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Article

Conditioning of ^{60}Co spent sealed sources

M.A. HASAN[1], Y.T. SELIM[1], Y.F. LASHEEN[1]

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ABSTRACT **As part of their functions, the Hot Laboratories Center, EAEA, are responsible for conditioning of different spent sealed sources which received from different end users. Regarding to the inventory of spent sealed sources in Egypt, thirty eight ^{60}Co spent sealed sources with low radioactivity levels are needed to be conditioned according to IAEA guidelines. The method of conditioning of ^{60}Co spent sealed sources can be achieved using 200 l drum with special lead shield and concrete as matrix material. The surface dose rate calculation reveals that when lead shield of 6.5 cm thickness (density 11.35 gm/ml), concrete shield at least 18 cm thickness (density 2.35 gm/cm^3) and 200 l drum thickness is 0.14 cm, (density 7.86 gm/cm^3), the surface dose rate is 0.86 mSv/ h. The data are complied with the regulation of IAEA safety standard (surface dose rate doses not exceed than 2.0 mSv/ h).**

Keywords: Conditioning / ^{60}Co / spent / sealed / sources

RÉSUMÉ **Le conditionnement des sources scellées de ^{60}Co périmées.**

Entre autres fonctions, le centre des laboratoires chauds de l'AEAE est responsable du conditionnement des diverses sources scellées périmées en provenance de différents utilisateurs. En ce qui concerne l'inventaire des sources scellées hors d'usage en Égypte, trente huit sources scellées de ^{60}Co périmées, présentant des niveaux faibles de radioactivité, doivent être conditionnées en conformité aux directives de l'AIEA. La méthode de conditionnement des sources scellées de ^{60}Co périmées emploie un fût de 200 litres, blindé par un écran spécial de plomb, placé lui-même dans une matrice de béton. Le calcul du débit de dose à la surface de l'ensemble indique qu'avec un écran de plomb épais de 6,5 cm (masse volumique : 11,35 g/ml), un écran de béton épais d'au moins 18 cm (masse volumique : 2,35 g/ml) et un fût de 200 litres, épais de 0,14 cm (masse volumique : 7,8 g/cm^3), le débit de dose sur la surface extérieure est de 0,86 mSv/h. Ces données respectent les normes de sûreté de l'AIEA (la dose à la surface ne doit pas dépasser 2,0 mSv/h).

1. Introduction

Although the immediate objective of conditioning is to facilitate interim storage, it should also facilitate transport when the disposal facility is off-site. However, the most important thing is to produce a package likely to be suitable and acceptable at a final disposal site. Therefore it is very important to follow international guidance and regulations for the conditioning and disposal of radioactive wastes. For these reasons, the conditioning-process involves:

- the production of a package type that is recognized in and conforms to IAEA transport regulations (1);

[1] Hot Laboratories and Waste Management Center, Egyptian Atomic Energy Authority, Abu Zabul, Egypt.

TABLE I
Type A1 and A2 containers used in Egypt.
Conteneurs de types A1 et A2 utilisés en Égypte.

Radionuclide	A1/ Ci	A2/ Ci	Specific activity Ci/g
^{137}Cs	54.1	13.5	87
^{60}Co	10.8	10.8	1100
^{241}Am	54.1	5.41×10^{-3}	3.4
^{90}Sr	5.41	2.7	140
^{192}Ir	27	13.5	9200
^{85}Kr	541	270	390

* 1 Ci = 3.7×10^{10} Bq.

- the use of an immobilization matrix (cement mortar) that is already widely accepted in many countries for interim storage and disposal sites.

It may be considered that a package that is safe to transport through the public sector will be also suitable for interim storage at a secure site. The method of conditioning described in this short report is based on the Type A package as defined in IAEA transport regulation (IAEA, 2000a). A Type A package is defined as a packaging, tank or freight container containing an activity up to the A2 value or up to A1 if the contents meet the definition of "special form" radioactive material.

"Special form" radioactive material means an in-dispersible, solid radioactive material or sealed capsule containing radioactive material. It may be expected that many sealed sources meet these requirements and thus are classed as special radioactive material (IAEA, 1991). To be considered "special form", the sealed source must have a certificate from the manufacture company. This documentation must include an expiration date if it has been withdrawn or expired without extension. In this case, the source cannot be considered "special form" so the IAEA recommends the use of A2 values for Type A containers. A list of the values of A1 and A2 are given in Table I for radionuclide commonly used in Egypt.

