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The dose of gamma radiation from building materials and soil

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Abstract. The radioactivity of some structural building materials, rows, binders, and final construction products, originating from Serbia or imported from other countries, was investigated in the current study by using the standard HPGe gamma spectrometry. The absorbed dose in the air was computed by the method of buildup factors for models of the room with the walls of concrete, gas-concrete, brick and stone. Using the conversion coefficients obtained by interpolation of the International Commission on Radiobiological Protection (ICRP) equivalent doses for isotropic irradiation, the corresponding average indoor effective dose from the radiation of building materials of 0.24 mSv·y^{-1} was determined. The outdoor dose of 0.047 mSv·y^{-1} was estimated on the basis of values of the specific absorbed dose rates calculated for the radiation of the series of ^{238}U, ^{232}Th and ^{40}K from the ground and covering materials. The literature values of the effective dose conversion coefficients for ground geometry were applied as well as the published data for content of the radionuclides in the soil.

Key words: natural radioactivity • building materials • soil • dose

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Introduction

The use of materials with elevated radioactivity in building construction can increase the indoor exposure to ionizing radiation. The content of natural radionuclides in common structural NORM (naturally occurring radioactive material) building materials – rows and products: lime, gypsum, sand, gravel, stone, concrete, brick, reflects the geological origin of their mineral components. In accordance with the mechanisms of formation and conditions for migration of rock substances, the quantities of ^{238}U, ^{232}Th and ^{40}K vary within the same species of a material [1].

In general, due to the abundance of accessory minerals, the highest concentration of radionuclides is found in igneous granites, while the amount of these elements in other stones depends on the type of protolite and weathering and are mainly determined by the content in lithophile minerals. The lowest radioactivity is exhibited in gypsum and lime, formed by chemical sedimentation, in which the presence of radionuclides occurs from non-carbonate constituents [2]. Clastic sediments, sand and gravel, with domination of quartz, do not contain great amounts of the radioactive elements – they are affected by the transport of different rock materials from distant locations. In sedimentary clay deposits, raw materials for producing ceramic, high adsorption capability of the layers of clay minerals can contribute significantly to the uranium and thorium concentration [3].

The content of ^{232}U, ^{226}Ra, ^{232}Th and ^{40}K, was previously measured in typical rocks, soil and common building materials in Serbia [4–10]. The results are around the averages of the worldwide ranges [1, 11–13].

Since new and imported products are in use, in order to consider the fulfilment of the effective dose criterion (1 mSv) [14–16], the exploration of the radioactivity of building materials continues to be carried out in this study. Also, for the purpose of radiological protection, the aim of this research is an estimation of the average indoor and outdoor gamma dose for the population in Serbia.

The effective dose, E (in Sv), corresponding to gamma radiation of a building material is usually determined from the absorbed dose in the air D (in Gy): $D = q_U A_U + q_{Th} A_{Th} + q_K A_K$, produced by the activities (A) of the series of ^{238}U, ^{232}Th and ^{40}K in a standard room [17], where the parameters q are the activity concentration – absorbed dose conversion factors. Also, the UNSCEAR [12] conversion coefficient $D \rightarrow E$, calculated for the plate of ground and rounded to the unique value of 0.7 Sv/Gy for all three radionuclides, is typically applied.

As the density and thickness of the materials used for room construction belong to a wide range (and can differ significantly from the standard room values) in this study, for the purpose of the calculation of the conversion factors q, we took into account the effect of these parameters' variability as discussed in [18, 19]. Furthermore, the effective dose was evaluated by applying the conversion coefficients for room geometry obtained by interpolation of their published values for isotropic (ISO) irradiation.

Different values of the absorbed dose conversion factors for terrestrial ^{238}U, ^{232}Th and ^{40}K radiation have been published [20–22] depending on the calculation parameters and computation method. So, the conversion factors in this study were evaluated for variable parameters of the plate model of the ground (obtaining the functional dependences on the thickness and surface), considering also the covering effect of concrete layer. The mean outdoor dose is then assessed from measured values of soil radioactivity [23, 24], averaged for the country.

