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Original Article

Evaluating internal exposure due to intake of ^{131}I at a nuclear medicine centre of Dhaka using bioassay methods

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A R T I C L E I N F O

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A B S T R A C T

Handling of radioisotopes may cause external and internal contamination to occupational workers while using radiation for medical purposes. This research aims to monitor the internal hazard of occupational workers who handle ^{131}I. Two methods are used: *in vivo* or direct method and *in vitro* or indirect method. The *in vivo* or direct method was performed by assessing thyroid intake with a thyroid uptake monitoring machine. The *in vitro* or indirect method was performed by assessing urine samples with the help of a gamma-ray spectroscopy practice using a High-Purity Germanium (HPGe) Detector. In this study, fifty-nine thyroid counts and fifty-nine urine samples were collected from seven occupational workers who were in charge of ^{131}I at the National Institute of Nuclear Medicine and Allied Sciences (NINMAS), Dhaka. The result showed that the average annual effective dose of seven workforces from thyroid counts were 0.0208 mSv/y, 0.0180 mSv/y, 0.0135 mSv/y, 0.0169 m Sv/y, 0.0072 mSv/y, 0.0181 mSv/y, 0.0164 mSv/y and in urine samples 0.0832 mSv/y, 0.0770 mSv/y, 0.0732 mSv/y, 0.0693 mSv/y, 0.0715 mSv/y, 0.0662 mSv/y, 0.0708 mSv/y.The total annual effective dose (*in vivo* and *in vitro* method) was found among seven workers in average 0.1039 mSv/y, 0.0950 mSv/y, 0.0868 mSv/y, 0.0862 mSv/y, 0.0787 mSv/y, 0.0843 mSv/y, 0.0872 mSv/y. Following the rules of the International Commission on Radiological Protection (ICRP), the annual limit of effective dose for occupational exposure is 20 mSv per year and the finding values from this research work are lesser than this safety boundary.

Contributions

Sharmin Jahan - Literature Review, Experiment, Result Analysis, Formatting and Manuscript Writing.

Jannatul Ferdous - Concept, Literature Review, Experiment, Result Analysis, Supervision, Formatting and Correction.

Md. Mahidul Haque Prodhan: Concept, Literature Review, Experiment, Result Analysis, Main Supervision, Manuscript Writing, Formatting and Correction.

Ferdoushi Begum - Literature Review, Experiment, Result Analysis and Supervision.

1. Introduction

Radionuclides are used in medical purposes to diagnose, treat, and evaluate several complicated diseases. In cases of diagnosis and therapy to identify and cure some specific diseases, a substance containing a radioactive isotope or radionuclide is administered to a patient, and that nuclide goes to that particular tissue. The radioactivity will be emitted from that nuclide will help to locate the radionuclide in case of diagnosis of the disease. To cure some diseases, radioactivity will be used to trigger surrounding cells to deteriorate the condition of certain toxic cells. The radionuclides are selected based on the nature of the emitted radiation, especially for their physical properties and chemical properties, and these properties will determine the final purpose of the molecule as a radiopharmaceutical [1–3]. The manipulation of many radiopharmaceuticals accompanied by various unsealed sources of radionuclides, which are used for both diagnostic and therapeutic purposes, may give rise to remarkable risks of internal contamination to the workers because of inhalation [4–9]. The scrutiny of the radiation dose that nuclear medicine worker receives internally is equally important as the dose externally received by them in terms of radiological protection.

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It is also found that internal doses in nuclear medicine centres are usually lower than doses from external exposure. However, this risk of radiation absorption should be measured for every case [10]. A Regular investigation should be performed for the workers who are involved in the managing of unsealed sources of radionuclide to validate that the doses that workers inhale are lower [11].

Radionuclides which are taken into a body, can be estimated by both direct and indirect measurement methods. The amount of emitted Gamma or X-ray photons including bremsstrahlung radiation from the deposited radionuclides in the body is estimated through direct measurements by body activity counters and whole-body counting system [12,13]. Moreover, indirect measurements are the method in which measurements of activity in samples have been measured by either biological or physical processes [14]. Every method of measurement has pros and cons, and so the selection of one over another is mostly dependent on the kind of radiation to be measured. In the case of direct methods, those radionuclides are used, which emit photons of sufficient energy, and those escape from the body and, after that, easily can be measured by an external detector [15]. These two methods can be defined as *in vitro* and *in vivo* bioassays as well. *In vivo* involves a direct method that is performed in a living organism. *In vitro*, the bioassay method involves indirect measurement of radionuclide concentrations in materials separated from the body, like urine, to assess intakes and internal doses [16–19].

