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ORIGINAL ARTICLE

Radiological characterization of the irt-5000(14-Tammuz) research nuclear reactor at Al-Tuwaitha nuclear center in Iraq

Iman Tarik Al-Alawy[1] · Osama Abdulameer Mzher[2]

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Abstract

Measurements were made at the destroyed nuclear reactor IRT-5000 (14-Tammuz research nuclear reactor at AL-Tuwaitha Nuclear Center in Iraq) to provide a basic comprehensive radiological characterization and to assess risk and dose for the workers in the workplace. Samples were collected from the site and analyzed, as well as using a portable survey meter to determine the external exposure dose rates. The quantity and quality of radionuclides were determined using gamma spectrometry techniques. The dose rate measured within the reactor core body ranged from 55 to 1250 mSv/h. The maximum dose was recorded in the middle of the corner near the horizontal experimental channel number seven, with activity concentration of 19.97 GBq estimated from Co-60 isotope. Most samples were contaminated with Cs-137, Co-60, and Eu-152 isotopes. The highest activity concentration of Cs-137 is 14772.41 ± 99.91 Bq/L and Co-60 is 7642.22 ± 40.02 Bq/kg, were found in slag from reactor tank. Two scenarios were developed based on the water level of the reactor tank. Assuming there are three locations for workers on the reactor surface. The annual dose of workers on the surface of the reactor (when the reactor tank is empty) ranges from 116 to 153 mSv, which is higher than the annual dose limit for workers. Therefore, workers will be subjected to the principle of As Low As Reasonable Achievable (ALARA) during all phases of dismantling nuclear reactor IRT-5000 (14-Tammuz) as recommended by International Atomic Energy Agency (IAEA).

Keywords Radiological characterization · Risk assessment · Dose rate · IRT-5000 reactor

Introduction

Workers at the decommissioning of nuclear facilities are exposed to many radiation and non-radiation hazards because these activities are accompanied by great exposure to radiation and occupational hazards associated with work. Hence, the exposure dose rate of the workers has to be analyzed and assessed under the principle of As Low As Reasonably Achievable (ALARA) (Jeong et al. 2014). Therefore, a complete radiological characterization of the nuclear facilities is required.

The radiological characterization of nuclear reactor facilities is to provide a credible database on the quantity, type, location, and physical and chemical status of radionuclides, including activation or contamination products generated by neutron activation of materials during the reactor operation. The characterization includes a survey of existing data, calculations, measurements on-site, and/or sampling and analyses. Radiological characterization is one of the recommended requirements in IAEA safety standards WS-R-5 (IAEA 2006) to prepare a plan for the dismantling of reactor facilities (Nonova et al. 2014; IAEA 1997).

The IRT-5000 is one of nuclear research facilities at Al-Tuwaitha site in Iraq. The research reactor IRT-5000 can be considered as a special case because this facility was very destructive as many parts of it were lost and stolen as well as the loss of all records in the operating period. Figure 1 describes the building layouts in Al-Tuwaitha and the associated sectors names (Chesser et al. 2009).

This research describes the radiological characterization of the IRT-5000 research reactor in Iraq and assesses the exposure dose to workers during decommissioning of

