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Calculation of Dose Distribution Around a Clinical ^{252}Cf Source for Neutron Therapy Based on AAPM, TG-43 Protocol

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ABSTRACT

Determining dose distribution around the brachytherapy sources is crucial to establish an accurate treatment planning. In this study dosimetric parameters of ^{252}CF as neutron brachytherapy source were calculated using Monte Carlo simulation method.Physical and geometrical parameters of ^{252}CF source were simulated by MCNPX (2.6.0)modeling code. Air kerma strength, S_k, of source positioned inside a vacuum sphere was calculated. Recommended dosimetric parameters by AAPM, TG-43 protocol were calculated by centering source in a homogeneity phantom.The air kerma strength of ^{252}CF source was estimated 0.345 (cGycm^2/¼gh). Dose rate constant was 7.014 cGyh^{-1}U^{-1}. The Radial dose function with 5 degree equation was estimated by $g_N(r) = 1.1745+0.2298r+0.061r^2+-0.0109r^3+0.0009r^4+-3E10^{-5}r^5$. Numerical amounts of the anisotropy dose functions and the related equations were calculated.Despite low-energy emission photons and high dose gradients with radial distance, dosimetric parameters of the model 6711 I-125 source can be calculated by MCNPX Monte Carlo code with acceptable accuracy. Calculated parameters of the model 6711 I-125 brachytherapy source can be used by treatment planning systems for brachytherapy.

Key words: Brachytherapy, Neutron dosimetry, ^{252}CF Source, Monte Carlo Simulation.

INTRODUCTION

Compared with photon and electron beams,fast neutrons have high therapeutic ratio for tumor cells treatment. This can be attributed to the relative high biological effect, high radiation weighting factor,low oxygen enhancement ratioand high linear energy transfer of neutrons and fission fragments[1, 2]. Delivering maximum and minimum dose respectively to the tumor and healthy surrounding tissues is possible through Boron Neutron Capture Therapy (BNCT) by adjusting fast neutrons before the depth of the tumor and also targeting tumor with boron element (due to the high cross section, 3837 Barn, in the reaction with thermal neutrons)[3].Despite the potential and high accuracy of BNCT method in treating various tumors, especially hypoxic tumors and advanced cancers (particularly brain tumors),high costs and the need for reactors to produce a neutron beam, has limited the widespread use of this technique[4].

Accordingly, the use of radioisotopec alifornium-252 (^{252}CF) as neutron emitter is proposed recently in radiation treatment, based on Neutron Brachytherapy (NBT) as a cost effective treatment method[5]. Using^{252}CF in radiation therapy was introduced for the first time by Schlea and Stoddard[6]. The half-life of ^{252}CF is approximately 2.645 years and in 96.6% cases decays with alpha particles irradiation and 3.1% of the disintegration will lead to nuclear fission. In each fission, on average, 3.768 neutrons are released;1 gram ^{252}CF, irradiates 2.314×10^6 neutrons per second, with a most probable energy in 0.7 MeV (The neutron energy spectrum follows the Watt function, from 1eV to 20MeV) (7) which provide appropriate flux for brachytherapy[4,8].

Performing neutron brachytherapy accurately andachieving the maximum tumor dose and the minimum dose of normal tissues require appropriate determination of dose distribution around the ^{252}CFsource;the results will be subsequently used in the treatment designing software. In the present study, dosimetry parameters of ^{252}CFsource is calculated using Monte Carlo calculations and based on the proposed protocol AAPM-TG 43 (9).

MATERIALS AND METHODS

Simulation code MCNPX (2.6.0) (8) was used for the simulation calculations. The energy of neutron cutoff was considered equal to 0.1 eV. No method was applied to reduce the error of neutron transport in the programs. The neutron scattering cross sections in the solid state, S(α, β), named lwtr.01t, was used in order to precisely calculate the transport of slow neutrons from the cross sections of thermal neutrons.In the simulation programs of this study, the number of transported neutron from neutron source was considered equal to 10^8resulting in less than 5% error in the farthest distance from the source (11 cm).

