ASSESSING RADIATION HAZARDS ASSOCIATED WITH NATURAL RADIOACTIVITY IN BUILDING MATERIALS IN HO CHI MINH CITY, VIETNAM

by

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This study is aimed at assessing radiation hazards associated with natural radioactivity in common building materials used in Ho Chi Minh City, Vietnam. Thirty-six samples from eighteen types of building materials were collected to measure activity concentrations using the gross alpha/beta counting system and gamma-ray spectrometry. The gross alpha and gross beta activity concentrations ranged from 94.7 31.3 to 1045.1 112.3 Bqkg^{-1} and 104.9 4.7 to 834.4 37.1 Bqkg^{-1}, respectively. In addition, the activity concentrations of ^{226}Ra, ^{232}Th, and ^{40}K were also determined, which ranged from 4.1 0.1 to 53.5 0.4 Bqkg^{-1} 0.1 to 83.6 0.8 Bqkg^{-1}, and 14.9 0.8 to 664.9 10.6 Bqkg^{-1}, respectively. The indices including radium equivalent activity, external and internal radiation hazard, gamma and alpha indices, activity utilization index, and annual effective dose, were calculated to evaluate the radiological hazards of natural radioactivity. The results showed that these indices were below the recommended safety limits for most investigated samples except six brick samples, whose activity utilization indexes are slightly higher than the safety limit. Even so, all annual effective doses of the samples were found to be below the world average.

Key words: building material, gross alpha/beta, gamma spectrometry, natural radioactivity, radiological hazard

INTRODUCTION

Nowadays, most building materials are made from the natural resources from the environment, so they contain natural radionuclides, mainly ^{40}K, and the series of ^{238}U and ^{232}Th. The use of such materials in construction may lead to long-term exposure to hazardous radiation that poses potential radiological risks to human health. Exposure to a low level of gamma rays can cause stochastic effects, and together with inhaling radon and its progeny in indoor environments increases the risk of lung cancers [1]. Therefore, assessing the radiation hazards due to natural radioactivity in building materials is of great importance for the safety and health of the general public. Over the years, numerous studies have been conducted to evaluate radioactivity in many localities around the world [2-13]. These studies have indicated that most conventional building materials possess radiological hazard indices within the maximum acceptable limits set by international standards. However, significantly higher levels of radioactivity have also been detected in some cases [10, 14-18]. The use of materials with radioactivity exceeding the acceptable limit poses a considerable radiation hazard to individuals. This implies that there may be health risks related to radiation for the general public if radiation levels in building materials are not regularly monitored and evaluated. Furthermore, these results showed that the variability of radioactivity is mainly dependent on the geochemical characteristics of those materials. It is evident that collecting local data is crucial to augment the international natural radionuclide database of building materials, thereby enriching our understanding of their distribution.

Ho Chi Minh City is by far the most populous city in Vietnam and has undergone significant devel-

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opment in building construction in recent years. The city's growing population and increasing urbanization have driven the construction boom for housing and public spaces. To fulfill the construction requirements, there are numerous stores throughout the city that provide a wide range of building materials from diverse origins. Therefore, it is crucial to conduct frequent assessments and monitoring of radiation levels in the building materials used in the city's construction sector. This process will provide valuable data that can be used to develop regulatory guidelines aimed at ensuring the safe utilization of these building materials. Moreover, such measurements would help ensure compliance with the regulations and standards of local government, as well as guarantee the safety of the general public. However, as far as we know, there is a scarcity of literature that offers available and detailed information about radiation levels in the building materials.

This work presents the current state of radioactivity in several kinds of common building materials in Ho Chi Minh City and evaluates potential radiation hazards. To perform the study, the different materials including cement, brick, sand, and rock were collected from various locations in Ho Chi Minh City. Firstly, the gross alpha/beta counting system was used for screening radioactivity in these samples. Then, the activity concentration of ^{226}Ra, ^{232}Th, and ^{40}K was measured by a gamma-ray spectrometer using the high-purity germanium detector. The measured radioactivity was used to calculate the radium equivalent activity, the external and internal radiation hazard indices, the alpha and gamma indices, and the annual effective dose. Finally, the potential radiological hazards were compared with the recommended values.

