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Radioactivity Investigation of Sand from the Northern Region of Tlemcen-Algeria, Using Well-Shape NaI(Tl) Detector

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Abstract.

The presence of natural radioactivity in sand and other building materials results in internal and external exposure to the general public. Therefore, it is desirable to determine the concentration of naturally occurring radionuclides. Sand is one of the main components in building construction beside cements, granites and bricks. Thus, this research has been carried out in order to investigate the levels of natural radioactivity and associated radiation hazard in some Algerian sand. The natural radioactivity due to the presence of ^{226}Ra, ^{232}Th and ^{40}K in sand samples used as building materials in Tlemcen province - Algeria was measured by gamma spectrometry using NaI(Tl) scintillation well-shaped detector. In this context, sand samples were collected from four different locations of northern areas of Tlemcen city, Behira seabeach sand, Targa sand, Boukdasen sand, Sidi Bourzin sand. The measured activity in the sand samples ranged from 4.70 to 7.85 Bq.kg^{-1}, 0.80 to 3.00Bq.kg^{-1} and 22 to 53 Bq.kg^{-1} for ^{226Ra}, ^{232}Th and ^{40}K, respectively. The concentrations of these natural radionuclides were compared with the reported data for other countries and were found significantly lower than the worldwide average as reported by United Nations Scientific Committee on the Effects of Atomic Radiation (1,2). Radium equivalent activities were calculated (9.25 to 16.39) Bq·kg^{-1}for the analyzed samples to assess radiation hazards arising due to the presence of these radionuclides in the samples. Most of the calculated radium equivalent activities are lower than the limit set in the OECD report at 370Bq.kg^{-1}(3). The measured representative level index values for the investigated samples varied in the range (0.068 to 0.118) Bq.kg^{-1}. External and internal hazard index (H_{ex},H_{in}), the specific dose rates indoor (D) and the annual effective dose (DE) due to gamma radiation from building materials was calculated.

Keywords. Sand, natural radioactivity, gamma radiation, absorbed dose, radiation exposure, Potassium, Thorium, Uranium, NaI(Tl) Detector, Tlemcen.

1.Introduction.

Naturally occurring radioactive materials is a widespread substance that can be found everywhere in the environment including soil, rocks, water, air and also tissues of living things. There are no ways to avoid the presence of natural radionuclide since it presence from the formation of earth. Sand is one of the main components in building constructions are known to contain with naturally occurring radioactive materials. Sands are mineral deposits formed through weathering and erosion of either igneous or metamorphic rocks. Natural radioactivity in sands contributed radiation dose to dwellers that originate from ^{238}U, ^{232}Th and their progeny and ^{40}K. For building construction purposes, sand is not just drawn out from river and sea but it is also extracted from the abandoned mines. As these radionuclides (^{238}U,^{232}Th, and ^{40}K) are not uniformly distributed, the knowledge of their distribution in soil, sand and rock play an important role in radiation protection and measurement(1,2). The present work investigates the concentrations of radioisotopes such as ^{232}Th, ^{226} Ra and^{40}K, in beach sand and carrier sand samples from extreme northern of Tlemcen province in Algeria, also this study is estimate the radiological hazard, the radium equivalent activity, the external hazard index, the absorbed dose rate, and the effective dose rates were calculated and compared with internationally approved values. The radiological survey is important for each country, to establish a data base for environmental purposes, and for future variation in radiation level due to one reason or another. The radiological impact from the natural radioactivity is due to radiation exposure of the body by gamma-rays and irradiation of lung tissues from inhalation of radon and its progeny. From the natural risk point of view, it is necessary to know the dose limits of public exposure and to measure the natural environmental radiation level provided by ground, air, water, foods, building interiors, etc., to estimate human exposure to natural radiation sources .Low level gamma-ray

spectrometry is suitable for both qualitative and quantitative determinations of gamma-ray-emitting nuclides in the environment. The concentration of radioelements in building materials and its components are important in assessing population exposures, as most individuals spend 80% of their time indoors. The average indoor absorbed dose rate in air from terrestrial sources of radioactivity is estimated to be (70nGyh^{-1}).Great attention has been paid to determining radionuclide concentrations in building materials in many countries .In Algeria the information about the radioactivity of building materials is limited (2,3).

