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Measurement of Radiation Background and Estimation of the Annual Effective Dose for Workers in the Radiochemistry Laboratories at the Al-Tuwaitha Site

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Abstract

This study was achieved to calculate the annual effective dose equivalent (*AEDE*) in units of ($mSv.y^{-1}$), and the average radiation dose rate (*ADR*) in units of ($\mu Sv.h^{-1}$) which were measured by portable devices. The study was carried out on the workers of the destroyed radiochemistry laboratory located at Al-Twuitha nuclear site (south of Baghdad). Radiation background was determined for comparison with the radioactive dose of soil samples measured with HPGe detector and portable devices type LUDLUM. The radioactivity levels of the area around the radiochemistry laboratory building were within the limits of radiation background. The result showed a significant increase of the annual effective dose of C1 laboratory workers, as the annual effective dose of the lysate cell 1 (AHC1)in the lab was about 18.995 mSv/y, with an occupancy factor of 0.042, for an average working hours of one hour per day. An annual effective dose of 24.073 mSv/y was also recorded in hot cell 2 (HCL2), an increase of more than 4 mSv/y, for an occupancy factor of 0.083, equivalent to 2 hours of work per day. The glove boxes 11 (GB11) of Laboratory C2 recorded an annual effective dose of 19,720 mSv/y for an occupancy factor of 0.125, equivalent to 3 working hours per day. The C3 Laboratory and the rest of the laboratories and the health physics rooms recorded an annual effective dose within the workers' allowable limits of 20 mSv/y.

Keywords: Background Measurement radiation, absorbed dose rate, annual effective dose equivalent (AEDE), Occupational exposure, occupancy factor.

قياس الخلفية الإشعاعية وتقدير الجرعة الفعالة السنوية في مختبرات الكيمياء الإشعاعية في موقع التويثة

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الخلاصة

نفذ هذه الدراسة لحساب مكافئ الجرعة الفعالة السنوية للعاملين (AEDE) (ملي سيفرت/ سنة) ، متوسط معدل جرعة الإشعاع (ADR) (مايكروسيفرت /ساعة) المقاسة بواسطة الأجهزة المحمولة في مختبر الكيمياء الإشعاعية المدمرة الواقع في موقع التويثة النووي (جنوب بغداد). تم تحديد الخلفية الإشعاعية للمقارنة مع قياس الجرعة المشعة باستخدام HPGe لقياس عينات التربة والأجهزة المحمولة من نوع LUDLUM كانت المنطقة حول مبنى مختبر الكيمياء الإشعاعية ضمن حدود الخلفية الاشعاعية. اظهرت النتائج زيادة كبيرة في الجرعة الفعالة السنوية للعاملين في المختبر C1 ، وكانت الجرعة الفعالة السنوية للخلية التحليلية الاولى (AHC1)حوالي 18.995 ملي سيفرت / سنة ، مع عامل إشغال قدره 0.042 ، بمتوسط عمل قدره ساعة واحدة في اليوم. كما تم تسجيل جرعة فعالة سنوية تبلغ 24.073 ملي سيفرت / سنة في الخلية الحارة الثانية (HCL2)، بزيادة تزيد عن 4 ملي سيفرت / سنة عن المحددات العالمية مع عامل إشغال يبلغ 0.083 ، أي ما يعادل ساعتين من العمل في اليوم. بينما سجل المختبر C2 في علب القفازات 11 (GB11) جرعة فعالة سنوية قدرها 19.720 ملي سيفرت / سنة وعامل إشغال 0.125 ، أي ما يعادل 3 ساعات عمل في اليوم.في حين سجل مختبر C3 وبقية المعامل وغرف الفيزياء الصحية جرعة فعالة سنوية ضمن الحدود المسموح بها للعاملين والتي هي 20 ملي سيفرت / سنة.