2. Spent radioactive sealed sources (SRSS) acceptance criteria for conditioning

There are some conditions and waste acceptance criteria for any source will be conditioned by this method:

1. only ^{60}Co sources will be approved to be conditioned by this method;
2. all ^{60}Co sources must have a leak test record with the test results not exceeding 0.005 Ci. If it cannot pass the test, the operator has to perform the leak test according to IAEA guidance (see Appendix 1);

3. the total activity in each drum is not exceeding 1 Ci per drum;
4. only the 200 l drum is approved for use in this method;
5. the density of the cement mortar is not less than 2.35 g/cm^3 and the density of lead is not less than 11.3 g/cm^3;
6. the drum's surface dose rate is not to exceed 2 mSv/h (200 mrem/h).

3. Quality assurance

Quality assurance means all those planned and systematic actions necessary to provide adequate confidence that an item, process or service will satisfy given requirements for quality that means the operator (HLWMC) has to make all actions and safety measures necessary to keep the contamination level, surface dose rate, transportation index, personnel certification, documentation and record keeping within the IAEA, EAEA and/or US standards. The operator has to develop organization and responsibility structure within the radioactive waste facility QAMP and also develop procedures for handling QAP1, recovery QAP2, conditioning QAP3, procurement QAP4, WAC for storage and disposal QAP5, instruments calibration QAP6, personnel training and certification QAP7, non-conformance, QAP8 and corrective actions QAP9 and record control QAP10.

4. Conditioning procedure

The conditioning procedure is based on the immobilization of the sources within a Type A package (mild steel 200 l drum) with concrete thickness of 20 cm. The sources encased within a special lead container are placed in the center of a 200 l drum filled with cement mortar. The activity in the package must not exceed the A1/A2 level as set out in the IAEA transport regulations. Conditioning in this way prevents unauthorized removal of the source. The bulk, weight and robust nature of the package also provide a barrier against the loss of containment of the radioactive material. Such a package would have a weight of about 450 kg and so removal and transportation would require the use of mechanical equipment *e.g.* fork lift (IAEA, 2000b).

4.1. Calculation the shielding at the top cap and surface

For metric distance the exposure rate for any radionuclide can be calculated as follow:

$$I = 0.5\ C\ E\ N / D^2$$

where I = the exposure rate in R/h, C = source activity in Curies (Ci), E = gamma energy in MeV, N = number of gammas per disintegration (photon yield), D = distance in meter.

For 1 Ci of ^{60}Co as a maximum radioactivity level to be conditioned, the exposure rate (at 25 Cm = 0.25 m) = 20 rem/h = 0.2 Sv/h.

The exposure rate 0.2 Sv/h of 1 Ci ^{60}Co should be reduced to be less than 2 mSv/h using lead and concrete shielding conditioned in 200 L steel drum according to the above criteria.

To prove these criteria we will define the half value layers (HVL) of the shielding materials (lead, concrete and steel).

One half value layer: is defined as the amount of shielding material required reducing the radiation intensity to one half of unshielded value.

The energies of gray of Co-60 are 1173.2 and 1332.5 keV.

The HVL for lead (density ρ = 11.35 gm/ml) is 1.3 cm, concrete (density ρ = 2.4 gm/ml) is 6.4 cm and iron (density ρ = 7.86 gm/cm^3) is 2.4 cm (IAEA, 1995). The basic equation for using HVL concept is:

$$I = Io\ (1/2)^n$$

where I = shielded exposure rate, Io = unshielded exposure rate, n = number of HVL = shield thickness (cm)/ HVL thickness (cm).

According to the above equation :

- for 6.5 cm thickness of lead (shield-1), the dose rate will be reduced from 200 mSv/h to 6.25 mSv/h;
- for 18.5 cm thickness of concrete (shield-2), the dose rate will be reduced from 6.25 mSv/h to 0.89 mSv/h;
- for 0.14 cm thickness of steel drum (shield-3), the dose rate will be reduced from 0.89 mSv/h to 0.86 mSv/h.

4.2. Calculation the shielding at the bottom

For 2 cm thickness of lead (shield-1), the dose rate will be reduced from 200 mSv/h to 68.87 mSv/h.