2. Materials and Method.

2.1. The Study Area.

Northern region of Tounan is the area of interest in the present study, which is located at Tlemcen district in Algeria. The study area is situated at about 80 kilometer away and to the north of Tlemcen. The locations of sample collection are shown in **Figure 1**. The sea beach sand samples were collected from the Behira beach area (Oulad ben ayed) while the other sand samples were collected from sand carriers adjacent the locality around the Tounan-Ghazawat region (Tlemcen).

Figure 1: The northern region of Tlemcen province – Algeria

2.2. Collection and Preparation of Samples.

For the measurement of natural radionuclides at the study area, a total of 25 sand samples were collected from within and around the sea beach areas and sand carriers, for our four kinds sand samples. Each sample was collected from each location and each of the samples was placed in plastic packet and transported to the laboratory. The collected sand was first weighed, then dried at 100- 110° C in an oven for 24 hours and there after ground into a fine powder with a grinder and collected after passing through a 2 mm mesh size. Thus, homogenized sample was transferred to sealable cylindrical plastic container with dimensions fit to the well (hole) of our detector NaI(Tl),and all the sample containers were sealed tightly, the samples were stored for four weeks prior to counting, allowing establishment of secular equilibrium between the long lived ^{238}U, ^{232}Th and their decay products. The containers were full filled for uniform distribution of ^{220}Rn and ^{222}Rn daughter products and to avoid accumulation at the top.

2.3. Experimental Procedure.

Before determining of the radioactivity concentration in samples, an empty cylindrical plastic container was counted for 24h under identical geometry to measure the background spectrum in the laboratory of measurement. This spectrum is necessary to establish a high confident background level to be used for determination of the specific activities of the analyzed samples. To determine the radioactivity concentration in the sand samples, the sample was placed on the well-shaped 2x2 NaI(Tl) detector and counted for the same counting time (24 h), and its spectrum was stored in a PC-based multichannel analyzer (MCA). Radiometric measurements were performed for qualitative identification as well as quantitative determination of radionuclides present in sand. The gamma-spectrometric measurements were performed with NaI (Tl) well detector 2x2 inch with its electronic circuits Canberra Inc.(4). The emphasis was on the determination of specific activity concentration of ^{226}Ra,

^{232}Th and ^{40}K. The analysis of ^{40}K was based upon its single peak of 1460.8keV, whereas the analysis of ^{226} Ra and ^{232}Th depended upon the peaks of the daughter products in equilibrium with their parent nuclides, the concentration of ^{226}Ra was determined from the average concentrations of ^{214}Pb (352keV) and ^{214}Bi (609, 1120 and 1765keV), and that of ^{232}Th was determined from the average concentrations of ^{212}Pb (239keV), ^{208}Tl (583,2615keV) and ^{228}Ac (338.3,911,969.11keV) in each sample under study (**Table 2**). In the uranium series the decay chain segment starting from radium (Ra) is radiological the most important and, therefore, reference is often made to radium instead of uranium.

2.4. Method of calculations.

The efficiencies for each radionuclide were calculated and used to estimate the activity concentration of each of the radionuclide in the samples. The detection efficiency of the system was determined using the several calculations including linear attenuation coefficient, geometric and intrinsic efficiencies for well type 2x2 NaI(Tl). The well shaped detectors are of higher efficiency for the same volume of detector. This particular characteristic allows almost a100 percent efficiency (so called 4π geometry) for low gamma-emitting test sources that can fit the well shape (5, 6).

2.4.1. Calculating of detector counting efficiency Dε.

There are three factors, G, I and M, their affect the efficient absorption of the photons emitted by the source. Their product is the detector counting efficiency **Dε.**

$$\mathbf{D\varepsilon = G \times I \times M} \qquad \mathbf{(1)}$$

G = fraction of all space that the detector subtends. Unless the detector completely surrounds the source, the geometrical solid angle factor is less than 1.

I = fraction of the photons transmitted by the intervening materials that reach the detector surface. There are losses due to absorption by material in the path of the photon. Air, detector housing materials and light reflectors around the detector are possible absorbers.

M = fraction of the photons absorbed by the detector. The detector material is not always sufficiently thick to stop the radiation.

