Report Series

Dossier 2005 Granite

Tome

Architecture and management of a geological repository

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The present English version is a translation of the original “*Dossier 2005 Granite*“ documentation written in French, which remains ultimately the reference documentation.

In order to be consistent through the various documents, while the word ''storage'' (''*entreposage*'' in French) refers only to temporary management (in terms of concept and facility), ''disposal'' (in term of concept) and ''repository'' (in terms of facility or installation) refers to long term management of high level long lived radioactive waste (''*stockage*'' in French for these words).

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1 Study approach

Granite is one of the geological formations studied by Andra in application of the 30 December 1991 Act, to assess the potential for a geological repository for High-Level Long-Lived waste (HLLL waste) [1].

Since no underground laboratory was available in French granite, the studies have been carried out with a generic approach. They are based on knowledge of the national geological context and analysis of the properties of the granite medium. Supplementary data comes from experiments carried out in foreign underground research laboratories. They have enabled the major issues to be identified relating to this medium, notably the allocation of safety functions to the architectural elements of the repository.

This document concerns the study of the architecture and management of a reversible disposal system in a granite medium. It exposes the various isssues that are specific to granite and proposed solutions. It demonstrates that the granite medium cannot be ruled out for the installation of a reversible disposal system, according to the architectural solutions envisaged.

1.1 The main stages since the 30 December 1991 Act

From 1994-1996, Andra carried out reconnaissance work with a view to installing an underground research laboratory on four sites designated by the consultation mission headed by French MP, Christian Bataille. The granite site was located in the south of the Vienne district. The selected granite massif was a granite formation overlaid by sedimentary formations, generally delimited from geophysical and geological data.

In 1994 preliminary reconnaissance work on the Vienne site resulted in selecting an area of approximately 30 km2 to the east of the Chapelle-Bâton village. In 1995 and 1996, a second phase of work on the selected site completed the acquired data.

As a result of all the work carried out, applications were filed with the regulatory authority to install and operate underground laboratories on three different sites: the Vienne granite site, the Gard and Meuse/Haute-Marne clay sites.

The National Review Board, in its Report no. 3 of 1997, reported unfavourably on the Vienne site, in particular on the risks of fluids circulating between the granite massif and the aquifers exploited in the sedimentary overlying formations, while it underlined the interest of "outcropping" granites that would have more favourable characteristics.

The government decided not to retain the Vienne site on 9 December 1998 and planned for the research into other potential sites for a research laboratory in a granite medium. A consultation mission was organised in 1999 to present this project and assess public opinion on fifteen sites selected on the basis of geological criteria. These fifteen sites, submitted to a committee of national and international experts, were identified from previous selection approaches and advances in knowledge of the granite medium in France and abroad. The mission report in July 2000 highlighted the difficulties to achieve the consultation process.

According to government expectations, Andra has designed in 2000 a research programme taking stock of current knowledge acquired in foreign underground research laboratories and in various geological environments, in order to assess the assets of granite medium for a high-level, long-lived waste repository.

The contextual differences for the studies between the clay and granite formations lead Andra to organise its research into two distinct projects. The repository design project for the clay medium is based on the Meuse/Haute Marne underground laboratory, while the granite medium repository design project is based on the results of foreign research laboratories (Sweden, Switzerland and Canada).

Some studies were common to both projects, primarily the waste package aspects, whose results have been used by both projects.

The "Dossier 2002 Granite" presented a preliminary overview of the designs and research about the possibility of a repository in granite medium.

The Dossier 2005 Granite draws on the conclusions of many studies carried out since 1991. Accordingly it aims to assess the assets of granite medium for the disposal of High-Level Long-Lived waste.

1.2 Andra's research programme on disposal in a granite formation

1.2.1 A generic study approach

Andra's study approach consists of identifying the major issues relating to disposal in a granite medium in order to assess the assets of granite medium for a repository and to check that granite medium is not ruled out. It also aims to study the technical options that would ease repository adaptation to a specific French site.

Accordingly, understanding of the long-term behaviour of a repository and safety assessments are decisive in defining the generic characteristics of a repository adapted to the variability of French granites.

The research programme comprises four complementary areas of study, incorporated in an iterative approach:

- the granite medium studies involve both improving understanding and modelling of the medium and analysing the variability of the properties of French granites;
- the design studies for a generic reversible disposal system are based on granite-specific elements, primarily safety considerations, the design of underground structures and the repository operating and closure mode. Furthermore they are based on data common to the clay repository design project, primarily those relating to disposal packages and materials;
- the purpose of the long-term repository behaviour studies is to understand and model the thermal, mechanical, chemical and hydraulic phenomena that come into play in a granite medium repository, on the basis of the proposed options;
- lastly, safety analyses aim to identify the major factors of repository performance in granite medium with respect to the objective of human and environmental protection. They also assess the robustness of the proposed design options.

1.2.2 The support of international co-operation and mobilisation of the national scientific community

Andra's programme has extensively relied on foreign studies. Indeed, Andra has taken an active part in experimentation programmes carried out in Swedish, Swiss and Canadian underground laboratories.

The main themes of co-operation involved issues concerning the structuring of a granite massif and its fracturing, survey methods, underground natural water flow and radionuclides retention capacity of the rock (Figure 1.2.1).

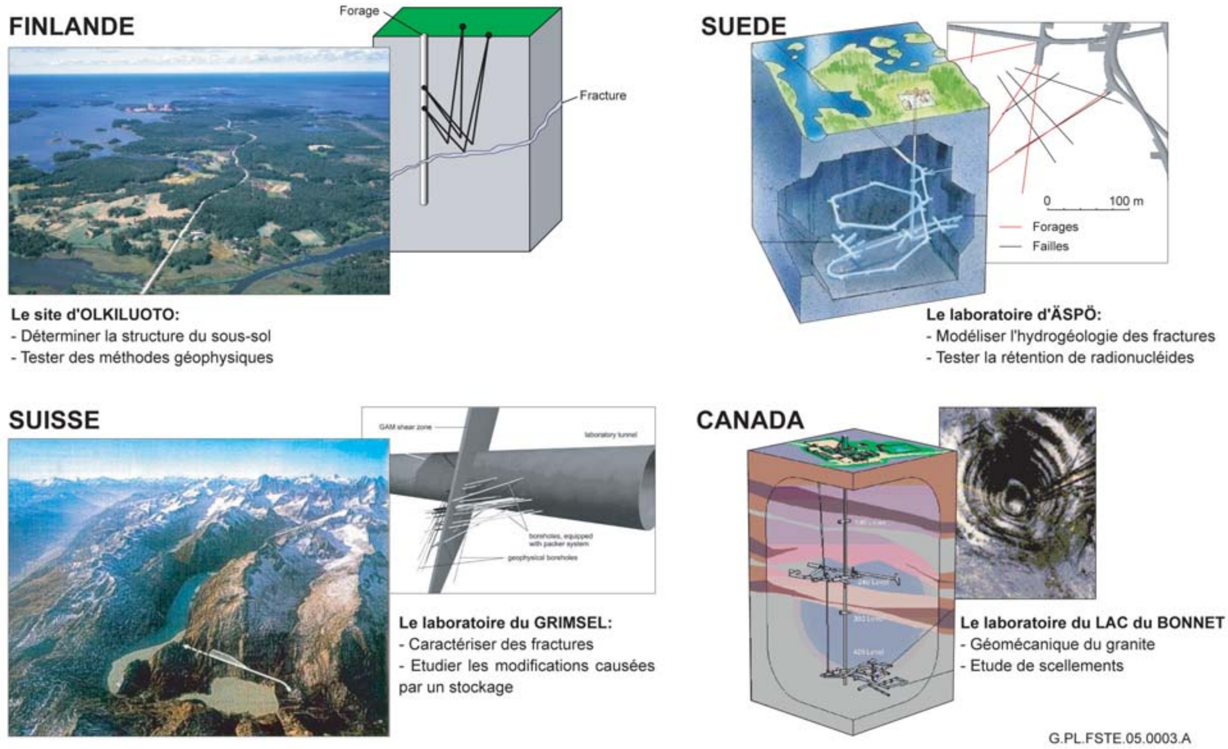

Figure 1.2.1 The study of the granite medium in the context of international co-operation

Demonstration elements have also been acquired in foreign laboratories. These elements relate to the installation and behaviour of engineered components of a repository such as seals, backfill, engineered barrier, etc.

Lastly, the study approach has benefited from feedback acquired for the safety analysis of a repository in a granite medium, particularly from Sweden and Finland.

This approach has therefore made the most of the extensive knowledge acquired internationally on the studies into a repository in a granite medium.

Andra has also set up scientific partnerships with national institutions (the Atomic Energy Commission "CEA", the French Geological Survey "BRGM", the research group "GdR Forpro" set up with the French National Research Centre "CNRS" and Paris School of Mines (ecole des Mines de Paris"). The participation of French research teams in foreign programmes has allowed the careful examination of the transposition to the French geological context of the results obtained abroad.

1.3 Tome structure

This document describes at first HLLL waste followed by the inventory model that forms the basis for the repository study. These elements are set out in chapter 2.

Chapter 3 provides a presentation of the functions of a repository in relation to the safety and reversibility objectives, and a presentation of the major characteristics of the granite medium. It then sets out the main technical orientations that form the basis of the design of a reversible disposal sytem in a granite medium.

The general architecture of a repository is described in chapter 4. This description highlights the link with the structure and fracturing of granite, and shows that this design is compatible with the implementation of various activities throughout the disposal process.

Chapters 5 and 6 analyse the issues relating to the design of repository zones for intermediate-level long-lived waste as opposed to high-level long-lived waste (B and C waste respectively).

In France spent fuel is not considered as waste. Nevertheless Andra has considered spent fuel disposal for cases where it is not reprocessed. Chapter 7 presents the possibilities of transposing the Swedish and Finnish designs to the French context, as these countries have opted for direct disposal.

The presentation highlights the respective functions of the various components (engineered elements and granite medium) and the proposed technical arrangements, for each repository zone described (B, C and CU). Implementation of the disposal process is also demonstrated as consistent with a reversibility rationale.

To conclude, chapter 8 summarises the main lessons drawn from the study. It emphasises the possibility of incorporating repository architecture in a granite formation and proposing a limited number of options to deal with the variability of characteristics presented by this medium in the French context.

2 High-level and long-lived waste (HLLL waste)

This chapter describes the primary HLLL waste packages that are considered for the repository feasibility study. It draws particular attention to their diversity. It is based on the results of the work carried out jointly between Andra and the producers concerning (i) surveying, and (ii) collecting and structuring the knowledge.

Firstly it sets out the waste production scenarios underpinning the inventory considered. The survey of existing waste is based on knowledge of past and present processes, production reports for each facility, identification of storage sites and control of their contents. In considering future waste, hypotheses have been formulated concerning the continuation of production by the various facilities. For waste from nuclear power plants, several scenarios have been selected to cover the various possible situations: ongoing reprocessing of spent UOX fuel consistent with current industrial practice, reprocessing of URE and MOX fuel, possible increase in the heat rating of vitrified C waste and the exploratory hypothesis of direct disposal of UOX, URE and MOX fuel.

This chapter then presents the two categories of waste that fall into the framework of the 30 December 1991 Act. It also provides an inventory model [3] which forms the basis for constructing all the design and dimensioning studies for the repository. The model brings together all the various waste families by defining waste "reference packages" (or package types) covering each a more or less important range, varying in extent, of primary waste packages. The notion of waste reference package is an essential element structuring the technical options considered in response to the diversity of primary waste. It is therefore a key to reading the following chapters. The inventory model stipulates which hypotheses are adopted for the number of primary packages to be incorporated for each study scenario.

2.1 The production of HLLL waste, study scenarios

The activity sectors producing the greatest volume of HLLL waste come within the nuclear power industry (EDF electricity-generating reactors, COGEMA fuel reprocessing plants, MELOX plant producing MOX fuel) or research and national defence activities (CEA centres).

The study must also consider waste produced upstream of the cycle, during uranium ore processing operations, and end-of-life radioactive objects from various industrial and medical activities.

Currently, spent fuels removed from "PWR" pressurised water reactors (58 of these are currently operated) are reprocessed in the La Hague plants, except for URE and MOX fuels, prepared from reprocessed uranium and plutonium respectively, which at present are stored in pools [4].

Reprocessing operations produce various types of waste, either directly resulting from spent fuel (fission product solutions and minor actinides, fuel assembly cladding waste), or linked to the use of facilities for maintenance operations (technological waste resulting from replacement of parts and other equipments) or radioactive effluent treatment (sludge). Currently, waste is conditioned in-line in the UP2-800 and UP3 plants at La Hague. In previous-generation plants (UP2-400 at La Hague and UP1 at Marcoule, now shut down), where fuels from various reactor generations were reprocessed, especially the first-generation NUGG (Natural Uranium-Graphite-Gas), part of the waste was stored in unconditioned form in specific facilities. However, with the exception of the so-called "UMo" solutions currently stored at La Hague, it should be noted that all fission product solutions, as well as effluent sludges at Marcoule, have been conditioned.

In addition, the operation of electricity-generating nuclear reactors requires systems for starting up and controlling the reactors. After a certain time these are replaced and become waste.

This mainly concerns neutronic poison and control rod assemblies and, to a lesser extent, waste such as source clusters and metal parts (thimbles and pins for example). All waste currently produced is stored in pools close to the reactors.

Research carried out at the CEA, especially on behalf of the French nuclear power programme, and the routine operation and maintenance of its facilities, are other sectors that have produced a wide range of waste. Most of this waste, made up of intermediate-level solid and liquid effluent waste, has been conditioned using immobilisation materials and packages of various types and geometry.

Finally, activities linked to national defence produce intermediate-level technological waste.

For the repository studies, the package inventory (in terms of type and quantity) includes all waste already produced as well as waste that may be produced through operating existing nuclear facilities. With regard to future production, this implies the need to formulate waste production and conditioning hypotheses, especially concerning management of nuclear power plants fleet.

Currently 58 pressurised water reactors, commissioned between 1977 and 1999, are operated to produce electricity. The tonnage of nuclear fuels removed from these reactors over their total operating period is estimated at 45,000 metric tons of heavy metal (tHM). This estimation is based on a combination of hypotheses concerning (i) the average lifetime of units (forty years), (ii) power production (16,000 terawatt-hours total production), (iii) the gradual increase of nuclear fuel "burn-up" in the reactors[1].

The fuel types considered and the corresponding average fuel burn-up is as follows:

- three generations of uranium oxide fuels: UOX1, UOX2, UOX3, irradiated respectively at 33 gigawatt-days per metric ton of fuel (GWd/t), 45 GWd/t and 55 GWd/t, on average;
- fuels containing recycled uranium (URE) irradiated on average at 45 GWd/t;
- mixed uranium oxide and recycled plutonium oxide fuels (MOX) irradiated at 48 GWd/t on average.

On this basis, four nuclear fuel management scenarios were selected for the studies. The principle behind these scenarios is to include various possible industrial strategies without singling out any of them. This process makes it possible to consider a very wide range of waste types and examine the technical aspects of the various packages.

The first three scenarios, designated as S1a, S1b and S1c, correspond to an ongoing reprocessing of spent fuel removed from EDF reactors. Scenario S1a supposes that all these fuels (UOX, URE and MOX) are reprocessed. This scenario includes the hypothesis of incorporating fission product mixtures and minor actinides from UOX and MOX fuels in glass. Also, for study purposes, it is assumed that a very small part of the plutonium from reprocessed UOX fuels is incorporated in some packages. This scenario therefore covers a variety of vitrified C package typologies.

In scenarios S1b and S1c, MOX fuels are not reprocessed, allowing the hypothesis of their direct disposal to be explored.

Scenarios S1b and S1c have been separated in order to study, in scenario S1b, the possibility of increasing the waste concentration in glass, compared with the packages currently produced; this greater concentration would result in a slightly greater release of heat from the packages.

Finally, a fourth scenario, designated as S2, which supposes that reprocessing is stopped, is used for the exploratory study of direct disposal of UOX and URE fuels, as well as the MOX fuels considered in scenarios S1b and S1c. In this scenario the fuels are considered to be waste, which, we should recall, is not the present case.

To be able to estimate the quantity of waste produced, scenarios S1a, S1b and S1c are based on the following distribution of various types of fuels removed from existing reactors: 8,000 tHM of UOX1 (33 GWd/t), 20,500 tHM of UOX2 (45 GWd/t), 13,000 tHM of UOX3 (55 GWd/t), 800 tHM of URE (45 GWd/t) and 2,700 tHM of MOX (48 GWd/t). In scenarios S1b and S1c, the direct disposal study concerns all the 2,700 tHM of spent MOX fuels.

[1] The burn-up of a nuclear fuel assembly expresses the energy produced in the reactor by the fissile material that it contains (uranium oxide or mixture of uranium and plutonium oxides)

Scenario S2 takes the hypothesis that reprocessing of some of the UOX fuels will continue until 2010 (8,000 tHM of UOX1 and 8,000 tHM of UOX2), but will cease after this date. Suspending uranium and plutonium recycling changes the overall distribution of the types of fuel removed from the reactors. Direct disposal of non-reprocessed fuels will then involve 29,000 tHM, including 12,500 tHM of UOX2, 14,000 tHM of UOX3, 500 tHM of URE and 2,000 tHM of MOX.

The studies concern conditioned waste. Conditioning processes have therefore been defined for existing unconditioned waste as well for future waste. The hypotheses adopted take the industrial processes currently implemented by the producers: vitrification, compaction, cementation and bituminisation.

The scenarios considered also allow a robust approach for the repository study in light of possible management changes of the nuclear cycle back-end.

In addition to these scenarios, the issue of management of spent fuel from French reactors other than the EDF pressurised water reactors (especially research and National Defence reactors) was also considered. In all events, their reprocessing will only produce a marginal quantity of waste compared with the waste from reprocessing EDF fuel. On an exploratory basis, the possibility of direct disposal of these fuels was given special attention, without predicting the choices concerning their management.

2.2 The study considers two waste categories and spent fuel

Intermediate-level long-lived (B waste) and high-level (vitrified C waste) are the waste categories included in the study. Together with spent fuel, they form distinct categories because of the different challenges they raise for repository functions.

2.2.1 High-level C waste (or vitrified waste)

This waste corresponds to the non-reusable materials contained in the solutions resulting from the reprocessing of the spent fuel: fission products, minor actinides and activation products.

Their high-level β-γ activity causes a *large thermal release* which decreases over time, mainly with the radioactive decay of the intermediate radioactive half-life fission products (caesium137, strontium90). *They are incorporated today in a matrix made of borosilicate glass (glass R7T7*[2]*)*, whose confinement capacity is particularly high and durable (several hundreds of thousands of years) when it is under favourable physico-chemical environmental conditions.

The radionuclides are thus spread uniformly in the vitreous matrix. This vitrified waste is poured into stainless steel drums, to make up C waste (vitrified) primary packages. C waste primary packages are described in chapter 6.

2.2.2 Intermediate-level long-lived B waste

This comes mainly from nuclear fuel manufacturing and processing plants and research centres. It therefore includes a large variety of items such as structure elements of fuel assemblies (fuel rod claddings called "hulls", endpieces called "end-caps", assembly holding grids, etc.), effluent reprocessing sludge, miscellaneous materials (filters, pumps, etc.). They are for the most part metals, but organic and inorganic compounds are also present (plastics, cellulose, etc.).

They have low or intermediate β-γ activity, consequently, they release no heat or hardly any. However, the quantity of long-lived elements that it contains justifies a very long-term containment, similarly to C waste.

Depending on its kind, *B waste is conditioned in bitumen* (sludge from effluent reprocessing), *in concrete or by compacting* (hulls and end pieces and technological waste). The thus-conditioned waste is placed in concrete or steel drums. These make up the B waste primary packages which are both

[2] R7 and T7 identify the two COGEMA vitrification plants at La Hague facility.

more numerous and more diverse through their conditioning. B waste primary packages are described in chapter 5.

2.2.3 Spent fuel

In terms of mass and quantity, spent fuel as studied in the exploratory direct disposal hypothesis essentially originates from fuel removed from EDF PWR reactors fleet. The radiological inventory of this spent fuel generates significant heat release as it does for C waste. However this heat release takes longer to decay because of the contribution of plutonium and isotopes produced by decay (daughter products). Other specific characteristics of this spent fuel are the large dimensions of the assemblies and their fissile material content in relation to the issue of criticality risk.

As previously indicated, fuels from former cycle back-ends (NUGG, EL4), research reactors and National Defence activities are to be added to the spent fuel from the PWR reactors. A description of these spent fuels is available in chapter 7.

2.3 The inventory model

The review of waste and the definition of their conditioning mode lead to a very wide variety of HLLL waste primary package families (total of 61). They differ in chemical and radiological content, their thermal and radiation output according to the presence of certain radionuclides, their nature, packaging geometry and quantity.

All the preceding data and hypotheses on the primary waste packages have been collated in an "inventory model" [3].

This inventory model, on the basis of which both the clay and granite medium repository studies are grounded, is organised using a tree-diagram structure. The tree-diagram structure groups primary packages showing similar issues in terms of waste management.

To cover the characteristics of the primary packages thus grouped, the inventory model defines "waste reference packages" (package types) which are representative of these groups.

The tree-diagram nomenclature adopted to identify the waste reference packages will be used throughout the following chapters. It has three levels. Level 1 differentiates in traditional fashion between the various categories (B waste, C waste and spent fuel identified by the letters "CU"). Various types of packages are identified within each of these categories, particularly for category B waste, based on waste type (sludge, technological waste, cladding waste from fuel assemblies, etc.) and conditioning methods (compacting, bituminisation, cementation). Breaking down level 1 of the inventory model into one or even two additional sub-levels, when necessary, provides a more detailed description on the variability of primary waste packages for design study, modelling and repository safety evaluation purposes. The following criteria are taken into account to differentiate between level 2 or level 3 waste reference packages: physico-chemical characteristics of the conditioned waste (in connection with the waste material and conditioning matrices), heat rating and irradiation levels of the packages (in connection with the radiological inventory) and container characteristics (dimensions, materials). Thus, for waste package types encompassing a wide range of primary package families, such as the B3 waste reference package, a second level in the tree-diagram is created to group packages initially based on the type of package materials (concrete, steel) and whether or not the conditioned waste is homogenous.

The physico-chemical characteristics of the conditioned waste have tremendous influence on the design choices, so that the packages are placed in favourable environmental conditions to limit their alteration over time. They determine the initial and long-term containment capability of the packages and disturbances potentially caused by their degradation. These disturbances include (i) the release of products likely to increase radionuclides solubility or to complex a significant number of radionuclides, (ii) gas production by radiolysis or corrosion of materials and (iii) the formation of potentially aggressive species for the surrounding materials. In particular, packages containing organic waste are singled out. Gaseous releases are input data for the study of the ventilation of both facilities and disposal packages, and for the risk analysis in the operating and observation phases.

Container characteristics (dimensions, weights, gripping systems) are important parameters in designing disposal packages, architectures and operating methods.

Thermal data are used for (i) repository thermal design and (ii) to assess behaviour. The irradiation level of the packages plays a part in designing the radiological protection methods, based on the radiological protection objectives adopted.

The inventory model defines the number of primary waste packages for each waste reference package taken into account in the study, including their total volume, for every scenario S1a, b, c and S2 introduced above.

2.3.1 Choice of reference packages, nomenclature used subsequently

The various groups of primary packages presented in Section 2.2 differ in (i) the thermal release level of the packages (B waste with nil to moderate thermicity, vitrified C waste and spent fuel with greater thermicity) and (ii) the waste kinds and conditioning methods.

Sixteen reference packages have thus been identified at level 1 of the inventory model tree-diagram by the package grouping possibilities. These include:

- eight B waste reference packages, graded B1 to B8, while differentiating between cemented cladding waste and compacted cladding waste;
- five vitrified C waste reference packages, C0 to C4 ;
- two PWR spent fuel reference packages, CU1 and CU2 (respectively UOX and MOX fuels), to which is added reference package CU3 (grouping all the other fuels with a far lower thermal release).

As indicated above, some of these reference packages are sub-divided, into level 2 and level 3 reference packages.

Thus, B2.1 and B2.2 reference packages present different geometries.

B3 reference packages, grouping a wide range of package families, are listed on two levels. Package groupings at level 2 have been defined on the basis of both the materials used for the containers and the homogenous or heterogeneous nature of the conditioned waste:

- B.3.1: heterogeneous waste contained in concrete envelopes;
- B.3.2: homogenous waste contained in concrete envelopes;
- B.3.3: heterogeneous waste contained in metallic envelopes;

The level 3 reference package listing corresponds to the chemical nature of the waste, the risk of hydrogen production and the package dimensions (level 3 reference packages associated respectively with level 2 reference packages: B.3.1, B3.2 and B.3.3 are classified by increasing size order):

- B3.1.1, B3.1.3, B3.2.1, B3.2.2, B3.3.2: packages potentially generating hydrogen;
- B3.1.2, B3.3.1, B3.3.3, B3.3.4: packages containing organic matter and generating hydrogen.

Distinction is made between the B5 reference packages based on the type of waste and associated characteristics (chemical, radiological, heat transfer):

- B5.1/B5.2: these reference packages differ from other cladding waste conditioned in CSD-C by their higher thermicity and also by the presence of technological waste. These reference packages can therefore be used to study two potentially different CSD-C populations according to their technological waste content: B5.1 reference packages cover technological waste containing organic matter while B5.2 reference package contains none;
- B5.3: this is cladding waste only, with no organic matter, very low thermicity of the waste given its age;

- B5.4: this reference package stands out due to the nature of the waste (magnesium), no organic matter and no thermicity.

Distinction is made in B6 reference packages in terms of waste type and materials and envelope geometries:

- B6.1: different envelope geometry from the B6.2 to B6.5 reference packages (B6.2 to B6.5 envelopes are identical);
- B6.2: cladding waste made up of steel, zircaloy and nickel alloy;
- B6.3: magnesium cladding waste;
- B6.4: packages containing organic technological waste and generating hydrogen;
- B6.5: packages containing metallic technological waste only.

Distinction is also made between B7 and B8 reference packages based on the waste type and materials and the envelope geometries.

Reference packages C0.1, C0.2 and C0.3 separate the C0 reference packages based on the type and chemical composition of the vitrified waste, its radiological and heat transfer characteristics and the package dimensions.

The reference packages and their titles are given in table 2.3.1

Table 2.3.1 List of model inventory reference packages

Reference packages	Cat.	Level 1	Level 2	Level 3	Types of waste contained in the reference packages
Activation product waste	B	B1			CSD-C containing activation product waste from PWRs and fast reactors
Bituminised waste		B2	B2.1		238- and 245-litre bituminisedn drums
			B2.2		428-litre bituminised waste drums
Technological and miscellaneous cemented or compacted waste		B3	B3.1	B3.1.1	1000-litre concrete containers reconditioned or non-reconditioned in metal containers
				B3.1.2	Concrete containers (CAC and CBF-C'2) containing miscellaneous technological waste
				B3.1.3	1800-litre concrete containers containing miscellaneous waste
			B3.2	B3.2.1	500-litre concrete containers (sludges and concentrates)
				B3.2.2	1200-litre concrete containers (CBF-C'2) containing CEDRA and AGATE waste
			B3.3	B3.3.1	Standardised containers for compacted waste (CSD-C) containing alpha waste
				B3.3.2	Multipurpose storage drums (EIP) containing powdered cemented waste
				B3.3.3	500-litre steel containers containing miscellaneous waste
				B3.3.4	870-litre steel containers containing miscellaneous waste
Cemented cladding waste		B4			Drums of cemented hulls and end caps
Cladding waste with or without technological, cemented waste		B5	B5.1		CSD-C containing a mixture of hulls and end-caps and technological waste (including organic waste)
			B5.2		CSD-C containing a mixture of hulls and end-caps and metallic technological waste
			B5.3		CSD-C containing cladding waste from PWRs and fast reactors with no technological waste
			B5.4		CSD-C containing magnesian cladding waste
Cladding waste and technological waste in drums		B6	B6.1		180-litre steel containers containing AVM operating waste
			B6.2		EIP drums containing metallic cladding waste
			B6.3		EIP drums containing magnesian cladding waste
			B6.4		EIP drums containing technological and organic waste
			B6.5		EIP drums containing metallic technological waste
Sources		B7	B7.1		Source blocks
			B7.2		CSD-C containing PWR primary and secondary source fuel rods
			B7.3		EIP drums containing sealed sources
Radium- and americium-bearing waste		B8	B8.1		EIP drums containing radium-bearing lead sulphate
			B8.2		870-litre steel containers containing radium- or americium-bearing lightning conductor tips
			B8.3		EIP drums containing ORUMs (objects containing radium for medical use)
Vitrified waste	C	C0	C0.1		Vitrified PIVER waste
			C0.2		Vitrified UMo waste
			C0.3		Vitrified AVM waste
		C1			"Current thermal" UOX/enriched recycled uranium vitrified waste
		C2			"Future thermal" UOX/enriched recycled uranium vitrified waste
		C3			UOX/MOX vitrified waste
		C4			UOX +Pu vitrified waste

The study of the following reference fuels is added to these waste reference packages:

Reference packages	Cat.	Level 1	Level 2	Level 3	Types of waste contained in the reference packages
EDF PWR Fuel	CU	CU1			PWR uranium oxide and enriched uranium spent fuels
		CU2			Spent MOX PWR fuel
CEA fuels		CU3	CU3.1		UNGG and EL4 spent fuel
			CU3.2		Célestin spent fuel
			CU3.3		Nuclear propulsion spent fuel

2.3.2 Number and volume of primary packages considered

In the framework of the scenarios presented in section 2.1, the number of reference packages is quantified by the inventories and waste production forecasts established by producers and assessed by Andra based on data provided.

For forthcoming waste, excluding spent fuel reprocessing, Andra has added dimensioning margins to take uncertainties into account. Note that disposal possibilities for some waste packages under other disposal solutions have not been taken into account so as to have cautious estimations available.

For past production, inventories are based on data established by the producers. Inventories for reprocessing waste are deduced from the hypothetical electricity production by the facilities.

The numbers and volumes of packages counted into the studies for B and C waste are presented in Table 2.3.2. The indicated volumes correspond to the conditioned waste volumes with the hypotheses formulated below.

Table 2.3.2 Overall quantitative data, in number and volume of packages, for B and C waste reference packages

Reference package	Production site	Scenario S1a		Scenario S1b		Scenario S1c		Scenario S2	
		Number	Volume (m^3)	Number	Volume (m^3)	Number	Volume (m^3)	Number	Volume (m^3)
B1	EDF	2 560	470	2 560	470	2 560	470	2 560	470
B2	COGEMA La Hague	42 000	10 000	42 000	10 000	42 000	10 000	42 000	10 000
	COGEMA Marcoule	62 990	26 060	62 990	26 060	62 990	26 060	62 990	26 060
Total of B2		**104 990**	**36 060**	**104 990**	**36 060**	**104 990**	**36 060**	**104 990**	**36 060**
B3	CEA	15 060	13 370	15 060	13 370	15 060	13 370	15 060	13 370
	COGEMA La Hague	9 890	10 470	9 890	10 470	9 890	10 470	7 340	7 750
	COGEMA Marcoule	7 990	3 420	7 990	3 420	7 990	3 420	7 990	3 420
Total of B3		**32 940**	**27 260**	**32 940**	**27 260**	**32 940**	**27 260**	**30 390**	**24 540**
B4	COGEMA La Hague	1 520	2 730	1 520	2 730	1 520	2 730	1 520	2 730
B5		42 600	7 790	39 900	7 300	39 900	7 300	13 600	2 490
B6	COGEMA Marcoule	10 810	4 580	10 810	4 580	10 810	4 580	10 810	4 580
B7	EDF CEA Andra	3 045	1 440	3 045	1 440	3 045	1 440	3 045	1 440
B8	CEA Andra	1 350	775	1 350	775	1 350	775	1 350	775
C0.1	CEA	180	10	180	10	180	10	180	10
C0.2	COGEMA La Hague	800	140	800	140	800	140	800	140
C0.3	COGEMA Marcoule	3 140	550	3 140	550	3 140	550	3 140	550
Total of C0		**4 120**	**700**	**4 120**	**700**	**4 120**	**700**	**4 120**	**700**
C1	COGEMA La Hague	4 640	810	4 640	810	38 350	6 710	4 640	810
C2		990	170	27 460	4 810	0	0	5 920	1 040
C3		13 320	2 330	0	0	0	0	0	0
C4		13 250	2 320	0	0	0	0	0	0

Table 2.3.3 shows the state of existing B and C waste volumes at the end of 2003, whether or not conditioned and waste volumes still to be produced under scenario S1a.

Table 2.3.3 State of B and C waste volumes produced at end of 2003, conditioned or not, and still to be produced (scenario S1a)

	Reference packages	Volume of waste produced at year-end 2003 (m^3)		Volume of waste to be produced (m^3)
		Conditioned	Unconditioned	
B	B1	0	250	220
	B2	27 790	7 620	650
	B3	13 895	4 910	8 455
	B4	2 730	0	0
	B5	135	530	7 125
	B6	0	4 580	0
	B7	155	1 285	0
	B8	15	760	0
Total volume of B waste (m^3)		**44 720**	**19 935**	**16 450**
C	C0	540	140	20
	Other glasses	880	0	4 750
Total volume of C waste (m^3)		**1 420**	**140**	**4 770**

Quantitative data relating to PWR fuels are found in Table 2.3.4.

Table 2.3.4 Number of PWR fuel assemblies

	Production sites	Number of PWR fuel assemblies			
		Scenario S1a	Scenario S1b	Scenario S1c	Scenario S2
"Short" UOX AFA-2GE assembly, type CU1	EDF	0	0	0	27 200
"Long" UOX AFA-2LE assembly, type CU1		0	0	0	26 800
Total UOX assemblies, type CU1		**0**	**0**	**0**	**54 000**
"Short" MOX AFA-2GE assembly, type CU2	EDF	0	5 400	5 400	4 000
Total MOX assemblies, type CU2		**0**	**5 400**	**5 400**	**4 000**

Furthermore, 5,810 primary claddings are to be considered for type CU3 fuels, if appropriate.

3 Study for the design of a repository in a granite medium

This chapter sets out the functions that a repository in a granite medium must fulfil over the course of time. Essentially they derive from the objective of protecting the environment and future generations from the risks likely to be caused by the waste. Reversibility is also taken into account in the formulation of these functions. The safety functions are identified from a functional analysis as part of Andra's safety approach adapted to the studies of the granite medium.

This chapter continues with the properties of the granite medium on which the repository design is bases. The properties are expressed in terms of general characteristics common to all granites and also in terms of possible variations of the different types of granite found in the French geological context.

The chapter closes with the elements of repository design bases on the capacity to take the most of the favourable properties of granite over long periods.

3.1 Definition of the geological repository functions

3.1.1 Functional analysis: repository design is driven by safety

The fundamental objective of long-term management of High Level and Long Lived waste (HLLL) is to protect, in the extremely long term, man and the environment from waste-related risks. The response provided by a repository consists of confining this waste in a deep geological formation to prevent dissemination of radionuclides in the waste.

This confinement is passively ensured over large timeframes (up to several hundreds of thousands of years) without eventually requiring any maintenance and monitoring, as indicated in the Basic Safety Rule RFS III.2.f.[2].

The RFS identifies confinement barriers: waste packages, engineered barriers (materials placed between the package and the rock) and the repository host formation which protect the waste from water circulation and intrusive human actions and limit and delay radionuclide transfer in the geological medium and biosphere.

In line with an iterative approach for design and safety, Andra has assigned safety functions to all repository components with a significant role (host formation, waste package and engineered barrier). The characteristics of these components (for example type of materials and waste package thickness, cell dimensions, etc) have been determined for safety by incorporating the potential disturbance caused by the environment and uncertainties.

Design of a multi-function system thus completes the notion of the multi-barrier system. Certain components contribute to fulfilling the same function (complementarity) and to maintaining the function in the event of failure of one of them (redundancy). This safety function-based approach associated with checking of the level of performance of these functions is common to operational safety.

3.1.2 Long-term safety functions of a repository

Firstly, an underground repository protects the waste from erosion phenomena and main human activities which after hundreds of thousands of years only affect a superficial ground thickness.

In this context, controlling dispersion of the radionuclides contained in waste relies on three major functions that must be performed by the repository:

- Preventing water circulation,
- Limiting radionuclides release and immobilising them within the repository,
- Delaying and attenuating migration of radionuclides which may have been released by the waste.

Eventually, these three functions must be passively fulfilled (without human intervention). Some are only implemented at a late stage. For example, the repository's ability to limit radionuclide migration does not become operational until the waste packages begin to release radionuclides. Such functions are said to be latent during the period when they are available but not yet operative.

- **Preventing water circulation within the repository**

Confinement of radioactivity contained in the packages firstly consists of maintaining it immobilised there. The repository must therefore:

- limit water renewal around the packages, which is the main factor liable to alter package envelopes,

- prevent advective transport of the radionuclides in order on the other hand to restrict their possibility of migration through only diffusion, a very slow phenomenon, by limiting both water flow reaching the repository and water circulation velocity between the disposal cells and the water conducting faults of the granite medium.

- **Limiting radionuclides release and immobilising it within the repository**

Water arrival at the waste packages which constitute an initial radionuclide confinement barrier cannot be ruled out over time. Under these conditions, a repository function is to limit radionuclides release in the water and immobilise them in the waste or as close as possible.

By creating beneficial physico-chemical environmental conditions, repository installations can limit water alteration of the waste containers and ,within these containers, of the matrices (glass, bitumen and cement) in which the radionuclides are incorporated.

Once water has started to alter the waste packages, the role of the repository is to limit the mobility of radionuclides likely to be dissolved in the water by creating reducing geochemical conditions (completed with pH control) in order to maintain and re-precipitate these radionuclides in solid form (only some radionuclides, such as iodine 129 and chlorine 36, remain unaffected by these beneficial geochemical conditions).

- **Delaying and attenuating radionuclide migration**

One of the repository functions is to delay and disperse, within space and over time, the migration of radioelements and chemical toxics, which may have been released by the waste, in order to attenuate it:

- Migration of radionuclides dissolved in the water is controlled by diffusion, dispersion and retention in the granite, the host formation. Dissolution in water of radionuclides likely to be released in gaseous form enables these elements to be controlled in a similar way,

- Radionuclides migration can be also contained within certain repository components (engineered barriers and bentonite seal body, etc), and therefore delayed. This will contribute as well to reduce any possible impact on man and the environment.

3.1.3 Waste package reception function in a reversible disposal system

Would a repository be decided, its first operational phase would be to construct surface installations, access structures to the geological host formation and the first underground repository installations.

It would be followed by an operational phase during which the waste packages would be emplaced and the structures observed. In principle, no time limit is pre-set for this phase according to the reversibility approach. The likely timescale ranges from one to several centuries.

In practical terms during this phase, the repository offers package management flexibility, similarly to a storage.

However a repository is also designed to be sealed and thus made passive[3]: closure primarily entails backfilling and sealing the underground installations.

Reversibility implies flexible management and gradual implementation of the closure process, thereby gradually reducing the disposal system reversibility level as choices are made.

Thus repository construction, operation and then closure are organised into a sequence of stages, which can be carried out independently for each waste category.

Observation of the structures during the implementation periods of these processes, helps understanding the phenomenological evolution of the repository. It provides useful data when moving on from one stage to the following one and also in the event of a decision to reverse the process, up to retrieving the emplaced packages.

3.1.4 Safety approach during the construction-operation-closure phase

The industrial activities in terms of waste management call for similar "operational safety functions" to those practiced in nuclear installations.

Occupational safety and protection of the public and the environment during repository operation phases are essential elements in installation design. This includes assessment of the main risks that notably workers face due to radioactive waste and underground working conditions.

Even if there is only one example of an operational geological repository of long-life radioactive waste throughout the world (the “WIPP” medium-level long-lived waste repository[4] in the USA), there is a quantity of operating experience feedbacks on underground structures and handling of high level waste and spent fuel packages.

The hazards are thus well identified and arrangements to prevent them and mitigate their gravity are routinely implemented.

At the generic stage of the Dossier Granite 2005, operational safety studies were mainly based on those developed for the clay medium repository studies. They involved initial identification and ranking of hazards and a preliminary outline of associated management techniques. Granite context specificities were also checked (for example, the more likely exposure to radon risks) as well as particularities of the concepts proposed for granite compared to those defined for clay to ensure that they did not cause any specific problem for the initial approach.

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3 After closure, the safety functions mentioned earlier will be provided without the need for direct human intervention.

4 Waste Isolation Pilot Plant

3.2 The granite medium

Repository design initially aims to make the most of geological formation properties beneficial to underground disposal so that it fulfils the various safety functions assigned to it.

In the absence of a specific study site, the repository design study cannot be based on the description of a specific granite massif. Design principles adopted by Andra are mainly based on properties common to all granites (cf. § 3.2.1).

However, French granites have specific characteristics which the design studies must consider in order to propose relevant options. Andra has therefore drawn up a reference document on French granite details to identify the granite characteristics which could affect repository design. (cf. § 3.2.2).

3.2.1 Granite: its properties for radioactive waste disposal

The word granite has two meanings for a repository study: it designates both a rock and a geological formation. Generally granite, the geological formation, is organised into massifs[5]. Thus, the possibility of a repository in granite medium depends on rock properties and characteristics and geological context of the massif studied.

3.2.1.1 Granite rock: a hard, resistant rock

The common perception of granite as a stone used for a long time as a lasting ornament is that of a hard rock, of very low porosity and very low permeability.

Rock mechanical resistance is naturally of interest for construction of underground structures. This rock can be excavated without any significant ground support being required, over volumes compatible with repository dimensions and depth. This mechanical resistance is due to rock texture composed of quartz (crystallised silica) and feldspars (aluminium silicates). Quartz also contributes to the usually high thermal conductivity of the rock, which makes it a formation likely to easily dissipate heat emitted by radioactive waste.

Granite rock contains very little water: its water porosity is usually less than 0.5%. Rock permeability is very low and can be on the limit of accessibility to *in situ* measurements.

These characteristics constitute interesting properties a priori for a radioactive waste repository.

[5] Unlike sedimentary geological formations which are usually arranged in superimposed layers (for example: the clay formation of the Callovo-Oxfordian of the Meuse/Haute-Marne site), magmatic formations such as granites often have bulkier geometries (three-dimensional) than planar. In the case of granites the word massif is generic and applies to most of the arrangements likely to be met .

Figure 3.2.1 Left: unfinished obelisk at the Aswan granite quarry. Right: Vermont quarry

3.2.1.2 A granite massif: a formation of vast dimensions and properties caused by its geological history

A granite massif suitable to host a repository is a thick geological formation usually of vast dimensions, which given rock resistance offers great flexibility for repository architectural design.

However, on the scale of a massif, granite is not a monolithic homogenous geological formation. It is essential to have a thorough understanding and modelling of its structure with enough details to study how repository design could suit or not inside.

This understanding is based on detailed characterisation of the massif studied thanks to methods applied during successive stages of site surveying. The complementarity of these methods enables a gradual surveying approach to be defined and adopted to the site studied.

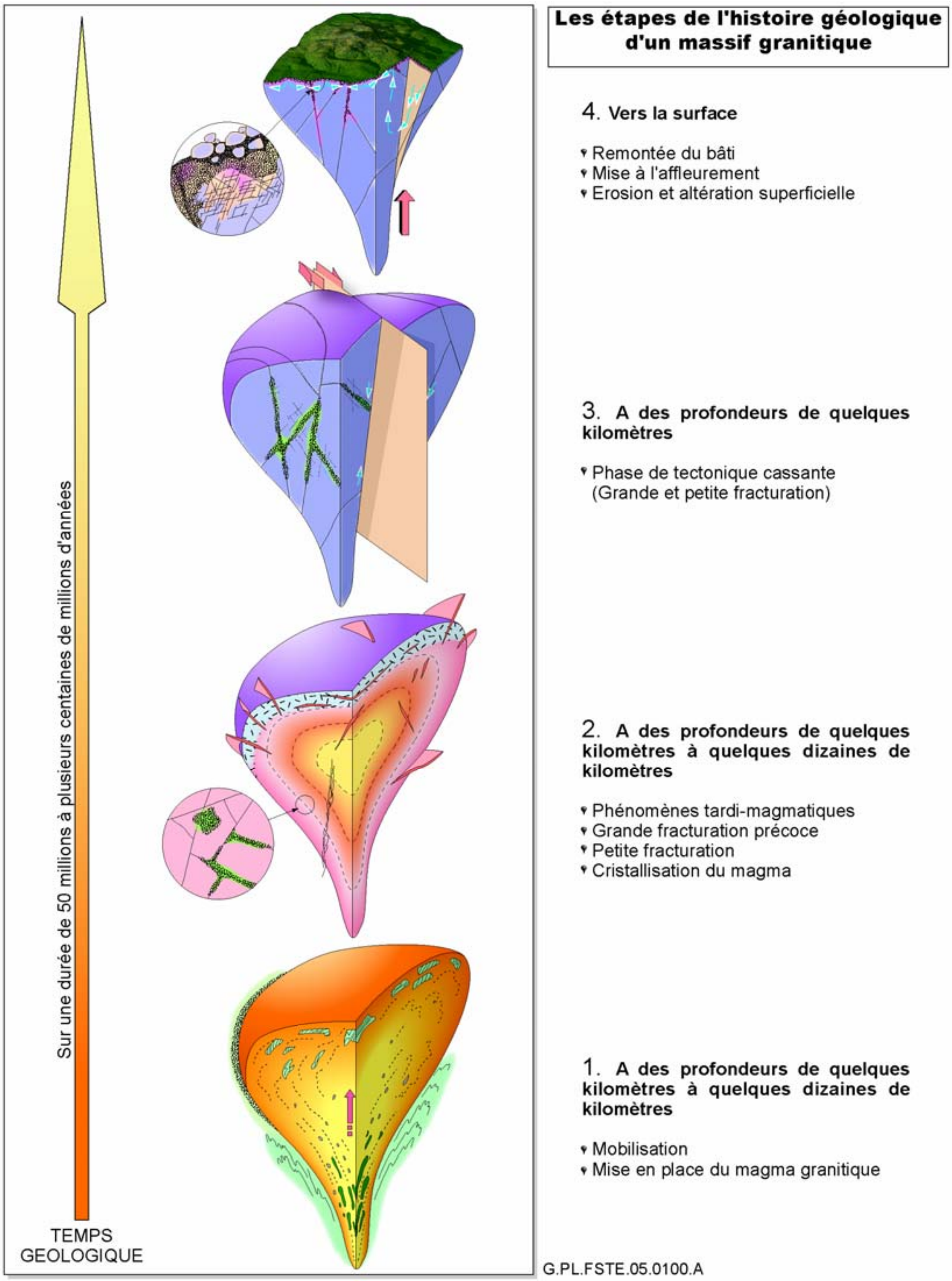


Figure 3.2.2 The geological history of a granite massif in stages

Interpretation of data collected is mainly based on reconstitution of the geological history of the granite massif. Compiling the geological history of a granite massif means understanding the phenomena which have produced and structured it over the course of time; it means as well integrating the various components of a massif according to a consistent and common historical rationale .

On granite outcrops, rock fractures outline is emphasised by the effects of alteration, decompression and then surface erosion of the rock. The effects of this surface erosion become less marked between surface and a depth of a hundred metres or so: the rock is "sound" and the variations in composition are the result of the granite's original geological history.

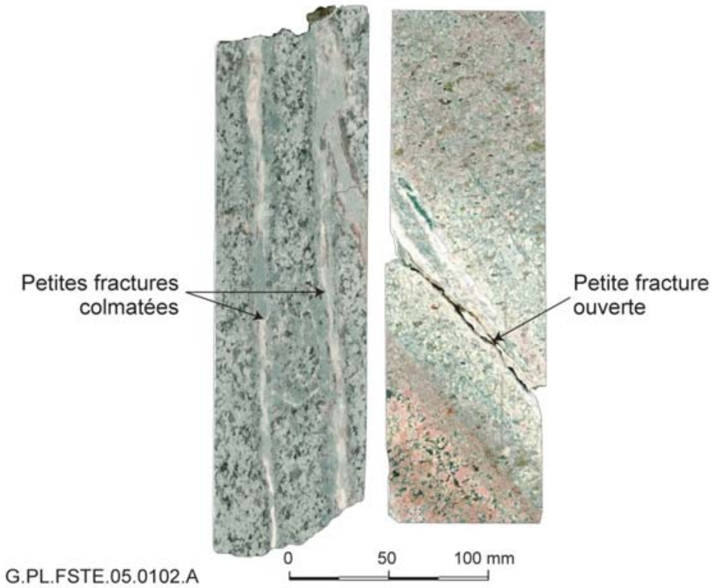
The lithological type and mineralogical composition of the rock can also change from one point to another in the massif, depending on the mode of granite formation. As a general rule, these variations are not of a type to significantly modify rock mechanical properties and permeability

More important is the pattern of granite fracturing resulting from its geological history. These fractures mean more or less marked discontinuity in rock properties which have to be taken into account for repository design.

Generally the size and number of fractures are correlated. Small-sized fractures, of metric to decametric dimension are much more numerous than large-sized fractures, of kilometric to pluri-kilometric extent. Medium-sized fracturing of decametric to hectometric dimension makes the transition.

Small-sized fractures can affect rock permeability where repository structures are to be constructed. Rock permeability depends on the properties of each of the small-sized fractures, their density and their extent. Small-sized fractures, which can be more or less connected are usually poorly water conducting. Granite permeability apart from large and medium-sized fractures is mainly low or very low and greatly limits water circulation.

Figure 3.2.3



Large-sized fractures, or faults, are the preferred pathways for water circulation in the granite which does not however mean that they contain large quantities of water. These are the largest faults, if they are not clogged with clay minerals, which store most of the granite water.

- **Slow underground hydro-geological flows**

Water present in granite faults moves very slowly underground. Movement is driven by hydraulic gradients related to topography. Schematically, the more contrasting the topography, the larger the gradients. However, the driving force which tends to move massif water is inhibited by high losses of hydraulic head in granite fractures. Irregularities in the detailed geometry of the fractures and the state of connections between fractures prevent water movements.

- **Underground chemical environment beneficial to a repository**

In underground granite, water usually has a chemical composition balanced with the rock and with fracture minerals. The chemical environment is thus of a reducing nature. Water pH balanced with granite rock normally approaches neutrality or is slightly basic.

These conditions are beneficial both to durability of the materials which can be used in a repository and immobilisation of most of the radionuclides.

3.2.1.3 Granite faults and fractures: ability to delay radionuclide migration

If granite fractures potentially constitute places of water circulation, therefore a potential means of transfer of radionuclides released by the repository, they are also the seat of phenomena likely to immobilise and delay this migration. This major aspect is subject to important studies at international level.

In particular, experiments carried out *in situ*, notably in the underground laboratory in Äspö (Sweden) have especially identified the various phenomena involved in delaying radionuclide migration in fractures and have led to understand clearly their nature.

Determination of relations between these phenomena and geological and mineralogical characteristics allow experimental results extrapolation to various types of granite depending on their own characteristics.

3.2.2 Variability of granites within the French geological context

The favourable properties of the granite medium likely to be considered for repository design differ from one granite massif to another one: they relate to its geological history in a given environment.

Although granite repository design is based on generic properties of the granite medium, it also incorporates specificities of a particular massif. Understanding the differences between granites enables them to be incorporated in the design and in the safety analysis in order to ensure that design options proposed fulfil the various safety functions. In the absence of a specific study site, Andra has carried out a typological analysis in order to collect data on the variability of French granites.

3.2.2.1 An analysis method adapted to a “site-less” study context

Variability of French granite properties has thus been understood through typological analysis based on a large sample. The analysis focussed on the Massif Central and the Massif Armoricain, the two largest areas in the French territory of outcropping crystalline basement.

Figure 3.2.4 Granites in France

Granite areas which, obviously, could not meet the main criteria of the Basic Safety Rule, RFS III.2.f, were ruled out of this study. The areas to be considered have a surface above 20 km^2 and are located away from large faults[6]. An inventory of 78 granite areas was thus taken into account for the study. [5].

- The analysis initially consisted of describing granite characteristics by evaluating their variations, proven and potential, and the way in which they can affect repository design options. As granite massifs had seldom been surveyed *in situ* underground, excluding few specific mining areas, their description was based on their outcrops mapping. Extrapolation of geometric characteristics and fracturing to the underground granite was carried out on the basis of geological arguments.
- Thermal and hydrogeological characteristics were defined from modelling and extrapolation. These were based on the borehole data available for aspects concerning hydrothermal "alterations" characterisation, thermal flux determination and transmissivity measurements. Sensitivity analyses corresponding to the level of uncertainty observed were also carried out by modelling.
- At the second stage, once the main characteristics of the inventory of granite massifs considered were sufficiently known, statistical analysis of their variability within the French geological context could be undertaken. This analyses lead to a granite classification (granite typology), with respect to each property studied, and to the appraisal of the breakdown of variations in properties of French granites.
- Moreover, a comparison of properties of the massifs studied and those of foreign granites provided validation of the use of the data collected in foreign underground laboratories.

[6] A "buffer" distance of 1.5 or 3-km has been adopted depending on fault size.

3.2.2.2 Main analysis information: variability of French granites properties for the repository design and safety analysis

- **Variability of mechanical properties**

Mechanical resistance of granite rock differs according to the types of granite, mainly depending on their hydrothermal alteration. However, generally these variations are not likely to cause any significant differences in the granite response to structure excavation. Differences for repository architectural design are thus minor and only affect detailed dimensioning of underground structures.[6].

- **Considerations relating to the variability of thermal properties**

The initial temperature of underground granite cannot be directly measured without a borehole. Modelling was thus based on thermal flux mapping in France and thermal conductivity of rocks depending on their quartz content. Estimations made for a depth of 500 metres have led to uncertainties concerning initial temperatures of between ± 2.8°C and ± 3.5°C depending on the massifs [7].

The graph (Figure 3.2.5) illustrates the relatively contrasting situations between the different types of French granite on this issue. Initial temperatures at a depth of 500 metres vary between 17 and 30°C. Rock conductivity has a value of between 2.4 and 3.8 W/m.K.

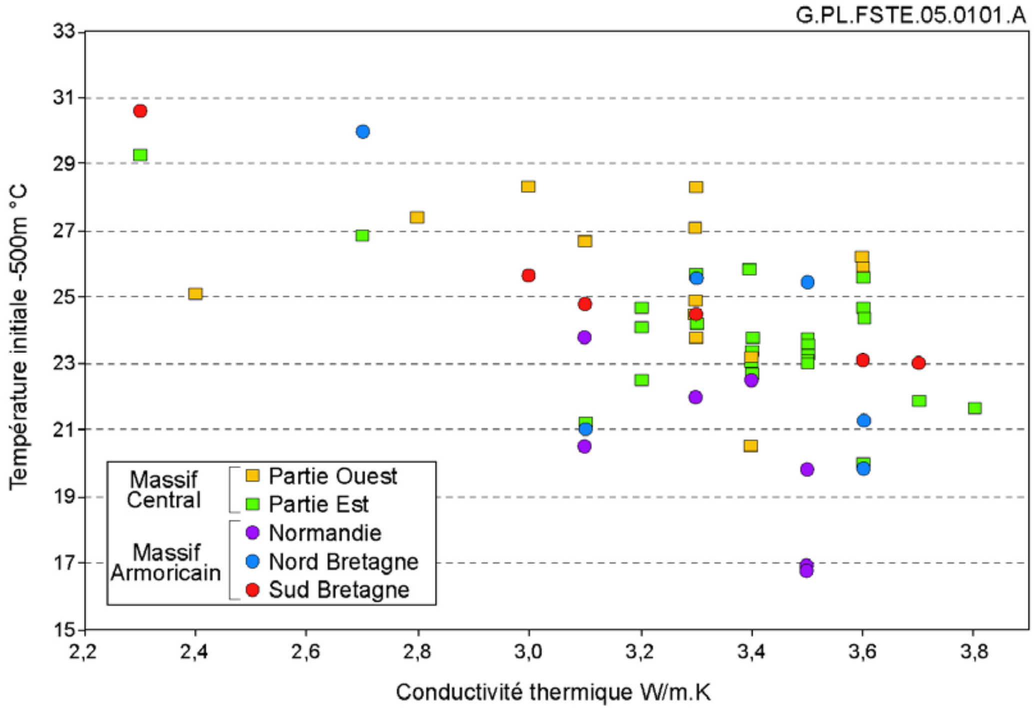


Figure 3.2.5 Correlation between thermal conductivity and temperature for granite massifs in France

Variability in the thermal properties of French granites influences the dimensioning of underground repository installations for highly exothermic waste (C waste and spent fuel). From this standpoint, the difference is significant with respect to Fenno-Scandinavian granites where underground temperature is lower by around 10°C.

For the least exothermic waste such as B5 and B1, this variability has little influence on repository dimensioning.

- **Variability of the hydraulic properties of the fractures**

Geometry of large-sized fracturing in granite is a significant element for repository architectural design on a specific site. On a particular site, the design of a repository depends on the major fracturing geometry. Depending on the tectonic history of the massif, the pattern is more or less regular and massif splitting is more or less pronounced. Analysis of a large number of French granites shows that even if the pattern of large-sized fracturing varies between massifs, the distribution of granite "blocks" where the repository could be constructed complies with rules relatively common to the French massifs studied. [8].

Small-sized fracturing of granite also has a significant impact for repository design. Rock ability to delay and attenuate radionuclide migration will mainly depend on the characteristics of small-sized fractures.[9]. Hydraulic conductivity of small-sized fractures is usually low or very low (less than 10^{-9}m/s). In the range of low permeability, the values can however vary mainly depending on the granites and types of fracture [10]. They depend on their geometry, orientation and potential natural clogging with minerals (Figure 3.2.3) and the same goes for radionuclide retention properties by fractures.

- **Considerations relating to the variability of morpho-structural contexts**

The morpho-structural context of granite has also been analysed. Site topography and morphology determine hydraulic gradients, which are the driving force of underground water flows. Differences between French massifs are significant. Typological analysis has identified three main morpho-structural granite arrangements which are taken into account in safety analysis: granite massifs in topographical depression compared to the surrounding geological formations, domed massifs and sloping massifs. Each type can correspond to more or less accentuated topographies.[11] (Figure 3.2.6).

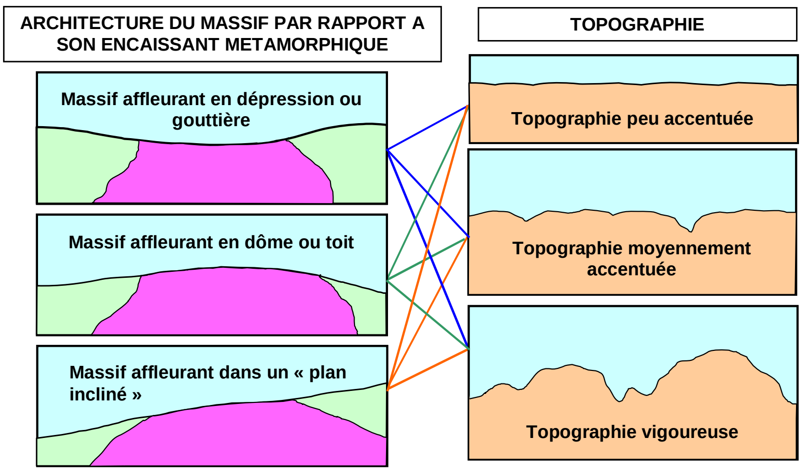


Figure 3.2.6 Role of the morphostructural context

- **Importance of hydro-geochemical variability**

The available inventory and analysis of chemical composition of groundwater in French granites show that different types of water are encountered: so called "alkaline" groundwater and carbo-gaseous one (Table 3.2.1).

Carbo-gaseous water is present in the Massif Central and can be linked to the geodynamic context and more or less old volcanic activity. Alkaline water corresponds to composition close to equilibrium with the granite medium.

Table 3.2.1 Examples of groundwater chemical compositions in various French contexts (contents in mg/l)

Type of water	Sites	pH	Na	K	Ca	Mg	Li	SiO2	Cl	SO4	HCO3
Alcaline	Site 1	8,8	101,16	5,55	5,13	0,02	0,54	50,26	8,05	96,99	109,76
	Site 2 .	8,86	124,84	4,54	34,07	0,08	0,38	23,87	148,89	129,63	167,70
Carbo-gaseous	Site 3	6,64	2195,54	170,46	62,92	47,15	9,85	47,73	3725,79	624,16	1798,93
	Site 4	6,80	958,68	24,83	142,28	85,07	4,10	21,62	42,54	21,13	3482,02
	Site 5	6,67	899,91	94,62	605,18	391,31	5,34	58,97	2006,47	249,67	2451,44

These differences in composition are not of a type to modify principles for the design options proposed. For certain granite massifs, they could lead to adjusting engineered barrier formulations to chemical composition of the water.

- **Considerations relating to the long-term geological evolution of French granite sites**

In terms of long-term geological evolution of a site, typological analysis of the granite massifs studied confirms that most of them are located away from active geodynamic areas, which means unlikely significant modifications in the long term to their geological arrangements, especially concerning underground fracturing [12]. Climatic changes and erosion can also alter the hydro-geological and topographical context of a site in the long term. Variations exist between massifs mainly due to differences in morpho-structural context. The analysis has thus identified the main arrangements encountered within the French context and phenomena which could come into play within a timeframe of 10,000, 100,000 and 1,000,000 years. It should be pointed out that, on a scale of 100,000 years, the models do not show any significant differences in evolution between the massifs. Beyond 100,000 years, the situation of each massif is to be specifically taken into account for the study of a particular site.

3.3 General options for the design of a repository in a granite medium

In order to fulfil long-term safety functions, the design proposed for a repository in granite medium consists of:

- using a variety of technical arrangements to make the most of the beneficial properties of the granite medium, its low permeability and mechanical resistance of granite rock in particular,

- designing engineered repository components (disposal packages, engineered barriers, backfills, and seals) so that they contribute to safety functions in terms of complementarity or redundancy with the granite medium.

- adopting design options aiming at limiting granite disturbances by the repository.

In addition to the long term and operational safety, design must meet reversibility requirement, closely linked to application of the principle of precaution provided for in the 30 December 1991 Waste Act. Beyond the possibility of removing emplaced packages (retrievability), reversibility is based on cautious and staged management of a repository, which, given the timeframes under consideration, leaves the options open for future generations.

These principles lead to adopting various technical arrangements for repository architecture and dimensioning, choice of materials for engineered components, and disposal processes. Some measures, architectural in particular, that are common to different categories of waste (B and C) and to spent fuel are presented below. Others, as for instance design of engineered components for example, that are specific to each waste categories are described in more detail in chapters 5, 6 and 7.

3.3.1 Making the most of favourable granite properties

The granite medium is characterised by rock of extremely low permeability, with high capacity for radionuclides retention and mechanically resistant. As granite is intersected with fractures liable to conduct water, making the most of its beneficial properties entails adapting repository architecture to this fracturing.

3.3.1.1 A compartmentalised architecture, adapted to the granite fracturing

Repository architecture is organised into different zones by major categories of packages: B waste, C waste and spent fuel. These zones are sufficiently far apart to avoid interaction between different types of waste, particularly from a thermal or chemical view point. Module separation in each repository zone also reduces quantities of waste and radioelements which would be affected in cases of system failure or intrusion.

Apart from these design principles, repository architecture and module separation is imposed by granite fracturing.

- **Construction away from the faults**

At repository scale, repository zones for the various categories of waste are constructed away from major faults of the granite massif.

Each repository zone is divided into repository *modules* grouping together a series of cavities the disposal cells) for the same type of waste. Modules are located in granite “blocks” not intersected by large- or medium-sized faults, considered as significant water conducting. One of the basic principles of a repository in a granite medium is *to construct disposal cells in very low permeability rock*. This does not mean that there can be no fracturing whatsoever in the rock but that small-sized fractures which may exist in cell walls do not conduct ground water or in small quantities. Therefore water which might come into contact with packages is minimal.

Disposal cells are of a “dead-end” type (therefore with only one access to repository drift), thus limiting possibilities of circulation of water coming from the drifts.

These principles regarding repository location and architecture meet the requirements laid down in Basic Safety Rule III.2.f, which states:

”Repositories in geological formations must be located

- *in crystalline mediums, within a host-block exempt from major faults, as the latter are likely to be preferential pathways of hydraulic movement. Disposal modules must be protected from medium-sized fracturing, although this may be intersected by access structures”.*

- **Construction of modules with respect to waste thermicity**

At the module scale, the principle of adapting architecture to fracturing works differently for each type of waste. The footprint required for different types of packages may necessitate implementation of different options (Figure 3.3.1).

The inexistent (or low) heat transfer by B waste makes possible a design with compact disposal cells and therefore requiring only a low volume of granite. B waste disposal thus requires a small footprint and adjustment of architecture to granite fracturing is then eased. Disposal modules are built so as to avoid fractures likely to cause water circulation.

C waste and spent fuel thermal characteristics define disposal modules dimensioning and lead to such large footprints that it is not possible to avoid the intersection of a module by a fracture, potentially water conducting[7].

Thus, repository architecture must be adapted to two levels of fracturing. Disposal cell size allows them to be built in very low permeability granite rock with minimal fracturing. Modules are installed in granite blocks, avoiding fractures which would too water-conducting (the so-called medium-sized fracturing as mentioned in RFS III.2.f). The large volume of granite rock available underground for a repository between 300 and 1000 metres deep, allows for flexibility in adapting architecture to granite fracturing. General repository architecture can then be designed on one or more levels.

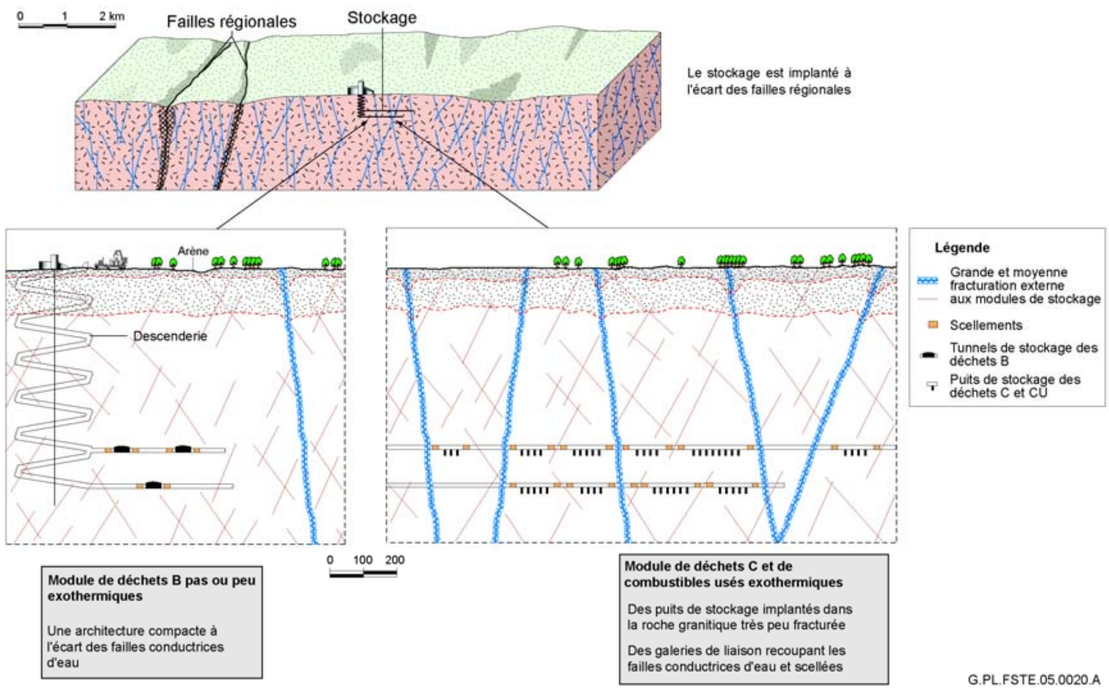


Figure 3.3.1 Architecture adapted to granite fracturing

From a functional point of view, such architectural arrangements enable the repository to fulfil a primary objective of preventing advective water flow within the underground installations. As regards disposal cells, this helps to facilitate a diffusion transfer system. Water flow rates are limited in the module drifts. In addition, the repository is located away from regional faults, safe from major water circulation.

Such architectural arrangements also facilitate other repository functions by limiting radionuclides release by disposal cells and their migration towards the environment.

- **A disposal process enabling "ongoing" surveying and characterisation of granite blocks where modules are constructed**

Adapting repository architecture to fracturing means knowing, accurately enough and without too much uncertainty, the granite host rock characteristics.