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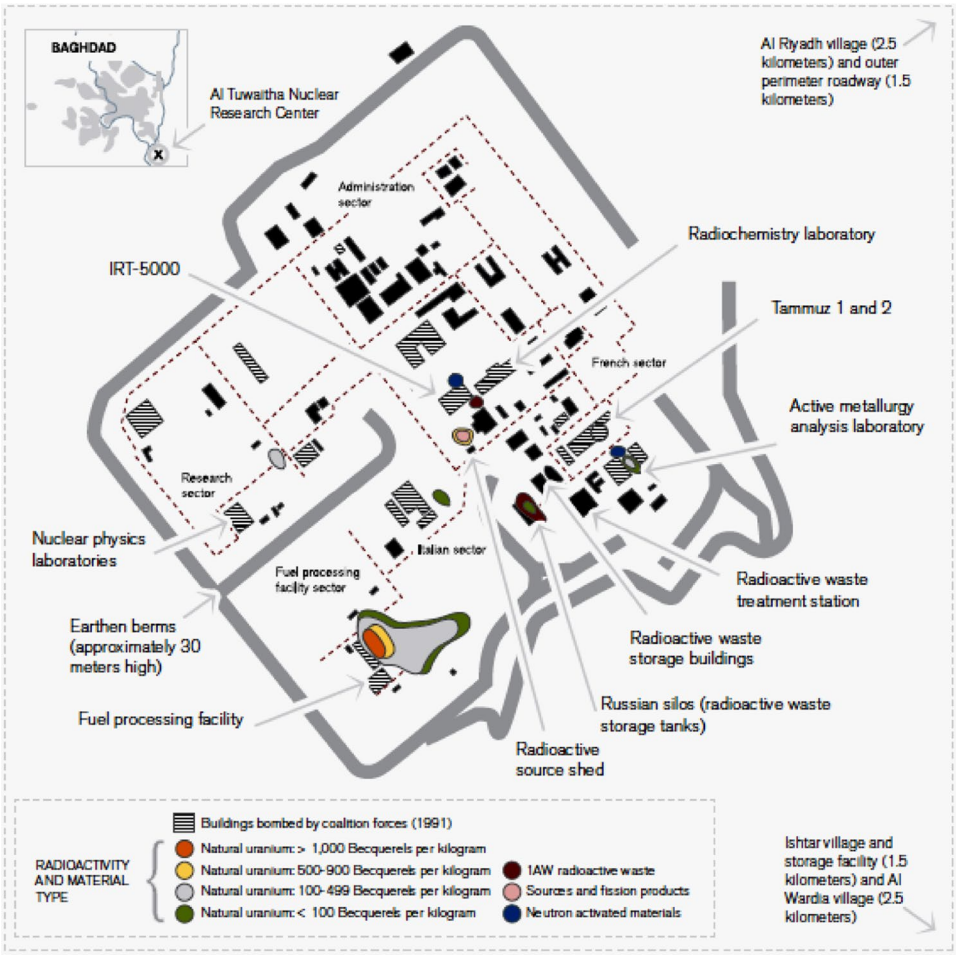

Fig. 1 Al-Tuwaitha nuclear site facilities map in Iraq (Chesser et al. 2009)

the reactor. The dose rate was measured directly in the reactor core body and inside the reactor tank, as well as analysis of samples taken from different parts of the reactor (reactor reservoir, first cooling cycle, ion filters, etc.) (Hulubei 2012; Nonova et al. 2014).

The RESRAD-BUILD code (computer code intended to assess the radiation dosages from RESidual RADioactivity in BUILDings) was used to assess the dose and examined the workers' hazards working on the surface of the reactor. The RESRAD-BUILD computer code is a pathway analysis model putting to evaluate the possible radiological dose to persons who work or live in a building contaminated with radioactive material. In RESRAD-BUILD computer code various scenarios are put for the possibility of exposure of individuals inside or outside a contaminated building. It gives importance to a description of building, air exchange as well as type, condition, and location of the source of contamination. RESRAD-BUILD computer code is also considered for exposure pathways whereas it analyzes inevitably the potential dose and shows results in both text and graphic reports (Yu et al. 2003).

IRT-5000 (14-Tammuz) reactor

The nuclear research reactor IRT-5000 (14-Tammuz) is located at Al-Tuwaitha nuclear center. Al-Tuwaitha nuclear center, which is heavily fortified, is located 18 km south of Baghdad, about 1 km east of the Tigris River, with an area of 3000 m^2. Al-Tuwaitha covers many facilities including research reactors, a fuel manufacturing facility, plutonium separation facilities, uranium enrichment, and other purposes. The IRT-5000 research reactor and other facilities were damaged and destroyed during the Gulf War (Cochran et al. 2006). Figure 2 shows a satellite view of Al- Tuwaitha nuclear research center (Google Map 2018).The GPS Coordinates are (North = 33°12′16.9″, East = 44°30′54.1″).

The IRT-5000 research reactor was built by the former Soviet Union in 1967 and started operation at full power (2 MW) in 1968 with a maximum thermal neutron flux 2.5×10^{13} n/cm^2 sec. It was built with a similar design to the Soviet IRT-2000 research reactor. Both of them are pool-type reactors using water as a moderator,

Fig. 2 Satellite view of Al-Tuwaitha Nuclear Research Center (Google Map 2018)

reflector, top shielding, and coolant. It was used pin-type fuel assemblies, the fuel enriched with 10–36% U-235.

