Fully Ceramic Microencapsulated Fuel in FHRs: A Preliminary Reactor Physics Assessment

Jeffrey J. Powers
Oak Ridge National Laboratory, P.O. Box 2008, Oak Ridge, TN 37831, powersjj@ornl.gov

INTRODUCTION

Fully ceramic microencapsulated (FCM) fuel consists of microencapsulated coated fuel particles embedded in a silicon carbide (SiC) matrix [1–3]. FCM technology development efforts have been conducted mostly at Oak Ridge National Laboratory (ORNL) as part of the United States Department of Energy Office of Nuclear Energy Advanced Fuels Campaign. Numerous organizations have investigated applications of FCM fuel in light water reactors (LWRs) [4]. A couple of efforts began considering FCM fuel in advanced reactors, but these studies were largely conceptual [5,6]. This work provides a preliminary reactor physics feasibility assessment of FCM fuel in a fluoride-salt-cooled high-temperature reactor (FHR).

FCM FUEL

The microencapsulated fuel in FCM is generally tristructural isotropic (TRISO) coated fuel particles. As shown in Fig. 1, TRISO particles contain a fuel kernel surrounded by four coating layers: a porous carbon buffer, a dense inner pyrolytic carbon (IPyC) layer, an SiC layer, and a dense outer pyrolytic carbon (OPyC) layer. The buffer attenuates fission fragments, holds gases released from the kernel, and accommodates dimensional changes in the particle. The SiC layer acts as a pressure vessel and diffusion barrier that prevents the release of fission products (FPs). The PyC layers protect the SiC layer from chemical attack, provide FP diffusion barriers, and form a thermo-mechanical system along with the SiC layer. Detailed descriptions of TRISO fuel properties, behavior, and failure mechanisms may be found elsewhere [7,8].

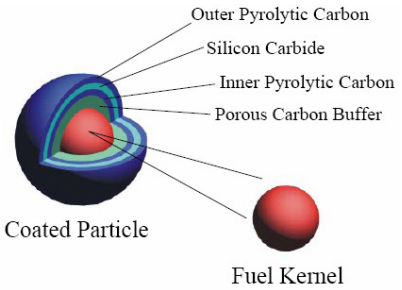

Fig. 1. Cutaway diagram of a TRISO fuel particle.

Historically, TRISO particles have been used in fuel elements consisting of spherical pebbles or hexagonal prismatic blocks, with graphite used as a matrix and coating for the fuel element. FCM fuel replaces the graphite matrix with an SiC matrix, which improves radiation tolerance and enhances FP retention. Most FCM fabrication work has focused on fabricating fuel pellets that will be stacked and placed inside fuel cladding tubes that are similar to standard LWR fuel rods that contain UO_2 fuel pellets stacked inside a zirconium alloy (Zircaloy) cladding tube (Fig. 2). Other FCM fuel forms could also be fabricated, including pebbles, plates, or fuel compacts in prismatic blocks. Extensive details about early processing techniques for FCM fuel are available elsewhere [3,9,10].

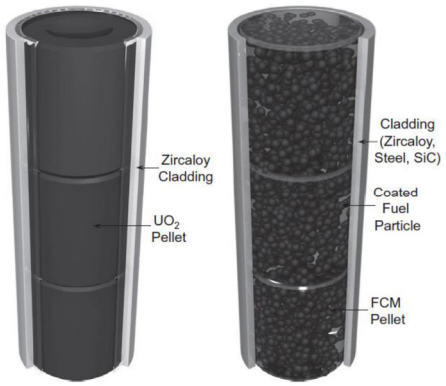

Fig. 2. Comparison of standard UO_2/Zircaloy fuel (left) and an FCM fuel pin (right) [3].

Engineered TRISO fuel particles being investigated for FCM differ substantially from historic TRISO particles. The particles undergoing qualification as part of the Next Generation Nuclear Plant (NGNP) program use UCO fuel kernels with a 500-μm diameter, a buffer PyC layer thickness around 100 μm, IPyC and OPyC thicknesses around 40 μm, and an SiC layer thickness around 35 μm [11]. UCO is a mixture of urania (UO_2) and uranium carbide (UC) fuel. Some FCM concepts use similar particle designs, but others use larger kernels and substantially thinner coatings: 25–75 μm buffer layer thicknesses, 15–20 μm IPyC layer thicknesses, and SiC layer thicknesses as low as 30 μm. Not all of these thicknesses will be feasible for all particle designs or

irradiation applications, but the full range is being investigated to understand the available design space. FCM fuel kernel material options include mixtures of UCO or uranium mononitride (UN). UN offers a higher heavy metal (HM) loading density than UCO, and the geometric changes (increased kernel diameter and decreased coating thicknesses) result in a larger fuel volume fraction in the particle. Both of these changes increase the effective HM loading density in FCM fuel. UN kernels have been fabricated at approximately 90% of the theoretical density of UN with kernel diameters around 820 µm [12,13].

