EXTERNAL RADIATION DOSES FROM PATIENTS ADMINISTERED WITH RADIOPHARMACEUTICALS
Measurements and Monte Carlo Simulation

by

Abdul Raheem KINSARA [1], ***Samir ABDUL-MAJID*** [1*], ***Wael EL-GAMMAL*** [2], ***Tarik ALBAGHDADI*** [3], ***Abdulraof MAIMANI*** [3], ***and Waleed H. ABULFARAJ*** [1**]

[1]Faculty of Engineering, King Abdulaziz University, Jeddah, Saudi Arabia
[2]Permanent Radiation Safety Committee, Jeddah, Saudi Arabia
[3]Faculty of Medicine, King Abdulaziz University, Jeddah, Saudi Arabia

Scientific paper
DOI: 10.2298/NTRP1403199K

Monte Carlo simulations and dose measurements were performed for radionuclides in the whole body and trunks of different sizes in order to estimate external radiation whole body doses from patients administered with radiopharmaceuticals. Calculations were performed on cylindrical water phantoms whose height was 176 cm and for three body diameters: of 24 cm, 30 cm, and 36 cm. The investigated radionuclides were: ^{99m}Tc, ^{131}I, ^{23}I, ^{67}Ga, ^{201}Tl, and ^{111}In. Measured and MCNP-calculated values were 2-6 times lower than the values calculated by the point source method. Additionaly, the total dose received by the public until a radionuclide is completely disintegrated was calculated. The other purpose of this work is to provide data on whole body and finger occupational doses received by technologists working in nuclear medicine. Data showed a wide variation in doses that depended on the individual technologist and the position of the dosimeter.

Key words: radiation dose, radiopharmaceuticals, MCNP

INTRODUCTION

Patients administered with radioactive materials are retained in hospitals after diagnosis or treatment; the length of their retention depends on the level of radiation so as to ensure that the whole body radiation doses that are likely to be received by the public do not exceed the established limits. While patients administered very short-lived diagnostic radiopharmaceuticals like ^{99m}Tc may be released immediately, those with therapeutic sources of longer effective half-lives and a high specific gamma ray constant like ^{131}I are retained for a few days. Patients with sources of intermediate half-lives and intermediate specific gamma ray constant, such as ^{67}Ga, ^{111}In, and ^{201}Tl, may or may not be retained and the decision is made on a case-by-case basis [1].

A recent ICRP publication [1] recommends that young children, visitors and individuals not engaged in the direct care of patients should be treated as members of the public where a whole body dose limit of 1 mS per year is applied. Accordingly, a good estimate of the dose from the patient must be made for dose assessment. Overestimations of the radiation dose can lead to an unnecessarily prolonged stay in the hospital, while the extra cost can put a burden and cause discomfort to the patient. Moreover, overestimation can be associated with an extra dose to the medical staff. The recommendation is that the decision to hospitalize or release a patient should be made on an individual case basis [1, 2].

The calculation method suggested in US Regulatory Guide 8.39 [3] on the release of radiation patients administered with radioactive materials, based on NRC Regulation 10 CFR 35.75 [4], views all activity inside the patient as a point source subject to the inverse square law; physical half-life is considered instead of the effective half-life. This method of dose calculation used by some researchers [5, 6], has been questioned in a recent IAEA safety report [2] and by many other investigators [6-9], as well. In point source calculations, self-absorption and buildup of radiation within the patient's body are neglected. Buildup and self-absorption depend mainly on the radiation energy and body size. The reference values for the body height of European adults were taken as 176 cm for males and 163 cm for females [10, 11]; data for some Asian populations [12, 13] are also available.

Local customs should also be considered when the protection of the public is concerned. In breastfeed, an infant will receive a major dose due to close contact with

* Corresponding author; e-mail: salzaidi@kau.edu.sa
** Currently at King Abdullah City for Atomic and Renewable Energy, Riyadh, Saudi Arabia

the mother administered with radiopharmaceuticals [14]. In order to provide appropriate instructions and a better design of hospital facilities such as waiting rooms, the dose received by the public was studied by many researchers [5, 7, 14, 15]. Because ^{131}I is more harmful than other radioactive elements, it has been subjected to more investigation [7-9, 16, 17]. The guidance for working with patients administered with unsealed radioactive materials was given in ICRP 94 [1].

In this work, MCNP codes were used for whole body dose calculations to be compared with the measured values, so as to study the effect of the body size on the dose from the patient and for comparison with point source calculations. This approach can lead to a better understanding and assessment of the dose to workers and the public. It can also provide appropriate instructions for caregivers and medical staff and result in a more economical practice.

