PACKING FOR RADIOACTIVE WASTE TRANSPORT

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ABSTRACT

ENEA and NUCLECO have established at the Casaccia Research Centre a technical unit called "Integrated Services" which takes care of all radioactive waste produced by hospitals, research centres and industry, and, waiting for the national disposal facility, remains owner for them in the temporary storage.

NUCLECO covers all the operative phases, including packaging, transport, volume reduction treatment, conditioning for nuclear power plant waste too.

At the end of the conditioning procedures there are, for the scope of this paper, two types of waste packages: industrial ones for LSA-II and packages for high activity sources.

Packages containing LSA-II are produced by solid volume reduction, performed through 220 l drums supercompaction, stacking of the pellets inside a metallic 380 l overpack and then pouring of a concrete compound.

Same packages are generated by treatment of liquid wastes, performed by a biological and chemical-physical process, mixing the sludge with cement inside a 200 l drum and get rid of the clear water.

The paper gives details on packages qualification referring to "the waste form", with compression test, thermal cycling, radiation resistance test, biodegradation resistance test, immersion test, water permeability, free liquids test, and referring to the "transport" with long term corrosion test, free drop, stacking and penetration tests.

Packages containing disused high activity sources must be mainly of Type B(U) and ENEA is studying several types of shielded containment systems for gamma and neutron sources, to ensure their characteristics over a long period.

The design of containment system, as regards the choice of materials for the construction and the thicknesses used, is aimed to comply with transport regulations, taking into account the characteristics of mechanical and thermal protection guaranteed by external part of the package, a model of which was approved by the Competent Authority as Type B (U) for different contents.

The paper shows how we can guarantee packages complying with the waste characteristics as approved by the competent authorities for the storage facility and with the transport regulations, at present and in the next 30 years.

INTRODUCTION

Packing for radioactive waste have to satisfy the requirements established by national regulations for the interim storage, transport and final disposal.
The first section of the paper, will present the qualification tests for the waste packages "MOWA" and "OVERPACK" produced and improved for many years.
The qualification of the packages "MOWA" and "OVERPACK" has been based on a first distinction between the waste form, the packaging and the package. The waste form satisfies the requirements for the disposal, while the packaging and the package complies with the specifications for the handling, temporary repository and the final transport.
The MOWA drums (volume of about 200 l) are the final waste of the conditioning process of liquid radioactive waste (the resultant sludge from the reduction treatments of the liquid waste) mixed with cement mortar. The OVERPACK drums (volume of about 380 l) are the final result of the conditioning process of the solid radioactive wastes that are compressed and stored in packages of about 220 l of volume, or low activity sources in a concrete matrix.

Figure 1: Mowa and Overpack drums

The second section of this work proposes a shielding and containment system for gamma high-activity sources, one of the several steps to analyse the average cost to take care of disused sources as fixed by Italian law implementing the Council Directive 2003/122/Euratom . The final package could be produced in the future only when all the authorizations and approval will be obtained by the different competent authorities.

1. TRANSPORT AND CONDITIONING QUALIFICATION FOR WASTE

The final product, obtained by cementation of solid, liquid and sludgy low radiation waste in treatment plants, can be considered as a LSA-II materials (Low Specific Activity material). The final radioactive materials are solid objects in a compact matrix (heterogeneous waste form) or is uniformly distributed in case of conditioned liquid waste (homogeneous waste form). The average specific activity of the solid final waste form does not exceed 10^{-4} A2 / g and the type of packaging must be an IP-2, in according to IAEA Safety Standards Series No. TS-R-1.

1.1 Conditioning Processes

In NUCLECO plants, the conditioning processes are divided into liquids or sludge treatment and solids treatments.
The conditioning process of the solid radioactive waste includes the volume reduction and the heterogeneous immobilization of the waste in a concrete matrix to preserve the waste form for a long period of time. The system includes the following steps: introduction of 220 litre characterized drums, dose and weight control, short storage; super-compaction of 220 litre drums, dimensional check; overpack filling with compacted drums and with concrete matrix; weight, external dose and contamination controls; overpack labelling and storage.
The liquid conditioned waste is composed by liquid and sludge produced by the liquid treatment plant. The waste solidification is made by mixing the sludge with the cement inside a 220 liters drum equipped with an agitator (MOWA). The conditioning of the sludge is done through the

following steps: connection of the MOWA drum to the adduction pipes in the cementing box; sludge and water in defined weights transferred in MOWA and agitator start; concrete mixing to a perfect homogenisation; stop mixer, supply pipes disconnection and package removal; package weighing and labelling.

1.2 Waste Form

The qualification referring to "the waste form" includes compression test, thermal cycling test, radiation resistance test, biodegradation resistance test, immersion test, water permeability, free liquids test.