The pool of the reactor IRT-5000 is an aluminum tank with a thickness of 6 mm surrounded by 1.8 m thick biological shielding from concrete which has different densities in different parts of the structure. The total volume of the pool is about 60 m^3 with height 7.6 m, width 1.9 m, and length 4.5 m. It has ten horizontal experimental channels, eight of which have the diameter of 100 mm and two channels of 150 mm diameter. The first channel was called a thermal channel containing graphite to moderate the fast neutrons (Abramidze et al. 2000; IAEA 1999).

In 1978, the 2 MW reactor was upgraded to 5 MW using new fuel assemblies' sandwich-type with 80% enriched U-235. As well as changed the aluminum tank and the installed equipment tank (a body of the reactor core, initial parts of experimental channels, ejection pump, hold-up tank, etc.) to a new one made of stainless steel. The cooling system has been developed by building new circular pumps to increase the power, and the aluminum heat exchangers have also changed to a new one with increased heat exchanger surfaces.

In 1991 during the Gulf War, the IRT-5000 reactor facilities were destroyed and the nuclear reactor was shut down. In 1998 the residual parts of spent fuel assemblies (five pieces) were removed and transported to the reprocessing facility by the USA (Cochran et al. 2006).

Materials and methods

The radiological characterization of nuclear facilities involves a survey of radiation measurements in situ, sampling, and analyses. The radiological characterization of the nuclear facilities has been carried out in accordance with the IAEA safety standards and recommendations (IAEA 1998, 2007). Different methods and devices have been used to obtain high accuracy and confidence for dose rate results, as follows (Holden et al. 2004; Nuccetelli et al. 2017):

Dose rate measurement

The dose rate at the site, inside the reactor tank, was measured using a portable survey meter (MIP 10, Canberra) with a waterproof detector (STTC wide range gamma probe, Canberra) which was designed to measure the equivalent dose rate of gamma. The STTC has a compact and robust metal case including the Geiger Mueller detector, high-voltage power supply, and the pulse-shaping circuits. This probe extends the instrument measurement range and supports remote measurements up to 20 m (65 ft) to detect X-ray and gamma radiation in the range of 58 keV–1.5 MeV. The response of this detector is to

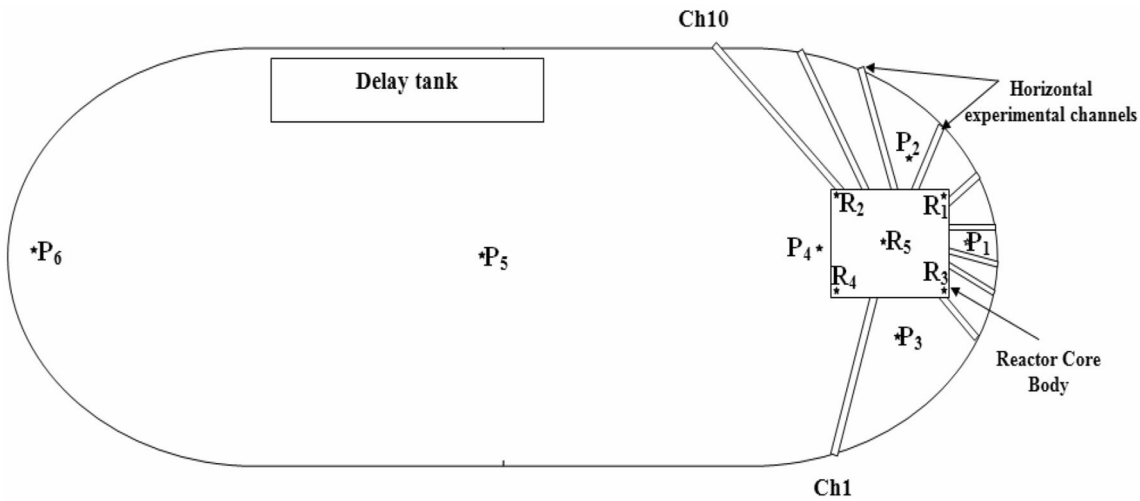

Fig. 3 Reactor tank with P_x site and reactor body with R_x site of dose rate measurement inside IRT-5000 reactor core body

measure the dose rate from 0.1 µSv/h to 10 Sv/h (Canberra 2016, 2017a). This detector has been calibrated using standard source Co-60. The dose rate was measured at six selected points (Px) inside the reactor tank and five points inside the reactor core body (Rx), at different altitudes as shown in Fig. 3.