MATERIALS AND METHODS

Sampling and sample preparation

The samples investigated in this work have been randomly collected from different stores supplying raw building materials and construction dwellings in Ho Chi Minh City, Vietnam. Thirty-six samples including cement (sample codes 1-8), brick (sample codes 9-16), sand (sample codes 17-34), and rock (sample codes 35, 36) were packed in proper plastic bags, labeled, and then transported to the laboratory as soon as possible after collection. All collected samples underwent grinding and milling until they reached a fine powder consistency. The obtained powders were homogenized by a sieving process to ensure a particle size of less than 200 μm. Then they were dried in a furnace at 105 °C until their weight stabilized to ensure that all moisture had been eliminated. Each sample was divided into three parts, namely:

– The first part was used to examine the concentration of elements by using the X-ray fluorescence system (EDX 8000 model [19]) at the Center of Analytical Services and Experimentation in Ho Chi Minh City, Vietnam. The results are shown in the *Supplementary information*.
– The part with approximately 250 g of dry mass was deposited on a stainless steel planchet (diameter of 2 inches, depth of 1/8 inch) and measured the gross alpha and gross beta activity concentrations.
– The third part was used to analyze the radioactivity by using gamma-ray spectrometry. The samples were packed into cylindrical polyethylene plastic containers with dimensions of 47 mm in height, 75 mm in external diameter, and 1 mm in wall thickness. The height of the powder sample to be filled in the container is 20 mm. The containers were tightly sealed with a plastic stopper of 1 mm thickness so that no radon escaped from the samples. Finally, they were preserved in the laboratory for at least 30 days to ensure radioactive equilibrium between ^{226}Ra and ^{232}Th and their respective daughter nuclides. This process has been proven to be reliable for preparing geological samples [20-22]. Information about the material type and mass density of the samples after being filled in the container is presented in tab. 1.

Gross alpha/beta measurements

The gross alpha and gross beta activity concentrations were measured by using a low-background XLB-S5 (Canberra). This detector includes a mixture of 10 % methane (P-10) and 90 % argon (type of gas flow proportional counter) and operates with a high voltage of 1,515 V. The efficiency calibration for detectors was performed by using ISO 17025:2017 standards [23]. Three soil calibration samples containing the isotope of ^{241}Am

Table 1. Information about the material types and mass density of the samples

Sample code	Materials	Density [gcm^{-3}]	Sample code	Materials	Density [gcm^{-3}]
1	Cement	1.40	19	Sand	1.94
2	Cement	1.40	20	Sand	1.94
3	Cement	1.45	21	Sand	1.76
4	Cement	1.46	22	Sand	1.91
5	Cement	1.51	23	Sand	1.78
6	Cement	1.50	24	Sand	1.77
7	Cement	1.36	25	Sand	1.89
8	Cement	1.37	26	Sand	1.67
9	Brick	1.05	27	Sand	1.99
10	Brick	1.24	28	Sand	1.94
11	Brick	1.23	29	Sand	1.92
12	Brick	1.18	30	Sand	1.85
13	Brick	1.12	31	Sand	1.86
14	Brick	1.19	32	Sand	1.89
15	Brick	1.59	33	Sand	1.90
16	Brick	1.61	34	Sand	2.06
17	Sand	1.87	35	Rock	1.52
18	Sand	2.01	36	Rock	1.47

(activity of 5.1 ± 0.3 Bq) were measured to determine the efficiency of the gross alpha measurement. Also, three soil samples containing the isotope of ^{90}Sr (activity of 5.3 ± 0.2 Bq) were used to determine the efficiency for the measurement of gross beta. The acquisition time of the sample was set at 300 minutes while the measurement time for the background of each detector was 1440 minutes by counting empty planchets. The efficiencies for the alpha and beta measurements were 7.3 ± 0.3 % and 42.2 ± 1.5 %, respectively.

Gamma spectrometry

The activity of radionuclides is measured by using an HPGe detector with a relative efficiency of 50 % (supplied by ORTEC, GEM50P4-83 model). This detector is a coaxial p-type with crystal dimensions of 77 mm in length and 65.9 mm in diameter. It is installed within a cylindrical low-background lead chamber, which includes the low-carbon steel, lead, tin, and copper layers with thicknesses of 13 mm, 101 mm, 0.5 mm, and 1.6 mm, respectively. Maestro software [24] was used to acquire photon energy up to 3000 keV in the gamma spectrum with 16,384 channels. An acquisition time for the gamma spectra of the background and samples was set at 86,400 seconds. Colegram software [25] was used to analyze the peak and the overlapping peaks of the measured gamma spectrum.

