CERMET TRANSPORT, STORAGE, AND DISPOSAL PACKAGES USING DEPLETED URANIUM DIOXIDE AND STEEL

C. W. Forsberg, L. B. Shappert, P. Byrne, and B. Broadhead

Oak Ridge National Laboratory*
P.O. Box 2008
Oak Ridge, Tennessee 37831-6179
Tel: (865) 574-6783
Fax: (865) 574-9512
Email: forsbergcw@ornl.gov

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C. W. Forsberg, L. B. Shappert, P. Byrne, and B. Broadhead
Oak Ridge National Laboratory
Oak Ridge, TN, 37831, USA; Tel: 865/574-6783; Email: forsbergcw@ornl.gov

ABSTRACT

It is proposed that the steel components of spent nuclear fuel (SNF) storage casks, transport casks, and repository waste packages (WPs) be replaced with a depleted uranium dioxide (DUO_2)-steel cermet consisting of DUO_2 particulates embedded in a continuous-steel phase. Typical cermets use a sandwich-type construction with clean uncontaminated steel layers on both sides of the cermet.

Cermets have several potential advantages over other materials of construction: (1) better gamma and neutron shielding than steel; (2) ability to withstand extreme conditions (fire, accident, sabotage); (3) potential to improve repository performance when used in WPs; and (4) use of excess DUO_2 and recycled steel from nuclear facilities, thereby avoiding disposal costs for these materials. New methods of manufacture and other factors may provide economic incentives for cermet packages when large numbers of casks are manufactured.

INTRODUCTION

Cermets can be used to construct cask and WP shells (Fig. 1). In this application, the cermet is used to meet structural and shielding requirements. Cermet baskets can be constructed that meet the structural, thermal, and criticality requirements. When baskets are constructed, rare-earth oxides or other nuclear absorbers may be added to the DUO_2 cermet to enhance operational criticality control. When used in WPs [1, 2], cermets may (1) improve the long-term WP performance, (2) minimize repository post-closure criticality issues, and (3) beneficially use excess DU.

DUO_2-steel cermets offer several advantages over uranium metal in cask applications. Unlike uranium metal, which rapidly corrodes under moist anoxic conditions, cermets are stable in many environments. Moreover, DUO_2 is less expensive than uranium metal. Finally, in terms of (1) repository applications or (2) transport or storage cask decommissioning, the U.S. Nuclear Regulatory Commission [3] has stated that uranium oxides are an acceptable chemical form for disposal but that significant uncertainties exist in terms of disposal of DU metal.

INCENTIVES FOR USE OF CERMETS IN SNF PACKAGES

Several incentives exist for using cermets in storage, transport, and WPs.

- *Shielding*. DUO_2-steel cermets provide excellent gamma shielding—up to 30+% greater density than steel, with resultant reductions in cask size and weight. Cermets also provide some neutron shielding. The high oxygen content in DUO_2 (1.3 g/cm^3) provides a neutron moderator to slow down fast neutrons and allow capture by thermal neutron absorbers. This quality is unique among high-density shielding materials.

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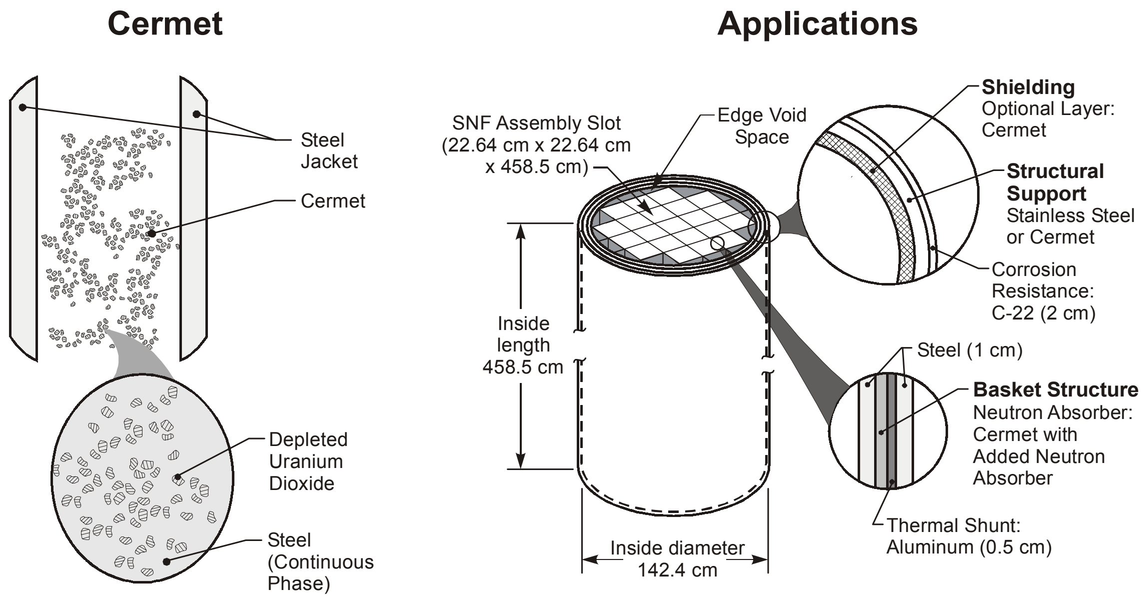

