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Monte Carlo Investigation on Neutron Dosimetry Parameters of ^{252}Cf Isotron Brachytherapy Source

(Kajian Monte Carlo ke atas Parameter Dosimetri Neutron Sumber Brakiterapi Isotron ^{252}Cf)

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ABSTRACT

In the present study, all dosimetry parameters around the ^{252}Cf Isotron source for neutron brachytherapy were calculated. The ^{252}Cf neutron energy spectrum was modeled using the Maxwellian spectrum. The calculations were performed using MCNPX.2.6.0 code based on the TG-43U1 protocol in the water medium. The neutron air kerma strength for the ^{252}Cf Isotron source was computed as 0.3098 ± 0.0002 cGy cm^2/μgh and the neutron dose rate constant was 6.559 ± 0.003 cGy/Uh. The anisotropy functions were calculated at intervals of 1, 2, 3, 4, and 5cm at different polar angles in the range of $\theta = 0^\circ$ to 90^o in respect to the source long axis in a water Phantom. The uncertainty of the calculated data at all distances is less than 1%. The radial dose function at different distances from the source for the Isotron model was compared with the theoretical HDR ^{252}Cf source model. The discrepancies between the results were the highest at the distance of 0.5 cm, with relative differences of 5.64 %. For the ^{252}Cf Isotron source, the data presented in this study provides better understanding on the dosimetric distribution of a theoretical neutron brachytherapy source, for potential clinical application.

Keywords: Brachytherapy; Isotron; neutron dosimetry; ^{252}Cf

ABSTRAK

Dalam kajian ini, semua parameter dosimetri sekitar sumber Isotron ^{252}Cf untuk brakiterapi neutron dihitung. Spektrum tenaga neutron ^{252}Cf dimodelkan menggunakan spektrum Maxwellian. Pengiraan dilakukan menggunakan kod MCNPX.2.6.0 berdasarkan protokol TG-43U1 dalam medium air. Kekuatan kerma udara neutron untuk sumber Isotron ^{252}Cf dihitung sebagai 0.30986 ± 0.0002 cGy cm^2/μgh dan kadar dos neutron tetap ialah 6.559 ± 0.003 cGy/Uh. Fungsi anisotropi dihitung pada selang 1, 2, 3, 4 dan 5 cm pada sudut polar yang berbeza dalam julat $\theta = 0^o$ hingga 90^o berkenaan dengan paksi panjang sumber dalam Phantom air. Ketidakpastian data yang dihitung pada semua jarak adalah kurang daripada 1%. Fungsi dos radial pada jarak yang berbeza daripada sumber untuk model Isotron dibandingkan dengan model sumber HDR ^{252}Cf teoritis. Perbezaan antara keputusan adalah yang tertinggi pada jarak 0.5 cm, dengan perbezaan relatif 5.64%. Bagi sumber Isotron ^{252}Cf, data yang dibentangkan dalam kajian ini memberikan pemahaman yang lebih baik mengenai pengagihan dosimetri sumber brakiterapi neutron teoritis, untuk aplikasi klinikal yang berpotensi.