The radioactivity is calculated by using the following equation

$$A_{\mathrm{i}} = \frac{N_{\mathrm{P}}(E_{\mathrm{i}})}{\varepsilon_{\mathrm{P}}(E_{\mathrm{i}})\,\mathrm{I}_{\gamma}(E_{\mathrm{i}})mt}\,\mathrm{C}_{\mathrm{i}} \qquad (1)$$

where $N_{\mathrm{P}}(E_{\mathrm{i}})$ are the net peak areas, $I_\gamma(E_{\mathrm{i}})$ – the emission probability of separate gamma-rays, $\varepsilon_{\mathrm{P}}(E_{\mathrm{i}})$ – the full energy peak efficiency, m [kg] – the dry mass of sample, t [s] – the acquisition live time, and C_{i} are the self-absorption and coincidence summing corrections (the change of self-absorption correction and coincidence summing factors *vs*. energy were shown in figs. 1(a) and (b). The self-absorption coefficient was determined using the XCOM program with information on the density and elemental composition of the sample referenced in tab. 1 and the *Supplementary information*. The coincidence coefficient was determined using the MCNP-CP program. The relative uncertainty of the activity concentration is evaluated by using the law of uncertainty propagation with the combined standard uncertainties ($k = 1$) [26].

The average activity concentration and its uncertainty are calculated in the following formulas

$$\overline{A} = \frac{\sum_{i=1}^{n} \frac{A_{\mathrm{i}}}{u_{\mathrm{i}}^2}}{\sum_{i=1}^{n} \frac{1}{u_{\mathrm{i}}^2}} \qquad (2)$$

and

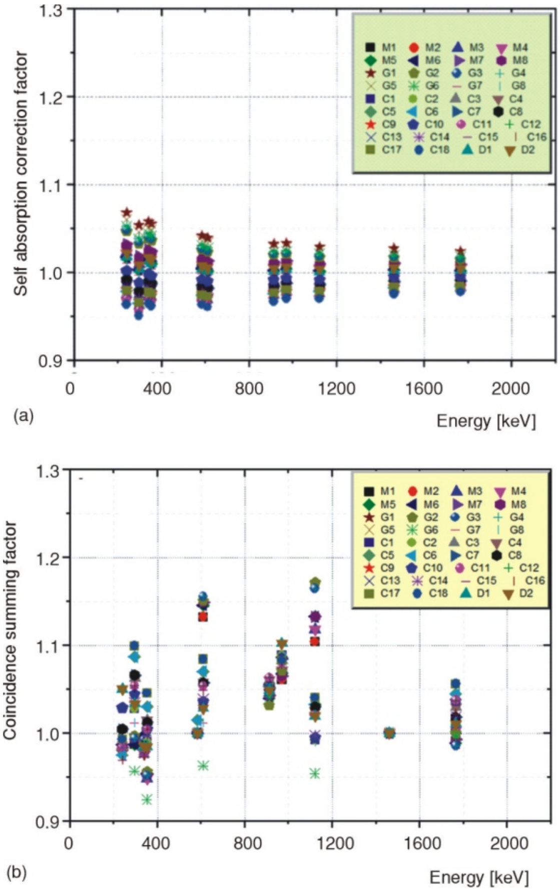

Figure 1. The self-absorption factor (a) and the coincidence summing factor (b) for the building materials

$$\overline{u} = \frac{1}{\sqrt{\sum_{i=1}^{n} \frac{1}{u_{\mathrm{i}}^2}}} \qquad (3)$$

where A_i and u_i are the activity concentration and absolute uncertainty of i^{th} isotopes, respectively.

In this work, the efficiency calibration for the HPGe detector in the energy range from 46.5 keV to 2204.2 keV was obtained by using the RGU reference sample. The coincidence summing correction factors for radionuclides emitting cascade gamma rays were computed by using the MCNP-CP code [27]. Furthermore, utilizing the XCOM database [28], the self-absorption correction factors were also calculated as in our previous study [20].

Radiological hazards variables

The radium equivalent activity index $\mathrm{Ra_{eq}}$ is used to depict the specific activities of ^{226}Ra, ^{232}Th, and ^{40}K isotopes and is calculated by the following equation [29, 30]

$$\mathrm{Ra_{eq}} = A_{\mathrm{Ra}} + 1.43A_{\mathrm{Th}} + 0.077A_{\mathrm{K}} \qquad (4)$$

where A_{Ra} (Bqkg^{-1}), A_{Th} (Bqkg^{-1}), and A_K (Bqkg^{-1}) are the activity concentrations of ^{226}Ra, ^{232}Th, and ^{40}K, respectively.

