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Tally and geometry definition influence on the computing time in Radiotherapy Treatment Planning with MCNP Monte Carlo code.

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***Abstract*— The present work has simulated the photon and electron transport in a *Theratron 780®* (MDS Nordion) ^{60}Co radiotherapy unit, using the Monte Carlo transport code, MCNP (Monte Carlo N-Particle), version 5. In order to become computationally more efficient in view of taking part in the practical field of radiotherapy treatment planning, this work is focused mainly on the analysis of dose results and on the required computing time of different tallies applied in the model to speed up calculations.**

I. Introduction

ALTHOUGH the Monte Carlo radiation transport technique is an accurate way of assessing radiotherapy dose treatments, its required processing time is considered prohibitive [1], [2]. This is the fact that difficults the introduction of Monte Carlo simulations in clinical practice to determine dose distributions in patients.

In order to make the calculation feasible in a reasonable amount of time, not only the parallelization of the MCNP5 code in an SGI Altix 3700, using the MPI parallel protocol with 16 processors has been applied [3]. Other acceleration techniques have been also utilized in this work. They consist of testing different tally cards, in order to study dose results variations in a heterogeneous water phantom and to conclude which of them has the best ratio of results accuracy versus computational speed.

As a novelty in this kind of studies, the FMESH tally has been applied to our simulation, proving that this new MCNP feature is a powerful tool for dose calculation in operational radiation therapies simulations.

II. Methodology

In this section, the experimental procedure and the Monte Carlo simulation model are described.

A. Experimental procedure

The experimental part of this research project was performed at the *Hospital Provincial de Castellón*, Spain, which has provided the access to a *Theratron 780®* and its facilities.

The experimental data were obtained in a 3D motorized water phantom ("RFA-300 Water Phantom") [4]. This cube-shaped water tank (with a scanning volume of 50 cm x 50 cm x 50 cm) consists of a solid assembly with steel reinforced toothed belts for exact positioning of a *Scanditronix-Wellhofer RK* thimble chamber with an active volume of 0.12 cm^3 [5]. This device is placed on a radiation resistant detector holder and is able to measure pulsed photon and electron radiation (continuous) from the ^{60}Co radiation therapy unit at 80 cm from the source to obtain the dose rates.

Most of the deterministic algorithms employed by current commercial computerized treatment planning systems are not satisfactory enough to account for the heterogeneities and dose variations. Since these commercial systems may significantly alter the dose in discontinuous mediums, we have studied the accuracy of Monte Carlo techniques in a heterogeneous medium.

For this, a 30 cm x 10 cm x 8 cm extruded polystyrene brick (97% air and 3% polystyrene) has been inserted into the tank, which was otherwise filled with water. This material, with a density of 0.0311 g/cm^3, is used as a lung tissue equivalent material in the analysis, due to its similar density and behavior concerning radiation absorption. The heterogeneity in the cube-shaped water-filled phantom was placed at 10 cm depth, measured in the direction of the entrance beam.

The Cobalt source activity corresponding to the measurement date was $7.261 \cdot 10^{13}$ Bq.

B. Simulation

The Monte Carlo code MCNP5 has been used in this work to simulate radiation transport through the Cobalt Therapy Unit [6]. The geometric drawings provided by the manufacturer *Nordion* were used as templates in designing the geometry for the input file of each component of the unit. The cylindrical geometry of the ^{60}Co source and its entire housing, the collimator system, jaws and water phantom as well as material composition of each component has been described in detail to perform the input file.

Figure 1 shows the generated model of the *Theratron* head and phantom used in the simulation.

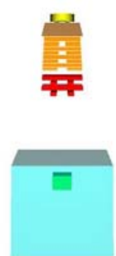

Fig. 1.- *Theratron* head and phantom. MCNP model.

Concerning radiation transport in this work, a detailed physics treatment for a coupled photon and electron mode simulation has been considered, taking into account the following physical processes: photoelectric effect with fluorescence production, Compton and Thomson scattering, and pair production in the energy range between 0.001 and 2.6 MeV.