Kata kunci: Brakiterapi; dosimetri neutron; Isotron; ^{252}Cf

INTRODUCTION

Due to greater relative biological effectiveness (RBE), neutron brachytherapy is more effective than conventional photon brachytherapy in treating tumors resistant to radiation such as bulky, late-stage tumors; melanomas; and cancers of the cervix (Ghassoun et al. 2010; Khosroabadi et al. 2016). Californium brachytherapy was introduced for cervical cancer in the 1970s. At the same time, the first long-term clinical research study began and was conducted on the usage of Californium-252 (^{252}Cf). The results showed that when ^{252}Cf was applied in tumor brachytherapy, it often produced better results than prevalent brachytherapy (Melhus et al. 2013). Later, Rivard (2000) performed a study on neutron dosimetry for a general ^{252}Cf brachytherapy source. The results showed that encapsulation thickness and composition of the source did not affect the ^{252}Cf thermal neutron fluence rate. The applicator tube (AT) model of the ^{252}Cf source is widely used in low-dose-rate (LDR) brachytherapy (Paredesa et al. 2010). Due to the dimensions of this source, it could only be used in intracavitary brachytherapy. Therefore, the advent of new sources could be the mark of a new era for ^{252}Cf based neutron brachytherapy. In October 2002, under the Cooperative Research and Development Agreement (CRADA) with Isotron Inc., the Oak Ridge National Laboratory (ORNL) encapsulated the Isotron ^{252}Cf sources, and shipped them to Georgia Tech in October of 2007. Because of the small size of this model, it could be used with remote the high-dose-rate (HDR) remote-afterloading system, comparable to the available commercial interstitial gamma brachytherapy systems of ^{192}Ir (Wang & Kelm 2009a). For Isotron model, the neutron and gamma dose rates in the water medium at various distances from the source have been determined by both the Monte Carlo N-Particle MCNP5 calculations and ion-chamber measurements (Wang & Kelm 2009b). The results of the neutron absorbed dose rates indicated that the data of the MCNP code were 25% lower than experimental data. Also, the experimental gamma absorbed dose rates in a water medium were higher than the MCNP data. Fortune et al. (2011) obtained gamma and neutron dose profiles near the Isotron brachytherapy source. The results showed that, by including the bremsstrahlung x-rays in gamma dose profile, the total gamma dose rates with MCNP code and experimental data were in a good agreement. The neutrons and gamma dose rates in axial and radial distances from Isotron source in a water medium were calculated using Maxwellian neutron energy spectrum and Maienshein's prompt fission gamma rays data (Al-Saihati & Naqvi 2013). The American Association of Physicists in Medicine (AAPM) TG-43 recommends that before using each new source, the dosimetric characteristics of the source must be determined to provide acceptable data for use in treatment planning calculations and dose prescription. This report was updated by AAPM (TG-43U1) in 2004 and included several modifications (Rivard et al. 2004). On the other hand, experimental dosimetry measurement near brachytherapy sources is a complicated process but the application of the Monte Carlo method is an effective technique for solving the dosimetry problems and can improve our understanding of all processes associated with radiation emission and transport by using random numbers. So far, no information has been published on the neutron dosimetry parameters of the Isotron brachytherapy source. The purpose of this research was to investigate the neutron dosimetric parameters of Isotron source, in comparison with other models of ^{252}Cf brachytherapy source. The MCNPX code (version 2.6.0) with histories of 5×10^7 particles were used for the calculation of the neutrons dosimetric parameters of the Isotron ^{252}Cf source based on the AAPM TG-43U1 protocol (Rivard et al. 2004). The neutron cross-section library was used from the ENDF/B-VI (Pelowitz 2008). The transport of the low energy neutrons was calculated by the thermal neutron scattering library (Lwtr.01t).

Materials and Methods

SOURCE GEOMETRY

Figure 1 illustrates the geometry of the ^{252}Cf Isotron source simulated by the MCNPX in this study. The dimensions of the Isotron source are 1.1 mm in diameter and 8 mm in length. The active length is a free-floating, 0.5 mm in diameter and 5 mm long Pd-Cf_2O_3 cermet wire with a mass density of 12.0 g/cm^3. The thickness of the capsule wall is 0.2 mm and consists of a Pt/Ir 10 % with a mass density of 21.51 g/cm^3. The radioactive substance was uniformly distributed within the volume of the active source. The average content of ^{252}Cf was estimated to be approximately 90 µg (Fortune et al. 2011; Rivard et al. 1999).

DOSIMETRY PROTOCOL

The neutron dose calculation formalism based on the AAPM TG-43U1 in a water medium around the source is as follows:

$$\dot{D}(r,\theta)=\Lambda S_k \frac{G(r,\theta)}{G(r_0,\theta_0)} g(r)F(r,\theta) \quad (1)$$

where $\dot{D}$ (r, θ) is the dose rate in the point of arbitrary r(cm) and the polar angle θ; r_0 = 1 cm and $\theta_0 = 90^0$ are the reference positions; the geometry factor is G(r,θ); S_K (U = cGy cm^2 h^{-1}) is the air-kerma strength; Λ (cGy h^{-1} U^{-1}) is the dose rate constant; g(r) is the radial dose function; and F(r,θ) is the anisotropy function (Rivard et al. 2004).

