Cask Size and Weight Reduction Through the Use of Depleted Uranium Dioxide–Concrete Material

M. Jonathan Haire, Ph.D.
Oak Ridge National Laboratory
P.O. Box 2008
Oak Ridge, TN 37831-6166
Tel: (865) 574-7141
Fax: (865) 574-0382
E-mail: hairemj@ornl.gov

Sergiy Y. Lobach, Graduate Student
Oregon State University
2207 NW Fillmore, Apt. 31
Corvallis, OR 97330
Tel: (541) 758-0216
E-mail: lobachs@engr.otst.edu

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Sergiy Y. Lobach, Student
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ABSTRACT

Newly developed depleted uranium (DU) composite materials enable fabrication of spent nuclear fuel (SNF) transport and storage casks that are smaller and lighter in weight than casks made with conventional materials. One such material is DU dioxide (DUO_2)–concrete, so-called DUCRETE™. This work examines the radiation shielding efficiency of DUCRETE as compared with that of a conventional concrete cask that holds 32 pressurized-water-reactor SNF assemblies. In this analysis, conventional concrete shielding material is replaced with DUCRETE. The thickness of the DUCRETE shielding is adjusted to give the same radiation surface dose, 200 mrem/h, as the conventional concrete cask. It was found that the concrete shielding thickness decreased from 71 cm to 20 cm and that the cask radial cross-section shielding area was reduced ~50%. The weight was reduced ~21%, from 154 tons to ~127 tons. Should one choose to add an extra outer ring of SNF assemblies, the number of such assemblies would increase from 32 to 52. In this case, the outside cask diameter would still decrease, from 169 cm to137 cm. However, the weight would increase somewhat from 156 tons to 177 tons. Neutron cask surface dose is only ~10% of the gamma dose. These reduced sizes and weights will significantly influence the design of next-generation SNF casks.

INTRODUCTION

Newly developed depleted uranium (DU) composite materials enable fabrication of spent nuclear fuel (SNF) transport and storage casks that are smaller and lighter in weight than casks made with conventional materials. One such material is DU–steel cermets [1]; another is DUCRETE™.

This work examines the possible use of DUCRETE as a shielding material in SNF storage casks. DUCRETE consists of a DU ceramic aggregate (DUAGG™), which replaces the coarse aggregate used in standard concrete. DUAGG briquettes are DUO_2 particles that are pressed and solidified by liquid-phase sintering. DUAGG is a very dense (>95% of theoretical UO_2 density), stable, low-cost, coarse aggregate that is combined with portland cement, sand, and water in the same volumetric ratios used for ordinary concrete. DUCRETE can have a density ranging from ~6.0 to 7.2 g/cm^3. This material efficiently shields gamma radiation because of the uranium, and shields neutrons because of water bonded in the concrete. Figure 1 shows the effectiveness of using DUCRETE as a gamma-shielding material compared with the use of competing materials. Only uranium metal is a better gamma-shielding material; however, uranium metal is much more expensive to fabricate. DUCRETE casks have smaller size, lower weight, and higher heat conductivity, therefore allowing for a higher decay heat content than conventional concrete casks and greater resistance to assault [2]. This work describes a radiation shielding evaluation where ordinary concrete shielding in a cask, such as that shown in Fig. 2, is replaced with DUCRETE. The outside-surface gamma-radiation dose is held the same. The radii of the DUCRETE casks are adjusted to give an outside-surface dose of 200mR/h. One-inch stainless steel inside and outside liners for DUCRETE are used for all calculations.

