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Evaluation of potential exposure risks of natural radioactivity levels emitted from building materials used in Adana, Turkey

S. Solak[1], Ş. Turhan[2], F. A. Uğur[3], E. Gören[1], F. Gezer[1], Z. Yeğingil[1] and İ. Yeğingil[1]

Abstract

In this paper, the natural radioactivity levels in a total of 117 samples of 14 different building materials collected from building construction sites and from the retailers in Adana were studied by means of gamma-ray spectrometer with HPGe detector. The mean activity concentrations of ^{226}Ra, ^{232}Th and ^{40}K measured in the studied building material samples ranged from 2.1 to 88.2 Bq kg^{-1}, 1.8 to 52.7 Bq kg^{-1} and 68.1 to 847.5 Bq kg^{-1} for ^{226}Ra, ^{232}Th and ^{40}K radionuclide, respectively. The external and internal index, the indoor absorbed dose rate and the corresponding annual effective dose were evaluated for potential exposure risks from the usage of the building material samples. The evaluated values of the external and internal index were below the recommended upper level. All the values of effective annual dose determined were lower than recommended exemption level of 0.3 mSv. The results suggest that the use of the studied building material samples in the construction of buildings is unlikely to give rise to any significant radiation exposure to the residents.

Keywords

Radioactivity, Building material, Gamma-ray spectrometer, External index, Internal index, Annual effective dose

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Introduction

The natural occurring radionuclides such as uranium-radium (^{238}U-^{226}Ra) and thorium (^{232}Th) and their decay products and the radioactive isotope of potassium (^{40}K) in building materials are the main sources of radiation. These radiations bring about external and internal exposure to the occupants. The external exposure is caused by direct gamma radiation emitting these radionuclides while the internal exposure is caused by the inhalation of the radioactive inert gas radon (^{222}Rn) and its short-lived decay products. As a consequence of the possible radiological hazards to human health caused by external and internal exposures of people of buildings, many countries, especially European countries, and international organizations such as NEA-OECD, ICRP, UNSCEAR etc. have adopted strong measures for minimizing such exposures.

The study of the natural radioactivity levels in building materials is not only needed to assess the possible radiological hazards to human health but also to develop standards and guidelines for the use and management of these materials.[1] In recent years, there has been an increasing interest in the study of the radioactivity in various building materials,[2–25] although

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there are a few studies on the radioactivity of building materials used. Some cities in Turkey[26–29] require detailed information of the activity concentrations of ^{226}Ra, ^{232}Th and ^{40}K in building materials used but these are not available in literature.

The purpose of the present paper is to study the natural radioactivity level in commonly used building materials in Adana, which is a city in Mediterranean region of Turkey and a major agricultural and commercial centre. Adana is the Turkey's fifth largest city (a population of 1,630,710) and has 250,447 buildings.[30] The studied radioactivity levels of ^{226}Ra, ^{232}Th and ^{40}K were compared with the reported data of other countries and the median values measured in the earth's crust. The results of the activity concentrations of these radionuclides were used to evaluate the potential exposure risks from the usage of the building material samples by calculating the external (radium equivalent activity and the radiation index) and internal (alpha index) hazard indices, the indoor absorbed dose rate and the corresponding annual effective dose.

Materials and methods

Sample collection and preparation

A total of one hundred seventeen building material samples used in the construction sectors in Adana were collected randomly from different building construction sites and from the retailers. Building material samples studied include structural (cement, concrete, gas concrete, clay brick, pumice brick, sand and aggregate) and covering (granite tile, marble tile, ceramic floor tile, lime-limestone, gypsum, grouting and adhesive) materials. The masses of the samples ranged from to 0.5 to 1 kg. The samples were catalogued and coded properly. The samples were grounded and crushed to a fine dust size. The samples were then dried at 110°C for 15–20 h to ensure that moisture is completely removed. Each powdered sample was then packed in cylindrical plastic containers (5 × 6 cm), weighed and hermetically sealed. The geometrical dimensions of the samples were kept identical to those of the reference materials which were used for the efficiency calibration of the gamma spectrometry system. The sealed samples and the reference materials were stored for more than 4 weeks before counting by gamma spectrometer to allow ^{226}Ra and its short-lived decay products to reach the secular equilibrium.