MONTE CARLO EVALUATION

In the present study, the Monte Carlo simulation is used to calculate the neutron dosimetry parameters in a water medium at various distances from the source by using the MCNPX.2.6.0 Code (Pelowitz 2008). For the simulation, the source is located at the center of a spherical water medium with a radius of 20 cm. The energy of neutrons emitted from ^{252}Cf is often modeled as either a Maxwellian or Watt fission spectrum. Maxwellian spectrum:

$$N(E)\ e^{-E/1.42}\ E^{0.5} \quad (2)$$

Watt fission spectrum:

$$N(E) = C\ e^{-aE} \sinh (bE)^{0.5} \quad (3)$$

E (MeV) in both equations is neutron energy, while a (MeV) and b (MeV^{-1}) are coefficients and N (E) is the Number of neutrons in particular energy. Prior to the simulations, each of these spectra was investigated. In different studies, the watt fission spectrum has been used with various coefficients (Table 1).

The number of neutrons in the energy range from 0 to 20 MeV for the watt fission spectrum with different coefficients (Table 1) and the Maxwellian spectrum were obtained. The results were compared with the neutron spectrum values by the Marten and Seeliger (IAEA 1987) in Figure 2.

According to Figure 2, the Maxwellian spectrum is in good agreement with the obtained spectrum by Marten and Seeliger (IAEA 1987). Therefore, for the energy of a neutron emitted from the ^{252}Cf source the Maxwellian spectrum used to carry out simulations.

F6 (MeV/g) tally is an estimate of the energy deposition in the track length of a cell. In the calculation of the F6 tally assumes all energies transferred to charged particle are deposited locally that it is called a kerma tally. Due to the fact that in the transport of neutrons, the range of produced charged particles is

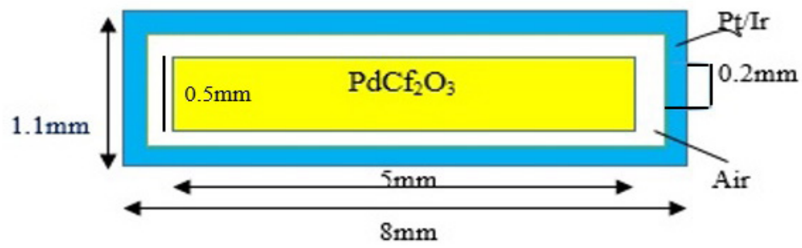

FIGURE 1. Simulated geometry of the ^{252}Cf Isotron source by the MCNPX in the present study

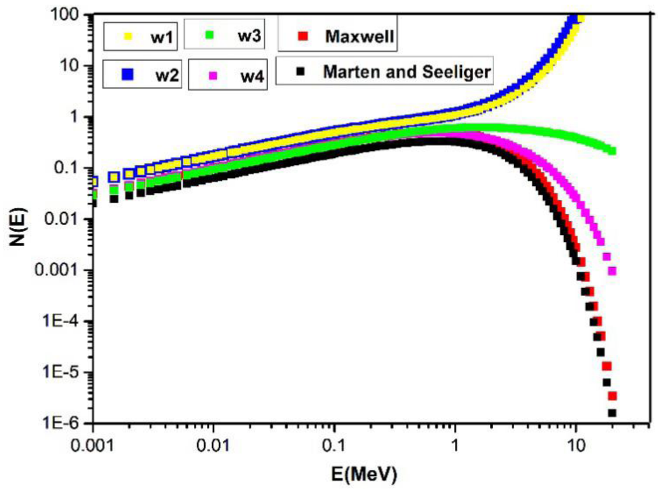

FIGURE 2. The ^{252}Cf source neutron spectrum, using the watt fission spectrum with different coefficients, the Maxwellian spectrum and the neutron spectrum obtained by Marten and Seeliger

TABLE 1. Different coefficients of the Watt fission spectrum used in various studies