Fig. 1. Characteristics and Uses of Cermets in WPs for PWR SNF

- *Strength*. Cermets are strong, tough materials that can operate at higher temperatures and under more severe conditions than the metal included within their structure. Cermets have been used as nuclear fuels, brake shoes, machine tools, and in other applications. Excellent performance would be expected in the event of an accident, a fire, or sabotage.

- *Repository benefits*. For repository applications, DUO_2 cermets provide a method to add DU to the repository WP in a form that has no significant impact on operations because there is no significant potential for DU contamination. The long-term repository benefits include:

  – Criticality control. The addition of DU to the WP helps ensure post-closure repository criticality control [2] by isotopic dilution of the enriched uranium from the SNF with the DU from the cermet.

  – Radionuclide releases. The addition of DUO_2 to the WP is expected to slow the degradation of SNF and thus reduce the releases of radionuclides from the WP and repository [1]. The mechanisms that help to lower the rate of SNF radionuclide releases include (1) maintaining chemically reducing conditions near the SNF for extended periods of time, (2) sorption of radionuclides on the hydrated oxidized DUO_2 formed by cermet degradation over time, (3) blocking of groundwater flow, and (4) saturation of the local environment with uranium.

- *Resource utilization*. The United States has in excess of >500,000 tons of DU and >2,000,000 tons of potentially contaminated metals from decommissioning uranium enrichment plants and similar facilities. The metals include very large quantities of nickel and stainless steel. The beneficial use of these materials would avoid disposal costs and promote more efficient use of resources. These factors may significantly impact the economics of cermet manufacture.

PROPERTIES

Cermet properties depend upon the choice of metal and the ratio of DUO_2 to metal. The specific application determines the type of cermet to be used. In the 1950s, cermet fuels were extensively investigated and used in 11 research and test reactors in the United States. As a consequence, a significant number of physical property measurements have been made. Experimentally measured and calculated properties of cermets as a function of composition are described herein.

Density

The density of a cermet is determined by the relative proportions of the respective ceramic and metal constituents and can be expected to fall somewhere between the densities of the two. Based on the theoretical densities of UO_2 (10.96 g/cm^3), aluminum (2.70 g/cm^3), and A.I.S.I. type 316 SS (7.92 g/cm^3), cermet density as a function of the UO_2 content and choice of metal can be determined (Fig. 2). Because the density of carbon steel [7.8 g/cm^3 (<1% C)] is similar to stainless steel, a carbon steel–UO_2 cermet will have a density [4] close to that of a stainless steel–UO_2 cermet.

