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Estimation of the Effective Radiation Dose in Nuclear Medicine Workers from Dhaka by Measuring Radioactivity in Urine Sample Resulting from Internal Exposure

Research Article

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ABSTRACT

This study estimates the effective doses and activity concentration from 151 urine samples which are collected from 15 (Fifteen) occupational workers at National Institute of Nuclear Medicine and Allied Science (NINMAS) in Dhaka, Bangladesh. The radioactivity concentration in urine sample is due to intake of I-131, Tc-99m and F-18. The samples have been analyzed using High Purity Germanium (HPGe) detector. The radioactivity of I-131 and Tc-99m was found 0.91 ± 0.26 BqL^{-1} to 504.49 ± 6.03 BqL^{-1} and 0.15 ± 0.21 BqL^{-1} to 191.19 ± 6.98 BqL^{-1} respectively. Radioactivity of F-18 was found from 0.031 ± 0.022 BqL^{-1} to 0.282 ± 0.065 BqL^{-1}. The effective doses of occupational workers have been also calculated using the radioactivity concentration and the dose coefficient according to ICRP publication 78.

Keywords: *Radiation Dose, Urine sample, Internal exposure, Nuclear medicine worker*

Introduction

Over the last few decades, it is seen that the use of radioisotopes in nuclear medicine is increasing continuously. Among the most widely used radionuclides in the field of nuclear medicine, Tc-99m, I-131 and F-18 are widely applied for diagnostic and therapeutic purposes, I-131 is highly volatile and radiotoxic, which may pose an occupational radiological hazard. In such situations,

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staff handling unsealed sources of I-131 may be internally exposed by inhalation. Moreover, due to short half-life the internal radiation risk that is poses may sometimes be overlooked [1]. F-18 has a short half-life of 109.8 minutes which makes rigorous inventory tracking unnecessary. F-18 may pose a radiological hazard by external and internal exposure [2].

The radioisotopes used in Bangladesh are I-131, Tc-99m and F-18. Tc-99m and F-18 is used in diagnostic purpose for 3D imaging gamma camera (SPECT, PET-CT) and I-131 is used for therapeutic purpose for treating thyroid diseases. Iodine is rapidly absorbed into the circulation following inhalation or ingestion. It is concentrated in the thyroid and excreted predominantly in urine [3].

An unsealed source is radioactive material that is not sealed or is in a form whereby some radioactive material could be lost under normal use (e.g., a powder, a liquid or a gas). When radioactive material is sealed, it may present an external radiation hazard to the people nearby. However, when it is unsealed, it can present an internal radiation hazard as there is a chance that this material may enter the body. In addition, unsealed sources may also be an external radiation hazard, although usually to a lesser extent than sealed sources. Internal exposures arise when radiation is emitted from radioactive materials present within the body. Once these particles get into the body, damage can occur since there is no protective dead skin layer to shield the organs and tissues. Internal exposures are not limited to the intake of large amounts at one time (acute exposure). Chronic exposure may arise from an accumulation of small amounts of radioactive materials over a long period of time [4].

For this study Nuclear Medicine Center was selected (NINMAS) among 18(Eighteen) Nuclear Medicine Centers in Bangladesh. The present work was taken up to measure the contamination of the occupational staff because of the above radiopharmaceuticals. In this study, we have considered bioassay method of urine samples. The aim of this study is to analyze the internal radioactivity in occupational workers due to handing of unsealed sources while treating the patient for diagnostic and therapeutic purposes.

Some cases of internal contaminations have already been identified at hospitals located in different countries [Hong-Bo Wang et al**.** 2016; Lucena et al, 2015; Departments of Nuclear Medicine, SGPGIMS Lucknow, Uttar Pradesh, 2013 and A. Robert Schleipman et al. 2015.]. Individual monitoring procedures of internal exposure for nuclear medicine workers were reported based on practical screening implemented for most radionuclides used in nuclear medicine, including gamma emitters and beta emitters [researchgate.net/ publication/264691947; Gauri S. Pant et al. 2006 and Vidal MV, Dantas Al and Dantas BM. 2007.]

Experimental Methods

The present experiments were carried out with a coaxial type HPGe (ORTEC) detector for measuring the levels of radioactivity in urine samples of occupational workers collected from National Institute of Nuclear Medicine and Allied Science (NINMAS), Dhaka.

A computerized multichannel pulse height analyzer (Emcaplus emulsion software) is used to measure rapidly the spectrum of pulse heights emerging from the spectroscopy amplifier. It is capable of analyzing pulses simultaneously in many different intervals or channels. Multichannel analyzer (MCA) performs for the essential functions of collecting the data providing a visual monitor and producing output either in the form of final result or as raw data for further analysis.

The shielding of the detector from the environmental radiation is an absolute for low-level measurement of activity. The shielding arrangement surrounding the detector has been designed and fabricated by using lead and steel material, available in local market. Because of high density (11.4 gm/cc) and large atomic number (Z=82) of lead, it is widely for the construction of radiation shielding. Hard gamma ray from external background source (such as 1.46 MeV from 40 K) can be absorbed efficiently by lead. Moreover, it is reasonably effective for removing many of the cosmic ray components of the background radiation. The photoelectric absorption of gamma rays in the lead shielding around the detector can lead to the generation of characteristic X-ray, which is covered by steel. Steel has much lower atomic number (39) compared to that of lead.