According to the UNSCEAR [1], the dose conversion coefficients, which were assessed for the standard room model (A standard room model is assumed as a room with the dimensions of 4 m × 5 m × 2.8 m, the wall thickness of 20 cm, and the density of uniformity of material for the whole structure of 2.35 gcm^{-3}), are 0.92 for ^{226}Ra, 1.1 for ^{232}Th, and 0.08 (nGyh^{-1} per Bqkg^{-1}) for ^{40}K. The absorbed dose rate (D) was calculated using the equation below [31]

$$D = 0.92 A_{Ra} + 1.1 A_{Th} + 0.08 A_K \tag{5}$$

The annual effective dose (AED) was calculated based on the absorbed dose rate as follows [1, 31]

$$\mathrm{AED} = D \times 0.7 \times 0.8 \times 8760 \times 10^{-6} \tag{6}$$

The external radiation hazard index H_{ext} was determined by using the eq. [31]

$$H_{ext} = \frac{A_{Ra}}{370} + \frac{A_{Th}}{259} + \frac{A_K}{4810} \tag{7}$$

The internal hazard index H_{int} controlled the hazard due to inhalation of radionuclides in building materials and was calculated by the eq. [31]

$$H_{int} = \frac{A_{Ra}}{185} + \frac{A_{Th}}{259} + \frac{A_K}{4810} \tag{8}$$

The external level index I_γ for the interior and the alpha radiation (I_α) was calculated according to eqs. (9) and (10) [31]

$$I_\gamma = \frac{A_{Ra}}{300} + \frac{A_{Th}}{200} + \frac{A_K}{3000} \tag{9}$$

$$I_\alpha = \frac{A_{Ra}}{200} \tag{10}$$

where $I_\gamma = 1$ as an upper limit, $I_\gamma \leq 1$ corresponds to 0.3 mSv per year, $I_\gamma \leq 3$ which corresponds to 1 mSv per year [1].

An activity utilization index (AUI) was calculated based on the dose rates in the air from naturally occurring radionuclides in building materials [13, 32]

$$\mathrm{AUI} = \frac{A_{Ra}}{50} f_{Ra} + \frac{A_{Th}}{50} f_{Th} + \frac{A_K}{50} f_K \tag{11}$$

where $f_{Ra} = 0.462$, $f_{Th} = 0.604$, and $f_K = 0.041$ are the fractional contributions to the total dose rate in the air due to gamma radiation.

RESULTS AND DISCUSSION

Gross alpha and gross beta activity concentrations

The results of gross alpha (GA) and gross beta (GB) activity concentrations for thirty-six samples of common building materials in Ho Chi Minh City, Vietnam are shown in fig. 2. The GB results for sample codes 27, and 28 are lower than the minimum detectable activity. The GA ranged from 94.7 ± 31.3 Bqkg^{-1} to 1045.1 ± 112.3 Bqkg^{-1} with an average of 409.5 ± 62.2 Bqkg^{-1}. The GB ranged between 104.9 ± 4.7 Bqkg^{-1} and 834.4 ± 37.1 Bqkg^{-1} with an average of 472.0 ± 21.0 Bqkg^{-1}. The International Atomic Energy Agency (IAEA) recommends a limit of 450 Bq kg^{-1} and 1000 Bqkg^{-1} for GA and GB in building materials, respectively [1]. In Vietnam, the maximum acceptable limits of GA and GB in building materials are regulated by the Ministry of Health. These limits are 500 Bqkg^{-1} for GA and 1000 Bqkg^{-1} for GB [33]. All GB results are lower than the recommended values fig. 2(b). However, the GA activity concentrations for the bricks (sample codes 9-16) were higher than the limit values, fig. 2(a).

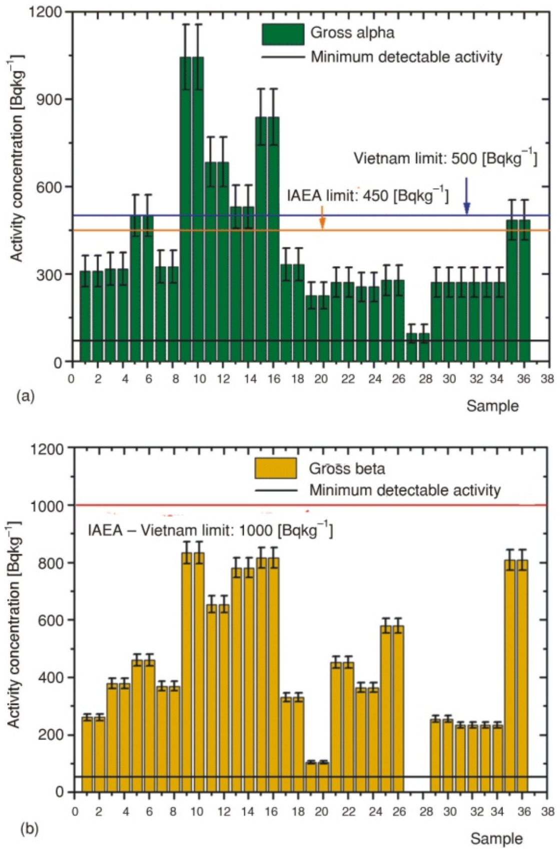

Figure 2. The gross alpha (a) and gross beta activity (b) in the building material samples