In our well detector, hence the sample placed in the hole of detector, we have specific conception

for dealing with this fractions. The dimensions of 2x2 NaI(Tl) detector in 2-inch diameter with 2 inches high(crystal) and a 0.75 inch diameter by 1.44 inch deep well (hole), for these properties of well-shape detector the previous fractions seen as following:

To calculate the fraction of space not subtended and then to subtract that value from 1 to get the fraction G subtended. The fraction not subtended is the area of the hole of 0.75 inch diameter at the end of the well a distance of 1.44 inches. The (absolute) total efficiencies for a right cylinder and a well-type are presented as functions of the source position and the photon energy when the sources are located on the surface, the total efficiencies for low energy photons are 0.5 and ~ 1, respectively. This means every photon incident on the detector produces an output pulse considering the solid angles of both geometries (2p for right cylinder, ~ 4p for well type), regardless of the energy deposited (7).

$$\mathbf{1 - G = (\pi r^2) / (4\pi R^2)} \quad \text{where:}$$

$\mathbf{\pi r^2}$ = area of hole in detector face, and $\mathbf{4\pi R^2}$ = area of sphere with a radius equal to the distance from the source to the hole.

1 - G = (π x 0.375 inch x 0.375 inch) / (4 x π x 1.44 inch x 1.44 inch) = 0.017, and G = 0.983

This detector subtends or intercepts 98% of all space (A great advantage of the well geometry is, of course, the large solid angle (~ 4π sr), which leads to a high efficiency.

To calculate **I** we have $\mathbf{I = \exp^{-(\mu l \times d)}}$ where:

μ_l = the linear attenuation coefficient for gamma ray in aluminum.

d = 0.025 cm (0.010 inch), the thickness of the aluminum container.

The fraction of the photons absorbed by the detector M is calculated by subtracting the fraction that pass through the detector from 1:

$$\mathbf{M = 1 - \exp^{-(\mu l \times d)}}$$

μ_l = the linear attenuation coefficient for gamma ray in NaI(crytal).

d = 1.422 cm (0.56 inch), the minimum distance traveled in NaI(Tl) at the bottom of the well,

2.4.2. Linear attenuation coefficient calculations.

For calculation detecting efficiencies we try to find the values of μ_l for each aluminum and NaI(crytal).Firstly we investigate the references in this item and do an comparison between them to take the main values of its, then we calculate the μ_l for mixture NaI using the following formulas:

$$\mu_{m(NaI)} = \sum \mu_i . W_i = (\mu_1 . W_{1)Na} + (\mu_2 . W_{2})I .$$

$$\mu_{l\,(NaI)} = \mu_{m(NaI)} . \rho \quad \text{where } \rho \text{ is the density of NaI } = 3.7 g/cm^3$$

the calculation result of $\mu_{l\,(NaI)}$ table (1) are compared with that values from references (6,7,8), to view the fit value with the graph of linear attenuation coefficient, this lead us to chose proper solution to this calculations. Finally we calculate the **Dε** of the our detector for each gamma energy in the cement samples under study table (2).Using above work, the activity concentrations for the ^{40}K, ^{232}Th, ^{238}U and ^{226}Ra radionuclides were calculated using the detected photopeaks in the spectra.

$$A(Bq/Kg) = N/(T.\varepsilon.I.W) \quad (2)$$

Where **N** is net peak counts (background subtracted), **T** is the measured time (sec.), **ε** is the efficiency of detector, **I** is the branching ratio of gamma emission for decay mode and W is the sample weight.

2.4.3. Radiological Hazard Assessment.

Radium Equivalent Activities:

The radium equivalent activity is an index that represents the specific activities of ^{226}Ra, ^{232}Th and ^{40}K by a single quantity which takes into account the radiation hazards associated with them. This can be calculated using the equation below.

$$Ra_{eq} = C_{Ra} + 1.43C_{Th} + 0.077C_K \quad (3)$$

Where, C_{Ra}, C_{Th} and C_K are the specific activities of ^{232}Th, ^{238}U and ^{226}Ra respectively (9). The maximum value of Ra_{eq} in building materials must be < 370 $Bq \cdot Kg^{-1}$ for safe use.

