Report Series

Dossier 2005 Granite

Tome

Safety analysis 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 The Safety Approach

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1.1 Foreword

The Law of 30 December 1991 conferred to Andra the mission of assessing the feasibility of high-level and long-lived (HLLL) waste deep geological disposal.

This volume of "Dossier 2005 Granite" describes the results of the safety analysis of the evolution of a repository. With no particular study site (see inset 1), the study approach is somewhat generic in character. It is largely based on international cooperation, especially for data collection in underground laboratories.

Inset 1 Background to the granite studies

In application of the Law of 30 December 1991, Andra carried out from 1994 to 1996, exploratory works with a view to installing an underground laboratory in the south of the Vienne department in western France. The selected granite massif was granite under a sedimentary cover, delimited using geophysical and geological data.

In its 1997 report, the National Review Board advised against the Vienne site, particularly due to the risks of fluid circulation between the granite and the worked aquifers in the sedimentary cover; it stressed the advantage of "outcropping" granites, which were likely to have more favourable characteristics.

In its decision on December 9th 1998, the Government did not select the Vienne site and began the search for another site capable of accommodating a research laboratory in a granite medium. A consultation mission was named in 1999 to present this project and gather the opinions of the concerned populations at fifteen geologically favourable sites. The identification of these fifteen sites, submitted to a committee of national and international experts, was the result of earlier selection procedures and the progress made in terms of knowledge of the granite medium, both in France and other countries. In July 2000, the mission's report highlighted the difficulties in bringing the consultation process to a conclusion.

In order to meet the Government's guidelines, Andra devised in 2000 a research programme enhancing all data acquired in foreign underground laboratories and in various geological contexts in order to assess the assets of granite medium for the deep disposal of high-level, and long-lived waste.

Due to the differences in context for the studies owing to the two formations (clay and granite), Andra organised the research into two distinct projects: study of disposal in a clay medium, supported by the Meuse/Haute-Marne underground laboratory, and study of disposal in a granite medium.

Some of the studies were common to both projects, notably those relatives to waste packages, with the results being used by each of the project.

In this context, the "Dossier 2002 Granite" provided an initial overview of the studies and researches about the possibility of a repository in a granite medium.

The Dossier 2005 Granite draws conclusions from the numerous studies conducted since 1991. On this basis, it aims at assessing how the granite medium is of interest for a high-level and long-lived waste repository.

1.1.1 The Andra research programme on disposal in a granite formation

Along with clay, granite is one of the geological formations studied by Andra in application of the Law of 30 December 1991. In the absence of a laboratory site to date, the studies conducted on the granite medium were not aimed at assessing the feasibility of a repository which would be designed to meet the specific features of a particular site.

1.1.2 A generic study approach

The programme that was built had the objective of assessing the assets of a granite medium for a repository. The study approach adopted by Andra, in this perspective, aimed at identifying and dealing with each of the major issues relatives to a disposal in a granite medium in order to ensure that none of them would rule out such an eventuality and to examine what technical options would be possible for a repository.

The approach therefore consisted of studying the design of generic repository architectures, based on the properties of the granite medium. The proposed options supported the various analyses aimed at understanding the behaviour of a repository during its operation and in the long term, as well as for safety assessments. Consistent with this approach, the research programme was composed of four complementary areas of study.

- **Study of the granite medium**

The generic design of a repository is based on the properties of the granite. Studies on the granite medium therefore consisted on the one hand of general studies aimed at understanding and modelling the granite medium and, on the other, of an analysis of the variability in the properties of French granites so that the design studies would be suited to them and the safety analyses would take them into account in the various assessments.

- **The generic design of a repository in a granite medium**

Based on design principles founded on safety, the studies aimed at proposing generic repository architectures, waste conditionings, operational methods and possibilities of the repository closure. The objective was to propose repository concepts taking into account the reversibility. Wherever necessary, the studies were supported by data shared with the study project of a disposal in a clay medium, especially those relative to packages and materials.

- **Repository behaviour and its long-term evolution**

The studies consisted of the analysis of the long-term repository behaviour based on the options proposed. The objective was to understand and model thermal, mechanical, chemical and hydraulic phenomena occurring in a repository in a granite medium.

- **Long-term safety analyses**

As the studies had a generic character, the safety analyses did not aim at assessing the performance of a repository at one or more specific granite sites. They aimed at identifying the major factors determining the performance of a repository in a granite environment with regard to the objective of protecting humans and its environment, and at assessing the robustness of the design options proposed.

1.1.3 The backing of international cooperation and the mobilisation of the French scientific community

The programme conducted by Andra was largely supported by studies performed in other countries. Andra has, in this respect, played an active role in the experimental programmes carried out at underground laboratories in Sweden, Switzerland and Canada.

The main topics of cooperation have focused on *study of the granite medium:* questions relative to the structuring of a granite massif and its fracturing, exploratory methods, the way in which natural water flows occur at depth and the radionuclide retention capacity of the rock (see Figure 1.1-1).

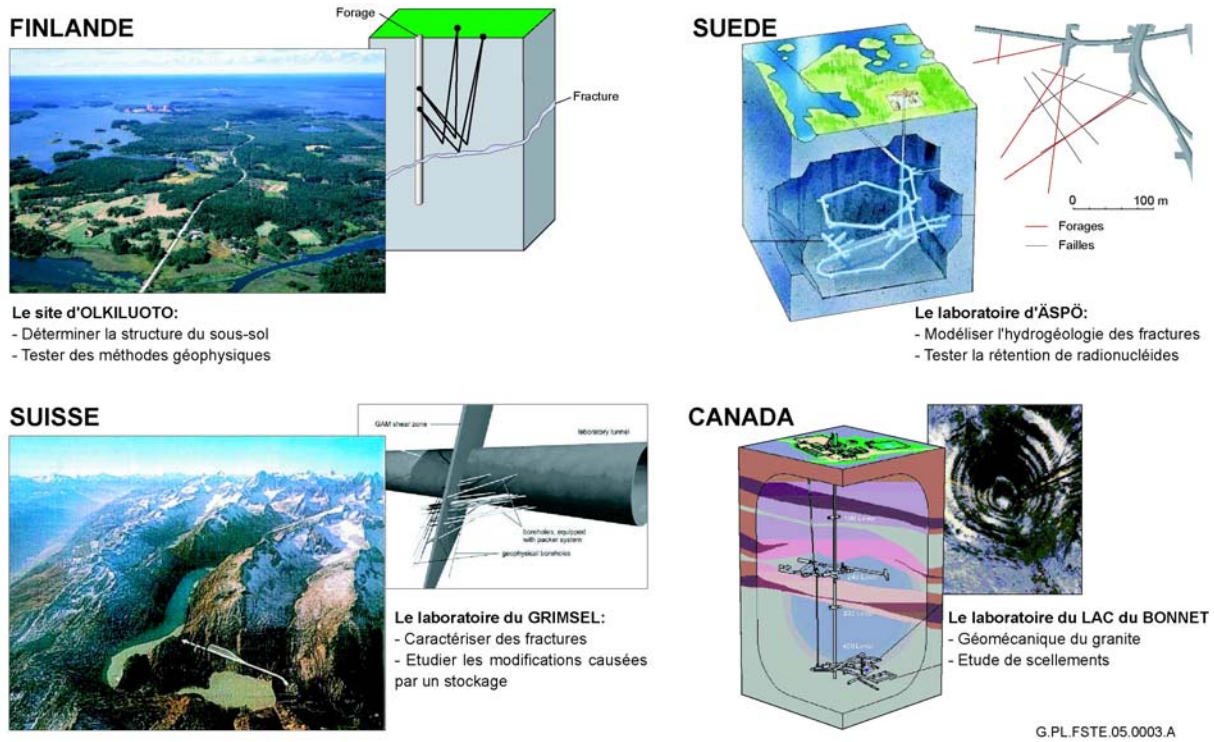

Figure 1.1-1 Study of the granite medium: international cooperation

The repository design studies were also supported by some demonstration elements acquired in underground laboratories and relative to the implementation and behaviour of engineered components of a repository: seals, backfill, engineered barrier, etc.

Finally, the study approach benefited the feedback from the experience acquired abroad in the areas of safety analysis of a repository in a granite medium, notably in Sweden and Finland.

This approach therefore put in good use the significant progress achieved at the international level in terms of the study of disposal in a granite medium.

Andra also set up scientific partnerships on a national level (CEA, BRGM, the Forpro Research Group at the CNRS and the École des Mines in Paris). In addition to the participation of French research teams in foreign programmes, this ensured that the results obtained abroad were not simply transposed to the French geological context without due precaution.

1.1.4 The scope of the approach

In a context where there was no specific study site, the approach adopted by Andra investigated all of the various aspects to be considered in designing a repository and in its assessment, notably on the basis of the large amount of data acquired in other countries.

This led to the proposal of "generic" repository concepts likely to ensure in the French geological context the suitability of a repository with regard to long-term safety objectives.

This approach does not however claim to draw the same conclusions as an approach supported by surface exploratory work at a granite site and then in an underground laboratory. Only such work would provide sufficient knowledge of the properties of a granite, to draw up a fair assessment of the feasibility of a repository.

The specific characteristics of a site would indeed make it possible to adjust the design of the various repository components to the properties of the granite studied, to adapt the architecture of the repository facilities to the structure of the massif, and to precise the phase and organisation of its construction and operation according to the context of the site. This would therefore, allow an accurate assessment of the specifically selected options to the site studied with regards to the long-term safety objectives.