1. Introduction

The Radiochemistry Laboratories (RCL) were established in 1978 by the SNIA TECHINT–Italy firm as part of the Chemical Research Centre at the former Iraqi Atomic Energy Commission (IAEC nuclear), situated at Al-Tuwaitha. The building, which covers an area of roughly 1000 m^2, was destroyed during the second Gulf War in 1991. The RCL goals were to extract plutonium isotopes (^{239}Pu) from spent fuel received from the IRT-5000 (Tammuz-14) reactor on a laboratory scale (dissolution, separation, purifying, and other chemical research and analysis). Since 1973, the waste fuel from this facility has been stored. As a result of the fission of the Uranium-235 (^{235}U) isotope, which belongs to the fuel rod utilized in these procedures, there have been significant amounts of radioactive liquid waste composed of various radionuclides [1]. Because of the impact on the health and safety of the operating employees and the general public in addition to environmental protection, radiological characterization must be carried out to estimate the danger of nuclear accidents and of nuclear sites decommissioning. “Radiological characterization” refers to designation of the nature, location, and concentration of radionuclides in nuclear power plants in general[2]. The International Atomic Energy Agency(IAEA) safety standards establish fundamental safety principles, conditions, and measures to control radiation exposure of people, the release of radioactive material into the environment and to limit the likelihood of events that could lead to a loss of control over a nuclear reactor core, nuclear chain reaction, a radioactive source, or any other source. Radiation and radioactive sources, radioactive material transit, and radioactive waste management[3]. The accident management program should be designed and maintained following plant’s present configuration and design. It could be completed as part of the plant’s systematic safety review[4, 5]. The term ‘radiation risks’ is used broadly to refer to radiation-induced adverse health consequences; radiation exposure of human tissues or organs can result in cell death on a scale large enough to affect the function of the exposed tissue or organ.

The type of consequences, known as ‘deterministic effects,’ are clinically detectable in a person only when the radiation dose surpasses a particular threshold level. A deterministic effect is severe for a dose higher than the threshold level[6, 7]. Three categories of exposure situations must be distinguished in order to create practical criteria for protection and safety:

planned exposure situations, emergency exposure situations, and existing exposure situations[8]. These three types of exposures, when aggregated, cover all exposures for which radiation safety standards apply[6].

2. Background Measurement

Because the limits for residual radioactivity at decommissioned facilities are defined in terms of radiation levels or activity levels above background levels, it is essential to conduct a radiation background survey for the area or institution. This background survey necessitates measuring direct radiation levels (often Gamma-ray exposure rates) and radionuclide concentrations. Pollutants in construction materials and the soil background is determined by on-site measurements and sampling or in the immediate area of the site (a few kilometers from the site limits), which is unaffected by site activity. In a similar way, interior background selections are located within the buildings. Because of naturally radioactive compounds in building materials and the shielding effect that building materials provide, radiation levels inside structures may differ from those in open land areas. Background samples and land area measurements at places unaffected by effluent emissions (upwind and upstream) and other site operations (upgrade from disposal areas) should also be considered. To calculate the net residual radioactivity from approved operations., radiation background levels are subtracted from total radiation or radioactivity levels[9, 10].

3. Occupational exposure limits

Radiation exposure to occupational workers has a specific dose limit; this limit is the maximum permissible limit which should not be exceeded otherwise health effects on radiation workers are induced. So exposure to radiation above the allowable limit must be avoided to prevent its immediate and future biological effects [11]. International Commission on Radiological Protection (ICRP) has set standards for radiation protection based on critical basic principles that are:

1- Justification - it is not permissible to carry out any activity that causes individuals to be exposed to radiation unless it is of sufficient benefit.

2- Improvements - all exposures within the practice should be kept to a minimum and reasonably achievable considering economic and social factors.

3- Dosing determination - Individuals should be given the exposure dose within the recommended limits [12].

Workers in the radiation field should not be exposed to radiation dose close to the specified annual dose limit. It should be kept as low as possible “as low as reasonably achievable,” and this is the principle of “ALARA”, which is the principle of radiation protection of individuals working in nuclear facilities. This principle ensures the reduction of the harmful risks of ionising radiation[13]. The International Atomic Energy Agency (IAEA) has adopted the ICRP’s recommended exposure limits outlined in the ICRP’s Basic Safety Standards. Individual exposure should be limited so that the total effective dose and the equivalent dose to the relevant organs and tissues do not exceed the dose limits[14]. Table 1 shows the dose limits for occupational exposure.

Table 1: ICRP Recommendations for the annual dose limit for occupational exposure[15].

Application	Dose limit
Whole-body	20 mSv per year, averaged over five years, with no single-year maximum dose over 50 mSv
The lens of the eye	150 mSv per year
Skin	500 mSv per year
Hand and feet	500 mSv per year

The (IAEA) offers advice on whether should regulate radioactive elements. The values of exclusion and exemption levels of activity concentration for specific radionuclides of natural and artificial origin are presented in Table 2 according to IAEA essential safety criteria [16].