Representative Level Index Values:

Another radiation hazard index called the representative level index, used to estimate the level of gamma radiation associated with different concentrations of some specific radionuclides, can be defined as follows:

$$I_r = (1/150)\, C_{Ra} + (1/100)\, C_{Th} + (1/1500)\, C_K \quad (4)$$

Where C_{Ra}, C_{Th} and C_K are the specific activities of ^{232}Th, ^{238}U and ^{226}Ra in Bq/Kg were calculated for the samples under investigation to indicate different levels of external gamma radiation due to different combination of specific natural activities in other materials (3). This index can be used to estimate the level of gamma radiation hazard associated with the natural radionuclide in the materials.

Dose calculation:

The total air absorbed dose rate ($nGy \cdot h^{-1}$) 1 m above the ground due to the specific activities of ^{232}Th, ^{238}U and ^{226}Ra in Bq/Kg was calculated using the equation:

$$D = 0.427C_{Ra} + 0.662C_{Th} + 0.0432C_k \quad (5)$$

To estimate the annual effective dose rates, the conversion coefficient from absorbed dose in air to effective dose (0.7 $Sv \cdot Gy^{-1}$) and outdoor occupancy factor 0.2, and for indoor is 0.8 (1,2).The effective dose rate in units of $mSv \cdot y^{-1}$ was calculated by following Equation

$$DE = DTF \quad (6)$$

Where D is the calculated dose rate in (nGy/h); T is the outdoor occupancy time. The annual effective dose equivalent in the outdoor environment is given by the following equation:

$$DE\ (mSv/y) = D\ (nGy/h) \times 8760h/y \times 0.2 \times 0.7\ (Sv/Gy) \quad (7)$$

And for indoor environment the above formula are seen as:

$$DE\ (mSv/y) = D\ (nGy/h) \times 8760h/y \times 0.8 \times 0.7\ (Sv/Gy) \quad (8)$$

External hazard index (H_{ex}):

The external hazard index (H_{ex}) is a radiation hazard index defined by UNSCEAR (1) to evaluate the indoor radiation dose rate due to the external exposure to γ-radiation from the natural radionuclides in the construction building materials of dwellings. This index value must be less than unity to keep the radiation hazard insignificant, i.e. the radiation exposure due to the radioactivity from construction materials to be limited to 1.5 mSv/year or 1.0mSv/year based on the formula:

$$\mathbf{Hex = (C_{Ra}/370) + (C_{Th}/259) + (C_K/4810) \leq 1} \quad \mathbf{(9)}$$

The maximum value of H_{ex} equal to unity corresponds to the upper limit of Ra_{eq} (370 Bq/kg).

Internal hazard index (H_{in}):

The internal exposure to 222Rn and its radioactive progeny is controlled by the internal hazard index (H_{in}) which is given by:

$$\mathbf{H_{in} = C_{Ra}/ (185) + C_{Th}/ (259) + C_K/(4810) \leq 1} \quad \mathbf{(10)}$$

For the safe use of a material in the construction of dwellings, index (H_{in}) should be less than unity (10).