In addition, Human beings and other living beings are also exposed to radiation from various sources and methods. Ionizing radiation is considered an unpleasant form of radiation for human beings at all levels. Therefore, the International Basic Safety Standard for Protection (BSS) sets regulatory limits on effective doses for the population. Moreover, the effective dose is not a physical quantity; rather, it is a derived quantity which is developed by the International Commission on Radiological Protection (ICRP) [20]. The staff and patients in nuclear medicine centres should be regularly monitored to sustain their safety.

Currently, eighteen (18) Nuclear Medicine institutes are running in Bangladesh, and the NINMAS is one of them. This institute is situated in Dhaka and is designated for this research, wherever employees are exposed to internal radiation exposure when handling and labelling the samples. It is important to conduct a study on the radionuclide's contents, which are customarily used for the diagnosis and treatment of critical diseases. applications. ^{131}I is one of the radioisotopes that is widely used for thyroid-related diseases in nuclear medicine [21–25]. ^{131}I was brought to light by Glenn Seaborg and John Livingood, two scientists at the University of California, Berkeley in 1938 [26]. In 1942, this radioisotope at first used to treat hyperthyroidism, and after that, it became popular to treat thyroid-related illnesses [27,28]. Additionally, it is utilized in kidney and bladder function tests [29,30]. ^{131}I is a volatile, radiotoxic isotope, and it exhibits a half-life of 8 days. Manipulating this isotope can be responsible for radiological hazards for the workers. Therefore, the staff who have taken responsibility for unsealed sources such as ^{131}I can be affected internally by the inhalation process [31]. Generally, the manipulation of all sources of radioisotopes in nuclear medicine creates conspicuous risks of internal exposure to the occupational staff [32–39]. In the process of radioiodine distribution rate in the body, most of the radioiodine accumulates in the thyroid gland, and the rest of the isotopes spread into the whole body [40]. So, there have been performed several studies on occupational workers by using a thyroid uptake monitoring system (*in vivo*) or gamma-ray spectrometry system (measurement of radioactivity in urine samples which is an *in vitro* method). However, for monitoring occupational staff more accurately, it is necessary to perform a study using *in vivo* and *in vitro* methods concomitantly for the staff who handle ^{131}I independently to observe internal exposure. The main objective of the present study is to analyse the intake activity and evaluate the annual effective dose of occupational workers in thyroid and urine samples due to the intake of ^{131}I.

2. Materials and methods

2.1. Preparing the samples after collection

The Employees who are involved with the lab of the nuclear medicine unit may be exposed to radiation in many ways, such as inhalation, absorption in intact skin, or open wounds. During the research period, seven occupational workers of NINMAS who are handling ^{131}I were taken for an *in vivo* count of the thyroid. In the case of an *in vitro* count, workers were guided to follow the collecting instructions provided to them with the vessels. Any type of external contamination of the vessels was carefully reduced as this would badly affect the possibility of getting exact activity. A Total of 59 thyroid counts and 59 urine samples from seven professional medical workers were collected after 3–4 h of handling ^{131}I from the NINMAS. These urine samples were kept in conventional plastic vessels covered with tissue paper and air-tight packets, which were distinguished by identification name, time of collection and radioactivity of managing radioisotope beforehand. Then, the urine samples were taken for analysis after collection as soon as possible and measured with the HPGe detector at the Division of Health Physics, Atomic Energy Centre (AEC) in Dhaka. The urine samples' volumes were estimated and were below 350 ml. After that, the samples were kept at the HPGe instrument to execute the measurements. Due to some constraints, and unfavourable working situations, it was not viable to collect urine samples periodically.

2.2. Methods of measuring radioactivity

2.2.1. In vivo bioassay method

In this study, the amount of activity due to use of ^{131}I deposited in the thyroid gland of nuclear medicine workers in the institute was investigated. The investigation had been performed for this research after the workers were administrating radioiodine (^{131}I) to the patients for thyroid uptake monitoring. If there was any leftover activity from the previous performance, that activity was deducted from the original measurement. The thyroid measurement was conducted by a thyroid monitor, which is a NaI detector (NATS3, USA) for the 60s per person. During taking their radioactivity, background radiation was taken, and this value was deducted to calculate the actual result.