Data recently released by the International Commission of Radiological Protection on the radiation protection of nuclear medical staff was provided by Vano [18]. The occupational dose significantly increases when the technologist works in close proximity to the patient during the injection or transfer to-and-from a trolley [19]. Another recent study on the radiation dose to the medical staff provided by Studbrock *et al*., [20], concluded that the upper limit dose may be reached due to close contact with ^{131}I patients. The dose to the hands has been assessed by many investigators [21-24], and many studies have reported high doses. Furthermore, contamination in nuclear medicine, even if it is minor, can lead to a major dose at an extremity [25].

There is relatively few data on occupational doses from developing countries [26-29], meaning that more data is needed, especially from the Middle East. Data presented here refer to the doses received by radiation workers in nuclear medicine.

METHODS

MCNP calculations of external doses using phantoms

To evaluate the external whole body dose from the patient, calculations were performed on cylindrical water phantoms with a height of 176 cm [10] and for three body diameters: 24 cm, 30 cm, and 36 cm. The multipurpose MCNP-5 code [30] was used for ^{99m}Tc, ^{131}I, ^{23}I, ^{67}Ga, ^{201}Tl, and ^{111}In sources. All of the cited sources are diagnostic imaging sources, with the exception of ^{131}I which is both a therapeutic and a diagnostic source. Because in many diagnostic procedures radiopharmaceutical elements accumulate in trunk organs, calculations were also performed at the height of 65 cm (37% of the total height, according to ICRP 89 [10]), for diameters identical to those cited above. The radioactive source was assumed to be homogeneously distributed inside the phantoms in the first model and as a point source at the center of the cylinder in the second model. Dose rates were calculated at 0.05 cm and 100 cm away from the surface of the phantom and along the line perpendicular to the symmetry axis of the phantom from its middle. Comparisons were made with bare-point-source-calculated values [32] and with measured values. The ANSI/ANS flux-to-dose rate conversion factors were considered [31]. Gamma ray energies and their corresponding intensities were obtained from Unger and Trubey [32]. A total of 10^7 histories were considered, which leads to a maximum relative standard deviation of 1.5%.

Measurements of the external dose from patients

The measured values were obtained after adult male patients were administered with radiopharmaceuticals; the external doses were measured in contact (1-5 cm from the surface of the body) at several locations of the body and at 1 meter away. The direct radiation dose rate was measured using GR 130M, supplied by Exploronium, Canada.

Occupational dose measurements

The integrated external doses to technologists were measured using LiF TLD-100 dosimeters which were used for both body monitoring and finger dose measurements. A Harshaw-4000 system chip reader was used for the TLD evaluation. Three TLD cards were supplied to each technologist. One TLD card was worn on the upper front (chest), one at the waist and one on the upper back (rear neck). The technologists were also given finger TLD chips to measure the finger doses.

The measurements were performed at King Abdulaziz University Hospital in Jeddah, Saudi Arabia, which has a capacity of 500 beds. Radiation and radionuclides were used for both diagnosis and therapy. The preparation room was equipped with safety equipment that included body shields, bench shields, vial shields, syringe shields and holders, waste modules and disposable containers. The room was also equipped with a calibrator and a fume hood with proper ventilation.

RESULTS AND DISCUSSION

Calculated and measured doses from patients

Figure 1 shows the calculated dose rates in (Sv/h)/MBq at 1 m for the cylindrical body phantom,

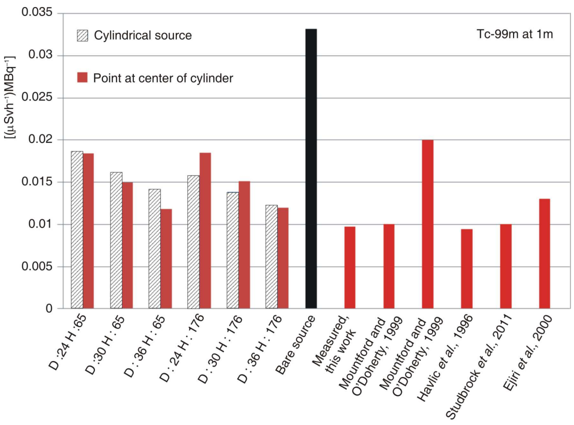

Figure 1. MCNP-calculated, bare-source-calculated, and measured Tc-99m dose rates at 1 m