1.1.5 The structure of the "Dossier 2005 Granite"

This Dossier 2005 comprises on the one hand a three-"volume" summary of the design options proposed and the associated scientific and safety analyses, and, on the other one, a presentation in the form of "reference documents" of the knowledge supporting the design of a repository and its assessment.

- **Three "volumes"**

Three volumes summarise the knowledge acquired in each area of the study programme:

- One volume "Architecture and management of a geological repository"

  Andra proposes generic options of repository architectures which are not only possible particularly in terms of safety and reversibility expectations, but also realistic from an industrial point of view. Based on available knowledge and technology, the technical options studied, chosen to be as simple and robust as possible, show that solutions do exist. These options have also formed the basis on which the safety of the repository has been analysed, especially its behaviour and its evolution over the various timescales. This analysis is covered in the other two volumes of the dossier;
- One volume "Phenomenological evolution of a geological repository"

  The design and safety assessment of a repository depend on the understanding of the phenomenological evolution of the repository and its environment. This understanding must ultimately enable to explain the processes which condition or control the behaviour and migration of radionuclides in the environment to be assessed on a million-year scale;
- One volume "Safety analysis of a geological repository"

  This volume, which is the subject of this report, describes the safety analysis approach for a repository in a granite medium in a generic study context. It presents the corresponding safety analyses in two sections: a qualitative analysis and a quantitative analysis.

- **"Reference knowledge" documents**

Andra has structured the acquisition of knowledge around reference documents. Three are shared with the "clay" dossier:

- reference document on repository materials which brings together data on the behaviour of materials (steels, concrete, etc.) other than the rock of the repository host formation;
- reference document on the behaviour of high-level, and long-lived waste packages which summarises the knowledge and models on the behaviour of waste in a repository environment;
- reference knowledge document and inventory and dimensioning model, which covers all the HLLL waste produced and to be produced by existing nuclear facilities.

A fourth reference document specific to the granite medium collates in the form of a typological analysis, the data available on French granites.

1.2 Context and general objective of the safety approach

At every stage in the definition, construction and operation of a repository, the objecive of protecting the workers, the public and the environment must guide the design options, the scientific data acquisition programme, the dimensioning of engineered structure elements and the methods of operation. In this respect, the safety approach in repository design does not differ from the design of a conventional nuclear facility. What is specific about a repository facility, however, is the need to take account of its long-term evolution, over very long periods, without assuming that a hypothetical human action can remedy a malfunction. This particular factor has a strong bearing on the design of the facilities (choice of the type of rock in which the repository may be installed, initial architectural definitions). For this reason, research programmes regarding HLLL waste management are structured by a safety approach. This obviously applies to studies on disposal possibilities in a granite formation.

However, the structuring of such an approach must not lead to any confusion regarding the objective sought. In the absence of any institutional processes aimed at defining one or more sites for detailed investigations, it is not the purpose of safety to ensure that particular granite massifs conform to preset criteria such as, for example, those defined in Basic Safety Rule III.2.f. [i]. Given the fragmented nature of the data available in the bibliography, such an *ab initio* analysis would obviously be inconclusive. Above all, it is not the aim of the safety approach to ascertain the conformity of a particular massif and repository concept with regard to impact objectives either.

The safety analysis of the Dossier 2005 Granite intends to identify the important processes with regard to safety which govern the evolution of a repository. These processes may concern the long-term evolution phase (thermal, mechanical and chemical interactions between the exogenous elements provided by the repository and the rock itself, near or far-field radionuclide transfer mechanisms, for example), or the construction, operating and closure phase (impact of excavation techniques on rock damage, management of major personnel and system safety issues in a granite context). The identified processes must be compared with the characteristics of French granites, taking account of the variability of these granites, in order to assess their importance and propose initial design measures to control them. This assessment is carried out in the scope of defined repository architectures suited to the specific features of the granite and to their potential variability. It enables the initial dimensioning elements of such architectures to be identified. It does not reach a definitive conclusion regarding the possibility of installing a repository on a particular granite site, but identifies the major themes to be tackled with a view to such a study and can be used to ascertain whether any of them already presents a redhibitory character. This is where the purpose of studying the interest of granite as a suitable formation to host a HLLL waste repository lies.

Another dimension of the safety approach of the the Dossier 2005 Granite is its methodological character. This dossier is not the first assessment made by Andra in a granite context [ii]. The Agency has however sought to extend the spectrum of the methods that it had already adopted, develop some of them and pay particular attention to controlling the tools available internationally to carry out safety assessments in a fractured medium. An important aspect of this programme is the participation of the Agency in the experimental and theoretical work conducted by its counterparts studying crystalline media (laboratory experiments at Äspö in Sweden, Grimsel in Switzerland and Lac du Bonnet in Canada).

Now that this framework has been defined, the purpose of the following paragraphs is to describe the various aspects of this approach.

1.3 References of the safety approach

As the safety approach of Dossier 2005 Granite falls outside the scope of both a site selection process and the assessment of a given site, it cannot refer in its entirety to the Basic Safety Rule (RFS) III.2.f which aimed at such approaches. However, RFS III.2.f proposes a number of methodological principles and design options which are suitable even to an earlier stage. For example, in the area of impact assessments, quantified radiological protection objectives are premature in a context where a sufficiently accurate assessment, based on site data, is not accessible. On the other hand, the general objective requiring all measures to be taken to limit repository impact, and to take this impact into consideration in both "normal" and incidental situations, is a relevant consideration to the studies.

As a general rule, the recommendations of the RFS III.2.f. are therefore taken into account for the studies, with their interpretation being adapted where required to the positioning of the dossier [iii]. The qualitative objectives set by the rule, and the proposed safety options, are taken as a reference. Clearly, for the part of the study which is common to granite and clay (control of the package inventory and material studies), consistency with RFS III.2.f. is ensured by the clay HLLL project.

In addition to the French rule, principles developed internationally (the IAEA "safety requirement" project [iv] and the OECD/NEA "safety case" definition [v]) have inspired the development of the safety approach of the Dossier 2005 Granite. As in the case of the basic safety rule, strict compliance with this reference document is not an aim. The Dossier 2005 Granite is not a "safety case" in the sense defined by the NEA. However, the main aspects dealt with in these documents (importance of clear, and transparent arguments, need to trace the origin of data, insistence on the management of uncertainties, etc.) are relevant to the dossier.

1.4 Design approach using Safety function

RFS III.2.f. recommends that repository design be centred on the definition of a confinement system, "consisting of a series of means or devices preventing or limiting (…) the transfer of radioactive materials to the biosphere". It consists of "the following three barriers: waste packages (…), engineered barriers (…) and the geological barrier". The evolution of the thinking on a domestic and international scale tends to privilege the notion of "safety functions" as a complement and extension to the "barrier" notion. A safety function is an action fulfilled by a component of the repository system which contributes to human or environmental protection. It enables the "confinement barrier" notion to be extended to types of functions other than the sole physical confinement of radioactivity in a repository. It allows differentiating the roles fulfilled by the components over time and highlights redundancies and complementarities between components, in that guaranteeing the robustness of the repository in case of possible malfunctions.

The expected safety functions, and the systems which may be proposed in order to fulfil these functions, are covered in chapter 3. These functions call widely on the notion of controlling water-based transfer, the main vector in a repository, and on RFS recommendations. In this respect, they are broadly common, in terms of their principle, with the functions developed in the context of a repository in clay formations. A certain number of more transverse safety functions, designed to protect the favourable properties of the natural environment against disturbances caused by the repository, are also distinguished.

These functions are considered with respect to various kinds of design system (general organisation of the repository, contribution of the favourable properties of the medium, use of specific engineered structures designed to complement or provide redundancy, etc.). Only passive arrangements, i.e. assuming no human action, are proposed for the post-closure phase. These systems are generic at this stage; pre-dimensioning is proposed for the most important components, to ensure that the expected functions are realistic. Pre-dimensioning is itself dependent on specific conditions (geochemical, geomechanical, etc.) within the rock. It is therefore based on characteristics considered as representative of the French granite context, with sensitivity on the parameters most likely to vary. These elements are still indicative and should be reviewed in the event of the development of a project at a specific site.

As a general rule, repository design is based on two principles:

- **robustness**, in other words, preference is given in terms of architectures to arrangements enabling the functions to be best maintained in spite of disturbances or uncertainties;
- **demonstrability**, in other words, the concepts are chosen so that their safety properties can be checked as easily as possible and do not require complex demonstrations, subject to caution.

An analysis of the repository functions is presented in its main conclusions in chapter 3.

1.5 Safety approach in the construction-operation-observation-closure phase

Worker security and the protection of the public and the environment during the phases of active repository operation are an important element to be considered in the design of a facility. It includes an analysis of the risks to which workers may be subjected in particular, due to the presence of radioactive waste and to underground working conditions.

However, even if there is just one example of deep geological disposal of long-lived, intermediate-level waste in the world [vi], there is considerable operational feedback from underground structures and from the handling of high-level waste packages or spent fuel. The risks posed by either of these contexts are well known, and suitable arrangements to prevent them or reduce their seriousness are commonly implemented. A the generic stage of the Dossier 2005 Granite, these aspects do not therefore constitute a focal point.