^{252}CFNeutron Source

The dimensions and materials of ^{252}CF neutron source utilized in this study are shown in Figure 1. The active element of cylindrical source is made of californium oxide, Cf_2O_3, with 12 gr/cm^3 density. The cylinder height and radius is1.5 and 0.615cm respectively, which is located in aplatinum - iridium 10% capsule (Pt/Ir-10%), with following characteristics:density= 21.55 gr/cm^3, the inner diameter= 0.135 cm, the external diameter= 0.175 cm, internal length= 1.55 cm and the external length= 1.77 cm. The internal and external diameter and the internal and external length of the outer capsule is 0.18, 0.28, 1.782 and 2.314 cm, respectively. As a part of the design, a spherical eye letwith the diameter of 0.0635 cm is embedded at the end of the source. ^{252}CF fission energy spectrum is considered as a function of W (4, 8), with the following equation:

$$N(E) = Ce^{-0.9756E}\sinh(2.926)^{1/2} \quad ...(1)$$

Protocols AAPM-TG.43 and Neutron Source Dosimetry of ^{252}CF

Based on the proposed protocol AAPM-TG 43(9), the absorbed dose rate of ^{252}CFneutron source is calculated any where around the source. This computing is performed in the water phantom D (r,θ), in terms of the radial distance from the central source, r, and the angle from the central axis of the source, q, and with the following equation:

$$\dot{D}(r,\Theta) = \dot{S}_{KN}.\Lambda_N.g_N(r).F_N(r,\Theta).[\frac{G(r,\Theta)}{G(r_0,\Theta_0)}] \quad ...(2)$$

Where, S_{KN}, the air kerma strength in a vacuum environmentis the product of air kerma rate and the square of the distance from the source center:

$$\dot{S}_{KN} = \dot{K}_N(d).d^2 \quad ...(3)$$

Where K_N is the air kerma rate and d is the distance from the source. Centering the source inside the vacuum sphere with the radius of 1.5 m, neutron air kerma was calculated using f6 tally in terms of MeV/gr in the concentric spherical shells with the thickness of 1 cmand up to 1 cm distance. The average of the calculated air kerma is multiplied by the square of the distance to report the kerma strength.

Λ_N, the constant of neutron dose rate is calculated by dividing the source neutron dose rate (in terms ofcGy/µghin the perpendicular axis reference point on the central axis of the source), θ_0=90°, r_0=1 cm, in a homogeneous water phantom by air kermastrength, :

$$\Lambda_N = \frac{\dot{D}_N(r_0,\theta_0)}{\dot{S}_{KN}} \quad ...(4)$$

Neutron absorbed dose is calculated positioning the source in the center of the water phantom with the dimensions 30×30×30cm^3 (using *f8 tally on perpendicular-axis to the central axis, 1 cm, and the sphere voxel with the radius of 0.5 cm) and is divided by the air kerma strength. The calculated simulation outputs are converted to Gyabsorbed dose unit applying relevant coefficients.

The radial dose function, $g_N(r)$, considers the dependence of neutron absorption and photons scattering along the transverse axis in the medium and is expressed as follows:

$$g_N(r) = \frac{\dot{D}_N(r,\theta_0)G(r_0,\theta_0)}{\dot{D}_N(r_0,\theta_0)G(r,\theta_0)} \quad ...(5)$$

According to the practical calculations, the best equation to express the radial dose function is the fifth grade function:

$$g_N(r)=a_0+a_1r+a_2r^2+a_3r^3+a_4r^4+a_5r^5 \quad ...(6)$$

The radial dose function, $g_N(r)$, is calculated by *f8 tally in the cylindrical shell around the central axis of the source, with 0.4cm thickness, at a distance of 0.2cm to 11cm from the source center and is converted to dose rate applying appropriate coefficients.

