# Determination of Dose Factors for External Gamma Radiation in Dwellings

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## INTRODUCTION

The largest contribution for the global population exposure to ionizing radiation arises from natural sources, especially from radionuclides present in terrestrial crust. Certain human activities may eventually increase such exposure to significant levels, from the point of view of radiological protection (1, 2).

The presence of natural radionuclides in building materials may lead to an increment of both external and internal radiation exposure of the inhabitants of dwellings built with such materials (3).

External exposure inside dwellings arises from gamma-emitter radionuclides existing in the walls, floor and ceiling of the building. Mathematical models can be used to predict external dose rates inside the room, provided the radionuclide concentration activities in dwelling constituents are known (4, 5, 6).

This paper presents a methodology for theoretical evaluation of external gamma doses due to radionuclides present in the walls of an hypothetical dwelling.

Assuming the dwelling as one compartment of rectangular section and the radionuclides uniformly distributed within each wall, the most commonly cited expression in the literature for the evaluation of the dose is expressed as a linear combination of values of radionuclide concentrations and corresponding dose factors.

Evaluation of dose factors is carried out by varying parameters of the model within a physically reasonable range. Results are presented in a graphical form.

## METHODOLOGY

The dwelling is modeled as three pairs of rectangular sheets with finite thickness. Assessment of doses was performed through the application of photon transport model, taking into account self-absorption and buildup factors. As the external dose due to a particular radionuclide is proportional to its activity concentration, results are presented as dose factors, defined as the quotient of absorbed dose rate ($nGy.h^{-1}$) at a defined point inside a compartment, due to the presence of a given radionuclide and its daughter products, by the activity concentration ($Bq.kg^{-1}$), for such radionuclide in the walls.

The radionuclides are assumed to be uniformly distributed in the building materials. It was assumed attenuation and radiation buildup factors of standard concrete.

The studied nuclides are $^{40}K$, $^{226}Ra$, and $^{232}Th$, taking in account, for dose calculations, all gamma emitters from $^{226}Ra$ and $^{232}Th$ decay chains.

Sensitivity of the model is estimated by varying four of its input parameters within a reasonable range of applicability, while leaving all other parameters at fixed selected values. The studied parameters and respective ranges of variation were: for thickness, 5 to 60 cm; for density, 0.5 to 4 $g.cm^{-3}$; for the room length, 1.5 to 10 m; and for the distance of the point of calculation to the nearest wall, 10 cm to the center of the room. Selected values left constant, for each parameter, were: thickness, 20 cm; density, 2.35 $g.cm^{-3}$; room length, 5 m; and for the position of the calculation point, the center of the room.

## GEOMETRIC MODEL

In order to calculate the dose rate at a given point $O$, it is assumed an uniform distribution of radionuclides within a finite rectangular sheet with thickness $t$. The point $O$ is in the origin of the system of Cartesian coordinates, at a distance $h$ of the inferior surface of the sheet. In the present case, the analysis is limited to the situation in which the perpendicular to the sheet that crosses the point $O$ intercepts the sheet. The model geometry is illustrated in Figure 1.

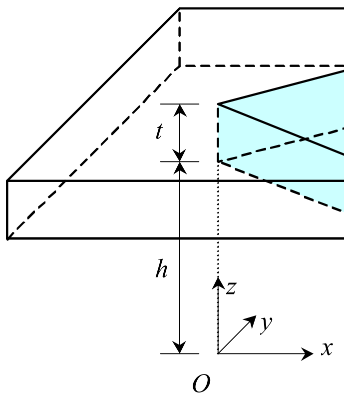


Figure 1. Geometry for the calculation of the dose rate due to radionuclides distributed in a rectangular sheet

The geometric modeling of a compartment can be made combining different sheets appropriately. The dose rate function is addictive, that is, the final dose is the sum of the doses of the sheets considered separately.

The absorbed dose is calculated by using attenuation coefficients and taking into account the contribution of the scattered radiation by means of expressions for the buildup factor.