Activity concentrations of ^{226}Ra, ^{232}Th, ^{40}K in the building material

The ^{226}Ra activity concentration was calculated based on the activities of ^{214}Pb (295.2 keV, 351.9 keV) and ^{214}Bi (609.3 keV, 1120 keV, 1764.5 keV) assuming the secular equilibrium exists in the analysis/analysed sample. These results ranged from 4.4 ± 0.1 Bqkg^{-1} to 53.5 ± 0.4 Bqkg^{-1} as shown in tab. 2. In addition, tab. 2 also shows the activities of ^{212}Pb (238.6

Table 2. Results of the activity concentration of ^{226}Ra, ^{232}Th, and ^{40}K for the variety of building materials

Sample code	Activity concentration [Bqkg^{-1}]		
	^{226}Ra	^{232}Th	^{40}K
1	38.3 0.3	19.6 0.3	94.7 2.5
2	37.6 0.3	19.0 0.3	102.6 2.6
3	25.9 0.3	23.8 0.3	248.9 4.8
4	28.1 0.3	23.3 0.3	241.9 4.7
5	26.9 0.3	26.6 0.3	226.4 4.4
6	28.8 0.3	28.8 0.3	239.9 4.7
7	53.5 0.4	32.0 0.4	277.9 5.3
8	49.8 0.4	34.5 0.4	263.5 4.9
9	51.2 0.5	83.6 0.8	516.6 9.4
10	48.4 0.4	72.6 0.7	512.0 9.0
11	38.9 0.4	59.1 0.6	457.4 8.3
12	42.1 0.4	55.9 0.6	468.9 8.5
13	32.9 0.3	46.5 0.5	600.1 10.4
14	33.1 0.3	43.9 0.5	611.7 10.4
15	48.7 0.4	71.7 0.7	449.3 7.7
16	48.5 0.4	67.4 0.6	472.2 7.9
17	14.2 0.2	18.7 0.2	269.2 4.9
18	12.3 0.2	17.2 0.2	258.0 4.7
19	10.4 0.1	12.1 0.2	35.1 1.3
20	9.7 0.1	12.5 0.2	34.5 1.3
21	19.8 0.2	25.9 0.3	354.7 6.2
22	16.9 0.2	21.6 0.3	328.7 5.7
23	16.3 0.2	20.5 0.3	240.8 4.6
24	16.4 0.2	21.2 0.3	245.6 4.7
25	20.0 0.2	23.9 0.3	467.7 7.6
26	19.5 0.2	24.1 0.3	440.6 7.4
27	5.2 0.1	5.8 0.1	14.9 0.8
28	4.4 0.1	6.9 0.1	15.8 0.9
29	14.5 0.2	12.6 0.2	203.4 4.0
30	13.8 0.2	14.9 0.2	209.3 4.1
31	12.2 0.2	14.8 0.2	184.3 2.0
32	13.9 0.2	14.0 0.2	179.4 3.7
33	9.6 0.2	9.9 0.2	397.1 6.7
34	8.0 0.1	10.2 0.2	376.9 ± 6.3
35	26.7 0.3	38.1 0.4	664.9 10.6
36	26.7 0.3	37.2 0.4	648.2 10.5

keV, 338.2 keV), ^{228}Ac (911.2 keV, 969.0 keV), and 208Tl (583.2 keV). The average of the ^{232}Th activity concentration was determined by using eq. (2) and ranged from 5.8 0.1 Bqkg^{-1} to 83.6 0.8 Bqkg^{-1}. The ^{40}K activity concentration ranged from 14.9 0.8 Bqkg^{-1} to 664.9 10.6 Bqkg^{-1}. From the measured activities in tab. 2, the minimum radioactivity was found in the sand samples with codes 27, and 28. The gross alpha and gross beta activities of such two samples are also the smallest of the investigated samples.

Table 3 presents a comparison between the measured results and the results from previous studies in Iraq [34], Japan [35], Egypt [36], China [9,10], India [11], Iran [31, 37], Ethiopia [13], Poland [38], Spain [39], Serbia [40], Canada [41], Europe [42], world average [43]. These results show that the level of natural radioactivity for the building materials is lower than the world average, especially the result of ^{40}K is very low when compared to Iraq, China, Iran, Poland, Spain, and Serbia. However, due to the sample size in this work not being large enough, an increase with a greater sample size should be considered in further work.

Assessment of radiological hazards

The results calculated of the radiological hazard variables for the building materials are illustrated in fig. 3. They were below 1 and had no marked effect on health. The AUI was found to be in the range between 0.12 0.01 and 1.52 0.03. The AUI for the bricks (sample codes 9-12, 15-16) were higher than 1 and it would be the subject of future research.