Total 151 Urine samples of a group of fourteen workers, who manipulate radio-nuclides at the National Institute of Nuclear Medicine and Allied Science (NINMAS) were collected using the following procedure. A standard plastic pot with a polythene bag, tissue paper, and a rubber band were provided for collecting urine samples. A mark point was given on these pots so that the volume of urine sample was below than two-third of pot volume (The volume of urine samples was below 350 ml). The name, date and time of collection (initial and final) were written on the plastic pots. To evaluated an average effective radioactive dose, multiple samples were collected from the same worker. Because, not all nuclear medicine workers had to handle the same amount of radioactive source everyday. The Number of patients treated by them varied day by day. Therefore, multiple samples are collected from a worker to evaluated the average amount of radiation inhaled by a worker over a certain number of days. These samples were collected on different days from occupational workers during 3 to 4 hours after manipulation of I-131, Tc-99m and F-18 in standard plastic pots. Workers should handle the radioactive nuclides maintaining all the radioactive shielding. Finally, sample was positioned on the HPGe detector to perform the measurement in the Health Physics Division Laboratory, Atomic Energy Center, Dhaka.

Radioactivity Measurement

The experimental procedure to perform the measurement of each sample is as follows:

The liquid nitrogen drawer of the HPGe detector was filled with liquid nitrogen at least 12 hours before the measurement was started. The high voltage bias supply to the detector was gradually raised to the operating voltage (+3200 Volt), the amplifier coarse gain, fine gain and peak shaping time were also adjusted to the desired values. The counting time in the MCA was adjusted to 5000 seconds to obtain statistically significant counts. Then the energy calibration of the detector was cheeked by placing Cs-137 and Co-60 point sources at the detector axis with a source-to-detector distance of about 10 cm for 100 seconds. Then a background spectrum was obtained by placing an empty plastic container as the same geometry of the sample plastic container at the top of the well-shielded detector head for 5000 seconds. The sample's counts were read out for 5000 seconds. The computer software "Silena - EMCAPLUS version 1.01.2" created the data of each sample. The analysis sheet contained the energy and corresponding counts per second of emitted gamma photons including the statistical error, FWHM, etc. after searching for the particular radionuclide's peak and taking the corresponding cps, the total activity for particular sample was calculated by the equation stated below. Background reading of the detector was taken by placing an empty container on the detector head as describe earlier. The mean background reading from a set of measurements was subtracted from each of the sample reading.

Efficiency Calculation of Urine Samples

The efficiency for this urine sample was calculated by using the efficiency equation [12].

$$Y = 346.47X^{-0.85} \quad (1)$$

Where, Y is for detector efficiency in percentage and X is for Gamma ray energy in keV.

Radioactivity Calculation

The background counts were then subtracted from the observed sample counts [12-13].

Urine radioactivity

$$A_u = \frac{C\times100\times1000}{\varepsilon(E)\times P_\gamma\times V}\ [BqL^{-1}] \qquad (2)$$

Where,

C = Net counts per second [net counts from computer], Ɛ(E) = Efficiency of the detector at the energy E (keV) of the peak earlier, V = Volume of urine in millimeter (mL), P_γ= The photon energy emission probability at energy E (keV).

Using the above-mentioned equation, the radioactivity of all urine samples was calculated.

Measurement of MDL of HPGe Detector

The method detection limit (MDL) is defined as the minimum concentration of a substance that can be measured and reported with 99% confidence that the analytic concentration is greater than zero. MDL can be written as

$$MDL = \frac{4.66S_b}{\varepsilon(E)\times P_\gamma\times W} \qquad (3)$$

Where,

S_b = Estimated error of the net count rate, Ɛ(E) = Counting efficiency at the desired energy of the radionuclides, P_γ = Absolute transition probability of gamma decay, W = Weight of the urine sample in kg.

Error Analysis

Experimental observations may be inaccurate due to the systematic and random errors. The systematic errors can be minimized by employing sophisticated equipment and techniques and by improving personal skill.

The error calculation was done by the following formula [14],

$$\sigma = \sqrt{\frac{S+B}{T^2}+\frac{B}{T^2}} \qquad (4)$$

Where,

S+B = Total sample with background counts, B = Total background counts, T = Time in second.

Effective Dose Calculation

There are two method of effective dose calculation. These are given below

Direct Method of radiation dose calculation.

Indirect Method of radiation dose calculation.

Direct Method of Radiation Dose Calculation

The committed effective dose and the committed equivalent dose can be calculated as follows

$$H_A = \sum_j I_{AJ}\, h_{AJ} \qquad (5)$$

Where,

$H_A$is the committed effective dose or the committed equivalent dose (S_V) by inhalation.

$h_{AJ}$is the inhalation dose efficiency (S_V/Bq) for the effective dose or for the target organ considered for the radionuclide j.

For an acute intake at the time *t* before the measurement, the activity of the radionuclides inhaled can be calculated by

$$I_{AJ} = \frac{R_j}{r_{aAj}(t)} \qquad (6)$$

Where,

I_{AJ} is the activity (Bq) of the radionuclide *j* inhaled by a member, R_j is the whole body the activity of the radionuclides *j* or the activity of the radionuclides *j* in the organ considered, obtained by indirect measurement (Bq), $r_{aAj}(t)$is the fractional activity in total body or in the organ considered at time *t* after an acute inhalation of a radionuclide *j* by a member.

Values of $r_{aAj}(t)$ have been calculated for urine and in some cases for faces using the bio kinetic models [8].

For a chronic intake at time *t* before the measurement, the activity of the radionuclides inhaled can be calculated by

$$I_{AJ} = \frac{R_j t}{r_{cAj}(t)} \qquad (7)$$

Where,

$r_{cAj}(t)$is the fractional activity in total body or in the organ considered at time *t* after an acute inhalation of a radionuclide *j* by a member.

Indirect Method of Radiation Dose Calculation

The committed effective dose and the committed equivalent dose can be calculated as follows

$$H_A = \sum_j I_{AJ}\, h_{AJ} \qquad (8)$$

For an acute intake at the time *t* before the measurement, the activity of the radionuclides inhaled can be calculated by

$$I_{AJ} = \frac{E_j}{e_{aAj}(t)} \qquad (9)$$

Where,

$E_j$tis the whole body the activity of the radionuclides *j* or the activity of the radionuclides *j* in the organ considered, obtained by indirect measurement (Bq), $e_{aAj}(t)$is the fractional activity in total body or in the organ considered at time *t* after an acute inhalation of a radionuclide *j* by a member.