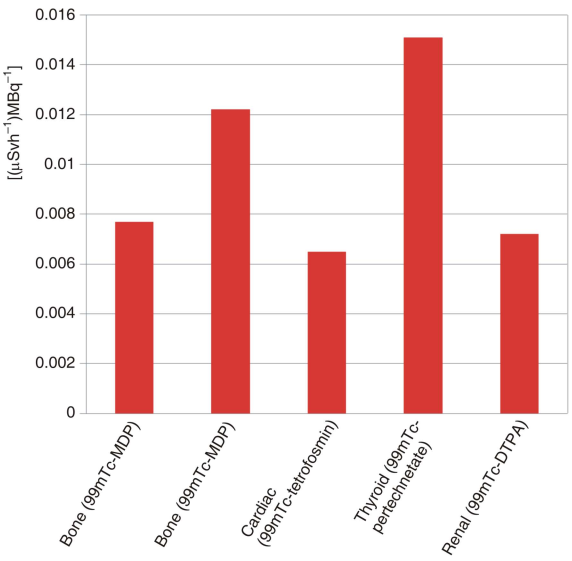

Figure 2. Measured Tc-99m dose rates at 1 m

cylindrical trunk phantoms, a point source at the center of the phanton, a bare point source and the measured values for ^{99m}Tc. No significant difference was observed when the source was a whole body or a trunk of uniform distribution or a point source at the center of same diameters. The dose decreased when the diameter (thickness) increased, because more absorption occurred in the body. The measured value was the average of the values obtained for the bone, thyroid, renal, and cardiac examinations shown in fig. 2. The values of some measured doses obtained from literature are also shown in fig. 1 [20, 33-35]. The work by Mountford and O'Doherty [35] summarized the results of the other authors; accordingly, two dose values in the graph are from these authors. The value given by Ejiri [33] was the average of many values taken at different positions of the dosimeter, with variations of 35%.

Except for a single measured value shown in fig. 1, dose calculations of a bare point source are more than 3 times the measured values and the MCNP-calculated ones. Measured and MCNP-calculated values are close, indicating that the models used for the calculations were appropriate. The difference between the bare source and MCNP calculations or measured values strongly depends on gamma ray energy. With lower energy, more absorption occurs in the body. The correct dose assessment of individuals near the patient should also consider radionuclide half-life. ^{99m}Tc is widely used in nuclear medicine, but because of its relatively short half-life of 6 hours, the hazard associated with it is more significant to radiation workers than the public.

The doses at 1 m for ^{67}Ga and ^{201}Tl are shown in figs. 3 and 4. Generally, similar conclusions were ob-

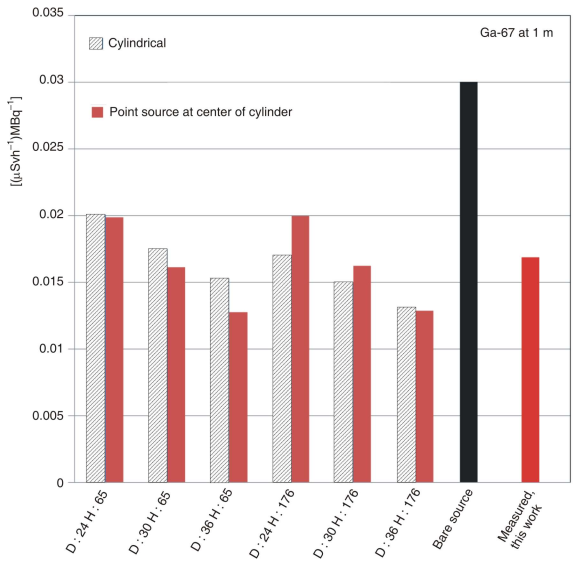

Figure 3. MCNP-calculated, bare-source-calculated, and measured Ga-67 dose rates at 1 m

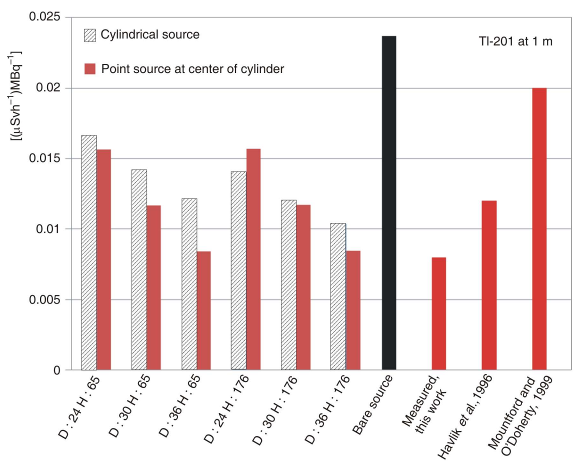

Figure 4. MCNP-calculated, bare-source-calculated, and measured Tl-201 dose rate at 1 m

