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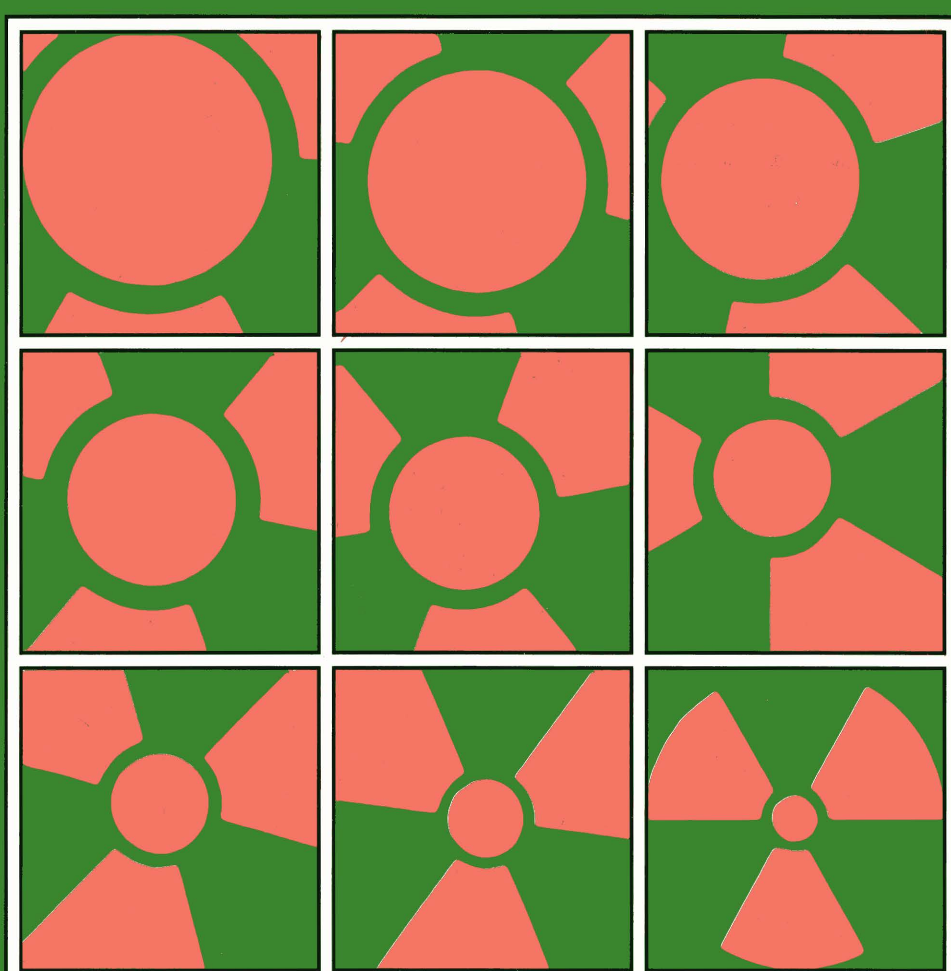

Report

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(Volume 7)

Cost and radiological impact associated with near-surface disposal of reactor waste (French concept)

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Final report

Work performed as part of the shared cost programme (1985-89) on management and disposal of radioactive waste of the European Communities

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FOREWORD

This report deals with the determination of the cost and the radiological impact associated to a near surface disposal of reactor waste based to a large extent on French practices in this field. This study is part of an overall assessment study aiming at evaluating a selection of management routes for LWR waste based on economical and radiological criteria.

Actually the assessment study was implemented through complementary contributions provided by nine organisations and companies, i.e.
CEN - Fontenay-aux-Roses, INITEC - Madrid, KAH - Heidelberg, BELGATOM - Brussels, TASK R&S - Ispra, SGN - St. Quentin-en-Yvelines, EDF/SEPTEN - Villeurbanne, FRAMATOME - Paris-la-Défense, GNS - Essen, co-ordinated by the Commission of the European Communities (Brussels).

The main achievements of the assessment study have been summarised by BELGATOM-Brussels.
These different contributions are published as EUR Reports in 1992 (listed as below):

VOLUME N°	MAIN AUTHORS	ORGANISATION	TITLE	EUR REPORT N°
1	R. Glibert	BELGATOM	Assessment of Management Alternatives for LWR Wastes : Main achievements of the joint study	14043 EN/Vol 1
2	E. de Saulieu C. Chary	SGN EDF	Assessment of Management Alternatives for LWR Wastes : Description of a French scenario for PWR waste	14043 EN/Vol 2
3	S. Santraille K. Janberg H. Geiser	FRAMATOME - GNS	Assessment of Management Alternatives for LWR Wastes : Description of German scenarios for PWR and BWR wastes	14043 EN/Vol 3
4	J. Crustin R. Glibert	BELGATOM	Assessment of Management Alternatives for LWR Wastes : Description of a Belgian scenario for PWR waste	14043 EN/Vol 4
5	B. Centner	BELGATOM	Assessment of Management Alternatives for LWR Wastes : Assessment of the radiological impact to the public resulting from discharges of radioactive effluents	14043 EN/Vol 5
6	G.M. Thiels S. Kowa	TASK R & S KAH	Assessment of Management Alternatives for LWR Wastes : Cost determination of the LWR waste management routes (Treatment/Conditioning/Packaging/ Transport Operations)	14043 EN/Vol 6
7	J. Malherbe	CEA	Assessment of Management Alternatives for LWR Wastes : Cost and radiological impact associated to near surface disposal of reactor waste (French concept)	14043 EN/Vol 7
8	N. Sanchez-Delgado	INITEC	Assessment of Management Alternatives for LWR Wastes : Cost and radiological impact associated to near surface disposal of reactor waste (Spanish concept)	14043 EN/Vol 8

C O N T E N T

1. INTRODUCTION :

In this present report are presented studies done in the frame of the third EC Programme on radioactive waste management and disposal. Particularly, this work is related to the management of radioactive waste of nuclear reactors.

The storage has been sized and evaluated on the basis of a park of 22 PWRs of 900 MWe. This park of 20 GWe is assumed to operate for 30 years. However, in France, the Near Surface Storage is used also for storing technological reprocessing waste of low level activity. That has led to dimension the capacity of the site for receiving both types of waste, reactor and low level activity technological reprocessing waste; but the radiological impact will be evaluated, in this report, for reactor waste only.

In this study, option considered for disposal of reactor wastes is the near surface disposal which corresponds to the French choice for disposal of low or medium level and short or medium lived waste, commonly named as LLW. Then main characteristics of the disposal facility considered are those of the Centre de Stockage de l'Aube, commonly named as CSA.

The reason is that this site has been chosen for geological and hydrogeological reasons and thus its characteristics approach the model ANDRA kept in mind to select a new site. An other reason is, as a new facility, the CSA allows ANDRA to use its last technology without any restriction and in a standardized manner. All these reasons allow to take into account standardized data in the study.

2. SAFETY PERFORMANCE OBJECTIVES AND TECHNICAL REQUIREMENTS FOR A LLW DISPOSAL FACILITY :

2.1. Safety performance objectives :

They are 2 in number. They are defined by the Fundamental Safety Rule 1.2 (F.S.R.) (Règle Fondamentale de Sûreté - RFS) promulgated by the Central Service for Nuclear Facilities Safety (CSNFS) (Service Central de Sûreté des Installations Nucléaires - SCSIN), a specialized service within the Ministry in charge of Industry (Ministère de l'Industrie, des PTT et du Tourisme) and which plays the part of the safety authorities in siting, licensing and inspecting nuclear facilities.

The First performance objective is to ensure the immediate and deferred protection of people and environment. The immediate protection occurs during the facility operation period, when the waste is being disposed of. The deferred protection concerns the institutional control period which extends from the closure of the facility to the moment the site is free of access. During this period, the release of radioactive from the disposal may present a radiological hazard.

Second performance objective : the institutional control period must not exceed 300 years, at the end of which the site is free of access.

The compliance of these two performance objectives leads to two safety design basis :

Containment of the radioactivity during the operational and institutional control period by the means of a multibarrier system which also prevents the adverse agents, mainly man and water to reach the radionuclides.

These barriers are three in number :

- the waste form including the physical form of the waste itself, the stabilizing or immobilizing material, the package and the possible overpack;

- the engineered features, including the disposal structures and the final disposal cap;

- the disposal site's natural characteristics, which can also act as a barrier, but only in the case of an accident.

The Second safety design basis is the limitation of the initial activity of radionuclides which are present in the waste packages so that, on one hand, during the operational and institutional control period, the radiological impact of disposal be acceptable in any circonstances, and, on the other hand, at the end of the institutional control period, the residual activity be compatible with free access.

Calculation shows that there are transfers of radioactivity by water which lead to the most stringent limitations for beta gamma emitters, i.e., for the quasitotality of short lived emitters which are present in the disposal. Taking into account each radionuclide decay law, it is possible to determine the maximum quantity of each radionuclide admissible in the disposal at the beginning of the institutional control period, so that free access may occur after the prescribed period of time.

Furthermore, in the transfer of radioactivity by air, these are the alpha emitters, which are present in small quantity in LLW and which activity will have decreased only a little within 300 years, which play an important part. Exhaustive surveys have shown that, whatsoever the site chosen for disposal, maximum specific activity limits must be applied for alpha emitters. They are :

- mean specific activity for the whole disposal lower than 0.01 alpha Ci/t

- specific activity for any package $<$ 0,1 Ci alpha/t,

- exceptionnally after specific review and agreement some packages with specific activity $<$ 0,5 Ci alpha/t may be accepted for disposal.

These requirements mixed with other technical requirements appear in the FSR I.2.

2.2. Technical requirements.
These technical requirements concern mainly :

- efficiency of the first two barriers until the end of the institutional control period (ICP),

- monitoring of barriers to the end of ICP,

- site selection criteria such as seismicity ans stability and hydrogeological characteristics,

- Quanlity Assurance Program for waste production processes and engineered features of the disposal facility.

Furthermore the FSR I.2. requires the determination of the nature and quantity of radionuclides to be disposed of. This must be carried out by classical radioactive element pathways analysis in which possible pathways for migration of radionuclides from the disposal facility to humans and environment is studied. Important inputs to the pathways analysis include the physical and chemical form of the waste, the distribution of the radionuclides in the waste and the process of immobilization of the waste.

As last, the FSR I.2. sets that the Preliminary Safety Analysis Report (PSAR) (Rapport Préliminaire de Sûreté - RPS) which is part of the technical report file submitted to the Safety Authorities for the construction permit (Décret d'Autorisation de Création) of a disposal facility, must, in particular, detail the safety measures taken to assure the intrinsic safety of the disposal during the operational and institutional control period and contain the surveys aimed to determine the nature and quantity of radionuclides to be accepted in the disposal.

From the results of the various Safety Authorities reviews come certain safety prescriptions which must be followed and which are particular to the disposal facility. These site specifications accompany the construction permit of the disposal facility.

Among them are waste acceptance criteria such as maximum specific activity limits for the packages to be disposed of at the facility.

3. FACILITY CONCEPTUAL DESIGN :

The waste containment is ensured by a set of provisions which prevent water to reach the waste in normal situation and which limit the quantity of radioactive substances carried away by water in case of accidental infiltration to a level weak enough so that radiological consequences are negligible.

