casks will not overheat if subjected to normal (average T = 80 °F) and off-normal (average T = 100 °F) temperatures. If the normal or off-normal temperature conditions are satisfied, then the cask may be used in that location. This is similar to the approach for the CoC earthquake and tornado conditions, but with one important difference: individual casks may interact with each other, causing temperature conditions above ambient temperature conditions. As a result, the Holite neutron absorbing material

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16 *Id.* at *xiv*.
17 Sandoval, RP *et al*, *An Assessment of the Safety of Spent Fuel Transportation in Urban Environs*, SAND82-2365, prepared for DOE by Sandia Labs, June 1983.

and the cladding may degrade due to excessive heat. In the HI-STAR 100 TSAR, the presence of adjacent casks and the concrete pad may not be correctly taken into account, as far as one can determine from Holtec's sketchy nonproprietary analysis. This should be properly addressed in the SER and CoC.

If the center-to-center distance between adjacent HI-STAR 100 casks is too small, casks may thermally interact with each other, effectively increasing the ambient temperature. According to Holtec's TSAR, the overpack shell outside surface temperatures are 229 °F and 249 °F under normal and off-normal temperature conditions.[18] In the most extreme example, if adjacent casks are in immediate contact, instead of the ambient temperature being 80 °F under normal conditions, it would be 229 °F. As the casks are moved away from each other, at some distance the casks become thermally independent of each other. Holtec attempts to calculate this distance in Fig. 4.4.5 by assuming a radiative blocking factor due to the presence of other casks. But the situation at an ISFSI is far more complicated. It is not a blocking factor so much as the presence of adjacent heat sources at 229 °F. The effective ambient temperature will be raised as the casks interact with each other. The distance at which casks will act independently of each other must be calculated by Holtec and included in the CoC. For the HI-STAR 100 cask, the critical temperature is 300 °F for the inner surface of the Holite neutron absorbing material that surrounds the metal cask. The maximum temperatures of the Holite under normal and off-normal conditions are 274 °F and 294 °F, respectively. That is, the HI-STAR 100 is already operating with a thin safety margin, not accounting for the interaction between casks.

To take into account the interaction of casks, the following factors must be incorporated into the calculation. As a first approximation, Holtec could assume adjacent casks at the same temperature, $T_0 = 229$ °F. Insolation, average pad temperature, external convective air currents, and wind speed must also be incorporated into the model. The surface temperature of the center cask, T_1, could then be calculated. In the next iteration, the adjacent casks could be taken at temperature T_1 and a new temperature for the center cask could be calculated T_2. Holtec could then determine whether the series T_0, T_1, T_2 is converging to some asymptotic value. If the value for the inner surface of the neutron shield exceeds 300 °F, the casks must be spaced further apart.

As the situation presently stands, the SER and CoC are deficient. The maximum cladding temperature or temperature of the neutron shield inner surface has not be correctly calculated. NRC staff and Holtec are assuming there is no interaction between the casks. This assumption is not conservative.

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[18] Holtec, Topical Safety Analysis Report for the HI-STAR 100 Cask System, Holtec Report HI-941184, NRC Docket No. 72-1008, Table 11.1.

Removable Surface Contamination

The TSAR includes Technical Specifications for removable surface contamination.[19] If the smearable contamination exceeds 2200 dpm/100 cm^2 from gamma and beta emitting sources, the Technical Specifications require that the accessible surface be flushed or pressure washed. "If the smearable contamination limits still cannot be reduced to acceptable levels, evaluate and perform alternative actions up to, and including, removal of the MPC from the HI-STAR 100 overpack after removing the spent fuel from the MPC."[20] These conditions cannot be met at the proposed off-site Skull Valley ISFSI in Utah. No provisions exist for decontaminating casks under PFS's "start clean, stay clean" philosophy. PFS's proposed policy is to return casks that are contaminated above regulatory limits back to reactor sites. No provisions would exist at PFS for removing the MPC from the HI-STAR 100 overpack. In recognition of the conflict between the Tech Specs for the HI-STAR 100 and design of the PFS facility (and possibly other ISFSIs), the NRC should specify that all users of the HI-STAR 100 have the capability to remove smearable contamination onsite.

Future Rulemaking Procedures

The State of Utah strongly disagrees with any proposal by the NRC to approve future additions and revisions to the list of approved spent fuel storage casks as direct final rules. Under such a procedure there would be no proposed rule. Instead, the rule would become final within 75 days after publication unless NRC receives "significant adverse comments on the direct final rule within 30 days after publication." 64 Fed. Reg. at 1543. On receipt of such significantly adverse comments, NRC would withdraw the rule, address the comments, then publish a final rule. First, the premise underlying NRC's proposed procedural change -- that "additions and revisions to the list of approved spent fuel storage casks are noncontroversial and routine" -- is inaccurate. The above comments show that NRC's approval is not "routine." Moreover, given the past problems, such as hairline cracks associated with dry storage casks, it is imperative that future approval or revision to the list of approved casks be subject to adequate and rigorous public scrutiny. Second, a direct final rule reduces to 30 days the period of time for effective public comment. This is an insufficient time period to review and prepare comments that may be "significantly adverse" to cause NRC to withdraw the published final rule. Third, a direct final rule will diminish the public role in commenting and affecting the outcome of rulemaking procedure. It is more likely that NRC will give due consideration to comments at the propose rule stage; any comments at the final rule stage

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[19] *Id.* at 12.3-16.

[20] *Id.*

would have to be "significantly adverse" for NRC to reverse course and withdraw the direct final rule.

Safety considerations are too important for NRC to expedite the approval process at the expense of diminishing the public's role in commenting on the approval of spent nuclear fuel casks.