DRAFT 12 August, 2005

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

For the
Waste Management 2006 Conference
Tucson, Arizona
February 26–March 2, 2006

*Oak Ridge National Laboratory, managed by UT-Battelle, LLC, for the U.S. Department of Energy under contract DE-AC05-00OR22725.

Cask Size and Weight Reduction Through the Use of Depleted Uranium Dioxide (DUO_2)–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, Student
Oregon State University
2207 NW Fillmore, Apt. 31.
Corvallis, OR 97330
Tel: (541) 758-0216
E-mail: lobachs@engr.otst.edu

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^{TM}. This work examines the radiation shielding efficiency of DUCRETE^{TM} as compared with a conventional concrete cask that holds 32 pressurized water reactor (PWR) SNF assemblies. In this analysis, conventional concrete shielding material is replaced with a DUCRETE^{TM}. The thickness of the DUCRETE^{TM} shielding is adjusted to give the same radiation surface dose, 200 mrem/hr, as the conventional concrete cask. It was found that the concrete shielding thickness decreased from 71 to 20 cm; the cask radial cross-section shielding area is 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 SNF would increase from 32 to 52. In this case, the outside cask diameter would still decrease, from 169 cm to137 in. But, the weight would increase somewhat from 156 to 177 t. 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 (ref. WM'05 conference), another is DCURETE^{TM}.

This work examines the possible use of DUCRETE^{TM} as a shielding material in SNF storage casks. DUCRETE^{TM} consists of a DU ceramic, depleted uranium aggregate (DUAGG^{TM}), that replaces coarse aggregate used in standard concrete. DUAGG^{TM} briquettes are DUO_2 particles that are pressed and solidified by liquid phase sintering. A photograph of DUAGG briquettes is shown Figure 1. DUAGG^{TM} 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^{TM} 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 neutrons because of water bonded in the concrete. Figure 2 shows the effectiveness of using DUCRETE^{TM} as a gamma shielding material compared to competing materials. Only uranium metal is a better gamma shielding material, but, uranium metal is much more expensive to fabricate. DUCRETE^{TM} casks have smaller size; lower weight; higher heat conductivity and therefore allows for a higher decay heat content than conventional concrete casks; and greater resistance to assault (Berlin conf.). This work describes a radiation shielding evaluation where ordinary concrete shielding in a cask, such as that shown in Figure 3, is replaced with DUCRETE^{TM}. The outside surface gamma radiation dose is held the same. Radii of DUCRETE^{TM} casks are adjusted to give an outside surface dose of 200mR/hr. One inch stainless steel inside and outside liners for DUCRETE^{TM} are used for all calculations.

