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Fully ceramic microencapsulated fuel in prismatic high temperature gas-cooled reactors: Analysis of reactor performance and safety characteristics

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ABSTRACT

Advanced nuclear reactor technologies have the potential to expand the missions of nuclear energy while reducing carbon emissions. This paper presents scoping reactor physics and thermal hydraulics analysis of a high temperature gas-cooled reactor (HTGR) using the fully ceramic microencapsulated (FCM) fuel form, and demonstrates the feasibility of FCM fueled HTGRs. FCM fuel consists of tristructural isotropic (TRISO) coated fuel particles embedded in a matrix of silicon carbide (SiC). The potential advantages of FCM fuel, which uses a monolithic SiC matrix, over conventional HTGR fuels with a carbon-based matrix include: a long refueling interval; high stability of the SiC matrix under irradiation with limited swelling; high fission product retention of the fuel form, with the SiC matrix acting as an additional barrier to fission product release; and enhanced oxidation resistance during normal operation and air ingress accidents. In addition, the literature shows that the effective thermal conductivity of SiC fuel compacts and conventional HTGR compacts are expected to be similar.

The key finding of this study is that FCM fuel, within the form factor of a typical General Atomics prismatic graphite block, exhibits similar fuel cycle performance to conventional HTGR fuel. The reactor cycle length, discharge burnup, and natural resource utilization are similar. However, the reduced moderation in the FCM designs considered here does marginally reduce the discharge burnup, and therefore natural resource utilization, versus the reference HTGR design.

The hardened neutron flux spectrum resulting from the SiC matrix, which displaces carbon from the core, requires a slightly higher packing fraction of conventional uranium oxy-carbide (UCO) fuel kernels or the use of higher density uranium mononitride (UN)-based fuel kernels. These options will marginally increase the decay power, because they harden the neutron flux energy spectrum and increase the density of ^{238}U in the fuel. In one case considered, this will increase the absorption of neutrons in ^{238}U, and the resultant impact of ^{239}Np isotope on the decay power. The Doppler coefficients normalized per total fuel heat capacity are weaker in the FCM-fueled designs than in the reference HTGR design. This impacts the energy deposition in a control rod ejection accident, and hence the design of potential transient tests of these fuel forms. In addition, analyses of loss-of-forced cooling accidents indicate that the fuel temperature during these design basis accidents are up to ~30 °C higher with FCM fuel than with conventional HTGR fuels due to the increased decay power.

1. Introduction

To meet global energy demand while simultaneously reducing greenhouse gas emissions we must realize the potential of highly innovative advanced nuclear reactor technologies. Novel modular reactors with gas coolants are potentially well suited to unique applications or flexible missions, including efficient and cost-effective integration with non-electricity generating applications. Examples include desalination, synfuels production, hydrogen production, and process heat for petrochemical and related industrial processes that require operating temperatures up to 900 °C (Brown and Revankar, 2012a, 2012b; Idaho National Laboratory, 2016).

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The High Temperature Gas-cooled Reactor (HTGR) was identified by the U.S. Department of Energy Office of Nuclear Energy (DOE-NE) Advanced Demonstration and Test Reactor (ADTR) Options Study as the most promising reactor technology option for high temperature process heat applications (Petti et al., 2016, 2017). Specifically, the ADTR Study identified HTGRs as the most promising option for: "a high-temperature process heat application (e.g., synfuels production) for industrial applications and electricity demonstration using an advanced reactor system to illustrate the potential that nuclear energy has in reducing the carbon footprint in the U.S. industrial sector." Petti et al. (2016, 2017) concluded that HTGRs have a high Technology-Readiness-Level (TRL) which will enable near-term deployment in flexible energy missions. FCM fueled HTGRs would require more research and testing to achieve a similar TRL as HTGRs with conventional fuel compacts. However, the required research and development would benefit from the significant knowledge base related to conventional HTGRs.

This paper explores the reactor performance and safety characteristics of a modular HTGR using the FCM fuel form within the form factor of the conventional General Atomics prismatic block. FCM fuel consists of tristructural isotropic (TRISO) coated fuel particles embedded in a matrix of silicon carbide (SiC) (Terrani, 2012a, 2012b). FCM fuel development has been conducted mostly for light water reactors under the DOE-NE Advanced Fuels Campaign (Brown et al., 2013a, 2013b; Carmack et al., 2013; Snead et al., 2014; Terrani et al., 2012a, 2012b, 2015a; Zinkle et al., 2014).

Owing to its carefully processes SiC matrix, FCM fuel exhibits neutron irradiation-stability (Terrani et al., 2018) with a potential for high fission product retention. This is due to multiple barriers to fission product release, both the TRISO particles and SiC matrix. FCM fuel also features the exceptional mechanical robustness of the SiC matrix as compared to graphite and graphitic materials (Terrani et al., 2018). Given the exceptional oxidation resistance of SiC, the FCM concept is an excellent candidate for an HTGR fuel that has enhanced tolerance to the impact of an air ingress accident (Terrani et al., 2014; Terrani and Silva, 2015b).