2.2.2. In vitro bioassay method

The radioactivity of each sample was measured with the help of the HPGe detector. However, all necessary quality assurance of the detector had been performed before taking radioactivity measurements. Moreover, the liquid N_2 dewar portion of the HPGe should be kept within full of liquid N_2 for minimal half a day (12 h) after taking measurements. After that, the HPGe instrument had been switched on, and it was given a quarter hour (Fifteen minutes) warm-up phase beforehand taking the 1st measuring method of every day. Operating voltage of HPGe progressively increased to (+)1600 V. Moreover, coarse gain, peak shaping time and fine gain of amplifier had been corrected as well. Then, this detector was allowed for half an hour to stabilize the system. After the equilibrium of this instrumental arrangement, the HPGe's calibration of energy was ensured to appear 661.66 keV, 1170 keV, & 1332 keV peaks in proper directs by using ^{137}Cs and ^{60}Co point sources. These sources were kept at the HPGe detector axis at the source-to-detector separate of approximate 10 cm with a time duration of 100 s. Additionally, an empty plastic container was put for 5,000s to determine a background spectrum. Then, the amount of background radiation was measured and each of the plastic containers filled with samples was kept on the vertex of the HPGe's head. Entering door of the frontage of the shielding system was locked. All samples were taken count of 5000 s. In this study, software named "Mestro-32" had been operated for data transformation of every single sample [20]. The amount of energy emitted from gamma photons with counts per second (cps), carrying the statistical error was contained in an analysis sheet. The total radioactivity was calculated by getting a distinct radionuclide peak and taking the parallel counts per second. The calculation was performed using the equation written below,

$$A = \frac{cps \times 100}{\mathcal{E}(E) \times p_\gamma} \tag{1}$$

Here, A is radionuclide activity present within samples in Bq, cps defines counts/second, p_γ is the particular gamma-ray energy fraction, Ɛ (E) is the particular gamma-ray energy efficiency of HPGe for which is discharged from an explicit radionuclide of attention [41].

2.2.3. Minimum Detectable Level

Minimum Detectable Level (MDL) is a term which is related to the detection limit of the detector. It originates when the samples are subjected to the gamma measurement system [42]. The term MDL can be calculated by the following equation-

$$MDL = \frac{4.66 S_b}{\varepsilon(E) \times p_\gamma} \tag{2}$$

Where, S_b = Estimated error of the net count rate, Ɛ(E) = Counting efficiency at the desired energy of the radio nuclides and P_γ = Absolute transition probability of gamma decay.

2.3. Error analysis

The error related to the analysis of samples should be calculated during the completion and interpretation of experimental data and results. Therefore, deviation from normal values can be calculated by using the expertise of statistical error consideration. After that, the value of error from final result can be abated. Precisely, the error was calculated by the following formula [43],

$$\sigma = \sqrt{\frac{S+B}{T^2} + \frac{B}{T^2}} \tag{3}$$

Where, (S + B) is Sample's counts including background counts, B is Total background counts and T is duration of time in second.

2.3.1. Assessment of committed effective dose

Committed effective dose of the samples of NINMAS were evaluated using the following equation,

$$H_A = \sum_j I_{AJ} h_{AJ} \tag{4}$$

Where, for the radionuclide j, H_A = committed effective dose/committed equivalent dose (Sv) through inhalation for mature person, I_{Aj} = activity in Bq of the radionuclide, h_{Aj} = dose coefficient (Sv/Bq) of effective dose/targeted body part.

At time t, for an acute intake, the activity I_{Aj} (Bq) of the radionuclide inhaled can be calculated by following equation,

$$I_{Aj} = \frac{E_j}{e_a A_{j(t)}} \tag{5}$$

In this equation, E_j is considered as whole-body radionuclide activity j, measured by *in vivo* or *in vitro* methods and $e_a A_{j\,(t)}$ at time t is total body fractional activity (Bq) or at targeted organ following a severe inhalation 'a' of radionuclide j by an adult. Moreover, the estimation of $e_a A_j$ at t have been taken by using the bio-kinetic models and data described in ICRP-78 [44].