The operational safety studies presented in this volume are therefore general in nature and are essentially inspired by the studies developed in the scope of research into disposal in a clay medium. They consist of initial identification and classification of the risks, and an initial overview of the techniques used to deal with them. It is ensured that the specificity of the granite context (for example, the greater probability of exposure to the radon risk) or the particular features of the concepts proposed for granite relative to the ones defined for clay (for example, the disposal borehale (or pit) shaft concepts for C-waste and spent fuels) do not generate any specific issues on first approach.

If disposal in a granite formation were to be developed further by Andra, the applicable radiological protection objectives would be those applied by the Agency to all of its repositories:

- 0.25 mSv per year for the public in a normal operating situation, this being the objective for an ALARA (As Low As Reasonably Achievable) dose reduction approach. This limit is already stricter than the statutory limit of 1 mSv per year;
- 5 mSv per year for exposed workers, again within the scope of an ALARA approach. This objective corresponds to a quarter of the statutory limit.

However, considering the objectives of the Dossier 2005 Granite, no personnel dosimetry assessments or impact studies have been carried out.

Safety studies in the operating phase are presented in chapter 4.

1.6 Safety analysis in the post-closure phase

The long-term safety analysis at the preliminary stage of the Dossier 2005 Granite aims at identifying the main processes governing the long-term evolution of a repository facility in granite and to ensure that adequate design arrangements and/or a data acquisition programme are in place or could be implemented in order to understand their effects and control their consequences.

1.6.1 Qualitative analysis

The analysis is essentially supported by an approach to the structure and properties of French granites and, in particular, fracturing in the granites. The RFS III.2.f. recommends that "*the implementation of the repository in a geological formation [is located] in crystalline media, within a host block free from large failures*" and that "*the repository modules [be] built away from medium fracturing, although this one could be traversed by the repository structures*". The question of the nature and spatial arrangement of the fractures is a central issue of the safety analysis in the post-closure phase. In the absence of investigations at actual sites, and on the basis of the only data available in the bibliography, it can only be tackled generically. Over and above the mere structural analysis of granite, it is also the case of questions relative to geochemistry, thermal and mechanical behaviour, etc.

A prior analysis of French granites to highlight their main structural, chemical, mechanical and thermal characteristics is therefore necessary. This is presented in [vii]. Its sole purpose is to make an assessment of the properties of each of the granite massifs in France, which is not possible from the available data, especially at depth. It aims to highlight the major characteristics of these granites and the variations liable to be encountered in the French geological context.

The nature of the granites present in France thus being identified and the generic concepts suited to these massifs being defined elsewhere [viii], the qualitative safety analysis consists of comparing this context with all of the phenomena liable to influence the evolution of the repository (intrinsic evolution of the components, interactions between repository components and radionuclide and toxic chemical transfer phenomena). Identification of the phenomena is built on the bases on FEP's (Features, Events and Processes) available internationally. Indeed these bases are suited to a generic context, such as Dossier 2005 Granite, and are used by many of Andra's counterparts in support of their research programmes and safety analyses. These events, identified in an general, all-encompassing manner, are screened against the specific features of French granites and the architectures proposed by Andra to distinguish the phenomena which are expected to occur ("FEP's in a normal evolution"), those which would occur in the event of a malfunction ("FEP's in an altered evolution") and those which it would be irrelevant or premature to consider.

1.6.2 Evolution scenarios and performance calculations

This initial qualitative safety analysis can be used to define a set of "normal" situations, in the sense indicated in RFS III.2.f. ("certain or highly probable events"), which is possible to assess in the form of a normal repository evolution scenario that encompasses all of these situations and proposes a simplified vision of them. This scenario is not in itself the representation of the "likeliest evolution of the repository" insofar as it can include a number of penalising choices, with the purpose of enriching the analysis which can be drawn from it. By way of an example, an isolated disposal container failure was considered in a normal evolution situation, although special attention would be paid to the quality of their construction and, in theory, this should eliminate such a failure. Apart from the fact that this choice takes on board failures which are random yet still possible, it means that radionuclide migration possibilities in the various types of disposal cell can be assessed and compared.

The analysis can also be used to define altered evolution scenarios, illustrating the main malfunctions possible and enabling the robustness of the repository to be tested in the event of the loss of a barrier or a safety function. The altered scenarios considered at this stage do not claim to cover all conceivable malfunction situations. They do however cover the main safety issues: the role of packages in repository safety ("package failure" scenario), the role of seals ("seal failure" or "plug failure" scenario), the role of the geological medium and, in particular, the role of medium fracture characterisation ("fracture characterisation failure" scenario). At this stage, the "human intrusion by boring" scenario, traditionally considered in geological repository safety assessments, was only dealt with on a qualitative basis and was not considered in the safety calculations. It should be considered quantitatively if studies are pursued on the granite medium.

The detailed qualitative analysis approach and the main results, in the form of a description of the phenomena to be considered and the scenarios adopted, are presented in chapter 5.

These scenarios are then calculated with the aid of safety models. They must be supported by a representation of the architectures and the geological medium.

As the architectures are subject to pre-dimensioning, it is this which is represented (geometry of the reference cells and representation of the structures considering the characteristics of the materials envisaged). Depending on the level of knowledge attained and uncertainties regarding the chemical, hydrological and mechanical behaviour of the materials used, their characteristics are represented in the calculations by more or less pessimistic parameters and models.

For the representation of the geological medium, the objective is not to "test" actual sites, which it is not possible to define to date, but to assess the behaviour of the architectures in representative contexts.

Models of sites are built from geological configurations representative of the French context. These configurations are notably based on characteristics identified from surface mapping documents and available data on depth (mining and borehole data). The values of the parameters to be introduced into the calculations are "test" values, chosen as being representative of the French context. From this, an initial hierarchy of their respective importance with regard to radionuclide transfer can be established. The purpose of representing the geological medium is not to present a "conservative" character, in the sense of safety. Such a character would clearly be indemonstrable in the absence of site data in sufficient quantity.

If a repository project were to be developed in a granite formation, the radiological protection objectives would be the targets adopted by Andra for all of its repositories and consistent with those proposed in RFS III.2.f: a committed individual dose at the outlets of no more than 0.25 mSv per year in a normal situation up to 10 000 years, with the reference to this value being maintained beyond this period. For altered situations, the value of 0.25 mSv per year may be exceeded according to the likelihood of the situation and the extent of the resulting exposure. A value of a few millisieverts per year is generally considered acceptable.

However, at this stage of the dossier, insofar as the purpose of the calculations is to assess the performance of the concepts in various granite contexts and not to assess whether a given site meets radiological protection targets, the calculations call on various safety indicators (radionuclide flow through various barriers, essentially) but not at the committed dose, which would only provide global information which is not usable at this stage of the project. It will also be noted that the environmental conditions, which will need to be known in order to define a typical biosphere and conduct a dose calculation, vary according to the sites and can only be roughly defined in a generic context.

The calculation models, the techniques used, the results obtained and their interpretation are presented in chapter 6.

Before detailing the various volumes of this safety analysis, some basic data of the study project is presented by way of an introduction. This is the subject of chapter 2.

2 General description

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2.2 The granite medium..46
2.3 General structure of the repository architecture 56

This chapter provides an overall view of the architecture and management of a possible repository facility. At this stage, the geological formation, repository architectures and management methods are only presented in a succinct manner, with no particular justification: the aim is merely to introduce the essential notions for the purpose of understanding the rest of the document.

This section begins by recalling the waste inventory to be considered (see paragraph 2.1) and the geological context for which its feasibility is studied (see 2.2). Paragraph 2.3 then introduces the technical options, in terms of the general architecture of the repository, selected by Andra for the studies.

2.1 High-level long-lived waste

This paragraph summarises the data relating to HLLL waste. It is placed here merely to assist with the reading of the rest of this volume. Readers with a special interest in package inventory matters are invited to consult the corresponding reference document [ix].

High-level and long-lived waste (HLLL waste) contains both short-lived radionuclides (generally in large quantities and hence with a high level of radioactivity) and long-lived radionuclides (in quantities ranging from moderate to very large).

Due to its long-lived radionuclide content[1] the waste has a long period of radiotoxicity combined with the risk of ingestion leading to the exposure of living tissues to α radiation; the half-life[2] of some isotopes can exceed a hundred thousand years.

A large part of HLLL waste also has a high level of radioactivity in γ radiation which implies protecting mankind from external radiation exposure.

The β-γ radioactivity present in HLLL waste decays relatively quickly over time: therefore, after a few decades, nuclear fuels contain no more than a small percentage of the radioactivity that they had when the reactor was unloaded.

The energy produced by radioactivity is converted essentially into heat: the radiation is absorbed into the actual material forming the waste package and, to a lesser extent, into the material in its immediate vicinity. When the β-γ radioactivity has decayed considerably (after a few centuries), the residual radioactive energy, combined with the long-lived isotopes, is very low and the heat produced then becomes insignificant.

For the most active packages, the decay in β-γ activity over time may require an intermediate holding period between production and disposal. This holding period can be spent in storage facilities. This helps reduce the heat produced by the waste and has an effect on the dimensioning of the repository facilities and the footprint in the host formation.

2.1.1 Nature and origin of HLLL waste

HLLL waste is produced by the nuclear power industry, as well as by research and national defence activities. In order to study disposal possibilities, Andra has drawn up an "inventory model" consolidating the data and hypotheses on HLLL waste [ix]. It takes account of the waste already produced and currently stored at production sites in addition to future waste.

