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# Calibration of gamma cameras for the evaluation of accidental intakes of high-energy photon emitting radionuclides by humans based on urine samples

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## ABSTRACT

**The prompt response to emergency situations involving suspicion of intakes of radionuclides requires the use of simple and rapid methods of internal monitoring of the exposed individuals. The use of gamma cameras to perform in vivo measurements was investigated by the US-CDC in 2010. The present study aims to develop a protocol for the use gamma cameras to be applied on internal monitoring based on urine samples to evaluate intakes of high-energy photon emitting radionuclides in emergency situations. A gamma camera available in a public hospital located in the city of Rio de Janeiro was calibrated using a standard liquid source of $^{152}$Eu supplied by the LNMRI of the IRD. "Efficiency vs Energy" curves at 2 and 10 cm were obtained. Calibration factors, Minimum Detectable Activities, Minimum Detectable Intakes and Minimum Effective Doses were calculated for $^{131}$I and $^{137}$Cs. It has been concluded that the equipment evaluated in this work presents enough sensitivity for use in accident situations involving intakes of such radionuclides.**

*Keywords:* Internal dosimetry, in vitro measurements, gamma camera.

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## 1. INTRODUCTION

In Brazil, as in many countries, there is a significant number of radioactive sources and, in many situations access control is still ineffective, giving rise to accidents involving the loss and theft of sources. The Brazilian Nuclear Regulatory Board (CNEN) is responsible for authorizing and auditing industrial and medical facilities as well as research and educational institutions where a variety of open sources of radionuclides are routinely handled [1].

The feasibility of the use of gamma cameras for the evaluation of the incorporation of radionuclides by nuclear medicine workers was studied by Dantas et al. [2] in the scope of the IAEA-TC-RLA 049 Project. The US-CDC (Centers for Disease Control and Prevention) also published a document containing instructions for the use of gamma cameras for calculating radioisotope activity in the human body [3]. The calibration of this type of equipment requires the determination of the detector efficiency and the estimation of the minimum detectable activity for each radionuclide of interest.

A preliminary evaluation of the severity of the accident may be carried out in nuclear medicine services participating in a network to be established for this purpose. Thus, the aim of this study is to provide a protocol for the evaluation of intakes of radionuclides by humans in emergency situations through the application of an *in vitro* bioassay method.

## 2. MATERIALS AND METHODS

The evaluation of internal exposure in practices with risk of intakes of radionuclides requires the application of *in vivo* and *in vitro* monitoring techniques, as well as methodologies for the interpretation of bioassay data [4]. Measurements are usually performed with scintillation or semiconductor detectors depending on radionuclide emissions, and require the use of radioactive standards for the calibration and quality control of the detection systems.

The evaluation of the sensitivity of the detection system is based on its minimum detectable activities for the radionuclides of interest in comparison to the expected activity in the organs and tissues to be measured. Such evaluation relies on retention and excretions factors in the body compartments as a function of the time elapsed between intake and measurement. The AIDE Software [5] performs the calculations necessary to provide activities in organs of interest as well as the committed effective dose for a wide variety of radionuclides in different intake patterns, chemical and physical forms.

This work was performed in a public military hospital where a gamma camera Phillips BrightView-XCT was calibrated with a standard liquid source of $^{152}$Eu provided by the National Laboratory for Metrology of Ionizing Radiation (LNMRI-IRD). The source was uniformly distributed in an acid solution contained in a 1L volume polyethylene bottle.

Five consecutive measurements were performed at 2 and 10 cm distance from the source to the front face of the detector, and the count rates were recorded in the regions of interest corresponding to the photons of 40.1, 344.3 and 778.9 keV of $^{152}$Eu. The detection efficiency in each ROI was calculated as follows:

$$Eff = (C/T) / (A \times Ig) \quad (1)$$

where *Eff* is the detection efficiency in the ROI of the photon of interest; *C* is the total counts in the ROI; *T* is the count time; *A* is the activity of the standard source of $^{152}$Eu and *Ig* Ig is the photon yield at the measured energy.

