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SHIELDING OF RADIATION FIELDS GENERATED BY ^{252}Cf IN A CONCRETE MAZE PART II - SIMULATION

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Abstract

A streaming experiment performed in a concrete maze of shape and size typical of a radiotherapy room was simulated with the Monte Carlo program FLUKA. The purpose of the calculation was to test the performance of the code in the low energy neutron range, and at the same time to provide additional information which could help in optimizing shielding of medical facilities.

Instrument responses were calculated at different maze locations for several experimental configurations and were compared with measurements. In addition, neutron and gamma fluence, ambient dose equivalent and effective dose were calculated at the same positions. Both sources used in the experiment, namely a bare ^{252}Cf source and one shielded by a tungsten shell 5 cm thick, were considered in the simulation.

1. INTRODUCTION

Neutron and photon attenuation in mazes has been the object of extensive experiments and calculations, and various empirical recipes and formulas have been proposed to help in the design of access ways through the shielding of accelerators and of other sources of radiation. A comprehensive review can be found in [1, 2], and more recent studies are mentioned in [3].

However, as pointed out in [4], most of the results obtained in those studies refer to typical conditions of high-energy accelerators, and are only partially applicable to the design of radiation therapy accelerator rooms. According to [4], maze streaming in medical accelerator rooms differs from that of most other accelerators in three main respects, namely lower neutron energy, wider angular distribution and maze geometry (length to width ratio). A further difference is the use of shielded doors to compensate for the insufficient attenuation of single-legged labyrinths.

In order to better understand radiation streaming at medical irradiation facilities, a mock-up radiotherapy room and its access maze were built by means of concrete blocks by McCall and collaborators. The radiation field produced by a ^{252}Cf source at different maze locations and in different configurations was exhaustively investigated, as described in a companion paper presented at this Conference [5] to which we refer for details.

The experiment was very comprehensive and its accuracy was more than adequate for practical purposes, but it cannot be considered a real benchmark for a transport code. There are uncertainties concerning the actual composition and density of the concrete (and especially of the macadamized floor), and also the response of the instruments used is only known with a certain degree of approximation. However, it may be useful to compare the measured data with the predictions of a simulation code: if a reasonable agreement is found (within the mentioned uncertainties), that code can be safely used to get complementary information about conditions, geometries and radiation quantities which were not addressed in the experiment.

Therefore, several of the experimental conditions studied in [5] were simulated with the Monte Carlo program FLUKA [6]. In addition to instrument response, neutron and gamma fluence and ambient dose equivalent were calculated at different maze locations. The two sources used in the experiment were considered in the simulation, namely a bare ^{252}Cf source and one shielded by a tungsten shell 5.08 cm thick. The shielding effect of doors of borated polyethylene (BPE) of different thickness was calculated for both sources at two door positions.

2. THE FLUKA CODE

FLUKA was originally a high-energy hadron transport code, but in the last 10 years it has been extended to handle more than 40 different particles and to cover energies spanning at least 15 orders of magnitude, thermal neutrons included. The good performance of the code in the low-energy neutron range has already been demonstrated implicitly in a series of experiments with high-energy hadron beams [7], where the response of various neutron detectors has been predicted with good accuracy. However, there have been so far only few opportunities to use FLUKA as a "pure" low-energy neutron transport code, and the SLAC experiment can serve as a good basis for comparison with actual data.

In FLUKA, low-energy neutron transport below 20 MeV is based on a multi-group approach. Capture gammas are generated according to the appropriate group probabilities, but are transported using continuous energy-dependent cross sections as any other photon of more than 1 keV. The FLUKA neutron cross section library, in a modified ANISN format which allows for kerma factors and partial cross sections, has 72 neutron groups and 22 gamma groups and is based on a P5 Legendre angular expansion. It has been expressly prepared for FLUKA by ENEA (Bologna) [8], mainly from ENDF/B-VI, JEF-2.2 and JENDL-3.2 evaluated data files, and contains more than 100 elements and isotopes at different temperatures.