3. Results and Discussions.

Sand is an important construction material for houses and buildings in urban areas of Algeria. It is used for blocks and concrete manufacturing as well as for plastering the buildings walls. However, detailed information of the specific activities of ^{226}Ra, ^{232}Th and ^{40}K in sand and other building materials used in Algeria is not available in literature except the work of Amrani and Tahtat (11). This study is a continuation of our ongoing project related to the measurement of specific activity of ^{238}U (^{226}Ra),^{232}Th and ^{40}K in environmental samples from northern Tlemcen city in Algeria using gamma-ray spectrometric technique. Using the formula (2) of specific activity with values of efficiency (table 2) for well-shaped 2x2 NaI(Tl) detector for each gamma ray emitted by radionuclide under the study, the activity concentrations due to ^{226}Ra,^{232}Th and ^{40}K have been determined as present in table(3). As can be seen from table (3), the **average** activity concentrations of three radionuclides (Ra,Th,and K) are 7.85Bq.kg^{-1}, 3.0 Bq.kg^{-1} and 55Bq.kg^{-1} for beach sand sample (Bhira-Oulad Ben Aid), for Targa sand samples were (5.0, 2.6,and 31) in Bq.Kg^{-1}, for Boukdasen sand sample were (4.7,2.2,and 22) in Bq/Kg and for Sidi Bourzin sand sample were (6.4, 0.8, and42) in Bq.Kg^{-1} , these values were for (^{226}Ra**,** ^{232}Th and ^{40}K) respectively. It was important to point out that these values were not the representative values for the countries mentioned but for the regions from where the samples were collected. Radium, thorium and potassium are not uniformly distributed in soil or rocks, from which building materials are derived, but the radioactivity varies, often greatly, over a distance of some meters. The measured values of radium and thorium contents show only the average radioactivity in building materials (sand) used in province of Tlemcen. The mean values are lower than the corresponding world-wide average values which are 35, 30 and 400 Bq.kg^{-1} for ^{226}Ra**,** ^{232}Th and ^{40} K respectively (1).The specific activity of ^{40}K,^{226}Ra and ^{232}Th determined in the present study for sand have also been compared with values reported for other countries as show below. Gamma irradiation hazard indices, radium equivalent activity, dose rates and effective dose rate equivalent were calculated using the formulas (3-10), the values of these indices are listed in table (3). The experimental results of radium equivalent activity which indicate radiation hazards arising from the various sand samples studied show that the average Ra_{eq} values are below the internationally acceptable value of 370 Bqkg^{-1} .The estimated external hazard indices were all also less than unity, in order to keep the radiation hazard insignificant. The obtained D*E* values for all the analyzed sands were lower than the worldwide outdoors annual effective dose average of 0.07 mSvy^{-1}, and also below the value of 1.0 mSv/y, recommended by the International Commission on Radiological Protection (12) as the maximum allowed annual dose to member of the public. Comparison the results(C, Ra_{eq}, H,D, and DE values) of this study with values of these indices in another studies for countries around Mediterranean sea, are very closely, for examples; Libya(13,14). Egypt(15), Spain(16), Bulgaria(17). The best studies in the world to approach our results are for countries such as India(18), Bangladesh, Brazil, Jordan, SouthAfrica, Thailand, Byers Peninsula Western Antarctica(19) Malaysia(20), Ghana(21)) Ireland, Canary Islands, and India-Bombay, Bangladesh(17).

4. Conclusion.

The gamma spectroscopy method was used for assessment of the U-238 and Th-232 series and K-40 concentration in many sand samples collected from northern region of Tlemcen province in Algeria, and they are compared with the results from other countries. The average activity concentrations of ^{226}Ra, ^{232}Th and ^{40}K in the sand samples were 4.70-7.85 Bq.kg^{-1}, 0.80-3.00 Bq.kg^{-1}and 22-55Bq.kg^{-1}, respectively. The results obtained in this study compares well with data from most countries, (Egypt, Spain, India, Libya, Bulgaria, Turkey, Cameron,

Malaysia, Slovakia, Montenegro, Ireland, Canary Islands, Bangladesh, Brazil, South Africa) but also showed some variations with values from other countries. These indicated considerable variations in the activity concentration are due to the varying amounts of Uranium, Thorium and K contents as a result of different geological formations under the earth crust from where the raw material for particular kind of sand was obtained. From this research, we deduce the following:

1- Performance new table for linear attenuation coefficients of gamma ray in Al and NaI (for efficiency calculation).(**Table 1**)
2- Calculation the efficiencies of well-shaped 2x2 NaI(Tl) detector. The values of DE were showing in **table 2** and **figure 2**, these values are closely to that in **figure 3** exactly between line (1.5) and (2.5), thus our work show new line (for 2) which can add to this figure and literature.
3- This paper used gamma spectrometry to assessment the activity concentrations of ^{226}Ra, ^{232}Th, and ^{40}K, and calculate the (Ra_{eq} , I_r, H_{ex}, H_{in}, D, DE_{ex}, and DE_{in}).The mean value for all indices obtained in this study are at low level (**Table 3**) and less than world average, thus this material (sand) can be used for building in the Tlemcen (Algeria) and another regions.

