POINT-KERNEL METHOD FOR RADIATION FIELDS SIMULATION

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Point-kernel source method for radiation field calculation using Mercure-3 code is considered. Calculation results are shown to be in perfect agreement with those obtained using MCNP code.

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1. INTRODUCTION

Design and maintenance of nuclear cycle facilities mandatory imply implementation of efficient radiation protection. Radiation environment assessment and shielding optimization require a considerable amount of numerical calculations. Nowadays mathematical modelling methods and corresponding software are widely used for such kind of problems. All the software used for radiation fields modelling falls into two main groups. To the first belongs software based on Monte Carlo methods, such as MCNP [1], Geant [2], Penelope, Fluka, etc. Modern implementations of Monte Carlo method provide consistent accounting of radiation transport effects. This leads to perfect accuracy of the calculated values even for complex models. At the same time Monte Carlo calculations require considerable amount of computing time. Computation burden increases rapidly for complex geometries, multiple sources and thick shielding.

Another group contains software implementing analytical methods. The examples of such programs are Microshield, QAD, Mercure. All this programs use point-kernel method for doze calculations. This method is much more less computationally intensive than Monte Carlo method. Series of calculations necessary for shielding optimization could be conducted at reliable time using point-kernel method. But due to macroscopic approach to radiation transport this methods lucks consistency. The main problem point-kernel method encounters is account for scattered radiation which is usually implemented through semi-empirical approximation. Additional "build-up" factor must be introduced as a multiplier to the attenuated doze. Determination of the appropriate build-up factor can be rather complex as it depends upon the energy, the thickness and type of material. Uncertainties in determining build-up factor essentially limit the accuracy of point-kernel method.

In our paper we considered some aspects of point-kernel method and its implementation by Mercure-3 program code [4]. Results of doze calculations for some typical cases are presented. Also point-kernel method verification by the MCNP code [1] is discussed.

2. POINT-SOURCE KERNEL METHOD

Point-kernel method is macroscopic approach used for gamma radiation exposure rate calculations. Within this approach gamma radiation propagation is assumed beam-like. Effects of radiation interaction in matter are described using macroscopic linear attenuation factors. Consistent scattered radiation accounting could not achieved within macroscopic approach. So common practice is to use semi-empirical relations, such as Berger formula, Taylor formula [3], etc.

According to the main idea of point-kernel method radiation source volume is cut up into elementary cells (point kernels). Each point kernel gives contribution to the doze rate at the detecting point for radiation energy E

$$A(\vec{r},\vec{r}',E) = C(E)B(t,E)\frac{\exp(-t(E))}{4\pi(\vec{r}-\vec{r}')^2}, \qquad (1)$$

where $C(E)$ is gamma flux density to doze rate conversion factor, $B(E)$ – build-up factor, t(E) – path length along $\vec{r}-\vec{r}'$ line measured in mean free pathes (mfp)

$$t = \sum_{i=1}^{n} \mu_i(E) X_i, \qquad (2)$$

where i is index of the space region, n – number of regions, $\mu_i(E)$ – linear attenuation factor for i-th region and X_i – length of the section of $\vec{r}-\vec{r}'$ line in i-th space region. Geometry used for point-kernel calculations is shown on the Fig.1.

Relation (1) corresponds to beam-like radiation propagation similar to that used in geometrical optics. Factors μ_i describe radiation attenuation along

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the given path. From the microscopic theory it follows that attenuation coefficients μ_i depend on matter composition and on photon energy.

Build-up factor $B(E)$ accounts for scattered radiation. Mercure-3 implements Taylor formula

$$B(t) = \beta(E)\exp(-\alpha_1(E)t(E)) + \\ + (1-\beta(E))\exp(-\alpha_2(E)t(E)), \quad (3)$$

where α_1, α_2, β – parameters of Taylor formula [3, 4]. This parameters depend on material composition and on radiation energy E. Mercure-3 code contains its own library of build-up factors for commonly used materials.

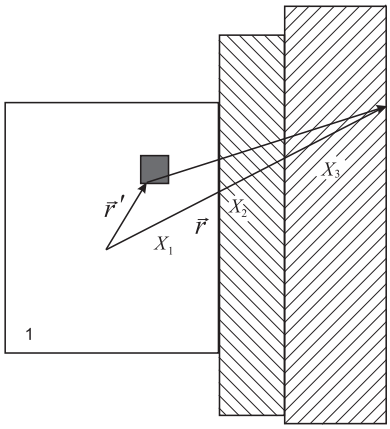

Fig.1. *Geometry of point-source kernel method*

Point-source kernels are assumed to be independent. So total doze rate at detection point is obtained by integration of (1) over the source volume and summation over the energies E_k of radiation spectrum

$$A(\vec{r}) = \sum_{k=1}^{N} \iiint_V d\vec{r}' A(\vec{r}, \vec{r}', E_k). \quad (4)$$

For simple geometries integration in (4) could be carried out analytically. A number of corresponding relations could be found in handbook [3]. In general case this integration is performed numerically.