Sample counting system

The natural radioactivity levels of ^{226}Ra, ^{232}Th and ^{40}K in the studied samples were determined with a gamma-ray spectrometry system equipped with a coaxial p-type HPGe detector (GX5020) with a relative efficiency of 50% relative to a 7.62 cm (diam.) × 7.62 cm cylindrical NaI (Tl) detector, with an energy resolution of 2.0 keV at 1332.5 keV and a peak-to-Compton ratio of 60:1. For gamma-ray shielding, a front opening split-top shield (Canberra Model 767) was used to reduce the background. It features 100 mm lead thickness, which is jacketed by a 9.5 mm steel outer housing. The graded liner comprises 1-mm-thick tin layer and 1.5-mm-thick copper layer to prevent interference by lead X-rays. To minimize scattered radiation from the shield, the detector was centred in the liner. The detector was interfaced to the digital spectrum analyzer (DSA-1000), which was a full-featured 16K channel multichannel analyzer on advanced digital signal processing (DSP) techniques. DSA-1000 operates through Genie-2000 gamma spectroscopy software including peak searching, peak evaluation, energy/efficiency calculation mode, nuclide identification etc.

The samples containers were placed on top of the detector for counting. The same geometry was used to determine peak area of samples and references. Prior to sample measurement, gamma-ray background at the laboratory was determined with an empty container under the same conditions of sample measurements and subtracted in order to get net counts for the sample.

Efficiency calibration of detector

The efficiency calibration procedure contains the calculation of the absolute efficiency of the HPGe detector as a function of gamma-ray energy. The absolute efficiency for gamma rays is dependent on the source-detector geometry. Therefore, the absolute efficiency calibration needs to be performed for each source-detector combination. In the ideal case, it can be calculated as the ratio of the number of detected photons of a particular energy to the total number emitted by the source at that energy as shown in equation (1):[31]

$$\varepsilon(E_\gamma) = \frac{N}{P_\gamma \cdot t \cdot A} F_C F_g F_t F_t \qquad (1)$$

where N is the corrected net peak area of the corresponding gamma-ray photopeak at energy E_γ, emitted by a radionuclide; A, the activity of the calibration source or reference material (in Bq), P_γ, the emission probability; t, live time of the sample spectrum in seconds; F_c, coincidence-summing correction factor needed for small distance source-detector; F_g, the geometry correction factor; F_t and $F_{\Delta t}$, the decay correction factors for the time difference between the

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reference date and the beginning of the measurement and the measurement duration, respectively.

In this paper, the absolute efficiency calibration of the HPGe detector was performed using the radionuclide specific efficiency method in which the efficiency values of gamma-ray lines belonging to the specific radionuclide existing in both the reference material and sample were only used. Thus, the uncertainty in gamma-ray intensities, the influence of coincidence-summing correction and self-absorption effects of the emitting gamma photons were avoided for the close geometry conditions. The reference materials RGU-1 (U-ore), RGTh-1 (Th-ore) and RGK-1 (K_2SO_4) used for the radionuclide-specific efficiency calibration of the counting system were counted by placing them in the sample position. The measured efficiency values were fitted to function as shown in Equation (2):

$$\varepsilon(E_\gamma) = \exp(a + b \cdot \ln(E_\gamma) + c \cdot \ln(E_\gamma)^2 + d \cdot \ln(E_\gamma)^3 \quad (2)$$

where E_γ is gamma-ray energy and a, b, c and d are fitted parameters and correspond to −4.43859323038, −0.619432249235, 0.208636210601 and 0.026544495811, respectively. Figure 1 shows the full energy peak efficiency as a function of gamma-ray energy for the HPGe detector.

Calculation of the activity concentration

The activity concentration (A_C) of a gamma-ray emitting radionuclide in the sample is calculated using equation (3), as:

$$A_C(\text{Bq kg}^{-1}) = \frac{N_{CR}}{\varepsilon(E_\gamma) \cdot P_\gamma \cdot m} \quad (3)$$

where N_{CR} is the net count rate (counts per second) of the gamma-ray photopeak in the sample spectrum subtracted from the corresponding gamma-ray photopeak area in the background spectrum; $\varepsilon(E_\gamma)$, the absolute efficiency of the detector given by equation (1); P_γ, the emission probability and m the mass of the sample (kg).