Figure 1. DUAGG^{TM} Briquettes

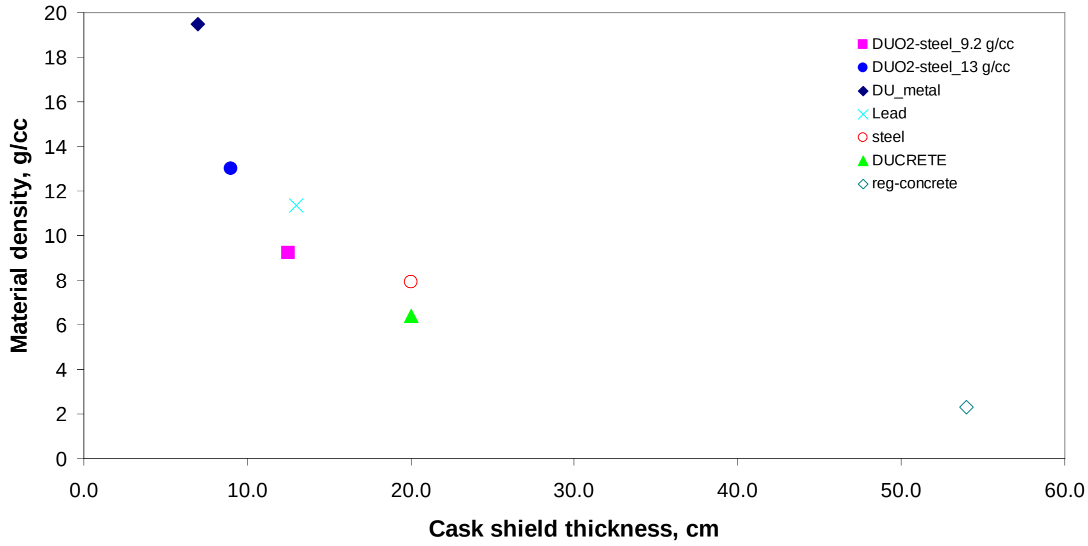

Fig. 2. Comparison of Cask Wall Thicknesses Required to Attenuate Gamma Dose from Pressurized Water Reactor (PWR) SNF to 200 mrem/hr Surface Dose at Cask Mid-Point.

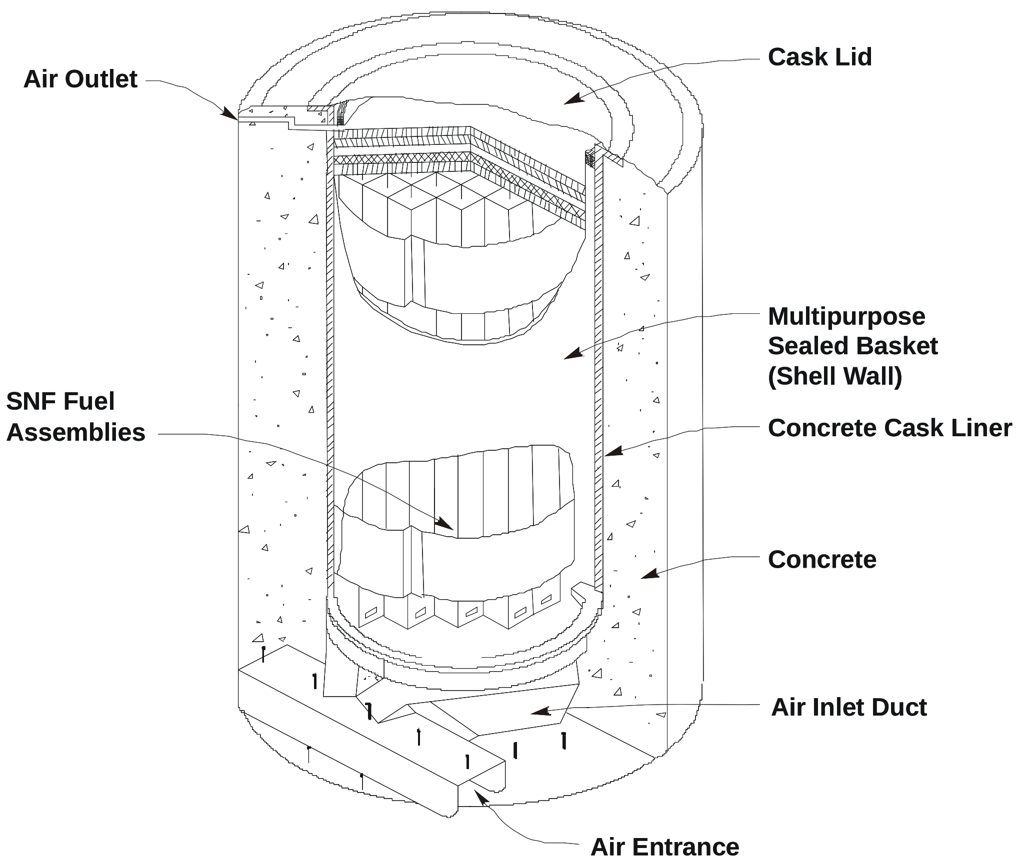

Fig. 3. Schematic of Conceptual SNF Concrete Storage Cask.

ASSUMPTIONS AND METHODOLOGY

Cask Description.
The SNF storage cask shown in Figure 3, containing 32 PWR SNF, was modeled using the radial dimensions are given in Figure 4. The height of the cask was assumed to be 336 cm in all cases. One inch thick inner and outer steel liners surround and seals the DUCRETE™. A DUCRETE™ density of 6.4 g/cm^3 was used in all calculations. DUCRETE™ with a density of more than 7 g/cm^3 has been fabricated. 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 steel liners. It is believed that DUCRETE™ cask licensing will be easier if no credit is taken for concrete strength; the cask rely solely on the stainless steel liners for cask mechanical, structural properties.