For this purpose, the surveying strategy includes several stages:

- Surveying and characterising operations from the surface or from underground structures (geological medium qualification underground facility), in order to define the granite structure where the repository is to be built. An iterative approach between safety analysis and the various phases of granite site survey, on the surface and then underground, defines criteria for exclusion of faults (or

[7] Given the waste volumes and their variations in thermicity, spent fuel disposal requires a larger footprint than C waste.

fractures) which may or may not be intersected by i) repository connecting drifts, ii) access drifts to modules and iii) disposal cells,

- On this basis, the process includes *in situ* characterisation of host-granite blocks for disposal modules before package emplacement. This stage of granite "ongoing" characterisation during the staged repository construction finalises module architecture and distribution of disposal cells in the granite according to fracturing.

Such a strategy aims at adapting repository architecture as best as possible to granite fracturing and to ensure that proposed design concepts fulfil their functions effectively as regards control of water circulation in the repository.

3.3.2 Design of engineered components, complementary and redundant with the granite medium for long-term safety

At the scale of both the repository as a whole, and the disposal cells, several arrangements are possible to ensure complementarity and redundancy between the granite medium and repository engineered components with respect to long-term safety. They particularly concern repository architecture and choice of materials for engineered components (disposal packages, engineered barriers, backfills and seals).

- **Multiple sealing of underground installations**

Connecting drifts and access drifts to modules and disposal cells are likely to intersect water-conducting fractures. In order to limit water circulation within the repository, seals are installed at various levels of the underground installations.

In the case of disposal cells, water may come from drifts serving them. Drifts are likely to be intersected by a more water-conducting fracturing than the cell rock wall one. The 'dead-end' architecture of cells, their construction in granite rock of very low permeability, and very low permeability 'plugs' at cell head limit water circulation and aim at establishing a transfer system in the cells governed by diffusion phenomena.

At the repository module scale, water circulation is limited by:

- Very low permeability seals installed in drifts to cut off modules from water coming from any possible intersecting faults,

- Backfills of sufficiently low permeability in module drifts.

Disposal cell seals and plugs are made of swelling clay (bentonite), which are very low permeability over long periods of time. Backfills may also incorporate clay materials to ensure sufficiently low permeability.

At the repository scale, connecting drifts between modules as well as structures between the surface and underground are backfilled. Seals are installed in access structures where they intersect water-conducting faults.

- **A physical and chemical environment suitable for waste packages**

Disposal cell design seeks to provide a suitable physico-chemical environment for waste and packages in order to control changes in state over time and help limit release of radionuclides. Such an environment is ensured i) by the materials used for waste over-packs, which are selected according to type, volume, radiological inventory and chemical nature of the waste, and ii) as well as by engineered barriers.

For B waste containing metal elements (B1, B3, B4 and B5 reference packages), the aim is to limit corrosion by providing a favourable chemical environment (reducing potential, pH 10 to 12.5), in particular by using concrete for waste over-packs. For bituminised B waste, the aim is to maintain, on the long-term, bitumen confinement properties (B2 reference package) by controlling chemical conditions and temperature (between 20 and 30°C).

For C waste and spent fuel, emplacing clay buffers between package and granite rock attenuates chemical interaction between packages and granite groundwater.

- **Leak-tight or very low permeability disposal packages over a sufficiently long period of time**

In order to ensure complementarity with the geological barrier, primary packages are inserted in additional containers, to constitute disposal packages. A study has been carried out in order to ensure leak-tightness or very low permeability over sufficiently long periods of time which depends on kinds of waste and their radiological inventories.

A concrete disposal package has been chosen for B waste. For some types (B1 and B5 packages, which have major radioactive content and do not release gas), disposal packages have long-term confinement properties (around ten thousand years). This performance is achieved by using a specially adapted concrete formulation (with very low permeability and porosity) and a specific design (closing method). This type of container limits the amount of water reaching primary packages as well as radionuclides release for this period of time.

For C waste packages, the aim is to prevent water from coming into contact with the glass for several thousand years. This period concerns the thermal phase (i.e. the period when the temperature at the heart of the glass is over 50°C) during which glass alteration phenomena are accelerated. The proposed design is based on a very thick steel container.

For spent fuel, a copper container is proposed, with long term leak-tightness property (up to several hundred thousand years). In contrast to C waste, radionuclides are not trapped in a confinement matrix (a fraction of the radionuclides is released upon contact with water, and the remainder is released gradually as the uranium oxide matrix dissolves). This option is based on the ‘KBS-3’ copper container a concept adopted in Sweden (SKB) and Finland (Posiva). It was adopted by Andra at this generic design phase. Site data could justify revision of this option if architecture adaptation to granite massif fracturing and engineered structures (backfills and seals) provide a sufficiently long transfer time in the geological medium to ensure radioactive decay of radionuclides.

3.3.3 Limiting granite disturbances by the repository

While the repository design aims to take the most of the favourable properties of granite, it should be ensured that repository construction and its long-term evolution do not aversely affect the properties of the granite medium. The various arrangements studied involve structure dimensioning, choice of materials for engineered components and the disposal process.

- **Design limiting mechanical and thermal disturbances**

Granite is a mechanically resistant rock. The structures (drifts and cells) are dimensioned to ensure mechanical stability in the long term.

Heat released by C waste and, if need arises, also by spent fuel means a temperature rise in the disposal cells and surrounding granite. In order to control the thermal phenomena incurred, the aim is to keep the temperature in the cells lower than 100°C (and therefore in the rock). In practical terms, a maximum temperature of 90°C has been adopted for the hottest point in the swelling clay buffers for C waste cells and at the surface of spent fuel copper containers.

The essential parameters for repository architecture dimensioning in order to limit the temperature are on one hand the number of disposal packages per cell and on the other hand the spacing in between disposal cells. The C waste and spent fuel repository zones footprint is mainly subordinate to these thermal considerations, which depend on the thermal power released by the packages once emplaced in the repository.

- **Disposal process limiting hydrogeological and hydrogeochemical disturbance of the granite hosting the underground installations**

Excavation of underground installations drains off the granite groundwater and disturbs initial hydrogeology. As granite is only slightly permeable, this disturbance mainly affects the most water-conducting faults and fractures. In order to limit water drainage from the granite and by extension the quantities of pumped water while excavating the underground installations, it can be envisaged to call for to injection techniques for the most water-conducting faults and fractures intersected by the structures.

After a transient phase of disturbance related to excavation of underground installations, equilibrium between water drainage and re-supply is established within the granite massif.

Managing appropriately, according to granite hydrogeological context, excavation of repository zones, their operation and then their closure constitutes a mean of limiting hydrogeological and hydrogeochemical disturbance of the granite.

3.3.4 Adaptation of design arrangements to long-term safety functions

The various options proposed contribute to one and/or another of the main functions of a repository, as summarised in Table 3.3.1:

- the function "preventing water circulation in the repository" is mainly fulfilled by architectural and sealing arrangements. Repository architecture is adapted to granite fracturing,
- the function "limiting radionuclides release and immobilising them within the repository" is mainly fulfilled by systems implemented near the packages in order to permanently ensure favourable environmental conditions to the protection of waste and immobilisation of radionuclides released,
- the function "delaying and attenuating radionuclides migration" makes the most of all technical arrangements adopted within the design options: structure dimensioning, choice of structure and packages materials.

Table 3.3.1 Technical arrangements and long-term safety functions for a repository in granite medium. Contribution of each technical arrangement to each function: XXX: essential; XX: significant; X: secondary

Design principles	Technical provisions	Safety functions: Preventing water circulating in the repository	Safety functions: Limiting radionuclides release and immobilising them in the repository	Safety functions: Delaying and attenuating radionuclide migration
Making the most of favourable properties of granite	**Architectural** Constructing cells in very low permeability granite rock	XXX	X	XX
	Constructing modules in the "blocks" apart from water-conducting faults	XXX	X	XX
	Constructing the repository away from regional faults	XXX	X	XX
	Disposal process Ongoing characterisation of granite "blocks"	XXX	X	XX
Designing engineered components complementary to and redundant with granite medium	**Architectural** Multiple sealing of structures (cells, modules, drifts, surface-bottom connecting structures)	XXX	XX	XXX
	Materials A physico-chemical environment beneficial to packages and waste: suitable engineered barriers and over-packs adapted waste types		XXX	X
	Disposal packages (containers, over-packs, etc.) leak-tight or only slightly permeable for sufficiently long periods of time depending on waste types		XXX	XX
Limiting granite disturbances induced by the repository	**Design basis** Structure dimensioning ensuring long-term mechanical stability	XXX		XXX
	Structure thermal dimensioning for control of phenomena caused by structure temperature rise		X	
	Materials Disposal packages and engineered barriers whose alteration does not significantly alter granite retention properties			XXX
	Disposal process Disposal process managed in order to limit disturbances of granite hydrogeological and hydro-geochemical characteristics	XXX	XX	XXX

3.3.5 Integrating reversibility

In addition to operational safety and, in the long term, repository design must fulfil reversibility requirements. As mentioned previously, reversibility is related to a cautious and staged management of a possible repository which, given the timeframes considered, leaves the options open for future generations.

Beyond the ability to withdraw the emplaced packages (retrievability), reversibility can be defined as the possibility of gradual and flexible management of the repository which leaves future generations free to decide. With this aim in mind, the disposal process can be broken down into a succession of stages to be undergone which provides from the construction of the initial modules up to the closure of a module or of a repository zone, the possibility of waiting and observation time, before deciding to go on to the next stage or reverse the process. Completing a step is thus a thought-out choice, exercised in full knowledge of the scientific, technical and economic parameters.

Reversibility requirement involves, over the course of time, human presence, monitoring and maintenance which do not put at stake whatsoever long-term safety, the primary aim of the repository. On the contrary, reversibility through a cautious and gradual management of the disposal process is at the core of the long-term safety foundations in managing uncertainties and safeguarding the rights of future generations. In any event, allowing for reversibility must not endorse any compromise on safety objectives. Thus no provision, that could significantly disturb a safety function, is added for the sake of reversibility.

The design approach adopted by Andra aimed at putting forward generic design options that meet the requirement for reversibility.

This covers three aspects:

- architectural arrangements beneficial to a gradual repository management,
- technical measures for going backward to the various disposal stages,
- means of observing repository status and its evolution at any time in the process.

- **Repository architecture incorporating and fostering reversibility**

The repository architecture options put forward by Andra build in the reversibility requirement and make its implementation easier.

Simple, robust repository concepts

Concepts proposed by Andra at this stage of the study are, by principle, simple and robust. Simplicity is based on the concern for technical feasibility and performance control. Making the most of mechanical resistance of granite rock especially allows to limiting support systems in structure design.

Simplicity of options proposed by Andra facilitates the description of their evolution over time as well as their modelling. Robustness is based on resistance of the concepts in terms of safety and necessary scientific knowledge.

Durable materials and systems to facilitate potential withdrawal of the packages

The possibility of future generations retrieving the packages is helped by the choice of durable materials for the packages and self-stable geometries for the underground excavations.

These choices contribute to maintaining the handling spaces between the disposal packages and the disposal cells, thus allowing the use of identical handling equipment to that used for emplacement, in case of retrieval.

Modular design of underground installations for flexible management and changes in design

The proposed architecture options are modular: they provide a flexible and gradual management of the development of a repository promote the staged management of the disposal process and allow differentiated management methods according to the various types of waste.

Each package category (B, C and CU) is to be hosted in a dedicated repository zone, constructed, operated and closed independently. Each repository zone is designed to be built and operated gradually as successive cell sub-assemblies. Closure designed in a gradual manner is organised into several stages: closure of cell sub-assemblies, which can be carried out at the same time as the creation of new sub-assemblies, closure of access to this sub-assembly, and then of the repository installations specific to this waste category and lastly of all installations.

As the repository is being developed in stages, new structures can be designed taking advantage of the lessons learnt and knowledge acquired during operation and observation of previous structures, as well as of technical progress carried out otherwise. It is as well possible to incorporate data from social, technical and scientific backgrounds.

- **Technical feasibility of reversing the process**

Andra has studied the technical feasibility of reversing the process for the various repository stages: technological resources, operating conditions and necessary precautions.

The repository is therefore designed to allow packages to be retrieved in the first stage by simply reversing the process of their emplacement in the cell (as easily as in a storage facility). For later stages, Andra has incorporated arrangements in order to access again to the installations which were closed and install the equipment required to retrieve the packages, would such a decision occur.

- **An observation programme supporting reversible repository management**

Keeping options open during the repository process implies knowledge of its evolution and situation at all times: therefore it requires constant observation and implementation of the necessary measurement resources and systems. Andra has studied the possibilities of integrating measuring sensors in the structures without disturbing repository operating and safety.

Beyond the monitoring related to operational safety, this observation programme reinforces understanding of the phenomenological evolution of the repository and provides important data for decision-making. It also draws on feedback from the operation of the first structures in order to improve repository design and management.

4 General architecture of the repository in a granite medium

A reversible repository comprises both underground repository infrastructures and installations constructed on the surface.

This chapter starts by presenting the surface installations. It provides a succinct description of their main characteristics and the general principles of organisation. It describes the still preliminary lay-out of the nuclear zone, at this very early design stage, based on the hypothesis that primary package reception and disposal package conditioning operations will be carried out on the repository site. It highlights the analogies between the planned installations and other existing nuclear installations.

A presentation of underground installations follows. The modularity of the architecture adapted to the fracturing of each repository zone is shown to be based on current and proven technical underground construction work solutions. While aiming at adapting the repository design to the various configurations of French granites, the principle of functional redundancy between the disposal packages and the engineered barrier has been adopted in the absence of a designated site. This arrangement limits the required criteria of the geological medium to those common to granites: slightly permeable rock that has retention capability and mechanical strength.

Lastly this chapter presents how the disposal process is implemented through the construction and operation aspects of the structures. It emphasises that these processes are based on proven nuclear industry and underground construction work practices (mines and underground infrastructures). It demonstrates that repository architecture allows these activities to be carried out simultaneously while process phasing offers flexible management according to the reversibility rationale.

4.1 Surface installations

The essential functions of the surface installations are: receiving the primary waste packages, conditioning the disposal packages and supporting the underground installation construction operations.

These installations are not specific to a repository: hence there are no new technical feasibility issues. They are very similar to existing industrial installations, which explains why at this stage, the description of them is limited to a preliminary outline.

Potential surface installation lay-outs are only given in this section by way of illustration. Likewise it should be noted that these lay-outs do not allow for site-related constraints such as topography. However the order of magnitude of the surface installation dimensions can be assessed from them, especially the footprint, and an overall estimate of their sensitivity to possible waste management scenarios.

4.1.1 General organisation of surface installations

The surface installations are divided into four main zones:

- the nuclear zone, with a surface area of around 25 ha, where the primary waste packages are received and the disposal packages are prepared,
- the industrial zone, with a surface area of around 35 ha, grouping the workshops and facilities required in support of the work underground,
- the administrative zone, with a surface area of around 20 h, consisting of offices, car parks and personnel related buildings (canteens, etc),
- the broken rock storage dump, with a surface area of between 120 and 250 ha depending on the scenarios.

Note that the surface area allocated to the industrial and administrative zones (55 ha) is around the same size as the area occupied by the surface installations for a major underground works site, such as the Channel Tunnel.

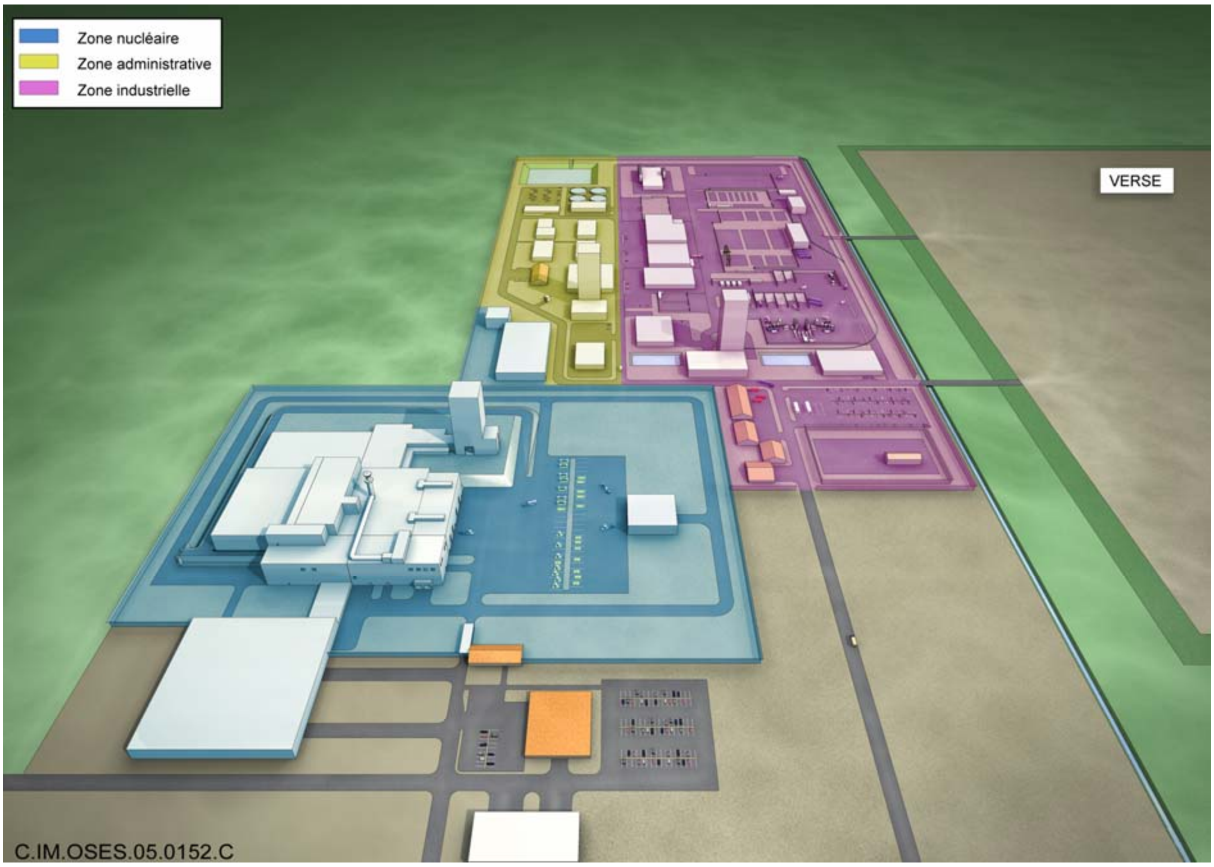


Figure 4.1.1 Example of surface installation layout

4.1.2 Nuclear zone

4.1.2.1 Overall rationale for the nuclear zone

The "nuclear" activities zone within the surface installations is a specific industrial zone, with controlled access. It includes:

- a “traditional” controlled zone (without specific risk linked to the nuclear activity), comprising the support installations (fire service building, equipment warehouses, etc.). The manufacture and procurement of the components needed to produce disposal packages needs at least a warehouse tin order to store these components,
- a "sensitive" zone, corresponding to nuclear facilities complying to the specific regulations concerning “Nuclear Basic Facility”. This zone is fully protected by a physical fence and a second level of access control. It includes the unloading and temporary storage facilities for the transport transfer casks, the buildings where waste packages are received and then prepared for diposal, the package transfer shaft to the underground installations and the air exhaust shaft.

The functional features of the nuclear "sensitive" zone in terms of waste types to be handled depend on the study scenario considered. It includes B and C waste-specific buildings. In the study scenarios including the hypothesis of spent fuel disposal, a dedicated building would be added.

4.1.2.2 Building for receiving B and C waste and for preparing disposal packages

The building for receiving the B and C waste and for preparing disposal packages has three schematic sections:

- a zone for unloading the primary waste transport transfer casks,
- a zone for manufacturing and storing the disposal packages,
- a zone for transferring the disposal packages to the transfer shaft.

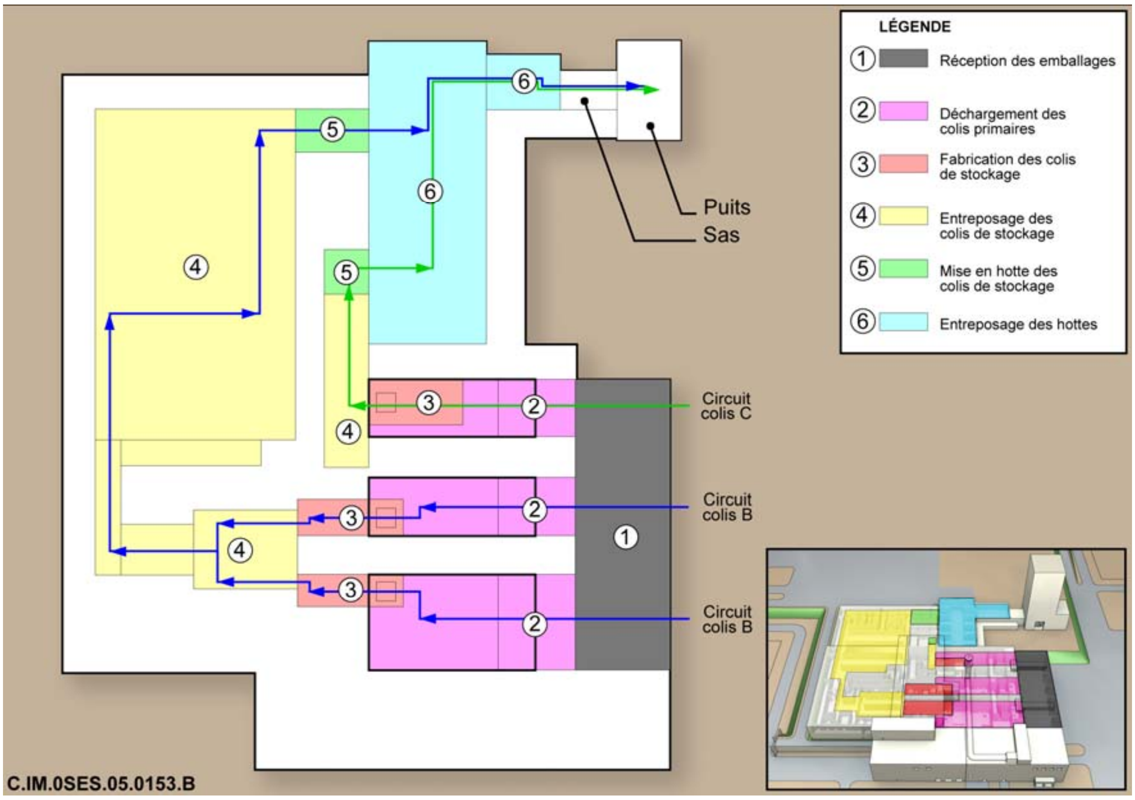


Figure 4.1.2 Schematic organisation of the B and C waste building

The transport transfer cask unloading zone is fitted with a travelling crane and shielded cells. The primary packages are extracted from the transfer casks and placed in buffer storage before being retrieved and placed in disposal packages.

The zone for manufacturing and storing the disposal packages consists of several manufacturing lines. Each line is made up of a succession of shielded cells. Two manufacturing lines are considered for B waste disposal packages and one line for the C waste disposal packages.

The disposal packages are routed from the end of the line to the cask loading station. The casks are then conveyed into the shaft building for transfer to the underground installations.

All types of activities planned for this building are already implemented in similar industrial nuclear facilities.

4.1.2.3 Surface building for spent fuel reception and containerisation

Should spent fuel disposal be considered (UOX and MOX), a specific additional facility will be built to receive and condition it before transfer to the underground installations. This facility could be hosted in extension linked to the previous building.

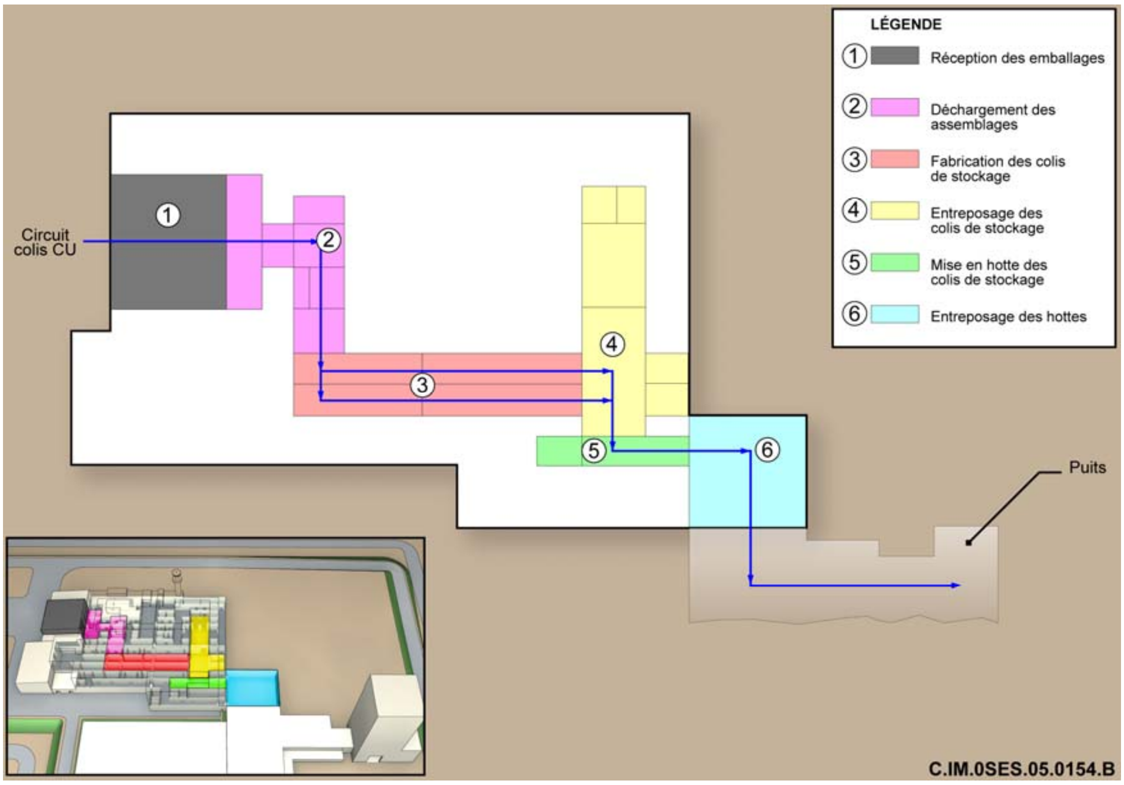


Figure 4.1.3 Schematic organisation of a spent fuel building

4.1.3 Industrial and administrative zone

The industrial zone occupies an area of 30-40 ha. It is organised around the construction shaft.

The main component buildings are:

- the construction shaft operations buildings;
- construction material (engineered barrier, concrete, backfill, etc.) preparation shops;
- an explosive warehouse;
- equipment maintenance and repair shops (electrical, hydraulic, mechanical, electromechanical, boiler-making, cabling, etc.);
- service buildings (warehouses, stores, core library, concrete laboratory, etc.).

This industrial zone is gradually built up as dictated by the underground works requirements. Thus for instance, until the decision to close the repository is made, there is no need to build installations to increase backfill material production capacities.

The surface installations for site personnel management and administration are grouped around the personnel transfer shaft. These are basically office buildings, a central emergency building and a "living" space for the personnel working on the site (cloakrooms, showers, canteens, etc.)..

4.1.4 Broken rock storage dump

The excavation site broken muck is stored in an area, designated by the mining term "dump", constructed near the construction zone. While trying to recycle a maximum amount of the broken muck as repository backfill, a third to a half of the extracted volume could remain on the surface if no other recycling solution is found.

- **Summary assessment of the dump footprint**

The broken rock storage dump represents a volume of several million cubic metres; this volume (and even more) is frequently found in open-cast mining. As backfilling date is not fixed in advance, the surface area for a dump capable of accommodating all the broken rock has been estimated.

As there is no study site, only the general dump management principles can be given, that will subsequently be adapted to the site topography. On the basis of a 10 meters high dump, the area required could vary from roughly one hundred hectares for scenario S1a to about 250 ha for scenarios S1b, S1c and S2.

- **General dump design and operation principles**

The dump project should be studied according to the site for its best integration with the landscape and the hydrographic network.

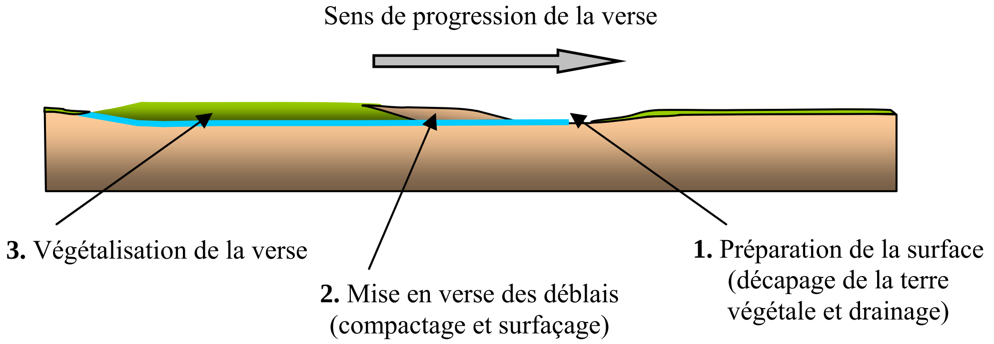


Figure 4.1.4 Phased dump operating principle

On a rather flat site (without any specific topography), a dump is generally operated in sections. Scraping and surfacing operations, constructing the drainage network, placing the broken rock and capping follow on from each other and progress with the development of the dump.

Apart from the geometric aspects, the dump is mainly designed to ensure the stability of the broken rock and prevent rainwater from percolating into the emplaced materials.

Preparatory earthworks, scraping off the soil, shaping dump sections, building ditches and slopes, and storm and settlement basins are necessary.

Lastly, the topsoil from the scraping of the various zones where the surface installations and dump are built is assembled and stored near the dump. It is used as and when necessary to cover the dump before vegetalisation.

- **Recovering broken muck from the dump for backfilling operations**

Broken rocks can also be retrieved for backfilling operations by section, similar to the way in which dumping was first organised.

The backfill materials recovered from a dump must be conditioned before their re-use. Conditioning may involve crushing and screening the retrieved materials and possibly mixing with added swelling clay.

4.2 Underground installations

4.2.1 Design principles

With respect to long-term safety, underground installation design aims at firstly making the most, through architectural arrangements, of the favourable properties of the granite medium: the mechanical strength of the rock, its radionuclide retention capacity and its low permeability in "blocks" characterised by rock with little or no fracturing. Making the most of these favourable properties entails designing repository architecture in modules that can be adapted to the fracturing of a specific site, since granite is intersected by potentially water-conducting fractures.

This modularity underlies the possibility of connecting drifts crossing these water-conducting fractures. With regards to long-term safety this possibility relies on inserting seals in addition to the backfill to fulfil functions that are redundant with those of the granite.

Furthermore the underground installations are dimensioned to limit performance-altering disturbances to the properties of the granite medium and the seals.

Lastly, underground installation design incorporates the reversibility requirement as indicated in chapter 3.

- **Modular architecture, adapted to granite structure and its fracturing**

The principle of modular architecture has been adopted in order to adapt the location of the repository structures to the fracturing. Repository structures (drifts and cells) installed in a low-permeability granite "block" at a distance from water-conducting faults (§ 3.3.1) constitute disposal modules. Architecture geometry will therefore depend on "block" distribution in a granite massif.

Typological analysis of French granites shows that the largest granite "blocks" generally tend to be grouped by sectors. Thus pre-defining repository architecture in granite medium is impossible, but must result from both surface exploratory work on a site and underground laboratory studies.

Furthermore the layout and geometry details of the repository structures are based as well on completion of "on-going" exploratory work during repository construction. This exploratory work consists of geological surveys and geophysical and hydro-geological measurements performed in drifts and boreholes.

Apart from the issue of siting repository structures in fracture-free granite blocks, a working failure of part of the repository must have limited impact.

Accordingly the architecture is divided into repository zones and modules. Thus distinct repository zones accommodate the repository modules of a single broad category of waste: B waste and vitrified C waste. A specific repository zone must be constructed if UOX and MOX spent fuel is to be hosted in the repository.

The repository modular character and compartmentalisation (waste of the same type is grouped together) contribute to prevent chemical and thermal interactions in particular, between waste packages of different kinds. For instance, bituminised B wastes are disposed of in specifically dedicated modules away from modules that contain exothermic wastes that are likely to significantly modify the temperature of the medium.

- **The possibility of multi-level architecture**

A granite massif generally represents a vast volume of rock available for a repository 300 to 1000 metres deep. Incorporating the "depth" dimension offers additional flexibility in adapting the architecture, not only to granite fracturing but also to the temperature of the medium accordint to waste thermicity. Thus the studies have examined the advantages and limitations of multi-level repository architecture to deal with the variability of the characteristics of the French granites identified in the typological analysis, in particular their thermal and structuring properties (§ 3.2.2).

Firstly, suitable adjustments to massif fracturing through two (or even three) repository levels provide a reduction of the underground installations footprint. Furthermore, in the case of French granites this arrangement enables the knowledge acquired at a given depth to be extrapolated to a volume of rock that can accommodate two repository levels.

There are essentially two limitations to multi-level architecture. From a design viewpoint, A two-level repository requires a thick enough massif in order to avoid any The rock that has to separate two superposed levels, must be thick enough to avoid the risk of harmful interferences.

From an operational viewpoint, the issue of sharing access structures must also be analysed. However these limitations do not appear to rule out such a two-level concept as the vast volume of available rock enables the “buffer” space between the two levels to be adjusted, and through construction of surface-underground access structures that serve only one repository level.

Two-level architecture has likewise been envisaged in Sweden, for the "SR 97" safety assessment dossier for a theoretical "Aberg" site in a granite medium, modelled from Äspö laboratory data (§6.2).

- **Surface-underground access structures (shafts or ramps) installed upstream of the repository modules from a hydraulic viewpoint**

In the interests of long-term safety, the access structures between the surface and the underground repository levels are located upstream of repository modules to limit potential water circulation and radionuclides transfer from the repository to the connecting structures. Likewise, in order to limit hydraulic gradients in between structures, access structures are grouped according to site configuration.

Shafts or a combination of shafts and ramps may be envisaged to provide the connection between the surface and the underground repository levels. Dimensioning is adapted to the various transfer functions assigned to them and the corresponding throughputs: excavated rock removal, ventilation, transfer of repository construction materials, disposal package and personnel transfer.

- **Multiple seals adapted to repository modularity**

The surface-to-underground access workings and repository drifts are likely to intercept water-conducting fractures. Seals and backfill are designed to protect the repository modules from any water flow resulting from intersected fractures.

From this viewpoint, backfill and seal functions are complementary and redundant in long-term safety terms. The exact specifications depend on the characteristics of a granite site and waste types.

In Fenno-Scandinavia, the KBS-3 spent fuel disposal concept (copper container) does not include any redundancy in terms of seal or backfill. Only low permeability swelling clay will backfill drifts. However both provisions are envisaged in the generic context of Andra’s studies.

Thus very low permeability long-lasting seals are to be constructed systematically between the water-conducting faults and the repository modules installed in low permeability granite "blocks".

Furthermore, depending of the granite site configuration, limiting direct water arrival from the most permeable surface part of the granite, can be achieve through seals installation between the surface-underground access structures and the connecting drifts.

Backfilling repository drifts are also planned to limit water circulation and contribute to delay any radionuclides migration towards the environment.

By principle, at this generic stage of the studies, the backfill are assumed to be more effective the closer the drifts are to the waste. In repository modules, drifts are backfilled with low-permeability material. Outside repository modules, the permeability level of the backfill emplaced in the drift and surface-underground access structures is set according with both the conductivity of the fractures likely to be intersected and the hydraulic properties of the excavation damage zone.

- **General architecture and dimensioning limiting repository disturbances**

Modular design of the repository architecture, if it is to be effective in terms of long-term safety, must ensure that the repository itself does not interfere ultimately the proper working of the granite medium, seals and repository modules.

Thus the hydro-geological context of a granite site and its fracturing govern the horizontal distances between modules and the vertical distances between levels. Possible interactions between modules are limited by “buffer” distances of about one hundred metres between modules and water-conducting faults.

By principle, modules with B waste, since systematically including (high pH level) concrete for waste conditioning, are not located right above or below modules of other repository zones. This arrangement avoids risk of interactions between, on one hand, the alkaline concrete water and, on the other hand, C waste and spent fuel repository modules that could be sensitive to it (§ 3.3.3).

For similar reasons seal and backfill designs also use materials that by composition tend to limit any potential disturbances to fractures properties, as fractures are the preferred place for radionuclides transfer and retention.

In particular, the concrete seal abutments may be designed with low pH concrete formulation in order to limit repository water alkalinity. The same goes for the cementitious material injected into the water-conducting faults intersected by the connecting structures during excavation work, to limit water arrival and hydraulic disturbances in granite.

From a thermal viewpoint, repository dimensioning for exothermic packages (vitrified C waste and potentially spent fuel) aims at limiting temperature in repository modules and granite. The maximum temperature criterion of 100°C in the modules results in a maximum temperature of 50-60° in the granite, for an original temperature of 25°C at a depth of 500 metres. In the event of multi-level architecture, the distance between two levels is set at about one hundred metres to prevent this temperature level being exceeded.

- **A staged management process conducive to disposal reversibility**

The repository is managed with a reversibility rationale meaning a step-wise approach, with no pre-set time limits, for the construction, operation and closure of the underground installations. This staged approach enables the repository operator to make the most of the first stages feedback for possible adjustment and modification of repository design. Furthermore this staged approach combined with repository modularity adapted to the fracturing is conducive to reversibility. They provide flexibility in terms of repository management, similarly to storage management, meaning that packages can be retrieved if needed or desired (§ 3.3.5).

4.2.2 Description of repository architectures

The architecture of a repository is designed to be adapted to the highest number of possible configurations in the French context. Architecture designed on two levels separated by one hundred metres or so is adopted as the reference [13]. From this stage of the studies, this architecture presents many adaptation possibilities that offer multi-level architectural solutions and identify the requirements likely to be dealt with when compared with single-level installation solutions.

The remaining part of this chapter describes the architectural components of a repository in terms of the repository zone exploration, construction, operation and closure process requirements. The latter are described in detail in chapters 5, 6 and 7 of the present volume, which cover B waste, vitrified C waste and spent fuel repository structures respectively.

4.2.2.1 General organisation

Repository architecture is an organisational solution of the following components:

- the access structures to the underground installations (or "surface-underground connecting" structures) are the shafts or a combination of shafts and ramps. They are used for transferring all the incoming and outgoing throughputs to and from the underground installations (materials, ventilation air, disposal packages, etc.);
- underground infrastructures constructed in the shaft zone, at each repository level, are devoted to housing the installations and support equipment for the exploration, construction, operation and closure activities at the corresponding repository level. They aim at distributing and organising the various throughputs;
- connecting drifts for the survey, construction, operation and closure of repository zones installed on a single level;
- exploratory drifts to characterise module host blocks in detail;
- repository zones, constructed on two levels are dedicated to each waste category. They accommodate the modules, each module including several disposal cells.

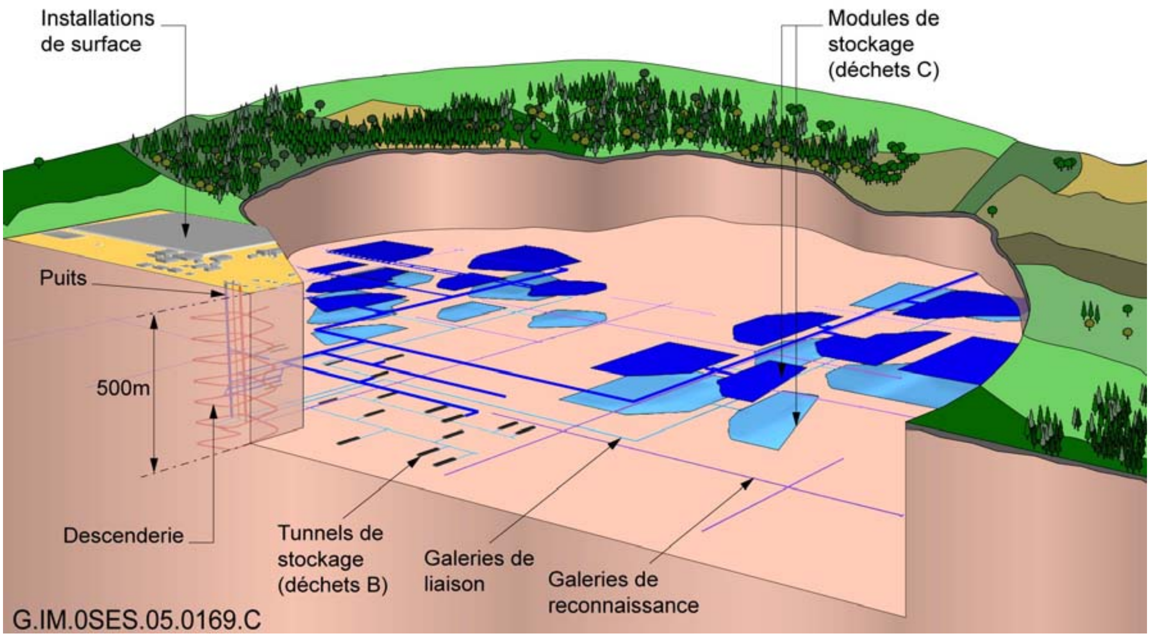


Figure 4.2.1 View of two-level repository architecture

4.2.2.2 The access shafts and ramp: description and construction

Two types of structure are considered for access to the underground installations of the repository: shafts and ramps.

- shafts are vertical structures, generally equipped with machinery to lower and raise loads in cages suspended by steel cables;
- ramps are inclined drifts, with a 10-15% gradient, that are used by vehicles with tyres[8].

These two types of structure are commonly used in mining and underground engineering (sediment or metals mines, underground subways, deep tunnels, underground hydrocarbon storage facilities, etc.). Shafts have been in use since the 19th century and ramps since the second half of the 20th century.

Studies carried out in Sweden and Finland for underground repositories in a granite medium conlude with hybrid solutions combining several shafts and one ramp. In Sweden, the ramp is designed for materials (broken muck, backfill, etc.) and also for disposal package transfers. However the elements that led to this choice cannot be directly transposed to the French context for the following reasons:

- firstly, a higher daily disposal package throughput is considered in this study than in the Swedish and Finnish projects. This throughput makes a strong case for using shafts for package transfer;
- furthermore, it is considered that physical separation of nuclear operation and construction throughputs means their easier co-activity.

In contrast, the "all shafts" solution has been adopted in the existing WIPP facility (United States) and the Gorleben repository project (Germany) where water-bearing overlying formations would have hindered ramp construction.

To sum up, it appears that technically both the "all shafts" and hybrid "shafts-ramp" solutions are suitable for an outcropping granite context.

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[8] There are also so called “inclined” (sub-vertical) shafts dipping up to 20%. This type of structure is only used in specific conditions (dipping deposits, mountainous topography, etc.).

In an "all shafts" configuration, an initial shaft (namely “construction shaft”) is dedicated to the repository construction and closure throughputs. It houses the broken muck extraction and backfill (loose materials) lowering equipment, the heavy machinery lowering and lifting equipment, the drainage water discharge systems. The construction shaft may also serve as an emergency route if the personnel transfer shaft becomes unusable. A second shaft (“personnel transfer shaft”) is exclusively dedicated to personnel transfers. The construction and personnel transfer shafts supply fresh air to all the underground installations.

A third shaft (“package transfer shaft”) is allocated exclusively to lowering the packages in their protective casks and returning the empty casks back to surface. The fourth and last shaft (“air exhaust shaft”) is dedicated to air removal.

- **Shaft dimensioning and description**

Both mechanical and technological parameters are input for shaft designs. The access trsuctures are designed for a minimum service life of one hundred years (in the mining industry, centuries-long service lives for this type of structure are current).

Circular section is taken as the reference design for all the shafts.

For safety reasons the shafts are concrete-lined from top to bottom. In all cases, both shaft linings and equipment located inside the shafts are accessible throughout the repository operational phase. The ability to carry out maintenance and repair work as needed enhances the service life durability of these access structures.

The movement of loads inside the shafts is based on using "Koepe" friction systems, each made up of a motorised pulley either side of which a transfer cage and its counterweight are suspended. This system has been adopted as the reference solution since it has been proven by the mining industry all over the world for a century.

Construction shaft

The construction shaft has a useful diameter of 10-12 metres. It is as deep as the deepest repository level, with an additional length of thirty metres or so at the bottom.

The shaft section may be compartmentalised in the interests of managing the main construction throughputs independently, with:

- a compartment equipped with two skip hoists (mobile hoppers for transporting loose materials) raises the broken muck to the surface and lowers the backfill;
- a compartment equipped with a large 40-tons capacity cage for transferring construction and heavy machinery.

An emergency cage is fitted to each compartment for rescuing any personnel inside the shaft.

Personnel transfer shaft

The personnel transfer shaft has a useful diameter of 6-7 metres and is as deep as the construction shaft. The equipment dimensions and ventilation requirements essentially govern its dimensioning. It is equipped with a main cage, that can carry a few tens of people and an emergency cage.

Package transfer shaft

The package transfer shaft is reserved exclusively for lowering disposal packages in the shielding casks from the surface to the underground installations and returning the empty casks to the surface. There are two reasons for this exclusive use: i) the kind of the packages being transferred, ii) the daily package throughput is close to maximum shaft capacity.

This shaft offers a useful diameter of 11-12 metres. The diameter results directly from the size of the package transfer equipment and the auxiliary equipment (personnel emergency cage for the maintenance and inspection staff).

In addition the shaft has an independent ventilation system. It is connected to the surface and underground installations by airlocks (dual-gate system).

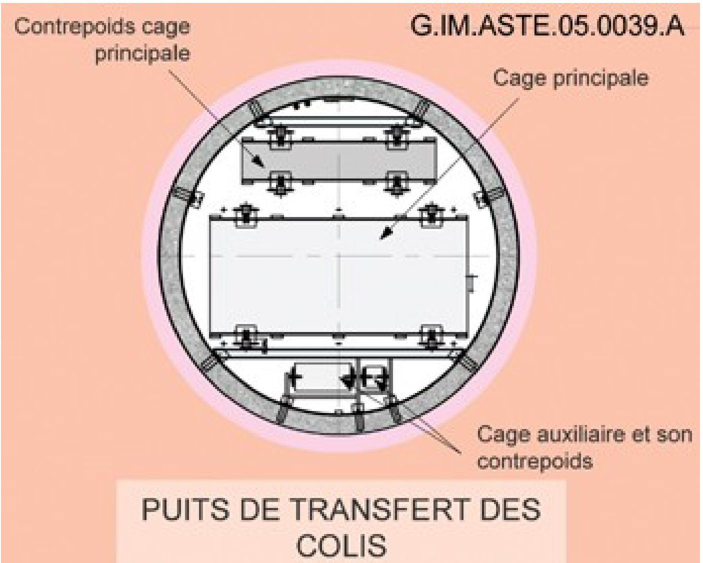

Figure 4.2.2 Diagram of the equipment layout in the package transfer shaft

The shaft equipment and its safety devices are dimensioned for a 110-tons net load transfer (which means a total suspended load of 300 tons) at reduced speed (of about 1 m/s). The suspended load of about 300 tons is spread over some ten cables, giving a load of about 30 tons per cable.

Table 4.2.1 presents shafts data with comparable mass transfer capacities.

Table 4.2.1 References of shafts transporting heavy loads

Location	Use	Depth (m)	Load suspended per cable (tons)	Number of cables	Speed (m/s)
Sweden	Ore extraction, Kiruna mine	802	32	6	17
Poland	Mining extraction, Pniowek	1160	57	4	18
Germany	Nuclear waste repository project, Gorleben	870	27	8	5
United States	WIPP nuclear waste repository	698	20	6	2.5

Air exhaust shaft

The general exhaust air of the repository, blasting fumes and if necessary fire smoke, are extracted through this shaft. Its useful diameter of about 10 metres can be adapted according to the repository process management choices (simultaneous construction of several repository levels, maintaining the installations open and ventilated).

- **Shaft construction**

The conventional method for constructing a large diameter shaft in the 6-12 metre range is to sink the shaft from the surface by drilling and blasting the rock and gradually installing the rock support and shaft liner as excavation progresses. At this stage this method which is being well-suited to the considered diameters and with a large feedback, is adopted as the reference solution.

However mechanised excavation methods, developed in the mining industry (such as raise-boring or shaft-drilling) could be considered. For example at the end of the chapter, the option of the "raise-boring" method to create a 2-4-metre diameter broken muck ejection chimney , followed by traditional sinking to enlarge the final shaft diameter, is presented.

The first two shafts are sunk simultaneously so that they can be linked at the bottom as soon as possible to create a ventilation circuit.

These shafts are constructed in the following stages:

- preparatory detailed survey of the formations to be intersected;
- platform and foreshaft construction;
- sinking then lining the shaft.

Detailed survey

In and near the shaft axis, boreholes are drilled to survey the geological column to be intersected and to appraise the amount of possible water arrival.

This survey ensures notably that the rock is conducive to the construction of underground stations at the planned depths and, when necessary, informs about necessary that injections to be carried out in order to limit any water arrival of water between these points and the surface.

The foreshaft

After possibly scraping surface-altered granite and making a reinforced concrete slab, the foreshaft is sunk to a depth of several tens of metres (crossing the decompressed and generally permeable upper granite formation). Sinking entails a number of successive drilling and blasting phases, and reinforcement with bolts and mesh (with shotcreting if necessary).

Then the shaft wall is totally lined with formed concrete, making a leak-tight lining associated with possible ground injections.

Shaft-sinking

Shaft-sinking is a cycle of operations. Its sequence is common to all drilling and blasting works:

- blast holes are bored then filled with explosives (2-4 metres depending on the shaft diameter);
- blasting, extracting the blast gases and inspecting the walls (dislodging unstable blocks if necessary: purging);
- broken muck removal and geological survey of the walls;
- installing temporary ground support (bolts, shotcrete, etc.)

The final lining is poured (formed concrete) every two to four cycles, some tens of metres above the bottom of the shaft.

The use of smooth blasting techniques using pre-cutting and/or post-cutting, minimises granite where the future seals will be sited.

Shaft-sinking requires equipment made up of two sub-assemblies: moving equipment lowered through the shaft and static equipment located on the surface.

The moving equipment is made up of a metal structure, the shaft-sinking platform. Work in the shaft is carried out from the shaft-sinking platform. This platform, suspended by cables, is lowered down into the shaft as work progresses.

Figure 4.2.3 Shaft-sinking platform (Konradsberg - Heilbronn salt mine - Germany)

The static equipment is made up of a head-frame and a set of winches for the shaft-sinking platform movements. Once shaft construction is completed, this equipment - specifically installed for shaft construction - is replaced by the permanent operating equipment.

Shaft-sinking combining the "raise-boring" and shaft-sinking techniques.

A "raise-boring" and shaft-sinking combined solution could be envisaged for sinking the third and/or fourth shaft.

A "pilot borehole" roughly 30 cm in diameter is bored from the surface to an existing drift. The pilot hole is reamed from the bottom to the surface (raise-boring method) to create a 2 to 3 m diameter “chimney” (in all cases maximum diameter less than 5 m). The reaming broken muck is recovered at the underground station and transported to surface via an existing shaft or the ramp. The chimney is then widened to the required final shaft diameter by sinking from the surface down to the drift. The broken muck is recovered at the chimney bottom and transported to surface ejected via another shaft or the ramp.

The presence of unstable or permeable zones along the planned course of the shaft governs the feasibility of this solution. This explains why the "raise-boring" technique does not generally apply to shafts requiring rock supports during the construction process.

- **Ramp dimensioning and description**

The ramp accommodates two tracks dimensioned to enable vehicle passing. A dedicated underground installation construction work ramp is a sloping drift comprising a series of straight sections dipping at about 15% and curved sections with curvature radius of over 30 metres. The ramp has an independent ventilation system.

The "inverted U" geometry of the ramp section is 5-6 m high and 6-8 m wide. It has recesses approximately every 250 metres, to park vehicles waiting to be repaired or for turning manoeuvres. Laterally, about every hundred meters refuge bays are constructed. The ramp is connected to a ventilation shaft by a small drift (with a section of about 5 m^2) at roughly one kilometre intervals. The ramp is about 4 km long.

The slope of a dedicated package transfer ramp would be limited to 10% and the length of the straight sections to about 80 metres. An escape lane is sited in the axis of the descending sections at each bend, dimensioned to absorb the potential energy of an uncontrolled vehicle. The package transfer ramp is 5-6 kilometres long.

Ramp construction

Ramp construction is also based on drilling and blasting (Figure 4.2.4).

Detailed surveying of the volume of rock required for installing the ramp has to be carried out (boreholes, geophysics, hydro-geological tests, etc.) just as it does for shafts.

This survey work serves to specify the ramp course and to refine the design of the surface trench. The whole ramp is lined with formed concrete. Once this trench is constructed, ramp sinking starts by full face drilling and blasting.

The trench and ramp floors comprise a concrete slab. The ramp walls are simply fitted with safety mesh bolted to the rock. Only zones that present mechanical instability or water ingress are injected with a cementitious grout and lined with formed concrete as required.

These injections do not aim at making waterproof the rock volume to be crossed, but at maintaining favourable conditions for continued excavation work and at limiting excavation-related disturbances on the hydro-geological medium.

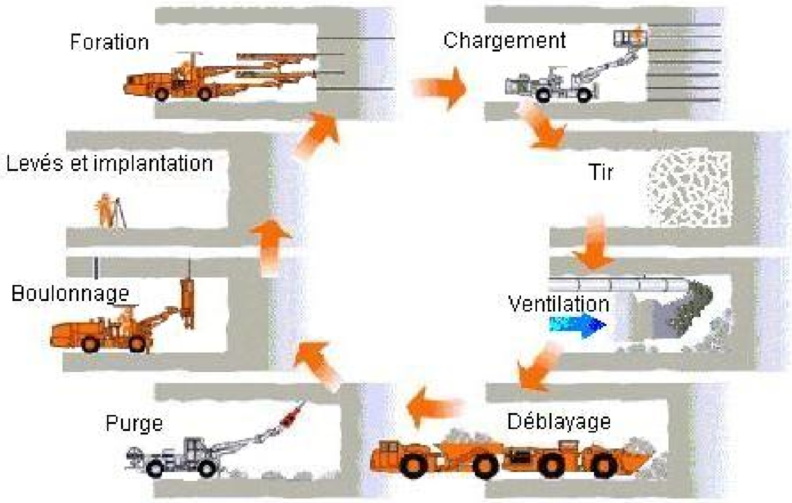

Figure 4.2.4 Schematic drilling and blasting cycle

Grout injection is carried out, through boreholes configured in a conic arrangement, ahead of the working face. Ramp excavation is performed in the volume of the nested injection cones. After each blast, the ground is inspected in order to decide on whether injection must be carried on or not.

The residual drainage water is collected behind the working face and pumped to a water treatment plant on the surface.

4.2.2.3 Shared infrastructures of the shaft zone

The shaft zone infrastructures, developed on two levels, provide the interface between the access structures (shaft and the possible ramp) and the connecting drifts that serve each repository zone.

They have four main functions:

- enabling construction throughputs to be managed (ejection of broken muck, delivery of concrete and backfill supplies, movement of equipment and miscellaneous materials) and the drainage water to be collected and discharged;
- providing a similar function for operational throughputs (supply of casks containing disposal packages and removal of empty casks);
- supporting also the underground installation operations (stores, oil and gas-oil distribution units, minor repair workshops, refectories, washrooms, emergency stations, fire refuge bays, etc.);
- managing the exhaust air (general exhaust, smoke extraction circuits).

4.2.2.4 Surveying structures

The purpose of the underground exploratory work is to define the precise location of the modules in the volumes of granite away from the faults of the massif which were identified during previous investigation phases (chapter 4.2.1). They enable the repository architecture to be adjusted to the real site conditions, as results are obtained.

Performing these works calls for the construction of exploratory drifts in order to conduct the detailed exploration and characterisation programme of the granite blocks where the repository modules will be installed.

These programmes basically entail coring long boreholes and carrying out *in situ* (hydro-geological and geophysical) tests similar to those conducted by Posiva for exploration of the Onkalo site in Finland.

The exploratory drifts, 4-6 m wide, form a network with a 400-500 m mesh.

These drifts are developed in a "comb pattern" with dead-ends on either side. The aim of this arrangement is to limit any disturbance to the hydro-geology of the medium that could be caused by a large system of interconnected drifts.

The exploration programme of the upper level in the context of dual-level repository architecture also includes downward cored boreholes. The exploration programme of the lower level, the layout and number of exploration drifts are adjusted to the validation elements remaining to be acquired.

At both repository levels, the access and connecting structures needed to construct the repository provide support to the exploration drifts construction.

4.2.2.5 Connecting drifts

An organised network of connecting drifts at each repository level links the modules (and their associated disposal cells) to the shared infrastructures of the shaft zone. This network has a tree structure from the shaft zone to give access to more or less dispersed blocks on a horizontal plan of the granite massif.

It may be a simple tree structure with a main cluster of connecting drifts, giving access to blocks on either side, or a more ramified tree structure with secondary clusters.

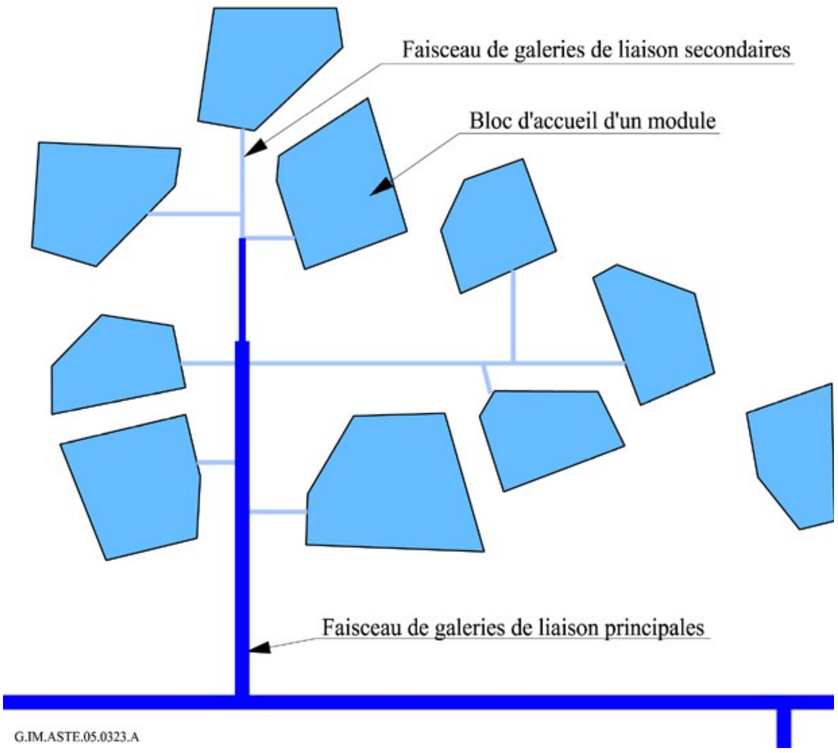

Figure 4.2.5 Tree structure adjusted to the distribution of blocks in the granite

- **General organisation**

- The *connecting drifts* linking the shaft zone infrastructures with the various repository zones are organised in clusters (or bundles) and enable exploration, construction, operation and possible closure activities to be carried out simultaneously in each repository zone. The number of drifts per cluster varies from five to three, depending on the distance from the shared infrastructures and throughput convergence in this cluster. At the current stage of the studies, one drift is dedicated to the movement of package transfer vehicles. Exhaust air also has its own dedicated drift. This drift houses the smoke extraction ducts. When far from the shaft zone infrastructure, and would ventilation requirements be minor, this drift is not anymore necessary and can be eliminated; then exhaust air extraction is performed by the ducts installed in a construction drift. The other drifts (two to three drifts, depending on the distance from the zone served) are dedicated to construction work or repository module closure. In the portions with large throughput, construction drifts are preferentially designed with one-way circulation along two tracks;
- The *secondary connecting drifts* serve a set of two to four neighbouring modules within a single repository zone. They are in turn served by the main connecting infrastructure. The cluster of secondary connecting drifts comprises two to three drifts, whose specialised function lasts only during the active performance period for this given activity (as construction, operation or closure). The drifts are re-equipped between each period and modified for the purposes of the following activity.

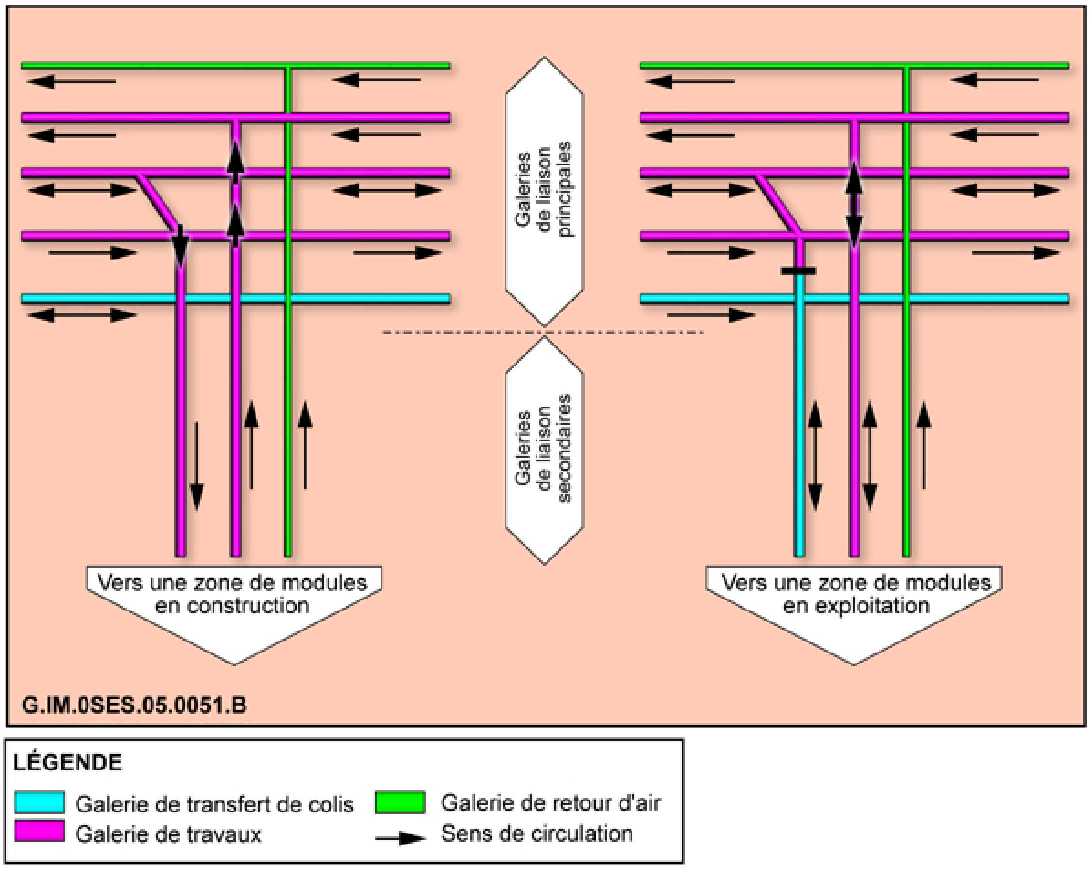

Figure 4.2.6 Diagram of the organisation and circulation in the connecting infrastructures

- **Description of the connecting drifts**

The geometry of the connecting cluster drifts is of the "inverse U" type. The favourable mechanical properties of the granite allow such a shape without any ground support.

In mechanically unstable zones or sections with significant water arrival, local bolting, injection or ground support may be required.

Package transfer drifts and construction drifts

These connecting drifts are fitted with a track for vehicles on tyres to provide suitable driving conditions preferably for one-way traffic.

Drift width ranges from 6-8 m for a height of around 6 m depending on the types of vehicles and required ventilation flow rate.

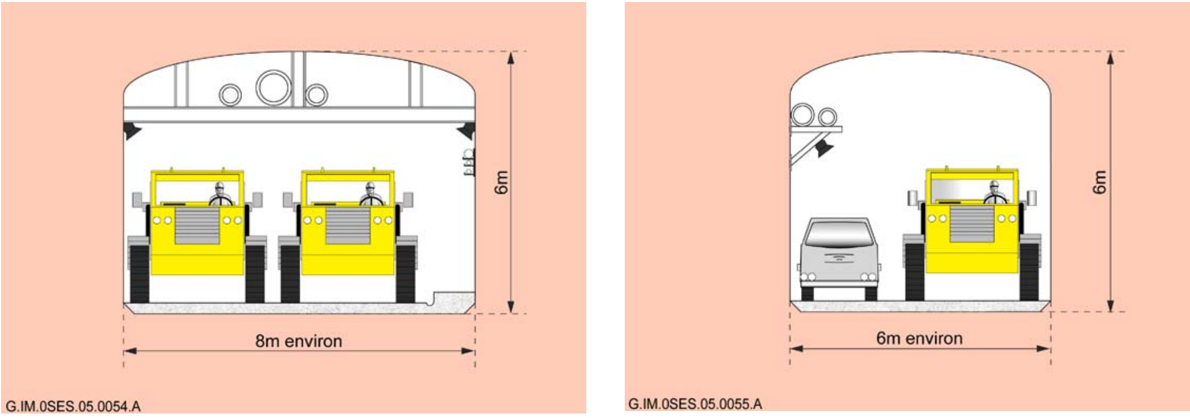

Figure 4.2.7 Typical section of construction drifts

Two-way traffic can be planned in the package transfer drifts, given the forecast level of traffic (ten convoys per day at the most).

The installation of a rail track is as well an hypothesis in one of the construction drifts to increase the transport throughput of broken muck (construction phase) or engineered barrier elements (operation phase) or backfill (when closure is decided).

Exhaust drifts

The exhaust drifts are 25-50 m^2 in section depending on their location in relation to the shaft zone infrastructures. The exhaust drifts accommodate the smoke extraction ducts. These ducts are fastened to the drift wall.

Interconnecting drifts

The drifts of a single cluster are interconnected, at the most every 400 m by perpendicular drifts: *the interconnecting drifts*. They have the same dimensions as the construction drifts so that site construction machinery, personnel evacuation and emergency vehicles can pass through them if required. They accommodate smoke extraction ducts connected to the exhaust drift. The drift intersections are designed to enable the bulkiest heavy machinery to manoeuvre.

- **Construction of the main and secondary connecting drifts**

Two excavating techniques can be envisaged for the connecting drifts: drilling and blasting, or mechanical excavation by tunnel boring machine (TBM).

At the current stage, the drilling-blasting method is taken as the reference because of the flexibility it offers, on the basis of comparative studies conducted by SKB [14].

This is because it enables sections of different shapes to be constructed and lends itself well to the gradual adaptation of the detailed repository architecture in response to the exploratory findings as construction progresses. However and whenever the connecting drift lengths are long (several kilometres), using tunnel borers is considered as an alternative, regardless of the architecture.

Mechanical excavation using a tunnel borer presents the advantage over the drilling-blasting method as it only generates slight damage to the excavation walls. Furthermore, in certain conditions (long stretches to be excavated in a straight line, homogenous geology, etc.) excavation productivity may be better than that achieved with drilling-blasting.

Given the dimensions of the main, secondary drifts and modules, solid seam blasting may be carried out.

The excavation sequencing is identical to that presented for ramp construction, namely in four stages:

- developing the blasting pattern, drilling the round and charging the mines;
- blasting the round and extracting the blasting gases by ventilation;
- removing the broken muck (mucking) and removing the unstable blocks in the roof (purging);
- making the walls safe (occasional or routine ground support).

As in ramp construction, pressure-grouting is used where water-conducting faults are crossed.

4.2.2.6 Drift seals and backfilling

It may be decided to close parts of the repository and lastly the access openings as part of the staged management of the disposal process. This essentially entails emplacing backfill and seals in the drifts to protect the repository modules from water circulations inside them (cf. chapter 4.2.1: design principles).

- **Seals in the repository drifts**

Very low permeability seals are systematically planned in the connecting drifts between the water-conducting faults and the repository modules installed in low permeability granite "blocks". They are made up of a swelling clay core mechanically confined by two concrete abutments anchored to the rock.

The seals are installed in portions of the drifts where the rock is not or slightly fissured and where excavating has been performed with methods limiting wall damage.

At the seal level, the damaged zone is removed by cutting an over-excavation using ornamental masonry techniques (diamond-tipped cable or disk sawing) [15]. Cutting is continued in the granite beyond the damaged zone through a thickness of about 50 cm.

