Radioactive Safety Assessment for Surface Contamination by using SAFRAN Tool

Mezher Abed Gatea[*], Zaidoon Hafiz Ibrahim[**], Saad Jundee Kadhum[*]
[*]Iraqi Decommissioning Directorate, Ministry of Science and Technology
[**]Central Laboratories Directorate, Ministry of Science and Technology
Corresponding author: mizher.abed@gmail.com

Abstract

This research presents the deterministic safety assessment for work activities carried out to remediate the surface radioactive contamination which found during the decommissioning of Radioisotopes Production Laboratory (RPL) at Al-tuwiatha nuclear site, to demonstrate that the dose acceptance criteria and the safety goals are met with a high degree of confidence. Work activities comprised characterizing, removal, packaging and relocating of the generated waste into specified zone. The physical status of the affected area is soil and debris and radiologically contaminated with (Cs-137, Eu-152, Co-60) and small amounts of natural Uranium. Safety assessment calculations have done by using SAFRAN (Safety Assessment Framework) version 2.3.2.7 software. The radiation exposure for workers in the affected area is considered as an endpoint to be compared to the worker dose limit. Dose to the public is considered to be negligible and is not numerically assessed in the SAFRAN file due to that RPL is located in a restricted zone far away from the public, low level radioactivity for the affected area and 30m berm surrounded Al-tuwiatha site. Assessment for accident conditions, were also considered to be negligible because no accident occurs in all activities of work. Safety assessment calculations based on maximum external dose rate (2.233μSv/h) and maximum air contamination (0.001Bq/m^3). Safety assessment results proved that the sum of external and internal doses to the workers for all work activities were 1.6mSv/y is less than 10% of the 20mSv/y dose limit. Hence, there are no activities that have been assessed to present a risk rating higher than low and the radiological risks remain below the relevant prescribed dose limits through implementing effective safety programs into remediation process of the surface contamination.

Key Words: Surface Contamination, SAFRAN Tool, Safety Assessment, Al-tuwiatha site.

تقييم السلامة لتلوث اشعاعي سطحي بأستخدام البرنامج الحاسوبي SAFRAN

مزهر عبد كاطع*، زيدون حافظ ابراهيم**، سعد جندي كاظم*

*مديرية تصفية المنشآت والمواقع النووية المدمرة/وزارة العلوم والتكنولوجيا

**مديرية المختبرات المركزية/وزارة العلوم والتكنولوجيا

الخلاصة

يقدم هذا البحث تقدير السلامة الاشعاعية لفعاليات العمل التي جرت لمعالجة التلوث الاشعاعي السطحي والذي وجد خلال تصفية مختبر انتاج النظائر المشعة (RPL) الواقع في موقع التويثة النووي، لأظهار تحقق معيار الجرعة المقبولة واهداف السلامة وبدرجة عالية من الثقة. اشتملت فعاليات العمل على توصيف ورفع ورزم ونقل النفاية الناتجة الى المكان المخصص. ان الحالة الفيزياوية للمنطقة المتأثرة بالتلوث هي تربة وانقاض ملوثة اشعاعيا بنظائر (Cs-137 و Eu-152 وCo-60) وكميات قليلة من اليورانيوم الطبيعي. تمت حسابات تقدير السلامة بأستخدام البرنامج الحاسوبي (SAFRAN) الاصدار 2.3.2.7. اعتبر التعرض الاشعاعي للعاملين في المنطقة المتأثرة كنقطة معايرة نهائية للمقارنة مع الحدود المقبولة للجرعة الاشعاعية للعاملين. تم اهمال الجرعة الاشعاعية لفرد الجمهور ولم يتم تقديرها عدديا في ملف برنامج الـ (SAFRAN) بسبب وقوع مختبر انتاج النضائر المشعة في المنطقة المقيدة والنشاط الاشعاعي الواطىء للمنطقة الملوثة ووجود حاجز ترابي يحيط موقع التويثة وبأرتفاع 30م. ايضا تم اهمال تقدير السلامة للحوادث بسبب عدم حدوث اي حادث عرضي في كافة فعاليات العمل. اعتمدت حسابات تقدير السلامة على اعلى معدل جرعة اشعاعية خارجية (2.233μSv/h) واعلى تركيز للملوثات خلال العمل ($0.001Bq/m^3$). اثبتت نتائج تقدير السلامة أن مجموع التعرض الاشعاعي الخارجي والداخلي للعاملين ولكافة فعاليات العمل كان (1.6mSv/y) وهو اقل من 10% من الجرعة الاشعاعية المقبولة للعاملين (20mSv/y). لذا فأن المخاطر الاشعاعية تكون واطئة، وتحت مستوى حدود الجرعة المقبولة والمتعلقة بالعاملين في حقل الاشعاع، وهذا ناتج من خلال تطبيق برامج سلامة فعالة في عملية المعالجة لمنطقة التلوث السطحي.