Method

Calculation

The conversion factors, q_U, q_{Th} and q_K in both cases, the room and ground radiation, were calculated in kerma approximation, for National Bureau of Standards (NBS) [25] air composition, by the method of buildup factors with geometric progression (GP) parametric form [26]. The integration over the volumetric radiation source was performed using the program Mathematica [19], which was applied for all interpolations in this study. Specific absorbed dose rates of ^{238}U, ^{232}Th and ^{40}K were determined for the detection point placed at the centre of the room with standard dimensions: $4 \times 5 \times 2.8$ m^3, without doors and windows [17] and with variable thickness of the walls: from 10 to 25 cm. The use of the materials with variable density was assumed, in the range from 0.6 to 2.6 g·cm^{-3} ($\rho = 1.6$ g·cm^{-3} for the soil), disregarding the differences in their chemical composition – the buildup factors [27] and density weighted attenuation coefficients for NBS concrete were applied. The effect of covering material with the layer of another material was considered using the buildup factor for the combination of the layers [28].

The indoor effective dose was calculated by weighting of the coefficients for the equivalent dose H/K_a (H – equivalent dose, K_a – air kerma), as in the study [29], but in accordance with the ICRP [30] recommendations for tissues and organs weighting factors. These coefficients, for radiation of the radionuclide series of ^{238}U, ^{232}Th and ^{40}K, were determined by Lagrange interpolation of ISO values in [31–33], given for particular discrete energies.

Measurement

The types of explored building materials, available at the market, originated either from Serbia or from imports are listed in Table 5, in the frame of the measurement results. The investigations were carried out using the standard gamma spectroscopy system with Canberra HPGe detector calibrated in the energy range 40–3000 keV with relative efficiency 26% and full width at the half of maximum FWHM = 1.8 at 1332 keV (^{60}Co). The activity concentrations, A [Bq·kg^{-1}], of natural radionuclides: ^{226}Ra, ^{232}Th and ^{40}K in the samples were measured with the associated combined standard uncertainty at 1σ confidence level of 10%.

Results

Conversion factors

Building materials

The specific absorbed dose rates in the room $5 \times 4 \times 2.8$ m^3, due to the ^{238}U, ^{232}Th and ^{40}K gamma radiation from certain building materials, assuming their application only in the walls construction, are given in Table 1. The contribution of the floor and ceiling of the concrete structure with the same thickness, $d = 20$ cm, is also presented.

In order to fit the results of the calculation where the combinations of the parameters ρ and d have been applied, we obtained (in Mathematica) the formula (1):

$$q = a_0 + a_1 \rho d + a_2 (\rho d)^2 + a_3 (\rho d)^3 \tag{1}$$

for dose conversion factors corresponding to radiation only from the walls with arbitrary density and

Table 1. The values of q_U, q_{Th} and q_K corresponding to the walls $d = 20$ cm, from some building materials and concrete floor and ceiling, $d = 20$ cm (in the room $4 \times 5 \times 2.8$ m^3)

Material	Density, ρ [$g \cdot cm^{-3}$]	Conversion factor, q [$nGy \cdot h^{-1}/Bq \cdot kg^{-1}$]		
		^{238}U	^{232}Th	^{40}K
			Walls	
Aerated concrete	0.8	0.193	0.227	0.0169
Gypsum	1.2	0.259	0.306	0.0230
Brick	1.6	0.307	0.365	0.0277
Concrete (standard room)	2.35	0.364	0.436	0.0335
Stone	2.6	0.376	0.452	0.0348
			Floor and ceiling	
Concrete (standard room)	2.35	0.393	0.474	0.0366

Table 2. The values of the parameters in the fitting formula (1)

Fit parameter	Conversion factor, q [$nGy \cdot h^{-1}/Bq \cdot kg^{-1}$]		
	^{238}U	^{232}Th	^{40}K
a_0	$-3.150 \cdot 10^{-3}$	$-7.687 \cdot 10^{-3}$	$-3.1836 \cdot 10^{-4}$
a_1	$1.556 \cdot 10^{-2}$	$1.865 \cdot 10^{-2}$	$1.3440 \cdot 10^{-3}$
a_2	$-2.167 \cdot 10^{-4}$	$-2.592 \cdot 10^{-4}$	$-1.7000 \cdot 10^{-5}$
a_3	$1.095 \cdot 10^{-6}$	$1.336 \cdot 10^{-6}$	$7.8240 \cdot 10^{-8}$

Table 3. The effective dose conversion coefficients for radiation of the series of natural radionuclides in standard room geometry

Radionuclide	Conversion factor (kerma in air → effective dose) E/K_a [$Sv \cdot Gy^{-1}$]
^{238}U	0.734
^{232}Th	0.748
^{40}K	0.753

thickness. The fitting formula for the radiation of floor and ceiling was published in Ref. [34].