The coupling between the seal core and the "sound" granite develops as clay swells. This coupling is favoured by the presence of retaining abutments which confine the clay into the drift axis.

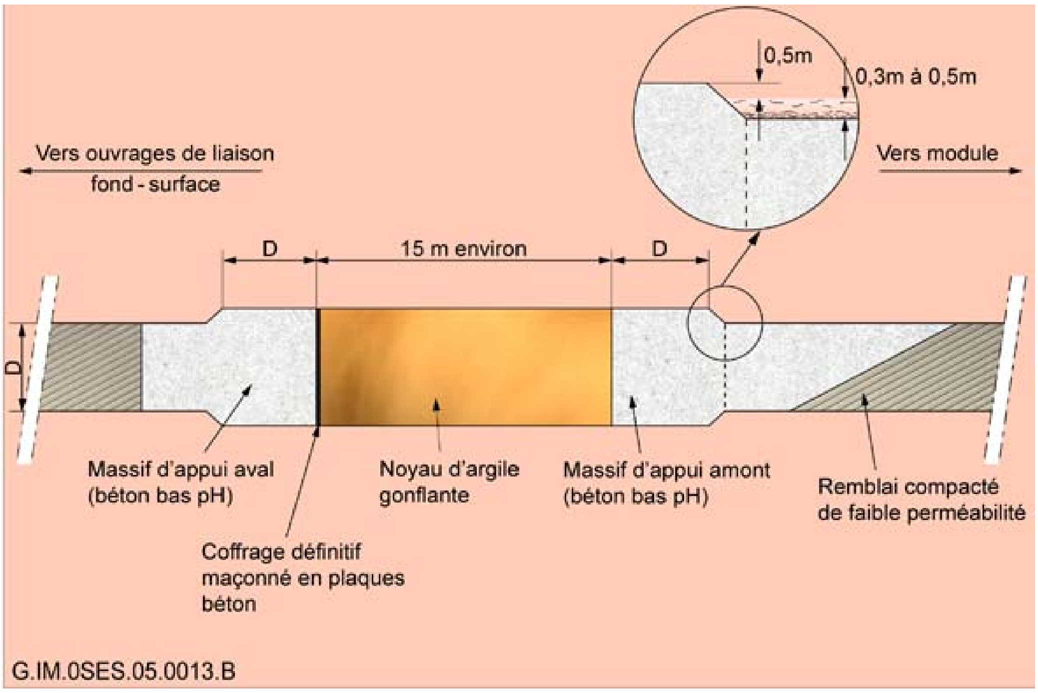

Figure 4.2.8 Basic diagram of a drift seal

A swelling clay core

The confinement of the clay core by the concrete abutments keeps the swelling pressure of the clay over 1 MPa. The hydraulic conductivity target for the clay is about 10^{-11} m/s. This may be achieved by adopting dry density of about 1.5 for the swelling clay if the clay is pure, and 1.75 in the case of a clay (70%) and sand (30%) mix.

The choice between pure clay of the MX80 type, Kunigel V1, used for the TSX experiment or Deponit CA-N examined by SKB or a blend of that sort of clay with sand, does not appear to be a decisive factor at this time [16] [17].

At this stage, it is considered that a core length of 10-15 m should enable sufficient non-transmissive contact to develop between the swelling clay and the undamaged (sound) granite.

A concrete abutment

An un-reinforced concrete abutment is constructed at each end of the core. The two abutments are dimensioned to withstand the swelling clay pressure by taking support on the over-excavation shoulders. Transferring the abutment pressure to the compacted backfill emplaced in the drifts upstream and downstream of each seal ensures redundancy in the core's mechanical confinement mechanism.

Each of these two abutments is equivalent in length to the width of the drift in which it is constructed, and a shoulder that incorporates a support surface some fifty centimetres wide on undamaged granite is planned.

High performance low pH concrete whose formulation enables thermal shrinkage to be minimised at the interface between the abutments and the surrounding granite, may be used.

Concrete of this type was used in the TSX experiment. Its main characteristics were:

- pH lower than 10 with little or no free calcium hydroxide ($Ca(OH)_2$);
- high single compression strength (of about 70 MPa);
- very low permeability of about 10^{-14} m/s in the concrete mass;
- low exothermic level when setting (temperature rise of about 25°C)

Construction of the seals in the repository drifts

Seal construction is carried out in stages with waiting time to let the concrete abutments set and partially dry (Figure 4.2.10).

The upstream abutment is constructed in two phases:

- in the first phase, slurry is poured between temporary formwork and the emplaced and compacted backfill in the drift upstream of the seal. The over-excavation is sawn at the edge of the sealing zone. Mechanical methods are used for surface-grinding of the over-excavation wall. The shoulder shape is adjusted. The formwork is taken down and readjusted to the over-excavated section right above the point where contact will be made between the swelling clay core and the upstream abutment;
- then the second part of the abutment is poured *in situ*. Once the concrete has set, contact inspection by core sampling is carried out so that it can be decided whether or not to inject grouting between the abutment concrete and the surrounding granite.

The swelling clay core is constructed by assembling a base of pre-fabricated bricks. Final core density can be improved by using large dimension bricks. Swelling clay powder can fill the annular spaces between the brick base and the drift walls to keep the dry density as high as possible.

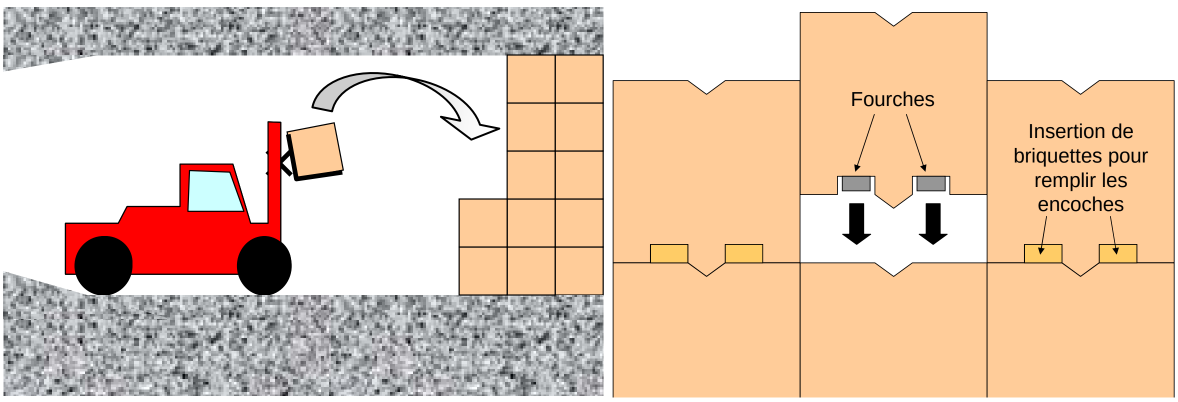

Figure 4.2.9 Principle of emplacing large dimension bricks in the swelling clay core

Other solutions may also be envisaged such as filling with pellets or a mixture of pellets and pure clay powder.

Permanent formwork made of concrete slabs is bonded with the last clay bricks to avoid hydration of the clay core when the downstream abutment concrete is poured. Temporary formwork is placed downstream of the second abutment which is poured *in situ*. Once the temporary formwork of the downstream abutment has been removed, the decision can be made whether or not to make grouting injections on the edge of the abutment concrete.

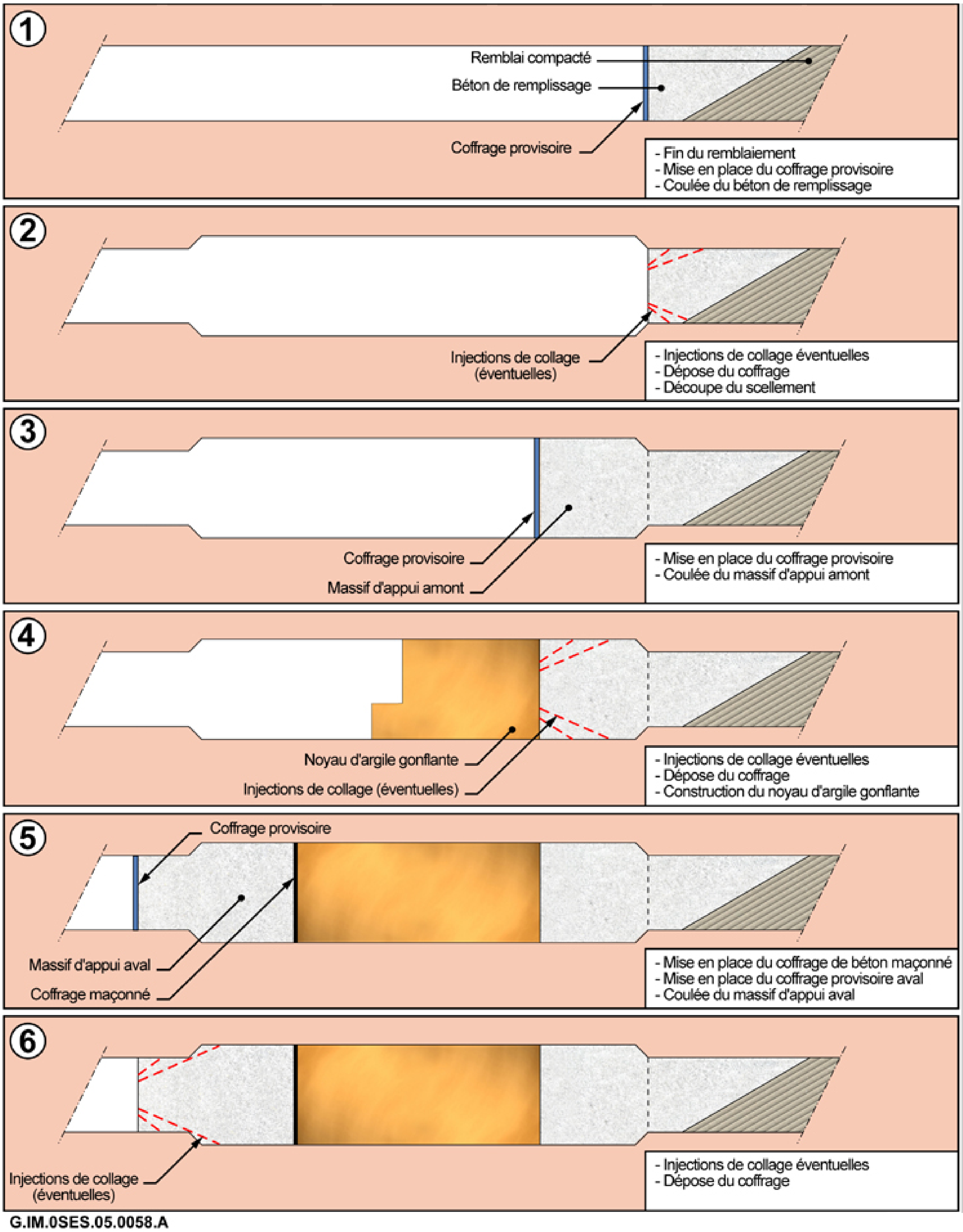

Figure 4.2.10 Construction principle of a drift seal

- **Backfilling the repository drifts**

Upon drift closure decision, they are fully backfilled. The current thinking is that in principle the low permeability performance of the backfill must be at its most efficient when emplaced in drifts close to the disposal cells or in the vicinity of the seals. The low permeability backfill considered is made up of granite aggregate and swelling clay, and mechanically compacted while emplaced. It should be possible to use standard backfill with or without a small amount of swelling clay in those drifts located far away from the disposal cells and if, in between, are already present, in redundancy, low permeability backfills and seals.

Interruption of the damaged zone at the excavation wall during the excavation works can be envisaged as additional means for enhancing backfill performance in the drifts where low permeability backfill is to be emplaced. This interruption could consist of cutting trenches on the edge of the drift using non-damaging methods (cf. excavation of seal zones).

The backfill specifications including the addition of swelling clay (granular/clay fraction, type of clay) and the emplacement and compacting methods envisaged at this stage are based on the solutions tested by SKB in Sweden [18].

By using a clay fraction (between 10% and 30% of the volume) the backfill permeability level as measured in the laboratory can be brought down (10^{-10} and 10^{-12} m/s respectively), while providing at the same time swelling properties. The granulated fraction adds mechanical strength to the backfill.

SKB has already resolved most of the difficulty of backfilling the upper part of a drift by developing a machine fitted with a vibrating plate mounted on a mobile arm. This machine can emplace and compact the backfill in continuous layers dipping from the drift vault down to the slab floor (Figure 4.2.11).

The main purpose of compacting is to increase backfill density to enhance its permeability performance. During the "Backfill and Plug Test" experiment, the densities obtained after compacting a backfill with 30% bentonite content, varied from 1.7 in the drift centre to 1.5 at the edge.

Future measurements will verify the permeability obtained by the backfilling operation for which correlations have been established in the laboratory (density ≈ 1.7/permeability ≈ 1.10^{-12} m/s; density ≈ 1.5/permeability ≈ 5.10^{-10} m/s).

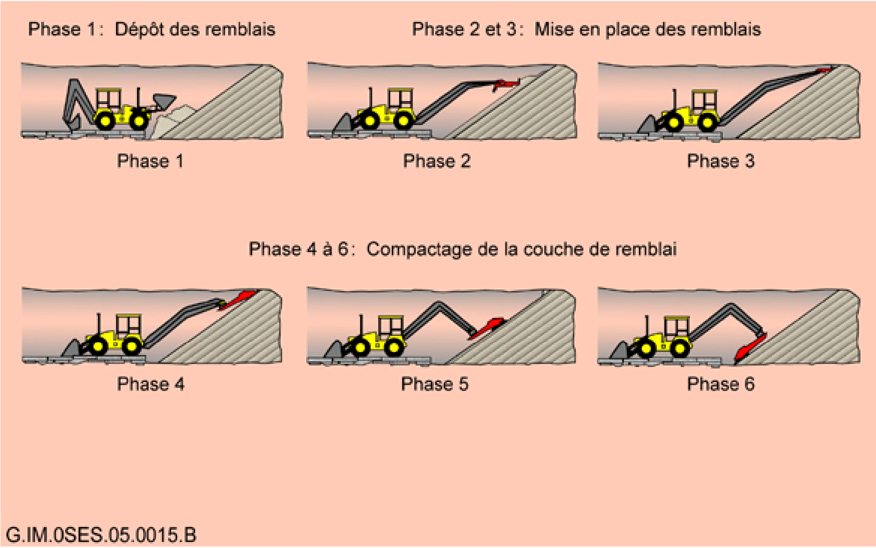

Figure 4.2.11 Emplacing and compacting low permeability backfill

4.2.2.7 Seals and backfilling of the surface-underground connecting structures

At the final stage of the repository closure process, the aim of backfilling and sealing the shafts and ramps is to limit the possibilities of direct water circulation between the potentially water-bearing superficial formations and the underground structures developed in the shaft zone at both repository levels.

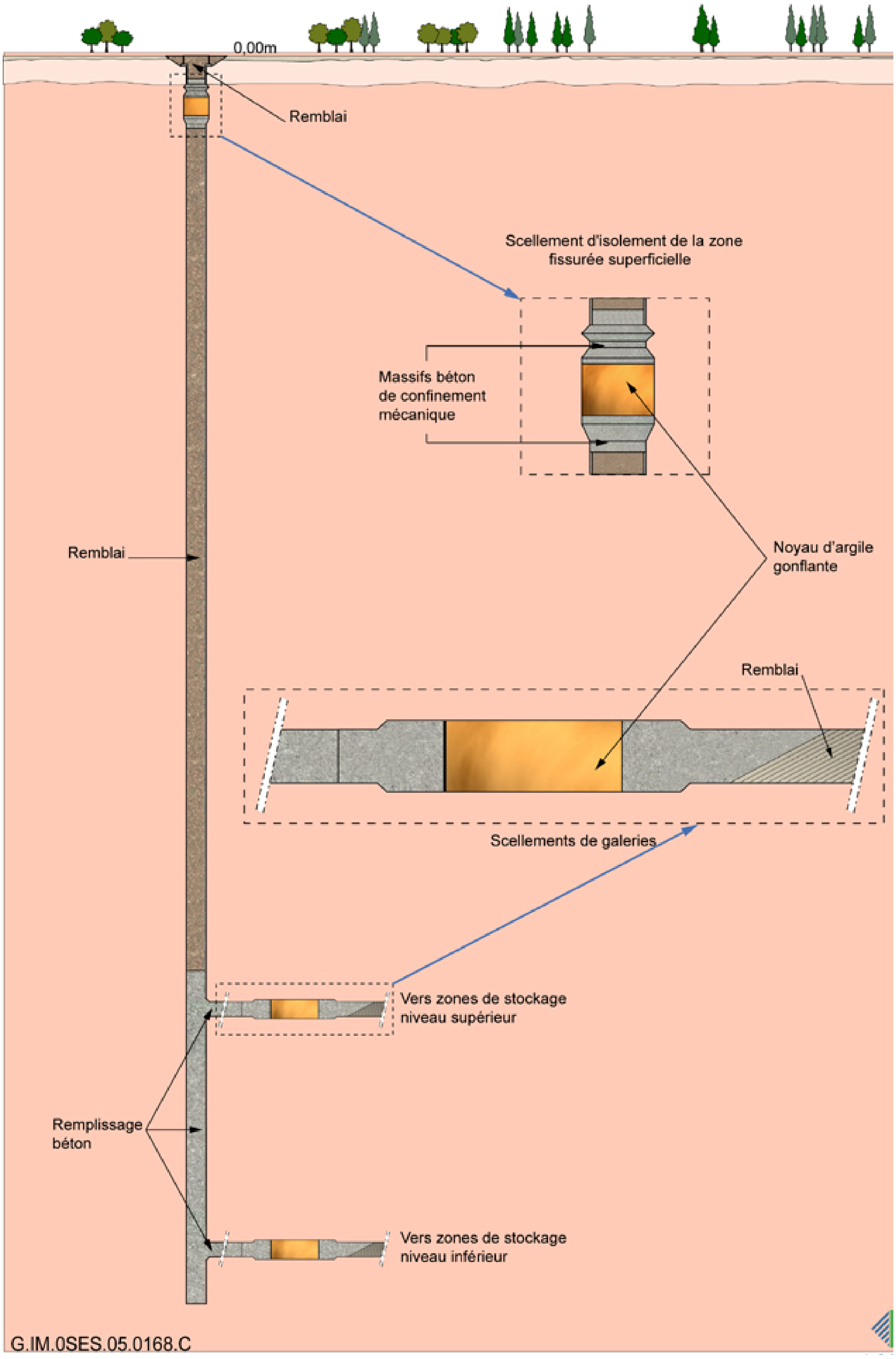

Figure 4.2.12 Shaft closure

Among available technical solutions, construction of low permeability seals combined with the use of slightly "tamped" backfilling materials is currently considered.

Seals of a similar design to those used in the repository connecting drifts are installed in all drifts serving an access structure (shaft and/or ramp). The shaft base is cemented to a height of about twenty metres above the upper level of the repository. The volume over this concrete filling is entirely backfilled up to a level about twenty metres below the leak-tight lining emplaced at the top of the shaft where a seal is constructed. The backfilling of the upper part, above the seal, completes the closure operation.

- **Seals in the access structures**

At the current stage of the study the seals comprise a swelling clay core mechanically confined by two concrete abutments anchored in the formation.

The seals are installed in the portions of the shaft or ramp excavated with methods limiting wall damage.

The concrete lining and damaged zone at the edge of the drifts and right above each seal are cut out (diamond-tipped cable or disk sawing). Cutting is continued in the granite beyond the damaged zone through a thickness of about 1 m.

The coupling between the seal core and the "sound" granite develops as the clay swells. The seal core, tens of metres high, also includes a buffer thickness to compensate for any deferred tamping of the backfill, in the event of concrete abutment failure. Two solutions are envisaged for clay core emplacement: tailored stacking of pre-compacted bricks or *in situ* filling with compacted pellets. In both cases, provision is made for inserting a device to collect any sweeping water on the shaft wall (gutters, weepholes), upstream of the sealing zone, in order to maintain favourable conditions for the construction of the clay core.

The core is mechanically limited by two concrete abutments, dimensioned to withstand the swelling pressure of the clay and the effective mass of the backfill materials emplaced higher up inside the shaft. Dimensioning these retaining plugs (shape and height) has yet to be made.

The seal for the ramp is similar to those planned for the repository drifts

- **Backfilling the access structures**

Only mechanical performance is envisaged at this stage for the access structures backfill. The particle size of the granite aggregate should minimise the voids after *in situ* compacting.

The addition of a fraction of swelling clay to compensate for the deferred tamping efects may be envisaged. Resort to this option should consider a draining water collection system upstream of the compacting worksite.

4.3 Disposal process

The industrial commissioning of the installations starts with the reception of the first packages on the site and their emplacement in the first constructed disposal cells. Simultaneously, construction work of new structures could gradually start while packages are emplaced in disposal cells. Operation of a reversible disposal facility is characterised by the coexistence of construction activities (including on-going geological surveying) and package emplacement, and also by the progressive feature of these activities.

Throughout this period, which could last several centuries, the reversible disposal system can be managed as a storage facility. Underground structures are continuously observed, which provides understanding of the phenomena concerning the installation, and solid knowledge of the state of the cells and their waste packages. Retrieving the waste packages in this configuration, if so decided, is easy.

The gradual nature of repository development also means making the most of the experience and knowledge acquired during the construction and operation of the first cells in order to adapt and develop the design of future structures.

At the end of an observations phase, which in principle has no set duration, the decision to close the repository cells is the starting point of the closure process. This is also gradually implemented through a series of successive stages, which can be separated by observation periods. Would package retrieval be decided at the start of the closure process, seal deconstruction and backfill removal would call for conventional civil engineering techniques; indeed, granite mechanical stability does not require complex technologies aiming at emplacing ground support while excavating.

Thus the staged nature of the repository process enhances the management flexibility offered by the modular design of the underground installations, combined with adapting the architecture to the fracturing of the host massif.

These elements ensure that at each stage:

- the installation can be kept as such and can be observed;
- going backward is possible, up to the retrieval of all or some of the emplaced packages;
- or keeping on with the disposal process is possible by deciding to move on to the following stage characterised by a greater passivity of the installation.

Likewise, while surveying and characterisation activities are going on, this capacity for action means pursuing, reorienting, suspending or abandoning the investigations of part of a repository zone, or even giving up the installation of disposal cells in an identified granite block.

4.3.1 On-going exploration and characterisation

On-going exploration and characterisation of the geological medium belong fully to the repository process in the granite context. They are performed according to the staged approach as described in the introduction.

The first two stages carried out from the surface and then in the laboratory, aim mainly at defining the general structure of the granite and identifying the criteria for excluding faults and fractures that may or may not be intersected in the connecting drifts and disposal cells. This identification results from an iterative approach phased by safety analyses.

The last stage aims at surveying and characterising the host blocks in detail prior to repository module construction, followed by the validation of the repository cell layout within each constructed module.

The small dimension geological objects addressed by this stage can only be detected using small-scale exploration techniques (geophysical tomography, geological correlations between core samplings, borehole logging, hydro-geological tests, etc.). Implementation of these exploration techniques calls for the construction of drifts and core samplings in the volumes of the massif which, on the basis of previous exploration phases appear to be suitable for accommodating repository structures.

The technical feasibility of exploration drifts and cored boreholes does not raise any particular issue with regards to construction feasibility of other repository infrastructures.

- **On-going block characterisation**

The first on-going exploratory phase concerns characterising the modules host blocks. Surveying activities are included in the general repository construction planning.

Hence the first surveys can be carried out during the construction of the last common infrastructures of the repository, in the shaft and main connecting drift zone. They concern granite massif zones where the first repository modules are to be installed.

Subsequent surveying are carried out in other volumes of the massif, while repository module construction work is going on in the blocks that have already been surveyed and qualified. The successive worksites can be accessed by the construction and commissioning of the first main connecting cluster drifts. These connecting drifts are constructed ahead of the surveying sites. Their lay-out is defined on the basis of previous surveys that have pinpointed the zones conducive to accommodating the repository structures in the granite massif.

The exploratory drifts are likely to intersect faults or water-conducting fractures (with hydraulic head at initial hydrostatic pressure). Dealing with the most water-conducting fractures by grout injection ahead of the working face or after excavation, and selecting the cementitious grout will be suited to the position of the permeable zone in relation to the blocks to be surveyed.

These adjustments aim at not disturbing any measurements to be performed while still maintaining favourable conditions for continuing excavation work (wall stability, checking drained water volumes, etc.).

These structures are considered temporary with the exception of the exploratory drifts (cf. § 4.2.2.4) whose installation in a block makes them suitable for conversion into connecting drifts in the module to be constructed. They are backfilled and sealed at the edge of the block at the end of the surveying work. All boreholes drilled in the block are plugged prior to exploratory drift closure operations.

- **On-going validation of the cell layout**

The second on-going exploratory phase entails identifying the disposal cells location in the module. This work is essentially conducted by boreholes. The distance in between boreholes and their length depend on the geometry of the cells to be constructed (cf. § 5.6 for B waste, and § 6.6 for waste disposed of in borehole).

This exploration work is carried out between the end of connecting drifts construction and disposal cells excavation. Carrying out surveying in part of a module while simultaneously excavating in neighbouring drifts must be assessed in terms of possible perturbations on measurements.

As in the case of exploratory drifts, boreholes are considered as temporary. Those drilled within the volume of a suitable location for disposal cell construction are not plugged (as their bored volume is included in the volume of the cell to be excavated).

Boreholes that have identified fractures not meeting the selection criteria defined at the end of previous exploration phases are plugged (low pH cementitious grout) or sealed in the vicinity of the drift from where they were drilled (swelling clay plugs and low pH cement plugs).

4.3.2 Nuclear operation

Nuclear operation is characterised by the sequence of operations from receiving primary waste packages and preparing disposal packages in surface installations, followed by package transfer and until emplacement in the disposal cells.

4.3.2.1 Primary package reception and disposal package conditioning

The primary waste package reception and disposal package conditioning activities are carried out in the surface installations.

Similar installations are operating in some existing nuclear installations such as the COGEMA reprocessing centre at La Hague and the Dutch COVRA storage facility. Thus the principles of operation and their associated means presented hereafter result largely from the transposition of industrial feedback.

The primary waste packages may be shipped by road or rail convoy from the production sites to the repository site in transport casks. Once on site, they are stored in a dedicated area of the surface installations, according to nuclear practices.

After storage, the packages are transferred to a so-called "conditioning building", comprising a succession of shielded cells for remote-controlled operations.

The packages are docked to an unloading cell and the primary packages are extracted by a travelling crane equipped with a specific grip for each package type. The primary packages are then transferred to a zone dedicated to their storage.

Disposal packages are prepared in a second phase. The primary packages are placed in disposal containers (concrete for B waste, steel for C waste and copper containers for spent fuel). The containers are closed using cementitious grout for B waste packages, and by welding for C waste and spent fuel. Then disposal packages are controlled then stored in a dedicated zone of the building.

Figure 4.3.1 illustrates the above process with C waste example.

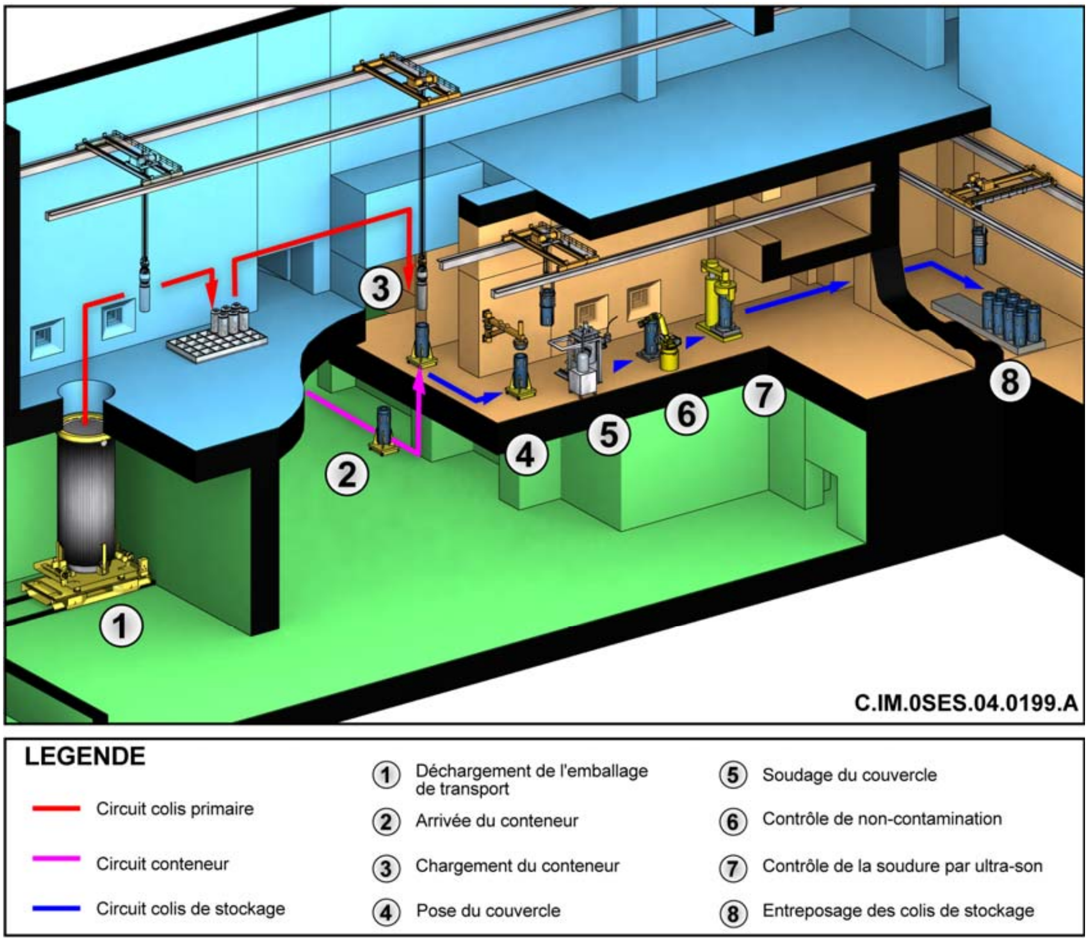

Figure 4.3.1 C waste disposal package preparation line

4.3.2.2 Disposal package transfer

The principle of transferring disposal packages between the surface installations and the disposal cells is mainly governed by radiological protection considerations. The residual equivalent dose rate of the majority of disposal packages (all types taken together: B, C and spent fuel) is such that they cannot be contact-handled. Thus the principle adopted consists of introducing the packages into a shielded cask (or radiation protection cask) in the surface installations. This cask is then transferred to the disposal cells.

The transfer cycle of the protection casks with their disposal packages from surface installations to the disposal cells can be broken down into the following three stages:

- loading the cask on the surface and transferring it to the package transfer shaft;
- transferring the cask from surface to underground by the shaft;
- transferring the cask in the drift up to a disposal cell

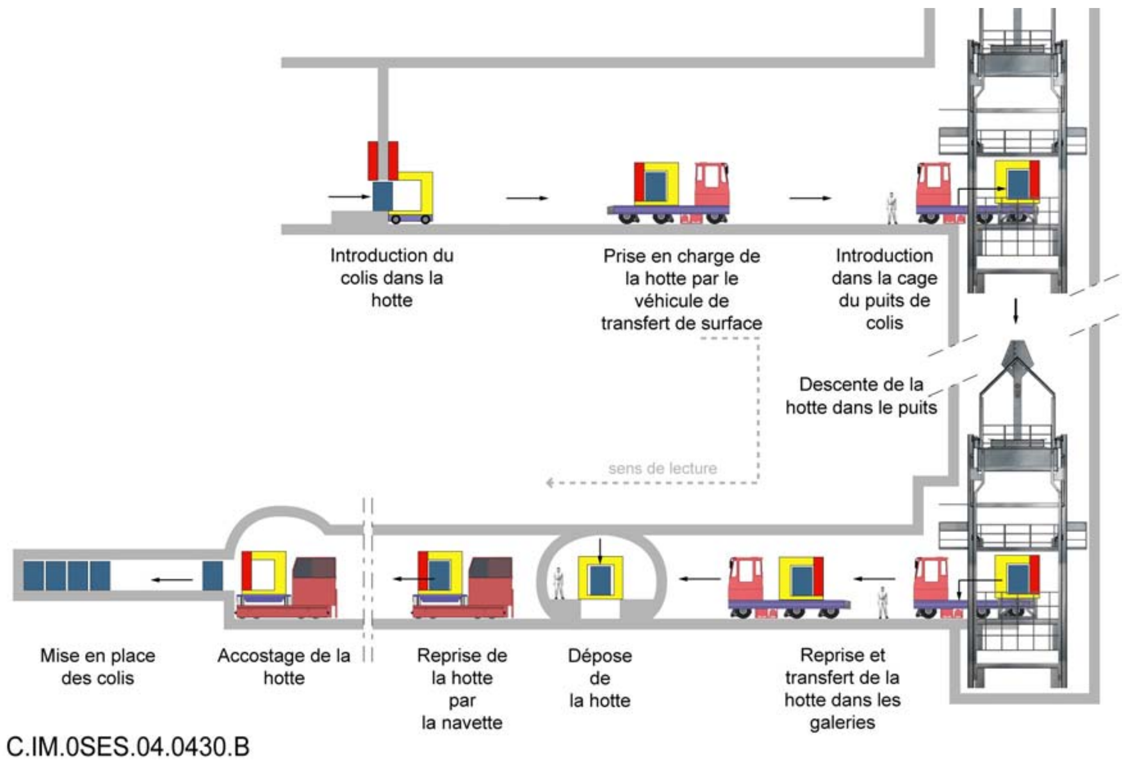

Figure 4.3.2 General diagram of transferring disposal packages from surface up to the cell

Cask docking and package emplacement operations are described in chapter 5 for B waste, chapter 6 for C waste and chapter 7 for spent fuel.

4.3.3 Repository module construction

Construction activities include excavation, ground support and equipment of the underground installations. The organisation of these activities is similar to underground workings (tunnels) in terms of work type, and to mine workings in terms of depth, ventilation and century-long duration of the process.

These activities call for permanent transportation of materials, equipment, ventilation air and operators in the underground installations.

Among these activities throughputs, removal of excavated materials and ventilation are dimensioning.

The organisation of broken muck transport from the blasting site to the broken muck dump for this type of underground site can be split into several stages:

- At first, the broken muck is collected at the work face by a load-haul-dump unit that tips the broken muck into a truck or a mine trolley. Generally rail transport is the adopted solution whenever a high throughput of broken muck takes place underground over distances of more than 2-3 km;
- the broken muck is then tipped into skip hoists and raised to surface level through the construction shaft or by trucks using the ramp;
- finally at the surface, the broken muck is transferred to trucks or conveyor belts and routed to a broken muck dump

The logic of this process is illustrated by Figure 4.3.3

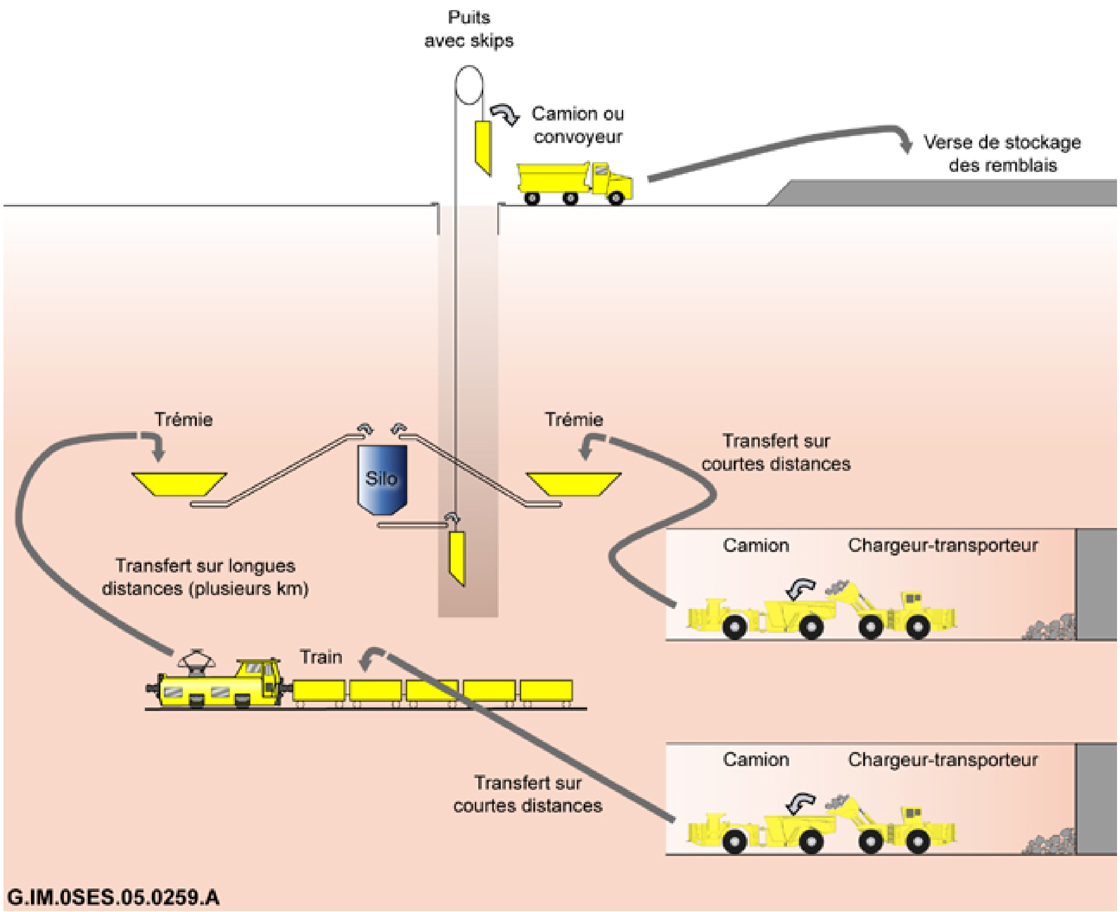

Figure 4.3.3 Broken muck transport diagram

4.4 Operational safety

As with all other industrial installations, the various repository activities (construction, operation, closure, and any package retrieval) may induce risks for man and the environment.

Incorporating these risks from the installation design stage leads to defining the operational safety functions and the suitable technical means to minimise the risks of occupational and public exposure.

Any potentially dangerous situations should be identified as part of an in-depth defence approach, at the same time as the installations are being designed. Measures should be put forward to prevent their occurrence and limit their effects, even if they appear fairly unlikely. This analysis benefits from feedback from the nuclear and mining industries.

Identifying the risks for a repository study in a granite medium is carried out in a generic context by highlighting the main risks. The studies carried out for research into a repository in a clay medium can largely be transposed to the case of granite[9]. The research results obtained for the repository studies in a clay medium are quoted as often as necessary.

9 The risk analysis for a repository in a granite medium is comparable to the analysis carried out for a repository in a clay medium **[19]**. A few additional features relate to the geological formation: risk of blocks falling, not only during construction but also during operation (in general, drifts will not be lined with concrete), water arrival (during the construction phase), etc.

4.4.1 Human protection

Repository-related activities call for the definition of safety functions that are specific to nuclear installations. Furthermore, protective measures for operators against the usual occupational risks of surface and underground construction work must also be taken into account.

4.4.1.1 Definition of the safety functions relating to repository operation

The functions of nuclear safety are as follows:

- **Confining radioactivity**

The radioactive materials have to be confined to prevent their dissemination. In particular, gaseous radionuclides likely to be released by certain packages will be limited as much as possible, and controlled ([20]). Checking the absence of package surface contamination also participates in radioactivity confinement.

Meeting this safety function limits the risks of radionuclide inhalation and ingestion by workers and the general public in the immediate vicinity of the repository installations.

- **Protecting humans from radiation**

This function consists of protecting operators and the general public from radiation emitted by packages.

This entails inserting fixed or removable shields, keeping operators at a safe distance from the sources and managing their exposure time.

- **Controlling the criticality risk**

This function consists of avoiding a criticality accident[10], whose consequences could notably undermine the radiation confinement and protection functions [22].

This objective entails preventing the reactivity by controlling the fissile materials, packages geometry, their distribution in the repository installations and, possibly by installing neutron-absorbing materials (neutron poison) between them[11].

- **Removing the residual thermal output**

This function consists of limiting the temperature levels inside the installations by dissipating the thermal output emitted by certain packages.

- **Removing radiolysis gases**

The explosive gases primarily arising from a radiolysis phenomenon[12] specific to certain packages have to be removed. Ventilating the installations participates in achieving this objective during the repository operating phase.

4.4.1.2 Human protection objectives

The public and personnel must be protected from the radiological risks associated with the industrial activities.

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[10] A criticality accident is an uncontrolled nuclear chain reaction (fission) between neutrons and fissile materials (uranium-239, plutonium-239 and 241).

[11] The role of these types of materials capable of capturing neutrons, is to limit the nuclear reaction.

[12] The radiolysis phenomenon is linked to the effect of ionising rays (β, γ) emitted by the radioactive materials on hydrogenated products present in certain B waste disposal packages (organic matter, matrix water, concrete container water). It mainly takes the form of the release of hydrogen, and also methane to a lesser extent.

The risk reduction measures during the operating phase means essentially radiological protection shields and measurements monitoring the absence of contamination and should limit human exposure below the limits set by Andra in its objectives for protecting humans from radiation [23] : 5 mSv per annum for personnel working in the nuclear zone, 0.25 mSv per annum for the public at large.

Humans must also be protected from other potential nuisances generated by the repository construction, operation and closure activities [24], such as:

- inherent physical risks (falling, crushing, electrical hazard, etc.) throughout the various repository activities and particularly during construction:
- fire risk;
- explosion risk by radiolysis gases (hydrogen, etc.) generated by certain B waste packages;
- risks inherent to the working environment (noise, dust, toxic gases emitted by heavy machinery, heat released by the exothermic packages, etc.).

Environmental protection requirements (as a reminder: underground and surface water [25] air [26], landscape, vicinity, soil, fauna and flora) also need to be taken into consideration throughout the repository activities, including during the post-operation monitoring phase. This monitoring should also include parameters likely to be involved in long-term safety functions.

4.4.2 Radiological risks during operation

The operation of nuclear installations induces waste package-related radiological risks. Design measures can reduce these risks to levels below those set in the objectives and ensure operation of the installations is safe.

4.4.2.1 Risk of external exposure

Waste packages are sources of external exposure (through β, γ rays and neutrons), from the time of their reception in the surface installations until they are disposed of in the underground installations.

The transport casks containing the primary packages delivered to the repository site have a radiological protection function, and their structure is designed for the radiological characteristics specific to the waste conveyed. Once they have been removed from their packaging, the primary packages are handled and conditioned inside cells made inaccessible to the operators working on them behind radiation shields (walls, viewing ports).

Management of external exposure risk during transfer operations and until the packages are emplaced in the disposal cells, is controlled by placing radiation shields[13] between the radioactive sources and the operators to attenuate the radiation flows. This is achieved by the disposal package transfer casks, the operating plugs of the C and CU disposal boreholes and the B waste tunnel gates. Protecting or distancing the control unit of the machinery used to transfer or insert the packages into the disposal cells will also contribute to the operator dose reduction, therefore meeting the safety objective set by Andra.

A preliminary estimate of doses received at the various workstations has been made for the repository project in the clay medium [27], which has very similar operating processes.

[13] The material used for these shields depends on the type of radiation emitted by the radioactive source:
- for gamma rays, the materials used are heavy materials such as steel, concrete and leaded glass;
- for neutron rays they are specific materials (with boron or cadmium, etc.) or hydrogenated materials;
- alpha and beta rays do not need any particular shielding because they are stopped by the package envelope.

The external exposure dose is the main component of the radiological risk. The maximum dose received by the operators per annum would be of the order of 4 mSv. These doses could be optimised as part of an ALARA ("as low as reasonably achievable") approach in the later development stages of the repository project

4.4.2.2 Risk of internal exposure by ingestion or inhalation of radioactive materials in the form of aerosols

This risk is most likely to occur in the surface installations. It could be linked to the dispersion of radioactive particles from the transport packaging, the packages (primary packages, disposal packages) or transfer casks.

Management of this risk depends on how the primary package reception and disposal package conditioning installations are organised into confinement systems[14] to avoid radionuclide dispersal into the operator circulation zones or the environment. Furthermore the ventilation systems of these installations need to be equipped with filtering devices, as they are in other existing nuclear installations of the same type[15]. Lastly, it must be stressed that routine contamination checks[16] would be made on transport packaging, packages and casks.

4.4.2.3 Risk of internal exposure by inhalation of radioactive gases emitted by disposal packages

Certain B waste disposal packages B (B2 and B5) emit small quantities of radioactive gases (tritium, carbon-14, etc.).

In the surface installations, the limited number of waste packages present would make the total quantity of gases released in to the atmosphere negligible. In the repository tunnels where a high number of packages are stored, ventilation with air extraction via ducts to the repository exhaust shaft would remove these gases without affecting the operators working in the underground installations.

Atmospheric releases could be assessed along with their potential impact as part of the studies carried out on a real site. In the light of the ventilation flows and the negligible quantity of radioactive gases released, the calculated doses would be very low.

4.4.2.4 Risk of internal exposure in the underground installations of the repository by inhaling radon gas emitted by the ground

This risk, present from the start of construction activity, is linked to the natural exhalation rate of the granite on the site of the underground installations.

Data relating to mining operations in this type of environment [29] shows that management of this risk requires the underground drifts to be permanent ventilated to dilute and remove the radon emitted into the outer atmosphere.

4.4.2.5 Summary

Preliminary dosimetry estimates indicate that the doses received, taking all possible invasive pathways together, would be below Andra’s annual limits set for workers and the general public.

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[14] The principle of a confinement system is to create a difference in air pressure between adjacent premises.

[15] These filtering devices are as well justified by planning for accidental situations, in particular for receiving and conditioning bare spent fuel contaminated by corrosion products deposited and activated as the fuel assemblies is present into the reactor.

[16] The acceptance thresholds could be those laid down by French Transport Regulations, namely labile surface contamination (not set) limited to 4 Bq/cm^2 for β, γ emitters and 0.4 Bq/cm^2 for α emitters

4.4.3 Risk analysis in accidental situations

This paragraph presents the main risks likely to occur carrying out repository activities and the associated risk reduction measures.

The risk analysis is related to the inherent risks of the repository ("internal" risks) at this stage of the study. It covers the "conventional" risks found in any industrial installation and the risks stemming from the presence of waste packages, that are essentially radiological (risk of external exposure, risk of internal exposure and even criticality risk).

However, since there is no determined site, the "external" risks relating to the repository environment (seismic risk, climate risks, risk of aircraft crashing, etc.) are not developed. They would mainly affect the surface installations that could be damaged by this type of event and could also lead to degradation or loss of safety functions ("radiological protection" function, "radioactivity confinement" function, etc.). Consequently, the allowance made for these risks will be based on the general rules applied to basic nuclear installations (BNI) [30] [31].

4.4.3.1 Conventional risks

The behaviour of the terrain may have an impact on disposal process risks: the risk of operators being crushed by block falls in the drifts (notably while excavating), risk of water ingress in the shafts or drifts. They are also associated with equipment (loads dropping during handling operations, objects dropping in a shaft, operators falling from elevated work platforms, crushing by moving equipment, etc.) or the use of heavy machinery (knocking over an operator, colliding vehicles).

Maintenance personnel may get electric shocks while working on the equipment and vehicles or the latter may cause a fire with consequences for many workers, especially when underground.

It should be noted that there are minor risks associated with the package characteristics, that would be caused by temporary loss of ventilation in the installations: this relates to the risk due to the presence of exothermic packages (C and CU packages) and the risk of explosion arising from the release of small quantities of explosive gases[17] by certain B waste packages (B2, B5).

Prevention is the key to risk reduction. It involves choosing materials that do not propagate fire, limiting the calorific load present, the use of suitable equipment and vehicles that are reliable and fitted with all the safety devices necessary, properly serviced and furthermore personnel training, awareness of the various types of risks incurred, observation of driving procedures and rules, and wearing individual protection gear[18].

As-you-go exploration at the working face of the excavations can help prevent risks relating to poor state of the ground or ingress of water in the underground installations. The risks run by the staff can also be reduced by installing physical protection devices (particularly for work in the shafts), the use of heavy machinery that can be controlled away from the working face, installing thermal shields (firebreak doors, fireproof materials), equipping with site safety systems (fire smoke extraction systems, fire protection water circuits, communications networks, etc.) as the work progresses..

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[17] These gases (mainly hydrogen) are caused by a radiolysis phenomenon that is due to the effect of ionising rays (β, γ) emitted by the radioactive materials on hydrogenated products present in the disposal packages (organic matter and conditioning matrix water).

[18] This protection gear includes safety shoes, ear protectors, dust masks, self-contained breathing apparatus (self-rescuers) to be used in the presence of fumes…

Simulations of the risks from packages made during the studies for a repository in a clay medium [27] show that in both cases the risk is controlled in the operating phase by ventilating the various repository installations, and that if ventilation of the installations is interrupted there are no real dangers as there is plenty of time to intervene. These conclusions can be transposed to a repository in a granite medium.

4.4.3.2 Radiological risks

The various radiological risks arising from dangerous situations during the repository operation activity have been identified. The findings are presented by type emphasising those identified risks that appear to be the potentially most harmful to mankind or the environment. As with the study carried out in clay [19], this analysis has the benefit of feedback from nuclear installations that are comparable to a repository.

These risks and their main associated risk reduction measures are as follows:

- **Risks arising from radiological protection failures**

An exposure risk may be linked to a failure in the protection provided by the cell doors, covers and viewing ports, the transfer casks, C waste and CU disposal borehole plugs and B waste disposal tunnel gates. This event could be due to play between moving parts that does not comply with the initial dimensioning. The measures associated to this risk would be: installing radiation detectors on the cells, transfer casks and disposal cells and also workers wearing suitable dosimeters.

Accidental exposure can also be envisaged during a maintenance operation in a room adjacent to a radiating cell. The maintenance operations will be preceded by a radiation check in the room. The maintenance zone could be sited away from the potential radiation field of the radioactive sources in the cell.

Malfunctioning of equipment while handling or transporting a package could lead to it immobilisation and require the intervention of maintenance personnel. The preventive measures would be adequate maintenance of the equipment used and redundancy of some of its components (motorisation, etc.). This intervention would expose the personnel operating close to the radiating source. The most effective measures to suppress or limit personnel exposure would be to provide means to remove the load and bring back the empty equipment to its maintenance shop, minimising the time that operators would be present by accurate intervention planning, and dimensioning radiological protection thicknesses.

- **Risks relating to the consequences of a fire**

A fire in the nuclear installations of the repository could have radiological consequences if a package of radioactive material is involved. The use of package carriers on the surface or underground, would call for specific precautionary measures as fire could degrade the radiological protection in place or worse could lead to the loss of confinement of radioactive material and their dispersion into the atmosphere.

At package reception, there should be no radiological consequences of a fire on the carrier with packages placed in over-packs that would withstand a fire breaking out at 800°C during 30 minutes [28].

Studies carried out for the repository in a clay medium on cask carriers [27] show that temperature rises are likely to affect the metal envelope of the casks without undermining the structure of the cask itself.

The waste packages would not be subject to a temperature rise likely to damage their packing or degrade their matrix. However once the fire were put out, the radiological protection level of the cask would have to be inspected, so that if needed, appropriate measures (for example mobile radiological protection devices) could be taken to protect those people called on to work on them.

Similarly, simulations of a fire on a machine used to emplace B waste packages in their repository tunnel [27] show that there would be no consequences for the waste packages, in the sense that the vehicle is designed with a built-in thermal shield between the motorisation originating the fire and the handling unit holding the waste package.

These various results hold true for the repository installations in a granite medium because identical vehicles would be used.

- **Risks of packages falling in the surface installations**

Falling packages are generally events routinely examined in installations where radioactive waste packages are handled.

In surface installations, the handling of primary packages by travelling cranes, transfer trucks, etc. could result in falls and damage in the primary package and reception disposal package conditioning installations. The possible consequences could be that the envelope of one or more primary packages opens followed by radioactive material dispersion inside the installations then outside into the atmosphere via the air extraction circuits.

The first preventive measure is to limit package handling heights to their drop resistance height. Furthermore as far as the equipment is concerned, prevention calls for dimensioning with margins for the gripping units, possible redundancies on certain components, systems for maintaining the grapnel closed in the event of an electrical power failure, etc. Personnel training and equipment maintenance are also extremely important for managing this risk.

These various measures should reduce the risks of packages dropping and eliminate the risk of loss of confinement. However, on a par with practices in similar nuclear installations, using static and dynamic confinement devices could be one way of eliminating any risk of radioactive material dispersing into the atmosphere in the event of a package drop.

A disposal package containing the primary package(s) could also drop during a handling and transfer operation.

The prevention measures are identical to those given above. Loss of confinement can be ruled out since maximum drop heights are of the order of two metres and the disposal package envelope and cask (during the transfer to the shaft) protect the primary packages

- **Risks of packages falling during disposal package transfers between the surface and underground installations**

The transfer of casks containing the waste packages between the surface and underground installations could either be carried out by a vehicle travelling in a sloping drift (ramp), or by a cage in a vertical shaft. This drift or shaft would be specifically allocated to package transfer.

The associated risks would be limited in the case of package transfer via a ramp. A cask dropping by breaking away from its fastenings on the vehicle could not have radiological consequences for same reasons as during surface transfer. If failure of the various vehicle braking systems were to cause impact to the cask, the impact speed would be an aggravating factor

Protection measures such as installing an energy absorber on the carrier structure, limiting lengths of straight drifts and possibly escape bays for emergency braking areas, should enable this risk to be managed and avoid cask breakage.

In the case of package transfer via a shaft, dangerous situations would be the uncontrolled drifting of the cage, or possibly free-falling in the very improbable event of its suspension cables snapping[19] off.

The prevention measures for these situations (cf. Figure 4.4.1) are drawn from feedback acquired in deep underground mines. They cover first equipment design (independent braking devices on the drive pulley, independent cage suspension cable clusters, etc.) and as well maintenance, inspection and operation procedures.

An additional safety system independent of the cage's control system could be a cage drop-prevention system that would be triggered in the event of excess speed.

These systems could be supplemented by other protection measures:

- an end-of-travel braking system, similar to the statutory type installed in mines, a few metres below the underground station, would limit the mechanical consequences of a drift;
- an energy absorber located in the lower part of the shaft would appear to be a solution to deal with a free-falling cage. Simulations on a similar system, made as part of the repository study in a clay medium [27], show that the cage, the shock absorber and the transfer cask would absorb practically all the kinetic energy of the travelling body. C waste and CU packages should withstand the impact without breaking, although loss of confinement cannot be ruled out for the most fragile B waste packages (B2 bituminised sludge packages);
- these preliminary results would lead to providing risk reduction devices, with primarily the possibility of installing filtration on the shaft air extraction circuit (cf. Figure 4.4.1)) to control releases into the atmosphere following a cage fall[20].

[19] In Germany, a study carried out for the Gorleben radioactive waste repository project [32] estimates that the probability of a cage in a shaft dropping is 5.10^{-7}/year (operating at 5000 hours per annum), for a similar installation to the planned Andra repository.

[20] This measure has already been adopted in the American WIPP transuranic waste repository, in New Mexico. It should be noted that this filtration unit, installed on the exhaust air shaft, only operates in the event of an incident.

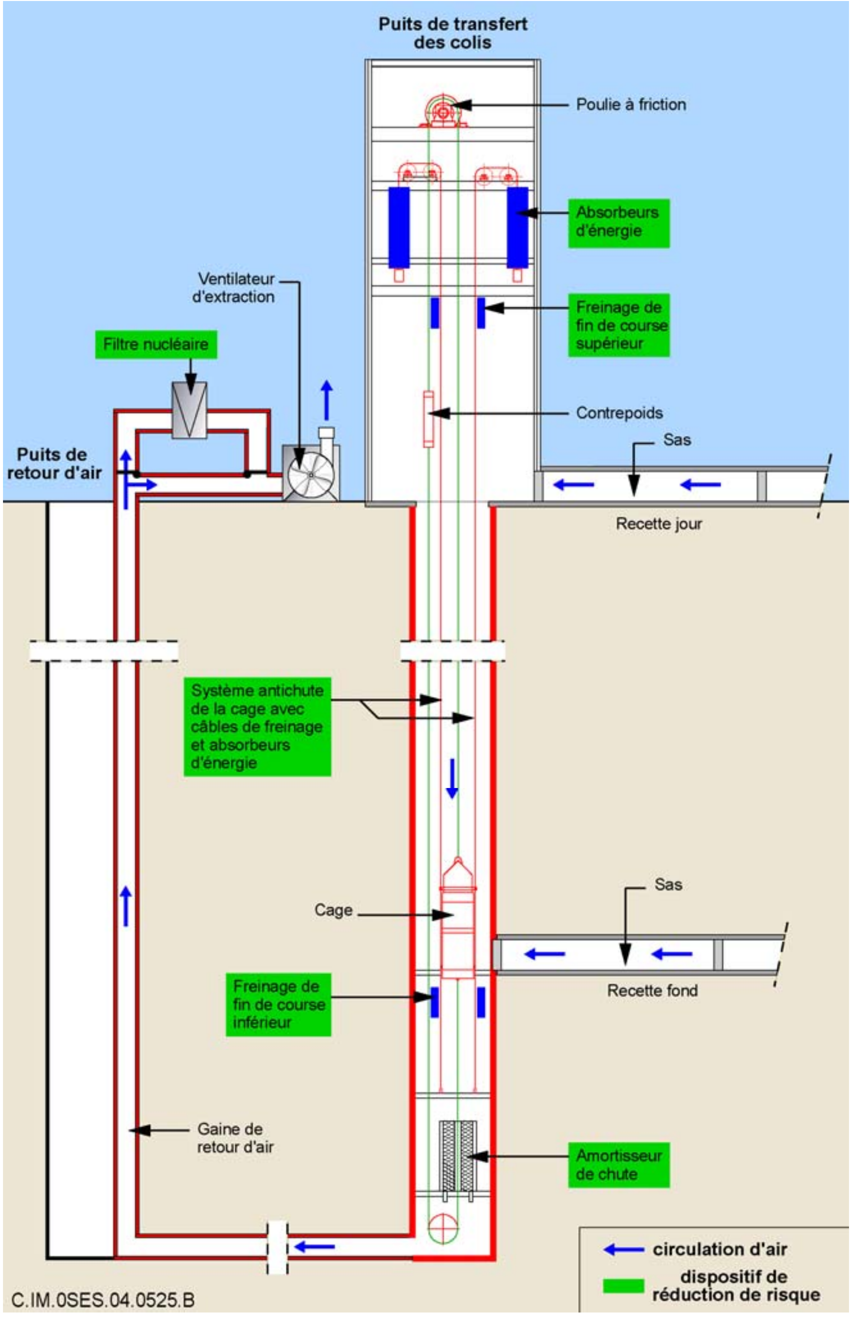


Figure 4.4.1 *Diagram of the risk reduction devices envisaged for package transfer via a shaft*

- **Risks of packages falling in the underground installations**

As on the surface, would a cask containing the package fall off from a moving carrier, there should be no radiological consequences because of the cask's mechanical strength, the short potential drop height and the carrier's low travelling speed.

As for the risk of dropping when the package is emplaced in the disposal cell, the analysis varies according to the package type in question. A number of simulations have been made to estimate the consequences of a B waste package falling during emplacement in horizontal tunnels, and those of a C waste or spent fuel package at the time of insertion into a vertical shaft. These studies presented in fuller detail in chapters 5 (B waste packages), 6 (C waste packages) and 7 (spent fuel), have not brought to light any risk of breakage of these different waste packages.

4.4.3.3 Criticality risk

The criticality risk corresponds to an uncontrolled nuclear chain reaction. This is initiated by increased neutron activity on fissile materials (uranium-235, plutonium-239 and plutonium-241).

B and C packages do not contain a sufficient quantity of fissile material (critical mass) for this type of reaction to occur. Only spent fuel packages are concerned by this risk [34].

In the case of spent fuel nuclear surface installations[21], water ingress must be prevented in spent fuel reception and disposal package conditioning cells to rule out the criticality risk, in line with the practices in similar existing waste storage installations.

In underground installations, package transfer and emplacement must be undertaken in dry conditions. There is no criticality risk associated with this operation.

During disposal package transfer via the shaft between the surface and underground installations, the hypothesis of a scenario combining serious damage to the spent fuel package (alteration of its internal geometry, fuel assemblies breaking and coming closer together, etc.) and ingress of water could lead to a criticality risk.

Given the planned risk reduction devices for the risk of falling in a shaft (cf. Figure 4.4.1), damage at such a level to a package seems unlikely. However to eliminate this risk altogether, an additional precaution would be to ensure that there is no water (or other hydrogenated fluid) in the shaft. That would entail prohibiting the installation of pipes in the package lowering shaft while at the same time installing a water extraction system at the bottom of the shaft.

The other dangerous situations envisaged [34] do not appear to induce a criticality risk.

4.4.4 Conclusion

The analysis presented in this chapter has highlighted the main risks associated with the disposal process for mankind and the environment, and proposed suitable prevention and protection measures. It is based on feedback from existing nuclear installations and lessons learnt from the clay dossier studies. It is not intended to be exhaustive at this stage of the studies.

The analysis has distinguished conventional risks, found in all industrial installations, from the risks associated with the waste packages. The risks relating to the outer environment of the repository have not been covered as there is no specific site.

The conventional higher and lower level risks have been identified for all the repository activities carried out in the surface installations. They are essentially risks of crushing (handled loads dropping, being knocked over by a vehicle, etc.) risks of falling through aerial work, risks of electric shocks, fire, etc. These risks do not call for additional investigations at this stage. Nonetheless, they should be delved into at later development stages of the studies.

In the underground installations, the risk of granite blocks falling during the construction of structures, and to a much lesser degree, that of water ingress, are additional to those risks mentioned above.

[21] The presence of water, that attenuates neutron energy and slows down their speed, makes them more reactive to fissile material and results in increasing system reactivity. Therefore, the disposal package conditioning processes adopted are dry processes, with no addition of water.

At a later stage of the project, the fire risk will have to be specifically examined, given its influence on the design of the installations, to ensure that recommended solutions achieve suitable safety conditions for evacuating personnel.

The risks relating to waste packages are essentially radiological risks. They are inherent to the repository operating activity, and to a lesser degree to the closure activity. These risks could be combined with exposure to radiation (radiological protection defects, interventions close to a radioactive source, etc.) in addition to fire or fall affecting the packages themselves. The provisions envisaged that have the benefit of feedback from similar industrial installations, should enable these risks to be managed.

The risks relating to the repository environment (earthquake, meteorological conditions, aircraft crashes, etc.) that would be assessed on the basis of usual practice in French nuclear installations, taking into account the local characteristics of the site retained for the repository, do not appear to raise specific difficulties.

At this stage the analysis conducted on operational safety has not revealed any elements that would undermine the technical feasibility of repository construction, operation and closure with its reversible staged management (including in particular the possibility of back-tracking).

5 B waste repository zone

This chapter describes the proposed design for the B waste repository zone. It takes into account the diverse nature and specific characteristics of B waste primary packages. This chapter indicates how the proposed options fulfil the safety functions defined in chapter 3.1. It then goes into detail on these options for the disposal packages, disposal cells and repository zone architecture respectively.

This chapter also shows how these options can be implemented for a flexible, reversible disposal process that meets operational safety and security requirements.

5.1 B waste primary packages

A few reference waste packages have been identified in the inventory model to cover the wide diversity of the nature of existing and forecast B waste. They are organised in a tree structure (cf. § 2.3). The following paragraphs provide a description of the primary package groupings into reference packages.

Table 5.1.1 List of reference packages in the inventory model

Reference packages	Cat.	Lev.1	Lev.2	Lev.3	Titles of waste grouped in the reference packages	Description
Activation product waste	B	B1			CSD-C containing activation product waste from PWR and fast reactors	§ 5.1.1
Bituminised waste		B2	B2.1		238- and 245-litre bitumen drums	§ 5.1.2
			B2.2		428-litre bitumen drums	
Technological and miscellaneous waste cemented or compacted		B3	B3.1	B3.1.1	1,000-litre concrete containers, reconditioned or not in metallic containers	§ 5.1.3
				B3.1.2	Concrete containers (CAC and CBF-C'2) containing miscellaneous technological waste	
				B.3.1.3	1,800-litre concrete containers containing miscellaneous waste	
			B3.2	B3.2.1	500-litre concrete containers (sludge and concentrates)	
				B3.20.2	1,200-litre concrete containers (CBF-C"2) containing CEDRA and AGATE waste	
			B3.3	B3.3.1	Standardised container for compacted waste (CSD-C) containing alpha waste	
				B3.3.2	EIP drums containing cemented pulverulent waste	
				B3.3.3	500-litre steel containers containing miscellaneous waste	
				B3.3.4	870-litre steel containers containing miscellaneous waste	
Cemented cladding waste		B4			Drums of cemented hulls and end pieces	§ 5.1.4
Compacted cladding waste with or without technological waste		B5	B5.1		CSD-C containing a mixture of hulls and end pieces and technological waste (including organic waste)	
			B5.2		CSD-C containing a mixture of hulls and end pieces and metallic technological waste	
			B5.3		CSD-C containing PWR (HAO) cladding waste, with no technological waste	
			B5.4		CSD-C containing magnesium cladding waste	
Cladding and technological waste placed in drums		B6	B6.1		180-litre steel containers containing AVM (Marcoule vitrification shop) operating waste	§ 5.1.5
			B6.2		EIP drums containing metallic cladding waste	
			B6.3		EIP drums containing magnesium cladding waste	
			B6.4		EIP drums containing metallic and organic technological waste	
			B6.5		EIP drums containing metallic technological waste	
Sources		B7	B7.1		Source blocks	§ 5.1.6
			B7.2		CSD-C containing PWR primary and secondary source rods	
			B7.3		EIP drums containing sealed sources	
Radon and americium waste		B8	B8.1		EIP drums containing radium-bearing lead sulphate drums	§ 5.1.7
			B8.2		870-litre steel containers containing lightning rod heads with radium or americium	
			B8.3		ORUM EIP drums	

*EIP = The French acronym for multipurpose storage facility (Marcoule)

5.1.1 Activated metal waste from nuclear reactors

The first set of B waste comes directly from electricity-generating reactors: operating waste from the existing pressurised water reactors (PWR) and activated waste from the SUPERPHENIX fast neutron reactor.

PWR neutronic poison and control rod assemblies represent more than eighty percent of the total weight of activated waste. Each assembly contains twenty-four fuel rods suspended from a support system which fits into the locations left for this purpose in the fuel assemblies (see Figure 5.1.1).

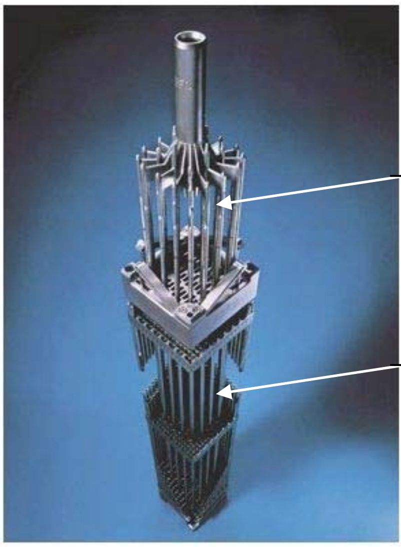


Figure 5.1.1 PWR fuel assembly with its rods

The neutronic poison rods are used during reactor start-up (first operating cycle) to control excess reactivity due to the use of entirely new fuel; they are removed during the following cycles and therefore become waste. The control rods are used to control the reactor power level and its immediate shut-down if necessary. They are replaced after several cycles in the reactor and also become waste.

Some rods contain neutron-absorbing materials: boron, in the form of PYREX glass for the neutronic poison rods, boron carbide (B_4C) and/or an alloy of silver, indium and cadmium (SIC) for control rods. The number of rods containing these materials depends on the reactor.

Other activated wastes from PWRs include metal waste, mainly dead-end tubes, known as core instrumentation system (CIS) thimbles fitted on the underside of the reactor vessel. These tubes are used to insert the neutron probes required to control the nuclear reaction. They are replaced, if necessary, after a certain period of use and then become waste.