Gamma spectrum was obtained for each studied building material samples using gamma-ray spectrometer mentioned above. Under the assumption which secular equilibrium was reached between ^{226}Ra and ^{222}Rn and ^{232}Th and ^{228}Ra, the activity concentrations were averaged from gamma-ray photopeaks at several energies. The activity concentration of ^{226}Ra was derived from the average of the activities of the gamma-ray line of 609.3 keV from ^{214}Bi and 351.9 keV from ^{214}Pb, while the gamma-ray lines of 911.2 keV from ^{228}Ac and 583.2 keV from ^{208}Tl were used to determine the activity concentration of ^{232}Th.

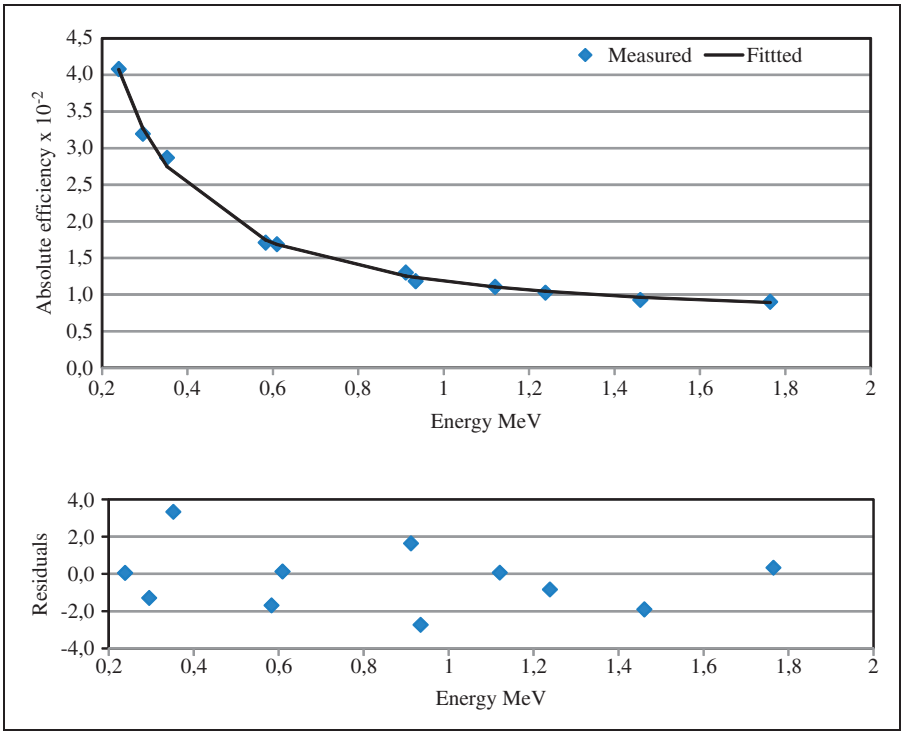


Figure 1. Efficiency curve performed for the HPGe detector used in this study.

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The activity concentration of ^{40}K was obtained using its 1460.8 keV gamma-ray line.

The minimum detectable activity (MDA) of the gamma-ray measurements was calculated as shown in equation (4):[22]

$$MDA(Bqkg^{-1}) = \frac{C_F \cdot N_B}{\varepsilon(E_\gamma) \cdot P_\gamma \cdot t \cdot m} \quad (4)$$

where C_F is the statistical coverage factor which is equal to 1.64 for confidence level 95%; N_B, the standard deviation of the background in the region of interest; $\varepsilon(E_\gamma)$, the absolute efficiency of the detector; P_γ, the emission probability; t, the measurement time in seconds and m the mass of the sample (kg). The mean values of MDA of ^{226}Ra, ^{232}Th and ^{40}K estimated for the studied building materials are presented in Table 1.

