DOI: 10.24996/ijs.2018.59.1C.3

ISSN: 0067-2904

Hypothetical Method for Gamma Dose Rate Assessment to Conditioned Radioactive waste Container

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Abstract

Metallic solid radioactive waste class low level - short lived Radioactive Waste (LL-SL RW) is the main type of radioactive waste generated from decommissioning operations. Transport, storage and disposal regulations require for gamma emitting radioactive waste (mainly by ^{137}Csisotope), that the dose rate in the proximity of the container should stand below a certain threshold. Also, the conditioning technique (using cementation technique) based on certain matrix with specific ratios should be able to attenuate the gamma radiation activity to the minimum level or to acceptable dosage rate at distance of 1m from the container. In this paper ,in absence of suitable labs for waste package assessment ,hypothetical method present to assess dose rate in safe way, assumption based on metallic waste pieces contaminated with (^{137}Cs), were conditioned with cement matrix and contained in carbon steel drum volume 220 liter ,60cm diameter then dose rate measurement applied in vicinity of the container. Instead of real contaminated metallic waste, (^{137}Cs,D_o=20mR/hr) gamma radioactive point source was positioned in different places in front of cross section of the cemented free metallic waste and gamma dose rates were measured on the outer side of the drum sample using NaI dose meter device. The readings showed good attenuation of gamma radiation activity (low dose rates), efficiency of the cement matrix to decrease the dose rate of (^{137}Cs,0.662Mev) gamma radiation lower to acceptable values and with waste acceptance criteria and regulation.

Keywords: Dose assessment, Radioactive waste, Linear attenuation coefficient

طريقة افتراضية لتقييم معدل جرعة اشعة كاما لحاوية نفايات مشعة مكيفة

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الخلاصة

تصنف النفايات المشعة الموضوعة داخل حاويات معدنية والناتجة من عمليات تصفية المنشأت النووية بانها نفايات منخفضةالنشاط الاشعاعي وذات عمر نصف قصير .(LL–SL RW)تتطلب أنشطة نقل وخزن وطمر النفايات المشعة الباعثة لاشعة كاما (مثلا نظير (Cs–137ان يكون معدل الجرعة الاشعاعية على سطح الحاوية ذات قيم محددة. يتم معاملة النفايات المشعة باستخدام تقنية السمنته باستخدام مواد اوليةبنسب معينة تحقيق عملية توهين نشاط اشعة كاما الى تلك الحدود المسموح بها على سطح الحاوية وعلى بعد متر واحد . في هذه الدراسة ولغياب المختبرات المناسبة لتقييم رزم النفايات المشعة استخدمت طريقة افتراضية لتقييم جرع التعرض الاشعاعي السطحي وبطريقة تحقق مبادئ السلامة الاشعاعية. بدلا عن استخدام حاوية

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نفايات معدنية مكيفة وملوثة بنظير السيزيوم تم اعداد حاوية نفايات سعة 220 لتر وبقطر 60 سم حاوية لقطع معدنية غير ملوثة جرت معاملتها بالسمنته وتقطيعها الى عدة مقاطع . تم وضع مصدر مشع نظير Cs-137ذو معدل جرعة اشعاعية (Do=20mR /hr) في منتصف المقطع العرضي لحاوية النفايات المتعرضة لنظير السيزيوم و اخذقياس الجرع الاشعاعية السطحيه وعلى اماكن مختلفة من المحيط الخارجي للمقطع العرضي باستخدام جهاز قياس الجرع الاشعاعية.(NaI) أظهرت القراءات توهينا جيدا لنشاط إشعةكاما (معدلات جرع اشعاعية منخفضة)، وكفاءة مصفوفة الأسمنت في تقليل معدل جرعة اشعةكاما لنظير (Cs-137) ذي الطاقة 0.662 (Mev)إلى قيم مقبولة تتوافق مع معايير قبول النفايات.