The dose rate $\dot{D}$ in the air in a certain point due to the primary and scattered photons, originating from a source uniformly distributed within a generic volume $V$, is given by

$$\dot{D} = \frac{KS}{4\pi}\sum_{i=1}^{n} I_i \frac{\mu_{en}(E_i)}{\rho} E_i \int_V dV \frac{1}{r^2} B(E_i, \mu_i r_m) e^{-\mu_i r_m}$$

where

| | | |
|---|---|---|
| $\dot{D}$ | : | absorbed dose rate ($Gy.s^{-1}$) |
| $K$ | : | constant for units conversion, from ($Bq.g^{-1}$).keV to $Gy.s^{-1}$, equal to $1.602 \times 10^{-13}$. |
| $S$ | : | concentration of a given radionuclide, in the volume $V$ ($Bq.cm^{-3}$) |
| $n$ | : | number of gamma transitions being considered |
| $I_i$ | : | gamma photons per disintegration for the $i$-th gamma transition |
| $\mu_i$ | : | gamma attenuation coefficient corresponding to the energy $E_i$ ($cm^{-1}$) |
| $\frac{\mu_{en}(E_i)}{\rho}$ | : | mass energy-absorption coefficient corresponding to the energy $E_i$ ($cm^2.g^{-1}$) |
| $\rho$ | : | density of the material ($g.cm^{-3}$) |
| $E_i$ | : | energy of the $i$-th gamma transition (keV) |
| $r_m$ | : | distance traveled in the absorber material (cm) |
| $r_a$ | : | distance traveled in the air (cm) |
| $r$ | : | total distance, from the point of interest to each point of the volume $V$ (cm), equal to $r_m+r_a$ |
| $B$ | : | buildup factor |
| $\mu_i \ r_m$ | : | number of mean free paths in the absorber material corresponding to the energy $E_i$ |

The previous expression should be applied to the geometric model described. By using spherical coordinates, with $O$ in the system origin and the sheet's sides defined by $x1$, $x2$, $y1$, $y2$, the absorbed dose rate is calculated in the point $O$, for a given radionuclide or a radioactive series.

Dividing the sheet volume in 4 parts, each one being a prism with triangular base, it is obtained the variation range for the integral in the expression for the dose, as shown in Figures 2 and 3.

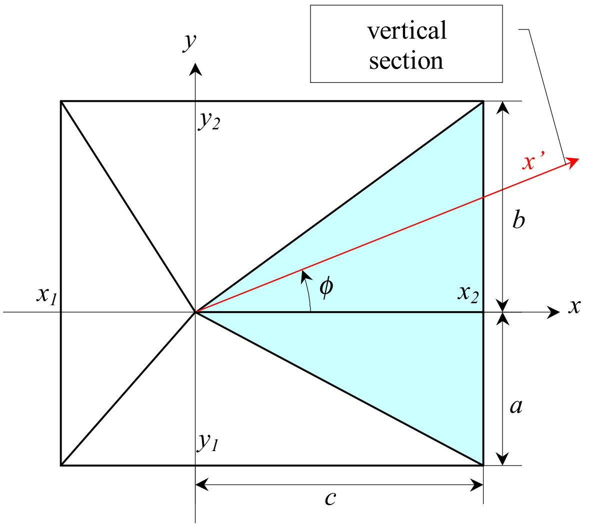


Figure 2. Top view of the sheet, showing its division in four parts for the integration.

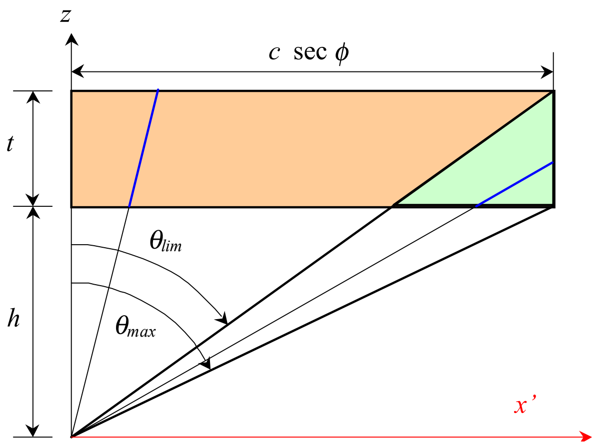


Figure 3. Vertical section x' at a generic angle $\phi$, illustrating the variation range of $r$ (in blue) and angle $\theta$.

Thus, the absorbed dose rate is given by

$$\dot{D} = \frac{KS}{4\pi} \sum_{i=1}^{n} I_i \frac{\mu_{en}(E_i)}{\rho} E_i \sum_{V=1}^{4} A_{i,V}$$

where $A_{i,V}$ is defined, for each gamma transition $i$ and part $V$ of the sheet, by the expression

$$\int_{\phi=\arctan\frac{-a}{c}}^{\arctan\frac{b}{c}} \left[ \int_{\theta=0}^{\theta_{\lim}(\phi)} \int_{r=h\sec\theta}^{(h+t)\sec\theta} f(r,\theta)\,dr\,d\theta + \int_{\theta_{\lim}(\phi)}^{\theta_{\max}(\phi)} \int_{r=h\sec\theta}^{c\sec\phi\,\mathrm{cosec}\theta} f(r,\theta)\,dr\,d\theta \right] d\phi$$

where

$$\theta_{\lim}(\phi) = \arctan\frac{c\sec\phi}{h+t}\,; \qquad \theta_{\max}(\phi) = \arctan\frac{c\sec\phi}{h}$$

$$f(r,\theta) = B(E_i, \mu_i r_m)\exp(-\mu_i r_m)\sin\theta$$

$$r_m = r - h \sec\theta$$

and the $a$, $b$, $c$ values are defined for each part $V$.

