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Natural Radioactivity in Red Clay Brick Manufactured in Tlemcen-Algeria, Using Well-Shape NaI(Tl) Detector

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Abstract.

The presence of natural radioactivity in brick 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. Bricks are one of the main components in building construction beside cements, granites and sand. Thus, this research has been carried out in order to investigate the levels of natural radioactivity and associated radiation hazard in Algerian red brick. The natural radioactivity due to the presence of ^{226}Ra, ^{232}Th and ^{40}K in brick samples used as building materials in Tlemcen province - Algeria was measured by gamma spectrometry using NaI(Tl) scintillation well-shaped detector. In this context, brick samples were collected from two manufactories Tafna and Tounan. The mean values of activity concentrations for ^{226}Ra, ^{232}Th and ^{40}K were found to be in the main value of 15.5Bq.kg^{-1}, 11Bq.kg^{-1}, and 196.5Bq.kg^{-1}, respectively. The concentrations of these natural radionuclides were compared with the reported data for other countries and were found significantly lower than the world wide average (1,2). Radium equivalent activities were calculated (41.3 to 51.4) Bq.kg^{-1} for the analyzed samples to assess the radiation hazards arising due to the presence of these radionuclides in the samples. The calculated radium equivalent activities are lower than the limit set by the OECD report 370 Bq.kg^{-1} (3). The measured representative level index values for the investigated samples varied in the range (0.31 to 0.38) Bq.kg^{1}. External and internal hazard index (H_{ex},H_{in}), the specific dose rates in door (D) and the annual effective dose (DE) due to gamma radiation from building materials was calculated.

Keywords: Brick, 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 living tissues. There are no ways to avoid the presence of natural radionuclide since it originates from the formation of earth. Brick, one of the main components in building constructions, is known to contain naturally occurring radioactive materials. Natural radioactivity in brick contributed radiation dose to dwellers that originates from ^{238}U, ^{232}Th and their progeny and ^{40}K. As these radionuclides (^{238}U, ^{232}Th, and ^{40}K) are not uniformly distributed, the knowledge of their distribution in soil, brick 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 red brick produced in two factories Tafna and Tounan, which supply all northwest regions of Algeria. In this study estimates of 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 database 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 (2, 3). 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 radio elements 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(2,3). In Algeria the information about the radioactivity of building materials is limited.

2. Materials and Method.

2.1. The Study Area.

The brick used in this study as the sample for investigation obtained from the Tafna and Tounan factories, which are located at Tlemcen province in western region of Algeria. The Tounan factory situated at about 70 kilometer away and to the north of Tlemcen, and the second manufacturer (Tafna) is about 10 Km to the north of Tlemcen city **Figure 1.**

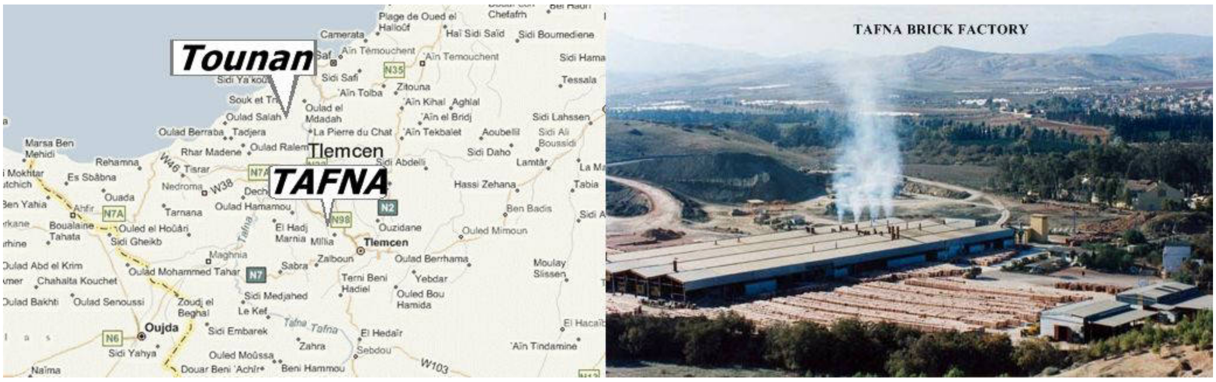

Figure 1: Sites of Tafna and Tounan bricks manufacturers in Tlemcen province of Algeria.