The parameters a_0-a_3 of expression (1) are shown in Table 2. For each q (q_U, q_{Th} or q_K), the coefficient of determination of the fit is: $R^2 \approx 0.9999$.

The values of the conversion coefficients: absorbed dose (air kerma) → effective dose, E/K_a (in $Sv \cdot Gy^{-1}$), calculated for ^{238}U, ^{232}Th and ^{40}K, in the geometry of a standard room, are given in Table 3. The results were obtained assuming the equilibrium in the series, using the values of q_U, q_{Th} and q_K in the standard room: 0.757, 0.910 and 0.0700 $nGy \cdot h^{-1}/Bq \cdot kg^{-1}$, respectively [10].

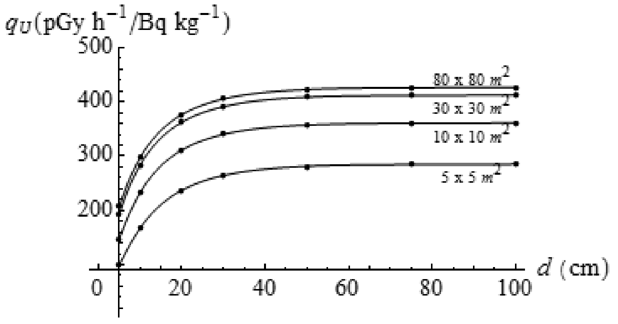

Fig. 1. The dependence of absorbed dose conversion factor q_U, on the thickness of the ground for four values of the surface (with the same form of the curves for q_{Th} and q_K).

Ground

The absorbed dose conversion factors were calculated for variable thicknesses, d (5–150 cm), and surfaces, S (5×5 – 100×100 m^2), of the ground modeled as the quadratic plate, with density of 1.6 $g \cdot cm^{-3}$ [20], at the point $h = 100$ cm above its centre. The results, shown in Fig. 1 for q_U (with the same form of the curves for q_{Th} and q_K) were indicative for the estimate of the degree of dependence saturation for published parameters $d = 100$ cm, and $S = 80 \times 80$ m^2 [21]. Since 99% of asymptotic q was attained, the values of the conversion factors for these parameters, presented in Table 4 (where q_U, q_{Th} and q_K in the case of the application of covering materials are also given), were used in further calculation of the doses.

For a more detailed insight, Fig. 2 shows the conversion factors per emitted photon, $q(E)/Y(E)$, resolved into the spectrum according to the [35] energy grouping and compared for the radiation of the ground and in the standard room. The lower values of the ground than the room conversion factors are significantly caused by the distribution and attenuation of photon fluence in different geometry rather than being the consequence of the density differences. The fitting formulae, derived in Mathematica:

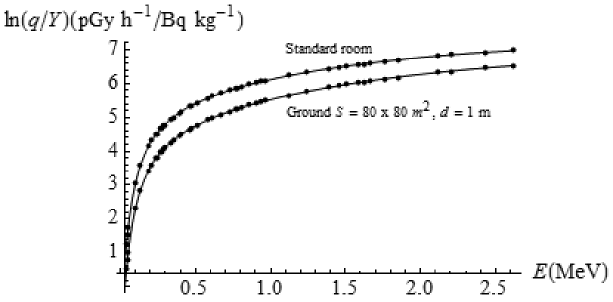

Fig. 2. The dependence of absorbed dose conversion factor per photon yield, q/Y, on the photon energy, E, for the emission of the ground and in the standard room.

Table 4. The values of q_U, q_{Th} and q_K for the radiation of the ground, $\rho = 1.6$ g·cm^{-3}, at the point $h = 100$ cm above the plate: $S = 80 \times 80$ m and $d = 100$ cm

Geometry	Coverage	Density, ρ [g·cm^{-3}]	Conversion factor, q [nGy·h^{-1}/Bq·kg^{-1}]		
			^{238}U	^{232}Th	^{40}K
Ground (80 × 80 m^2, $h = 100$ cm)	No	1.6	0.427	0.528	0.0405
Covered ground $d = 100$ cm	Yes	1.6	0.133	0.176	0.0137
Covering material (asphalt, stone) $d = 6$ cm		2.6	0.294	0.352	0.0268

– for ground

$$\ln(q/Y)(E) = 5.527 + 1.103\ln E - 0.00522\ln^2 E - 0.00579\ln^3 E - 0.0510\ln^4 E - 0.0119\ln^5 E \tag{2.1}$$

– for standard room

$$\ln(q/Y)(E) = 6.120 + 0.988\ln E - 0.0393\ln^2 E - 0.0101\ln^3 E - 0.0499\ln^4 E - 0.0108\ln^5 E \tag{2.2}$$

with the coefficients of determination of $R^2 = 0.99999$ for both fits, and with a similar form as the point source dose equivalent in [36], enable the calculation of conversion factors for an arbitrary approach to energy grouping.

