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Calculation of adjoint-weighted kinetic parameters with the Reactor Monte Carlo code RMC

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ABSTRACT

In this work, the capability of computing adjoint-weighted kinetic parameters, including effective delayed neutron fraction and neutron generation time, was implemented in the Reactor Monte Carlo (RMC) code based on the iterated fission probability (IFP) method. Three algorithms, namely, the Non-Overlapping Blocks (NOB) algorithm, the Multiple Overlapping Blocks (MOB) algorithm and the super-history algorithm, were implemented in RMC to investigate their accuracy, computational efficiency and estimation of variance. The algorithms and capability of computing kinetic parameters in RMC were verified and validated by comparison with MCNP6 as well as experimental results through a set of multi-group problems and continuous-energy problems.

1. Introduction

The Reactor Monte Carlo (RMC) code (Wang et al., 2015a; 2015b), developed by the Department of Engineering Physics, Tsinghua University, is a modern Monte Carlo particle transport code mainly designed for nuclear reactor modeling and simulations. In the previous work, a capability of quasi-static dynamic calculation (Xu and Wang, 2014) was developed in RMC. However, it should be noted that RMC computes the point reactor kinetic parameters by weighting a constant instead of adjoint flux due to the difficulty of obtaining adjoint flux in continuous-energy Monte Carlo transport calculations. Recently, based on the concept of iterated fission probability (IFP), capabilities of computing adjoint-weighted kinetic parameters have been implemented in several Monte Carlo codes including MCNP5 (Kiedrowski and Brown, 2009; Nauchi and Kameyama, 2010), McCARD (Choi and Shim, 2014), TRIPOLI-4 (Truchet et al., 2013), SERPENT2 (Leppänen et al., 2014), etc. In previous works, the IFP method was adopted by RMC to compute eigenvalue sensitivity coefficients with regard to nuclear data (Qiu et al., 2015). Therefore, these experiences should be used to compute adjoint-weighted kinetic parameters and improve the dynamic simulation capability in RMC. In this work, three algorithms based on the IFP method were implemented in RMC to investigate their accuracy and computational efficiency of computing adjoint-weighted kinetic parameters. These algorithms can apply in continuous-energy Monte Carlo simulations as well as multigroup Monte Carlo simulations. Additionally, as several studies have found that the correlation of fission source between cycles may cause underestimation of variance of k_{eff} and fluxes (Gelbard and Prael, 1990), the differences in estimation of variance of k_{eff} and kinetic parameters among different algorithms were compared.

2. Theory

2.1. Adjoint-weighted kinetic parameters

The adjoint-weighted kinetic parameters this work is concerned with are the effective delayed neutron fraction (β_{eff}) and the neutron generation time (Λ_{eff}) which are defined as

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$$\beta_{eff} = \frac{\left\langle \Psi^*, F^D \Psi \right\rangle}{\left\langle \Psi^*, F\Psi \right\rangle}, \tag{1}$$

$$\left\langle \Psi^*, \frac{1}{\nu}\Psi \right\rangle = \int_V \int_0^\infty \int_0^{4\pi} \Psi^*(r,\Omega,E)\frac{1}{\nu}\Psi(r,\Omega',E')d\Omega dE dV, \tag{5}$$

$$\left\langle \Psi^*, F^D \Psi \right\rangle = \int_0^\infty \int_V \int_{4\pi} \Psi^*(r,\Omega,E)\beta\frac{\chi^D(r,E)}{4\pi}\int_0^\infty\int_0^{4\pi}\bar{\nu}\sum_f(r,E')\Psi(r,\Omega',E')d\Omega' dE' d\Omega dV dE, \tag{6}$$

$$\left\langle \Psi^*, F\Psi \right\rangle = \int_0^\infty \int_V \int_{4\pi} \Psi^*(r,\Omega,E)\frac{\chi(r,E)}{4\pi}\int_0^\infty\int_0^{4\pi}\bar{\nu}\sum_f(r,E')\Psi(r,\Omega',E')d\Omega' dE' d\Omega dV dE, \tag{7}$$

$$\Lambda_{eff} = \frac{\left\langle \Psi^*, \frac{1}{\nu}\Psi \right\rangle}{\left\langle \Psi^*, F\Psi \right\rangle}, \tag{2}$$

where

Ψ is the neutron flux $\Psi(r,\Omega,E)$ which is a variable of position r, direction Ω, energy E,
ν is the speed of neutron,

$F^D\Psi$ is the delayed neutron production term which can be expressed as

$$F^D\Psi = \beta\frac{\chi^D(r,E)}{4\pi}\int_0^\infty\int_0^{4\pi}\bar{\nu}\sum_f(r,E')\Psi(r,\Omega',E')d\Omega' dE', \tag{3}$$

$F\Psi$ is the total neutron production term which can be expressed as

$$F\Psi = \frac{\chi(r,E)}{4\pi}\int_0^\infty\int_0^{4\pi}\bar{\nu}\sum_f(r,E')\Psi(r,\Omega',E')d\Omega' dE', \tag{4}$$

χ^D is the delayed neutron emission spectrum,
χ is the total neutron emission spectrum,
β is the total delayed neutron fraction,
Ψ^* is the adjoint flux,
and $\langle\rangle$ is the integration over all space, angle, and energy variables.

where V means the entire space of the system in question.

As it is known, the flux Ψ is the solution of the Boltzmann Equation

$$B\Psi = \frac{1}{k}F\Psi, \tag{8}$$

where

$B\Psi$ is the transport operator which can be expressed as

$$B\Psi = \Omega\cdot\nabla\Psi(r,\Omega,E) + \sum_t(r,E)\Psi(r,\Omega,E) - \int_0^\infty \times \int_{4\pi}\sum_s(r,\Omega'\to\Omega,E'\to E)\Psi(r,\Omega',E')d\Omega' dE', \tag{9}$$

and k is the effective multiplication factor.

And the adjoint flux Ψ^* is the solution of the adjoint Equation

$$B^*\Psi^* = \frac{1}{k}F^*\Psi^*, \tag{10}$$

where

$B^*\Psi^*$ is the adjoint transport operator which can be expressed as

$$B^*\Psi^* = -\Omega\cdot\nabla\Psi^*(r,\Omega,E) + \sum_t(r,E)\Psi^*(r,\Omega,E) - \int_0^\infty\int_{4\pi}\sum_s(r,\Omega\to\Omega',E\to E')\Psi^*(r,\Omega',E')d\Omega' dE', \tag{11}$$

The terms on the right hand side of Eqs. ((1)–(2)) can be expressed in the form of

and $F^*\Psi^*$ is the adjoint neutron production term which can be expressed as

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