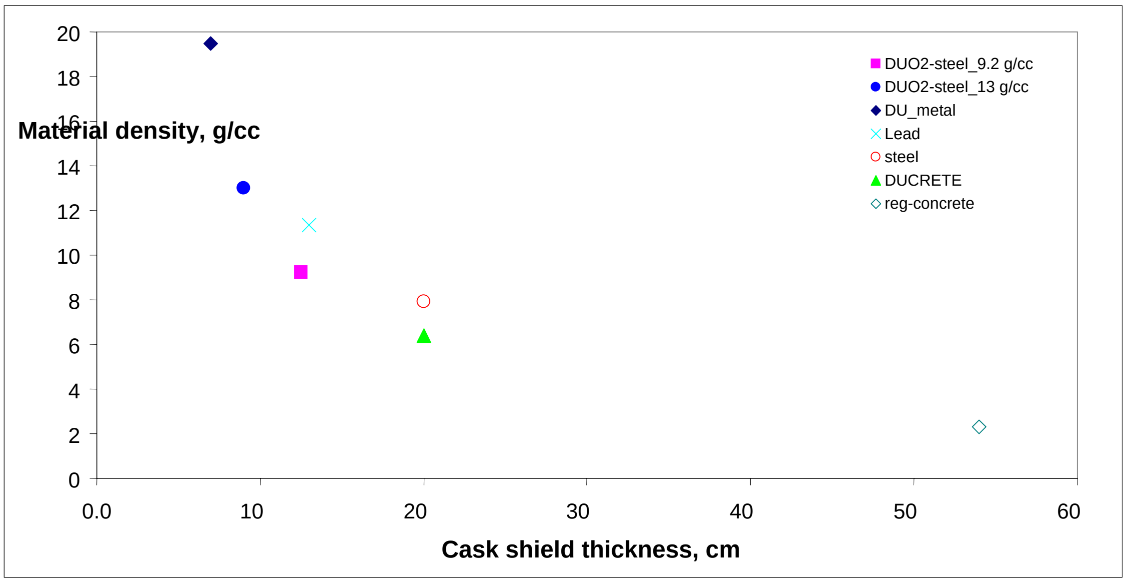

Fig. 1. Comparison of cask-wall thicknesses required to attenuate gamma dose from pressurized-water-reactor (PWR) SNF to 200-mrem/h surface dose at cask midpoint.

ASSUMPTIONS AND METHODOLOGY

Cask Description

The SNF storage cask shown in Fig. 2, containing 32 PWR SNF assemblies, was modeled using the radial dimensions as given in Fig. 3. The height of the cask was assumed to be 336 cm in all cases. One-inch-thick inner and outer steel liners surround and seal the DUCRETE. DUCRETE with a density of >7 g/cm^3 has been fabricated. However, a DUCRETE density of 6.4 g/cm^3 was used in all the calculations. Although DUCRETE at a density of 6.4 g/cm^3 has a compression strength of 670 kg force/cm^2, no credit is taken for this strength; that is, no rebar or structural steel is present between the steel liners. It is believed that DUCRETE cask licensing will be easier if no credit is taken for the concrete strength; the cask relies solely on the stainless steel liners for its mechanical and structural properties.