Indoor irradiation due to the building materials

Radionuclides in building materials; fulfilment of dose criterion

The range and the average values of ^{226}Ra (^{238}U), ^{232}Th and ^{40}K activity concentrations in the investigated 137 building materials classified by the type are shown in Table 5. The results are in accordance with the geological characteristics of the samples: the lowest content of the radionuclides, comparable with the minimum detectable activities, are measured in sediments lime and gypsum, while the highest activity concentrations are exhibited by some granites (African red, new imperial red).

Table 5. The ranges and the average values with standard deviation (calculated for five and more samples), of activity concentrations, A [Bq·kg^{-1}], of radionuclides in building materials and the corresponding values of the effective dose, E [mSv·y^{-1}]

Building material (number of samples)	Activity concentration, A [Bq·kg^{-1}] (range) average value with SD			Effective dose, E [mSv·y^{-1}] (range) average value with SD
	^{226}Ra	^{232}Th	^{40}K	
Sand (5)	(11–13) 12 ± 1	(13–19) 16 ± 2	(323–382) 348 ± 23	(0.023–0.028) 0.025 ± 0.002
Gravel (3)	(14–26) 19	(12–17) 14	(305–354) 333	(0.25–0.28) 0.26
Quartz sand (7)	(2–14) 5 ± 4	(<1–8) 4 ± 3	(7–422) 99 ± 156	(0.020–0.21) 0.073 ± 0.073
Lime (3)	(1–4) 2	(1–2) 1	(2–13) 6	<0.01
Gypsum (15)	(1–39) 8 ± 9	(1–5) 2 ± 1	(2–63) 29 ± 17	(<0.01–0.059) 0.018 ± 0.012
Clinker (8)	(13–19) 17 ± 2	(12–20) 17 ± 3	(194–263) 216 ± 25	(0.18–0.26) 0.22 ± 0.026
Portland cement (2)				
I (ordinary)	20	16	45	0.17
II SR (sulfate resistent)	43	19	197	0.33
White cement (10)	(15–25) 18 ± 3	(2–19) 12 ± 5	(13–120) 55 ± 37	(0.092–0.20) 0.15 ± 0.031
Brick (24) bulk (10), block (14)	(6–42) 31 ± 9	(11–54) 40 ± 11	(108–583) 460 ± 118	(0.046–0.25) 0.19 ± 0.049
Concrete (8) (*Manić et al.* [10])	(8–32) 18 ± 8	(8–24) 14 ± 6	(157–313) 251 ± 53	(0.061–0.15) 0.11 ± 0.032
Gas-concrete (2)	(12–36) 24	(1–2) 2	(90–148) 119	(0.021–0.051) 0.036
Stone (50) marble (16), sandstone (5), dolomite (3), granite (26)	(<1–162) 23 ± 30	(<1–128) 22 ± 29	(5–1282) 327 ± 375	(<0.01–0.61) 0.15 ± 0.16

SD – standard deviation.

Table 5 summarizes also the effective doses corresponding to the materials of the same type evaluated from:

$$E\ [\mathrm{mSv \cdot y^{-1}}] = D\ [\mathrm{nGy \cdot h^{-1}}] \cdot f \cdot 8760\ [\mathrm{h \cdot y^{-1}}] \cdot E/K_a\ [\mathrm{Sv \cdot Gy^{-1}}] \times 10^{-6}\ [\mathrm{mSv \cdot nGy^{-1}}] \tag{3}$$

for indoor occupational factor $f = 0.8$ and coefficients E/K_a rounded to the unique value of 0.740 $\mathrm{Sv \cdot Gy^{-1}}$ for all radionuclides.

In this relation, the absorbed dose, D, was calculated using the conversion factors q_U, q_{Th} and q_K in Table 1, that is, distinguishing the structural materials: brick, concrete, gas-concrete, gypsum and stone only based on density and assuming their application was only in the walls of a standard room. For all raw materials and binders: sand, gravel, quartz sand, clinker and cement, we assumed a 100% application in the walls, floor and ceiling of the standard room (concrete), while lime is treated as the component of gypsum walls, with full utilization percentage of 50%.