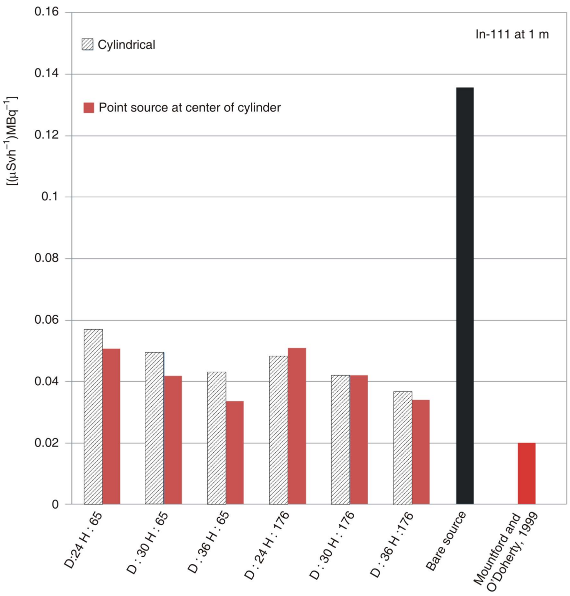

Figure 5. MCNP-calculated, bare-source-calculated, and measured In-111 dose rate at 1 m

tained: the dose decreased when the cylinder thickness increased, but was almost independent of the cylinder height. Except for one measured value, the bare source provides approximately twice the dose as that of a cylindrical phantom of 30 cm and more than that of a thicker phantom. The effective half-lives of ^{67}Ga and ^{201}Tl are approximately 3 days. Accordingly, they stay longer in the body and give more doses.

For ^{111}In, fig. 5 shows that the bare source dose value was approximately 3.5 times the MCNP values and more than that for the measured values. The source emitted 245 keV at 94%, 171 keV at 90% and 23 keV at 69%; the last gamma energy of 23 keV easily absorbed into the body gave a dose lower than those from a bare source. The source effective half-life is 2.8 days, necessitating realistic calculations for dose assessments.

The dose values for ^{131}I are shown in fig. 6. The source emits 365 keV gamma ray at 82%; the MCNP value was approximately half that of a bare source. The source is widely used for therapy with a long effective half-life of approximately 7.6 days. It is extremely important that a realistic estimate of the dose from patients is made in order to avoid longer than necessary hospitalization or a premature release.

The dose rate of the ^{123}I bare source is almost 4 times the value calculated using MCNP (fig. 7) or the value measured by Ejiri *et al.*, [33]. The source emitted 159 keV at 89% and lower gamma rays of 27 keV and 31 keV at 70% and 16%, respectively. The lower gamma rays were absorbed into the body before reaching the outside, which made the bare source value much higher. Because of this big difference in dose values and the effective half-life of approximately 12 hours, a realistic dose estimate is necessary. Figure 8 shows the comparison between the MCNP-calculated and bare source calculated dose rates.

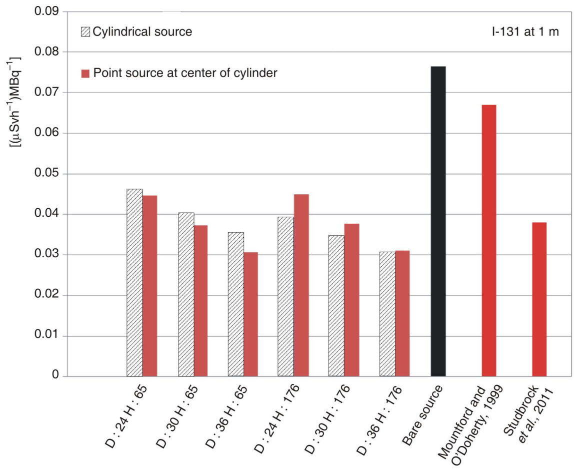

Figure 6. MCNP-calculated, bare-source-calculated, and measured I-131 dose rate at 1 m

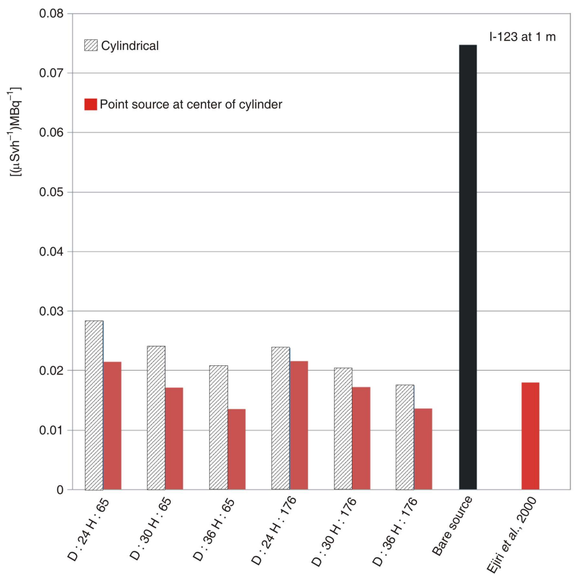

Figure 7. MCNP-calculated, bare-source calculated, and measured I-123 dose rate at 1 m

Contact dose rates are important for assessing the doses to infants [15], patient spouses and other members of the family [8, 9]. The measured values showed significant variation, depending on the measured position on the body. Measured contact dose rates for ^{99m}Tc relative to this work and other studies are shown in fig. 9. The values obtained in this work were acquired at the highest contact point for different diagnostic examinations. The thyroid had the highest measured value because the neck has the smallest body attenuation. Dose values given by Ejiri *et al.*, [33] were the average of four values at a point close to the spine for ^{99m}Tc-ECD. Dose rates measured by Mountford and O'Doherty [35] were obtained at 10 cm from the body.