The dose rate calculated taking in account only non-scattered photons would lead to an underestimate of the real dose. A commonly used way for taking into account the scattered radiation contribution is the introduction of the so-called buildup factor, $B(E, \mu\, r)$, where $B$ is the buildup factor, $E$ is the energy of the primary photon and $\mu\, r$ is the number of mean free paths in the scattering material.

Eisenhauer & Simmons (6) presented, only for concrete, calculated values of $B$ for discrete energy values, in the 15 keV—15 MeV range, and for mean free paths from 0 to 30. In order to obtain $B$ values for both generic energy values and mean free paths within those ranges, in the present paper the evaluation of the buildup is made by means of an interpolation applied to the logarithms of the energy and to the mean free paths, on the $B$ values supplied in that paper.

So, given the values of $B_{11} = B(E_1, \mu\, r_1)$, $B_{12} = B(E_1, \mu\, r_2)$, $B_{21} = B(E_2, \mu\, r_1)$ and $B_{12} = B(E_2, \mu\, r_2)$, the buildup factor $B$ for $E$ and $\mu\, r$, for $E_1 < E < E_2$ and $\mu\, r_1 < \mu\, r < \mu\, r_2$, is fitted by

$$B(E, \mu\, r) = B(E, \mu\, r_1)\exp\left[\ln\frac{B(E, \mu\, r_2)}{B(E, \mu\, r_1)}\frac{\mu\, r - \mu\, r_1}{\mu\, r_2 - \mu\, r_1}\right]$$

The values of $E_i$ that are the values of energy of the primary gamma transitions, compose a spectrum of discrete values, while the values of $r$ vary continuously when one performs the integration of the function that contains the $B(E, \mu_i\, r_m)$ factor. Thus, the interpolation is made first on the logarithms of the tabulated values of energy, in order to obtain a set of buildup factors for the tabulated values of $r$, for each value of primary gamma energy; when it is made the calculation of the dose factor, that data set is used to obtain the values of $B$ for $r$ varying continuously.

It was obtained values for the mass attenuation and for energy-absorption coefficients for discrete values of energy, within the 1 keV to 20 MeV range (8). Again, it is used the interpolation in order to obtain the coefficients for a generic energy $E$. Thus, the coefficients for $E_1 < E < E_2$ were obtained by means of the expression

$$\mu(E) = \mu(E_1)\exp\left\{\ln[\mu(E_2)/\mu(E_1)]\frac{\ln(E/E_1)}{\ln(E_2/E_1)}\right\}$$

Defined the methodology for the calculation of the dose, the dose factor $q$ for each radionuclide or radioactive series is given by the expression

$$q = \frac{\dot{D}\rho}{S}$$

In the previous expression, it is assumed the use of an unique type of material in the walls, with both homogeneous density $\rho$ and activity by volume $S$.

## RESULTS AND DISCUSSIONS

In each simulation is studied the variation of one parameter of the model, in a physically reasonable range, keeping all other parameters at selected values. Table 1 shows the studied parameters, where the first line shows the constant values and the second line the respective variation ranges.

Table 1: variation ranges and constant values used for each model parameter

| | Thickness (cm) | Density ($g.cm^{-3}$) | Dimensions (m) | Distance from the point of interest to the nearest wall (m) |
|---|---|---|---|---|
| Fixed value | 20 | 2.35 | L x W x H [a]: 5 x 4 x 2.8 10 x 4 x 2.8 | Center of the compartment |
| Range | 5 – 30 | 0.5 – 4 | L [a]: 1.5 – 10 | 0.1 – 2.5 |

[a] L: length; W: width; H: height.

The graphs show the dose factors for each radionuclide.

By varying the thickness of the walls in the range from 5 cm to 60 cm, it is observed that the dose factors initially increase rapidly and then tend to an upper limit. For a 35 cm thickness, the dose factor already reaches a value only 2% inferior to the corresponding factor for a 60 cm thickness (Figure 4).