The use of all these materials would lead to additional doses that are below the reference level of 1 $\mathrm{mSv \cdot y^{-1}}$, and would meet the dose criterion. Still, for the quantification of the natural radiation background in the room and estimation of the exceeding of that level, the dose was calculated on the assumption that the content of the radionuclides in the walls is equal to their activity concentration in the local terrestrial environment, figuratively, if 'walls were made of the soil'. Using the averaged results of measurements of the ground in Serbia [23, 24] and conversion factors for the standard room, the effective dose from the background: $E = 0.46$ $\mathrm{mSv \cdot y^{-1}}$, was obtained, close to the world average level of indoor gamma irradiation [12]. Obviously, in accordance with the low radionuclide content, all of the investigated building materials additionally fulfil the stricter condition for indoor utilization of $\Delta E < 0.3\ \mathrm{mSv \cdot y^{-1}}$ [14].

The combination of building materials; average dose in the room

Based on the activity concentrations of the radionuclides in building materials, using the appropriate conversion factors, that is, data from Table 2 and formula (1), the indoor dose was estimated for several common models of the room. The results for the rooms whose walls were made from concrete, brick or gas-concrete with floor and ceiling of concrete shown in Table 6, were obtained taking into account the presence of the voids in 30% volume of the blocks and assuming the mass fractions of 35 : 50 : 15% of sand, gravel and cement, respectively, in concrete. The application of the 2-cm thick stone as a covering material of the floor and half surface of the walls was also considered.

In calculation of the average doses, the typical room dimensions (thickness of the walls are 15 and 20 cm) were assumed and the usage weighting factors of 40, 50 and 10% for the models of concrete, brick and gas-concrete, respectively, were taken into account. Further, the occupational factor f for such a 'room' was additionally weighted in percentage ratio of 25 : 75 with regard to the application or no of coverring of 50% surfaces of all walls (by stone). Thus, the average absorbed dose of approximately 46 $\mathrm{nGy \cdot h^{-1}}$ and effective dose of 0.24 $\mathrm{mSv \cdot y^{-1}}$ were estimated for the population in consideration, the values which are in the range of lower indoor gamma doses evaluated for EU countries [37].

The outdoor dose

For the assessment of the outdoor dose from terrestrial radionuclides, the averaged results (Table 7) of previous measurements [23, 24] of the radionuclide content in the ground in Serbia were used and conversion factors from Table 2 were applied. The absorbed dose of $D = 51\ \mathrm{nGy \cdot h^{-1}}$ was calculated, similar

Table 6. The range and the average value with standard deviation of absorbed dose, D [$\mathrm{nGy \cdot h^{-1}}$], for the standard room and for the room $4 \times 5 \times 2.8\ \mathrm{m^3}$ with concrete floor ($d = 20$ cm) and ceiling ($d = 15$ cm), where the thickness of opposite walls are $d = 20$ cm and $d = 15$ cm. The model utilization average absorbed dose and effective dose are also presented

Model of the room	Absorbed dose, D [$\mathrm{nGy \cdot h^{-1}}$] (range) average value with SD		
	Concrete	Brick	Gas-concrete
Standard room dimensions			
Uncovered walls	(31–49) 40 ± 4	(29–75) 58 ± 9	(26–36) 30 ± 1
100% Covered walls	(22–86) 46 ± 14	(20–113) 60 ± 14	(17–75) 38 ± 14
Typical room: walls 15 and 20 cm			
Uncovered walls	(30–47) 38 ± 4	(28–70) 54 ± 6	(24–34) 29 ± 1
50% Covered walls	(25–66) 41 ± 10	(23–88) 56 ± 11	(20–54) 33 ± 10
Averaged according to the application of the covering (25%) and materials utilization (concrete : brick : gas-concrete = 40 : 50 : 10%)			
Absorbed dose, D [$\mathrm{nGy \cdot h^{-1}}$] (range) average value with SD	(27–62) 46 ± 7		
Effective dose, E [$\mathrm{mSv \cdot y^{-1}}$] (range) average value with SD	(0.14–0.32) 0.24 ± 0.036		

SD – standard deviation.