الكلمات المفتاحية: التلوث السطحي، اداة الـ (SAFRAN)، تقييم السلامة، موقع التويثة.

Introduction

There are a number of sites in Iraq which have been used for nuclear activities and which contain potentially significant amounts of radioactive materials [1]. Many of these sites suffered substantial physical damage during the Gulf War in 1991. Secret operations at Al-tuwaitha site, combined with the bombing of nuclear facilities and the subsequent looting by local residents, have contributed to the perception that the site and nearby residents suffer widespread radioactive contamination [2].

A general requirement in decommissioning is the development of a decommissioning plan which includes, or has associated with it, an evaluation of the potential radiological consequences to the public and workers during planned decommissioning activities and as a result of any credible accidents that might occur during these activities [3]. The primary purpose of the safety assessment is to identify hazards during normal and potential accident conditions, and then to identify engineered and administrative control measures to mitigate the hazards and their consequences [4]. As a part of this process, it should be demonstrated that risks have been reduced to meet As Low As Reasonably Acheivable (ALARA) principle [5] and to within nationally prescribed safety criteria.

The SAFRAN tool allows the user to visibly, systematically and logically address pre-disposal radioactive waste management and decommissioning challenges in a structured way. It also records the decisions taken in such a way that it constitutes a justifiable safety assessment of the proposed management solutions.

A safety assessment is a systematic process to verify that applicable safety requirements are met in all decommissioning works. Safety analysis is a key component of a safety assessment. It incorporates both probabilistic and deterministic approaches, which complement each other [6].

Probabilistic safety analysis was attached with the RPL decommissioning plan to demonstrate that the safety goals are met for work and potential accidents within the decommissioning activities. It identifies vulnerabilities not necessarily accessible through deterministic safety analysis alone.

The deterministic safety analysis is used here to verify that the dose acceptance criteria and safety goals are met with a high degree of confidence for all works. The Safety Assessment Framework (SAFRAN) software tool was implemented for safety analysis [7]. It developed to apply the methodology developed within the Safety Assessment Driving Radioactive Waste Management Solutions (SADRWMS) project. The International Atomic Energy Agency (IAEA) organized the International Project on SADRWMS to examine international approaches to safety assessment for predisposal management of all types of radioactive waste, including disused sources, small volumes of waste, legacy and decommissioning waste, operational waste, and large volume naturally occurring radioactive material residues. The initial outcome of the SADRWMS Project was achieved through the development of a series of flowcharts which were intended to improve the mechanisms for application of safety assessment methodologies for predisposal management of radioactive materials [8].

The evaluation of all decommissioning works and the preliminary safety assessment has been undertaken with the best available data and applying a mixture of quantitative and qualitative approaches based on site characterization.

Materials and Methods

The Instruments and equipment used are:-

(1) Ludlum (type 2241) [figure (1-a)] used in field measurement with two probes. The first was Geiger-Muller (GM) detector (type 44-9) used for detecting surface contamination by count per second (cps) unites. And the second was (NaI) detector (type 44-10) used for measuring dose rate by micro-Sievert per hour (µSv/h) unites.

(2) Interceptor identifier (thermo scientific type) used in field measurement to identify radioisotopes [figure (1-b)].

(3) Radeye-sx [figure (1-c)] with100cm^2 scintillation probe model DP6BD for measuring (α β γ) contamination. A zinc sulfide (ZnS(Ag)) scintillator is used for detecting alpha particles, and a thin plastic scintillator is employed for detecting beta particles with gamma sensitivity approximately 15-20 cpm/µR/hr for Cs-137.

