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Recent developments of JAEA's Monte Carlo code MVP for reactor physics applications

Yasunobu Nagaya *, Keisuke Okumura, Takamasa Mori

Japan Atomic Energy Agency, Nuclear Science and Engineering Directorate, Tokai-mura, Naka-gun, Ibaraki 319-1195, Japan

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ABSTRACT

This paper describes the recent development status of a Monte Carlo code MVP developed at Japan Atomic Energy Agency. The basic features and capabilities of MVP are overviewed. In addition, new capabilities useful for reactor analysis are also described.

1. Introduction

A general-purpose continuous-energy Monte Carlo code MVP for neutron and photon transport calculations has been developed since the late 1980s at Japan Atomic Energy Agency (JAEA, formerly JAERI). The MVP code is designed for nuclear reactor applications such as reactor core design/analysis, criticality safety and reactor shielding. The main application field is reactor physics: analysis of critical experiments, calculation of reference solutions for reactor core design, etc. The code has been widely in domestic use since the first release in 1994. (Mori and Nakagawa, 1994) The second version was released in 2005 with new capabilities. (Nagaya, 2005) Modifications and enhancements have been made with advanced Monte Carlo methodology for reactor physics applications. This paper overviews the MVP code and the recent work for the code.

2. General description of MVP

MVP implements basic Monte Carlo capabilities for neutron/photon transport based on the evaluated nuclear data. The following summarizes the features of MVP.

Problems to be solved
MVP can solve eigenvalue and fixed-source problems for neutron, photon and neutron-photon coupled transport. MVP can also solve time-dependent problems.

Geometry representation
MVP employs combinatorial geometry with multiple-lattice capability to describe the calculation geometry. Fig. 1 shows an example of an MVP calculation model for the High Temperature Engineering Test Reactor (HTTR) (Saito, 1994) at JAEA. (The figure was drawn with the CGVIEW code bundled in the MVP source package.) Furthermore, the statistical geometry model is available as described later.

Particle sources
A lot of sampling functions are available for flexible source specification; for example, the `UNIFORM` function samples a value from a uniform distribution in a certain interval and the `SPHERE` function samples from a uniform distribution in a sphere. The data transform functions are also available; for example, the `ROTATE2D` function rotates a two-dimensional vector. Conditional sampling can be done together with the above sampling functions; namely a value is sampled only if specified conditions are satisfied.

Cross sections
MVP uses the specific cross section libraries, which are generated from the evaluated nuclear data (JENDL-4.0, ENDF/B-VII.1, JEFF-3.1.2, etc.) with the LICEM code. (Mori, 2004) The neutron cross sections in the unresolved resonance region are

* Corresponding author.
E-mail address: nagaya.yasunobu@jaea.go.jp (Y. Nagaya).

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described by the probability table method. The neutron cross sections at arbitrary temperatures are available for MVP by just specifying the temperatures in the input data (See Fig. 2).

Variance reduction techniques

The basic variance reduction techniques of Russian roulette kill and splitting are implemented. In addition, importance and weight window based on them are available. Path stretching and source biasing are also available.

Estimators

The track length, collision, point and surface crossing estimators are available. The eigenvalue is estimated with the track length, collision and analog estimators for neutron production and neutron balance methods. The most probable value and its variance are also calculated by the maximum likelihood method with the combination of the estimators. Real variances are estimated with Ueki's method (Ueki, 1997) in eigenvalue calculations.

Symbolic parameters

User-defined variables can be used in the input data. Fig. 2 shows a part of the material specification in an MVP input data. Symbolic parameters are used to define the atomic density of chromium and its abundance in the lines that begin with "%". The atomic density of each chromium isotope is then calculated in the angle brackets.

Vectorized algorithm

MVP employs the stack-driven zone-selection method (Mori, 1992) which is a variant of the standard event-based stack-driven algorithm. A speedup of more than 10 times has been achieved on vector supercomputers for the wide range of applications compared with conventional scalar codes.