Table 7. The ranges and the average values with standard deviation of activity concentration, A [Bq·kg^{-1}], of radionuclides in outdoor building materials and ground, and corresponding effective doses E [mSv·y^{-1}]

Building material (number of samples)	Activity concentration, A [Bq/kg] (range) average value with SD			Effective dose, E [mSv·y^{-1}] (range) average value with SD
	^{226}Ra	^{232}Th	^{40}K	
Asphalt (5)	(6–25) 17 ± 9	(<2–26) 11 ± 12	(29–585) 244 ± 288	(<0.01–0.039) 0.019 ± 0.018
Concrete (7)	(3–45) 13 ± 15	(<2–54) 13 ± 18	(18–260) 129 ± 80	(<0.01–0.043) 0.015 ± 0.014
Stone (9) sandstone (3), granite (26)	(2–89) 29 ± 23	(1–128) 36 ± 32	(5–1143) 545 ± 359	(<0.01–0.11) 0.044 ± 0.029
Ground	34	34	461	0.063

SD – standard deviation.

to the world population weighted average value of outdoor gamma irradiation [12], whereas taking the conversion coefficient (E/K_a) of 0.7 Sv·Gy^{-1} and $f = 0.2$ in formula (3), corresponding annual effective dose of $E = 0.063$ mSv·y^{-1} was evaluated [12].

The contribution to the exposure due to radiation of building materials for outdoor usage was estimated based on the content of radionuclides measured in asphalt, concrete and stone, and the conversion factors calculated in this study for 6 cm of the layer covering the soil (Table 2). The obtained results presented in Table 7 are far below the reference level.

In assessment of the average outdoor dose from the ground (Table 8), it was assumed that nearly 70% of the surfaces are covered in the percentage relationship of 60:30:10 for the application of asphalt, concrete or stone, respectively. The absorbed dose of 39 nGy·h^{-1} was calculated, corresponding to the effective dose of 0.047 mSv·y^{-1}, the value which is the lower range of the average dose worldwide distribution [12]. However, the quotient between the indoor and outdoor absorbed dose, 1.2, are close to the world average value of 1.4.

Summary and conclusions

The activity concentrations of ^{238}U, ^{232}Th and ^{40}K in building materials that are used in Serbia (concrete: 18, 14, 251 Bq·kg^{-1}; brick: 31, 40, 460 Bq·kg^{-1}) are in general, lower than the average content of these radionuclides in building materials in European Union (concrete: 40, 30, 400 Bq·kg^{-1}; brick 50, 50, 670 Bq·kg^{-1}, for ^{228}U, ^{232}Th and ^{40}K, respectively [14]). The dose from all the materials are below the reference level of 1 mSv, so, consequently, the excess of this quantity with respect to the natural radiation background meets the dose criterion.

According to the applications of the materials, the population weighted average absorbed dose indoor, 46 nGy·h^{-1}, is below the most of corresponding values for the EU countries reported in [37], and is also lower than the world averaged indoor dose of 84 nGy·h^{-1}. The outdoor dose (39 nGy·h^{-1}) arising from radionuclides in the ground is below the global population average of 59 nGy·h^{-1}, but the quotient between the indoor and outdoor dose in Serbia (1.2) is near the worldwide average value of 1.4 [12].

To check the possibility of the application of building materials, the gamma index (index of activity concentration) is a widely used approach. The gamma index I is defined as:

$$I = \frac{A(\mathrm{Ra})}{300} + \frac{A(\mathrm{Th})}{200} + \frac{A(\mathrm{K})}{3000} \le 1 \tag{4}$$

This is a simple criterion of the applicability of a building material, whose derivation is based on the dose criterion and the activity concentrations in different materials [38], could be adopted in accordance with the type and utilization fraction of the material. Namely, if a material was applied in a small part of a whole construction, it follows that the maximal radionuclide concentrations that comply with the dose criterion of 1 mSv·y^{-1} are several times higher than the values in the equation for the gamma index. Nevertheless, due to such a limited application of the material, the gamma index should be weighted by actual volume fraction of the used material.

However, a mass fraction of a building material (with respect to the standard room), determined by its density, could change the expression for I through

Table 8. The averaged absorbed dose D and effective dose E due to the radiation of the ground (d = 100 cm) with covering layer (d = 6 cm) (averaged)

Geometry	Absorbed dose, D [nGy·h^{-1}] (range) average value with SD	Effective dose, E [mSv·y^{-1}] (range) average value with SD
Ground (d = 100 cm) + covering layer (d = 6 cm) (averaged)	(30–55) 39 ± 7	(0.036–0.067) 0.047 ± 0.008

SD – standard deviation.

corresponding differences in the conversion factors: for the materials with lower density, the permissible activity concentrations are proportionally higher. So, in accordance with the density of the building materials, values of I could be roughly divided into at least two ranges of values: for densities of up to 2 g·cm^{-3} (gas-concrete, brick) $I = 2$, and for these greater than 2 g·cm^{-3} (concrete, stone) $I = 1$, with unchanged expression for I of given EC [14] form.