All simulations have been initiated from a surface source input generated in previous studies [1] tracking one hundred billion particles.

To obtain 3D dose distribution, a voxel water tank model has been developed. This model has been split in numerous cells, considering a 0.5 cm x 0.5 cm x 0.5 cm *voxel* size, except the first 2 cm in the direction of the entrance beam in which the depth of *voxels* has been reduced to 0.1 cm with the purpose of better reproducing the build-up region. The total number of *voxels* was 3648.

In Figure 2, it can be seen the exact location of the scoring points in a horizontal plane of the water tank.

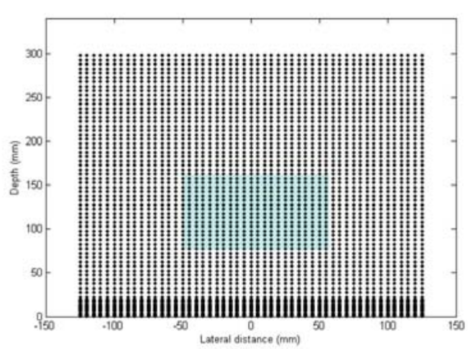

Fig. 2.- Location of scoring points in the phantom.

The registration of particles in all the *voxels* from surface to the bottom phantom, allows us to see how the energy deposition varies inside the water phantom and to develop the 3D dose distribution map or to obtain depth dose and dose profile curves.

Unfortunately, there is no standard tally in MCNP that calculates the dose in such small volumes in a perfectly accurate way. Despite this fact, we have run the MCNP simulation with different tally methods, with the aim of comparing the accuracy results and computing time. F8, F6 and F4 have been selected for dose calculation.

The first tally used in the simulation is the energy deposition in units of MeV in each *voxel* and subsequently converted to relative dose. In MCNP, this tally is the pulse-height distribution registered in a detector, modified to energy units (designated in MCNP as *F8:E card) and provides the energy deposition by both photons and electrons in each cell.

On the other hand, F6 outputs the energy deposition averaged over each cell in MeV/gram.

Moreover, advantages of the new feature in Version 5 of MCNP, called FMESH, "Superimposed Mesh Tally" have been used in this simulation, where a complex *voxelization* is necessary in terms of memory. MCNP has this special tally type, which allows the user to tally particles on an independent mesh of the current problem geometry, but only track-length (type 4) mesh tally has been implemented. The FMESH card allows to define a mesh tally superimposed over the problem geometry and to obtain the track length estimation of the particle flux. By default, this tally is averaged over a mesh cell, in units of particles/cm^2. Therefore, the application of this feature in the simulation has been made in combination with the dose energy (DE) and dose function (DF) cards to transform the output results into Grays.

FMESH, in comparison with traditional tallies, offers three main advantages: First, the easiness of the input description. In order to characterize the grid, it is only needed the origin and location of the mesh, and the number of fine meshes within a corresponding coarse mesh in each direction. The second advantage is the usefulness of results, which are displayed in a separate output file and can be formatted in a series of matrices that are easy to analyze. Finally, the exceptional short calculation time achieved, allowing Monte Carlo methods to become more practical to routine treatment planning dose calculations.

One of the main reasons for speeding up the calculation in such a considerably way by this tool is the fact that, with the use of this feature, cross sections do not need to be re-evaluated each time a particle crosses a surface.

III. Results

Experimental measurements have been used to validate the different Monte Carlo simulations. Depth dose curves obtained at the *Theratron* facilities using a 10 cm x 10 cm field size and including the heterogeneity slab inside the water phantom have been compared with the calculated results obtained with each tally.

Dose curves have been generated using three photon tally cards, F8, F6 and FMESH4 tally, with the aim of studying the variations on computing time and the accuracy in relative dose calculation.

*F8 is the first studied tally. In this case, dose parameters were calculated from the recorded deposited energy (MeV), dividing this value by its corresponding cell mass. Figure 3 compares *F8 tally results with the experimental data in a relative depth dose curve. It can be stated the high level of accuracy obtained with this model. This figure presents also the error bar in percentage associated to tally results. An average uncertainty of 2σ is shown in the following figures. As can be seen, all values respect the permissible rank of error. The calculation time for this simulation was 4630 min.

