

Committed Effective Doses Received by Occupational Workers Handling Radioisotopes (^{131}I and $^{99\text{m}}\text{Tc}$) at INMAS, as Assessed from Urine-Samples

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Abstract

This study estimates the potential health risks attributed to the internal contamination of occupational workers at the Institute of Nuclear Medicine and Allied Sciences (INMAS) located at Dhaka Medical College and Hospital, Dhaka, during nuclear medicine practices involving the radionuclides ^{131}I and $^{99\text{m}}\text{Tc}$, using *in vitro* methods from urine samples. A total of 55 urine samples from 6 occupational workers are collected over a period of about 11 months. These samples are analyzed using a High Purity Germanium (HPGe) detector coupled with a multichannel analyzer (MCA). The radioactivity of the isotopes present in each urine sample is measured based on the detector efficiency, and the committed effective dose due to each intake is calculated from this activity. The average annual doses of individual workers found in this study range from 4.57×10^{-5} mSv to 9.72×10^{-3} mSv. Although these doses are considerably below the International Commission on Radiological Protection (ICRP) recommended annual dose limit of 20 mSv, efforts to abide by the ALARA principle should continue.

Keywords: Radioactivity, effective dose, bioassay, nuclear medicine, occupational worker, INMAS, dose limit, cancer

I. Introduction

Humans, like all other living beings, are constantly exposed to “background radiation”, which refers to a level of ionizing radiation of various kinds from both natural and artificial sources.¹ The average level of background radiation, which obviously varies moderately from one region of the world to another, is considered low. However, exposure to ionizing radiation at any level is considered in a conservative way, and is generally regarded as undesirable; even low levels of exposures are associated with stochastic health effects like cancer and genetic damages in a human body, while high levels can cause various deterministic health effects including death.² Hence it is of utmost importance to estimate these effects, to compare them to international standards of safe limits,³ as set by International Commission on Radiation Protection (ICRP), and to control the amount of radiation one is exposed to if and when required.

One of the many areas in which radioactive sources are widely used at present is that of medical procedures. Unsealed radioactive sources are used for diagnostic and therapeutic purposes, and the radioactive nuclides from these sources are inhaled by the workers when they handle them. Radionuclides one thus intakes, or is exposed to internally, are partially excreted through excreta (urine, feces, perspiration, etc.) and partially deposited in various organs in the body for a considerable period of time.³ One such intake hence continues to be a source of radiation incorporated inside the body, albeit with its effect diminishing with time. And there are not one but many such intakes for an occupational worker who handles sources repeatedly at their workplace. Assessment of the health risks incurred by an internally-contaminated person can be done using in-vivo (direct) methods involving measurements from an organ, or in-vitro (indirect) methods

involving measurements from an excreta collected after manipulation of radioisotope source(s).³

The present study was performed, using in-vitro measurements from urine samples, on the workers at the Institute of Nuclear Medicine and Allied Sciences (INMAS), one of more than 20 nuclear medicine facilities all over Bangladesh, which mostly use the radionuclides ^{131}I and $^{99\text{m}}\text{Tc}$ for medical practices. Both isotopes are used in the form of liquid compounds that are volatile at temperatures typical in the labs they are handled in. Also, workers under this study typically do not wear appropriate protective gears, eg., masks, gloves etc. Thus they end up inhaling these isotopes and hence get internally exposed thereto. The physical half-lives of the isotopes ^{131}I and $^{99\text{m}}\text{Tc}$ with which they disintegrate are about 8 days and 6 hours respectively, while the respective biological half-lives in human bodies are about 80 days and 1 day.

Theory behind the work

The stochastic health risk of an exposure to radiation of low levels is represented by the quantity *effective dose*, E defined by

$$E = \sum_{T,R} W_T \cdot W_R \cdot D_{T,R}. \quad (1)$$

Here $D_{T,R}$ is the *mass-averaged absorbed dose*, that is, the average energy imparted to unit mass of organ T due to an exposure to the ionizing-radiation of type R (alpha, beta, gamma, neutron, etc.); its unit is a *Gray (Gy)* = J/kg in the SI system. W_R and W_T are two dimensionless quantities called, respectively, the *radiation weighting factor*⁴ for radiation type R and the *tissue weighting factor*⁴ for the tissue type (i.e., organ) T . The summation is over all the types of radiation one is exposed to, and all the organs exposed. The biological quantity E has the same dimension as that of the physical quantity $D_{T,R}$, but its unit is named a

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Sievert (Sv), which represents an amount of health risk, namely, a 5.5% chance of developing cancer.