Nuclear power industry waste originates essentially from spent fuel unloaded from nuclear power plant reactors. This fuel is currently being reprocessed by Cogema at its La Hague facilities. Residues such as fission products and minor actinides, as well as the mechanical structures of fuel assemblies (cladding sections and end cap parts) are separated from the uranium and plutonium at this point.

For the repository study, all of the waste to which current nuclear power plants are committed is considered on the basis of a hypothetical average reactor lifecycle of forty years.

1 Long-lived isotopes include (i) fission or activation products, resulting respectively from the splitting of heavy atoms such as uranium and plutonium during fission reactions in the reactor, and from the absorption of neutrons by materials present in the reactor (mainly metals), and (ii) actinides, composed of uranium and heavier atoms formed from uranium, by capturing neutrons.

2 The half-life of an isotope is the time required for the disintegration (i.e. spontaneous transformation into another radioactive or stable element) of 50% of the present quantity of this isotope. An isotope is termed "long-lived" when its half-life is strictly in excess of 30 years.

Various production scenarios were considered in the study. They were chosen so that a wide range of waste types, even hypothetical, could be taken into account and the various issues for the study of waste disposal could be tackled. A first family of scenarios considers the continued reprocessing of spent fuel unloaded from reactors. The second type of scenario assumes that reprocessing will be discontinued. The purpose of these scenarios is not to predict an overall industrial system, but to examine how repository architecture can take into account various inventories and possible methods of managing the back end of the nuclear power cycle. These scenarios were established in liaison with waste producers (EDF, CEA and Cogema).

The scenarios considered take account of spent fuel which is not considered as waste. If spent fuel were not to be reprocessed, it would need to be considered in studies aimed at identifying suitable waste management techniques. This may include MOX spent fuel (mixed uranium and plutonium oxides) originating from plutonium recycling, or enriched uranium fuel (UOx).

In addition to spent fuel and the residues from processing spent fuel, waste from the operation of nuclear reactors (rod cluster control assemblies) and the operation and maintenance of reprocessing plants (so-called "technological" waste (replaced or obsolete parts, parts contaminated by the processed radioactive waste and materials, etc.), liquid effluents, etc.) is considered.

It will be noted that waste also comes from the Marcoule plant, now shut down, where fuel from the former GCR (gas cooled reactor) system was reprocessed.

HLLL waste from activities other than nuclear power generation (research and defence) is generally technological waste.

The existence of a small quantity of spent fuel from research or military reactors will also be noted. Without trying to anticipate how these fuel elements will be processed in the future, the possibility of their disposal has been explored.

2.1.2 HLLL waste conditioning

The conditioning of waste consists of (i) solidifying and immobilising the waste produced in a dispersible form, especially liquid, and (ii) placing the waste in a container while facilitating handling and storage in industrial facilities.

The HLLL waste inventory includes two types of waste:

- waste already produced and currently stored at production sites in conditioned and unconditioned states;
- waste to be produced in the future, either with conditioning by continuing the operation of existing nuclear facilities, or with adaptations not yet precisely identified and which will depend on future power generation and fuel cycle strategies.

In establishing the inventory, the waste has only been considered in a conditioned form[3]. This requires knowing or assuming the nature and methods of conditioning and packaging existing unconditioned waste and future waste, as well as the numbers and volumes of so-called "primary" packages to be considered; primary packages are the items which would be delivered to a repository site.

The identification of the various wastes and the definition of their conditioning method (irrespective of whether it exists already or whether it is adopted as a reference hypothesis) have mobilised the waste producers considerably. A relatively wide variety of primary package families has emerged, differing in terms of radiological content, the heat given off as a result of the presence of certain radionuclides, the physical and chemical nature of the waste or conditioning materials, and dimensions.

[3] However, for spent fuel assemblies, the option of conditioning directly at the repository site is studied.

Traditionally, a distinction is made between the following categories of HLLL waste packages, with each one presenting its own specific issues:

- So-called category B waste is characterised by low or medium β-γ activity and, as a result, by the fact that it gives off little or no heat. This represents the largest number of packages and the widest variety of types of conditioning. The total long-lived radionuclide inventory, relatively lower than that of other packages, is spread throughout the large volume represented by this type of waste.
- Category C waste consists of fission products and minor actinides separated during fuel reprocessing. Due to its high β-γ activity, considerable heat is given off although this decreases over time, mainly with the radioactive decay of the medium-lived fission products (caesium 137, strontium 90). This waste is conditioned by incorporating it into a glass matrix; the confinement capacity of this material is particularly high and durable when in a favourable physical and chemical environment.
- Spent fuel (identified by the letters "CU") also has a high level of activity and consequently gives off a notable amount of heat. The quantity of heat given off is due to the spent fuel containing medium-lived fission products, plutonium and americium (originating mainly from plutonium disintegration); these two elements lead to slower decay over time. Other specific features include the large size of the fuel unloaded from the nuclear power reactors, if it has been decided to dispose of it "as is", and a greater content of fissile material, combined with the matter of a criticality risk.

Within each of the categories introduced above, the various families of HLLL waste packages have been grouped into a more restricted number of representative "reference packages". The purpose of this is (i) to develop the studies in greater detail by limiting the number of cases to be dealt with specifically, without neglecting the diversity of the packages, and (ii) to standardise as far as possible the structures and means which would be used in a repository facility. As a result of this approach, a possible disposal solution has been able to be studied for each of the packages recorded in the inventory, independently of whether other categories of waste are to be emplaced in the repository or not.

2.1.3 Study scenarios

Fifty-eight PWRs commissioned between 1977 and 1999 are currently in operation. The mass of nuclear fuel unloaded from these reactors during their entire period of operation is estimated at 45,000 tonnes of heavy metal (tHM). This estimate is based on a combination of hypotheses relating to (i) the average lifetime of the units (forty years), (ii) energy production (16,000 terawatt-hours of cumulated production) and (iii) the progressive rise in the "burnup" of reactor fuels[4].

The fuel types considered and the corresponding average burnup are as follows:

- three generations of uranium oxide fuels: UOx1, UOx2 and UOx3, irradiated on average at 33 gigawatt-days per tonne of fuel (GWd/t), 45 GWd/t and 55 GWd/t respectively;
- 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 of these scenarios consists of encompassing various possible industrial strategies, without seeking to give priority to any particular one. This process makes it possible to consider a very wide range of waste types and examine the technical aspects of the various packages.

[4] The burnup fraction of a nuclear fuel assembly conveys the energy generated in a reactor by the fissile material contained in the fuel (uranium oxide or a mixture of uranium and plutonium oxides).

The first three scenarios – referred to as S1a, S1b and S1c – correspond to the continued reprocessing of spent fuel unloaded from reactors. Scenario S1a assumes that all of these fuels (UOx, URE and MOX) are reprocessed. It is combined with the hypothesis of incorporating mixtures of fission products and minor actinides from UOx and MOX fuels into glass; furthermore, in one aim of the study, a very low percentage of the plutonium originating from reprocessed UOx fuel is assumed to be incorporated into certain packages. This scenario therefore covers a variety of vitrified C package typologies. In scenarios S1b and S1c, MOX fuel is not reprocessed and, as a result, the hypothesis of direct disposal is 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, S2, which assumes that reprocessing is discontinued, supports the exploratory research into the direct disposal of UOx and URE fuels, as well as MOX fuels as in scenarios S1b and S1c. This scenario assumes that fuel is considered as waste which is not the case today.

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 2,700 tHM of spent MOX fuels.

Scenario S2 hypothetically considers continuing the reprocessing of a part of UOx fuel until 2010 (8,000 tHM of UOx1 and 8,000 tHM of UOx2), then stopping the reprocessing. The suspension of uranium and plutonium recycling changes the overall distribution of the various types of fuel unloaded from the reactors. The study into the direct disposal of unprocessed fuel then relates to 29,000 tHM comprising 12,500 tHM of UOx2, 14,000 tHM of UOx3, 500 tHM of URE and 2,000 tHM of MOX.

As recalled in the introduction, the studies refer to conditioned waste. Conditioning processes have therefore been defined for existing unconditioned waste as well for future waste. The retained hypotheses adopt the industrial processes currently used by the producers: (vitrification, compaction, cementation and bituminisation).

The various scenarios considered for the study of disposal also make it possible to adopt a robust approach in relation to the various evolutions possible in terms of managing the back end of the cycle.

In addition to these scenarios, the management of spent fuel originating from French reactors other than EDF's pressurised water reactors was examined (research and military reactors in particular). Reprocessing this fuel will, in any case, only produce a marginal quantity of waste compared with the waste originating from the reprocessing of EDF fuel. For exploratory purposes, we have particularly studied the possibility of direct disposal for these fuels without trying to anticipate the management choices which could be made.

2.1.4 Description of reference packages

The following paragraphs provide a description of the waste which is grouped into reference packages within the dimensioning inventory model (MID).

2.1.4.1 Primary reference packages for B waste

- **Reference package B1**

Reference package B1 contains waste from the operation of the pressurised water reactors (PWRs) currently in service as well as some activated waste from the SUPERPHÉNIX fast neutron reactor.

The PWR control clusters and neutron poisons represent over eighty percent of the total mass of activated waste. All consist of twenty-four fuel rods, suspended from a support system, which are introduced into slots left free for this purpose in fuel assemblies.