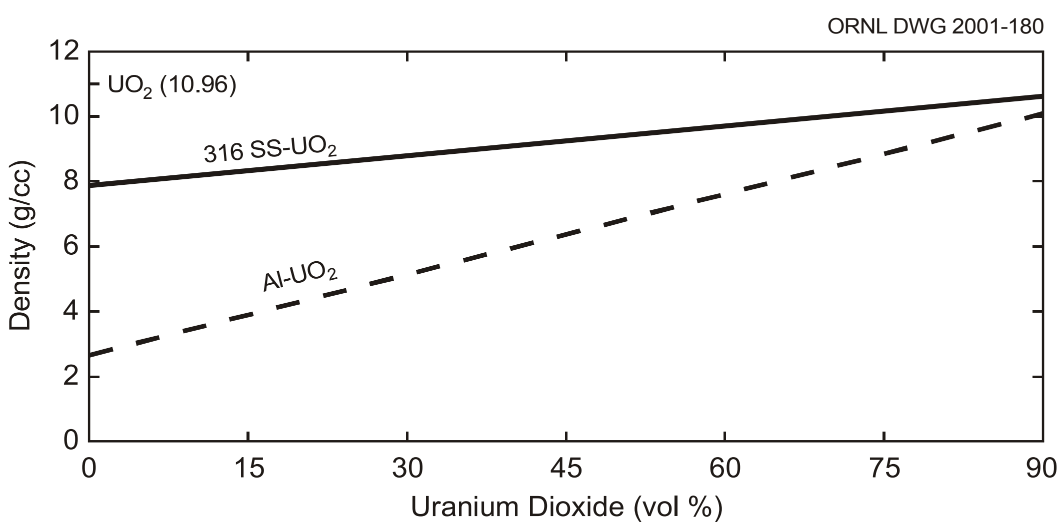

Fig. 2. Cermet Density as a Function of UO_2 Volume Fraction.

Thermal Conductivity

Because these cermets were first considered for use as nuclear fuels, the thermal conductivity of UO_2 cermets has been previously investigated. As a ceramic material, UO_2 possesses a thermal conductivity lower than that of the partnered metal in a cermet. A cermet's overall thermal conductivity will fall between the values of its constituent ceramic and metallic phases. Experimentally measured values [5] of the thermal conductivity (Fig. 3) for different cermets confirm the expected result.

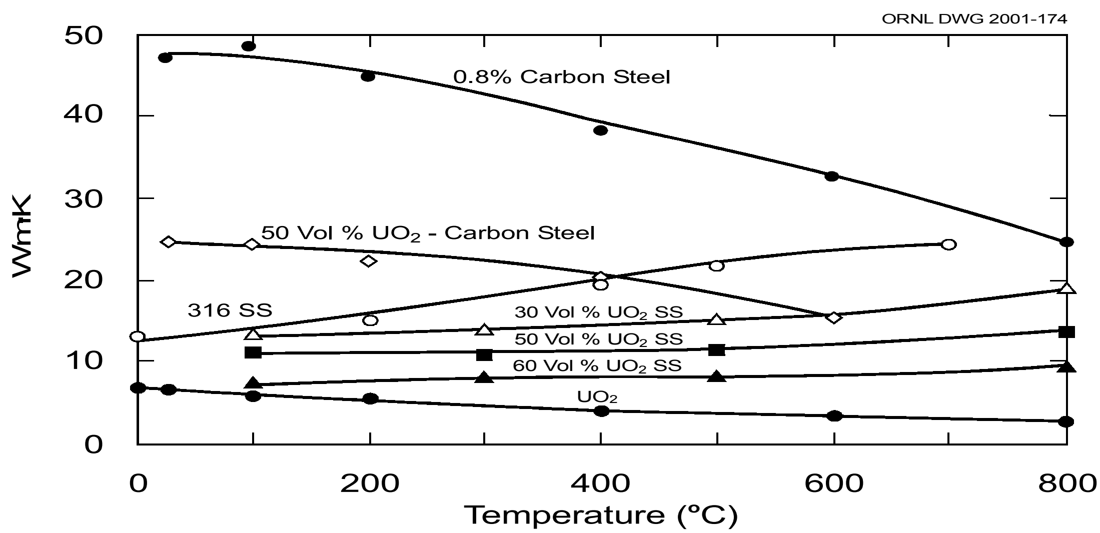

Fig. 3. Thermal Conductivity as a Function of Temperature for Cermets and Other Materials.

Strength

The mechanical strength of UO_2–stainless steel cermets has been studied by a variety of researchers. Particle size, density, temperature, and fabrication techniques have each been shown to affect the mechanical strength of the finished cermet. Figure 4 shows some measured values of the mechanical strength of several cermets. Unlike density and thermal conductivity, the mechanical strength of two cermets with the same volume fraction of UO_2 can vary widely because of the aforementioned factors. As a consequence, generalized curves of strength vs volume fraction of UO_2 should be used with extreme care.