Table 2: Activity concentration values determined from the exclusion principle for various radionuclides[16].

Radionuclide Activity concentration	($Bq.gm^{-1}$)
^{40}K	10
All other radionuclides of natural origin	1
^{137}Cs	0.1
^{60}C	0.1
$^{152}Eu, ^{154}Eu$	0.1
^{3}H	100

4. Materials and Methods

4.1. Samples Collection and Preparation

Ten surface soil samples were collected from an area several kilometers away from the Al-Tuwaitha site(of the coordinates N=33°.12’38, E=44°.32’50) to determine the radiation background, as shown in Figure 1. Four samples were also collected from the radiochemistry facility area, as shown in Figure 2. Samples were collected with a shovel at a depth of 15 cm from the topsoil layer so as to have approximately 1 kg of weight per sample. To avoid cross-contamination, each soil sample was packaged in an airtight sealable plastic bag and sent to the laboratory for measurement. Data cards were used for soil samples tracking. The data card for each plastic bags specify code, sample type, data collected, dose rate at a touch, and 1 meter

Figure 1: The chosen area for background measurement (indicated in blue color).

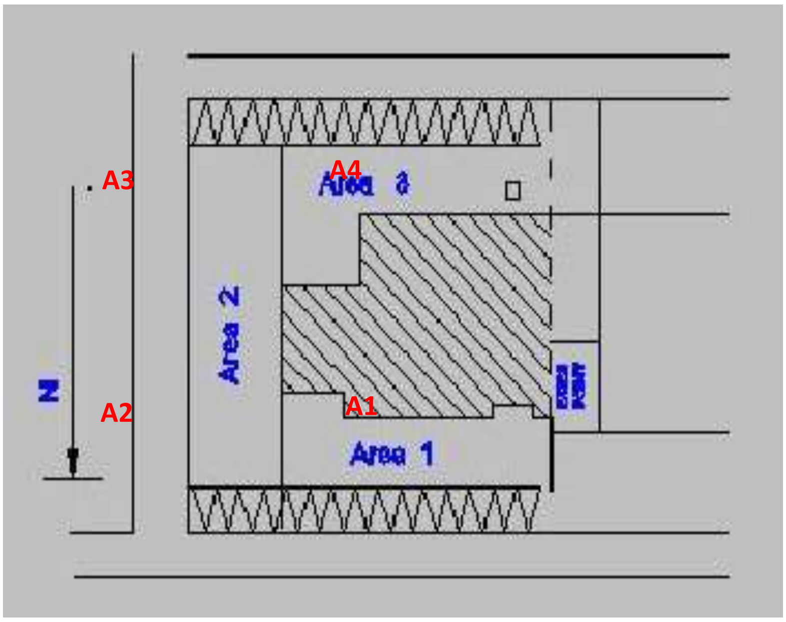

Figure 2: Radiochemistry laboratories building (surrounding area and sampling sites).

4.2. Sample Preparation

Soil samples were prepared for measurement with high purity germanium (HPGe) detector. The samples were dried in an electric oven at a temperature of 80 ° C for 3 hours to remove any moisture. The dried soil was grounded with an electric mill and sieved using a 500µm sieve to obtain homogeneous particles. This was put into a Marinelli beaker, as shown in Figure 3, weighed and kept for about a month to get a radioactive secular equilibrium between mother and daughter radionuclides. The preparation of the soil samples and their measurements were carried out in the Ministry of Science and Technology in Baghdad (MoST) in the Central Laboratory Directorate (CLD), Department of Radiation Measurements, Gamma Emitters Division.

Figure 3: A prepared soil sample filling in a 250 ml Marinelli beaker.