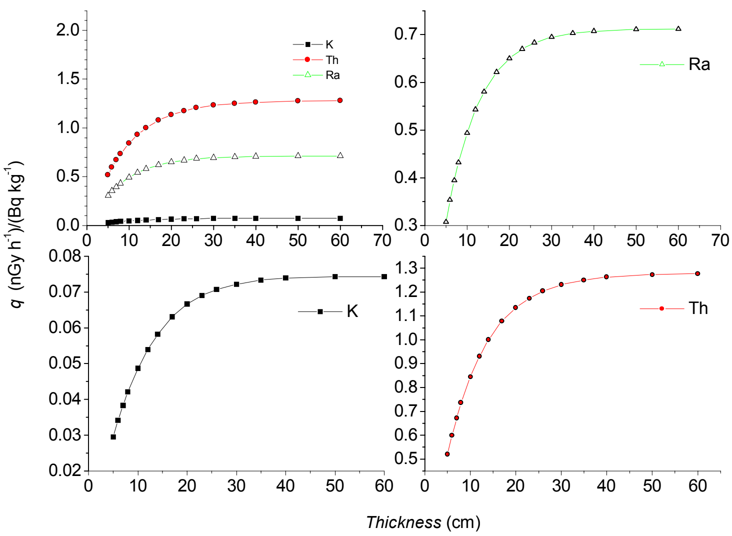


Figure 4. Graph of the dose factors $q$ in function of the thickness of the walls of the compartment.

By varying the density of the material, the dose factor increases linearly with the density of the wall material (Figure 5). In fact, this property can be directly deduced from the expression for the dose factor.

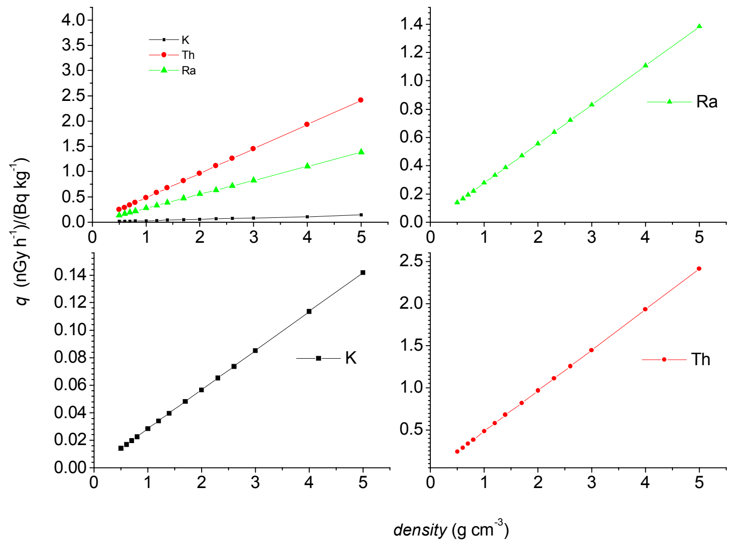


Figure 5. Graph of the dose factors $q$ in function of the material density

Keeping constant both the height, at a value of 2.8 m, and the width, at 4 m, and varying the length within the 1.5 m—10 m range, the dose factor slightly decreases (2%), from 1.5 m to 3 m, and slightly increases from 3 m to 10 m until a value 4% greater than the minimum at 3 m (Figure 6).