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

Other activated waste from PWRs is the metallic waste consisting mainly of dead-end tubes, called CIS (in-Core Instrumentation System) thimbles, equipping the reactor vessel (they are situated under the vessel). These tubes are used to insert the neutron probes required to control the nuclear reaction. They are replaced, if necessary, after a certain time and are then considered as waste.

The conditioning hypothesis considered in the study is the compacting of waste placed in holders[5], followed by emplacement in a small stainless steel container called a "Standardised Container for Compacted Waste or CSD-C" (see Figure 2.1-1).

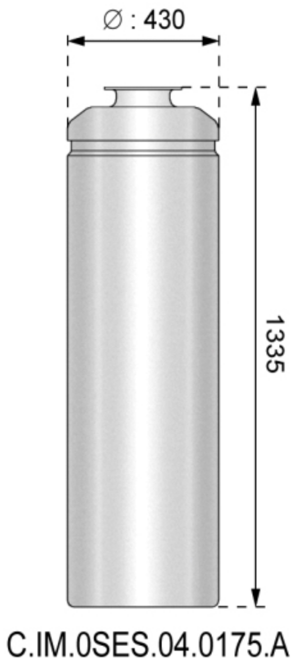


Figure 2.1-1 Standardised Container for Compacted Waste (CSD-C)

It will be noted that these packages do not contain any organic material and are not liable to produce gas (hydrogen) by radiolysis.

- **Reference package B2**

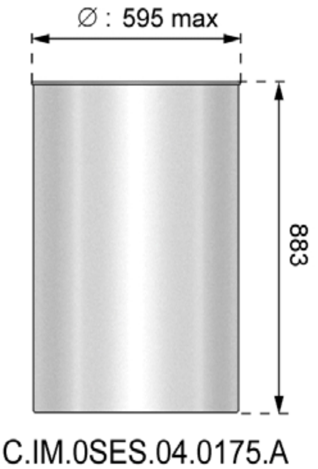
Reference package B2 describes sludge waste originating from the chemical treatment of radioactive effluents, which is dried and embedded in bitumen. The effluents considered here are produced at various stages of fuel reprocessing and during operations carried out on equipment and facilities (decontamination and rinsing). 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 sludges.

At the STEL (Marcoule) and STE3 (La Hague) effluent treatment plants, started up in 1966 and 1989 respectively, these sludges have been conditioned by embedding in bitumen poured into steel drums. However, sludges from the effluents produced and chemically treated at STE2 (effluent treatment plant nº 2) at La Hague, from 1966 to 1990, have been gradually stored in tanks and silos at the facility, pending conditioning. The planned conditioning method for these sludges is also embedding in a bitumen matrix.

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. Radiolysis of the bitumen therefore leads to the production of gas, mainly hydrogen, as well as traces of carbon monoxide and dioxide and methane (for hydrogen, 1 to 2 litres at atmospheric pressure per year for STE3 and STEL packages, and 9 to 10 litres for STE2 packages).

[5] Large-sized RIC fuel bundles and thimbles are cut in pieces before emplacement in their compacting case.

Furthermore, the packages of bitumen-embedded waste do not all have the same geometry. A first group of packages (reference package B2.1), representing 45% of the packages recorded in the inventory, corresponds to stainless steel primary drums with a capacity of 238 litres (STE3/STE2) and 245 litres (STEL from October 1996). These packages are illustrated in Figure 2.1-2.



Fût en inox de 238 litres (STE3/STE2)

Fût en inox de 245 litres (STEL)

Figure 2.1-2 STE3/STE2 and STEL stainless steel drums

The second group of packages (reference package B2.2, 55% of the bitumen packages recorded in the inventory) corresponds to 428 litre stainless steel drums. These drums (see Figure 2.1-3) are used for reconditioning non-alloy steel primary drums produced at STEL between 1966 and October 1996.

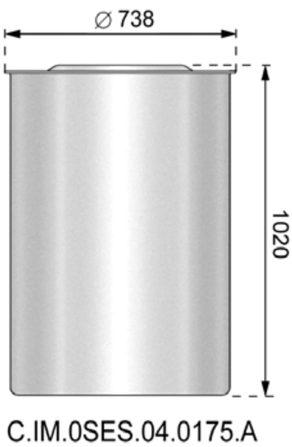


Figure 2.1-3 Stainless steel drum used for reconditioning primary drums in non-alloy steel

The hypothesis adopted for the studies is the complete filling of the interstitial gap with an incompressible material, such as mortar for example, in order to limit long-term mechanical deformations in the repository.

- **Reference package B3**

Reference package B3 groups together technological waste resulting from the operation and maintenance of the nuclear facilities operated by COGEMA and the CEA. This waste mainly consists of various types of solid waste (various metals and organic matter), but also includes filtration sludges and evaporation concentrates. This assembly also includes various waste produced at Marcoule such as graphite, ion-exchanger resins and zeolites. The radiological activity of the waste, especially technological waste, is most often due to the presence of contamination on the surface of the waste by fission products and/or activation products and/or actinides.

This waste is conditioned differently according to the origin of its production and/or its nature. The issues raised by these waste packages are therefore mostly linked to the diversity of (i) their chemical content, in connection with the types of waste and the conditioning matrices used, and (ii) the forms and materials of the containers. Owing to their chemical nature, some packages are also liable to produce gases by radiolysis, essentially hydrogen. These packages do not generate heat

For the needs of the study, the various existing and future technological waste packages are described by reference packages which are broken down on two levels (see Table 2.1-1). Package groupings at level 2 have been defined based on the materials used for the containers and the homogeneous or heterogeneous nature of the conditioned waste:

- B3.1: heterogeneous waste contained in concrete envelopes;
- B.3.2: homogeneous waste contained in concrete envelopes;
- B3.3: heterogeneous waste contained in metallic envelopes.

The breakdown into level 3 reference packages corresponds to the consideration of the chemical nature of the waste, the risk of hydrogen production and the dimensions of the packages (level 3 reference packages combined respectively with level 2 reference packages: B.3.1, B3.2 and B.3.3 are classified by ascending 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 producing hydrogen

	Reference packages	Container material	Embedding or immobilising matrix	Presence of metallic materials	Presence of organic matter
B3.1	B3.1.1.	Concrete	Cement-bitumen	None	x
	B3.1.2	Fibre-reinforced concrete or asbestos cement	Cement	x	x
	B3.1.3	Concrete or non-alloy steel	Cement-bitumen or mortar	x	x
B3.2	B3.2.1	Concrete	Cement	None	x
	B3.2.2	Fibre-reinforced concrete	Mortar	x	x
B3.3	B3.3.1	Stainless steel	None	x	x
	B3.3.2	Stainless steel	Cement	None	x
	B3.3.3	Fibre-reinforced concrete or asbestos cement	Cement-bitumen	None	x
	B3.3.4	Concrete or non-alloy steel	Ciment-bitume ou liant hydraulique	x	x

Table 2.1-1 Summary of the main characteristics of the reference packages for cemented or compacted technological waste

- **Reference packages B4 and B5**

These two reference packages contain waste resulting from the reprocessing of spent fuel at COGEMA plants, corresponding to the elements forming the metallic framework of the fuel assemblies. The waste is separated from recyclable nuclear materials (uranium and plutonium) and fission products and minor actinides at the start of the reprocessing stage, during fuel shearing and dissolving operations.

This waste is commonly known as "hulls and end caps" in pressurised water reactor fuel assemblies. The hulls correspond to the cladding of the fuel rods, recovered in sections approximately three centimetres long, from which the nuclear material has been extracted by dissolving in acid. The end caps correspond to the parts at each end of the fuel assembly.

The cladding waste under consideration here comes from reprocessing operations in the COGEMA plants at La Hague. It includes (i) the waste produced during earlier reprocessing of GCR and PWR fuels, stored today in silos and pools, and (ii) the waste resulting from current and future treatments on the various types of UOx and MOX PWR fuels, defined in the study scenarios set out in paragraph 2.1.3.

There are several types of materials in cladding waste: magnesium-zirconium and magnesium-manganese alloys for the GCR fuels; zirconium-tin (zircaloy 4) or zirconium-niobium (M5) 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.

The initial method of conditioning the cladding waste of PWR fuel assemblies consisted of cementation in large stainless steel drums (reference package B4, see Figure 2.1-4). This process was applied between 1990 and 1995, and then replaced by a waste compacting process, implemented in the cladding hull compacting facility (ACC) at La Hague, from 2002 (reference package B5).

Figure 2.1-4 1800-litre stainless steel drum containing cemented cladding

The compacting process applies to cladding waste originating from GCR and PWR fuels, previously reprocessed but today stored in silos and pools, and cladding waste resulting from the processing of fuels unloaded at present and in future years from PWRs. As mentioned above, some packages also contain compacted technological waste from the operation and/or maintenance of the site's facilities. Given the diversity and nature of the flows of waste in question, distinction is made between four sub-assemblies of compacted cladding waste packages (CSD-C). They are summarized in Table 2.1-2.

