PHOTONEUTRON AND CAPTURE GAMMA DOSE CALCULATIONS FOR A RADIOTHERAPY ROOM MADE OF HIGH DENSITY CONCRETE

by

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Neutron and capture gamma ray dose equivalent along the maze and entrance door of a radiation therapy room made of high density concrete was calculated using analytical and Monte Carlo methods. The room geometry and the 18 MV photon beam of a Varian 2100C/D linac were simulated using MCNPX MC code. Four analytical methods including Kersey, French, McCall, and Wu-McGinley methods were used in the current study. Average difference of 13-30% was seen between analytical and MC methods along the maze for photoneutron calculations. The difference between Wu-McGinley and MC methods was about 17% for capture gamma ray calculations. It was concluded that the analytical methods overestimate both neutron and capture gamma ray dose equivalents compared to MC. Moreover, it was shown that the analytical methods can be used as conservative estimators for neutron and capture gamma calculations.

Key words: photoneutrons, Monte Carlo modeling, radiation therapy

INTRODUCTION

Photoneutrons and capture gamma photons are produced in high energy photon beams as consequences of photoneutron interactions between high energy photons and linac head components, concrete walls and patient itself [1, 2]. Several methods have been proposed and currently being used to calculate the dose equivalent of these harmful radiations in the maze inner entrance and at the maze entrance door [3, 6]. The design and geometry of a radiotherapy room depends strongly on used photon energy, workload and other installation site characteristics. Radiotherapy rooms are normally designed with standard dimensions proposed by linac manufacturer and are built with normal concrete with the density of 2.35 g/cm^3. While space is the first priority and a room cannot be built in its standard size, high density concrete with different densities can be employed to create required shielding with lower thickness in comparison to ordinary concrete. There are several compositions of high density concrete that are commercially available in the market [7]. Their compositions entail some high atomic number materials such as iron and barium which give higher density to concrete with comparable structural properties. To evaluate the shielding efficacy against neutrons and capture gammas, several analytical formulas have been proposed. Recently the report of International Atomic Energy Agency (IAEA) No. 47 has proposed analytical methods to calculate the neutron fluence in different points of radiotherapy rooms [8].

Total photoneutron fluence φ_A at the inner maze entrance per Gy X-ray at the isocenter [n^{0}m^{-2} Gy^{-11}] can be calculated by [8-10]

$$\varphi_A \quad \frac{Q_N}{4\pi d^2} \quad \frac{5.4 Q_N}{2\pi S} \quad \frac{1.26 Q_N}{2\pi S} \qquad (1)$$

where d [m] is the distance from the isocenter to point A in inner maze (see fig. 1), S [m^2] – the inner surface area of the treatment room , and Q_N – the photoneutron source strength in terms of n^0/Gy. Photoneutron source strength, Q_N, is the number of produced photoneutrons per 1 Gy photon dose absorption at the isocenter. Additionally, Q_N for different models of accelerators has been published in the literature [11, 12].

In all analytical methods, it is considered that concrete walls of radiotherapy room are made of ordinary concrete with the density of 2.35 g cm^3. The effect of concrete wall on neutron production, scattering

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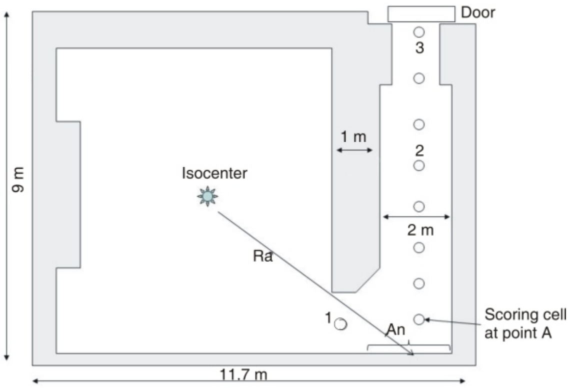

Figure 1. Geometry of the bunker used in the current study; the scoring cells with radius of 10 cm were located at height of 100 cm from the floor

