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Neutronic evaluation of a PWR with fully ceramic microencapsulated fuel. Part II: Nodal core calculations and preliminary study of thermal hydraulic feedback

Nicholas R. Brown *, Hans Ludewig, Arnold Aronson, Gilad Raitses, Michael Todosow

Brookhaven National Laboratory, PO BOX 5000, Upton, NY 11973, United States

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ABSTRACT

The fully ceramic microencapsulated (FCM) fuel is based on the tri-isotropic (TRISO) carbon coated fuel particles. These particles were developed and demonstrated for use in high temperature gas reactors. It has been proposed to use these particles in light water reactors to provide potential operational and safety benefits. The reference fuel in this case assumes TRISO-like particles with a ~20%-enriched uranium-nitride kernel embedded in a silicon carbide (SiC) matrix. The fuel particles are contained in a "compact" which is then inserted into a cladding. The fuel assembly features the same dimensions as a standard 17 × 17 Westinghouse fuel assembly. FCM fuel requires fission products to traverse several barriers in the proposed fuel design before reaching the cladding. FCM fuel may also reduce fuel-cladding interaction and fuel pellet swelling while enabling higher fuel burn-up. This study is a neutronic evaluation of the use of FCM fuel in an advanced pressurized water reactor (PWR). On the lattice level, the SERPENT Monte Carlo and TRITON deterministic tools were used, while the whole core simulation was based on the three-dimensional PARCS nodal code.

The present paper focuses on two of the issues associated with this proposed implementation: specifically the development of a reasonable reference full-core model of an advanced PWR with FCM fuel and the response of the PWR to a reactivity insertion accident (RIA). This work addresses the issues of the increased power density and transients that occur on short time-scales in a PWR. In this case, the RIA takes the form of a control rod ejection for a typical PWR reactor. This results in a sudden increase in power and a corresponding increase in fuel kernel temperature. In the case of a PWR, this response is more demanding than in the case of a gas-cooled reactor, because the kinetic parameters and feedback coefficients of the two reactors are quite different. The parameters for the fuel and matrix material in the PARCS thermal–hydraulic module were modified to reflect the different geometry and materials. Preliminary data for both un-irradiated and irradiated SiC were obtained from the literature and included in the analyses. A super prompt critical RIA produces an average energy deposition (<124.6 J/g) that is estimated for different simplified thermal representations of the FCM fuel pin.

1. Objective of the present study

This is the second part of a two part paper. Part I of the paper consists of lattice-level analysis, determination of cycle lengths, and calculation of reactivity coefficients. Part II of the paper consists of nodal analysis of an advanced pressurized water reactor (PWR) core with fully ceramic microencapsulated (FCM) fuel utilizing the PARCS code.

A full-core nodal model of an advanced PWR with FCM fuel is developed and utilized to assess the response of the fuel to a reactivity insertion accident (RIA). PARCS was selected as the core simulator in this study (Downar et al., 2002); PARCS is the US Nuclear Regulatory Commission (NRC) sponsored core simulator and is a state-of-the-art nodal code. PARCS is a three-dimensional simulator that solves the multi-group neutron diffusion equation in a variety of geometries. PARCS features built-in spatial kinetics capability for transient analysis and integrates a standalone thermal hydraulics module for single-phase coolant flow conditions with uranium oxide (UO_X) fuel. For this work, the PARCS thermal hydraulics module was modified to provide appropriate thermal hydraulic feedback for the FCM fuel. PARCS is an acronym for Purdue Advanced Reactor Core Simulator, although the present code development takes place at the University of Michigan.

Many lattice physics tools, which are used to generate diffusion parameters for PARCS, do not have the capability to accurately model FCM fuel due to the inherent double heterogeneity (DH)

* Corresponding author. Tel.: +1 (631) 344 5204.
E-mail address: nbrown@bnl.gov (N.R. Brown).

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in the pin geometry. Additionally, it is beyond the ability of most lattice physics codes to perform branch calculations where only the fuel particle kernel temperature is perturbed. However, the code SERPENT (Leppänen, 2007), when coupled to a modified version of the cross-section generation tool SerpentXS (Herman et al., 2011), can be utilized to explicitly model the tri-isotopic (TRISO) particles within the FCM pins. TRITON, a module from the SCALE 6.1 package, was utilized as a comparison tool for the few-group homogenization capability (Oak Ridge, 2011). The US NRC has extensively utilized the TRITON code for generation of few-group parameters in regulatory applications. However, TRITON cannot presently calculate thermal hydraulic branches for explicit FCM particles (Gentry and Godfrey, 2011). Additionally, the few-group parameters generated with TRITON have not been validated for geometries with DH.

The Westinghouse AP1000 was selected as the reference core for analyzing the operational and safety performance of FCM fuels in PWRs (Westinghouse, 2009). In this document, the AP1000 is henceforth referred to as the reference PWR. The reference PWR is a Generation 3+ reactor based on the proven performance of Westinghouse designed PWRs, and incorporating enhanced plant safety and operations with reduced construction costs. The reference PWR utilizes the well-known 17×17 fuel assembly design consisting of 264 fuel rods and 25 guide tubes. The reference PWR serves as a representative reactor configuration to study the compatibility of FCM fuel within the constraints of a present generation PWR.

2. Few-group parameter generation capability of the SERPENT code

In part I of this article, a detailed benchmark calculation of FCM assembly reactivity, burn-up, material inventories, and pin power was performed with the SERPENT and TRITON codes. However, for application in a nodal core simulator the main function is the generation of few-group parameters. The purpose of this section is to assess the few-group parameter generation capability of the SERPENT code by comparison with the TRITON code.