METHODOLOGY

FHR concepts use fluoride salt coolants such as Flibe (2 LiF + BeF_2) with outlet temperatures near 700°C; they typically use graphite as a neutron moderator material. Core geometry options include hexagonal prismatic blocks with cylindrical fuel compacts, packed beds of spherical pebbles such as the pebble-bed advanced high-temperature reactor (PB-AHTR) [14], or hexagonal fuel assemblies with planks as in the AHTR [15].

A simple unit cell model with TRISO particles packed into a pebble surrounded by coolant, shown in Fig. 3, was used to assess FCM fuel in an FHR environment. Depletion calculations were performed using SCALE/TRITON from SCALE 6.1 [16] with the KENO-VI Monte Carlo neutron transport code using 238-group cross-sections based on ENDF/B-VII.0 nuclear data. The DOUBLEHET calculation type in SCALE was used to accurately capture the double-heterogeneity of the fuel. The pebble outer radius was 3.0 cm, but the fuel particles were restricted to an inner region with a 2.5-cm radius, leaving an unfueled 0.5-cm thick shell on the surface. The coolant region was sized to yield a pebble packing fraction near 61%, which approximated a packed bed. Further details of this unit cell model were published as part of a study evaluating the impact of switching the fuel kernel from UCO to UN while retaining the standard particle geometry and graphite moderator [17].

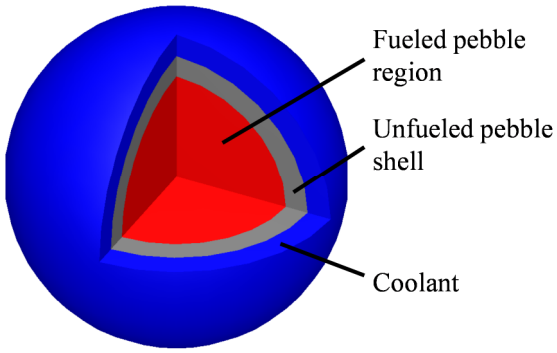

Fig. 3. Cutaway view of pebble unit cell geometry.

Baseline results were established for an FHR case using a graphite pebble matrix and shell (1.75 g/cm^3), UCO kernels (10.9 g/cm^3), 19.75% enriched ^{235}U, the standard TRISO particle geometry (500-µm kernel diameter), and a 34.6% TRISO packing fraction (PF). These values are from the AHTR design [15] but represent many FHRs, which tend to maximize enrichment and TRISO PF due to volume constraints. Flibe was modeled at 675°C with lithium enriched to 99.9% ^{7}Li, which is typical for FHR concepts.

All UN calculations assumed natural nitrogen; nitride fuel studies often use nitrogen that is enriched in ^{15}N, but the reduced fuel loading of particle fuels compared to solid pellets should limit parasitic capture in ^{14}N and consequent production of ^{14}C. Enriched nitrogen would improve neutronic performance but increase fuel costs.

RESULTS FOR FCM FUEL IN AN FHR

Previous efforts established the feasibility of using UN fuel kernels in FHR fuel [17]; however, FCM fuel involves more drastic changes. FCM calculations used the FHR unit cell described above but replaced graphite with SiC in the pebble matrix and shell, used an advanced TRISO particle geometry (800-µm kernel diameter, 75/20/40/20-µm coating thicknesses), and UN fuel kernels (13.76 g/cm^3). Several combinations of TRISO PF and ^{235}U enrichment levels were studied in addition to the reference values of 34.6% TRISO packing and 19.75% ^{235}U enrichment; all other model parameters remained unperturbed. Initial calculations focused on beginning of life (BOL) effects before considering depletion; parameter values that do not produce sufficient reactivity at BOL could then be discarded. Figures 4 and 5 show the infinite multiplication factor (k_{inf}) for FCM fuel in an FHR unit cell at several TRISO PFs and enrichment levels, respectively.

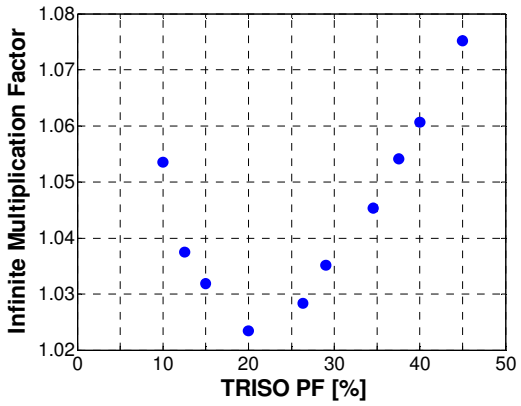

Fig. 4. BOL infinite multiplication factor as a function of TRISO PF for FCM fuel in an FHR unit cell with enrichment held constant at 19.75 % ^{235}U.