Table 1: Linear attenuation coefficients of gamma ray in Al. and NaI

U-238 **(A)**	Gamma Energy KeV	Probab. Of γ Emission%	μ_1(Na) Cm^2/g W_1=0.153	μ_2 (I) Cm^2/g W_2=8.46	$\sum \mu_i.w_i$ μ_m.cal. (NaI)	μ_m.(NaI) Ref.aver.	μ_m.(NaI) average	μ_l(NaI) $=\mu_m.\rho$ $\rho = 3.67$ g/cm^3	μ_l. (Nal)for low enegy.ref.	μ_l(Al) 1/cm Ref.	μ_l(Al) 1/cm average
226Ra	186.10	03.51	0.123	0.500	0.4188	0.425	0.420	1.541	1.450	0.343	0.343
	241.98	**07.12**	0.110	0.250	0.2173	0.278	0.245	0.899	0.860	0.310	0.310
	295.21	18.15	0.102	0.163	0.1486	0.153	0.151	0.554	0.629	0.290	0.290
	351.92	03.51	0.095	0.130	0.1151	0.130	0.122	0.448	0.480	0.260	0.260
	609.31	44.10	0.077	0.079	0.0762	0.078	0.077	0.283		0.205	0.205
	768.63	04.76	0.069	0.070	0.0675	0.068	0.068	0.249		0.188	
238U											
	49.50		0.244	11.20	9.1874	10.40	9.89	36.30		0.960	
214Pb	295.10	19.24	0.102	0.160	0.1456	0.153	0.149	0.547	0.629	0.290	0.280
	325.00	37.20	0.099	0.145	0.1331	0.142	0.138	0.506	0.530	0.280	0.270
	351.93	35.34									
214Bi	609.30	46.36	0.077	0.079	0.0762	0.078	0.077	0.283		0.205	0.205
214Bi	1764.5	15.80	0.046	0.043	0.0420	0.043	0.043	0.158		0.126	0.126
214Bi	1120.3	15.1 0	0.057	0.053	0.0520	0.051	0.051	0.187		0.156	0.156
234Th	63.280	04.47	0.200	07.32	6.016	6.100	6.05	22.20	20.94	0.710	0.710
	92.370	02.60	0.155	2.600	2.1477	2.220	2.18	8.000	7.400	0.490	0.450
235U	185.70	57.25	0.125	0.510	0.4360	0.425	0.43	1.578	1.460	0.345	0.345
	143.70	10.96	0.128	0.600	0.5096	0.516	0.51	1.872	2.400	0.385	0.385
(B)	**232Th**										
228Ac	338.30	11.40	0.097	0.151	0.1379	0.140	0.139	0.510	0.500	0.274	0.265
	911.20	27.70	0.065	0.060	0.0589	0.062	0.060	0.220		0.174	0.174
	969.80	05.20	0.062	0.058	0.0565	0.059	0.058	0.213		0.171	0.171
212Bi	727.00	11.80	0.075	0.072	0.0709	0.071		0.257		0.197	0.193
212Pb	115.18	0.62	0.146	1.800	1.4913	1.500	1.495	5.487	4.300	0.430	0.430
	300.09	03.40	0.101	0.162	0.1475	0.153	0.150	0.550	0.600	0.285	0.275
	238.60	43.60	0.117	0.150	0.2200	0.190	0.205	0.752	0.866	0.293	0.300
208Tl	583.20	84.50	0.079	0.080	0.0731	0.081	0.077	0.283		0.215	0.215
	2615.0	99.79	0.038	0.039	0.0378	0.038	0.038	0.140		0.054	0.054
228Ra	338.32	11.26	0.097	0.151	0.1372	0.140	0.139	0.510	0.500	0.274	0.274
	911.07	26.60	0.065	0.060	0.0589	0.062	0.060	0.220		0.175	0.175
	969.11	16.23	0.062	0.058	0.0565	0.059	0.058	0.213		0.172	0.172
(c)											
60Co	1173.0	100	0.057	0.055	0.0527	0.054	0.0530	0.195		0.156	0.156
60Co	1332.0	100	0.052	0.050	0.0488	0.050	0.0495	0.182		0.145	0.145
134Cs	604.70	97.10	0.077	0.079	0.0763	0.079	0.078	0.286		0.210	0.210
	795.50	85.40	0.067	0.065	0.0649	0.065	0.065	0.239		0.183	0.183
137Cs	661.60	85.00	0.070	0.075	0.0720	0.075	0.0735	0.270		0.196	0.196
(D) k-40	1460.8	10.66	0.050	0.045	0.0384	0.042	0.040	0.147		0.137	0.137

Table 2: Efficiencies of well-shaped 2x2 NaI(Tl) detector.