4.3 Instrument and Techniques

This study was conducted in the radiochemistry laboratories of the Tuwaitha Nuclear Research Center in Iraq. In this investigation, ORTEC's high-purity germanium detector HPGe (model GEM65P4-95)(shown in Figure 4), with a relative efficiency of 65 % and a sensitivity of 1.9 keV, was utilized The positive working voltage is 1500 volts. The HPGe detector has a diameter and length of 71.9 and 73.1mm, respectively. It is encircled by a 10 cm thick armored lead to shield it from background radioactivity. The gradient shield is made up of cadmium (Cd) and copper (Cu) layers, and the detector's copper layer is immediately exposed to absorb the lead (Pb) shield's distinctive plates and diminish their peaks in the background spectrum. The HPGe detector is chilled with liquid nitrogen at 77 K (-196 °C) to lower leakage current. A portable device (Ludlum model (2241-2)) of Sodium Iodide (NaI (TI)) scintillation detector (model 44-10) (shown in Figure 5) with a scalar/rate meter unit (μ $Sv.hr^{-1}$) was used for the detection of gamma radiation (γ) in the range of 60 keV-2 MeV with a sensitivity of approximately 900 CPM per ($μR.hr^{-1}$) for ^{137}Cs. The detector is made up of a NaI crystal with a diameter of 5.1 cm and a thickness of 5.1 cm, which is connected to a photomultiplier tube and housed in 0.16 cm thick aluminum housing. The detector operating voltage ranges from 500 to 1200 volts. The instrument operational check was done before each usage by exposing the detector to the supplied check source of ^{137}Cs with activity 0.250μCi to validate the correct reading.

Figure 4: Gamma-ray spectroscopy system

Figure5: The LUDLUM device

4.4 Description of radiochemistry laboratories

Through this study, radiological measurements of radiochemistry laboratories. According to the engineering description of the building and radiochemistry laboratories, RCL, which consists of health physics laboratories (B5, B6), consist of an entrance and a private bathroom to remove radioactive contamination for workers. Health physics laboratories are connected to it and a corridor of length 30 m, roughly joined by a group of rooms and small corridors, Figure 6.

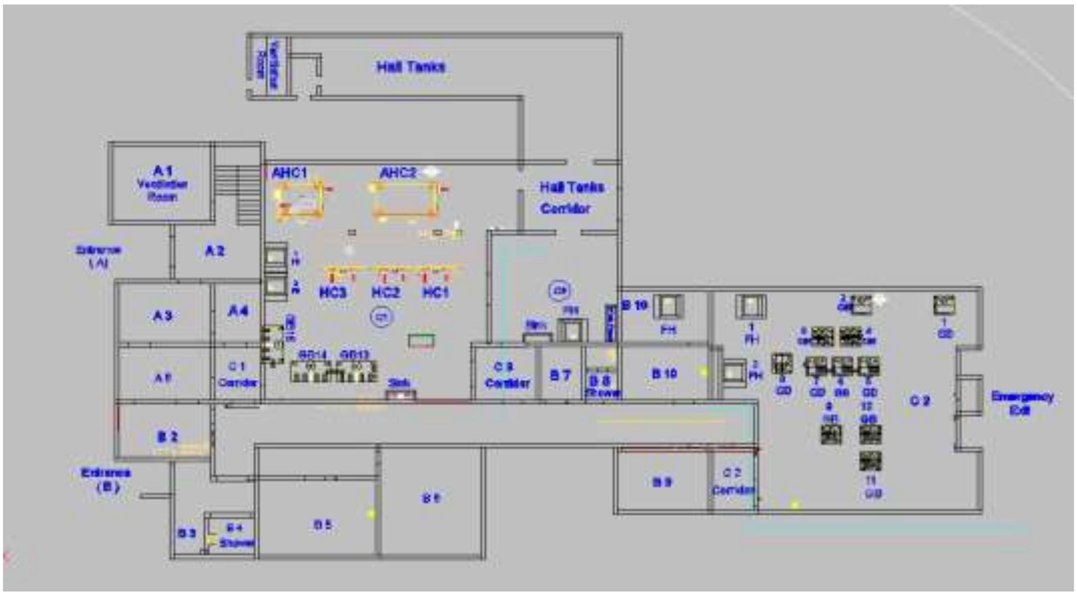

Figure 6: Block diagram of the Radiochemistry Laboratories

Laboratory C1 contains three glove boxes (GB13, GB 14, GB 15) and three hot cells (HC1, HC2, HC3), in addition to 2 lysate cells (HC1, HC2) with 2 fume hoods (FH1, FH2), Laboratory C2 contains 12 glove Boxes (GB1,2,3,4,5,6,7,8,9,10, and 11) and 2 fume hoods. Laboratory C3 contains one fume hood.