The software used to control the gamma cameras acquisition allows to record counts in the ROIs using the “spectrum mode”.

As shown in Figure 1, a spacer made of Styrofoam was used to allow correct positioning of the plastic bottle in relation to the front face of the gamma camera. It should be highlighted that, in order to increase the sensitivity, all measurements where performed after removing the high energy collimator of the gamma camera.

**Figure 1:** *Calibration geometry: Plastic Bottle containing standard source of $1^{52}$Eu positioned at 10 cm distance to the gamma camera front face*

The *Efficiency x Energy* curves obtained at 2 and 10 cm were used to calculate the efficiencies of the gamma camera to specific radionuclides of interest. In this work efficiencies were calculated for the 364 and 662 keV of $^{131}$I and $^{137}$Cs respectively. Subsequently the Minimum Detectable Activities (MDA) were calculated for the same radionuclides, which are assumed to be of concern in situations related to radiological and nuclear accidents.

Minimum Detectable Activities for $^{131}$I and $^{137}$Cs were calculated at each geometry based on respective room background of the gamma camera at the ROIs and efficiency values as follows:

$$\mathrm{MDA}_{Bq} = (4{,}65\ .\sqrt{\mathrm{N}})/Eff\ x\ Ig \tag{2}$$

where MDA is the Minimum Detectable Activity; *N* is the total counts of the background in the ROI in 1 minute; *Eff* is the detection efficiency (cps/dps) and *Ig* is the photon yield at the measured energy.

The AIDE software allows calculating the activity present in the compartments of interest of the human body as a function of the intake scenario. In this work it has been simulated the inhalation of 1 Bq of $^{131}I$ and $^{137}Cs$. Based on the excretion fractions – m(t) - provided by the software it was calculated the Minimal Detectable Intake for those radionuclides assuming that the activity measured in the urine equals the MDA of the technique.

$$\mathrm{MDI}_{Bq} = \mathrm{MDA} / \mathrm{m(t)}_{inh} \tag{3}$$

where **MDA** is the minimum detectable activity and **m(t)** is the excretion fraction urine.

Using the MDI value and the dose coefficient given by AIDE software, it was calculated the Minimum Detectable Effective Dose to both radionuclides.

$$MDED = MDI * e(50) \tag{4}$$

## 3. RESULTS AND DISCUSSION

Figures 2 and 3 show the *efficiency vs energy* curves obtained using three ROIs of $^{152}Eu$ at 2 and 10 cm distance source-detector respectively.

**Figure 2:-** *Efficiency curves of $^{152}Eu$ at 2 cm distance from source to detector.*

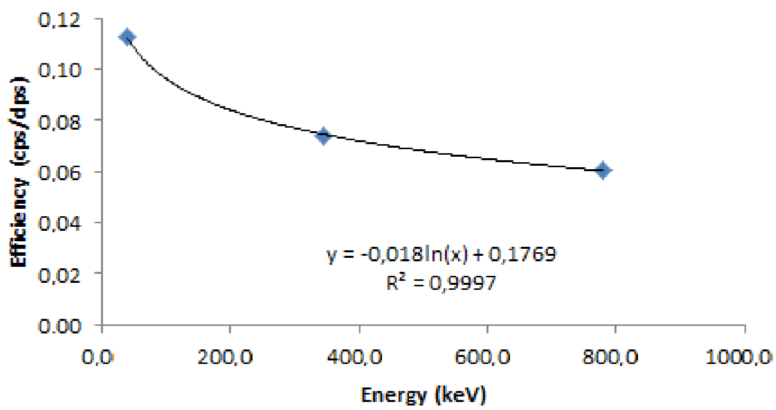

Figure 3: *Efficiency curve of $^{152}$Eu at 10 cm distance from source to detector.*

