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 some Algerian’s 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 15.5 Bq kg$^{-1}$, 11 Bq kg$^{-1}$, and 196.5 Bq kg$^{-1}$, respectively. The concentrations of these natural radionuclides were compared with the reported data for other countries and significantly lower than the world wide average as reported by United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) (2000, 2008) 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 in the OECD report ($370$ Bq kg$^{-1}$). 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.

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. Brick is one of the main components in building constructions are known to contain with naturally occurring radioactive materials. Natural radioactivity in brick contributed radiation dose to dwellers that originate 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. 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 are supplied all northwest regions of Algeria by red brick, 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 (70 nGy h$^{-1}$). 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. 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 is situated at about 70 kilometers away and to the north of Tlemcen, and the second manufacturer (Tafna) is about 10 km to the north of Tlemcen city Figure 1.

![Fig.1 sites of Tafna and Tounan bricks manufacturers in Tlemcen province of Algeria.](image)

2.2. Collection and Preparation of Samples.
As shown above all brick samples are collected for two factories (Tafna and Tounan), the obtained samples, which chose for study are produced in different times. Each of bricks fabricated by 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 remove the moisture), 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 238U, 232Th and their decay products (exact, the radon). The containers were full filled for uniform distribution of 220Rn and 222Rn 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). The emphasis was on the determination of specific activity concentration of 226Ra, 232Th and 40K. The analysis of 40K was based upon its single peak of 1460.8 keV, whereas the analysis of 226Ra and 232Th depended upon the peaks of the daughter products in equilibrium with their parent nuclides, the concentration of 226Ra was determined from the average concentrations of 214Pb (352 keV) and 214Bi (609, 1120 and 1765 keV), and that of 232Th was determined from the average concentrations of 212Pb (239 keV), 208Tl (583,2615 keV) and 228Ac (338.3, 911, 969.11 keV) in each sample under study (table-1). In the uranium series the decay chain segment starting...
from radium (Ra) is radiologically 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 radionuclides 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 (Saint Gobain, detectors efficiencies). The well-shaped detectors are of higher efficiency for the same volume of detector. This particular characteristic allows almost a 100 percent efficiency (so called 4π geometry) for low gamma-emitting test sources that can fit the well shape.

2.4.1. Calculating of detector counting efficiency \(D\epsilon\):

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 \(D\epsilon\).

\[
D\epsilon = G \times I \times M
\]  

(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 these 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 \(\sim 1\), respectively. This means every photon incident on the detector produces an output pulse considering the solid angles of both geometries (2π for right cylinder, and nearly 4π for well type), regardless of the energy deposited.

\[
1 - G = \frac{\pi r^2}{4\pi R^2}
\]

where:

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

\[
1 - G = \frac{\pi \times 0.375 \text{ inch} \times 0.375 \text{ inch}}{4 \times \pi \times 1.44 \text{ inch} \times 1.44 \text{ inch}} = 0.017, \text{ and } G = 0.983
\]

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

To calculate \(I\) we have

\[
I = \exp - (\mu_1 \times d)
\]

where:

\(\mu_1\) = 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:

\[
M = 1 - \exp - (\mu_1 \times d)
\]

\(\mu_1\) = the linear attenuation coefficient for gamma ray in NaI( crystal).

\(d = 1.422\) 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 $\mu$ for each aluminum and NaI(crystal). 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 $\mu$ for mixture NaI using the following formulas:

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

$$\mu_{l(NaI)} = \mu_{m(NaI)} \cdot \rho$$  

where $\rho$ is the density of NaI =3.7g/cm$^3$

the calculation result of $\mu_{l(NaI)}$ table (2) are compared with that values from references, 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 $D_{\epsilon}$ of the our detector for each gamma energy in the brick samples under study table (1).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.

$$A \ (B \ q / K g N / (T \cdot \epsilon \cdot I \cdot W)$$  

Where N is net peak counts (background subtracted), T is the measured time (sec.), $\epsilon$ 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 ($Ra_{eq}$)

