A Comparison of Deterministic and Monte Carlo Depletion Methods for HTGR Fuel Elements

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INTRODUCTION

High-temperature reactors (HTRs) use graphite-moderated fuel forms and helium gas as a coolant. There are two main forms of the HTR fuels: pebbles are used in the pebble bed high-temperature reactor (PBR), and rods are used in the modular high-temperature gas-cooled reactor (MHTGR). In PBRs the fuel elements are 6 cm diameter spheres. In MHTGRs they are cylindrical graphite compacts that are inserted into hexagonal graphite blocks.

The fuel element consists of a double-layered geometry: (1) small fuel particles (generally comprised of a fuel grain ~0.5 mm in diameter surrounded by a 0.25 mm thick graphite shell) and (2) the macroscopic fuel elements, which are large spheres or "pebbles" in a PBR or cylindrical compacts in an MHTGR. This double heterogeneity must be explicitly treated because the fuel particles are closely packed, and the interactions between the fuel particles, as well as the slowing down within them, cannot be ignored. The doubly-heterogeneous (DH) nature of the fuel lattice cannot be modeled directly in resonance processing codes that rely on the collision probability technique. Instead, one must use either the Dancoff factor approach or another method that does not rely on a collision probability technique to account for the lattice effects.

DEPLETION OF DH MEDIA

A new DH unit cell type was introduced in SCALE 5.1 [1] that uses CENTRM, PMC, and additional modules to generate problem-dependent multigroup cross sections [2]. When the DH unit cell type is selected, first the pointwise (PW) flux disadvantage factors in the fuel grains (coated particles) are calculated. Then, these factors are used to generate the cell-weighted PW cross sections for the homogenized fuel region in the fuel pebble. Finally, these spatially averaged PW cross sections are used to calculate the flux distribution in the fuel element, which is then used in PMC to generate the multigroup problem-dependent cross sections. This cross-section process has been folded into the TRITON depletion sequence [3] to allow depletion of DH fuels. Because the SCALE DH treatment generates effective multigroup cross sections for a homogenous fuel cell, only the homogenized region is included in the transport model; it is not necessary to explicitly model the grain level of heterogeneity. However, the ORIGEN depletion step focuses on only the contents of the fuel grain. TRITON is responsible for conversion of discrete composition number densities used in the first pass of cross-section processing and in ORIGEN calculations to the homogenized number densities used in the second cross-section processing pass and transport calculation.

During the depletion calculation, grain-level number densities are updated by ORIGEN calculations. TRITON is able to deplete multiple grain types within one or more separate fuel elements by independent ORIGEN calls using the flux and number densities that characterize each grain. As noted above, the cross sections used by TRITON and collapsed for use in ORIGEN have been homogenized using disadvantage factors for each grain. Therefore, calculations using the homogenized cross sections and resultant homogeneous media flux will reproduce the correct reaction rates in ORIGEN to model local burnup.

VALIDATION OF THE DEPLETION SEQUENCE

Although the DH cross-section processing methodology has been tested in criticality calculations for code-to-code benchmarks [2, 4], the nature of the depletion process introduces additional complexities that must be validated. A lack of experimental measurements for the depletion of high-temperature gas-cooled reactor (HTGR) fuel elements makes direct validation

difficult at present; however, a computational benchmark has been developed to prescribe simple infinite lattice fuel configurations for code-to-code testing [5, 6]. TRITON calculations have been performed for each of the three benchmarks configurations: (1) an infinite lattice of grains (no double heterogeneity), (2) a three-dimensional (3-D) infinite lattice of pebbles with embedded fuel grains, and (3) a two-dimensional (2-D) infinite hexagonal fuel element loaded with DH compacts and coolant holes. Benchmark solutions are currently being performed by a number of international participants. One of these solutions has been performed using the Serpent package [7]. Serpent is a 3-D continuous-energy Monte Carlo reactor physics burnup calculation code, developed at VTT Technical Research Centre of Finland. A new explicit particle fuel model has been developed within Serpent to account for the heterogeneity effects of randomly distributed fuel grains within HTGR fuels.