Code validation

The code validation has been performed mainly with the ICSBEP (OECD/NEA Nuclear Science Committee, 2008) and IRPhEP (OECD/NEA Nuclear Science Committee, 2012) handbooks. Recent works (Okumura and Nagaya, 2011; Chiba, 2011) have been done with the latest Japanese evaluated nuclear data library JENDL-4.0 (Shibata, 2011).

3. Useful capabilities for reactor analysis

3.1. Statistical geometry model

The design for high-temperature gas-cooled reactors (HTGRs) adopts coated-particle-type fuel. Coated fuel particles (CFPs) are randomly packed in compacts or pebbles and are then loaded in blocks or a reactor core. It is very important to treat the heterogeneity effect of the randomly distributed coated fuel particles in reactor analysis. MVP can take the effect into account directly in

```
& IDMAT(1) /* material ID
  TEMPMT(900.0) /* Kelvin
% CR50=4.345, CR52=83.789, CR53=9.501, CR54=2.365
% CRN=1.399E-2
  CR0500J40(<CRN*CR50/100>)
  CR0520J40(<CRN*CR52/100>)
  CR0530J40(<CRN*CR53/100>)
  CR0540J40(<CRN*CR54/100>)
```

Fig. 2. Example of the usage of symbolic parameters.

full-core modeling of HTGRs with the statistical geometry model (Murata, 1996a).

In this model, the location of spheres (CFPs) is determined probabilistically along a particle flight path. Fig. 3 illustrates the tracking sequence of the statistical geometry model. When a neutron enters a stochastic mixture region, the distance to a sphere is sampled from a nearest neighbor distribution (NND) of the spheres. If a sampled flight path is larger than the distance, the neutron comes into the sphere and takes normal random walk in the sphere. Otherwise, the neutron makes a collision in the matrix region of the mixture region and then repeat sampling from the NND. This sequence is repeated until the neutron exits the mixture region.

The statistical geometry model requires the NND as the input data. MVP can employ not only an accurate NND obtained with the Monte Carlo hard sphere packing simulation code MCRDF (Murata, 1996b) but also the theoretical NND based on the assumption of statistical uniformity.

The statistical geometry model was originally developed for the analysis of the HTTR (See Fig. 1). Later, the statistical geometry model has been extended to treat multiple kinds of spherical geometries for pebble bed reactors (Nagaya, 2004).

3.2. Calculations at arbitrary temperatures

MVP can perform realistic calculations for power reactors at arbitrary temperatures with cross section data generated its master cross section library (Mori, 1999). The master library is produced with the LICEM system and includes temperature-dependent data such as the cross section probability tables and thermal scattering data at several different temperatures. MVP obtains cross sections at a specified temperature by interpolating the temperature-dependent values and Doppler broadening for pointwise cross sections.

3.3. Burnup calculations

The MVP source package includes a burnup calculation module to perform burnup analysis for various types of reactors (Okumura,

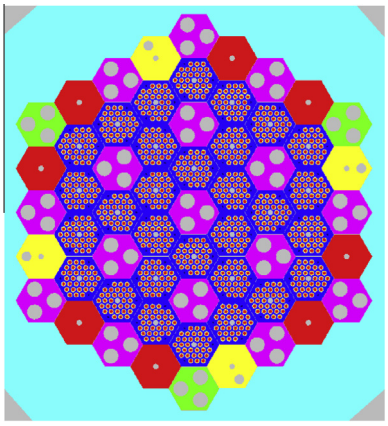

Fig. 1. Cross-sectional view of the HTTR core drawn with CGVIEW.

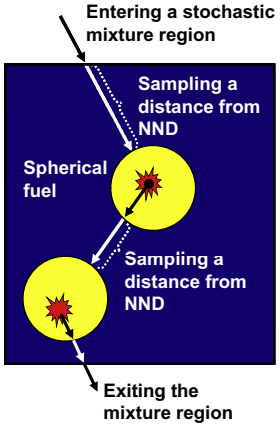

Fig. 3. Schematic view of the statistical geometry model.

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