The combined standard uncertainty of the activity concentration ΔA_C is calculated by the next formula, shown in equation (5):

$$\frac{\Delta A_C}{A_C} = \sqrt{\frac{\Delta N_{CR}}{N_{CR}} + \frac{\Delta P_\gamma}{P_\gamma} + \frac{\Delta \varepsilon}{\varepsilon} + \frac{\Delta m}{m}} \quad (5)$$

where ΔN_{CR} is the count rate uncertainty; ΔP_γ, the emission probability uncertainty found in the nuclear data tables; $\Delta\varepsilon$, the efficiency uncertainty and Δm, the weighing uncertainty.

Table 1. The mean values of minimum detectable activity measured for ^{226}Ra, ^{232}Th and ^{40}K in the studied building material samples.

		Minimum detectable activity ($Bq\,kg^{-1}$)		
Sample	*N*	^{226}Ra	^{232}Th	^{40}K
Concrete	5	0.4	0.4	4.0
Gas concrete	5	0.5	0.5	4.5
Cement	35	0.4	0.4	3.8
Clay brick	4	0.3	0.3	3.2
Pumice brick	5	0.3	0.3	2.7
Sand	5	0.3	0.3	2.8
Aggregate	4	0.4	0.3	5.1
Granite	5	0.3	0.3	3.2
Marble	5	0.4	0.4	4.1
Ceramic floor tile	4	0.4	0.4	4.1
Lime–limestone	11	0.6	0.6	5.5
Gypsum	19	0.5	0.5	4.7
Grouting	5	0.4	0.4	3.8
Adhesive	5	0.3	0.3	2.6

Results

Radioactivity levels

The range (maximum and minimum) and mean of the activity concentrations of ^{226}Ra, ^{232}Th and ^{40}K in the studied building material samples together with the statistical uncertainty (1 σ) and standard error (SE) are shown in Table 2 (structural building materials) and Table 3 (covering building materials). As can be seen from Tables 2 and 3, the highest mean values of the activity concentration of ^{226}Ra ^{232}Th and ^{40}K are 88.2 ± 18.0 (granite), 52.7 ± 2.9 (pumice brick) and 847.5 ± 24.2 (pumice brick) $Bq\,kg^{-1}$, respectively, while the lowest mean values of the activity concentration of the same radionuclides are 1.7 ± 0.5 (adhesive), 1.3 ± 0.4 (lime) and $72.5 \pm 17.7\,Bq\,kg^{-1}$ (gypsum), respectively. The mean activity concentrations of ^{226}Ra, ^{232}Th and ^{40}K measured in the studied building material samples with the exception of pumice brick and granite tile are compared with the average (population-weighted) values in the earth's crust, which are 32, 45 and 420 $Bq\,kg^{-1}$ for ^{226}Ra, ^{232}Th and ^{40}K, respectively.[32] It can be seen that the mean activity concentrations of ^{226}Ra, ^{232}Th and ^{40}K measured in aggregate, marble, lime-limestone, gypsum, grouting and adhesive samples were significantly lower than the median values in the earth's crust. Table 4 compares the mean values of the activity concentration of ^{226}Ra, ^{232}Th and ^{40}K for selected building materials measured in other countries with the corresponding values measured in the present study. It is concluded from Table 4 that the activity level in the different and also same type building materials varied from one country to another. In general, the mean activity levels of the studied building material samples used in Adana were comparable with those from other countries.

Discussion

Evaluation of potential exposure risks

In the present study, external hazard index (gamma activity concentration), internal hazard index (alpha index), absorbed gamma dose rate in indoor and the corresponding annual effective dose were calculated to evaluate potential exposure risks from usage of the studied building material samples used in Adana.

