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A Precise Assessment of Gamma Ray Absorbed Dose Rates Using Hyper Pure Ge, Thermoluminescence and Monte Carlo Simulation Methods in Girls' Branch, King Saud University, Riyadh

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In the present work, gamma spectroscopy technique has been applied to determine the outdoor and indoor absorbed dose due to background gamma radiations outside and inside King Saud University campus in Malaz (Girls' *Branch*). Thermoluminescence technique along with a theoretical model based on Monte Carlo simulation and point Kernel dose equation have been applied to determine the gamma rays indoor absorbed dose rates in the same place. The absorbed dose of each gamma line was calculated and estimation of the total absorbed dose for the detected gamma lines were obtained and found to be 44.73 and 55.03 nG/h for outdoor and indoor respectively. Furthermore, TLD dosimeters were employed for two consecutive 60 days measuring periods. The obtained TLD results provide an estimation of outdoor and indoor average dose rates of values 45 and 63.21 nGy/h respectively. Additionally, the mean value of gamma rays indoor absorbed dose rates obtained by means of the applied theoretical model was 58.49±2.1 nGy/h.

Keywords: Absorbed dose; HPGe; TLD Dosimeters

Introduction

For the purpose of determining the contamination levels in the environment and assessing the radiation dose to man due to man-made sources of radioactivity, as well as for comparing man-made radiation with natural background radiation, a set of coordinated surveillance program all over the world has been developed. [1- 5]

The majority of radionuclides of significance in environmental surveillance can be determined by γ-spectrometric measuring methods. By the use of highly effective detectors, γ-spectrometric activity determination has become the dominant method in nuclear activity surveillance. In conditions of increase in the environmental contamination, its importance is growing considerably.

Measurements of natural background radiation are important in the assessment of exposure of the population due to radionuclides. Uranium and thorium are widely distributed in the earth's crust. Except in geologically recent sediments, there is equilibrium between parent and daughter nuclides in decay chains.

Natural indwelling radioactivity could lead to the increase of human absorbed dose due to the γ ray emitters in floor, wall and ceiling materials. Therefore, theoretical models were proposed for adopting conversion factors for different exposure geometries and the radionuclide distribution. Therefore, it is so difficult to have certain indoor conversion factors analogous to those of outdoors

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for estimation of the human absorbed dose rates. Calculation of the conversion factors depends on the dwelling design and constructing materials [2, 3 ,6].

A precise analysis for the radioactive absorbed dose conversion factors has been demonstrated by Koblinger [7] and Stranden [3] taking into considerations the position inside the room, dimension of the room, the high variability of the wall thickness and density. Risica (2001) [8] has improved the calculations, making use of the γ line 1461 keV for ^{40}K, weighted mean energy of 810 keV for ^{238}U and the 2614 keV γ line for ^{232}Th. Allam (2009) [9] has calculated the indoor absorbed dose conversion factors making use of 24 γ lines in the ^{238}U chain, 32 γ lines in the ^{232}Th chain and the 1460 keV γ lines for ^{40}K applying the same markkanen's room model dimensions which are 5 x 4 x 2.8 m and floor, wall and ceiling thickness of 0.2m.

Accordingly, the aim of the present work is to obtain precise values of the outdoor and indoor absorbed doses using γ-spectroscopy and thermo luminescence techniques, taking into considerations all the detected γ lines attributed to the natural radioactive series and ^{40}K along with the application of a convenient theoretical model to evaluate the dependence of the γ indoor absorbed dose conversion factors on the configuration of the room.

Experimental Procedure

The experimental arrangement consists of a liquid nitrogen cooled ORTEC p- type hyper pure germanium detector of an active volume of 156.6 cm^3 with a photo- peak efficiency ≈ 38% relative to a 3in x 3in NaI(Tl) detector and an energy resolution of 1.85 keV at 1.33 MeV γ line. The energy calibration of the spectrometer was performed using the well-known standard sources (^{22}Na, ^{88}Y, ^{137}Cs, ^{226}Ra and ^{241}Am) to cover an energy range from 60 keV to 3 keV. To evaluate the true intensities of the background gamma radiations at different energies, it is necessary to have an accurate detector efficiency as gamma radiations hit the surface of the detector from all directions. Also, it is very well known that the detector efficiency is a function of the three main processes that characterize interaction of radiation with matter. Accordingly, to overcome these difficulties in determining realistic absolute efficiency in gamma ray background measurements, ^{226}Ra, ^{137}Cs and ^{60}Co sources were placed around the detector at several positions at a constant distance (25 cm) from the detector in order to find out the dependence of the detector efficiency on the position or angle and gamma ray energy. It is found that the detector efficiency at 352 keV (^{226}Ra), 662 keV (^{137}Cs) and 1332 keV (^{40}K) has its maximum value in front of the detector window while its minimum value is behind the detector or at 180^o from the detector window at the three considered gamma ray energies.