Values of $e_{aAj}(t)$ have been calculated for urine and in some cases for faces using the bio kinetic models [9].

For a chronic intake at time *t* before the measurement, the activity of the radionuclides inhaled can be calculated by

$$I_{AJ} = \frac{E_j t}{e_{cAj}(t)} \qquad (10)$$

Values of $e_{aAj}(t)$ have been calculated for urine and in some cases for faces using the bio kinetic models and data described in ICRP publication 78.

Results and Discussion

Among the 151 urine samples 123 samples were collected from 12 male workers and rest of the 28 samples were collected from 3 female workers. As it is seen in the Table 1, the concentration of Tc-99m radionuclide was observed in 77 samples, whereas presence of I-131 was detected in 60 samples which is shown in table 2. The concentration of F-18 radionuclide was observed in 14 samples which is shown in table 3.

Radioactivity Concentration

The concentration ranges of I-131 was from 0.91 ± 0.26 BqL^{-1} to 504.49 ± 6.03 $BqL^{-1}$and the concentration ranges of Tc-99m were from 0.15 ± 0.21 BqL^{-1} to 191.19 ± 6.98 BqL^{-1} which is shown in table 2 and 1 respectively. The concentration ranges of F-18 were from 0.031 ± 0.022 BqL^{-1} to 0.282 ± 0.065 BqL^{-1} which is shown in 3.

The highest and lowest detected radioactivity concentrations of Tc-99m in urine samples were 191.19 ± 6.98 BqL^{-1} and 0.15 ± 0.21 BqL^{-1} respectively, whereas the highest and lowest detected radioactivity concentrations of I-131 in urine samples were 504.49 ± 6.03 BqL^{-1} and 0.91 ± 0.26 BqL^{-1} respectively. The highest Radioactivity of F-18 was 0.031 ± 0.022 BqL^{-1} found in the urine samples of occupational workers who handled F-18 sources, and the lowest detected radioactivity concentrations of F-18 in urine samples were 0.282 ± 0.065 BqL^{-1} due to handle low activity of unsealed F-18 sources and for proper shielding.

It was inferred that the worker with I-131 concentration in urine might inhale I-131 radionuclide which was diffused into the air because of its volatile nature, during the distribution job of I-131. The possible reason for Tc-99m and F-18 concentration in urine is for unconscious manipulation in the workplace.

Table 1: Average Radioactivity Concentration of Tc-99m in Urine Samples.

Name of the Worker	Net Sample Weight (ml)	Net count per second (cps)	Energy (keV)	Activity (BqL^{-1})	Error Activity (cps)	Activity ± Error Activity (BqL^{-1})	Average Activity of Tc-99m in BqL^{-1}
A	200	0.012	140.47	16.82455	2.172040455	16.82 ±2.17	53.486
	185	0.022		33.34596	3.179412626	33.35 ± 3.18	
	220	0.15		191.1881	6.981202432	191.19 ± 6.98	
	230	0.092		112.1637	5.229658863	112.16 ± 5.23	
	200	0.058		81.31867	4.775196897	81.31 ± 4.77	
	200	0.058		81.31867	4.775196897	81.31 ± 4.77	
	195	0.018		25.88393	2.728405547	25.88 ± 2.72	
	213	0.0063		8.293794	1.477739744	8.29 ± 1.48	
	210	0.011		14.6881	1.980543249	14.69 ± 1.98	
	210	0.0031		4.139374	1.051402087	4.14 ± 1.05	
	190	0.013		19.18589	2.379717245	19.19 ± 2.38	
B	200	0.015	140.47	21.03069	2.428415054	21.03 ± 2.43	37.46
	160	0.011		19.27813	2.599463014	19.28 ± 2.60	
	193	0.059		85.72095	4.990866802	85.72 ± 4.99	
	203	0.061		84.2609	4.824764969	84.26 ± 4.82	
	203	0.0033		4.558376	1.122194704	4.56 ± 1.12	
	170	0.006		9.896796	1.806899453	9.90 ± 1.81	
C	171	0.022	140.47	36.07604	3.439715414	36.08 ± 3.44	31.29
	190	0.016		23.61341	2.640059245	23.61 ± 2.64	
	200	0.11		154.2251	6.576179054	1564.22 ± 6.58	
	200	0.11		154.2251	6.576179054	154.22 ± 6.58	
	220	0.073		93.04488	4.870191427	93.04 ± 4.87	
	220	0.0021		2.676633	0.826027011	2.68 ± 0.83	
	220	0.007		8.922111	1.508112091	8.92 ± 1.51	
	190	0.0015		2.213757	0.808349755	2.21 ± 0.81	
	160	0.0219		38.38101	3.667831343	38.38 ± 3.67	
	160	0.00031		0.543293	0.290418455	0.54 ± 0.29	