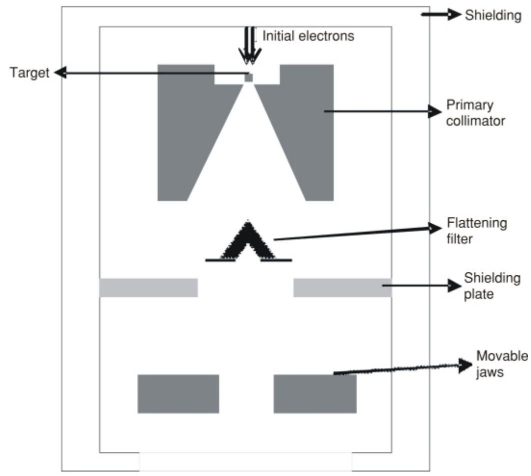

Figure 2. Schematic representation of the simulated head of Varian 2100 C/D

and absorption can be influenced by its atomic composition because the cross-section of photoneutron interactions varies with atomic number of concrete elements [5]. Consequently, a question arises: how well the neutron dose can be calculated by the proposed formula if the density and composition of concrete are different from ordinary concrete?

A radiotherapy bunker was made using high density concrete composed of hematite in the busy part of a cosmopolitan city with limitation in required space for standard linac installations. In order to fulfill the requirements for high energy photons shielding, the shielding barrier calculations for primary and secondary photons, as well as neutrons and neutron capture gamma ray were performed based on IAEA report No. 47. However, the current Monte Carlo (MC) study was conducted to verify the photoneutron and capture gamma ray shielding calculations using the above mentioned report. Moreover, the accuracy of some analytical methods was assessed comparing to the MC results.

MATERIAL AND METHODS

Monte Carlo simulation

In the current study we used the MCNPX MC code (2.4.0) with LA150U library file to simulate the 18 MV photon beam of Varian 2100 C/D linac and the treatment room made of a high density concrete (hematite) (figs. 1 and 2) [13]. The MCNPX, a general purpose MC code is capable to simulate the photoneutron generation from photon interactions as well as capture gamma from photoneutron interactions within treatment room and maze. Linac head components including target, primary collimator, flattening filter, and secondary collimator jaws were simulated using the data provided by linac manufacturer (fig. 2). The radiation beam direction was downward for all simulations. The model was validated by comparing the calculated and measured percent depth dose and beam profiles. It should be noted that the model was used in our previous study [14]. The modeling procedure was in accordance to other published works on linac MC modeling [15-17].

Application of full MC model of linac to perform photoneutron calculations requires a long time and the statistical uncertainty of the results is not acceptable in most cases. So, to speed up the photoneutron calculations within the maze, the full model was run and Q_N value and photoneutron spectra around the head were calculated. The method used for Q_N calculation was identical to our previous study [14]. The calculated neutron spectrum is shown in fig. 3. The MC calculated Q_N value was used for other MC calculations. An isotropic photoneutron source was defined at the target position and linac head components were deleted from treatment room geometry. Using this simple photoneutron source, photoneutron and capture gamma ray doses were tallied at points along the maze (fig. 1). For photoneutron and

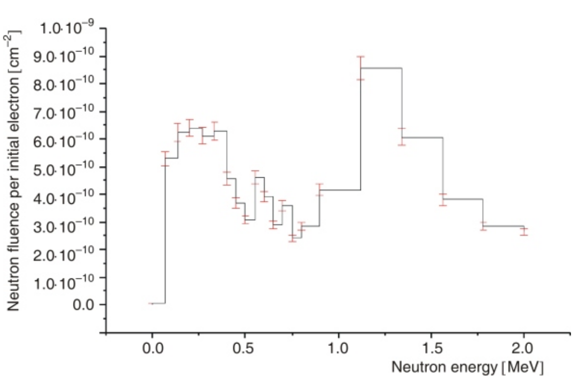

Figure 3. Photoneutron spectrum calculated by MCNPX MC code at the distance of 1 m from the target; the spectrum was used in our previous study [14]

capture gamma ray dose calculations, spheres with the diameter of 10 cm were defined along the maze and were filled with water. The photoneutron and capture gamma absorbed doses were calculated in terms of MeV/g per initial neutron and then their values changed to Gy per initial neutron. By multiplying the MC calculated dose by Q_N value, the absorbed dose from photoneutrons in different points per 1 Gy X-ray at the isocenter were calculated.

The statistical uncertainty of MC results was less than 2% in its worst case for point 3 at the maze entrance door.