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 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 of $^{226}$Ra, 260 Bq/kg of $^{232}$Th and 4810 Bq/kg of $^{40}$K produce the same gamma ray dose rate. Thus the radium equivalent activities (Ra) were estimated using the equation (3).

$$Ra_{eq} = C_{Ra} + 1.43C_{Th} + 0.077C_{K}$$  

Where $C_{Ra}$, $C_{Th}$ and $C_{K}$ are the specific activities of $^{232}$Th, $^{238}$U and $^{226}$Ra respectively. 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:

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

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. 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·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:

$$D = 0.462C_{Ra} + 0.604C_{Th} + 0.0417C_{K}$$  

To estimate the annual effective dose rates, the conversion coefficient from absorbed dose in air to effective dose (0.7 Sv·Gy$^{-1}$) and outdoor occupancy factor (0.2) for indoor is (0.8) proposed by [UNSCEAR, 2000] were used. The effective dose rate in units of mSv·y$^{-1}$ was calculated by following Equation

$$DE = DTF$$  

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:

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

And for indoor environment the above formula are seen as:

$$DE (mSv/y) = D (nGy/h) \times 8760h/y \times 0.8 \times 0.7 \ (Sv/Gy)$$
External hazard index (H_{ex}): 

The external hazard index (H_{ex}) is a radiation hazard index defined by (UNSCEAR, 2000) 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 1.0 mSv/year based on the formula:

\[ H_{ex} = \frac{C_{Ra}}{370} + \frac{C_{Th}}{259} + \frac{C_{K}}{4810} \leq 1 \] (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:

\[ H_{in} = \frac{C_{Ra}}{185} + \frac{C_{Th}}{259} + \frac{C_{K}}{4810} \leq 1 \] (10)

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

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 226Ra, 232Th and 40K in brick and other building materials used in Algeria is not available in literature except the work of (Amrani & Tahtat, 2001). In this study is a continuation of our ongoing project related to the measurement of specific activity of 238U (226Ra), 232Th and 40K 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 226Ra, 232Th and 40K 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 Bqkg^{-1}, 10 Bqkg^{-1} and 182 Bqkg^{-1} for Tounan brick samples, and for Tafta brick samples are 18 BqKg^{-1}, 12 BqKg^{-1} and 211 BqKg^{-1} for (226Ra, 232Th and 40K) 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 226Ra, 232Th and 40K respectively UNESCO (2000). The specific activity of 40K, 226Ra and 232Th 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 to 51.4) 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 DE values for all the analyzed bricks were lower than the worldwide outdoors annual effective dose average of 0.07mSv y^{-1}, and also below the value of 1.0 mSv/y, recommended by the International Commission on Radiological Protection(ICRP-60 1990) 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, Germany, Greece, Ireland, Netherlands, Poland, Slovakia, Sweden, United Kingdom(Cristina Nuccetelli, et al.2011, Brick table), Egypt (Kamal M.K et al.2011, table2), Turkey (Arman Erkan.2007. table 1, 2, 3), India (Ravisankar, R. et.al.2013. table1), Sri Lanka (Hwannna, R. et al. table 1- brick sample, 1, 2, 3), India, Egypt, Kuwait (Uosif, M.A.M.2011.3. with references), Iran (Mehdizadeh, S.et al.2011. table1), Estonia (Merle, L. & Enn, E. 2012 table2), China (Ding, Xiang, et al.2013.)
Table 1  Efficiencies of well-shaped 2x2 NaI(Tl) detector.