4.5 Determination of the background radiation

Radiation background was determined in the first stage by conducting a radioactivity survey using portable devices and taking soil samples with two scenarios. The first scenario was selecting open land outside the Al Tuwaitha site and using a portable device to take direct measurements. At each location, three readings were recorded each for one minute duration and the averaged was calculated. The purpose of this process is to determine the radiation background and whether there is a radioactive contamination around the building of the radiochemistry laboratories. Radiation background is specified in Table 3. The second scenario: a similar structure was chosen in terms of building materials and shielding to determine the radiation background of mobile devices only without taking samples inside the building due to the effect of building materials on the radiation background, as shown in Table 3.

Table 3: Dose rate and specific activity background sample measurements.

Scenarios no. 1					**Scenarios no.2**	
Code site	**Dose Rate µSv/hr**	**Specific activity (Bq.kg^{-1})**			**Code site**	**Dose Rate µSv/h**
		^{232}Th(^{228}Ac)	**^{226}Ra(^{214}Bi)**	**^{40}K**		
S1	0.064	16.19±1.4	18.01±1.1	391.23±22.1	*G1*	0.048
S2	0.055	14.25±1.7	17.23±1.4	370.43±19.8	*G2*	0.053
S3	0.059	15.33±1.6	18.13±1.2	381.21±22.3	*G3*	0.045
S4	0.065	14.16±1.6	16.14±1.6	402.38±19.6	*G4*	0.048
S5	0.061	15.34±1.1	17.12±1.1	401.3±16.7	*G5*	0.055
S6	0.058	14.22±1.3	13.72±2.1	411.4±17.33	*G6*	0.053
S7	0.063	16.13±0.9	15.24±1.2	374.4±21.21	*G7*	0.053
S8	0.053	14.71±1.6	12.31±1.1	369.4±21.11	*G8*	0.056
S9	0.058	15.83±1.9	14.71±1.3	396.6±16.76	*G9*	0.049
S10	0.053	15.63±1.1	15.28±1.4	376.6±15.66	G10	0.045
Max	0.064	16.19±1.4	18.13±1.2	411.4±17.33	Max	0.056
Min	0.053	14.16±1.6	12.31±1.1	369.4±21.11	Min	0.045
Average	0.0589	15.179	15.789	387.495	Average	0.0505

5. Results and Discussion

The annual effective dose equivalent *(AEDE)* in-suit (mSv.y^{-1}) was calculated for workers based on the average radiation dose rate (*ADR*) (µSv.h^{-1}) measured by the portable devices. The AEDE was calculated from the following formula:

$$AEDE\ (\mathrm{m}Sv.\ y^{-1}) = ADR\ (\mu Sv.h^{-1}) \times T \times OF \times 10^{-3} \quad (1)$$

Where: T is the total exposure time in hours per year (8760 h), OF stands for occupancy factor and represents the actual number of hours workers spend near the radioactive source [17]. More than one scenario was adapted in this study depending on the worker’s working hours near the radioactive source. The number of hours adopted in this study were (1, 2, 3, 4 and the 4.8 hours) as recommended by UNSCEAR, 2008; five different values of the occupancy factor (0.042, 0.083, 0.125, 0.166, and the 0.2) were considered as recommended by UNSCEAR, 2008 for the world. The measured absorbed dose rates and the calculated annual effective dose equivalent are shown in Table 4. The radioactivity of the area around the radiochemistry laboratories building, which is open ground surrounded by a concrete fence, were also measured. Table 5 shows the results of the examination of soil samples collected from this area, which are, as noted from the table, within the limits of the radiation background.

Table 4: Annual Effective Dose Equivalent (AEDE) for land around radiochemistry laboratories with different values of occupancy factor