1. Introduction

RW handling activities are hazardous as concern for both contamination and external exposure. Therefore strict regulations are applied for radiation protection in this field [1]. Dose rate in the vicinity of waste drum has to be kept below certain constraints throughout entire treatment and conditioning process and for the final storage; the dose rate should meet waste acceptance criteria (WAC)[2]. Many experiments and tests applied to conditioned radioactive waste drum to check their purposes such as quality of encapsulation process or evaluation the attenuation of gamma activity to minimum level. Also to determine and assess the dose rate for conditioned RW drums to storage or disposal [2]. Here one of the test that should be taken in RW management include cut open the package RW (cemented drum) in horizontal direction in safe –secured conditions to examine the enteric of the cemented RW drum and for gamma attenuation determination (shielding function), The cemented bulk will be head to gamma ray detector or dose rate device and will be taken in contact to the outer side of the drum for quality and safety assessment . In such tests risks of highly contamination and exposure will be high (radioactive waste dust dispersed in environment). In the absence of controlled lab for such test; hypothetical system present in this work ; using metallic waste free of contamination having conditioning process with cementation technique and ^{137}Cs gamma point source was used . This estimation method was necessary in absence of suitable labe for cemented waste as sample test while in waste management activities are very costly and thus the (hypothetic) tests could incorporate as much activity as possible to optimize these cots .In addition many studies could be followed in this field for labs poor in RW drums characterization. The estimation of gamma dose for homogenous waste containers are widely field because of variety of radioactive waste materials for each country and with variety of their activities [3, 4].

2. Method and Results

The container consist of 15-20 % free metallic waste and 80-85% of encapsulation material .This drum waste cut open to get two halves piece of cement – waste bulk be headed to ^{137}Cs gamma source pointed in many places across the diameter of the sample (R) and dose rate detection held in contact to the bulk outer surface using dosimeter device

To achieve the objectives of the present study, Iraq Portland cement and additives are prepared and mixed in certain ratios as Tables-(1, 2) show the specification of cementation matrix ratios while Table-3 represent the chemical analysis of container alloy components that used in the waste management facility .

Table 1-Specification of preliminary mix

- ❖ water/ cement ratio : 0.5
- \- flyash/ cement ratio :0.3
- \- Additives : SP703, 1 Liter / 100 Kg cement
- \- Time of mixing :30 min
- ❖ Density of cement matrix : 1.675 g/m^3
- \- Compressive strength : 22 N/mm^2

Table 2-Mass Components for Portland cement .

Components	CaO	SiO_2	Al_2O_3	Fe_2O_3	MgO	SO_3	K_2O and Na_2O
Mass content (percent)	58-66	18-66	4-12	1-6	1-3	0.5-2.5	1

Table 3-The chemical analysis of container waste alloy using AAS technique.

Element	Fe	Mn	Ni	Cr	Cu	Zn	Mg	Al	Mo
Concentration	96%	0.11%	42 ppm	0.02%	65 ppm	68 ppm	10 ppm	288 ppm	0.21p pm

The cement matrix and additives was mixed for 30min. which were considering enough to achieve good homogeneity. The density (ρ) about 1.675g/cm^{3}dropped into the drum where the clean solid waste was collected in iron basket and centered in the container and left for few moment on the vibrate stage to let gas bubbles escaped from the surface. The sample left for 28 days where the solidification of the cement - waste mixture are completed. Figure-1, show conditioned free waste package after cutting process using electrical machine to get half shape pieces. For gamma dose rate validation, a sample of half -half package waste form was prepared and head to ^{137}Csgamma source to assumed the waste has been exposed to such gamma radiation rays from many distance (R) along the bulk diameter and that head to different thickness (x) or depth has gamma radiation penetrate through cement matrix shield and reach the detector window. Used (NaI) crystal type Ludlum held device with 40% efficiency of gamma ray 0.661Mev to measurement of dose rate, the background was 9µR/hr. and used at contact with the outer side of waste package. Dose rate measurement and apparatus design in Figure-2 which represents schematic representation of method design.

Figure 1-Low magnification optical graphs of cut open conditioned solid wasteContainer.