Following the methodology given by El Kameesy et.al [10, 11], the average value of the efficiency dependence on position or angle was taken as a favored method for the estimation of the absolute efficiency of the detector in background measurements. In pervious investigations [10, 11], only the 1332 keV gamma line was utilized in the treatment ignoring that the efficiency dependence on position or angle depends on the gamma ray energy which is directly affected by the three main interaction processes of gamma radiation with matter. To overcome that difficulty, the aforementioned gamma lines (352, 662, 1332 keV) were ultilized to take into consideration the domination of each of the three main processes at different gamma ray energies.

As a consequence, the correction factors at 352, 662, and 1332 keV were found to have the values 0.76±0.04, 0.82±0.03 and 0.88±0.05 respectively. The average value of these corrections (0.82±0.03) was taken as an overall correction factor for the whole range of γ rays in background measurements.

In the 2nd technique, the thermoluminescence phenomena have been utilized where 50 commercial LiF: Mg, Ti (TLD100) phosphors were used with two thermoluminescent elements in each. The dimensions are normally 3.175 mm x 3.175 mm x 0.883 mm encapsulated between two sheets of Teflon 0.0635mm thick. The readings were achieved using TL-reader (Harshaw- 4000 TLD) which was connected to a pico-processor glow curve. The maximum tray temperature was 400^{o}C, with a heating rate of 5^o C/sec.

Further investigation of the external indoor dose rate has been performed utilizing the activity concentrations of the internal walls, ceiling and floor of the location under investigation. Therefore, samples were collected from each of them and thoroughly mixed together to form one sample. The samples were taken at depths 0-10 cm from the surface. The samples were sieved through 2 mm mesh size. The collected samples were dried in a vacuum drier at 105°C, weighted and transferred to 300 ml Marinelli beakers and then stored for four weeks to reach the known state of secular equilibrium. Gamma ray analysis was carried out by means of the aforementioned HPGe detector. The specific activity of ^{238}U (^{226}Ra) was determined using the 295.1(19.2%), 352(37.1%) from ^{214}Pb, 609.3 (46.1%) and 1120.3 (30%) keV gamma rays from ^{214}Bi. ^{232}Th specific activity was determined from the gamma rays 238.6 (43.6%) from ^{212}Pb, 338.4 (12%), 911.2(29%), 969(17%) keV from ^{228}Ac and 583.6(86%), 2614 (36%) keV gamma rays from ^{208}Tl. The specific activity (Bq/Kg) is given by: [12, 13]

$$A = \frac{\mathrm{Np}}{\mathrm{tc}.I\gamma(E\gamma).\varepsilon(E\gamma).M} \quad (1)$$

Where N_p is the net counts in a given peak area at energy E, $\varepsilon(E\gamma)$ is the detection efficiency at energy E, t_c is the counting time in seconds, I_γ (E_γ) is the branching ratio of the gamma line and M is the mass in Kg of the measured sample.

Results and Discussion

Gamma dose rate measurements using a gamma spectrometer

Measurements of the γ background radiation in a wooden port-cabin situated in the King Saud University courtyard have been repeated several times during a period of three months. The time of each measurement was 90 hours. It was found that the fluctuations in the measurements do not exceed the inherent statistical fluctuations of the measurement itself.

In the spectra, well-resolved γ lines were observed. The energy of each line, as well as its respective photo peak intensity and its parent radioactive isotope, are tabulated in Table (1). These gamma lines are belonging to the ^{238}U-series, ^{232}Th- series and ^{40}K.

The absorbed dose rate for each γ line was calculated using the equation [10]:

$$\text{Dose rate} = \frac{\emptyset E(\frac{\sigma}{\dot{\rho}})}{\frac{62500\mathrm{MeV}}{\mathrm{g}}.\mathrm{mrad}} \quad \text{mrad/sec,}$$

$$\varphi = \frac{\text{I abs}}{\text{Area of the detector}} \quad (2)$$

Where: φ, E, σ, ρ̇ are the flux density (photons/ sec.cm^2), energy, absorption coefficient and the density of the media respectively.