(4) RADeCO instrument [figure (1-d)] model H-809VII with cellulose filter paper type 0750-029 used for air sampling.

(5) Ludlum (type 3030) Alpha Beta radiation sample counter [figure (1-e)]. It has radiation detector ZnS(Ag) adhered to plastic scintillation material with 0.4 mg/cm^2 aluminized window.

(6) Vacuum cleaner [figure (1-f)] to suck up and containerize the generated dust and aerosols.

(7) Gamma spectrometer [figure (1-g)] with semiconductor detector of high purity germanium used for laboratory analyzing of homogeneous contaminated materials (debris and soil) samples.

(8) Barrels of 200 liter in volume, made of carbon steel, painted with brightly colored (yellow) and have closed sealed, used to containerize the homogeneous contaminated materials (debris and soil).

(9) Freight container with dimensions (6×2.5×2.5m) with closed sealed, used as accumulation zone for containers containing radioactive waste [figure (1-h)].

a-Ludlum2241 b-Interceptor c-Radeye-sx

d-RADeCO e-Ludlum3030 f- Vacuum cleaner

g-Gamma meter h-freight container

Figure (1) the used instruments

The research has done in Radioisotopes Production Laboratory (RPL) which is located at Al-tuwaitha nuclear site. It used to produce the radioisotopes kits for medical and industrial uses after being irradiated in IRT-5000 reactor [9]. The decission was taken to decommission the facility as a part of Iraqi Decommissioining Project and the work began in 2010 and finshed in 2014. The surface contamination is found during decommissioning of RPL. The affected area accupied 600m^2 and laid out in five hot spots (figure 2) HS3, HS4, HS5, HS6 and system 6100. The contaminated materials in the affected area are soil and debris [10].

HS3 HS4 and HS5

HS6 system6100

Figure (2) the affected area

Passive safety and defense in depth ensure that radiation protection is optimized and doses are kept within appropriate limits. With respect to the need for radiation protection during work activities, certain points considered and tabulated in table (1) below.

Table (1) indicates the engineered safety features

Safety feature	**Safety function**	**Passive/active**
Site fencing and gates	To separate the facility from normal access areas, providing physical access barriers	Passive
Radiation protection detectors	To detect and alarm in case of high dose rates within the work area.	Active
Ventilation system	To suck and mitigate the aerosols with activities	Active
caravanes, access control	To control radioactivity movement	Active
Work areas classification	To control radiological risk and segregate the scattered material in affected area	Active
Security system	To prevent unauthorized access to the site (non-radiological function).	Passive and active
Decontamination equipment	To minimize generated waste and time of handling of radioactive materials	Active
Respiratory protection	To control the concentrations of radioactive material in the air	Passive

Regulatory limitations which implemented here are (20mSv/y) maximum dose to the workers from all pathways; (0.4Bq/cm^2) clearance levels for surface contamination of radioisotopes have β and γ emitters, (0.04Bq/cm^2) clearance levels for surface contamination of radioisotopes have α emitters, (0.1Bq/g) clearance levels for radioisotopes (^{137}Cs, ^{152}Eu and ^{60}Co) in bulk materials and (1Bq/g) clearance levels for radioisotopes has natural origin [11-13].

The assessments covered work took place over 2.5 years period. The activities characterization, removal, packaging and relocating were practiced to manage radioactive waste according to calculated work time for each activity 180h/y, 300h/y, 160h/y and 80h/y respectively. Work time was 3h/day, 5day/week, 4week/month and 12month/year.