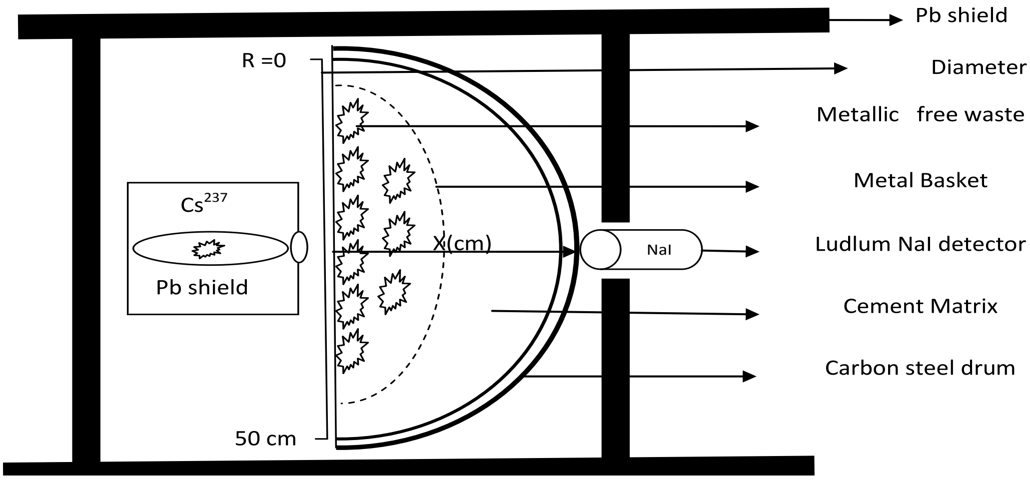

Figure 2-Schematic representation of dose rate measurement to waste container.

Table 4-Dose rate measurement and Calculated of linear Attenuation Coefficient (**μ**) in terms of Dose rate.

^{137}CsPosition (R) (cm)	(D∘) (μSv/hr.)	(D) (μSv/hr.)	Thickness of Cemented Waste body (x) (cm)	ln(D∘ / D)	**μ** (cm^{-1})
5	26.4	7.62	19.5	1.24	0.063
10	20	4.86	24	1.41	0.058
15	18.1	3.30	24.5	1.70	0.069
20	16.9	3.21	26.5	1.66	0.062
25	15.9	3.26	28.5	1.58	0.055
30	15	3.11	29.5	1.57	0.053
35	15.9	3.23	28.5	1.59	0.055
40	16.9	3.76	26.5	1.50	0.056
45	18.1	4.58	24.5	1.37	0.056
50	20	5.80	24	1.23	0.051
52	26.4	7.55	19.5	1.25	0.064

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The well-known photon linear attenuation coefficient (**μ**) or shielding equation may calculated using equation below:

$$\boldsymbol{\mu} = \frac{ln\frac{A_\circ}{A_i}}{X} \quad (1)$$

[5].

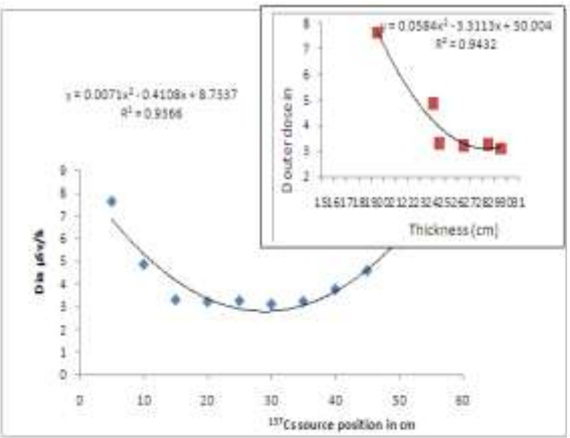
Where A_0 is the incident gamma ray activity , A_i is the the incident photons obtained for the cement waste bulk of thickness(x). using narrow collimated mono-energetic beam of ^{137}Csgamma source. The linear attenuation coefficient μ was obtained by the following formula:

$$\boldsymbol{\mu} = \frac{ln\frac{D_\circ}{D_i}}{X} \quad (2)$$

[5].

D_0 is the dose rate which obtained without inserting any sample between the detector and the source, D_i is the dose rate value after the cement waste bulk.