The radium equivalent activity indicator Ra_{eq} for all studied building materials was under the limit value (370 Bqkg^{-1}), ranging from 14.9 0.8 to 210.5 4.3 Bqkg^{-1} . Its average was 92.4 1.8 Bqkg^{-1} and lower than the results from Moharram *et al*. [36], Lu *et al*. [10], Ding *et al*. [9], Ravisankar *et al*. [11], Mas *et al*. [39], but higher than the results from Buranurak and Pangza [44]. Moreover, the absorbed dose rates were in the range from 12.3 0.7 nGyh^{-1} to 180.4 3.7 nGy^{-1}. The calculated annual effective dose ranging from 0.060 0.003 mSv to 0.885 0.018 mSv is shown in fig. 4. These results were lower than the recommended value of 1 mSv [1]. A positive correlation between ^{226}Ra, ^{232}Th, and ^{40}K activity concentrations and radiological hazards for the building materials (0.75 *r* 0.97) is illustrated in tab. 4. These results fit very well with previous research [31, 44-46]. In addition, the ^{226}Ra activity concentration is strongly correlated with ^{232}Th ($r = 0.83$) and moderately correlated with ^{40}K ($r = 0.41$).

The strength of the correlation for the absolute value of *r*: 0-0.19 *very weak*, 0.20-0.39 *weak*, 0.40-0.59 *moderate,* 0.60-0.79 *strong*, 0.80-1.0 *very strong*, minus is a negative correlation and plus is a positive correlation [45, 46].

The *p*-value is calculated by Excel software, where ^{a}p value < 0.01, ^{b}p value < 0.05, and ^{c}p value > 0.05. Bold values indicate strong and very strong correlations. Italic values indicate weak and very weak correlations.

CONCLUSION

In this work, the radioactivities for various building materials and their potential radiological hazards in Ho Chi Minh City (Vietnam) were determined. The observation values of the H_{ext} and H_{int} ranged from 0.038 0.002 to 0.568 0.012 and from 0.052 0.003 to 0.707 0.015, respectively. For the gamma index (I) and the alpha index (I), the calculation val-

Table 3. Comparison of activity concentration of ^{226}Ra and ^{232}Th series and ^{40}K for samples in the present study to the other research

Research	Activity concentration [Bqkg^{-1}]		
	^{226}Ra	^{232}Th	^{40}K
Present work	(4.4 0.1) – (53.5 0.4)	(5.8 0.1) – (83.6 0.8)	(14.9 0.8) – (664.9 10.6)
Iraq [34]	(33.0 1.9) – (179.3 2.9)	(1.9 0.2) – (17.43 0.47)	(108.7 3.2) – (977.8 10.1)
Japan [35]	0.8 – 320	0.4 – 200	1 – 1100
Egypt [36]	(14.2 0.3) – (60.6 1.1)	(2.75 0.01) – (84.7 0.5)	(7.35 0.03) – (554.4 3.9)
China [9]	19.8 – 87.4	11.6 – 47.7	273.3 – 981.2
China [10]	(32.5 21.6) – (233.1 8.6)	(17.2 6.8) – (49.3 10.8)	(249.6 22.8) – (795.5 29.1)
India [11]	2 – 89	25 – 359	103 – 634
Iran [37]	23 – 93	24 – 118	462 – 1190
Iran [31]	(6.7 1.0) – (43.6 9.0)	(5.9 1.0) – (60.1 11.0)	(28.5 3.0) – (1085 113)
Ethiopia [13]	(20.2 0.9) – (35.6 4.9)	(10.1 0.6) – (50.3 6.4)	(44.1 1.0) – (406.8 6.7)
Poland [38]	5 – 52	7 – 71	520 – 1560
Spain [39]	(1.8 0.3) – (181 10)	(3.5 2.1) – (185 10)	(67 11) – (4530 200)
Serbia [40]	(5 2) – (311 43)	(7 2) – (887 18)	(17 6) – (2643 264)
Canada [41]	28	28	641
Europe [42]	7 – 272	4 – 138	17 – 805
World average [43]	35	30	400