D	210	0.033	140.47	44.06431	3.430401533	44.06 ± 3.43	32.58
	182	0.0096		14.79082	2.134870377	14.79 ± 2.13	
	219	0.077		98.59137	5.024681408	98.59 ± 5.02	
	209	0.043		57.69185	3.93455141	57.69 ± 3.93	
	189	0.0031		4.599305	1.168224541	4.60 ± 1.17	
	189	0.003		4.45094	1.149227754	4.45 ± 1.15	
	218	0.003		3.858842	0.996348833	3.86 ± 0.99	
E	197	0.029	140.47	41.27851	3.427994018	41.27 ± 3.43	23.34
	207	0.008		10.83707	1.713491406	10.84 ± 1.71	
	200	0.072		100.9473	5.320390816	100.95 ± 5.32	
	220	0.038		48.43432	3.513794835	48.43 ± 3.51	
	230	0.0023		2.804092	0.82688167	2.80 ± 0.83	
	200	0.0027		3.785524	1.030289251	3.79 ± 1.03	
	200	0.007		9.814323	1.6589233	9.81 ± 1.66	
	200	0.007		9.814323	1.6589233	9.81 ± 1.66	
	170	0.005		8.24733	1.649465983	8.25 ± 1.65	
	181	0.0034		5.267355	1.277521348	5.27 ± 1.28	
	199	0.011		15.50001	2.090020514	15.5 ± 2.09	
F	201	0.02	140.47	27.90141	2.790141463	27.90 ± 2.79	13.02
	180	0.003		4.673487	1.206689142	4.67 ± 1.21	
	194	0.0045		6.504338	1.408809779	6.50 ± 1.41	
G	171	0.018	140.47	29.51676	3.111339659	29.52 ± 3.11	54.89
	190	0.125		184.4797	7.379189922	184.48 ± 7.38	
	160	0.061		106.906	6.121420554	106.91 ± 6.12	
	190	0.049		72.31606	4.62010368	72.32 ± 4.62	
	190	0.049		72.31606	4.62010368	72.32 ± 4.62	
	180	0.008		12.46263	1.843248111	12.46 ± 1.84	
	180	0.0018		2.804092	0.894904619	2.80 ± 0.89	
	170	0.008		13.19573	2.273631952	13.19 ± 2.27	
	170	0		0	0.164946598	0 ± 0.16	
H	221	0.006	140.47	7.61292	1.389922656	7.61 ± 1.39	16.65
	200	0.032		44.86547	3.546927211	44.87 ± 3.55	

	199	0.032		45.09093	3.564750966	45.09 ± 3.56	
	199	0.008		11.27273	1.782375483	11.27 ± 1.78	
	210	0.008		10.68226	1.689012958	10.68 ± 1.69	
	189	0.0085		12.611	1.934438563	12.61 ± 1.93	
	189	0.0001		0.148365	0.209819322	0.15 ± 0.21	
	190	0.00061		0.900261	0.515488047	0.90 ± 0.52	
I	210	0.032	140.47	42.72902	3.378025915	42.73 ± 3.38	14.67
	210	0.024		32.04677	2.925456257	32.05 ± 2.92	
	210	0.024		32.04677	2.925456257	32.05 ± 2.92	
	210	0.003		4.005846	1.034304979	4.01 ± 1.03	
	199	0.008		11.27273	1.782375483	11.27 ± 1.78	
	199	0.005		7.045458	1.409091543	7.05 ± 1.41	
	199	0.005		7.045458	1.409091543	7.05 ± 1.41	
	169	0.006		9.955357	1.817591166	9.96 ± 1.82	
	199	0.008		11.27273	1.782375483	11.27 ± 1.78	
	199	0.0008		1.127273	0.563636617	1.23 ± 0.56	
	180	0.0018		2.804092	0.93469739	2.80 ± 0.93	

From Table 1, it is observed that the highest average activity, 54.89BqL^{-1} was that of worker G who contaminated by highest radioactivity due to handling of unsealed radioactive source of Tc-99m which has the activity about 735 mCi. Worker F was contaminated by the lowest average activity 13.02 BqL^{-1}. Even though both the worker G and F handle the same unsealed source, their exposure levels are different because the number of patients treated by both of them is different. Beside this in Bangladesh, workers are only concern about whole radiation measured by TLD badge but they hardly think about the internal exposure. In this case, may be F worker were more cautious and always tried to maintain proper radioactive shielding. From figure 1, we can easily notice that.

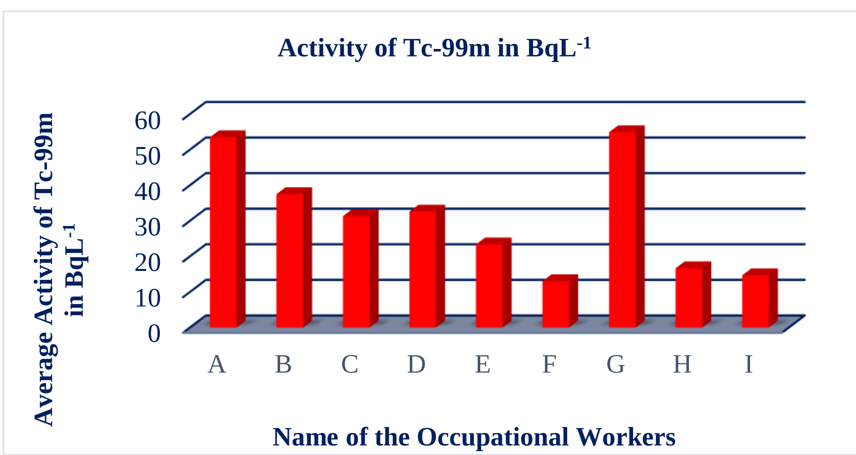

Fig. 1: Activity Concentration of Tc-99m of the Occupational Workers.

Table 2: Average Radioactivity Concentration of I-131 in Urine Samples.