Table 1: Waste form tests

TEST	ACCEPTABILITY	TEST RESULT
Compression test	$R_c \geq 5$ N/mm^2	$R_c = 60$ N/mm^2
Thermal cycling test	No cracks or damaged surfaces $R_c \geq 5$ N/mm^2	No cracks or damaged surfaces at the end of the thermal cycling test $R_c = 62$ N/mm^2
Radiation resistance test (10^6 Gy)	No cracks or damaged surfaces $R_c \geq 5$ N/mm^2	No cracks or damaged surfaces at the end of the irradiation exposure $R_c = 74$ N/mm^2
High temperature resistance test	Absence of structural failure or deformation	The specimens show no structural failure and are stackable
Water permeability test		Total impermeability
Free liquids test	No liquid inclusions	Total absence of free liquids and uniform distribution of concrete in the voids
Biodegradation resistance test	No cracks or $R_c \geq 5$ N/mm^2	No cracks or damaged surfaces R_c (bacteria) = 64 N/mm^2 R_c (mold) = 60 N/mm^2
Immersion resistance test	No bumps or damaged surfaces $R_c \geq 5$ N/mm^2	No bumps or damaged surfaces at the end of immersion $R_c = 67$ N/mm^2
Gas generation test (Radiolysis 10^6 Gy)		Presence of hydrogen with magnesium waste

Waste form tests must be performed whenever the mortar formula or the liquids/sludges physical property is changed.

1.3 Packaging Qualifications

The preliminary checks are: identification of the steel sheets; dimensional checks; welds inspection; leaking control by pressurizing air; leaking control by immersion of the empty packaging in water; coating cycle control and film thickness; functionality check (MOWA agitator).
The leaking control is done on 10% of packaging for each supply. If bubbling or loss of pressure is detected, the drum is repaired and the check is extended on more elements.

The coating treatment includes a pre-treatment with nano-technology, application of zinc oxide and polyester coating for a thickness of 200 micron.
Accelerated corrosion test in neutral salt spray (NSS) fog chamber is a process of artificial aging used to evaluate the corrosion resistance of coated steel sheets with the same characteristics used in the manufacture of packagings.
To evaluate the duration of the test in salt spray, in the absence of specific reference standards, it is considered a corrosive and polluted environment; and it is chosen the parameter of durability "high", corresponding to a period of resistance to corrosion more than 15 years (maximum limit of regulations). The resulting period of the test is equal to 720 hours. Test demonstrates that the real environmental conditions are better than the considered ones; in fact, the 30 years old packages of NUCLECO storage has not the same corrosion of the NSS specimens.
The European Standard EN ISO 4628 defines a system for designating the quantity and size of defects and the intensity of changes in appearance of coatings. This system is used for defects caused by ageing and weathering.
The quantity of defects in the form of discontinuities or other local imperfections in the coatings, scattered over the test area is designed in accordance with a rating from 0 (no detectable defects) to 5 (dense pattern of defects). The average size (magnitude) of defects is designated from 0 (not visible under x 10 magnification) to 5 (larger than 5 mm).

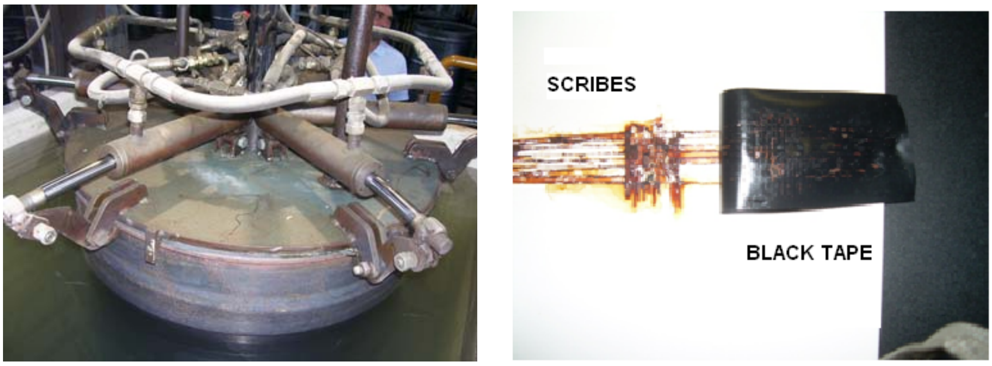


Figure 2: **Leaking test and specimen after NSS**

Table 2: **Corrosion analysis**

ANALYSIS	NOTES	ANALYSIS ASSESSMENT	ISO 4628 ASSESSMENT
Blistering	No evidence of blistering Density=0, Size=0	**< 2(S2)**	**3(S3)**
Rusting	Edge rusting is allowed	**Ri 0**	**Ri 1**
Cracking	No evidence of cracking	**0(S0)b**	**2(S3)b**
Flaking	No flaking up to 1 mm	**< 1(S1)a**	**2(S2)a**
Chalking by tape method (Fig. 2)	No evidence of chalking (absence of coating on tape)	**Rate 0,0**	**Rate 1,0**
Delamination and corrosion on the scribe	Absence of delamination around a scribe	**Grade 1** Very slight	**Grade 2** slight
Filiform corrosion	Absence of filiform corrosion around a scribe	**L1/M1**	**L3/M2**
Other notes	Corrosion in the scribes and the edges is allowed		