Sampling and analyses

The nuclear facility consists of many different parts that can be sampled (such as, water and slag from reactor tank, debris, concrete block using for shielding and soil samples taken from different depths near the reactor's biological shield) and another parts that cannot be sampled such as pipes, reactor tank plates, etc. Samples were prepared and analyzed in the Radioactive Waste Treatment and Management Directorate (RWTMD) laboratory, as follows:

a. Sample preparation

The solid samples (soil, slag, debris, concrete) were prepared by placing each sample in the oven to dry at 100 °C for 4–6 h, depending on the moisture content in the sample to reach a constant weight to ensure complete removal of any residual moisture. The dried samples were grinded into a fine powder and passed through a standard 300 µm sieve (ASTM No.18) (Lloyd et al. 2009). The homogenized samples were filled in 550 ml Marinelli beaker and then sealed hermetically with the aid of PVC tape. While liquid samples were filled in 550 ml Marinelli beakers after removing the impurities using a 0.5 mm filter paper.

b. Analyzing samples

Sampling analysis was achieved using a high-performance digital signal processor gamma-ray spectrometry system (DSPEC LF ORTEC) with low background configuration to measure radionuclides activity concentrations in samples. The gamma-ray spectrometry system consists of a P-type high-purity germanium (HPGe) detector with dimensions of 65 mm diameter × 50 mm length with passive shielding (ORTEC 2014). The photo-peak efficiency and energy calibrations of the gamma spectroscopy system were performed using a standard mixed source, 550 ml Marinelli beaker, containing Am-241, Cd-109, Co-75, Co-60, Sn-113, Hg-203, Y-88, and Cs-137. The counting geometry of the samples and standard source used for efficiency calibration was kept constant.

c. Gamma spectroscopy measurement

Portable high-purity germanium detector (Micro-Detective-HX ORTEC) was used to detect and identify the radioactive isotopes inside the reactor tank, cooling cycle pipes, and horizontal experimental channels at the situ. Micro-Detective-HX contains P-type high-purity germanium (Crystal Dimensions 50 mm diameter × 30 mm length), cooling system with low-power sterling cooler, and neutron detector (ORTEC 2014). The In Situ Object Counting System (ISOCS) calibration software was used to assay gamma spectra at the site. The ISOCS program has the capability to verify the gamma spectrum by eliminating the need for traditional calibration sources to perform the efficiency calibration process by merging the detector characterization produced by Monte Carlo N-Particle (MCNP) model, mathematical geometry templates, and a few physical parameters (Venkataraman et al. 1999; Canberra 2017b).

Results and discussion

For creating the radiological characterization, more than 150 measuring point doses were taken at different locations and about 50 samples of solid and liquid materials were collected

Table 1 Dose rate measurement inside the reactor tank, in μSv/h

Depth (m)	P_1	P_2	P_3	P_4	P_5	P_6
0	1.7	1.35	1.58	1.2	1.16	0.7
1	3	2.8	2.06	1.52	1.3	0.85
2	5.9	4.5	4.2	1.94	1.95	1.15
3	8.5	6	4.86	4.8	2.9	1.3
4	20	12.1	7.5	5.1	3.25	1
5	23	28	8.2	27.4	5.9	1.5
6	1120	2220	1200	606	150	0.5
7	106×10^3	600×10^3	95×10^3	27×10^3	113	3.8
Bottom	12×10^3	318×10^3	18×10^3			

Table 2 Dose rate measurement inside the reactor core body, in mSv/h

Location	R_1	R_2	R_3	R_4	R_5
Above	99	65	55	55	75
Middle	1250	650	520	590	750
Bottom	115	98	92	91	100

Table 3 Isotopes founded at different locations using high-purity germanium detector

Location	Isotopes			
Reactor tank	Cs-137	Co-60	Eu-152	
Experimental horizontal channel 7		Co-60		
Primary cooling system pipe	Cs-137	Co-60	Eu-152	Eu-154
Reactor pipe	Cs-137	Co-60	Eu-152	Eu-154

from the reactor systems, as well as about 20 measurements at the gamma spectroscopy situ. The results of this work are the preliminary radiological characterization of the IRT-5000 research reactor. The results of characterization can give an initial perception of using the necessary equipment and specify the type of the implementation technique in the dismantling, decontamination, and the choice of appropriate methods to assure the radiation protection, safety assessment and risk assessment of the employees carrying out the dismantling and the environment. The characterizations at Al-Tuwaitha are to cleanup radioactively contaminated facilities and safely dispose of the radioactive wastes. Part of the characterization task, provided by this work, will involve Al-Tuwaitha IRT-5000 research reactor uncharacterized radioactive wastes. Results of the dose rate measurement inside the reactor tank and reactor core body are tabulated in Tables 1 and 2, respectively.