2.2. Collection and Preparation of Samples.

As shown above, all brick samples were collected from two factories (Tafna and Tounan). The samples, which we chose for this study, were produced at different times. Each of the bricks collected from the factories was cleaned and then crushed into smaller pieces. In order to eliminate any water content, the samples were dried at temperature 110° C in an oven for 24 hours (until sample weight became constant and the moisture removed), and there after ground into a fine powder with a grinder and collected after passing through a 1 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 (exact, the radon). 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 brick samples, each 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 brick. 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.

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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) detector .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, that affect the efficient absorption of the photons emitted by the source. Their product is the detector counting efficiency **Dε**.

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

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, and nearly 4p for well type), regardless of the energy deposited (6, 7).

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

$\mathbf{\pi r^2}$ = 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:

$$\mu_{m(NaI)} = \sum \mu_i \cdot w_i = (\mu_1 \cdot w_1)_{Na} + (\mu_2 \cdot w_2)_I .$$

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

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 brick 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 activity (Raeq):

Distribution of ^{238}U, ^{232}Th and ^{40}K in environment is not uniform, so that with respect to exposure to radiation, the radioactivity has been defined in terms of radium equivalent activity (Ra_{eq}) in $Bq.kg^{-1}$ to compare the specific activity of materials containing different amounts of ^{238}U, ^{232}Th and ^{40}K. An index called the 'radium equivalent activity' to obtain the sum of activities for comparison of the specific radio activities of materials containing different radionuclides like Ra, Th and K. It has been estimated that 370 $Bq.kg^{-1}$ of ^{226}Ra, 260 $Bq.kg^{-1}$ of ^{232}Th and 4810 $Bqkg^{-1}$ of ^{40}K produce the same gamma ray dose rate. Thus the radium equivalent activities (Ra) were estimated using the equation (3).

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

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 $Bq.Kg^{-1}$ 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:

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

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 $BqKg^{-1}$ was calculated using the equation:

$$D = 0.462C_{Ra} + 0.604C_{Th} + 0.0417C_k \quad (5)$$

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 ($nGyh^{-1}$), 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 1.0mSv/year based on the formula:

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

The maximum value of H_{ex} equal to unity corresponds to the upper limit of Ra_{eq} (370 $Bq.kg^{-1}$).

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

Brick is an important construction material for houses and buildings in urban areas of Algeria. However, detailed information of the specific activities of ^{226}Ra, ^{232}Th and ^{40}K in brick and other building materials used in Algeria is not available in literature except the work of Amrani and Tahtat,(11).In this study is a continuation of our ongoing project related to the measurement of specific activity of ^{238}U (^{226}Ra), $^{2\,32}Th$ and ^{40}K in environmental samples from brick manufacturers in Tlemcen province in Algeria using gamma-ray spectrometric technique. Using the formula (2) of specific activity with values of efficiency (**Table 1**) 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 13 $Bq.kg^{-1}$,10 $Bq.kg^{-1}$and 182 $Bq.kg^{-1}$ for Tounan brick samples, and for Tafna brick samples are 18 $Bq.Kg^{-1}$,12 $Bq.Kg^{-1}$and 211 $Bq.Kg^{-1}$ 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 (brick) produced 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 brick 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 two sources brick samples studied show that the average Ra_{eq} values (41.3 to51.4) $Bqkg^{-1}$ 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 bricks were lower than the worldwide outdoors annual effective dose average of $0.07mSvy^{-1}$, and also below the value of $1.0mSvy^{-1}$, 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 are very closely specially in C_{Ra},C_{Th}, for examples, Austria, Cyprus, Denmark, Finland, France, Germant, Greece, Ireland, Netherland, Poland, Slovakia, Sweden, United Kingdom(13), Egypt (14), Turkey (15), India (16), SriLanka (17), India, Egypt, Kuwait (18), Iran(19), Estonia (20), China (21).