The external index (gamma activity concentration index)

So far, a number of indexes dealing with the assessment of the excess gamma radiation originating from

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building materials such as radium equivalent activity (Ra_{eq}) index, external hazard index (H_{ex}), the representative level index ($I_{\gamma r}$) and gamma activity concentration index have been proposed by several investigators.[36,37,38] In the present study, the gamma activity concentration index (I_{γ}) is defined to use only as screening tools for ready-to-use building materials. It was calculated using equation (6) proposed by the European Commission:[33]

$$I_{\gamma} = \frac{A_{CRa}}{C_{Ra}} + \frac{A_{CTh}}{C_{Th}} + \frac{A_{CK}}{C_{K}} \quad (6)$$

where A_{CRa}, A_{CTh} and A_{CK} are the measured activity concentrations of ^{226}Ra, ^{232}Th and ^{40}K, respectively,

Table 2. The values of the activity concentrations of ^{226}Ra, ^{232}Th and ^{40}K in the structural building material samples.

			Activity concentration ($Bq\,kg^{-1} \pm 1\sigma$)		
Sample	N		A_{CRa}	A_{CTh}	A_{CK}
Concrete	5	Range	5.9 ± 0.4–38.9 ± 1.9	1.2 ± 0.2–25.4 ± 1.8	150.8 ± 11.5–676.7 ± 47.1
		Mean ± SE	20.6 ± 5.5	14.6 ± 4.1	288.5 ± 97.7
Gas concrete	5	Range	10.5 ± 0.8–35.1 ± 2.4	6.2 ± 0.4–45.6 ± 2.7	107.3 ± 8.9–475.1 ± 14.5
		Mean ± SE	24.5 ± 5.5	28.4 ± 9.0	238.4 ± 62.3
Cement	35	Range	8.7 ± 0.4–131.2 ± 3.9	5.1 ± 0.3–65.5 ± 8.2	31.8 ± 1.8–431.7 ± 8.1
		Mean ± SE	49.8 ± 5.8	17.3 ± 2.2	246.2 ± 20.8
Clay brick	4	Range	19.1 ± 1.0–82.3 ± 2.8	15.9 ± 1.6–38.5 ± 3.4	311.5 ± 15.9–490.2 ± 18.3
		Mean ± SE	35.3 ± 15.7	25.8 ± 4.7	404.8 ± 43.7
Pumice brick	5	Range	50.3 ± 6.2–64.7 ± 4.0	38.4 ± 2.4–60.5 ± 3.9	779.4 ± 20.4–889.8 ± 30.3
		Mean ± SE	56.7 ± 2.9	52.7 ± 4.3	847.5 ± 24.2
Sand	5	Range	3.2 ± 0.2–62.4 ± 3.8	2.1 ± 0.2–53.4 ± 3.7	213.1 ± 17.0–671.7 ± 23.3
		Mean ± SE	38.8 ± 10.0	29.5 ± 11.3	471.4 ± 101.2
Aggregate	4	Range	<MDA–24.4 ± 3.8	<MDA–23.7 ± 3.7	16.2 ± 3.9–492.4 ± 32.3
		Mean ± SE	10.1 ± 1.4	9.5 ± 1.3	178.9 ± 23.2

Table 3. The values of the activity concentrations of ^{226}Ra, ^{232}Th and ^{40}K in the covering building material samples.

			Activity concentration ($Bq\,kg^{-1} \pm 1\sigma$)		
Sample	N		A_{CRa}	A_{CTh}	A_{CK}
Granite	5	Range	18.6 ± 1.1–114.9 ± 3.2	1.4 ± 0.2–52.0 ± 1.9	92.1 ± 3.1–551.9 ± 22.1
		Mean ± SE	88.2 ± 18.0	17.3 ± 9.3	551.9 ± 22.1
Marble	5	Range	<MDA–7.8 ± 1.3	<MDA–11.2 ± 0.9	12.3 ± 0.7–191.2 ± 13.2
		Mean ± SE	2.3 ± 1.8	3.7 ± 2.5	238.4 ± 62.3
Ceramic floor tile	4	Range	26.7 ± 1.2–38.4 ± 1.6	42.5 ± 2.9–56.1 ± 2.4	329.8 ± 27.3–580.4 ± 41.5
		Mean ± SE	33.1 ± 2.5	49.5 ± 3.3	459.1 ± 51.3
Lime–limestone	11	Range	<MDA–47.6 ± 2.9	<MDA–4.7 ± 0.3	5.5 ± 0.3–487.5 ± 36.5
		Mean ± SE	12.9 ± 5.1	1.3 ± 0.4	122.8 ± 43.6
Gypsum	19	Range	<MDA–27.3 ± 1.6	<MDA–10.4 ± 0.7	4.7 ± 0.3–294.7 ± 12.6
		Mean ± SE	5.4 ± 1.8	2.6 ± 0.3	72.5 ± 17.7
Grouting	5	Range	<MDA–5.4 ± 0.3	<ÖEA–16.3 ± 1.1	97.5 ± 7.6–185.9 ± 12.6
		Mean ± SE	4.6 ± 2.9	3.8 ± 3.1	141.5 ± 19.8
Adhesive	5	Range	<MDA–3.4 ± 0.4	<MDA–2.5 ± 0.2	75.1 ± 6.8–237.7 ± 18.2
		Mean ± SE	1.7 ± 0.5	1.5 ± 0.2	158.9 ± 27.6