For photoneutron simulations, the energy cut-off of 7 MeV was used for both electrons and photons, because the threshold energy of photoneutron reactions for main components of linac is higher than 7 MeV. For photon absorbed dose calculations at d_{max} the photon and electron energy cut-offs were set to be 10 keV and 500 keV, respectively. The energy cut-off of photons was not applied for capture gamma dose calculation to score full energy range of capture gammas.

ANALYTICAL METHODS

For neutron dose calculation we used four analytical methods including French, McCall, and Kersey and Wu-McGinley methods. Capture gamma dose was also calculated in maze using Wu-McGinley method which was recommended by IAEA report 47.

Neutron dose calculation methods

(1) In Wu-McGinley method which is used for single-bend mazes, exponential attenuation of photoneutrons is considered by [18]

$$D_n \quad 2.4 \ 10^{15} \varphi_A \sqrt{\frac{A_r}{S_1}} (1.64 \ 10^{\frac{d_2}{1.9}} \quad 10^{\frac{d_2}{T_N}} \qquad (2)$$

where D_n is the photoneutron equivalent dose at the maze entrance [$SvGy^{-1}$] and A_r and S_1 are cross-section areas [m^2] of inner maze entrance and the maze, respectively. d_2 is the distance from point A_r to the entrance door of maze (point 3) (fig. 1). In the current calculations, φ_A derived from formula (1) and the Q_N value of 1.2 10^{12} [neutrons per 1 Gy X-ray] recommended for Varian 2100 linac were used [14].

(2) According to Kersey method [19], the neutron dose equivalent at the entrance door of the maze is given by

$$H \quad H_0 \frac{T}{T_0} \frac{{d_0}^2}{{d_1}^2} 10^{\frac{d_2}{5}} \qquad (3)$$

where H_0 [Sv] is the dose equivalent due to neutrons , measured at the isocenter, d_0 [m] – the distance from the target to the isocenter, d_1 [m] – the distance from the isocenter to a point at the central line of the inner maze entrance (point 1). d_2 [m] is the distance from point 1 to the entrance door of the maze (point 3). T/T_0 is the ratio between the smallest and biggest cross-sectional area of the maze. The value of H, is the dose equivalent at the maze entrance door from neutrons per absorbed dose of X-rays at the isocenter. It considers only the contribution of direct neutrons to the dose, which means that scattered and thermal neutrons are not considered.

(3) French proposed a method for neutron dose calculation at the maze entrance door of radiotherapy room which can be explained by this [20]

$$H \quad \sum^{n} \frac{H_0}{R_a^2} \frac{A_n a_{d_n}}{R_b^2} \qquad (4)$$

where A_n [m^2] is the area of scattering surface n, H [Sv] – the neutron dose equivalent at the maze entrance door , H_0 [Sv] – the neutron dose equivalent at 1 m from the source , R_a [m] – the distance from the target to the mid-wall for surface n (fig. 1), R_b [m] – the distance from the mid-wall to the maze entrance door for surface n , and α_{dn} [m] – the dose albedo for surface n .

(4) McCall method [21] in the current study was explained in the recent previous study of Waller *et al* [22] and the following formula was used

$$H \quad \frac{\Phi_0}{2\pi} C \frac{\alpha_c A' A''}{R_a^2 R_b^2} \qquad (5)$$

where H [Sv] is the neutron dose equivalent at the doorway, Φ [ncm^{-2}] – the neutron fluence at 1 m from the source, A' [m^2] – the area of maze illuminated by the source, A'' [m^2] – the cross-sectional area at the end of the maze, C – the fluence to dose equivalent conversion factor, α_c – the current albedo, R_a [m] – the distance from the target to point 1, and R_b [m] – the distance from point 1 to the entrance door of the maze .

Capture gamma dose calculation method

To determine the dose of capture gamma rays, the proposed method by Wu-McGinley was used [18]

$$H \quad 5.7 \ 10^{16} \varphi_1 \ 10^{\frac{d_2}{6.2}} \qquad (6)$$

where φ_1 is the total photoneutron fluence at point A, d_2 [m] – the maze length , and H [Gy] is in terms of Gy per photon dose at the isocenter.