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

U-238 **(A)**	Gamma Energy KeV	Probab. of γ Emission %	μ_1(Na) Cm²/g W_1=0.153	μ_2 (I) Cm²/g W_2=8.46	$\sum \mu_i.w_i$ μ_m.cal. (NaI)	μ_m.(NaI) ref.	μ_m. (NaI) average	μ_l(NaI). =$\mu_m.\rho$ $\rho = 3.67$ g/cm^3	μ_l. (NaI)for low enegy.ref.	μ_l(Al) 1/cm Ref.	μ_l(Al) 1/cm average
226Ra	186.10	3.510	0.123	0.500	0.4188	0.425	0.420	1.541	1.450	0.343	0.343
	241.98	7.120	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	3.510	0.095	0.130	0.1151	0.130	0.122	0.448	0.480	0.260	
	609.31	44.10	0.077	0.079	0.0762	0.078	0.077	0.283		0.205	
	768.63	04.76	0.069	0.070	0.0675	0.068	0.068	0.249		0.188	
238U											
	49.500		0.244	11.20	9.1874	10.40	9.890	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	15.10	0.057	0.053	0.0520	0.051	0.051	0.187		0.156	0.156
234Th	63.280	04.47	0.200	7.320	6.0160	6.100	6.050	22.20	20.94	0.710	0.710
	92.370	02.60	0.155	2.600	2.1477	2.220	2.180	8.000	7.400	0.490	0.450
235U	185.70	57.25	0.125	0.510	0.4360	0.425	0.430	1.578	1.460	0.345	0.345
	143.70	10.96	0.128	0.600	0.5096	0.516	0.510	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	0.620	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
(D)											
60Co	1173.0	100	0.057	0.055	0.0527	0.054	0.053	0.195		0.156	0.156
60Co	1332.0	100	0.052	0.050	0.0488	0.050	0.050	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	0.074	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	Dε
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 bricks: 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 (outdoor and indoor)

Factory Brick	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 **outdoor**	DE mSv/ yr **indoor**
Tounan	13	10	183	41.314	0.31	0.111	0.146	19.64	0.024	0.096
Tafna	18	12	211	51.407	0.38	0.139	0.187	24.36	0.030	0.120

$DE_{outdoor}$ (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 $SvGy^{-1}$ $x10^{-6}$

4. Conclusion:

The gamma spectroscopy method was used for assessment of the U-238 and Th-232 series and K-40 concentration in many brick samples obtained from two factories Tounan and Tafna inTlemcen 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 brick samples were 13-18 Bq.kg^{-1}, 10-12 Bq.kg^{-1}and 182-211Bq.kg^{-1}, respectively. The results obtained in this study compares well with data from most countries, as explain in discussion 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 potassium contents as a result of different geological formations under the earth crust from where the raw material for particular kind of brick was manufactured.

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 typical world average, thus this material (brick) can be used for building in the Tlemcen (Algeria) and another regions.