The ordinary concrete shielding material in Figure 3 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/hr. Next, an additional ring of SNF was placed in the multi-purpose canister to give a total of 52 pressurized water reactor (PWR) SNF assembles, and the wall thickness again calculated while holding the cask surface dose constant.

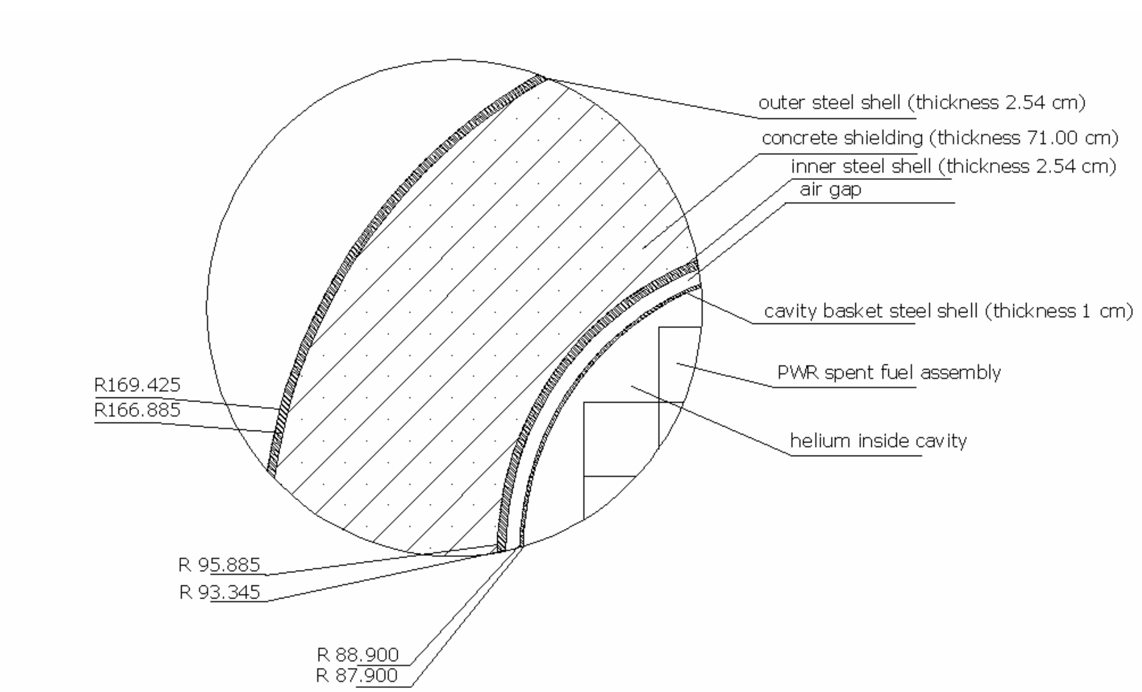

Figure 4: Radii and Thicknesses of Materials of the Inner Cavity and Shielding Layers

Shielding Calculations.
The source radiation term for these shielding calculations was obtained from an output file of tehSAS2 code program in the SCALE 5 package. The SCALE code system (ref. SCALE) is available from the Radiation Safety Information Computational Center at ORNL. The multi-

purpose canister (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).

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 the SCALE 5 was used for all shielding calculations. The SAS2 module of SCALE use 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 5 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. Similarly, Figure 6 shows how DUCRETE™ shielding wall thicknesses vary with different DUCRETE™ densities.