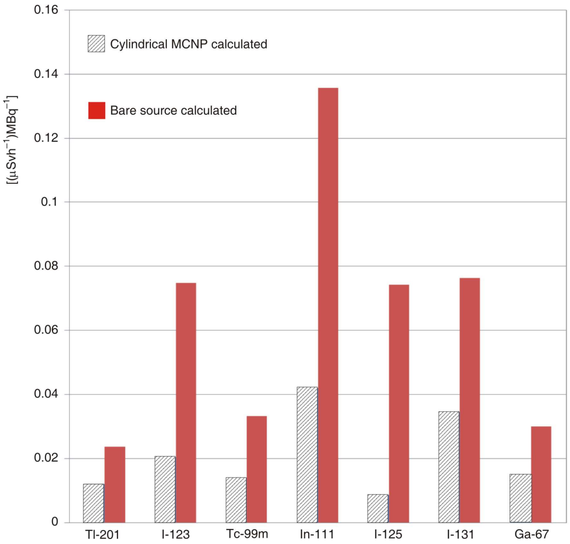

Figure 8. MCNP-calculated and bare-source calculated dose rates at 1 m

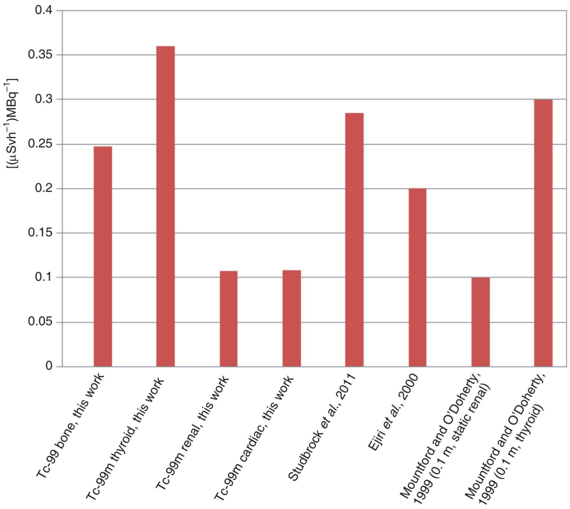

Figure 9. Tc-99m measured contact dose rates

Calculated public and occupational doses

The period of time during which a patient must stay in the hospital can be decided based on dose commitment, which is the total dose received until the radionuclide is completely disintegrated, given by

$$D \quad \frac{D_0}{\lambda_E} \tag{1}$$

where D is the total dose, D_0 – the initial dose rate, and λ_E – the effective disintegration constant. If a patient remains hospitalized for a time t and is discharged from the hospital afterwards, the total dose received by others is given as

$$D \quad \frac{D_0 \exp(\ \lambda_E t)}{\lambda_E} \tag{2}$$

The total dose at 1 m and the total contact dose in eq. (1) are shown in tab. 1 by taking a round figure dose rate from the graphs. This round value is close to the measured value or/and MCNP values. The indicative example activity is also provided in the table. For a patient administered with 700 MBq of ^{99m}Tc, the total dose at 1 m is approximately 90 Sv and the total contact dose is approximately 1.8 mSv. A patient administered with 200 MBq of ^{131}I for imaging will have a total dose of 2.1 mSv at 1 m and 79 mSv in total contact; contact doses were estimated from data found in literature [20, 35]. High doses were observed for ^{131}I and ^{111}In. However, a realistic dose estimate can only be obtained by understanding the specific habits, social life, transportation and work of the patient. A close contact with a ^{131}I or ^{111}In patient can easily lead to a dose that is above the constraint [1]. The recommendation is that the dose received by young children, infants and individuals not engaged in the direct care of the patient should not exceed that of the public, *i. e*, that of 1 mSv per year. A dose can easily exceed this limit if proper precautions are not taken [8, 9].

