Annals of Nuclear Energy 118 (2018) 81–91

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

A direct calculation method for subcritical multiplication factor in Reactor Monte Carlo code RMC

Xiaotong Shang *, Ganglin Yu, Jing Song, Kan Wang

Department of Engineering Physics, Tsinghua University, Beijing 100084, China

ARTICLE INFO

Article history:
Received 16 April 2017
Received in revised form 2 April 2018
Accepted 4 April 2018
Available online 10 April 2018

Keywords:
Subcritical multiplication factor
Modified source iteration method
RMC
ADS

ABSTRACT

The subcritical multiplication factor k_s with the existence of external source neutrons is important for the evaluation of the neutron multiplication M, especially in the accelerator-driven system (ADS) performance assessment. The effective multiplication factor k_{eff} calculated from the traditional source iteration method in Monte Carlo codes can't fully describe the subcritical system, acquiring the spurious neutron flux distribution. A direct Monte Carlo method called modified source iteration method by external source is introduced to calculate the subcritical multiplication factor k_s directly, acquiring the real neutron flux at the same time. Compared with Monte Carlo fixed source calculation, the modified source iteration method has a great advantage in calculating the subcritical multiplication factor k_s and its statistical error directly, especially for systems with complex geometry and a variety of fissile materials where much post-processing work on the tally results from fixed source calculation is needed. Moreover, much calculation time can be saved through this method when obtaining the subcritical multiplication factor k_s with similar precision. This method has already been implemented in Reactor Monte Carlo code RMC including parallel calculation capability and can be effectively applied to the analysis of accelerator driven subcritical system (ADS).

© 2018 Elsevier Ltd. All rights reserved.

1. Introduction

Accelerator driven system (ADS) is being developed by many countries, with the advantage on transmutation of the high-level radioactive waste as well as energy production. The external source neutrons play a vital role in the ADS system. However, the effective multiplication factor k_{eff} takes no account of the position and energy of external source neutrons and is not capable of describing the multiplication effect of external source neutrons. So another parameter that is the subcritical multiplication factor k_s was introduced to express the subcriticality for the ADS system (Kobayashi and Nishihara, 2000; Gandini and Salvatores, 2002; Nishihara et al., 2003).

The subcritical multiplication factor k_s can be calculated by the standard source iteration method in Monte Carlo codes by combination of k_{eff} estimates from each successive source iteration step (Tucek et al., 1997). However, the resulting neutron flux distribution is unrealistic, which may be quite different from the one obtained in experiments. The calculation of subcritical multiplication factor k_s can also be achieved by the fixed source calculation mode in Monte Carlo codes, obtaining the real neutron flux distribution and has been applied to the analysis of the Kyoto University Critical Assembly experiments (Shahbunder et al., 2010a,b,c), but post-processing with the tally results is needed especially for systems with complex geometry and a variety of fissile materials, and the statistic error could not be acquired directly. Based on Reactor Monte Carlo code RMC (Wang et al., 2015; Qiu et al., 2015; She et al., 2014), this paper introduces a direct Monte Carlo method, which is called modified source iteration method by external source, to solve the subcriticality factor k_s iteratively, providing the statistical error directly and the real neutron flux distribution at the same time. Compared with Monte Carlo fixed source calculation, much time can be saved without loss of accuracy.

This paper is organized as follows: Section 2 presents two traditional Monte Carlo methods in details and introduces the modified source iteration method by external source. Section 3 describes the modified source iteration method's implementation and parallelism in Reactor Monte Carlo code RMC (Wang et al., 2015; Qiu et al., 2015; She et al., 2014). Section 4 describes the numerical verification of this method with ADS benchmarks. Finally, the conclusion and future work are presented in Section 5.

* Corresponding author at: LIUQING Building Room 901, Tsinghua University, Beijing 100084, China.
E-mail address: sxt14@mails.tsinghua.edu.cn (X. Shang).

https://doi.org/10.1016/j.anucene.2018.04.006
0306-4549/© 2018 Elsevier Ltd. All rights reserved.

2. Methodology in Monte Carlo codes

The effective multiplication factor k_{eff}, which is defined as:

$$k_{eff} = N^{g+1}/N^g \tag{1}$$

is usually used to describe the properties of the nuclear reactor system. In Formula (1) N^g stands for the number of neutrons in Generation $g + 1$ while N^g is the number of neutrons in Generation g. If k_{eff} is equal to 1.0, the reactor system is critical. If it is above 1.0, the reactor is supercritical and if it is below 1.0, the reactor is called subcritical. k_{eff} is an important quantity in reactor physics to estimate whether the reactor is in critical state. Most Monte Carlo codes use source iteration method (Brown, 2005) to solve Boltzmann neutron transport equation and obtain its eigenvalue (k_{eff}). However, in subcritical systems, k_{eff} does not take into account the effect of external source neutrons, and the real neutron flux distribution could not be obtained through the calculation of k_{eff}.

Thus the subcritical multiplication factor k_s was introduced to describe the ADS system (Kobayashi and Nishihara, 2000; Gandini and Salvatores, 2002; Nishihara et al., 2003) and it is the ratio of all the fission neutrons to the sum of all the fission neutrons and external source neutrons in the system:

$$k_s = F/(F + S_e) \tag{2}$$

F represents the number of fission neutrons while S_e stands for the number of external source neutrons. The subcritical multiplication factor k_s is important for the evaluation of the neutron multiplication M (Shi et al., 2005) of each external source neutron which can be expressed as:

$$M = (F + S_e)/S_e = 1/(1 - k_s) \tag{3}$$

The subcritical multiplication factor k_s can be calculated by the standard source iteration method (Tucek et al., 1997) and fixed source calculation method (Shahbunder et al., 2010a,b,c) in Monte Carlo codes. It can be also calculated effectively and efficiently by the modified source iteration method introduced by this paper.