For 42.5 cm thickness of concrete (shield-2), the dose rate will be reduced from 68.87 mSv/h to 0.69 mSv/h.

For 0.14 cm thickness of steel drum (shield-3), the dose rate will be reduced from 0.69 mSv/h to 0.662 mSv/h.

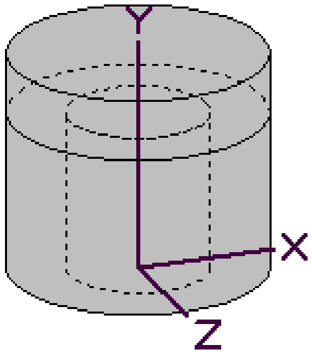


Figure 1 – ***Lead shield where X = 6.5 cm, Y = 26.5 cm, Z = 6.5 cm.***
Écran de plomb.

4.3. Equipment and supplies required

- Special lead shield (Fig. 1) with a thickness of 6.5 cm, cap thickness of 6.5 cm and bottom thickness of 2 cm (shown below);
- 200 l drum free from rust spots or other defects (inside and outside);
- mold for preparing a cavity inside the drum and for the cap of the drum;
- cement mixer or even manual mixing;
- handling tools (*e.g.* Forceps, tongs 60 cm long with parallel grip 100 cm long);
- a fork lift;
- labels for storage and transportation;
- plastic sheets to prevent against contamination during work.

4.4. Procedure steps

Assemble all equipment and supplies required to proceed with the conditioning process as listed above (Fig. 2).

- Open and visibly inspect the Type A 200 l drum for rust spots or other defects. Don’t use the defected drums.
- Mix sufficient concrete to fill the drum to approximately half of its volume.
- Place the concrete mixture in the drum and shake the drum to assure that no void has been formed in the concrete.
- Insert the mold into the container and the container casket, place the cement mortar around the mold and then let the concrete set to harden. Ensure that the thickness of the concrete in the cavity section is not less than 18 cm and the cavity diameter is at least 19 cm.
- Using appropriate radiation protection and handling procedures, remove the sources from their shields and place them in the prefabricated lead shield as prescribed above. The maximum activity in prefabricated shield is 1 Ci for ^{60}Co.