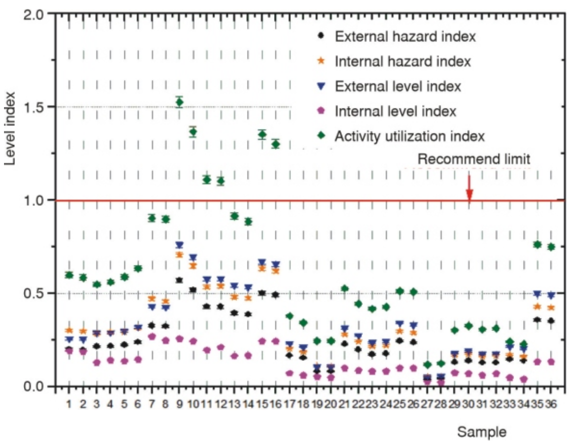

Figure 3. Comparison of the external hazard index, the internal hazard index, the external level index, the internal level index, and the activity utilization index with the limit value (represented by a line)

ues ranged from 0.049 0.003 to 0.760 0.016 and from 0.022 0.001 to 0.268 0.001, respectively. Although the estimated activity utilization index AUI had a maximum value of 1.52 0.03, the average of the AUI was 0.62 0.01. Most of the indices H_{ext}, H_{int}, I , I , and AUI for the building materials were below 1 and were within the recommended safety limit. The maximum value of the Ra_{eq} was 210.4 4.3 Bqkg^{-1} which was lower than the limit value (370 Bqkg^{-1}). The absorbed dose rate of all investigated building materials ranged from 6.2 0.4 nGyh^{-1} to 95.6 2.0 nGyh^{-1} with an average of 43.3 0.9 nGyh^{-1}. The calculated values of the AED ranged from 0.0076 0.0005 mSv to 0.1173 0.0024 mSv with an average of 0.040 0.001 mSv. These results were lower than the world average value [1] and were a good agreement with the relevant regulations of the Ministry of Health in Vietnam [33].

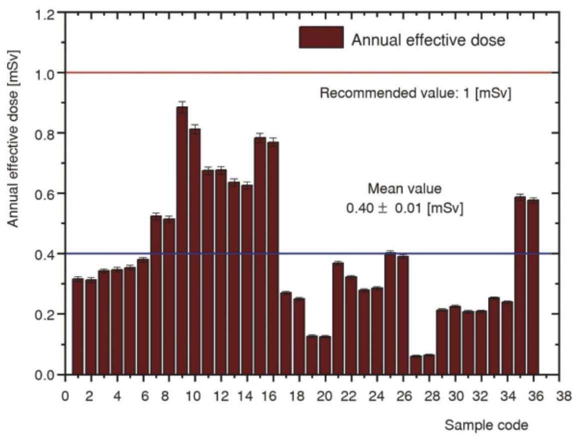

Figure 4. The annual effective dose for natural radioactivity of the building materials

Table 4. Pearson correlation coefficient of parameters (*n* = 36)

	Gross alpha	Gross beta	^{226}Ra	^{232}Th	^{40}K	Ra_{eq}	Dose
Gross alpha							
Gross beta	**0.80**^{c}						
^{226}Ra	*0.31*^{a}	*0.05*^{a}					
^{232}Th	*–0.19*^{a}	*–0.35*^{a}	**0.83**^{c}				
^{40}K	*–0.32*^{c}	*–0.56*^{b}	*0.41*^{a}	**0.64**^{a}			
Ra_{eq}	*–0.10*^{a}	*–0.33*^{a}	**0.86**^{a}	**0.97**^{a}	**0.75**^{a}		
Dose	*–0.11*^{a}	*–0.35*^{a}	**0.85**^{a}	**0.97**^{b}	**0.77**^{a}	**1.00**^{a}	
AED	*–0.11*^{a}	*–0.35*^{a}	**0.85**^{a}	**0.97**^{a}	**0.77**^{a}	**1.00**^{a}	**1.00**^{a}

AUTHORS CONTRIBUTIONS

L. Q. Vuong: Conceptualization, methodology, formal analysis, writing – original draft. H. D. Chuong: Conceptualization, methodology, formal analysis, writ-