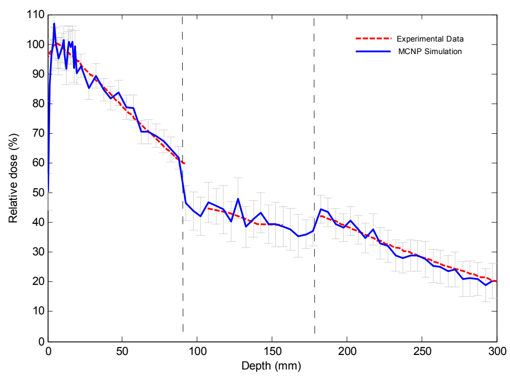

Fig 3. Comparison of the calculated and experimental relative depth dose using *F8 tally. Mode PE. 10x10 cm^2 field size.

Since the F6 tally type is not available for electrons, and the F4 tally can not be used to score photons and electrons simultaneously, in order to compare computing time and results between the three tallies, we have also calculated dose depth curves with a *F8 tally in an 'only photon' mode.

The possibility of avoiding the electron tracking reduces the computing time in an important way, see Table I. At first sight it appears to be an interesting possibility of speeding up the simulation, but as Figure 4 shows, avoiding the electrons trajectory has an important effect on dosimetric results, since the values in the heterogeneity region are not reliable and neither those of the build up region, where the MCNP simulation does not reproduce the real dose behavior precisely.

Dose is explicitly calculated in the case involving the F6 tally, because this card provides the energy deposition averaged over a cell in MeV/g. However this tally does not allow considering the electron contribution to the dose.

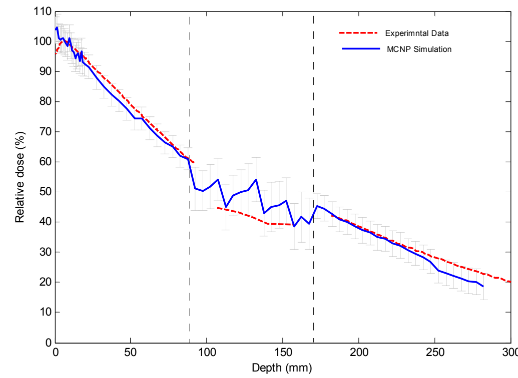

Fig 4. .Comparison of the calculated and experimental relative depth dose using *F8 tally. Mode P. 10x10 cm^2 field size.

It can be noted in figure 5 how the F6 tally results does not give the heterogeneity dose absorption with a similar accuracy level than that achieved with the *F8 tally, considering the contribution of both photon and electrons (Figure 3).

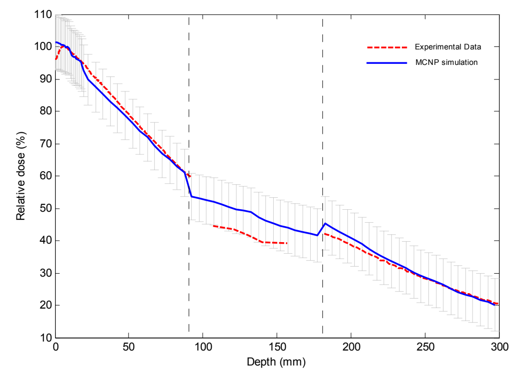

Fig 5. Comparison of the calculated and experimental relative depth dose using F6 tally. Mode P. 10x10 cm^2 field size.

A different approach has been used to calculate the dose using the F4 tally. The F4 tally is a track-length estimator that can be used to determine the average particle flux in a volume. Although this tally provides, by default, flux averaged over a cell, we have modified it adding a dose-energy/dose-function (DE/DF) card which gives results in terms of MeV/cm^2. This is an easy way to convert the average particle flux in a volume into absorbed dose. When a photon produces a F4 score for the tally, this value is multiplied by the value of the dose response function at the energy of the photon, and gives the absorbed dose results.