Treatment room simulation

The walls, roof and the maze's wall were built from heavy concrete with the density of 4.2 g/cm^3 achieved by adding 78% of hematite mineral to ordinary concrete composition. The composition of the used concrete was: 0.23% – H, 13.27% – O, 1.54% – Si, 5.6% – Ca, 0.63% – Mn, 1.03% – Al, 75.67% – Fe, 1.59% – Ti, and 0.64% – Va. This composition was used in material definition of walls in MC simulations. The height of the room was 3.65 m and the distance from X-ray source to the roof was considered 0.75 m. Dimensions of the simulated geometry are shown in fig. 1.

The maximum resistance that a concrete structure will sustain, when loaded axially in compression in a testing machine, at a specified rate, is measured as the compressive strength. The compressive strength of our concrete was provided by the producer and it was 299 kg/m^2. Its collapse load was determined to be 55000 kg.

RESULTS AND DISCUSSION

The comparison between MC and four different analytical methods for neutron dose equivalent calculations along the maze is shown in fig. 4. In this study, the MC method was considered to be more accurate method, as well as the reference for all comparisons. As can be seen, there is a close agreement between MC and Wu-McGinley method in all calculation points within the maze. However, a noticeable difference exists near the inner entrance of the maze. Comparing other methods with MC results shows that the French method acts better than McCall and Kersey methods. The results of Kersey method were better than the other methods near the maze entrance door, but in other points it overestimated the neutron dose equivalent in comparison to all other methods. To have more quantitative comparison between studied methods, the results in three points and the difference of calculated doses relative to MC method were tabulated in tabs. 1 and 2. Three points were selected for comparison including points at inner maze (A), middle maze (2) and at the maze entrance door (3). It is seen that among the studied analytical methods, the Wu-McGinley and then French methods provide better results comparing to MC results for neutron dose equivalents at the maze entrance door. The results were very similar to the results of Muller-Runkel *et al*, who showed that the French method acts better than Kersey and McCall methods in a radiation therapy room with 20 MeV photons [23]. Monte Carlo and experimental studies have shown that analytical methods overestimate the neutron dose equivalent at the maze entrance door [14].

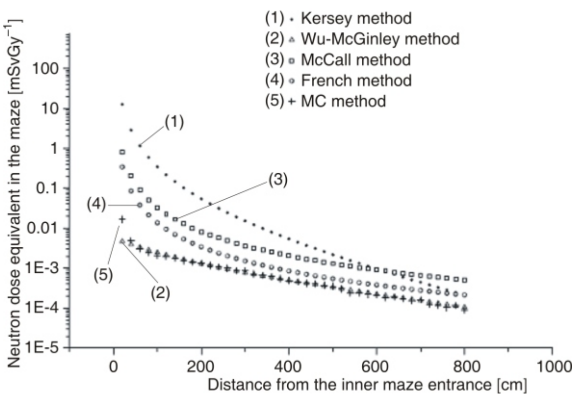

Figure 4. Comparison of calculated neutron dose equivalents between analytical methods and MC method

Table 1. Calculated neutron dose equivalent [mSvGy^{-1}] in three points including inner maze entrance, middle of the maze, and entrance door

Method	Point A	Point 2	Point 3
Kersey	$6.75\ 10^{-2}$	$8.85\ 10^{-3}$	$1.71\ 10^{-3}$
French	$6.35\ 10^{-2}$	$8.09\ 10^{-3}$	$1.48\ 10^{-3}$
McCall	$8.16\ 10^{-2}$	$9.80\ 10^{-3}$	$2.86\ 10^{-3}$
Wu-McGinley	$6.10\ 10^{-2}$	$7.81\ 10^{-3}$	$1.08\ 10^{-4}$
MCNPX	$5.63\ 10^{-2}$ $2.9\ 10^{-4}$	$6.5\ 10^{-2}$ $3.6\ 10^{-5}$	$9.48\ 10^{-4}$ $1.3\ 10^{-5}$

Table 2. The difference in calculated neutron dose equivalent between analytical methods and MC method

Method	Difference at the door (Point 3)	Average difference along the maze
Kersey	44%	30%
McCall	65%	53%
French	35%	23%
Wu-McGinley	12%	13%

Capture gamma dose equivalent was calculated by MC and Wu-McGinley methods. Figure 5 shows that Wu-McGinley method overestimates the capture gamma dose along the maze. The difference remains constant from the inner maze (point A) toward the entrance door and its average difference along the maze was 17% compared to MC method. Additionally, at the maze entrance door the difference was also 17%.