Supplementary information

Sample code	Concentration element [%]																						
	Ca	Si	Al	S	Fe	Mg	K	Ti	Mn	Sr	Cu	Zn	Zr	V	Y	Rb	Nb	Cr	Ga	Ni	Ag	Ir	O
1, 2	48.693	7.954	2.923	1.455	1.769	1.207	0.433	0.211	0.062	0.049	0.018	0.017	0.011	0.007	0.003	0.004							35.186
3, 4	31.709	14.145	5.381	1.394	4.285	1.838	0.844	0.620	0.160	0.046	0.007	0.008	0.014		0.002	0.005	0.003	0.070					39.469
5, 6	35.550	13.068	4.839	1.052	4.434	1.069	0.860	0.487	0.192	0.036	0.008	0.010	0.013	0.016	0.002	0.005		0.087					38.272
7, 8	37.281	13.197	4.510	1.518	1.566	1.962	0.950	0.177	0.084	0.091	0.007	0.004	0.010	0.007	0.004	0.005							38.627
9, 10		31.437	13.034	0.052	3.670		1.421	0.497	0.017	0.004		0.004	0.018	0.017			0.002	0.009	0.001				49.818
11, 12	0.363	29.274	14.231	0.085	4.859		1.218	0.668	0.027	0.004		0.005	0.016	0.024	0.003			0.014		0.004		0.005	49.200
13, 14	0.181	29.048	14.438		5.160		1.664	0.431	0.019	0.004		0.017	0.013	0.021	0.005			0.013		0.003			48.983
15, 16	0.137	31.990	12.242	0.042	3.299	0.392	1.348	0.544	0.022	0.004		0.006	0.018	0.019	0.004	0.001	0.002	0.012	0.001	0.003			49.913
17, 18	1.617	39.601	4.307	0.085	1.817		1.280	0.159	0.060	0.008		0.003			0.001	0.003		0.021					51.038
19, 20	0.029	43.065	3.493	0.042	0.316		0.235	0.113	0.005		0.004							0.021			0.009		52.668
21, 22	0.367	38.423	6.728	0.051	1.513		1.455	0.173	0.031	0.005	0.005	0.002	0.011		0.001	0.005		0.018			0.009		51.203
23, 24	2.205	36.527	6.317	0.110	3.247		1.051	0.208	0.077	0.010		0.004	0.010	0.006	0.002			0.027					50.199
25, 26	0.239	38.832	6.100	0.071	1.553		1.744	0.200	0.026	0.005	0.005	0.002	0.010		0.001	0.006		0.021					51.185
27, 28		43.712	2.959	0.055	0.265		0.064	0.050	0.003		0.002		0.002					0.039					52.849
29, 30		42.139	3.643	0.260	0.284		1.274	0.091	0.005	0.002	0.005		0.007		0.001	0.003		0.027			0.012		52.249
31, 32	0.081	41.889	4.081	0.126	0.440		1.016	0.107	0.011	0.003	0.004		0.008			0.003		0.023					52.209
33, 34	0.073	40.370	4.529	0.183	1.382		1.754	0.115	0.015	0.003	0.004	0.002			0.002	0.005		0.036			0.009		51.519
35, 36	2.718	29.300	11.163	0.140	3.929	1.854	2.030	0.388	0.059	0.015		0.005	0.019	0.017	0.003			0.012					48.348

ing – original draft. L. D. Nhat: Formal analysis. H. D. Tam: Conceptualization, formal analy – is, project administration, writing – review & editing. T. T. Thanh: Formal analysis. V. T. Minh: Investigation. L. D. Hung: Investigation. P. L. Ho: Formal analysis. Tao C. V.: Formal analysis.

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ПРОЦЕНА ОПАСНОСТИ ОД ЗРАЧЕЊА ПОВЕЗАНИХ СА ПРИРОДНОМ РАДИОАКТИВНОШЋУ У ГРАЂЕВИНСКИМ МАТЕРИЈАЛИМА У ХО ШИ МИНУ, ВИЈЕТНАМ

Рад има за циљ процену опасности од зрачења повезаних са природном радиоактивношћу у уобичајеним грађевинским материјалима који се користе у Хо Ши Мину, Вијетнам. Сакупљено је 36 узорака из 18 врста грађевинског материјала да би се измериле концентрације активности коришћењем укупног алфа и бета система бројања и спектрометрије гама зрачења. Укупне концентрације алфа и бета активности кретале су се од 94.7   31.3 до 1045.1   112.3 Bqkg^{-1} и 104.9   4.7 до 834.4   37.1 Bqkg^{-1}, респективно. Поред тога, утврђене су и концентрације активности ^{226}Ra, ^{232}Th и ^{40}K које су се кретале у распону од 4.1   0.1 до 53.5   0.4 Bqkg^{-1}, 5.7   0.1 до 83.6   0.8 Bqkg^{-1} и 14.9   0.8 до 664.9   10.6 Bqkg^{-1}, респективно. Индекси који укључују еквивалентну активност радијума, опасност од спољашњег и унутрашњег зрачења, гама и алфа индексе, индекс коришћења активности и годишњу ефективну дозу, израчунати су да би се процениле радиолошке опасности природне радиоактивности. Резултати су показали да су ови индекси испод препоручених граница безбедности за већину испитиваних узорака осим шест узорака цигле, чији су индекси искоришћења активности нешто већи од границе безбедности. Чак и тако, утврђено је да су све годишње ефективне дозе узорака испод светског просека.

Кључне речи: грађевински материјал, укупно алфа и бета зрачење, гама спектрометрија, природна радиоактивност, радијациона опасност