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Application of a wrist dosimeter prototype for radiation monitoring (^{153}Sm) during a therapeutic procedure simulation

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Abstract
Gamma and beta radiation emitting radiopharmaceuticals are handled in nuclear medicine services, and in many cases there is only individual monitoring of gamma radiation. In this paper, the results obtained using a wrist dosimeter prototype ($CaSO_4$:Dy + Teflon pellets) show that the doses for workers occupationally exposed to beta radiation from ^{153}Sm are not negligible. It is important that this dose is evaluated, and it has to be taken into consideration in the individual monitoring system.

1. Introduction

The use of radionuclides in medical procedures has grown steadily in recent years due to the introduction of new radiopharmaceuticals and equipment. In nuclear medicine the radiopharmaceuticals are prepared in a radionuclide dispensary, and they are administered to patients for a variety of diagnostic, palliative and therapeutic procedures (Vanhavere *et al* 2006).

The nuclear medicine personnel typically receive radiation exposure to the whole body and hands resulting from dose preparation and injection of radiopharmaceuticals, as well as blood pressure monitoring tasks during quantitative studies. All these procedures are carried out manually; therefore, the hands receive a much higher radiation dose than other parts of the body (Tsopelas *et al* 2003, Chruscielewski *et al* 2002). This necessitates proper attention to the extremity doses of radiopharmacy staff.

The most frequently utilised radiopharmaceuticals in nuclear medicine are those labelled with ^{99m}Tc. Among the most common radiopharmaceuticals in Brazil are ^{201}Tl, ^{131}I, ^{18}F, ^{153}Sm, ^{51}Cr and ^{67}Ga. After injection of a radiopharmaceutical, each patient becomes a radiation source for workers in his or her proximity.

The monitoring of workers constitutes a very important task for any radiological protection programme (Wrzesien *et al* 2006). Thermoluminescent (TL) dosimeters are usually considered to be suitable for workers occupationally exposed to ionising radiations (Vanhavere *et al* 2006, Gilvin *et al* 2006, Rimpler and Barth 2007, Brasik *et al* 2007).

A wrist dosimeter prototype for monitoring beta radiation was proposed in a previous work, where the dosimetric features of $CaSO_4$:Dy were studied, and the dose–response curves were established for ^{153}Sm in an acrylic phantom and in the wrist dosimeter prototype (Cecatti and Caldas 2006). As nuclear medicine service workers complain about the discomfort of using finger dosimeters during medical applications (because they make handling syringes less efficient), the option in this work was to use a wrist dosimeter. Moreover, as workers must use gloves during these kinds of operations, loss or contamination of finger dosimeters sometimes occurs when the gloves are removed. The dose at the finger tips is higher than at the wrist, but the wrist is located about 10 cm from the position of the finger dosimeter, and a calculation may give the dose at the finger tips. Syringes are taken to patients in special shielding; the syringe is withdrawn from the shielding only at the moment of application to the patient, to minimise the worker exposure to radiation.

Sm-153-EDTMP has the commercial name Quadramet (EUSA PHARMA, USA). It is produced in Brazil by the Institute for Energy and Nuclear Research (IPEN-CNEN/SP) and is utilised in oncology treatments for pain palliation from bone metastases due to breast, prostate or lung cancer. Its properties are: low medullar toxicity, a half-life of only 46.7 h, beta emission of 284 keV (72%) and gamma emission of 103 keV (28%), providing verification of its biodistribution through the bone scintigraphy. The objective of this paper was to test a wrist dosimeter prototype to monitor workers exposed to beta radiation of ^{153}Sm during simulation of a therapeutic procedure.

2. Materials and methods

Two types of TL dosimeters were utilised in this study: $CaSO_4$:Dy + Teflon pellets (6 mm diameter, 0.8 mm thickness and 50 mg mass) and $CaSO_4$:Dy + Teflon pellets (6 mm diameter, 0.2 mm thickness and 20 mg mass), both produced by the Laboratory of Dosimetric Materials, IPEN (Campos and Lima 1986, 1987).

The beta irradiations were carried out using an unsealed reference source of ^{153}Sm (517.2 kBq), produced by the Center of Radiopharmacy, IPEN, and calibrated in activity by the Nuclear Metrology Laboratory, IPEN. This source presents a short half-life (46.7 h) and mean beta energy of 290 keV. It has also a gamma component of 103.2 keV (28%).

The TL pellets were irradiated in position on a polymethylmethacrylate (Lucite) phantom (120 mm × 120 mm × 15 mm) and on a wrist dosimeter prototype. The detectors were always covered with a plastic foil of 1.20 × mg cm^{-2} area density during the irradiations. Under certain circumstances, in addition to the plastic film, a foil of black polystyrene with a superficial density of 64.0 mg cm^{-2} was also used, in order to avoid the incidence of ambient light.

