HTGR Reactor Physics and Burnup Calculations Using the Serpent Monte Carlo Code

Jaakko Leppänen

VTT Technical Research Centre of Finland, P.O. Box 1000, FI-02044 VTT, Finland, Jaakko.Leppanen@vtt.fi

Mark DeHart

Oak Ridge National Laboratory,[*] P.O. Box 2008, Bldg. 5700, Oak Ridge, TN 37831-6170, USA, dehartmd@ornl.gov

INTRODUCTION

One of the main advantages of the continuous-energy Monte Carlo method is its versatility and the capability to model any fuel or reactor configuration without major approximations. This capability becomes particularly valuable in studies involving innovative reactor designs and next-generation systems, which often lie beyond the capabilities of deterministic light water reactor transport codes.

In this study, a conceptual prismatic high-temperature gas-cooled reactor (HTGR) fuel assembly was modeled using the Serpent Monte Carlo reactor physics burnup calculation code, under development at VTT Technical Research Centre of Finland since 2004 [1].[a] A new geometry model was developed for the Serpent code in order to account for the heterogeneity effects encountered in randomly dispersed particle fuels. The results are compared to other Monte Carlo and deterministic transport codes, and the study also serves as a test case for the modules and methods in SCALE 6 [2].

DESCRIPTION OF THE ACTUAL WORK

HTGR Calculations

Three HTGR fuel models consisting of coated fuel particles and graphite moderator were constructed based on the data in two technical reports [3,4]:

1. single fuel particle in regular lattice,
2. fuel compact in regular lattice, and
3. prismatic fuel block with and without burnable poison (boron).

The same material compositions and main geometry dimensions were used for each model, and the local particle packing fraction was preserved in the single-particle case.

Criticality calculations were carried out using three Monte Carlo neutron transport codes (Serpent, MCNP5 [5], and KENO-VI) and one deterministic code (NEWT). KENO-VI and NEWT are included in the SCALE 6 code system as part of the TRITON sequence [6]. All codes used cross-section libraries based on ENDF/B-VII data. The KENO-VI calculations were repeated using both continuous-energy and 238-group cross sections. The same 238-group library was used in the NEWT calculations. Fuel burnup in the prismatic block model was simulated using Serpent and the results compared to TRITON-NEWT calculations.

Heterogeneity Effects

The simplest approach to modeling HTGR fuel as a regular lattice of fuel particles fails to account for certain heterogeneity effects. The local fuel-to-moderator ratio is generally not preserved, and the streaming paths in the lattice planes between particles are longer compared to a randomly dispersed fuel. The CENTRM/PMC multigroup cross-section preprocessor sequence in SCALE has the capability to homogenize this type of geometry using the "double-het" option, but continuous-energy Monte Carlo codes need to rely on more explicit models.

For the purpose of this study, a new particle fuel model was developed for the Serpent code. The random coordinates of each particle are read from a separate pregenerated input file into a three-dimensional search mesh used by the geometry routine. This model is capable of handling more than 100,000 fuel particles explicitly, with only about 20-50% increase in the overall CPU time. The model also works for double-stochastic pebble-bed geometries.

RESULTS

Table I shows the results of the criticality calculations. All values in each case are within less than 1% of each other, and the best agreement is found between Serpent and MCNP5, both using the same ACE format cross-section libraries. Differences between Serpent and continuous-energy KENO-VI are larger, but only on the order of a few hundred per cent mille (pcm). Comparison between continuous-energy and multigroup codes becomes more complicated, but the validity of the double-het preprocessing method is clearly shown. The explicit model in Serpent results in a systematically higher k_{eff} compared to the regular lattice calculation. This is due to the various heterogeneity effects, but the differences are not easily traced back to a single factor.

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Table I. Comparison of criticality calculations among Serpent, MCNP5, KENO-VI, and NEWT using different particle fuel options. Statistical uncertainties of Monte Carlo calculations are in units of pcm.