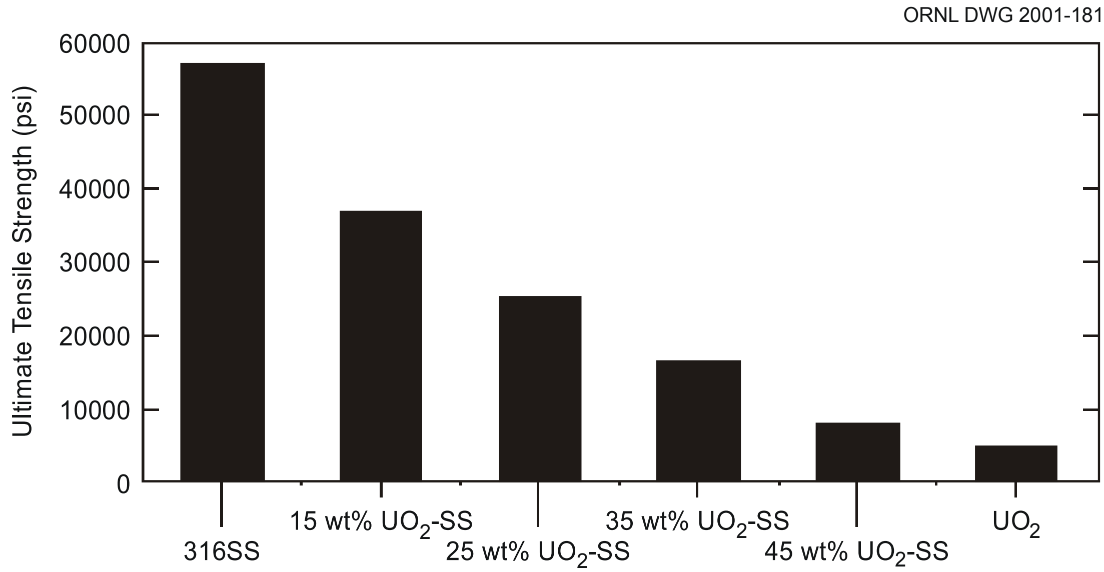

Fig. 4. Strength of Cermets and Other Materials.

Because of UO_2's lower tensile strength (- 5000 psi), the overall cermet tensile strength decreases as the UO_2 concentration increases. The ultimate mechanical strength of a UO_2–stainless steel cermet lies between the respective values of the mechanical strength for UO_2 and stainless steel [6].

For the applications under consideration, the cermet is between layers of clean steel that are bonded with the cermet. Similar in principle to an I beam, the strongest sections of the composite structure are the zones of clean metal on the outer surfaces. The weakest sections of the composite structure is the cermet in the middle of the composite structure. From a structural perspective, this construction maximizes the overall strength.

Shielding

Shielding calculations were undertaken to allow the relative effectiveness of cermets to be compared with other shielding materials. The SNF source term for the proposed Yucca Mountain (YM) WP for 21 pressurized-water reactor (PWR) fuel assemblies was used [7]. The dose rates were calculated using the SCALE SAS1 sequence [8], the same methods used in the YM study. The basket diameter is 1.424 m. The SNF is assumed to have an average burnup of 40,000 MWd and is loaded into the WP 25 years after reactor discharge. In all cases, the metal is assumed to be 316 SS—the repository WP material of choice. Because the densities of carbon steel and stainless steel are almost identical, similar gamma shielding would be expected for a carbon steel cermet.

Three materials were compared: steel, 50 vol % DUO_2, and 90 vol % DUO_2. The neutron, gamma, and total radiation doses at the surfaces were calculated for different shielding thicknesses. The results are shown in Table 1.