3. Result and discussion

In this study, a total of 118 samples of 7 radiation workers from NINMAS were examined during the period of December 2019 to February 2020. Among these samples, ^{131}I activity in 59 thyroids counts were taken by thyroid uptake monitor using direct measurement method, and 59 urine samples were counted by gamma spectrometry system using indirect measurement method. The results of thyroid counts for the radiation workers who are contaminated by ^{131}I are found. Although staff share the same type of work and the same workplace, it is found that the diagnosed concentration of radioactivity for some workers is excessive in number. Moreover, after taking daily thyroid counting for these workers, the activity of ^{131}I is decreasing. ^{131}I is detectable in the thyroid and urine samples of all the radiation workers. As the types of duties performed by workers are different, the distribution ^{131}I can be explained from that. So, a precise assessment should be done to estimate the doses that nuclear medicine workers are subjected to perform.

Table-1
Average Intake Activity of thyroid and urine samples of workers.

Subject Code	Type of Radio-isotope	Energy of the gamma-ray detected in keV	Average Intake in Bq (in Thyroid)	Average Intake Activity in Bq (in Urine)
A	^{131}I	364.48	314 ± 2.17	1260 ± 63.18
B	^{131}I	364.48	273 ± 2.27	1167 ± 60.24
C	^{131}I	364.48	205 ± 2.21	1109 ± 59.54
D	^{131}I	364.48	257 ± 2.16	1049 ± 57.85
E	^{131}I	364.48	110 ± 2.21	1083 ± 58.53
F	^{131}I	364.48	274 ± 2.21	1004 ± 56.73
G	^{131}I	364.48	248 ± 2.30	1073 ± 58.58
Range			110 ± 2.21–314 ± 2.17	1004 ± 56.73–1260 ± 63.18

3.1. Radioactivity measurement

While handling this radioisotope, workers have been contaminated because iodine is spread out into air for the reason that of hazardous disposition of air. Therefore, the worker was contaminated by ^{131}I and the concentration of ^{131}I was detected in thyroid and urine samples because they inhaled this radionuclide. In this study, radiation workers are coded by A, B, C, D, E, F, and G, who are contaminated by ^{131}I. In the present study, urine samples from the selected workers were collected and analyzed for radioactive nuclides, and ^{131}I was detected in the urine samples.

The average intake activity estimated in thyroid and urine samples are within the range from 110 ± 2.21 to 314 ± 2.17 Bq and 1004 ± 56.73 to 1260 ± 63.18 Bq respectively, which is shown in Table-1 and Figure-1. According to the biokinetic model, the amount of ^{131}I is absorbed in the body, 30 % of it goes to the thyroid gland, and the rest of 70 % excretes from the body with urine [45]. Our findings from the samples also show that the intake activity found in urine samples is more than the activity of thyroid glands. To find more accuracy, we need to take samples of a whole year to estimate the result using samples of all seasons. Furthermore, the uncertainty of the stated results is about one sigma.

3.2. Dose assessment

The next step of the study is to estimate doses for the nuclear medicine workers who are involved with ^{131}I. According to the publication of ICRP-66 and ICRP-78, exposures that occur due to the radioisotope ^{131}I can be calculated by following lung absorption type F and elemental

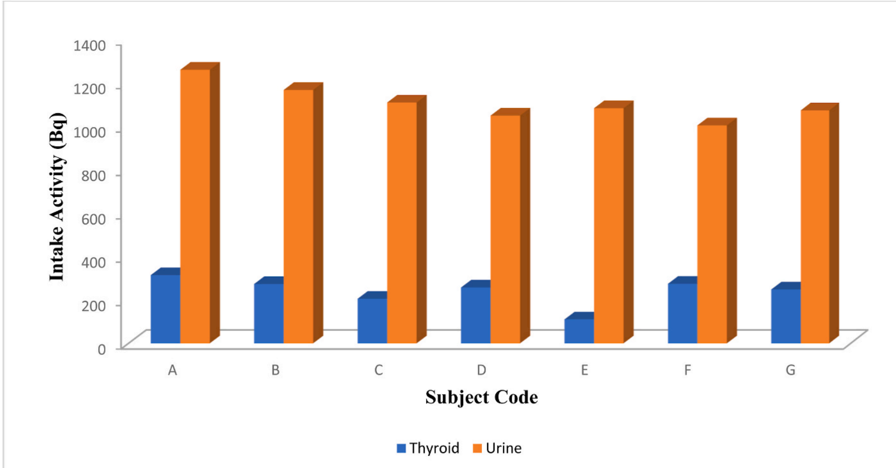

Figure-1. Comparison of Average Intake Activity of thyroid and urine samples among workers.