Watt fission spectrum	References	a(MeV) and b(MeV)$^{-1}$
W1	Rivard et al. (1999)	a=0.97 b=2.92
W2	Ghassoun (2013)	a=1.02 b=2.92
W3	Krishnaswamy (1974)	a=0.88 b=2.00
W4	Radev (2014)	a=1.18 b=1.03

short (in cell dimensions), hence kerma can be a good estimate for the calculations of the neutron absorbed dose (Pelowitz 2008). For calculations of neutrons dosimetry parameters of ^{252}Cf source, the isotopic compositions, mass densities, and neutron cross-section libraries were utilized as shown in Table 2. Neutron cross-section library identifiers generally contain the atomic number Z, mass number A, and library specifier ID.

To estimate the neutron dose rate distribution in liquid water, the ^{252}Cf Isotron source was situated in the center of a spherical 20 cm radius water phantom. The neutron dose rate was determined in water in a cylindrical annulus with a size of 0.2 mm × 0.2 mm. It was deeply positioned along the transverse axis at distances ranging from 0.5 to 10cm from the source center. The neutron absorbed doses in water medium were calculated using F6 tally of MCNPX code. Air-kerma strength is calculated as follows (Rivard et al. 2004):

$$S_k = K_\delta \times r^2 \tag{4}$$

where K_δ is the air kerma rate in a vacuum. The neutron air kerma strength is computed by placing the source at the center of a sphere with a radius of 200 cm. The neutron air kerma rate is stored in a cylindrical annulus that is 0.5 cm diameter × 0.5 cm length in the transverse axis at distances ranging from 10 to 120 cm from the source filled with dry air. The kerma rates were obtained using an F6 tally. The geometry function G(r, θ), was determined using the AAPM TG-43U1 approximation for a line source and F4 tally of the MCNPX code. The particle fluence (1/cm^2) in each cylindrical in a vacuum was calculated in order to neglect the absorption and the scattering in the

TABLE 2. Neutron cross-section library, mass density and isotopic compositions for brachytherapy source and water medium (Rivard 2000)

Neutron cross-section library		Water	$Pd\text{-}Cf_2O_3$	Pt/Ir 10%	dry air
^{1}H	1001.60c	0.11186			
^{2}H	1002.60c	0.000034			
C	6000.60c				0.000086
^{14}N	7014.60c				0.77777
^{15}N	7015.60c				0.00309
^{16}O	8016.60c	0.887751	0.1741		0.209625
^{17}O	8017.60c	0.000355			0.000089
Ar	18000.59c				0.00934
^{105}Pd	46105.50c		0.49		
^{108}Pd	46108.50c		0.49		
Ir	77000.55c			0.1	
Pt	78000.35c			0.9	
^{252}Cf	98252.60c		1.8259		
Mass density(g/cm3)		0.998	12.0	21.5	0.001197

source and the medium; therefore, only the effect of activity distribution is represented. The radial dose function g(r), was obtained according to the instructions of the AAPM TG-43U1 (Rivard et al. 2004):

$$g(r) = \frac{\dot{D}(r,\theta_0)G(r_0,\theta_0)}{\dot{D}(r_0,\theta_0)G(r,\theta_0)} \tag{5}$$

The anisotropy function was obtained according to the instructions of the AAPM TG-43U1 (Rivard et al. 2004):

$$F(r,\theta) = \frac{\dot{D}(r,\theta)G(r,\theta_0)}{\dot{D}(r,\theta_0)G(r,\theta)} \tag{6}$$

The values of the anisotropy function were calculated at different radial distances from r = 0.5 to 10 cm relative to the source center, and the different polar angles were in the range of $\theta = 0^o$ to 90^o with respect to the source long axis. The dose rate constant Λ (cGy h^{-1} U^{-1}), is obtained according to the instructions of the AAPM TG-43U1 (Rivard et al. 2004):

$$\Lambda = \frac{\dot{D}(r_0,\theta_0)}{S_k} \tag{7}$$

Results and Discussion

NEUTRON ABSORBED DOSE RATES

A comparison of the reference neutron dose rate (r = 1 cm, $\theta = 90^o$) in water medium for this study with other studies as shown in Table 3.