Name of the Worker	Net Sample Weight (ml)	Net count per second (cps)	Energy (keV)	Activity (BqL^{-1})	Error Activity (cps)	Activity ± Error Activity (BqL^{-1})	Average Activity of Tc-99m in BqL^{-1}
P	160	0.0164	364.48	17.77777	1.388210388	17.78 ± 1.39	18.83
	198	0.0154		13.4899	1.087047493	13.49 ± 1.09	
	180	0.0109		10.50285	1.005990993	10.50 ± 1.01	
	205	0.00437		3.700652	0.55954997	3.70 ± 0.56	
	195	0.023		20.45722	1.348909009	20.46 ± 1.35	
	143	0.0154		18.67833	1.505142682	18.68 ± 1.51	
	180	0.049		47.21467	2.132941201	47.21 ± 2.12	
Q	200	0.0113		9.799452	0.921854913	9.80 ± 0.92	81.59
	172	0.014		14.11734	1.19313317	14.12 ± 1.19	
	188	0.0112		10.33269	0.976347707	10.33 ± 0.98	
	209	0.0074		6.140996	0.713876231	6.14 ± 0.71	
	178	0.0941		91.69021	2.989015437	91.69 ± 2.99	
	187	0.0091		8.440208	0.884774176	8.44 ± 0.88	
	200	0.0051		4.422762	0.619310497	4.42 ± 0.62	
	210	0.189		156.0975	3.590582881	156.09 ± 3.59	
	146	0.0033		3.920256	0.682428982	3.92 ± 0.68	
	209	0.0101		8.381629	0.834003262	8.38 ± 0.83	
	230	0.191		144.032	3.295658446	144.03 ± 3.30	
	241	0.241		173.4416	3.533009679	173.44 ± 3.53	
	241	0.701		504.492	6.025530859	504.49 ± 6.03	
	187	0.007		6.492467	0.775998286	6.49 ± 0.78	
	170	0.013		13.26318	1.163258148	13.26 ± 1.16	
	200	0.078		67.64224	2.421980567	67.64 ± 2.42	
	231	0.32		240.2655	4.247333449	240.27 ± 4.25	
R	203	0.0147		12.55957	1.035895575	12.56 ± 1.04	63.5
	175	0.007		6.937665	0.829209597	6.94 ± 0.83	
	193	0.0132		11.86233	1.032483227	11.86 v 1.03	
	271	0.0065		4.160039	0.515989348	4.16 ± 0.52	
	211	0.0175		14.38497	1.087401475	14.38 ± 1.09	

	188	0.109		100.5592	3.045851659	100.56 ± 3.05	
	190	0.0021		1.916986	0.418320738	1.92 ± 0.42	
	230	0.183		137.9992	3.225901165	137.99 ± 3.23	
	176	0.0021		2.069474	0.451596251	2.07 ± 0.45	
	198	0.0073		6.394565	0.748427244	6.39 ± 0.75	
	214	0.114		92.39414	2.736478251	92.39 ± 2.74	
	241	0.149		107.2315	2.777982082	107.23 ± 2.78	
	241	0.431		310.1798	4.724707652	310.18 ± 4.72	
	241	0.00491		3.533603	0.504286163	3.53 ± 0.50	
	167	0.007		7.270008	0.868932213	7.27 ± 0.87	
	190	0.091		83.06941	2.753724216	83.07 ± 2.75	
	199	0.203		176.9279	3.926885977	176.93 ± 3.93	
S	210	0.0103		8.506899	0.838209669	8.51 ± 0.84	22.1
	205	0.004		3.384227	0.535093263	3.38 ± 0.54	
	157	0.009		9.942514	1.048032983	9.94 ± 1.05	
	192	0.0137		12.37578	1.057334501	12.38 ± 1.06	
	209	0.0011		0.912851	0.275234839	0.91 ± 0.28	
	185	0.104		97.50232	3.023418665	97.50 ± 3.02	
T	201	0.002		1.725787	0.385897788	1.73 ± 0.39	105.08
	204	0.0123		10.45751	0.942921931	10.46 ± 0.94	
	204	0.0441		37.494	1.785428553	37.49 ± 1.79	
	241	0.111		79.8839	2.397716298	79.88 ± 2.40	
	241	0.691		497.2953	5.982398373	497.29 ± 5.98	
	241	0.005		3.598374	0.508886943	3.60 ± 0.51	
U	241	0.098		70.52813	2.252938537	70.53 ±2.25	50.93
	241	0.119		85.6413	2.482617353	85.64 ± 2.48	
	241	0.005		3.598374	0.508886943	3.60 ± 0.51	
	220	0.014		11.03719	0.932813205	11.04 ± 0.93	
	187	0.041		38.02731	1.878034951	38.03 ± 1.88	
	190	0.106		96.76217	2.972026519	96.76 ± 2.97	

From Table 2, it is observed that the highest average activity, 105.08 BqL^{-1} was that of worker T who inhaled the highest radioactivity due to handle highest unsealed radioactive source of I-131 which has the activity about 2280 mCi. Worker P was inhaled the lowest average activity 18.83 BqL^{-1} due to handle very low radioactive source of I-131 and always tried to maintain proper radioactive shielding. From the figure 2, we can easily notice that.

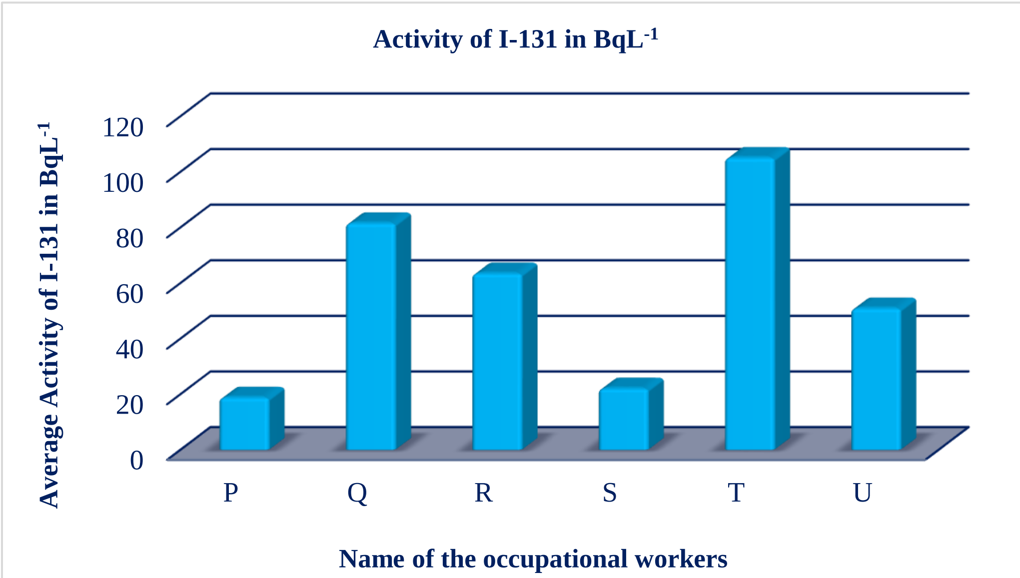

Fig. 2: Activity Concentration of I-131 of the Occupational Workers.