3.1. The waste packages and the disposal modules

The packages and the engineered features must present an sufficient intrinsic safety for at least 300 years and for this purpose fulfill the following conditions :

- have a mechanical strength sufficient to ensure for 300 years a good steadiness of the disposal modules and the stability of the support on which the cap is laid,

- don't have a yearly leachable activity quantity (YLAQ) (Quantité d'activité lixiviable annuellement - QAL) greater than a fixed limit, for each radionuclide likely to be present. These YLAQ limits are the results of the radiological impact assessment of the disposal.

If the waste package offers by itself this intrinsic safety, it can be directed to a module where the waste packages are stacked on a pad (plateforme) to get, when the module is filled and closed, a tumulus. If the waste package does not bring alone this intrinsic safety, it will be directed to a module where the waste packages are disposed of in cell (alvéole) to get, when the module is filled and closed a monolith which will bring the complementary features to insure this intrinsic safety.

Practically, waste which present itself in the form of solid or solidified waste (ion exchange resins grains, chemical precipitates, evaporator concentrates) or in the form of various materials (tissues, scrap iron, plastics...) on which radioactive particles are fixed, are generally either stabilized or immobilized in a matrix (cement, bitumen, resin ...) inside either a concrete or a metallic container to constitute a package ready to be disposed of.

It is the combination of the waste, its stabilizing or iimmobilizing matrix and, under certain conditions, the waste container and the disposal module engineered features which allows to meet the safety requirements.

Practically, concrete blocks containing immobilized waste, as well as the drums and metallic boxes containing stabilized very low activity waste are disposed of in tumulus; other types of packages (immobilized in a perishable container or non immobilized waste) are generally disposed of in monoliths.

3.2. The disposal cap

The disposal modules once filled and covered are protected from rainwater by a cap which must be stable and impervious enough for at least 300 years.

Its maintenance must be as low as possible in normal situation until the end of the institutional control period.

It will be composed of several alternating impermeable and draining layers in order to prevent water from infiltrating the disposal module and to drain off any water that may have infiltrated before it can come in contact with the module.

3.3. The leachate collection system

The barriers described above are present to prevent water from infiltration the disposal system, contacting the waste, leaching out radionuclides and possibly transporting radionuclides into the environment. Since the ability to prevent water from entering the disposal system cannot be absolutely guaranteed, even though all measures will be taken to minimize the likelihood of occurrence, the facility conceptual design calls for the use of a leachate collection system (Réseau Séparatif Gravitaire Enterré - RSGE) that underlies and is integral to each disposal module.

Should water infiltrate the disposal module it is drained into a tank where it can be monitored for activity to determine if it has leached out radionuclides from the waste. Since it has been collected in this manner, the leachate can be processed and no leachate will be released in an uncontrolled fashion to the environment. The leachate collection system will also provide early indication of problems with the disposal cap. This will permit remedial action much sooner than would be possible from relying on visual inspection for indications of problems.

3.4. Disposal modules description

At the end of the operating period, once the buildings and facilities dismantled, only the disposal modules will continue to exist on the disposal facility. They consist of the following features :

- the raft :

First, the disposal consists of the juxtaposition of rafts, each one supporting a disposal module. These rafts are a few hundred square meters large, calculated so that the effects of a earthquake be minimized.

The raft is laid at such an altitude that the water table cannot reach the waste packages, even at its highest level.

- The tumulus :

A tumulus is a disposal module constituted by the stacking of packages on the raft playing the part of a platform.

The packages which are disposed of in tumulus must have a sufficient intrinsic safety with regard to the safety requirements defined above in paragrap 2. : long term stability and radionuclide containment.

Generally they are low medium level activity waste package immobilized in concrete containers, or very low level activity waste package stabilized in metallic containers.

The space between the packages is filled with gravel allowing a good stability while giving a free way to water, should water infiltrate the tumulus.

- The monolith

The low and medium level activity waste packages which do not offer by themselves a sufficient intrinsic safety with regard to the safety requirements cannot be disposed of in tumulus; the safety of the disposal of these packages is insured during the necessary period of time by strengthening the second barrier with the adding of concrete between them and environment.

In this prospect, the disposal module is constituted of a raft on which is laid the disposal structure consisting of concrete bottom and walls and forming an alveole.

The alveole containing the waste packages and the space filling material constitutes by itself a disposal module which is intrinsically safe with respect to the safety requirements and called a monolith.

If the packages are irradiating, the alveole walls and the space filling material of the monolith are determined so that they constitute an efficient biological shielding against external irradiation.

3.4.1. Platform/tumulus - Design basis

The disposal modules thus consists of a reinforced concrete pad (the platform raft or the bottom of the alveole) and four reinforced concrete walls supported on the pad. The pad is slopped from the outside to a drain in the center so that any water that may have infiltrated the module is directed to a leachate collection system. Waste packages are placed on the pad inside the walls, in 8 layers for the concrete blocks.

The modules are built in rows. A disposal unit consists of several rows of modules.

While in operation, the module is covered with a mobile Butler-type shelter which incorporates handling equipment such as jib cranes, lifting beams... The shelter covers the entire volume where disposal operations are being conducted, as well as part

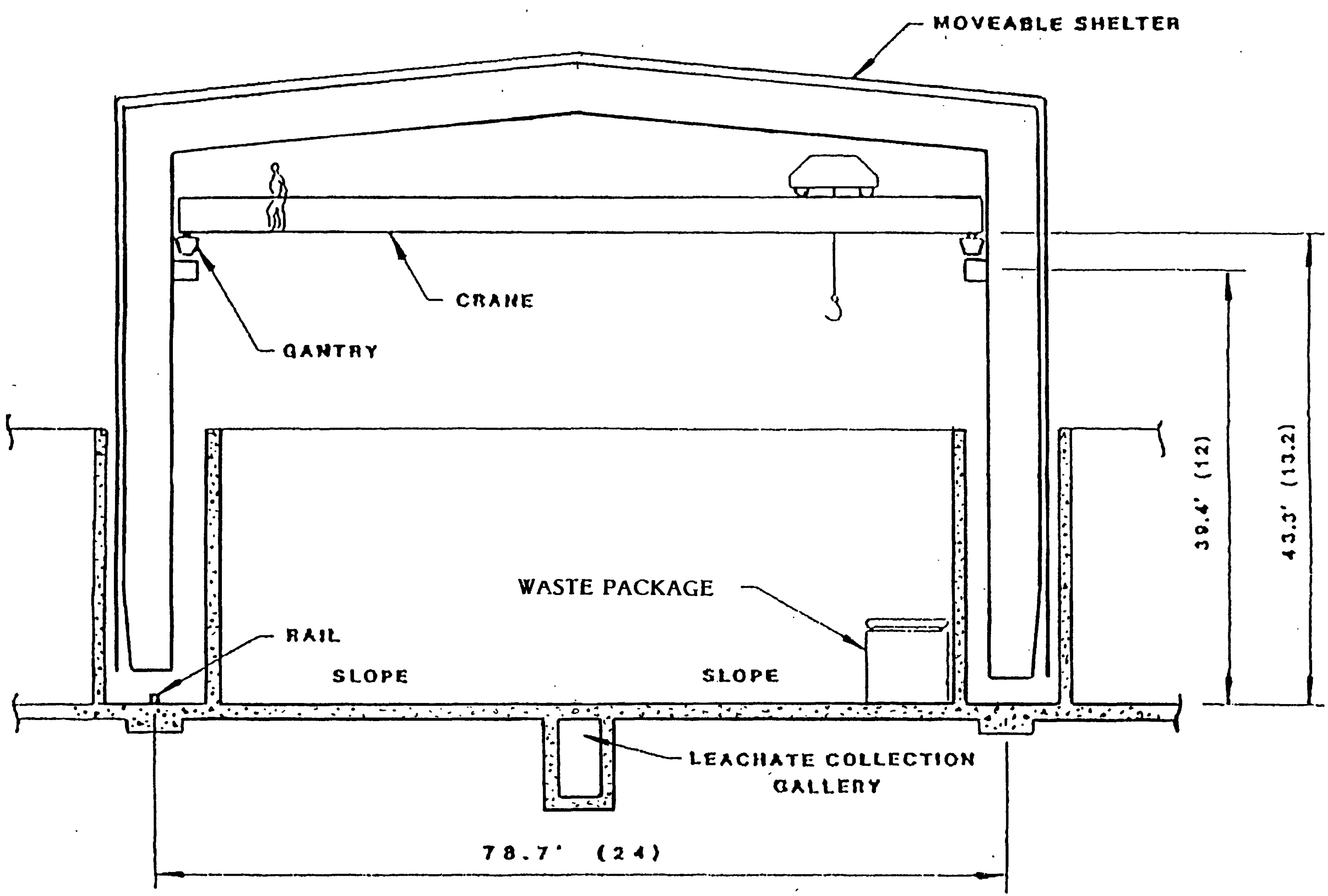

FIGURE 1 : DISPOSAL MODULE SIDE SECTION VIEW

of the adjacent module, where the packages are unloaded. Once the disposal module has been filled and the concrete roof poured in place, a temporary cover is put over the module and the shelter is moved to the next module. The shelter is moved from one module to the next on rails that are built into concrete tracks which extend the entire length of a row of modules.

The final unit cap is put over disposal every 5 years.

The disposal module is schematically shown in Figures 1, 2 and 3.

3.4.1.1. Concrete Pad, Walls and Leachate Collection System

The floor of the disposal module is a reinforced concrete pad measuring approximately 82 feet x 71 feet (25m x 21,5m). The thickness of the pad varies from about 20 inches (50cm) on the edges to about 16 inches (40 cm) in the middle. The portion of the pad that is located inside the module walls is covered with a waterproof material that rises to about 20 inches (50cm) on the module walls.

The walls of the module have been sized so as to provide sufficient shielding to limit the contact dose rate on the outside face of the wall as well as to suport the lateral pressure of the module contents (backfill gravel).

A 20 feet (6m) opening will be left in the wall of the next empty module to allow the waste packages to be brought in, unloaded and transferred into the module. This opening is filled in once the module has been filled.

Water that may have infiltrated the disposal module is directed toward a central drain by the 1% slope of the concrete pad. The drainpipe goes to a monitoring tank in an underground gallery that can be accessed by operating personnel.

3.4.1.2. Mobile shelter

The main purpose of the mobile shelter is to protect the waste from rainwater during handling and disposal operations and until the temporary cover disposal cap is in place.

The shelter measures approximately 131' L x 85' W x 52' H (40m x 26m x 16m). It sits on four articulated wheels assemblies which are used to move the shelter along rails from one module to the rainwater collection channels that direct the water to a retention pond.