The neutron absorbed dose rates on the transverse axis compared with previously published studies are shown in Table 4. The statistical uncertainty for the neutron absorbed dose rates at $r \leq 3.5$ cm is less than 0.3% and at $4 \leq r \leq 10$ cm is less than 0.6%. The neutron absorbed dose rate compared to the transverse distance is shown in Figure 3.

It can be seen from this plot that the neutron dose drops markedly as the transverse distance increases. The neutron dose rate gradient is exceedingly sharp at the distances near the source and drops off rapidly as the distance increases. The neutron dose rate results obtained from the simulations were compared with the results by Wang and Kelm (2009a), Al-Saihati and Naqvi (2013) and Krishnaswamy's calculation for a ^{252}Cf source with 0.4 cm active length (Krishnaswamy 1974). There is good agreement between the results.

Neutron absorbed dose rates due to different in thickness of the source casing were calculated as well. In this calculation, the thickness of the casing varied from 0.10 to 0.25 mm. The variant thickness did not affect the neutron dose rates. This is shown in Figure 4. Note that at this point, we are limited to the air gap space between the source and the casing. Therefore, the thickness varied between 0.10 and 0.25 mm.

RADIAL DOSE FUNCTION g(r)

The neutron radial dose function describes the effect of water attenuation on the neutrons emitted from a ^{252}Cf source and accounts for the dose fall-off along the source transverse axis due to the neutron scattering and attenuation. The neutron radial dose function in liquid water was calculated using (5) for distances between 0.50 and 10.0 cm. Table 5 lists the values of g(r) for the ^{252}Cf Isotron source in water that was calculated for this work. The computed radial dose functions g(r) for the source were evaluated and are shown graphically in Figure 5.

As TG43-U1 recommendation, g(r) for ideal orientation in water was fit to a fifth order Polynomial function. A fifth-order polynomial fit into the MCNP radial dose function in water for a range of r values between 0.5 and 10 cm yielded the following relationship:

TABLE 3. Comparison of neutron reference dose (cGy/μgh) in water medium for The Isotron source with other works

References	$\dot{D}(r_0,\theta_0)\left(\frac{cGy}{\mu gh}\right)$	Method
Rivard et al. (1999)	1.873 1.863	MCNP4B Experimental
Paredes et al. (2007)	$1.916 \pm 2\times10^{-4}$	MCNPX
Ghassoun et al. (2010)	$1.868 \pm 3\times10^{-4}$	MCNP5
This study	2.032±0.003	MCNPX

TABLE 4. ^{252}Cf transverse axis neutron absorbed dose rates, $\dot{D}(r,\theta_0)\left(\frac{cGy}{\mu gh}\right)$

r(cm)	Present study	Krishnaswamy (1974)	Al-Saihati and Naqvi (2013)	Relative differences with Al-Saihati and Naqvi %	Wang and Kelm (2009a)	Relative differences with Wang and Kelm %
0.50	7.756	8.772	7.803	-0.565	7.814	-0.710
0.75	3.582	--	3.60	-0.438	-	-
1.00	2.032	2.196	2.044	-0.587	2.047	-0.724
1.50	0.902	0.953	0.904	-0.263	0.905	-0.422
2.00	0.495	0.524	0.497	-0.384	0.498	-0.597
2.50	0.308	0.331	0.308	-0.087	0.309	-0.295
3.00	0.207	0.216	0.207	-0.030	0.208	-0.391
3.50	0.146	0.109	0.146	0.013	0.146	-0.048
4.00	0.106	0.083	0.107	-0.452	0.107	-0.410
4.50	0.081	0.064	0.080	0.270	0.081	-0.409
5.00	0.062	-	0.062	-0.374	0.062	-0.60
5.50	0.049	--	0.048	0.943	0.055	-11.13
6.00	0.039	-	-	-	-	-
7.00	0.025	-	-	-	-	-
8.00	0.017	-	-	-	-	-
9.00	0.012	-	-	-	-	-
10.0	0.009	-	-	-	-	-