A realistic approach is taken in respect of data used in the assessment, with real measured values taken from characterization results. The assessment based on the maximum external dose (2.233µSv/h) to the worker and maximum air contamination (0.001Bq/m^3) which came from the arising dust in work area. Calculations of air contamination have done by using air sampler device (RADeCO) type (H-809VII) with cellulose paper filter type (0750-029) then, filter measured in Ludlum type (3030) Alpha Beta radiation sample counter in unit (dpm). The radiation measurement to this filter refers to the amount of radiological activity in volume by comparison with radiation background. Then, the following equation was used to transfer (dpm) units into (Bq/m^3) units [14].

$$\text{Beta (pCi/cm}^3) = B / (2.22xDxV)$$

Where:-

pCi (picocurie) = 0.037 Bq (Becquerel); B= net beta count rate; D= beta efficiency factor (cpm/dpm); V= volume of sample; 2.22= conversion factor from dpm/pCi

One type of endpoints is considered in exposure assessment of normal operation scenarios. It refers to workers in the affected area. In this assessment the worker endpoint is defined as a cumulative endpoint in SAFRAN. The worst case is a generic worker who charged with different activities. The annual dose for this worker is then calculated as the sum of all exposures for all the mentioned activities. Dose to the public is considered to be negligible and is not numerically assessed in the SAFRAN file due to that RPL is located in a restricted zone far away from the public, low level radioactivity for the affected area and 30m berm surrounded Al-tuwiatha site. Assessment for accident conditions, were also considered to be negligible because no accident occurs in all activities of work.

Workers under control of personal radiation protection segregated and picked up radioactive from non-radioactive waste (soil/debris) [15-17] and containerized it in the proper container carefully [18]. The engineered safety feature, in this manner, had done by establishing a boundary around the suspected contaminated area. The boundary should be as small as possible, but large enough to allow workers and equipment to access the area and to allow work to be accomplished with the safe manner. Sufficient ground cover should be placed below suspected items (soil/debris) in the work area. The ground cover should be made of thick nylon, waterproof and capable of withstanding work activities without tearing or ripping. The ground cover should be sized enough to prevent contaminates dispersion. Then, segregation process was allowed to segregate contaminated from noncontaminated waste at point of waste generation (figure 3).

Surrounded area | Carpeting with ground cover | Pick up of contaminated soil

Fig. (3) Segregation and pick up processes

Homogeneous samples were sent to the Central Laboratories Directorate (CLD) to recognize the type and concentration of radioisotopes in the waste material. The analysis procedure has done according to IAEA-TECDOC-1092 [19]. Samples were dried in oven at temperature (80-100) degree Celsius for 12 hours milled by milling machine then sifted by specific sieve after that sample volume were (500ml) kept in locked (marinelli beaker) and stored for one month in order to let the chains of U-238 and Th-232 can reach to the radiological equilibrium finally.

Gamma spectroscopy system (Canberra) was used to measure the samples. This system consist of detector, preamplifier, pulse-height analyzer (DSA1000), lead shield and vertical high purity germanium (HPGe) detector with relative efficiency 40% and resolution (<1.8KeV) based on measurements of 1.332MeV gamma ray at photo peak of Co-60 source and multichannel analyzer (MCA) with 8192 channel. Both high voltage supply and amplifier device are compact in one unit (DSA1000), detector shield with a cavity adequate to 10cm lead, absorbed grid from Cadmium 1.6mm and Copper 0.4mm to reduce radiological back ground. System calibration efficiency is carried out by using multi gamma ray standard source (MBSS-2 Canberra) of marinelli beaker geometry. A library of radionuclides which contained the energy of the characteristic gamma emissions of each nuclide was analyzed and their corresponding emissions probabilities were built from the date supplied in the software (Genie-2000). The radioactivity concentration of U-238 can be determined by Gamma energy (1001KeV) which is belong to isotope (Pa-234m) for high radionuclide concentration samples and by (Bi-214) to low concentration, U-235 determine by (185KeV) of Gamma energy which is belong to the same isotope while (Th-232) is determine by (911.7KeV) of Gamma energy which is belong to (Ac-228), (K-40) can be determined by peak energy (1460.8KeV). Cs-137 and Eu-152 can be determined at (662KeV and 344.3KeV) peak energy respectively. The radionuclide concentrations determined in (Bq/kg) units.

Results and Discussion

The physical and radiological characteristics of the waste generated during remediation of the contaminated area are indicated in table (2).