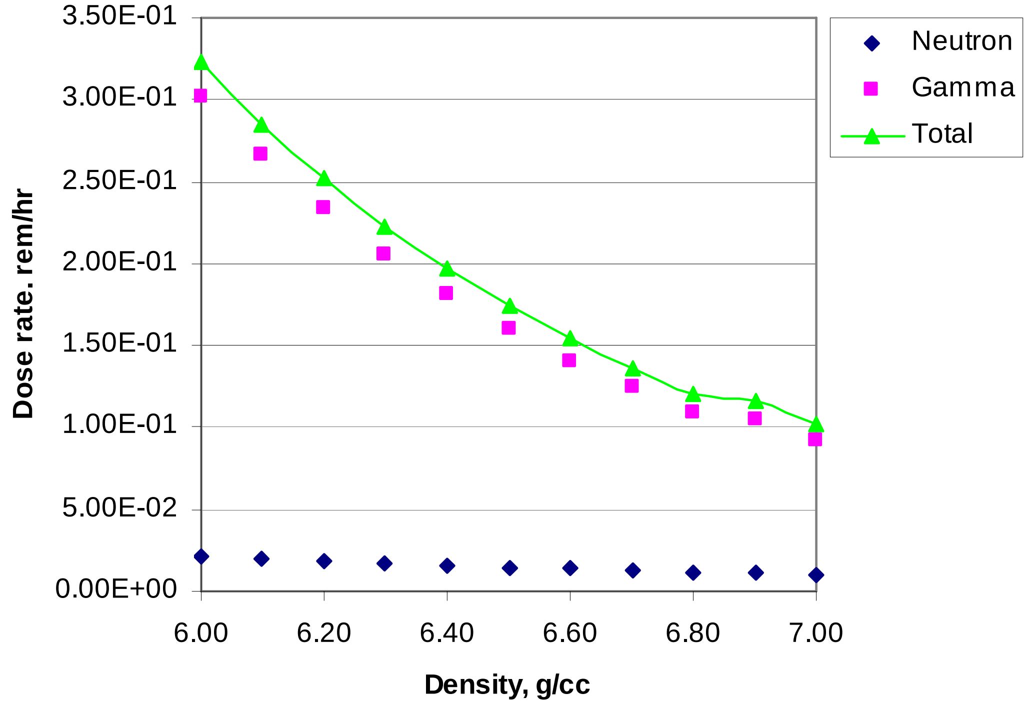

Fig. 5. Surface dose for various DUCRETE™ densities for a constant wall thickness of 20 cm.