G (r, θ) is a geometric factor that considers the reduction of the neutrons flow, at any distance from the source and based on the geometry of the source; this is dependent on the distribution of radioactive material in the source. Considering the point source, G (r, θ) = $1/r^2$applies and for a line source with uniform distribution of geometric factor the equation is:

$$G(r,\theta) = \frac{\arctan\left[\frac{L}{2r\sin\theta}+\cot\theta\right]+\arctan\left[\frac{L}{2r\sin\theta}-\cot\theta\right]}{Lr\sin\theta} \quad ...(7)$$

Where L is the length of the source active element. For example, for θ = 90, the geometric factor to calculate the radial dose function is expressed as follows:

$$G(r,_0) = \frac{2\arctan\frac{L}{2r}}{Lr} \quad ...(8)$$

In this study, the more stringent assumption of source linearity has been used and these coefficients were calculated separately for the studied points in the radial distances and at different angles.

Anisotropy function, $F_N(r, \theta)$, takes into account non-homogeneity, angular dependence of neutron absorption and scattering within the source capsule andis expressed in the Eq. (8) as follows: (10):

$$F_N(r,) = \frac{\dot{D}_N(r,)G(r,\theta_0)}{\dot{D}_N(r,\theta_0)G(r,)} \quad ...(9)$$

Anisotropy function, $F_N(r, \theta)$, was calculated using Eq. 8 at intervals of 2, 3, 5, 8, 10 cm and at angles of 0, 5, 10, 20, 30, 40, 50, 60, 70, 80, 90 degrees in a water phantom.

RESULTS

Air Kerma Rate and Dose Rate Constants

Air kerma strength of neutrons, θ is calculated and with a 3% error is equal to 0.345 (cGy cm²/µgh) or 0.345 U/µg. Dose absorbed rate in water at a distance of 1cm from the source center, θ is 2.075 cGy/µgh;the constant of neutron dose rate, Λ_N, was estimated 6.014 cGy/Uh,dividing dose absorbed rate by the air kerma strength of the neutrons in the air.

Radial Dose Function

The calculated values for the radial dose function, $g_N(r)$, are presented in Table 1. The resulted coefficients from 5 fitted variables on the values of radial dose function based on the radial distance, r, is obtained as follows:

A_0=1.17545, a_1=-0.2298, a_2=0.061, a_3=-0.1090, a_4=0.9000, a_5=-3×10^{-5}

Anisotropy Dose Function, $F_N(r, \theta)$

The obtained values for dose anisotropy function, $F_N(r, \theta)$ are shown in Table 2. As expected, increasing the radial distance and angle to the source central axis leads to enhancingthe calculated values for the anisotropy dose.

DISCUSSION

In all discussed studies at following, dose calculations and measurements were reported based on protocols AAPM, TG-43 (7) and the length of source's active element was equal to 1.5 cm. The computed air kerma strength for ^{252}CF, , is equal to 0.3450 (cGy cm2/µgh) or 0.3450 U/µgwhich

Table 1: The calculated values of the radial dose function, $g_N(r)$, based on the radial distance, r, from the source center of ^{252}CF

Distance from the source center (4)	radial dose function, $g_N(r)$	Distance from the source center (4)	radial dose function, $g_N(r)$	Distance from the source center (4)	radial dose function, $g_N(r)$
0.6	1.050	3.8	0.755	7	0.548
0.8	1.030	4	0.742	7.2	0.536
1	1.000	4.2	0.728	7.4	0.524
1.2	0.975	4.4	0.716	7.6	0.513
1.4	0.449	4.6	0.712	7.8	0.501
1.6	0.920	4.8	0.689	8	0.490
1.8	0.900	5	0.764	8.2	0.478
2	0.882	5.2	0.662	8.4	0.467
2.2	0.867	5.4	0.648	8.6	0.457
2.4	0.852	5.6	0.634	8.8	0.447
2.6	0.837	5.8	0.622	9	0.438
2.8	0.824	6	0.610	9.2	0.427
3	0.810	6.2	0.597	9.4	0.418
3.2	0.796	6.4	0.584	9.6	0.409
3.4	0.782	6.6	0.571	9.8	0.399
3.6	0.769	6.8	0.560	10	0.389