These various types of reactor operation waste have specific chemical and radiological characteristics distinguishing them from other types of B waste. Their specific chemical nature is due to the nature of the materials making up some of the waste. More specifically, control rods add significant quantities of SIC alloys and B_4C, whereas other types of B waste generally contain none of these. The radiological activity is due solely to activation products formed by neutronic activation of the elements and impurities contained in the waste materials during their time in the reactor. These activation products are located inside the materials and are therefore unlikely to be dispersed. Among them, the activation products that contribute most to the radioactivity of the waste are, in decreasing order, nickel 63 (^{63}Ni) which is a long-lived isotope (half-life 100 years), then iron 55 (^{55}Fe) and cobalt 60 (^{60}Co) which are short-lived isotopes.

The high radioactivity of ^{60}Co has thermal consequences. This waste therefore belongs to category B, with the highest heat rating in relative terms (roughly 20 Watts per package when the packages are produced, taking the conditioning hypotheses set out below). Since this is largely due to cobalt, the residual heat rating of the package drops rapidly as the cobalt decays. As an illustration, the heat rating is divided by 2 after 5 years of cooling, by 3.5 after 10 years of cooling and by 6 after 15 years of cooling. Another consequence of this radiological inventory is the high level of radiation of the packages. Thus, the equivalent ß-γ dose rate in pseudo-contact with the package (i.e. at a distance of 5 centimetres) is roughly 50 sieverts per hour (Sv/h), at the time of its production. This is mainly attributable to ^{60}Co, but also to silver-108m (^{108m}Ag) whose half-life is 420 years, and remains at a relatively high level even after 10 years' cooling (around 15 Sv/h).

The conditioning hypothesis considered in the study [22] is compaction of the waste placed in holders[23], then transfer to stainless steel containers known as "Standard Compacted Waste Containers" (CSD-C) in small format (see Figure 5.1.2).

Figure 5.1.2 Standard compacted waste containers (CSD-C)

The conditioning hypothesis assumes a weight of 400 kilograms of waste per package, made up of a mixture of different types of activated waste in the following proportions: 5% neutronic poison rods, 78% control rods and 17% various metal wastes. The weight of the finished package is around 510 kilograms.

It should be noted that these packages contain no organic materials and are not liable to produce gas (hydrogen) by radiolysis.

5.1.2 Bituminised liquid effluent processing sludge

The second group of B wastes comes from radioactive liquid effluents resulting from the operation of fuel reprocessing facilities. The effluents considered here are generated at various stages of fuel reprocessing and during work carried out on equipment and facilities (decontamination, flushing). These effluents are collected in treatment stations where they are decontaminated by chemical processes before discharge. The residual waste is then recovered in the form of sludge.

In the “STEL” effluent treatment station at Marcoule and STE3 at La Hague, opened in 1966 and 1989 respectively, this sludge has been conditioned by embedding in bitumen, which is then put in steel drums. On the other hand, sludge from effluents produced and chemically treated at La Hague STE2 (Effluent Treatment Station No. 2) from 1966 to 1990 have been gradually stored in tanks and silos at the plant, awaiting conditioning. The planned conditioning method for these sludges is also embedding in a bitumen matrix.

22 An alternative conditioning mode, not dealt with here, is being studied at EDF

23 Large rods and CIS thimbles are cut into sections before being placed in compaction holders

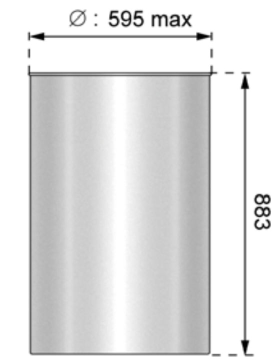
The bitumen-embedded materials consist of dry extract (obtained by drying the sludge), 70/100 direct refinery bitumen, tensioactive additive, and a small quantity of residual water. The average composition of the embedded waste, expressed as percentage weight, is as follows:

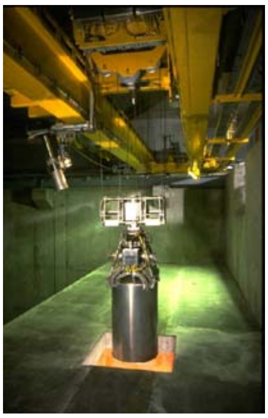
- Dry extract of sludge: 39% of the weight. This dry extract is itself made up of a mixture of insoluble and soluble salts, in proportions that vary according to the source of the sludge;
- Bitumen: 58% of the weight;
- tensioactive agent: 1% of the weight;
- Water: 2% of the weight.

The radioactivity of the waste comes from traces of activation products, fission products and actinides evenly distributed in the mass of the embedded waste. Among these, short- or intermediate-lived radionuclides represent a large part of the radioactivity of the packages. However, they release much less radiation that those described in the previous paragraph: the equivalent ß-γ dose rate in pseudo-contact with the package is one sievert an hour at most. As a result of their relatively low radioactivity, they do not release heat.

The specific nature of these packages derives mainly from the chemical nature of the conditioned waste, which contains a high load of salts and organic matter. Also, radiolysis of the bitumen results in the production of gases, mainly hydrogen, as well as traces of carbon monoxide and dioxide and methane (for hydrogen, 1 to 2 litres[24] per year for STE3 and STEL packages, 9 to 10 litres a year for STE2 packages).

Another difference is that the bitumen-embedded packages do not all have the same geometry. A first group, representing 45% of the inventoried packages, consists of stainless steel primary drums of 238 litres (STE3/STE2) and 245 litres (STEL from October 1996 onwards). These packages are illustrated in Figure 5.1.3.





Fût en inox de 238 litres (STE3/STE2)

Fût en inox de 245 litres (STEL)

Figure 5.1.3 STE3/STE2 and STEL stainless steel drums

The second group of packages (55% of the bituminised packages in the inventory) is made up of 428-litre carbon steel drums, also known as EIP over-drums[25]. These drums (see *Figure 5.1.4*) are used for reconditioning primary non-alloy steel drums produced at STEL between 1966 and October 1996.

[24] At atmospheric pressure

[25] The term EIP designates the multipurpose storage facility on the Marcoule site that will eventually group all the B waste from the site.

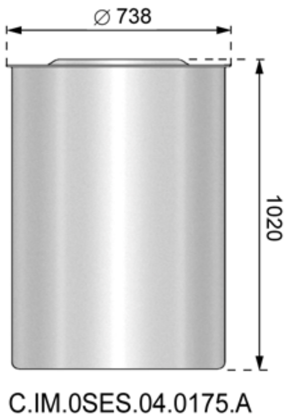


Figure 5.1.4 Stainless steel drum used as over-drum for primary non-alloy steel drums

The study takes the hypothesis that the interstitial gap between the two drums is completely filled with a non-compressible material, such as mortar, in order to limit the long-term mechanical deformation in the repository.

On average, the weight of conditioned waste per package is 220 to 240 kilograms, including roughly 90 kilograms of dry extract and 130 kilograms of bitumen. The average weight of the finished packages is 240 kilograms for STE3/STE2 and STEL packages and 330 kilograms for STEL packages with over-drum..

5.1.3 Cemented or compacted technological waste

A third group of waste (reference package B3) is technological waste resulting from the operation and maintenance of nuclear facilities by COGEMA and the CEA. This consists mainly of various kinds of solid waste (various metals, organic materials), but also includes filtration sludge and evaporation concentrates. This group also includes various waste produced at Marcoule such as graphite, ion-exchanger resins and zeolites. The radiological activity of waste, especially of technological waste, is usually due surface contamination of the waste by fission products and/or activation products and/or actinides.

The conditioning process for this waste depends on its production site and/or its type. The problems posed by these waste packages are therefore linked essentially to the diversity (i) of their chemical content, itself linked to the type of waste and the conditioning matrices used, and (ii) of container shapes and materials. The chemical nature of some packages also makes them liable to produce gases, chiefly hydrogen, by radiolysis. These packages do not generate heat.

The various existing and planned technological waste packages can be grouped into nine subsets, taking into account the types of waste, the conditioning processes and the containers.

A first subset of these packages comprises 1000-litre concrete containers, manufactured by the CEA, containing low-contamination sludge, debris, earth and sands immobilised in a cement-bitumen matrix. Following degradation, some of these concrete containers have been installed in non-alloy steel containers (see Figure 5.1.5).

These two package shapes are considerably larger than those described in earlier paragraphs; on average, they weigh around 3.2 metric tons. There are very few of these packages (90 in total). As a precaution, with no precise data available, the possibility of hydrogen production by radiolysis of the matrix water was considered.

Figure 5.1.5 *Non-alloy steel container used as over-drum for 1000-litre concrete containers*

A second subset of packages, from the COGEMA La Hague site, contains technological or pulverulent waste consisting of a mixture of resins, zeolites, diatoms and graphite, conditioned in a cement matrix inside cylindrical fibre-reinforced concrete containers, known as CBF-C'2 (see Figure 5.1.6).

It should be noted that technological waste was conditioned between 1990 and 1994 in asbestos-cement containers, which were replaced from 1994 onwards by fibre-reinforced concrete containers. Both types of container have the same geometry. The weight of the finished containers varies from 1.5 to 3 metric tons, with an average of around 2.4 metric tons. The technological waste is made up of metal or organic materials. The equivalent ß-γ dose rate in pseudo-contact (5 cm) with the packages is around 0.5 Sv/h, for the packages releasing the most radiation. For these packages, production of hydrogen by radiolysis of the organic matter and cement matrix water is taken into account; this production is from 1 to 2 litres[26] a year.

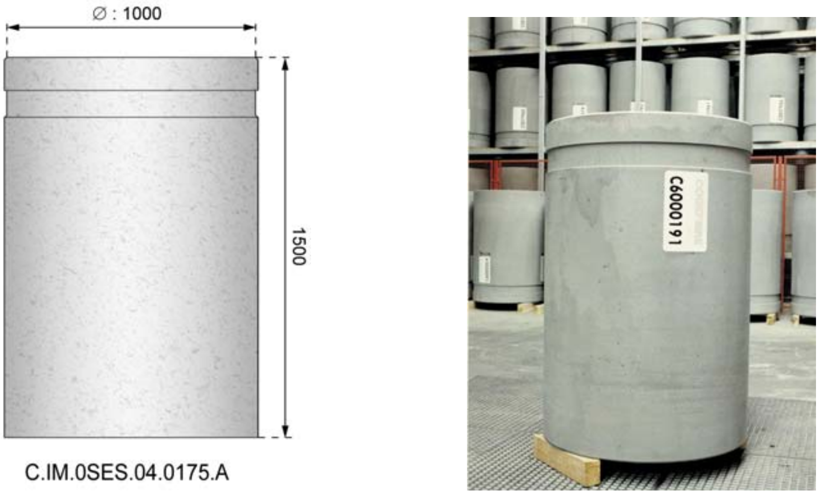


Figure 5.1.6 *Cylindrical fibre concrete containers (CSF-C'2)*

A third subset of packages, manufactured by the CEA, consists of 1800-litre concrete containers filled with various metallic and organic waste conditioned in a cement-bitumen matrix or in mortar. All of these packages were produced between 1964 and 1987: 25% of the concrete containers have been

[26] At atmospheric pressure.

repaired (resurfacing) and following degradation 75% have been installed in non-alloy steel containers (see Figure 5.1.7). These packages are also very large.

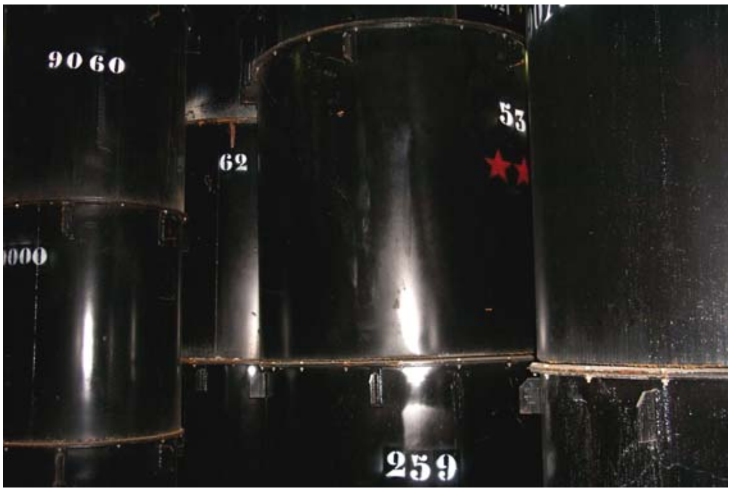


Figure 5.1.7 Non-alloy steel container used as over-drum for 1800-litre concrete containers

The weight of the packages varies from 2.7 metric tons to 6.8 metric tons. The number of these packages is still low (180 in total). As with the previous subset, potential production of hydrogen by radiolysis of the organic matter and matrix water also needs to be considered. It is important to note that these packages do not release radiation; they can therefore be handled with operators present.

A fourth subset of packages, manufactured by the CEA, comprises drums of filtration sludge (still being produced) or of cemented evaporation concentrates (production stopped), conditioned in 500-litre concrete containers (see Figure 5.1.8). As things currently stand, this packaging process is taken as the reference hypothesis for the drums produced at the CEA Valduc centre containing either sludge, either concentrates or a mixture of cemented sludge and concentrates. It should be noted that the use of mortar to immobilise waste drums in concrete containers was stopped 1996.

The weight of the finished packages varies from 770 to 920 metric tons.

As with the waste referred to in Paragraph 5.1.2 (bituminised sludge), these packages differ from the other technological waste packages by the chemical nature of the conditioned waste, especially the presence of salts. Once again, potential production of hydrogen by radiolysis of the matrix water has to be considered. These packages, like those of the previous subset, do not release radiation.

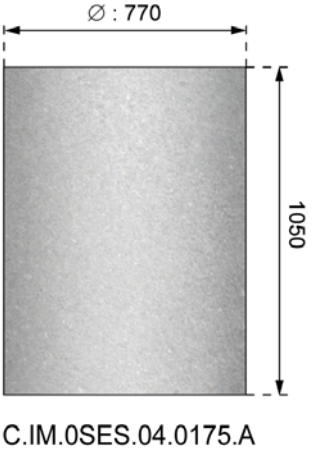


Figure 5.1.8 500-litre concrete container

A fifth subset of packages consists of solid waste and sludge that will be conditioned in the future CEDRA and AGATE facilities of the CEA. The hypothesis currently adopted is conditioning of the waste in CBF-C'2 fibre-reinforced containers. It is also supposed that mortar will be used to immobilise the waste in the containers. As a precaution, the potential hydrogen production by radiolysis of water in the mortar is considered. Once again, these packages do not release radiation.

A sixth subset of packages contains various technological waste, known as alpha waste, mainly contaminated by plutonium during the manufacture of MOX fuels or reprocessing of spent fuels (especially for the conditioning of plutonium). The most representative waste, in terms of nature and flow, is waste from the manufacture of MOX fuels in the MELOX plant (Marcoule). This is composed of organic waste, filters and various predominantly metallic waste from glove boxes. The planned conditioning process is compaction of the waste then transfer to standard compacted waste containers (CSD-C) in the ACC shop of the La Hague plant. The exact mixture of the waste mentioned above in the packages depends on the hypotheses selected. The average weight of the finished packages is 635 kilograms. As with other containers, the presence of organic matter in the packages means that the production of hydrogen by radiolysis of this matter must be considered. These packages do not release radiation.

A seventh subset of packages contains pulverulent waste from the COGEMA plant at Marcoule. The waste comes either from the water filtration systems (ion-exchanger resins, zeolites, diatoms and pool sludge), or from mechanical processing of fuels (graphite). The planned conditioning method is embedding in a cement matrix and transfer to 428-litre stainless steel drums (EIP drums; see Figure 5.1.9).The exact mixture of the waste mentioned above in the drums depends on the hypotheses selected. The average weight of the finished packages is 720 kilograms. Potential production of hydrogen by radiolysis of the water in the matrix concrete must be taken into account. The equivalent ß-γ dose rate in pseudo-contact (5 cm) with the packages is low at roughly 0.05 Sv/h.

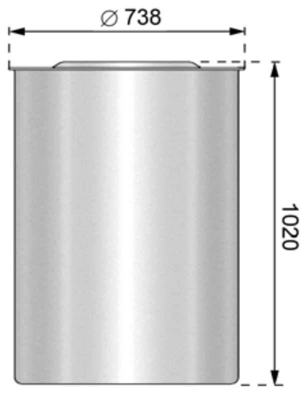


Figure 5.1.9 EIP stainless steel drum

An eighth subset of packages, manufactured by the CEA, contains intermediate-level solid technological waste from shielded cells, mainly contaminated with ß-γ emitters. The waste is conditioned in compacted form in 500-litre steel containers (see Figure 5.1.10).). It should be noted that changes have been made since the packages were first produced, concerning (i) the conditioning matrix used to immobilise the waste in the containers: initially a cement-bitumen matrix, then from 1990 a cementitious material, (ii) the material used for the container: the non-alloy steel used initially was replaced by stainless steel from 1994.

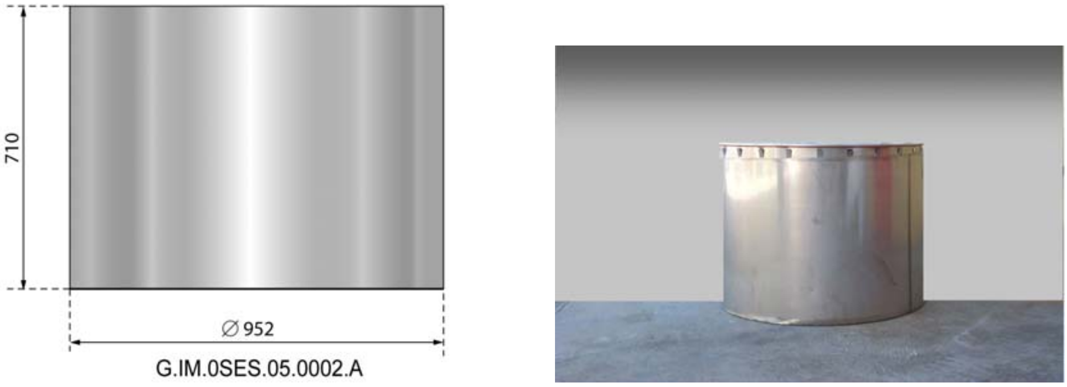


Figure 5.1.10 500-litre stainless steel container

In this subset, the waste consists of various materials, especially metals, organic matter and glassware, mixed in the same package. The average weight of the finished packages is 925 kilograms for those immobilised by a cementitious material and 850 kilograms for a cement-bitumen matrix. Given the chemical content of the packages, potential production of hydrogen by radiolysis of the organic matter and conditioning matrix water also needs to be considered. The equivalent ß-γ dose rate in pseudo-contact (5 cm) with the packages is roughly 0.2 Sv/h.

The ninth and last subset of technological waste packages, manufactured by the CEA, contains solid waste contaminated mainly by α emitters. Depending on its dimensions, the waste is conditioned, in compacted or non-compacted form, in 870-litre non-alloy steel containers (see Figure 5.1.11).

As with the 500-litre steel containers above, two waste conditioning matrices – initially a cement-bitumen matrix, then from 1990 a cementitious material – have been used successively since the packages were first produced. It should be noted that this package subset also includes some packages of cemented concentrates initially conditioned in 700-litre steel drums, then reconditioned in 870-litre containers.

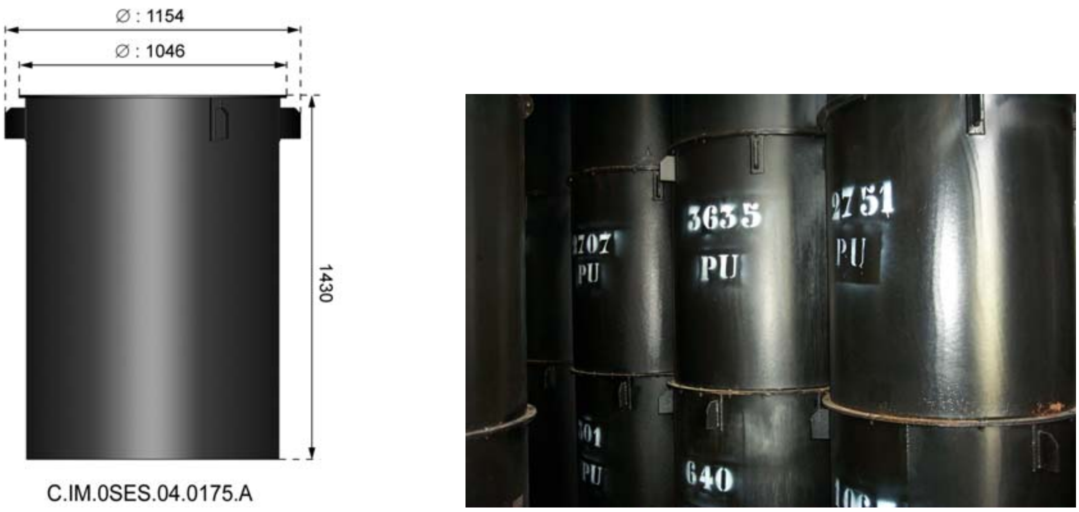


Figure 5.1.11 870-litre non-alloy steel container

The conditioned waste comprises various materials (metals, organic matter, glass, debris, etc.) mixed in the same package. The average weight of the finished packages varies between 1.6 and 2.3 metric tons, depending on the content (compacted or non-compacted, conditioning with over-drums). Given the chemical content of the packages, once again potential production of hydrogen by radiolysis of the organic matter and conditioning matrix water must be considered. These packages do not release radiation.

The main characteristics of the nine subsets of cemented or compacted technological waste packages are summarised in Figure 5.1.2.

Table 5.1.2 *Summary of the main characteristics of subsets of cemented or compacted technological waste packages*

Package subset	**Overall volume of the package (m^3)**	**Container material**	**Embedding or immobilising matrix**	**Presence of metallic materials**	**Presence of organic matter**
1	1	Concrete	Cement-bitumen	None	x
2	1.2	Fibre-reinforced concrete or asbestos cement	Cement	x	x
3	1.8 3.2 or 3.8 with over-drum	Concrete or non-alloy steel	Cement-bitumen or mortar	x	x
4	0.5	Concrete	Cement	None	x
5	1.2	Fibre-reinforced concrete	Mortar	x	x
6	0.18	Stainless steel	None	x	x
7	0.428	Stainless steel	Cement	None	x
8	0.5	Non-alloy or stainless steel	Cement-bitumen or cementitious material	x	x
9	1.22	Non-alloy steel	Cement-bitumen or cementitious material	x	x

5.1.4 Cemented or compacted cladding waste

This type of waste comes from spent fuel reprocessing in the COGEMA plants; it relates to the metallic framework components in the fuel assemblies. This waste is separated from recyclable nuclear materials (uranium, plutonium) and from fission products and minor actinides when reprocessing commences during fuel shearing and dissolution operations.

This waste is commonly known as "hulls and end caps" in pressurised water reactor fuel assemblies. The hulls are the cladding from the fuel rods, recovered in lengths of around three centimetres long, from which the nuclear material has been extracted by being dissolved in acid. The end caps are the parts located at both ends of the fuel assembly.

The cladding waste under consideration here has come from reprocessing operations in the COGEMA plants at La Hague. They include (i) waste produced during previous reprocessing operations of NUGG (Natural uranium graphite gas reactor) and PWR fuels, today stored in silos and pits, and (ii) waste from current and future reprocessing operations of the various types of PWR UOX and MOX fuels, defined in the design scenarios given in chapter 2.

There are several types of materials in cladding waste: magnesium-zirconium and magnesium-manganese alloys for the NUGG fuels; zirconium-tin (zircaloy 4) or zirconium-niobium (M5 alloy) alloys, stainless steels and nickel alloy for the PWR fuels. Following the conditioning hypotheses indicated below, some packages also contain technological waste formed of metal only (non-alloy and stainless steels) or a metallic-organic mixture. The mass of this technological waste represents around ten per cent of the total mass of conditioned waste per package.

Radiological activity is due (i) to activation products from neutron activation of the alloy components and impurities making up the component materials of the assembly structures during their time in the reactor and spread in the mass of these materials, and (ii) contamination by fission products, impurity activation products and actinides found in the oxidised layer covering the internal surface area of the cladding lengths that held the fuel. This contamination is also caused by traces of undissolved substances that may be left after the waste has been rinsed. The radiological contribution from technological waste found in some packages is negligible compared with that of the cladding waste.

Package thermicity is mainly caused by cobalt-60. A decrease in cobalt results in some packages losing their thermicity given the age of the waste. Other packages have a heat rating in the order of around thirty watts when they are produced, but this drops rapidly after a few years, in conjunction with the decrease in cobalt. These changes are illustrated in Figure 5.1.12.

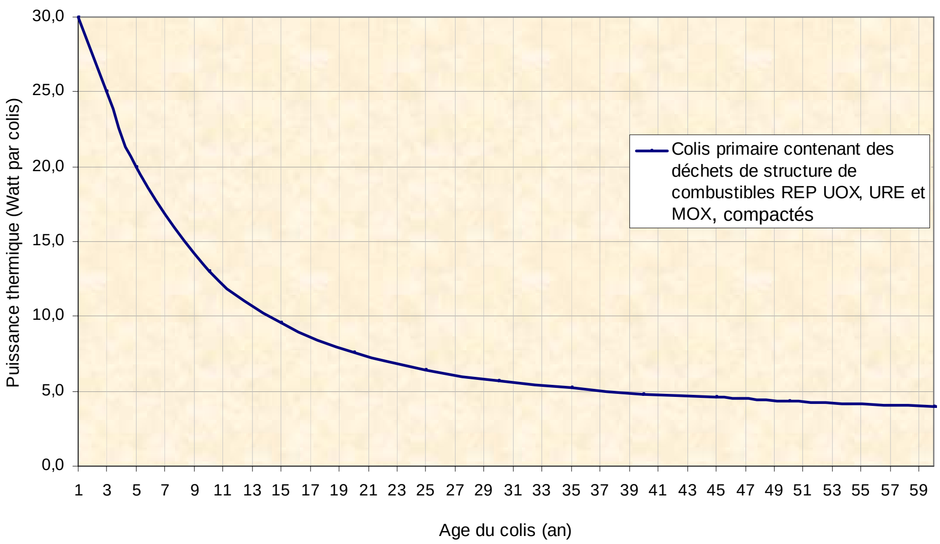


Figure 5.1.12 Changes in residual heat rating in cladding waste primary packages from PWR UOX, URE (enriched recycled uranium) and MOX fuels

- **Cemented cladding waste (reference packages B4)**

The cladding waste from the PWR fuel assemblies is initially cemented into huge stainless steel drums. This process was applied between 1990 and 1995, then replaced in 2002 by waste compacting, in the hull compacting workshop (ACC) at La Hague.

The packages produced between 1990 and 1995 are called drums of cemented hulls and end caps (Figure 5.1.13).

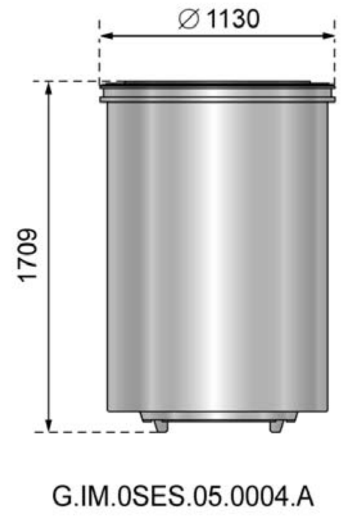


Figure 5.1.13 1800-litre stainless steel drum containing cemented cladding waste

They contain an average of 776 kilograms of cladding waste, made up of 80.7% zirconium-tin (zircaloy 4) alloy, 15.9% stainless steel and 3.4% nickel alloy. The package weight (drum + waste + matrix) is 3.5 tonnes on average. The residual heat rating in the packages from the cobalt-60 is today in the order of around ten watts. It will be about three watts by 2025. The equivalent β-γ dose rate in pseudo-contact with the packages (5 cm) is currently in the order of 4 Sv/h; it will be in the order of 0.5 Sv/h by 2025. Water radiolysis in the cementing matrix produces hydrogen.

- **Compacted cladding waste (reference packages B5)**

As indicated previously, a new method of conditioning cladding waste was introduced on the La Hague site in 2002. This involves compacting waste placed preliminarily in claddings, before being moved to stainless steel containers (CSD-C). The compacting process is applied to cladding waste produced from NUGG and PWR fuels reprocessed previously and now stored in silos and pits, and cladding waste produced from current and future reprocessing operations of fuel unloaded from the PRW reactors. As mentioned above, some packages also contain compacted technological waste from the site's operating and/or maintenance shops. Given the diversity and nature of the waste throughputs in question, distinction is made between four sub-assemblies of compacted cladding waste packages (CSD-C).

The first package sub-assembly contains cladding waste from UOX, enriched recycled uranium and MOX fuel reprocessing mixed with metallic and organic technological waste. The hypotheses adopted are (i) fuel reprocessing on average eight years after unloading from reactors and (ii) average weight of 420 kilograms of conditioned waste per package (including compacting claddings). Based on these hypotheses, the heat rating of the packages, calculated for an envelope radiological inventory of the various waste flows mentioned above, is around 30 watts. Thermal decay of these packages is given in Figure 5.1.12.

Package irradiation level, initially around fifty sieverts per hour (Sv/h) is in the order of 15 Sv/h after a ten-year cooling period. The radiolysis of the organic waste in the packages made up of technological waste produces hydrogen. Note also that radioactive elements such as tritium (with the symbol T or ^{3}H), carbon-14 (^{14}C), chlorine-36 (^{36}Cl), argon-39 (^{39}Ar) and krypton-85 (^{85}Kr) can be released in a gaseous form. The finished package weighs approximately 520 kilograms.

Like the previous packages, t*he second package sub-assembly* contains a mixture of UOX, enriched recycled uranium and MOX cladding waste and technological waste. It differs from the first in the type of technological waste, here formed of metallic materials only. Unlike the first sub-assembly packages, it does not therefore generate hydrogen through radiolysis. However, there is a risk of the waste releasing traces of radioactive gases (^{3}H, ^{14}C, ^{36}Cl, ^{39}Ar and ^{85}Kr). This raises the question of their containment as close to the waste as possible; this issue is dealt with in section 5.3. The other package characteristics (heat rating, equivalent dose rate, weight) are otherwise identical to the first sub-assembly.

The third package sub-assembly only contains cladding waste from PWR fuels reprocessed in the past and now stored. The packages weigh 725 kilograms on average. Given the age of the waste, the packages do not transfer heat. Their irradiation level is around 5 Sv/h; it will be around 1 Sv/h by 2025.

The third package sub-assembly only contains cladding waste from NUGG fuels reprocessed in the past and now stored. The finished packages weigh 350 kilograms on average. Given the age of the waste, the packages do not transfer heat. Their irradiation level is also lower than for the previous sub-assemblies, namely around 0.4 Sv/h.

Table 5.1.3 summarises the principal characteristics of the four sub-assemblies of compacted cladding waste packages.

Table 5.1.3 *Summary of principle characteristics of sub-assemblies of compacted cladding waste packages*

Package sub-assembly	**Cladding waste materials**	**Presence of technological waste**	**Presence of organic matter**	**Production of gas by radiolysis (H_2)**	**Thermicity, irradiation level when packages produced**
1	Zirconium-tin or zirconium-niobium alloys, stainless steels, nickel alloy	x	x	x	Packages transfer little heat, extremely irradiating
2	Ditto sub-assembly 1	x	None	None	Packages transfer little heat, extremely irradiating
3	Zirconium-tin alloy, stainless steels, nickel alloy	None	None	None	Packages do not transfer heat, moderately irradiating
4	Magnesium-zirconium or magnesium-manganese alloys	None	None	None	Packages do not transfer heat, only slightly irradiating

5.1.5 Cladding and technological waste placed in drums

This groups waste (reference package B6) produced on the COGEMA Marcoule site and currently stored, excluding bitumen waste and cemented pulverulent waste described above. It includes (i) operating waste from the Marcoule vitrification shop, (ii) cladding waste from fuels reprocessed in the UP1 plant and (iii) operating and maintenance technological waste from the Marcoule site facilities.

The waste is placed in stainless steel drums. Note that the modalities for limiting the residual voids inside the primary packages have yet to be defined.

A first package sub-assembly contains technological waste from operations in the Marcoule vitrification shop (AVM). The waste formed of equipment, tools and various steel parts is deposited in a stainless steel container of similar geometry to the AVM vitrified waste containers. The packages weigh 160 kilograms on average and may weigh as much as 320 kilograms (excluding waste immobilisation material). Radiological activity corresponds to contamination of the waste surface area. These packages do not transfer heat and have an irradiation level of around 0.05 Sv/h.

A second package sub-assembly contains fuel cladding waste. The waste is temporarily conditioned in stainless steel drums known as EIP drums (from the French acronym for multi-purpose storage facility). The packages hold aluminium and stainless steel cladding waste or magnesium-alloy cladding waste. They weigh less than 300 kilograms on average (excluding waste immobilisation material). Packages containing aluminium and steel cladding waste have a heat rating in the order of 10 watts, mainly attributable to cobalt-60; this will be 0.5 watts at most by 2025. The package irradiation level is 25 Sv/h, becoming 2 Sv/h by 2025.

The packages containing the magnesium alloy cladding waste do not transfer heat. Their irradiation level, attributable to two isotopes, barium-137m (^{137m}Ba) and europium-154 (^{154}Eu), is 3 Sv/h, becoming around 2 Sv/h by 2025.

A third package sub-assembly contains technological waste made up of a mixture of metallic materials and organic matter or metallic materials alone. The temporary conditioning method is also to place the waste in EIP drums.

The packages containing metallic and organic waste weigh 90 kilograms on average (excluding waste immobilisation material). They do not transfer heat and are not irradiating. Hydrogen release from radiolysis of the organic matter should be taken into account.

The packages containing the metallic technological waste do not transfer heat nor produce gas. Their irradiation level is around 0.05 Sv/h, becoming around 0.02 Sv/h by 2025. The finished packages weigh 240 kilograms on average, excluding waste immobilisation material.

Table 5.1.4 summarises the principal characteristics of the three sub-assemblies of the cladding and technological waste packages..

Table 5.1.4 *Summary of the characteristics of the cladding and technological waste placed in drums*

Package sub-assembly	Overall package volume (m^3)	Presence of metallic waste	Presence of organic waste	Production of gas by radiolysis (H_2)	Thermicity, irradiation level when packages produced
1	0.175	x	None	None	Packages do not transfer heat, only slightly irradiating
2	0.428	x	None	None	Nil or average thermicity, average or high irradiation level depending on packages
3	0.428	x	x	x	Packages do not transfer heat and are not irradiating

5.1.6 Sources

This waste groups PWR source rods and sealed sources for industrial use.

Source rods are operating waste from PWR reactors, similar to the variety of active metallic waste described in paragraph 5.1.1. Forming part of the rods contained in the primary and secondary source clusters, they are used to raise the flow level to a threshold that may be detected by neutron counters during reactor start-up. Primary source rods containing a californium capsule are unloaded at the end of the first cycle, whilst secondary source rods, made up of an antimony-beryllium mixture, go through several irradiation cycles before being scrapped. The primary source clusters unloaded from 900 MW reactors are reprocessed to recover the californium capsules and are therefore not considered waste (and are therefore not included in the inventory). The total weight of waste for conditioning is less than two tonnes.

The conditioning hypothesis retained for the study is, like for the other PWR activated waste, shearing then compacting of the source rods before being placed in a CSD-C container. Note that the source rod conditioning will produce a maximum of four CSD-C.

The sealed sources for industrial use contain radioactive material with very different properties, activities and periods. Several thousand sources were conditioned in concrete containers between 1972 and 1984, which were then reconditioned into metallic containers. The packages known as "source blocks" are currently stored on the CEA site at Cadarache (see Figure 5.1.14). These are huge packages weighing between 6.0 and 9.2 tonnes.

Figure 5.1.14 Source blocks

Several thousand other sealed sources are also stored today at various facilities. They cover a very wide range of radioactive isotopes, activities and varying periods. All sources for a period higher than or equal to that of cesium-137 (equal to thirty years) have been adopted for consideration in the study, consistent with the waste accepted for surface disposal at the Aube facility. The conditioning hypothesis envisaged at the moment is to cement the sources into EIP drums.

5.1.7 Radium- and americium-bearing waste

This set (reference packages B8) groups various types of waste including radium-bearing lead sulphate, items for medical use and lightning rods. Taking this waste into account in the HLW/ILW-LL inventory remains exploratory, however. *Radium-bearing lead sulphates* come from uranium ore processing in the Bouchet plant. The waste is placed initially in metallic drums that have been reconditioned successively for storage purposes. The hypothesis adopted for the studies is the recovery of primary radium-bearing lead sulphate drums for conditioning in EIP drums. Note that the modalities for limiting the residual voids inside the primary packages have yet to be defined.

Items for medical use are needles and very small metallic tubes, each containing a few milligrams of radium. The radium is incorporated in a solid, insoluble but pulverulent chemical form (sulphate or chloride). The history of the radium industry shows that about a hundred grams of radium have been extracted, including around fifty grams used in the manufacture of items for medical use. Note that the items for medical use (a total of some 5,000) can be conditioned in a single EIP drum.

As a precaution, consideration is also given to lightning rods that contain radium or americium. Already used for a few radium lightning rods, compacting has been adopted as the conditioning solution, followed by cementing the lightning rod heads in 870-litre, non-alloy steel containers. The packages contain around 200 lightning rod heads on average, both radium and americium, and have an activity in the order of 10 gigabecquerels (GBq). They weight 2 tonnes on average..

5.1.8 Production scenarios

The four production scenarios for B waste presented in chapter 2.2.2 are converted into number and volume of primary reference packages and illustrated in Table 5.1.5.

Table 5.1.5 Number and volume of B waste primary reference packages

Reference packages				**Scenario S1a**		**Scenario S1b**		**Scenario S1c**		**Scenario S2**	
Lev.1	**Lev.2**	**Lev.3**	**Production sites[27]**	**Number**	**Volume (m^3)**	**Number**	**Volume (m^3)**	**Number**	**Volume (m^3)**	**Number**	**Volume (m^3)**
B1			EDF								
Total for B1 "Activated waste"				2560	470	2560	470	2560	470	2560	470
B2											
	B2.1		COGEMA (La Hague, Marcoule)	46930	11210	46930	11210	46930	11210	46930	11210
	B2.2		COGEMA (Marcoule)	58060	24850	58060	24850	58060	24850	58060	24850
Total for B2 "Bituminised waste"				104990	36060	104990	36060	104990	36060	104990	36060
B3											
	B3.1	Total for B3.1		8960	11030	8960	11030	8960	11030	6710	8370
		B3.1.1	CEA	90	90	90	90	90	90	90	90
		B3.10.2	COGEMA (La Hague)	8690	10250	8690	10250	8690	10250	6440	7590
		B3.10.3	CEA	180	690	180	690	180	690	180	690
	B3.2	Total for B3.2		6990	4290	6990	4290	6990	4290	6990	4290
		B3.2.1	CEA	5730	2800	5730	2800	5730	2800	5730	2800
		B3.20.2	CEA	1260	1490	1260	1490	1260	1490	1260	1490
	B3.3	Total for B3.3		16990	11940	16990	11940	16990	11940	16690	11880
		B3.3.1	COGEMA (La Hague)	1200	220	1200	220	1200	220	900	160
		B3.30.2	COGEMA (Marcoule)	7990	3420	7990	3420	7990	3420	7990	3420
		B3.30.3	CEA	1700	850	1700	850	1700	850	1700	850
		B3.3.4	CEA	6100	7450	6100	7450	6100	7450	6100	7450
Total for B3 "Technological and miscellaneous cemented or compacted waste"				32940	27260	32940	27260	32940	27260	30390	24540
B4			COGEMA (La Hague)								
Total for B4 "Cemented cladding waste"				1520	2730	1520	2730	1520	2730	1520	2730
B5											
	B5.1		COGEMA (La Hague)	7940	1450	7400	1350	7400	1350	2140	390
	B5.2			31760	5810	29600	5420	29600	5420	8560	1570

[27] Or storage sites

Reference packages				Scenario S1a		Scenario S1b		Scenario S1c		Scenario S2	
Lev. 1	Lev. 2	Lev. 3	Production sites[27]	Number	Volume (m^3)	Number	Volume (m^3)	Number	Volume (m^3)	Number	Volume (m^3)
	B5.3			2500	460	2500	460	2500	460	2500	460
	B5.4			400	70	400	70	400	70	400	70
Total for B5 "Cladding waste with or without technological, compacted waste"				42600	7790	39900	7300	39900	7300	13600	2490
B6											
	B6.1		COGEMA (Marcoule)	180	30	180	30	180	30	180	30
	B6.2			930	400	930	400	930	400	930	400
	B6.3			7550	3230	7550	3230	7550	3230	7550	3230
	B6.4			1200	510	1200	510	1200	510	1200	510
	B6.5			950	410	950	410	950	410	950	410
Total for B6 "Cladding waste and technological waste in drums"				10810	4580	10810	4580	10810	4580	10810	4580
B7											
	B7.1		EDF, CEA, Andra	41	155	41	155	41	155	41	155
	B7.2			4	0.7	4	0.7	4	0.7	4	0.7
	B7.3			3000	1285	3000	1285	3000	1285	3000	1285
Total for B7 "Sources"				3045	1440	3045	1440	3045	1440	3045	1440
B8											
	B8.1		CEA, Andra	1100	470	1100	470	1100	470	1100	470
	B8.2			250	305	250	305	250	305	250	305
	B8.3			1	0.4	1	0.4	1	0,4	1	0,4
Total for B8 "Radium- and americium-bearing waste"				1350	775	1350	775	1350	775	1350	775

5.1.9 Throughput hypotheses

Throughput hypotheses for the packages received on a repository site have been drawn up in line with the total quantities of inventoried packages so that the operating methods for a repository along with its drifts and access shafts can be studied.

In the case of B waste, the annual reference primary package acceptance figure has been set at 5,000. It would roughly correspond with a forty-year accumulated disposal period for the counted packages. All or the majority of the other reference packages will already have been produced by the 2020 dateline, with the exception of certain packages produced at a later date, particularly reference package B1.

5.2 Safety options for the design of a B waste repository

The general principles underlying repository design in a granite medium have been described in section 3.3. They refer to long-term safety functions, of waste disposal and management of the installations with a view to reversibility.

A number of measures relating to the repository architecture, its dimensioning, the choice of materials used for the openings and the disposal process fulfil the following three principles for long-term safety:

- Making the most of the favourable properties of the granite medium;
- designing engineered components that are complementary and redundant to the granite medium;
- limiting disturbances to the granite by the repository.

Application of these principles to the B waste repository takes account of the characteristics of the inventory presented earlier primarily in terms of:

- volume and diversity;
- radiological content (this is mainly expressed by the irradiating nature of most of the waste packages, the low thermal rating of certain packages (B1, B4 and B5), traces of fissile isotopes and the potential release of radioactive elements in gaseous form);
- the physical and chemical nature of the waste and its conditioning.

Furthermore the technical options proposed at the current generic stage of the studies also take into account the main requirements for operational safety and the reversibility of the disposal process.

5.2.1 Making the most of the favourable properties of the granite medium

Making the most of the very low permeability properties of the granite, its radionuclide retention properties and its mechanical strength calls for installing the disposal cells in granite blocks away from water-conducting faults.

In the case of B waste, this architectural arrangement involves making allowance for the volume and diversity of B waste.

With respect to the rather large volume of B waste, among the various architectures possibilities, the one achieving repository compactness is preferred. It will limit the number of cells and the volume of excavated rock in turn. It will also limit the number of very low permeability blocks of granite needing characterisation.

However the diversity of waste implies that only packages that are mutually compatible, for instance through the presence of organic matter, can be placed in a single cell. By compartmentalising the repository, the quantity of waste and radionuclides that would be affected in the event of a failure or intrusion is also reduced.

- **The reference cell: disposal tunnels**

Various architectures have been envisaged for the B waste cells. The desire for compactness has led to studying a very tall (about thirty metres) vertical "silo-cylindrical" type concept.

Silos of this type have been adopted for the short-lived waste repositories in Finland (Olkiluoto and Loviisa) and Sweden (Försmark).

However the option creates difficulties such as the risk of packages dropping, the stability of the stacks in the long term and the need for significant head room on top of the silo to accommodate the handling equipment (overhead crane) in the case of intermediate-level long-lived waste.

This explains why the "vertical" silo cell has not been adopted as the reference option. Horizontal architectures of cells in tunnels have been chosen. The option has also been studied by Japan and Sweden for long-lived waste equivalent to B waste [35] (Figure 5.2.1).

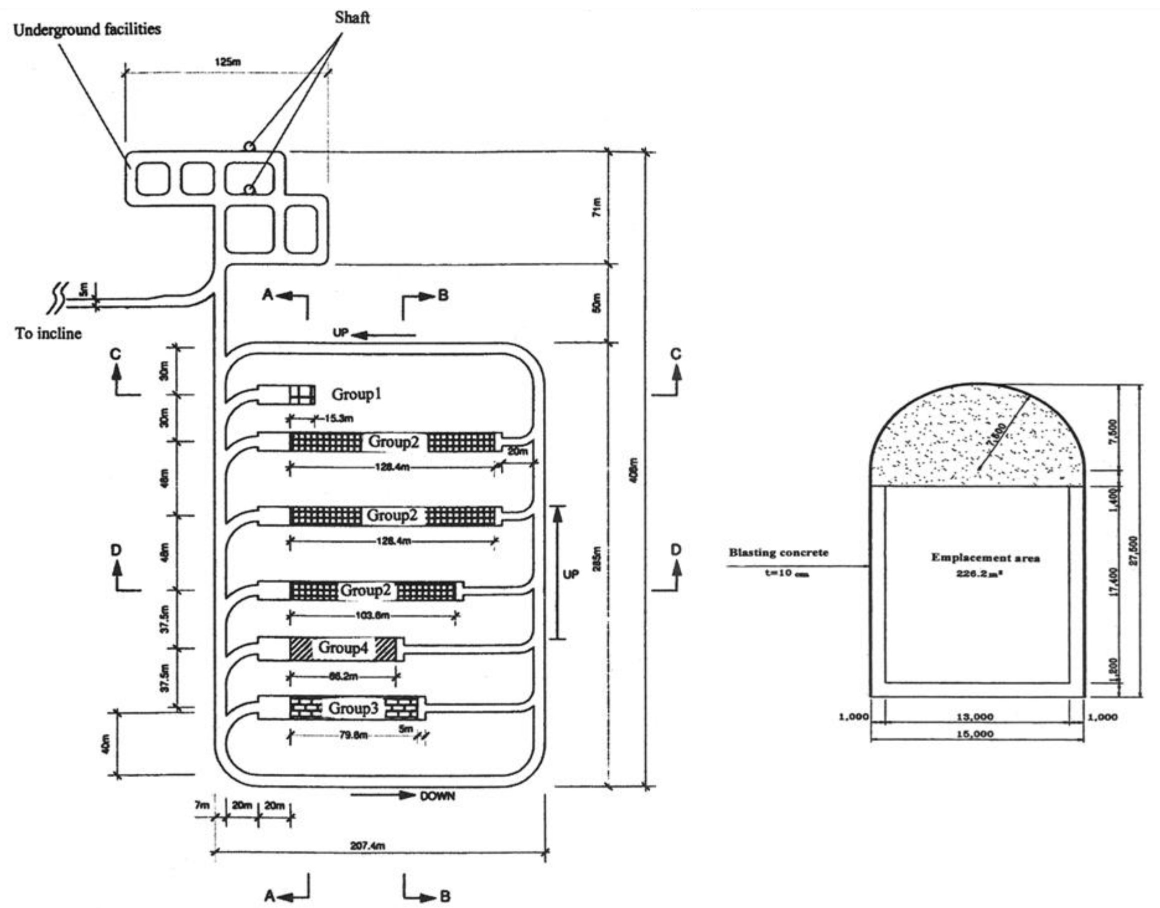


Figure 5.2.1 *Japan: general architecture of a deep TRU waste in crystalline rock and a bituminised or low-level waste storage tunnel*

The horizontal package handling option is adopted as the reference in the studies out of the same concern for compactness and reducing the clearances left over after disposal in the tunnels. By opting for this choice the overall volume in the cell can be reduced, avoiding the need for a clearance (that the installation of an overhead crane requires in the upper part of the tunnel) that cannot be used for disposal.

Given the mechanical properties of the granite, the dimensions of the tunnels can be quite large, which contributes to the objective for repository compactness [35].

Dimensioning for all B waste is essentially governed by the operational safety requirements in relation with package emplacement and package drops prevention (cf. § 5.3). In the case of slightly exothermic waste (B1 and B5), dimensioning takes also into account temperature criteria for the long-term behaviour of the concrete of the packages and the radionuclide behaviour inside the cells (cf. § 5.2.2).

Thus the cell architectures studied are disposal tunnels about ten metres high, and ten to twenty metres across (cf. § 5.4). The tunnel lengths can be adjusted to both the waste characteristics and inventory and the granite fracturing. They vary from 70-200 metres.

- **Installing disposal tunnels in "blocks" of very low permeability granite**

Installing disposal tunnels in blocks of very low permeability granite implies that these blocks have been delimited and characterised to ensure they meet the safety requirements. This characterisation is accomplished on an "as-going" basis during repository construction.

Module architecture will thus be governed by the geology and particularly the geometry of the "blocks" and faults surrounding them (Figure 5.2.2). The properties of the rock dictate the clearance length (some tens of metres) to be allowed for between these faults and the disposal tunnels.

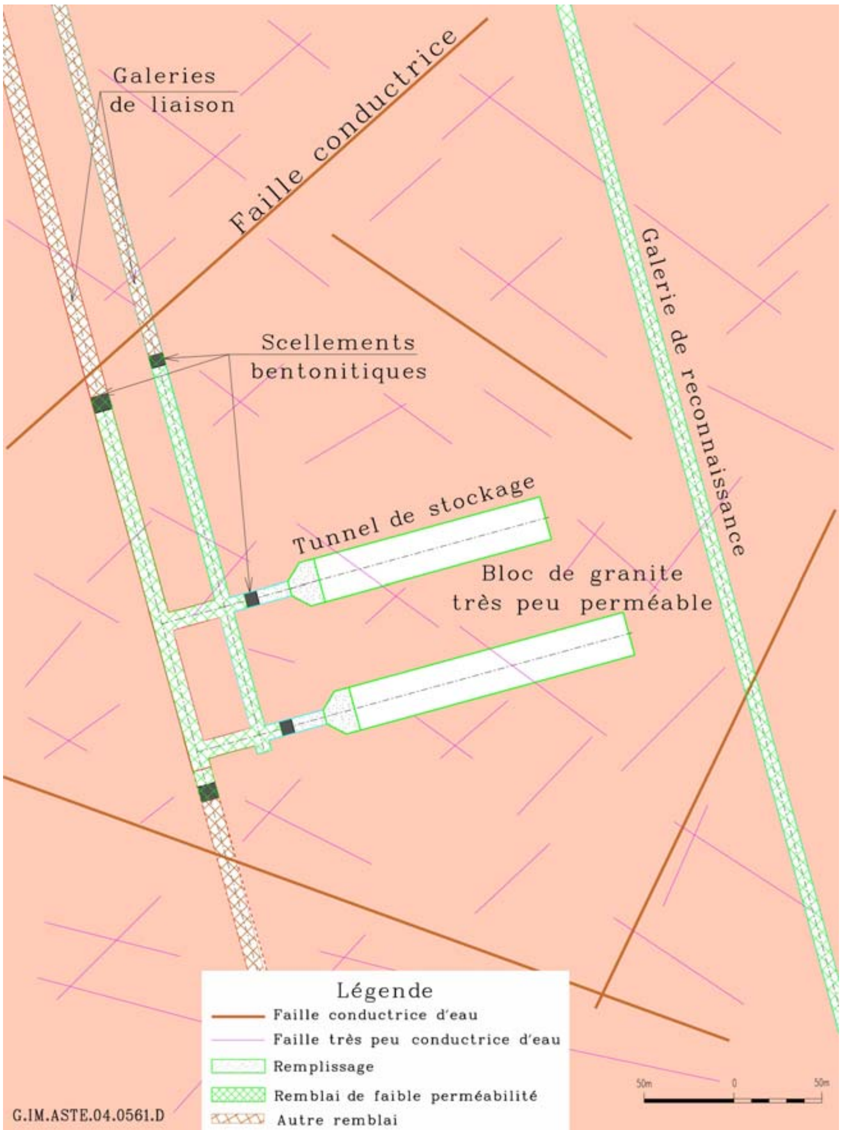


Figure 5.2.2 Principle of disposal tunnel installation in a granite block

Furthermore, the architecture inside a block is also governed by the splitting up induced by the nature of the waste. Sufficient clearance is maintained between the disposal tunnels holding different kinds of waste in relation to their organic matter content or their sensitivity to temperature (bituminised waste).

Thus a block can accommodate one to several disposal tunnels (Figure 5.5.1) on one or more levels.

- **"Dead-end"tunnels**

"Dead-end" cell architecture, with one end bound by the granite rock which means without an exit to the disposal drifts, complements the installation of tunnels in very low permeability granite blocks. It tends to reduce water renewal in the disposal cells, even in the event of seal failure. This principle is applied to all the disposal tunnels.

5.2.2 Designing engineered components that are complementary and redundant to the granite medium

Making the most of the favourable properties of the granite mainly entails installing the repository modules in very low permeability granite rock. However the access drifts leading to repository modules in granite are likely to intersect water-conducting faults. Thus the various engineered module components (disposal packages, backfill, seals) must complement the granite by acting as a barrier firstly to the circulation of advective water in the modules, and secondly to radionuclide release and migration.

These provisions also provide redundancy with the granite in the event of failures or external attacks.

5.2.2.1 Multiple sealing of the repository modules

Very low permeability seals are inserted in each access drift to protect the repository modules from water flows that could originate from the faults intersected by the access drifts. Their precise location is governed by the geometry of the "blocks" of granite accommodating the modules and their surrounding faults (Figure 5.2.2).

Low permeability backfill is placed in the drifts redundant to these seals, as a measure for dealing with seal failure.

The repository tunnels are also closed by very low permeability "plugs". These seals isolate the tunnels hydraulically from water flows in the drifts and thus form barriers that limit radionuclide migration from the waste to the access drifts and into the environment to diffusion.

To ensure that seals remain effective in the long term, their *dimensioning* take into account the potential disturbance to their properties by the alkaline waters that would result from the concrete alteration of the waste package disposed in the tunnels.

5.2.2.2 Disposal packages in concrete

The installation of repository tunnels in the very low permeability rock and the emplacement of seals protect the waste packages from flows of water once the repository has closed, which slows down the kinetics of package deterioration and thus radionuclide release. Furthermore, underground the granite water - an element conducive to radionuclide immobilisation - is reducing.

The choice of disposal packages in concrete also has a hand in limiting radionuclide release and immobilising them by providing an alkaline environment (pH level of 7-12.5) inside the repository tunnels [36].

The precise formulation of the concrete will be determined in line with the exact characteristics of the groundwater in the granite hosting the repository: concrete formulation possibilities mean that adjustments can be made in the French geological context as dictated by the salinity of the granite water and their pH level [37].

Concrete disposal packages need to be given additional confinement performance for the most radioactive B waste (B1, B5), for the activation products and long-lived fission products (^{94}Nb, ^{93}Zr, ^{135}Cs, etc.) in the form of low permeability. Furthermore the packages need to last at least several millennia.

Certain types of waste produce hydrogen. The hydrogen is generated by the radiolysis of materials, primarily the organic matter contained in certain technological waste or embedding materials such as bitumen. Thus the disposal package options studied for the waste in question provide for permeability to this gas.

5.2.2.3 Thermal criterion of design

The concern for compactness and cell temperature criteria must be compatible from a dimensional angle.

There is a dual requirement for limiting temperature:

- to control alteration of the disposal package concrete;
- to control radionuclide behaviour inside the cell.

A temperature threshold that affects the chemical change processes of the material (carbonation, hydrolysis) emerges from studying the physical and chemical properties of the cementitious materials being considered. As long as temperature remains below this threshold, the physical and chemical properties, especially permeability, change in a reversible manner or remain constant. Above this threshold, some cement paste hydrates become unstable: major mineralogical modification would therefore lead to irreversible changes in concrete physical and chemical properties. Additionally there are uncertainties surrounding the chemical properties at these temperatures. A maximum temperature of 70°C has been adopted.

The temperature is limited to below 50-60°C on the basis of the available data, and below 80°C with a controlled margin of uncertainty, to ensure that the right conditions are in place to control radionuclide solubility and sorption behaviour.

Thus in terms of dimensioning, 70°C is the temperature limitation criterion in the disposal tunnels.

Furthermore the temperature must be below 30°C for bituminised waste (type B2), to maintain the rheological properties of the embedding bitumen.

5.2.3 Limiting disturbances to the granite by the repository

Increasing the depth generally leads to reducing the hydraulic gradients in the structures, which is likely to enhance the robustness of the repository in the event of seal failure. By limiting the hydro-geological and hydro-geochemical disturbances, it is a potential parameter for adapting the architecture of a B waste repository to a granite site.

The other thermal or chemical disturbances likely to affect the properties of a B waste repository are managed either by dimensioning the structures and the various disposal tunnel components or by managing the disposal process.

5.2.3.1 Limiting chemical disturbances

Excavating and operating a disposal tunnel and its long term evolution induce chemical disturbances at the structure walls. An initial disturbance is caused by introducing air into the tunnel during the operating phase. Once the repository is closed and water returns to the tunnels, gradual alteration of the concretes leads to alkaline disturbance at the rock wall.

Installing the tunnels in "blocks" of very low permeability and slightly fissured granite ensures that the oxidising or alkaline disturbances essentially affect the granite rock itself rather than the granite fracture network.

Thus given the buffering capacities of the rock [38], these disturbances are unlikely to significantly affect neither the long-term prediction of tunnel evolution nor the radionuclide transfers in the surrounding granite massif and its fractures.

5.2.3.2 Managing the mechanical stability of the disposal tunnels over the long term

Dimensioning manages the mechanical stability of the disposal tunnels over the long term by allowing for the state of the natural underground stress field [39]. The excavation method is chosen to limit the creation of a fissuring zone around the disposal tunnels (a "damaged" zone).

5.3 Disposal packages in concrete

This section specifies the design options adopted for disposal packages of B waste and succinctly describes the associated manufacturing techniques.

The options retained for packing in concrete are the same as those considered for the clay medium. They have been studied in conjunction with the French Atomic Energy Commission (CEA) [40]. This is because the CEA has a comparable issue to deal with for the study of a long-term storage facility.

5.3.1 Selected design principles

The simplification offered by a disposal package of standardised dimensions containing several primary packages enhances the reliability of disposal operations and potential package retrieval in view of the diversity of the shape and size of primary packages.

Grouping several primary packages together reduces the throughput of handled packages in the underground installations. Standardisation of the packages reduces the physical diversity of the handled objects and simplifies their arrangement in the disposal cell.

Compact packing solutions, with no excess thickness to attenuate radiation from the waste have been adopted to limit the number of disposal cells. The B waste repository zone is thus made up of a low number of high-capacity disposal chambers, that each holds several layers of disposal packages in stacks (several hundred to a few thousand packages per chamber). These chambers release radiation. They must therefore be automatically or remotely operated, which is additional justification for simplifying operating processes.

- **Stackable disposal packages**

Each disposal package in the cells must withstand the mechanical load of the masses of packages stacked on top of it. The mechanical dimensioning of the package must also allow for a potential vertical “misalignment” that could lead to uneven distribution of efforts. Furthermore, the behaviour of the disposal package must be assessed in accidental situations such as being dropped during emplacement in the cell. Consequently the deformations induced on primary packages must be kept sufficiently small in order to limit radioactive material dispersion.

- **Criticality**

Some B waste (particularly reference packages B3, B4, B5 and B6) comes from spent fuel reprocessing and therefore contain traces of fissile materials. The repository design must ensure sub-criticality in all circumstances. The B5 reference packages therefore seem to be the most constraining with respect to their potential fissile material content. Package design is linked to that of the cell, especially in terms of spacing between packages to maintain sub-criticality.

- **Management of the gases produced by the waste**

The organic matter contained in some waste, the embedding bitumen of reference packages B2 (§ 5.1) and the water used to make up the mortars and concrete of many primary packages, are subject to the phenomenon of radiolysis. This phenomenon takes the form of slow production of gas, mainly hydrogen. Reference packages B2, B3, B4, B5.1, B6.4, B7.1, B8.1 and B8.2 all produce gas.

The hydrogen produced must be prevented from building up inside the disposal package. Accordingly the disposal package must enable the hydrogen produced by the particular waste to be expelled.

Moreover, some types of waste can release traces of radioactive gas such as tritium (^{3}H), krypton-85 (^{85}Kr), chlorine-36 (^{36}Cl), radon 222 (^{222}Rn), iodine-129 (^{129}I) argon 39 (^{39}Ar) and possibly carbon-14 (^{14}C). The reference packages primarily concerned are B1 and B5 (§ 4.1). Dissemination should be prevented as far as possible.

Once there is no risk of the package producing hydrogen by radiolysis (like B5.2 reference packages especially), these gases should be confined as close as possible to the source. Hence some primary containers such as the CSD-Cs (B1 and B5.2) may be made gas-tight prior to being placed in a disposal package, by welding a plug over the vent hole[28].

- **Long-term control of chemical interactions**

In the longer term the chemical interactions between the constituent materials of the over-packs and the waste itself must be controlled.

In B2 reference packages, the waste in the form of salts is closely mixed with a bitumen matrix. Hence the radionuclides they contained would be released concomitantly with the take up of water by the embedding bitumen, salt solubilisation and salt diffusion in the permeable zone. A pH level of 7-12.5 is sought to extend the embedding bitumen durability [36].

Some of the radionuclides present result from the activation of the metals that make up the waste for other reference packages B1, B4, B5 B6 and B7. These activation products are found physically inside the metal; they are released from the waste concomitantly with the corrosion of the metal. Basic pH could slow down this corrosion [36].

- **Temperature**

Certain types of B waste present low heat release, that is basically due to activation products. The thermal rating of reference packages B4 and B6 at the time of emplacement is less than 5 watts per primary package (this could be very much less, particularly for some of the B3 and B2 packages). B1 and B5 reference packages are more exothermic. The rating at the time of emplacement can vary from 5-15 watts per primary package. Thus it should be verified that the repository compactness provided by disposal package design combined with their layout in the cells does not lead to excessive temperature, in relation to the ability to describe the alteration of the materials (§ 5.4.2.3).

Accordingly in the case of B2 type waste, confinement capacity involves maintaining the mechanical integrity of the embedding bitumen. Now the rheological characteristics of the embedding bitumen are only maintained if its temperature remains below 30°C. However this criterion has actually little influence on disposal package design as these packages release very little heat. On the other hand, this criterion must be fulfilled by the repository zone architecture and the temperature evolution in the host medium at disposal cell depth.

28 This may be done as the package leaves the storage facility or when entering the repository surface installations. It is compatible with the low level of gas expected to be produced in these packages (no organic matter).

- **Overpack materials with durable retention capacities**

For certain types of B waste, the disposal package needs to retain the activation products or long-lived fission products they contain (niobium-94 [^{94}Nb], zirconium-93 [^{93}Zr], caesium-135 [^{135}Cs], etc.). This long term function (at the scale of 10 000 years) applies to reference packages B5.2 (hulls and end caps with no organic matter) and B1 (activated waste), that represent a major proportion of the B waste for the above-mentioned radionuclides. As these primary packages do not contain any organic matter they do not need gas-permeable disposal containers (§ 5.1).

5.3.2 Description and dimensioning

A standardised parallelepiped concrete container is the adopted option. It will hold up to 4 primary packages per disposal package while weighing less than roughly 25 tonnes. Package retrieval can be achieved from this container using the same handling equipment as for its emplacement.

In the case of primary packages that release hydrogen through radiolysis, making vents in the over-pack will avoid the risk of excessive concentration inside the container. As long as the concrete is in a dry atmosphere, its porosity would also allow sufficient hydrogen dissipation.

The retention properties of concrete provide a delayed radionuclide migration from those packages which must achieve confinement of their activation products or long-lived fission products.

Most of the disposal packages (over 80%) are of standard length and height between 1.5 and 2.1 metres. They hold four primary packages.

The other disposal packages (less than 20%) are larger. They are between 2.5 and 2.9 metres long and 1.7 and 2.4 metres high. These disposal packages contain one to four primary packages depending on their dimensions.

- **Standard disposal packages**

One option for manufacturing disposal containers is to use two prefabricated concrete elements: a body and a lid.

The body has internal partitions forming the housings that suit the primary package shape. Clearance between the internal diameter of each housing and the external diameter of the primary packages enables the latter to be emplaced.

The handling principle for emplacing the disposal containers in the cells optimises the cell utilization ratio. It is based on gripping from below using a machine of the "fork-lift truck" type.

Prefabricated high-performance reinforced fibre concrete (HPC) is used for the body and lid. The concrete used for the body offers 75 Mpa compression strength (HPC75) and for the lid 60 MPa (HPC60). The lid is made up of a plate placed on the inner rim of the body side walls, with five openings (one central and four at the sides) for the tie rods fixed to the body. Both the side walls and the lid are at least 110 mm thick.

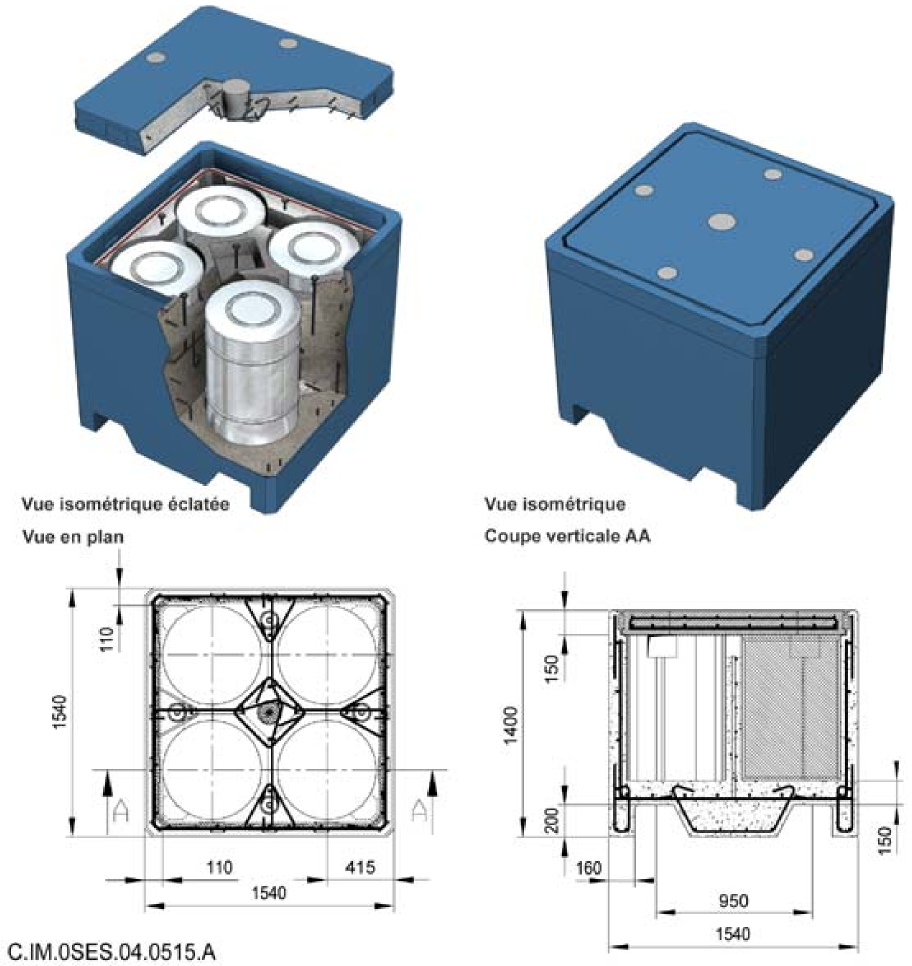


Figure 5.3.1 Example of a standard disposal package

The addition of fibres (25-30 kg/m^3) prevents heterogeneous fissuring of the concrete as it shrinks during curing. Shrinkage can thus only cause evenly spread heterogeneous micro-fissures and thus not weaken the object. To limit risks of corrosion and improve the mechanical durability of the container, stainless steel has been adopted for the reinforcements and fibres.

- **The disposal package with greater retention capacity**

The disposal package envisaged for B1 and B5.2 type waste has durable confinement capacity. It takes stock of the concrete's retention capacity. It comprises a prefabricated body with housings adjusted to the primary package dimensions, as the standard package. It differs from the standard package in that the lid is made up of four independent elements poured directly into the upper part of the body housings. One single concrete formulation is used for all package components (HPC 90 MPa minimum) and package fabrication does not include reinforcements, which entails increasing the stainless steel fibre content (between 55 kg/m^3 and 90 kg/m^3).