Reference package	Cladding waste materials	Presence of technological waste	Presence of organic matter	Production of gas by radiolysis (H_2)	Heat transfer and irradiation level on date of package production
B5.1	Zirconium-tin or zirconium-niobium alloys, stainless steels, nickel alloy	x	x	x	Packages transfer little heat, high level of irradiation
B5.2	As subassembly B5.1	x	None	None	As B5.1
B5.3	Zirconium-tin alloy, stainless steels, nickel alloy	None	None	None	Packages transfer no heat, moderate level of irradiation
B5.4	Magnesium-zirconium or magnesium-manganese alloys	None	None	None	Packages transfer no heat, low level of irradiation

Table 2.1-2 Summary of the main characteristics of the reference packages for cladding waste with or without compacted technological waste

- **Reference package B6**

This reference package contains waste produced at the COGEMA Marcoule site, stored at present, comprising (i) operating waste from the Marcoule vitrification facility, (ii) cladding waste from the fuel reprocessed at the UP1 plant, and (iii) technological operating and maintenance waste from the Marcoule site facilities. It is broken down into five reference packages taking account of the specific nature and characteristics of the waste and dimensions of the envelopes.

Reference package B6.1 contains technological waste linked to the operation of the Marcoule vitrification facility (AVM). The waste, consisting of various equipment and parts in steel, are placed in a stainless steel container of similar geometry to the AVM vitrified waste containers. The packages have an average mass of 160 kilograms but can reach 320 kilograms (excluding the waste immobilisation material). The radiological activity corresponds to contamination at the surface of the waste. These packages do not give off heat.

Reference packages B6.2 and B6.3 contain 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 are composed either of aluminium and stainless steel cladding waste (reference package B6.2), or of magnesium alloy cladding waste (reference package B6.3). The packages have an average mass of less than 300 kilograms (excluding the waste immobilisation material). The heat energy of the packages containing aluminium and steel cladding waste is of the order of 10 watts, but will be a maximum of 0.5 watt by the 2025 dateline. Packages containing magnesium alloy cladding waste do not give off heat.

The other two reference packages contain technological waste consisting either of a mixture of metallic and organic materials (reference package B6.4), or of metallic materials only (reference package B6.5). The waste is also temporarily conditioned in EIP drums.

The packages containing metallic and organic waste have an average mass of 90 kilograms (excluding the waste immobilisation material). They do not transfer heat and are not irradiating. The release of hydrogen formed by radiolysis of the organic matter is to be considered.

The packages containing just metallic technological waste do not transfer heat or produce gas.

The main characteristics of the five cladding and technological waste reference packages are summarised in Table 2.1-3.

Reference packages	Presence of metallic waste	Presence of organic waste	Production of gas by radiolysis (H_2)	Heat transfer and irradiation level on date of package production
B6.1	x	None	None	Packages transfer no heat, low level of irradiation
B6.2 B6.3	x	None	None	Packages transfer no or medium heat, irradiation level medium or high depending on the packages
B6.4 B6.5	x	x	x	Packages do not transfer heat and are not irradiating

Table 2.1-3 Summary of the characteristics of reference packages B6 for cladding and technological waste in drums

- **Reference package B7**

Reference package B7 contains PWR source fuel rods in addition to industrial-use sealed sources.

The industrial-use sealed sources contain radioactive materials of a highly diverse nature, activity and half-life. Several thousand sources were conditioned in concrete containers between 1972 and 1985, these concrete containers subsequently being reconditioned in metallic containers. These so-called “source blocks” are currently being stored by the CEA at the Cadarache site (reference package B7.1, see Figure 2.1-5). These packages are large and weigh between 6 and 9.2 tonnes.

Figure 2.1-5 Source blocks

Source fuel rods (reference package B7.2) are operating waste from PWRs, like the various activated metallic waste described by reference package B1. 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 capsule of californium are unloaded at the end of the first cycle, whereas secondary source rods, comprising an antimony-beryllium mixture, undergo several irradiation cycles before they are discarded. The primary source clusters unloaded from 900 MW reactors have been processed to retrieve the californium capsules and are not considered as waste (and are therefore excluded from the inventory). The total weight of waste for conditioning is less than two tonnes.

For the study, the conditioning hypothesis adopted, as in the case of other PWR activated waste, is the shearing and compacting of the source rods followed by emplacement in a CSD-C container. It will be noted that a maximum of four CSD-C containers will be produced from the conditioning of the source rods.

Moreover, several thousand other sealed sources are today being stored at various facilities. They cover a very broad range of radioactive isotopes of variable activity and half-life. For the study, it was agreed to take account of all sources with a half-life greater than or equal to that of caesium 137 (30 years), in keeping with the range of waste accepted for surface disposal at the Aube facility. The conditioning hypothesis envisaged at the present stage is that the sources will be cemented in EIP drums (reference package B7.3).

- **Reference package B8**

Reference package B8 contains various types of waste comprising radiferous lead sulphates, radium-laden items for medical use (ORUMs) and lightning rods. However, the inclusion of this waste in the HLLL inventory remains exploratory.

Radiferous lead sulphates (reference package B8.1) originate from the processing of uranium ore at the Le Bouchet plant. The waste was initially placed in metallic drums which have successively been reconditioned with a view to storage. For the studies, the hypothesis adopted is the collection of the primary drums of radiferous lead sulphates for conditioning in EIP drums. It will be noted that the methods of limiting residual voids within the primary packages have not been defined at this stage.

Lightning rods (reference package B8.2) are objects containing either 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. On average, the packages contain around 200 radium or americium lightning rod heads and their activity level is of the order of 10 gigabecquerels (GBq).

ORUMs (reference package B8.3) 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 around 100 grams of radium were extracted, including approximately 50 grams used for ORUM production. Note that all of the ORUMs (a total of some 5,000) can be conditioned in a single EIP drum.

2.1.4.2 Primary reference packages for vitrified C waste

Vitrified waste is a product of spent fuel reprocessing. It mostly includes fission products and minor actinides (neptunium, americium and curium) formed by nuclear reaction and contained in the spent fuel separated of its uranium and plutonium on reprocessing. It is calcinated and incorporated into a glass matrix. The manufactured glass is poured at temperature into a stainless steel container. The radiological activity is spread evenly throughout the vitrified waste.

In France, vitrification was developed at several pilot facilities operated by the CEA, including the PIVER pilot facility which has since been shut down, and then implemented on an industrial basis at three facilities 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) the initial characteristics of the solutions of fission products and minor actinides from the fuel processed at these facilities, (ii) the various levels of concentration of fission products in the glass, and (iii) the age of the waste.

It was therefore decided to differentiate between several sets of vitrified C waste packages containing respectively (i) older glass production (reference package C0), (ii) current glass production or that envisaged in the short term (reference packages C1 et C2), (iii) forecast glass production, including UOx/MOX glass and UOx glass with a hypothesis of incorporating a small fraction of plutonium (reference packages C3 and C4 respectively).

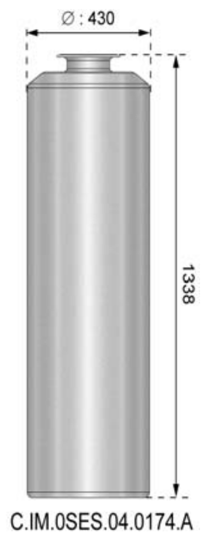
- **Reference package C0**

The reference package comprises (i) glass packages containing solutions of fission products originating from the reprocessing of fuel from graphite-moderated, gas-cooled reactors (Sicral-type GCR fuel) and fuel from the Phénix fast neutron reactor at the PIVER facility, (ii) glass packages containing solutions of fission products, known as UMo, originating from GCR fuel previously reprocessed at the COGEMA La Hague site and now stored, and (iii) glass packages produced since 1978 at the COGEMA Marcoule vitrification facility (AVM glass) containing fission products and actinides mostly from the reprocessing of GCR fuel. 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.

The PIVER packages (reference packages C0.1), produced between 1969 and 1981, are mainly composed of GCR glass. The vitrified waste is conditioned in stainless steel containers of the same diameter, but different heights.

The UMo glass packages (reference package C0.2) correspond to the future conditioning of existing solutions of fission products; these solutions are derived from GCR fuel reprocessed at the COGEMA UP2-400 plant in La Hague. Due to the chemical nature of the solutions, a specific glass formulation needs to be developed and the process equipment adapted, especially with regard to the vitrification furnace. Following the hypotheses adopted, the average weight of the conditioned waste is 400 kilograms per package. This waste will be conditioned in a stainless steel container identical to the one used today at the COGEMA R7 and T7 vitrification facilities in La Hague. This container, called a Standardised Container for Vitrified Waste (CSD-V), is illustrated in Figure 2.1-6.



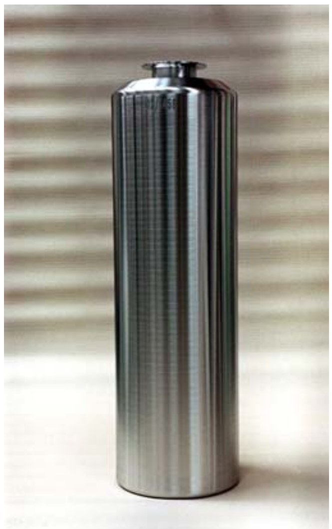


Figure 2.1-6 Standardised Container for Vitrified Waste (CSD-V)

The AVM glass packages (reference package C0.3) contain all vitrified waste produced since 1978 at the COGEMA vitrification facility in Marcoule. As stated above, the vitrified solutions mainly originate from GCR fuel reprocessed at the UP1 plant at the site. It will be noted that four different glass formulations are used for one or more vitrification campaigns. The vitrified glass is conditioned in a stainless steel container.