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in Bq kg^{-1} and C_{Ra}, C_{Th} and C_K are the activity concentrations assumed to produce the same annual effective dose, i.e. 300, 200 and 3000 Bq kg^{-1} for ^{226}Ra, ^{232}Th and ^{40}K, respectively. For structural materials used in bulk amounts, e.g. concrete $I_\gamma \leq 1$ corresponds to an annual effective dose less than or equal to 1 mSv, while $I_\gamma \leq 0.5$ corresponds to an annual effective dose less than or equal to 0.3 mSv. For covering and other materials with restricted use, e.g. tiles $I_\gamma \leq 61$ corresponds to an annual effective dose less than or equal

Table 4. Comparison of the mean values of activity concentrations measured in the present study for the samples examined with those obtained in other countries.

Materials	Country	Activity concentration (Bq kg^{-1})			Reference
		^{226}Ra	^{232}Th	^{40}K	
Concrete	Cameron	13	18	306	[20]
	EU	40	30	400	[33]
	Israel	18	5	51	[7]
	Bangladesh	37	49	250	[2]
	Nigeria	66	126	589	[34]
	Turkey, Ankara	16	9	151	[27]
	Turkey, Adana	21	15	289	This study
Gas concrete	Turkey, Ankara	17	25	339	[27]
	Turkey, Adana	25	28	238	This study
Cement	Bangladesh, Dhaka	61	80	1133	[10]
	Israel	43	48	870	[7]
	Pakistan	26	29	273	[35]
	Algeria	41	27	422	[6]
	Turkey, Ankara	40	26	317	[27]
	Turkey, Isparta	26	10	130	[28]
	Turkey, Adana	50	17	246	This study
Clay brick	EU	50	50	670	[33]
	Egypt	32	26	298	[5]
	China, Xi'an	59	50	714	[19]
	Algeria	65	51	675	[6]
	Turkey, Ankara	31	37	776	[27]
	Turkey, Isparta	59	12	249	[28]
	Turkey, Adana	35	26	405	This study
Sand	Algeria	12	7	74	[6]
	China, Xi'an	41	22	303	[19]
	Bangladesh	14	25	158	[2]
	Turkey, Ankara	23	26	527	[27]
	Turkey, Adana	39	30	471	This study
Lime–limestone	Algeria	16	13	36	[6]
	Israel	12	4	51	[7]
	China, Xi'an	20	13	63	[19]
	Turkey	20	5	55	[36]
	Turkey, Adana	13	1	123	This study
Gypsum	Pakistan	6	13	174	[35]
	Israel	11	6	51	[7]
	EU	10	10	80	[33]
	Turkey	9	4	41	[36]
	Turkey, Isparta	28	16	200	[28]
	Turkey, Adana	5	3	73	This study

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to 1 mSv, while $I_\gamma \leq 2$ corresponds to an annual effective dose less than or equal to 0.3 mSv.