The proposed concrete formulation aims at high performance levels (permeability of less than 2.10^{-13} m/s, diffusion coefficient of 2.10^{-13} m²/s and 10% porosity at the most). The thickness is set at 150 mm.

A prefabricated conical plug (conical shape aiming at plugging the functional clearance) is placed over the primary package of each housing prior to pouring the lids. The use of prefabricated plugs allow an easy removal of the primary packages if required and provides a good mechanical link over the residual clearances of the housing. Particular attention must be made to the coupling, for each housing, between the container body and the concrete cast in the second phase.

Design measures have been retained to ensure that the body-lid coupling is good: limiting the cast lid diameter, reducing shrinkage that might generate microfissuring, circular shape to avoid singular zones, interface profile chosen on the basis of feedback on existing disposal packages (low- and intermediate-level waste packages disposed of at the CSFMA Centre).

A testing programme is currently being run to check that this coupling is not a weak zone.

This container is illustrated below (Figure 5.3.2). Its dimensions are roughly 1.50 metres wide and long by 2 metres high. Its total mass (full container) is about 12 tonnes.

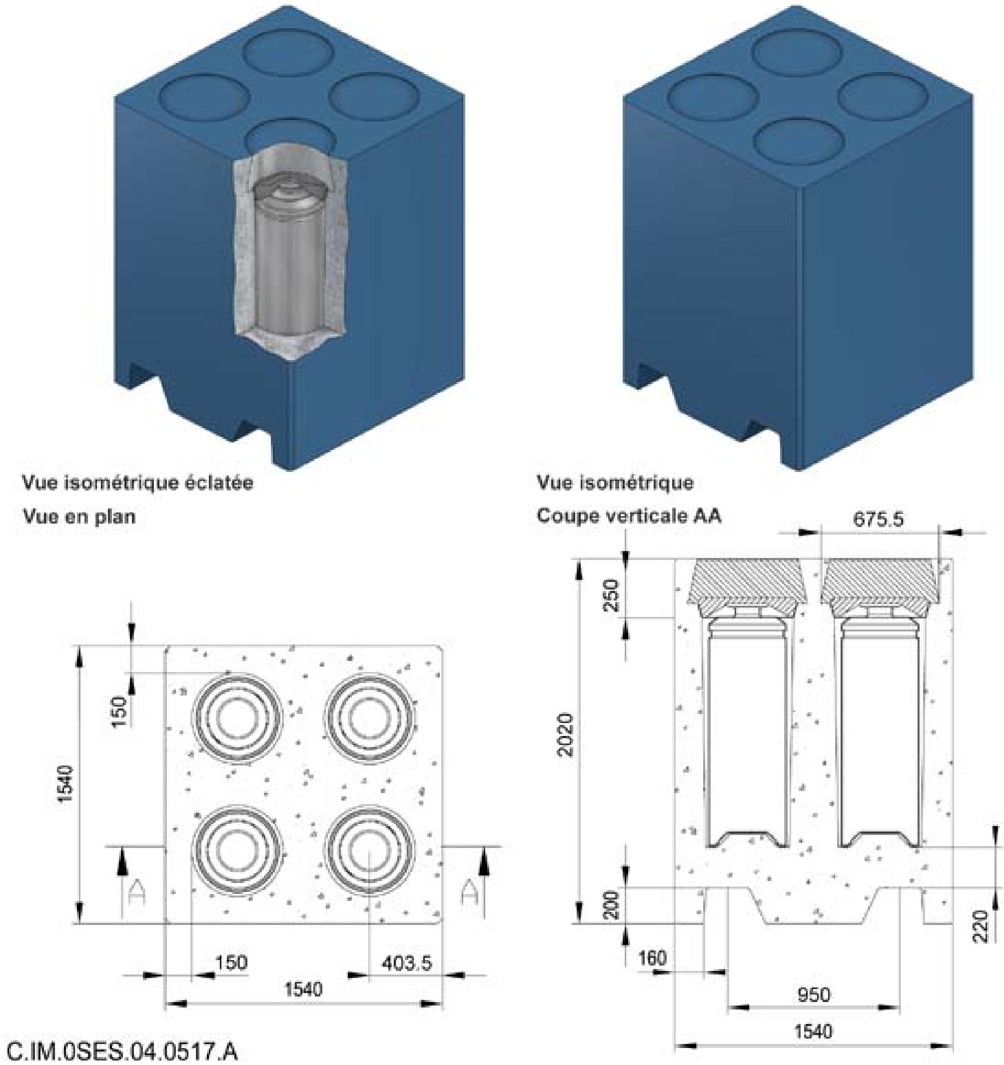


Figure 5.3.2 Example of a disposal package with reinforced retention capacity

Other design options may be envisaged to obtain the performance levels sought for durable confinement. By way of illustration, the Japanese agency RWMC has examined a solution in which the primary packages are embedded in a “single-cast” monolithic concrete matrix and then has successfully carried out with a ¼-scale demonstrator. The concrete used is Ductal® high-performance, fibre-reinforced concrete.

5.3.3 Concrete, a constituent of disposal containers

The material requirements deduced from current standards in the concrete construction field generally include mechanical considerations and secular durability of the structures.

Within the context of container studies, Andra and the CEA have followed these standards; they have also completed them with stricter requirements. The solutions given here are therefore based on scientific studies, laboratory tests and industrial suitability tests performed over several years. Some of them are in synergy with Andra's international peers, particularly the Japanese agency RWMC.

The concrete used for disposal packages consists of four groups of materials:

- cement, mineral additions and water;
- adjuvants;
- granulates (sands and gravels);
- reinforcements and/or metal fibers.

This section is devoted to defining the characteristics of each of them. It presents the respective contributions to global package performance.

5.3.3.1 Technical translation of expected performances

It will be noted first of all that there is strong coherence between the durability of concrete and its retention capacity. Indeed, to meet its objectives, concrete needs to be very compact (low porosity), with a very low diffusion coefficient and reduced permeability (no open porosity nor cracking). These qualities promote concrete durability because they protect the concrete from penetration by water and hence prevent chemical alteration reactions engendered by the presence of water. They are also favorable to retention potential.

The ability to use the material and obtain the best possible quality initial condition (i.e. with cracking as limited as possible in aperture and in length) depends on:

- low hydration heat;
- limited endogenous shrinkage;
- appropriate relaxation of constraints generated early on by deformations blocked in the structure.

Apart from the above characteristics, concrete durability also depends on preventing concrete pathologies such as alkaline silica and alkaline granulate reaction, sulfate attack, hydrolysis and carbonation[37].

The choice of concrete constituents is based on all the above elements. However this choice does not apply to the formulation of the standard container lid defined specifically to provide hydrogen diffusion.

5.3.3.2 Cement, mineral and water additives

- **Cement and mineral additives**

The recommended cement is the "mixed" type. It is characterized by a ternary composition (e.g.: CEM V or CEM I with added silica fume and fly ash). This choice reduces hydration heat and produces concrete with a better developed microstructure. This produces more compact, less porous, less permeable concretes.

The concentration of tricalcium aluminate clinker (C_3A) is limited to less than 5%. This choice[29] minimizes physical disorder (risk of cracking) due to exothermicity of the tricalcium aluminate (C_3A) hydration reaction.

[29] The cements selected to meet this objective are PMES type (Cured sulfated seawater*)* which also prevent sulfate attack.

The quantity of alkaline elements (K and Na) in the cement, adjuvants and other constituents is as low as possible, to limit the risk of silica alkaline reaction (level of alkaline Na_2O not exceeding 0.6%).

The class of cement resistance to obtain a high performance concrete (HPC) class of concrete must be at least 42.5 Mpa.

- **Water dosage and ratio of total water to total cementitious material**

The weight ratio of total water (W) to total cementitious material (C = cements plus mineral additives) must be less than 0.40. To ensure that this is the case, apart from using a reasonably maximized dose of cementitious material, water (W) must also be minimized. A reasonable quantity of cementitious material, taking into account an aggressive environment and current normalization, is within the range of 300 to 550 kg/m3.

This choice makes a significant contribution to obtaining highly compact concrete, i.e. minimizing porosity. This provides better protection from chemical attack and improves mechanical and confinement performances.

The low dose of water also helps limiting the hydrogen source term by radiolysis of the envelope, particularly for disposal containers holding primary irradiating packages type B1 and B5 in particular.

5.3.3.3 Adjuvants

For implementation, adjuvants are necessary. They compensate for the low water content, providing the concrete with the fluidity required while pouring. However, since adjuvants are mostly organic materials, the dosage must be adjusted to limit the quantity of complexing species likely to influence radionuclide water transfer mechanisms (solubility).

5.3.3.4 Granulates

The choice of granulates (sands and gravels) results from the objective of preventing alteration phenomena due to alkali silica reaction.

Considering the time scales considered, particularly for the variant with reinforced retention capacity, granulate must be a physico-chemically stable limestone. Dolomitic limestone (containing magnesium), or limestone with a high alkaline content, must be avoided. The limestone granulates envisaged must contain at least 95% calcium carbonate ($CaCO_3$). The impurities constituting the remaining 5% must also be free of any alkali silica reaction.

The diameter of granulates must be limited (dimensional classification less than 12.5 millimeters) to improve the concrete's mechanical strength and density. The granular section can also be refined to improve the concrete's density therefore enhancing performance and facilitate concrete casting (injection, possibility of pumping). Finally, granulates must be non-reactive.

5.3.3.5 Possible fibres and reinforcement

Any fibres and reinforcement contribute to the need of mechanical reinforcement and durability of the container.

The use of stainless steel reinforcement rods and/or fibres avoids the risk of cracking linked to the expansion of non-alloyed steel corrosion products.

The addition of fibres to the concrete improves the quality while reducing the risk of microcracking and provides characteristics of ductility and tensile strength. A tensile strength of 5 MPa has therefore been targeted.

5.3.3.6 Formulation, industrial implementation tests and performance obtained

Several concrete formulations have been developed in the laboratory, then tested using industrial suitability tests. During these tests, particular attention was paid to the "pumpability" of the concrete, linked to granulate size and dimension and shape of the fibres. A self-placing concrete was therefore sought, which does not need vibration to position it.

Two types of formulations were developed:

- an initial formulation basis was defined using CEM I with the addition of 10% silica fume on the one hand and limestone filler or fly ash on the other, for a total mass of cementitious material close to 500 kg/m^3 and between 170 and 200 kg/m^3 of water. The W/C ratio obtained is less than 0.35. Granulates and sands are Boulonnais limestone for a total mass of around 1700 kg/m^3. The fibres are stainless steel hooked fibres for 30 kg/m^3 (or 56 kg/m^3 for the variant without reinforcement).
- a second formulation basis was defined using CEM V with the addition of 5% silica fume for a total mass of cementitious material close to 500 kg/m^3 and a mass of water between 170 kg/m^3. The W/C ratio obtained is less than 0.35. The fibres are stainless steel straight fibres for 85 kg/m^3.

Furthermore, the specific formulation for the lid of the standard package (hydrogen porosity) was developed on the basis of CEM I with limestone filler but no silica fume and only 25 kg of stainless steel hooked fibres.

The formulations developed and used to produce demonstration models allowed to reach the fixed objectives.

5.3.3.7 Evolution of pH in a repository situation

The choice of materials and development of the formulations presented above means that a target pH for fresh concrete can be set at 13, which will rapidly fall to around 12.5 once the concrete has matured. Then, depending on the evolution of the concrete and its chemical breakdown, particularly by hydrolysis, the pH will fall very slowly to a pH of about 10.5 in degraded state [41].

5.3.4 Manufacturing techniques

This section presents the envisaged manufacturing techniques [40].

In 2004 and 2005 joint work by Andra and CEA resulted in manufacturing full-scale container demonstrators: two variants of the standard package for B2.1 primary packages (fibred and reinforced body, and fibred plain body) and a variant with improved retention capacity for B5.2 primary packages.

A program for assessing the performance of the manufactured objects has also been set up, particularly for early characterisation, the study of the containers-lids coupling (concreting joints and cast lid) and dropping resistance.

5.3.4.1 Manufacture of standard prefabricated disposal packages

The principle for manufacturing standard packages consists of maximising the number of stages involved in a non-irradiating context, so that manufacturing quality can be checked as simply as possible and manufacturing costs optimised.

Two types of element are factory-manufactured in a non-irradiating environment: bodies and lids. The last operations, final primary package emplacement and disposal package closure, are carried out in irradiating cells.

The bodies and lids are prefabricated in steel moulds (cf. Figure 5.3.3).

Figure 5.3.3 Core emplacement in the mould of the standard B2 package prior to casting

Figure 5.3.4 Demonstrator of a standard disposal package

The procedure chosen to seal the lid onto the body for the package closure is the "hydraulical concrete jointing". The purpose of concrete jointing is to provide a mechanical and leak-tight joint between the lid and the body and prevent the lid being torn off. A groove a few centimetres thick between the lid and the cover is used with the right profile to prevent the lid from being torn off. Fresh concrete is cast in this groove.

Concrete hydraulic jointing is a common process used on the fibre-reinforced concrete packages currently at the CSFMA disposal facility in the Aube district (CBF-C1C, CBFK, etc.).

5.3.4.2 Manufacture of disposal packages with reinforced retention capacity

As in the previous case, the container is put together in two stages. In the first step, the body is manufactured outside a nuclear context. In the second step, the primary waste packages are loaded and closed in an irradiating cell.

Once primary packages are installed in the body, a packing plug is placed in each individual housing. Each one is then closed by casting concrete. These small closure lids are matured in a buffer storage area (for 28 days) to obtain adequate performances and stabilisation of the concrete so that the container can undergo quality control before being transferred to the repository.

The process of casting lids onto prefabricated bodies benefits from a great deal of industrial feedback; it has been in use since 1990 by COGEMA and EDF for packages showing low and medium activity disposed of at the CSFMA Centre (Aube district).

Figure 5.3.5 Demonstrator of a disposal package with reinforced retention capacity

5.4 The disposal cells

This section gives a general description of a B waste disposal cell (§ 5.4.1), and then specifies the architecture of the repository tunnels and their dimensioning, taking in account on one hand the grouping possibilities for standardised reference packages and on the other hand granite properties (§ 5.4.2).

5.4.1 The disposal cell

The disposal cell adopted as reference is a horizontal dead-end tunnel in which disposal packages are stacked in several layers. More precisely, the disposal cell is made up of a disposal chamber whose length is governed by the extent of the host block, a cell head and an access drift.

The disposal chamber, once with packages, releases radiation. Package handling and stacking are carried out by a remotely-controlled fork-lift truck type of vehicle. The cell head is fitted with a radiation protection air-lock (dual-gate system) to protect operators during package removal operations from the transfer casks and emplacement in the disposal chamber.

The access drift links the cell head with the disposal module connecting drifts. A seal is constructed in the access drift when it is decided to close the cell (Figure 5.4.1).

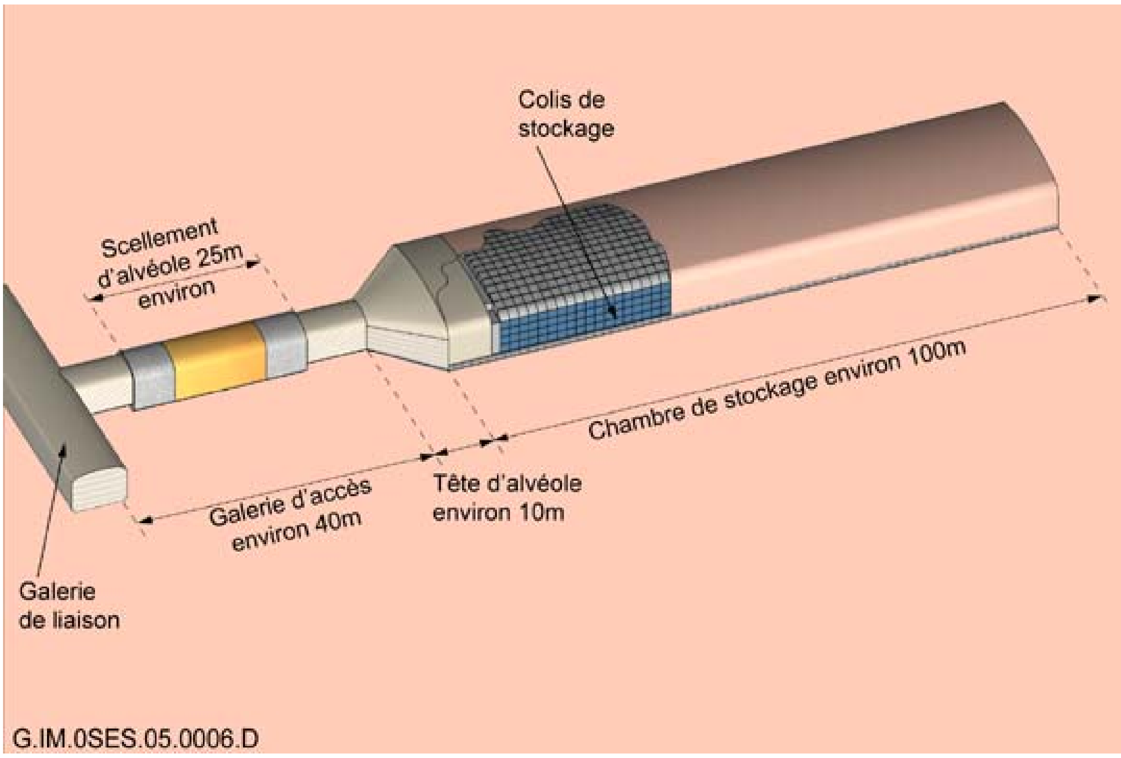


Figure 5.4.1 View of a disposal cell

5.4.1.1 The disposal chamber

Large-dimension mechanically stable sections can be envisaged underground in a granite medium.

Thus disposal chamber sections correspond to a compromise between the need for compactness[30], operational safety considerations, primarily regarding package drop hazards and temperature control for slightly exothermic B waste.

This compromise leads to distinguishing two types of disposal chamber: large-section chambers, about ten metres high and up to 25 m wide, and smaller width chambers (about 10 m across) that accommodate slightly exothermic B waste packages.

The disposal chamber profile has the dual aim of stability without the need for rock support and a shape that reduces the excavated volumes not utilised for disposal. Thus the envisaged shape is an "inverted U", with a very slim arch.

The chamber slab comprises a plain concrete layer on top of which the final layer rests, incorporating rails for the package handling fork-lift trucks. The slab incorporates the ventilation exhaust air and drainage water pipes.

The cell end has a perforated concrete wall behind which arrive return air pipes.

5.4.1.2 The cell head

The cell head at the entrance to the disposal chamber has a radiological protection air-lock (dual-gate system) to allow disposal tunnel operations. This airlock comprises two gates through which the disposal packages are transferred between the operator-accessible access drift and the disposal chamber, which is inaccessible to personnel as it is an irradiating zone.

30 The compactness mentioned here refers to the ratio between emplaced waste volume and disposal chamber volume.

On the access drift side, the shielded cask (that delivers each disposal package from the surface installations) docks at the first gate. This gate protects the operators working in the access drift from radiation.

The dual-gate system is fitted with a mobile travelling floor and a rotary table to enable the fork-lift truck to retrieve the disposal package out of the cask and move it into the disposal chamber (see section 5.6.3). The sliding door on the disposal chamber side allows this fork-lift truck through to place the packages in the disposal tunnel.

The space between the system's two gates is about ten metres long – enough to park the fork-lift truck during maintenance operations (the sliding door thus offers radiological protection for the maintenance staff).

The cell head accommodates the drainage water collection and pumping installation either under the slab or in a side recess.

Ventilation in the dual-gate system zone is provided firstly by ducts linked to the disposal tunnel return air pipes, and secondly by shielded twists & turns at the gates.

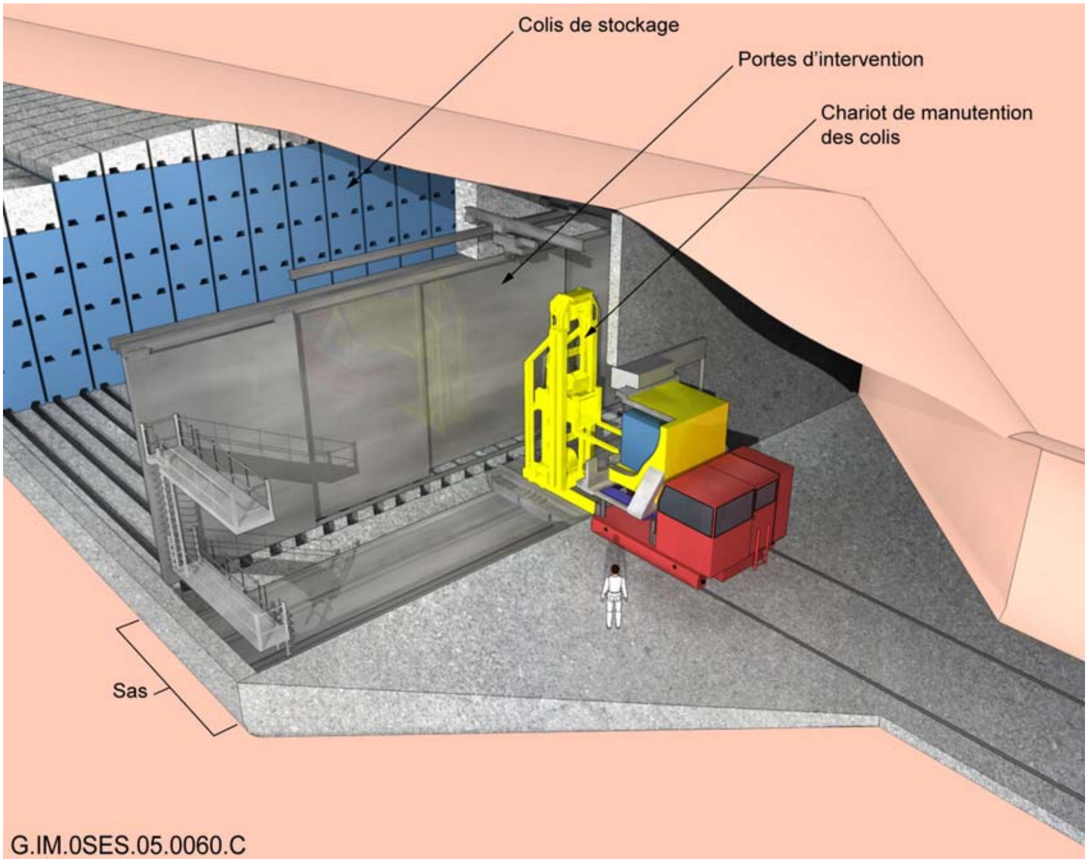


Figure 5.4.2 Cell head

5.4.1.3 The access drift

The access drift links the module connecting drifts to the cell head. Its useful section, approximately 30m^2, enables the tunnel construction machinery and then the package transfer cask trolley to access the disposal chamber.

The entrance drift length is adapted so that a seal can be constructed in a zone that is slightly damaged or undamaged by the excavations works. This zone is located some ten metres from the connecting drift and the cell head. The access drifts are designed to be about forty metres long.

5.4.1.4 The cell seal

The cell seal comprises a swelling clay core about ten metres long that is mechanically confined between two plain concrete abutments. The seal is constructed in an over-excavated part of the access drift.

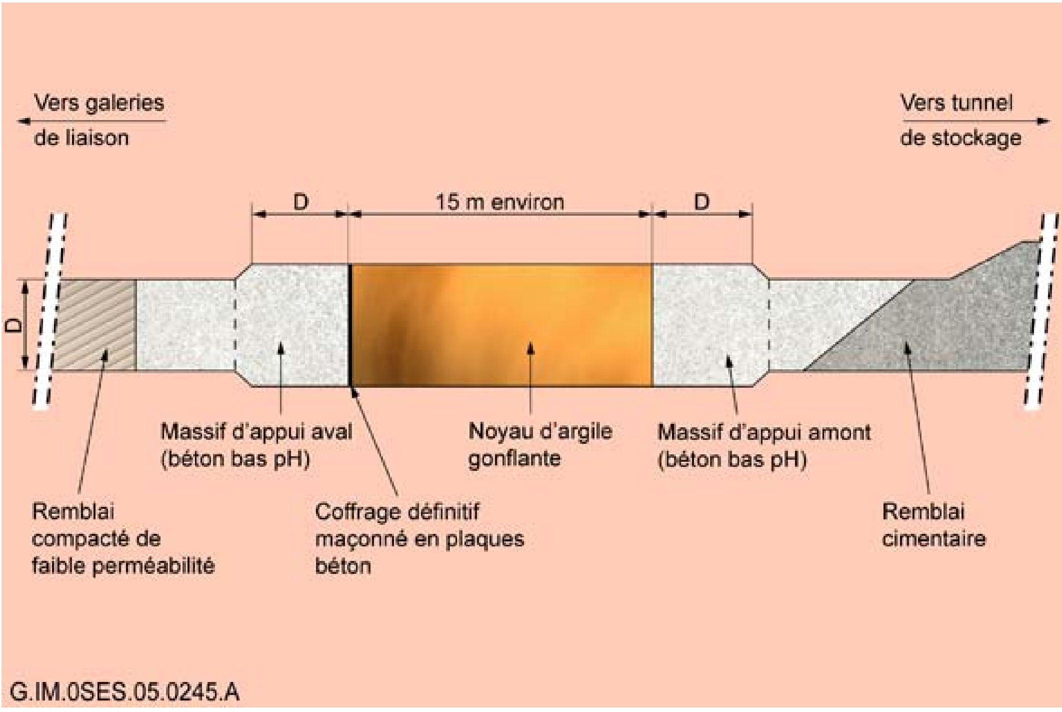


Figure 5.4.3 Basic diagram of a B waste cell seal

5.4.2 Dimensioning elements of the disposal cell

The elements taken into account for dimensioning the B disposal cell sections relate to:

- the possibilities of grouping different types of waste together (standardised disposal package dimensions and physical and chemical compatibility of the waste);
- geo-mechanical aspects (mechanical characteristics of the granite, geometry of the discontinuities, etc.);
- thermal aspects (temperatures of the geological medium and thermal conductivity);
- technological aspects (package stacking heights, etc.).

5.4.2.1 Grouping the reference packages

Grouping several types of different packages together in a single cell is sought to improve repository compactness and make repository management more flexible in terms of package supply.

However at the current state of knowledge, some groupings of packages need to be avoided: degradation, through low heat release, of packages emplaced together should be limited or prevented, and the impact of complexing species or acids forming during the alteration of the organic matter contained in some packages should be limited too. The main outcome is avoiding grouping together slightly exothermic packages (reference packages B1 and B5) and packages that are sensitive to temperature (reference packages B2). A co-disposal scenario of reference packages in some cells has been drawn up, that both observe the separation constraints of certain reference packages and also the geometric dimensions of the standardised packages. This scenario also enables those packages that produce gases to be separately managed.

Thus allocation to distinct cells has been made for:

- packages that do not contain organic matter or release hydrogen;
- packages that do not contain organic matter but release hydrogen;
- packages that contain organic matter (and release hydrogen);
- reference packages B2 (bituminised waste).

Furthermore disposal packages of identical dimensions or those that provide stacks of very similar widths and heights to be built up, are emplaced in each cell.

Accordingly ten different types of cell have been identified.

5.4.2.2 Dimensioning and description of non-exothermic waste tunnels

From a mechanical point of view, preliminary dimensioning calculations [39] at the planned construction depths have been made both to check the stability of the large diameter tunnels and make sure that the fissured zone[31] around the structures remains limited.

These calculations have been based on the selection of mechanical properties considered as the median of a comparison of six granite sites in France and abroad, in the absence of specific site.

The results of these calculations show that tunnels about ten metres high by about 25 m wide with an "inverted U" section are "self-stable" in the event of stress isotropy along the horizontal plane. The calculations indicate that there is little likelihood of a fissured zone appearing. In the event of anisotropy (with a hypothesis of the horizontal stress being twice the vertical stress), even if the mechanical stability is confirmed, a fissured zone extending several tens of metres may appear.

From a technological point of view, the design of mechanical handling equipment for the B waste, which is the same as for the studies for the clay medium, enables piles of waste packages to be stacked to a height of some ten metres.

The sections envisaged for the non-exothermic disposal cells are about 200-260 m^2.

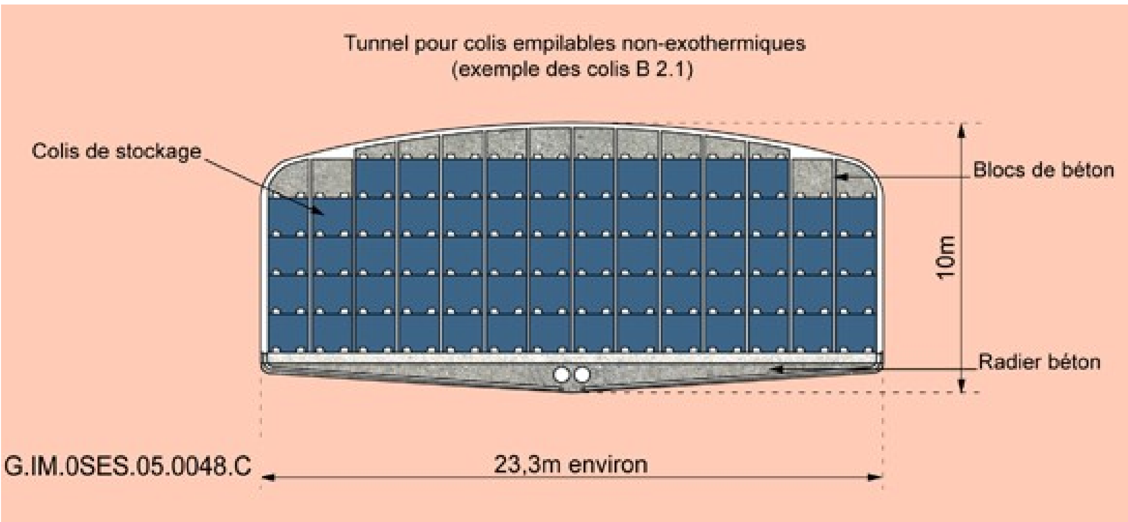


Figure 5.4.4 Cross section of a disposal tunnel of non-exothermic B waste

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[31] The fissured zone, in this case, refers to the damaged zone in the geo-technical sense. It is the zone where irreversible displacements are likely to take place along existing fractures (natural or generated by the excavation works).
In the rest of this report the term "damaged zone" is used exclusively to refer to the zone impacted by the excavation technique used, to avoid ambiguity.

The application of light rock support (bolts and shotcrete) may be envisaged to ensure that safe conditions are provided on the construction site and for the operating activities.

Filling the clearance space between the packages and the granite roof may require the installation of suitably shaped concrete elements.

The slab is cut out with a slight slope towards the centre of the disposal tunnel to facilitate cleaning operations prior to the construction of the final concrete slab.

Lateral spacing between the columns of stacked packages is 10 cm. The clearance between cell walls and packages is 10-20 cm. The space between each row of packages in the longitudinal plane is minimised.

The assessed accumulated length of the disposal chambers produces the following data for each waste grouping and per inventory model scenario:

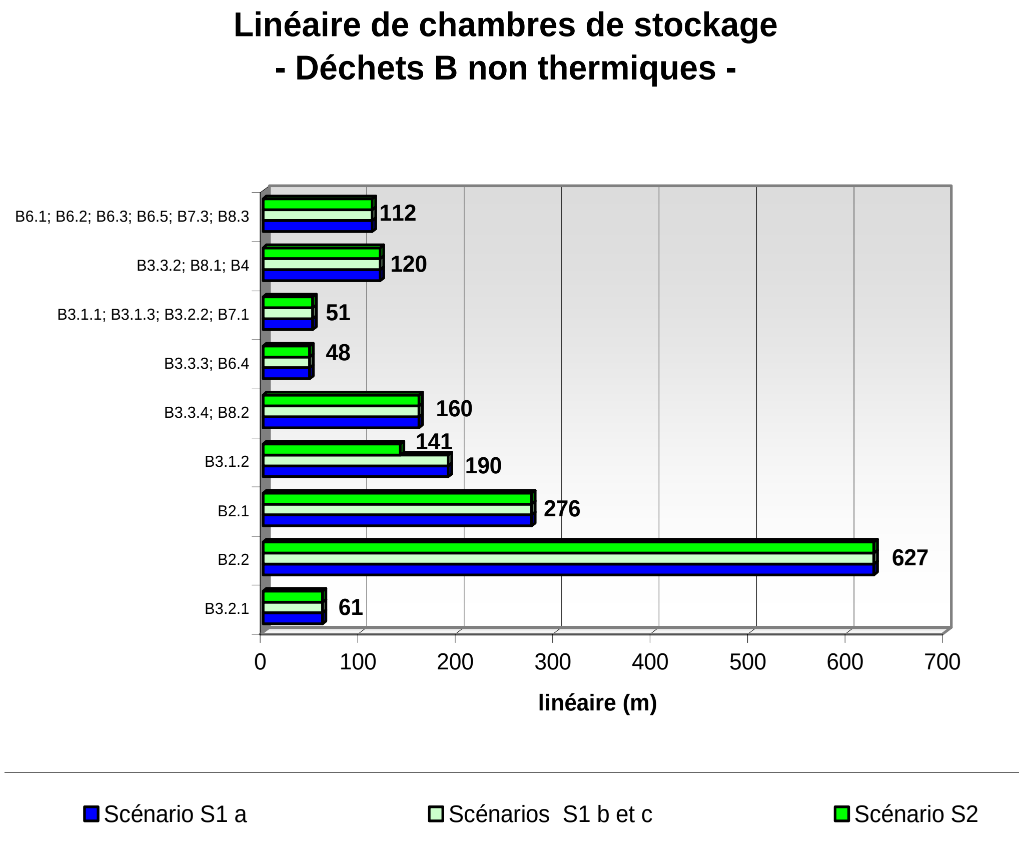

Figure 5.4.5 *Assessment of the length in metres of the disposal chambers for non-exothermic B waste*

5.4.2.3 Dimensioning and description of slightly-exothermic waste tunnels

In the slightly exothermic B waste disposal tunnels (types B1 and B5), the packages must be arranged so as is not to exceed the temperature criterion of 70°C in the concrete (cf. 5.2.2.3) in order to control the behaviour of the packaging concrete and radionuclides inside the cell.

Dimensioning calculations [42] have been made to analyse the sensitivity of the potential arrangements to the variability of thermal characteristics presented by French granites [7] (Table 5.4.1).

Table 5.4.1 Thermal characteristic input data for granite massifs

	Temperature at a depth of 500 m	Thermal conductivity (W/[m.°C])
"Cold" granite	18°C	3.5
"Medium" granite	25°C	3.3
"Hot" granite	28°C	2.5

Calculations have shown that it is possible to adjust the arrangement of the disposal packages, and more particularly the number of packages that can be stacked per tunnel section, (Table 5.4.2), to meet the temperature criterion mentioned above.

Preliminary calculations have been made for a disposal cell with reference package B5 (in relative terms the most exothermic of the B waste packages that accounts for 85-95% of the total inventory of slightly exothermic B waste). They are based on 500 m deep cell, disposal of the packages some fifteen years after being conditioned and a buffer distance of about one hundred metres between adjacent cells. The favourable effect of cell ventilation is not considered into the calculations.

Table 5.4.2 Number of disposal packages of B5.2 waste per tunnel section

	Number of packages per tunnel section	Maximum temperature in the concrete packaging (°C, +/- 2.5°C)
"Cold" granite	20	66
"Medium" granite	18	66
"Hot" granite	12	65

In the case of a "medium" granite the package temperature (Figure 5.4.6) reaches its maximum value less than 10 years after emplacement then drops back below 50°C, about 400 years later.

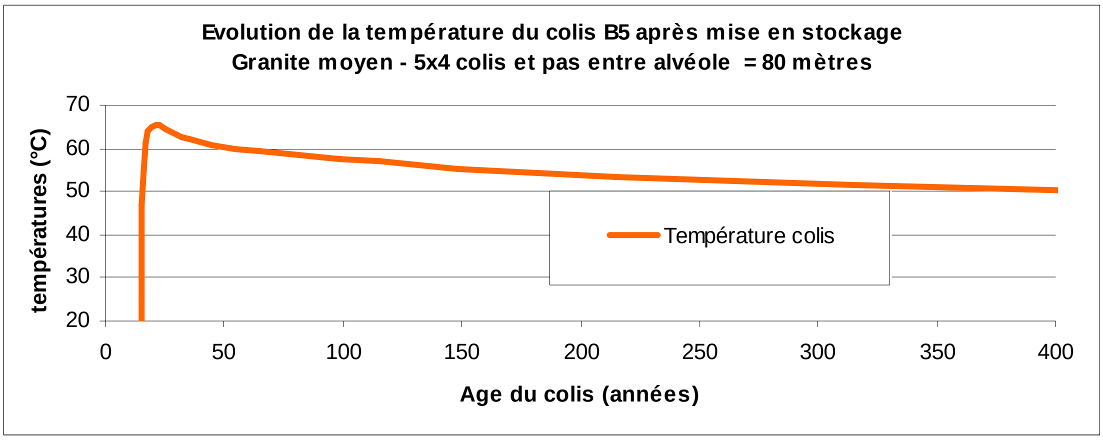

Figure 5.4.6 Temperature evolution of the package after emplacement

These preliminary calculations show that the possibilities of arranging slightly exothermic B packages are more sensitive to the initial temperature of the massif than the thermal conductivity of the granite.

In two-level repository architecture, installing the thermal package tunnels on the upper level (where the temperature of the massif is cooler) is preferred for a compact repository. This upper level in the reference architecture is about 400 m deep where the temperatures of the massif should be 2-4°C lower than those considered in the calculations.

Disposing of 18 packages per section of exothermic waste tunnels at a depth of about 400 m in granite with "medium" thermal characteristics is thus adopted as the reference.

Adapting this reference to real site conditions that deviate from those considered here as average, will affect the number of packages per tunnel section, which finally will have a consequence on tunnel length to be constructed for all the slightly exothermic waste.

The reference tunnel comprises disposal chambers with an excavated section of 80-85 m^2, for stacking 18 disposal packages in 5 columns, with 4 levels in the central part and 3 levels along the side walls[32]

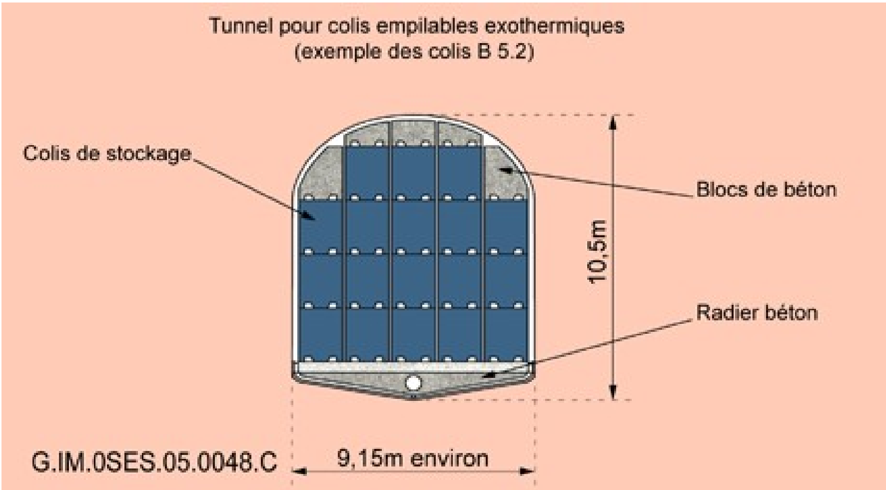

Figure 5.4.7 Cross-section of a disposal tunnel of slightly exothermic waste

The construction details are similar to those of the large diameter tunnels (bolts and shotcrete used to ensure excavation site safety, slab cut to slope slightly towards the centre of the tunnel, spacing between the columns of packages).

Assessment of overall disposal chamber length results in the following figures for the groupings of waste and for each scenario:

32 In the event of significant stress anisotropy ($\sigma_H/\sigma_V > 2$), geo-mechanical stability calculations could examine geometry enabling the packages to be stacked on 7 columns with only 3 levels in the central part.

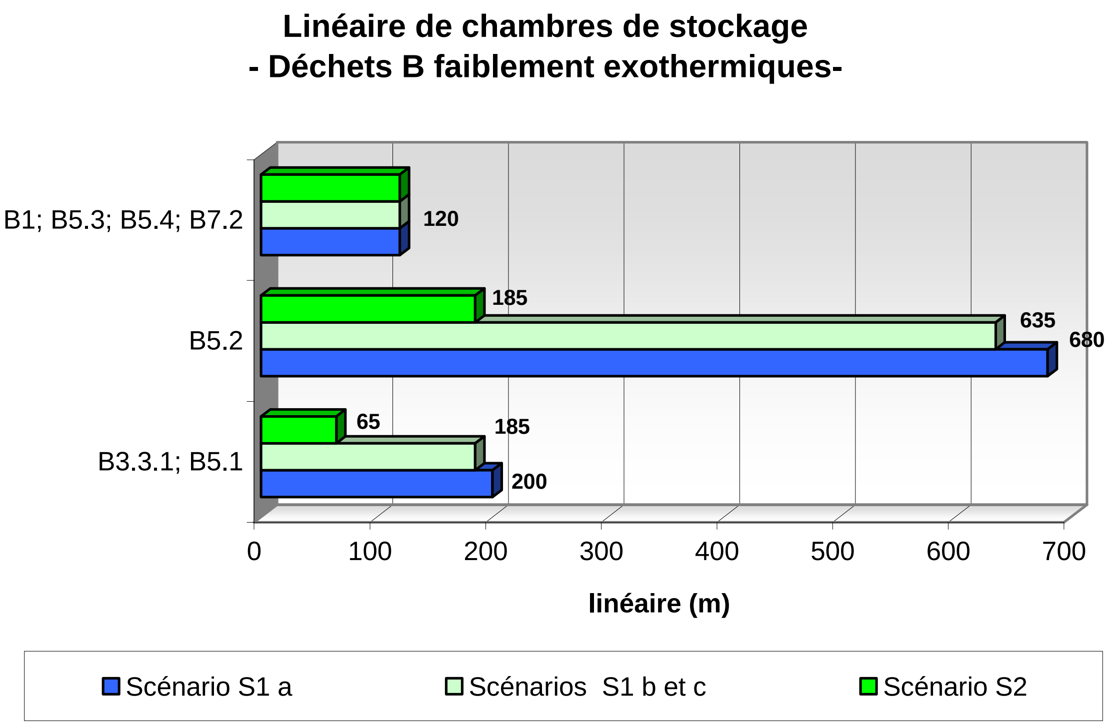

Figure 5.4.8 *Assessment of the disposal chamber length in metres for slightly exothermic B waste*

5.4.2.4 Disposal chamber length

The disposal chamber lengths are in the range 50-150 m. They are individually adjusted:

- to the grouping scenario of the different types of package;
- to the distance between the cell and the edge of the host block – the potential site for water circulation. Thus the end of the chamber is sited with a buffer distance of several tens of metres from the edge of the block. This buffer distance is specified according to the hydraulic characteristics of the block and its environment just above the cell, on the basis of the hydraulic criteria defined during the site qualification exploration works. These characteristics are surveyed during the "on-going characterisation" work.

5.4.2.5 Assessment of the number of B waste disposal cells

The B waste inventory repository requires about thirty cells in the case of the S1 (a, b and c) scenarios and about twenty in the case of the S2 scenario with average chamber length at about 100 m.

5.5 Architecture of the B waste repository zone

5.5.1 Dimensioning elements of a disposal module

A disposal module is made up of a set of structures (disposal cells and associated drifts) installed inside a block of low permeability granite. Exploration, construction followed by operation of the module cells are achieved through the connection drifts. When the decision is made to close a module, backfill and seals are emplaced in the connecting drifts. They isolate the module cells from the rest of the repository zone.

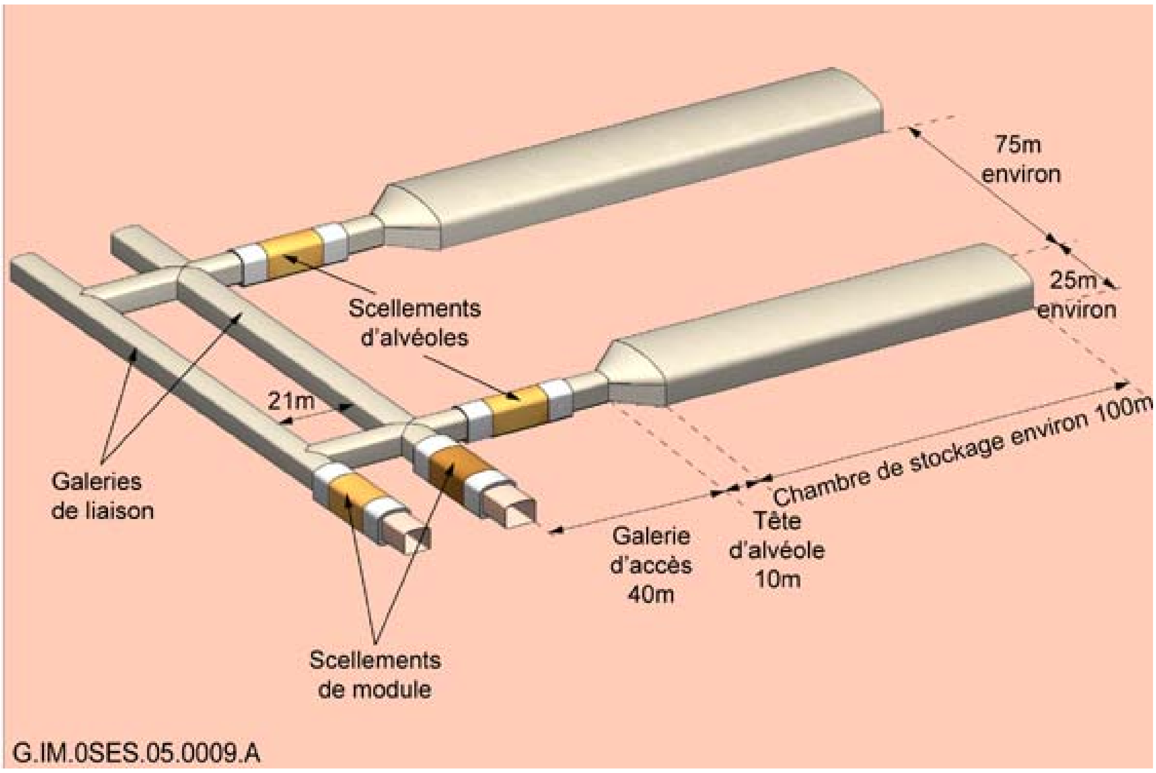

Figure 5.5.1 3D view of a module of non-exothermic waste cells

In the case of two-level architecture, the module refers to disposal cells located at the same depth and sharing common connecting infrastructures.

Disposal module dimensioning primarily depends on the definition of rock volumes to be preserved around each cell. The previous section introduced a "hydraulic" buffer distance between a disposal chamber and potentially water-conducting faults that delimit the host block. Two other types of buffer need to be defined in order to control the disturbance to the medium induced by the medium:

- "thermal" buffer, between disposal chambers of slightly exothermic packages ;
- "mechanical" buffer, between disposal chambers (pillar), and between a chamber and faults at the edge of the blocks.

- **Dimensioning of a "thermal" buffer**

The compactness of the repository with modules that accommodate slightly exothermic waste (types B1 and B5) depends on a defined buffer distance to limit thermal disturbance between the disposal cells.

Accordingly an analysis has been made of the design sensitivity to the various clearance values for the three types of granite modelled using preliminary disposal cell dimensioning calculations (§ 5.4.2.3).

For buffer values close to 60 m, thermal interference is not perceptible at the core of the pillar between two cells until after a few centuries. When this interference is observed, the temperature of the packaging concrete is less than 50°C.

- **Dimensioning of a "mechanical" clearance**

Taking into account mechanical criteria for dimensioning disposal modules plays a part in defining the pillars between two large diameter tunnels, and in the buffer distance between a cell and a fault delimiting the module host block.

Pillar thickness equivalent to three times the width of the widest tunnel, in the absence of parameters for a particular granite, would seem sufficient to guarantee that there is no mechanical interaction between two neighbouring cells.

The calculations for the buffer to be provided in mechanical terms between a cell and a fault on the edge of a module, indicate that a distance of about 10 m makes the risks of interaction between a cell and a fault negligible [39].

- **Dimensioning and description of the connecting drifts**

The connecting drifts are long "inverted-U" structures 6-7 m wide and about 6 m high. They are dimensioned for cell construction machinery traffic, allowing for vehicles crossing, package transfer and the installation of all the utilities and systems (water, air, energy and instrumentation) during module construction and operation phases.

At least two parallel drifts are required for the construction followed by the operation of a disposal module. They are linked by an interconnecting drift at the level of each disposal cell and no less than every 400 m. The pillar between two connecting drifts is about twenty metres wide (about three times the width of a drift). The width of a cluster of two parallel drifts is about thirty metres.

At the time of closure, the connecting drifts at the edge of the module are sealed with a swelling clay core placed between two concrete abutments of a similar design to the cell seals.

5.5.2 Assessing the number of granite blocks to be identified in the disposal zone

The B waste repository zone includes all the repository modules arranged on one or more levels, and the drifts that link them to the underground-to-surface connecting structures.

The repository modules are distributed on two levels in the reference architecture. Structural elements (as faults) that define the block geometry of most of the French granite massifs surveyed at one level can be transposed to the other level at the scale of one hundred metres.

It can be envisaged that the construction of two to four 100 meter-long cells per module is feasible according to the site configurations. The exploratory work thus required should identify some twelve blocks.

In the case of a site configuration with geological constraints by fracturing, the number of blocks to be characterised is close to the number of cells to be built.

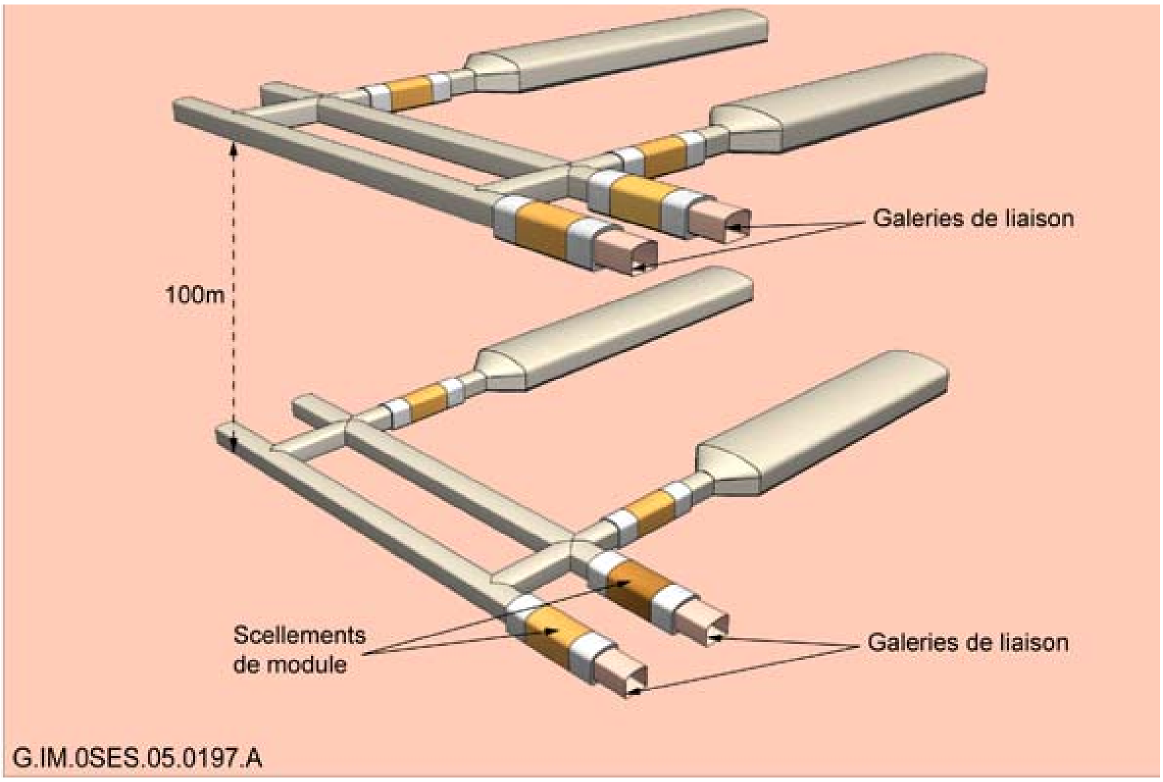

Figure 5.5.2 3D view of modules on 2 levels

5.6 Disposal process and feasibility elements

This chapter distinguishes the activities simultaneously implemented during the B waste disposal process. The activities vary in kind:

- on-going exploration aiming to define the disposal tunnel layout inside the granite blocks (cf. § 4.3.1);
- construction of tunnels and disposal modules;
- nuclear operation in the disposal tunnels, for package emplacement and reversible management of the structures.

Moreover when the choice is made to close the disposal modules, the work to be carried out is similar to the construction activities.

Architectural measures in the B waste repository zone are such that these various activities can be flexibly and simultaneously carried out.

Along these architectural measures, the proximity of the modules to the main connecting drift allows the throughputs required by each of these activities to be organised at all points of the repository zone.

5.6.1 Siting the disposal tunnels

The B waste cells are installed in very low permeability granite blocks that favour a mainly diffusive transfer mode in the near-field granite, and far enough away from the water-conducting faults – some several tens of metres away.

The envelope of the blocks and their characteristics have been subject to detailed survey during a preliminary phase of the works (cf. § 4.3.1).

This envelope defines a volume inside of which the properties of the fractures meet the criteria laid down during previous exploratory work, basically in the underground laboratory.

In principle installation of the cells in this volume is validated for the most stringent criteria, concerning properties of minor fracturing that must be avoided by the disposal tunnels.

The construction programme is carried out in several stages. The sequence of these works aims to gradually acquire the elements needed to validate each arrangement.

This works programme consists of:

- Drilling boreholes, performing hydro-geological tests and geo-physical measurements between boreholes in the volume of the blocks. The distance between boreholes is adjusted to the desired resolution when interpreting the geophysical measurements. It can be envisaged from currently available technologies, that the spacing between boreholes will be about fifty metres. Borehole length is adjusted to the block dimensions to avoid intersecting its envelope.
- The results of this work should allow accurate cell installation keeping a minimum distance of about one hundred metres between them, and a minimum distance of several tens of metres from the water-conducting faults. Thus a cell may be installed on a borehole or between two boreholes.
- This arrangement may be confirmed by measurements in the boreholes either side of the cell to complete near-field characterisation, particularly its very low permeability.
- Once the cell site has been determined, a small diameter exploratory drift is excavated in the borehole axis up to the planned end point. Final validation of the cell site is acquired by comparing the minor fracturing on the walls against the criteria defined in the underground laboratory.
- When a cell is sited between two boreholes, a borehole may be drilled in the axis of the exploratory drift.
- Cell qualification is finally acquired following detailed geological surveys of the fracturing carried out on the excavation walls at the end of excavation work.

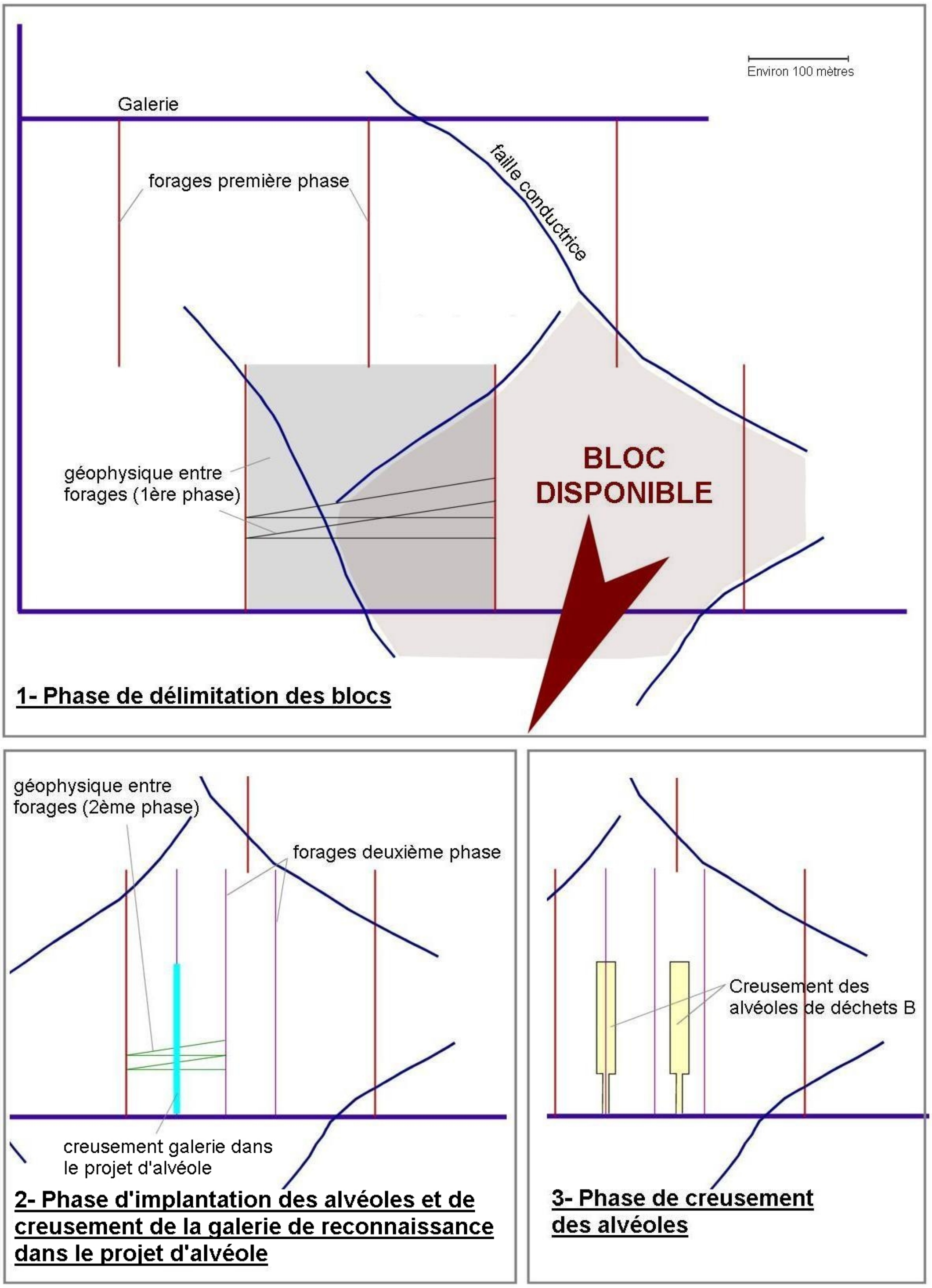

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Figure 5.6.1 *Basic diagram of exploration and characterisation for a B waste module and cell*

5.6.2 Constructing a disposal cell

Disposal cell excavation is carried out by the drill& blast method. However the implementation of this method must be based on suitable boring and blasting methods to limit damage to the rock at the excavation wall.

- **Excavating the access drift**

The access drift is built in two stages:

- excavating an exploratory drift;
- then over-reaming it to the useful section of the access drift.

The exploratory drift whose section (of about 14 m^2) is part of the future section of the access drift, is continued to the end of the cell. Full section blasting is made to excavate the exploratory drift. Blasting technique involving smaller section to limit the damaged zone at the excavation walls may be used until the cell seal construction zone is positioned. Light safety ground support is applied to the drift arch (essentially shotcrete).

Over-reaming the exploratory drift takes place in a post-excavation phase. To minimize the risk of damage, the use of sawing techniques (diamond-tipped disks or cables) at the level of the future seal zone may be envisaged.

- **Excavating the disposal tunnel**

In the case of planned heights of 10 meters or so, excavating the full of the disposal chamber is carried out in two drill & blast phases: excavating a "top heading" followed by excavating a “stross”.

Limiting the extent of the damaged zone may be achieved by combining blasting techniques of smaller sections and adjusting the blasting patterns (boreholes, explosives and firing sequences).

This combination results in the following excavation sequencing:

- Excavating the top heading over the whole disposal tunnel length:
    - excavating a production section (at least 1 m) that is inside the useful section;
    - excavating the useful section and purging towards the production section;
- then excavating the stross;
    - excavating the stross to a production section (at least 1 m) that is in the useful section;
    - excavating the slab;
    - excavating the stross side walls.

At the end of the excavation, a clean-up/purging operation takes place to prepare for casting the concrete slab.

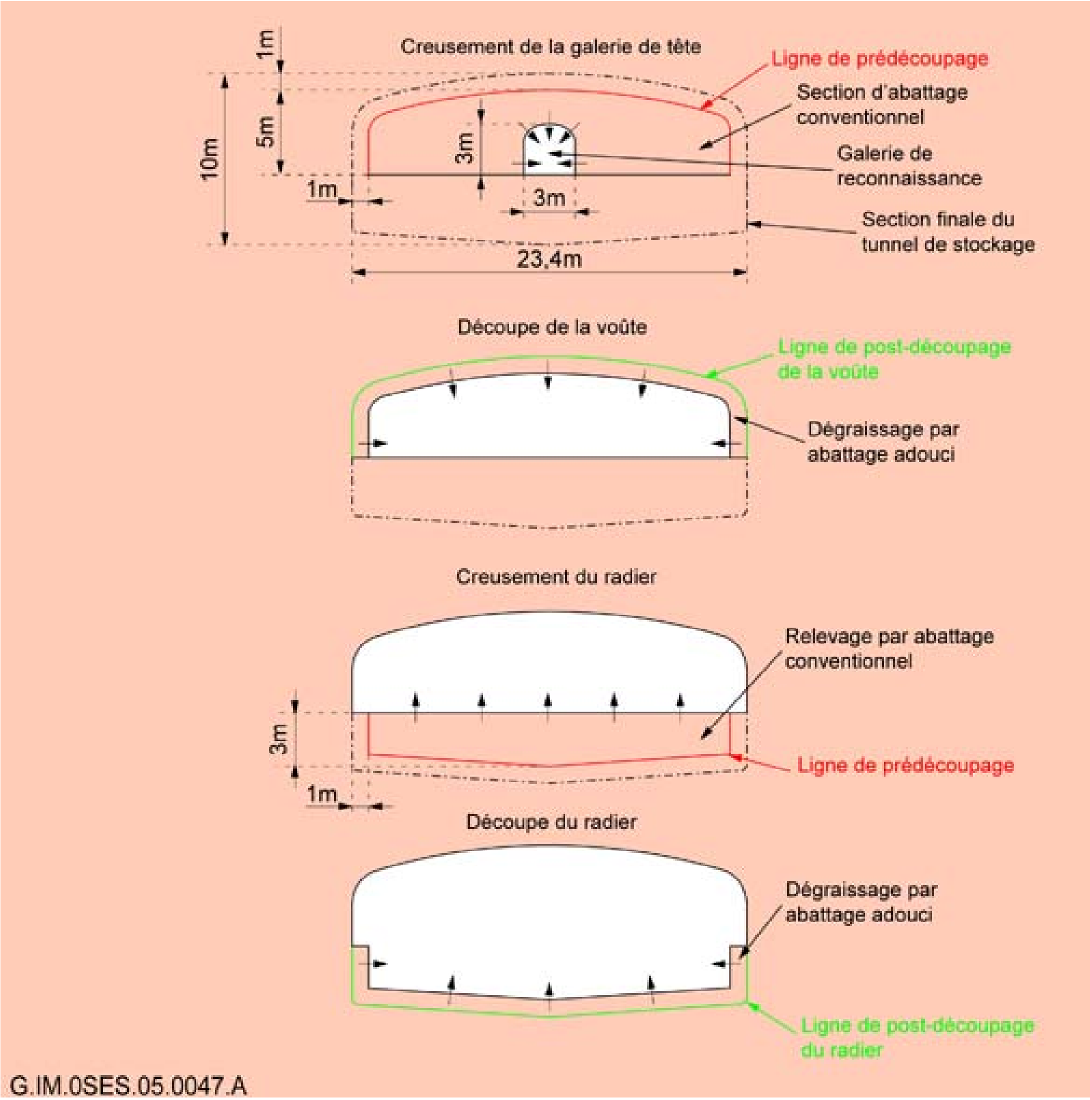

Figure 5.6.2 Basic diagram of blasting in smaller sections

- **Cell construction**

Once the tunnel slab and access drift have been cleaned up, the concrete slab is cast over the whole length of the cell, starting from the cell end towards the connecting drifts. Utility networks are gradually installed (ventilation, drainage and possible instrumentation).

Once the slab concrete has cured, the concrete wall at the cell end and the dual-gate system are constructed. The final slab, that incorporates the rails for the package handling machinery, is cast at the end of the construction phase.

5.6.3 Reversible operation of a disposal cell

This section presents the methods and procedures for transferring B waste disposal packages from the surface installations to the disposal cells, and the methods and procedures for emplacing these packages in a disposal cell. By reversing the process, these methods and procedures are also used to retrieve the emplaced packages.

The package transfer operations are carried out using a radiological protection cask comprising a base plate, a parallelepiped shielded vessel and a double door. It has no onboard mechanical devices, since the loading and unloading operations are carried out by separate equipment at the docking station. It comprises motorised doors that engage the cask doors for opening and closing. Cask wall design uses materials as dictated by the types of radiation emitted by the waste packages carried (lead or steel for γ rays, steel composite and neutron-absorbing materials for neutrons).

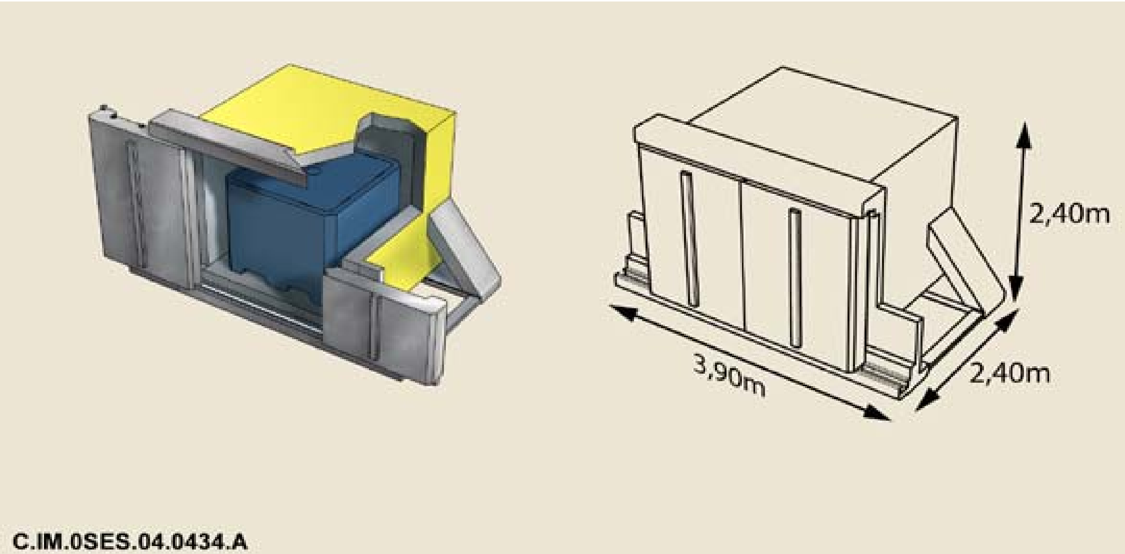

Figure 5.6.3 Biological protection cask for the most irradiating B waste packages

Several families of casks have had to be defined to deal with the variety of dimensions, masses and equivalent dose rates of the B waste disposal packages. Their loaded masses vary from 40-100 tonnes for 7-25-tonne waste packages. The thickness of the biological protection varies from 50-300 mm of steel or composite material depending on the dose rate of the packages transported.

The cycle to transfer the protective casks holding disposal packages from the surface installations to their emplacement in a disposal cell can be broken down into a number of stages:

- loading the cask on the surface;
- transferring from the cask into the package-shaft;
- transferring the cask through the drift-network;
- docking the cask with the cell head;
- extracting and transferring the package into a disposal cell.

5.6.3.1 Transferring the cask into the shaft and drifts

The cask loading and unloading procedures in the shaft cage at the surface and in the underground installations are similar.

To lower a disposal package down to the underground installations, a transfer vehicle lifts the cask, takes it from the storage building to the shaft via a dual-gate system, and places it into the cage on a skid.