Considering the density of tissue to be approximately equal to that of water and using the data of the absolute counts, the surface area of the detector and the energy absorption coefficient curve [14] the outer door dose rate was calculated and the obtained results are shown in Table (1). The sums of the outdoor absorbed dose rates in units of nGy/h of the detected γ lines belonging to the elements of each of the two naturally occurring series as well as that for ^{40}K obtained in the present work are given in Table (2) along with those obtained by the same procedure for the indoor absorbed dose inside a room at 1m height from the floor. The central point (1m above the floor) was considered as the origin p (0, 0, 0). Farther measurements were performed regularly at 1m and 2m distance from the central point. The deviation in the calculated values of the absorbed dose rates doesn't exceed 13% from that at p (0, 0, 0).

It is clear from Table (2) that the total dose rates inside and outside the building indicate that the contribution of wall constituents (bricks and concrete) to background radiation is about 23% of the total dose rate. Also, the same high contribution of the uranium series inside and outside King Saud University was observed.

A rather well-known method has been applied to evaluate the external indoor gamma dose rate. In this method, the contribution of natural radionuclides to the absorbed rates depends on varying concentrations of radionuclides in the wall concrete samples [15, 16].

The indoor gamma absorbed dose rate (D) can be calculated by using the following equation;

$$D = C_1 A_{Ra} + C_2 A_{Th} + C_3 A_K \quad (3)$$

Where A_{Ra}, A_{Th} and A_K are the activity concentrations of ^{226}Ra, ^{232}Th and ^{40}K, respectively in (Bq/Kg) and C_1, C_2 and C_3 are their dose conversion factors (nGy/h per Bq/Kg).

To take into consideration the exposure geometries, building materials, and position in the room, several works have been performed [6, 7, 17 and 18] to determine the dose conversion factors making use of the point Kernel technique expressed by the following relation [19, 20].

$$D(r, \Gamma) = A\Gamma_0 \int^V e^{\mu b\, rb - \mu a\, ra} / r^2 \quad dV \qquad (4)$$

Where D (r, Γ) is the dose rate (nGy/s); A is the activity concentration (MBq/m^3); Γ is the specific gamma emission rate ($C.K\ G^{-1}.m^2.\ MBq^{-1}.h^{-1}$); μ_b and μ_a are linear energy absorption coefficient (m^{-1}) for wall material and air respectively. r_b, r_a and r are the gamma rays travel distance in wall, air and the total distance between the point source and calculation point, respectively.

In a previous work [8] the conversion factors C_1, C_2 and C_3 at the center point of space of the room were determined where their value are found, to great extent, similar to those predicted by Isinkaye and Emelue, (2015) [21] based on the formula given by the European Commission, 1999 [22]. The applied approach proved that there is a significant variation corresponding to building composition (18%), position inside the room (17%) and density of the building materials (up to 32%). In the present work, calculations of the estimated indoor gamma dose rate are performed following the method applied in reference [8] at the central point of the room at 1m from the floor.

Taking into consideration the experimentally adopted mean values of ^{226}Ra, ^{232}Th and ^{40}K activity concentrations (16.5, 12.7 and 320 Bq/Kg respectively), the adopted, conversion factors are found to be close to those obtained previously [8, 21, 23 and 24] for a standard room of dimensions 4m x 5m x 2.8m and wall, ceiling and floor thickness of 0.2 m. The conversion factors have been found to be 0.917, 1.214 and 0.076 for ^{226}Ra, ^{232}Th and ^{40}K respectively. As a consequence, the values of gamma indoor absorbed dose are ranging from 52.24 – 68.26 nGy/h with an average of 58.49±2.8 nGy/h which is very close to the world average value (57nGy/h) [25]. Therefore, the percentage variation in the value of gamma indoor absorbed dose due to position is up to about 15%.

Gamma dose using LiF: Mg,Ti (TLD100) Dosimeter

The measurements of the γ dose rate using LiF: Mg,Ti (TLD100) dosimeter was performed after studying its dosimetric properties taking into account its energy dependence on γ ray energies [3]. The chips were annealed for 45 minutes in an oven at 400^{o}C and the zero readings of the chips were checked. The dose response and linearity were studied by irradiating the annealed LiF: Mg,Ti dosimeters to different doses in the range 10- 200 μGy using a dose rate calibrated Cs^{137} source.