1.4 Package

A Type IP-2 is required for our present waste, but in the future we would use the same packaging as Type IP-3 for LSA-III too, so we performed the free drop test the stacking test and the penetration test. The free drop tests onto a flat target, horizontal surface with a steel thickness of 40 mm, demonstrate that there are no evidence of opening and loss of simulate radioactive contents. Several falls with different orientation (60°, 26°, 54°, 48°) assured us that we have found the most damaging free drop.

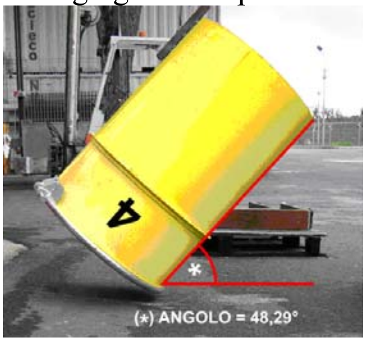



Figure 3: **Penetration and free drop tests**

The specimens are not deformed by the load in stacking test.
The bar of the penetration test doesn't break the drum in every impact point, even close to closure system.

2. SPENT HIGH ACTIVITY SOURCES CONDITIONING.

The aim of this study is to design a shielding and containment system for spent high activity sources, limited to ^{60}Co and ^{137}Cs. This it is the first step to fix the cost from collecting to the final disposal for the user so that he can know how much will pay to get rid of a source at the end of use.

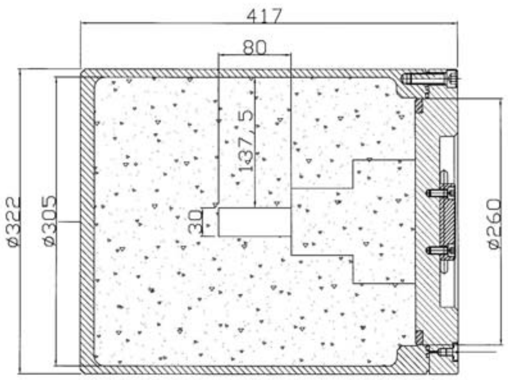
2.1 Description of shielding and containment systems for spent high activity sources

2.1.1 Lead containment system.

The choice of a shielding materials was limited by the dimension and allowed mass content of the external component and by cost: Therefore we started with lead and then we check the tungsten, the material that now is substituting depleted uranium. A first design of the containment system for gamma spent high activity sources appropriate to transport and containment for disposal is shown in Figure 4.
The size of the inner cavity for the source, was taken from the sources collected in the past, some used in Italy at the present and from the IAEA International Catalogue of Sealed Radioactive Sources and Devices (Sources Catalogue). The container is totally realized in lead surrounded by a AISI 304 stainless steel liner with thickness of 17 mm.

2.1.2 Lead - Tungsten containment system.

An alternative design of containment system for disposal of gamma spent high activity sources is shown in Figure 5.
This system is similar to the previous for external and internal dimensions of the cavity, but the new containment system is constituted by a primary shield realized in tungsten, followed by a secondary shield of lead. The choice of tungsten material as an alternative to lead is due mainly to the increased shielding obtained with the same thickness, even if the cost of the system increases.



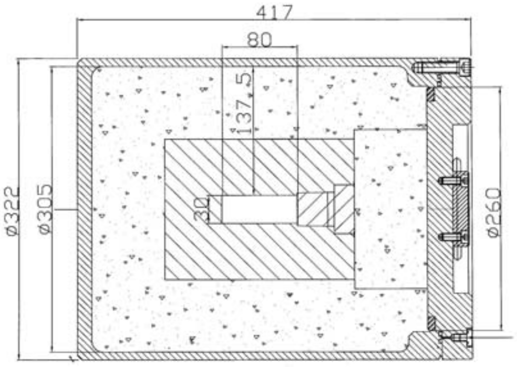


Figure 4: Lead containment system section **Figure 5: Lead - Tungsten containment section.**

3. SHIELDING CALCULATION.

The containment systems, lead or lead-tungsten, were inserted in an another component to form a package, named CF66, approved in the past as Type B (U) by the National Competent Authority (ISPRA) for the transport of nuclear material, shown in Figure 6. It consists of a steel structure surrounded by an absorption system made of a vermiculite layer contained in a steel shell.