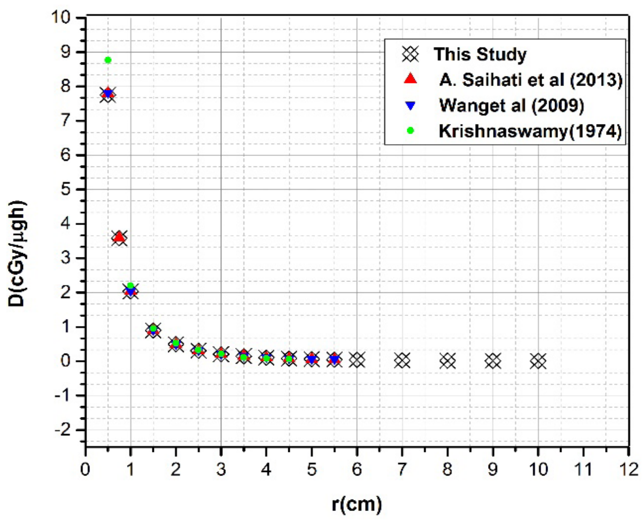

FIGURE 3. Neutron dose rate results obtained by the MCNPX in this study and other works (Al-Saihati & Naqvi 2013; Krishnaswamy 1974; Wang & Kelm 2009a)

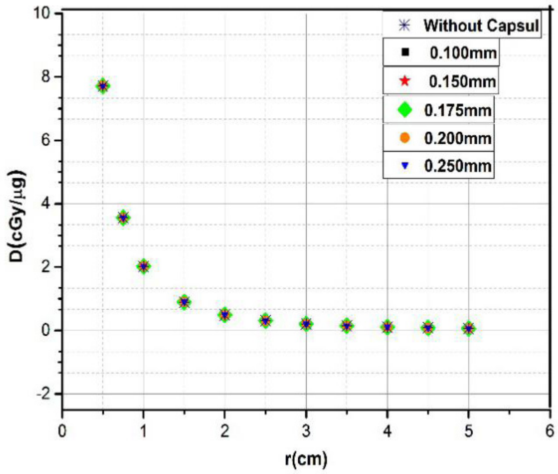

FIGURE 4. Neutron absorbed dose rates in water medium with different casing thicknesses of the source

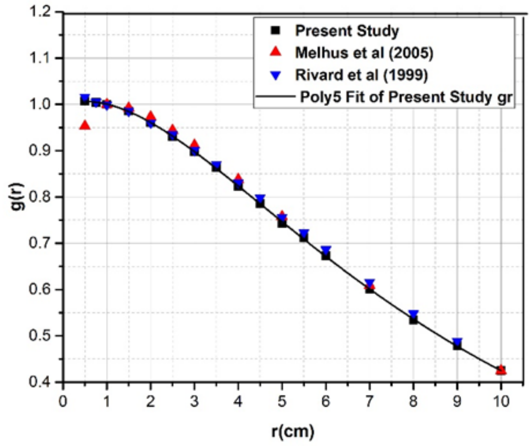

FIGURE 5. Comparison of the simulated radial dose function of the ^{252}Cf source in the water medium with other works (Melhus et al. 2005; Rivard et al. 1999)

$$g(r) = a_0 + a_1 r + a_2 r^2 + a_3 r^3 + a_4 r^4 + a_5 r^5 \quad (8)$$

$a_0 = 0.99949 \pm 0.00428$, $a_1 = 0.02846 \pm 0.008$, $a_2 = -0.03149 \pm 0.00458$, $a_3 = 0.00434 \pm 0.0011$, $a_4 = -2.75786 \times 10^{-4} \pm 1.16203 \times 10^{-4}$, $a_5 = 7.0736 \times 10^{-6} \pm 4.45127 \times 10^{-6}$, Adj. R-Square = 0.99.

ANISOTROPY DOSE FUNCTION F(r, θ)

The angular variations of neutron absorbed dose rate at different distances of the source are calculated by the anisotropy function F(r, θ). The anisotropy function was calculated at the intervals of 1, 2, 3, 4, and 5 cm at the angles from 0^o to 90^o in a water medium. The calculated values of F(r, θ) for the Isotron source in water are presented in Table 6. Uncertainty in calculated data at each distance is lower than 1%.