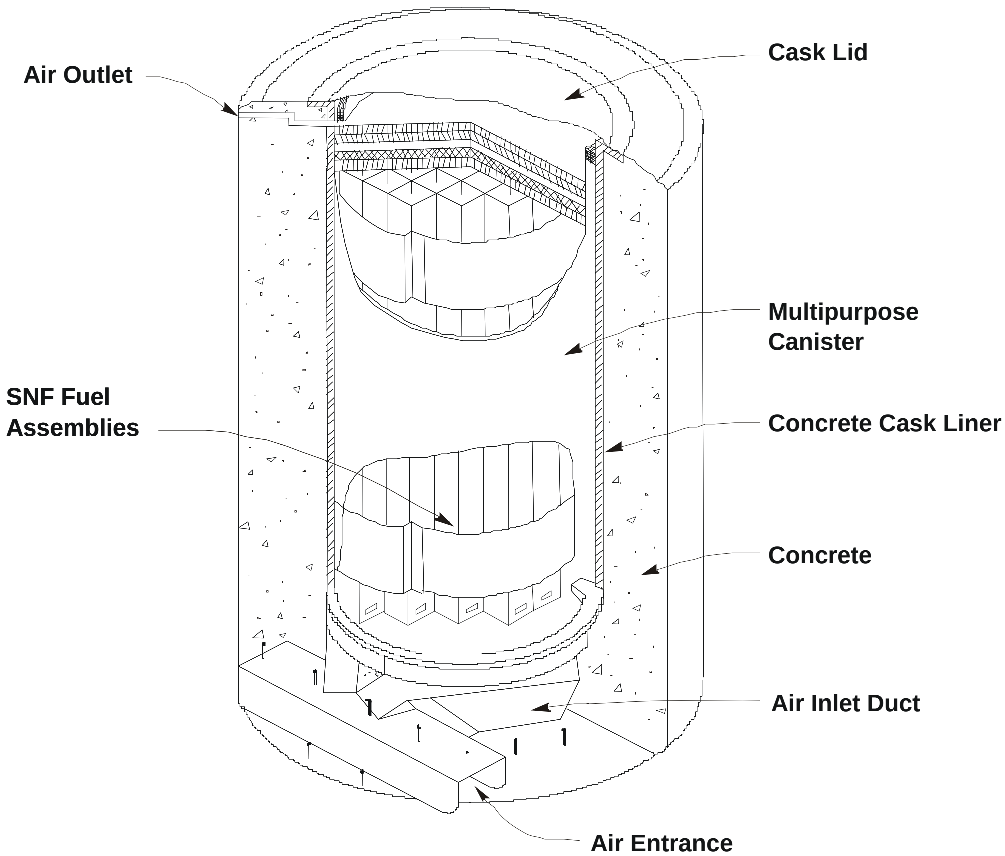

Fig. 2. Schematic of conceptual SNF concrete storage cask.

The ordinary concrete shielding material shown in Fig. 2 was replaced with DUCRETE in this work. The resulting smaller shielding-wall thickness (outside radius) was calculated while holding the surface radiation dose to a constant 200 mrem/h. Next, an additional ring of SNF was placed in the multipurpose canister (MPC) to give a total of 52 PWR SNF assembles; the wall thickness was then again calculated while holding the cask surface dose constant.

Shielding Calculations

The source radiation term for these shielding calculations was obtained from an output file of the SAS2 code program in the SCALE 5 package. The SCALE code system [3] is available from the Radiation Safety Information Computational Center at Oak Ridge National Laboratory. The MPC is assumed to generate 34 kW of heat. All calculations performed for this project assumed that the storage cask contained 32 PWR spent fuel assemblies (see Table I).