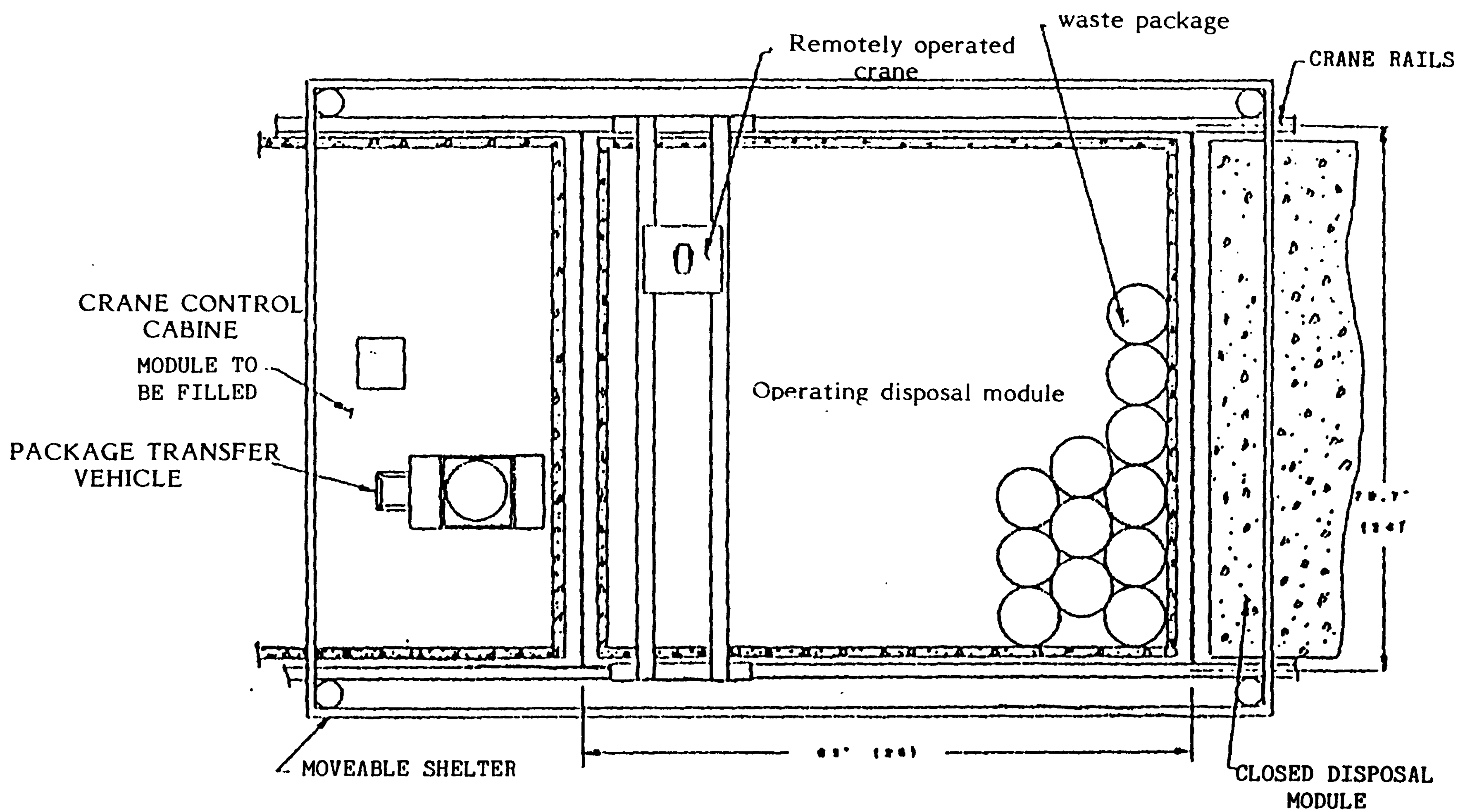

FIGURE 2 : DISPOSAL MODULE PLAN

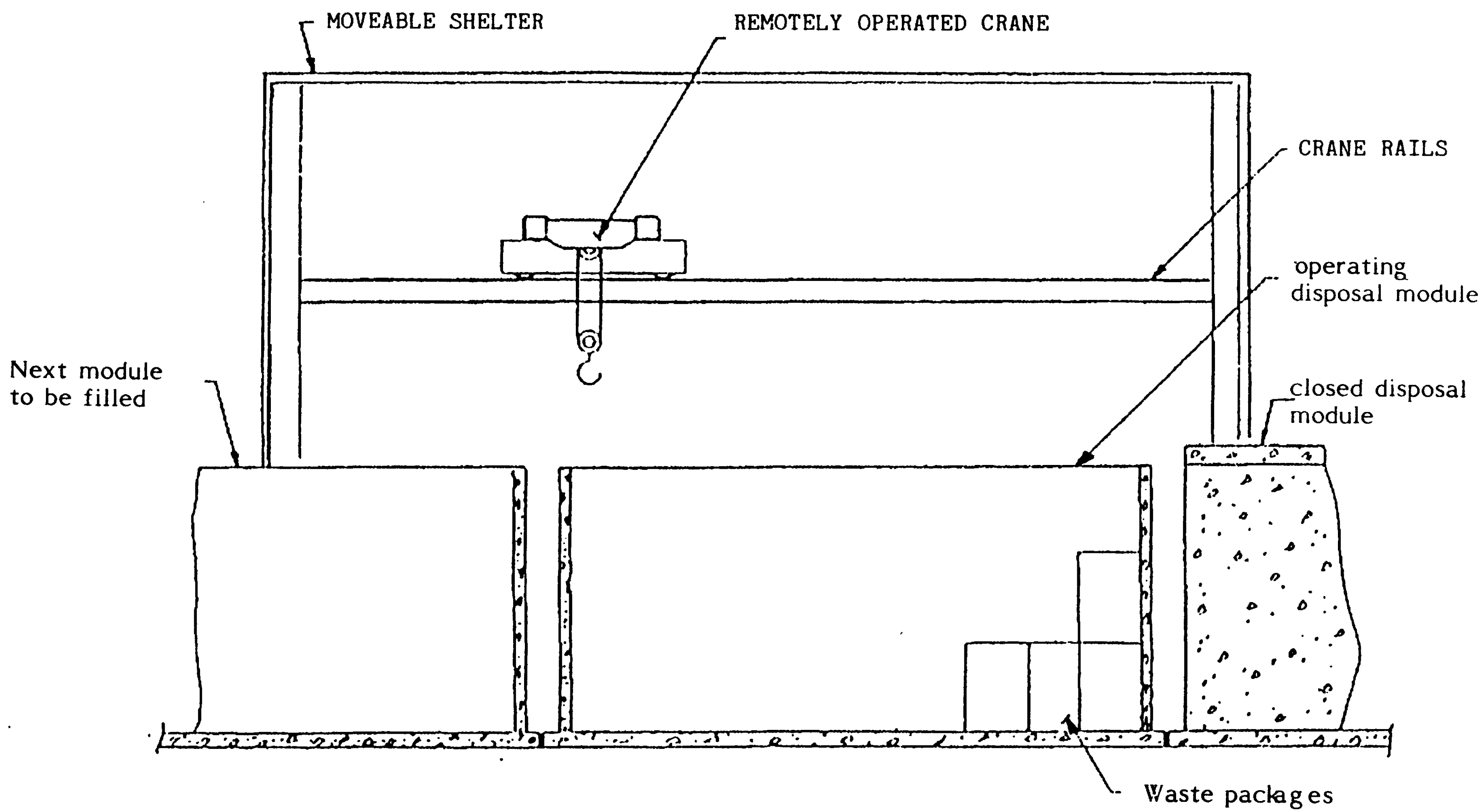

FIGURE 3 : DISPOSAL MODULE FRONT SECTION VIEW

3.4.1.3. Operation of the Disposal Module

The principle operations conducted at the disposal module are described here in sequential order.

Unloading and disposal of packages.

The truck carrying the packages is brought inside the mobile shelter to the unloading station in the module next the module where the packages are to be disposed.

Unloading is performed using a crane with the appropriate lifting beam. The crane is operated using a control panel and video-monitors from a shielded cab that is inside the shelter.

The location of the packages in the disposal module is then recorded in the computerized waste tracking system of the site.

Backfilling of the Disposal Modules.

The space between packages is filled with gravel up to 1,64 ft (0,5m) beneath the top of the packages. The backfill is brought into the unloading station by truck and is spread over the module with a clamshell bucket attached to the overhead crane. It will take approximately 5 to 6 working days using 2 shifts to fill the module.

Placement of the Disposal Module Roof

Prefabricated concrete slabs are put in place on top of the packages to support the construction of the monolithic concrete roof and to provide radiation shielding for the workers. The roof will be poured using rebar that ties in with the rebar which is left exposed at the top of the walls.

Placement of the Temporary Cover

The entire closed disposal module is covered with a waterproof synthetic material. This cover will be left in place when the final earthen cap is placed over the disposal unit.

Closure of the Module Access Wall

The next module, which was temporarily opened to enable the packages to be brought in, will be closed to allow the module to be filled with packages. This closing will be accomplished by emplacement of rebar and forms, and by concrete pouring.

Transfer of the Mobile Shelter

The shelter is moved 82 ft (25m) longitudinally to the next disposal module in the row. This operation requires about 8 hours. It is accomplished by disconnecting the tie-downs between the shelter and the track, jacking up the shelter a maximum of 4 inches (10 cm) by means of hydraulic jacks in the feet of the main gantries, connecting the wheel assemblies to the shelter and guiding the shelter along the rails using a hydraulic winch that has been anchored to the track. Once in position over the next module the shelter is lowered and anchored in place.

The moveable shelter is moved laterally after the filling of a row of disposal modules. This requires one day to move. This involves the same operations described in the preceding paragraph, but the wheel assemblies are rotated 90 degrees so the shelter can be guided onto the perpendicular tracks that run the width of the disposal unit. The wheels are totated a second time to direct the shelter onto the next row of disposal modules.

3.4.1.4. **Water Management System**

There are 2 water management systems : one for leachate collection and one for rainwater diversion. The purpose of the leachate collection system is to monitor the efficiency of the barriers in preventing water from coming in contact with the waste and to collect any leachate that may arise so that it is not released to the environment. The purpose of the rainwater diversion system is to direct rainwater away from the disposal module and the waste, during both loading operations and after closure, so that the amount of water that may potentially infiltrate the disposal module is minimized. With the first system (leachate collection), the water if any, may have come in contact with the waste and therefore may be contaminated; in the second system (rainwater diversion), the water is not contaminated.

The leachate collection system is presented, schematically in Figure 4; it is composed of drainpipes which collect the leachate from the drain in the center of the module and carry it to an individual monitoring tank. The monitoring tanks from a row of modules are connected to a header that carries the leachate to an underground tank. This tank has a level detector and a sampling mechanism. The entire system is located inside the undergroung gallery that lies directly below the modules.