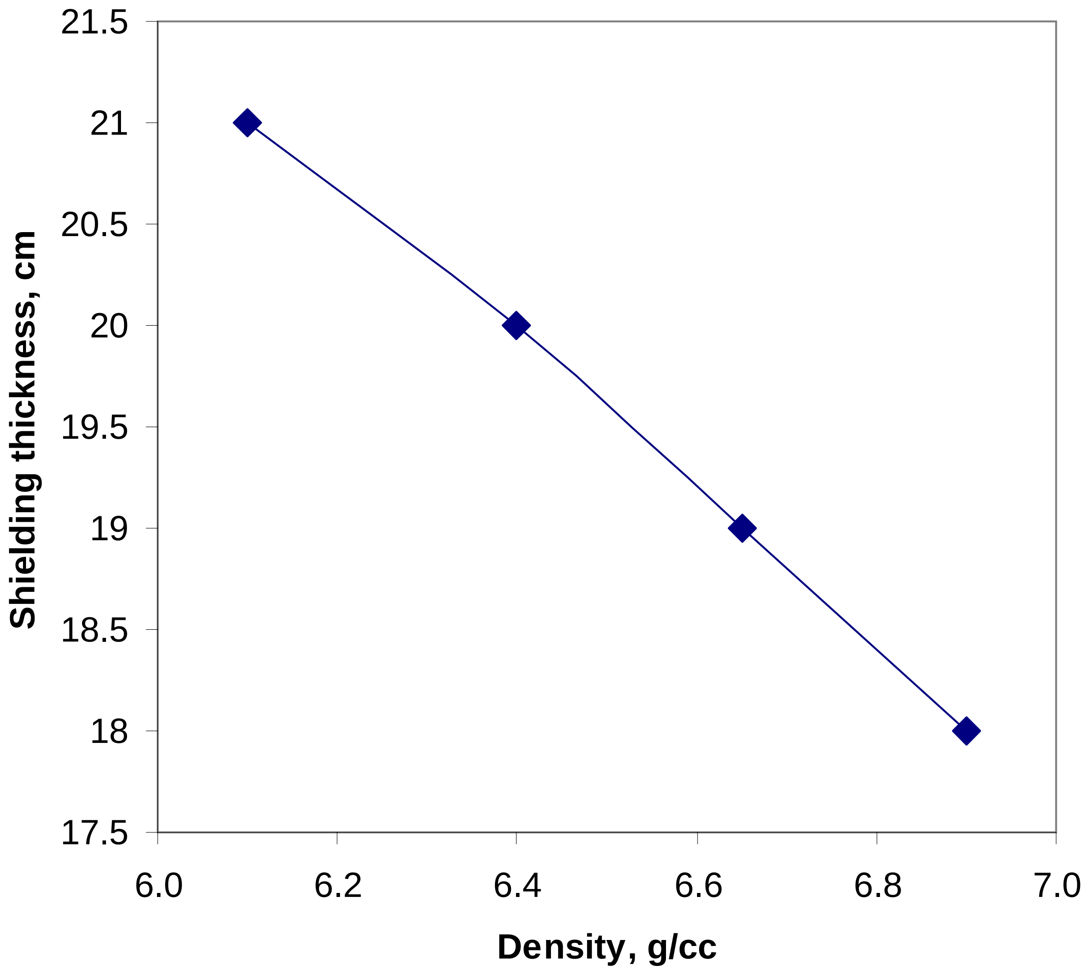

Fig. 6. Shield Thicknesses for Various DUCRETE™ Densities at a Constant 200 mrem/hr Surface Dose

Radiation Shielding Analysis.

Figure 7 summarizes the radiation shielding analysis of this work.

Figure 7a. shows the cross-section for the reference conventional concrete storage cask for PWR fuel. It contains neutron flux traps. If the U.S. Nuclear Regulatory Commission (NRC) allows burn-up credit for PWR fuel, then the cask represented by cross section shown in Fig. 7b. becomes applicable. Most electric utilities who own PWRs are buying casks such as that shown in Fig 7b. in anticipation of NRC approval of burn-up credit for PWR fuel during storage. Fig. 7c. is the cask shown in 7b. when DUCRETE™ replaces the conventional concrete. Note the large reduction in shielding thickness from 71 cm to 20 cm. Similarly, weight is reduced from 155 to 126 tons. There is no difference among inner stainless steel liner radii in cases 7a., 7b. and 7c. In case 7d. an additional ring of SNF is added to the reference case (Fig. 7b.) and the cavity radius increase 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 to 137 cm. However, the total weight of the cask increase from 155 tons to 200 tons.