The internal index (alpha index)

A few indexes dealing with the assessment of the excess alpha radiation due to inhalation of radon escaped from building materials such as alpha index and internal health index were developed.[15] In the present study, the alpha index was calculated using equation (7):

$$I_\alpha = \frac{A_{CRa}}{200, Bq, kg^{-1}} \quad (7)$$

where A_{CRa} is the activity concentration of ^{226}Ra in Bq kg^{-1}. The design level for indoor radon exposure in future buildings set by Turkish Standard TS 12614[39] corresponds to an annual mean radon gas concentration of 200 Bq m^{-3}. When the activity concentration of ^{226}Ra in a building material exceeds the value of 200 Bq kg^{-1}, it is possible that the radon exhalation from this material could cause indoor radon concentration exceeding 200 Bq m^{-3}. It is very important to remark that there is no distinction between materials used in bulk materials and covering and other materials with restricted use in the alpha index determination.[15,40]

The range (minimum-maximum) and mean values of the I_γ and I_α calculated for the studied building material samples are presented in Table 5. It is shown in Table 5 that all value of I_γ and I_α did not exceed the recommended upper level of 1. Also, the mean values of I_γ are lower than or slightly above the recommended exemption level of 0.5 corresponding to an effective dose of 0.3 mSv, except that pumice brick samples.

Indoor absorbed gamma dose rate and annual effective dose

Indoor absorbed gamma dose rates due to gamma-ray emissions from ^{226}Ra, ^{232}Th and ^{40}K in the studied structural and covering building material samples were evaluated using formula provided by European Commission Report.[33] The specific dose rates in nGy h^{-1} per Bq kg^{-1} were evaluated for different type of building materials. Indoor absorbed gamma dose rates (DR_{in}) were calculated for:

1. Structural building materials in a standard room calculated using equation (8) with dimension 4 m × 5 m × 2.8 m, wall thickness 20 cm and a density of 2350 kg m^{-3}

$$DR_{in}(nGy, h^{-1}) = 0.92 \times A_{CRa} + 1.10 \times A_{CTh} + 0.08 \times A_{CK} \quad (8)$$

2. Covering building materials in a standard room calculated using equation (9) with dimension 4 m × 5 m × 2.8 m; 3 cm thickness of tiles on all walls; a density of 2600 kg m^{-3}

$$DR_{in}(nGy, h^{-1}) = 0.12 \times A_{CRa} + 0.14 \times A_{CTh} + 0.0096 \times A_{CK} \quad (9)$$

where A_{CRa}, A_{CTh} and A_{CK} are the activity concentrations of ^{226}Ra, ^{232}Th and ^{40}K, respectively, in Bq kg^{-1}.

Table 5. The values of the external (I_γ) and internal (I_α) index calculated for the studied building material samples.

		I_γ		I_α	
Sample	*N*	Range	Mean ± SE	Range	Mean ± SE
Concrete	5	0.18–0.31	0.24 ± 0.02	0.03–0.19	0.10 ± 0.03
Gas concrete	5	0.11–0.42	0.30 ± 0.06	0.05–0.18	0.12 ± 0.03
Cement	35	0.09–0.91	0.33 ± 0.03	0.04–0.66	0.25 ± 0.03
Clay brick	4	0.25–0.63	0.38 ± 0.09	0.10–0.41	0.18 ± 0.08
Pumice brick	5	0.68–0.78	0.74 ± 0.02	0.25–0.32	0.28 ± 0.01
Sand	5	0.09–0.70	0.43 ± 0.11	0.02–0.31	0.19 ± 0.05
Aggregate	4	0.01–0.36	0.19 ± 0.13	0.002–0.122	0.062 ± 0.042
Granite	5	0.16–0.79	0.52 ± 0.01	0.09–0.57	0.44 ± 0.09
Marble	5	0.02–0.09	0.07 ± 0.02	0.002–0.039	0.0012 ± 0.009
Ceramic floor tile	4	0.43–0.56	0.51 ± 0.03	0.13–0.19	0.17 ± 0.01
Lime–limestone	11	0.01–0.19	0.09 ± 0.02	0.003–0.238	0.065 ± 0.025
Gypsum	19	0.01–0.15	0.05 ± 0.01	0.003–0.137	0.027 ± 0.009
Grouting	5	0.01–0.18	0.06 ± 0.03	0.002–0.077	0.023 ± 0.015
Adhesive	5	0.03–0.10	0.07 ± 0.01	0.002–0.017	0.009 ± 0.003

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Table 6. The values of the indoor absorbed gamma dose rate (DR_{in}) and the corresponding annual effective dose (E_{in}) evaluated for the studied building material samples.