A similar vehicle in the underground installations takes over the cask and transfers it through the drifts to approach the disposal cells.

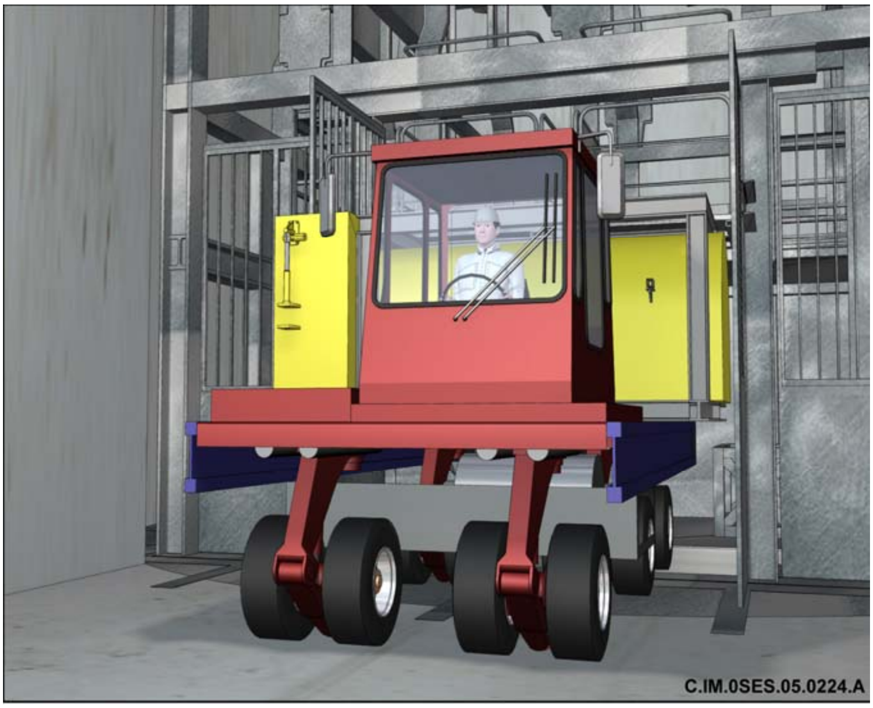

Figure 5.6.4 Principle of loading and unloading the cask in the shaft cage

The transfer vehicle is a low-profile jack-up electrically self-propelled trolley on tyres. Electrical energy is preferred over a diesel engine, in view of the inherent fire risks associated with this type of motorisation. The machine comprises a loading deck with independently motorised multi-directional wheels giving a very tight turning circle. This type of vehicle uses widely proven technology, primarily in the nuclear context. The technology has been used at COGEMA's La Hague site for many years.

This design also benefits from the transposition of industrial practices in the mining industry. Its main aim is to minimise the dimensions of the drifts and crossing structures. It is based on the principle of separating the transport functions over long distances that can be provided by industrial machines, from the docking function, that calls for more specific equipment to couple the cask accurately with the cell head when unloading.

5.6.3.2 Docking the cask with the cell head

Docking the casks with the cell heads is carried out by a specific rail-borne vehicle (the "docking shuttle"). This vehicle is also of the electrically self-propelled jack-up type. This design enables the cask to be docked from a narrow access drift located in front of the disposal cell.

After being unloaded by the transfer vehicle, the cask is taken up by the shuttle, which travels along the access drift to the disposal cell and docks against the "docking port" of the radiological protection dual-gate system at the cell head.

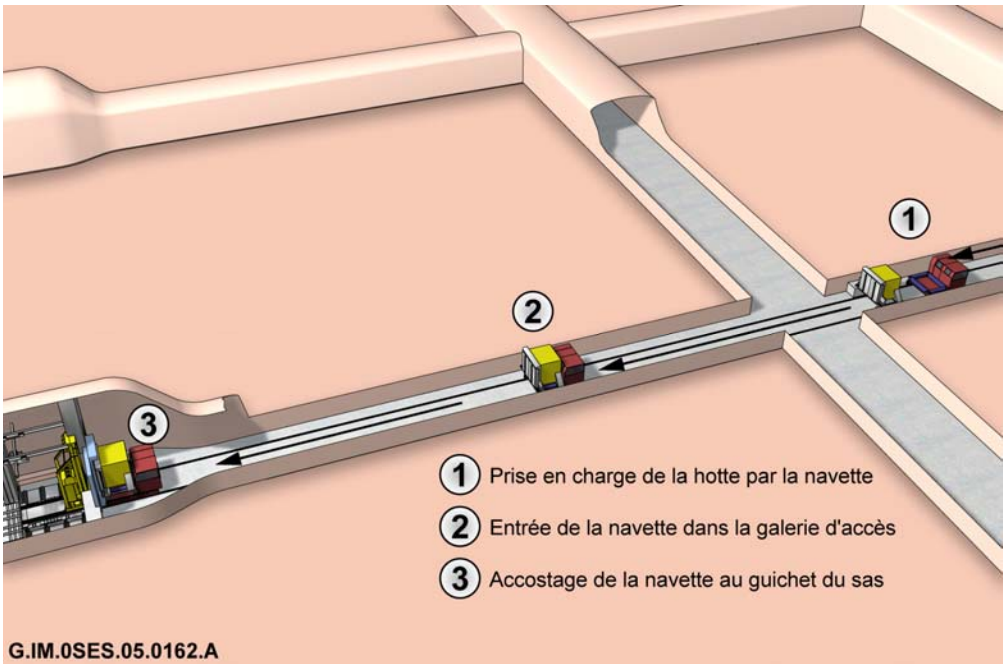

Figure 5.6.5 Docking the B waste cask

5.6.3.3 Extracting and transferring the package into a disposal cell

The sequence of disposal package extraction from the cask and emplacement in the cell is illustrated by Figure 5.6.6.

Two types of equipment are needed to carry out these operations:

- a radiological protection dual-gate system;
- a fork-lift trolley (disposal trolley).

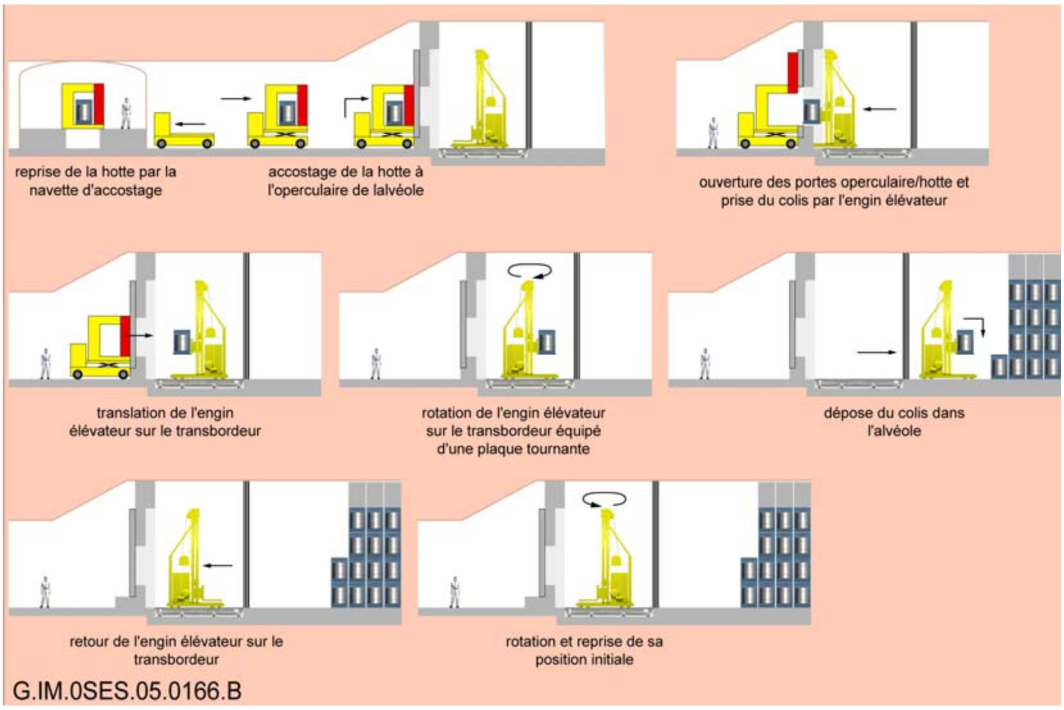

Figure 5.6.6 Synoptic scheme of B waste package emplacement in a disposal cell

- **The radiological protection dual-gate system**

The radiological protection system dual-gate system at the cell head comprises two doors that enclose enough space to accommodate the disposal trolley (Figure 5.4.2).

The transfer cask docking port comprises two motorised sliding steel panels 300 mm thick.

The door opening onto the handling chamber comprises two full cell-height sliding panels (steel plate about 300 mm thick); these panels let air through to ventilate the disposal chamber.

The dual-gate system has a moving floor that carries the disposal trolley. By rotating through 180° and through its sideways sliding capability, it can position the disposal trolley opposite the cask to extract the packages or opposite one of the rows of packages for the transfer operation in the disposal chamber.

- **The disposal trolley**

The disposal trolley used to extract the waste packages and place it in the cell is a fork-lift type trolley. A preliminary feasibility study has resulted in an assessment of its dimensions, which are approximately 2 metres wide, by 4 metres long by 6 metres high. Its mass is about 25-30 tonnes. It can be powered by on-board batteries.

The main structure of the disposal trolley comprises two metal beams fitted with wheels. The two lifting forks slide vertically along a beam that interlinks these two beams. The mast is positioned so that the forks are at the vertical to the beams. Thus, when handling a waste package, its centre of gravity is inside the base formed by the wheels.

This truck combines translation inside the cell with lifting for package handling. The disposal trolley moves along rails placed in the grooves built over the whole length of the cell slab floor. In the interest of compactness, this principle has been favoured over a system with tyres.

The package is kept at low level throughout the way inside the disposal chamber, to avoid any risk of dropping and to offer stability to the moving unit.

When it approaches the wall of emplaced packages, the disposal trolley gradually slows to a halt to start package emplacement. The package is lifted to its final deposition height. The disposal trolley moves slowly forward and positions the package above the placement line formed by the top of the upper package of the row. The package is then lowered onto it very slowly.

The positioning precision of the packages when being deposited or eventually removed, raises a major technological issue. There are several technologies available to provide a solution. For instance, it is possible to measure the distance travelled by the disposal trolley using a "laser distance meter", which is commonly used in industry for ranges of up to 500 metres with a precision level to a few millimetres. Use of this type of process is compatible with package-to-cell wall and package-to-package functional clearances of about 10 cm.

With regards to risk of package drops, the disposal trolley is equipped with safety devices current in nuclear equipment design. For example a load lowering velocity reducing device is envisaged in the event of a malfunction or break of an element of the lifting kinetic process.

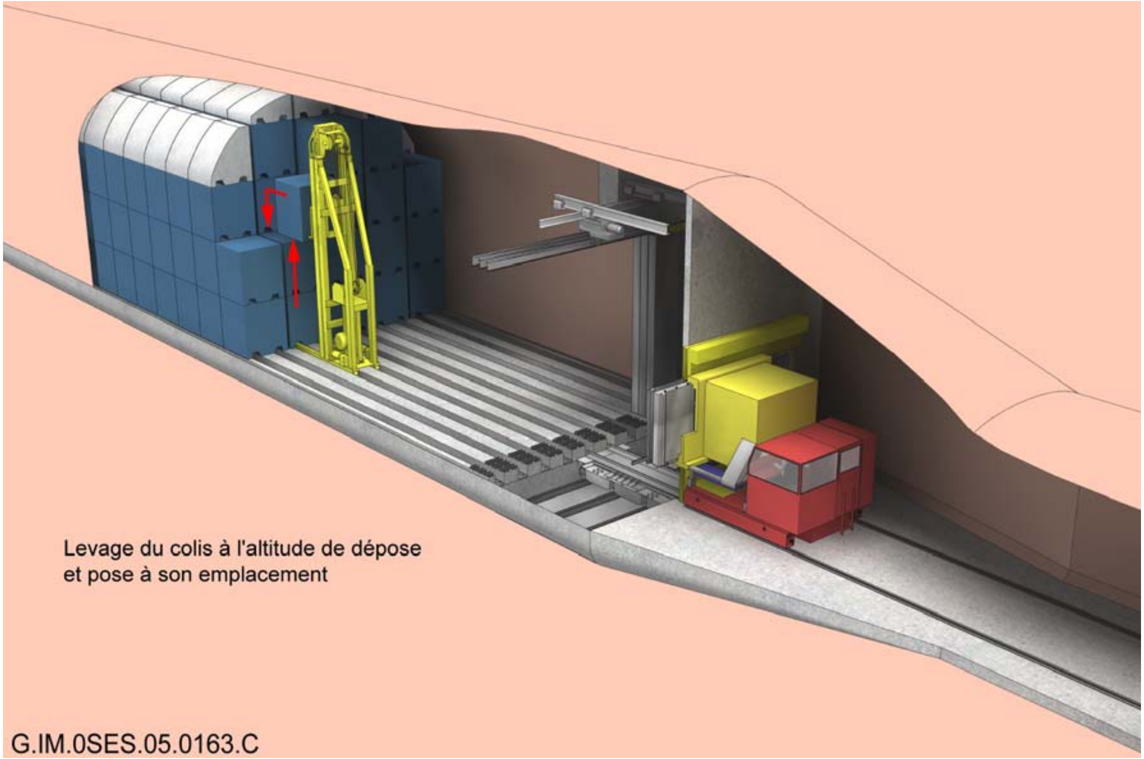

Figure 5.6.7 Lifting the package to the deposition height and placing at its location

5.6.4 Risks associated with B waste package emplacement in the cell

In spite of the safety systems planned, the risk of the package handling equipment catching fire or a package dropping during emplacement (or future retrieval) cannot be ruled out. Such events would be likely to have radiological consequences if they were to affect the confinement maintenance function of the radioactive materials provided by the primary package(s). Simulations of such events have been carried out as part of the repository studies for a clay medium [27]. They can be transposed to the repository in a granite medium as the trolley and waste packages would be identical[33].

5.6.4.1 Consequences of a fire of the handling trolley

Simulation has been made of a fire breaking out on the handling equipment while emplacing a disposal package. The fire's characteristics (power, duration) have been defined on the basis of an electrically powered vehicle equipped with a thin heat shield between the motor compartment (the fire source) and the handling compartment. The package being considered for this case study is a B2 package because its bituminous matrix is the most vulnerable to temperature rise.

The simulation demonstrates that in these conditions, the disposal package envelope would be subject to a maximum temperature of about 500°C, which would lead to the concrete suffering superficial flaking without bursting. The primary packages would be subjected to temperatures below 100°C for the two most exposed primary packages and this would not damage their metal envelope or affect their bituminised waste content. Their average temperature would only be about forty degrees (cf. Figure 5.6.8).

[33] The only difference would be the package handling height, which would be slightly higher in the granite medium disposal tunnels.

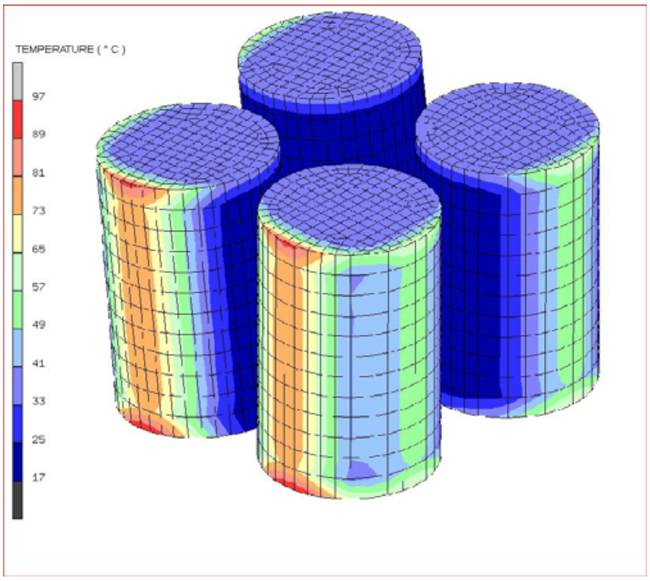

Figure 5.6.8 *Temperature fields of primary packages of bituminised waste (B2.1) in the event of a fire in the disposal cell*[34]

The risk of direct combustion of the waste would appear to be ruled out in view of the protection provided by the concrete envelope of the disposal package. Likewise, spontaneous combustion of the embedding bitumen, which takes place at about 350°C, could not occur. These results lead us to rule out the hypothesis of radioactive material release as a result of a fire on a handling vehicle.

5.6.4.2 Consequences of a disposal package falling

The drop scenario applied is based on the hypothesis that a disposal package tilts and turns over, followed by a 6-metre drop on to the rim of the lid (cf. Figure 5.6.9) for different types of B[35] waste packages (B2, B5). This dropping configuration is more severe than "flat drop" or “edge drop" scenarios.

The simulations set out firstly to estimate the mechanical consequences of the drop for the primary package before assessing its potential radiological consequences. Their results are due to be validated by full-scale disposal package drop tests during 2005.

Of all the drop scenarios examined, a B2 disposal package drop is the one that leads to the greatest deformations for the primary packages contained, with deformation factor values that may peak at 20% for the primary package closest to the point of impact (cf. Figure 5.6.10). The primary package deformation factors for B5 disposal packages are below 10%.

34 The ignition source is placed laterally to the left of the packages and the concrete envelope of the disposal package is not illustrated.

35 The B2 disposal package is a standard container in the simulations, while the B5 disposal package is a container with greater retention capacity (cf. section 5.3).

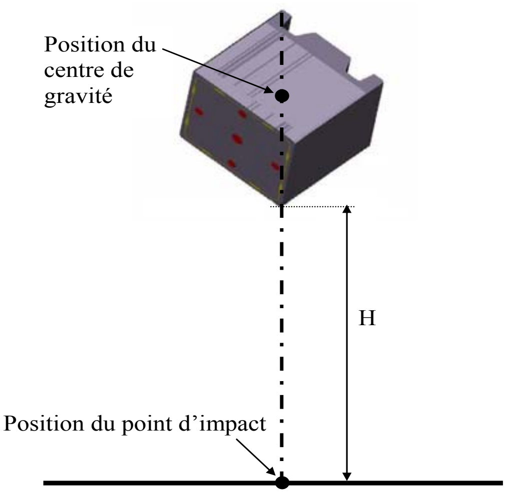

Figure 5.6.9 Scenario of a disposal package dropping onto the cell floor

Simulations demonstrate that the concrete envelope of the disposal package absorbs about 90% of the kinetic energy generated by the package drop. The primary packages take the remaining 10%.

In both instances, these values are below the primary package envelope burst level (35%).

However this conclusion needs to be tempered by the fact that simulations cannot predict with certainty the behaviour of the primary package lid crimping (B2 packages) and the behaviour of the seal between the container's body and lid. Real drop tests will remove these uncertainties and if necessary indicate the required adaptations to disposal package design.

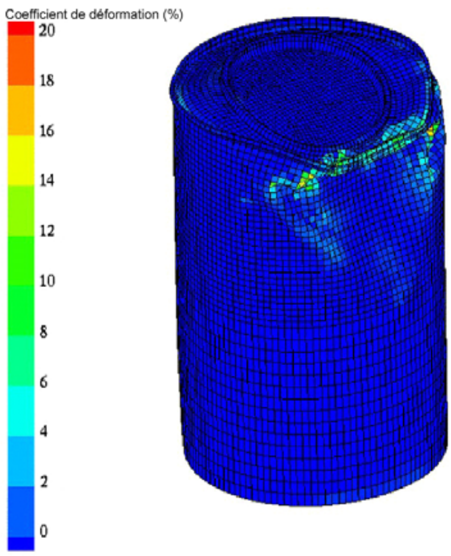

Figure 5.6.10 Plastic deformations of the B2.1 primary package (6-metre drop on an angle[36])

Given the results of the above-mentioned simulations and modelling uncertainties, radioactive material release scenarios have been envisaged for disposal packages containing B2 waste.

They show that the dose received at the surface installations, 500 m from the air extraction shaft would be less than 0.001 mSv. This negligible dose would not have any consequences for man and the environment.

[36] The primary package is shown in its initial pre-drop position, that is, vertical with the lid upwards to enhance visualisation of the deformations.

5.6.4.3 Summary

The design measures made for the B disposal package handling vehicles should avoid fire outbreak on these vehicles or packages from dropping during emplacement. Simulations have confirmed that if such events were to occur they would have no radiological consequences. Real drop resistance tests are planned to corroborate these results.

5.6.5 Closing the structures

The closure operations can be carried out in stages, in line with a reversibility rationale. This scheme provides an adaptable and flexible process management.

The closure operations consist of constructing seals and backfilling to limit water circulations (and compartmentalising the repository).

A first stage consists of closing the disposal cells. The following stage consists of closing the modules.

Than subsequent decisions to close the whole of the repository zone, connecting drifts and shafts may follow (see chapter **4**).

5.6.5.1 Closure of a disposal cell

The closure of a disposal cell comprises the following staged sequence:

- emplacement of a radiation protection shield in front of the last stack of packages in the disposal chamber;
- dismantling all the cell head installations;
- cutting out the sealing zone;
- backfilling the cell head and constructing the internal concrete abutment;
- emplacement of the swelling clay seal core and construction of the external concrete abutment.

These stages may take place discontinuously.

- **Emplacement of the radiation protection shield**

The radiation protection shield is made up of a double row of concrete blocks to avoid diffused radiation. The blocks are emplaced by the package handling disposal trolley. Thus the cell can continue being ventilated as before during and after emplacement of this shield.

- **Dismantling all the cell head and entrance drift installations**

Once the radiological protection screen has been put in place, work can start on dismantling the equipment at the cell head while continuing the benefit of cell ventilation. This dismantling covers all the mechanical installations of the dual-gate system and may be phased. The equipment used to collect and pump the drainage water is maintained in running order.

- **Cutting out the sealing zone**

The section of the entrance drift right above the seal construction zone is over-reamed to cut out and extract the volume of rock damaged by the excavation works. Cutting out the granite may be carried out using ornamental masonry work methods adapted to the underground situation (diamond-tipped disk sawing and jack rock-splitting or boreholes and diamond-tipped cable sawing (Figure 5.6.11, [15]).

Close-up of diamond-tipped cable

Cutting machine

Figure 5.6.11 Cutting ornamental granites with a diamond-tipped cable

- **Backfilling the cell head**

The volume that corresponds to the cell head and the portion of the access drift upstream of the seal is backfilled (cementitious backfill). The backfill is supported by a permanent masonry form constructed in front of the radiological protection.

- **Construction of the cell seal**

The cell seal is made up of a swelling clay core confined by two concrete abutments. Cell seal construction is similar to that of a drift seal (cf. § 4.2.2.6). The clay core is about fifteen metres long.

5.6.5.2 Closure of a disposal module

The closure of a disposal module comprises the following staged sequence:

- the connecting drifts and access drifts downstream of the cell seals are entirely backfilled. It is planned to use low permeability backfill made up of granite aggregates and swelling clay, and to compact it mechanically into place (cf. § 4.2.2.6);
- dismantling all the module operating installations (ventilation, drainage and energy);
- constructing module seals in the connecting drifts downstream of the faults that delimit the host blocks (cf. § 4.2.2.6).

5.7 Functions of the repository components over time

The previous sections have described the various components of a B waste repository, presenting the provisions proposed to create a reversible disposal system, the principles behind their design and dimensioning with a view to long-term safety.

Table 5.7.1 summarises how the various components contribute to the main functions of a repository during the various repository phases.

Table 5.7.1 Repository components during the various repository phases

COMPONENT	PERIOD	MAIN REPOSITORY FUNCTIONS	PROPERTIES HARNESSED
1. GEOLOGICAL MEDIUM: GRANITE			
Part of the granite rock where a disposal cell is installed	During operation	Emplacing (and being able to retrieve) the packages in the cells	Mechanical strength Very low permeability
	After closure	Protecting the disposal cells from the circulation of water Delaying and reducing radionuclides migration to the environment	Chemical properties of the granite water that tend to delay radionuclide migration (reducing environment, etc.)
Granite "block" where the disposal module is installed	After closure	Delaying and reducing radionuclides migration to the environment Protecting also the disposal cells from water circulation	Low permeability (at a distance from the significant water-conducting fractures) Low underground hydraulic gradients Chemical properties of the granite water that favour the delay of radionuclide migration (reducing environment, etc.)
The repository hosting part of the granite massif	After closure	Isolating the waste, protecting the disposal cells from water circulation Protecting also the disposal cells from the circulation of water Delaying and reducing radionuclides migration to the environment	Underground installation of the repository (500 m taken as reference) Low permeability (at a distance from the major faults, the main vectors of water in the massif) Low underground hydraulic gradients Chemical properties of the granite water that tend to delay radionuclide migration (reducing environment, etc.) Geo-dynamic context that ensures favourable repository conditions are maintained in the long term, particularly erosion phenomena

COMPONENT	PERIOD	MAIN REPOSITORY FUNCTIONS	PROPERTIES HARNESSED
		2. ENGINEERED COMPONENTS	
Connecting drifts between access structures from the surface and repository modules	Before and during operation, reversible repository management	Emplacing (and being able to retrieve) the packages in the granite	Dimensions adjusted to package throughput Drift equipment that ensures safe operation
Drifts in the modules (cell access drifts and exploratory structures)	Before and during operation, reversible repository management	Emplacing (and being able to retrieve) the packages in the granite	Suitable dimensions for package throughput and granite characteristics Drift equipment that ensures safe operation
Backfill and seals of the disposal module drifts	After closure	Protecting the disposal modules from the circulation of water Delay and reduce radionuclides migration to the environment	Low permeability Swelling capacity Radionuclide retention
Cell: **disposal cavity** **B waste:** **disposal tunnel**	During operation and reversible repository management After closure	Emplacing (and being able to retrieve) the packages in the granite Restricting radionuclides release by the primary packages and immobilise them	Dimensions adapted to the granite fracturing and handling technologies Properties of the various components: packages, engineered barriers, backfill
Concrete B waste disposal packages	During operation and reversible repository management After closure	Emplacing (and being able to retrieve) the packages in the granite Restricting radionuclide release by the primary packages and immobilising them	Mechanical strength Suitable dimensions for handling them Suitable permeability and concrete formulation for the different types of waste: type B2 hydrogen-producing waste or type B5.2 waste that does not produce hydrogen gas Concrete alkalinity conducive to immobilising the radionuclides
Backfilling the B waste disposal tunnels	After closure	Protecting the disposal modules from water circulation	Capacity to fill the tunnel voids (characteristics as yet undefined)

6 C waste repository zone

This chapter describes the design of the vitrified C waste repository zone. It specifies how this design makes allowance for the physical characteristics of C waste, in particular its thermal and chemical characteristics, and fulfils the safety functions defined in chapter 3.1

It then goes into detail on the options for the disposal packages, disposal cells and repository zone architecture.

Finally it describes how these options can be implemented for a reversible disposal process that meets operational safety and security requirements.

6.1 Primary C waste packages

Vitrified waste is a product of spent fuel reprocessing. This includes mainly fission products and minor actinides (neptunium, americium and curium) formed by nuclear reaction and contained in the spent fuel, which are separated from the uranium and plutonium during the reprocessing operation. They are separated and incorporated into a glass matrix. The manufactured glass is poured at temperature into a stainless steel container. Radiological activity is spread homogenously through the mass of vitrified waste.

Vitrification has been developed in several pilot French facilities operated by the CEA, including the pilot PIVER facility now shut-down, then implemented industrially in three shops operated by COGEMA: Marcoule vitrification shop (AVM), started in 1978, and vitrification shops R7 and T7 at La Hague, started in 1989 and 1992 respectively.

Vitrified waste characteristics, particularly their activity and heat rating, depend on several parameters: (i) initial characteristics of solutions of fission products and minor actinides from fuel reprocessed in these facilities, (ii) varying degrees of concentration of fission products in the glass and (iii) the age of the waste.

It was therefore decided to make a distinction between several vitrified C waste package groups, respectively (i) the oldest glass productions, (ii) current glass productions or those planned in the short term, (iii) provisional glass productions, including UOX/MOX glass and UOX glass with the hypothesis of incorporating a small fraction of plutonium.

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The heat rating levels associated with these different package sets are given in Figure 6.1.1.

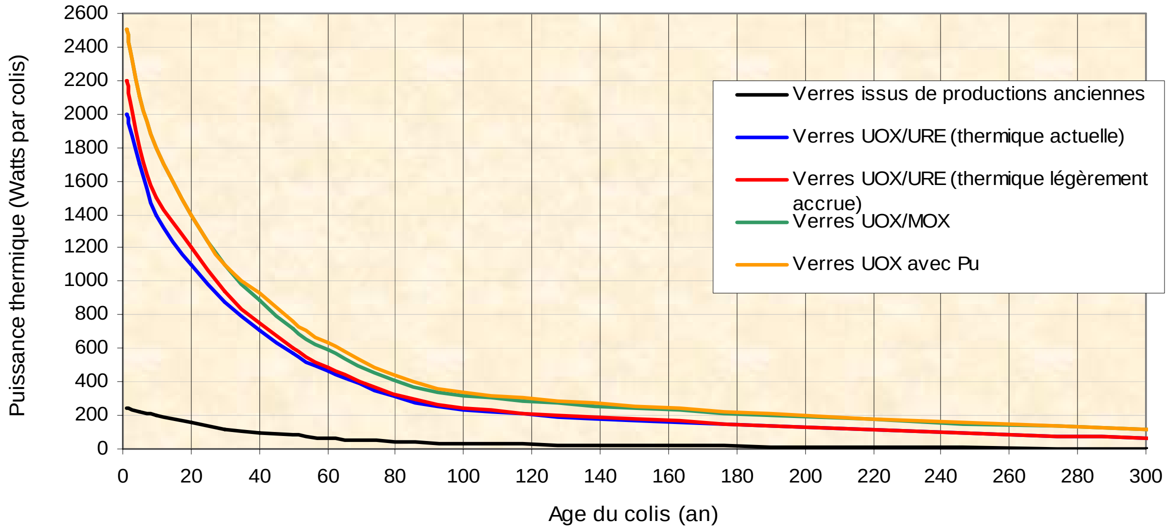


Figure 6.1.1 Heat ratings of vitrified waste packages versus time

The inventory model, presented in the following table, is founded on these elements.

Table 6.1.1 Vitrified C waste reference packages

Reference packages		Package families included in the reference packages and reference sub-packages		
NLev. 1	**Lev. 2**	**Lev. 1**	**Lev. 2**	**Lev. 1**
C0	C0.1	Vitrified PIVER waste	Glasses of legacy fission products and actinides	§ 6.1.1
	C0.2	Vitrified R7T7 UMo waste		
	C0.3	Vitrified AVM waste		
C1		"Current thermal" vitrified waste	Glasses of fission products and actinides from UOX/enriched uranium fuel reprocessing	§ 6.1.2
C2		"Future thermal" vitrified waste	Glasses of fission products and actinides from UOX/enriched uranium fuel reprocessing with higher heat rating	
C3		UOX/MOX vitrified waste (being considered)	Glasses of fission products and actinides from UOX and MOX fuel reprocessing	§ 6.1.3
C4		Vitrified Pu waste (being considered)	Glasses of fission products and actinides incorporating Pu	

6.1.1 Glass packages from former productions

This package group (reference package C0) includes (i) glass packages containing solutions of fission products from the reprocessing of fuel from Natural Uranium Graphite Gas reactors (Sicral-type UNGG fuels) and fuels from the Phenix fast neutron reactor in the PIVER facility, (ii) glass packages containing solutions of fission products known as UMo, produced by UNGG fuels recovered previously on the COGEMA La Hague site and now stored and (iii) glass packages produced since 1978 in the COGEMA Marcoule vitrification shop (AVM glasses), containing fission products and actinides produced mainly by reprocessing of UNGG fuels. The majority of packages are of the latter type.

These packages differ in their chemical content, depending on the composition of the glass matrix used, their radiological content and subsequently their heat rating, and in container geometry.

PIVER packages, produced between 1969 and 1981, are predominantly formed of UNGG glasses. The vitrified waste is conditioned in stainless steel containers of the same diameter but with different heights (see Figure 6.1.2).

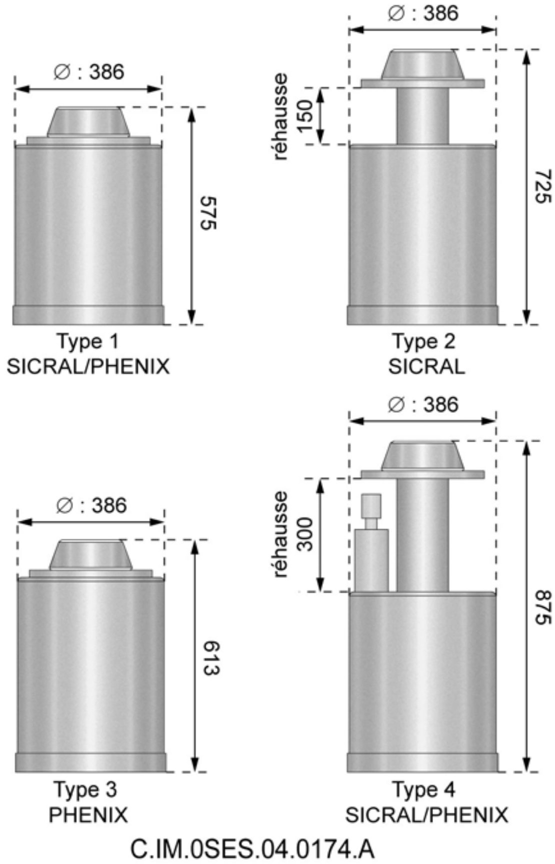


Figure 6.1.2 PIVER containers

These are low-capacity packages (39 or 45 litres depending on the type of container) containing glass weighing between 20 and 120 kilograms. An average weight of 72 kilograms of vitrified waste per package has been adopted for the study, with a total average weight of 90 kilograms for a full package (vitrified waste + container).

The radiological activity of the packages has decreased considerably given their date of production. It continues to be dominated by two medium-life fission products - strontium-90 (^{90}Sr) and caesium-137 (^{137}Cs). These generate residual thermal release, currently on the order of about thirty watts per package. The equivalent β-γ dose rate in pseudo-contact (5 cm) with the package is around 70 Sv/h, becoming around 45 Sv/h by 2025.

The UMo glass waste packages correspond to future conditioning of solutions of existing fission products; these are produced by UNGG fuels recovered in the COGEMA UP2-400 plant at La Hague. The chemical nature of the solutions requires a specific glass to be formulated and modifications to process equipment, particularly the vitrification furnace. Following the hypotheses adopted, the average weight of the conditioned waste is 400 kilograms per package. Strontium-90 and caesium-137 also play an important role here in the radiological activity of the packages, although this has decreased tremendously given the age of the solutions. Subsequently, the residual thermal release is currently around 70 watts per package. The equivalent β-γ dose rate in pseudo-contact (5 cm) with the package is around 15 Sv/h, becoming around 10 Sv/h by 2025.

This waste will be conditioned in a stainless steel container identical to the one used today in the COGEMA R7 and T7 vitrification shops at La Hague. This container, known as the Standard Vitrified Waste Container (CSD-V), is presented in Figure 6.1.3.

Figure 6.1.3 Standard vitrified waste container (CSD-V)

The AVM glass waste packages group all the vitrified waste produced since 1978 in the COGEMA vitrification shop at Marcoule. As indicated above, the vitrified solutions come in the main from reprocess UNGG in the site's UP1 plant. Note that four different glass formulations exist for one or more vitrification campaigns.

The vitrified waste is conditioned in a stainless steel container presented in Figure 6.1.4

Figure 6.1.4 AVM vitrified waste containers

The average weight of each package is 410 kilograms, including 360 kilograms of vitrified waste. The packages' heat rating comes principally from medium-lived fission products, ^{90}Sr et ^{137}Cs. A heat rating of 155 watts by 2025 has been adopted for the study. Note that this heat rating is higher than for the previous packages; it is nevertheless significantly less than those of the C packages presented below. The equivalent β-γ dose rate in pseudo-contact (5 cm) with the package, around 235 Sv/h, will be around 150 Sv/h by 2025.

6.1.2 Vitrified waste packages from current or planned productions in the short term

These packages contain solutions of fission products produced by reprocessing PWR UOX/enriched recycled uranium fuels in the COGEMA La Hague plants, conditioned as glass in a so-called CSD-V stainless steel container (see Figure 6.1.3). It is assumed that production and conditioning of the waste will occur after an average fuel storage time of eight years, after unloading from the reactors. The conditioned waste weighs 400 kilograms on average per package.

A first group of waste packages relates to current industrial productions from a heat transfer viewpoint. Following the hypotheses adopted, the vitrified waste is made up of a mixture of solutions of fission products from UOX1 (average combustion rate of 33 GWj/t), UOX2/enriched recycled uranium (average combustion rate 45 GWj/t) and UOX3 (average combustion rate 55 GWj/t) fuels.

A second group of waste packages relates to packages with a slightly-increased heat rating. The vitrified waste is made up of a mixture of solutions of fission products from UOX2/enriched recycled uranium and UOX3 fuels, with, as previously, average combustion rates of 45 GWj/t and 55 GWj/t respectively.

The heat ratings of these packages are illustrated in Figure 7.1.1 (blue and red curves). They raise the issue of storage time before being placed in a repository and of repository module design, to limit the temperature to acceptable levels

The equivalent β-γ dose rate on pseudo contact (5 cm) with the packages is about 240 Sv/hr after 60 years' cooling.

6.1.3 Future hypothetical packages

These waste packages relate to potential glass productions on the COGEMA La Hague site. These packages have been defined in the scenarios adopted for the study on the assumption that production and conditioning of the waste will take place, as for the previous glasses, after an average fuel storage period of eight years, after unloading from the reactors. Note that other possibilities could be envisaged.

Here also, the conditioned waste weighs 400 kilograms on average per package. The containers are similar to the ones presented above (CSD-V).

A first waste package group describes glasses produced by conditioning solutions of fission products from UOX and MOX fuels. Their make-up is defined as a mixture of 15% MOX and 85% UOX2.

A second waste package group describes vitrified waste from reprocessing UOX fuels, which contains a small additional plutonium load. The plutonium incorporation rate in the glass is fixed specifically at one percent, about 4 kilograms per package. The incorporated plutonium comes from the UOX2 fuels.

The heat ratings produced by these packages are illustrated above in Figure 3.2.15 (green and orange curves).

The equivalent β-γ dose rates in pseudo-contact (5 cm) with the packages will be 235 Sv/h for the UOX/MOX glasses and 265 Sv/h for the glasses with plutonium respectively, after a sixty-year cooling period.

6.1.4 Production scenarios

Table 6.1.2. Table 6.1.2 gives the numbers and volumes of C waste primary packages arising from the study scenarios presented in Table 6.1.2.

Table 6.1.2 *Number and volume of C waste primary packages*

Reference package	Production sites	Scenario S1a		Scenario S1b		Scenario S1c		Scenario S2	
		Number	Volume (m^3)	Number	Volume (m^3)	Number	Volume (m^3)	Number	Volume (m^3)
C0.1	CEA	180	10	180	10	180	10	180	10
C0.2	COGEMA La Hague	800	140	800	140	800	140	800	140
C0.3	COGEMA Marcoule	3 140	550	3 140	550	3 140	550	3 140	550
Total of C0		**4 120**	**700**	**4 120**	**700**	**4 120**	**700**	**4 120**	**700**
C1	COGEMA La Hague	4 640	810	4 640	810	38 350	6 710	4 640	810
C2		990	170	27 460	4 810	0	0	5 920	1 040
C3		13 320	2 330	0	0	0	0	0	0
C4		13 250	2 320	0	0	0	0	0	0

6.1.5 Throughput hypotheses for primary package reception

Throughput hypotheses for the C waste primary packages received on a repository site have been drawn up in line with the total quantities of inventoried packages so that the operating methods for a repository along with its drifts and access shafts can be studied.

Chronologically, the C0 reference packages will be the first vitrified C waste packages available technically for repository disposal in view of their heat rating. 400 packages per year is the rate adopted for the hypothesis, corresponding to a resorption of storage facilities over ten years or so.

It is presumed that there will be an interim storage period for the other vitrified C packages prior to disposal, to reduce their heat rating. It emerges that a reasonable interim storage period would be about sixty years for the least exothermic waste (for example C1 and C2) and about a century for the most exothermic waste (such as C4), in the context of the designs studied and thermal criteria currently envisaged to allow for the variability in the temperature of French granites.

Whatever happens, the hypothesis of an annual reception rate of 600 reference packages C1 to C4 (in the case of scenario S1c, 700 packages per annum for reference packages C1) would enable the stockpiles in interim storage to gradually be reduced as and when the packages present the same heat rating through radioactive decay

6.2 Safety options for the design of a C waste repository

The general principles underlying repository design in a granite medium have been described in section 3. They call for long-term safety functions, of waste emplacement and management of the installations with a view to reversibility. The following principles are adopted for long-term safety:

- harnessing the favourable properties of the granite medium;
- designing engineered components that are complementary and redundant to the granite medium;
- limiting disturbances to the granite by the repository.

Allowance must be made for the radioactivity and thermicity of vitrified C waste and the particular nature of their conditioning material - glass - in order to apply these general principles to their disposal.

Managing the heat released by C waste is a fundamental element when designing for its disposal. To control repository behaviour, allowance must be made for the temperature criteria that limit the number of packages per cell (management of heat rating) and the spacing between cells (managing heat dissipation).

The vitreous nature of the packages is another element governing the design of a C waste repository. This is because the glass matrix needs to be mobilised to retain the radionuclides. Thus the glass needs to be protected, namely set in conditions that limit the dissolution of its constituent silica. Accordingly the design is based on a watertight over-pack made of steel with low alloy content, that prevents the water from coming into contact with the glass while the temperature at the core of the glass is over 50°C, since glass alteration is enhanced above this limit. This over-pack is surrounded by a swelling clay-based engineered barrier.

These design options for vitrified waste are similar to those studied abroad. For the two geological media in question (granite and clay) the Japanese have adopted one design in which the primary package (CSD-V) is protected by an over-pack in steel with low alloy content [43]. The packages are disposed of in horizontal or vertical cells approximately 2.20 m in diameter. An engineered barrier made up of prefabricated swelling clay blocks (with 30% sand) is placed between the disposal package and the rock (Figure 6.2.1). The engineered barrier may be 30-70 cm thick depending on the geology in question and the strength assigned to the over-pack.

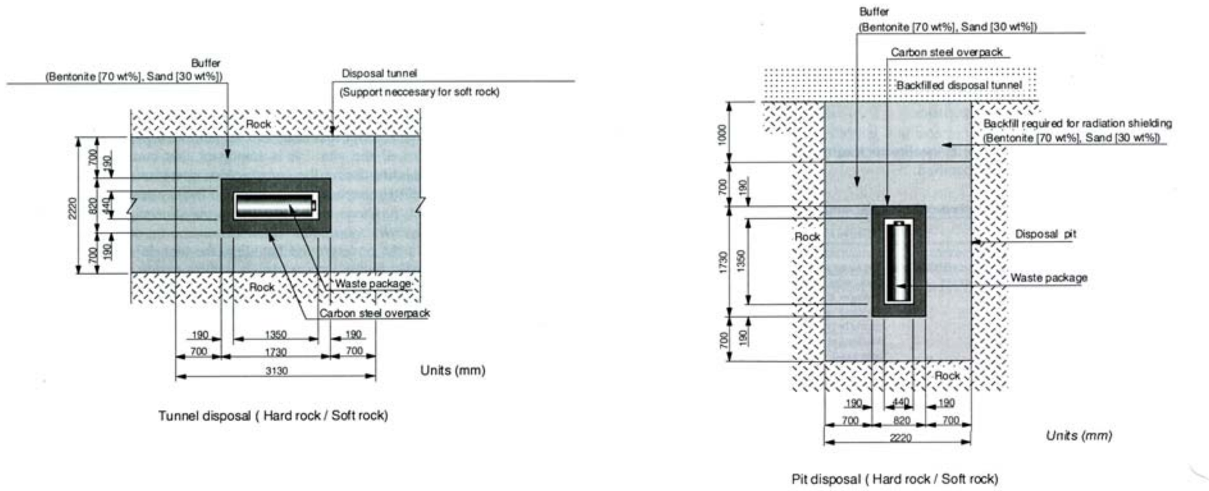


Figure 6.2.1 JNC (Japan) designs for the disposal of C [43]

6.2.1 Harnessing the favourable properties of the granite medium

Harnessing the very low permeability properties of the granite, its radioactive element retention properties and its mechanical strength (cf. § 3.3) call for adapting the repository architecture to the granite fracturing. This adaptation is on two levels: firstly on the scale of the cell by aiming for installation in those parts of the rock where there is very little fracturing, and secondly at the scale of the repository module, by developing it at a distance from water-conducting faults.

6.2.1.1 The reference cell: small disposal boreholes

The principle of installing the disposal cells in a very low-permeability and slightly fissured granite rock avoids cell-bisection of the fractures that could conduct water.

The orientation and distribution of minor fracturing likely to locally affect the very low permeability of the rock govern the choice between horizontal (tunnels) or vertical (shaft) cells in the granite. The analysis of minor fracturing in the French geological context shows that fracturing is very often vertical [44].

Hence the vertical borehole design, which statistically limits the number of potential minor fractures at the walls of the disposal cells, has been retained as the reference.

A horizontal design could also be envisaged for a site where fracturing distribution turns out to be mainly horizontal. It would be based on the same basic design principles, namely an over-pack and a clay engineered barrier.

As short boreholes statistically bisect fewer fractures than longer ones, disposal borehole length is another element that can be adapted to the fracturing of the granite.

The disposal boreholes are dead-end. There is no access from the borehole bottom to the repository module drifts. This arrangement reduces the possibilities of water circulating in the disposal boreholes. Towards the surface they open out onto a handling drift dimensioned for package emplacement in the disposal boreholes.

Thus the option proposed for the C waste cell is a borehole with a limited length of about 12 metres, less than 2 metres in diameter, that enables a clay engineered barrier to be placed between the disposal packages and the rock (cf. § 6.4).

6.2.1.2 Installing repository modules in "blocks" of granite away from water-conducting faults

The repository modules are installed in the granite away from the water conductive faults which cannot be bisected by the module drifts and primarily the handling drift (Figure 6.2.3). These faults generally extend over several hundred metres or more.

The installation of the modules in the granite blocks away from the water-conducting faults does not rule out the presence of minor rock fracturing inside the module. However its hydraulic conductivity must be low enough not to jeopardise the safety functions.

The application of this principle calls for defining criteria concerning the properties of these faults and their *in situ* surveying and characterisation prior to installing the modules in the repository host massif (cf. § 4.3.1).

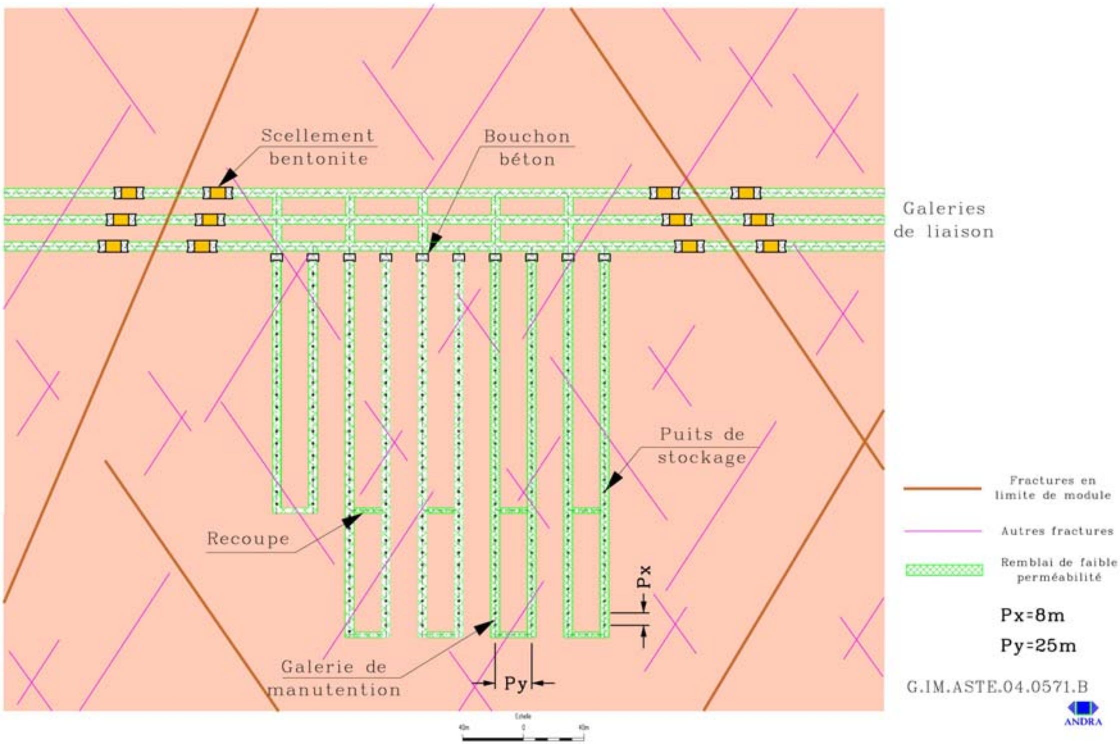


Figure 6.2.2 Principles of installing a repository module away from water-conducting faults

This principle, which results in dividing up the C waste repository zone architecture in line with the fracturing, is along the same lines as those considered abroad (Figure 6.2.3).

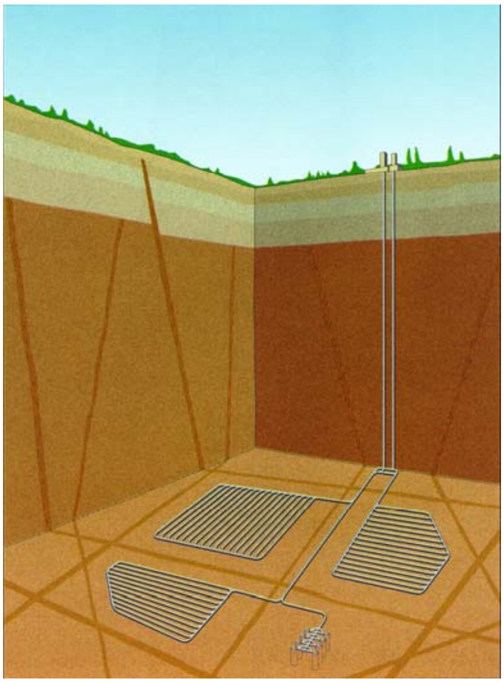


Figure 6.2.3 *Switzerland – Architectural plan of a repository in the granite adapted to the fracturing (Nagra data [45])*

6.2.2 Designing engineered components that are complementary and redundant to the granite medium

The architectural options described above aim to make the most of the favourable properties of the granite medium. Dividing up the repository zone may lead to the connecting drifts between modules crossing water-conducting faults. At a smaller scale, the options studied do not rule out the presence of minor fracturing inside the module.

With respect to the main long-term safety functions, it means that engineered barriers must be provided to be complementary and redundant with the granite medium.

Thus low-permeability backfill is planned for the handling and connecting drifts, and seal emplacement at the module edges will complement the cell and module installation principles.

Furthermore, a steel over-pack, that will remain watertight for a long enough period, surrounded by a clay engineered barrier, will contribute towards the primary waste package borosilicate glass performance.

6.2.2.1 Redundant backfill and drift seals

Two complementary and redundant devices can be envisaged to protect the repository modules from circulations of water emanating from the faults bisected by the drifts: backfilling the drifts with a low-permeability material and the emplacement of very low permeability seals.

The hydro-geological context of the granite site and the permeability of the rock govern the requirement for backfill permeability.

A value of 10^{-10} m/s is the common reference value for all backfill emplaced in the drifts regardless of waste type in the modules, (B waste: see section 5.2; C waste and if relevant spent fuel: see chapter 7).

In Sweden, dedicated laboratory experiments on backfill emplacement and *in situ* performance assessments have been conducted [18]. These experiments have demonstrated the technical feasibility of using low permeability backfill partly made up of swelling clay in the drifts (cf. § 4.2.2.6).They also enhance our understanding of how these performances are acquired as the water returns to the backfilled drifts and particularly when it comes into contact with the backfill and the drift roof.

In situ laboratory experiments have been carried out on very low permeability seal emplacement in drifts in Canada [16]. This programme of experiments has demonstrated the possibility of constructing seals from very low permeability swelling clay (equal to or less than 10^{-11} m/s).

The precise positioning of the seals will be governed by the geometry of the "blocks" and faults surrounding them.

6.2.2.2 A watertight over-pack during the thermal phase

The main function of the over-pack is to protect the glass package from water in the cells during the thermal phase. Glass dissolution models demonstrate the importance of temperature on the kinetics of alteration [47]. Between 50 and 90°C the dissolution rates increase by a factor of 15-30 (depending on the model and reference package in question).

Furthermore, because of our limited knowledge of the thermodynamic data that govern chemical equilibriums as indicated in chapter 5, control of radionuclide behaviour in the water following release by the packages depends on temperature: in practice this behaviour can only be described with a margin of manageable uncertainties for temperatures below 80°C [47].

Consequently, the design principle for the over-pack relies on it being watertight for at least the whole period when the temperature is in over about 50°C (cf. § 6.3.2), that is roughly one millennium.

The choice of a metallic material guarantees confinement at high temperature and limits the chemical impact on the glass (as opposed to concrete whose behaviour is hard to control at temperatures over 70-80°C - cf. chapter 5- - whose alkaline pH may alter glass matrix confinement performances in the long term).

When a corrosion "consumable" thickness is incorporated into the over-pack design, steel is a suitable material for the desired period of water-tightness. One millennium does not call for the use of materials with greater corrosion resistance (i.e. passive or thermo-dynamically stable such as copper [37]). Additionally, steel provides the disposal package with mechanical strength (cf. 6.3.1).

6.2.2.3 A swelling clay engineered barrier

The emplacement of a swelling engineered barrier in the disposal cells is adopted in all the C waste repository options for a granite medium studied abroad (Figure 6.2.1 and [45]). This is because it can back up the role of the other components in the following ways:

- swelling provides a close interface between the granite rock and the engineered barrier, that tends to sharply reduce the water-conducting properties of the damaged zone created by cell excavation [47];
- the swelling clay engineered barrier prevents water circulating in the cells regardless of whether it comes from the granite or the access drifts; thus transfers of species in solution will be restricted to diffusion;
- the swelling clay engineered barrier limits mechanical deformations in the cell that would derive from the long-term alteration of the metal components, primarily the over-packs
- the swelling clay engineered barrier contributes to maintaining a favourable chemical environment for the glass, primarily by buffering chemical interactions firstly between the various cell components and secondly with the granite.

Knowledge of a swelling clay engineered barrier behaviour leads to limiting the temperature at the hottest point of the engineered barrier during the thermal phase to 90°C. This allowance is built into repository module dimensioning by adjusting the spacing between disposal cells to the thermal characteristics of French granites (cf. § 6.5).

In addition, various possibilities of clay engineered barrier formulation are another way of managing the variability of the chemical compositions of French granite water.

6.2.3 Limiting disturbances to the granite by the repository

The hydro-geological and hydro-geochemical disturbances of a granite rock, arising from the excavation of the underground installations are managed as part of the general disposal process (cf. § 6.3).

The other mechanical, thermal or chemical disturbances likely to affect the properties of a C waste repository are limited either by structure dimensioning or by disposal process management.

6.2.3.1 Limiting mechanical damage to the rock during excavation work ("EDZ")

Excavating the underground structures is likely to incur damage to the granite rock at the wall by creating fissures. This damage creates a potential path for water circulation. It is technically possible to limit its extent.

Experiments in foreign underground laboratories ([14] [46]) have demonstrated the possibility of controlling this damage. Drilling techniques in particular result in very slight damage and may be used to excavate the small disposal boreholes. At this stage the choice of drift excavation techniques is open, if the possibilities of adapting blasting methods to the context are exploited. Limiting the damage incurred by blasting methods involves using familiar techniques such as smooth blasting, pre-cutting and excavating in divided sections. These low-damage techniques are based on reducing the instantaneous quantity of energy liberated in the rock by the explosives (spreading the firing sequence) and distributing blasting pattern explosives in line with their impact (shattering) and gas energy.

Added to this, sawing methods for cutting out the damaged zone such as those used in ornamental stone quarries can be used at the sites of the seals.

6.2.3.2 Limiting thermo-mechanical deformations in the granite

The granite that hosts C waste is likely to warm up and thus be subjected to thermo-mechanical stresses, primarily during the early phases of the repository when temperatures and temperature gradients are at their highest [47]. These stresses may result in granite medium deformations primarily where it is weakest, namely at the fractures or faults. Thus these deformations may trigger off modifications to fracture hydraulic and transfer properties. The intensity of these deformations must therefore be controlled by suitable design measures

The presence of a swelling clay engineered barrier contributes in limiting both the maximum temperature in the rock at the wall of the shaft and the thermal gradients between cells. The engineered barrier characteristics and the spacing between the cells are thermal dimensioning parameters of the repository.

6.2.3.3 Limiting chemical disturbances

The corrosion of the metallic components in the cells, and in particular the over-packs may in the long term cause interactions with the granite water in the fractures. These chemical modifications may be limited by always placing a swelling clay engineered barrier around the packages

6.3 C waste over-packs

6.3.1 Selected design principles

The choice of the constituent metal of the over-pack uses the same analysis as for the clay medium. Thus the reference is non-alloy steel.

The corrosion process of non-alloy steel is well known. Concerning aqueous corrosion, a set of experimental results and models show that generalised corrosion is the dominant mechanism in the medium and long term. The rate of generalised corrosion, reflecting the thickness of corroded metal over time, can be quantified on the basis of models that have been validated experimentally. Localised corrosion, by pitting or fissuring, may be observed on these materials, especially in the presence of oxygen. However the localised corrosion rate observed experimentally over short periods decreases faster than that of generalised corrosion, hence its relative importance reduces with time. Finally, the risks of specific corrosion such as corrosion under stress or hydrogen embrittlement, are of secondary importance when compared to other corrosion mechanisms. Iron archaeological analogies over 2000 years old back up the assessment of the corrosion processes and provide a time reference for the durability of the steels over a significant time scale.

Furthermore the non-alloy steel corrosion models are tolerant to the water chemistry. They are also tolerant with respect to composition of the metal, its structural status and surface condition. This element limits the risk of consequent defects degrading the water-tightness and durability of the over-packs, especially at the welds. Moreover, very good welding results are achieved on non-alloy steels and there is ample feedback on proven welding techniques on heavy thicknesses.

They are preferable to passive alloys. Although they offer good corrosion resistance, these materials are sensitive to environmental conditions and could thus be subject to pitting, which is harder to control over the long term.

6.3.2 Description and dimensioning

This section describes the disposal packages comprising a non-alloy steel over-pack that holds a C waste primary package. Its design aims at achieving the expected durability performance and is consistent with the handling processes.

6.3.2.1 Description of the disposal packages

Two disposal packages with standardised dimensions deal with all C waste primary packages:

- an external length of 1291 millimetres and a diameter of 615 millimetres for both the C0.1 (PIVER vitrified waste) and C0.3 (AVM vitrified waste) reference packages;
- a length of 1607 millimetres and a diameter of 550 millimetres for the C0.2 (CSD-V R7/T7 /UMo), and C1-C4 reference packages produced at La Hague.

The mass of these disposal packages (over-pack and primary package) varies from 1500 kg (type C0.1) to 1800 kg (types C0.2 and C1-C4)

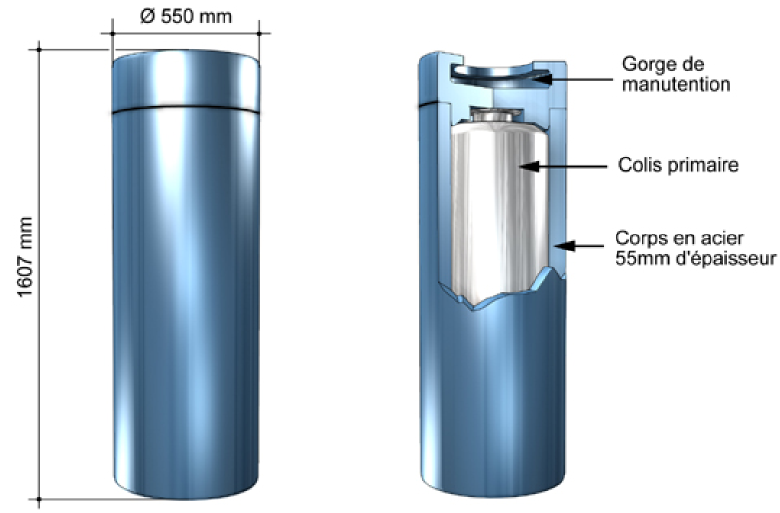


Figure 6.3.1 Principle of the C waste over-pack

The over-pack comprises a body and a lid, welded together through their whole thickness once the primary package has been inserted in the body.

- **The body**

The over-pack body comprises a cylindrical shell 55 millimetres thick (see below). The base has an effective thickness of 77 or 83 millimetres depending on the type of reference package. It can be manufactured in a single piece, without welding, so that the body has a built-in base.

Having compared the various possible grades of steel[37], preference is for the P235 type. Its low carbon content gives it good weldability. Its structure is not very sensitive to cold-cracking during welding, because of its low elastic limit. Furthermore it does not require any preheating or thermal treatment before or after welding respectively.

Incidentally, although this steel is in widespread use, its mechanical characteristics can offer the over-pack the required mechanical strength. Lastly its mechanical characteristics are favourable to controlling corrosion.

- **The lid**

The lid is made from the same grade of steel as the body. Its thickness, about 180 mm, includes a handling interface.

This handling interface comprises a machined groove to allow the grapnel to grip the over-pack

6.3.2.2 Dimensioning

- **Envelope thickness and service life**

The over-pack offers water-tightness for at least one millennium in the repository situation. This period includes the thermal phase when the temperature at the core of the glass may exceed 50°C.

[37] In particular 16MnD5 light-alloy steel used for pressurised water reactor vessels.

A thickness of 55 mm has been retained at this stage in order to ensure this service life, fixed by adding together the following two terms [48] :

- thickness of 28 millimetres for the cylindrical shell and 50 millimetres for the base and lid, to provide enough mechanical strength to withstand a load of 12 MPa (evenly applied); this load covers the interstitial water pressure and the clay engineered barrier swelling pressure;
- over-thickness of 27 millimetres, that is a penalising modelling assessment of the loss of substance through generalised corrosion. This thickness could be adjusted according to water chemistry of a specific granite.

- **Compatibility with the handling facilities**

Once the disposal packages have been manufactured in the surface installations, they are stored upright. During these operations the handling height is about 1.60 metres. Once the packages have been transferred to the underground installations, they are placed upright in disposal boreholes that are about 9 metres deep.

Dimensioning for these operations entails calculating the thickness of the handling interface and checking package resistance to dropping.

The admissible stress on the material that acts as the over-pack handling interface is adopted from the European Federation of Materials Handling (FEM) calculation rules for lifting apparatus, namely 2/3 of the elastic limit. The dimensions of this interface provide robustness with regard to all the demands that will be made of it during and after the handling operations.

As regards package resistance to dropping, a number of cases have been simulated, such as: (i) dropping on a corner from a height of 1.6 metres and (ii) a vertical drop from a height of 20 metres onto a support presumed to be infinitely rigid. The materials are modelled using a Johnson Cook type law that expresses the plastic behaviour of the materials. The resulting deformations are restricted to the base of the disposal package envelope and will not affect the primary packaging of the waste.

The case of a C disposal package drop (while in its cask) has also been considered during shaft transfer from surface to underground: the maximum deformation of the package envelope would not lead to loss of confinement in the event of the cage dropping 500 m onto a shock absorber at the bottom of the shaft.

Generalised corrosion of the disposal package and cell sleeve has been considered as an impediment to package retrieval. For the purpose of dimensioning the handling pocket in this situation, the hypothesis is made that generalised corrosion (2 mm)[38] would affect all the outer surfaces of the handling interface. To allow for this situation in the dimensioning it has to be checked that the package can be gripped and significant traction effort exerted to overcome the friction and sticking forces caused by the corrosion. The calculation gives a maximum applicable effort equivalent to the elastic limit of about 40 tonnes. Theoretically reaching breaking strength would enable a traction effort of about 85 tonnes with however the concomitant risk of damaging the package handling head up its total deterioration.

6.3.3 Manufacturing techniques

This section covers the disposal package manufacturing techniques that may be envisaged on the basis of comparable existing industrial techniques.

[38] 2 mm of corrosion would lead to product a 3 mm thickness being built up over the original dimensions (the total thickness of the corrosion products being then 5 mm). The geometrical dimensioning of the groove makes allowance for this corrosion thickness so that the residual central space can accommodate the fingers of a specific package retrieving tool.

In a two phase manufacturing, the first one would be factory manufacturing of over-pack components. The second phase includes installation of the primary package of vitrified waste in the over-pack and closure of the disposal package by full thickness welding of its lid. This second phase must be carried out in a series of shielded cells to protect the operators from exposure to the radiation emitted by the waste.

6.3.3.1 Factory manufacturing of the container and its lid

A presentation of the components to be manufactured has already been made. There are a number of possible techniques. They have been proven industrially for the dimensions and steel thicknesses at least equal to those of the over-pack [48]. Andra has conducted a comparative analysis of possible manufacturing techniques and two of them have emerged as particularly suitable:

- the first consists of forging a shell then joining a base to it. To manufacture the shell, a first blank (rough form) is obtained from an ingot, then pierced and finally forged. This technique has been used to produce a spent fuel container demonstrator studied for the clay medium. The base is manufactured separately. The shell and the base are then assembled by welding through the full thickness to form the container body;
- The second technique consists of producing the body in a single part, by boring and drawing. This technique has the advantage of avoiding welding the base. The internal cavity of the container is obtained by deformation of the metal. Boring – drawing consists of obtaining a tube from an ingot. The ingot which is removed from the oven is hot-drilled using a boring machine with a vertical press to obtain the inner diameter, then drawn by rolling, using a horizontal press in an extrusion die to obtain the required outer diameter and length (Figure 6.3.2 and Figure 6.3.3). The part then undergoes heat treatment and is machined to achieve the final dimension.

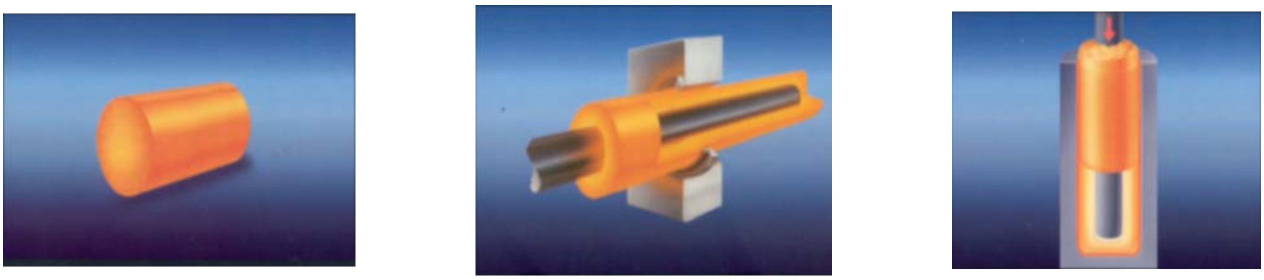


Figure 6.3.2 Ingot – Piercing - Drawing

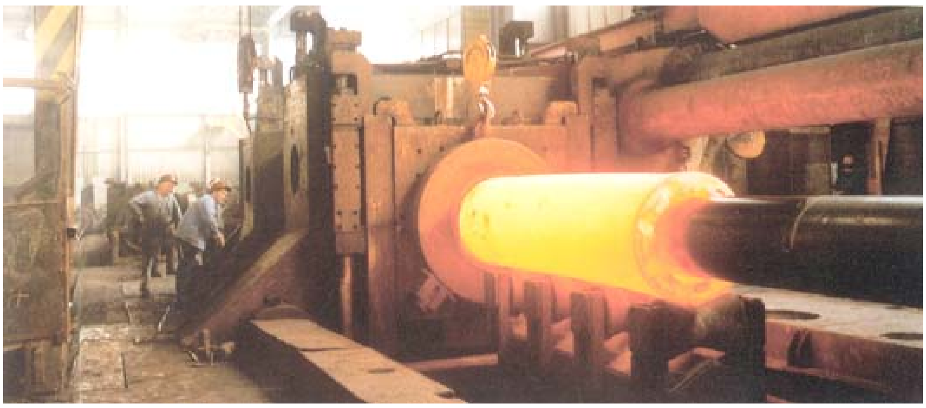


Figure 6.3.3 Drawing operation in a die using a mandrel and a horizontal press

6.3.3.2 Disposal package assembly

Disposal packages are assembled in a shielded cell:

The first operation consists of loading the primary package of vitrified waste into the container. Functional clearance provides an easy vertical loading by getting rid of container manufacture and primary package tolerances. The lid is then placed on the container body. The assembly is then inserted into a vacuum chamber as vacuum is required for electron beam welding. The container is then rotated and orbital welding of the lid is carried out.