2.1. Standard source iteration method

Most Monte Carlo codes use source iteration (power iteration) method to solve the Boltzmann neutron transport equation in neutron multiplying systems for obtaining the eigenvalue k_{eff}:

$$[\Omega\cdot\nabla+\Sigma_t(\vec{r},E)]\Psi(\vec{r},E,\Omega) = \iint \Psi(\vec{r},E',\Omega')\Sigma_s(\vec{r},E'\to E,\Omega'\to\Omega)dE'd\Omega' + \frac{1}{k_{eff}}\frac{\chi(E)}{4\pi}\iint \nu\Sigma_f(\vec{r},E')\Psi(\vec{r},E',\Omega')dE'd\Omega' \tag{4}$$

where $\Psi(\vec{r},E,\Omega)$ is the neutron angular flux at position $\vec{r}$ with energy E and the direction of flight Ω, $\Sigma_t(\vec{r},E)$ is the total cross section, $\Sigma_s(\vec{r},E'\to E,\Omega'\to\Omega)$ is the scattering cross section, $\frac{\chi(E)}{4\pi}$ is the angular distribution of fission neutrons, ν is the average number of fission neutrons per fission reaction and $\Sigma_f(\vec{r},E)$ is the fission cross section. It can be written in a simplified form:

$$(L+T)\Psi = S\Psi + \frac{1}{k_{eff}}F\Psi \tag{5}$$

where L is the neutron leakage operator, T is the neutron collision operator, S is the neutron scattering operator, and F is the fission neutron production operator. The process of source iteration method to obtain the effective multiplication factor k_{eff} is as follows (Brown, 2005):

$$\Psi^{(n+1)} = \frac{1}{k^{(n)}}(L+T-S)^{-1}F\Psi^{(n)} \tag{6}$$

$$k^{(n+1)} = k^{(n)}\frac{\langle F\Psi^{(n+1)}\rangle}{\langle F\Psi^{(n)}\rangle} \tag{7}$$

where the diamond brackets denote integration over the system space $(\vec{r},E,\Omega)$. Another multiplication factor k_f was defined in Reference Tucek et al. (1997) for the subcritical system:

$$N_f = \frac{k_0}{\nu_0}\frac{1}{1-k_f} \to k_f = 1 - \frac{k_0}{\nu_0 N_f} \tag{8}$$

where k_0 is defined as:

$$k_0 = n_1/n_0 \tag{9}$$

n_1 is the number of neutrons released in fissions induced by external source neutrons and n_0 is the number of external source neutrons. ν_0 is the average (prompt + delayed) neutron yield in these fissions and N_f is the total number of fissions induced by one external source neutron. Thus k_f expresses the ratio of fissions induced by fission neutrons to the total fissions induced by external source neutrons. In standard source iteration method of Monte Carlo codes, N_f can be calculated from the sum of all the fission neutrons divided by the average neutron yield, or alternatively as:

$$N_f = \frac{k_0}{\nu_0}(1 + k_1 + k_1k_2 + k_1k_2k_3 + \cdots) \tag{10}$$

where

$$k_i = n_{i+1}/n_i, i > 0 \tag{11}$$

k_i is the effective multiplication factor of fission generation i in the standard Monte Carlo source iteration process. This method can be developed to calculate the subcritical multiplication factor k_s. And the neutron multiplication of the source neutrons in subcritical system can be written as:

$$M = 1/(1-k_s) = 1 + k_0 + k_0k_1 + k_0k_1k_2 + k_0k_1k_2k_3 + \cdots \tag{12}$$

Rewrite Formula (3) with Taylor expansion as:

$$M = 1/(1-k_s) = 1 + k_s + k_s^2 + k_s^3 + k_s^4 + \cdots \tag{13}$$

Formulas (12) and (13) have the same expression form. According to Formula (12), the subcriticality factor k_s can be calculated through the standard source iteration method in Monte Carlo codes by setting the initial fission source as same as the external source neutrons. However, real neutron flux distribution of the subcritical system with external source neutrons could not be obtained through the standard source iteration method in Monte Carlo codes, because the standard source iteration method takes no consideration of the external source neutrons.

2.2. Fixed source calculation method

Fixed source calculation mode in Monte Carlo code such as MCNPX (Pelowitz et al., 2011) was selected to make calculations and analysis of accelerator driven system (ADS) (Shahbunder et al., 2010a,b,c). Neutron flux and reaction rates distribution can be obtained through fixed source calculation in Monte Carlo codes and match well with the experiment data (Shahbunder et al., 2010a). The subcriticality factor k_s can also be calculated from the tally results of some parameters in the way described below.

The steady state transport equation with the existence of external source neutrons can be expressed as:

$$[\Omega\cdot\nabla+\Sigma_t(\vec{r},E)]\Psi(\vec{r},E,\Omega) = \iint \Psi(\vec{r},E',\Omega')\Sigma_s(\vec{r},E'\to E,\Omega'\to\Omega)dE'd\Omega' + \frac{\chi(E)}{4\pi}\iint \nu\Sigma_f(\vec{r},E')\Psi(\vec{r},E',\Omega')dE'd\Omega' + S_e(\vec{r},E,\Omega) \tag{14}$$

where $S_e(\vec{r},E,\Omega)$ stands for the external source neutrons and $\Psi(\vec{r},E,\Omega)$ is the neutron angular flux in steady state with external source neutrons. The subcriticality factor k_s and neutron multiplica-

Download English Version:

https://daneshyari.com/en/article/8066953

Download Persian Version:

https://daneshyari.com/article/8066953

Daneshyari.com