<table>
<thead>
<tr>
<th>Radio nuclides</th>
<th>Decay series</th>
<th>Photopake Energy</th>
<th>I</th>
<th>M</th>
<th>G</th>
<th>$D_\varepsilon$</th>
</tr>
</thead>
<tbody>
<tr>
<td>$R \ ^{22} \beta \ ^1$</td>
<td>$B \ ^2 \ ^1$</td>
<td>609.3</td>
<td>0.975309912</td>
<td>0.336304179</td>
<td>0.983</td>
<td>0.32236311</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1120.3</td>
<td>0.996107595</td>
<td>0.246494955</td>
<td>0.983</td>
<td>0.24113213</td>
</tr>
<tr>
<td></td>
<td></td>
<td>1764.5</td>
<td>0.996854956</td>
<td>0.185569411</td>
<td>0.983</td>
<td>0.18182091</td>
</tr>
<tr>
<td>$P \ ^{21} \beta$</td>
<td></td>
<td>295.2</td>
<td>0.992776217</td>
<td>0.542600005</td>
<td>0.983</td>
<td>0.52957105</td>
</tr>
<tr>
<td></td>
<td></td>
<td>351.9</td>
<td>0.993024400</td>
<td>0.47247744</td>
<td>0.983</td>
<td>0.46110032</td>
</tr>
<tr>
<td>$T \ ^{23} \beta$</td>
<td>$P \ ^{21} \beta$</td>
<td>238.6</td>
<td>0.992701762</td>
<td>0.656766395</td>
<td>0.983</td>
<td>0.64088961</td>
</tr>
<tr>
<td></td>
<td>$A \ ^{22} \beta$</td>
<td>338.3</td>
<td>0.993173407</td>
<td>0.51378197</td>
<td>0.983</td>
<td>0.5015525</td>
</tr>
<tr>
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<td></td>
<td>911.6</td>
<td>0.995649447</td>
<td>0.268633078</td>
<td>0.983</td>
<td>0.26292012</td>
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<tr>
<td></td>
<td></td>
<td>969.1</td>
<td>0995734124</td>
<td>0.261316696</td>
<td>0.983</td>
<td>0.25577852</td>
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<tr>
<td></td>
<td>$T \ ^{20} \beta$</td>
<td>583</td>
<td>0.994639419</td>
<td>0.343270417</td>
<td>0.983</td>
<td>0.33512621</td>
</tr>
<tr>
<td></td>
<td></td>
<td>2614</td>
<td>0.99865091</td>
<td>0.180515668</td>
<td>0.983</td>
<td>0.17487558</td>
</tr>
<tr>
<td>$K \ ^{4} \alpha$</td>
<td></td>
<td>1460.8</td>
<td>0.996854956</td>
<td>0.223101565</td>
<td>0.983</td>
<td>0.20225655</td>
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<tr>
<td>$C \ ^{13} \alpha$</td>
<td></td>
<td>661.7</td>
<td>0.995111985</td>
<td>0.318827703</td>
<td>0.983</td>
<td>0.31187569</td>
</tr>
<tr>
<td>$C \ ^{13} \beta$</td>
<td>$\alpha$</td>
<td>1173.2</td>
<td>0.996107595</td>
<td>0.242165306</td>
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<td>0.23712191</td>
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<td></td>
<td></td>
<td>1332.5</td>
<td>0.996381562</td>
<td>0.228025684</td>
<td>0.983</td>
<td>0.22333897</td>
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<tr>
<td>$U \ ^{23} \beta$</td>
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<td>143.8</td>
<td>0.990421172</td>
<td>0.930190419</td>
<td>0.983</td>
<td>0.90561851</td>
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<tr>
<td></td>
<td></td>
<td>185.7</td>
<td>0.991412088</td>
<td>0.893957571</td>
<td>0.983</td>
<td>0.87121311</td>
</tr>
</tbody>
</table>


Table-2  Linear attenuation coefficients ($\mu$) of gamma ray in Al. and NaI.