Occupational radiation dose to nuclear technologists

Table 2 shows whole body integral doses received in 19 weeks by each of the 3 nuclear medicine technologists; the doses were recorded using TLD chips at 3 positions of the body. Approximately 25-30 patients were treated per week. The differences between the obtained values can be attributed to many factors that should include the personal habits and skills of the technologist administering the radiopharmaceuticals and caring for the patients, their body size (in particular, height of the technologist), the position of the dosimeter and his/her

Table 1. Dose commitments at 1 m and contact dose commitment

	Effective half-life [h]	Initial dose rate at 1 m [(Svh^{-1})MBq^{-1}]	Dose commitment at 1 m [SvMBq^{-1}]	Initial contact dose rate [(Svh^{-1})MBq^{-1}]	Contact dose commitment [SvMBq^{-1}]	Indicative activity [MBq]	Dose commitment at 1 m [Sv]	Contact dose commitment [Sv]
^{99m}Tc	6	0.015	0.13	0.3	2.6	700	91	1820
^{131}I (imaging)	182.4	0.04	10.5	1.5	395	200	2100	79000
^{67}Ga	70	0.015	1.5	0.45		300	46	
^{201}Tl	57.6	0.01	0.83	0.3		200	166	
^{111}In	67.2	0.04	3.9	1.5		200	776	

Table 2. Whole body occupational doses for nuclear technologists over a period of 19 weeks and finger doses received in 8 weeks

Occupational doses for 19 weeks			
Technician no.	Upper back [mSv]	Waist [mSv]	Upper front [mSv]
1	0.52	0.912	0.27
2	0.49	0.28	0.33
3	1.00	0.63	0.33
Occupational finger doses for 8 weeks			
Technician no.	Finger dose [mSv]		
1	10.77		
2	2.76		
3	0.778		

ability to communicate with the patient. All technologists included in this study were of international origin and did not speak the language of the patients entrusted to their care.

Table 2 shows the doses obtained from using finger TLD chips for over 8 weeks. The first technician (no. 1) received much higher doses than the others. The annual value if the same dose rate was to remain would amount to approx. 67 mSv. If the actual dose to the fingertip is 5 times the values recorded using the finger dosimeter [23], the actual annual dose would amount to approximately 335 mSv.

CONCLUSIONS

Figure 1 and figs. 3-7 show that the measured values of the whole body dose rates in this and other relevant studies are close to MCNP-calculated values. Figure 8 shows the significant difference between the MCNP-calculated and bare source calculated doses. Point source calculations [3] highly overestimate the dose. The data presented here can be used for a better estimation of the dose to the public and radiation workers, particularly since patient body size was taken into consideration. The dose values at 1 m depended mainly on body thickness instead of height; better dose estimation can be obtained based on the specific size of individual patients.

The committed dose per MBq at 1 m (tab. 1) for ^{131}I and ^{111}In was 10.5 and 3.9 Sv/MBq , respectively. These are high doses and special attention needs to be paid to patients administered with the said radionuclides. A typical indicative committed total dose received by a caregiver in prolonged close contact to a patient during a diagnostic procedure will be approximately 2 mSv and 0.8 mSv for each of the mentioned radionuclides, respectively; the actual dose received will depend largely on the time spent near the patient. Accordingly, staying in close proximity to the patient for a long time must be avoided. The other radionuclides rendered less significant total doses.

The body and finger occupational doses for technologists provided in tab. 2 showed a wide variation that depends on the individual technologist and the position of the dosimeter. These doses can be significantly affected by the practice, skill and habits of individual technologists. Lower doses were recorded by the dosimeter placed in the upper chest, while higher doses were recorded by the dosimeters placed near the waist and the upper back, which can be attributed to the practice followed. Receiving the highest dose of approximately 1 mSv in 19 weeks, the annual dose can amount to approximately 2.6 mSv which is well below the occupational limit.

The highest finger dose recorded was approximately 11 mSv for 8 weeks; the annual dose is approximately 67 mSv or 335 mSv after a multiplication of 5 for the fingertip dose [23]; although it is below the extremities dose limit of 500 mSv per year, it is still much too close to it. Proper monitoring of technologists' practices is needed; those with higher doses compared to others assigned the same tasks should be closely monitored.

ACKNOWLEDGMENT

The authors are thankful to King Abdulaziz University for the financial support of this work. Our gratitude also goes to Dr. Samara Alzaidi from the Royal Prince Alfred Hospital, Camperdown, Australia, for the scientific advice and valuable discussion.

AUTHOR CONTRIBUTIONS

Experiments and analysis of results obtained were carried out by A. R. Kinsara, S. Abdul-Majid, T. Albaghdadi, A. Maimani, and W. H. Abulfaraj. Literature research and figure preparation were carried out by A. R. Kinsara and S. Abdul-Majid. MCNP calculations were carried out by W. El-Gammal. The manuscript was written by S. Abdul-Majid.