		DR_{in}		E_{in}	
Sample	*N*	Range	Mean ± SE	Range	Mean ± SE
Concrete	5	43.9–75.8	58.2 ± 5.6	0.03–0.19	0.10 ± 0.03
Gas concrete	5	26.3–99.9	72.8 ± 14.1	0.05–0.18	0.12 ± 0.03
Cement	35	21.1–227.0	84.6 ± 8.4	0.04–0.66	0.25 ± 0.03
Clay brick	4	59.9–157.3	93.3 ± 21.9	0.10–0.41	0.18 ± 0.08
Pumice brick	5	165.7–190.0	177.9 ± 5.5	0.25–0.32	0.28 ± 0.01
Sand	5	22.3–169.9	105.9 ± 27.0	0.02–0.31	0.19 ± 0.05
Aggregate	4	2.0–87.9	45.0 ± 30.4	0.002–0.122	0.062 ± 0.042
Granite	5	5.0–25.2	17.1 ± 3.3	0.025–0.124	0.084 ± 0.016
Marble	5	0.5–2.6	1.5 ± 0.5	0.003–0.013	0.007 ± 0.003
Ceramic floor tile	4	12.9–16.7	15.3 ± 0.9	0.063–0.082	0.075 ± 0.004
Lime–limestone	11	0.3–6.7	2.9 ± 0.6	0.001–0.033	0.014 ± 0.003
Gypsum	19	0.2–4.9	1.7 ± 0.3	0.001–0.024	0.008 ± 0.002
Grouting	5	0.1–5.5	1.9 ± 0.9	0.001–0.027	0.009 ± 0.005
Adhesive	5	0.8–3.0	1.9 ± 0.4	0.004–0.015	0.010 ± 0.002

In order to determine the corresponding annual effective dose, a value of $0.7\,Sv\,Gy^{-1}$ was used for the conversion factor from absorbed dose in air to effective dose received by adults. The indoor occupancy factor was 0.8, implying that the occupants spent 80% of time indoors.[40] The corresponding annual effective dose (E_{in}) was calculated using Equation (10):

$$E_{in}(mSv) = DR_{in} \times 0.7, Sv, Gy^{-1} \times 0.8 \times 365.25, d, y^{-1} \times 24, h, d^{-1} \times 10^{-6} \quad (10)$$

where DR_{in} is the indoor absorbed gamma dose rate given by equation (9).

The range (minimum-maximum) and mean values of the DR_{in} and E_{in} evaluated for the studied building material samples are presented in Table 6. The worldwide average values of DR_{in} range from 20 to 200 $nGy\,h^{-1}$ with a population-weighted average of $84\,nGy\,h^{-1}$.[32] The mean value of the DR_{in} evaluated for covering building material samples was significantly lower than the quoted population-weighted average value. The mean value of the DR_{in} for clay brick, pumice brick and sand samples was higher than the quoted value. The values of E_{eff} for all estimated samples were lower than the exemption dose criterion of 0.3 mSv.

Conclusion

The natural radioactivity levels of ^{226}Ra, ^{232}Th and ^{40}K in the studied building material samples used in Adana were measured by means of gamma-ray spectrometer. The obtained activity results were tabulated and compared with the values in other countries for selected building materials. The natural radioactivity levels of ^{226}Ra, ^{232}Th and ^{40}K measured in the samples with the exception of pumice brick and granite tile were compared with the earth's population-weighted mean values. The external and internal index, the indoor absorbed gamma dose rate and the corresponding annual effective dose were calculated to evaluate the potential exposure risks from the usage of the building material samples. The mean values of I_γ, I_α, DR_{in} and E_{in} calculated for the structural building material samples are lower than measured in the EU, Bangladesh and China. The calculated values of the external and internal indexes are lower than the recommended upper levels. All values of the annual effective dose were below the criterion limit of the exemption corresponding to $0.3\,mSv\,y^{-1}$.

The results show that the building material samples studied can be safely used in construction of dwellings, work place and schools in Adana according to the international and national recommendations.

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