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# Technical Considerations for Radiation Detection Equipment Specifications

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***Abstract:***
*Many types of radiation detection instruments are widely used for nuclear security purposes. These instruments are used either only to detect radioactive sources or they can identify existing isotopes as well. Usually, manufacturers of radiation detectors provide some specifications for the detectors that allow users to determine the required ones. However, the user may need first to draw some figures for quantities like flux or dose rate due to specific radiation sources with targeted radioactivity in order to select the suitable detector that meets his requirements. This article presents a set of fitted equations that could be directly used to calculate radiation flux and dose rates due to radioactive sources and nuclear materials which are most commonly encountered in nuclear security events. The equations could be applied for a range of distances between 50 to 800 cm with an accuracy of better than 3%. Radiation shielding due to the existence of different thicknesses of iron metal are also considered. The calculations were performed using the general Monte Carlo Code MCNP5. This work may help users to determine the instrument with the desired specification.*
***Key Word****:Radiation detection instrument, nuclear security, flux, MCNP5*

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## I. Introduction

Large numbers of radioactive sources and large amounts of nuclear material are used worldwide in areas such as scientific research, health, agriculture, education and industry. If such material falls out of regulatory control, there is a probability that they could be used in criminal or unauthorized acts. The potential consequences of such acts depend on the material's amount, form, composition and activity. A nuclear security regime should be established to deter and detect these acts. The design and implementation of sustainable nuclear security systems and measures should be based on a process of identifying and assessing threats and risks and developing and evaluating alternatives to meet nuclear security application requirements [1]. Radiation detection instruments are used to detect any unauthorized movement of radioactive materials. The technical specifications of these instruments describe how they should operate, and should be based on a State's requirements for nuclear security applications. The detection or identification capabilities of an instrument are normally expressed in terms of the instrument's ability to make a successful detection or identification given a minimum source activity or dose rate. Specifically, the international standard for radiation portal monitors is expressed in terms of source activity [3], and the international standard for handhelds is expressed in the dose rate received by the instrument [4]. However, both of these values need to be converted into flux in order to be used as a basis for instrument specifications. This paper discusses the relevant calculations and provides fitted equations for several example threat materials that are most commonly encountered in nuclear security events. The equations are based on numerical calculations that allow the planner to easily calculate the flux and the dose rates at a range of distances due to some radioactive sources a. Effects of source moving and radiation background levels are not explicitly considered in this work.

## II. Material And Methods

### 1. Calculation of Flux and Dose Rates

### 1.1 Direct use analytic forms for gamma and neutron point sources

Analytic equations could be directly used to calculate radiation flux due to point radioactive sources. For example, the following equation can be used to calculate radiation flux $\boldsymbol{\varphi}$ (*γ/(second*area)*) at distance "R" from a gamma point source [6]:

$$\varphi = \frac{b \cdot A}{4 \cdot \pi \cdot R^2}, \qquad (1)$$

where:
***A*** is the activity of the source (Bq); and ***b*** is the branching ratio for the gamma ray of interest.
This equation ignores the effects due to attenuation and build-up factors of air. If a shielding material of thickness "*t*" exists, attenuation and build-up factors should be multiplied by the right-hand side of equation (1). In this case the following equation can be used to calculate the flux [6]:

$$\varphi = \left(\frac{b \cdot A}{4 \cdot \pi \cdot R^2}\right) e^{-\mu t} . B \ , \qquad (2)$$

Where:
$\boldsymbol{\mu}$ is the linear attenuation coefficient of the shielding material at the specific gamma energy and ***B*** is the build-up factor.
For isotopes with multiple gamma energy lines, summation over all lines must be taken in order to obtain the total flux. Similar equations are also available for neutron flux calculations. The flux $\boldsymbol{\varphi}$ (*n/(second*area)*) at a point distant ***R*** from the spontaneous fission source is [6]:

$$\varphi = \frac{m \cdot A_S \cdot Y_n}{4 \cdot \pi \cdot R^2} \ ; \qquad (3)$$

where:
***m*** is the mass of the isotope (***g***);$\boldsymbol{A_S}$ is the specific activity of the isotope (***Bq/g***), and$\boldsymbol{Y_n}$is the neutron yield per activity (***n/s*(1/Bq)***).
While the flux at a point distant ***R*** from due to alpha-n reaction source is [6]:

$$\varphi = \frac{m \cdot A_S \cdot Y_\alpha \cdot Y_I}{4 \cdot \pi \cdot R^2} \ ; \qquad (4)$$