Table 1. Shielding Effectiveness (R/h) of Different Materials

Width	Steel (316 SS)			Cermet (50% DUO_2)			Cermet (90 vol % DUO_2)		
cm	γ	n	total	γ	n	total	γ	n	total
0	14700	3.18	14703	14700	3.18	14703	14700	3.18	14700
5	836	2.27	838	91.6	2.12	93.7	28.4	2.01	30.4
10	64.1	1.66	65.8	2.34	1.37	3.71	.586	1.20	1.79
15	5.27	1.20	6.47	.122	.846	.968	.0245	.675	.700
20	.469	.854	1.32	.00951	.512	.522	.00293	.372	.374
25	.0473	.607	.654	.00168	.306	.308	.00089	.202	.203
30	.00645	.430	.436	.00077	.181	.182	.00052	.109	.110
35	.00179	.305	.307	.00047	.107	.107	.00030	.0587	.0590
40	.00099	.216	.217	.00030	.0633	.0636	.00017	.0315	.0317
45	.00070	.153	.154	.00019	.0372	.0374	.00010	.0169	.0170
50	.00052	.109	.110	.00012	.0219	.0220	.00006	.00901	.00907

Compositions are volume %.

Two important observations may be made:

- *Gamma shielding*. The gamma-shielding performance of the cermets is considerably better than that of steel because of the higher densities of the cermet. The densities of the three materials are 7.92 g/cm^3 (stainless steel), 9.44 g/cm^3 (50 vol % DUO_2 cermet), and 10.66 g/cm^3 (90 vol % DUO_2 cermet).

- *Neutron shielding*. Cermets have better neutron-shielding capability compared with steel. The DUO_2 has an oxygen density of 1.3 g/cm^3 and is a moderator—similar to carbon.

MANUFACTURE

UO_2–stainless-steel cermets have been produced with loadings up to 90 vol % of UO_2. Cermets have been used as nuclear fuels in at least 11 research, test, and demonstration reactors in the 1950s and 1960s. They are currently being investigated in Europe [9] for use as very-high-burnup power-reactor fuels. Non-uranium-oxide cermets are produced in large quantities (>100,000 tons/year) for a variety of commercial applications.

Multiple cermet manufacturing methods are possible. The traditional method for manufacture of UO_2 cermets is to mix DUO_2 and metal powder and press the mixture into a flat compact (Fig. 5). "Picture frames" are fabricated from thick steel sections with the compact placed within the picture frame. A sheet of clean uncontaminated steel is placed above and below the picture frame. The edges are welded together, and the compact is vacuum degassed. The section is heated to a high temperature and rolled into a thinner plate to produce the final cermet.