Geometry	Code	Particle model		
Particle in cell	KENO-VI (CE)[a]	Single particle	1.07777	(45)
	KENO-VI (MG)[b]	Multiregion	1.07013	(43)
	MCNP5	Single particle	1.07711	(48)
	Serpent	Single particle	1.07768	(52)
Compact	KENO-VI (CE)	Regular lattice	1.42130	(40)
	KENO-VI (MG)	Double het	1.41493	(33)
	NEWT	Double het	1.41394	
	MCNP5	Regular lattice	1.42000	(38)
	Serpent	Regular lattice	1.42145	(35)
	Serpent	Explicit model	1.42260	(38)
Prismatic block without BP[c]	KENO-VI (CE)	Regular lattice	1.52985	(30)
	KENO-VI (MG)	Double het	1.53451	(31)
	NEWT	Double het	1.53449	
	MCNP5	Regular lattice	1.53189	(32)
	Serpent	Regular lattice	1.53160	(31)
	Serpent	Explicit model	1.53261	(34)
Prismatic block with BP	KENO-VI (CE)	Regular lattice	1.21975	(48)
	KENO-VI (MG)	Double het	1.21865	(38)
	NEWT	Double het	1.22282	
	MCNP5	Regular lattice	1.21558	(44)
	Serpent	Regular lattice	1.21545	(46)
	Serpent	Explicit model	1.21702	(45)

[a]CE = continuous energy
[b]MG = multigroup
[c]BP = burnable poison

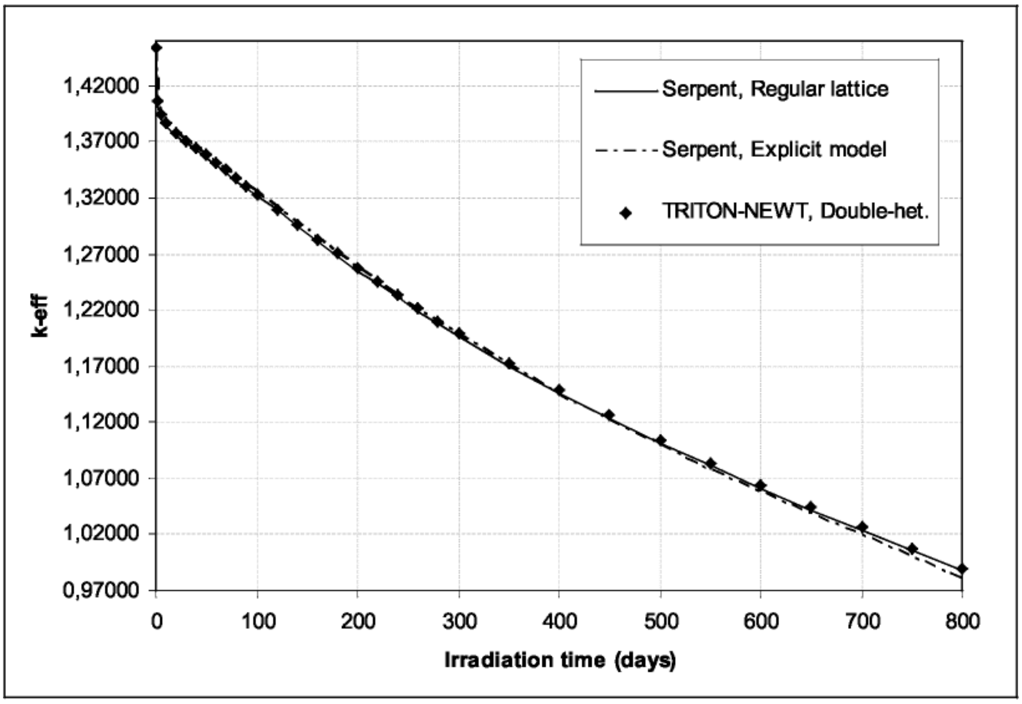

Fig. 1. Comparison of burnup calculations between Serpent and TRITON-NEWT. Prismatic block model without burnable poison.

Burnup calculations using Serpent and TRITON-NEWT are summarized in Fig. 1, which shows the results for the prismatic block model surrounded by reflective boundary conditions. All uranium in the fuel particles was treated as a single material zone. The results depend on several factors, such as the number of fission product cross sections included in the calculation. The results show remarkable consistency when 250 actinide and fission product nuclides are included.

The differences between the regular lattice and the explicit particle fuel model in Serpent are also seen in the burnup calculation. Since the heterogeneity effects are reflected in the overall reaction rate balance, the differences tend to accumulate as the fuel is burnt.

ENDNOTES

[a]A comprehensive and up-to-date description of the Serpent project can be found at the code website http://montecarlo.vtt.fi.

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