- **Reference packages C1 and C2**

These packages contain solutions of fission products originating from the reprocessing of UOx/URE PWR fuel at the COGEMA plants in La Hague, conditioned in glass form in a CSD-V container (see Figure 2.1-6). The production and conditioning of the waste is assumed to occur after an average storage period of eight years once unloaded from the reactors. The conditioned waste weighs 400 kilograms on average per package.

Reference package C1 corresponds, from the point of view of heat, to current industrial production. Depending on the hypotheses adopted, the vitrified waste consists of a mixture of solutions of fission products derived from UOx1 (average burnup 33 GWd/t), UOx2/URE (average burnup 45 GWd/t) and UOx3 fuels (average burnup 55 GWd/t).

Reference package C2 corresponds to packages with slightly higher heat energy. The vitrified waste consists of a mixture of solutions of fission products derived from UOx2/URE and UOx3 fuels, the average burnup of which, as above, is 45 GWd/t and 55 GWd/t respectively.

- **Reference packages C3 and C4**

These packages relate to potential glass productions on the COGEMA La Hague site. In the scenarios adopted for the study, these packages were defined assuming that the production and conditioning of the waste occur, as for the above glass, after an average storage period of eight years once unloaded from the reactors. Note that other possibilities could be envisaged.

Reference package C3 describes glass resulting from the conditioning of solutions of fission products derived from UOx and MOX fuels. They are defined as consisting of a mixture of 15% MOX and 85% UOx2.

Reference package C4 describes the vitrified waste originating from the reprocessing of UOx fuel containing a low additional plutonium charge. The rate of plutonium inclusion in the glass is set at one per cent by mass, or approximately 4 kilograms per package. The incorporated plutonium comes from UOx2 fuel.

2.1.5 The case of spent fuel

The spent fuel taken into account in the study is fuel originating from PWR plants, plus fuel from channels which have been shut down, research reactors (GCR, EL4) and National Defence activities. This fuel will be included in the study if it is considered as waste, assuming that processing is discontinued, which is not the strategy currently adopted in France.

In the scope of scenario S2, PWR fuel is predominant in terms of the number of assemblies. The fuel types considered (see paragraph 2.1.3 are UOx2 and URE (45 GWd/t), UOx3 (55 GWd/t) and MOX (48 GWd/t).

- **Reference packages CU1 and CU2**

The reference fuel assembly corresponds to a Fragema-designed “second generation advanced” assembly with thicker guide thimbles and zirconium alloy cladding. It is illustrated in Figure 2.1-7. It is named AFA-2GE for 900 MWe PWRs and AFA-2LE for 1300 MWe and 1450 MWe PWRs

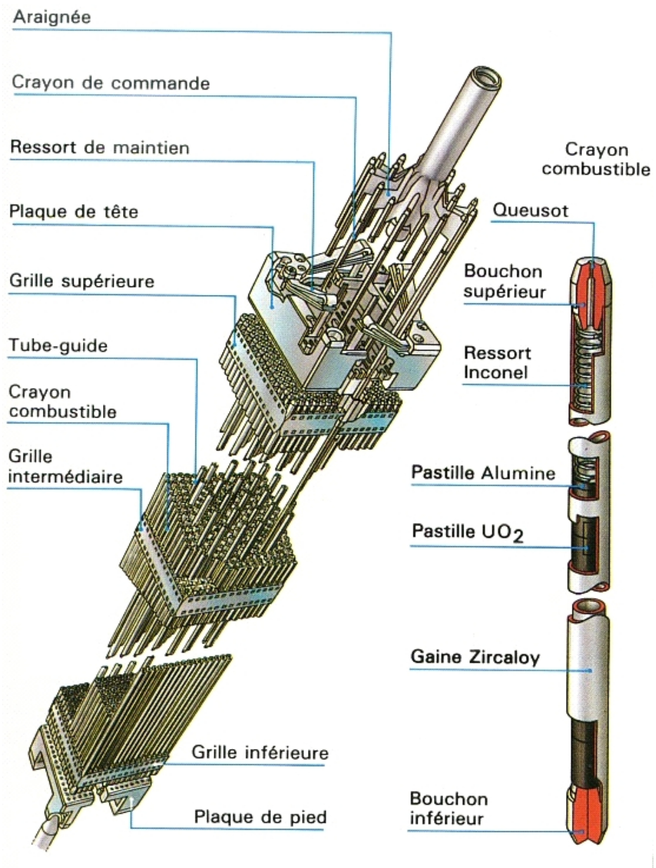


Figure 2.1-7 *PWR fuel assembly*

The assembly consists of a rigid metallic structure (skeleton) supporting 264 geometrically identical fuel rods distributed in a square 289-slot grid.

The fuel rod is made up of:

- a cylindrical metallic cladding closed at the top and bottom ends by two welded plugs (the upper plug has a pip which is sealed once the rod has been filled with helium);
- a stack of fuel pellets to approximately 95% of the height of the rod;
- a helical spring at the top of the fuel pellet stack to hold it axially during handling operations.

Reference package CU1 corresponds to enriched uranium oxide (UOx) or enriched recycled uranium oxide (URE) fuel. Reference package CU2 corresponds to mixed uranium and plutonium oxide (MOX) fuel.

- **Reference package CU3**

The fuels contained in reference package CU3 are quite varied in nature. They include (i) fuels from GCRs, (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 ships.

GCR fuels correspond to a residual tonnage of unprocessed fuel of approximately 15 tonnes. They are conditioned in cylindrical holders with a diameter of 88 mm or 130 mm and a height of 655.5 mm.

EL4 fuels represent about 50 tonnes of heavy metal. The fuel element takes the form of a cluster of 19 rods held in an ATR structure (zirconium alloy with copper and molybdenum).

These first two types of fuel make up reference package CU3.1, which breaks down into CU3.1.1 (GCR fuel) and CU3.1.2 (EL4 fuel).

Célestin fuel elements (reference package CU3.2) consist of metallic plates containing enriched uranium, mounted on a metallic structure. They are conditioned in stainless steel holders approximately 340 mm in diameter and 1100 mm long.

Nuclear propulsion fuels (reference package CU3.3 which breaks down into reference packages CU3.3.1 to CU3.3.6) consist of (i) oxide fuels based on platelets of sintered uranium oxide, and (ii) metallic fuels based on highly enriched uranium metal. These latter fuels are no longer used.

In both cases, the fuel takes the form of an assembly made up of several bundles. These bundles are separated from the assembly and conditioned in holders of the same diameter (approximately 340 mm, like the holders containing the Célestin fuel elements) but of variable length according to the dimensions of the bundles.

2.1.6 Number and volumes of primary packages considered

Within the framework provided by the scenarios presented above (see paragraph 2.1.3), la quantification du nombre de colis types s'appuie sur les inventaires et les prévisions de production de déchets établies par les producteurs, évaluées par l'Andra sur la base des données produites.

the quantification of the number of reference packages is based on the inventories and waste production forecasts drawn up by the producers and assessed by Andra on the basis of the data produced.

For the waste to be produced, except for spent fuel reprocessing, dimensioning margins were added by Andra in order to take account of uncertainties. It should be noted that the disposal possibilities of certain waste packages in the scope of other disposal solutions were not considered in order to obtain cautious estimates.

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

he numbers and volumes of packages considered in the studies, for B waste, are presented in Table 2.1-4 and in Table 2.1-5.

The numbers and volumes of C waste reference packages are given in Table 2.1-6.

Quantitative data relating to spent PWR fuel (reference packages CU1 and CU2) is provided in Table 2.1-7. Furthermore, the number of primary holders considered for CU3 fuels, where applicable, is 5810.

The volumes indicated in the tables correspond to the volumes of conditioned waste with the hypotheses formulated above.

Reference packages	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)
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 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	7990	3420	7990	3420	7990	3420	7990	3420
Total 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	**COGEMA La Hague**	**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**

Table 2.1-4 *Overall quantitative data, in terms of number and volume of packages, for B waste reference packages.*