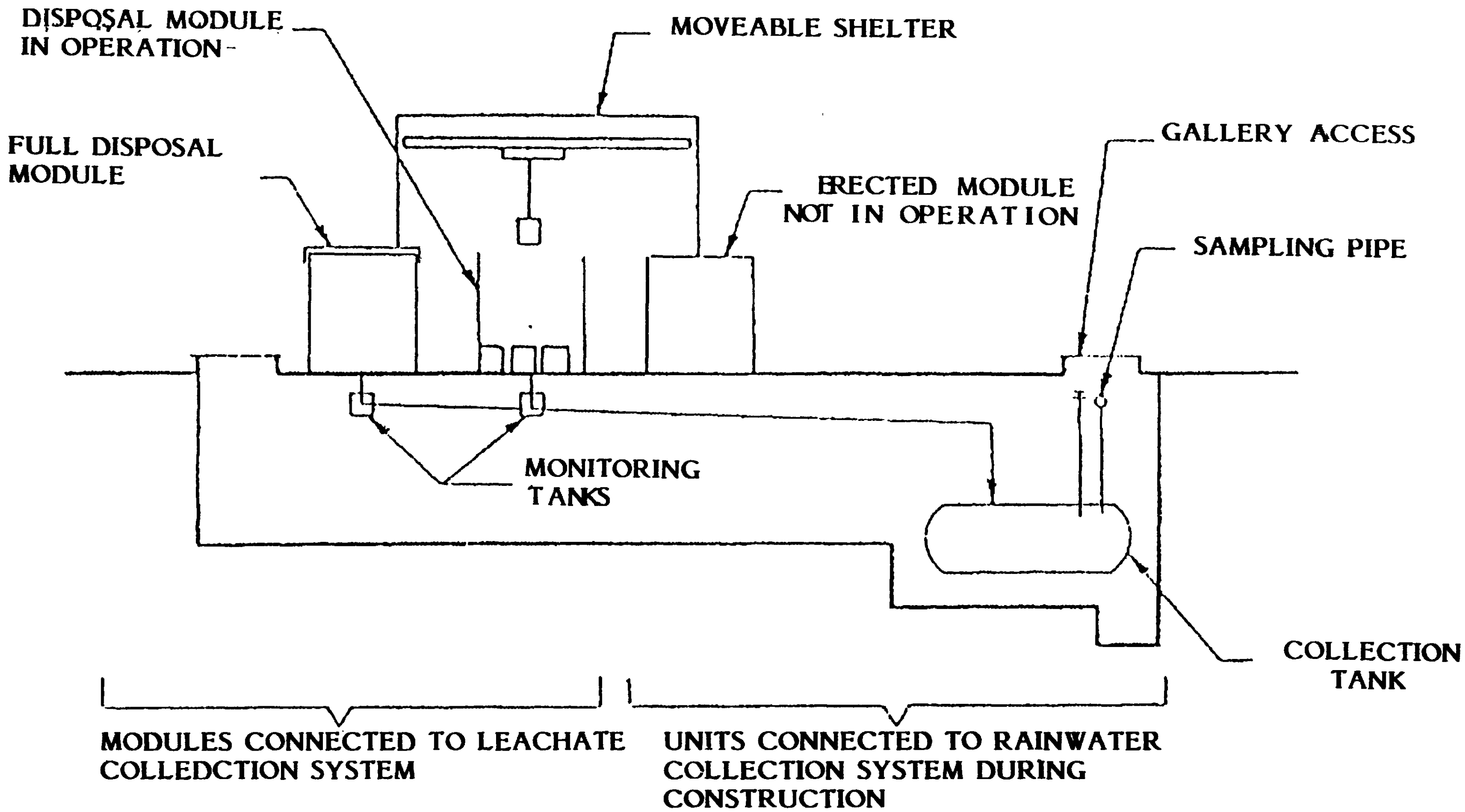

FIGURE 4 : SCHEMATIC OF LEACHATE COLLECTION SYSTEM

The gallery can be accessed by operating personnel to check the condition of the gallery materials and to sample the individual leachate monitoring tanks.

The leachate collection system enables the efficiency of the barriers to be continually monitored. The quantity of water collected gives an indication of the status of the secondary barrier (cap and disposal structures) and the activity of the leachate gives and indication of the status of the primary barrier (waste packages).

Leachate collected in the tank is sampled and sample is analyzed to determine activity level. If the leachate activity is below acceptable levels, it may be released to the rainwater diversion system using a pump. If the leachate is contaminated above acceptable release levels it may be transferred from the tank to an effluent treatment station.

During the institutional control period, the underground tanks are connected to a central tank to simplify monitoring operations.

The water in the rainwater diversion system is collected from modules in operation and from temporary covers by channels that run along the track that carries the shelter and from modules with final disposal caps by channels that are located at the base and at the top of the earthen cover. Water collected in this manner is sent to the rainwater retention pond of the site.

Before a module enters operation, rainwater from the module is collected in a drainpipe located in the gallery that leads to the rainwater diversion system. One the module enters operation, the drainpipe is collected to the leachate collection system and the pipe to the rainwater diversion system is capped.

3.4.2. Alveole monolith Design Basis

The design of the empty alveole is the same as the one of the platform, excepted that the pad with the walls are put on an other pad in order to get an alveole supported by a pad.

The design of the module shelter and the leachate and rainwater collection systems is the same for the alveole as for the platforms.

Only the loading operations change from one to the other.

The waste packages are immobilized in the grout layer after layer and, once filled with the last layer of packages and grout, the alveole is closed with a reinforced concrete roof.

3.4.3. Final cap design basis

The final cap design includes multiple layers as shown in Fig. 5 and described below, starting with the top layer :

Biological Barrier

Consisting of a layer of top soil and a layer of coarse material, the biological barrier fulfills 3 functions :

- to regulate the amount of water that reaches the layer of clay below,

- to enable vegetation to take root to promote evapotranspiration while limiting the development of deep-rooted plants,

- to generally protect the underlying impermeable layers below froms various threats, such as erosion, freeze-thaw cycles and animals.

Draining layer

This layer which may be composed of sand and sandy soil, help to drain off any water that may have infiltrated the biological layer and thus to prevent the clay layer below from being exposed to water.

Impermeable Barrier

Composed of compacted clay.

Draining Layer

Any water that may permeate the clay layer is drained off to the sides of the disposal module by a layer that is composed of sand or sandy soil.

Impermeable Barrier

This layer is an impermeable synthetic membrane.

FUNCTION	MATERIALS
BIOLOGICAL BARRIER	TOP SOIL
	UNSIZED GRAVEL
1ST DRAINING LAYER	SAND
1ST IMPERMEABLE LAYER	CLAY
2ND DRAINING LAYER	SAND
2ND IMPERMEABLE LAYER	GEOTEXTILE MEMBRANE
3RD DRAINING LAYER	SAND
FORM LAYER	UNSIZED MATERIAL
DISPOSAL MODULE	REINFORCED CONCRETE

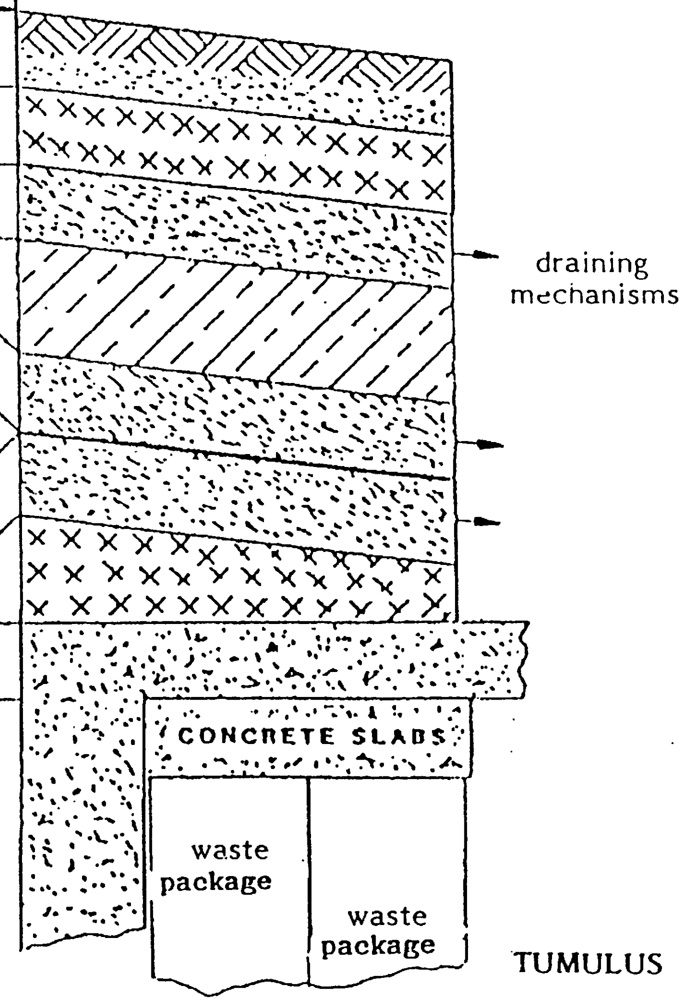

FIGURE 5 : MULTIPLE LAYER FINAL CAP

Draining Layer

A final layer, composed of sand or other material, is put in place to drain any infiltrated water to the side of the disposal module.

The thickness of the final disposal cap, which is several feet (a few meters), is determined after thorough site and safety analysis is conducted. The slope of the sides of the disposal cap is determined by an evaluation of the weather conditions of the selected site.

The performance of the final disposal cap is monitored by monitoring the water drained from the cap in a separate water collection system as well as by monitoring any leachate in the leachate collection system. In addition, a gutter on the upper edge of the disposal cap collects rainwater from the surface and carries it toward a drainpipe which directs it to the rainwater collection system. This system prevents rainwater from flowing down the sides of the mound, which would result in erosion of the biological layer of the disposal cap.

4. WASTE ACCEPTANCE CRITERIA

We have seen in 2. that among the technical prescriptions which accompany the construction permit of a particular disposal facility are acceptance criteria for the waste to be disposed of in this facility.

It exists an other Fundamental Safety Rule (FSR III.2.2) which fixes preliminary requisite conditions for the acceptance of <u>immobilized</u> solid waste intended for disposal in a near surface disposal facility.

In the following, we will find the CSM (Centre de Stockage de la Manche) Technical prescriptions :

- only solidified ($<$ 30 years) LLW and ILW short and medium lived are definitely admitted in the CSM.

- waste must not contain neither free liquid, nor organic liquid, nor biological toxical product, nor give chemical exothermic reaction,

- the specific activity is limited to the values indicated in tables 1 and 2 for the most important emitters.

Nuclides	Not imbedded wastes MBq/kg (Ci/t)	Imbedded wastes MBq/kg (Ci/t)
3 H	7,4 (0,2)	7,4 E1 (2)
60 Co	3,7 (0,1)	4,8 E4 (1300)
90 Sr	3,7 (0,1)	7,4 E2 (20)
137 Cs	3,7 (0,1)	4,8 E3 (130)
54 Mn	37 (1)	7 E5 (1,9 E4)
55 Fe	37 (1)	3 E6 (8,1 E4)
57 Co	37 (1)	2 E6 (5,4 E4)
65 Zn	10 (0,3)	1 E5 (2,7 E3)
93m Nb	37 (1)	3 E5 (8,1 E3)
110m Ag	20 (0,5)	2 E5 (5,4 E3)
106 Ru	9 (0,25)	8,8 E4 (2,4 E3)
125 Sb	37 (1)	8,1 E5 (2,2 E4)
134 Cs	3,7 (0,1)	3 E4 (8,1 E2)
137 Cs	3,7 (0,1)	4,8 E3 (1,3 E2)
144 Ce	9 (0,25)	8,8 E4 (2,4 E3)
152 Eu	30 (0,8)	3 E4 (8,1 E2)
154 Eu	20 (0,5)	2 E4 (5,4 E2)
TOTAL	37 (1)	$\sum_n \frac{a_i}{LMA_i} < 10$

1 t. = 1 000 kg
a_i = activity of nuclides in the package
LMA_i = maximum of mass activity for Nuclide i

TABLE 1 maximum of mass activity of waste package for Nuclides beta emitter with a half life between 0.5 and 30 years.

Nuclides	Not imbedded wastes MBq/kg (Ci/t)	Imbedded wastes MBq/kg (Ci/t)
226 Ra	3,7 x E-2 (0,001)	3,7 (0,1)
232 Th	3,7 x E-2 (0,001)	1,1 (0,03)
Total of actinides	1,9 x E-1 (0,005)	3,7 (0,1)

TABLE 2 Specific activity limitation for alpha emitter

The containment performances are assessed in annual released act fractions. The objective is the same as for homogeneous waste.