The depth of reactor tank is about 7.5 m at points P_1–P_3 which are about 0.5 m deeper than other points. Table 1 shows the measurement of dose rate within the reactor tank at different points with different depths. It can be seen that the highest value of the dose rate is at 6–7.3 m depth. This is the depth at which the reactor core body is located. The radiation dose in this depth, especially in points P_1–P_4 coming from the reactor core body. The difference in values is due to the radioactivity of the reactor core body which varies from point to point because the neutron activation of the reactor's core body is not equal at all points due to the way

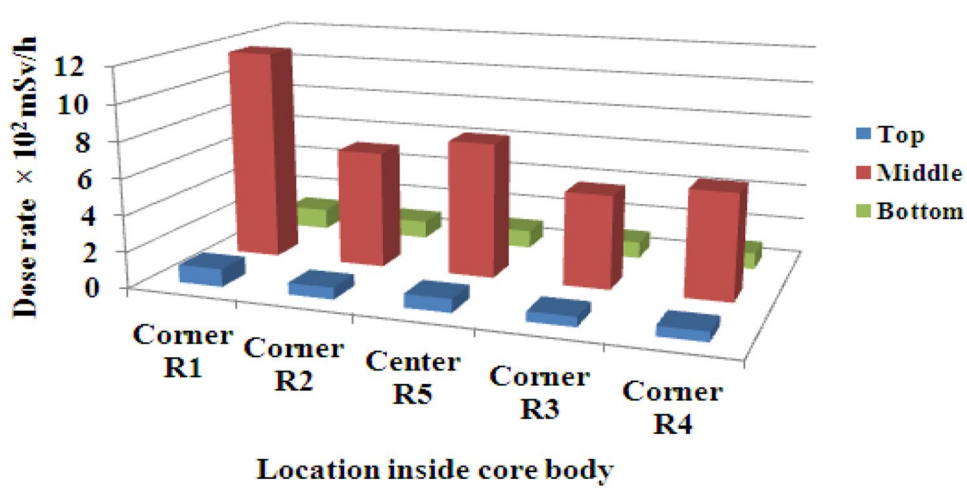

Fig. 4 Dose rate measurement inside reactor core body

Table 4 Activity concentrations at different locations in (Bq/cm^2)

Location	Cs-137	Co-60	Eu-152	Eu-154
Reactor pipe	3534.37 ± 17.98	56.71 ± 3.58	527.28 ± 26.54	36.03 ± 15.17
Primary cooling system pipe	4533.09 ± 23.42	271.60 ± 6.83	318.78 ± 40.45	
Heat exchanger	9547.03 ± 160.22	357.84 ± 24.43	679.90 ± 35.38	74.04 ± 10.13

Table 5 Activity concentrations of liquid samples in (Bq/L)

Sample	Cs-137	Bi-207	Na-22
Water from reactor tank	20.07 ± 0.93		5.02 ± 1.21
Liquid from ion filters	14772.41 ± 99.91	55.07 ± 9.87	

the reactor and the fuel locations operate during the period of operation. Table 2 clearly shows the measured dose rate inside the reactor core body. The highest value recorded at point R_1 in the middle of the inner surface of the reactor core body is 1250 mSv/h. Point P_2 is close to the point R_1 at a depth of 7 m (0.3 m from the bottom) outside the reactor core body with dose rate 600 mSv/h. Figure 4 shows the highest dose rate within the reactor body in the middle of the corner R_1.

The main radionuclide at the reactor core and pipelines is Co-60, which is responsible for the dose rate that prevailed in the first 50 years. As well as Eu-154 and Eu-155 could contribute less than 5% of the dose rate (Hulubei 2012; IAEA 2007; Mikhalevich 2000).