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References

1. International Atomic Energy Agency. (2003). *Guidelines for radioelement mapping using gamma ray spectrometry data*. Vienna: IAEA. (IAEA-TECDOC-1363).
2. Bell, K. G. (1963). *Uranium in carbonate rocks*. Geological Survey professional paper 474 – A. Geological Survey (U.S.).
3. Egidi, P., & Hull, C. (1999). Naturally occurring and technologically enhanced naturally occurring radioactive material. In Producers, users and proposed regulations, PEP Course 1.A Notes, Health Physics Society, 32nd Midyear Topical Meeting, Albuquerque, New Mexico.
4. Vesić, D. (1987). *The gamma spectrometry determination of uranium, thorium and potassium in geologically defined samples.* Doctoral dissertation, Faculty of Science, Belgrade, Serbia. (in Serbian).
5. Nikezić, D. (1990). *The contribution to the experimental and theoretical study of indoor irradiation of the population and individuals.* Doctoral dissertation, Faculty of Science, Kragujevac, Serbia. (in Serbian).
6. Pavlić, B., Čonkić, Lj., Bikit, I., Slivka, J., Krmar, M., & Žikić, N. (2000). Estimation of the indoor exposure to external radiation from natural radionuclides in building materials in Vojvodina. In Proceedings of the Fifth International Symposium and Exhibition on Environmental Contamination in Central and Eastern Europe. Prague.
7. Popović, D., & Todorović, D. (2006). Radon indoor concentrations and activity of radionuclides in building materials in Serbia. *Facta Universitatis, Series: Physics, Chemistry and Technology*, *4*, 11–20.
8. Krstić, D., Nikezić, D., Stevanović, N., & Vučić, D. (2007). Radioactivity of some domestic and imported building materials from south eastern Europe. *Radiat. Meas.*, *42*, 1731–1736.
9. Ujić, P., Čeliković, I., Kandić, A., Vukanac, I., Đurašević, M., Dragosavac, D., & Žunić, Z. S. (2010). Internal exposure from building materials exhaling ^{222}Rn and ^{220}Rn as compared to external exposure due to their natural radioactivity content. *Appl. Radiat. Isot.*, *68*, 201–206.
10. Manić, V., Manić, G., Nikezic, D., & Krstic, D. (2012). Calculation of dose rate conversion factors for ^{238}U, ^{232}Th and ^{40}K in concrete structures of various dimensions, with application to Niš, Serbia. *Radiat. Prot. Dosim.*, *152*, 361–368.
11. National Council on Radiation Protection and Measurements. (1994). *Exposure of the population in the United States and Canada from natural background radiation.* Bethesda, Maryland. (NCRP Report 94).
12. United Nations Scientific Committee on Effects of Atomic Radiation. (2000). *Sources, effects and risks of ionizing radiation.* UNSCEAR 2000 Report. New York: United Nations.
13. Nuclear Regulatory Commission, Environmental Protection Agency, Department of Energy. (2009). *Multi-Agency radiation survey and assessment of materials and equipment (MARSAME).* USA. (NUREG-1575, Suppl. 1, EPA 402-R-09-001, DOE/HS-0004).
14. European Commission. (1999). *Radiological protection principles concerning the natural radioactivity of building materials. Radiation Protection 112.* Luxembourg: Office for Official Publications of the European Communities.
15. Official Gazette of the Republic of Serbia. (2009). *The law on protection against ionizing radiation and nuclear safety*. Official Gazette of the Republic of Serbia no. 36/2009 of 15 May 2009. (in Serbian).
16. The Radiation and Nuclear Safety Authority (STUK). (2010). *The radioactivity of building materials and ash*. (Guide ST 12.2). Helsinki.
17. Koblinger, L. (1978). Calculation of exposure rates from gamma sources in walls of dwelling rooms. *Health Phys.*, *34*, 459–463.
18. Risica, S., Bolzan, C., & Nuccetelli, C. (2001). Radioactivity in building materials: room model analysis and experimental methods. *Sci. Total. Environ.*, *272*, 119–126.
19. Maduar, M. F., & Hiromoto, G. (2004). Evaluation of indoor gamma radiation dose in dwellings. *Radiat. Prot. Dosim.*, *111*, 221–228.
20. Beck, H. L. (1975). The physics of environmental gamma radiation fields. In *The natural radiation environment II*. Oak Ridge: United States Atomic Energy Commission.
21. Clouvas, A., Xanthos, S., Antonopoulos-Domis, M., & Silva, J. (2000). Monte Carlo calculation of dose conversion factors for external exposure to photon emitters in soil. *Health Phys.*, *78*, 295–302.
22. Krstic, D., & Nikezic, D. (2010). Calculation of the effective dose from natural radioactivity in soil using MCNP code. *Appl. Radiat. Isot.*, *68*, 946–947.
23. The Agency for Protection against Ionizing Radiation and Nuclear Safety of Serbia. (2011). *Report on the level of exposure of the population to ionizing radiation from the environment in the Republic of Serbia in 2010.* Belgrade (in Serbian). Available from www.srbatom.gov.rs.
24. Radiological Laboratory, Institute of Occupational Health "Niš". (2012). *Final report on monitoring the activities of radionuclides in the soil and the environment of the city of Nis for the period 2011/2012 year.* For the Directorate for Economy, Sustainable Development and Environmental Protection of the city of Niš, Niš. (in Serbian).
25. Hubbell, J. H. (1969). Photon cross sections, attenuation coefficients, and energy absorption coefficients from 10 keV to 100 GeV. National Bureau of Standards, Washington DC. (NSRDS-NBS 29).
26. Harima, Y., Tanaka, S., Sakamoto, Y., & Hirayama, H. (1991). Development of new gamma-ray buildup factor and application to shielding calculations. *J. Nucl. Sci. Technol.*, *28*, 74–84.
27. American National Standards Institute. American Nuclear Society. (1991). *Gamma-ray attenuation*