Table (2) characteristics of the generated waste

Physical Status	Weight (kg)	Radionuclide	Specific activity (Bq/kg)
Soil	2650.9	Cs-137	1290.13
		Eu-152	12.5
		Co-60	127.32
	287.8	Co-60	4277
		Eu-152	5.35
		Cs-137	3.82
	33.2	U-238	4382.53
		Cs-137	48.2
		Eu-152	114.5
		Co-60	24.1
Debris	1197	Cs-137	14478.3
	8.4	Eu-152	297.62
	150	Co-60	2700

Safety assessment calculations have done by using SAFRAN (Safety Assessment Framework) version 2.3.2.7 software that incorporates the methodologies developed in SADRWMS (Safety Assessment Driven Radioactive Waste Management Solutions) project.

SAFRAN calculations are tabulated in tables (3). The Assessment results for doses from decommissioning activities were shown in fig.(2). The doses come from different waste management activities which took place in field pretreatment for radioactive wastes generated during remediation of the affected areas.

Table (3) characteristics of waste processing activities

Impact	Exposure time (h/year)	Dose rate (µSv/h)	Annual dose (µSv/year)
Characterization	**180**	**2.23**	**401.4**
removal	**300**	**2.23**	**669**
Packaging	**160**	**2.23**	**356.8**
Relocating	**80**	**2.23**	**178.4**

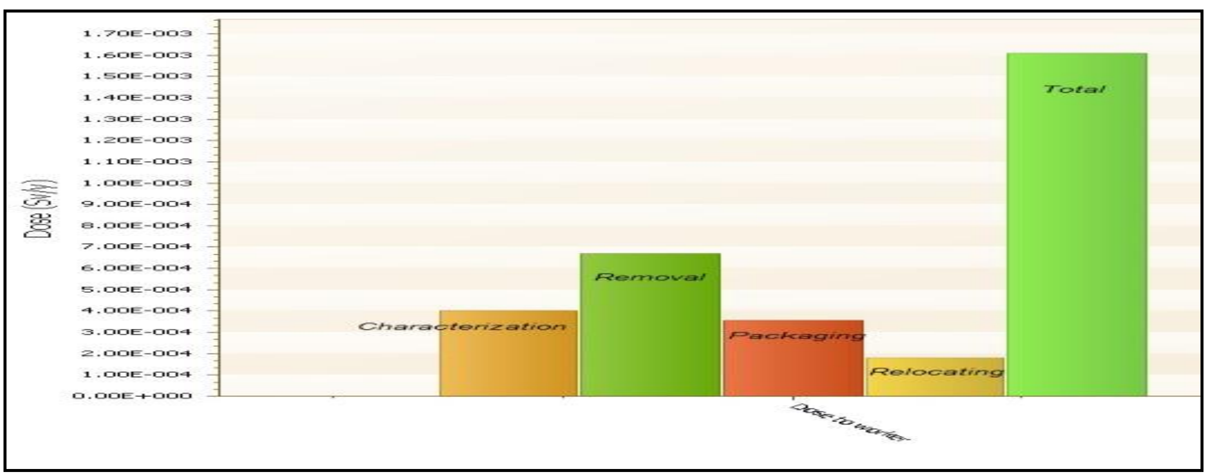

Fig. (2) SAFRAN dose calculation to workers from all activities

From table (2) we can note that the generated waste are physically segregated into soil and debris then radiologically classified according to content of radioisotope type in waste material. The dominated radioisotopes were Cs-137, Eu-152, Co-60 and small amounts of natural Uranium.

Table (3) named activities of waste processing which followed in field pretreatment to remediate the contaminated soil and debris. The duration, periodic and accumulative dose rate for each activity also have showed in table (3). The exposures are calculated (in SAFRAN) based on the annual durations of the different activities which are provided in table (3). The external dose rate is estimated based on results of the radiological measurements of hot spots by dividing the maximum of the recorded doses by the number of hours spent for all activities. This dose rate was then multiplied by the duration number of each activity to obtain the annual dose for each activity. Assessment takes into account the external and internal dose rate. The maximum concentration of airborne radioactivity under normal operation conditions was manually calculated after air sampling by RADeCO device and measured by Ludlum (3030) instrument.

Fig.(2) shows the total and dose rate for each activity which affecting to the worker from all pathways. There are no activities that have been assessed to present a risk rating higher than low and the risk remain below the relevant prescribed dose limits. From fig.(2), the sum of the doses to the workers for all activities, 1.6 mSv/y, is less than 10% of the 20 mSv/y dose limit.