Table 7 shows the 2D anisotropy function data for the neutron from the theoretical HDR ^{252}Cf source by Melhus et al. (2005). This model is the same in terms of active length as the Isotron source (same active length of 5 mm). The MCNP neutron scattering library (lwtr.60t) was used in the study of Melhus et al. (2005) but the neutron scattering library in this study is (lwtr.01t). In addition, the capsule thickness, capsule length, composition, density, and libraries are also different in the Isotron model. For the two HDR sources, all F(r, θ) values were within 1% of unity. This was expected due to the thin walls, low cross-sections, and high-Z materials comprising the encapsulation. Thus, the impact of anisotropy function was minimal and considered unity for practical purposes with no significant loss in accuracy for brachytherapy clinical treatment planning.

AIR KERMA STRENGTH S_K

In Table 8, the amount of air kerma strength obtained in this work is compared with the amount of air kerma strength in the AT model.

TABLE 5. Radial dose functions g(r) for the ^{252}Cf source

r(cm)	g(r) (Isotron Model)	g(r) (Melhus et al. 2005)	Isotron relative differences with (Melhus et al. 2005) %
0.50	1.007 ± 0.003	0.953	5.643
0.75	1.004 ± 0.003	-	-
1.00	1.000 ± 0.003	1.000	0.000
1.50	0.986 ± 0.003	0.992	-0.626
2.00	0.961 ± 0.003	0.973	-1.263
2.50	0.931 ± 0.004	0.944	-1.371
3.00	0.898 ± 0.004	0.912	-1.550
3.50	0.864 ± 0.004	-	-
4.00	0.822 ± 0.003	0.838	-1.851
4.50	0.786 ± 0.004	-	-
5.00	0.743 ± 0.003	0.758	-2.012
5.50	0.712 ± 0.003	-	-
6.00	0.673 ± 0.003	-	-
7.00	0.601± 0.003	0.609	-0.134
8.00	0.535 ± 0.003	-	-
9.00	0.479± 0.003	-	-
10.0	0.425 ± 0.001	0.425	0.032

TABLE 6. F(r, θ) for neutrons emitted from the ^{252}Cf Isotron source

angles	1 cm	2 cm	3 cm	4 cm	5 cm
θ=0°	0.99	0.99	1.01	1.00	1.01
θ=10°	1.01	1.00	1.00	1.01	0.99
θ=20°	0.99	0.99	1.01	1.01	0.99
θ=30°	1.00	1.00	1.00	0.99	1.00
θ=40°	0.99	0.99	1.00	1.00	1.00
θ=50°	0.99	0.99	0.99	1.00	1.01
θ=60°	1.00	1.00	0.99	0.99	1.01
θ=70°	0.99	1.00	0.99	1.01	1.00
θ=80°	0.99	1.00	0.99	0.99	1.01
θ=90°	1.00	1.00	1.00	1.00	1.00

TABLE 7. F(r, θ) for neutrons emitted from the theoretical HDR ^{252}Cf source (Melhus et al. 2005)

angles	1 cm	2 cm	3 cm	4 cm	5 cm
θ=0°	0.97	0.98	0.97	0.98	0.98
θ=10°	0.98	0.98	0.98	0.98	0.99
θ=20°	0.99	0.99	0.99	1.00	0.99
θ=30°	1.00	1.00	1.00	1.00	1.00
θ=40°	1.00	1.00	1.00	1.00	1.00
θ=50°	1.00	1.00	1.00	1.00	1.00
θ=60°	1.00	1.00	1.00	1.00	1.00
θ=70°	1.00	1.00	1.00	1.00	1.00
θ=80°	1.00	1.00	1.00	1.00	1.00
θ=90°	1.00	1.00	1.00	1.00	1.00

TABLE 8. Comparison of the air kerma strength obtained in this study and other studies