In this section, a set of simple benchmark problems is utilized to evaluate the ability of SERPENT to calculate an assembly-level k-infinity and homogenized few group parameters. A test problem allows for the study of few-group constant generation capabilities in a fuel region that is adjacent to a moderator region. Because the capability of the TRITON code to perform branching calculations with DH geometry has not yet been developed, it was decided to test the few-group parameter generation capability utilizing a conventional UO_X 17×17 PWR assembly and reflector.

The assembly level benchmark consists of a single assembly or quarter assembly with the geometry specifications given in the FSAR of the reference PWR (Westinghouse, 2009). There are no IFBA rods in the assembly and the ^{235}U enrichment is 2.35%. The boron concentrations are fixed at 0 and 1000 parts-per-million (ppm). The cladding material is taken as a generic alloy of zirconium with trace amounts of nickel and iron. Table 1 shows the infinite multiplication factor results from MCNP (Los Alamos, 2008), SERPENT, and TRITON. The reactivity difference for the SERPENT and TRITON models is shown with MCNP as the reference solution. The reactivity difference is calculated as $\Delta\rho = (k_2 - k_1)/(k_2 k_1)$. The NEWT module was utilized for the transport calculation in TRITON. The default convergence criteria were utilized as well as the default number of quadrature angles. Twenty-four polygon sides were utilized to approximate a fuel pin in NEWT. Two TRITON results are shown: a 238-group result and a 49-group result. The 238-group library self-shielding is determined utilizing the CENTRM module of the TRITON code for a square-pitch lattice cell. The problem-specific 49-group library is collapsed based on the flux-spectra in the 238-group result. For all of the benchmark calculations, the most recently available version of the ENDF/B-VII.0 library is utilized. The fuel temperature is fixed at 900 K and all other temperatures are fixed at 600 K. For the stochastic calculations, the unresolved resonance treatment is deactivated to ensure a one-to-one comparison.

Table 1
Lattice benchmark comparison.

	k-Infinity	σ	$\Delta\rho$ (pcm)
2.35% enriched, 0 ppm Boron			
MCNP5	1.28139	0.00010	–
MCNP5 U238 s(α,β)	1.28166	0.00011	16
SERPENT	1.28145	0.00008	4
SERPENT (DBRC ENDF/B-VII.0)	1.27952	0.00011	−114
SERPENT (DBRC LANL-T16 2003)	1.27970	0.00009	−103
TRITON/NEWT (238 g)	1.27568	–	−349
TRITON/NEWT (49 g)	1.27755	–	−235
2.35% enriched, 1000 ppm Boron			
MCNP5	1.12458	0.00012	–
MCNP5 U238 s(α,β)	1.12501	0.00012	34
SERPENT	1.12468	0.00009	8
SERPENT (DBRC ENDF/B-VII.0)	1.12303	0.00010	−123
SERPENT (DBRC LANL-T16 2003)	1.12322	0.00009	−108
TRITON/NEWT (238 g)	1.11994	–	−368
TRITON/NEWT (49 g)	1.12169	–	−229

The difference in k-infinity from SERPENT to MCNP is on the order of the statistical accuracy; there is essentially no difference in this result. The bias between the TRITON and MCNP codes is relatively consistent for the two cases. The bias is slightly smaller for the 49-group TRITON result than the 238-group TRITON result. Some of the bias between MCNP and TRITON comes from the CENTRM calculation of self-shielding, which is based on a cylindrical "equivalent" cell. This self-shielding calculation does not account for local variation of the neutron flux-energy spectrum within the assembly. The 49-group cross section set is based on a problem-specific collapse.

For heavy nuclides that have scattering resonances in the epithermal energy range (e.g., ^{238}U), resonance absorption is underpredicted when using the ENDF continuous energy cross sections (Becker et al., 2009). Thus, the MCNP calculation in Table 1 overpredicts k-infinity by a small margin. SERPENT has a built-in feature to correct for this, known as the Doppler broadening rejection correction (DBRC). This feature requires a very low temperature continuous energy evaluation of the cross section of ^{238}U. As a benchmark, the DBRC is utilized to compare with TRITON and MCNP5. Two low temperature ^{238}U ACE format cross-section files were utilized for the DBRC: the ENDF/B-VII.0 cross section evaluation at 1 K, generated using the processing code NJOY99.161, and the LANL-T16 (2003) evaluation at 77 K (Los Alamos, 2008).

For this geometry, it is apparent that the reactivity difference ($\Delta\rho$) with the DBRC is on the order of 100 pcm, with a raw difference (delta-k) on the order of 190 pcm. In the literature, for an LWR pin cell at 1200 K, the raw difference (delta-k) is on the order of 360 pcm (Becker et al., 2009). This difference from the literature may be due to the fuel temperatures utilized in the calculation (900 K vs. 1200 K) as well as increased thermalization due to the presence of moderator-filled guide tubes. The MCNP5 cases were also run using the ENDF/B-VII.0 S(α,β) library for ^{238}U in UO_2 at 800 K. These calculations are within 3σ (99.9% confidence interval) of the result reported in Table 1. It should be noted that in (Becker et al., 2009), two specialized S(α,β) libraries that include the epithermal energy range were developed and utilized for ^{238}U.

To test the ability of the SERPENT and TRITON codes to generate few-group cross sections, a simple test problem was developed.