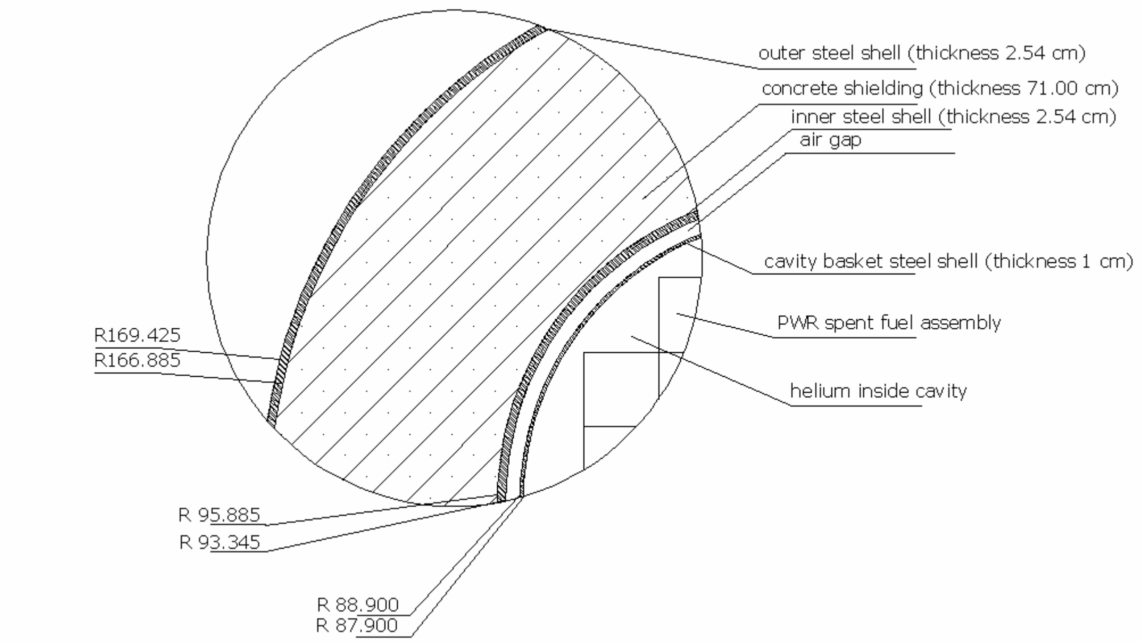

Fig. 3. Radii and thicknesses of materials of the inner cavity and shielding layers.

Table I. Input data for ORIGEN-ARP Express for 15 × 15 PWR spent fuel

Parameter	**Input**
Fuel type	15 × 15
Uranium	1.18916E+07 g
Enrichment	4.0%
Burnup	40,000 MWd/MTU
Cycles	3
Cooling time	5 years
Average power	40 MW/MTU

The SAS2 module from SCALE 5 was used for all shielding calculations. The SAS2 module of SCALE uses standard SCALE composition libraries and the functional modules BONAMI-S, NITAWL-S, XSDRNPM-S, and XSDOSE to perform one-dimensional shielding calculations. The radiation calculations used 27 energy groups of neutrons and 18 energy groups of gamma.

RESULTS AND ANALYSIS

Variations in Density

DUCRETE can be manufactured with a range of densities. Figure 4 shows how the surface dose varies for different DUCRETE densities at a constant 20-cm DUCRETE wall thickness. Note the low contribution of neutrons to the total dose.

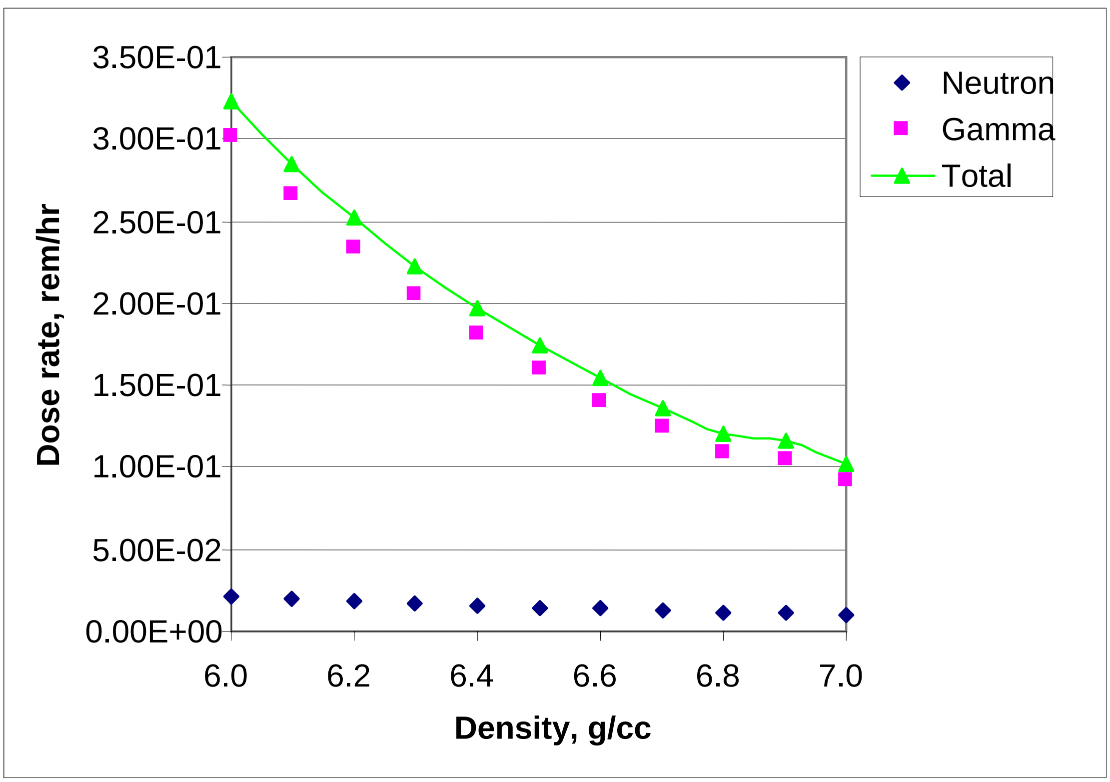

Fig. 4. Surface dose for various DUCRETE densities for a constant wall thickness of 20 cm.