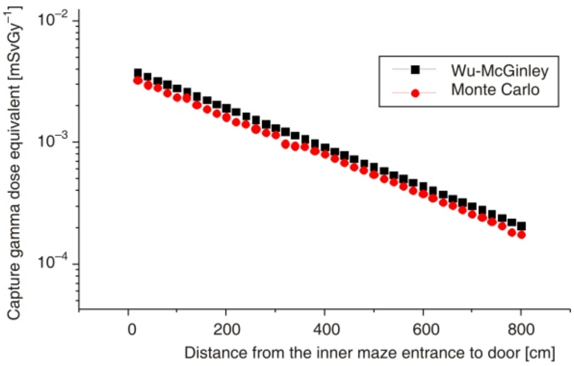

Figure 5. Comparison of MC and IAEA recommended method (Wu-McGinley Method) for capture gamma dose calculations along the maze

Our results were in close agreement with the recent study on application of MC method for calculating the neutron and gamma doses in different room geometry [14]. It was shown that the recently recommended Wu-McGinley method results in more accurate estimations in comparison to other analytical methods.

To have a practical application for our calculated data and evaluate the shielding requirements for maze entrance door, the total dose equivalent from neutrons and capture gammas was calculated for the studied case. It was calculated with assuming a workload of 600 Gy per week. The dose equivalent of 0.57 mSv per week was obtained at the maze entrance door. If the dose limit of 0.1 mSv per week is considered for controlled area adjacent to maze entrance door, it will be needed to take into account the neutron and capture gamma shielding in the maze entrance door design.

McGinley *et al*. [24] have tested the Kersey method in 13 accelerator rooms. Measurements were done by using both activation detectors and a neutron rem-meter. The results showed a discrepancy between measurements and Kersey method. Neutron dose level calculated for 7 of 13 facilities was within the range of 18 to 20% of the measured value and the calculated value exceeded the measured value by more than 20% for the six remaining accelerators.

It can be concluded that all the analytical methods, in general, provide conservative estimations of the maximum neutron dose equivalent at the entrance door of an accelerator maze. However, among the studied methods the Wu-McGinley method calculation was very close to the MC method.

CONCLUSIONS

In the current study the accuracy of different analytical methods in calculating neutron and capture gamma dose equivalents in a bunker made of high density concrete with hematite was evaluated and compared to MC results. The results showed that the analytical methods overestimate the neutron and capture gamma dose relative to MC results and they can be used as conservative estimators in designing maze outer door for radiation therapy with high density concrete. Finally, the use of Wu-McGinley for both neutron and capture gamma dose equivalent calculations is recommended for bunkers made of hematite.

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ПРОРАЧУН ЕКВИВАЛЕНТНЕ ДОЗЕ У РАДИОТЕРАПИЈСКОЈ СОБИ НАЧИЊЕНОЈ ОД ТЕШКОГ БЕТОНА

Применом аналитичких и Монте Карло метода израчунате су еквивалентне дозе од неутрона и пратећег гама зрачења дуж лавиринта и улазних врата собе за радиотерапију, начињене од бетона велике густине. Геометрија собе и сноп 18 MeV-ских фотона из линеарног акцелератора Varian 2100C/D, симулирани су коришћењем MCNPX Монте Карло програма. Четири аналитичке методе – Керсијева, Френчова, Мекколова и Ву-Мекгинлијева, коришћене су у овој студији. Разлике између аналитичких и Монте Карло метода у прорачуну фотонеутрона дуж лавиринта износила су 13-30%, у средњем. У прорачуну пратећег гама зрачења, разлика између Ву-Мекгинлијеве методе и Монте Карло симулације била је око 17%. Закључено је да аналитичке методе упоређене са Монте Карло симулацијом, прецењују еквивалентну дозу и неутрона и пратећег гама зрачења, те да могу бити употребљене у циљу конзервативне процене доза.

Кључне речи: фотонеутрони, Монте Карло моделовање, терапија зрачењем