The dose rates and attenuation coefficient**μ** were performed for cemented waste cross section along diameter (R) in relative to their thickness (x).While **μ** depend on density of absorber material and cross sections of gamma ray reactions with absorber material readings show ratios of errors as was expected .By assume equation (2) is valid since the photons in the incident beam are mono-energetic and the beam are narrow [4], so for simplicity of the test ,build up factor or the mean free path of photons travels were not considered in calculation . From dose rates measurement, activity values were obtained using computer program RadPro 3.2 calculator version 3.24 (2009). The results show in Tables-(3, 4) are displayed in Figures-(3, 4, 5 and 6) it can be seen the linear attenuation increased with cement –waste thickness. The plots in Figures-(5, 6) show the responding relation between the final dose rate of cemented waste bulk and the depth shield along the R axis of the sample or 662Kev gamma energy[6,7]. The dose rate of the ^{137}Cscontained by cemented waste and carbon steel container specimen was dropped to very low values which are within waste acceptance criteria (WAC)[8]. Figure-3 show discrepancies in the results which is mainly due to internal in homogeneity of the waste form which can alter the dose rate measurement results, To be noted that the main hypothesis of this estimation is the homogeneity of the waste form content and the uniformity of the activity distribution. By comparing the dose rate results which attenuated by the cement matrix thickness and path length of gamma ray are with the (WAC) for storage or transport seems acceptable and the conditioning mixture showed a very good suitability and efficiency for solid (liquid) radioactive waste management[9].



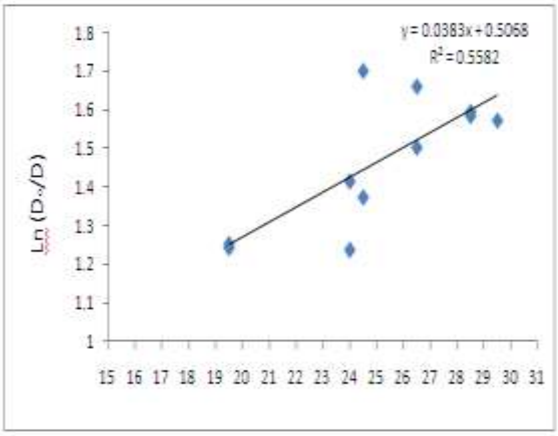


Figure 3-Relation of ln (D_o/D) against cemented waste specimen depth (thickness*).*

Figure 4-Variation of final Dose rate D relate to ^{137}Cs *gamma source location along the cement waste crosssection diameter, R (0-52cm).*

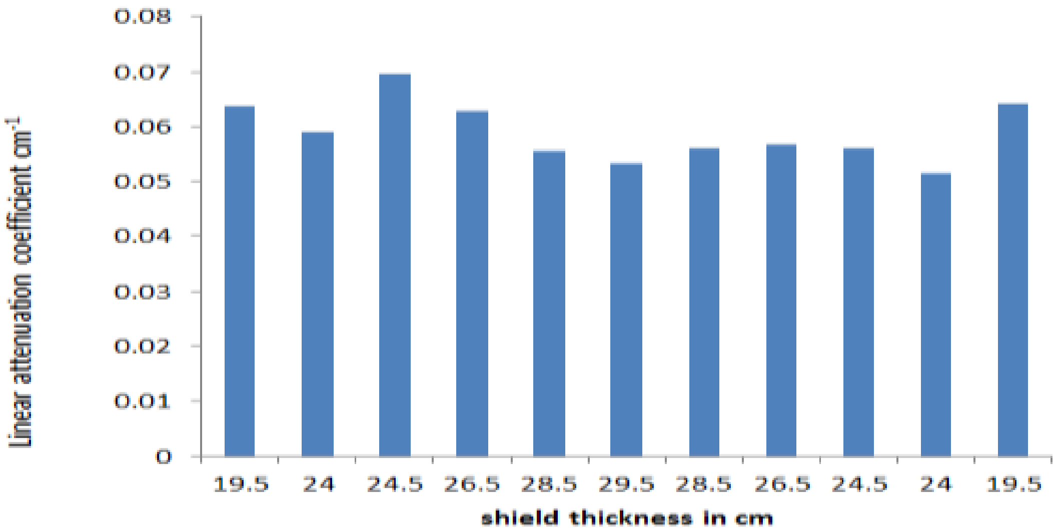


Figure 5-Linear attenuation coefficient (**μ**) against shield thickness (x).