Conversion coefficients from flux to dose in liquid water and extruded polystyrene material have been taken from NIST tables, where values of mass attenuation coefficient as a function of photon energy for compounds and Stopping Power tables for electrons in the desired energy range are available.

Since the F4 tally cannot be used for coupled photon and electron radiation transport, a tally for electrons (Fmesh4:e) and other for photons (Fmesh4:p) has been necessary in the simulation to take in conjunction both dose contributions.

The F6 tally is a derivative of the F4 tally in MCNP, so F4 tally results are approximately agree with the F6 ones, while differences between these tallies with *F8 results are larger. This is reasonable since the *F8 tally is based on a different physics that the F4 tally.

Figure 6 compares the depth dose curve obtained in a heterogeneous water phantom using the FMESH Independent Superimposed Mesh Tally versus the experimental data.

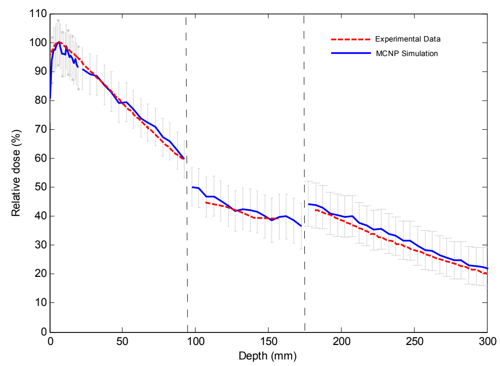

Fig 6. Comparison of the calculated and experimental relative depth dose using FMESH tally. Mode PE. 10x10 cm^2 field size.

Regarding to calculation speed, Table 1 shows the computing time in each case. This table displays the processing speed of various simulations using voxelized models or the FMESH card. Some of the cases involve a 'coupled photon and electron' transport simulation and others an 'only photon' one. Obviously, 'only photon' transport is faster in comparison to 'photon and electron' one.

TABLE I
COMPUTING TIME COMPARISON. DIFFERENT TALLY TYPES. VOLUMETRIC SOURCE . 10X10 CM^2

TALLY TYPE			CPU time (16 CPU's)
voxelization	*F8	Mode PE	4630 min
		Mode P	2835 min
	F6	Mode P	2760 min
	F4	Mode PE	3161 min
FMESH	F4	Mode PE	486 min

On the other hand, the noteworthy aspect pointed by this table is the amazing velocity achieved by the FMESH tool, with the results inside the permissible error bar rank in all cases, while having a calculation much faster than the rest of the cases.

The choice of the most appropriate tally depends on the objective of the problem to be solved. Using a heterogeneous water tank, it was found that the *F8 tally is the more accurate way of calculating dose variations in discontinuities, while the F4 and F6 photon tallies agreed reasonably only in the water region. Therefore, if it is necessary to study in depth heterogeneities dose variation, then the *F8 tally should be used.

From this table it can be stated that the FMESH tally card is formidably faster than any tally applied in a voxelized model, concluding that this new feature available in MCNP5 is an extremely powerful tool to be use in radiotherapy treatment planning systems.

IV. CONCLUSION

The use of a large number of scoring *voxels* dramatically slows down the MCNP calculation. Therefore, this Monte Carlo based treatment planning system makes use of not only parallel computing techniques, but also different scoring methodologies to achieve a reasonable computation time.

After comparing the MCNP calculated dose results using the *F8, F6 and FMESH4 tallies, to those obtained from the experimental measurements, we can state that although the first one has the longer computation time, its accuracy, specially in the heterogeneity region is essential in dose planning systems where a high level of precision is required.

This study has also demonstrated the advantage of the MCNP5 tool called FMESH tally. This feature makes our simulation lighter to obtain the 3D dose mapping in many operational radiation treatment conditions. With a complex *voxelization* of the water phantom configuration, MCNP FMESH tally can be used to predict doses that may be too computationally expensive to obtain without this tool, considering that in heterogeneous regions, dose estimation is as precise as the *F8 results, respecting the 2σ error bar.

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