Table-2
Average Annual Effective Dose of thyroid and urine samples of workers.

Subject Code	Type of Radio isotope	Energy of the gamma Ray detected in keV	Average Annual Effective Dose (mSv/y) (in Thyroid)	Average Annual Effective Dose (mSv/y) (in Urine)
A	^{131}I	364.48	0.0208	0.0832
B	^{131}I	364.48	0.0180	0.0770
C	^{131}I	364.48	0.0135	0.0732
D	^{131}I	364.48	0.0169	0.0693
E	^{131}I	364.48	0.0072	0.0715
F	^{131}I	364.48	0.0181	0.0662
G	^{131}I	364.48	0.0164	0.0708

when radionuclide is inhaled by a worker [44,45]. Moreover, class D (day) is considered for calculation from ICRP-30 and ICRP-54 publications [46,47]. Moreover, according to ICRP-68 and ICRP-78, dose--coefficients have been taken for Activity Median Aerodynamic Diameter (AMAD) 5 μm [45,48]. The reason behind this is that the dimension of 5 μm is appropriate for all sizes of ^{131}I particles [45,48]. According to the radiation work schedule, the amount of time between two intakes was calculated. To prevent errors in intake estimation and to find a more accurate calculation process, the intake should consider the ^{131}I activity in the air of the lab of the nuclear medicine centre and the respiratory rate of the staff should also be considered. Dose assessments have been performed using measurements resulting from thyroid and urine samples.

The average annual effective doses of thyroid and urine samples of workers A, B, C, D, E, F, and G are shown in Table-2 and Figure-2. The approximate average annual effective doses taken by the nuclear medicine worker in the study using the direct method are found to range from 0.0072 to 0.0208 mSv/y. Furthermore, the calculated average annual effective dose received by the nuclear medicine worker in the

Table-3
Average Annual Effective Dose (*in vivo* and *in vitro* measurements) of workers.

Subject	Average Annual Effective Dose (mSv/y)
A	0.1039
B	0.0950
C	0.0868
D	0.0862
E	0.0787
F	0.0843
G	0.0872
Whole Body Annual Effective Dose (WBAED) (Standard limit by ICRP)	20

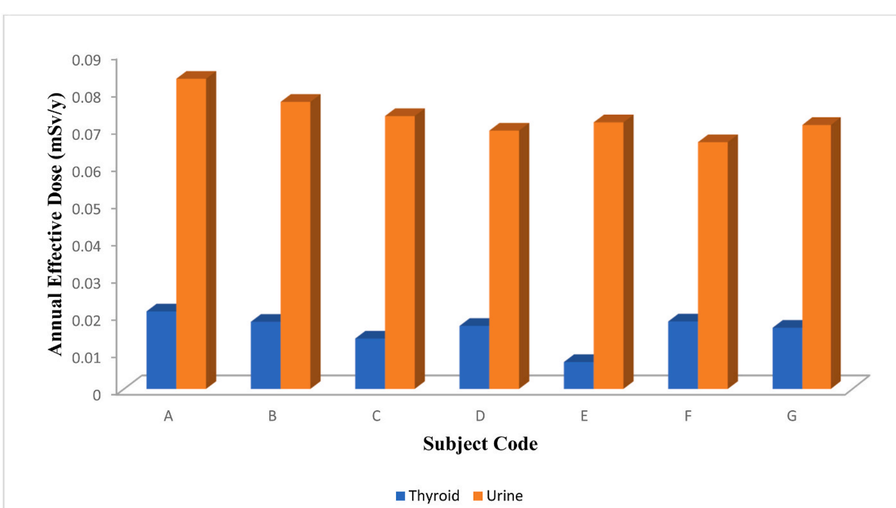

Figure-2. Comparison of Average Annual Effective Dose of thyroid and urine samples among workers.