RESULTS

Analyses of all three benchmark configurations have been completed with both TRITON and Serpent. Within TRITON, the 3-D KENO V.a Monte Carlo transport solver was used for the 3-D grain and pebble calculations; both the 2-D NEWT deterministic solver and KENO V.a were used for the prismatic calculation. TRITON calculations were performed using the 238-group ENDF/B-VII library with SCALE 6.0. Figure 1 shows outstanding agreement between multigroup TRITON/KENO and continuous energy Serpent calculations for the entire fuel burnup. Figure 2 illustrates the results of depletion calculations for the prismatic fuel model. Eigenvalues calculated by NEWT are shown; for clarity KENO V.a results are not shown, but are statistically identical to NEWT results. Results for the prismatic phase of the benchmark are also good, but a slight discrepancy is noted with TRITON/NEWT giving a slight overprediction of k_{inf} for most of the depletion relative to Serpent, with the two results crossing over at about 95 GWd/MTU burnup. The cause for the difference remains to be investigated, but it is believed that it could possibly result from the gas coolant hole in the prism, which is not considered in the cross-section processing.

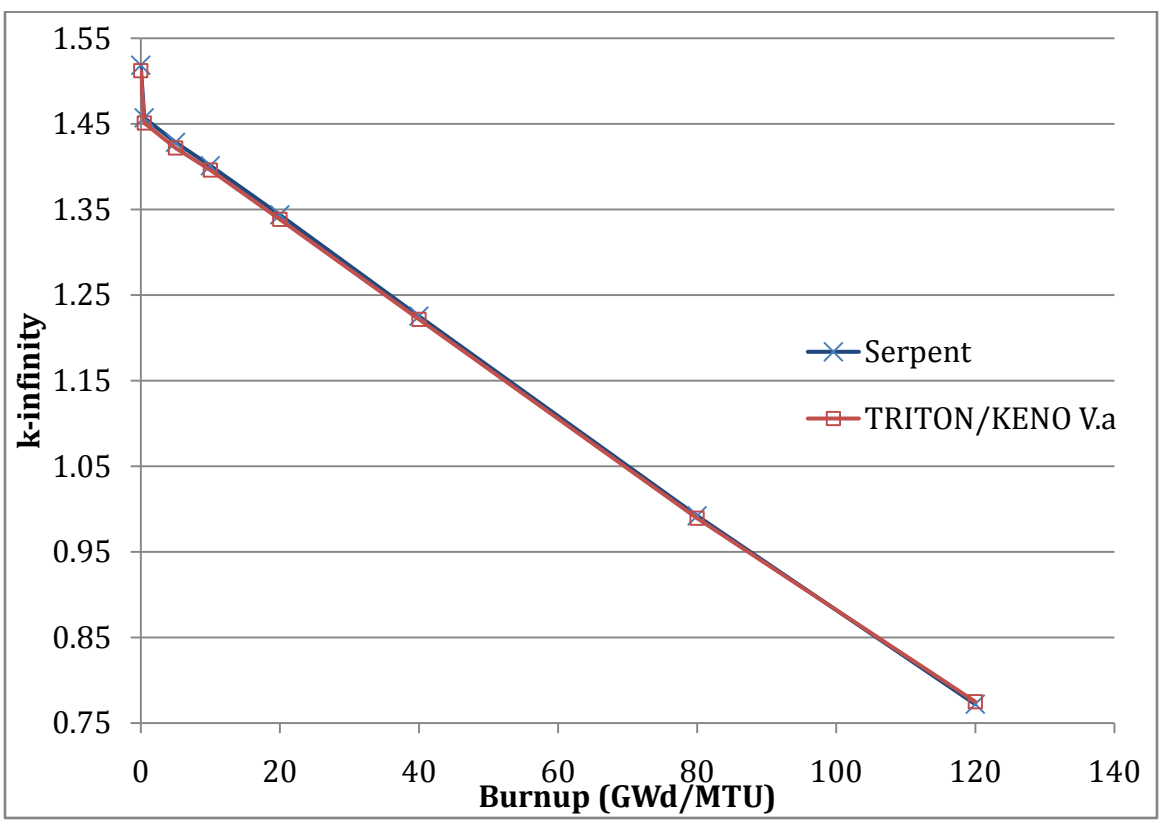

Fig. 1. Comparison of TRITON/KENO V.a and Serpent depletion results for pebble fuel.