The TL measurements were obtained using a Harshaw Nuclear System (model 2000A/B) with a linear heating rate of 10 °C s^{-1}. The reading cycle was performed within 26 s, with a constant N_2 flux of 4.0 l min^{-1}. Light emission was integrated in the temperature interval between 140 and 240 °C. The readings were always taken immediately after irradiation.

Figure 1. Wrist badge prototype.

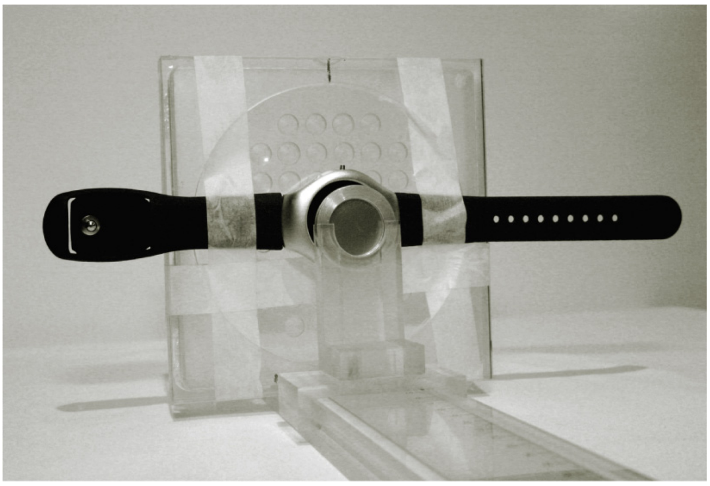

Figure 2. Wrist badge prototype in the experimental set-up during simulation of beta monitoring in the laboratory.

Intermediate thermal treatment of the pellets were performed in an electric oven, with temperature adjustment between 30 and 500 °C, with variation of 1 °C. The pellets were thermally treated at 300 °C for 1 h prior to each re-use.

The wrist dosimeter prototype (figure 1) had its watch core substituted by a plastic plate of expanded polyvinyl chloride (PVC) of diameter 32 mm and thickness 3 mm (Cecatti and Caldas 2006). Four prototypes were prepared for this study. The material of the metallic circular structure of the clock was characterised by an x-ray dispersive energy (EDAX) analyser coupled to a sweeping electronic microscope (Ametek) at the Center for Material Science and Technology, IPEN. This material comprises 82.62% nickel, 10.55% copper and 6.83% tin.

3. Results

3.1. Dose–response curves

In order to simulate the conditions of use of the proposed dosimeter, some applications were realised in the laboratory. During the irradiations the wrist dosimeter was fastened on the phantom so as to make the dosimeter core coincide with the centre of the radiation beam (figure 2). The irradiations were always realised using four TL pellets

Table 1. Relative TL response of beta and gamma radiations of ^{153}Sm. The uncertainties represent standard deviations of the measurements (1σ, $k = 1$).

		TL response (au)		
Detector	Al filter (0.5 mm)	Thin plastic foil	Thin plastic + black polystyrene foils	Radiation type
$CaSO_4$:Dy + Teflon (50 mg)	Yes	0.067 ± 0.021	0.085 ± 0.023	γ
	No	5.193 ± 0.018	6.473 ± 0.019	$\beta + \gamma$
$CaSO_4$:Dy + Teflon (20 mg)	Yes	0.137 ± 0.023	0.124 ± 0.021	γ
	No	9.581 ± 0.019	9.776 ± 0.018	$\beta + \gamma$

of each material at a distance of 10 mm from the source-surface of the wrist dosimeter, so that each TL measurement point would correspond to the average reading of the detectors. The calibration curves obtained under these circumstances, with and without the black polystyrene foil, were shown by Cecatti and Caldas (2006) for the ^{153}Sm source, using the $CaSO_4$:Dy + Teflon (50 mg) and $CaSO_4$:Dy + Teflon (20 mg) pellets, respectively.

Linearity was obtained in both cases. Comparing the calibration curves obtained when the detectors were directly irradiated on a simulator object without the wrist dosimeter prototype, with those obtained when the detectors were irradiated using the wrist dosimeter, an increase in the TL response was observed in the case of irradiation with the prototype. This fact may indicate the occurrence of scattering radiation by the metallic edges. This scattering radiation has to be taken into consideration for the determination of the absorbed dose, or the samples must always be calibrated directly on the prototype (Cecatti and Caldas 2006).

