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MONTE CARLO APPLICATIONS IN NUCLEAR MEDICINE

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1. THE CONCEPTUAL ROLE OF MONTE CARLO SIMULATIONS

The recent developments in the IT domain (increasing the speed, reducing the costs and the large scale accessibility) led to a rapid growth in the Monte Carlo method use. Considering the development perspectives of the calculus techniques, it is foreseen that this tendency would continue to manifest (for example, the Monte Carlo programs are some of the applications which would mostly benefit from the development of transputers and of parallel computers). This will also significantly increase the number of specialists that use the Monte Carlo method in order to resolve specific problems.

The Monte Carlo simulation of radiation transport through matter (photons, electrons, neutrons) represents a powerful method of resolving an impressive number of problems in medical physics, from modeling the nuclear apparatus used in the research area or industry, to calculating the dosimetric quantities used in predicting the effects of irradiation on the biological matter.

In essence, the Monte Carlo method applied to the radiation transport consists in the simulation, more or less imitative, of the successive interactions that occur when radiation passes through substance. The simulation is possible only by building an abstract model of the physical system. From the multitude of possible trajectories, only a number of representative cases are selected within the model. In consequence, solving a problem with the Monte Carlo method joins together general issues, concerning the simulation of random variables, with specific aspects of the given problem.

3. PRESENTATION OF THE MCNP5 CODE

This paper presents an application of the Monte Carlo method in radiotherapy using the MCNP5 code. MCNP5 is a general-purpose Monte Carlo N–Particle code that can be used for neutron, photon, electron, or coupled neutron/photon/electron transport. Specific areas of application in the medical physics domain include diagnostic, radiation protection, dosimetry, radiation shielding, radiography, detector design and analysis, etc.

The code treats an arbitrary three-dimensional configuration of materials in geometric cells bounded by first- and second-degree surfaces and fourth-degree elliptical tori. It also gives the possibility of using reflective surfaces, called "white" or periodical ones. The complicated geometries can be represented using repetitive structures. All types of interaction are described by using pointwise cross-section data, although group-wise data also are available.

Important standard features that make MCNP very versatile and easy to use include a powerful general source, criticality source, and surface source; both geometry and output tally plotters; a rich collection of variance reduction techniques; a flexible tally structure; and an extensive collection of cross-section data. The tallies have extensive statistical analysis of convergence. Rapid convergence is enabled by a wide variety of variance reduction methods. Energy ranges are 0-20 MeV for neutrons (with data available up to 150 MeV for many nuclides), 1 keV - 1 GeV for electrons, and 1 keV - 100 GeV for photons.

4. ANTHROPOMORPHIC MATHEMATICAL AND VOXEL BASED PHANTOMS

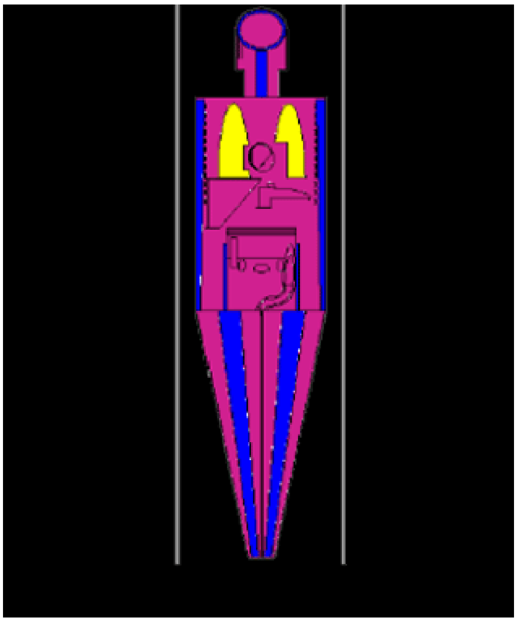
In order to simulate the radiation transport through the human body by using the Monte Carlo codes, the geometry of interest has to be properly described. Human phantoms or models have been used for long in nuclear medicine and radiation protection dosimetry. To date, two types of anatomical models have been developed: equation-based stylized models (analytical phantoms), where organs are delineated by a combination of simple surface equations, and image-based tomographic models (voxel phantoms), that contain large arrays of voxels that are identified in terms of tissue type (soft tissue, hard bone, air, etc) and unique organ identification (lungs, liver, skin, etc).

5. THE APPLICATION

To study the near-field dosimetry of a High Dose Rate (HDR) Ir^{192} source

The analysis of the angular anisotropy of radiation sources (Ir^{192} - known as "high dose rate") used in radiotherapy is of great interest. This gamma sources are placed within or in the near vicinity of the tumor with the purpose of obtaining a high dose in the malign tissue, reducing the exposure of healthy regions. The main objective of the application is to analyse the dose distribution in the vicinity of the source. In this region the dose distribution is no longer isotropic, presenting a highly angular dependency.

Analytical anthropomorphic phantom MIRD12 (ORNL)



ORNL 1996: 35 discrete cells; 3 matterials (soft, bone, lung).
Analytic phantoms consist of regularly shaped continuous objects defined by combinations of simple mathematical geometries.

GSF Voxel Models Representing ICRP Reference Man

3-dimensional view of Godwin (left) and Klara (right).
As a basis for the reference models, the male and female adult voxel models were constructed from whole-body CT.

6. GEOMETRICAL AND MATERIAL DATA OF THE PROBLEM:

The HDR source consists in a cylindrical pellet of pure Ir^{192};

The core encapsulation is stainless steel (ANSI Type 316L);

One end of the capsule is welded to a woven steel cable.