Radiation Shielding Analysis

Figure 5 summarizes the radiation shielding analysis of this work.

Figure 5a shows the cross section for the reference conventional concrete storage cask for PWR fuel that contains neutron flux traps. If the U.S. Nuclear Regulatory Commission (NRC) allows burnup credit for PWR fuel, then the cask represented by cross section shown in Fig. 5b becomes applicable. Most electric utilities that own PWRs are buying casks such as that shown in Fig. 5b in anticipation of NRC approval of burnup credit for PWR fuel during storage. Figure 5c represents the cask shown in Fig. 5b when DUCRETE replaces the conventional concrete. Note the large reduction in shielding thickness from 71 cm to 20 cm. Similarly, the weight is reduced from 156 tons to 127 tons. There is no difference between inner radii of stainless steel liners in cases 5a, 5b, and 5c. In case 5d, an additional ring of SNF is added to the reference case (Fig. 5b) and the cavity radius increases from 87.9 cm to 103 cm. Even when an extra ring of SNF assembles is added, the outside radius of the cask is reduced from 169 cm to 137 cm. However, the total weight of the cask increases from 156 tons to 177 tons.

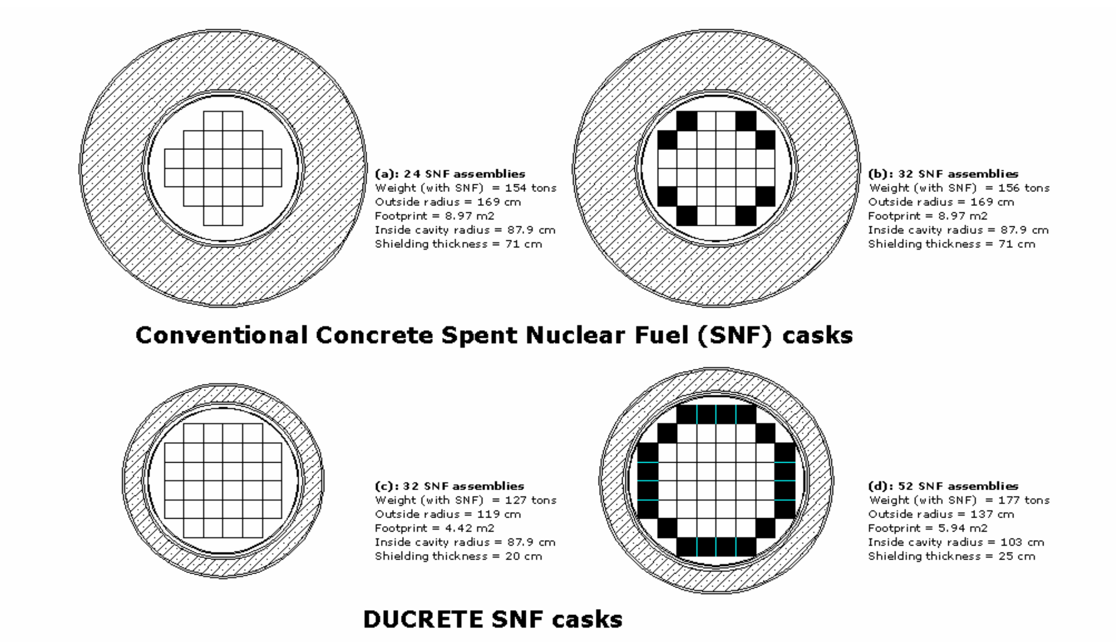

Fig. 5. Cask size and weight reduction through the use of DUCRETE.