3.2. Gamma and beta radiation of ^{153}Sm

In addition to beta radiation, sources of ^{153}Sm emit gamma radiation of 103.2 keV. For the purpose of determining the absorbed doses due to beta and gamma radiations, a study was carried out to identify these components.

A preliminary test was made using a portable radiation monitor (Geiger–Müller type, MIP 10A, Eurisys Mesures) and a source of ^{57}Co, the standard source with energy closest to the gamma radiation of ^{153}Sm. The purpose of this test was to determine the appropriate thickness of aluminium in the wrist dosimeter to bar the beta radiation of ^{153}Sm and to avoid attenuating the gamma radiation. Various aluminium plates of different thicknesses were used to determine an adequate thickness to avoid attenuating the gamma radiation of the ^{57}Co source in the wrist dosimeter. An Al thickness of 0.5 mm was considered sufficient to shield the beta radiation of the ^{153}Sm source without attenuating the gamma radiation.

To determine the contribution of the gamma radiation, the measurements were taken using the wrist dosimeter prototype and the $CaSO_4$:Dy + Teflon (50 mg) and $CaSO_4$:Dy + Teflon (20 mg) detectors during the irradiations. The detectors were placed inside the prototype on an expanded PVC plate; two of them were kept covered with the 0.5 mm thick Al plate. The irradiations were carried out using also the thin plastic and black polystyrene foils.

Table 1 shows the values of the relative TL response using the thin plastic and the black polystyrene foils to cover and keep the $CaSO_4$:Dy + Teflon (50 mg) and $CaSO_4$:Dy + Teflon (20 mg) detectors in the wrist dosimeter prototype, with the 0.5 mm Al filter.

The contribution to the dose due to gamma radiation (a maximum of 1.4%) is considered negligible; therefore, the use of the Al filter for monitoring the ^{153}Sm radiation is not necessary.

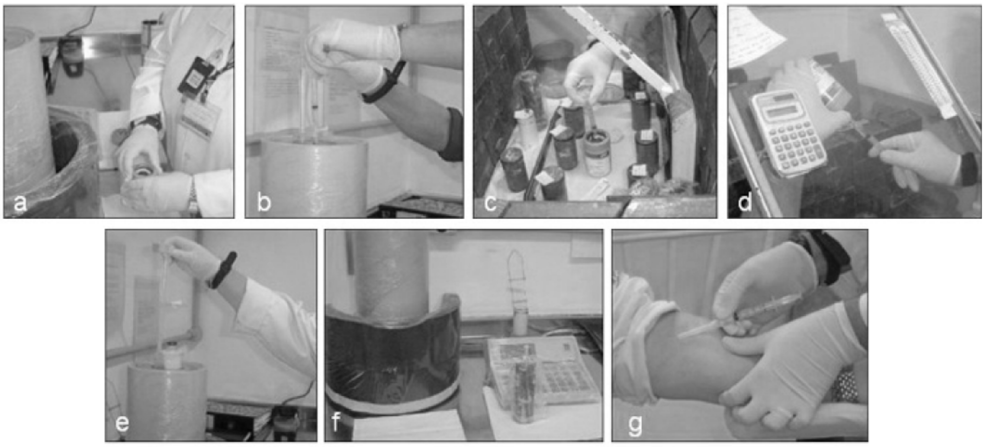

Figure 3. Sequential procedures for preparation of a therapeutic dose with ^{153}Sm in a nuclear medicine service setting: (a) opening of the lead shield to remove the flask containing the ^{153}Sm radiopharmaceutical; (b) determination of the total activity of the sample in the dose calibrator; (c) introduction of the syringe into the flask containing the sample; (d) removal of the portion to be administered to the 'patient'; (e) determination of the ^{153}Sm activity using a dose calibrator; (f) syringe with the ^{153}Sm prepared for administration to the 'patient'; (g) simulation of administration of the therapeutic dose to the 'patient'.

3.3. Application of the dosimeter prototype

The wrist dosimeter prototype was tested by workers at the Center for Nuclear Medicine, University of São Paulo, occupationally exposed to beta radiation of ^{153}Sm.

The field test consisted of the simulation of a therapeutic procedure with ^{153}Sm, without a real patient. Simulations of the preparation, handling and the administration of the radiopharmaceutical to the 'patient' were performed, in such a way as to correspond exactly to a therapeutic procedure.