Isotopes that were found at different locations of the reactor using portable high-purity germanium detector are shown in Table 3.

The activity concentration can be estimated, at the reactor core body, from dose rate as (Cember and Johnson 2009):

$$A = \frac{D \times d^2}{\Gamma}, \quad (1)$$

where A is the activity concentration in MBq, D is the dose rate in Sv/h, d is the distance from the source 0.05 m, and Γ is a constant 3.13×10^{-7} Sv.m^2/MBq.h for Co-60. Hence, the calculated activity concentration of Co-60 in the reactor body at highest dose rate is 19.584 GBq (0.54 Ci).

The data obtained from the gamma spectrometric analyses of material samples are given in Tables 4, 5 and 6.

The concentration of Cs-137 was found in all liquid and solid samples. The highest concentration of 14772.41 ± 99.91 Bq/L was found in liquid samples of ionic filters. Also, the highest concentration of solid slag (residue) samples was found in the reactor tank. These results can be seen in Tables 4, 5 and 6. Consequently, all reactor equipments are contaminated with Cs-137 relatively with high concentrations, especially the pipes and equipment of the reactor's first cooling cycle. Co-60 was also found in some samples with a high concentration of slag samples from the reactor tank up to 7642.22 ± 40.02 Bq/kg and also different concentrations were found in the primary cooling cycle (pipes and equipment). The reason for the high concentration of Co-60 in the slag samples in the reactor tank is due to rust and corrosion in the core body of the reactor. Other isotopes such as Eu-152 and Co-57, with the highest concentration of Eu-152 were found in the primary cooling cycle equipment up to 679.90 ± 35.38 Bq/cm^2. The presence of Na-22 in the water samples from the reactor tank and the presence of the Bi-207 in the liquid samples of the ion filters were also observed. The reactor has been shut down since 1991 and it is long enough time to decay many radioactive isotopes from the reactor's operation. In this case, the results of the sample analysis showed relatively long half-life isotopes, such as CS-137, Co-60, EU-154, and EU-152.

Dose and risk assessment

Most of the radiation dose in the reactor tank comes from reactor core body, where the risk of workers on the reactor surface increases when they are starting in dismantling the

Table 6 Activity concentrations of solid samples in (Bq/kg)

Sample	Cs-137	Co-60	Co-57	Eu-152
Lead	3.01 ± 0.25			
	2.60 ± 0.34			
Slag (residue)	3596.64 ± 20.11	7642.22 ± 40.02	32.51 ± 4.07	88.41 ± 12.03
Debris	5.21 ± 1.02			
	2.60 ± 0.90	1.81 ± 0.30		
Concrete shielding block	1.62 ± 0.31	1.24 ± 0.51		0.80 ± 0.02