Moreover, a diffusion testing is carried out for 1 year to appre the homogeneity and continuity of the containment barrier.

OTHER ACCEPTANCE CRITERIA ARE :

Size of packages * :	
drums of 100 l with diameter of	445 mm to 470 mm
height of	630 mm to 690 mm
drums of 200 l with diameter of	580 mm to 620 mm
height of	775 mm to 915 mm
Filling rate of drum	90 %
Weight of package	350 kg
Water content in bitumized wastes	5 %
Water exudation 1 h under 0.35 MPa	3 % in Volume
Inflating under water during 30 days	5 %
Softening temperature	60°C (Polymer) 40°C (bitumen)
Ignition temperature	300°C (Polymer) 250°C (bitumen)
Thermal cycling test	5 cycles between-20°C to+40°C
Test of fire resistance	30 mn at 800°C
Dropping test	1.2 m
Matrix deformation under compressive test at 0.35 MPa	1 % (concrete or polymer)
Rupture strenght of the matrix material at	8 MPa for polymer or bitumen 20 MPa for concrete at 90 days
Test under gamma irradiation at integrated dose	10^5 Gy
Tritium degasing	2.10^6 Bq t^{-1} d^{-1} (5.4. 10^{-5}Ci $t^{-1}d^{-1}$)
Radon degasing	5.10^3 Bq t^{-1} d^{-1} (1.4 10^{-7}Ci t^{-1} d^{-1})

(*) Heterogen wastes can be also imbedded in Polymer or Cement metallic caisson of 2.5 m^3(8 t) or 5 m^3 (16 t) or 10 m^3.

Surface contamination of the package $\alpha < 0.37$ Bq/cm^2

$\beta\gamma \leqslant 3.7$ Bq/cm^2

Dose at contact of package lower than 2 mGy/h

Leaching rate of package

Annual Leaching Fraction for alpha emitter	FAL $< 10^{-4}$year $^{-1}$
Annual Leaching Fraction for beta-gamma emitter	A $<$ 37 MBq/kg : FAL $<$ 0.1 year^{-1}
	37 $<$ A $<$ 370 MBq/kg : FAL $<$ 0.01 y^{-1}
	A $<$ 370 MBq/kg : FAL $<$ 0.007 y^{-1}
Annual Leaching Fraction for Tritium	FAL $< 5.10^{-2}$ year $^{-1}$

FAL : annual Leaching Fraction.

5. **WASTE MANAGEMENT**

5.1. LIST OF PACKAGES TO BE STORED ON THE SURFACE SITE

In the above chapters are described the storage installations, their operation and the specification of the packages stored there.

In this chapter are described the procedures for management of the packages arriving at the surface storage site : the various operations to which the packages are subjected depend, in reality, on the type and the form under which the wastes have been conditioned.

A list of these packages have been drawn up. They come from two different sources (see tables 3 and 4) : on one hand the packages(*) from nuclear power plants (PWRs), and on the other hand the wastes from reprocessing plants, meeting the specifications for storage at surface sites. Note that the packages from nuclear power plants represent two thirds of the total volume.

A large number of packages from nuclear power plants can be further compacted (density of about 0.45). An installation for conditioning these drums will thus be included in which they will be compacted and cemented in 400 l drums, six 200-l drums giving one 400-l drum. Only one of the technological wastes delivered in 200 l drums or in boxes (VHE filter), hundred 200-l drums for a 900 MWe reactor cannot be compacted.

(*) The volume of reactor waste have been defined by other participants of this EC PROGRAMME.

The values given in this report are issued from report SGN N° 3174.00.002 - Rev. 2, Juin 1989.

ORIGIN	PACKAGE	VOLUME	ACTIVITY
Resins			
TEP	14 C4 concrete boxes	17.3 m3/year	14 x 52 Ci = 728 Ci/year
TEU	8 C4 concrete boxes	9.9 m3/year	8 x 1.3 Ci = 10.4 Ci/year
RCV	5 C4 concrete boxes	6.2 m3/year	5 x 130 Ci = 650 Ci/year
PTR	10 C4 concrete boxes	12.4 m3/an	10 x 13 Ci = 130 Ci/year
Filters			
TEP	5 C1 concrete boxes	10 m3/year	5 x 50 Ci = 250 Ci/year
TEU	3 C4 concrete boxes	3.7 m3/year	3 x 5 Ci = 15 Ci/year
TEG	1 C4 concrete boxes	0.6 m3/year	1 x 0.6 Ci = 0.6 Ci/year
RCV	10 C1 concrete boxes	20 m3/year	10 x 50 Ci = 500 Ci/year
PTR	20 C4 concrete boxes	24.7 m3/year	20 x 1 Ci = 20 Ci/year
Concentrates			
TEU	9 C1 concrete boxes	18 m3/year	1.6 Ci/year
Technological wastes			
(*) Compactible	600x200 l drums	120 m3/year	3.6 Ci/year
Non-compactible	100x200 l drums	20 m3/year	0.2 Ci/year
Non-compactible	100x200 l drums	20 m3/year	4 Ci/year
VHE filters	7 cardboard boxes	4 m3/year	0.12 Ci/year

(*) Initially 360 m^3 with a density of 0.15 before compaction.

TABLE 3a : WASTE PACKAGES FROM NUCLEAR POWER PLANT (from one 900 MWe power plant).

PACKAGE	NUMBER OF PACKAGES	VOLUME	ACTIVITY
Compactible cardboard boxes	154	88 m^3/year	2.64 Ci/year
Compactible 200 l drums	13.200	2.640 m^3/year	79 Ci/year
Non-compactible 200 l drums	4.400	880 m^3/year	92.4 Ci/year
C1	528	1 056 m^3/year	16 535 Ci/year
C4	1.342	1.645 m^3/year	34.188 Ci/year
		6.309 m^3/year	
Hence after compaction			
Non-compactible 200 l drums	4.400	880 m^3/year	92 Ci/year
Compacted 400 l drums	2.225	909 m^3/year	82 Ci/year
C1	528	1 056 m^3/year	16.535 Ci/year
C4	1.342	1.645 m^3/year	34.188 Ci/year
S/TOTAL		4.490 m3/year	

TABLE 3b : WASTE PACKAGE ANNUAL INVENTORY FROM A PARK OF 20 GWe (twenty two x 900 MW Reactors).

PACKAGES	TOTAL VOLUME m^3/year	α ACTIVITY Ci/year	β ACTIVITY Ci/year	ORIGIN
12 packages	27 m^3/year	-	9 600	Nympheas
3 150 single CACs				
499 single CACs	2 914 m^3/year	133.9	11 262	Technological wastes
46 double CACs				
309 x 220 l drums	68 m^3/year	1.4	1 761	TEO
	3 009 m^3/y	135.3	22 623	

TABLE 4 : ANNUAL INVENTORY OF WASTE PACKAGE FROM THE REPROCESSING PLANT (600 t/YEAR), WHICH CAN BE STORED AT THE SURFACE SITE.

5. 2. DIAGRAM FOR MANAGEMENT OF THE WASTES AT THE STORAGE SITE

We can then describe the routing of the packages from their entry at the storage site, which includes an access control with external inspection of the load, i.e. inspection of the transcontainer or of the platform on the trailer, followed by routing of the packages to their destination.

If the packages are not in conformity with the specifications, the non-conformity is studied and sometimes an additional packaging is necessary before treatment. They can also be returned to the sender. But this should happen only very rarely.

If the packages are compactible, they are sent to the conditioning workshop for compacting and cementing. This workshop is fully automatic. The packages are identified and registered, taken to the press, which is simultaneously supplied with 400 l drums. At the press output, the prenumbered barrel is cemented and stowed for drying. From there it will be picked up and placed on the storage platform.

If the packages are in conformity with the acceptance specifications, the load is conveyed to the platform or alveole meant for it. An internal inspection of the load is performed on the spot, i.e. inspection within the container or platform on the trailer. Then the package is unloaded under the shelter of the mobile hangar by a gantry crane which picks up the package in the truck and places it directly in its final location. Simultaneously, this package is identified and registered with its final storage location.

All these operations can be schematically represented in table 5.

Temporary stowage is also possible if the storage modules are not available because of freezing temperatures or to be able to store the packages systematically by batches, etc.

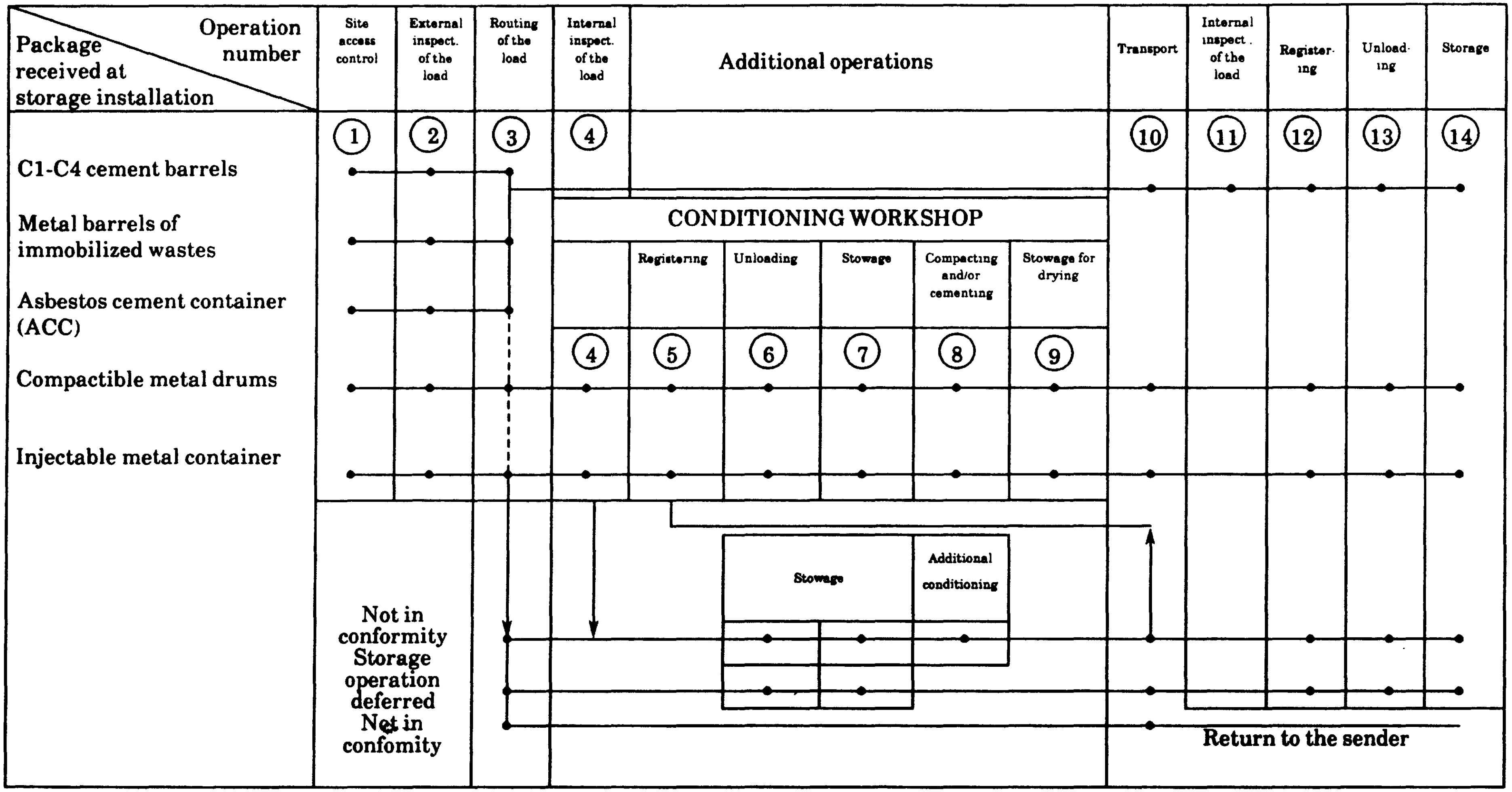

TABLE 5 : Management flowchart of low activity wastes

Operators \ Operation number	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15
Team leader							1						1		+ 1 Foreman for storage module conctruction
Employee	Same as in 3		1					Drums	Container	Same as in 3					
Machine operation						1		1							+ 5 Masons for module construction
Gantry operator						1		400 l drums	Cement				3		
Handler								1					1		+ 4 for installing sealing membrane
Trolley operator										1			Crane operator	cement	
Radiation inspector		1				1		1					1		

TABLE 6 : Flowchart of low activity wastes - Personnel assigned to the various operations

5. 3. DISTRIBUTION OF THE PACKAGES AND STORAGE DIMENSIONING

There are three possible types of storage modules depending on the mechanical stability, the specific activity of the waste package and the coating.