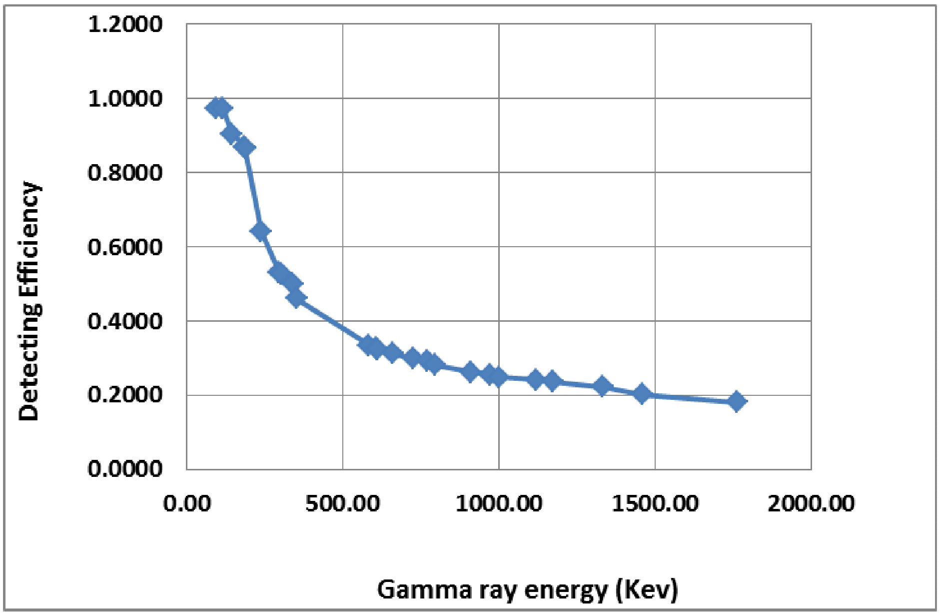

Figure 2: Efficiency of well-shaped 2x2 NaI(Tl) detector

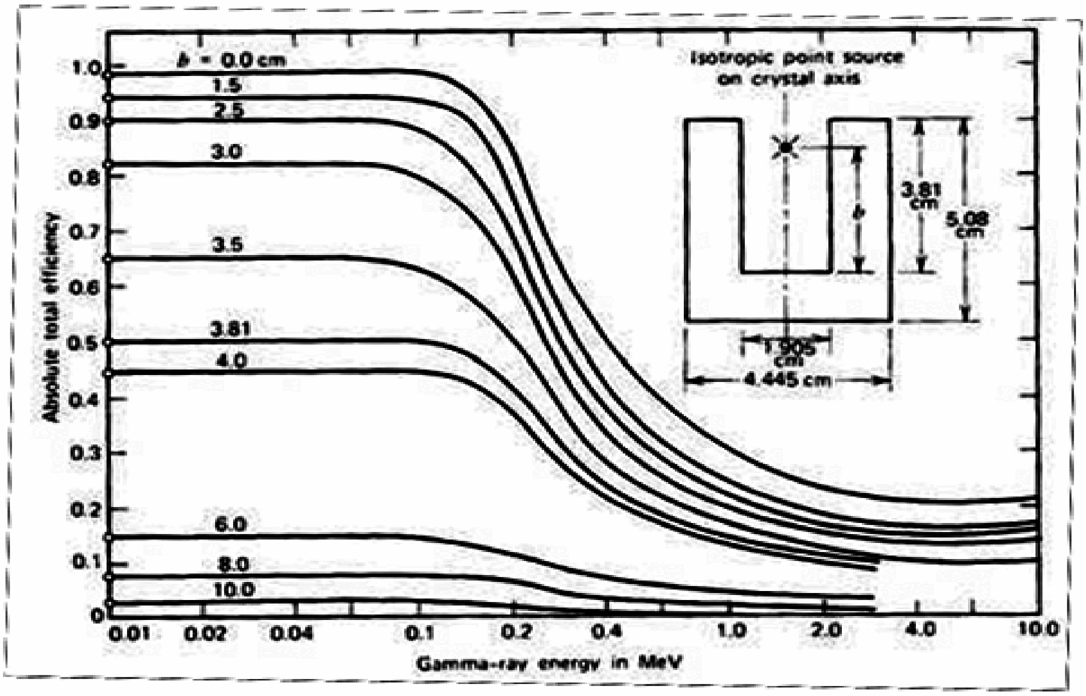

Figure 3: Absolute total efficiency for a well-type NaI(Tl). (22)

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