Spatial integration in (4) presents a problem for detection points located too close to source. This is a common drawback of point-source kernel methods due to integral divergence at such points. One of the possible solutions is to exclude from integration kernel points that fall within some predefined region near the detection point. This procedure provides stabilization of integration procedure in (4) but leads to some loss of accuracy. At the same time for the most cases concerning radiation protection this problem does not arise at all. The matter is that detection points are usually separated from the source by shielding layers and no divergence take place.

3. CALCULATION RESULTS AND DISCUSSION

There exist several software implementations of point-kernel method. Our choice of Mercure-3 code was stipulated by several reasons. The main was the highly verified code, as the program kernel has been developed as early as in 1967 for three dimensional gamma shielding calculations on IBM mainframes and due to sufficient accuracy has been recommended by EURATOM for general use [5]. Another valuable feature of Mercure-3 code is rather powerful and flexible geometry module capable of handling complex structures with multiple sources or inhomogeneous source. Also Mercure-3 contains gamma cross section library for commonly used materials and a set of Taylor coefficients of buildup factor (3). Addition of new materials to the library is possible.

One of the nuclear cycle most important components is radioactive waste management and storage. Effective personnel radiation protection is necessary during spent fuel processing. So we have considered a problem of doze rate calculations for VVER-1000 nuclear reactor fuel assembly.

Technological operations with spent fuel assembly are usually performed after cooling period. Photon spectra of the spend fuel assembly depends on cooling period duration. For our calculations photon spectrum of spent fuel assembly for 3 year cooling period was used (see table).

Fuel assembly photon spectra after 3 year cooling period

Energy, MeV	Activity, ph/s
0.015	$3.27 \cdot 10^{11}$
0.02	$1.04 \cdot 10^{11}$
0.03	$1.96 \cdot 10^{14}$
0.04	$2.09 \cdot 10^{14}$
0.05	$9.27 \cdot 10^{12}$
0.06	$8.35 \cdot 10^{12}$
0.08	$3.73 \cdot 10^{12}$
0.1	$6.28 \cdot 10^{13}$
0.15	$1.60 \cdot 10^{14}$
0.2	$1.51 \cdot 10^{13}$
0.3	$1.05 \cdot 10^{13}$
0.4	$2.57 \cdot 10^{13}$
0.5	$3.33 \cdot 10^{14}$
0.6	$3.88 \cdot 10^{15}$
0.8	$1.43 \cdot 10^{15}$
1.0	$1.61 \cdot 10^{14}$
1.5	$1.59 \cdot 10^{14}$
2.0	$1.15 \cdot 10^{13}$

Real fuel assembly is constituted by a number of construction elements with 312 fuel elements inside. Detailed description of the fuel assembly internal structure is a rather complicated procedure. At the same time the very common considerations show details of internal structure do not influence significantly on doze calculations. Thus for modelling purposes we have neglected assembly internal structure. Fuel assembly geometry was described by cylindrical volumes - one for steel shank and another for bun of fuel elements. Model geometry of fuel assembly used for calculations is shown on the Fig.2. Bun of fuel ele-

ments was replaced by homogenous media composed by uranium dioxide, zirconium and air. Volumetric concentrations were chosen in accordance to assembly technical specification: 0.3186(Zr), 0.205(UO_2), 0.477(air).

Exposure doze rate calculations results are shown on the Fig.3. Exponential doze rate attenuation with distance is observed. Doze rate values are close to those obtained earlier using Microshield code, which is verified commercial product for such calculations.

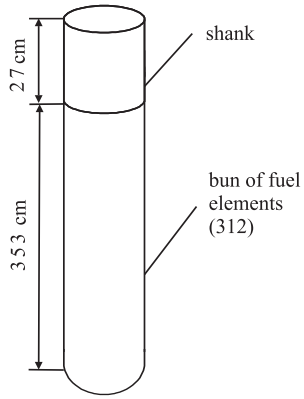

Fig.2. *Fuel source assembly geometry*

Monte Carlo calculations for fuel assemblies and similar objects are impeded by strong photon absorption in UO_2 contained in fuel elements. Another important factor is large geometrical size of the VVER fuel assembly. An intensive calculations are necessary to achieve appropriate accuracy in this case.