Figure 6.3.4 shows a welding chamber under vacuum conditions and Figure 6.3.5 the EBW electron beam welding process.

Figure 6.3.4 Vacuum welding chamber

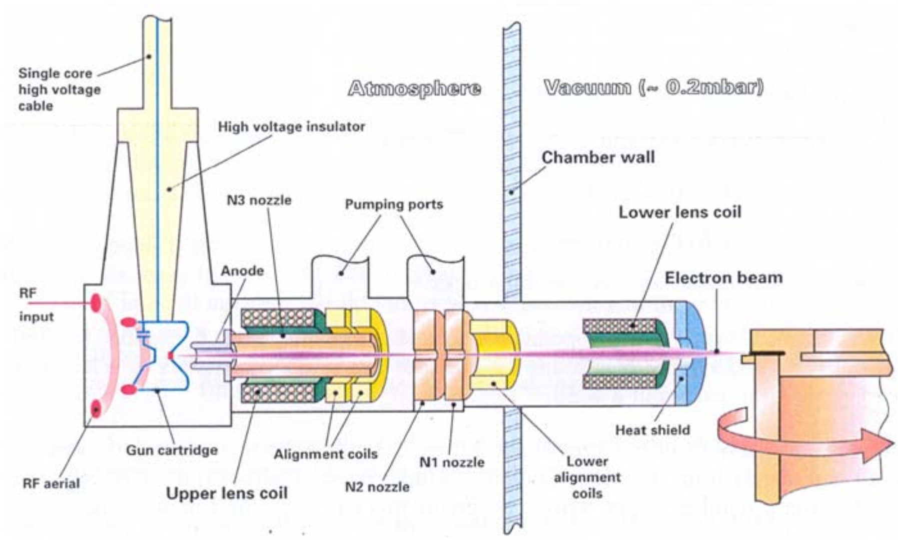


Figure 6.3.5 EBW process diagram

For welding process control, important parameters can be recorded during the operation. Once completed, welding is then tested, with the disposal package placed in another shielded cell.

An appropriate weld surface test procedure, routinely used by manufacturers, consists of visual inspection with video camera. Other procedures being considered, namely liquid penetrant testing (ressuage) and magnetoscopy , are more difficult to apply in an irradiating cell.

A proven process by volumetric examination is based on use of ultrasound.

Electron beam welding and the welding inspection methods have been used for the spent fuel container demonstrators studied for the clay medium using the same grade of steel as envisaged here.

6.4 The disposal cells

In the granite medium, the orientation and distribution of minor fracturing likely to locally affect the very low permeability of the rock govern whether horizontal or vertical cells are chosen for the granite.

The analysis of minor fracturing in the French geological context shows that it is more often represented by sub-planar structures and generally sub-vertical or vertical.

Also in reference it has been decided to study the vertical disposal borehole design, which as it is close to the line of the highest dip of most of these structures, statistically limits the number intersected by the disposal cells.

These disposal boreholes are installed in handling drifts. This chapter describes both these two structures.

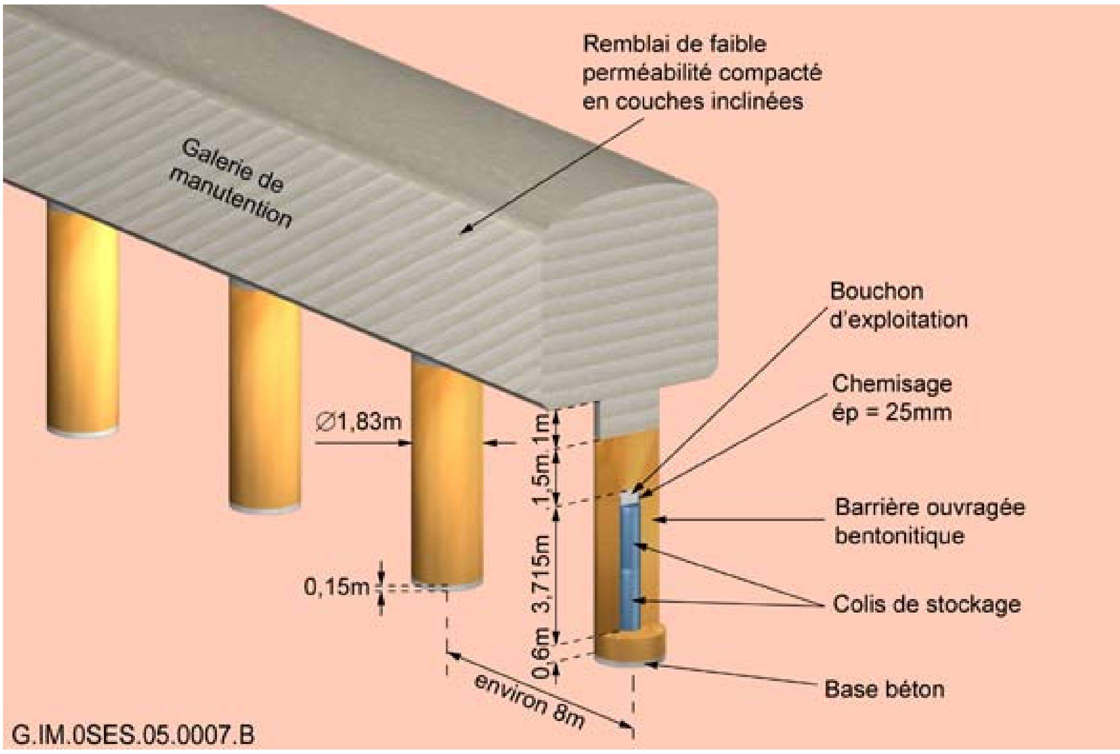


Figure 6.4.1 View of a C waste disposal borehole

However for a site where the geometries of minor fracturing are predominantly horizontal, a horizontal tunnel cell design could be applied. It would be based on the same broad design principles, with primarily a steel over-pack inserted in a sleeve and a clay engineered barrier placed between the sleeve and the wall of the horizontal tunnel.

For the concepts studied, a package cooling period is required prior to acceptance by the repository so that the temperature criterion is not exceeded (cf. § 6.2.2.3). The interim storage period has little to do with the thermal characteristics of the granites as envisaged in the French context [42].

6.4.1 Description of a disposal cell

The reference disposal cell for repository studies in a granite medium comprises a vertical disposal borehole with an engineered barrier. The disposal borehole diameter varies from about 1.80 m (reference packages C02 and C1-C4) to 1.90 m (reference packages C0.1 and C0.3). The depth of such a disposal borehole for two C packages (C1 to C4) is about 7 m. In the case of a borehole accommodating 5 C0 packages, this depth is about 10.2 m. It increases to 11.8 m for the disposal of C0.2 packages, which are the tallest.

A cell comprises two parts:
- the useful part of the cell is equipped from the outside inwards, with a 60-cm thick swelling clay engineered barrier and a permanent metal sleeve containing two to five disposal packages depending on their heat rating;
- the borehole head will accommodate the cell seal, comprising from bottom to top, a 1.5-m high swelling clay plug and 1 m of slightly permeable compacted backfill.

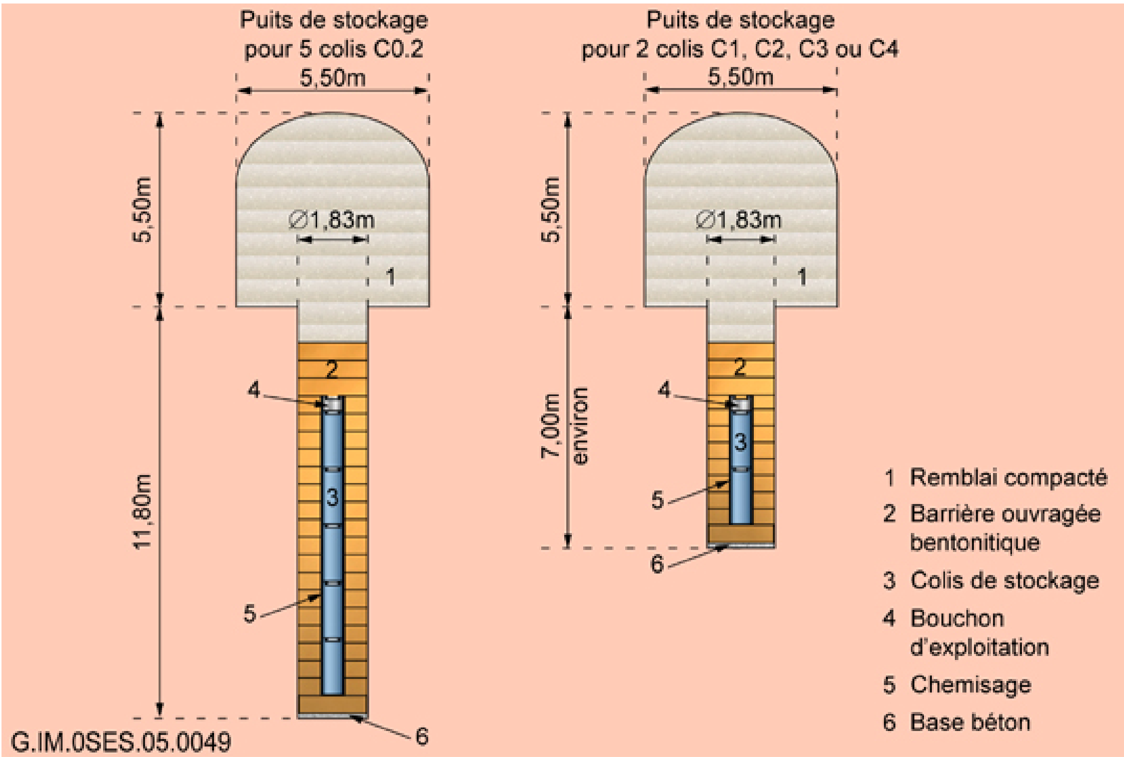


Figure 6.4.2 General dimensions of C waste cells

- **The swelling clay engineered barrier**

The following main properties are expected of the engineered barrier:
- low permeability with a target of about 10-11 m/s;
- the highest possible thermal conductivity; over 1.5 W/m/°C in the long term (resaturated material) and 1.2 W/m/°C during the thermal peak (if the material is not yet saturated);
- adequate mechanical strength (resaturated material) to prevent that the column of packages punches the engineered barrier at cell end.

To achieve such properties, the swelling clay formulation (percentage of "MX80" clay or equivalent to sand) must also take into account the site groundwater's geochemical properties. Andra's studies [37], along with those carried out for the Swedish KBS-3 project, open up a wide enough range of possibilities to adapt this formulation to the chemical composition of the waters of a specific granite massif.

Steel corrosion (over-pack and sleeve) produces iron which is likely to interact with the swelling clay: smectites (swelling) are transformed into chlorites (non-swelling). This transformation spreads radially in the clay engineered barrier from the surface where it is in contact with the sleeve. Chloritisation likely to disturb the swelling power of the clay only affects a few centimetres' thickness of the clay where it is in contact with the steel: further, the smectites are only very slightly transformed and the engineered barrier retains its swelling capacity [49].

Furthermore alkaline disturbance is avoided by the absence or very small amount of concrete in and around the cells.

- **Disposal cell end**

The borehole bottom comprises a flat surface that receives the first elements of the engineered barrier. The option under consideration at this stage is to cast a low pH concrete base about fifteen centimetres thick. Other materials that are inert to clay (sand, bentonite pellets or powder) may also be envisaged. An alternative is granite surfacing at the borehole bottom.

- **A steel sleeve**

A steel sleeve, of the same grade as the over-packs, is placed in the engineered barrier axis. Its purpose is to receive disposal packages. This sleeve protects the engineered barrier mechanically during package emplacement and makes package emplacement and retrieval easier. At this stage of the studies, sleeve thickness is set at 25 mm for S235-grade steel. It allows for both 12 MPa of pressure applied at the extrados (just like the over-pack, see above) and a corrosion allowance of about 5 mm.

The sleeve base consists of a welded blind flange. The upper part is designed to take a steel radiological protection plug. This plug, made from S235-grade steel or equivalent, is dimensioned to protect operators from external exposure while emplacing or retrieving packages from the sleeve. It has grooves at the top to enable the same grapnel to be used as for the package handling operations.

- **A limited number of packages**

At the time of disposal the thermicity of the C waste reference packages governs the number of packages to be emlaced in each disposal borehole.

For the most exothermic reference packages, C1 to C4, the main overriding factor is the temperature criterion of 90° at the hottest point of the clay engineered barrier (so that its mineralogical and geochemical characteristics remain unaltered). A design based on placing two superposed packages is suitable for adapting to the characteristics of most French granites with an interim storage period of 60-70 years for reference packages C1 and C2, and century-long for reference packages C3 and C4, before being accepted by the repository [42].

As C0 waste presents lower thermicity, a greater number of packages can be placed in each disposal borehole. However increasing cell capacity, which entails boring a deeper disposal borehole, is limited by the higher statistical risk of encountering minor potentially water-conducting fractures. At this stage, the number of C0 packages per disposal borehole is limited to five to allow for this risk.

- **Sealing the disposal cell**

The swelling clay cell plug shares the same functions as the 60-cm thick engineered barrier placed around the packages. Additional thickness is incorporated to mitigate any mechanical confinement defects in case of deferred settling of the backfill by swelling clay thrust. Thus a height of 1.5 m is taken as the reference.

Low permeability backfill is emplaced above the swelling clay plug. It has identical proportions of granite aggregates and swelling clay to those used in drift backfill. The swelling clay cell plug is mechanically confined as continuity prevails throughout from the backfill at the cell head to the handling drift. A metric thickness adopted is similar to the one considered by SKB in Sweden for its borehole design experiments.

6.4.2 Description of a handling drift

The cells are arranged along handling drifts, whose length is adapted to the geometry of the host block. A mean length of 250 metres may be envisaged as the reference.

The standard section of a handling drift is the outcome of two dimensioning elements: cell construction and package handling equipment profile and the need for long term geomechanical stability.

With regard to the first point, the element used for dimensioning is the machine that transfers and emplaces the packages in the cells (cf. § 6.6.4).

Just like the engineered structures described in chapter 5 for B waste and on the geomechanical viewpoint, the geometry of the handling drifts is set in line with the granite mechanical characteristics. They are stable in the long term without the need for additional rock support other than that required for operator safety (occasional ground support: shotcrete and rock bolts, etc.) Thus "inverse U" geometry is taken as the reference.

This results in useful sections of 28.5-30 m^2 for a width of 5.5 m and heights of 4-4.5 m on the sides and 5.5 m under the arch axis.

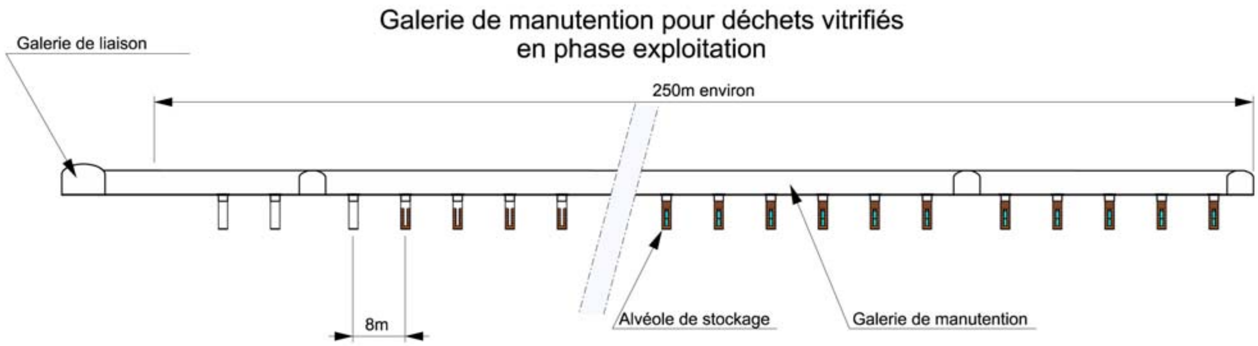


Figure 6.4.3 Cross-section of a handling drift during the operating phase

Interconnecting drifts, placed roughly every 150-200 m and at the ends of the handling drift, allow machinery traffic (one-way only), rapid evacuation from an incident zone and the implementation of a fresh air circuit.

6.4.3 Distance between handling drifts

C waste disposal compactness in a module is governed by the distance between two disposal cells in a drift and the distance between two parallel handling drifts. These distances depend on the application of mechanical and thermal criteria.

From the mechanical angle, the distance between two neighbouring cells must be enough to prevent disturbances occurring in the volume of rock separating them. In granite, this basically depends on the mechanical characteristics of the rock and the state of the natural stresses. At this stage a reasonable base would appear to be a distance two to three times the thickness of the disposal borehole diameter, that is 4-6 m.

From the thermal angle, the distance between two neighbouring cells must be enough to prevent thermal interference occurring which would probably damage the clay engineered barrier. Preliminary calculations to test the sensitivity of this distance at various foreseeable thermal characteristics for French granites have resulted in very close results: for interaxial distances between cells in the range 6-8 m and interim storage periods of 60-70 years (reference packages C1 and C2) to century-long (reference packages C3 and C4), the thermal interference is only at its maximum at the midpoint between two disposal boreholes (less than 60°C) after a century. When it is observed, the calculated temperature of the package is below 90°C [42].

At the current stage, the presence of minor fracturing of the rock inside the module cannot be ruled out, but its hydraulic conductivity must be low enough not to affect the confinement functions of the various repository components: the rock itself at the cell sites, the disposal packages, the engineered barriers and backfill. This potential fracturing presence is currently managed by considering that an average of 10% of all possible cell locations will not be used.

Thus about fifty reference packages C1-C4 (two packages per cell) and roughly 130 reference packages C0 (five packages per cell) can be accommodated by a configuration characterised by drifts averaging 250 m in length and with cells sited every 6-8 m

6.5 Architecture of the C waste repository zone

The C waste repository zone includes the repository modules for reference packages C0 and C1-C4 distributed over two levels and their connecting infrastructures.

Each repository module includes the handling drifts served by a shared cluster of connecting drifts. The number of handling drifts in a module depends both on the host block geometry and the distance that separates two neighbouring drifts.

This section presents the module dimensioning elements, followed by a description of them and the repository zone. It gives the preliminary assessments on the basis of the inventory model, of how many modules need to be constructed and the underground footprints they will take up.

6.5.1 Dimensioning elements of a repository module

- **Mechanical dimensioning elements**

As with the distance (pitch) between two disposal cells, there must be sufficient distance between two neighbouring drifts to prevent perturbations occurring in the volume of rock separating them. At this stage a pillar approaching three times the width of the drift, that is about fifteen metres wide, appears to be the right basis for the mechanical characteristics of all French granites.

- **Thermal dimensioning elements**

Thermally, the distance between cells installed in two neighbouring handling drifts, must be sufficiently far apart to avoid any thermal interference that might alter the evolution of a disposal cell in terms of the temperature criterion given earlier.

Very close results have emerged from preliminary calculations to test the sensitivity of this distance at various foreseeable thermal characteristics for French granites. Thermal interference is at its maximum at the core of the pillar (less than 50 °C) fifty to one hundred years after the temperature peak in the cell [42] for distances approaching twenty metres and interim storage periods of 60-70 years (reference packages C1 and C2) to century-long (reference packages C3 and C4).

- **Hydraulic dimensioning elements**

Architectural adaptation of the module to the geometry of the host granite block includes a clearance distance between the excavated openings and the potential water-conducting fractures at the block boundaries.

In principle this clearance distance cannot be predetermined. It essentially depends on the hydraulic characteristics of the fractures that delimit each block and their connectivity with the intermediate fracturing admitted in each block. At the current stage, it is considered that all foreseeable situations in the French context can be covered by clearance distances of several tens of metres (up to one hundred) adapted to suit the type of fractures that set host block limits.

6.5.2 Description of a repository module

The handling drifts of a module are served by a shared cluster of connecting drifts. The latter are dimensioned to enable successively exploration, construction and package emplacement operations and finally, when the decision is taken to close the module, backfilling and sealing operations.

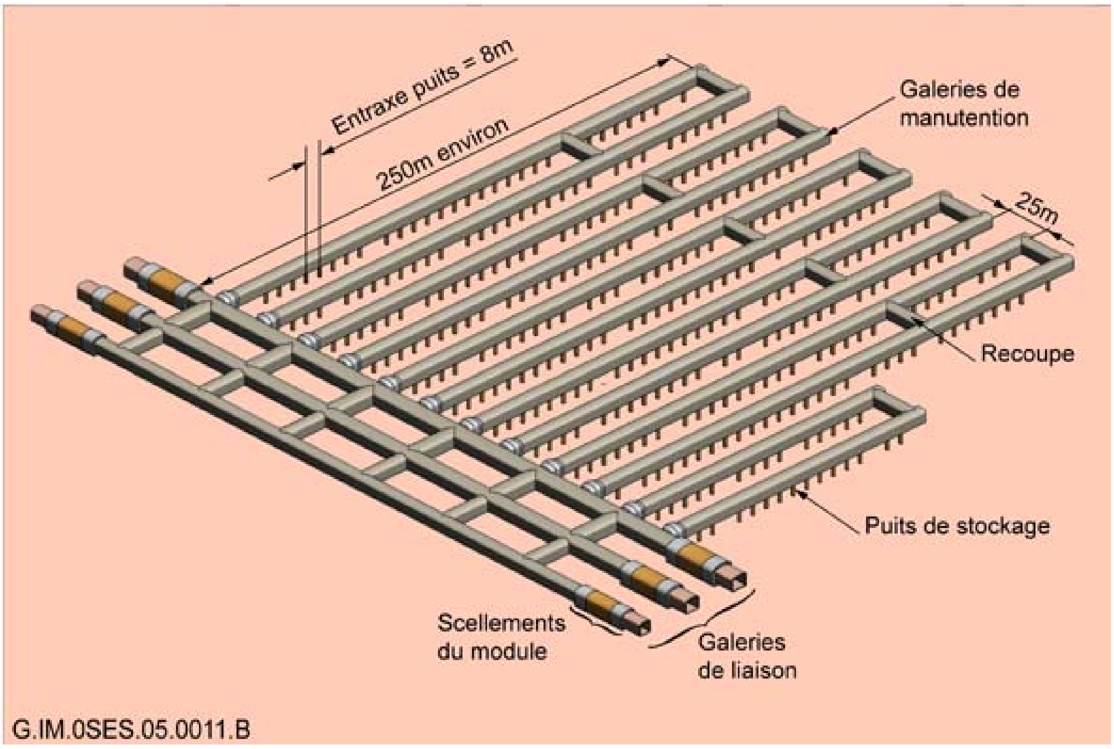


Figure 6.5.1 Schematic representation of a C waste module

The construction activity dimensions the connecting drift number and section in a repository module.

Two drifts are envisaged, each of which is dimensioned to enable two workface units to pass: thus they are 6-8 m wide. These drifts also enable fresh air ventilation (in drift full section) and its distribution towards the handling drifts.

During module operation, these drifts are reallocated to disposal package transfer.

Exhaust air may be ducted through one of the conduits installed in one of the construction drifts in specific configurations (small dimension module, proximity to main connecting cluster, etc.). In other configurations, a third dedicated exhaust air drift is constructed. Its dimensions are adjusted to module ventilation requirements.

Once again, the connecting drifts present "inverse U" sections to achieve stability without systematic ground support. Mechanical perturbation in between connecting drifts (in the pillar) is avoided by constructing them three times the widest connecting drift diameter apart.

A number of simulations have been carried out at this stage, to determine the module exploration and construction time. This appears to be longer than the package emplacement period. It takes three to four years to prepare a 160-180-cell module for reference packages C0, filling taking two years, while it takes four to five years to prepare a 600-660-cell module for reference packages C1-C4, filling taking two years too.

Module construction time does not increase in proportion to the number of packages to be emplaced: the larger a module, the more optimised can be construction work scheduling through various simultaneous work faces.

6.5.3 Assessing the number of C waste modules

In the absence of specific site data, an estimate of the required number of modules can be made by considering those modules whose mean capacity corresponds to filling in two years (about 600 cells): for scenario S1a (or S1b), 32 modules of this type are needed, including 5 for reference packages C0. This number increases to 37 for scenario S1c, and to 14 for scenario S2 (spent fuel reprocessing is stopped). About twenty blocks would have to be identified for two-level architecture.

In a more demanding site configuration on the fracturing viewpoint, smaller modules, operated for about eighteen months can be envisaged (about 450 cells). Thus the number of modules to be constructed would vary from about 40 for scenario S1c to about 20 for scenario S2.

C waste repository zone architecture entails organisation of clusters of primary and secondary connecting drifts suitable for dealing with the spread of ten to twenty modules on each repository level.

These more or less ramified solutions (cf. § 4.2), should provide a flexible and gradual management of the disposal process by enabling various simultaneous activities associated with exploration, construction, operation and closure of modules distributed in various points of the repository zone.

Assessments of the waste repository zone area developed on both levels for a defined site can be calculated for each study scenario in line with the mean distance between neighbouring modules. This distance is specified during the site exploration work (surface and laboratory work).

6.6 Disposal process as part of the reversibility rationale

6.6.1 Installation of the disposal cells

In the same way as for the B waste repository tunnels, the detailed exploratory work of a C waste module host block is carried out in stages (Figure 6.6.1), as excavation work progresses..

This program aims at defining:

- the position of the handling drifts on the basis of horizontal boreholes drilled from a drift accessing to the block and of geophysical measurements between boreholes;
- the layout of disposal boreholes using geological surveys at the handling drift walls, short vertical boreholes and geophysical and hydro-geological measurements;
- cell qualifications on the basis of geological surveys of the wall and additional measurements carried out in the disposal boreholes.

At each exploration stage, the information gathered is compared to the criteria as defined through underground laboratory studies and safety analyses.

Through these comparisons, module architecture is gradually adjusted to the characteristics of the granite in the block and disposal cell qualification rate by module can be managed.

6.6.2 Construction of the connecting and handling drifts

As for B waste module connecting drifts, full section excavation by drilling and blasting is taken as the reference method. Likewise the use of low-damage techniques such as "smooth blasting", is envisaged in the vicinity of the future module seal sites.

Two excavating techniques can be envisaged for the handling drifts:

- drilling and smooth blasting;
- a combination of mechanical excavating techniques using a tunnel boring machine (TBM) in the handling drifts and smooth blasting for the interconnecting drifts.

At this stage excavating with explosives has been taken as the reference method because it offers greater flexibility with today’s proven techniques. This method is the same as the one adopted by SKB for their spent fuel disposal borehole design: KBS-3V. This choice is based on results of the APSE experiment, which has demonstrated that it is possible to limit the development of a damaged zone to about thirty centimetres through blasting pattern adjustments.

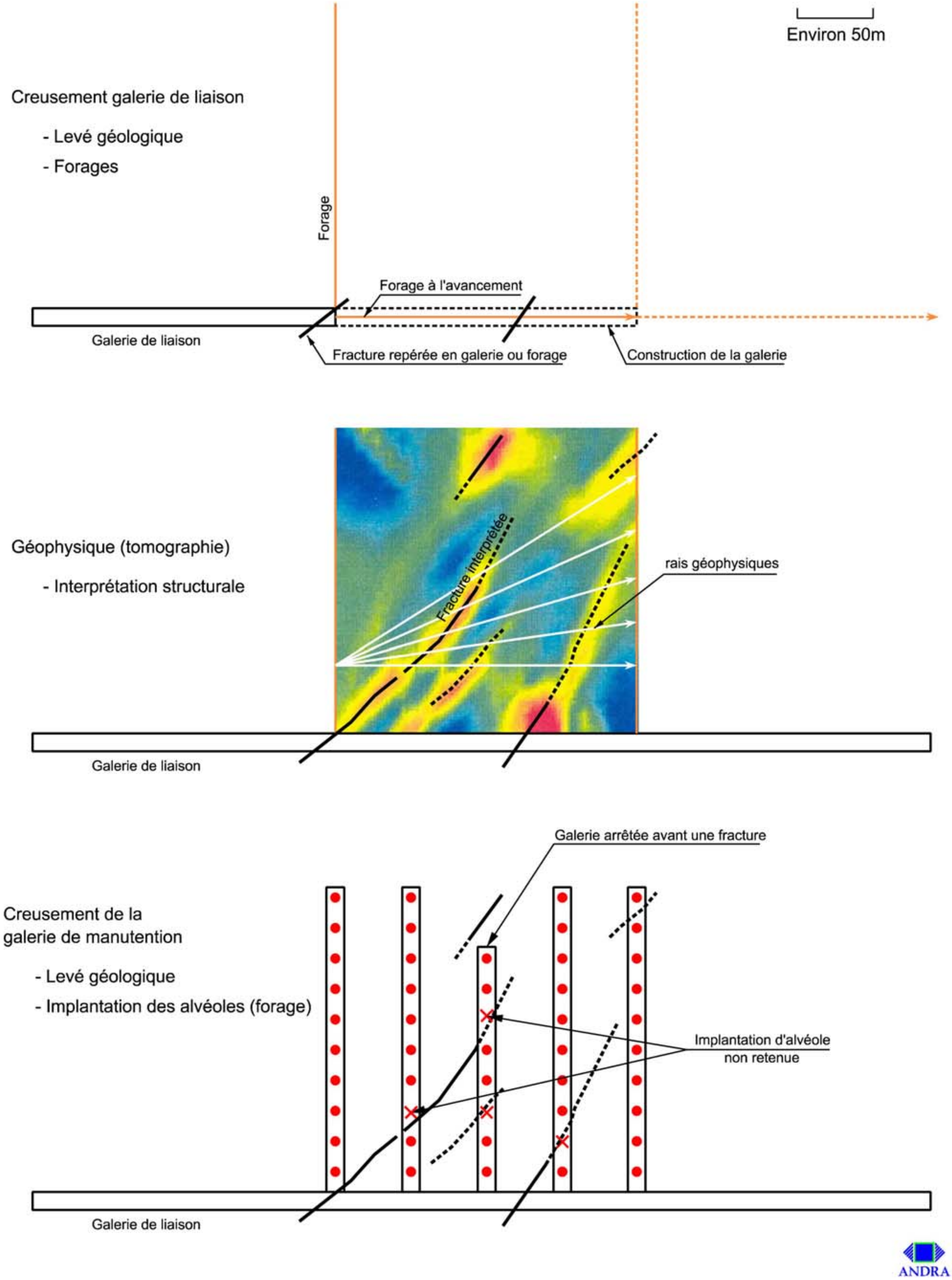


Figure 6.6.1 Basic surveying programme for cell installation

6.6.3 Construction of disposal boreholes

Disposal boreholes are bored using a micro-tunnel boring machine modified to operate vertically. Construction is based on the demonstrations carried out by SKB in the Äspö laboratory to excavate a disposal borehole for spent fuel.

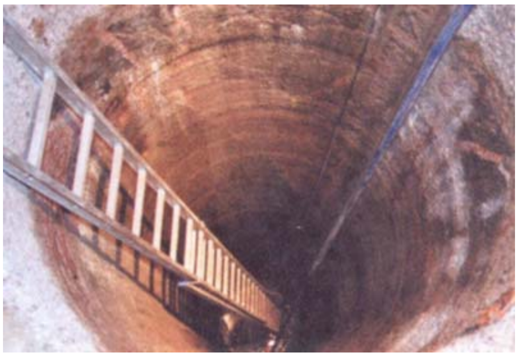


Figure 6.6.2 Disposal borehole in the Äspö laboratory (source: SKB)

This method has been adopted since it is suited for constructing small diameter vertical shafts from small section drifts and because it causes very slight damage at the excavation wall.

The drill head thrust system and the removal of broken muck (Figure 6.6.3) are the main areas modified to enable the micro-tunnel boring machine to operate vertically. The thrust is power-assisted by four independent hydraulic jacks. Steel casings, approximately the same height as the jack stroke, transmit the thrust to the drill head. By adding steel casings, boring is carried out. The jacks lean against the drift arch [50].

The broken muck is raised by reverse air circulation in a suction column installed in the body of the micro-tunnel boring machine. Suction is controlled by a vacuum pump installed in the handling drift. A filtration system removes dust from the air and collects the broken muck into a closed bin.

Improvements are planned so that boreholes can be excavated deeper than the Äspö laboratory ones. As well, a pilot hole is also envisaged to steer the drill head.

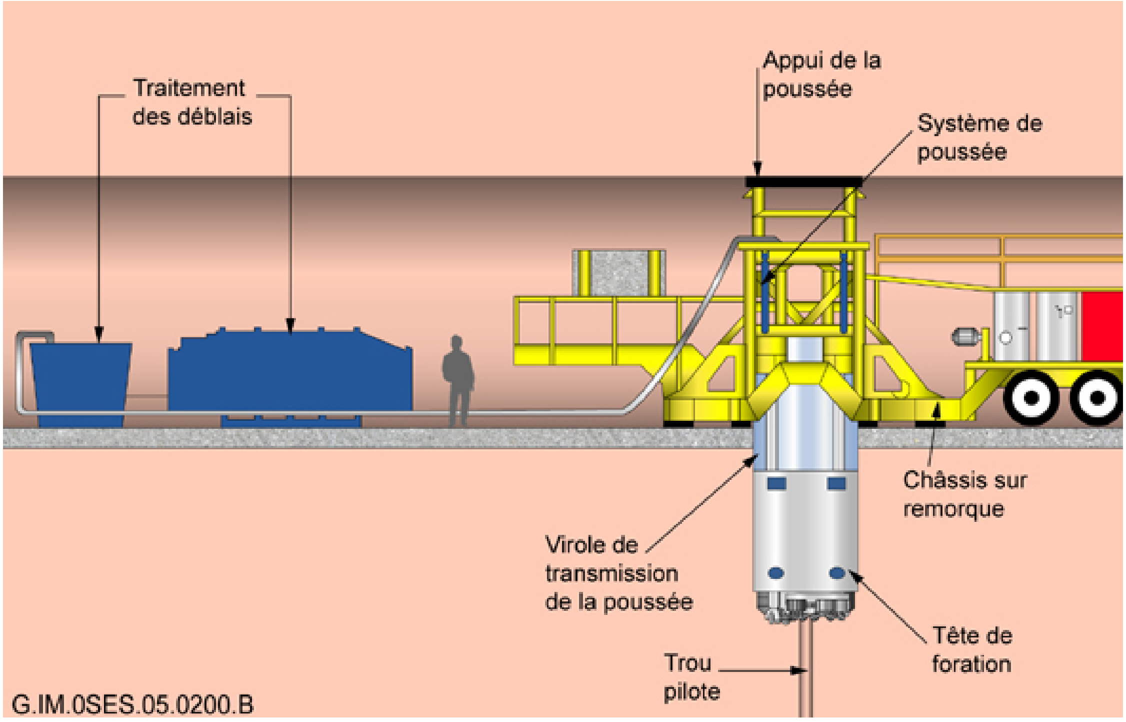


Figure 6.6.3 Operating principle of a vertical micro-tunnel boring machine

As indicated above, geological surveys of the drilled borehole (state of the walls, characterisation of any ingress of water) are performed to confirm its acceptability for conversion to a disposal cell. The concrete base can be shaped by an operator lowered in a cradle.

The engineered barrier is made up of rings whose internal and external diameters are adjusted to those of the sleeve and the borehole. Rings 50 cm high have been constructed by uniaxial pressing in Sweden for experimentation requirements in the Äspö laboratory with Andra's participation.

Figure 6.6.4 Emplacing an engineered barrier ring (source: SKB)

The engineered barrier elements are positioned using a conventional lifting device (Figure 6.6.3).

After each descent phase or less often, an inspection is made to confirm that the engineered barrier rings have been properly centred.

The sleeve, a set of tubes screwed together, is put in position either gradually as the engineered barrier construction advances to help guiding the rings, or once the engineered barrier column has been fully formed.

A mobile device is inserted to close and protect the disposal head outside disposal borehole intervention periods. This device is designed and dimensioned so that handling equipment can pass freely into the handling drifts.

6.6.4 Operation of a disposal cell as part of the reversibility approach

The waste package emplacement-retrieval equipment and process using a biological protection cask and concerning vertical disposal borehole can be transposed using industrial feedback from COGEMA’s La Hague centre. COGEMA uses a cask comprising a barrel that handles several primary packages of vitrified waste (CSD-V) in succession in a nine-package capacity storage pit.

The similarity between the operations in the storage pit operations at La Hague and those planned for the disposal boreholes (package emplacement and retrieval) makes a compelling case for adopting a similar approach in the granite formation [51].

The cask illustrated in Figure 6.6.5 is made up of a cylindrical shielded vessel. The walls, made from a steel composite and neutron absorbing material, are about 300 mm thick, to provide radiological protection of the operator.

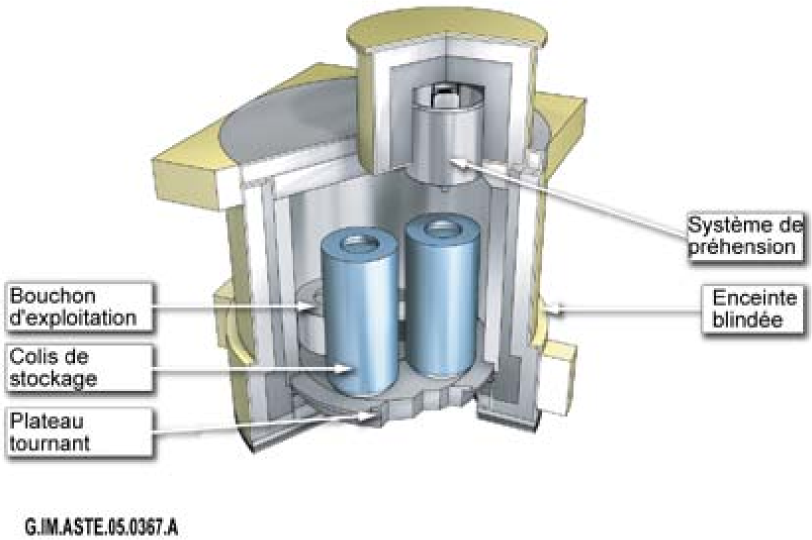


Figure 6.6.5 Radiological protection cask for C waste packages

Cask loading and unloading is carried out through a cylindrical opening on the underside of the vessel. Inside the vessel, the barrel is made up of a rotary table with three slots and an opening. By revolving the table the openings can be lined up to lower the disposal packages freely. Right above the cask opening, the cask is equipped with a winch that enables an integrated clamping arm to pick up a package and transfer it from the cask to the borehole.

The cask dimensions are roughly 2.5 m in diameter by 3.4 m high. The loaded mass will be about 50 tonnes. The cask can transport two upright disposal packages and also a radiological protection plug and/or engineered barrier plugs.

The disposal package emplacement cycle consists of loading the cask in the surface installations, transferring it to the disposal cell, emplacing the package in the cell and finally returning the cask to the surface.

6.6.4.1 Cask transfer in the underground installations

Cask transfer in the shaft between the surface and underground is similar to B waste cask transfer (chapter 5). Once the cask is removed from the cage at the repository level, it is placed on a transfer shuttle that takes it right above a disposal cell.

This type of shuttle technology has been used at COGEMA's La Hague site for many years.

The shuttle basically consists of a wheeled vehicle and a moving chassis that suspends the cask while travelling and adjusts the cask opening alignment to the borehole axis when docking onto the disposal cell.

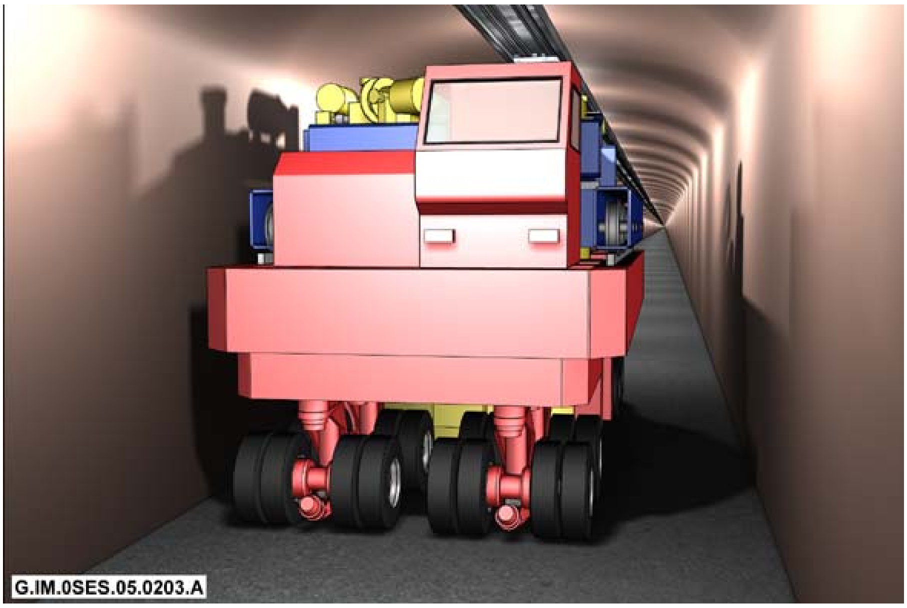


Figure 6.6.6 Shuttle in a handling drift

The vehicle is fitted with multi-directional wheels, each one with independent hydraulic motorisation, thus providing very tight turning circle. Each of these wheels has its own hydraulic lifting device which, in synchronisation, can lift or lower the loading deck with the cask. The total mass of the loaded shuttle is about one hundred tonnes.

6.6.4.2 Emplacing the packages in a disposal borehole

While docking the cask to a disposal cell (vertical borehole) head, additional radiological protection devices, similar to those at La Hague are used: a thick elastomer leaded seal fixed to the lower part of the vessel, and fixed or removable metal plates installed at the cell head.

Before the shuttle arrives, the cell head protection device is removed. Once the shuttle is positioned over the cell head, the loading deck is lowered. Final alignment of the cask opening with the cell axis is adjusted using the translation systems of the moving chassis that holds the cask. The cask is then docked by lowering the jacks integrated to the moving chassis.

Disposal package emplacement in the borehole is carried out in the following sequence (Figure 6.6.7) :

- grasping a package or plug in the cask;
- revolving the barrel to line up the deck opening and cask axes;
- emplacing the package in the borehole then retracting the empty clamp into the cask.

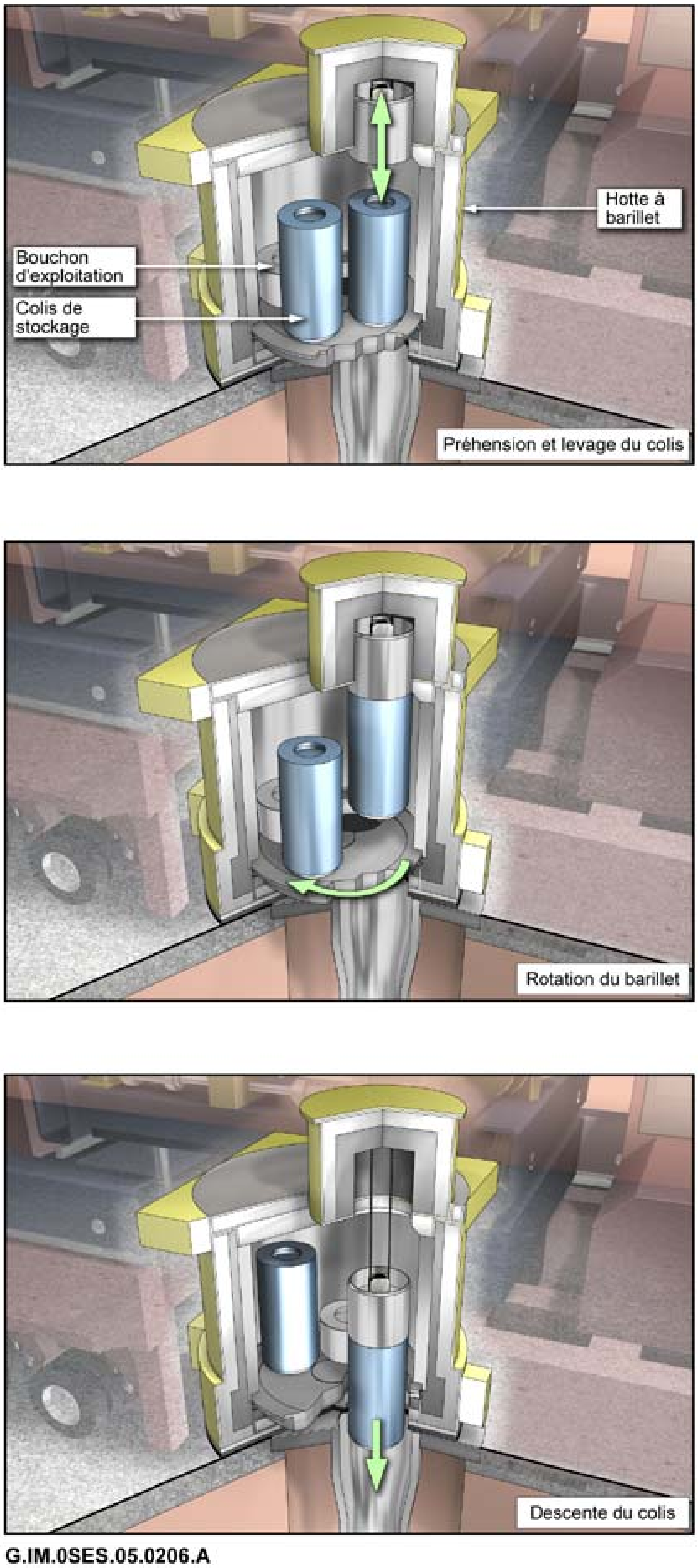


Figure 6.6.7 *Basic diagram of barrel operation*

The cask accommodation capacity (two packages and one plug) is such that a C1-C4 waste disposal borehole can be fully filled in a single shuttle trip. Package emplacement for cells that accommodate more packages starts by removing the borehole radiological protection plug and storing it in the cask. The cycle ends by emplacing the radiological protection plug in the borehole.

The lifting system is equipped with the usual package drop safety devices provided in nuclear equipment design. For example a load lowering speed reducing device is envisaged in the event of a malfunction or break in the lifting drive train. Likewise, the package grappling arm is designed with inherent security and the fingers can only open if the load is put down.

At the end of the cycle, the rotary table set at the position “closing the cask”, then the cask is undocked for the return trip to the surface. The disposal borehole head protection device is put back in position.

6.6.5 Risks associated with C waste package emplacement in the cell

A package grasping grapnel malfunction may cause a package to drop during emplacement (or retrieval as the case may be) and despite the designed safety systems cannot be ruled out

Specific simulations [52] have been made on the basis of two scenarios: the first scenario corresponds to the first disposal package falling from the cask to the bottom of the disposal borehole, the second is the second disposal package falling onto the top of the first disposal package. The height of 20 metres adopted for this study amply covers the planned disposal borehole heights as they will only be about ten metres deep.

The results obtained reveal that even with a drop height of 20 metres, the deformations would be limited to the disposal package envelope base. They are a little more severe in the first scenario studied (direct drop of the package to the bottom of the borehole) than in the second. The C waste primary package itself is hardly affected by the impact in either case.

Thus the risk of a C package falling during emplacement does not appear to be able to lead to the breaking of a package with loss of confinement of radioactive material.

6.6.6 Closing the structures

As in the case of B waste (chapter 5), the closure of the C waste repository zone can be staged, which provides potential observation time between each stage. Each stage of the process can be managed with considerable flexibility. Reversing the process by backtracking to an earlier stage, as far as retrieving the packages, is possible, even if it calls for the deconstruction of more substantial closure structures, as closure progresses further.

The successive closure stages for the disposal cells (see section 6.4.1), handling drifts, module or set of grouped modules, and disposal zone level call for the emplacement of backfill and seals that play a complementary and redundant role.

- **Closure of the handling drifts**

The closure of the drifts consists of removing the equipment from the structure, cutting a trench across the concrete roadway, emplacing continuous low permeability compacted backfill between the cell heads and the standard section of the drift, and constructing a concrete plug to confine the backfill mechanically.

The composition of the low permeability backfill and their emplacement method in inclined and compacted layers are described in chapter 3.1.4.

A concrete plug constructed at the entrance to each handling drift mechanically confines the swelling of the clay fraction of the backfill. In the event of the closure of one single drift, the construction of additional plugs in the interconnecting drifts is considered. These plugs are installed in drift sections where no cell has been constructed.

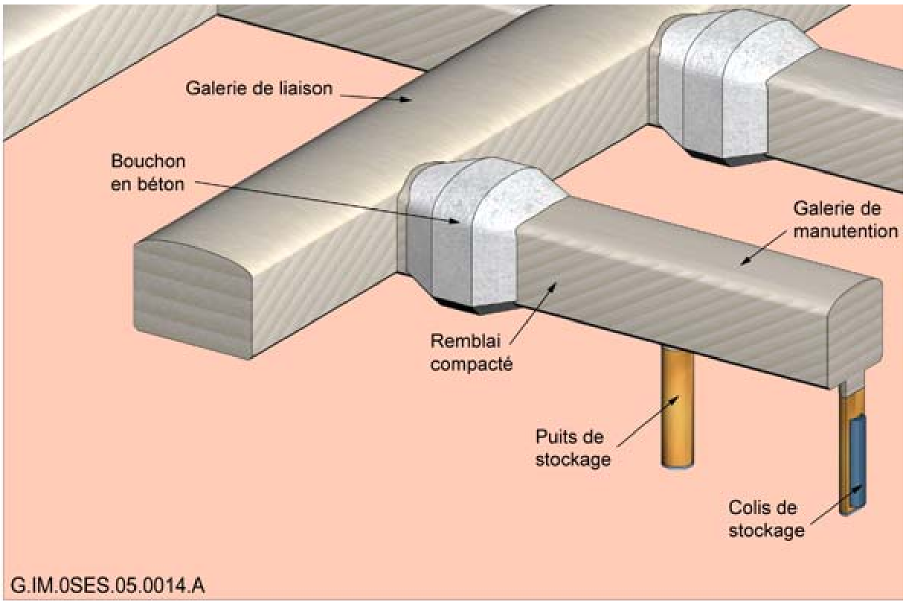


Figure 6.6.8 Closure of a handling drift

- **Closure of a repository module**

As in the case of closure of a handling drift, the closure of the connecting drifts of a module comprises removing the equipment from the structure, scraping the concrete roadway and emplacing compacted low permeability backfill. Seals incorporating a swelling clay core are built at the edge of the module in each connecting drift (chapter 6.1.4). These seals are installed in the host block away from water-conducting fractures that define their limits.

- **Closure of a repository zone level**

Connecting secondary and main infrastructure closure essentially consists of removing the equipment and backfilling the drifts and constructing concrete plugs, which enables the process to be phased.

To maintain an observed hydraulic compartmentalisation in the granite host massif, seals associated with low permeability backfill may be implemented.

6.7 Functions of the repository components over time

The previous sections have described the various components of a C waste repository. The following table summarises how these various components contribute to the main functions of a repository (cf. section 2.1) during its various phases

Table 6.7.1 *Functions of the repository components over times*

COMPONENT	PERIOD	MAIN REPOSITORY FUNCTIONS	PROPERTIES HARNESSED
1. GEOLOGICAL MEDIUM: GRANITE			
Part of the granite rock where a disposal cell is installed	During operation and reversible repository management After closure	Emplacing (and being able to retrieve) the packages in the cells Protecting the disposal cells from water circulation Delaying and reducing radionuclides migration to the environment	Mechanical strength Very low permeability Chemical properties of the granite water that tend to delay radionuclide migration (reducing environment, etc.)
Granite "block" where the repository module is installed	After closure	Delaying and reducing radionuclides migration to the environment Protecting the disposal cells from water circulation	Low permeability (at a distance from the significant water-conducting fractures) Low in-depth hydraulic gradients Chemical properties of the granite water that tend to delay radionuclide migration (reducing environment, etc.)
The repository host part of the granite massif	After closure	Isolating the waste from human activities Protecting the disposal cells from water circulation Delaying and reducing radionuclides migration to the environment	Underground installation of the repository (500 m taken as reference) Low permeability (at a distance from the major faults, the main vectors of water in the massif) Low underground hydraulic gradients Chemical properties of the granite water that tend to delay radionuclide migration (reducing environment, etc.) Geo-dynamic context that ensures favourable repository conditions are maintained in the long term, particularly erosion phenomena

COMPONENT	PERIOD	MAIN REPOSITORY FUNCTIONS	PROPERTIES HARNESSED
2. ENGINEERED COMPONENTS			
Connecting drifts between access structures from the surface and repository modules	Before and during operation and reversible repository management	Emplacing (and being able to retrieve) the packages in the granite	Dimensions adjusted to package throughput Drift equipment that ensures safe operation
Drifts in the modules (cell access drifts and exploratory structures)	Before and during operation and reversible repository management	Emplacing (and be able to retrieve) the packages in the granite	Suitable dimensions for package throughput and granite characteristics Drift equipment that ensures safe operation
Backfill and seals of the repository module drifts	After closure	Protecting the repository modules from water circulation Delaying and reducing radionuclides migration to the environment	Low permeability Swelling capacity Radionuclide retention
Cell: disposal cavity C waste: small vertical boreholes	During operation and reversible management After closure	Emplacing (and being able to retrieve) the packages in the granite Restricting radionuclide release by the primary packages and immobilising them	Dimensions adapted to the granite fracturing and handling technologies Properties of the various components: packages, engineered barriers, backfill
Steel over-pack for C waste	During operation and reversible management After closure (thermal phase)	Emplacing (and being able to retrieve) the packages in the granite (with sleeve and operating plug) Preventing water coming into contact with the glass (and the release of radionuclides)	Mechanical strength Suitable dimensions for handling them Water-tightness throughout at least the thermal phase (1000 years)
Clay engineered barriers of the C waste disposal boreholes	After closure	Protecting the disposal cells from water circulation Delaying and reducing radionuclides migration to the environment	Very low permeability Swelling capacity and plasticity (providing the long-term mechanical stability of the cell) Chemical buffer to interactions between the granite water and the component alteration products Radionuclide retention

7 Spent fuel repository zone

Spent fuel is not considered for the time being as waste. Nevertheless an examination of its potential disposal in a deep geological formation repository has been made on the basis of scenarios formulated from hypotheses on the back-end management of the nuclear energy cycle (cf. § 2).

This examination is based on the characteristics and properties of the fuel produced by the PWR nuclear power plants, reactor systems now shut down, research reactors (UNGG, EL4) and French National Defence activities and included the safety functions that must be ensured for their disposal.

This chapter describes the design for the spent fuel repository zone. It states how designs adopted abroad in the granite medium can be transposed to the French context to fulfil the safety functions defined in chapter 3.1.

The proposed cell design and repository zone architecture options are also based on the options already envisaged for the disposal of vitrified C waste.

Finally this chapter describes how these options can be implemented for a reversible disposal process and meet operational safety and security requirements.

7.1 Spent fuel assemblies

7.1.1 The different types of spent fuel

As in the case of B and C waste, the spent fuel inventory has been organised by type (cf. § 2.3).

This organisation is based on a tree structure that groups together fuel assemblies with similar thermal and radiological characteristics on one level, and then the assembly families differentiated by origin.

Table 7.1.1 Spent fuel reference packages and assembly families

Reference fuel			**Assembly families included in the reference packages and reference sub-packages**	
Lev. 1	**Lev. 2**	**Lev. 3**	**Description**	**Description**
CU1			PWR, UOX and enriched uranium fuel	7.1.1.1
CU2			PWR and MOX fuel	
CU3	CU3.1		UNGG and EL4 fuel	7.1.1.2
	CU3.2		CELESTIN fuel	
	CU3.3		Nuclear propulsion fuel Batch A	

7.1.1.1 PWR fuel assemblies

The reference fuel assembly corresponds to a second-generation advanced assembly designed by FRAGEMA with thicker guide thimbles and zirconium alloy cladding. It is illustrated in Figure 7.1.1. It is called AFA-2GE for the PWR 900 MWe reactors and AFA-2LE for the PWR 1300 MWe and 1450 MWe reactors.

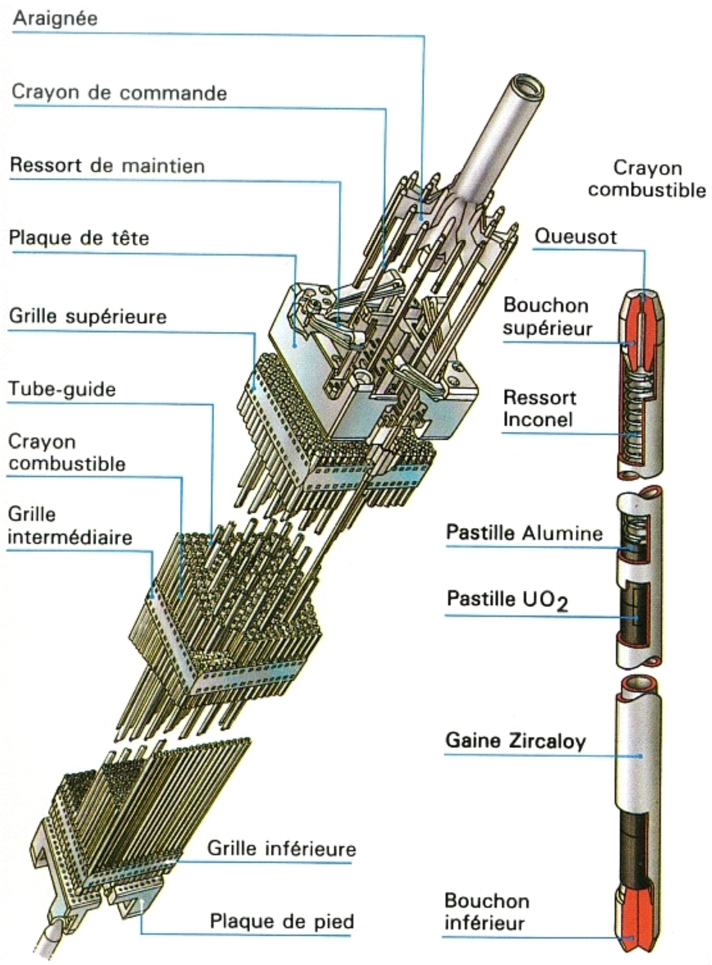


Figure 7.1.1 *PWR fuel assembly*

The assembly comprises a rigid metallic structure (skeleton) supporting 264 geometrically-identical rods distributed through a square lattice of 289 compartments.

The skeleton is made up of the following components:

- 2 fabricated end pieces, fixed by guide thimbles at each end, maintaining the position in the core;
- 24 tubes to guide the cluster rods (guide thimbles) and support the grids making up the structural framework;
- 1 central instrumentation tube with the sole task of guiding a flow measurement micro-chamber;
- 8 (AFA-2GE) or 10 (AFA-2LE) spacing grids to hold the rods axially and radially using a spring assembly.

The fuel rod is made up of:

- closed cylindrical, metallic cladding closed top and bottom by two welded plugs (the upper plug has a pressurisation hole which is plugged once the rod is filled with helium);
- stack of fuel pellets over roughly 95% of the rod length;
- a helical spring in the upper part of the fuel pellet stack, holding it axially during handling operations.

The AFA-2GE and AFA-2LE assemblies have identical configuration (lattice 17x17 with 12.6 mm intervals, section 214 mm x 214 mm), but different total lengths: 4.12 metres for AFA-2GE and 4.87 metres for AFA-2LE (dimensions after irradiation).

Material weights vary according to the type of fuel and type of assembly. They are given Table 7.1.2.

Table 7.1.2 Mass of PWR fuel assemblies (by material and total mass)

Assembly	Material weight per assembly (in kilograms)				Total weight of assembly) (in kilograms)
	Ceramic fuel	**Zirconium alloy (rod cladding)**	**Nickel alloy (end piece springs and grids)**	**Stainless steel (end pieces, rod springs, etc.)**	
AFA-2GE UOX/URE	521.2 (UO_2)	125.6	2.1	16.4	665
AFA-2LE UOX	608.1 (UO_2)	146.2	2.5	19.5	775
AFA-2GE MOX	513.9 ($(U\text{-}Pu)O_2$)	125.6	2.1	18.1	660

With respect to geological disposal study, one of the common issue to spent fuel and vitrified C waste is their considerable thermal release linked to their radiological inventory.

Nevertheless, the major contribution made by plutonium to fuel thermicity means a longer thermic phase; this difference is with vitrified waste due to the period of the involved isotopes, mainly americium-241 (^{241}Am), a daughter product of plutonium-241 (^{241}Pu).

Evolution over time in the residual heat rating of the UOX and MOX assemblies, after unloading from the reactors, is illustrated in Figure 7.1.2.

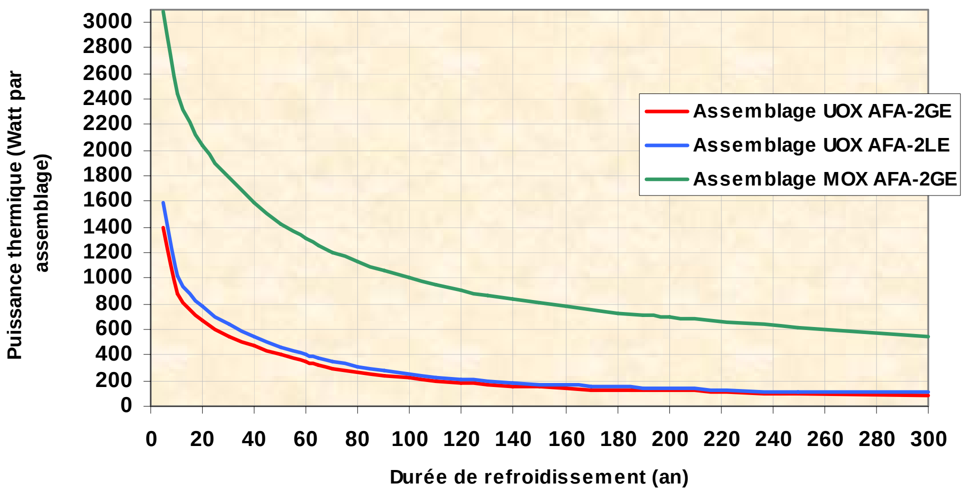


Figure 7.1.2 Evolution in residual heat rating of UOX and MOX fuel assemblies

In the previous figure, the heat ratings are deducted from the UOX and MOX fuel radiological inventories. For the specific UOX AFA-2GE and UOX AFA-2LE assemblies, an envelope inventory has been defined for a mixture of UOX2, UOX3 and enriched recycled uranium fuels.

7.1.1.2 CEA fuels (research reactors and National Defence)

Alongside the scenarios described above, CEA fuels are quite diverse. They include (i) fuels from UNGG reactors, (ii) fuels from the EL4 heavy water reactor, (iii) fuel elements from the Célestin reactors installed at Marcoule and (iv) nuclear propulsion fuels from land-based reactors or those on board of ships.

The UNGG fuels correspond to a residual tonnage of non-reprocessed fuels, around 15 tonnes. They are currently conditioned in cylindrical claddings 88 or 130 mm in diameter and equal to 655.5 mm high. These claddings contain a very small amount of waste, 9 or 18 kilograms on average depending on the type of cladding. They have a particularly low heat rating of around 3 watts maximum.

The EL4 fuels represent about 50 tonnes of heavy metal. The fuel element is in the form of a 19-rod cluster fitting tightly in an ATR structure (alloy of zirconium with copper and molybdenum). The rods are made up metallic cladding in a zirconium-copper alloy and contain uranium oxide pellets very slightly enriched with uranium-235 (1.28% or 1.41% depending on the rods). The initial weight of the uranium oxide is 10.6 kilograms per cluster. The EL4 fuel clusters are currently conditioned in stainless steel claddings about 100 mm in diameter and 1100 mm long. Each cladding contains two clusters placed on top of each other. Their heat rating is also very low (maximum 10 watts per cladding).

The Célestin fuel elements are made up of metallic plates containing enriched uranium, mounted on a metallic structure. They are conditioned in stainless steel claddings around 340 mm in diameter and 1100 mm long. Each cladding holds six fuel elements representing a total heat rating of 120 watts maximum.

The nuclear propulsion fuels are made up of (i) oxide fuels based on sintered uranium oxide plates and (ii) metallic fuels based on highly enriched metallic uranium. These latter fuels are no longer used.

In both cases, the fuel takes the form of an assembly made up of several bundles. The bundles are separated from the assembly and conditioned in identical diameter claddings (340 mm approximately, like the claddings containing the Célestin fuel elements), but of different lengths to suit the bundle dimensions. Each cladding contains four or six bundles from the same type of fuel assemblies. The heat rate here is at most in the same order of magnitude as for the C0 waste packages (155 watts) described in chapter 6.1.

7.1.2 Quantification of the number of spent fuel assemblies

The following table reproduces the hypotheses of the number of assemblies included in the study.

Table 7.1.3 Number of PWR fuel assemblies

	Production sites	Number of PWR fuel assemblies			
		Scenario S1a	Scenario S1b	Scenario S1c	Scenario S2
"Short" UOX AFA-2GE assembly, type CU1	EDF	0	0	0	27 200
"Long" UOX AFA-2LE assembly, type CU1		0	0	0	26 800
Total UOX assemblies, type CU1		**0**	**0**	**0**	**54 000**
"Short" MOX AFA-2GE assembly, type CU2	EDF	0	5 400	5 400	4 000
Total MOX assemblies, type CU2		**0**	**5 400**	**5 400**	**4 000**

Furthermore, 5,810 primary claddings are to be considered for type CU3 fuels, if appropriate.

7.1.3 Throughput hypotheses for spent fuel package reception

CU2 fuels identified in scenarios S1b, S1c and S2 would be received after a currently-estimated temporary storage period of 70 tot 90 years (according to granite temperature at disposal depth). At an annual rate of 150 assemblies, resorption of their stored inventory will take place over about forty years under scenarios S1b and S1c (or thirty for scenario S2).

Lastly, for CU1 fuels (scenario S1), the study considers the hypothesis of 1,650 assemblies per year.

Note that an annual reception rate of 400 primary claddings can be envisaged for CU3 fuels, corresponding to a resorption of their stored inventory over about fifteen years

These throughput hypotheses entail emplacing two disposal packages per day in the granite formation (see below).

7.2 Safety options for the design of a spent fuel repository

The general principles underlying spent fuel repository design in granite medium are the same as those formulated for the HLLL waste repository described in section 3. They entail long-term safety functions, waste disposal and management of the installations with a view to reversibility. The following principles are adopted for long-term safety:

- harnessing the favourable properties of the granite medium;
- designing engineered components that are complementary and redundant to the granite medium;
- limiting disturbances to the granite by the repository.

In the framework of these general principles, designing a spent fuel repository zone is similar to the vitrified C waste zone case. However it differs in that the assemblies release heat (§ 7.1) and that the radionuclides they contain have no primary conditioning. Managing the repository behaviour calls for incorporating the temperature criteria that restrict the number of packages per cell (heat rating management) and spacing between cells (heat dissipation management).

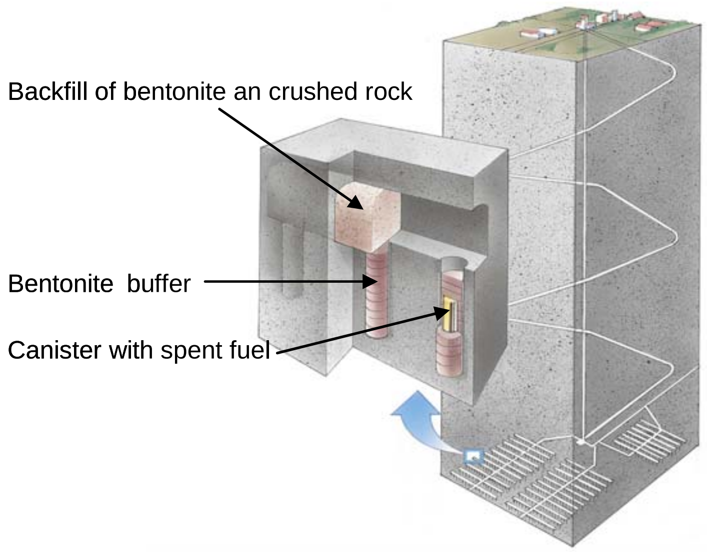
7.2.1 Harnessing the favourable properties of the granite medium

As in the case of C waste, harnessing the favourable properties of the granite medium happens on two levels, firstly on the scale of the cells by seeking sites in those parts of the rock where there is very little fracturing, and secondly at the scale of the disposal module, that are developed away from water-conducting faults (§ 6.2.1).