<table>
<thead>
<tr>
<th>U-238 (A)</th>
<th>Gamma Energy KeV</th>
<th>Probab. $g a m n$ ecmn.</th>
<th>$\mu_1$(Na) Cm$^2$/g $W_1=0.153$</th>
<th>$\mu_2$ (I) Cm$^2$/g $W_2=8.46$</th>
<th>$\sum \mu_i$, W$<em>i$ $\mu</em>{al}$, cal. (NaI)</th>
<th>$\mu_{al}$, ref. NIST</th>
<th>$\mu_{al}$, NIST average</th>
<th>$\mu_{al}$, (NaI)</th>
<th>$\mu$(NaI) for low energy ref</th>
<th>$\mu$(Al) 1/cm NIST</th>
<th>$\mu$(Al) 1/cm average</th>
</tr>
</thead>
<tbody>
<tr>
<td>226Ra</td>
<td>186.1 3.51%</td>
<td>0.123</td>
<td>0.50</td>
<td>0.488</td>
<td>0.425</td>
<td>0.420</td>
<td>1.541</td>
<td>1.450</td>
<td>0.343</td>
<td>0.343</td>
<td></td>
</tr>
<tr>
<td>228Ac</td>
<td>212Pb 115.18</td>
<td>0.163</td>
<td>0.2173</td>
<td>0.278</td>
<td>0.245</td>
<td>0.899</td>
<td>0.629</td>
<td>0.290</td>
<td>0.310</td>
<td>0.309</td>
<td></td>
</tr>
<tr>
<td>238Th</td>
<td>214Bi 15.1%</td>
<td>0.095</td>
<td>0.130</td>
<td>0.1151</td>
<td>0.132</td>
<td>0.495</td>
<td>0.489</td>
<td>0.205</td>
<td>0.203</td>
<td>0.205</td>
<td></td>
</tr>
<tr>
<td>214Pb</td>
<td>209.7 19.24%</td>
<td>0.102</td>
<td>0.160</td>
<td>0.1456</td>
<td>0.153</td>
<td>0.149</td>
<td>0.547</td>
<td>0.629</td>
<td>0.290</td>
<td>0.290</td>
<td></td>
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<tr>
<td>214Bi</td>
<td>1764.5 15.8%</td>
<td>0.046</td>
<td>0.043</td>
<td>0.0420</td>
<td>0.043</td>
<td>0.043</td>
<td>0.158</td>
<td>0.126</td>
<td>0.126</td>
<td>0.126</td>
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</tr>
<tr>
<td>214Bi</td>
<td>1120.3 15.1%</td>
<td>0.057</td>
<td>0.053</td>
<td>0.0520</td>
<td>0.051</td>
<td>0.187</td>
<td>0.156</td>
<td>0.155</td>
<td>0.156</td>
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</tr>
<tr>
<td>235U</td>
<td>185.70 57.25%</td>
<td>0.125</td>
<td>0.51</td>
<td>0.4360</td>
<td>0.425</td>
<td>0.43</td>
<td>1.578</td>
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<td>238 Th</td>
<td>228Ac 11.40%</td>
<td>0.097</td>
<td>0.151</td>
<td>0.1379</td>
<td>0.140</td>
<td>0.139</td>
<td>0.510</td>
<td>0.500</td>
<td>0.274</td>
<td>0.265</td>
<td></td>
</tr>
<tr>
<td>212Bi</td>
<td>727 11.8%</td>
<td>0.075</td>
<td>0.072</td>
<td>0.0709</td>
<td>0.071</td>
<td>0.257</td>
<td>0.197</td>
<td>0.193</td>
<td>0.193</td>
<td>0.193</td>
<td></td>
</tr>
<tr>
<td>212Pb</td>
<td>115.18 0.62%</td>
<td>0.146</td>
<td>1.800</td>
<td>1.4913</td>
<td>1.500</td>
<td>1.495</td>
<td>5.487</td>
<td>4.300</td>
<td>0.430</td>
<td>0.430</td>
<td></td>
</tr>
<tr>
<td>238 Th</td>
<td>208Tl 84.5%</td>
<td>0.079</td>
<td>0.080</td>
<td>0.0731</td>
<td>0.081</td>
<td>0.077</td>
<td>0.283</td>
<td>0.215</td>
<td>0.215</td>
<td>0.215</td>
<td></td>
</tr>
<tr>
<td>228Ra</td>
<td>338.32 11.26%</td>
<td>0.097</td>
<td>0.151</td>
<td>0.1372</td>
<td>0.140</td>
<td>0.139</td>
<td>0.510</td>
<td>0.500</td>
<td>0.274</td>
<td>0.274</td>
<td></td>
</tr>
<tr>
<td>60Co</td>
<td>1173 100 %</td>
<td>0.057</td>
<td>0.055</td>
<td>0.0527</td>
<td>0.054</td>
<td>0.0530</td>
<td>0.195</td>
<td>0.156</td>
<td>0.156</td>
<td>0.156</td>
<td></td>
</tr>
<tr>
<td>60Co</td>
<td>1332 100 %</td>
<td>0.052</td>
<td>0.050</td>
<td>0.0488</td>
<td>0.050</td>
<td>0.0495</td>
<td>0.182</td>
<td>0.145</td>
<td>0.145</td>
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<td></td>
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<tr>
<td>134Cs</td>
<td>604.7 97.1%</td>
<td>0.077</td>
<td>0.079</td>
<td>0.0763</td>
<td>0.079</td>
<td>0.078</td>
<td>0.286</td>
<td>0.210</td>
<td>0.210</td>
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</tr>
<tr>
<td>137Cs</td>
<td>661.6 85 %</td>
<td>0.070</td>
<td>0.075</td>
<td>0.0720</td>
<td>0.075</td>
<td>0.0735</td>
<td>0.270</td>
<td>0.196</td>
<td>0.196</td>
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<td></td>
</tr>
<tr>
<td>k-40</td>
<td>1460.8 10.66%</td>
<td>0.050</td>
<td>0.045</td>
<td>0.0384</td>
<td>0.042</td>
<td>0.040</td>
<td>0.1468</td>
<td>0.137</td>
<td>0.137</td>
<td>0.137</td>
<td></td>
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</tbody>
</table>
Fig-2 Efficiency of well-shaped 2x2 NaI(Tl) detector.

