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Program system for three-dimensional coupled Monte Carlo-deterministic shielding analysis with application to the accelerator-based IFMIF neutron source

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Abstract

A program system for three-dimensional coupled Monte Carlo-deterministic shielding analysis has been developed to solve problems with complex geometry and bulk shield by integrating the Monte Carlo transport code MCNP, the three-dimensional discrete ordinates code TORT and a coupling interface program. A newly-proposed mapping approach is implemented in the interface program to calculate the angular flux distribution from the scored Monte Carlo particle tracks and generate the boundary source file for the use of TORT. Test calculations were performed with comparison to MCNP solutions. Satisfactory agreements were obtained between the results calculated by these two approaches. The program system has been chosen to treat the complicated shielding problem of the accelerator-based IFMIF neutron source. The successful application demonstrates that coupling scheme with the program system is a useful computational tool for the shielding analysis of complex and large nuclear facilities.

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1. Introduction

Monte Carlo (MC) methods and discrete ordinates methods (i.e. S_N methods) are widely used for nuclear design and shielding analysis. However, their applications are limited due to the

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inherent weakness of both methods. The prominent advantage of MC method is the capability of treating complex geometry. However, it is recognised that the convergence of MC calculation for large-scale system is very time-consuming or even not achievable. For S_N method, it is suited to solve those deep-penetration-relevant problems with relatively simple geometry. Furthermore, most S_N transport codes (e.g. the S_N code packages DOORS [1] and DANTSYS [2]) are still not capable of handling problems with charged particle transport and MC methods are generally employed in this area [3,4]. Therefore, for the shielding analysis of a large and complex nuclear system, in particular when charged particle transport is of high concern, it is desirable to make a coupled MC–S_N calculation, in which MC simulation is performed around the source region with complex geometry, and S_N method is used to deal with the deep-penetration problem in the bulk shield region with relatively simple geometry.

In previous studies, several MC–S_N coupling schemes were developed and used in the radiation-shielding analyses of accelerator systems [5,6]. However, these schemes only enable the coupling of one- or two-dimensional deterministic codes to MC codes. Coupled analysis in full three-dimensional geometry is not supported by these schemes. In this work, a program system has been devised to make full three-dimensional MC–S_N coupling calculations by loosely integrating the Monte Carlo transport code MCNP [7], the three-dimensional S_N code TORT [8] and a coupling interface program with the implementation of a newly proposed mapping approach [9] for the evaluation of angular flux distribution. The test calculations and practical application to the IFMIF neutron source [10] shielding analyses are presented and discussed.

2. Methodology

To perform the coupled MC–S_N shielding analysis, the whole model of the problem needs to be separated into two parts: the source/target region and the bulk shield region. The geometry of the source/target region is generally complex. Consequently, the MC method is the natural choice to simulate the particle transport in this region. For the part of bulk shield region, the geometry is relatively simple. But the deep-penetration problem makes the shielding calculation in this region difficult. The discrete ordinates methods are the best choice to resolve this kind of problem.

The MC method to neutron transport problems is different from the S_N solution. In MC simulation, individual particles are simulated by tracking their life histories and recording events to obtain their average behaviour. In contrast, the S_N method solves the Boltzmann transport equation by using numerical methods. In order to couple these two different methods, a suitable mapping approach [9] has been proposed to hand the particle information from MC simulation over to the discrete ordinates codes. A common surface needs to be specified to link the geometrical models using by these two approaches for the mapping process.

In S_N solution, the scalar flux in a mesh cell (i, j, k) is approximated by the weighted sums of angular fluxes evaluated in particular directions and energy groups

$$\Phi^{S_N}_{i,j,k} = \sum_m \sum_g w_m \psi_{i,j,k,m,g}, \tag{1}$$

where $\psi_{i,j,k,m,g}$ represents the angular flux in the mesh cell (i, j, k) for direction m and energy group g. w_m is the weight associated with the discrete direction m.

In MC transport simulation, a particle weight represents the partial current of this strength across the given surface element. The scalar flux per source particle can be written as [9]

$$\Phi^{\mathrm{MC}}(\Delta A, \Delta E, \Delta \vec{\Omega}) = \sum_n \frac{\mathrm{weight}_n}{N \Delta A |\lambda_n|} \int_{\Delta E} \mathrm{d}E \int_{\Delta A} \mathrm{d}\vec{r} \times \int_{\Delta \vec{\Omega}} \mathrm{d}\vec{\Omega}\, \delta(E, E_n) \delta(\vec{r}, \vec{r}_n) \times \delta(\vec{\Omega}, \vec{\Omega}_n) \tag{2}$$

where weight_n is the MC weight of particle n. E_n, $\vec{r}_n$, Ω_n and λ_n are particle n's energy, position, flight direction and cosine between surface normal and particle trajectory. N is the total number of