Figure-3. Comparison of Average Annual Effective Dose (*in vivo* and *in vitro* Measurements) of workers with whole body Annual Effective Dose (in logarithmic scale).

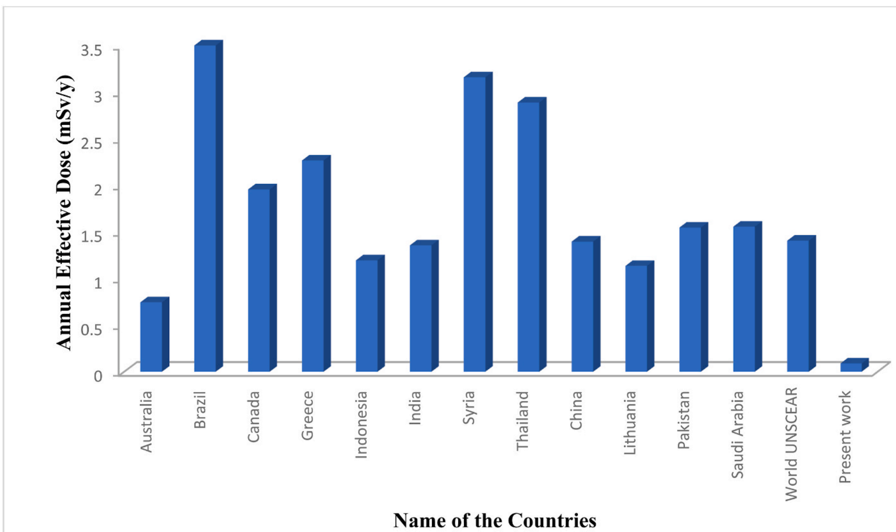

Figure-4. Comparison of Average Annual Effective Dose of other countries with present work.

study using the indirect method is found to differ from 0.0662 to 0.0832 mSv/y.

The total average annual effective dose of workers and whole body permissible annual effective dose according to ICRP 103 is shown in Table-3 [49].

Figure-3 shows the comparison of the average annual effective dose (*in vivo* and *in vitro* Measurements) of workers with that permissible limit [49]. The estimated values of this study are much lower than the permissible limit. Though all values are notably lower than the permissible limit, subject A seems to intake a higher dose rest of the workers.

Figure-4 provides information about the average annual effective doses of several countries and shows a comparison with the present work [50–53]. This figure represents that the value of this study is much lower than other countries.

The probability of internal exposure through the patients would be less than those who are directly handling ^{131}I. In spite of that, the workers who are regularly involved in manipulating radioisotopes should be taken for regular assessments. Moreover, they do sanitization after doing their jobs. Still, they should be trained properly to keep themselves safe from unreasonable amounts of radiation [54,55]. For this reason, it is important to establish stringent rules and regulations that will help to make sure this nuclear medicine branch is safe for workers and patients.

4. Conclusions

Internal exposure is one of the basic concerns in the nuclear medicine sector. It is significant, especially for occupational workers who spend their lives giving service to the patients of nuclear medicine institutes. Therefore, regular assessment of internal contamination is necessary to ensure their safety. Among all the unsealed radioisotopes, ^{131}I is one of the most commonly used radioisotopes. Moreover, this radioisotope is volatile and toxic and can be contaminated through injection, inhalation, and skin wounds. So, workers who handle ^{131}I should be monitored separately. The aim of this research was to evaluate the intake and effective dose (annual) of samples that were collected from workers who handle ^{131}I radioisotopes in the NINMAS, Dhaka. The total annual effective dose (*in vivo* and *in vitro* method) was found from seven workers in average 0.1039 mSv/y, 0.0950 mSv/y, 0.0868 mSv/y, 0.0862 mSv/y, 0.0787 mSv/y, 0.0843 mSv/y, 0.0872 mSv/y which were much lower than annual limit dose of 20 mSv per year recommended by ICRP 103 [49]. As values found in this study are much lower than the permissible value of ICRP, it can be concluded that the workers are out of risk of contamination. Nevertheless, the regular monitoring process should be continued for the safe handling process of radioisotopes in nuclear medicine centres. Moreover, the future goal of this study is to monitor every nuclear medicine centre in this country for the whole year and estimate an average annual effective dose.

Declaration of competing interest

There is no conflict of interest.

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