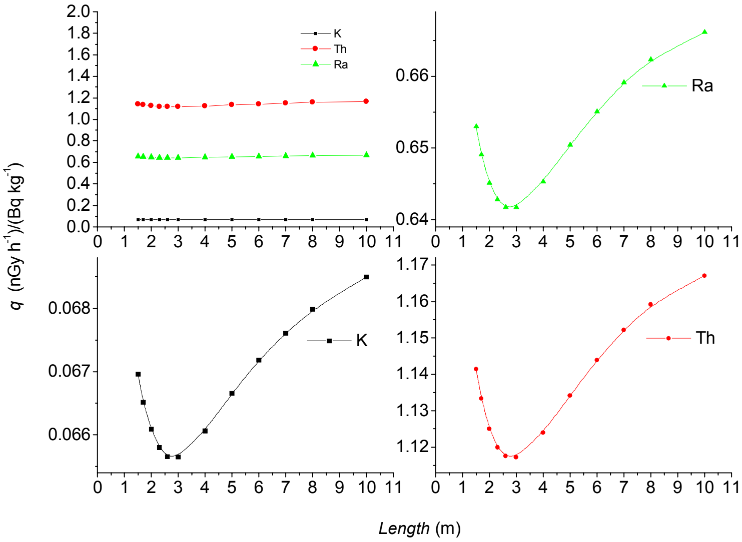


Figure 6. Graph of the dose factors $q$ in function of the compartment length

Calculating the dose factor in positions along the axis that crosses the centers of two parallel walls, starting from the center of the compartment and approaching to a position 10 cm off the wall, it is observed that, for a compartment of 5 m x 4 m x 2.8 m (Figure 7), the dose factor remains practically constant (within an 1% range) from the center of the compartment to approximately 1 m off the wall; for a distance from the wall varying in the 1 m—0.1 m range, the dose factor slightly increases, but no more than 6% of the value at the compartment center.

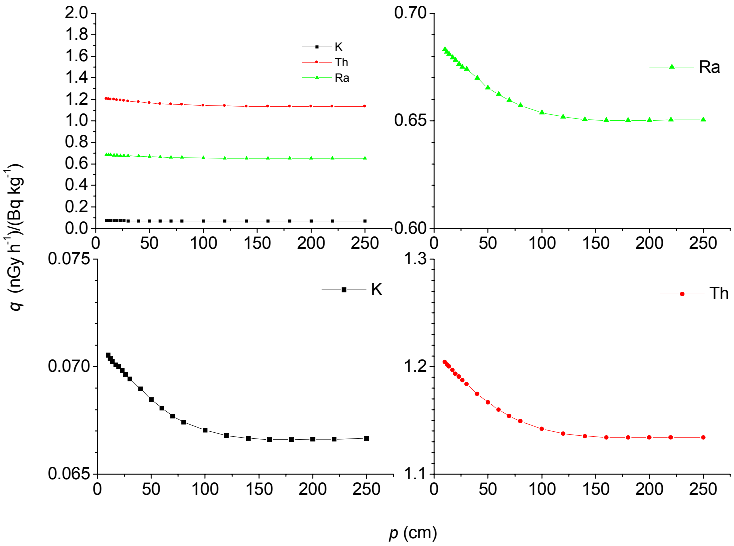


Figure 7. Graph of the dose factors $q$ in function of the distance $p$ from the point of interest to the closest wall (5 x 4 x 2.8 m compartment).

For a 10 m x 4 m x 2.8 m compartment (Figure 8), there is a change in the behavior of the curve.

From the center of the compartment (5 m apart from the wall) and approaching the wall, the factor decreases until a minimum value at a point 1.4 m off the wall (1.6% less than the value in the center). From 1.40 m to 10 cm off the wall, it increases until a value 3.5% greater than that in the center.

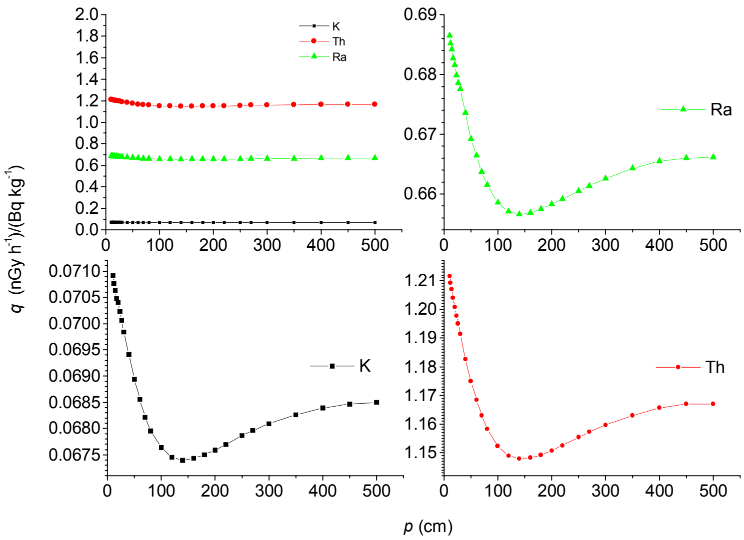


Figure 8. Graph of the dose factors $q$ in function of the distance $p$ from the point of interest to the closest wall (10 x 4 x 2.8 m compartment).

# CONCLUSIONS

It was found that the most sensitive parameters are the density, that shows a linear dependency to the dose factor, as expected, and wall thickness. Model sensitivity for the room length and distance from the calculation point to the walls was negligible in the studied range of variation. It was also observed that the qualitative behavior of the dose-parameters relations does not change significantly from one radionuclide to another, for all the studied parameters, although the gamma energy spectra of the considered radionuclides vary considerably.

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