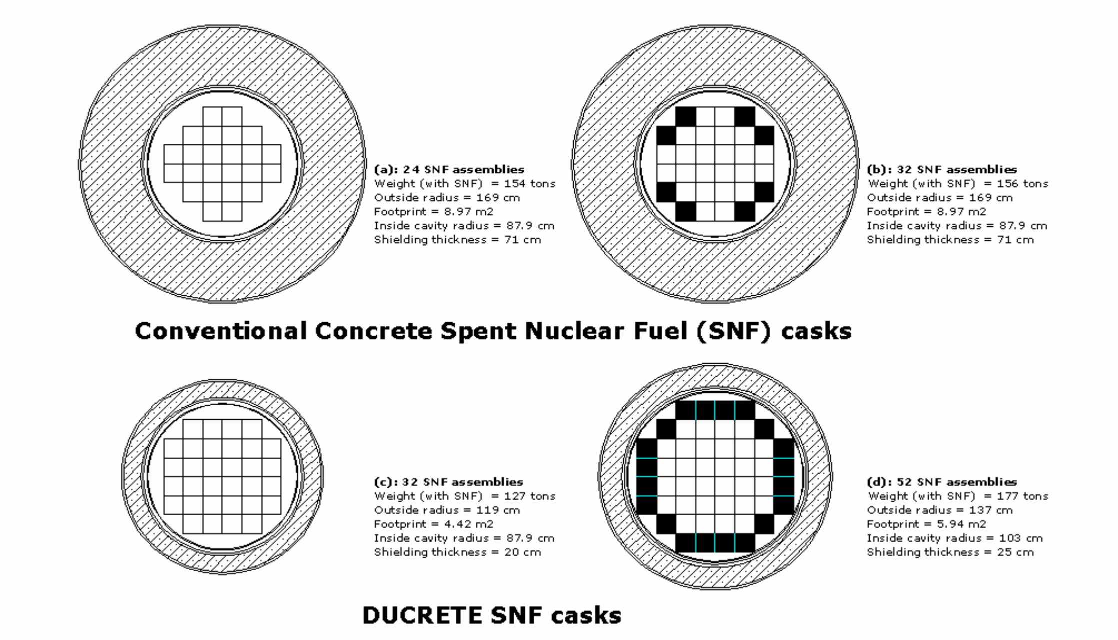

Fig.7. Cask Size and Weight Reduction Through the Use of DUCRETE™

Cask Weight Calculations.

Table 2 gives the weight of major components with SNF casks. If the individual component weight can be kept below 120 tons, DUCRETE™ cask components can be shipped via conventional rail car. Alternatively, if the component weight is below 36 tons, the component could be shipped via truck. Components could be manufactured at a central location, shipped individually to a site, and assembled into an integrated cask.

Table 2. Estimated Cask Component Weights

CASE		Cylinder	Top	Bottom	multipurpose canister (MPC) - empty	fuel assemblies **	Total weight	Cask weight w/o MPC and SNF
(a) 24 SNF assemblies with flux traps: conventional concrete	**kg**	87038.2	13620.0	13620.0	20430.0	19200.0	153908.2	114278.2
	lbs	*191714.2*	*30000.0*	*30000.0*	*45000.0*	*42290.7*	*339004.9*	*251714.2*
	*Metric tons****	**87**	**14**	**14**	**20**	**19**	**154**	**114**
(b) 32 SNF assemblies*: conventional concrete	**kg**	87038.2	13620.0	13620.0	16344.0	25600.0	156222.2	114278.2
	lbs	*191714.2*	*30000.0*	*30000.0*	*36000.0*	*56387.7*	*344101.9*	*251714.2*
	Metric tons	**87**	**14**	**14**	**16**	**26**	**156**	**114**
(c) 32 SNF*: DUCRETE	**kg**	58450.5	13620.0	13620.0	16344.0	25600.0	127634.5	85690.5
	lbs	*128745.5*	*30000.0*	*30000.0*	*36000.0*	*56387.7*	*281133.2*	*188745.5*
	Metric tons	**58**	**14**	**14**	**16**	**26**	**127**	**86**
(d) 52 SNF*: DUCRETE	**kg**	81123.9	15000.0	15000.0	25000.0	41600.0	177723.9	111123.9
	lbs	*178687.0*	*33039.6*	*33039.6*	*55066.1*	*91630.0*	*391462.4*	*244766.3*
	Metric tons	**81**	**15**	**15**	**25**	**42**	**177**	**111**

* without neutron flux traps
** The weight of an individual PWR fuel assembly, **800** kg
*** 1 metric ton = 2205 lbs

ASSAULT RESISTANCE

Russian analysis (ref. Alekseev) indicate 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, aircraft impacts will be improved by a factor of from two to three.
If the radius of the cask internal cavity is increased, through the use of DUCRETE™, to increase cask capacity, its defense (resistance) properties will increase by the 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 size, will be lighter weight for a given number of SNF assembles, have higher heat transfer rates thereby enabling the storage of higher energy (shorter cool-down time) SNF, and are 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 form 156 tons to 127 tons. Gamma radiation (not neutron) is the dominate contributor to total cask surface dose.

8

REFERENCES

1. M.J. HAIRE and P.M. SWANEY, "Cask Size and Weight Reduction Through the Use of Depleted uranium Dioxide (DUO2)-Steel Cermet Material," Waste Management 2005 Symposium, Tucson, Arizona, February 27-March 3, 2005, Oak Ridge National Laboratory (March 2005).

2. 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.

3. Alekseev, O.G., Matveev, V.Z. Morenko A.I., II'kaev R.I., Shapovalov V.I., "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, 20-24 September 2004 .