3. THE SIMULATIONS

A user code was written to sample neutrons from an isotropic source, with a spectrum as reported in [9], interpolated to coincide with the 72 energy groups of FLUKA (see Fig. 1). The inner geometry of the room and of the maze was described by means of 8 regions (cells), filled with air. Since no information was available on the actual composition of the floor, the latter was assumed to be made of concrete, 91.5 cm thick, identically to the ceiling and to the lateral walls. For biasing purposes, walls ceiling and floor were subdivided into 3 layers, each 10 cm thick, plus a fourth outer layer 61.5 cm thick, for a total of 104 concrete regions. Twelve more regions were needed to describe the source, the shielding door and a number of scoring detectors.

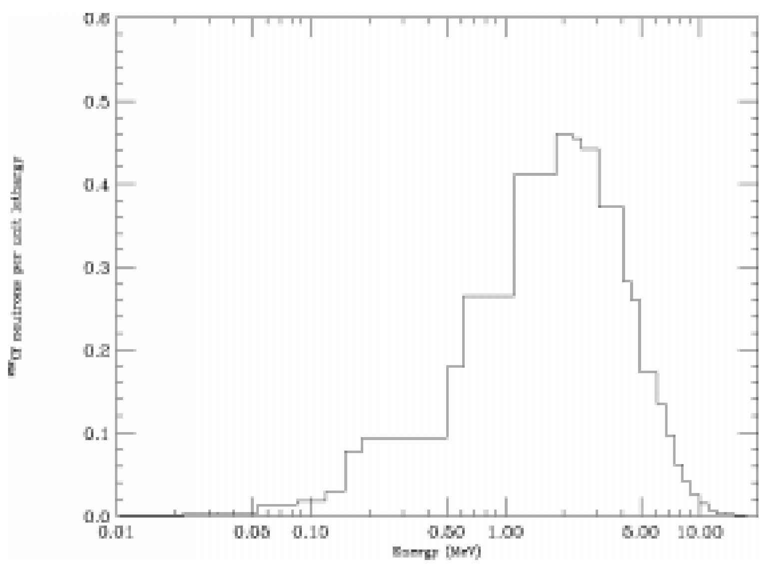

Fig. 1. ^{252}Cf spectrum in the 72-group energy structure of FLUKA, normalized to 1 source neutron

The concrete density and composition were unknown, but following a standard SLAC practice they were assumed to be respectively 2.436 g/cm^3 and the following percentage in weight: 0.945 H, 50.0 O, 3.20 Al, 38.6 Si, 7.27 Ca. Some tests were made using a density of 2.35 g/cm^3 and a different composition (Portland concrete as reported in [10]). No significant differences were found in the results, but a composition effect cannot be excluded in general, since neutron scattering may be affected by hydrogen content, which was about the same in both cases.

The shielding door was assumed to be made of polyethylene of density 0.934 g/cm^3 , with natural boron content 5% in weight. The first calculations were made using cross sections for gaseous hydrogen, and were then repeated with more appropriate data for hydrogen bound in polyethylene: however, no significant differences were found in the calculated quantities. The tungsten shell surrounding the neutron source was assumed to be pure tungsten, of density 19.3 g/cm^3 , although the actual shell, filled with W powder, might have been of lower density: this uncertainty adds to those listed above.

Particle transport was biased in several ways: splitting and Russian Roulette were implemented at boundary crossing, based on pre-defined region importances, and a weight window was also applied. The probability of non-absorption was forced to be 0.95 in superficial concrete layers and 0.85 in all deeper layers.