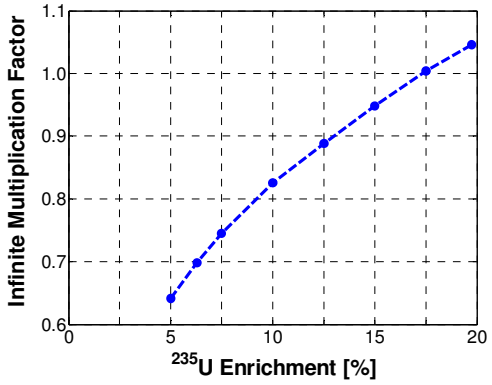

Fig. 5. BOL infinite multiplication factors as a function of enrichment for FHR FCM fuel with 34.6% TRISO PF.

When requiring BOL infinite multiplication factors above 1.03 to allow for neutron leakage and lifetime reactivity swing, FCM fuel may be feasible in an FHR using TRISO PFs less than 15% or greater than 30% and an enrichment above 17.5%. Infinite multiplication factors calculated during depletion studies are shown as a function of time in Fig. 6 and as a function of burnup (BU) in units of megawatt-days per metric ton of HM (MWd/tHM) in Fig. 7. TRISO PF and enrichment values remained at reference FHR values unless otherwise stated. Lifetimes in units of effective full-power days (EFPD) were estimated by defining end of life (EOL) as when the time-averaged reactivity reached a target value. This study used a value of 1.02 to account for 2% neutron leakage; future studies may increase this value but general trends should remain the same. Table I summarizes key results from these analyses. FCM options with high PFs exhibited different depletion trajectories than other cases due to increased self-shielding in larger kernels and higher total fuel masses. Overall, SiC moderates neutrons less effectively than graphite, but FHR FCM designs at or above the reference PF may achieve reasonable lifetimes.

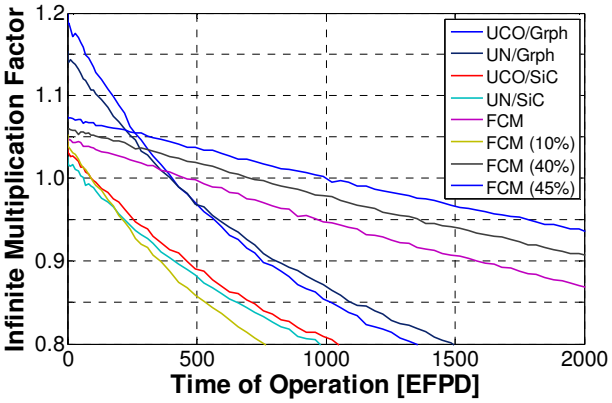

Fig. 6. Infinite multiplication factor as a function of time for several fuel options in an FHR.

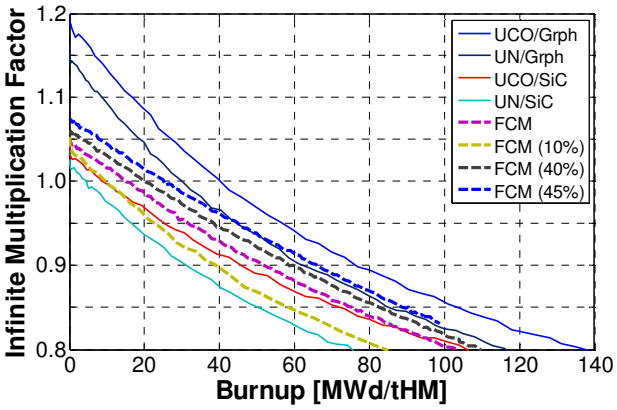

Fig. 7. Infinite multiplication factor as a function of burnup for several FHR fuel design options of interest.

Table I. Summary of key depletion results for FHR fuel design options of interest

Fuel	Particle Geometry	Matrix	Notes	BOL k_{inf}	EOL EFPD	EOL BU [GWd/tHM]
UCO	NGNP	Graphite	-	1.2086	774	78.2
UN	NGNP	Graphite	-	1.1546	718	55.3
UN	NGNP	Graphite	29% PF	1.1859	750	68.9
UCO	NGNP	SiC	-	1.0430	77	7.4
UN	NGNP	SiC	-	1.0265	6	0.4
UN	FCM	SiC	-	1.0453	538	17.6
UN	FCM	SiC	40% PF	1.0606	967	27.3
UN	FCM	SiC	45% PF	1.0752	1483	37.3
UN	FCM	SiC	10% PF	1.0535	93	10.0

CONCLUSIONS

A preliminary reactor physics assessment shows that FCM fuel designs appear feasible for application in an FHR though they do introduce some challenges due to reduced neutron moderation compared to graphite. The attractiveness of FCM fuels in FHRs depends upon the mission or goal of a specific FHR project; results indicate that FCM fuel could substantially extend the lifetime compared to the reference UCO/graphite design, but the discharge fuel burnup appears to be diminished, which means resource utilization has decreased.