When an internal exposure takes place, that is, when an intake of radionuclides are incorporated into the body through the pathway of inhalation, ingestion, injection or absorption, the total resulting dose received is that received over the period, say τ years, during which the nuclides remain inside the body. If dE is the dose received over the temporal interval between t and $t+dt$, then the rate at which dose is received at time t is $\frac{dE}{dt}$, and the total dose received between the intake time t_0 and $t_0+\tau$ is called the *committed effective dose*, $E(\tau)$,

$$E(\tau) = \int_{t_0}^{t_0+\tau} dt \left[\frac{dE}{dt} \right] \\ = \int_{t_0}^{t_0+\tau} dt \left[\sum_{T,R} W_{T,R} \cdot W_R \cdot \frac{dD_{T,R}}{dt} \right], \quad (2)$$

where the period τ is typically taken to be 50 years for an intake by an adult, and to the age of 70 years for an intake by a child. The units of $E(\tau)$ are *Sv* again. While the integrand in (2) is a complicated function of many physical and biological factors, it is obvious that the integral is directly proportional to the *initial intake* (or, simply the *intake*), $I_{pathway}$ of radioactivity (the number of disintegrations of the radioactive nuclides per unit time, expressed in *Becquerels (Bq)* = disintegrations/second), that is, $E(\tau) \propto I_{pathway}$. The proportionality constant to be introduced here as the coefficient of $I_{pathway}$ is called the *committed effective dose per intake*, or the *dose coefficient*, $e(g)_j, pathway$. This quantity has a unit of *Sv/Bq*, and depends on the age-group g of the exposed person, the species j of the radionuclide, the pathway of internal exposure, the biokinetic model for the intake of nuclide j through the specific pathway, etc. Hence $E(\tau)$ for an exposure to nuclide j is

$$E(\tau) = e(g)_j, pathway \cdot I_{pathway} \quad (3)$$

The $e(g)_j, pathway$ -values for varied combinations of the mentioned factors are given by ICRP in its publications, and hence enable one to calculate the *committed effective dose* due to an intake once the intake is known (measured directly, or estimated indirectly from other measured quantities).

The *total effective dose*, E_{tot} received by a person in the age-group g due to exposures over a period of time can be written as the sum of doses due to external and internal exposures over the same period:

$$E_{tot} = \sum_{external\ exposure} E + \sum_{internal\ exposure} E(\tau) \\ = \sum_{external\ exposure} E + \sum_j \sum_{pathway} e(g)_j, pathway \cdot I_{pathway} \quad (4)$$

The dose-limit on total annual occupational exposure is set at $20\ mSv$ averaged over 5 consecutive years, and at $50\ mSv$ in any individual year.³

II. Methodology: Sample Collection and Analysis

A total of 55 urine samples were collected from 6 occupational workers handling radioisotope sources at INMAS over a period of about 11 months between December 2018 and October 2019. All six workers were male, their ages ranging between 28 and 47. Each sample was collected 2-3 hours after the worker had handled unsealed sources of ^{131}I or ^{99m}Tc for diagnostic or treatment purposes.

After the collection of the urine samples they were analyzed by a High Purity Germanium Detector housed in the Health Physics Division of Atomic Energy Centre Dhaka, Bangladesh Atomic Energy Commission. This detector is basically a gamma-ray spectrometer, which analyzes the gamma-ray spectrum of the radiation from the sample, that is, shows the energies and intensities of the gamma rays emitted by the sample. Examples of such spectra are given in Fig. 1, which show the significant peaks of count C for gamma-rays from ^{131}I and ^{99m}Tc . The detector-efficiency, expressed in percentages, of such a spectrometer varies with the gamma-ray energy, and therefore the detector was calibrated against known gamma-sources.