Radio nuclides	Decay series	Photopeak Energy	I	M	G	DE
Ra^{226}	Bi^{214}	609.3	0.975309912	0.336304179	0.983	0.32236311
		1120.3	0.996107595	0.246494955	0.983	0.24113213
		1764.5	0.996854956	0.185569411	0.983	0.18182091
	Pb^{214}	295.2	0.992776217	0.542600005	0.983	0.529571059
		351.9	0.993024400	0.47247744	0.983	0.461100321
Th^{232}	Pb^{212}	238.6	0.992701762	0.656766395	0.983	0.640889613
	Ac^{228}	338.3	0.993173407	0.51378197	0.983	0.5015525
		911.6	0.995649447	0.268633078	0.983	0.262920121
		969.1	0995734124	0.261316696	0.983	0.25577852
	Tl^{208}	583	0.994639419	0.343270417	0.983	0.33512621
		2614	0.99865091	0.180515668	0.983	0.174875589
K^{40}		1460.8	0.996854956	0.223101565	0.983	0.20225655
Cs^{137}		661.7	0.995111985	0.318827703	0.983	0.31187569
Co^{60}		1173.2	0.996107595	0.242165306	0.983	0.237121914
		1332.5	0.996381562	0.228025684	0.983	0.22333897
U^{235}		143.8	0.990421172	0.930190419	0.983	0.905618519
		185.7	0.991412088	0.893957571	0.983	0.871213117

Table 3: Gamma radiation hazard indices for the analyzed sands: radium equivalent activity, Ra_{eq}, representative level index, I_r, external and internal hazard index, H, and the corresponding absorbed dose, D, and annual effective dose, DE (indoor and outdoor).

Sampling site	C_{Ra226} Bq/Kg	C_{Th232} Bq/Kg	C_{K-40} Bq/Kg	$(^{226}Ra)_{eq}$ Bq/Kg	I_r Bq/Kg	H_{ex} Bq/Kg	H_{in} Bq/Kg	D nGy/h	DE mSv/yr indoor	DE mSv/ yr outdoor
Bhira Oulad Benaid	7.85	3.00	55.00	16.385	0.118	0.044	0.065	7.72	0.038	0.0095
Targa	5.00	2.60	31.00	11.105	0.080	0.023	0.037	5.20	0.026	0.0064
Boukdasen	4.70	2.20	22.00	9.254	0.068	0.027	0.038	4.41	0.022	0.0054
Sidi Bourzin	6.40	0.80	42.00	10.778	0.079	0.029	0.046	5.07	0.025	0.0062

$DE_{outdoor}$ (mSv y^{-1}) = Absorbed dose rate in air (nGy h^{-1}) x8760 h x 0.2 x0.7 $SvGy^{-1}$ x10^{-6}
DE_{indoor} (mSv y^{-1}) = Absorbed dose rate in air (nGy h^{-1}) x8760 h x 0.8 x0.7 $SvGy^{-1}$ x10^{-6}