Code	(ADR) (µSv/hr)	(AEDE) (mSv/hr) With OF 0.042	(AEDE) (mSv/hr) With OF 0.083	(AEDE) (mSv/hr) With OF 0.125	(AEDE) (mSv/hr) With OF 0.166	(AEDE) (mSv/hr) With OF 0.2
Ar 1	0.066	0.024	0.048	0.072	0.096	0.096
Ar2	0.063	0.023	0.046	0.069	0.092	0.092
Ar3	0.061	0.022	0.044	0.067	0.089	0.089
Ar4	0.067	0.025	0.049	0.073	0.097	0.097
Ar5	0.073	0.027	0.053	0.080	0.106	0.106
Ar6	0.061	0.022	0.044	0.067	0.089	0.089
Ar7	0.061	0.022	0.044	0.067	0.089	0.089
Ar8	0.053	0.019	0.039	0.058	0.077	0.077
Ar9	0.063	0.023	0.046	0.069	0.092	0.092
Ar10	0.059	0.022	0.043	0.065	0.086	0.086
Ar11	0.069	0.025	0.050	0.076	0.100	0.100
Ar12	0.063	0.023	0.046	0.069	0.092	0.092
Ar13	0.076	0.028	0.055	0.083	0.111	0.111
Ar14	0.067	0.025	0.049	0.073	0.097	0.097
Ar15	0.077	0.028	0.056	0.084	0.112	0.112
Ar16	0.069	0.025	0.050	0.076	0.100	0.100
Ar17	0.071	0.026	0.052	0.078	0.103	0.103
Ar18	0.075	0.028	0.055	0.082	0.109	0.109
Ar19	0.074	0.027	0.054	0.081	0.108	0.130
Ar20	0.069	0.025	0.050	0.076	0.100	0.121

Table 5- Measurement of radiation dose and concentrations of radioactive isotopes for samples around radiochemistry and radiation dose laboratories

Code site	Dose Rate µSv/hr	Specific activity (Bq.kg^{-1})		
		^{232}Th(^{228}Ac)	**^{226}Ra(^{214}Bi)**	**^{40}K**
A1	0.054	15.1± 9	27 ± 5	275.2±31
A2	0.063	11.9±7	18.3±5	164.2±21
A3	0.052	21.8±2	27.2±6	247.5±32
A4	0.051	18.9±2	16.12±2.6	242.38±33
Max	0.063	21.8±2	27.2±6	275.2±31
Min	0.051	11.9±7	16.12±2.6	164.2±21
Average	0.0565	16.925	22.155	232.32

The measured absorbed dose rates(ADR) and the calculated annual effective dose equivalent(AEDE) are presented in Tables 6, 7, and 8. They include measurements of the C1, C2, and C3 laboratories and the other facilities that make up the radiochemistry laboratories.

Table 6: Annual Effective Dose Equivalent (AEDE) for laboratory C1 with different values of occupancy factor

Code	(ADR) (µSv/hr)	(AEDE) (mSv/hr) With OF 0.042	(AEDE) (mSv/hr) With OF 0.083	(AEDE) (mSv/hr) With OF 0.125	(AEDE) (mSv/hr) With OF 0.166	(AEDE) (mSv/hr) With OF 0.2
GB13	0.079	0.029	0.057	0.086	0.114	0.138
GB14	0.101	0.037	0.073	0.110	0.146	0.176
GB15	0.132	0.048	0.095	0.144	0.191	0.231
FH1	0.6	0.220	0.436	0.657	0.872	1.051
FH2	3.17	1.166	2.304	3.471	4.609	5.553
HCL1	5.1	1.876	3.708	5.584	7.416	8.935
HCL2	33.11	12.181	24.073	36.255	48.147	58.008
HCL3	6.13	2.255	4.457	6.712	8.914	10.739
AHC1	51.63	18.995	37.539	56.534	75.078	90.455
AHC2	1.5	0.551	1.090	1.642	2.181	2.628

Table 7: Annual Effective Dose Equivalent (AEDE) for laboratory C2 with different values of occupancy factor

Code	(ADR) (µSv/hr)	(AEDE) (mSv/hr) With OF 0.042	(AEDE) (mSv/hr) With OF 0.083	(AEDE) (mSv/hr) With OF 0.125	(AEDE) (mSv/hr) With OF 0.166	(AEDE) (mSv/hr) With OF 0.2
GB1	0.76	0.279	0.552	0.83	1.105	1.331
GB2	1.16	0.426	0.843	1.27	1.686	2.032
GB3	0.18	0.066	0.130	0.197	0.261	0.315
GB4	0.42	0.154	0.305	0.45	0.610	0.735
GB5	1.7	0.625	1.236	1.86	2.472	2.978
GB6	0.6	0.220	0.436	0.657	0.872	1.051
GB7	0.55	0.202	0.399	0.602	0.799	0.96
GB8	0.98	0.360	0.712	1.073	1.425	1.716
GB9	2.31	0.849	1.679	2.529	3.359	4.047
GB10	5.31	1.953	3.860	5.814	7.721	9.303
GB11	18.01	6.626	13.09	19.720	26.18	31.553
FH1	8.54	3.142	6.209	9.351	12.41	14.962
FH2	3.1	1.140	2.253	3.394	4.507	5.431