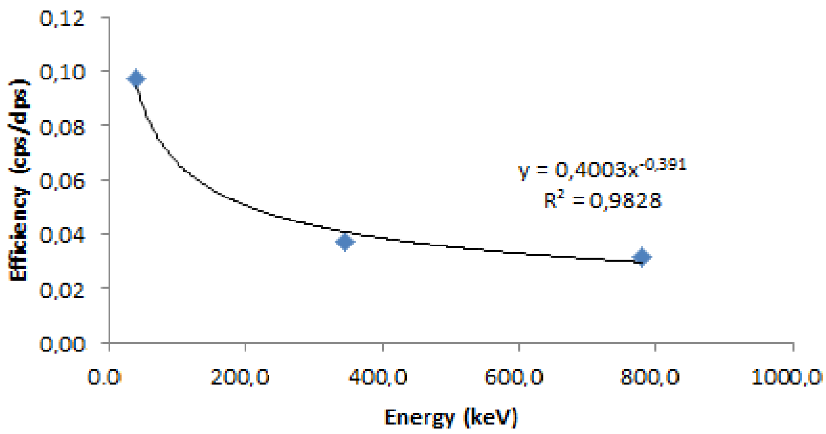


Table 1 presents Efficiency values of the gamma camera for the measurement of $^{131}$I and $^{137}$Cs in urine samples positions at 2 and 10 cm distances from source to detector, calculated based on the Efficiency x Energy curves obtained with the standard source of $^{152}$Eu. The minimal detectable activities for $^{131}$I and $^{137}$Cs are also shown in Table 1.

**Table 1:** Efficiencies and Minimum Detectable Activities of the Gamma Camera for the measurement of $^{131}$I and $^{137}$Cs in 1 L urine samples at 2 and 10 cm distances

| Radionuclide | Geometry (cm) | Efficiency (cps/dps) | MDA (Bq) |
|---|---|---|---|
| $^{131}$I | 2 | 0.07 | 55 |
| | 10 | 0.04 | 99 |
| $^{137}$Cs | 2 | 0.06 | 45 |
| | 10 | 0.03 | 86 |

The gamma camera proved to be a very sensitive device in terms of measurable activity since the MDA values in 1 minute count time are all below 100 Bq both for $^{131}$I and $^{137}$Cs at 2 and 10 cm distances adopted in this study.

Table 2 shows the corresponding MDIs and MDEDs values calculated as a function of the excretion rates provided by the AIDE software in the simulated intake scenario and the MDAs calculated based on the calibration results.

**Table 2:** Minimum Detectable Intakes (MDI) and Minimum Detectable Effective Doses (MDED) of the Gamma Camera for the measurement of $^{131}$I and $^{137}$Cs in 1 L urine samples at 2 and 10 cm distances

| Radionuclide | Geometry (cm) | MDI (Bq) | MDED (mSv) |
|---|---|---|---|
| I-131 | 2 | 105 | 0.002 |
| | 10 | 187 | 0.004 |
| C-137 | 2 | 5882 | 0.03 |
| | 10 | 11171 | 0.05 |

It is observed that the MDEDs to $^{131}$I and $^{137}$Cs at both geometries were far below 1 mSv, which is the annual effective dose limit for public individuals according to Regulation NN 3.01 of the Brazilian Nuclear Energy Commission (CNEN) [7]. Furthermore, according to the IAEA, in certain circumstances, in emergency situations it is tolerated that individual doses could reach higher values. Therefore, this equipment can be considered suitable for monitoring radionuclide intakes in such situations.

The *efficiency vs energy* curve obtained with $^{152}$Eu was used to calculate detection efficiencies of the gamma camera for $^{131}$I and $^{137}$Cs for evaluation purposes in this work. However it is also important to point out that it is possible to use the same curve to determine the activity content in urine samples provided by individuals potentially exposed to any photon emitting radionuclide in the energy range covered by this calibration curve.

## 4. CONCLUSIONS

This work provides a simple, fast and inexpensive protocol for the calibration and use of a gamma camera for bioassay of urine samples. The calibration results show that the gamma camera evaluated in this work presents enough sensitivity to detect $^{131}$I and $^{137}$Cs intakes in accident situations allowing a reliable assessment of internal doses associated to these radionuclides.

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