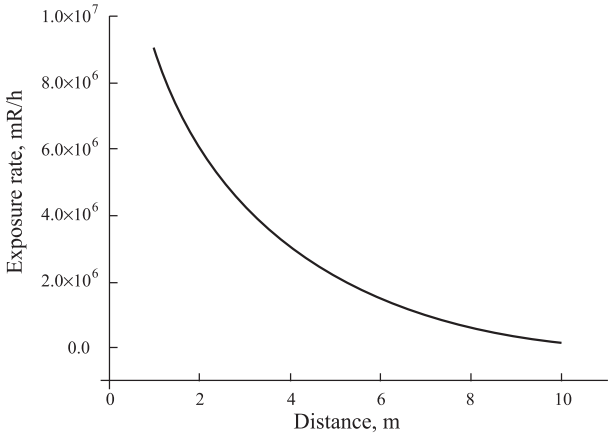

Fig.3. *Exposure doze rate for fuel assembly*

In order to decrease computational burden required for Monte Carlo simulation a small VVER-M assembly was taken as a source for verification calculations. Photon spectrum was the same as in previous case (see table). Problem geometry included VVER-M fuel assembly placed inside concrete shielding (see Fig.4). Exposure doze rate was calculated for points outside the shield to avoid computational instability of point-kernel method.

Calculations for this geometry were carried out both with Mercure-3 code and MCNP (Monte-Carlo method). The results are shown on the Fig.5. Perfect agreement between calculations using Mercure-3 and Monte-Carlo method (MCNP) proves reliability of point-kernel method for radiation fields modelling. At the same time computing speed of Mercure-3 code is considerably higher than those of MCNP (and other Monte-Carlo methods).

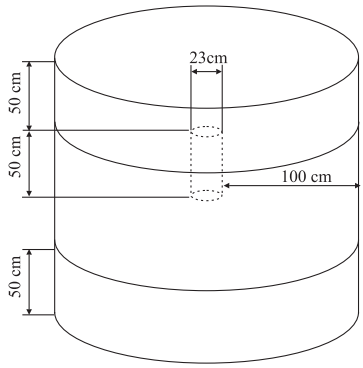

Fig.4. *VVER-M assembly in concrete well*

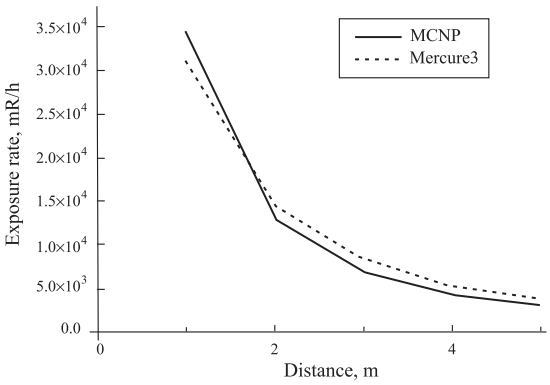

Fig.5. *Exposure doze rate for VVER-M fuel assembly*

So we can consider point-kernel method implemented by Mercure-3 code an efficient instrument for various dose rate calculations, including shielding optimizations. We have shown correct software implementation of this method provides accurate results for practical needs. More over, optimization scheme may include preliminary calculations using point-kernel method with consequent Monte-Carlo simulation.

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МАТЕМАТИЧЕСКОЕ МОДЕЛИРОВАНИЕ РАДИАЦИОННЫХ ПОЛЕЙ МЕТОДОМ ТОЧЕЧНОГО ИСТОЧНИКА

И.М. Прохорец, С.И. Прохорец, М.А. Хажмурадов, Е.В. Рудычев, Д.В. Федорченко

На примере программного кода Mercure-3 рассмотрено применение метода точечного источника для расчета радиационных полей. Показано, что результаты расчетов хорошо согласуются с расчетами по методу Монте-Карло, выполненными при помощи программного кода MCNP.

МАТЕМАТИЧНЕ МОДЕЛЮВАННЯ РАДІАЦІЙНИХ ПОЛІВ МЕТОДОМ ТОЧКОВОГО ДЖЕРЕЛА

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На прикладі програмного коду Mercure-3 розглянуто застосування метода точкового джерела для розрахунку радіаційних полів. Показано, що результати розрахунків добре узгоджуються з розрахунками за методом Монте-Карло, які виконано за допомогою програмного коду MCNP.