Table 8: Annual Effective Dose Equivalent (AEDE) for laboratory C3 and another laboratory with different values of occupancy factor

Code	(ADR) (µSv/hr)	(AEDE) (mSv/hr) With OF 0.042	(AEDE) (mSv/hr) With OF 0.083	(AEDE) (mSv/hr) With OF 0.125	(AEDE) (mSv/hr) With OF 0.166	(AEDE) (mSv/hr) With OF 0.2
C3FH	0.731	0.269	0.531	0.800	1.063	1.281
B5lab	0.11	0.040	0.080	0.120	0.160	0.193
B6lab	0.056	0.021	0.041	0.061	0.081	0.098
B10FH	0.122	0.045	0.089	0.134	0.177	0.214
B9room	0.087	0.032	0.063	0.095	0.127	0.152
B8room	0.208	0.077	0.151	0.228	0.302	0.364
B7room	0.076	0.028	0.055	0.083	0.111	0.133
Acoorid	0.065	0.024	0.047	0.071	0.095	0.114
C2coori	0.075	0.028	0.055	0.082	0.109	0.131
C2exit	1.12	0.412	0.814	1.226	1.629	1.962
A1room	0.075	0.028	0.055	0.082	0.109	0.131
A2room	0.068	0.025	0.049	0.074	0.099	0.119
A3room	0.072	0.026	0.052	0.079	0.105	0.126
A4room	0.069	0.025	0.050	0.076	0.100	0.121
A6room	0.082	0.030	0.060	0.090	0.119	0.144

As shown from Table 3, the radiation background limits of the open ground ranged between (0.053-0.061) µSv/h, while that of the buildings were recorded in the range of (0.045-0.056) µSv/h. Table 4 shows the radiation measurements of the average radiation dose to the area surrounding the radiochemistry laboratories. The measured dose limits are shown to be within the radiation background limits, which indicate that there is no pollution and that it is within the building limits. Table 5 shows that the annual effective dose for C1 laboratory workers recorded a significant increase: at an occupancy factor of 0.042 in AHC1 the annual effective dose was 18.995 mSv /y, a value close to 20 mSv/y. While for an occupancy factor of 0.083, both HCL2 and AHC1 exceeded the permissible exposure limits for workers, and it was impossible to work at these values. For laboratory C1, the annual effective dose was within the allowable limits for all values of the occupancy factor. Table 7 shows the radiation measurements of the annual effective dose for C2 laboratory workers which was recorded to be 19.720 mSv /y for GB11 for an occupancy factor of (0.125). This value is close to the permissible exposure limits for workers. The highest value was recorded at an occupancy factor (0.2) which was (31.55) mSv/y. The rest of the laboratory annexes recorded permissible exposure limits to work within all work values. Table 8 recorded the highest value of the annual effective dose for workers of (1.281) mSv /y for an occupancy factor of (0.2), which is a value less than the annual permissible limits for workers' exposure. Ugbede and Echeweozo[18] and Benson and Ugbede[19] calculated the annual effective dose for the public in different workplaces by measuring the effective dose in several locations and adopting exposure limits of 1 mSv/y.

This study calculated the annual effective dose of workers at a nuclear site based on an exposure limits of 20 mSv/y.

6. Conclusion

In this study, radiological measurements were carried out in the radiochemistry laboratories of the Tuwaitha Nuclear Center in Iraq for radiological characterization, calculation of radiation dose for workers and assessment of risks resulting from radiation exposure. Samples were taken from the site, analyzed, and a portable survey meter was used to determine external dose rates. The site for the radiation background measurement was chosen based on areas similar to the work environment. The area around the laboratories was measured and found that the place was free of any excess radioactivity. The results show different doses in the radiochemistry laboratories, ranging from low to high annual effective doses that exceeds the permissible limits. As in each of the laboratories, C1 and C2, based on these results, the radiation safety requirements for the workers must be applied to determine the doses and working hours that achieve the highest levels of safety.

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Conflict of interest

The authors declare that no conflict of interest exists.

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