Future work should include an economic assessment to weigh the fuel cost increase due to reduced discharged burnups against increased cycle lengths and capacity factors offered by the extended neutronic endurance of the fuel. Additional neutronics studies should be performed to investigate improved FHR FCM designs and to look at models for assemblies and entire reactor cores for FHRs instead of a simple unit cell. The thermal-hydraulic and fuel performance impacts of these FCM fuel designs also need to be assessed.

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REFERENCES

1. L. L. Snead et al., "Fully Ceramic Microencapsulated Fuels: A Transformational Technology for Present and Next Generation Reactors—Properties and Fabrication of FCM Fuel," *Transactions of the American Nuclear Society*, **104**, 668 (2011).
2. K. A. Terrani, L .L. Snead, and J. C. Gehin, "Microencapsulated Fuel Technology for Commercial Light Water and Advanced Reactor Application," *Journal of Nuclear Materials*, **427**, 209 (2012).
3. K. A. Terrani et al., "Fabrication and characterization of fully ceramic microencapsulated fuels," *Journal of Nuclear Materials*, **426**, 268 (2012).
4. J. J. Powers et al., "Fully Ceramic Microencapsulated Fuels: Characteristics and Potential LWR Applications," *Proc. of PHYSOR 2014*, Kyoto, Japan, September 28 – October 3, 2014.
5. F. Venneri et al., "Fully Ceramic Microencapsulated Fuels: A Transformational Technology for Present and Next Generation Reactors—A Preliminary Analysis of FCM Fuel Reactor Operation," *Transactions of the American Nuclear Society*, **104**, 671 (2011).
6. C. W. Forsberg et al., "Fluoride-Salt-Cooled High-Temperature Reactor (FHR) with Silicon-Carbide-Matrix Coated-Particle Fuel," *Transactions of the American Nuclear Society*, **107**, 907 (2012).
7. J. J. Powers and B. D. Wirth, "A Review of TRISO Fuel Performance Models," *Journal of Nuclear Materials*, **405**, 74 (2010).
8. D. L. Hanson, *A Review of Radionuclide Release From HTGR Cores During Normal Operation*, EPRI Report 1009382, Electric Power Research Institute, February 2004.
9. K. A. Terrani, L. L. Snead, and J. C. Gehin, "Fully Ceramic Microencapsulated Fuels for LWRs," *Transactions of the American Nuclear Society*, **106**, 1106–1107 (2012).
10. L. L. Snead et al., "FCM Fuel Development for LWR Applications," *Transactions of Reactor Fuel Performance 2012 (TopFuel 2012),* Manchester, UK, September 2–6, 2012, pp. 422–428 (2012).
11. J. D. Hunn and R. A. Lowden, *Data Compilation for AGR-1 Baseline Coated Particle Composite LEU01-46T*, ORNL/TM-2006/019, Oak Ridge National Laboratory, April 2006.
12. R. Hunt et al., "Preparation of $UC_{0.07–0.10}N_{0.90–0.93}$ spheres for TRISO coated fuel particles," *Journal of Nuclear Materials*, 448, 399–403 (2014).
13. T. B. Lindemer et al., "Carbothermic synthesis of 820μm uranium nitride kernels: Literature review, thermodynamics, analysis, and related experiments," *Journal of Nuclear Materials*, in press (2013).
14. M. Fratoni, E. Greenspan, and P. F. Peterson, "Neutronic and depletion analysis of the PB-AHTR," *Proc. of GLOBAL 2007 Conference on Advanced Nuclear Fuel Cycles and Systems (GLOBAL 2007)*, Boise, ID, September 9–13, 2007, American Nuclear Society (2007).
15. D. E. Holcomb et al., *Core and Refueling Design Studies for the Advanced High Temperature Reactor*, ORNL/TM-2011/365, Oak Ridge National Laboratory, September 2012.
16. *SCALE: A Comprehensive Modeling and Simulation Suite for Nuclear Safety Analysis and Design*, ORNL/TM-2005/39, Version 6.1. Available from Radiation Safety Information Computational Center at Oak Ridge National Laboratory as CCC-785. (2011).
17. J. J. Powers and K. A. Terrani, "Uranium Nitride: Enabling New Applications for TRISO Fuel Particles," *Transactions of the 2013 LWR Fuel Performance Meeting (TopFuel 2013),* Charlotte, NC, September 15–19, 2013.