Development and calibration of a portable detection device for in vivo measurement of high-energy photon emitters incorporated by humans

Soares[a] A. B., Arbach[a] M. N., Lucena[a] E. A., Dantas[a] A. L. A., Dantas[a] B. M.

[a] *Laboratório de Monitoração Interna, Instituto de Radioproteção e Dosimetria, 22780-160, Rio de Janeiro, Rio de Janeiro, Brazil*

alexandrebaso@globo.com

ABSTRACT

This work presents the evaluation of the applicability and sensitivity of a portable detection device specially designed for in vivo measurement of high-energy photon emitters in the human body. The calibration was performed at the In-Vivo Monitoring Laboratory of the IRD. The equipment consists of a lead-collimated NaI(Tl) 3"x3" scintillation detector assembled on a tripod. The detector and its compact associated electronics are connected via USB cable to a portable PC. Spectrum acquisition and analysis is controlled by specific commercially available software. The calibration was performed using a standard liquid source of ^{152}Eu contained in 3 L polyethylene bottles. The evaluation of the system is based on the estimation of the minimum committed effective doses associated to the minimum detectable activities, calculated using current biokinetic and dosimetric models available in the literature. The dose detection limits for selected radionuclides of interest in an emergency scenario have shown to be far below 1 mSv allowing the system to be useful in accident situations.

Keywords: Internal monitoring, Internal dosimetry, Bioassay.

1. INTRODUCTION

Many international organizations are worried about the nuclear terror events. In 2016, the International Atomic Energy Agency (IAEA) published the Information Circular in attention for Nuclear and Radiological Terrorism [1]. That document recognizes the need for ensuring adequate nuclear emergency preparedness and response capabilities. Radiologic terrorism events include Radiological Dispersal Devices (RDD), referred as *dirty bombs*, which employ conventional explosives to disperse radioactive materials [2]. In Brazil, a government body was created for this purpose including civil and military entities [3]. The hospitals also should be prepared to identify and quantify intakes of radionuclides of interest in order to assess the committed effective doses associated to the internal exposure. In such situations a fixed whole body counting system may not be suitable to attend an increased demand of in vivo measurements [4]. Therefore, portable detection devices are more versatile and effective equipments to be employed for prompt response to emergencies involving intakes of radionuclides.

2. MATERIALS AND METHODS

The detection system consists in a NaI(Tl)3"x3" detector assembled on a tripod and connected to the compact electronics. Spectrum acquisition and analysis is software-controlled. Firstly, the system was calibrated with a ^{152}Eu standard point source. The "Energy *vs* Channel" curve was obtained. After, an ^{152}Eu standard liquid source (7,177 kBq/g) was prepared to be apply in a thorax simulator composed by five 3 L polyethylene bottles. This simulator is low cost device, resistant and applicable to our purpose. The liquid source known mass (mass differences method) was transferred into a volumetric flask whose activity was calculated. Deionized water was introduced into this flask up to the 25 mL level being the activity per L calculated. The 4 mL volume was transferred to each polyethylene bottle contained 3 L deionized water volume. The bottle set activity was calculated which simulate a radionuclide incorporation. The bottle set contained radioactive liquid was positioned at 0,05 m distance from detector surface (Figure 1).

Figure 1*: Portable detection device and the thorax simulator.*

Six Regions-of-Interest (ROI) were defined and a series of five 900-seconds counts were performed. The average net count was recorded in each ROI. The efficiencies were calculated as follows:

$$\varepsilon = (N_{ctg} / \Delta t) / (A_c \times I_\gamma) \tag{1}$$

where N_{ctg} is the average count in each ROI, Δt is count time, A_c is the source activity and I_γ is the relative intensity of the [152]Eu photon. Then, "Efficiency *vs* Energy" curve was obtained. For evaluation purposes, ^{137}Cs was selected as radionuclide possibly present in an emergency scenario. Thus, the efficiency at 661.6 keV [5] in cps/dps was calculated from the Eff x Energy curve obtained previously. The calibration factor (CF) in cps/Bq was calculated from equation 2, where I_γ is the gamma relative intensity for ^{137}Cs at 661.6 keV:

$$CF = \varepsilon \times I\gamma \tag{2}$$

The evaluation of the system sensitivity is based on the estimation of the minimum committed effective doses associated to the minimum detectable activities [6]. The minimum detectable activity (MDA) in Bq was obtained from equation 3:

$$MDA = (4,65 \times \sqrt{N_{BG}}) \div (t_{BG} \times CF) \tag{3}$$

where N_{BG} is the total background counts (BG) and t_{BG} is the count time in the 661.6 keV region. It was considered 300-seconds to t_{BG}. The MDI is a function of the MDA and depends on the exposure scenario and time elapsed between intake and in vivo measurement. It was calculated as follows (equation 4):

$$MDI = MDA \div m(t) \tag{4}$$

where m(t) is the retention fraction in the compartment of interest, in Bq/Bq [7]. The minimum detectable effective dose (MDED) in µSv was obtained from equation 5:

$$MDED = MDI \times e(g) \tag{5}$$

where e(g) is the dose coefficient associated to the intake scenario adopted in the simulation [8]. In this work, a single intake of ^{137}Cs by inhalation was assumed and the *in vivo* measurement performed 1 day after incorporation.