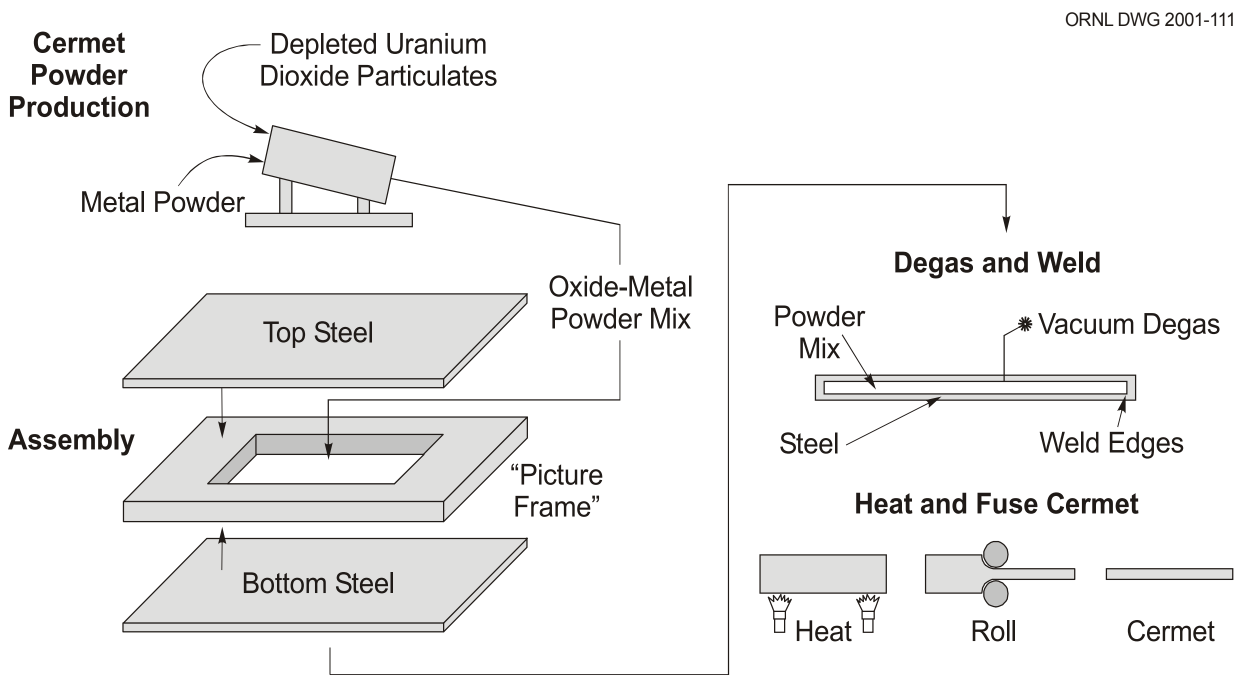

Fig. 5. Sample Method for Cermet Production.

There are potentially very low cost fabrication methods for the shells of casks and WPs. Aluminum-SiC cermets are made by casting molten aluminum with solid SiC particulates. The same approach may be viable for a DUO_2-steel cermet. For round objects, such as sewer pipe, centrifugal casting is the preferred casting technology. The mold is rotated at high speed, molten metal is added to a cylindrical mold, the metal is centrifugally held against the mold until the metal solidifies, and the final product is removed after metal cooldown. If the same process can be developed for casting cermet cask and WP bodies, very low fabrication costs are possible. Significant R&D is required before there would be assurance that such methods are viable.

Cermets for non-nuclear applications are produced in very large quantities at very low costs—including use in safety-critical applications. The UO_2-steel cermets that have been produced were fabricated on a small scale, primarily with enriched or highly-enriched uranium, for use as nuclear fuels. The challenge in fabricating DUO_2-cermets is to adopt the high-volume, low-cost methods used in other applications.

Cermets can be made using a variety of metals; however, not all fabrication methods can be used with all metals. Steel and UO_2 are thermodynamically stable with respect to each other. However, this is not the case for aluminum and UO_2. At high temperatures, the latter two materials will react with one another. Consequently, special low-temperature fabrication methods are required for the manufacture of aluminum-UO_2 cermets. This reactivity also limits some applications of aluminum-UO_2 and similar cermets.

The U.S. Department of Energy has about 2,000,000 tons of metal for recycle, primarily from decommissioning of gaseous-diffusion uranium-enrichment facilities. Because of unresolved issues associated with the recycle of metals from nuclear facilities into general commerce, it is unclear whether this material can be recycled in the open market. If it cannot be recycled, significant disposal costs will be incurred. In terms of DUO_2-steel cermet manufacture, the source of the material is not an issue. The inventory includes large quantities of nickel and stainless steels. The potential availability of this inventory at low or negative costs may make it practical to use what are traditionally considered "expensive, ductile high-performance" stainless steels for construction of cask bodies. The use of such steels may simplify fabrication (i.e., involve more-forgiving materials), improve cask performance, simplify licensing (i.e. reduce materials issues such as ductility), and facilitate quality assurance.

EXAMPLE DESIGN: YUCCA MOUNTAIN WP

The conceptual design of a self-shielded WP was developed that is a variant of the baseline unshielded YM WP [7, 10]. Proposed changes in the repository design are creating new incentives for a self-shielded WP. The YM project is evaluating alternative designs in which the repository remains open and is ventilated for 50 to 300 years. Ventilated repositories allow removal of the decay heat during the initial period of time when the decay heat is high. Ventilation for this limited period of time reduces the peak temperatures in the repository. Reducing temperatures reduces the uncertainties in long-term repository behavior, may simplify licensing, and may allow design simplifications with significant cost reductions. With an extended maintenance and inspection period, strong incentives exist for considering self-shielded WPs.

The existing unshielded YM WP has the following characteristics: (1) it is designed for 21 PWR SNF assemblies, (2) the SNF has an average burnup of 40,000 MWd and is loaded into the WP 25 years after discharge from the reactor, (3) the basket diameter is 1.424 m, (4) the internal cavity length is 4.585 m, (5) the inner WP shell is 5 cm thick and constructed of 316 SS, and (6) the inner WP is placed into a closely fitting 2-cm-thick outer WP. The outer WP is made of a corrosion-resistant nickel alloy (C-22) to ensure long-term integrity in the repository environment.