The same four wrist dosimeter prototypes were utilised during the whole experiment on therapeutic procedures with ^{153}Sm. A procedure was realised every day for six calendar days, always involving the same three professionals: a physician, who receives the radiopharmaceutical, determines the total activity of the sample in the penicillin flask and, thereafter, calculates the sample activity to administer to the patient; the biophysician, who removes the required amount of the sample from the flask directly into the syringe and, thereafter, checks the activity using the dose calibrator; and the nurse, who administers the therapeutic dose to the 'patient'. The handling sequence of the ^{153}Sm sample during a therapeutic procedure is shown in figure 3.

Each wrist badge dosimeter had two samples of $CaSO_4$:Dy + Teflon (50 mg) and two samples of $CaSO_4$:Dy + Teflon (20 mg). A control dosimeter was also delivered to the nuclear medicine service to be kept outside the area of radionuclide handling, to allow the determination of the ambient background radiation.

For the simulation of the therapeutic procedure, a source of ^{153}Sm was produced by the Center of Radiopharmacy, IPEN, calibrated with an initial activity of 3700 MBq (in order to have the necessary activity at the time of application), in a volume of 2.59 ml. The specific activity of the ^{153}Sm sample was 88.30 μBq mg^{-1}. An amount of 1.2 ml of ^{153}Sm was taken to carry out each simulation, suitable for the needs of a person weighing approximately 70 kg; the activity of this was determined using a CRC15 dose calibrator (Capintec) from the Center

Table 2. Values of personal dose equivalent, H_p (0.07), of the workers occupationally exposed to ^{153}Sm, during the simulation procedures carried out in this work. The uncertainties represent standard deviations of the measurements (1σ, $k = 1$).

User	Dosimeter $CaSO_4$:Dy + Teflon	Personal dose equivalent (mSv)
Dosimeter control	(20 mg)	0.17 ± 0.01
	(50 mg)	0.17 ± 0.01
Physician	(20 mg)	0.59 ± 0.02
	(50 mg)	0.61 ± 0.02
Biophysician	(20 mg)	0.45 ± 0.02
	(50 mg)	0.43 ± 0.02
Nurse	(20 mg)	1.93 ± 0.03
	(50 mg)	1.96 ± 0.02

for Nuclear Medicine, University of São Paulo. The activity of the ^{153}Sm samples handled on the first day of the procedure was 5560 MBq, and 350 MBq on the sixth day.

The TL pellets utilised in the therapeutic procedures with the ^{153}Sm radiopharmaceuticals were evaluated by a Harshaw Nuclear System reader; the TL response obtained was corrected for the calibration factor of each detector. There was further correction for the background radiation dose determined through the control dosimeters.

The values of the absorbed doses from the control dosimeter and from those used by the workers occupationally involved in the simulations with ^{153}Sm are shown in table 2. Both kinds of TL sample (20 and 50 mg) proved to be useful for this dosimetric application. The personal dose equivalent values are identical to those of the absorbed doses, because the weighting factor of beta radiation is equal to 1 (ICRP 2007).

Analysing the data shown in table 2, it can be observed that the nurse received a higher dose than the physician and the biophysician. This is because the nurse is closer to the radioactive material, injecting the radiopharmaceutical into the patient without any additional shielding for her extremities, and has a longer period of exposure, according to the procedure; both factors cause difficulties for optimisation of this procedure.

Comparing the values shown in table 2 with the annual personal dose equivalent limits of 500 mSv (41.67 mSv/month) recommended for extremities by the ICRP 103 Report (ICRP 2007), the average doses received by the physician, the biophysician and the nurse were 0.60 mSv, 0.43 mSv and 1.95 mSv, respectively, representing, 1.4%, 1.0% and 4.7%, respectively, of the monthly limit (ICRP 2007).

The nuclear medicine service usually uses protocols prescribing 37 MBq of Sm-153-EDTMP per kg of patient weight. The procedure is carried out once a week on one or two patients, depending on the medical situation. The nuclear medicine service uses a rotation system among the workers; rarely will the same worker apply the injection on two consecutive weeks. Usually the same worker will carry out this operation every 3 or 4 weeks.

For a complete monitoring programme for workers exposed to beta radiation in nuclear medicine, an even better dosimeter system would be a combination of a wrist dosimeter (facing the direction of incidence of the radiation field) and a finger dosimeter, to give a full assessment of the doses.

4. Conclusions

The results obtained in this paper indicate that the equivalent doses of workers occupationally exposed to beta radiation from ^{153}Sm are not negligible, and therefore they have to be taken into consideration. These values represent the doses received by workers in the handling of only ^{153}Sm.

The monthly equivalent doses to the extremities of workers in nuclear medicine services are normally underestimated dose values, determined by means of only individual gamma monitoring; the doses relative to the beta radiation emitting radiopharmaceuticals are not taken into consideration in the cases where they are also handled during medical procedures.

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