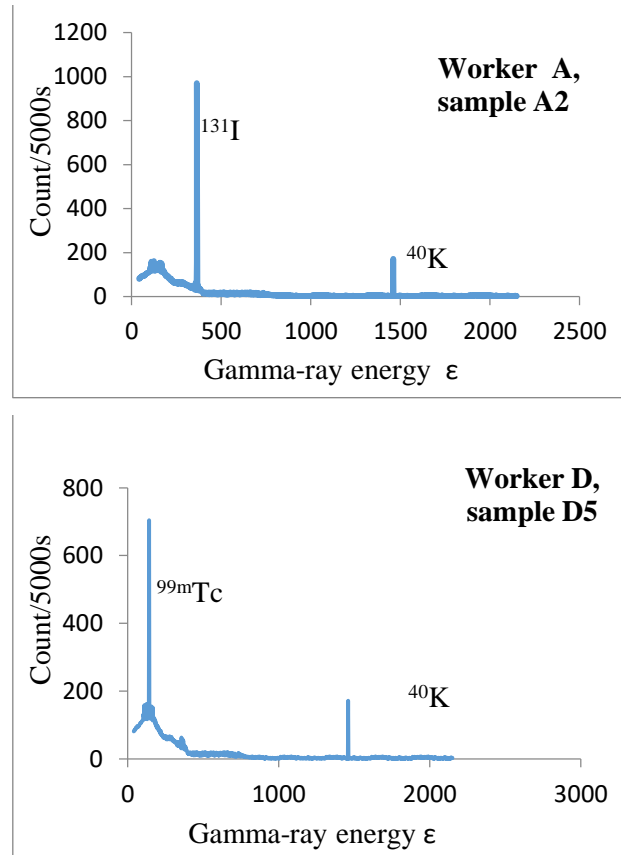


Fig. 1. Gamma-ray spectra of urine samples.

Upper: A2 collected from worker A. Lower: D5 collected from worker D [See Table 2]

III. Estimation of Committed Effective Dose from Initial Intake Based on Detector Count

The number of detector counts per second, *cps*, obtained from a detector for a gamma-ray of a particular energy ε from a specific nuclide is linked to the actual activity *A* present in the sample through the relation

$$cps = [P_\gamma \cdot h(\varepsilon)] \cdot A \Rightarrow A = cps / [P_\gamma \cdot h(\varepsilon)] \quad (5)$$

where P_γ is the emission probability of the gamma-ray of concern, and $h(\varepsilon)$ is the detector-efficiency at energy ε of the same ray [see Table 1].

Now, *A* is only a fraction of the initial intake (the only pathway of intake is assumed to be inhalation in this case), that is,

$$\frac{A}{I_{inhalation}} = a(\Delta t) \quad (6)$$

where the fraction $a(\Delta t)$ is sometimes called the “fractional activity”. Its value depends on the biokinetic model⁵ of the particular radionuclide inhaled by the worker (that is, the distribution of intake in the organs, the physical and biological half-lives of the nuclide, etc.), and on the time Δt between intake and sample collection/analysis. The combination of equations (3), (5) and (6) is used for

calculating the committed effective dose (using $\tau=50$ years) in terms of the experimentally determined quantity *cps* and other quantities pertinent to the given problem:

$$E(50) = e(g)_{j,inhalation} \cdot \frac{A}{a(\Delta t)} \\ = \left[\frac{1}{P_\gamma} \cdot \frac{1}{h(\varepsilon)} \cdot \frac{e(g)_{j,inhalation}}{a(\Delta t)} \right] \cdot cps \quad (7)$$

IV. Results and Discussion

The values of P_γ , $h(\varepsilon)$, $e(g)_{j,inhalation}$, and $a(\Delta t)$, that are appropriate for the cases studied here are described in Table 1. For both the nuclides the Activity Median Aerodynamic Diameter (AMAD) of the inhaled aerosol of nuclides was taken to be 1 μm , and the lung absorption type, fast (F).

The experimental results of this work are laid out in Table 2, where the detector-counts over 5000 seconds are tabulated for each urine sample studied, and the corresponding sample-activities and committed effective doses are calculated using the quantities described in Table 1. The individual workers are reported to handle radioisotopes around 3 times a week on an average, and, based on that was calculated the annual average committed effective dose, E_{ann} for each individual worker.

Table 1. Values of P_γ , $h(\varepsilon)$, $e(g)_{j,inhalation}$, and $a(\Delta t)$ that are relevant for the gamma-ray of concern and used in the calculation of $E(50)$ in Table 2.

