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Research Article

Assessment of radiological risks in the Eu-152 nuclides contaminated soil

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Abstract

The assessment of radiation hazards, whether for workers or the public is very important factor in terms of radiation protection for any activities in which nuclear material or radioactive waste is dealt with. A portable NaI (TI) used to measure gamma dose rate of the contaminated soil, the specific activity of ^{152}Eu in soil were determined using high purity germanium (HPGe) detector. The results of gamma dose rate and specific activity were range from 375.96 nGy h^{-1} to 23675.84 nGy.h^{-1}nGy.h^{-1} and 765.70 Bq.kg^{-1} to 354789.84 Bq.kg^{-1} respectively.

Keywords: *Specific activity .dose rate. Annual effective dose*

1. Introduction

The primary purposes of the safety assessment are to determine whether an adequate level of safety has been achieved for a facility or activity and whether the basic safety objectives and safety criteria established by the designer, the operator and the regulatory body have been fulfilled (Emad S. Shamsaldin. *et al*,2014). Radioactive waste generated by the operation of nuclear facilities, nuclear power plants, nuclear research in agriculture or industry, oil and gas extraction fields as well as through the increasing of natural radioactivity of natural radionuclides in the soil over permissible limits (ICRP,2012). These isotopes can seep into the soil as a result of natural or non-natural factors (negligence), or because of lack of knowledge of radiation protection and risk of it, generated such situations

During the removal of radioactive waste from areas contaminated with radioisotopes, radiological hazards are generated for the staff of the removal teams as well as for the general public who are close or present in those areas. The aim of the study is to calculate the effective radiation dose of the workers and its compatibility with the recommendations of the International Atomic Energy Agency (IAEA,1989). And the recommendations of the regulatory bodies within the applicable limits as well as calculation of the qualitative radioactivity of the 152Eu in the soil.

2. Experimental Work

2.1 Samples collection and preparation

Quantities of 14 samples were taken from contaminated site. The samples were dried in an oven at 80 ^{0}C for 48 hours. The samples were grinded to form powder and then sieved by a sieve of 315 µm. The samples of 500 g kept in a cylindrical container.

2.2 Measurements of specific activity

1) Calculate the background of activity for the same volume of cylindrical container and measure the time of analyzing samples.
2) The specific activity of ^{152}Eu, ^{214}Bi and ^{40}K were measured using spectrometer with high portable purity germanium detector (HX, Canberra. USA). It has a relative efficiency of 40%, and a resolution of 1.8 keV for 1332 keV gamma ray emission of ^{60}Co. Gamma vision version (6.8) software for gamma spectrum analysis a gamma spectrometer and relevant accessories were supplied by Canberra, USA (ORTEC® Micro-Detective).

The activity of ^{152}Eu was directly determined using its 344.6 keV, and 609.4keV ,1460.8 for ^{214}Bi and ^{40}K photo peak respectively. The activity concentration of each radionuclide calculated by the following equation (A. Mohanty, *et al*,2004).

$$A_c = \frac{CPS}{I * \epsilon * M} \qquad (1)$$

Where

Ac is the concentration of the radionuclide in the sample (Bq. kg^{-1}), cps is the counts of the sample per second, $I_{(\gamma)}$ is the emission probability of a specific energy photo peak. M is the weight of the sample (kg); ε is the absolute efficiency of the detector.

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The dose rate measured by Ludlum radiation detector by measuring the dose rate above 1 m from the sample (Ludlum Model 19 Micro R Meter) .

Conclusion and discussion

External gamma dose rates

Used two methods to evaluate external exposures of gamma dose rate in this study area. The first was direct measurement of external gamma dose rates; the reading was taken at 1 m above the ground using NaI gamma detector (Ludlum Model 19 Micro R Meter). Second through multiplying the dose rate by 0.00964 to convert dose rate to absorbed dose. The external gamma dose rate ranges from 375.96 nGy h^{-1} to 23675.84 $nGy.h^{-1}$as shown in Table 1.

Table 1: The direct measurements of gamma dose rate for the samples collected from a contaminated area surface (nGy/ h)

Sample	dose rate (μR/ h)	Measured gamma dose rate (nGy/h)	sample	dose rate (μR/ h)	Measured gamma dose rate (nGy/h)
1	109	1050.76	8	785	7567.4
2	133	1282.12	9	1278	12319.92
3	133	1282.12	10	1845	17785.8
4	39	375.96	11	2456	23675.84
5	927	8936.28	12	3200	30848
6	685	6603.4	13	870	8386.8
7	695	6699.8	14	1485	14315.4

Specific activity of ^{152}Eu, ^{214}Pb and ^{40}K

HPGe detector was used to measure the specific activity of ^{152}Eu, ^{214}Pb and ^{40}K samples. The specific activity varies from 765.70 $Bq.kg^{-1}$ to 354789.84 $Bq.kg^{-1}$ of ^{152}Eu as shown in Fig.1, and from (12.6 to 956.22) $Bq.kg^{-1}$ for ^{214}Pb, (241 to 542.6) $Bq.kg^{-1}$ for ^{40}K. The highest specific activity was found in sample 2 for ^{152}Eu and ^{40}K radionuclide and sample 6 for ^{214}Pb as shown fig(2 and 3) while samples 7, 2 and 4 have the lowest activity of ^{152}Eu, ^{214}Pb and ^{40}K respectively. And the results are illustrated in Table 2.