- the alveole (indicated by A in table 8) for the waste packages whose specific beta activity is higher than 1 Ci/t, unless they are immobilized, as is the case for cemented drums,

- the platforms for the waste packages whose specific beta activity is lower than 1 Ci/t and for the waste packages whose specific beta activity is higher than 1 Ci/t provided that the latter are immobilized.

Type 1 platforms (PF1) are used for packages of stabilized wastes and type 2 platforms (PF2) are used for packages of non-stabilized wastes.

Type 1 and type 2 platforms differ by the filling materials: cement filling for type 2 platforms and gravel for type 1 platforms.

Because of these criteria, the packages can be divided between three types of storage: PF1, PF2 or A.

It is assumed that all the packages have an alpha activity lower than 0.1 Ci/t and that the average for all the packages together does not exceed 0.01 Ci/t. This condition is indispensable for surface storage.

Tables 7 and 8 are drawn up according to these criteria. From these tables it can be seen that all the wastes from nuclear power plants are sent either to PF1 for concrete boxes (lasting) or to PF2 for metal drums (non-lasting or non-stabilized).

For the reprocessing wastes, the concrete boxes (lasting) are sent to PF1 and the non-lasting metal drums with an activity lower than the limit for coating are placed in the concrete compartments.

The hypotheses for filling the platforms are:

1 488 C1 packages	or	3 036 m^3 per platform,
or 2 460 C4 packages	or	2 976 m3 per platform,
or 7 280 x 200 l drums	or	1 456 m^3 per platform,
or 4 134 x 400 l drums	or	1 653 m^3 per platform,

For the CACs, a hypothesis similar to that for C type packages, i.e. 3026 m^3 per platform and 1456 m^3 per alveole loaded with 220 l drums.

PACKAGE		ORIGIN		MATRIX		USEFUL VOLUME m^3/ package	TOTAL ACTIVITY Ci/year	NUMBER OF PACKAGES per year	SPECIFIC ACTIVITY Ci/t	COATING	DESTINATION
				TYPE	DENSITY						
Concrete hull	C_4	TEP	Resin	Polymer	1	0,478	728	14	109	Yes	PF1
	C_4	TEU	Resin	Polymer	1	0,478	10,4	8	2,6	Yes	PF1
	C_4	RCV	Resin	Polymer	1	0,478	650	5	272	Yes	PF1
	C_4	PTR	Resin	Polymer	1	0,478	130	10	27,2	Yes	PF1
	C_1	TEP	Filters	Cement	2	0,916	250	5	27,2		PF1
	C_4	TEU	Filters	Cement	2	0,478	15	3	5,2		PF1
	C_4	TEG	Filters	Cement	2	0,478	0,6	1	0,6	Yes	PF1
	C_1	RCV	Filters	Cement	2	0,916	500	10	27,3	Yes	PF1
	C_4	PTR	Filters	Cement	2	0,478	20	20	1,0	Yes	PF1
	C_1	TEU	Concentrates	Cement	2	0,916	1,6	9	0,1	Yes	PF1
Metal drum	200 l	Precompacted technological wastes		Cement	2.45	0,200	4,2	200	0,04	No	PF2
	400 l	Supercompacted technological wastes		Cement	2	0,400	3,7	102	0,05	No	PF2

TABLE 7: DESTINATION OF THE WASTE PACKAGES ACCORDING TO THEIR SPECIFIC ACTIVITY

Annual waste for one 900 MWe reactor

PACKAGE	ORIGIN	MATRIX		USEFUL VOLUME m^3/ package	TOTAL ACTIVITY Ci/year	VOLUME OF PACKAGE m^3/year	SPECIFIC ACTIVITY Ci/t	COATING	DESTINATION
		TYPE	DENSITY						
Single CAC	Zone 4 Technological wastes	Cement	2	1,18				Yes	PF1
Double CAC	Zone 4 Technological wastes	Cement	2	2,36	α = 134 β = 11 262	2914	α = 0,023 β = 1,9	Yes	PF1
CAC	Zones 2 + 3 Technological wastes	Cement	2	0,67				Yes	PF1
220 l metal drum	TEO Mineralization ashes	Cement	2	0,22	α = 1,4 β = 1 761	68	α = 0,01 β = 12,95	Yes	A
	Ion exchanger nympheas	Resin	1	(2,259)	β = 9 600	27	β = 356	Yes	A

TABLE 8: DESTINATION OF THE WASTE PACKAGES ACCORDING TO THEIR SPECIFIC ACTIVITY

Reprocessing plant waste (600 tML/year)

The annual requirements for storage facilities are thus:

0.9	PF1 platform	for C1 and C4 type packages,
1	PF1 platform	for CAC packages,
1.25	PF2 platform	for 200 l and 400 l drums,
0.1	alveole	for TEO and Nympheas drums,

i.e. over 30 years:

56 PF1 platforms,
35 PF2 platforms,
2 alveoles.

The question can be posed whether it is not better to temporarily stow the packages meant for storage in concrete alveole until a sufficient volume is accumulated for storage rather than to keep the alveole work open continuously while using it only once per month.

The loading capacity of the trucks is 10 to 12 m3 per trip, i.e. 6 C4 packages or 4 to 8 (average 6) C1, C2, C3 packages, or 48 x 200 l drums.

This results in an annual rate of:

313	trucks loaded with C1 or C4 packages	to be stored at PF1
92	trucks loaded with 200 l drums	to be stored at PF2
278	trucks loaded with 200 l drums	to be compacted
182	trucks loaded with CACs	to be stored at PF1
10	trucks loaded with drums	to be stored at A

giving a daily rate of 875/220 or about 4 trucks/day, of which half will be travelling to the PF1 platforms.

The rate for loads in alveole is 1 truck per month on an average. This explains the remark made above concerning the possibility of stowing these packages until a sufficient number have been accumulated to justify the reserving of a gantry for filling the compartment or the construction of a mini-alveole at the centre of a PF2 platform.

The daily rate for storage of packages in the storage modules should be sufficient to handle:

PACKAGE	ORIGIN		NUMBER OF PACKAGES		NUMBER OF TRUCKS PER YEAR	NUMBER OF STORAGE MODULES	
			ANNUAL	OVER 30 YEARS		ANNUAL	OVER 30 YEARS
C_4	TEP	Resin	308	9 240	52	0,125	3,76
C_4	TEU	Resin	176	5 280	29,3	0,072	2,15
C_4	RCV	Resin	110	3 300	18,3	0,045	1,34
C_4	PTR	Resin	220	6 600	36,7	0,089	2,68
C_1	TEP	Filters	110	3 300	18,3	0,074	2,22
C_4	TEU	Filters	66	1 980	11	0,027	0,80
C_4	TEG	Filters	22	660	3,7	0,009	0,27
C_1	RCV	Filters	220	6 600	36,7	0,148	4,44
C_4	PTR	Filters	440	13 200	73,7	0,179	5,37
C_1	TEU	Concentrates	198	5 940	33,0	0,133	3,99
SUBTOTAL C_4			1 342	40 260	225	0,546	16,40
C_1			528	15 840	88	0,355	10,65
			1 870	56 700	313	0,901 PF1	27,05 PF1
200 l drums	Non-compactible technological wastes		4 400	132 000		0,604	18,3
400 l drums	Technological wastes: compacted drums		2 220	66 660		0,537	16,1
SUBTOTAL	Before compacting		17 754	532 600	370		
	After compacting		6 622	198 660		1,24 PF2	34,4 PF2

T A B L E 9 : **PRODUCTION OF PACKAGES AND STORAGE VOLUME FOR A PARK OF NUCLEAR POWER PLANTS PRODUCING 20 GWe OVER A PERIOD OF 30 YEARS**

PACKAGE	ORIGIN	NUMBER OF PACKAGES		NUMBER OF TRUCKS PER YEAR	NUMBER OF STORAGE MODULES	
		ANNUAL	OVER 30 YEARS		ANNUAL	OVER 30 YEARS
CACs	Zones 2 + 3 + 4	2 914 m^3	87 420 m^3	182	0,96 PF1	29 PF1
Drums	TEO (309 drums)	68 m^3	2 040 m^3	10	0,07 A	2,1 A
	Nympheas (12 containers)	27 m^3	810 m^3			

TABLE 10 : PRODUCTION OF PACKAGES AND STORAGE VOLUME FOR A 600 tML/YEAR PLANT

6. COST EVALUATION

6.1. Requirements in personnel

Depending on the characteristics of the installation and the routing in the management of the wastes which need to be stored there, it is possible to estimate the requirements in personnel.

Two categories of personnel can be distinguished: on one hand, the personnel directly assigned to the storage of wastes and to the conditioning, if necessary, and, on the other hand, the personnel of the general services responsible for management, administration, supplies, supervision and control and maintenance.

This estimation has been made on the basis of a park of nuclear power plants producing 20 GWe and resulting in the wastes listed and quantified above.

Workers directly assigned to the work (see table 6)

The conditioning workshop will operate with 10 persons:

- 1 foreman,
- 2 machine operators:
  1 for unloading, 1 for compacting and cementing,
- 1 operator for the unloading gantry,
- 2 handlers:
  1 for supplying the 400 l drums and 1 for supplying cement and rinsing,
- 1 trolley or trailer driver to transfer the packages between the conditioning workshop, the temporary stowage and the storage spot,
- 2 radiation inspectors for unloading and conditioning,
- 1 maintenance technician.

The storage insallations will be run by 8 employees:

- 1 foreman,

- 1 clerk for orienting the packages,

- 3 gantry operators,

- 1 handler for supplying and filling the modules with gravel and cement,

- 2 radiation inspectors one of whom has access to the centre and the other to the storage modules.