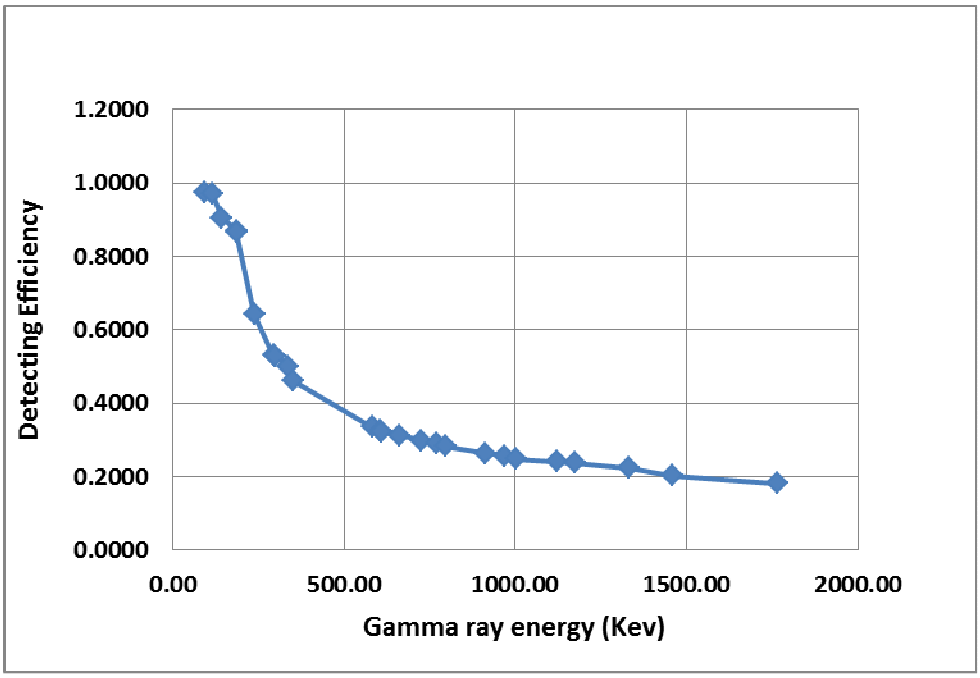

Figure 2: Efficiency of well-shaped 2x2 NaI(Tl) detector.

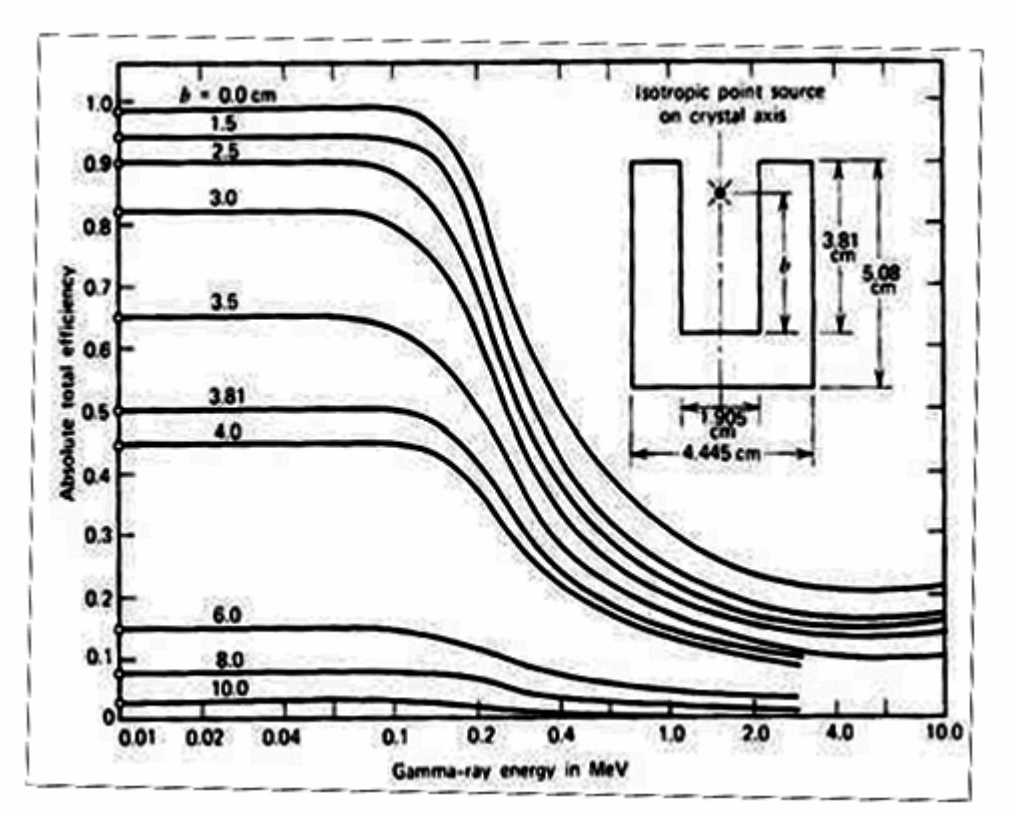

Figure 3: Absolute total efficiency for a well-type NaI(Tl).Ref.(22)

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