Table 2: The calculated values for dose anisotropy function, FN(r, θ)

Radial	Angle to the central axis of the ^{252}CF (degree)										
distance	0	2	10	20	30	40	50	60	70	80	90
2	0.92	0.943	0.960	0.960	0.974	0.987	0.989	0.999	0.999	1.010	1.001
3	0.93	0.954	0.97	0.970	0.977	0.988	0.989	0.999	0.998	1.101	1.008
4	0.95	0.97	0.95	0.987	0.983	0.989	0.994	1.004	1.023	0.988	1.007
5	0.967	0.987	0.94	0.988	0.985	0.995	0.995	1.01	1.020	0.998	1.008
8	0.998	0.989	0.96	0.991	0.984	0.997	0.996	1.008	1.004	0.989	1.001
10	1.01	0.99	0.986	0.999	0.988	0.999	0.998	1.070	1.008	1.001	0.999

Table 3: The calculated absorbed dose in water phantom, D_N (r_0,θ_0), at distance of 1cm from the source center on the central axis and perpendicular to the central axis of the ^{252}CFsource

Absorbed dose rate (cGy / µgh) in water, D_N (r_0,θ_0), at a distance of 1cm from the source center	
Krishnaswamy(13)	1.929
Colvett et al. (14)	2.093
Yanch and Zamenhof(3)	1.900
Wierzbicki et al. (15)	1.880
Rivard et al. (4)	1.873
Paredes et al. (12)	1.916
Ghassoun(11)	1.868
Current Research	2.075

Table 4: The constant of dose rate in ^{252}CF source, Λ_N, in comparison with other results reported

The constant of dose rate,	Λ_N, ^{252}CF source (cGy/Uh)
Rivard et al. (4)	5.676
Ghassoun(11)	5.579
Paredes et al. (12)	5.719
Current Research	6.014

is consent with other reported results, 0.3348, 0.3350, 0.3300 $cm^2/\mu gh$, respectively by Ghassoun *et al.*[11],Paredes *et al.*,[12] and Rivard *et al.*, [4].The calculated absorbed dose rate in the water phantom, at a distance of 1cm from the source centeron the central axis and perpendicular to the central axis of ^{252}CF,, are compared to other results (Table 3)[3,4,11-15]. The minimum calculated difference of is -0.86% with the measured data of Colvett *et al.*[14] and the greatest difference was found to be +11.81% with Ghassoun *et al.*[11].