For the construction of the storage modules and the installation of the sealing covers on the storage modules after filling, 6 operators will be required, including:

- 1 foreman,

- 5 masons for building new modules or installing the sealing cover.

We could consider, however, that these masons are hired from the firm entrusted with the construction work.

Indirect labour

For the organization of the storage installation, its management, the supplies, guarding and supervision and all the general services, the following posts will be counted (16 to 18 persons):

- 1 manager,

- 1 assistant manager who could ensure one of the following functions,

- 1 safety engineer,

- 1 administrative officer (finances and personnel),

- 3 secretaries,

- 1 supplies officer,

- 1 quality control engineer,

- 1 maintenance engineer,

- 1 health physicist (who could be the same as the safety engineer),

- 1 engineer for new construction work,

- 1 package follow-up officer,

- 2 employees for follow-up of the wastes,

- 1 quality control inspector,

- 2 clerks.

For the access control, the team will include 16 persons:

- 1 headkeeper,
- 15 keepers (5 groups of 3).

The maintenance team will consist of 3 persons:

- 1 foreman,
- 2 technicians.

For the laboratories and services, 3 persons will be required:

- 2 health physicist agents,
- 1 nurse.

For the information of the public, two persons:

- 1 public relations officer,
- 1 hostess.

The personnel required for the storage installation and working there permanently will thus include:

- 10 + 8 + 1 = 19 persons assigned directly to the work of storage of the packages,

- 16 + 16 + 3 + 3 + 2 = 40 persons for the tasks of management and the indirect work.

Other persons may have to work within the premises of the installation but will not remain there.

The latter will belong to the firms working under contract and their cost will be included in the cost of the subcontracted work and services.

6. 2. INVESTEMENT COST

From the design of the centre and the management principles described above, three zones or three types of activity will be distinguished on the site:

- Firstly, the storage platforms including the construction of modules (platform or compartment) as and when required, on a terrain already prepared by a drainage network and the casting of concrete slabs on which the structures will be built later.

The system for collection recovers, at the base of the structures, any water which may have leaked through the final covering of the storage space and drains it, if necessary, to a tank.

These construction works will thus be carried out away from the radiation of the packages since they will not be contiguous to the storage areas which are already full or being filled, or at least protected by walls from the alveoles or platforms.

- Secondly, a workshop for treatment and conditioning of packages (compacting, cementing, etc.) to complete or possibly redo the conditioning of a package.

These buildings will also include a zone for temporary stowage of packages which will also serve as a buffer area, as explained above, to adapt the arrival rate to the storage rate, whatever the problems encountered in the performance of the storage operations.

- And finally, the various administrative and service buildings including:

. the guard room,

. the service buildings, including the laboratories and the buildings for radiation protection and health services,

. the mechanical workshop buildings including the decontamination room,

. the power station,

. the water distribution and processing station,

. the fire-fighting station,

. the public information building.

All these storage installations are supplied by all the utilities and enclosed by a guarded fence.

The investment and operating costs of the installation have been estimated on this basis.

The cost of construction of the storage modules is included in the operating costs since these modules are built as required. This cost includes labour.

The rate of change used for the cost evaluation is 6.90 F/ECU.

INVESTMENT

DIRECT COST

Site improvement:		1 698 000 ECU
Civil works (all the buildings except the storage structures):		9 022 000 ECU
Major equipment of the storage modules:		3 311 000 ECU
2 movable hangars and movable gantries for PF1		
1 movable hangar and movable gantry for PF2 and alveole		
Concrete pumps		
Bulk materials		17 724 000 ECU
- General power supply:	3 865 000 ECU	
- Control and data processing (supervision and control, radiation protection, security, etc):	4 298 000 ECU	
- Mechanical:	6 660 000 ECU	
- Systems related to processes (effluent treatments)	830 000 ECU	
- Systems related to the ancillary services (ventilation, water supply, etc):	2 071 000 ECU	

- Quality assurance
  (included in the other costs)

- Indirect constructions (worksite structures, worksite equipment, storage and transport of materials, etc.): 821 000 ECU

- Labour
  (The labour costs for the personnel involved is included under each title).

INVESTMENT

Total Direct cost	32 576 000 ECU
Indirect cost	10 906 000 ECU
TOTAL INVESTMENT	**43 482 000 ECU**

6. 3. OPERATING COSTS

- Processing materials	1 719 000 ECU
- Fluids	129 000 ECU
- Maintenance materials	576 000 ECU
- Direct labour costs	
19 employees at 35 ECU/h for 1 694 hours/y :	1 127 000 ECU
- Overhead costs	
40 employees at 35 ECU/h for 1 694 hours/y :	2 372 000 ECU
Total annual operating cost	**5 923 000 ECU**

6. 4. COST OF NEAR SURFACE DISPOSAL PER M3 OF WASTE

The cost calculation of the waste m^3 is based on the following data :

Construction duration	n = 3 years
Annual rate of interest	i = 8.3 % y^{-1}
Annual rate of inflation	e = 2.2 % y^{-1}

Cost determination method given by TASK/KAH (*).

Time duration x between the start of the plant construction and the middle of the activity of the cost element for :

- Site improvement	x = 0.5 year	n-x = 2.5 year
- Civil Work	1.0	2.0
- Major Equipment	1.75	1.25
- Indirect Construction	1.50	1.50
- Laboratory and Health	2.50	0.50
- Architect - Engeneering	1.00	2.00

The resulting cost of waste storage is evaluted at 1 355 ECU/M^3.

(*) TASK/KAH - First progress Report period 01.10.87 31.03.88.
Proposed methodology for the cost evaluation of radioactive waste management route by S. KOWA, T.A. SHAMSI, F. STENERSEN, G. THIELS. Document N° 10 EN 114-115/023.

7. SENSITIVITY STUDIES :

The sensibility of the cost of storage on the type of package or quantity of waste to be stored depends primarily on the diversity of the waste packages to be stored.

The volume occupied by the stored packages per unit volume depends on the way they are arranged within the cell or monolith, the size of which can be optimized if there is a sufficient number of packages. In fact, a single size of cubic package would allow storage areas to be filled to maximum capacity since the storage area could be designed especially for this size of package.

It should be noted that the investment costs of fixed installations, administrative buildings, service buildings and equipment rooms is relatively constant, irrespective of the volume of packages being stored.

As regards the operating costs, the personnel will be little affected by the volume of packages, and in particular the administrative personnel which represents the majority will be unchanged. Only the cost of materials will vary in proportion to the storage areas.

Consequently, the cost estimation based on the assumption the site would only be used to store reactor waste representing an annual volume of 6,309 m^3 before supercompacting on the site has been evaluated.

The investment cost would be 39.41 MECU (million ECUS) compared with 43.5 MECU for 9,320 m^3/year, and annual operating costs will drop from 6 MECU to 5.4 MECU.

The resulting reduction in the cost of storage will only be 10% for a reduction of about 30% in the volume of packages being stored : that leads to increase in the cost per m^3 up to 1800 ECU/m^3.

8. EXPOSURE OF WORKERS TO RADIATION

8.1 Objectives

The design of storage areas and work stations is such that under normal operating conditions in the storage centre, a very few people receive an annual dose equivalent rate (D.E.R.) of more than 500 mrem (5 mSv).

Furthermore, nobody receives an annual D.E.R. in excess of 1,500 mrem (15 mSv) under normal operating conditions.

The personnel can be located:

- either at a permanent or temporary work station,

- or in transit between work stations or auxiliary buildings,

- or in the auxiliary buildings.

A list of the personnel, their work stations and the time spent at them must be established so that received radiation doses can be determined.

The work stations can be divided into:

- normal permanently or temporarily occupied work stations,

- work stations for scheduled preventive or corrective maintenance.

8.2. Definition of zones

The premises are divided into health physics zones so that the maximum dose equivalent rates can be calculated. The site can be divided into 3 zones

Zone 1 (surveillance zone) shall include all the normal work stations where the D.E.R. will be limited to 0.06 mrem/hour. (0.6 micro Sv/h). However, under certain circumstances a D.E.R. of up to 0.25 mrem/hour (2.5 micro Sv/h is accepted.

For personnel occupying normal work stations full time, which is the case for the majority of the operating personnel, the total exposure duration is 1,600 hours/year. The maximum D.E.R. gives an annual exposure of less than 1,600 x 0.06 = 100 mrem/year (1 m Sv/year).

In zone 2 (low-risk controlled zone), the maximum D.E.R. at normal work stations is in principle less than 0.25 mrem/hour, (2.5 micro Sv/h) but under certain circumstances it can be increased to 0.75 mrem/hour for limited periods of time.

This D.E.R gives an annual exposure of less than 400 mrem/year (4 m Sv/y) for permanent stations with an exposure duration of 1,600 hours/year.

In zone 3N, which a specially regulated medium-risk zone, no work stations are supposed to be occupied for more than 410 hours per year (2 hours/day for 205 days/year or 10 hours/week x 41 weeks/year). In principle there are no normal work stations in zone 3N. Personnel which has to work in this zone shall only enter the zone on an occasional basis for a limited period of time to fulfil a specific task.

It is therefore acceptable to have a radiation dose per person in this zone of less than 410 hours x 2.5 mrem = 1025 mrem/year (10.25 m Sv/y). The rest of the time workers shall be assigned work in zone 1 or 2, where the annual dose shall be less than 1200 hours x 0.25 mrem/hour, that is 300 mrem/year (3 m Sv/y). The cumulative annual radiation dose shall remain below 1325 mrem (13,25 m Sv/y).

8. 3 Classification of work stations

The installations can be classed according to the 3 previously defined zones:

Zone 1: gatehouse, administrative buildings, company canteen, electricity substation, service buildings including the health physics laboratory, the infirmary, the equipment rooms for distribution of drinking water, fire protection water and heating.

Zone 2 (controlled zone): transit buildings for temporary holding prior to conditioning or storage. The waste conditioning shops, the storage platform and cell, the mechanical shops. Certain buildings in zone 2 contain zone 3 areas.

Zone 3 (regulated area) comprises:

1) In the waste conditioning shop:

- the unloading hall, the conveying tunnel, the non-conforming materials processing shop with entrance lock chamber for reception of packages to be stored,

- the conditioning hall, corridors for transporting metal drums for compaction and metal boxes for injection, with their filling lock chamber, corridors for evacuating conditioned 400-litre metal drums and vessels with their evacuation lock chamber, drying and handling areas.

The control room is in zone 1.

2) Virtually all the transit buildings,

3) Virtually all the storage areas, with the exception of the control cabin which is classed as zone 2 because it has biological protection.

From these definitions, and without going into a detailed analysis of each work station, one can make an overall hypothesis as regards the distribution of personnel between the 3 zones:

- 48 people/year for zone 1,
- 13 people/year for zone 2,
- 3 people/year for zone 3.

9. ASSESSMENT OF THE RADIOLOGICAL IMPACT TO THE PUBLIC

9.1 Radionuclide Inventory

The aim of this chapter is to assess the effects on the health of those members of the public most exposed (reference group) to radioactivity through the storage of radioactive waste from the operation of PWR. The assessment is made for a number of PWR, corresponding to an output power of 20 GW(e) operating for 30 years.

The radioactivity characteristics of reactor waste used as a reference are taken from the NUREG-CR 1759 report. With the volumes given in table 11, the radionuclide inventory shown in table 12 is obtained.