Species of radionuclide inhaled,	Energy of the gamma ray detected,	Emission probability,	Detector efficiency,	Dose coefficient ⁶ ,	Time between intake and sample collection,	Fractional activity ⁷ ,
j	ε	P_γ	$h(\varepsilon)$	$e(g)_{j,inhalation}$	Δt	$a(\Delta t)$
	(keV)	(%)		(Sv/Bq)	(day)	
¹³¹ I	364.48	81.6	0.706572	7.60×10^{-09}	0.1	4.83×10^{-02}
^{99m} Tc	140.47	91.5	0.400834	1.20×10^{-11}		9.93×10^{-05}

Table 2. Assessment of average annual committed effective doses for individual workers.

Worker	Sample	Species of radionuclide inhaled,	HPGe detector count in 5000 seconds,	Activity in the sample,	Calculated committed effective Dose for the single intake,	Average annual committed effective dose for a worker, estimated assuming 3 exposures per week on an average,
		j	C	A	$E(50)$	E_{ann}
				(Bq)	(mSv)	(mSv)
A	A1	131I	12	0.00416	6.55×10^{-7}	2.96×10^{-3}
	A2	131I	907	0.31462	4.95×10^{-5}	
	A3	131I	78	0.02706	4.26×10^{-6}	
	A4	131I	441	0.15298	2.41×10^{-5}	
	A5	131I	79	0.02740	4.31×10^{-6}	
	A6	131I	777	0.26953	4.24×10^{-5}	
	A7	131I	83	0.02879	4.53×10^{-6}	
	A8	131I	518	0.17969	2.83×10^{-5}	
B	B1	^{99m} Tc	777	0.42371	5.13×10^{-7}	9.72×10^{-3}
	B2	^{99m} Tc	596	0.32501	3.94×10^{-7}	
	B3	131I	1036	0.35937	5.65×10^{-5}	
	B4	131I	7551	2.61932	4.12×10^{-4}	
	B5	^{99m} Tc	130	0.07089	8.58×10^{-8}	

	B6	131I	2590	0.89843	1.41×10^{-4}	
	B7	^{99m}Tc	648	0.35336	4.28×10^{-7}	
	B8	131I	3885	1.34764	2.12×10^{-4}	
	B9	131I	2070	0.71805	1.13×10^{-4}	
	B10	^{99m}Tc	826	0.45043	5.45×10^{-7}	
	B11	^{99m}Tc	294	0.16032	1.94×10^{-7}	
	B12	131I	1554	0.53906	8.48×10^{-5}	
	B13	^{99m}Tc	388	0.21158	2.56×10^{-7}	
	B14	131I	2640	0.91577	1.44×10^{-4}	
	B15	^{99m}Tc	311	0.16959	2.05×10^{-7}	
	B16	^{99m}Tc	80	0.04362	5.28×10^{-8}	
	B17	^{99m}Tc	210	0.11452	1.39×10^{-7}	
	B18	^{99m}Tc	254	0.13851	1.68×10^{-7}	
C	C1	^{99m}Tc	566	0.30865	3.74×10^{-7}	8.88×10^{-3}
	C2	131I	896	0.31081	4.89×10^{-5}	
	C3	131I	1483	0.51443	8.09×10^{-5}	
	C4	131I	5478	1.90023	2.99×10^{-4}	
	C5	131I	104	0.03608	5.68×10^{-6}	
	C6	131I	628	0.21784	3.43×10^{-5}	
	C7	131I	2445	0.84813	1.33×10^{-4}	
	C8	^{99m}Tc	370	0.20177	2.44×10^{-7}	
	C9	^{99m}Tc	244	0.13306	1.61×10^{-7}	
	C10	131I	480	0.16650	2.62×10^{-5}	
	C11	^{99m}Tc	52	0.02836	3.43×10^{-8}	
	C12	131I	5180	1.79686	2.83×10^{-4}	
	C13	^{99m}Tc	389	0.21213	2.57×10^{-7}	
	C14	131I	2756	0.95601	1.50×10^{-4}	
	C15	^{99m}Tc	2298	1.25313	1.52×10^{-6}	
	C16	^{99m}Tc	1451	0.79125	9.58×10^{-7}	
	C17	^{99m}Tc	401	0.21867	2.65×10^{-7}	
	C18	^{99m}Tc	342	0.18650	2.26×10^{-7}	
D	D1	^{99m}Tc	143	0.07798	9.44×10^{-8}	4.57×10^{-5}
	D2	^{99m}Tc	296	0.16141	1.95×10^{-7}	
	D3	^{99m}Tc	466	0.25412	3.08×10^{-7}	
	D4	^{99m}Tc	704	0.38390	4.65×10^{-7}	
	D5	^{99m}Tc	699	0.38117	4.62×10^{-7}	
E	E1	131I	66	0.02289	3.60×10^{-6}	9.93×10^{-4}
	E2	131I	57	0.01977	3.11×10^{-6}	
	E3	131I	241	0.08360	1.32×10^{-5}	
F	F1	^{99m}Tc	287	0.15650	1.90×10^{-7}	1.65×10^{-3}
	F2	131I	322	0.11170	1.76×10^{-5}	
	F3	131I	279	0.09678	1.52×10^{-5}	