where:
***m*** is the mass of the alpha emitter in **g**;
$\boldsymbol{A_S}$ is the specific activity of the alpha emitter (***Bq/g***);
$\boldsymbol{Y_\alpha}$is the alpha yield per activity (***α/Bq***);
$\boldsymbol{Y_I}$is the neutron yield per alpha on target material "***I***" (***n/α***); and
***R*** is the distance from the source
After flux is obtained, Tables for flux-to-dose conversion factors could be used to calculate the dose rate.Although equations (1-4) seem to be simple equations for users to calculate radiation flux, however they could be used as rough estimation. To obtain more accurate estimations, the effects of real setups and situations should be taken into consideration. For example, the given equations do not account for volume of the source, its geometry, and self-attenuation due to its material density. Also, neutron self-multiplication, matrix material effects and energy distributions of spontaneous and stimulated neutrons have to be considered. In addition, some physical constants that are used in the equations must be available. Consequently, it is a necessity to present a set of equations that could be directly used to estimate flux and dose rates more simply.

### 1.2 MCNP Calculation:

Flux and dose rates due to isotropic gamma point sources are calculated using the general Monte Carlo Code MCNP5 [7]. The calculations are performed taking into consideration all factors that may affect the results - including all types of interaction of radiation with matter, self and other-attenuation materials, shielding and radiation build-up. In addition, all gamma energy lines emitted from simulated isotopes with their corresponding branching ratios are considered [8]. The calculations are performed to cover different threat categories based on the information about the threat materials (isotope and amount) already identified in the IAEA Nuclear Security Series No. 24-G[1]. The results of calculations for both flux and dose rate as functions of distance are then fitted to a power function. The gamma sources that are mostly encountered in nuclear security events which are considered in this study are $^{137}$Cs, $^{60}$Co, $^{241}$Am, $^{192}$I and $^{75}$Se. Numerical equations for neutron flux due to the most commonly encountered neutron sources are provided as functions of distance as well. Monte Carlo calculations are used to calculate the flux due to threat quantities of nuclear materials. The nuclear materials are assumed to be metallic element in spherical shapes. For each nuclear material type a specific isotopic mixture is defined as presented in Tables 1 and 2 for uranium and plutonium, respectively.

**Table 1.** Uranium Isotope Distributions

| ***Material Type*** | ***Mass Fraction*** | | | | | |
|---|---|---|---|---|---|---|
| | $^{232}U$ | $^{233}U$ | $^{234}U$ | $^{235}U$ | $^{236}U$ | $^{238}U$ |
| ***HEU 90%*** | *3.01E-11* | *0* | *7.02E-03* | *0.90271* | *3.01E-03* | *8.73E-02* |
| ***LEU 19%*** | *0* | *0* | *0* | *0.19* | *0* | *0.81* |
| ***LEU 5%*** | *0* | *0* | *0* | *0.05* | *0* | *0.95* |
| ***U233*** | *0* | *1* | *0* | *0* | *0* | *0* |

**Table.2.** Plutonium Isotope Distribution

| *Material Type* | *Mass Fraction* | | | | | |
|---|---|---|---|---|---|---|
| | $^{236}Pu$ | $^{238}Pu$ | $^{239}Pu$ | $^{240}Pu$ | $^{241}Pu$ | $^{242}Pu$ |
| *WGPu 94%* | *1.50E-10* | *1.50E-04* | *0.93477* | *6.00E-02* | *5.00E-03* | *1.00E-04* |

Although the considered neutron sources have different energy distributions (and consequently different average neutron energies), calculations showed that the calculated flux in air in the range 50-800 cm for 0.5 and 10 MeV average neutron energies differ only within 2.7%. Therefore, as a good approximation, average neutron energy of 3.0 MeV is used in the calculations. The MCNP built-in F2 tally (flux averaged over surface) is used to calculate gamma and neutron flux. While dose rate due to gamma radiation is calculated using the dose energy (DEn) and dose function (DFn) cards in combination with F2. The flux-to-dose gamma conversion factors (ANSI /ANS-6.1.1-1977) are considered [9].

## III. Results And Discussion

### *1.* Gamma Flux Equation for Unshielded and shielded Sources

The following equation can be used to calculate gamma flux ($s^{-1}.cm^{-2}$) at any distance in the range of 50 – 800 cm due to an isotropic radioactive point source with activity $A$Bq,

$$Flux(A,x) = A\,.\,[\,a(1+x)^b - c.\,e^{-d.x}]$$

where *a, b, c* and *d* are fitting parameters given in Table .3.