Table 5-Calculated of linear attenuation coefficient (μ) in Activity terms (A).

^{137}Cs **Position R (cm)**	**(A_1) (Bq)**	**(A_2) (Bq)**	**Depth (cm)**	**ln(A_1 / A_2)**	**μ (cm^{-1})**
5	1.31E+07	3.79E+06	19.5	1.24E+00	6.37E-02
10	1.51E+07	3.67E+06	24	1.41E+00	5.89E-02
15	1.42E+07	2.59E+06	24.5	1.70E+00	6.95E-02
20	1.55E+07	2.95E+06	26.5	1.66E+00	6.27E-02
25	1.69E+07	3.47E+06	28.5	1.58E+00	5.56E-02
30	1.65E+07	3.44E+06	29.5	1.57E+00	5.32E-02
35	1.69E+07	3.44E+06	28.5	1.59E+00	5.59E-02
40	1.55E+07	3.46E+06	26.5	1.50E+00	5.67E-02
45	1.42E+07	3.60E+06	24.5	1.37E+00	5.61E-02
50	1.51E+07	4.38E+06	24	1.24E+00	5.16E-02
52	1.31E+07	3.76E+06	19.5	1.25E+00	6.42E-02

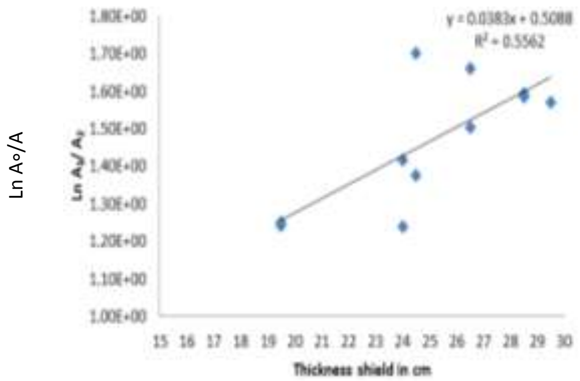


Figure 6-Relation between ln A∘/A and thickness shield(x).

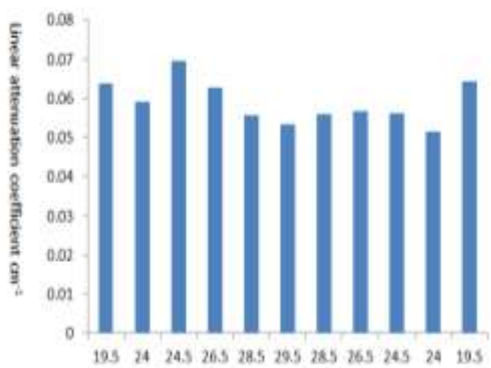


Figure 7-linear attenuation coefficient(**μ**) and thickness shield(x).

3. Conclusions

During this study, characterization of radioactive waste containers for low and intermediate level waste was implemented.

1- The maximum linear attenuation coefficient in terms of dose rate or activity were attained to 0.0531/cm cement matrix ratios incorporate with iron basket for 15-20 % waste ,80-85% cement matrix for ^{137}Cs gamma source of initial dose 15μSv/hr.

2- The minimum dose rate values was in contact of waste container bulk exposed to ^{137}Cs gamma source was (3.11μSv/hr.,) and the maximum dose values was (7.62 μSv/hr) related to the thickness of the penetration depth of the gamma radiation 29.5 and 19.5 cm Respectively .

3-The results proved that the cementation matrix (ratios) and the iron basket with the carbon steel container metals, have an impact on attenuation performance of gamma radiation beside the bulk density and porosity level of the solidified cement –waste mix.

4- Substantial improvement of about (28.86,20.73) % in attenuation performance in term of dose rate ratios at R(5 and 30) cm for initial dose rate D∘ (26.4 and 15 μSv/hr.) in air of cement waste sample depth was attained for solid waste and using Iraqi Portland cement local product.

5-Solid waste conditioning and treatment utilized successfully due to the structure of open cut cross section of cement waste bulk.

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