We can say that worker who have taken doses from normal operations remain within the legal annual limits. The assessment undertaken indicates that the works complied with international safety standards and meet the relevant dose limitation criteria with respect to workers.

Conclusions

Effective safety programs have been included into remediation process of the surface contaminated area. The required level of remediation was established on a site specific basis and in accordance with the radiation protection principles that apply to intervention situations.

The radiological risk associated with the remediation activities is assessed as low. Assessment taking into account specific aspects like contact dose rates, concentration of contaminants in air. The measures which were identified in the safety assessment are elicited from contaminated areas through detail characterization and formally laid down in operational procedures and work instructions.

The International Commission on Radiological Protection (ICRP) derive the limit of an average of 20 mSv per year over five years for the occupational dose limit and 1 mSv per year for the public dose limit. The maximum worker dose is 1.6 mSv per year, Public dose has been found to be negligible and no accident was mentioned during works. Thus, remediation works to complete surface contamination is considered to be adequately with the reduced associated risks As Low As Reasonably Achievable.

References

[1] Abbas, M., Al-Atia, M., Bushra, A., Helou, T., Al-Mubarak, M., Danneels, J., Cochran, J. R., Sorenson, K. and Coates, R. (2007) Decommissioning of the Iraq Former Nuclear Complex. Proceedings of the 11th International Conference on Environmental Remediation and Radioactive Waste Management ICEM2007. Bruges. Belgium.

[2] Kenagy, W. D. (2013) The Iraq Decommissioning Project –Eight Years of Accomplishments. the 57th IAEA General Conference. Vienna

[3] IAEA. (2005) standard format and content for safety related decommissioning documents. safety reports series No. 45. Vienna.

[4] IAEA. (2013) Safety Assessment for Decommissioning. safety reports series No.77. Vienna.

[5] IAEA. (2000) Predisposal Management of Radioactive Waste Including Decommissioning. Safety Standards Series No. WS-R-2. Vienna.

[6] IAEA. (2013) The Safety Case and Safety Assessment for the Predisposal Management of Radioactive Waste. Safety Standards Series No. GSG-3. Vienna.

[7] SAFRAN tool and SAFRAN User Guide (http://goto.iaea.org/safran).

[8] IAEA. (2011) Safety Guide DS 284 for Safety Case and Safety Assessment for Predisposal Facilities and Activities. IAEA. Vienna.

[9]IAEA. (2011) Al Tuwaitha-Italain Radioisotopes Production Facility. Report on decontamination activities in Iraq. Vienna.

[10] Abdulhadi, Q.A. (2009) Project Management Plan of RPL. Iraqi Decommissioning Program of Previous Nuclear Sites and Facilities. Baghdad-Iraq.

[11] Radiation Protection Center (RPC). (2012) Concentration Values for Clearance Levels of Radionuclides in Solid and Liquid Waste. Ministry of Environment. Baghdad-Iraq

[12] IAEA (2004) Application of the Concepts of Exclusion, Exemption and Clearance. Safety Guide No.RS-G-1.7. Vienna

[13] IAEA. (2005) Derivation of Activity Concentration Values for Exclusion, Exemption and Clearance. Safety Series No.44. Vienna

[14] Environmental Protection Agency (1980) Method 900.0:Gross Alpha and Gross Beta Radioactivity. www.epa.gov . USA

[15] IAEA. (2003) Predisposal Management of Low and Intermediate Level Radioactive Waste. Safety Standards Series No.WS-G-2.5. Vienna

[16] IAEA. (2007) Categorizing Operational Radioactive Wastes. TECDOC-1538. Vienna.

[17] IAEA. (2008) Managing Low Radioactivity Material from the Decommissioning of Nuclear Facilities. Technical Reports Series No.462. Vienna

[18] IAEA. (2006) Development of Specifications for Radioactive Waste Packages. TECDOC1515; ISSN 1011-4289. Vienna.

[19] IAEA. (1999) Generic Procedure for Monitoring in a Nuclear or Radiological Emergency. IAEA-TECDOC-1092. Vienna.