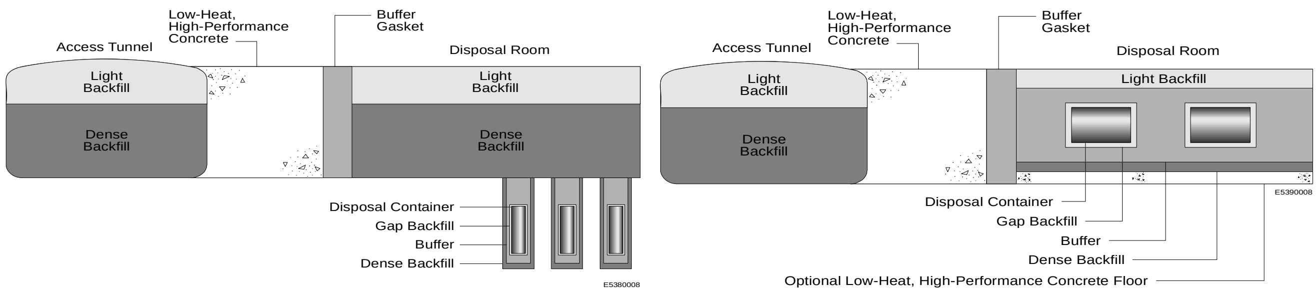
7.2.1.1 The reference cell: small disposal boreholes

International studies have been conducted on different types of cells, horizontal tunnels and vertical boreholes for spent fuel disposal (Figure 7.2.1 [53]).

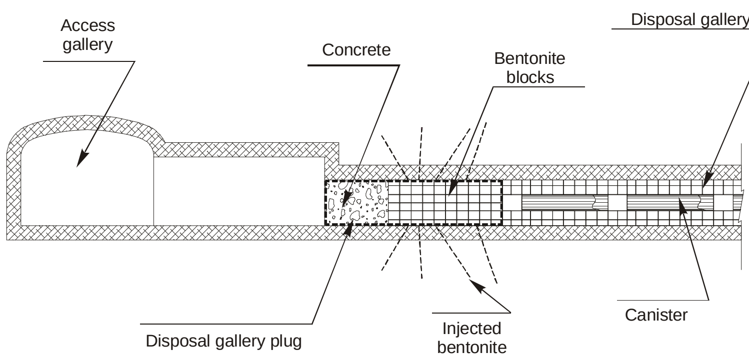
The Swedish KBS-3 reference design is based on the disposal of spent fuel in short vertical boreholes, called as well pits (KBS3-V) [54]. However the Swedish agency SKB, and the Spanish agency, ENRESA are also studying a horizontal design. In Canada both the horizontal and vertical options have been studied.



Sweden : KBS-3V design of a disposal cell in a borehole (source: SKB)



Canada: Repository design in vertical boreholes or horizontal tunnels (data: OPG, Canada).



Spain: Spent fuel disposal tunnel design with its seal
(ENRESA data)

Figure 7.2.1 *Types of cells studied abroad for a spent fuel repository*

At this generic stage of the studies for spent fuel disposal cells, after examining the available statistics on how minor fractures in French granites are organised, the most suitable option is the shallow vertical dead-end borehole as adopted for C waste. This option is again taken up as the reference design.

A single package per borehole is envisaged given the foreseeable dimensions of the spent fuel disposal packages. Furthermore limiting borehole capacity to a single package will also simplify control of cell and module alteration (primarily thermal). Thus the option proposed is a borehole with a limited length of about 8 metres, and a diameter of less than 2 metres, to enable a clay engineered barrier to be placed between the disposal package and the rock.

7.2.1.2 Installing disposal modules in "blocks" of granite away from water-conducting faults

The spent fuel disposal modules, like the C waste modules, are installed in the granite away from the faults usually of several hundreds metres in dimension; such faults are too water-conducting (Figure 7.2.2) to be intersected by module drifts and in particular by handling drifts. This principle, leads to module separation in the repository zone architecture according to granite massif fracturing. The adaptation of the architecture to this fracturing is once again based on the definition of criteria related to the properties of these faults and their on-going characterisation (cf. § 4.3.1).

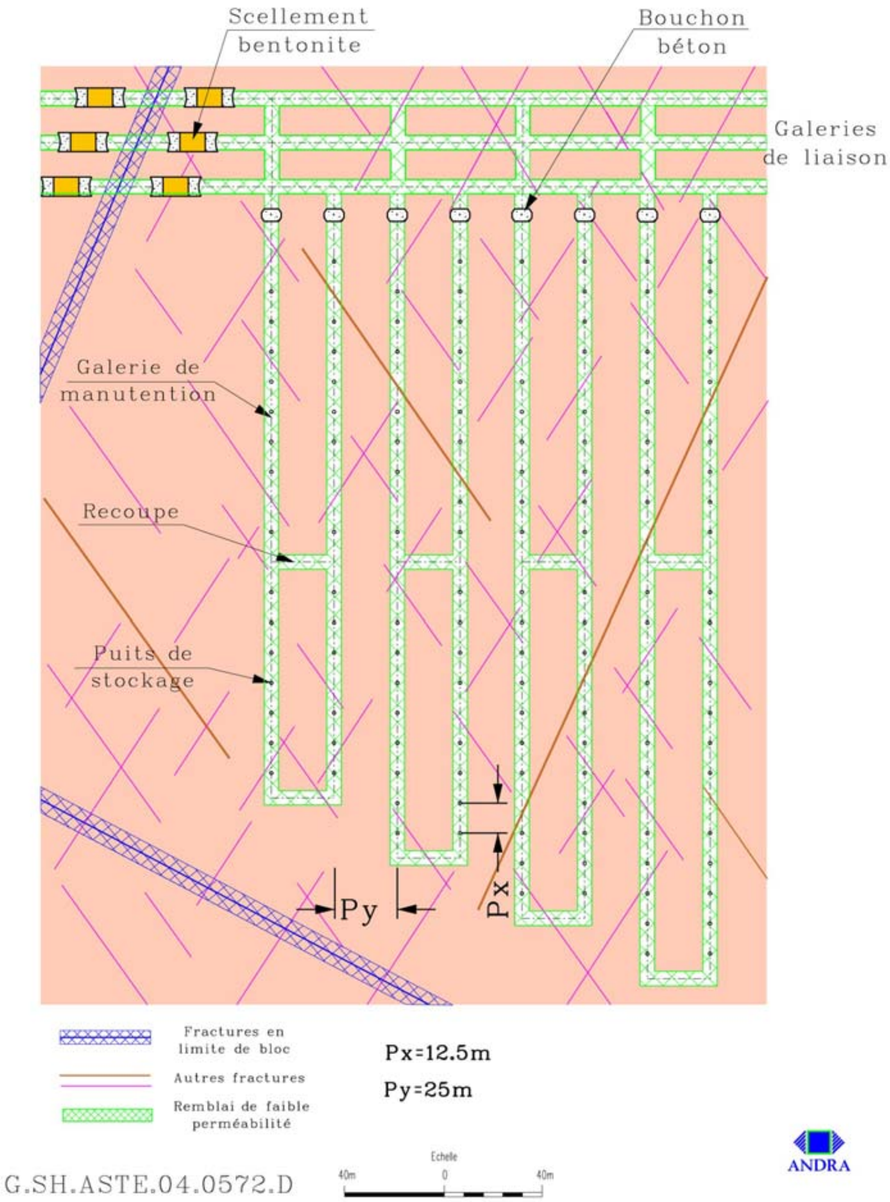


Figure 7.2.2 Principle of the arrangement of a spent fuel disposal module

Examples of module architecture resulting from the same approach in Sweden are illustrated in Figure 7.2.3 [54]. Swedish agency SKB envisages the possibility of two-level architecture in the case of more fractured geology.

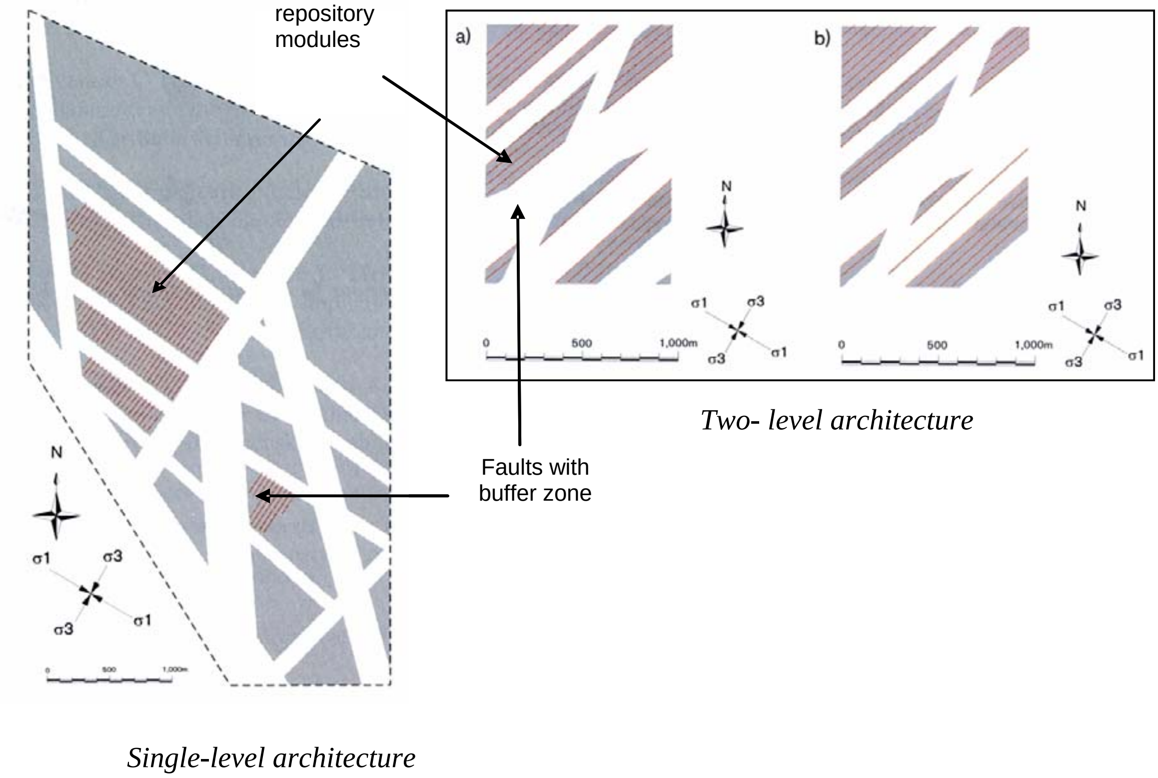


Figure 7.2.3 *Sweden: Examples of spent fuel repository architecture for two different granite sites*

7.2.2 Designing engineered components that are complementary and redundant to the granite medium

The intersection of faults by connecting drifts in the repository zone, as expected for the vitrified waste repository zone, cannot be ruled out. Hence both low-permeability backfill in the handling and connecting drifts, and seals at the module edges are similarly envisaged to complement the cell and module arrangement principles.

7.2.2.1 A container that must be watertight over long periods: the example of the Swedish KBS-3 design with a copper canister

The spent fuel assemblies are likely to release radioelements faster in the event of water ingress from the repository zone underground installations as they have no confinement matrix such as C waste glass. Consequently the disposal packages are designed to be watertight over very long periods.

Accordingly, given that the study has been conducted without a specific site, it has been decided to take stock of the Fenno-Scandian technical studies and particularly the knowledge acquired from the "KBS-3" design with the copper canister that offers long-term water-tightness. This design relies on copper's stability in the typical chemical environment conditions that prevail in deep granite. Thus the repository is designed to maintain these conditions at the timescale included in the safety analyses, namely several hundred of millennia.

Sweden opted for the "KBS-3" copper canister design back in the 1980s. Both SKB in Sweden and Posiva in Finland use this same reference for their repository studies in a granite medium (Figure 7.2.4).

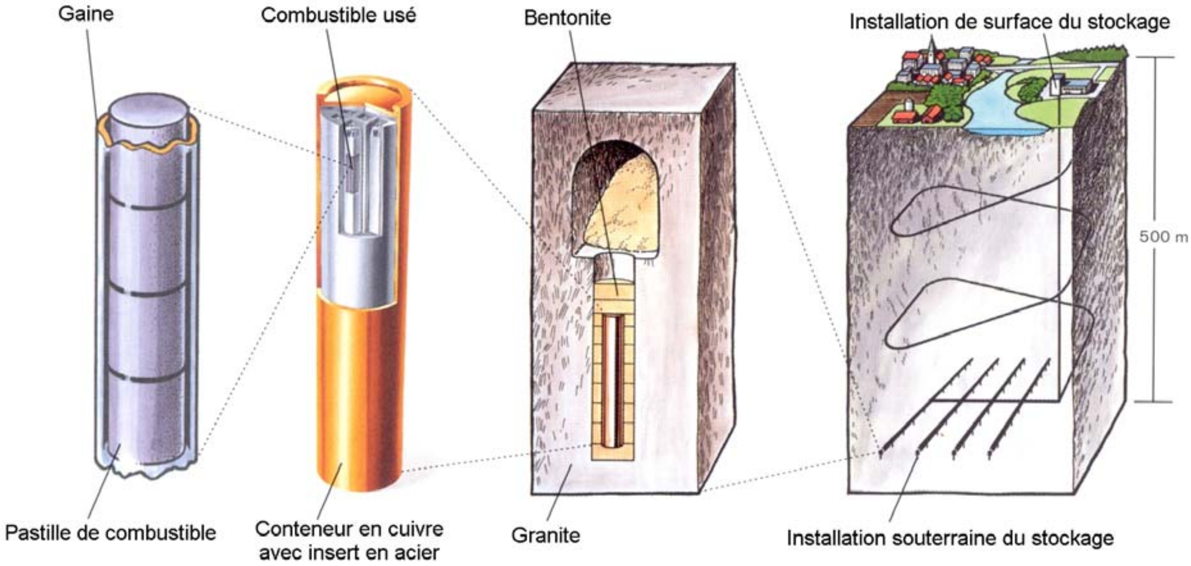


Figure 7.2.4 The Swedish KBS-3 spent fuel design (SKB data [58])

The issues concerning the canister's feasibility have essentially dealt with its manufacturing and the welding techniques to be used on the copper envelope. These issues have been covered by an extensive research programme which has demonstrated the possibility of manufacturing and welding the canisters (cf. § 7.3.3) [55].

Furthermore full-scale experiments are being conducted to test the main elements of the KBS-3 design in the Äspö underground laboratory (Hard Rock Laboratory) in Sweden.

The swelling engineered barrier placed around the disposal package contributes to maintaining compatible chemical environment conditions with the thermo-dynamic stability range of canister copper. In addition, the swelling clay engineered barrier provides mechanical stability for the disposal cell. Its contribution as chemical buffer affects both the water arriving from the granite and the water from the handling drifts. The possible short and long-term chemical interactions with the canisters are taken into consideration when formulating the engineered barrier.

From a thermal viewpoint, the management of the evolution of the copper canister and the engineered barrier imply that temperature of the canister surface must not exceed 100°C. This calls for the spent fuel to be stored on surface prior to disposal and for an appropriate dimensioning of disposal module in terms of distance between cells and of spacing between handling drifts (cf. § 7.4 and 7.5).

Looking at the granite hydro-geochemical characteristics and their long-term evolution, it must be point out that copper canister dimensioning in Sweden is based on conditions that are in principle more restrictive (with glacial context) than those likely to prevail in France.

Lastly the requirements of canister durability could be lower, in the case of certain granite sites that have more performing properties than those considered generically. The adaptation of the canister design could then be based on the studies conducted by the Spanish agency ENRESA that aims to place spent fuel in durable steel canisters in the disposal cells for several millennia (Figure 7.2.1).

7.2.2.2 Redundant backfill and drift seals

The two redundant measures envisaged to prevent water circulation in the C waste repository zone modules are considered again for the spent fuel disposal modules: backfilling the drifts with a low-permeability material and the emplacement of very low permeability seals (cf. § 6.2.2.1).

7.2.3 Limiting disturbances to the granite by the repository

The disturbances likely to occur to the granite as a result of adding a dedicated spent fuel zone to the repository do not differ from those that should be limited and controlled by vitrified C waste disposal process or structure dimensioning (cf. § 6.2.3).

7.3 The spent fuel canisters

7.3.1 Selected design principles

The spent fuel canister taken as the reference in the studies is the KBS-3 type canister designed on the following principles:

- a copper envelope thick enough in terms of corrosion, to provide water-tightness over long periods (one million years);
- an insert strong enough to provide the canister with long-term mechanical stability.

Figure 7.3.1 View of the copper envelope and an insert (source: SKB)

The copper envelope dimensions for French spent fuel are adapted to accommodate two types of four-housing inserts for CU1 assemblies (short: UOX AFA-2GE or long: UOX AFA-2LE) and the single-housing insert for CU2 (MOX AFA-2GE).

7.3.2 Description and dimensioning

7.3.2.1 A copper envelope

The copper envelope, comprises a cylindrical body onto which are welded a base and a lid. SKB has specified its thickness at 50 millimetres.

SKB has demonstrated the feasibility of manufacturing tubes 50 millimetres thick with no lengthwise weld, using a number of standard metal industry processes: extrusion, boring and drawing, forging.

Specification of the canister's thickness is based not only on mechanical dimensioning, but also corrosion studies of the copper envelope. The latter indicate that degradation of the copper's properties should not affect more than about fifteen millimetres of envelope thickness, given all the uncertainties surrounding the chemistry of the geological medium and the potential consequences of the disturbances generated by the disposal process (oxygen increase, bacterial contamination) [55]. Mechanical dimensioning is based on pressure resistance criteria relating to the pressures on the envelope after cell closure. The copper envelope should be able to deform around the rigid insert without jeopardising its integrity (water-tightness).

7.3.2.2 A cast iron insert

Inside the copper envelope, there is a cast iron insert comprising a one-piece body cast around one or more housings (planned to receive spent fuel assembly) and a bolt-on lid. The geometry of the housings is designed for bare assemblies, with a square base, or assemblies in cylindrical long-lasting cladding (ELD). They are dimensioned to minimise the voids.

The insert is dimensioned to take up the mechanical efforts borne by the disposal package after cell closure.

Spacing between the housings is determined so that the package should remain sub-critical from design stage without the need for adding a neutron-absorbing element. However it is limited so that if this option is adopted, the total mass of the package remains within the scope covered by industrial feedback for transfer to a borehole.

The minimum cast iron thickness between a housing and the insert outer wall is defined bearing in mind mechanical considerations and also implementation of the insert casting process. The reference thicknesses adopted are 50 mm for the sides and 100 mm at the base of the insert.

The upper part of the insert is machined to allow for the insertion and bolting of the 50-mm thick cast iron lid.

7.3.2.3 Description of the spent fuel canisters

The dimensions of the spent fuel disposal canisters, illustrated by Figure 7.3.2 are given in the following table:

Table 7.3.1 External dimensions of the CUI and CU2 disposal packages

Reference packages	Type of assemblies	Disposal package dimensions	
		Height	Diameter
CU1	AFA-2GE or "short" assembly	4.5 m	1.15 m
	AFA-2LE or "long" assembly	5.25 m	1.15 m
CU2	-	4.5 m	65 cm

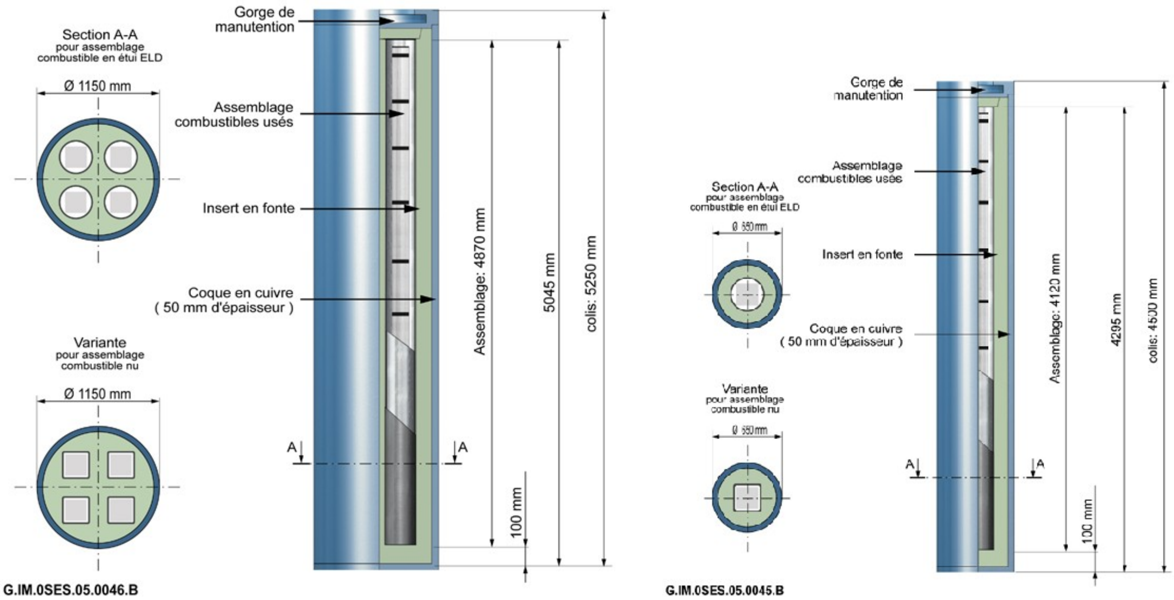


Figure 7.3.2 *Spent fuel canister. On the left the CU1 "long" (AFA - 2LE assembly) and on the right the CU2 (AFA - 2GE assembly).*

7.3.3 Manufacturing techniques

SKB initiated the feasibility demonstration for the manufacture of canisters with a copper envelope in 1994. The main elements of this study were as follows:

- manufacturing cylindrical copper envelopes, bases and lids;
- welding the base or lid onto the copper cylinder;
- manufacturing cast iron insert partitions.

7.3.3.1 Manufacture of thick, wide diameter copper tubes

While manufacturing inserts is based on current, established foundry techniques, SKB concentrated its studies from a very early date on the manufacture of copper tubes with no lengthwise weld. It tested three possible alternatives:

- manufacturing tubes by extrusion;
- manufacturing tubes by boring & drawing;
- manufacturing tubes by forging.

A feasibility demonstration was carried out in 2003 by manufacturing 22 weld-free tubes, including 15 by extrusion, 5 by boring & drawing and 2 by forging.

All these methods result in the production of copper cylinders ready for machining to the desired finished dimensions, both internally, externally and at both ends of the cylinder.

SKB has also tested the extrusion process presented below in relation to C waste repository package manufacturing techniques (chapter 7.3.3).

- **Extrusion process**

Essentially the extrusion process consists of constructing a hollow ingot then extruding it using a hydraulic press:

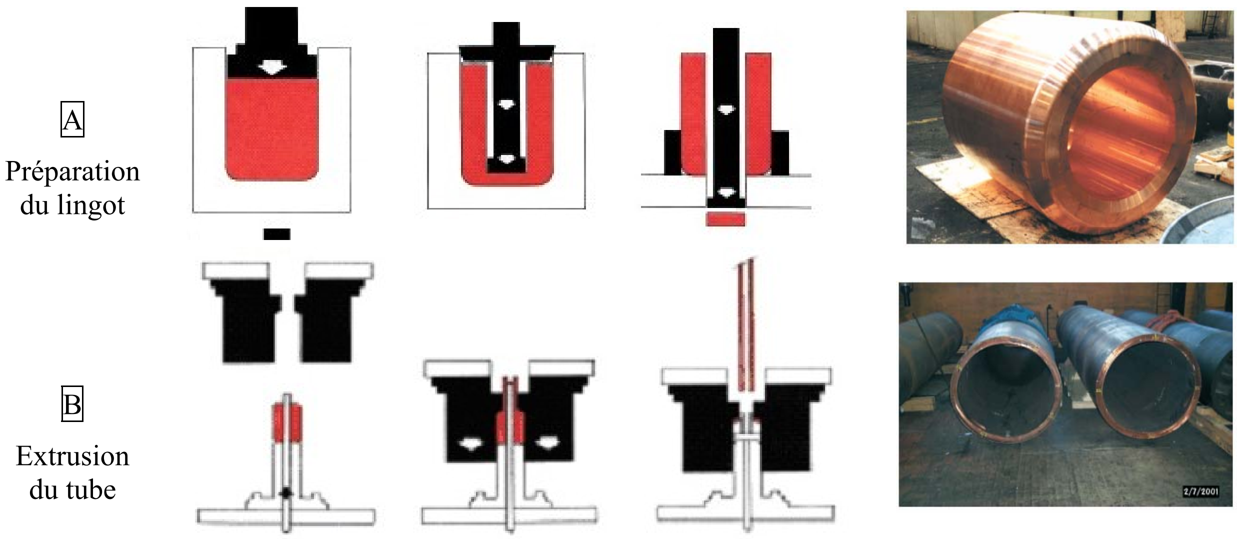


Figure 7.3.3 Principles of copper tube construction 50 mm thick by extrusion

Extrusion differs from the boring & drawing method in it that requires factory welding at the bottom of the copper envelope.

7.3.3.2 Welding processes in an irradiating atmosphere

Two methods have been tested for welding the lid of the copper envelope, which will be carried out in an irradiating atmosphere (as the CU assemblies will be already in position in the cast iron insert):

- the friction stir method;
- electron beam welding – the same method as used in the manufacture of C waste over-packs (chapter 5.3.3.2).

The two methods can be implemented on the tube construction site (in a non irradiating zone) for the copper canister base weld.

- **The friction stir welding method (FSW)**

The friction stir welding process (FSW) was invented in 1991 at the Swedish Welding Institute (The Welding Institute TWI). It is a thermo-mechanical solid phase process that combines extrusion and forging.

A cylindrical, profiled tool is rotated under pressure (friction) along the joint line between the two elements to be welded. The heat of the friction produced between the wear-resistant tool and the joint faces of the parts to be welded enables the tool to plunge slowly into the copper (stirring). By controlling the exerted pressure and the rotation speed, the copper does not melt. The tool is then moved along the weld line. The transfer of plasticised copper along the tool probe (non-consumable), either side of the two edges submitted to friction, creates a continuous solid phase bond between the two parts.

A diagram of this process is presented in Figure 7.3.4.

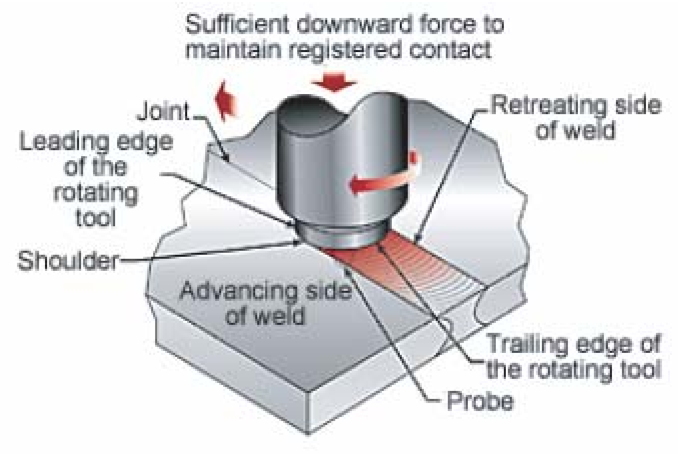


Diagram of process

Start and finish of weld line

Metallography in a cross section of the weld line (steady state zone)

Figure 7.3.4 Friction stir welding process

The results of the Swedish FSW process study program have so far demonstrated that this process can be used to seal 50-millimetre thick copper canisters producing welds with a high level of integrity.

- **The electron beam welding method**

Electron beam welding (EBW) consists of creating, accelerating then centring an electron beam on a weld line (cf. 6.3.3.2). Conversion of the kinetic energy of the electrons into heat causes localised melting of the metal either side of the weld line.

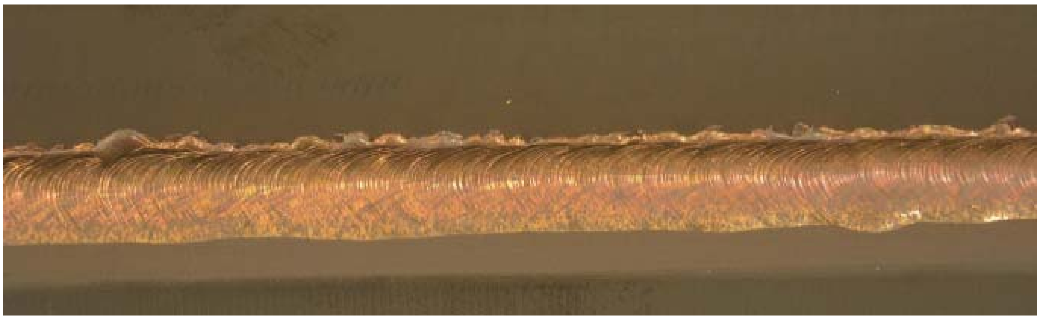


Figure 7.3.5 Welding surface with an oscillating electron beam

7.3.4 Non-destructive tests of the integrity of the copper envelope

The non-destructive testing methods (NDT) used to check the quality of the copper canister lid welding are:

- digital radiography testing;
- ultrasonic testing.

SKB's studies on the manufacture of copper envelopes included reliability testing of NDT systems and procedures on over twenty welded lids using radiographic and ultrasonic inspection, and on over one hundred test samples exclusively by radiography.

It should also be mentioned that an electromagnetic inspection system is currently in development.

- **Digital Radiography (DR) testing**

The main aim of this radiography system is to inspect any volumetric defaults presented by the lid weld.

The system designed by SKB enables 1-mm pores to be detected in a weld carried out over the whole circumference of a canister (Figure 7.3.6).

Figure 7.3.6 *Radiographic equipment for testing the electron beam weld of the copper canister lid*

The major advantage of the radiographic process is that it provides a full image of the weld with precise default identification, both in volume and position (on the circumference).

However, destructive testing carried out subsequently and randomly on the X-rayed welds has shown that this identification was sensitive to canister centring and ovalisation.

Furthermore this method presents the following limitations:

- positioning the discontinuities in the radial direction is delicate (in the weld thickness);
- small discontinuities are hard to detect in the axial direction of the canister.

- **Ultrasonic testing (UT)**

The ultrasonic testing consists of analysing the reflections of ultrasonic signals on the surface between two metal parts welded together. Interpreting the reflected signals pinpoints some of the non-volumetric faults, such as pockets where there is no bonding because the structure of the metal remains unmodified.

At the current stage this test is planned to supplement the radiographic test. Furthermore it can determine the thickness of the metal whose structure has been modified by the weld.

7.4 The disposal cells

The disposal cell is a vertical, dead-end borehole with a swelling clay engineered barrier. It is dimensioned to take one single container of spent fuel.

As in the case of the C waste package disposal borehole, the choice of layout and borehole depth is governed by the need to limit the statistical possibilities of intersecting minor sub-vertical and potentially water-conducting fractures.

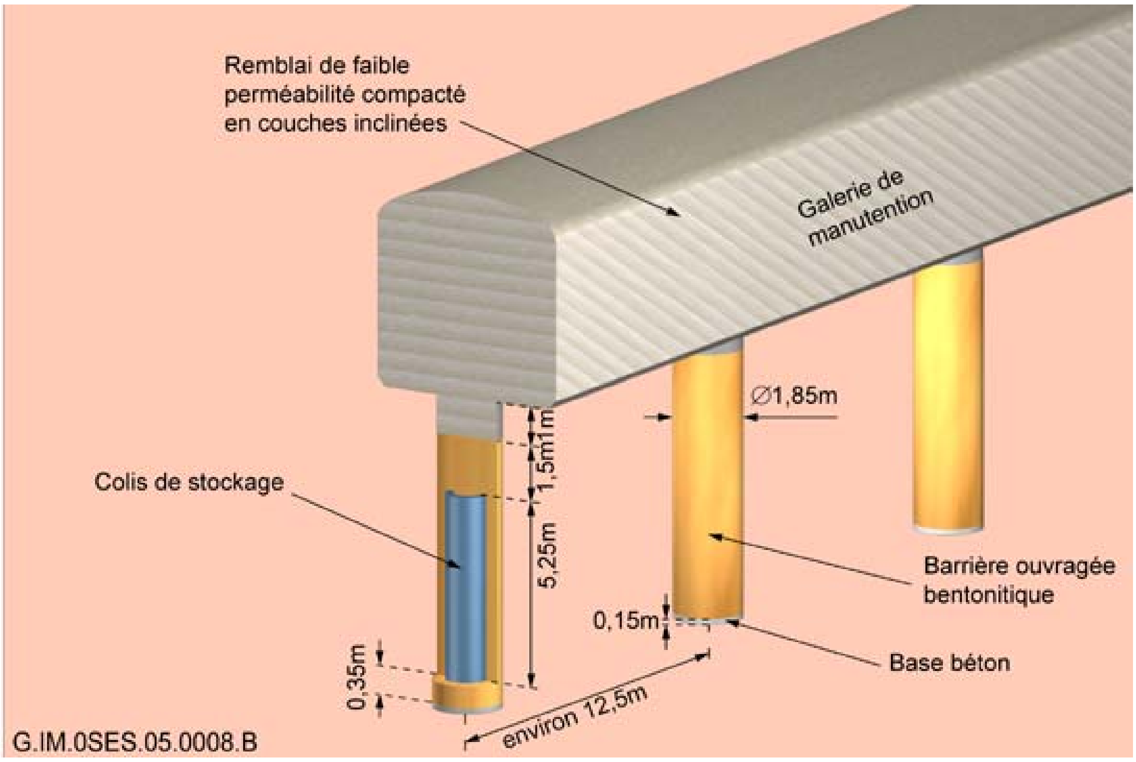


Figure 7.4.1 View of a spent fuel disposal cell

7.4.1 Description of the disposal cell

The design of the spent fuel disposal cells is very similar to C waste ones. However they differ in that there is no metal sleeve between the disposal package and the engineered barrier. This is because the proximity of an element in steel and the copper canister is likely to create galvanic corrosion conditions that could alter the canister's durability.

Thus the disposal package is emplaced directly in contact with the engineered barrier.

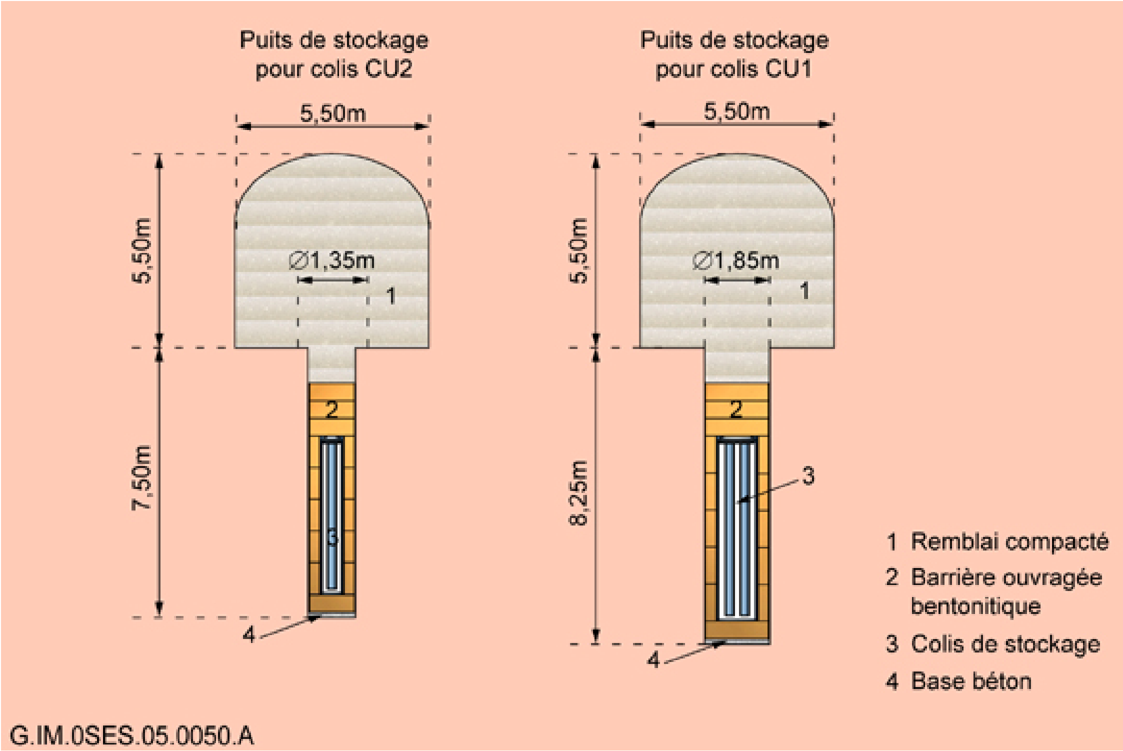


Figure 7.4.2 General dimensions of CU cells

- **The engineered barrier**

The swelling clay engineered barrier is expected to have identical properties to those of the C waste cells. Its function is similarly to protect the copper envelope from chemical (site water and work-induced oxygen) and biological (bacterial contamination during the disposal process) attack.

According to SKB studies, atmospheric oxygen is rapidly consumed after cell closure [55]. Controlling the presence of sulphur in the vicinity of the canister could result in selecting bentonite (such as MX80) with only some traces of pyrite. Likewise bacterial corrosion could be controlled by defining a criterion for the minimum engineered barrier density above which bacterial life would be unviable [17].

The choice of swelling clay and its formulation (as % of sand) can only specified for a specific site because French granite massif groundwaters vary so widely in chemical composition (cf. § 3.2.2.2) The specification could be based on the results of research and experiments carried out internationally and also for the repository study in the clay medium.

The engineered barrier is made up of prefabricated stacked bentonite elements. Its emplacement necessitates clearances between the borehole wall and the outer surface of the engineered barrier rings. Clearance is also provided between the engineered barrier rings and the disposal package for package lowering operations.

- **Disposal cell dimensions**

To maintain favourable long-term environmental conditions for the stability of the cell and the copper canister an engineered barrier of 35 cm thick around the disposal packages is required.

The resulting disposal cell dimensions vary from 1.35-1.85 m in diameter and 7.5-8.25 m deep, depending on the dimensions of the stored package. All these characteristics are summarise Table 7.4.1.

Table 7.4.1 Functional axial dimensions of disposal cells

Component	Description	CU2	CU1	
			"Short"	"Long"
			"Court"	"Long"
Cell head backfilling	Height of the backfill in the disposal borehole	1 m		
Cell plug	Plug thickness	1.5 m		
Disposal package	Total length	**4.5 m**	**4.5 m**	**5.25 m**
EB at the base	EB thickness under the package	0.35 m		
Concrete base	Thickness of the concrete base	0.15 m		
Disposal borehole	***Total "length"***	***7.50 m***	***7.50 m***	***8.25 m***

7.4.2 Description of a handling drift

The disposal cells are arranged along handling drifts, whose axial dimensions are governed by the geometry of the host block. These drifts are of the same design as the vitrified C waste repository module drifts.

They are dimensioned to be mechanically stable with no need for systematic rock supports. Drift section is adapted for use by similar construction and operating machinery to that planned by SKB. They are linked two by two by interconnecting drifts installed roughly at 200 m intervals.

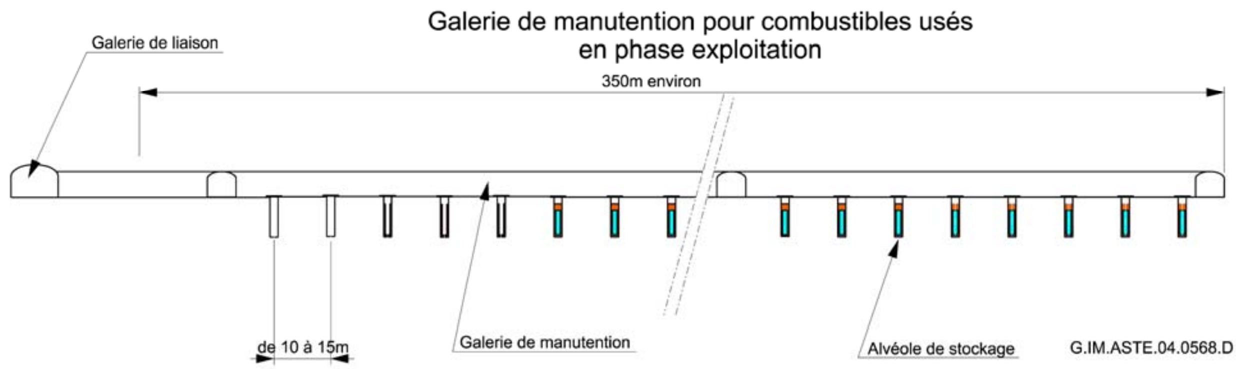


Figure 7.4.3 Cross section of a handling drift

7.4.2.1 Handling drift dimensioning elements

By using a tilting machine to emplace the packages in the cell, drift height can be limited to lower value. The required drift dimensions are about 4-6 m wide and 5.5 m high right above the appliance for a fairly similarly designed machine to the one selected by SKB (see chapter 7.6.2).

The geometry of the handling drift sections as in the case of the other drifts, is adapted to the mechanical characteristics of the granite for long term stability with no other rock support than the required one for operator safety. "Inverse U" geometry adapted to the technological dimensioning elements (heights at walls and at the arch, width) is adopted as the reference.

7.4.2.2 Assessment of the total running length of handling drifts

Spent fuel repository compactness in a handling drift depends on the distance (pitch) in between two neighbouring cells. As the spent fuel packages release heat, this distance is primarily subject to the application of the thermal criterion for the presence of the engineered barrier around the disposal packages (90°C).

Very close results have emerged from preliminary calculations to test the sensitivity of this distance at various estimated thermal characteristics for French granites.

For distances 12 and 15 m between two cells, and cooling periods of 50-70 years for reference packages CU1 and 70-90 years for reference packages CU2, the thermal interference reaches its maximum at the midpoint in between 2 disposal cells a few centuries after the package temperature has dropped below 90°C, [42].

7.5 Architecture of the spent fuel repository zone

The spent fuel repository zone includes the repository modules for reference packages CU1 and CU2 (distributed over two levels) and their connecting infrastructures.

A disposal module comprises a set of handling drifts served by a shared cluster of connecting drifts, as specified for the vitrified C waste repository zone.

The disposal capacity of each repository module depends on the number of handling drifts which can be sited in the host granite block, whose geometry has been determined by previous exploration works.

This section presents the dimensioning elements, followed by a description of the modules in the repository zone. It gives a preliminary assessment of how many modules need to be constructed and of the necessary area for spent fuel disposal on the basis of the inventory model.

7.5.1 Dimensioning elements of a repository module

Dimensioning a spent fuel disposal module is subject to identical considerations to those that prevail for C waste modules:

- the distance between two adjacent drifts equal to three times the drift width (that is about fifteen metres) to avoid any mechanical interference;
- the distance of 12-15 metres between adjacent cells to avoid thermal interference between cells;
- the buffer distance from several tens of metres up to one hundred metres, between the excavated structures and the potentially water-conducting fractures at the host block limits (this range covers all foreseeable situations in the French context).

7.5.2 Description of a repository module

A repository module is a set of handling drifts and connecting drifts.

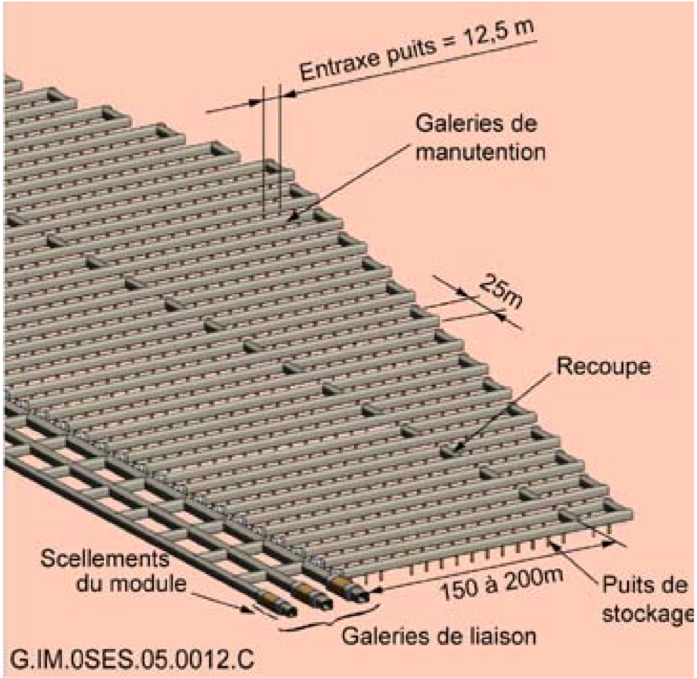


Figure 7.5.1 Schematic representation of a module for CU packages

The handling drifts are served by a cluster of two to three connecting drifts that are dimensioned in the same way as for C waste repository module to enable the module exploration, construction, operation and closure operations to be carried out successively (cf. § 6.5.2).

A number of simulations have been carried out at this stage, to determine the module construction and operating time. Estimated module exploration and construction time is two to three years longer than module operation time, just as it is for C waste repository modules. It takes four to six years to prepare a module for reference packages CU1 and about four years for a module for reference packages CU2, which will be operated respectively for eighteen months and two years [42].

7.5.3 Assessing the number of spent fuel modules

With no specific site data, an indicative estimate has been made of the number of modules in the spent fuel repository zone on the basis of the mean number of modules in operation over eighteen months for CU1 and two years for CU2. For scenarios S1b and S1c, 18 modules are needed to host the CU2 packages. This number would rise to 38 in the case of scenario S2 in which 24 modules would be exclusively dedicated to CU1.

About fifteen blocks will have to be identified for two-level architecture in the case of scenarios S1, and 20-25 blocks for scenario S2.

7.6 Disposal process as part of the reversibility rationale

7.6.1 Characterisation of the granite blocks and construction of the modules

The same host block exploration and module construction methods are used as for vitrified C waste repository structures given the close similarities between them and spent fuel repository structures.

Exploration relies on drift and borehole work. Modules are constructed using blasting techniques and drifts using drilling techniques with a micro-tunnel boring machine adapted to vertical operation (see chapters 6.6.1 - 6.6.4).

7.6.2 Operation of a cell as part of the reversibility rationale

The equipment design and spent fuel cell emplacement process have been transposed from SKB demonstration carried out in the Äspö laboratory [56].

7.6.2.1 Description of the transfer and cell emplacement equipment

- **The transfer cask**

The spent fuel transfer cask comprises a cylindrical shielded vessel ("steel and neutron-absorbing material"), with walls about 200 mm thick. It is made up of two parts that are screwed together for the time of transport [57].

The upper part (on the right of Figure 7.6.1) is fitted with a grapnel to grip the disposal package. The dimensions of the transfer cask are about 6 m long, a diameter of about 1.70 m with an unloaded mass of about 30 tonnes and 75 tonnes when loaded.

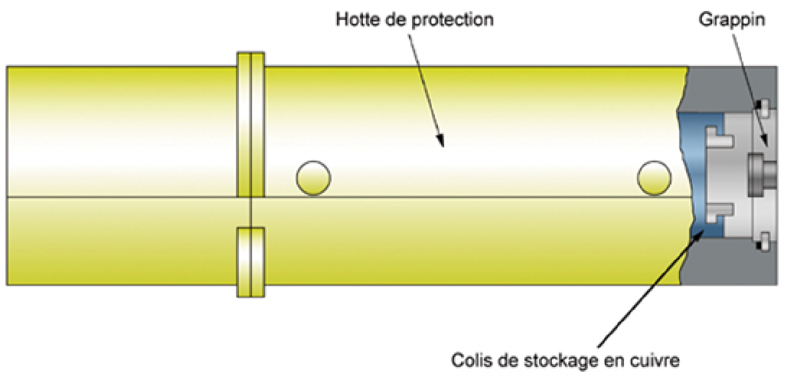


Figure 7.6.1 Radiological protection cask for CU packages

- **The transfer vehicle**

Transfer of the cask from the surface installations to the underground installations is carried out using exactly the same approach as described earlier for B and C waste. Once the transfer cask is at the repository level, it is placed on a transfer vehicle that carries it to the emplacement equipment located near the host cell.

The vehicle is fitted with independent multi-directional wheels giving a very tight turning circle. The vehicle's onboard mechanical equipment enables the transfer cask to be pulled or pushed horizontally inside the emplacing equipment. Its dimensions are roughly 12 m long by 4 m wide by 3.80 m high. Its unloaded weight is about 40 tonnes.

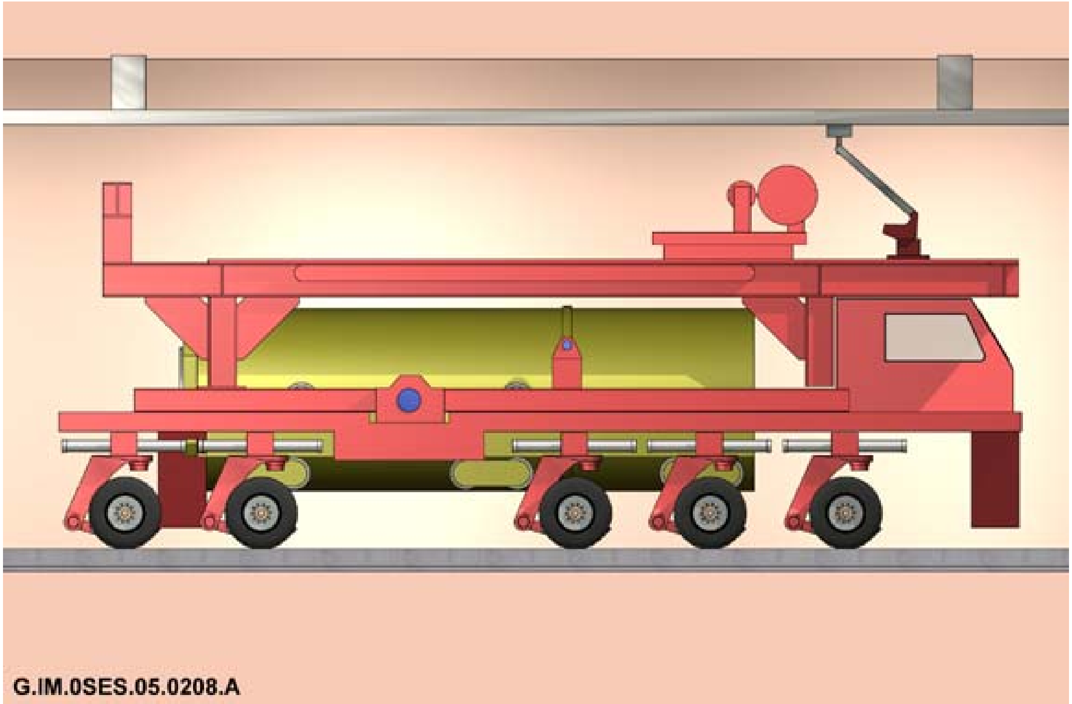


Figure 7.6.2 Transfer vehicle

- **Emplacing equipment**

The emplacing equipment is designed to take the transfer cask conveyed by the transport vehicle and then travel on tyres until reaching the position right above a disposal borehole, open the transfer cask, place the package in a disposal borehole by tilting the cask using a combined "rotation-shift" movement and finally place a bentonite plug on top of the package.

It includes a fixed radiological protection chamber and a mobile circular radiological protection device that accommodates the transfer cask (Figure 7.6.3).

The emplacing equipment dimensions are about 12 m long by 3.70 m wide by 4.60 high. Its unloaded weight is about 70 tonnes and 145 tonnes when loaded.

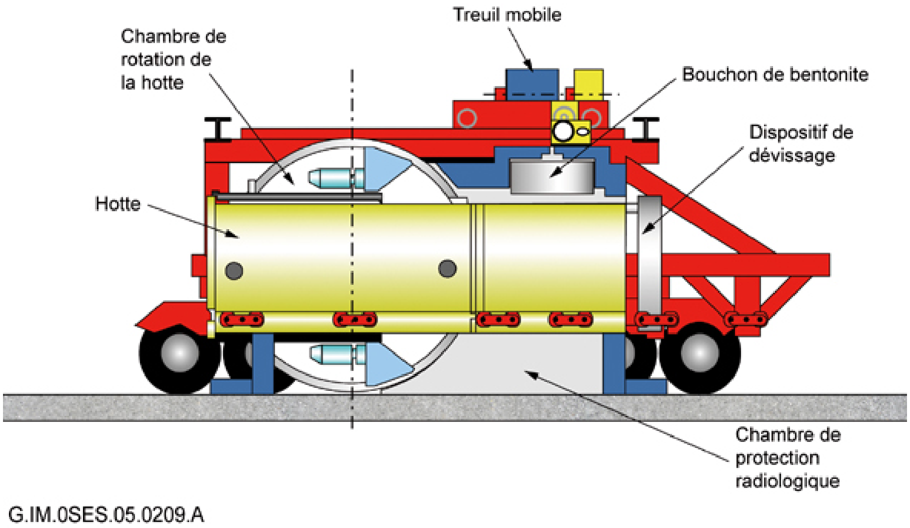


Figure 7.6.3 CU package emplacing equipment

7.6.2.2 Transfer and cell emplacing process

Loading the transfer cask is carried out vertically in the surface installations using its onboard clamp. Once closed, the transfer cask is lowered to the bottom via the package transfer shaft or the ramp. The transfer vehicle conveys it to the emplacing equipment.

The following operations form the package emplacing sequence, illustrated in Figure 7.6.4:

- transhipment of the cask into the emplacing equipment;
- simultaneous shifting and tilting of the emplacing equipment;
- package emplacement;
- clay plug emplacement.

The principle of simultaneously tilting and shifting of the emplacing equipment aims at minimising drift height.

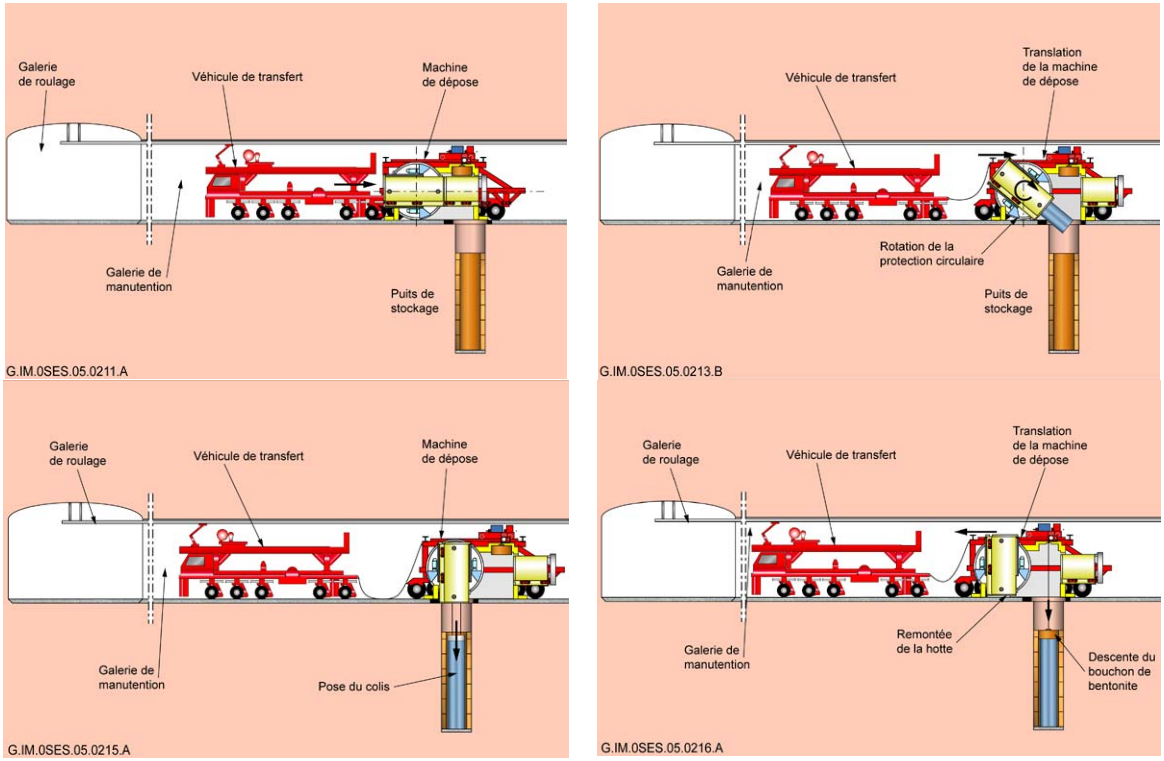


Figure 7.6.4 Sequence of CU package emplacement in a cell

In the event of package retrieval, the engineered barrier would have to be removed to extract the canister. This operation is under demonstration in the Aspö laboratory (Canister Retrieval Test). It consists of destroying the swelling clay by using processes tested during the "Slurrying Test" experiments in 2002 and 2003. They use high-pressure saline sprays to gradually erode the engineered barrier around the package and then pump out the resulting slurry.

7.6.3 Risks associated with spent fuel package emplacement in the cell

A package grasping grapnel malfunction may cause a drop of spent fuel package during emplacement (or retrieval as the case may be). Despite the designed safety systems, it cannot be ruled out.

Simulation of the consequences of a vertical drop has been performed a CU1 package as part of the repository study for a clay medium. That package has a similar structure to the one adopted for a repository in a granite medium but its envelope is made of steel rather than copper hull. This simulation, that applies the hypothesis of a 7-metre drop onto a deformation-resistant floor, has demonstrated that plastic deformations could affect the base of the package’s cast iron insert, but that the spent fuel and assembly housings themselves would not be ruptured. This would appear to be more pessimistic than the case of a container with a copper hull (whose envelope is considerably easier to deform) which would drop on the bentonite engineered barrier.

Thus the risk of a spent fuel package falling during emplacement apparently cannot lead to a package breaking and losing confinement of its radioactive material.

7.6.4 Closing the structures and observation

Closing the handling drifts, module or a group of modules, or a repository level requires the backfill and seals to play a complementary and redundant role in the same way as they do for C waste (chapter 7.6).

Likewise, the closure process can be conducted continuously or by successive stages, accompanied by the implementation of an observation programme for the structures as and when they are closed. In any event, it is possible to reverse the process, backtracking to an earlier stage, right back to package retrieval, even if the deconstruction of more substantial closure structures is required as closure progresses further.

7.7 Functions of the repository components over time

The previous sections describe the various components of a spent fuel repository. They specify the means used to construct a reversible disposal system. They expose the principles at the basis of their design and dimensioning for long-term safety.

The following table summarises how the various components contribute to the main functions of a repository during the various repository phases (§ 1.1).

Table 7.7.1 Functions of the repository components over time

COMPONENT	PERIOD	MAIN REPOSITORY FUNCTIONS	PROPERTIES HARNESSED
1. GEOLOGICAL MEDIUM: GRANITE			
Part of the granite rock where a disposal cell is installed	During operation and the reversibility phase After closure	Emplacing (and being able to retrieve) the packages in the cells Protecting the disposal cells from water circulation Delaying and reducing radionuclides migration to the environment	Mechanical strength Very low permeability Chemical properties of the granite water that tend to delay radionuclide migration (reducing environment, pH, etc.)
Granite "block" where the disposal module is installed	After closure	Protecting the disposal cells from water circulation Delaying and reducing radionuclides migration to the environment	Low permeability (at a distance from the significant water-conducting fractures) Low underground hydraulic gradients Chemical properties of the granite water that tend to delay radionuclide migration
The repository part of the host granite massif	After closure	Protecting the disposal cells from water circulation Delaying and reducing radionuclides migration to the environment	Underground installation of the repository (500 m taken as reference) Low permeability (at a distance from the major faults, the main vectors of water in the massif) Low underground hydraulic gradients Chemical properties of the granite water that tend to delay radionuclide migration Geo-dynamic context that ensures favourable repository conditions are maintained in the long term, particularly in relation to erosion phenomena Isolating the waste from human activities

COMPONENT	PERIOD	MAIN REPOSITORY FUNCTIONS	PROPERTIES HARNESSED
2. ENGINEERED COMPONENTS			
Connecting drifts between access structures from the surface and repository modules	Before and during repository operation During the reversibility phase	Emplacing (and being able to retrieve) the packages in the granite	Dimensions adjusted to package throughput Drift equipment that ensures safe operation
Drifts in the modules (cell access drifts and exploratory structures)	Before and during repository operation During the reversibility phase	Emplacing (and being able to retrieve) the packages in the granite	Suitable dimensions for package throughput Drift equipment that ensures safe operation
Backfill and seals of the disposal module drifts	After closure	Protecting the disposal modules from water circulation Delaying and reducing radionuclides migration to the environment	Low permeability Swelling capacity Radionuclide retention
Cell: disposal cavity **Spent fuel: small vertical borrehole**	During operation and the reversibility phase After closure	Emplacing (and being able to retrieve) the packages in the granite Restricting radionuclides release by primary packages and immobilising them	Dimensions adapted to the granite fracturing and handling technologies Properties of the various components: packages, engineered barriers, backfill
Spent fuel containers in copper	During operation and the reversibility phase After closure	Emplacing (and being able to retrieve) the packages in the granite (with sleeve and operating plug) Preventing radionuclides release by fuel assemblies and immobilising the radionuclides	Mechanical strength Suitable dimensions for handling them Water-tightness of the canisters (over several hundreds of millennia)
Clay engineered barriers of the spent fuel repository boreholes	After closure	Protecting the disposal cells from convective water circulation Delaying and reducing radionuclides migration to the environment	Low permeability Swelling capacity and plasticity (providing the long-term mechanical stability of the cell) Chemical buffer to interactions between the granite water and the component alteration products Radionuclides retention

8 Conclusions

The study shows, from the engineering viewpoint, the possibility of building and operating a high-level and long-lived waste repository in a granite formation.

The study covers all existing and future high-level long lived waste committed to by the French nuclear power plant fleet. Accordingly various production and conditioning hypotheses have been formulated for future waste. It has been possible to make allowance for the diversity of primary waste packages by organising the inventory model into "reference packages" that are representative of all the issues created by this diversity. This inventory model is common to both the studies carried out for the clay and granite media.

Hence the design of a HLLL waste repository in granite has points in common with the design for a clay medium, in particular the surface installations, the B and C disposal packages and B package handling facilities. The brief definition of these elements is based on the conclusions of the feasibility study conducted for the clay medium.

Over and above B waste (intermediate-level long lived) and C waste (high level), the study explores the case of spent fuel in case it is decided to stop reprocessing, without pre-empting the management choice. For this latter case, the study takes up a repository design developed in Sweden and Finland (these countries have opted for direct disposal in a granite medium) and ascertains that it can be transposed into the French context (French granites and spent fuel).

In the absence of a reference site, a generic architecture design is proposed, that is sufficiently robust and flexible to be adapted to the characteristics of a particular granite massif. This generic architecture illustrates that it is possible to find a solution to the main specific issues raised by the design of a repository in a granite medium.

The development of generic architecture is based on a typological analysis of French granites. In order to meet long-term safety objectives, the study aims at defining the technical measures (i) that will enable the characteristics common to these granites to be harnessed and (ii) to supplement them with engineered components. The options presented thus allow for the variability of the characteristics of French granites.

Special attention has been given to disposal reversibility, to fulfil the expectations that were first voiced in the Act of 30 December 1991 [1], and since then on several occasions. Thus the study demonstrates the assets of granite for constructing self-stable very long-lasting structures.

As regards aspects specific to granite, the study relies heavily on the results of the studies and work carried out in other countries looking into this medium and also the experiments conducted by Andra in conjunction with its counterparts in the foreign granite laboratories (the laboratories of Lac du Bonnet in Canada, Äspö in Sweden and Grimsel in Switzerland).

8.1 Generic architecture that can be adapted to the diversity of French granites

The existence of fracturing, which is the potential site for water circulation, raises the main issue for repository design in a granite medium. This fracturing varies from granite to granite in terms of hydraulic configuration and conductivity. Nonetheless the study shows that the generic concept envisaged enables a repository to be installed in the types of French granite that have been surveyed. Indeed, the adopted design approach consists of distributing the repository modules between blocks that are slightly or not fractured and apart from the major water-conducting faults.

The envisaged two-level architecture presents the range of possibilities offered by the vast volume of rock a granite massif represents to adapt modular separation to the fracturing of a particular granite rock. The study demonstrates that it is possible to implement on-going exploratory work during the construction phase, so that the geometric limits of the blocks identified during the surface surveys can be specified and that the locations of the structures can be more accurately defined at each repository level. This approach is similar to the approach adopted by the Swedish (SKB) and Finnish (POSIVA) agencies.

This compartmentalisation results in modular underground installations, which also fulfils the need for operational flexibility in repository management. It also limits the consequences of a failure of a structure or group of structures and avoids interferences between different types of waste by disposing of them in distinct zones.

Another important issue is the variability of the thermal properties of granite, particularly in term of its initial temperature. The study shows that the disposal cell design for the most exothermic waste (primarily vitrified C waste, and if necessary spent fuel) and their layout in the context of the proposed generic architecture are such that they can be installed in the types of various French surveyed granite massifs.

From this viewpoint, the good thermal conductivity offered by granite is a favourable element to the passive conduction of heat in the geological medium.

The study also shows that from a hydro-geochemical viewpoint, repository design is robust with respect to the variability of the composition of French granite water.

Finally the good mechanical properties offered by granite are a favourable element for the feasibility of constructing very long service life structures without the need for rock support or linings.

8.2 A robust design

The studied architecture seeks robustness in view of the uncertainties caused by not having a particular site, and the inevitable uncertainties relating to the long timescales implied by long-term safety functions. Hence a set of additional and redundant measures have been studied.

Thus cautious working hypotheses have been adopted at this stage for the hydraulic behaviour of the repository. By way of example it is envisaged that the drifts close to the cells and seals will be backfilled with low permeability material made up of granite aggregates with added swelling clay. This means redundancy with the multiple seals. Furthermore, just above the sealing devices, provisions have been proposed and studied to hydraulically interrupt the granite zone that could be fractured in the vicinity of the excavation works.

The realism of the technical options presented is founded on industry-proven procedures. Hence the study points out the existence of mining and nuclear industry construction and operating means similar to those envisaged.

Lastly cooperation with Andra’s foreign counterparts, both in underground laboratory experiments and repository design has contributed to strengthening the robustness of the options presented. Thus the envisaged generic repository design is based on the actual results of these experiments and feasibility demonstrations of designs made in other countries. The generic repository design presented is also largely a result of comparing it with the designs developed primarily in Sweden and Finland. By way of example we can mention the similarity of the repository concept in vertical boreholes and the underground installation closure mode with those of the Swedish KBS-3 design.

8.3 Architecture taking in account the reversibility rationale

The request for reversibility has been incorporated into the proposed generic architecture.

The considerable mechanical stability of excavations in the granite medium is generally conducive to package retrieval capability and management flexibility of the repository process over very long periods. Thus, structures, with very long service life (at least several centuries long) and without the need for human intervention, can be constructed.

Similarly reversibility is particularly enhanced by adopting long-lasting repository containers.

In a first disposal process phase, the disposal packages may be managed as flexibly as if they were in a warehouse. They can be retrieved using the same facilities as for their emplacement with no deterioration to the installations or the packages themselves.

At this preliminary stage of design, it appears that this phase will last at least a few centuries (typically 200-300 years), with no heavy maintenance.

Repository observation would provide an re-assessment of (i) the alteration and durability of the components (swelling of swelling clay, service life of cell sleeve) and (ii) the impact on the granite of operating the structures. However it must be noted that operations in ventilated drifts or structures (B waste tunnels, access drifts to various disposal cells, connecting drifts) entails hydro-geological and hydro-geochemical impacts on the granite. However these site-dependent disturbances appear to be reversible.

In contrast to storage, a repository may also be closed by passive means to ensure the confinement of the waste and long-term protection of man and the environment. The proposed options are consistent with a flexible management of the repository process by providing the possibility of progressive closure. As this closure progresses, the reversibility level gradually diminishes to the lowest level that coincides with total closure of the installation. However, after this closure, it remains technically possible to access the repository structures and retrieve the emplaced waste.

To conclude, it appears that for two to three centuries it is technically possible to manage the repository process with a reversibility rationale and without involving heavy operations. During this period, whatever the repository closure level, once access to the packages has been re-established, package retrieval from a disposal cell can be carried out using comparable handling equipment as that used for its emplacement.

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