REFERENCES

[1] ***, ICRP Publications 94, Release of Patients after Therapy with Unsealed Radionuclides, Annals of the ICRP, 34 (2), International Commission on Radiological Protection, 2004

[2] ***, IAEA Safety Reports Series No. 63, Release of Patients after Radionuclide Therapy, International Atomic Energy Agency, 2009

[3] ***, NRC Regulatory Guide 8.39, Release of Patients Administered Radioactive Materials, Nuclear Regulatory Commission, 1997

[4] ***, NRC Regulation 10 CFR 35.75, Release of Individuals Containing Unsealed Byproduct Material or Implant Containing Byproduct Material, Nuclear Regulatory Commission

[5] Harding, L. K., *et al*., The Radiation Dose to Accompanying Nurses, Relatives and other Patients in a Nu-

clear Medicine Department Waiting Room, *Nuclear Medicine Communications, 11* (1990), 1, pp. 17-22
[6] Chiesa, C., *et al.*, Radiation Dose to Technicians per Nuclear Medicine Procedure: Comparison between Technetium-99m, Gallium-67, and Iodine-131 Radiotracers and Fluorine-18 Fluorodeoxyglucose, *European Journal of Nuclear Medicine and Molecular Imaging, 24* (1997), 11, pp. 1380-1389
[7] Mathieu, I., *et al.*, Dose in Family Members after ^{131}I Treatment, *Lancet, 350* (1997), 9084, pp. 1074-1075
[8] Barrington, S. F., *et al*., Radiation Exposure of the Families of Outpatients Treated with Radioiodine (Iodine-131) for Hyperthyroidism, *European Journal of Nuclear Medicine, 26* (1999), 7, pp. 686-692
[9] Grigsby, P. W., *et al*., Radiation Exposure from Outpatient Radioactive Iodine (^{131}I) Therapy for Thyroid Carcinoma, *The Journal of the American Medical Association, 283* (2000), 17, pp. 2272-2274
[10] ***, ICRP Publication 89, Basic Anatomical and Physiological Data of Use in Radiological Protection: Reference Values, Annals of the ICRP 32, issue 3-4, International Commission on Radiological Protection, 2002
[11] ***, ICRP Publication 110, Adult Reference Computational Phantoms, Annals of the ICRP, 39 (2), International Commission on Radiological Protection, 2009
[12] Tanaka, G., *et al*., Reference Man Models for Males and Females of Six Age Groups of Asian Populations, *Radiation Protection Dosimetry, 79* (1998), 1-4, pp. 383-386
[13] Jain, S. C., *et al.*, Formulation of the Reference Indian Adult: Anatomic and Physiologic Data, *Health Physics, 68* (1995), 4, pp. 509-522
[14] Mountford, P. J., O'Doherty, M. J., Close Contact Doses to Children from Radioactive Patients, *Nuclear Medicine Communications, 18* (1997), 4, pp. 383-386
[15] Cormack, J., Shearer, J., Calculation of Radiation Exposures from Patients to Whom Radioactive Materials Have Been Administered, *Physics in Medicine and Biology, 43* (1998), 3, pp. 501-516
[16] Barrington, S. F., *et al*., Measurement of the Internal Dose to Families of Outpatients Treated with ^{131}I for Hyperthyroidism, *European Journal of Nuclear Medicine and Molecular Imaging, 35* (2008), 11, pp. 2097-2104
[17] Koshida, K., *et al*., Level of ^{131}I Activity in Patients to Enable Hospital Discharged, Based on External Exposure of Family Members of the Patients in Japan, *Radiation Protection Dosimetry, 83* (1999), 3, pp. 233-238
[18] Vano, E., ICRP and Radiation Protection of Medical Staff, *Radiation Measurements, 46* (2011), 11, pp. 1200-1202
[19] Smart, R., Task Specific Monitoring of Nuclear Medicine Technologists Radiation Exposure, *Radiation Protection Dosimetry, 109* (2004), 3, pp. 201-209
[20] Studbrock, F., *et al*., Dose and Dose Rate Measurements for Radiation Exposure Scenarios in Nuclear Medicine, *Radiation Measurements, 46* (2011), 11, pp. 1303-1306
[21] Chruscielewski, W., *et al*., Hand Exposure in Nuclear Medicine Workers, *Radiation Protection Dosimetry, 101* (2002), 1-4, pp. 229-232
[22] Vanhavere, F., An Overview of the Use of Extremity Dosemeters in Some European Countries for Medical Applications, *Radiation Protection Dosimetry, 131* (2008), 1, pp. 62-66
[23] Wrzesien, M., Olszewski, J., Jankowski, J., Hand Exposure to Ionizing Radiation on Nuclear Medicine Workers, *Radiation Protection Dosimetry, 130* (2008), 3, pp. 325-330
[24] Carnicer, A., *et al*., Hand Exposure in Diagnostic Nuclear Medicine with ^{18}F- and ^{99m}Tc-labelled Radiopharmaceuticals – Results of the ORAMED Project, *Radiation Measurements, 46* (2011), 11, pp. 1277-1282
[25] Covens, P., *et al*., The Contribution of Skin Contamination Dose to the Total Extremity Dose of Nuclear Medicine Staff: First Results of an Intensive Survey, *Radiation Measurement, 46* (2011), 11, pp. 1291-1294
[26] Sandouqa, A. S., Haddadin, I. M., Abu-Khalid, Y. S., Hand Equivalent Doses of Nuclear Medicine Staff in Jordan: Preliminary Experimental Studies, *Radiation Measurements, 46* (2011), 2, pp. 250-253
[27] Kopec, R., *et al*., On The Relation between Whole Body, Extremity and Eye Lens Doses for Medical Staff in the Preparation and Application of Radiopharmaceuticals in Nuclear Medicine, *Radiation Measurements, 46* (2011), 11, pp. 1295-1298
[28] Budzanowski, M., *et al*., Dose Levels of the Occupational Radiation Exposures in Poland Based on Results from the Accredited Dosimetry Service at the IFJ PAN, Krakow, *Radiation Protection Dosimetry, 144* (2011), 1-4, pp. 107-110
[29] Abdul-Majid, S., *et al*., Radiation Doses to Medical Personnel from Diagnostic Radiography in Saudi Arabia, 4th Saudi Conference on Medical Physics, May 23-25, Al-Khobar, Saudi Arabia, 2009
[30] ***, MCNP – X-5 Monte Carlo Team, A General Monte Carlo N-Particle Transport Code, Version 5, 2003
[31] ***, ANSI/ANS-6.1.1-1977 (N666), American National Standard Neutron and Gamma-Ray Flux-to-Dose Rate Factors, American Nuclear Society, LaGrange Park, Ill., USA, 1977
[32] Unger, L. M., Trubey D. K., Specific Gamma-Ray Dose Constants for Nuclides Important to Dosimetry and Radiological Assessment, Oak Ridge National Laboratory, Oak Ridge, Tenn., USA, ORNL/RSIC-45 / KJ, 1982
[33] Ejiri, K., *et al*., Radiation Dose Rates from Patients Administrated Radiopharmaceuticals Used for Brain Blood Flow Investigation, IRPA 10, International Radiation Protection Congress, Hiroshima, Japan, 2000, p-7-54
[34] Havlik, E., Kurtaran, A., Preitfellner, J., Radiation Exposure around Patients after Administration of Tc-99m-DPD or Tl-201-Chloride, IRPA 9, International Congress on Radiation Protection, *Proceedings*, Vol. 3, Vienna, 1996, pp. 550-552
[35] Mountford, P. J., O'Doherty, M. J., Exposure of Critical Group to Nuclear Medicine Patients, *Applied Radiation and Isotopes, 50* (1999), 1, pp. 89-111