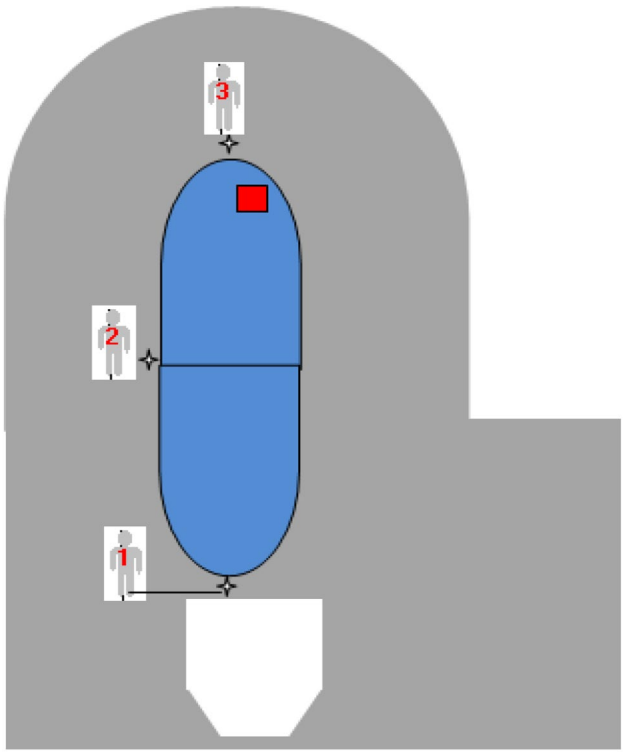

Fig. 5 The reactor surface and locations of workers

Table 7 Workers dose received for the exposure duration

Dose received/ Location of workers	mSv/day	mSv/week	mSv/month	mSv/year
First scenario (the reactor tank is filled within 1 m height of water)				
1	4.5×10^{-16}	2.2×10^{-15}	9×10^{-15}	1×10^{-13}
2	2.5×10^{-9}	1.2×10^{-8}	5×10^{-8}	5.7×10^{-7}
3	3.2×10^{-14}	1.6×10^{-13}	6.4×10^{-13}	7.3×10^{-12}
Second scenario (the reactor tank is empty)				
1	0.5	2	10	116
2	0.63	3.1	12.6	145
3	0.66	3.3	13.2	153

reactor. Hence, the RESRAD-BUILD computer code was used to assess the prospective dose and risk of workers on the reactor surface. Two scenarios have been created. The first is when the reactor tank contains 1.5 m height of water and the second scenario is when the reactor tank is drying up. Three locations for workers could be found on the surface of the reactor were also assumed. Figure 5 shows the suggested locations for workers on the reactor surface. The working time at the site was calculated at 5 h per day per worker and 240 days per year, and was entered within the data of the RESRAD-BUILD computer code. The results are tabulated in Table 7. Table 7 shows that in the first scenario, the annual dose of workers is well below the annual dose limit (50 mSv/y) (UNSCEAR 2000), while the annual dose was significantly higher in the second scenario. Figures 6 and 7 (RESRAD-BUILD output graphs) show that most of the workers' risk is due to the external dose. Figure 7 shows the risk comparison between the two scenarios, where the risk level is higher in the second scenario. As for the contaminants of radon gas, the levels of this gas are not calculated because of the absence of contaminants emitting radon and also because the site is not closed, therefore, a large exchange of air with the atmosphere occurs.

Conclusions

This study was conducted to provide basic data for a comprehensive radiological characterization by identifying the type, location, and levels of radioactive contamination at the IRT-5000 research nuclear reactor at AL-Tuwaitha Nuclear Center in Iraq. The results showed that the highest dose rate measured within the core body of the reactor was 1.25 Sv/h in the middle of the corner near the horizontal experimental channel 7. The spectrum analysis of the reactor core through experimental channel 7 showed that an estimated isotope is Co-60 with activity concentration 19.5 GBq or 0.54 Ci at high dose rate. The spectral analysis of gamma-ray spectroscopy for various samples taken from several sites showed that the presence of Cs-137, Co-60, and Eu-152 isotopes are the most common because of the longevity of their half-life's and the period length of the reactor shutdown since 1991. It has been concluded that the highest activity concentration for Cs-137 is 14772.41 ± 99.91 Bq/L found in ionization filters and 3596.64 ± 20.11 Bq/kg in slag taken from reactor tank. The highest activity concentration of Co-60 is 7642.22 ± 40.02 Bq/kg measured in slag from reactor tank. The annual dose of the workers on the surface of the reactor (when the reactor tank is empty) ranges from 116 to 153 mSv, which is higher than the annual dose limit for workers. Therefore, the principle of ALARA should be applied by reducing the period of work and increasing the number of workers in these conditions. In general, samples from the reactor's first cooling cycle show that all parts of the cycle are contaminated at different concentrations of isotopes, with the highest concentrations of Cs-137, Co-60, and Eu-152 being 9547.03 ± 160.22, 357.84 ± 24.43, and 679.90 ± 35.38 Bq/cm^2, respectively, in the heat exchanger.