3.1 Lead containment system.

The objective of this study was to evaluate the maximum activity, expressed in TBq, allowed by the shielding thickness of lead containment system.

The design parameter of the calculations was to have, on the outer surface of the shielding an equivalent dose rate of **2 mSv/h**, (dose rate allowed by international standards for the transport of nuclear material). The evaluation of the maximum activity, for a fixed thickness of lead shielding, for ^{60}Co and ^{137}Cs was performed by using the computer code **MCNP5 (Monte Carlo N-Particle** Transport Code), developed at Los Alamos National Laboratory (New Mexico, USA), a transport code for neutrons, photons and electrons using Monte Carlo method.

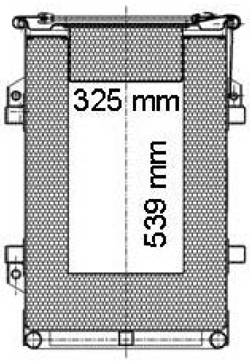


Figure 6: CF66 mechanical protection and thermal shielding system.

The maximum activity for ^{60}Co sources is reported in Table 3.

Table 3: Equivalent Dose Rate for ^{60}Co source placed in a lead containment system inserted in CF66 container.

Dose contact CF66: **2 mSv/h** →	Activity allowed in lead container: **19.30 TBq = 521.62 Ci**
Dose contact lead lateral surface	8.86 mSv/hr
Dose contact lead upper surface	0.97 mSv/h
Dose contact lead lower surface	0.47 mSv/h
Dose upper surface CF66	0.185 mSv/h

Dose lower surface CF66	0.118 mSv/h
Dose at 1 m from CF66	0.14 mSv/h

The maximum activity allowed by shielding thickness of the lead containment system for ^{137}Cs high activity sources is shown in Table 4. The same table shows the values of equivalent dose rate (mSv/h) at the lateral surface of the lead container and the value of equivalent dose rate at 1 meter from CF66, calculated for maximum activity.

Table 4: Equivalent Dose rate for ^{137}Cs source placed in a lead containment system inserted in CF66 container.

Dose contact CF66: **2 mSv/h** →	Activity allowed in lead container: **173529.8 TBq = 4690 kCi**
Dose contact lead lateral surface	8.54 mSv/hr
Dose at 1 m from CF66	0.13 mSv/h

3.2 Lead - Tungsten containment system.

The evaluation of the maximum thickness of the lead–tungsten containment system for ^{60}Co and ^{137}Cs high activity sources was performed using the computer code **MCNP**.
The design parameter of the calculations was to have, on the outer surface of the container shielding CF66, a dose rate equivalent of **2 mSv/h**.
The maximum activity allowed by shielding thickness of the lead-tungsten containment system for ^{60}Co high activity sources is reported in Table 5.

Table 5: Equivalent Dose rate for ^{60}Co source placed in a lead-tungsten containment system inserted in CF66 container.

Dose contact CF66: **2 mSv/h** →	Activity allowed in lead container: **183.2 TBq = 4951.3 Ci**
Dose contact Pb-W lateral surface	8.90 mSv/hr
Dose contact Pb-W upper surface	0.85 mSv/h
Dose contact Pb-W lower surface	0.41 mSv/h
Dose upper surface CF66	0.16 mSv/h
Dose lower surface CF66	0.12 mSv/h
Dose at 1 m from CF66	0.14 mSv/h

The maximum activity allowed by shielding thickness of the lead-tungsten containment system for ^{137}Cs high activity sources of is shown in Table 6.

Table 6: Equivalent Dose rate for ^{137}Cs source placed in a lead-tungsten containment system inserted in CF66 container.

Dose contact CF66: **2 mSv/h** →	Activity allowed in Pb-W container: **3991936 TBq = 107890 kCi**
Dose contact Pb-W lateral surface	6.51 mSv/hr
Dose at 1 m from CF66	0.15 mSv/h

At the present we do not have ^{137}Cs sources with an activity of $4\ 10^6$ TBq so the decay heat was not taken into account.

CONCLUSION

Conditioning of solid or liquid wastes, ready to be transported even in future, is an activity started in the past when the final disposal facility was foreseen in few years. At the present, 30 years later, the planning for the final disposal is the same but with higher probability to be implemented. In any case we are taking into account a long interim storage: a nuclear site is a difficult political decision.
High activity sources represent a real challenge since the present legislation asks the user to guarantee money for the final disposal but we are not planning a final disposal for high activity wastes and there is no International experience to estimate the cost for disposal and often it is impossible to sent back the source to the supplier.
ENEA and NUCLECO have found the package for the disposal and are still working on transport and disassemble old irradiators. All the job performed needs an agreement with many authorities before an estimated cost will be a real expenditure for the users of high activity sources.

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