3. RESULTS AND DISCUSSION

The ^{152}Eu standard liquid source activity per mass was 7.2 kBq/g. The liquid source mass transferred to a volumetric flask was 2.615818 g. This resulted in 18.8 kBq (activity in the flask). In 25 mL, the activity per mL was 0.8 kBq/mL. In each bottle was transferred 4 mL wich mean 3.2 kBq. The activity of the all bottles was 16 kBq (0.4 µCi). The average net count was recorded in each ROI. Knowing the average net count, count time 900-seconds, activity of the bottles and gamma relative intensity (^{152}Eu – each ROI) was calculated the efficiencies to each ROI, as follow in the Table 1.

Table 1: ROI Efficiencies.

	ROI 1	ROI 2	ROI 3	ROI 4	ROI 5	ROI 6
Average net count	15448	13344	3537	4281	5322	5150
Energy (keV)	121.78	344.28	778.90	964.08	1112.10	1408.00
I gamma	0.28596	0.27682	0.13616	0.16243	0.26048	0.21842
Efficiency (cps/dps)	0.00400	0.00357	0.00192	0.00195	0.00151	0.00174

The Efficiency *vs* Energy curve obtained resulted in equation 6, as Figure 2 illustrates.

$$\varepsilon(E) = 0.0042 \ cps/dps * e^{-(0.0008/keV * E)} \ (R^2 = 0.86) \tag{6}$$

Figure 2*: Portable detection device's Efficiency vs Energy curve.*
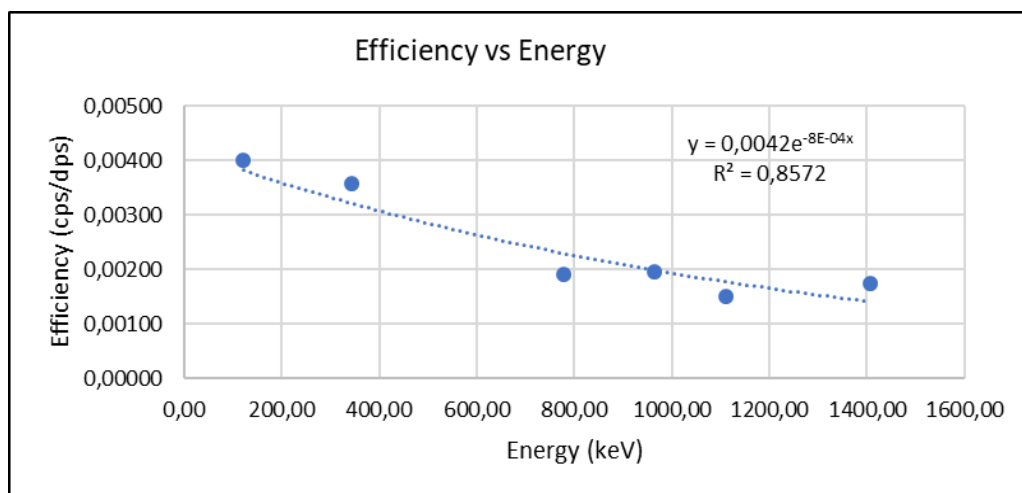
The efficiency at energy 661.6 keV (^{137}Cs), FC, MDA, MDI, MDED were obtained from the equations previously described, considering N_{BG} = 5411 ctg, t_{BG} = 300 s, I_γ = 0.851, m(t) = 1.70 x 10^{-1} Bq/Bq and e(g) = 8.84 x 10^{-6} mSv/Bq.

The results were CF = 0.0021 cps/Bq; MDA = 541 Bq; MDI = 3186 Bq and MDED = 28 μSv.

4. CONCLUSION

The dose detection limits for the selected radionuclide of interest in an emergency scenario have shown to be far below 1 mSv allowing the system to be useful in accident situations or unexpected event. It is recommended to evaluate this technique considering other radionuclides of interest that could be used in malevolent actions. The proposed calibration and measurement protocols could be easily applied to other detection devices owned by civil and military medical facilities in order to improve prompt response capabilities in case of emergency situations.

ACKNOWLEDGMENT

REFERENCES

[1] IAEA – International Atomic Energy Agency. **Communication dated 7 December 2016 received from the Permanent Mission of the Republic of Korea concerning a Joint Statement on Supporting Nuclear and Radiological Terrorism Preparedness and Response Capabilities**. **Information Circular - INFCIRC/904**. IAEA, 2016.

[2] BARNETT, D., J. et al. Understanding Radiologic and Nuclear Terrorism as Public Health Threats: Preparedness and Response Perspectives. **The Journal of Nuclear Medicine**, v. 47, 2006.

[3] EMAER. **Doctrine of the Preparation and Employment of the FAB in Transmission Missions in Chemical, Biological, Radiological and Nuclear Defense (CBRND)**. **DCA 1-6** - Nº 43/3SC2, EMAER, 2014.

[4] GALEEV, Roman et al. Suitability of Portable Radionuclide Identifiers for Emergency Incorporation Monitoring. **Radiation Protection Dosimetry**, v. 173, p. 145-150, 2017.

[5] Avaliable at: <http://www.nucleide.org/Laraweb>. Last Accessed: 2 May 2017.

[6] DANTAS, B. M. et al. A Mobile Bioassay Laboratory for the Assessment of Internal Doses Based on Vivo and in Vitro Measurements. **Health Physics Society**, v. 99, p. 449-452, 2010.

[7] IAEA - International Atomic Energy Agency. **Methods for Assessing Occupational Radiation Doses due to Intakes of Radionuclides**. **Safety Reports Series Nº 37**, IAEA, 2004.

[8] BERTELLI, L., Melo, D. R.; Lipsztein, J., Cruz-Suarez, R. AIDE – Internal Dosimetry Software. **Radiation Protection Dosimetry** 130 (3), p. 358-367, 2008.