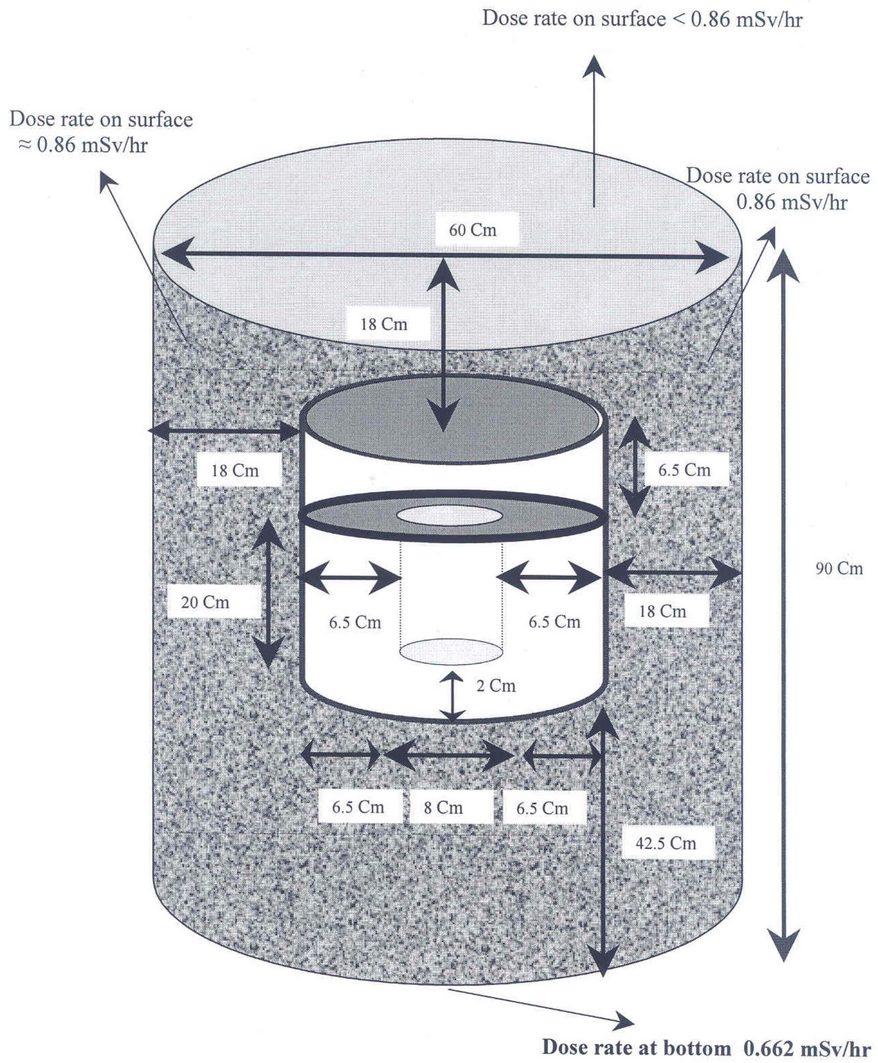


Figure 2 – ***200 l Drum designed for ^{60}Co conditioning using concrete and lead shield.***
Fût de 200 litres pour le conditionnement du ^{60}Co, associé à des écrans de béton et de plomb.

TABLE II
Contamination level on the drum.
Niveau de contamination sur le fût.

Contaminant	**Max. Permissible level**	
	Bq/cm^2	**μCi/cm^2**
Beta and gamma emitters	4.0	10^{-4}
Alpha emitters	0.4	10^{-5}

Table III
Drum number 1.
Fût n° 1.

I.D	**Activity/mCi/2005**	**Source Number**	**Total/mCi**
1	0.125	1	0.125
7	0.1	4	0.4
13	0.1	1	0.1
24	16.25	8	130
58	9.2	2	18.4
61	0.28	1	0.28
63	79	10	790
74	11.2	1	11.2
+02	0.38	2	0.76
	Total	**30**	**951.66**

- Place the prefabricated shield into the cavity in the drum.
- Install the drum head and bolt ring by screwing the closure device.
- Check the outside of the drum to ensure that no concrete has been spilled on the outside surface.
- Inspect the drum integrity and perform wipe tests (see Appendix 1) to determine presence of contamination on the drum. According to the IAEA safety series No. 6, “regulation of the safe transport of radioactive material” should be followed. These limits are summarized below in Table II.
- Post appropriate radiation labels on the drum to ensure proper identification for storage. Regulations for the safe transportation of radioactive material place a limit on surface radiation of 200 mrem/h (2mSv/h) (IAEA, 1996).

Tables III and IV reveal the total radioactivity levels of ^{60}Co and total number of sources per each drum.

TABLE IV
Drum number 2.
Fût n° 2.

I.D	Activity/mCi/2005	Source Number	Total/mCi
18	125	8	1000
	Total	8	1000

5. Conclusion

It is obviously clear to conclude that according to the spent radioactive sealed source (SRSS) inventory in Egypt (which reveals 38 known ^{60}Co spent sources already stored at Inshas storage facility), the two designed drums can be used for conditioning of thirty eight ^{60}Co spent sealed sources with a maximum radioactivity level about 1 Ci of each.

The surface dose rate calculation reveals that when lead shield of 6.5 cm thickness, concrete shield of at least 18 cm thickness, (density = 2.35 g/cm^3) and the 200 drum thickness is 0.14 cm, the surface dose rate is 0.86 mSv/h. The surface dose rate which is calculated by the basic equation is confirmed by micro shield software version 6.2. According to this procedure, the 38 ^{60}Co sources may be put into 2 drums without the surface activity of each drum exceeding 1 Ci = 1000 mCi.

Appendix 1: Leak testing of sealed sources

(A) Each licensee who uses a sealed source shall have the source tested for leakage periodically. The licensee shall keep a record of leak test results in units of microcuries and retain the record for inspection by the EORP office for three years after the leak test is performed. When the sources are declared as spent, the licensee must approve by documentation that the last leak test conformed with regulatory standards and did not exceed 0.005 microcurie.

(B) The wipe of a sealed source must be performed using a leak test kit or method approved by the Regulatory body according to Egyptian guidance and regulations. The wipe sample must be taken from the nearest accessible point to the sealed source where contamination might accumulate. The wipe sample must be analyzed for radioactive contamination. The analysis must be capable of detecting the presence of 185 Bq (0.005 microcurie) of radioactive material on the test sample and must be performed by a person approved by the Chairman of Radioactive Waste Management Division, and Egyptian regulatory body.