**Table 3**. Fitting Parameters for Flux Calculation Due To Point Sources.

| Isotope | Iron thickness (cm) | Fitting parameters | | | |
|---|---|---|---|---|---|
| | | a | b | c | d |
| $^{137}$Cs | 0 | 0.08038 | -2.0128 | 7.617E-7 | 0.01758 |
| | 0.5 | 0.0784 | -2.027 | 1.47E-6 | 0.0178 |
| | 1 | 0.07327 | -2.034 | 1.692E-6 | 0.01802 |
| | 2 | 0.05781 | -2.044 | 1.635E-6 | 0.01818 |
| | 4 | 0.0312 | -2.058 | 1.087E-6 | 0.0184 |
| $^{60}$Co | 0 | 0.1776 | -2.0146 | 1.523E-6 | 0.01677 |
| | 0.5 | 0.18674 | -2.0256 | 2.969E-6 | 0.0176 |
| | 1 | 0.18202 | -2.0318 | 3.591E-6 | 0.0178 |
| | 2 | 0.15928 | -2.0407 | 3.954E-6 | 0.01806 |
| | 4 | 0.109 | -2.0538 | 3.40E-6 | 0.017 |
| $^{241}$Am | 0 | 0.03613 | -2.0314 | -9.033E-8 | 0.00445 |
| | 0.5 | 3.741E-4 | -1.992 | -3.00E-7 | 0.07654 |
| | 1 | 5.7336E-6 | -2.0449 | -1.428E-11 | 0.00381 |
| $^{192}$Ir | 0 | 0.19646 | -2.0084 | 2.083E-6 | 0.01739 |
| | 0.5 | 0.18517 | -2.0254 | 3.855E-6 | 0.01799 |
| | 1 | 0.15819 | -2.0329 | 3.950E-6 | 0.01809 |
| | 2 | 0.10482 | -2.0427 | 3.119E-6 | 0.01818 |
| | 4 | 0.04038 | -2.0562 | 1.435E-6 | 0.01824 |
| $^{75}$Se | 0 | 0.15392 | -2.0052 | 1.825E-6 | 0.01782 |
| | 0.5 | 0.12022 | -2.0218 | 2.591E-6 | 0.01815 |
| | 1 | 0.08548 | -2.0295 | 2.178E-6 | 0.01824 |
| | 2 | 0.04373 | -2.0401 | 1.335E-6 | 0.01835 |
| | 4 | 0.01208 | -2.0538 | 4.365E-7 | 0.01834 |

### *2.* Gamma dose rate Equation for Unshielded point Sources

The following equation can be used to calculate dose rate (µSv/hr) at any distance in the range of 50 – 800 cm due to an isotropic radioactive point source with activity $A$Bq,

$$Dose\ rate(A,x) = A\,.\,[\,a(1+x)^b - c.\,e^{-d.x}]$$

where *a, b, c* and *d* are fitting parameters given in Table.4.

**Table 4**. Fitting Parameters For Dose Rate Calculation Due To Point Sources.

| Isotope | Fitting parameters | | | |
|---|---|---|---|---|
| | a | b | c | d |
| $^{137}$Cs | 0.00116 | -2.02169 | 8.45796E-9 | 0.01863 |
| $^{60}$Co | 0.00423 | -2.02266 | 2.86628E-8 | 0.0181 |
| $^{241}$Am | 1.09266E-4 | -2.03183 | -3.02066E-10 | 0.00464 |
| $^{192}$Ir | 0.00178 | -2.01971 | 1.41905E-8 | 0.01866 |
| $^{75}$Se | 8.70917E-4 | -2.01579 | 8.04041E-9 | 0.01874 |

### *3.* Neutron Flux equation for point sources

The following equation can be used to calculate neutron flux ($s^{-1}.cm^{-2}$) at any distance in the range of 50 – 800 cm due to an isotropic radioactive point source with activity *ABq*,

$$Flux(A, x) = A \times Y_i \times 0.09024 \times (1 + x)^{-2.0185}$$

Where $Y_i$ is the neutron yield per source activity as given in Table.5

Table A.Ii.5. Neutron Yield Per Source Activity

| Radioactive Source | $Y_i$ n/(s-Bq) |
|---|---|
| $^{252}$Cf | 1.17E-1 |
| $^{244}$Cm | 1.17E-1 |
| $^{241}$Am-Be | 1.17E-1 |
| Pu-Be | 5.40E-5 |
| $^{226}$Ra-Be | 4.0E-4 |

### *4.* Flux equations for nuclear material

#### 4.1 Flux as a function of U masse ( 90% enriched uranium) at different distances

The following equation can be used to calculate gamma flux ($s^{-1}.cm^{-2}$) for any mass (ranges between 15 to 5000 g) of metallic uranium sphere at different distances.