The quantities scored were neutron and gamma track-length in vertical air slabs 6 cm thick located at different locations in the maze, corresponding to actual measuring points. To avoid wall effects, a distance of 30 cm was left between such scoring slabs and floor, ceiling and lateral walls. The fluence spectra obtained were folded with energy-dependent responses and fluence-to-dose equivalent conversion factors in order to estimate quantities which could be compared to the measured ones. The energy response of the particular Andersson-Braun instrument used in the experiment is not known with great accuracy: therefore it has been necessary to use the response curve of the SNOOPY, an instrument with similar characteristics, but whose response is better known (see [10], p. 55). The numerical values were provided to us by PTB [11].

Gamma doses were calculated as the sum of dose from gammas produced by neutron capture in the concrete walls (or in the BPE door) and dose of photons emitted by the source itself. The primary photon spectrum of ^{252}Cf was taken from [12], as reported by [13]. The normalization factor, 1.3 photons of energy between 1 and 6 MeV per emitted neutron, was obtained from [9]. Primary photons of less than 800 keV were not considered.

4. RESULTS

Calculated Andersson-Braun responses are compared in Fig 2 to the values measured at 6 different maze locations, for both the bare and the W-shielded ^{252}Cf source, and for a BPE door of thickness 0 to 4 inch (10.16 cm) located at 5 m from the maze entrance. Similar data, but for a door located at 3 m from the maze entrance, are shown in Fig. 3.

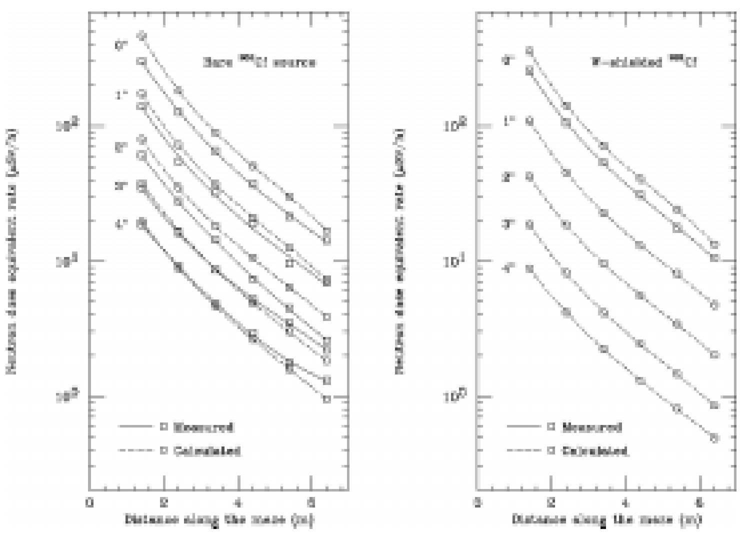

Fig. 2. Calculated and measured Andersson-Braun response at different maze locations, with and without a BPE shielding door of 1- to 4-inch thickness. Left: bare ^{252}Cf source, right: source covered with a W shell 5.08 cm thick. Door position: 5 m from maze entrance