Table II. Estimated Cask Component Weights

Case		Cylinder	Top	Bottom	Multipurpose canister (MPC) - empty	Fuel assemblies[a]	Total weight	Cask weight w/o MPC and SNF
24 SNF assemblies with flux traps: conventional concrete	kg	87,038	13,620	13,620	20,430	19,200	153,908	114,278
	lb	*19,1714*	*30,000*	*30,000*	*45,000*	*42,290*	*339,004*	*251,714*
	Metric tons[b]	87	14	14	20	19	154	114
32 SNF assemblies[c]: conventional concrete	kg	87,038	13,620	13,620	16,344	25,600	156,222	114,278
	lb	*191,714*	*30,000*	*30,000*	*36,000*	*56,387*	*344,101*	*251,714*
	Metric tons	87	14	14	16	26	156	114
32 SNF[c]: DUCRETE	kg	58,450	13,620	13,620	16,344	25,600	127,634	85,690
	lb	*128,745*	*30,000*	*30,000*	*36,000*	*56,387*	*281,133*	*188,745*
	Metric tons	58	14	14	16	26	127	86
52 SNF[c]: DUCRETE	kg	81,123	15,000	15,000	25,000	41,600	177,723	111,123
	lb	*178,687*	*33,039*	*33,039*	*55,066*	*91,630*	*391,462*	*244,766*
	Metric tons	81	15	15	25	42	177	111

[a]Weight of an individual PWR fuel assembly = 800 kg.

[b]1 metric ton = 2205 lb.

[c]Without neutron flux traps.

Cask Weight Calculations

Table II gives the weight of major components of SNF casks. If the weight of each individual component can be kept below 120 tons, DUCRETE cask components can be shipped via conventional railcar. Alternatively, if the component weight is below 36 tons, the component can be shipped via truck. Components can be manufactured at a central location, shipped separately as individual components, and assembled into an integrated cask.

ASSAULT RESISTANCE

Russian analysis [2] indicates that if ordinary concrete is replaced by DUCRETE without changing the size of the storage cask, the cask resistance to damage caused by hollow-charge shells, shock waves, and aircraft impacts will be improved by a factor of 2 to 3. If the radius of the cask internal cavity is increased, through the use of DUCRETE, to increase cask capacity, the defense (resistance) properties of the cask will increase by a factor of 1 to 1.5. It is concluded that when replacing ordinary concrete with DUCRETE, an optimum relationship between capacity and defense properties can be obtained. Further evaluations are planned as a prototype DUCRETE cask is designed, fabricated, and tested.

SUMMARY

Casks made with DUCRETE will be smaller in size and lighter in weight for a given number of SNF assembles and have higher heat transfer rates, thereby enabling the storage of higher-energy (shorter-cooldown-time) SNF. Such casks are also more assault resistant. For a 32-SNF-assembly cask, the shielding thickness of conventional concrete, 71 cm, is reduced to 20 cm when DUCRETE is used. The cask radial cross-section area is reduced from 8.97 m^2 to 4.42 m^2. The cask weight is reduced from 156 tons to 127 tons. Gamma radiation (not neutron) is the dominant contributor to total cask surface dose.

REFERENCES

1. M.J. HAIRE and P.M. SWANEY, "Cask Size and Weight Reduction Through the Use of Depleted uranium Dioxide (DUO_2)–Steel Cermet Material," Waste Management 2005 Symposium, Tucson, Arizona, February 27–March 3, 2005, Waste Management Symposia, Inc. (2005).

2. O. G. ALEKSEEV, V. Z. MATVEEV, A. I. MORENKO, R. I. II'KAEY, and V. I. SHAPOVALOV, "Estimation of Terrorist Attack Resistibility of Dual-Purpose Cask TP-117 with DU (Depleted Uranium) Gamma Shield," 14th International Symposium on the Packaging and Transportation of Radioactive Materials, Berlin, Germany, September 20–24, 2004, PATRAM 2004 Conference Agency (2005).

3. *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluations*, NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), Vols. I, II, and III (May 2000). Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-545.