coefficients and buildup factors for engineering materials. La Grange Park, Illinois, USA. (ANSI/ANS-6.4.3-1991).

28. Goldstein, H. (1959). *Fundamental aspects of reactor shielding.* Addison-Wesley Publishing Co; reprinted by Johnson Reprint Corp. (1971).
29. Al-Jundi, J., Ulanovsky, A., & Prohl, G. (2009). Doses of external exposure in Jordan house due to gamma-emitting natural radionuclides in building materials. *J. Environ. Radioact*., *100*, 841–846.
30. International Commission on Radiological Protection. (2007). *The 2007 recommendations of the Commission on Radiological Protection.* (ICRP Publication 103), *Ann. ICRP*.
31. International Commission on Radiological Protection. (1996). *Conversion coefficients for use in radiological protection against external radiation.* (ICRP Publication 74). Oxford, UK: Pergamon Press.
32. Zankl, M., Petoussi-Henß, N., Drexler, G., & Saito, K. (1997). *The calculation of dose from external photon exposures using reference human phantoms and Monte Carlo methods Part VII: Organ doses due to parallel and environmental exposure geometries.* Institut für Strahlenschutz. (GSF-Bericht 8/97).
33. Lee, C., Lee, L., Han, E. Y., & Bolch, W. E. (2007). Consideration of the ICRP revised tissue weighting factors on age-dependent values of the effective dose for external photons. *Phys. Med. Biol*., *52*, 41–58.
34. Manić, V., Nikezic, D., Krstic, D., & Manić, G. (2014). Assessment of indoor absorbed gamma dose rate from natural radionuclides in concrete by the method of build-up factors. *Radiat. Prot. Dosim*., *162*(4), 609–617. DOI: 10.1093/rpd/nct358.
35. Mustonen, R. (1984). Methods for evaluation of radiation from building materials. *Radiat. Prot. Dosim*., *7*, 235–238.
36. Unger, L. M., & Trubey, D. K. (1981). *Specific gamma-ray dose constants for nuclides important to dosimetry and radiological assessment*. Oak Ridge, Tennessee: Oak Ridge Laboratory. (ORNL/RSIC-45).
37. Nuccetelli, C., Risica, S., D'Alessandro, M., & Trevisi, R. (2012). Natural radioactivity in building material in the European Union: robustness of the activity concentration index *I* and comparison with a room model. *J. Radiol. Prot*., *32*, 349–358.
38. Markkanen, M. (1995). *Radiation dose assessments for materials with elevated natural radioactivity*. Helsinki: Finnish Centre for Radiation and Nuclear Safety. (STUK-B-STO 32).