FCM fuel has been considered with engineered TRISO fuel particles that differ substantially from historic TRISO particles. FCM concepts have explored larger fuel kernel diameters than traditional TRISO particles as well as fuel kernel materials that offer high heavy metal density, such as uranium mononitride (UN)-based kernels (Terrani et al., 2014; Terrani and Silva, 2015b). The enhanced heavy metal loading can achieve longer cycle lengths with UN kernels instead of the oxide or oxy-carbide (UCO) kernels used previously in HTGR applications.

Preliminary studies of FCM fuel in advanced reactors (Powers, 2014, 2016; Venneri et al., 2011) indicate the potential for longer reactor cycle lengths than reference TRISO fuel concepts from the Next Generation Nuclear Plant (NGNP) program. A modular FCM-fueled HTGR with long refueling intervals would be able to address operational missions other than the delivery of base load electric power and would also enable deployment flexibility as a mobile reactor.

FCM fuel compact production using Spark Plasma Sintering (SPS) has been demonstrated (Terrani et al., 2016). Terrani et al. (2016) produced FCM pellets with high-density and high-packing-fraction UN TRISO particles. The present focus is on analysis of the FCM concept in prismatic HTGRs. One challenge is that FCM fuel hardens the neutron energy spectrum in the HTGR by removing carbon and replacing it with SiC. To compensate for this, and maintain cycle length, an increased packing fraction of fuel particles may be used, or a higher density fuel kernel. Both of these options will increase the decay heat, because they increase the quantity of ^{238}U in the fuel. This will increase the resonance absorption, and the resultant impact of ^{239}Np isotope on the decay heat.

The objective of this paper is to perform a preliminary reactor physics analysis of FCM fuel in a typical modular HTGR. These scoping studies are not intended to be representative of all possible HTGR designs, but rather to elucidate information about potential impacts on reactor performance and safety characteristics.

2. Study approach

The approach taken in this study was to perform parametric reactor physics evaluations of the impact of several changes in fuel configuration on HTGR fuel cycle performance and reactor performance and safety characteristics. The modular HTGR fuel block baseline configuration in the study was based on Pope (2012) with 15.5%-enriched uranium oxy-carbide ($UC_{0.5}O_{1.5}$ or UCO) fuel. The configuration used in this study is similar to that adopted as part of an international benchmark problem (Strydom et al., 2016). Note that in UCO fuel, the uranium carbide is present to mitigate CO production and prevent resultant fuel failure (Olander, 2009). Determination of the optimal content of uranium carbide in HTGR fuel kernels is the subject of ongoing studies as part of the DOE-NE Advanced Reactor Concept Cooperative Agreement Award led by X-Energy, LLC (McMurray et al., 2017).

The reactor physics calculations in this paper were conducted using the ENDF/B-VII.0 libraries (Chadwick et al., 2006). It is noted that some improvements have been made to both the ^{12}C radiative capture cross section and fission yields in the more recent ENDF/B-VII.1 library (Chadwick et al., 2011), which is the cross-section library used in the Advanced Gas Reactor (AGR) fuel qualification program (Demkowicz et al., 2015). It will be shown in Section 4.3 that the cycle lengths calculated with ENDF/B-VII.0 would be slightly higher than that with ENDF/B-VII.1, but the differences were found to be small and very similar among all the cases studied. The calculations in this paper are thus expected to give reasonable results as they were performed using a consistent approach. The nuclear data for the configurations considered in this paper is all taken at a temperature of 900 K, with the exception of the thermal neutron scattering data for graphite, which is used at 1000 K, and data for 1330 nuclides (including fission and transmutation products) was included in the calculations. This approach is consistent with the calculations in McMurray et al. (2017).

To assess the impact on reactor performance and safety characteristics, reactor physics calculations are performed at both the single prismatic block and two-dimensional core level, as shown in Fig.1. The 2-D core model consists of 66 single blocks along with the inner and outer reflectors. Taking into account the scoping nature of the current study, the reflectors are simplified and modeled as graphite. The control rods and the lumped burnable absorbers are not modeled in the current study, but will be analyzed in the next paper in a more detailed 3-D model. Periodic boundary conditions are used for the single block configuration which assumes an infinite lattice and thus no neutron leakage is considered ($k_{eff} = k_{inf}$). Similarly, a periodic boundary condition is applied to the z-direction of the 2-D core configuration, and the axial neutron leakage is thus considered to be nil.

The general calculation approach followed that of Pope (2012), which was the basis used to evaluate the fuel cycle performance of HTGRs in the DOE-NE Fuel Cycle Evaluation and Screening Study (Wigeland et al., 2014). The overall matrix of scoping neutronic cases considered in this work include:

- A reference case consisting of a conventional prismatic block (and core) configuration with **carbon** matrix compacts and **15.5%**-enriched **UCO** fuel with a **35%** packing fraction and a typical TRISO particle geometry (referred to as REF-UCO),