Figure (1) shows the calibration results which prove that the phosphor response is linear along the chosen range. Calculated detection threshold [26] of the TLD chips was 8 μGy. When using different sensitivity corrections and background subtraction, the LiF: Mg,Ti pellets are suitable for environmental measurements when the measuring time is long enough and the environmental factors (temperature, humidity and UV light) are under control. The average temperature at the time of measurements was 24^{o}C and the mean relative humidity was less than 30%. Also, phosphors must be stored in good dark plastic envelopes to prevent sunlight. Under these conditions, the effect of environmental factors is negligible. The measurements were performed regularly at 1m and 2m from the central point of the room (1m above the floor) within a horizontal plane.

To use the tested TLD100 dosimeters in environmental outdoor dose rate evaluation, 50 annealed dosimeters were used and divided into two groups, each of 25 phosphors. One group input on air (1m above floor surface) for one month, and the 2nd group for two months. Taking into account the thermal fading. The results of the present study provide an estimation of the average dose rate of 45 (nGy/h) for outdoor measurements.

The indoor dose rate for the King Saud University environment was experimentally estimated and found to vary between 60.7 nGy/h and 72.4 nGy/h with an average value equal to 63.21 nGy/h where the TLD100 dosimeters were situated at different

places inside the room 1m above the ground level of the room.

change in the indoor absorbed dose values from that at the central point reaches about 20%.

The average indoor absorbed dose rates in the air that are previously reported to show a range of 23 – 120 nGy/h with most falling in the range of 60 – 95 nGy/h. The average value of the present work (63.21 nGy/h) is thus nearly close to the worldwide average (70 nGy/h) [27].

The adopted value at the central point 1m above the floor was found to be 61.1±3.2 nGy/h. Therefore the

In general, the obtained external indoor absorbed dose rate results of the gamma spectroscopy technique that depends on basic and/or fundamental knowledge along with those obtained by the thermoluminescence technique that depends on calibrated known radioactive sources support the validity of the simple Monte Carlo simulation method in the indoor dosimetry domain.

Table (1): Intensity (I=photons/ sec) and the outdoor absorbed dose (PGy/h) of the measured background spectra

	E_γ	Element	I	A.Dose
1	92	^{234Th}	3.5310	5.949
2	186.2	^{226}Ra	4.7648	20.196
3	209.5	^{228}Ac	4.1818	21.393
4	238.6	^{212}Pb	20.5922	121.905
5	241.9	^{214}Pb	7.1033	44.532
6	270.2	^{228}Ac	1.7787	12.393
7	277.4	^{208}Tl	0.1160	0.837
8	295.2	^{214}Pb	18.4334	140.364
9	300.1	^{212}Pb	1.1336	8.865
10	328.3	^{228}Ac	1.5655	13.671
11	338.7	^{228}Ac	5.184	47.646
12	352.3	^{214}Pb	38.5086	366.138
13	409.9	^{228}Ac	0.4778	5.337
14	438.9	^{228}Ac	0.7600	9.234
15	463.3	^{228}Ac	1.3086	16.776
16	487.2	^{214}Pb	0.3218	4.338
17	511	^{208}Ti +an	25.8453	367.101
18	562.6	^{228}Ac	0.6706	10.44
19	583.1	^{208}Tl	17.7213	285.993
20	609.3	^{214}Bi	81.5357	1334.529
21	662.6	^{214}Bi	10.824	192.654
22	665.5	^{214}Bi	2.0416	36.495
23	702.7	^{214}Bi	2.1199	38.763
24	727.3	^{214}Bi	3.9268	74.394
25	741.5	^{214}Bi	0.43907	8.478
26	754.1	^{214}Bi	0.22085	4.338
27	768.3	^{214}Bi	12.6049	252.261
28	771.8	^{218}Ac	2.2659	45.81
29	786.1	^{214}Pb	3.1837	65.007
30	795	^{228}Ac	2.5383	52.398
31	806	^{214}Bi	23.4899	479.439
32	860.4	^{208}Ti	3.6469	79.182
33	911.2	^{228}Ac	28.3862	631.647
34	934	^{214}Bi	9.3926	214.236
35	964.4	^{228}Ac	19.232	454.347