Table 3: Average Radioactivity Concentration of F-18 in Urine Samples.

Name of the Worker	Net Sample Weight (ml)	Net count per second (cps)	Energy (keV)	Activity (BqL^{-1})	Error Activity (cps)	Activity ± Error Activity (BqL^{-1})	Average Activity of Tc-99m in BqL^{-1}
V	163	0.0011	511	0.200098	0.060331833	0.20 ± 0.06	0.16
	200	0.0014		0.207556	0.055471742	0.208 ± 0.055	
	205	0.0005		0.072319	0.032342155	0.072 ± 0.032	
W	177	0.0008		0.134015	0.047381577	0.134 ± 0.047	0.15
	192	0.0015		0.231648	0.059811154	0.232 ± 0.060	
	210	0.0007		0.098836	0.037356617	0.099 ± 0.037	
X	191	0.0017		0.263908	0.064007207	0.264 ± 0.064	0.26
Y	188	0.0012		0.189261	0.054634951	0.189 ± 0.055	0.15
	200	0.0012		0.177905	0.051356854	0.178 ± 0.051	
	210	0.0009		0.127075	0.042358419	0.127 ± 0.042	
	190	0.0013		0.202875	0.056267272	0.203 ± 0.056	
	190	0.0002		0.031211	0.022069841	0.031 ± 0.022	
Z	200	0.0019		0.281683	0.064622624	0.282 ± 0.065	0.28
	194	0.0018		0.275111	0.064844375	0.275 ± 0.065	

From Table 3, it is observed that the highest average activity, 0.28 BqL^{-1} was that of worker Z who contaminated by highest radioactivity due to handle highest unsealed radioactive source of F-18 which has the activity about 200 mCi. Worker W & Y was contaminated by the lowest average activity 0.15 BqL^{-1} due to handle very low radioactive source of F-18 and always tried to maintain proper radioactive shielding. From the figure 3, we can easily notice that.

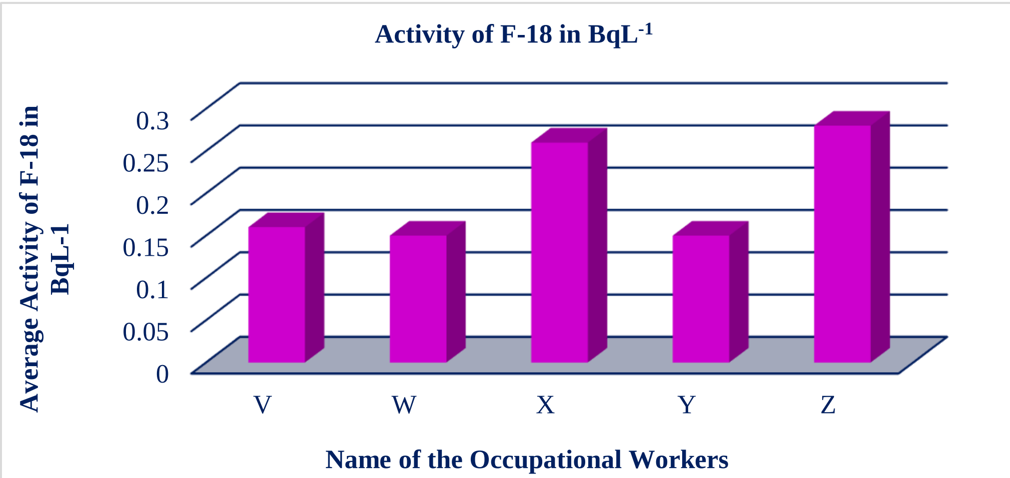

Fig. 3: Activity Concentration of F-18 of the Occupational Workers.

The result of committed effective dose of Tc-99m, I-131 and F-18 in urine samples of occupational staff is given in table 4, 5 and 6.

Table 4: Effective Dose ofTc-99m of Urine Samples.

Name of the Workers	Measured Average Activity (BqL^{-1})	Effective Dose (mSv)=Effective dose (Sv)*1000
A	53.486	0.0025
B	37.46	0.0018
C	31.29	0.0015
D	32.58	0.0015
E	23.34	0.0011
F	13.02	0.0006
G	54.89	0.0026
H	16.65	0.0008
I	14.67	0.0007

From figure 4, it is observed that the highest effective dose, 0.0026 mSv was that of worker G who contaminated by highest radioactivity due to handle highest unsealed radioactive source of Tc-99m. The lowest effective dose 0.0006 mSv was found from worker F due to handle very low radioactive source of Tc-99m.