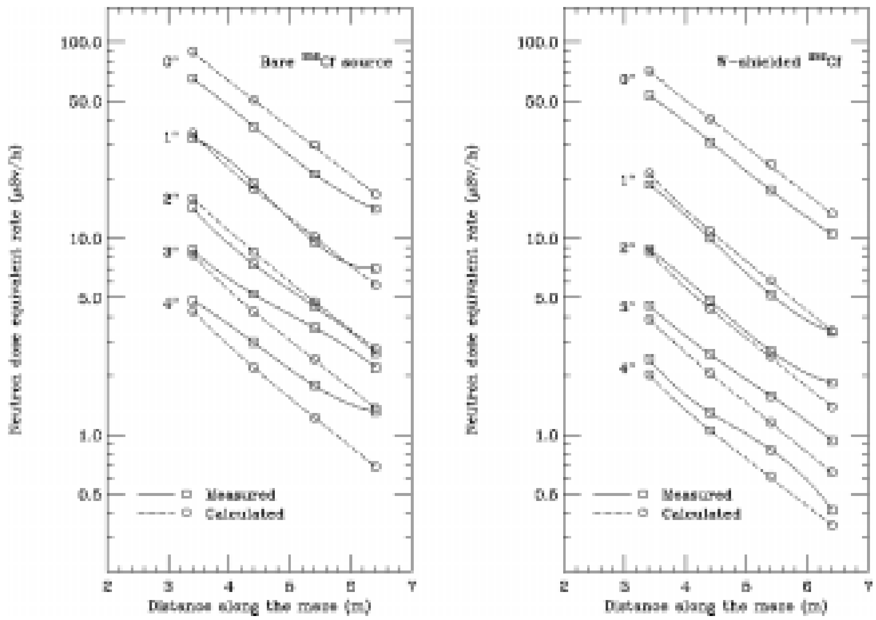

Fig. 3. As in Fig. 2, but with door position 3 m from maze entrance

Calculated gamma doses, total and from scattered primary ^{252}Cf photons only, are shown and compared to measured data in Fig. 4.

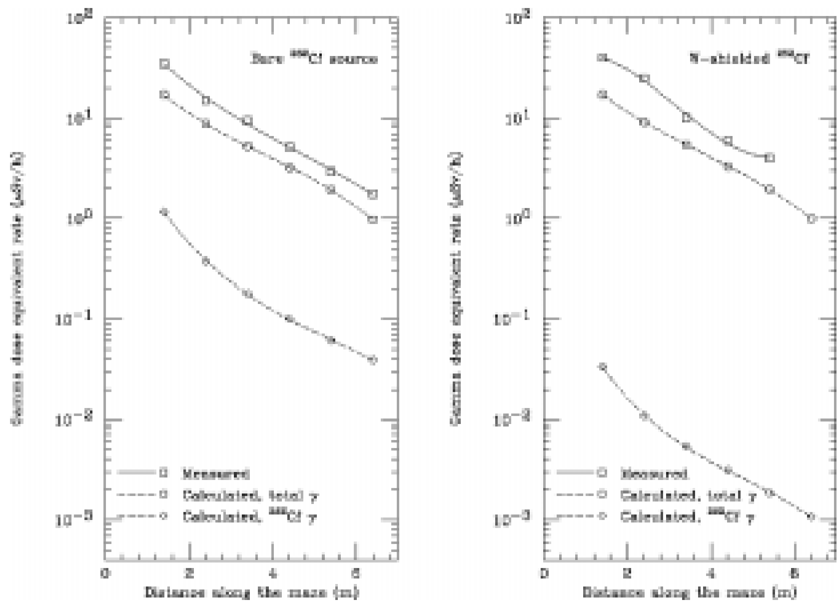

Fig. 4. Calculated and measuresd gamma dose at different locations in the maze, for a bare ^{252}Cf neutron source (left) and for one covered with a tungsten shell 5.08 cm thick (right). The upper calculated curves refer to the sum of capture gammas and gammas emitted by the ^{252}Cf source, the lower curves to the latter only.

In addition to quantities which can be directly compared with measurements, further useful information can be obtained from the simulations. As an example, in Fig. 5 neutron spectra, calculated at 4 positions in the maze without any door, are shown for the unshielded and for the W-shielded source.