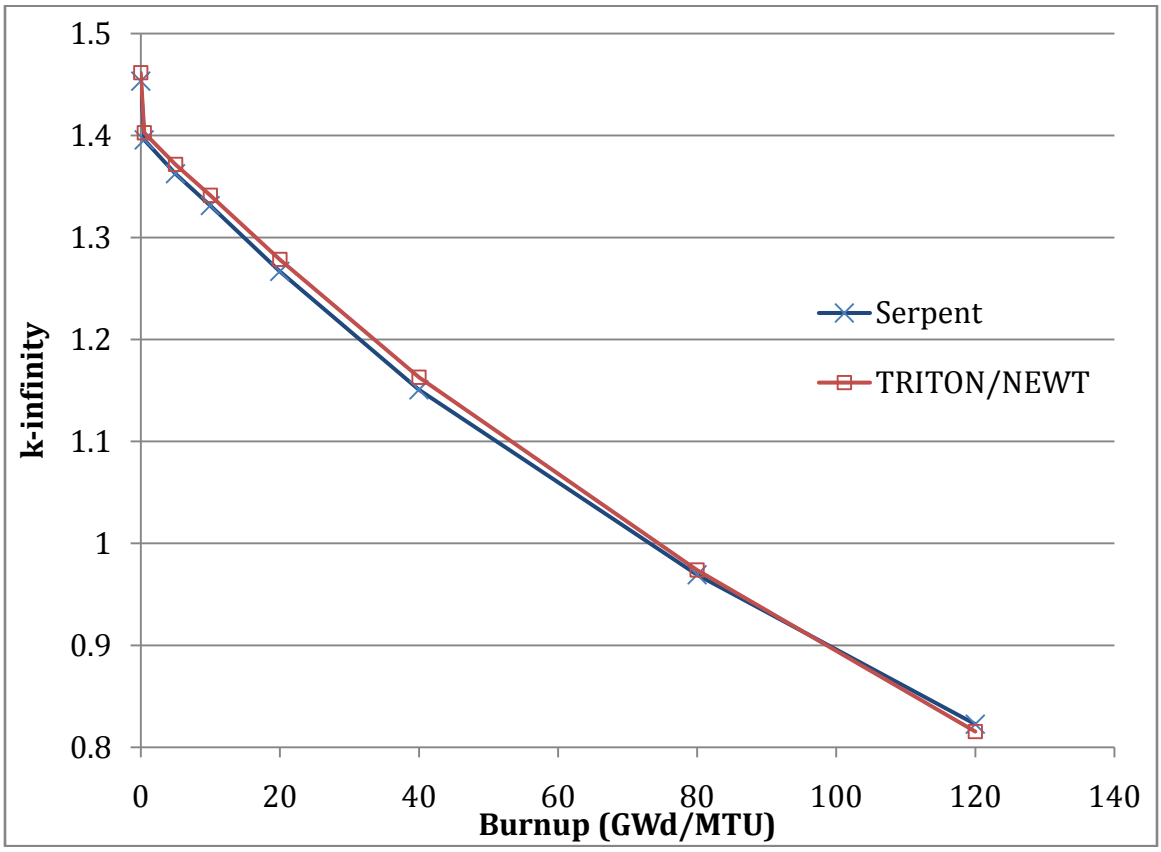

Fig. 2. Comparison of TRITON/NEWT and Serpent depletion results for prismatic fuel.

CONCLUSIONS

Two independent methods using both deterministic and Monte Carlo methods were used to perform depletion calculations for a set of simple HTGR-type configurations and were found to be in excellent agreement. The explicit representation of randomly dispersed particles used within a continuous energy Serpent solution represents the best estimate simulation of fuel particle interactions. The DH treatment in TRITON allows simulation of grain-to-grain effects without requiring the detailed representation of each individual grain and was found to be in very close agreement with Serpent results. In absence of qualified experimental data for spent fuel samples burned in a prototypic environment, this very close agreement of completely independent methods and data provides a great deal of confidence in the capabilities of both methods to represent the physics of HTGR fuel depletion.

Additional submissions of results for this benchmark from other organizations are expected, which will provide an even greater variety of methods and data for comparison. Compilation and comparison of results should provide considerable insight into strengths and limitations of various methods and data for HTGR analysis.

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