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DEVELOPMENT OF Be_2C-GRAPHITE-UC_2 FUEL FOR PULSED REACTORS*

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A recent program to upgrade the Annular Core Pulsed Reactor (ACPR) at Sandia Laboratories[1] resulted in the development of UO_2-BeO[2] and (U, Zr)C-graphite[3] fuels for pulsed reactor applications. The work described in this paper is an attempt to develop a fuel which is superior to either of these in a pulsed reactor environment. The main fuel requirements are high volumetric enthalpy (to absorb the nuclear pulse without excessive temperature rise), high thermal stress resistance (to withstand the thermal stresses caused by the radial temperature gradient during the 2 to 5 ms risetime pulse), and a simple pellet geometry. The fuel is a right circular cylinder whose outer diameter is 33.0 mm.

Beryllium carbide has a volumetric heat capacity which is greater than that for BeO. Therefore, a Be_2C matrix fuel can combine the high heat capacity of the BeO matrix fuel with the good thermal stress resistance of the graphite matrix fuel. This can be accomplished by introducing graphite into a carbide-based system containing Be_2C and UC_2. As in graphite matrix fuels, the graphite in the composite system should increase the thermal stress resistance by decreasing the elastic modulus and thermal expansion without severely

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decreasing the strength. Coobs and Koshuba[4] showed that small additions of graphite to Be_2C substantially decreased the elastic modulus and facilitated fabrication.

A comparison of the volumetric enthalpy of various types of fuel is given in Fig. 1. The BeO-UO_2 curve is limited to 1400°C because occasional pellet fracture occasionally occurs at this temperature for a 0.51 cm high, dual slotted annulus pellet geometry.[2] A maximum pulse temperature of 1800°C was chosen for the Be_2C fuels in order to eliminate possible decomposition of the Be_2C. A 30 vol. % Be_2C composite fuel at 1800°C has the same enthalpy as the BeO matrix fuel at 1400°C (but the pellet geometry for the Be_2C composite fuel will be much simpler). A 70 vol. % Be_2C composite fuel has a volumetric enthalpy which is potentially 35% higher than the BeO matrix fuel.

The feasibility of fabricating fuel-element samples of Be_2C-graphite-UC_2 has been studied and reactor test samples have been fabricated. The fabrication process involved milling UH_3 powder (or UO_2 powder), Be_2C powder, graphite flour and a carbonizable binder in a solvent. After milling, the mixture was evaporated to dryness and densified by conventional graphite die hot pressing at 12.5 MPa and 1850°C. The uranium hydride (or oxide) was carbided <u>in situ</u> during the hot pressing. A series of 12 developmental densifications were made using depleted uranium to determine the appropriate conditions necessary to achieve satisfactory densities (< 15% porosity), UC_2 particle size (< 10 μm) and dispersion, and mechanical integrity. Parameters varied included the source of uranium, type of graphite flour, volume percents of beryllium carbide and carbonizable binder, and total milling time. Uranium loading densities of 0.450 to 0.500 g/cm^3 and beryllium carbide contents of approximately 30, 50, and 70 vol. % were obtained.

It was concluded from the developmental hot pressings that $Be_2C/C/UC_2$ fuel pellets with a length-to-diameter ratio at least up to 0.75 could be made with good mechanical integrity and material dispersion, porosities < 10% and UC_2 particles size of < 10 μm. This was accomplished by: (1) using enriched UH_3 powder (1 m^2/g surface area) as the uranium source material; (2) using 10 wt. % acenaphthylene as a binder; (3) using < 45 μm Be_2C powder in quantities which would give 30 to 70 vol. % Be_2C in the finished specimens; (4) using graphitized coke (< 75 μm GL-1008 powder) as a carbon filler and source of carbon for the carbiding reaction of UH_3 to UC_2; (5) milling the starting materials slurried in toluene and evaporating the toluene while blending, and (6) using a pressure-temperature-time cycle during hot pressing which involved no pressure below 600°C and a 3 hour hold at 1850°C and 12.5 MPa pressure. Pressings 33 mm diameter by 25 mm long were made using this procedure and cut into reactor test pellets 25, 13 and 6.5 mm high. Samples containing 26 vol. % Be_2C (density = 2.45 g/cc, 11% total porosity, 4 vol. % UC_2) and 67 vol. % Be_2C (density = 2.58 g/cc, 4% total porosity, 2 vol. % UC_2) were prepared.