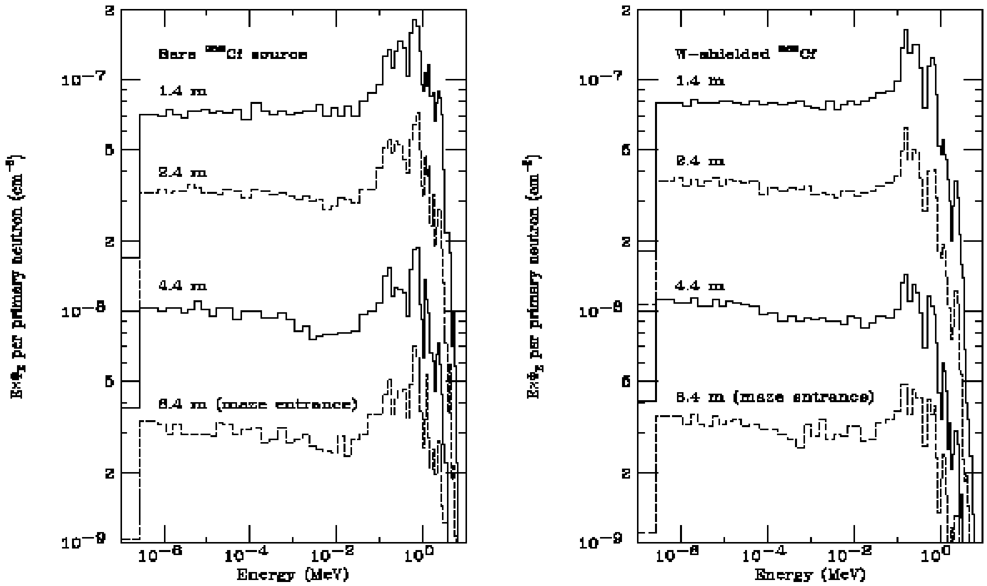

Fig. 5. Neutron lethargy spectra at different maze locations. Left: bare source, right: W-shielded source.

It is also interesting to compare attenuation data with the prediction of commonly used empirical formulae. In Fig. 6 and 7, the relative attenuation of several quantities along the maze is compared with the "Universal transmission curve" which has been used successfully at CERN for many years [1]. The curve is used to predict attenuation of neutron dose in the second leg of a labyrinth, as a function of a dimensionless parameter $d/A^{1/2}$, where d is the distance from the center of the corner at the beginning of the maze, and A is the area of the maze vertical cross section. The middle curve represents the expected attenuation to be used in maze design, while the other two curves are to be considered as confidence limits. The analytical dependence proposed by Dinter et al.[3] is also shown.

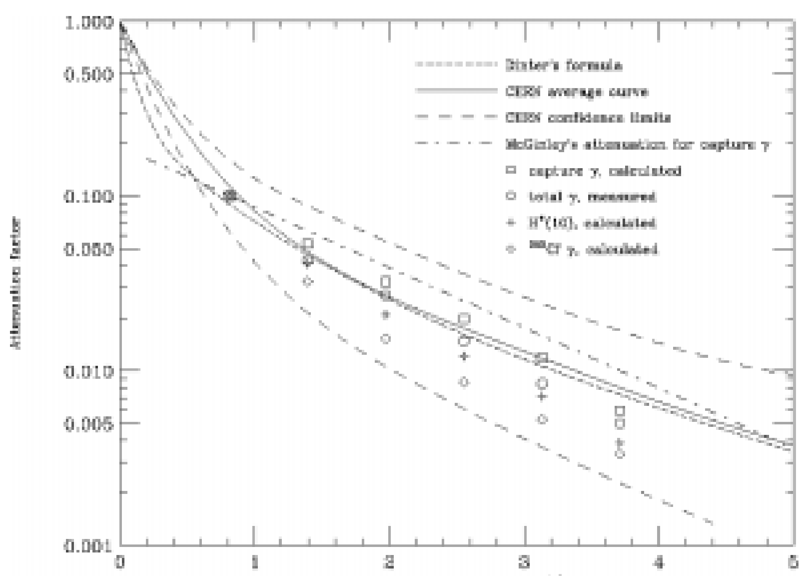

Fig. 6. Relative attenuation of neutron and gamma dose equivalent for a bare ^{252}Cf source. The meaning of the curves is explained in the text.