Origin	Annual Volume (m^3/y)	Total Volume (m^3)
Demineralizing resins	360	10800
Evaporation concentrates	2480	74400
Sludge	40	1200
Filter cartridges	220	6600
Compactable waste	4300	129000
Non-compactable waste	2220	66600
TOTAL	9620	288600

Table 11 : WASTE VOLUME OF A NUCLEAR PARK OF 20 GWe OPERATING FOR 30 YEARS.

All this waste material is stored in the Near Surface Site. It is assumed that it is stored in a site such as the Aube storage site

Radionuclide	Ci/year	Ci/ 30 years
H 3	1,27E+01	3,82E+02
C 14	9,80E-01	2,94E+01
Fe 55	5,21E+02	1,56E+04
Co 60	1,01E+03	3,02E+04
Ni 59	6,21E-01	1,87E+01
Ni 63	1,92E+02	5,76E+03
Nb 94	1,97E-02	5,92E-01
Sr 90	1,95E+00	8,84E+01
Tc 99	8,28E-03	2,48E-01
I 129	2,45E-02	7,36E-01
Cs 135	8,28E-03	2,48E-01
Cs 137	2,20E+02	6,60E+03

Table 12 REACTOR WASTE RADIOACTIVITY

9.2. Estimation of doses received by the reference group

The indicator used here is the effective dose equivalent ("dose") for the reference group. This corresponds to the maximum individual risk that can be associated with the utilization of outflow river water in which the different radionuclides will end up after transport in the geosphere.

The estimation of the transfer of the radionuclides through the geosphere up to the outflow river, and the calculation of the dose on the individual representative of the reference group as a function of all the possible transfer routes were done using GEOLE ([1]) a calculation code recently developed by the CEA for the work concerning the establishment of a preliminary safety report on the Aube storage centre.

The GEOLE code permits the simultaneous processing of about twenty radionuclides stored in barrels with a known damage function. The annual waste from each radionuclide in the outflow river is calculated taking into account stored activity, package damage laws, the annual fraction of lixiviated activity of each radionuclide, the geometry of the storage centre, and characteristics of the environment in which the packages are stored.

Typical outflow river water consumption scenarios can be considered to assess the total dose received by a representative individual from the reference group.

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1 P. GUETA, GEOLE version 1988. Code de calcul pour l'évaluation des transports de radionucléides à partir d'un stockage de surface. Manuel d'utilisation. Note SEPD N° 88/04 janvier 1988.

9.3. Basic Data

The annual lixiviated activity fractions are given for two periods :

- The first 330 years, which correspods to the period of storage and to the surveillance period during which coverage is assumed to be effective,

- Beyond 330 years, which corresponds to the danger-free.

The barrel damage law is determined from a gaussian with an average time of 500 years and a standard deviation of 170 years. However, in accordance with the recommendations set forth in the safety report, one assumes that the package is completely destroyed after the surveillance period, i.e. 330 years after the beginning of storage. The true and modified package damage curve is shown in figure 6.

The main characteristics used for calculations are :

- Length of centre in direction of flow:	700 m
- Distance from enclosure to outflow river:	250 m
- Kinematic porosity of ground	0.06
- Darcy's speed	6 m/year
- Coefficient of molecular diffusion	0
- Coefficient of longitudinal dispersion	50 m

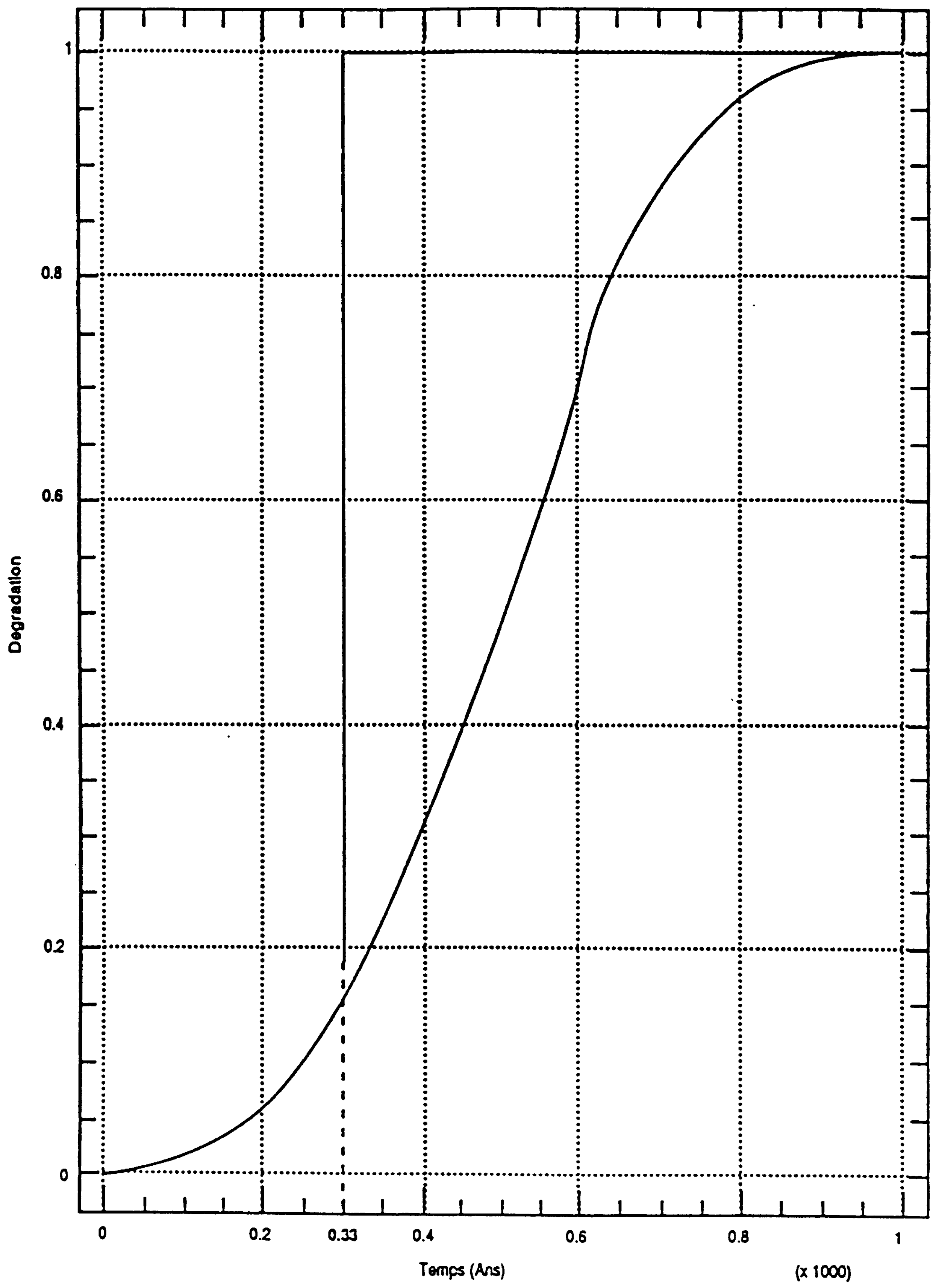

Figure 6 : Theoretical package damage law and the law used for the analysis (function of distribution of the normal law)

9.4. Results

After simulating the change in activity of the source and the outflow river, the GEOLE code provides the activity peak in the outflow river for each of the main radionuclides.

The calculation of the dose received by the individual representing the reference group is based on the various possible uses that could be made of the outflow river water. The scenario for consumption of this water takes several possibilities into account: produce from a garden watered with water from the river, meat and milk from animals drinking from the river for part of the year, fish from the river.

The values for the menus are given in the following table,

PRODUCT		MENU 1	MENU 2
LETTUCE	(kg/year)	20	-
CABBAGE	"	9	-
CARROTS	"	10	-
LEEKS	"	10	-
POTATOES	"	81	-
GREEN BEANS	"	15	-
RADISHES	"	4	-
MILK	(l/year)	88	88
BEEF	(kg/year)	5	5
MUTTON	"	1	1
FISH	"	5	-

TABLE 13 / ANNUAL CONSUMPTION OF AN INDIVIDUAL

The annual equivalent dose rates have been determined taking into account the consumption profile (assumed to remain constant through time) and the calculated activities in the river.

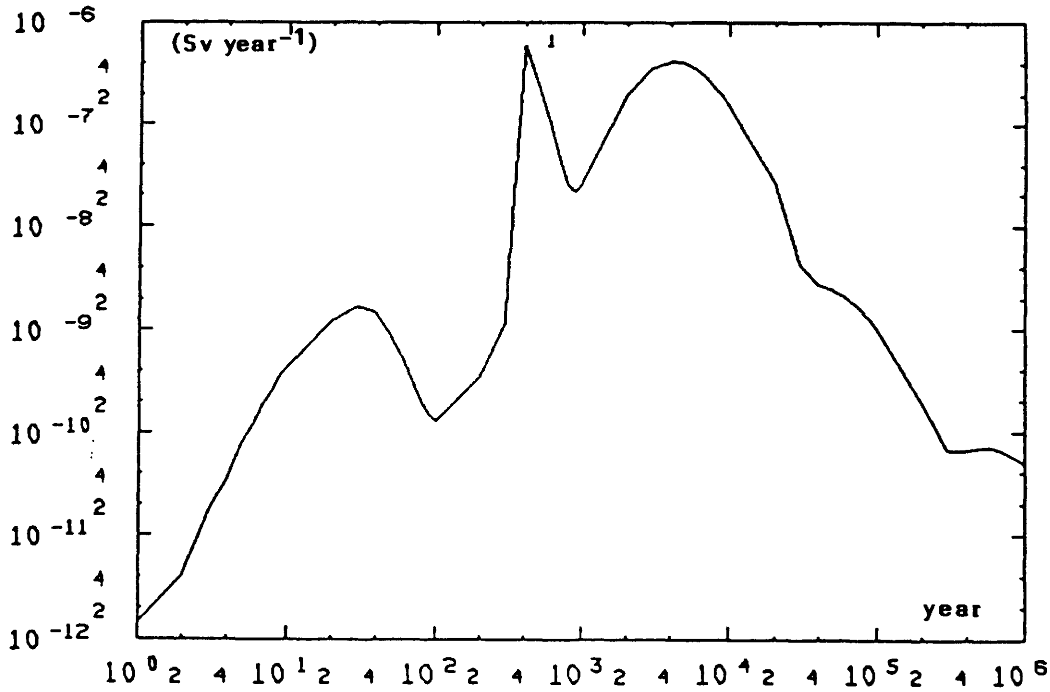

Figure 7 : TOTAL RADIOLOGICAL IMPACT OF REACTOR WASTE AS A FUNCTION OF TIME (20 GWe x 30 YEARS).

One can see (Fig. 7) doses of the order of 0.5 µSv/year, or 50 µrem/year. These values seem negligible when compared with the other levels of exposure that man is, and will be, subjected to.

These results show that the radiological impact of surface storage of reactor waste from nuclear power stations producing 20 Gwe (22 PWR's of 900 MWe) operating for 30 years produces a maximum annual dose of 0.5 microsievert for a consumption of menu 1 level. This value is expressed in effective dose equivalent for the reference population group in the vicinity of the surface storage site.

European Communities – Commission

EUR 14043 – Assessment of management alternatives for LWR wastes
(Volume 7)
Cost and radiological impact associated with near-surface disposal of reactor waste (French concept)

J. Malherbe

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