(C) Sealed sources must be tested at the following frequencies: each sealed source must be tested at intervals not to exceed six months. In the absence of a certificate from a transferor that a test has been made within the six months before the transfer, the sealed source may not be used until tested.

(D) If a sealed source shows evidence of leaking by revealing the presence of more than 0.005 microcuries, then after a confirmatory recount, a second swipe should be performed. If the test results are positive, the source will be immediately withdrawn from use, decontaminated, and/or repaired or disposed of as appropriate. In the event of the detection of a leaking sealed source, a report will be filed to Chairman of Radioactive Waste Management Division describing the source, the test results, and the corrective action taken.

(E) Each sealed radioactive source, except for hydrogen-3 (tritium), with a half-life greater than 30 days and in any form other than gas, will be tested for leakage by means of a swipe test. This action must be taken prior to initial use and at six-month intervals in accordance with IAEA guidance. The leak tests, to be performed by a member of the radiation safety office staff, will be capable of detecting the presence of 0.005 microcuries (185 Bq) of removable contamination. The results of all leak tests will be recorded in an appropriate record system in units of microcuries or Becquerel. Leak tests are required for sealed beta or gamma sources with an activity of 100 microcuries (3.7 MBq) or greater, and for alpha sources with an activity greater than 10 microcuries (370 kBq).

(F) **Special case: sealed sources in storage**. Old or spent sealed sources that are not being used may be placed in an appropriate sealed container and placed in a long-term storage vault. Sources not in use will be transferred to the radioactive waste management division. They are responsible for storage and record-keeping. Stored sources need only be inventoried once every year and must be leak-tested before being put back into use. However, a single swipe of the storage vault is still required on a six-month interval. This provision is to minimize needless radiation exposure during the swipe testing of unused sealed sources. A special record of all such sources will be maintained and reviewed each six months.

Test procedure:

- verify each source's serial number;
- swipe each source and/or container;
- obtain alcohol or other effective solvent;
- moisten a piece of filter paper, or other suitable material of high wet strength and absorbent capacity, with this solvent;
- wipe all external surfaces of the source thoroughly;

- count each swipe using a liquid scintillation counter, gas-flow proportional counting system or equivalent detection equipment.

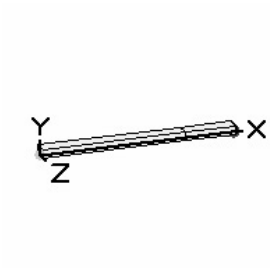
Appendix 2

MicroShield v6.02 (6.02-00229)
Egyptian Atomic Energy Authority

Page	:1
DOS File	:Lasheen.ms6
Run Date	: May 10, 2006
Run Time	: 12:22:39 AM
Duration	: 00:00:00

File Ref	:Conditioning of Spent Co-60 Sources
Date	: May 10, 2006
By	: Yasser T. Mohamed
Checked	: Yes

Case Title: Case 1
Description: Case 1
Geometry: 1 - Point



Dose Points

A	X	Y	Z
# 1	24.64 cm	0 cm	0 cm
	9.7 in	0.0 in	0.0 in

Shields

Shield N	Dimension	Material	Density
Shield 1	18.0 cm	Concrete	2.4
Shield 2	6.5 cm	Lead	11.35
Shield 3	.14 cm	Iron	7.86
Air Gap		Air	0.00122

Source Input : Grouping Method - Actual Photon Energies

Nuclide	curies	becquerels
Co-60	1.0000e+000	3.7000e+010

Buildup : The material reference is - Shield 2
Integration Parameters

Results

Energy MeV	Activity Photons/sec	Fluence Rate MeV/cm²/sec No Buildup	Fluence Rate MeV/cm²/sec With Buildup	Exposure Rate mR/hr No Buildup	Exposure Rate mrem/hr With Buildup
0.6938	6.035e+06	1.315e-02	3.609e-02	2.539e-05	6.967e-05
1.1732	3.700e+10	5.058e+03	1.578e+04	9.039e+00	2.821e+01
1.3325	3.700e+10	9.927e+03	3.097e+04	1.722e+01	5.372e+01
Totals	**7.401e+10**	**1.499e+04**	**4.675e+04**	**2.626e+01**	**8.193e+01**

Surface Dose rate = 0.82 mSv/h

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