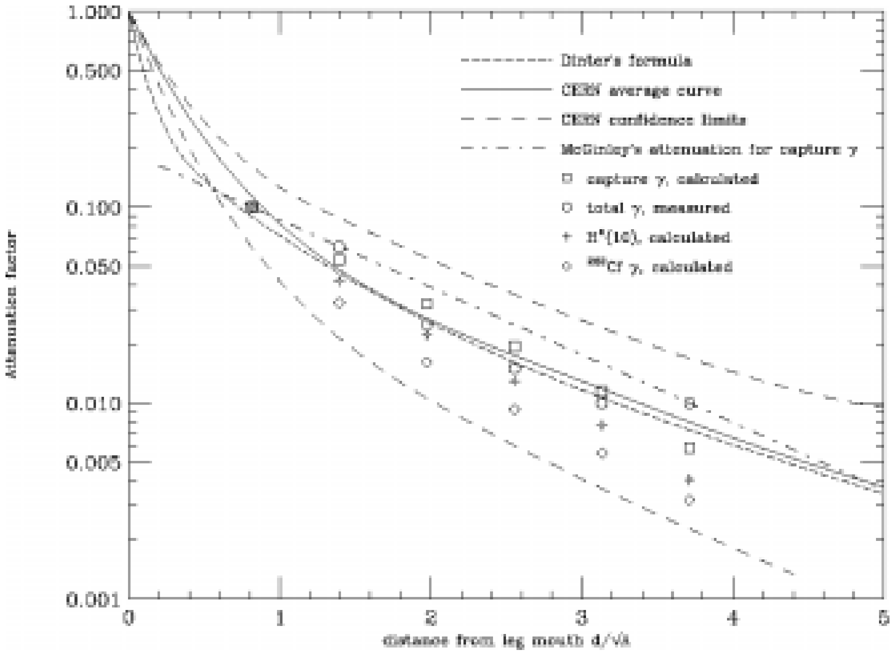

Fig. 7. Same as Fig. 6, for the W-shielded source.

The attenuation of Andersson-Braun response (measured and calculated) is indistinguishable from that of calculated ambient dose equivalent H*(10), although in absolute terms differences up to ±30% have been found; therefore, only H*(10) has been plotted.

In principle, the curves are supposed to be applied only to neutrons, but experience has shown that the attenuation curve for photon dose is typically situated close to the lowest curve[14, 15]. For capture gamma dose, McGinley et al.[16] have measured in different medical accelerator mazes a tenth-value length (TVL) ranging between 5 and 6.4 m. The exponential attenuation curve corresponding to TVL = 5 m has been plotted in Fig. 6 and 7, as the one which better agrees with the present measurements and calculations.

Since the values of the different quantities are not known at $d/A^{1/2} = 0$, all data have been normalized to the same value at the first measurement point (d = 1.405 m). The 0.1 value chosen is the average of those predicted by the CERN and by Dinter's curves, and is close to both.

5. DISCUSSION

The neutron dose equivalent as measured with an Andersson-Braun instrument is in fair agreement with the simulation, if the uncertainties about experimental conditions are taken into account. Differences up to about 40% exist in some cases, but differences of the same order have been found between values calculated using different available Andersson-Braun response functions.

A larger discrepancy (up to a factor 2) can be noticed between the absolute value of measured and calculated gamma doses. The possibility cannot be ruled out that such a difference could be due to the neutron sensitivity of the instrument used to measure gamma doses (Bicron remmeter by Harshaw/Bicron): indeed, a response to neutron *absorbed dose* (not dose equivalent!) equal to one third of that of gammas would be sufficient to explain the difference. However, since measurements made with a different instrument gave essentially the same results, no firm conclusion can be made.

From the practical point of view, empirical formulae and attenuation curves, although originally derived for very different accelerator energies and layouts, appear more than adequate for standard maze design, and in many cases there should be no need to resort to the use of a complex Monte Carlo program. However, special geometries or shielding doors are not easily handled by means of simple recipes. The present results confirm that FLUKA can be safely used in such cases, avoiding the need of an excessively conservative approach.

6. CONCLUSIONS

The results of the calculations were compared to the experimental data and to commonly used empirical formulae. The overall agreement is satisfactory, taking into account the limits of the simulation and the uncertainties of instrument response.

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