7. THE REQUIREMENT

To compile a table of anisotropy factors (in homogeneous liquid water – an approximation of the human tissue) in the following conditions:

Radii of 1 to 5 cm in 1cm increments (measured along the y-axis from the geometric centre of the source);

Angles of 0 to 180 degrees in 10 degrees increments.

The computations were performed using MCNP5 code (1.0e+7 histories):

The estimator type F5 (flux-point detector) has been used to evaluate the gamma flux in the points of interest;

The dose rates have been estimated by multiplying the flux values with the ANSI type flux to dose conversion factors.

Anisotropy factors (MCNP5 results)

Angle (grades)	R=1cm	R=2cm	R=3cm	R=4cm	R=5cm
0	8.34878E-01	7.96882E-01	7.31415E-01	6.89176E-01	6.32072E-01
10	9.16281E-01	9.72438E-01	9.70273E-01	9.68734E-01	9.64572E-01
20	9.23065E-01	9.70599E-01	9.80102E-01	1.00825E+00	9.74303E-01
30	9.23856E-01	1.33354E+00	9.76118E-01	9.78379E-01	9.67130E-01
40	9.29498E-01	9.57524E-01	9.70995E-01	9.66226E-01	9.58276E-01
50	9.53112E-01	9.70327E-01	9.75945E-01	9.72001E-01	9.56853E-01
60	9.75632E-01	9.86027E-01	9.90928E-01	9.86510E-01	9.73319E-01
70	9.88699E-01	9.93838E-01	9.94668E-01	9.93323E-01	9.79837E-01
80	9.96737E-01	1.00018E+00	9.98147E-01	9.93945E-01	9.87502E-01
90	1.00000E+00	1.00000E+00	1.00000E+00	1.00000E+00	1.00000E+00
100	9.96558E-01	9.97244E-01	9.97392E-01	9.95869E-01	9.82214E-01
110	9.87687E-01	9.92555E-01	9.94298E-01	9.88975E-01	9.84071E-01
120	9.72645E-01	9.83821E-01	9.87700E-01	9.83772E-01	9.70585E-01
130	9.51413E-01	9.68712E-01	9.79960E-01	9.70954E-01	9.57349E-01
140	9.28862E-01	9.63024E-01	9.68962E-01	9.73910E-01	9.60227E-01
150	9.32367E-01	9.73384E-01	9.83799E-01	9.85688E-01	9.73955E-01
160	9.38451E-01	9.85707E-01	9.97152E-01	9.99242E-01	9.89174E-01
170	9.46050E-01	9.93271E-01	1.00741E+00	1.00494E+00	1.00132E+00
180	9.51094E-01	1.00055E+00	1.01043E+00	1.01433E+00	1.00219E+00

8. CONCLUSIONS

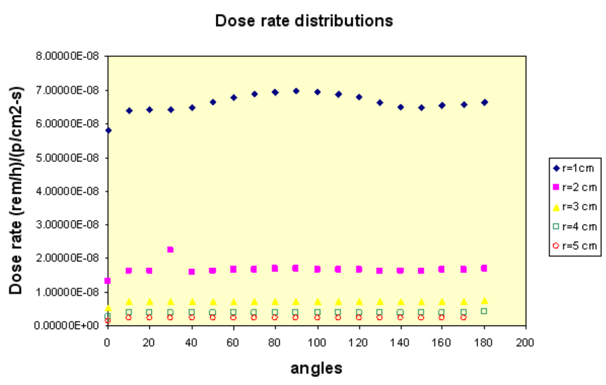
The Monte Carlo codes (referring mainly to the MCNP5 code):

Provide a full and detailed radiation transport simulation for neutral and charged particles;

Are used to investigate some of the most difficult problems associated with radiation therapy;

Offer the chance to apply a single code to all anatomical geometries, all modalities and all equipment specifications;

Are used to investigate the physics of various processes, for analyzing new treatment protocols, for investigating dosimetry problems, and general quality assurance.



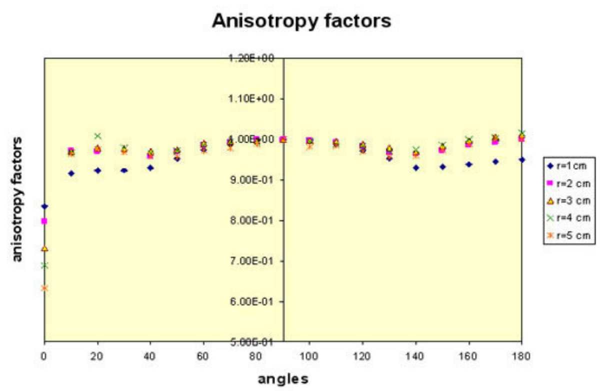
The photons were generated in the Iridium volume; the initial distances were generated randomly and uniformly in the solid angle.

For evaluating the relative dose produced by the source in the specific points of interest, the punctual flux estimator F5 was used multiplied with the conversion flux-dose factors.

The figure shows the dose anisotropy at various distances from the source (radii of 1 to 5 cm); the anisotropy decreasing with radius increasing.

For every radial distance, the anisotropy factors are calculated based on the dose rates obtained with the MCNP5 and divided to the calculated value for the 90^0 angle.

For every value of the radial distance and for the considered angle we evaluated the coordinates for the points of interest where the gamma flux calculation is made (F5 estimator) and the dose rate. The MCNP5 code was run for a number of 10^7 histories (photons) generated initially from the Iridium source.



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