Fig-3 Absolute total efficiency for a well-type NaI(Tl). Radioisotopes and Radiation Methodology (Med Phys 4R06/6R03) page 5-14.
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 in Tlemcen province in Algeria, and they are compared with the results from other countries. The average activity concentrations of $^{226}\text{Ra}$, $^{232}\text{Th}$ and $^{40}\text{K}$ in the brick samples were 13-18Bq kg$^{-1}$, 10-12Bq 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 K 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 2)

2- Calculation the efficiencies of well-shaped 2x2 NaI(Tl) detector. The values of DE were showing in table (1) 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 $^{226}\text{Ra}$, $^{232}\text{Th}$, and $^{40}\text{K}$, 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.

**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)

<table>
<thead>
<tr>
<th>Factory Brick</th>
<th>C$_{\text{Ra226}}$ Bq/Kg</th>
<th>C$_{\text{Th232}}$ Bq/Kg</th>
<th>C$_{\text{K-40}}$ Bq/Kg</th>
<th>$^{226}\text{Ra}_{eq}$ Bq/Kg</th>
<th>I$_{r}$ Bq/Kg</th>
<th>H$_{ex}$ Bq/Kg</th>
<th>H$_{in}$ Bq/Kg</th>
<th>D nGy/h</th>
<th>DE mSv/yr outdoor</th>
<th>DE mSv/yr indoor</th>
</tr>
</thead>
<tbody>
<tr>
<td>Tounan</td>
<td>13</td>
<td>10</td>
<td>183</td>
<td>41.314</td>
<td>0.31</td>
<td>0.111</td>
<td>0.146</td>
<td>19.64</td>
<td>0.024</td>
<td>0.096</td>
</tr>
<tr>
<td>Tafna</td>
<td>18</td>
<td>12</td>
<td>211</td>
<td>51.407</td>
<td>0.38</td>
<td>0.139</td>
<td>0.187</td>
<td>24.36</td>
<td>0.030</td>
<td>0.120</td>
</tr>
</tbody>
</table>

DE$_{\text{outdoor}}$ (mSv y$^{-1}$) = Absorbed dose rate in air (nGy h$^{-1}$) x 8760 h x 0.2 x 0.7 SvGy$^{-1}$ x 10$^{-6}$

DE$_{\text{indoor}}$ (mSv y$^{-1}$) = Absorbed dose rate in air (nGy h$^{-1}$) x 8760 h x 0.8 x 0.7 SvGy$^{-1}$ x 10$^{-6}$
REFERENCES:


