EL CABRIL LOW AND INTERMEDIATE LEVEL WASTE DISPOSAL FACILITY (SPAIN): NEW ACCEPTANCE CRITERIA

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ABSTRACT

This paper describes the main topics of the Cabril Disposal Facility (Spain) in operation since 1992. New acceptance criteria approach, determination of activity, control of the acceptance methodology, characterization methodology and the new project for disposal of very low level wastes are also described.

BASIC DESCRIPTION OF THE DESIGN OF THE EL CABRIL DISPOSAL FACILITY

The primary waste packages, for the most part 0.22 m^3 drums, are located inside concrete containers. The drums are immobilised inside these containers, constituting a compact block weighing some 24 tons and measuring externally 2.25x2.25x2.20 metres. This assembly is known as the Disposal Unit.

There are other configurations that may be disposed of directly at the Facility, following their acceptance by the Regulatory Authority, on a case by case basis, in response to proposals by ENRESA.

These containers are disposed of by piling them in the cells. Each of these has a capacity for 320 containers of the type described above and approximate external dimensions of 24x19x10 m. The containers are placed in such a manner that they are in contact with one another, except for a central cross or strip that is left to absorb container manufacturing or emplacement tolerances.

Both the containers and the disposal cells are designed to withstand extreme demands, including an earthquake implying an acceleration in the ground of 0.24 g. The definition of the concrete used for the cells and containers has been the subject of a research programme carried out by the Eduardo Torroja Institute (Spain), the objective of which was to optimise the durability of the concrete barriers. The concrete used has a high mechanical strength (48 MPa), compactness and resistance to sulphates.

On completion of the operational phase of each cell, the central strip is filled with gravel in order to make the assembly rigid and backfill any voids and an upper closure slab is constructed. Subsequently the structure is weatherproofed by means of a synthetic layer.

The lower slab is the main element of the disposal cell. It has a thickness of 0.6 m at the edges and 0.5 m in the centre, thus forming a slope, and is covered with an impermeable layer of polyurethane and a 10-20 cm-thick layer of porous concrete, providing a horizontal surface for emplacement of the containers. The function of this slab is to collect any water that might seep in and to channel it to a network of pipes installed in inspection galleries located beneath the disposal structures. Each structure is connected to this network, known as the Seepage Control Network, via a holding tank, such that if water is collected in the control network it is possible to determine from which structure it comes and then undertake repairs to the protective layer and take samples of the water.

This passively operating network of pipes flows into a final control tank which has a capacity for the collections corresponding one year, taking into account the rated seepage of the cover plus that which would occur in the event of its collapsing. This makes it possible to maintain adequate surveillance of the operation of the disposal system, detecting and determining the origin of abnormal quantities of infiltrating water and its possible contamination.

During the operating phase, and with the triple objective of protecting the waste containers against the weather, minimising the quantity of water collected in the seepage control network and supporting the container handling resources, each row of disposal structures is covered by a mobile roof mounted on rails. This auxiliary roof is positioned above the disposal structure currently in operation. Following the weatherproofing of this structure, the roof is moved on to the adjacent structure.

On completion of the operation of the installation, the overall assembly will be covered with a low permeability coating made up of alternating layers of impermeable and draining materials which will provide long-term protection for the containers with the waste packages and ensure their suitable durability.

This defines the multiple barrier system, made up of three barriers:

- The first is formed by the waste immobilisation matrix, which, even in the event of the packages being submerged in water, ensures that the radionuclide release rate would be minimised by the isolating container and immobilisation mortar.

- The second consists of the disposal structures, the cover and the seepage control network, limits the access of water to the waste packages and allows for the control, and treatment where necessary, of whatever water might have come into contact with them.

- The third is the geological barrier, which is made up of the surrounding terrain and which would limit the impact of an eventual release in the event of an accident or of the hypotheses of complete degradation of the two first barriers, adopted for the free use stage.

Given that the containers are not immobilised in the cells, the disposal system would allow any disposal unit to be retrieved, if this were necessary for the safety of the facility, during the period of operation or institutional surveillance.

ACCEPTANCE CRITERIA FOR CONDITIONED WASTES AT THE EL CABRIL FACILITY

General

The waste acceptance criteria were included in the documents making up the study for licensing of the facility and drawn up by ENRESA for approval by the Regulatory Authority. Any revision of these criteria was subject to approval by the said authority.

Summarised below are the basic aspects of the criteria that were in force from the start-up of the facility (1992) to the year 2003, when the new criteria were approved. The important changes that have taken place are described.

Acceptance criteria during the initial operating phase of the facility (1992–2003)

The most significant aspects of these criteria are listed below:

- They were applicable to primary packages as regards both activity limits, dose rate and the mechanical requirements of confinement and durability. Therefore, no credit was given to the existence of the container.

- The packages were classified on the basis of two criteria:

- Activity as of the date of conditioning (Levels 1 and 2). Table I
- Existence of ENRESA Specifications as of the date of generation (non-typified and typified)

- The methodology for the treatment of non-conformities was not included, in other words packages not complying with certain limits were excluded from the acceptance process.

Table I. Former Limits between Level 1 and 2

Total α	1.85 10^2 Bq/g
^{137}Cs	1.85 10^4 Bq/g
^{60}Co	1.85 10^4 Bq/g
^{3}H	7.4 10^3 Bq/g

Disposal container qualification process

The first objective mapped out by ENRESA in preparing the revision of the Acceptance Criteria was the study of the concrete container as a component of conditioning. As has been pointed out above, the primary function of this container was to provide the system with the capacity for recovery.

For this purpose a dossier of tests was prepared, the aim being to homologate the container to what in the French system developed by ANDRA is known as the "durable confining container", which is similar to the concept of the "High Integrity" container in the American approach.

The reference document selected by the Spanish Regulatory Authority was the French Fundamental Safety, RFS-III.2., in relation to the following requirements:

Mechanical and structural requirements

The container shall be capable of withstanding the combined actions due to permanent, variable and accidental loads deriving from the disposal system or facility.

For this purpose the mechanical and structural requirements to be met by the containers and the materials used in their manufacturing shall be determined.

The methodology to be used in performing the calculations shall be that established in the spanish structural concrete instruction.

In addition, the container shall be capable of withstanding a load of 0.35 MPa without presenting a vertical relative deformation of more than 3% in 24 hours.

Durability requirements

Requirements regarding resistance to thermal cycles

The material used in manufacturing the container shall be capable of withstanding a thermal cycle test (5 cycles between -20º and 5º C and 5 cycles between 5 and 40º C with water spray) without the following types of defects being observed:

- Appearance of a general system of fissures.
- Fissures of more than 0.3 mm in width.
- Initiation of local rupturing.

- Reduction of the resistance to flexotraction or indirect traction in excess of 20%, taking as a reference the values obtained from specimens not subjected to thermal cycles.

Requirements regarding resistance to irradiation

If the dose rate in contact with the internal wall of the container is equal to or higher than 500 mSv/h, the material used in manufacturing the container shall be capable of withstanding an irradiation resistance test, without the following types of defects being observed:

- Appearance of a general system of fissures.
- Fissures of more than 0.3 mm in width.
- Initiation of local rupturing.
- Swelling causing an increase of more than 5% in the volume of the specimen (prior to irradiation).
- Reduction of the resistance to flexotraction or indirect traction in excess of 20%, taking as a reference the values obtained from specimens not subjected to thermal cycles.

In the event of non-compliance with any of the criteria indicated above, a diffusion test will be performed on irradiated specimens with a view to verifying that the confinement capacity is maintained.

Requirements regarding resistance to aggressive chemical environments

A study shall be carried out by a specialist organisation to identify those processes that might cause significant degradation of the containers under disposal conditions.

Once the kinetics of these processes has been established, either by testing or by a study of the scientific-technical literature, the variables affecting durability shall be defined and the limit values of these variables shall be established.

By way of an example, and if carbonatation were the controlling process, it would be necessary to limit the coefficient of carbonatation of the concrete and the thickness of the concrete over the reinforcing elements.

Requirements regarding confinement capacity

The limits for the annual fraction of activity released derived from the French standard RFS III-2.e shall be applied to the container (Table 2).

Table II. Confinement Values FAR (*)

β-γ isotopes (except ^{3}H)	$7.3\ 10^{-3}$
^{3}H	$5.0\ 10^{-2}$

(*) Annual fraction of activity released

The values used in performing the calculations shall be those determined experimentally for the coefficients of diffusion of concrete and mortar. Compliance with the requirements applicable to coefficients of diffusion may be justified through the performance of diffusion tests for one radionuclide exclusively, the recommendation being that tritium or ^{137}Cs be used preferably.

Acceptance Criteria in force

Once the results obtained had been evaluated and Regulatory Authority had accepted that the concrete container not only provided the system with the capacity for recovery but also added an additional confinement capacity to the primary wastes themselves, ENRESA undertook the revision of the acceptance criteria.

This revision was performed with the objective of ensuring that, under the normal operating conditions of the conditioning systems of the different producers, the quality of the packages to be disposed of was not reduced and of using it as an element for the study only of non-conforming packages, giving credit under such conditions to the characteristics of mechanical strength, confinement capacity or durability provided by the container.

The most important aspects revised in the new criteria have been as follows:

- The activity limits apply to the overall assembly of the container and the waste packages immobilised in its interior, defining the Level 1 and Level 2 Disposal Unit (see tables III. and IV.)

Table III. Disposal Units Activity Limits for Level 1

Total Alpha	$1.85\ 10^{2}$ Bq/g
^{3}H	$7.40\ 10^{3}$ Bq/g
^{137}Cs	$3.70\ 10^{3}$ Bq/g
^{60}Co	$3.70\ 10^{3}$ Bq/g

Table IV. Disposal Units, Activity Limits for Level 2

Total Alpha	$3.70\ 10^{3}$ Bq/g
^{3}H	$1.00\ 10^{6}$ Bq/g
^{137}Cs	$3.30\ 10^{5}$ Bq/g
^{60}Co	$5.00\ 10^{7}$ Bq/g

The limits for primary packages are derived from this activity and from a series of factors of heterogeneity per 220-litre volume (see table V.).

$$A_{220} = \frac{A_{DU}^{\max} x 0,22}{V\,\mathrm{int}.(DU)} x\, K(\,factor) \qquad \text{(Eq. 1)}$$

Table V. Factor of Heterogeneity

Vol. of DU m^3	K
< 0,5	1
0,5 – 1	2
1 -5	3
> 5	4

- The new limits are higher for primary packages since in calculating specific activity it is possible to count as mass that due to the container and to the backfill mortar (in almost all configurations).

- The maximum dose rate limit is applicable to the overall disposal unit and not to the primary package. In practical terms the maximum dose rate per individual package has changed from 50 mSv/h to 100 mSv/h.

- The minimum wall thicknesses required by the acceptance criteria for packages containing filters, dispersible solids, etc. change from 10 cm to 5 cm, due to their being associated with the diffusion resistance of the material used.

- Primary packages not complying with the quality objectives established may be studied as non-conforming, with consideration given to the contribution of the container. These studies, which will cover operationally unforeseen situations or high activity historic packages (Level 2), will be subject to case-by-case approval by the Regulatory Authority.

DETERMINATION OF ACTIVITY

Assessment methods

Detailed below are different methods for assessment of the activity of low and intermediate level waste packages depending on the isotope to be determined and the nature of the waste.

Non-destructive analysis (NDA). Gamma Spectrometry

Used by the producer for measurement of both ^{60}Co and ^{137}Cs and other gamma emitters with significant levels of activity. This technique is applied fundamentally to the following:

- Samples of unconditioned liquid wastes. The activity of the final package is determined by the quantity of waste introduced in the package. A final correction for the Contact Dose Rate of each package is performed on all the packages making up the batch.
- This technique is applied only to packages of homogeneous moist wastes incorporated to cement matrixes.
- Determination on the whole package: there are currently various measurement techniques of this type. If the calibration of efficiencies is performed from a template, the essential difficulty is encountered in manufacturing the latter. If such calibration is carried out using statistical simulation methods (Montecarlo), the hypothesis of waste homogeneity is assumed.

This technique is applied fundamentally to homogeneous waste packages, although it may also be applied to other configurations with certain technical conditions.

Theoretical methods (TM)

Used by both the producer and ENRESA for assessment of gamma emitter activity.

The following generic expression is applied:

$$A = \frac{D}{\sum_{i=1}^{i=n} d_i \cdot f_i} \qquad \text{(Eq. 2)}$$

where:

A: Activity contained in the package (MBq/Package).
D: Average package dose rate (μGy/h).
n: Number of isotopes considered.
fi: Isotopic fraction of isotope i at so much per unit.
di: Activity-Contact dose rate conversion factor (μGy/h per MBq/Package).

For one same type of package, the Conversion Factor of an isotope will change only with its density, as a result of which if the isotopic value and the corresponding Conversion Factors for each isotope and density are available, it will be possible to calculate the activity of the package from its contact dose rate, through the use of shielding calculation codes.

The following expression is used to calculate the activity per individual nuclide:

$$A_i = f_i \ A \qquad \text{(Eq. 3)}$$

This technique is mainly applied to heterogeneous solid waste packages (compactable and non-compactable) and to solids immobilised by the wall.

Whole package gamma spectrometry is difficult to apply to these wastes and at most allows for qualitative determination.

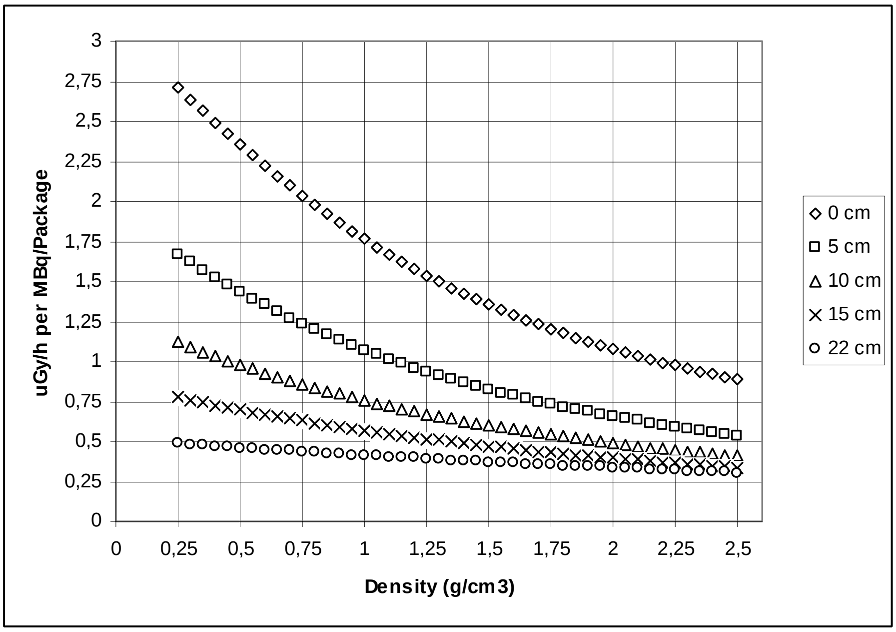

Fig. 1. Activity-Dose Rate conversion factors of ^{60}Co for different configurations.

Destructive analysis (DA). Radiochemistry

Used by ENRESA mainly for the measurement of pure Beta and Alpha emitters. This technique is applied fundamentally to:

- Samples of unconditioned wastes, used to create a database for application of the Scaling Factors methodology.

- Samples of dust from the "drilling" of conditioned waste packages.

Scaling Factors

Through the use of data compiled from radiochemical samples, correlations have been established between difficult and easy to measure isotopes, the aim being to establish a Scaling Factor to be applied a priori, from the measurement of the ^{60}Co or ^{137}Cs of a package. This technique is applied to practically all waste packages.

Nuclides considered

Two fundamental groups of isotopes (nuclides) to be determined are established:

- Evaluated by the producer, known as key nuclides:

  ^{60}Co, ^{137}Cs.

- Evaluated by ENRESA, difficult to measure isotopes:

  ^{3}H, ^{14}C, ^{59}Ni, ^{63}Ni, ^{94}Nb.
  ^{99}Tc, ^{90}Sr, ^{129}I.
  ^{238}Pu, ^{239}Pu, ^{241}Pu, ^{241}Am, ^{244}Cm.

There is the possibility of adding new isotopes of significance in subsequent Safety Assessment revisions.

Development of scaling factors

As has been indicated above, radiochemical techniques are required to assess the activity of isotopes of difficult to measure half lives.

Difficult to measure isotope Scaling Factors have been calculated for each waste stream and nuclear power plant, in accordance with the methodology currently approved by the Regulatory Authority.

This methodology is based on the apparent log-normal distribution of the concentrations of the isotopes in the waste streams. If the difficult to measure isotope correlates with the easily measured isotope, there is a linear relation between them and a Scaling Factor (SF) may be established, as shown in Figure 2.

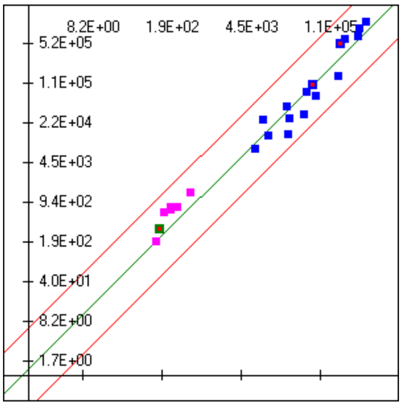

Fig. 2. $^{63}Ni/^{60}Co$ (Bq/g)

If, on the contrary, it is not possible to establish any relations between the difficult and easy to measure isotopes, the activity value proposed for the difficult to measure isotope in the package will be the mean activity concentration (MAC).

Statistical tests are used with a view to verifying the existence or otherwise of a relationship between correlated isotopes, for use of the Scaling Factor or Mean Activity Concentration.

$$LnY = a + bLnX \;;\; if\ b = 1\ statistica\ l\ test \Rightarrow$$

$$\Rightarrow a = \frac{1}{n}\sum_{i=1}^{i=n} Ln\frac{Y_i}{X_i};\ SF = antiLn\,(a) \qquad \text{(Eq. 4)}$$

$$SF = antiLn\left(\frac{1}{n}\sum_{i=1}^{i=n} Ln\frac{Y_i}{X_i}\right) = \left(\prod_{i=1}^{i=n}\frac{Y_i}{X_i}\right)^{\frac{1}{n}}$$

The Factor used (SF or MAC) corresponds to the geometric mean of the data used for calculation, which in addition to being a consistent estimator is a good approximation to the median, thus minimising the variation of individual data.

If the number of statistically established data available for a waste stream is insufficient (at least 15), the data should be linked to others on the same plant, using statistical stream clustering tests. If this test were unsuccessful, data on streams from other Nuclear Power Plants having similar characteristics should be used, and the clustering tests also applied. The final objective is to gradually cancel out clustered data as data on the stream under study become available.

APPLICABLE CONTROLS

These are a set of laboratory tests and checks on waste packages with an approved acceptance process, using non-destructive and destructive techniques for the determinations requested. The objectives to be met by the EVT's are as follows:

- Checking of package activity values against those declared by the producers.
- Compliance with the properties of the package associated with the generation methodology.
- Chemical checking of aspects of significance for the safety of the disposal facility (compatibility with the container, corrosion, gas generation, etc.).
- Compliance with objectives relating to the quality of the conditioned wastes.

The aspects relating to the quality of the characteristics of the conditioning undergone by the wastes have focused initially on packages of homogeneous moist wastes incorporated in a solid hydraulic conglomerant matrix.

A distinction is made between three types of EVT's (Figure 3):

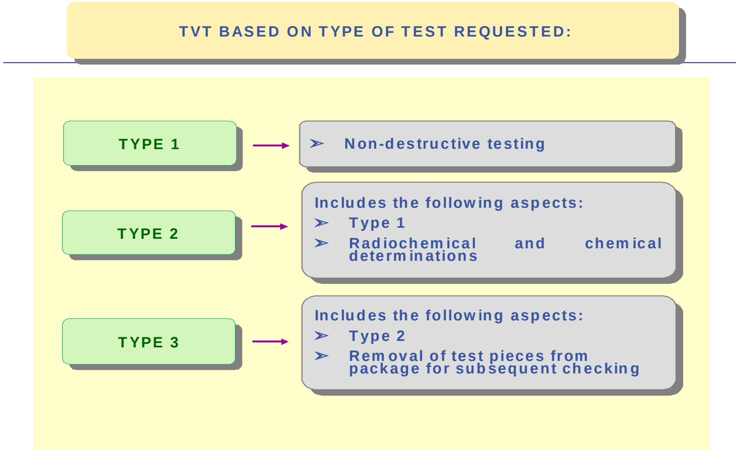

Fig. 3. Types of TVT

Type 1 EVT

The main objective is the verification of the physical and radiological parameters of the package with respect to what is declared by the producers.

a) Physical checks

- X-rays.
- Dimensional control.
- Weighing of the package.
- Visual and photographic inspection of the structural status of the packaging.

b) Radiological checks

- Determination of dose rate.
- Complete and segmented spectrometry.
- Determination of package homogeneity.

Type 2 EVT

In addition to what is indicated above for type 1, the type 2 EVT's are aimed at obtaining solid samples, using destructive techniques, for subsequent chemical and radiochemical checks. The tests additional to those performed for type 1 EVT's are as follows:

- Determination of weak β emitters
- Determination of α emitters
- Chemical determinations

Type 3 EVT

In addition to what is indicated above for types 1 and 2, the type 3 EVT's are aimed at obtaining specimens, using destructive package trepanation techniques, for subsequent mechanical checks. The tests additional to those performed for type 1 and 2 EVT's are as follows:

- Checking for the absence of free liquid
- Checking for correct setting
- Mechanical matrix resistance tests.

Figure 4 summarises the tests contemplated for each type of EVT.

The following tests are considered, depending on the type of TVT:

TESTS	TVT		
	Type 1	Type 2	Type 3
Visual and photographic inspection	×	×	×
Determination of dose rate	×	×	×
Determination of package weight	×	×	×
Checking of packaging	×	×	×
Dimensional control	×	×	×
Gamma spectrometry	×	×	×
Degree of homogeneity	×	×	×
Absence of free liquid	—	—	×
Correct setting	—	—	×
Radiochemical determinations	—	×	×
Chemical determinations	—	×	×
Mechanical tests	—	—	×
Other determinations	—	Eventual	Eventual

Fig. 4. Summary: tests contemplated for each type of EVT

CHARACTERISTION LABORATORY

The characterisation laboratory has been designed to the specific needs of the Spanish radioactive waste management programme. It is the main tool to perform the waste acceptance process, and to verify, in a random basis, the characteristics of the packages being actually delivered to ENRESA.

The laboratory has two buildings: the so-called “inactive laboratory”, where tests on simulated waste samples are carried out; and the “active laboratory”, with capability for testing real packages and radioactive samples.

The main equipment in the inactive building are the climatic chamber for preparation of the thermal cycling tests and the mechanical test equipment

The active building of the laboratory has been designed around a sample preparation cell. This hot cell houses:

- The drilling machine fit to drill out in a dry way, i.e. not disturbing the samples with the cooling water, cylindrical samples from the real radioactive packages).
- The saw (to cut the cylindrical samples to standard size specimens)
- The mechanical test equipment (a 50 t press and ancillary equipment)
- The cutting machine (to cut out the metallic skin of the packages).
- Handling equipment, including:

  – Two master-slave 200 kN tele-manipulators
  – Two leaded glass windows
  – One 3 t overhead crane, which can support:

    ⇒ One auxiliary tele-manipulator with 1000 kN capacity.

    ⇒ One turning-over grip for 0,22 or 0,48 m3 drums, to move and change from vertical to horizontal position the packages.

In addition, it has to be mentioned:

The operations room, from which the tele-manipulators are handled, and which has the control panels and the aliquot preparation cell.

The leaching tests room, with leaching tanks for the above-mentioned drilled out samples as well as for “unskinned real packages”.

The spectrometry assay room, providing means to carry out a complete beta-gamma spectrometry on homogeneous or heterogeneous 220 or 480 liters real packages or on standard size specimens drilled out from the packages. The main piece of equipment there is a segmented scanning system with a Germanium detector and two dose rate measuring devices together with the elevating turntable, collimator and supports necessary.

The laboratory is completed with classical radiochemical sample preparation and measurement rooms ancillary equipment room and ventilation systems to guarantee the safe and reliable performance of the laboratory.

OTHER DEVELOPMENTS. VERY LOW LEVEL WASTE DISPOSAL

General definition of very low level wastes

Solid or solidified materials that are contaminated or activated and whose radioactive content does not exceed the limit values determined may be considered very low level radioactive wastes (VLLW). These wastes are a sub-classification of low and intermediate level wastes and in general present specific activities of between 1 and 100 Becquerels per gram, although these may increase to several thousands in the case of certain radionuclides of low toxicity or existing in small quantities.

Table 6 shows the maximum provisional values of activity for each radionuclide, for the pre-classification of a radioactive waste as being of very low activity.

Characteristics

Radioactive wastes which, because of their physical and chemical characteristics and the process of conditioning to which they are subjected, may be considered very low level wastes may be classified in two groups:

a) Radioactive wastes that do not present hazardous characteristics other than those associated with the presence of radioactivity. These may in turn be divided into two sub-groups:

   - Inert wastes: by their very nature these do not evolve significantly with time and consist, for example, of earths, rubble from construction materials, dust from the cleaning of concrete walls, etc.

   - Wastes associated with non-hazardous industrial activities: these may evolve slowly with time and their release of contaminants to the atmosphere is very low. If these wastes were not radioactive they would be classified as non-hazardous industrial wastes.

b) Radioactive wastes that, in addition to the hazards associated with the presence of radioactivity, have other hazardous characteristics. In this respect they are comparable to wastes classified as hazardous.

Activity limit

When there is only one radionuclide present, the specific activity will be such that:

$$ai < Ai\max \qquad \text{(Eq. 5)}$$

Where Ai max is the maximum specific activity per unit of VLLW, for each radionuclide, as defined in table VI.

In the case of wastes in which the activity is distributed in basically a homogeneous manner, the specific activity may be calculated by averaging out through the entire mass of wastes, including where appropriate that of the packaging and backfill or immobilisation materials.

In the case of radionuclide mixtures, the following should be obtained:

$$\sum ai / Ai \max < 1 \qquad \text{(Eq. 6)}$$

Table VI. Values Proposed for VLLW Maximum Specific Activity

Nuclide	Maximum specific activity per D.U. (Bq/g)
^{60}Co	100
^{94}Nb	100
^{137}Cs	300
^{238}Pu	100
^{239}Pu	100
^{241}Am	100
^{244}Cm	100

CONCLUSIONS

These are the achievements reached by ENRESA for managing low and intermediate radioactive wastes in Spain. However, some gaps are still existing which need further developments, such as: disposal of spent sealed sources with nuclides between ^{60}Co and ^{137}Cs; management of big components (such as steam generators, reactor pressure vessel heads, etc.); and improvements of techniques for activity determination, specially the definition of uncertainties.