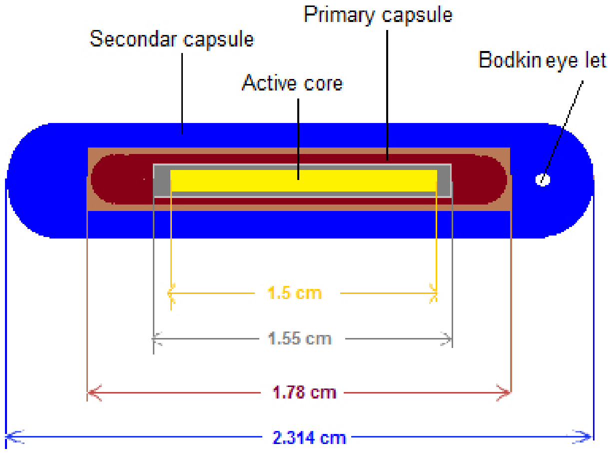

Fig. 1: The dimensions and materials of ^{252}CFneutron source

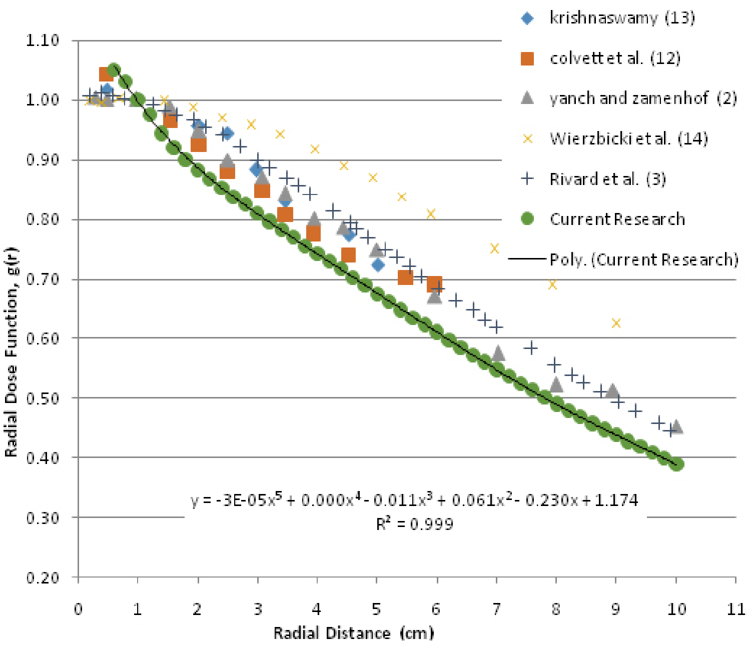

Fig. 2: Anisotropy functions in ^{252}CF source compared with other result

The constant of dose rate in ^{252}CF source, Λ_N, are compared in Table 4 with the previous results. The calculated constant of dose rate in ^{252}CFsource, Λ_N, is equal to 6.014 cGy/Uh and was consistent (+5.95%) with5.676 cGy/Uh,reported by Rivardet al. (4). Since the constant of neutron dose rate depends on the geometry of the source and the adsorbent environment, the constant of dose rate might change due to the geometrical characteristics, the source capsule material, phantom material, or on the clinical basis the tissue material the source has been planted in[4, 11, 12]).

Among the reviewed studies only Colvett *et al.*,[14] is based on the measurement; Colvett's dose radial function is reported to a depth of 6 cm and is in good agreement with the results of the present study[14]. The cross section of the neutrons interaction with material highly depends on the weights of the elements (especially hydrogen content), the density of the target material and also the neutron spectrum used.

The unacceptable differences between the constant of dose rate and anisotropy functions reported in Wierzbicki *et al.*[15] compared to the present and other studies is mainly due to the difference in the source radiation neutron spectrum, hydrogen content and water density as the corresponding tissue. For example, Krishnaswamy[13] considered the watt energy spectrum for the ^{252}CFsource, a=0.9756 and b=2.926,whichis 0.88 and 2 in this study.On the other hand, for example in the Colvettet al.,[14] and Krishnaswamy[13] studies absorbed dose is considered in tissue-equivalent material instead of water in which densities are 1.06 and 1.00 g/cm^{3}respectively, containing 10.3% and 10.5% hydrogen.Wierzbicki *et al.*[15] and Yanch and Zamenh of[3] used water phantom with density of1.00g/cm^3, containing 11.2% hydrogen.

Another reason for these discrepancies is probably taking into account different values for the neutron source flux. For example, the neutron flux considered in this study is equal to 2.314×10^6 n/¼gs, while Colvett *et al.*[14] used the flux of 2.339×10^6 n/¼gs. Based on these results appropriate correction factors must be applied to make different factors equivalent such as the energy spectrum, different materials in report environment, absorbed dose and neutron flux.

CONCLUSION

The presentstudy calculated dosimetry parameters of^{252}CF source as recommended by AAPM, TG-43 [9] for implementing Monte Carlo simulation (MCNPX)[16]. Air kerma strength computed for ^{252}CF source, SKN, is equal to 0.3450 U/μg, absorbed dose rate in water at a distance of 1cm from the source center, DN (r_0, θ_0) is 2.075 cGy/μgh and the constant of dose rate, ΛN, is 6.014 cGy/Uh; these calculated data are in good agreement with simulation studies and measurements have been reported . The calculated values are operative with a high confidence in brachytherapy software designing and numerical calculations in brachytherapy units with a ^{252}CF source.

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