Received on October 13, 2013
Accepted on June 19, 2014

Абдул Рахим КИНСАРА, Самир АБДУЛ-МАЏИД, Ваел ЕЛ-ГАМАЛ, Тарик АЛБАГДАДИ, Абдулраоф МАИМАНИ, Валид Х. АБУЛФАРАЏ

СПОЉАШЊЕ ИЗЛАГАЊЕ ДОЗАМА ОД РАДИОФАРМАКА АПЛИЦИРАНИХ ПАЦИЈЕНТИМА – МЕРЕЊА И МОНТЕ КАРЛО СИМУЛАЦИЈА

Обављене су Монте Карло симулације и мерења расподеле радионуклида у целом телу, као и у појединачним деловима различитих димензија, како би се проценило спољашње излагање дозама од радиофармака апликираних пацијентима. За прорачуне су коришћени цилиндрични водени фантоми висине 176 cm и три различита пречника: 24 cm, 30 cm и 36 cm. Испитивани радионуклиди су: ^{99m}Tc, ^{131}I, ^{23}I, ^{67}Ga, ^{201}Tl и ^{111}In. Измерене вредности и оне добијене MCNP симулацијом биле су два до шест пута ниже од вредности израчунатих методом тачкастог извора. Одређена је укупна спољашња доза до тренутка у којем је радионуклид потпуно дезинтегрисан. Други циљ овог рада је да се прикажу подаци за ефективну дозу и дозу за шаке техничара који раде на одељењима нуклеарне медицине. Резултати су показали широк распон доза, што зависи од вештине техничара и положаја ношења дозиметра.

Кључне речи: доза зрачења, радиофармак, MCNP