$$Flux(x_i, m) = a_i m^{b_i} ,$$

where $a_i$ and $b_i$ are the fitting parameters at the distance $x_i$ as given in

Table 6. Fitting Parameters For Flux Calculation At Different Distances Due To Different Masses Of U90%.

| Distance (xi, cm) | Fitting parameters | |
|---|---|---|
| | $a_i$ | $b_i$ |
| **50** | 0.38627 | 0.66796 |
| **100** | 0.09819 | 0.66661 |
| **200** | 0.02459 | 0.66825 |
| **400** | 0.00634 | 0.66629 |
| **800** | 0.00156 | 0.66539 |

#### 4.2 Flux due to different masses of LEU (19% enrichment) as a function of distance

The following equations can be used to calculate gamma flux ($s^{-1}.cm^{-2}$) for 1000 and 10000 g of metallic uranium sphere (19% enrichment) at different distances.

(a) For 1000 g of U19% enrichment (CAT3)

$$\mathrm{Flux(x)} = 19294\ \mathrm{x}^{-1.9877}$$

(b) For 10000 g of U19% enrichment (CAT2)

$$Flux(x) = 90408\ x^{-1.9893}$$

#### 4.3 Flux due to 10000g of LEU (5% enrichment) as a function of distance (CAT3)

The following equations can be used to calculate gamma flux ($s^{-1}.cm^{-2}$) for 10000g of metallic uranium sphere (5% enrichment) at different distances.

$$Flux(x) = 23812\ x^{-1.9893}$$

#### 4.4 Flux due to different masses of U233 as a function of distance

The following equations can be used to calculate gamma flux ($s^{-1}.cm^{-2}$) for different U233 masses in form of metallic sphere at different distances.

$$Flux(g_i, x) = a_i (1 + x)^{b_i} - c_i . e^{-d_i . x} \ ;$$

where $g_i$ (i=1, 2, 3) refers to the category ($CAT_i$) of the U233. The values of *a, b ,c* and *d* parameters for each category are given in Table.7

**Table7.** Fitting Parameters For Flux Calculation At Different Distances Due To Different Masses Of U233.

| Category | U233 mass (g) | Fitting parameters | | | |
|---|---|---|---|---|---|
| | | a | b | c | d |
| $g_1$ | 2000 | 880486 | -2.03242 | -1.73023 | 0.00401 |
| $g_2$ | 500 | 347901 | -2.03193 | -0.6924 | 0.00404 |
| $g_3$ | 15 | 33239 | -2.03187 | -0.06597 | 0.00405 |

### *5.* Gamma and neutron flux due to different masses of Pu as a function of distance

The following equations can be used to calculate gamma and neutron flux ($s^{-1}.cm^{-2}$) for different Pu masses in form of metallic spheres at different distances.

$$Flux(g_i, x) = a_i(1+x)^{b_i} - c_i . e^{-d_i . x} \quad ;$$

where $g_i$ (i=1, 2, 3) refers to the category ($CAT_i$) of Pu. The values of *a*, *b* ,*c* and *d* parameters for each category are given in Table 8.

**Table 8.** Fitting Parameters For Flux Calculation At Different Distances Due To Different Masses Of Pu.

| | *Category* | *Pu (g)* | *Fitting parameters* | | | |
|---|---|---|---|---|---|---|
| | | | *a* | *b* | *c* | *d* |
| **Gamma** | $g_1$ | 2000 | 1.36253E6 | -2.0088 | 16.86 | 0.01815 |
| | $g_2$ | 500 | 492136 | -1.9948 | -114045 | 0.0674 |
| | $g_3$ | 15 | 54514 | -2.03066 | -0.09383 | 0.00392 |
| **Neutrons** | $g_1$ | 2000 | 25795 | -2.0084 | 0.2112 | 0.0156 |
| | $g_2$ | 500 | 4096 | -2.011 | 0.017 | 0.0157 |
| | $g_3$ | 15 | 89.86 | -2.0098 | 0.001 | 0.0186 |

## IV. Conclusion

Equations for flux and dose rate calculations for radioactive and nuclear materials that are most likely encountered in nuclear security events are presented. All given numerical equations can be used to calculate flux and dose rate with an accuracy of better than 2%, except for $^{241}$Am source for which an accuracy of better than 2.8% as a result of its low gamma energy. The information provided in this article can now be combined with the planner's knowledge about the operational environment to determine the detector with suitable specifications. This work could be extended to include other types of radioactive sources and forms of nuclear materials.

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