36	1000	^{228}Ac	3.3566	81.675
37	1109	^{228}Ac	0.6415	17.37
38	1120.3	^{214}Bi	76.9060	2103.777
39	1155.2	^{214}Bi	8.4141	367.101
40	1238.1	^{214}Bi	42.3	976.266
41	1280.9	^{214}Bi	8.3451	199.485
42	1377.7	^{214}Bi	29.0991	10.44
43	1385.3	^{214}Bi	5.9092	285.993
44	1401.5	^{214}Bi	10.5271	348.741
45	1408	^{214}Bi	20.22	672.075
46	1461.2	^{40}K	402.7991	13923.38
47	1509.2	^{214}Bi	27.0237	926.937
48	1580.8	^{228}Ac	9.2903	334.35
49	1588.3	^{228}Ac	8.7578	317.043
50	1592	^{214}Bi	3.8751	140.913
51	1620	^{214}Bi	4.8689	179.775
52	1630.7	^{228}Ac	4.6711	173.619
53	1661.2	^{214}Bi	9.86	373.329
54	1729.6	^{214}Bi	25.8	1017.09
55	1764.5	^{214}Bi	139.7556	5620.608
56	1838.8	^{214}Bi	2.8764	116.397
57	1847.4	^{214}Bi	18.4911	750.798
58	2103.4	^{208}Tl	14.3089	636.993
59	2119.5	^{214}Bi	9.12	409.104
60	2204.2	^{214}Bi	41.8667	1877.994
61	2293.4	^{214}Bi	2.3675	110.808
62	2447.8	^{214}Bi	13.1044	652.788
63	2614.5	^{208}Tl	103.7556	5520.456

Table (2): Outdoor and indoor absorbed dose rates (nGy/h) of the present work

Series	Present results	
	Outdoor	Indoor at the origin p (0, 0, 0).
U- series	21.51	18.4
Th- series	9.27	14.04
^{40}K	13.95	22.59
Total	44.73±3.2	55.03±3.4

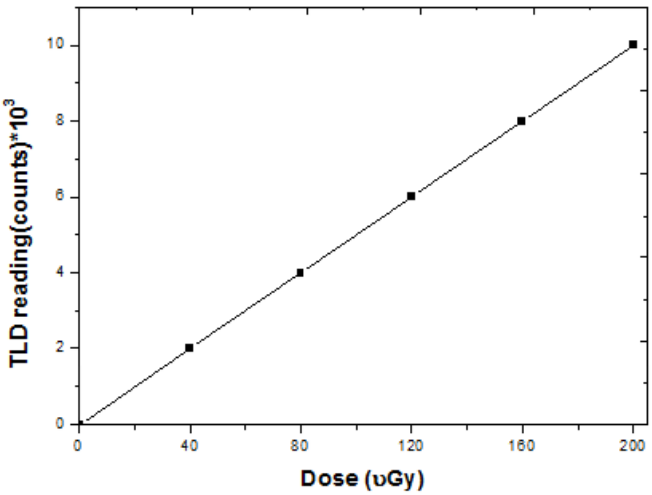

Fig. (1): The TL-Response of LiF: Mg, Ti (TLD100) phosphor as a function of radiation exposure from 40 up to 200 µGy

Conclusion

The present work is conducted to undertake an extensive study of both outdoor and indoor gamma rays absorbed dose rates. Great effort has been devoted to determine the indoor gamma rays absorbed dose by means of HPGe detector and TLD100 dosimeter. In the former method, the calculations are based on the individual gamma ray flux, energy, attenuation and absorption coefficients along with the efficiency of HPGe detector for each gamma ray. The latter method is based on a well-known sensitive TLD100 dosimeter. Furthermore, a theoretical approach has been applied to evaluate the indoor gamma rays absorbed dose where the position from walls, floor and ceiling is taken into consideration at different horizontal places inside the room 1m above the floor. The theoretical calculations are based on the so called point- Kernel dose equation for homogeneous distribution of gamma ray emitters in a medium of constant density. The mean values of the indoor gamma rays absorbed dose due to the theoretical prediction and TLD100 dosimeters are 58.49±2.8 and 63.21±3.2 respectively, accompanied by variations that do not exceed 20%. In the experimental method based in equation 2, the adopted values of the indoor gamma rays absorbed dose are comparable with those obtained by the remaining aforementioned methods.

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