References	S_k cGy cm²/µgh)
This study	0.3098 ± 0.0002
Ghassoun et al. (2010)	0.3348 ± 0.0003
Rivard et al. (1999)	0.33

TABLE 9. Constant of dose rate, Λ_N, of the ^{252}Cf source compared to other results reported

References	ΛN (cGyh-1U-1)
This study	6.559 ± 0.003
Paredes et al. (2007)	5.719 ± 0.005
Ghassoun et al. (2010)	5.579 ± 3.7368×10-4

DOSE RATE CONSTANT Λ

The dose rate constant of the ^{252}Cf source is compared with the AT model results in Table 9. Given that the constant of the neutron dose rate depends on the geometry of the source and the spatial distribution of radioactive matter within the source, it may change due to the geometrical characteristics, the source capsule material, or the phantom material.

Neutrons emitted from the ^{252}Cf Isotron source can be used in the treatment of cancer in brachytherapy (Venselaar et al. 2000). Hence, the recognition of the dosimetric characteristics of this source is important. In this study, simulations were performed based on the proposed protocol AAPM TG-43U1 (Rivard et al. 2004) and by using the Monte Carlo method to evaluate the neutron dosimetric characteristics of the ^{252}Cf Isotron brachytherapy source. The calculated results of the dosimetry parameters indicate that the air kerma strength for ^{252}Cf is equal to 0.3098 ± 0.0002 (cGy cm²/µgh) or 0.3098 U/µg, which is consistent with other reported results (0.3348 and 0.330 cm²/µgh, by Ghassoun et al. (2010) and Rivard et al. (1999), respectively). Ghassoun et al. (2010) and Rivard et al. (1999) had been used an AT source with active length 15 mm while in this study the Isotron active length was only 5 mm. In the AT model, the source is protected from two capsules. But in the Isotron model, the source has only one capsule. The simulations in Ghassoun et al. (2010) study and Rivard et al. (1999) study were performed using the MCNP5 and MCNP4B code, respectively. While in this study, the MCNPX code was used to perform radiation transport calculations. In both models (this work and other works), calculations are done with the assumption of 1 µg of ^{252}Cf with 1 decay per second that the good agreement between air kema strength in this work with other works indicates the veracity of the calculations in this study. Obviously, given the actual activity of the source, the air kerma strengh of each model can be obtained.

The neutron absorbed dose rate at a reference point in the water medium was determined to be 2.032 ± 0.003 cGy/µgh. The dose rate constant was estimated to be equal to 6.559 ± 0.003 cGy/Uh, dividing the absorbed dose rate at the reference point by the neutrons air kerma strength in the air medium.

The ^{252}Cf neutron radiation exhibited little anisotropy in relation to the capsule. Thus, the anisotropy function can be considered the unity factor for practical purposes with no significant loss in accuracy due to the thin walls, the low cross-sections, and the high-Z materials comprising the encapsulation. The impact on F(r, θ) was minimal and considered unity for practical purposes. There was no significant loss in accuracy for brachytherapy clinical treatment planning due to the small relative volumes subtended along the source long-axes. The uncertainty of the anisotropy function among the calculated data is less than 1%. The radial dose function at different distances from the source for the Isotron model (with an active length of 5 mm) was compared with the theoretical HDR ^{252}Cf source model (with an active length of 5 mm) in research by Melhus et al. (2005). The discrepancies between the results (especially at a distance of 0.5 cm with relative differences of 5.643%) can be attributed to differences in the models, material densities, casing thickness, different dimensions and application of the different versions of the MCNP. also, as the depth increased, the radial dose function decreased more slowly.

CONCLUSION

Neutrons can leave their energy at greater distances from the source. Hence, neutrons can be an appropriate choice to treat tumors due largely to their greater penetration depth. So far no information has been published on the neutron dosimetry parameters of the Isotron brachytherapy source. In the present study, the dosimetry parameters around the ^{252}Cf Isotron source for neutron brachytherapy were calculated using the Monte Carlo simulation method based on the AAPM TG-43U1 protocol. Thus, the calculated values are operative in brachytherapy software designing with the ^{252}Cf Isotron brachytherapy source.

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