Preliminary reactor pulse tests have been performed in the ACPR on the test pellets. A number of solid pellets with varying heights were subjected to pulses similar in severity to the pulses required to fail the BeO matrix fuel[2]. All the 26 vol. % Be_2C pellets survived, but in some of the 67 vol. % Be_2C pellets, material spalled off one edge (around the circumference). The spalling has been attributed to "capping" (a common problem in hot pressing) and does not indicate a basic material deficiency.

The usefulness of a beryllium carbide/graphite/uranium carbide composite fuel for pulsed reactors has been demonstrated. Test pellets have been successfully fabricated and tested. Further development of this type of fuel should further increase their usefulness in pulsed reactors with fast risetimes.

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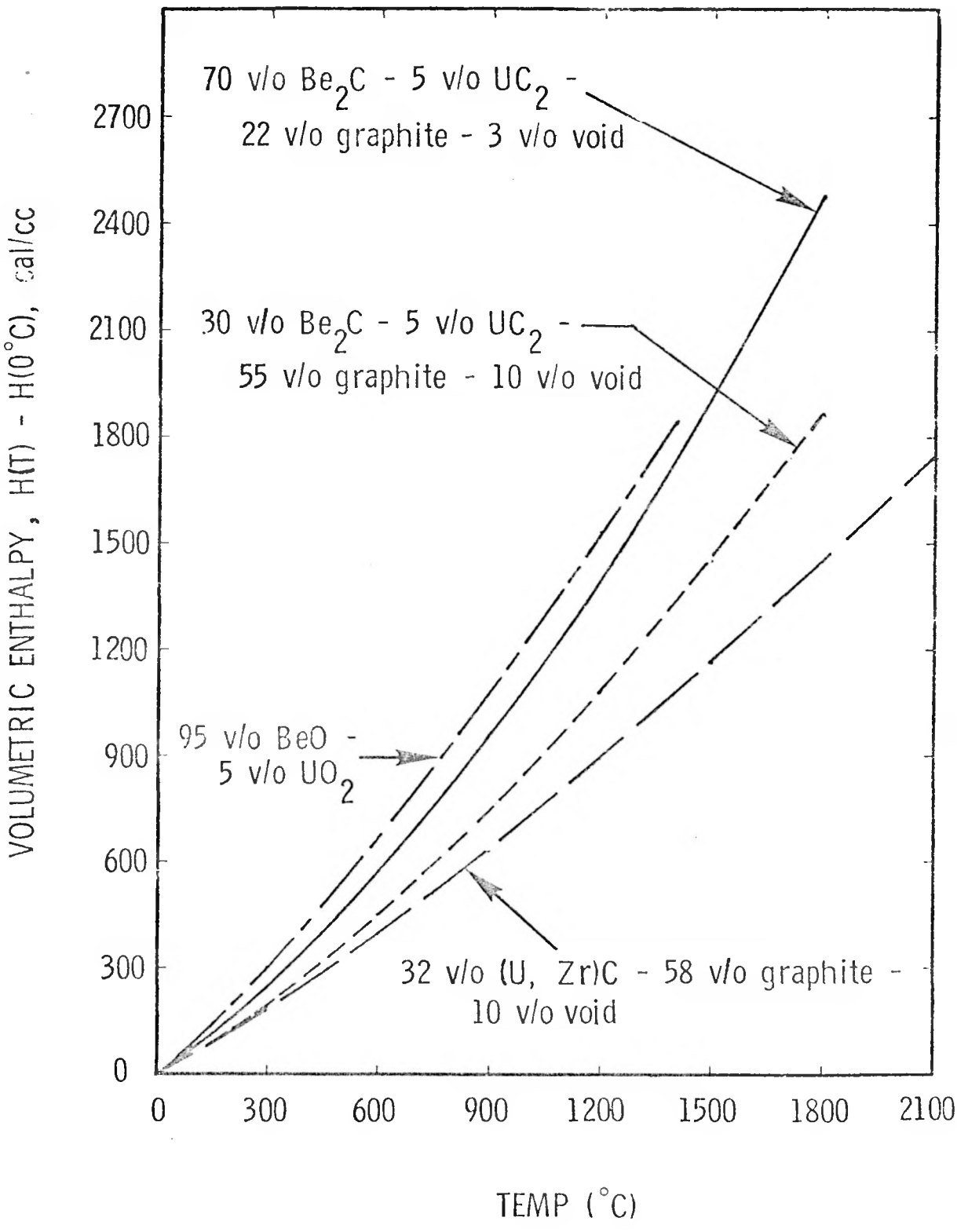

Figure 1. Volumetric enthalpy (cal/cc) for various types of pulsed reactor fuels. (v/o = volume percent) See text.