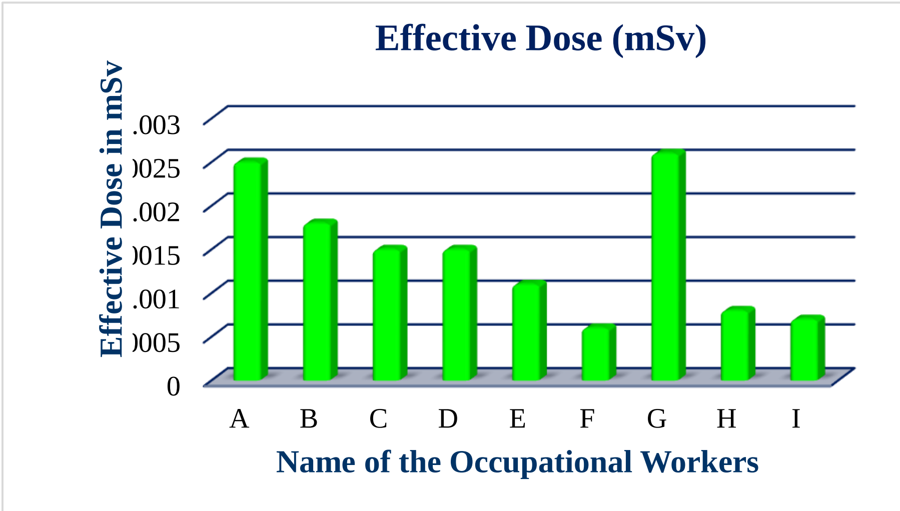

Fig. 4: Effect Dose (mSv) from Tc-99m of the Occupational Workers.

Table 5: Effective Dose from I-131 of Urine Samples.

Name of the Workers	Measured Activity (BqL^{-1})	Effective Dose (mSv)=Effective dose (Sv)*1000
P	18.83	0.0004
Q	81.59	0.0016
R	63.5	0.0013
S	22.1	0.0005
T	105.08	0.0021
U	50.93	0.001

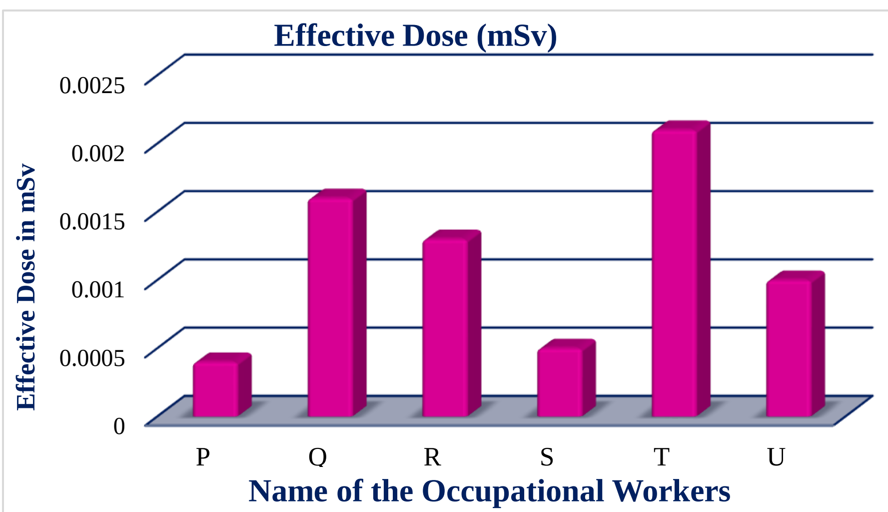

Fig. 5: Effect Dose (mSv) from I-131 of the Occupational Workers.

From figure 5, it is observed that the highest effective dose, 0.0021 mSv was that of worker T who inhaled highest radioactivity due to handle highest unsealed radioactive source of I-131. The lowest effective dose 0.0004 mSv was found from worker P due to handle very low radioactive source of I-131.

Table 6: Effective Dose from F-18 of Urine Samples.

Name of the Workers	Measured Activity (BqL^{-1})	Effective Dose (mSv)=Effective dose (Sv)*1000
V	0.16	0.0000002
W	0.15	0.0000002
X	0.26	0.0000003
Y	0.15	0.0000002
Z	0.28	0.0000004

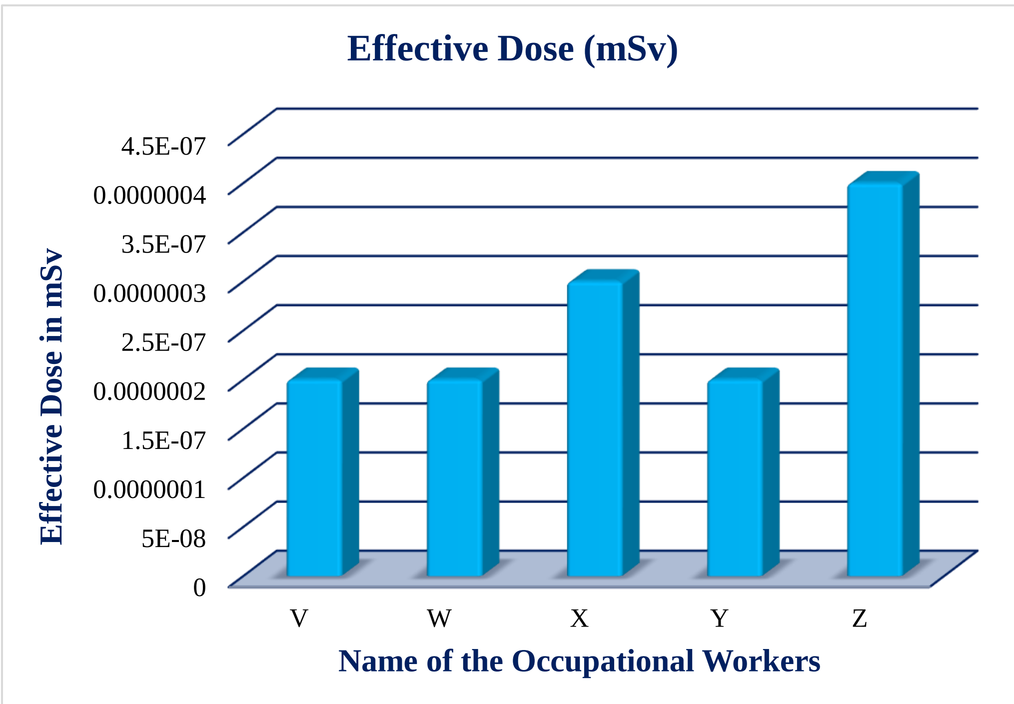

Fig. 6: Effect Dose (mSv) from F-18 of the Occupational Workers.