The self-shielded cermet WP (1) is designed for the same types and quantities of SNF, (2) has the same basket structure and internal cavity dimensions, (3) uses the same materials of construction (except for the addition of DUO_2 and neutron absorbers to form a cermet and possible addition of graphite as a neutron moderator), and (4) has the same corrosion resistant C-22-alloy layer. The cermet WP consists of up to three layers.

- *Inner layer*. The inner layer is a cermet that is assumed to average 50 vol % DUO_2 with 0.2 wt % ^{235}U, 1 vol % Gd_2O_3, and 49 vol % 316 SS. The center of the cermet may have a higher DUO_2 content, but it will be covered with clean steel so the average DUO_2 content is 50 vol %. The cermet provides most of the gamma shielding.

- *Middle layer*. The cermet provides significant neutron shielding but the neutron shielding capability is not as good as the gamma shielding capability. Depending upon design goals (see below), an additional middle neutron-shielding layer may be required. If required, a potential neutron shielding material is graphite with 1 wt % boron. The graphite provides added neutron moderation to slow neutrons down and allow their capture. For neutron shielding, the major advantage of graphite compared to a cermet is its lower weight. Traditional neutron moderators (organics, concrete, etc.) are not allowed in repository WPs. Graphite is extremely inert [11] and has been considered as a potential WP material. Graphite was ultimately rejected as the primary WP material because of concerns about ductility. This is not an issue when the graphite is sandwiched between ductile materials. The option exists to use the graphite as both a neutron shield and a secondary WP to extend the total WP lifetime.

- *External layer*. The external layer is composed of a 2 cm of C-22, identical in composition, thickness, and function to the external layer of the currently proposed WP.

The results of radiation shielding calculations are shown in Table 2 for different WP designs. Evaluations of repository self-shielded WPs [10] have considered allowable external-surface radiation levels between 0.0025 and 0.2 R/h. For an external radiation dose of 200 mrem/h, the WP would consist of slightly more than 25 cm of cermet and 2 cm of C-22. The cermet is a better gamma shielding material than a neutron shielding material. If very low external radiation levels are required, the neutron radiation dose is controlling. Under these conditions, there are incentives to adding a separate neutron-shielding material, such as graphite, to the WP.

In terms of operations, WP shielding can be chosen to allow limited access to the area with the requirement to use added temporary shielding if major work is being undertaken in a particular area. Temporary neutron shielding typically consists of light-weight plastics with boron whereas gamma shielding requires large masses of material. These considerations suggest that the external WP gamma dose should be significantly less than the external neutron dose. The cermet WP has this feature. Specific WPs are designed for a maximum burnup SNF. Most of the WPs will have SNF with significantly lower radiation levels; thus, the typical external WP radiation levels will be significantly below the design limits.

Table 2. External Radiation at the Surface of the WP for Different Designs

Shielding Materials (cm)			Surface Radiation Levels (R/h)		
Cermet	Graphite	C-22	γ	n	Total
0	0	0	14700	3.18	14703
10	0	2	1.11	1.12	2.23
15	0	2	0.063	0.675	0.738
20	0	2	0.00596	0.401	0.407
25	0	2	0.00159	0.236	0.238
25	10	2	0.00137	0.0419	0.0433
25	15	2	0.000923	0.0179	0.0188
25	20	2	0.000573	0.00776	0.00833
25	25	2	0.000340	0.00343	0.00377
25	30	2	0.000198	0.00154	0.00174

In the past, the YM repository has examined several types of self-shielded WPs. Most of the proposed designs for self-shielded WP designs were rejected. Many constraints exist in the design of a shielded WP [10]. Shield thickness is to be minimized to avoid the need for larger tunnels. Weight should be constrained to limit handling difficulties. Low radiation levels are required because the working space between the WP and tunnel wall in the repository is generally <2 m. Last, there are severe constraints on the allowable materials of manufacture. Unlike a surface storage facility or transport cask, the licensing period of the repository is 10,000 years. Materials that may accelerate WP corrosion or migration of radionuclides from the SNF are banned. Many traditional packaging materials (organics used in neutron shielding and concrete) are not allowed in repository WPs.

The pre-conceptual design herein avoids the factors that disqualified earlier designs. The primary uncertainties may be associated with manufacturing costs. No WP optimization was undertaken. Higher DUO_2 loadings in the cermet may be preferred and other neutron absorbers may be used. Detailed designs include complex trade-offs between costs, operational requirements (shielding, strength, thermal conductivity, etc.), and long-term performance.