This study addresses the safety and health concerns of the occupational workers at INMAS. Their average annual effective doses due to inhalation of ^{131}I and ^{99m}Tc radionuclides are found to vary between $4.57 \times 10^{-5} \text{ mSv}$ for worker *D* and $9.72 \times 10^{-3} \text{ mSv}$ for worker *B*. These values are much smaller compared to the permissible annual effective dose of an occupational worker, which is 20 mSv .³

There were a few factors which probably have led to results being deviated from accuracy. The urine samples collected over a period of 24 hours, instead of 2-3 hours, are required for a more accurate assessment of the initial intake, but the

workers under this observation were unwilling to provide samples after hours. Moreover, the samples collected from them were not analyzed immediately; it took some time to transport them from INMAS to AECD, and to prepare the detector for the analysis. This led to a count *C* smaller than that at the time of sample collection, and hence a smaller dose than the actual one, especially for ^{99m}Tc because of its short decay half-life (approximately 6 hours). For these reasons and more, in-vivo methods (thyroid monitoring, for example) are preferred over in-vitro methods;⁸ nevertheless, indirect measurements provide results that are reasonably

good and acceptable, especially when direct methods are not available, like at AECD.

Also, the ionizing radiations from radioisotopes other than ^{131}I and $^{99\text{m}}\text{Tc}$ (e.g., ^{40}K appearing in the example samples in Fig. 1) are ignored due to the low counts in the HPGe detector. This might have shrunk the calculated dose a bit, but the considerable small sizes of these counts make them quite insignificant compared to the dominant counts.

Since the dose values found here are much smaller than the permitted values,³ it appears reasonable to assume that even with the corrections made for the factors mentioned above, the accurate results, though a bit different from what was found, would still be very low compared to not only the safe limit for occupational exposure, but also the much lower limit for public exposure, namely 1 mSv .³

It is important to bear in mind that while there are threshold doses for deterministic health effects to take place, there is none for stochastic effects. The ICRP-recommended dose-limits on effective dose, which is a measure of stochastic health risks by definition, only represent the level of dose above which the risks of stochastic effects are considered unacceptable.³ These effects, for example cancer and heritable effects, can occur at any level of exposure, even much below the limits. The higher the exposure, the greater the risk. And hence the only consolation with a low effective dose is that the chances of getting effected are low.

Exposure to manmade sources of radioisotopes used in various areas of civilization has benefits just as well as risks. And the whole world is trying to balance these two by following the ALARA (As Low As Reasonably Achievable) principle, that is, by minimizing exposures while allowing reasonable amounts thereof for benefits that outweigh the associated risks.

V. Conclusion

We live in an inherently radioactive world. While natural radiation is something that cannot be controlled, radiation from manmade sources can be, and therefore should be, controlled and monitored continuously. The first step toward this goal is to assess the radiation doses imparted to those exposed to manmade sources. Like many other studies,^{9,10} the present study is performed with an effort to do the same, in this case on the occupational workers at Institute of Nuclear Medicine and Allied Sciences (INMAS), Dhaka, who handle unsealed sources of ^{131}I and $^{99\text{m}}\text{Tc}$ for medical purposes. Their average annual effective

doses due to inhalation of radionuclides from these sources are found to be smaller by at least 3 orders of magnitude compared to the permissible annual effective dose of an occupational worker, as recommended by ICRP. This means that the chances of stochastic effects to take place are very thin. However, since very thin chances still do not mean zero chances, protective measures should continue to be used anyway.

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