Table 2: The specific activity of ^{152}Eu in the samples

Sample	Calculated Sample Activity concentration(Bq/kg)		
	Eu-152	Pb-214	K-40
1	7523.92	49.41	265
2	354789.84	12.6	542.6
3	823.97	14.6	316
4	4261.33	68.38	241
5	1631.29	20.29	363
6	2713.27	956.22	267
7	765.70	235.7	273
8	10736.57	12	249
9	11659.05	145	320
10	16246.02	289	378
11	21188.94	375	453
12	27207.82	230	480
13	8358.38	67	245
14	13333.66	383	360

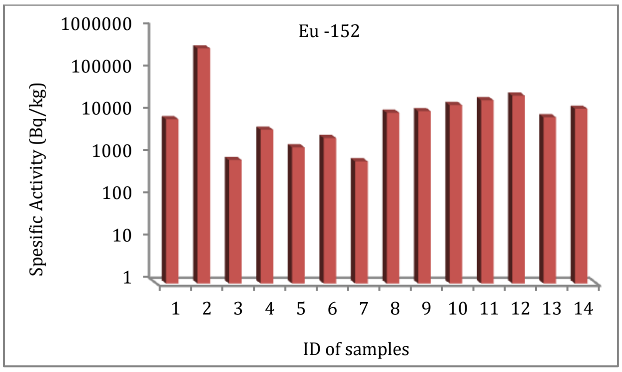

Fig.1: The specific activity of ^{152}Eu in the samples

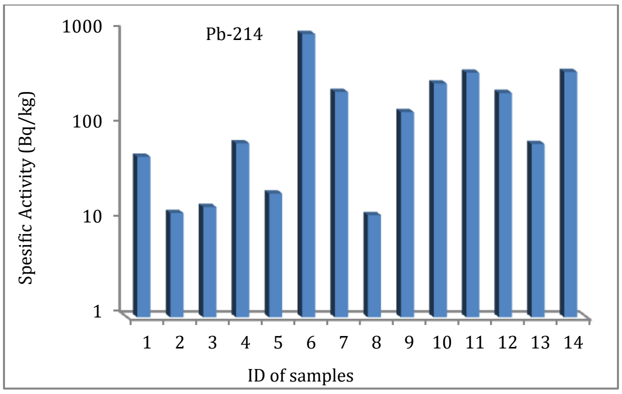

Fig.2: The specific activity of ^{214}Pb in the samples

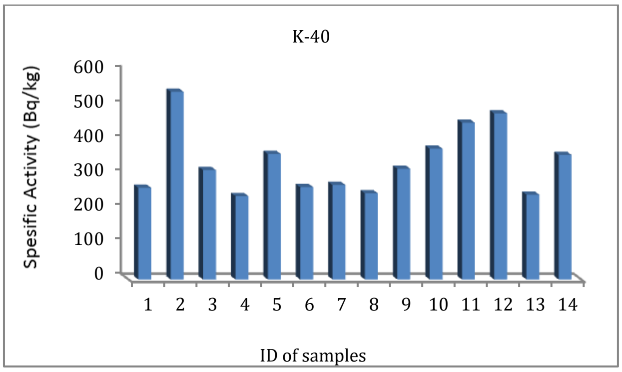

Fig.3: The specific activity of ^{40}K in the samples

Annual Effective Dose Equivalent (AEDE)

To assess the radiation risk of the workers, equation 2 was used to calculate the annual effective equivalent dose. In order to estimate the annual effective dose equivalent (ICRP,2007).

$$AEDE\ (mSv.y^{-1}) = D(nGy.h^{-1}) * 2000h * 0.7(Sv.Gy^{-1}) * 10^{-6} \quad (2)$$

Where

AEDE is the Annual Effective Dose Equivalent. D is the absorbed dose in air, 0.7 is the conversion coefficient. 2000 is the average working time annually were used (ICRP1990).

Table 3: The calculated gamma dose rate from ^{152}Eu and annual effective dose equivalent

Sample No.	Calculated gamma dose rate (nGy/h)	AEDE (mSv/y)
1	954.84	0.1336776
2	1165.08	0.1631112
3	1165.08	0.1631112
4	341.64	0.0478296
5	8120.52	1.1368728
6	6000.6	0.840084
7	6088.2	0.852348
8	6876.6	0.962724
9	11195.28	1.5673392
10	16162.2	2.262708
11	21514.56	3.0120384
12	28032	3.92448
13	7621.2	1.066968
14	13008.6	1.821204

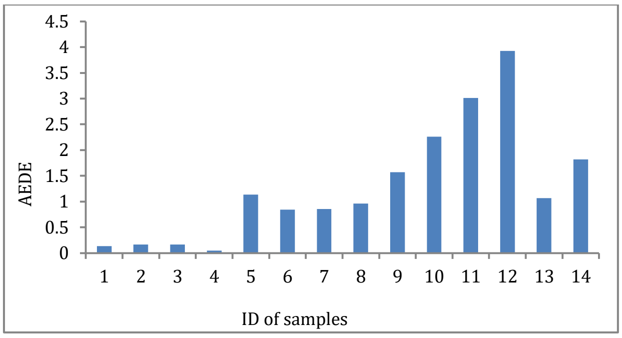

Fig.4: The value AEDE

The AEDE was calculated for the sample of soil according to the calculated gamma dose rate. The results were in the range from 0.0478296 mSv /y to 3.92448 mSv/y as shown in Table 4. Fig 4. One can note all values for the effective annual effective dose do not cause any radiological hazards to the workers compared with the annual effective dose set by the International Commission on Radiological Protection, ICRP.

One can note all radiation values for the annual effective dose as well as the specific activity of the models under study are within the limits of the International Commission on Radiological Protection, ICRP, the International Atomic Energy Agency and the Radiation Center.

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