From figure 6, it is observed that the highest effective dose, 0.0000004 mSv was that of worker Z who contaminated by highest radioactivity due to handle highest unsealed radioactive source of F-18. The lowest effective dose 0.0000002 mSv was found from workers V, W and Y due to handle very low radioactive source of F-18.

Table 7: Average Effective Dose of Tc-99m, I-131, F-18 and Annual Dose Limit of the Occupational workers.

Av. Effective Dose of Tc-99m in mSv	Av. Effective Dose of I-131 in mSv	Av. Effective Dose of F-18 in mSv	Annual Dose limit in mSv
1.46E-03	1.15E-03	1.30E-06	2.00E+01

Due to contamination of Tc-99m, the highest effective dose 2.6E-03 mSv and the lowest effective dose 6.1E-04 mSv. On the other hand, the highest effective dose 2.1E-03 mSv and the lowest effective dose 3.7E-04 mSv due to intake of I-131 and for F-18 the highest effective dose 3.6E-07 mSv and the lowest effective dose 1.9E-07 mSv. The effective dose of occupational staff due to inhalation and contamination of I-131, F-18 and Tc-99m radionuclides were found low in comparison annual effective dose of an occupational staff which is 20 mSv.

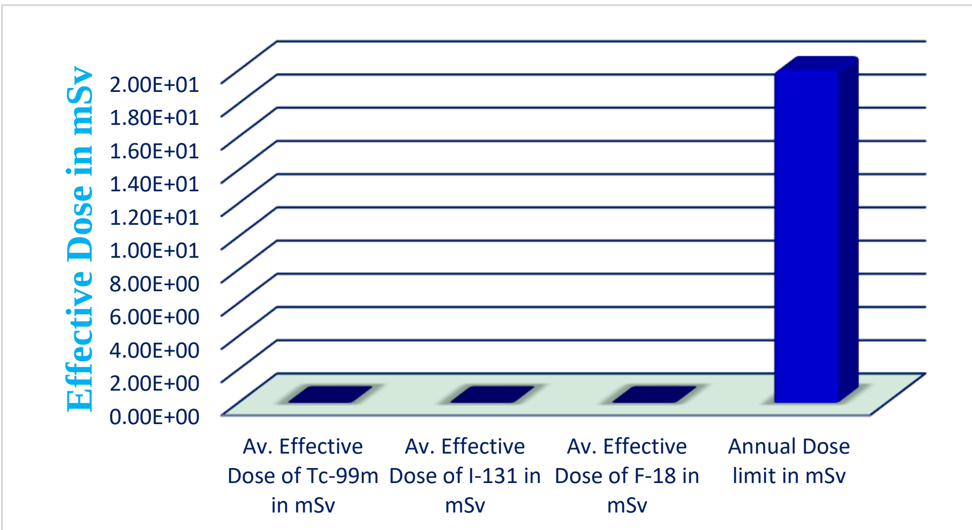

Fig. 7: Annual Dose Limit Comperes with Average Effective Dose of Tc-99m, I-131 and F-18.

From the Figure 7, it is understood that the nuclear medicine staffs in Bangladesh are not in risk of radiation hazards.

Conclusion

Quantitatively evaluating the internal doses expected under the tasks related to the handling of Tc-99m, I-131 and F-18 by the radiation staff is essential for ensuring their health. For internal radiation monitoring, 151 urine samples of 15 occupational staff were collected from NINMAS where there were three female workers and 12 male workers. These urine samples were analyzed to find the radioactivity of Tc-99m, I-131 and F-18 using HPGe detector coupled to a Multichannel Analyzer (MCA) provided by Health Physics Division, Atomic Energy Center, Dhaka under Bangladesh Atomic Energy Commission. After analyzing the samples, radioactivity concentrations were detected in 151 urine samples. The measured activity was then used to calculate the effective doses due to the intake of the radionuclides. The concentration ranges of I-131 was from 0.91 ± 0.26 BqL^{-1} to 504.49 ± 6.03 BqL^{-1}; on the other hand, the concentration ranges of Tc-99m were from 0.15 ± 0.21 BqL^{-1} to 191.19 ± 6.98 BqL^{-1} and of F-18 was from 0.031 ± 0.022 BqL^{-1} to 0.282 ± 0.065 BqL^{-1}. The highest and lowest detected radioactivity concentrations of I-131 in urine samples were 504.49 ± 6.03 BqL^{-1} and 0.91 ± 0.26 BqL^{-1} respectively, whereas the highest and the lowest detected radioactivity concentrations of Tc-99m and F-18 in urine samples were 191.19 ± 6.98 BqL^{-1} and 0.15 ± 0.21 BqL^{-1} and 0.031 ± 0.022 BqL^{-1} to 0.282 ± 0.065 BqL^{-1} respectively. The highest effective dose of Tc-99m was 2.6E-03mSv and the lowest effective dose was 6.1E-04 mSv and the highest effective dose of intake I-131 was 2.1E-03 mSv and the lowest was found3.7E-04 mSv. On the other hand, the highest effective dose of F-18 was 3.6E-07 mSv and the lowest was found 1.9E-07 mSv.

Occupational worker radiation protection cannot be overlooked and it is an important issue. In general, occupational worker can lower their own level of risk if they are aware of radiation protection. But in Bangladesh, workers are only concern about whole radiation measured by TLD badge but they hardly think about the internal exposure. It is necessary to take proper step to confirm occupational worker's safety which they get by internal radiation.

The experimental results showed that the internal monitoring is very useful for checking radiation risk of staff working with unsealed radioactive sources, in particular, for the case of inhalation of I-131 and contamination of Tc-99m and F-18. The results of these measurements can contribute in improvement of the regulations and in deciding the radiological policies for internal dosimetry of radiation for staff handling the radioisotopes in the nuclear medicine fields.

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