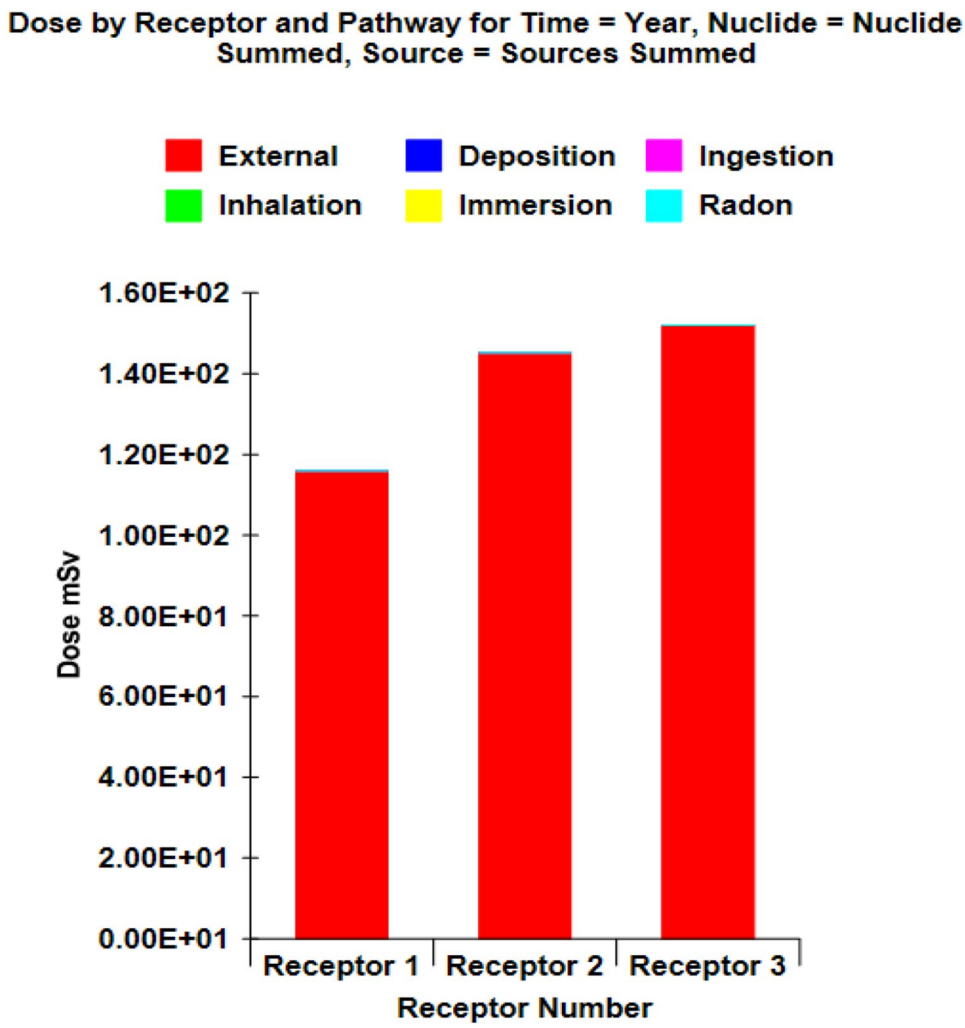

Fig. 6 RESRAD-BUILD output graph, annual dose by receptors, when reactor tank is empty

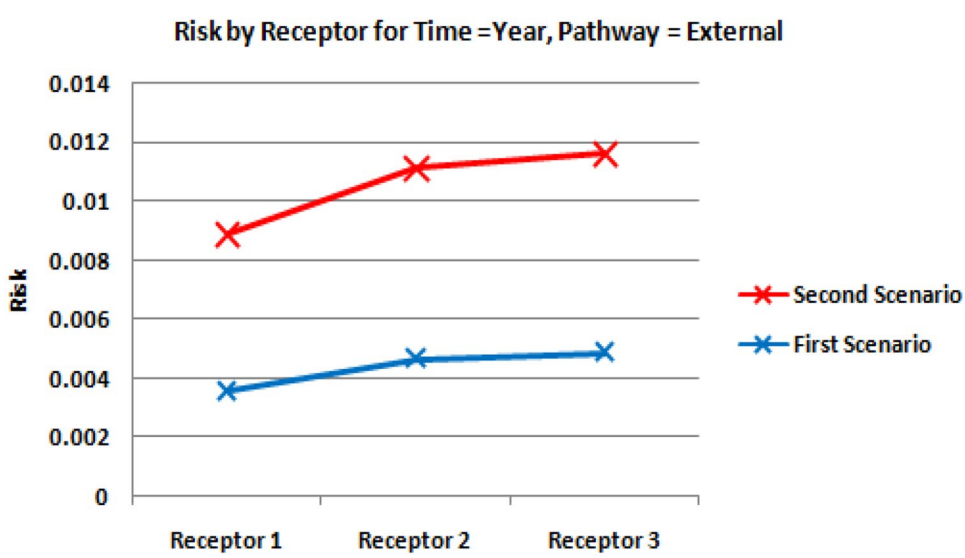

Fig. 7 RESRAD-BUILD output graph, annual risk by external dose

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