The cermet WP may enable the development of multipurpose canisters for storage, transport, and disposal of SNF. In a multipurpose canister system: (1) the SNF is loaded into a multipurpose canister at the reactor; (2) the loaded canister functions as a dry-storage cask at the reactor site; (3) the canister, with a transportation overpack, is used to transport the SNF to the repository; and (4) at the repository, the transport overpack is replaced with a repository overpack (C-22 outer package) that allows the canister and overpack to function together as a repository WP. The multipurpose canister would be the inner cermet package (as described earlier) that provides the basket, structural support, radiation shielding, and physical protection for the SNF. The multipurpose canister avoids the need for separate storage, transport, and disposal packages for SNF. No transfer of SNF between storage, transport, and disposal packages is required.

Multipurpose canisters were evaluated in the 1980s. It was determined that they had potential economic and safety advantages. However, multipurpose canisters were not adopted because the requirements for the repository WPs were not well understood or defined at that time. Once a repository is licensed, the specific requirements will be known and defined. At that time (probably within this decade in the United States), it will become feasible to design and consider implementing a multipurpose canister for the 40,000+ tons of SNF that will be generated after the repository license is issued. Thus, multipurpose canisters should be considered as a potential second-generation WP system.

CONCLUSIONS

Cermets offer (1) significant advantages for use in SNF transport and storage applications and (2) important benefits in repository applications. The evidence strongly supports the technical feasibility, but significant work is required before the tradeoffs between economics (cermet production costs) and performance are understood.

ACKNOWLEDGMENT

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REFERENCES

[1] C. W. Forsberg, “Effect of Depleted-Uranium Dioxide Particulate Fill On Spent Nuclear Fuel Waste Packages,” Nuclear Technology, Volume 131, 252, September 2000.

[2] C. W. Forsberg, “Repository Criticality Control With Depleted-Uranium-Dioxide Cermet Waste Packages,” *Topical Meeting on Practical Implementation of Nuclear Criticality Safety, Reno, Nevada, November 11–15, 2001*, American Nuclear Society, La Grange Park, Illinois, November 2001.

[3] E. J. Leeds, “Comments on DUF_6 Materials Use Roadmap,” Draft, Dated September 1, 2000, Letter to the U.S. Department of Energy from the U.S. Nuclear Regulatory Commission, Special Projects Branch, Division of Fuel Cycle Safety and Safeguards, Office of Nuclear Material Safety and Safeguards, ML003762080, October 18, 2000.

[4] G. W. C. Kaye and T. H. Laby, *Tables of Physical and Chemical Constants, 16th Ed.* New York: Longman, 1995.

[5] G. Arthur and J. A. Coulson, “Physical Properties of Uranium Dioxide–Stainless Steel Cermets,” Journal of Nuclear Materials, Volume 13, Number 2, 242–253, 1964.

[6] D. L. Keller, “Dispersion Fuels,” pp. 304–330 in *Reactor Handbook, 2nd Ed., Vol. 1*, edited by C. R. Tipton, Jr., New York: Interscience, 1960.

[7] J. S. Tang, “Shielding Characteristics of Various Materials on PWR Waste Packages,” Yucca Mountain Project, BBAC00000-01717-0210-00008 REV00, Las Vegas, NV, 1999.

[8] *SCALE: A Modular Code System for Performing Standardized Computer Analyses for Licensing Evaluation,* NUREG/CR-0200, Rev. 6 (ORNL/NUREG/CSD-2/R6), May 2000.

[9] J. Porta and Y. Y. Doriath, “Toward Very High Burnups, a Strategy for Plutonium Utilization in Pressurized Water Reactors,” Journal of Nuclear Materials, Volume 274, Number 153, 1999.

[10] U.S. Department of Energy, Yucca Mountain Project, “License Application Design Selection Feature Report: Waste Package Self Shielding Design Feature 13,” B00000000-01717-2200-00220 REV00, Las Vegas, NV, March 6, 1999.

[11] A. L. Lotts et. al., *Options For Treating High-Temperature Gas-Cooled Reactor Fuel for Repository Disposal,* ORNL/TM-12027, February 1992.