Reference Package	Scenario S1a		Scenario S1b		Scenario S1c		Scenario S2	
	Number	Volume (m^3)	Number	Volume (m^3)	Number	Volume (m^3)	Number	Volume (m^3)
B2.1	46 930	11 210	46 930	11 210	46 930	11 210	46 930	11 210
B2.2	58060	24 850	58 060	24 850	58 060	24 850	58 060	24 850
Total of B2	**104 990**	**36 060**	**104 990**	**36 060**	**104 990**	**36 060**	**104 990**	**36 060**
B3.1.1	90	90	90	90	90	90	90	90
B3.1.2	8 690	10 250	8 690	10 250	8 690	10 250	6 440	7 590
B3.1.3	180	690	180	690	180	690	180	690
Total of B3.1	**8 960**	**11 030**	**8 960**	**11 030**	**8 960**	**11 030**	**6 710**	**8 370**
B3.2.1	5 730	2 800	5 730	2 800	5 730	2 800	5 730	2 800
B3.2.2	1 260	1 490	1 260	1 490	1 260	1 490	1 260	1 490
Total of B3.2	**6 990**	**4 290**	**6 990**	**4 290**	**6 990**	**4 290**	**6 990**	**4 290**
B3.3.1	1 200	220	1 200	220	1 200	220	900	160
B3.3.2	7 990	3 420	7 990	3 420	7 990	3 420	7 990	3 420
B3.3.3	1 700	850	1 700	850	1 700	850	1 700	850
B3.3.4	6 100	7 450	6 100	7 450	6 100	7 450	6 100	7 450
Total of B3.3	16 990	11 940	16 990	11 940	16 990	11 940	16 690	11 880
Total of B3	**32 940**	**27 260**	**32 940**	**27 260**	**32 940**	**27 260**	**30 390**	**24 540**
B5.1	7 940	1 450	7 400	1 350	7 400	1 350	2 140	390
B5.2	31 760	5 810	29 600	5 420	29 600	5 420	8 560	1 570
B5.3	2 500	460	2 500	460	2 500	460	2 500	460
B5.4	400	70	400	70	400	70	400	70
Total of B5	**42 600**	**7 790**	**39 900**	**7 300**	**39 900**	**7 300**	**13 600**	**2 490**
B6.1	180	30	180	30	180	30	180	30
B6.2	930	400	930	400	930	400	930	400
B6.3	7 550	3 230	7 550	3 230	7 550	3 230	7 550	3 230
B6.4	1 200	510	1 200	510	1 200	510	1 200	510
B6.5	950	410	950	410	950	410	950	410
Total of B6	**10 810**	**4 580**	**10 810**	**4 580**	**10 810**	**4 580**	**10 810**	**4 580**
B7.1	41	155	41	155	41	155	41	155
B7.2	4	0,7	4	0,7	4	0,7	4	0,7
B7.3	3 000	1 285	3 000	1 285	3 000	1 285	3 000	1 285
Total of B7	**3 045**	**1 440**	**3 045**	**1 440**	**3 045**	**1 440**	**3 045**	**1 440**
B8.1	1 100	470	1 100	470	1 100	470	1 100	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 of B8	**1 350**	**775**	**1 350**	**775**	**1 350**	**775**	**1 350**	**775**

Table 2.1-5 Detail of the number and volume of reference packages B2, B3, B5, B6, B7 et B8

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**	**3 8350**	**6 710**	**4 640**	**810**
C2	**COGEMA La Hague**	**990**	**170**	**27 460**	**4 810**	**0**	**0**	**5 920**	**1 040**
C3	**COGEMA La Hague**	**13 320**	**2 330**	**0**	**0**	**0**	**0**	**0**	**0**
C4	**COGEMA La Hague**	**13 250**	**2 320**	**0**	**0**	**0**	**0**	**0**	**0**

Table 2.1-6 Overall quantitative data, in terms of number and volume of packages, for C waste reference packages

	Sites de production	Number of PWR fuel assemblies			
		Scenario S1a	Scenario S1b	Scenario S1c	Scenario S2
AFA-2GE “short” UOx assembly, type CU1	EDF	0	0	0	27 200
AFA-2LE “long” UOx assembly, type CU1		0	0	0	26 800
Total UOx assemblies, type CU1		**0**	**0**	**0**	**54 000**
AFA-2GE “short” MOX assembly, type CU2	EDF	0	5 400	5 400	4 000
Total MOX assemblies, type CU2		**0**	**5 400**	**5 400**	**4 000**

Table 2.1-7 Number of PWR fuel assemblies

2.1.7 Radiological inventory considered

The radiological inventory of the packages covers a long list of radionuclides comprising fission products (PF), activation products (PA) and high-level, long-lived actinides. It is described in the document “Reference knowledge document and inventory model of waste packages” [ix].

The various radionuclides are characterised by their half-life and are thus split up as follows:

- 44 short-lived radionuclides (31%), with a half-life not exceeding six years,
- 16 medium-lived radionuclides (11%), with a half-life of between seven and 30 years,
- 84 long-lived[6] radionuclides (58%), with a half-life in excess of 31 years.

6 Nickel-63 is a special case on account of its half-life (100 years) and its character as an activation product. It is present at a high level of activity in a large number of packages. Its activity over the first centuries is particularly high, dominant even, compared with that of other long-lived PA/PF in B reference packages, especially packages B1, B4, B5 (except B5.4) and B6.2. The activity of nickel-63 is comparatively lower in C packages and fuels due to the predominance of fission products. Nickel-63 is therefore presented separately in the radiological inventory.

Figure 2.1-8 and Figure 2.1-9 summarise the radiological inventory taken into account in the safety studies.

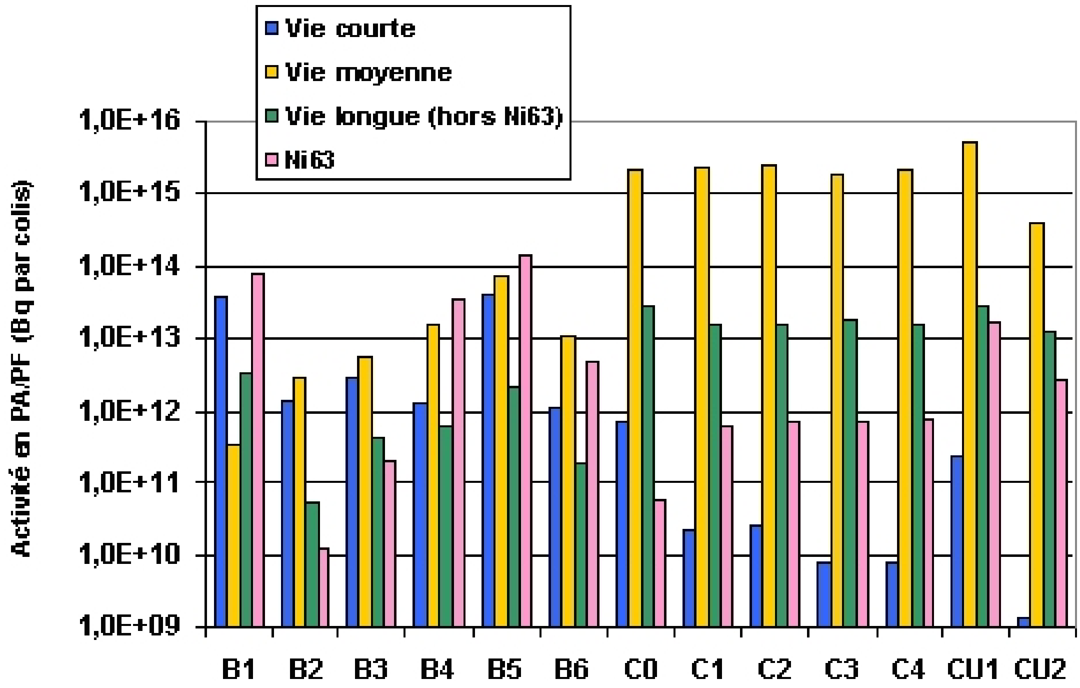


Figure 2.1-8 *PA/PF activity of the various reference packages*

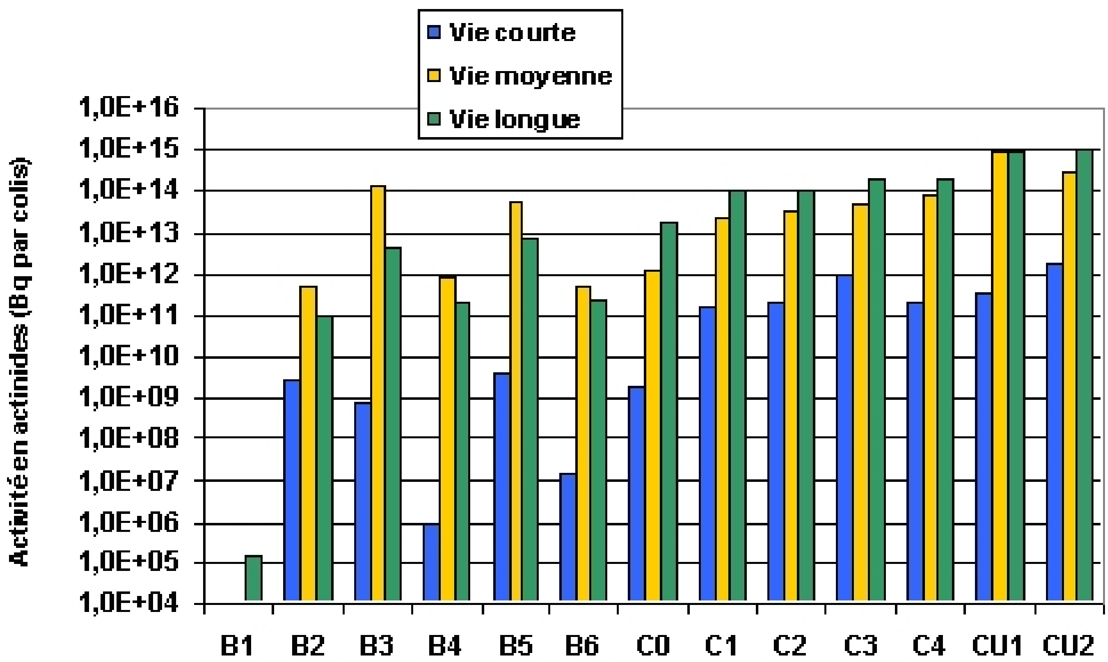


Figure 2.1-9 *Actinide activity of the various reference packages*

2.2 The granite medium

This general description of the granite medium is also to be found in the synthesis report.

The design of a repository aims first of all to harness the properties of a geological formation favourable to deep disposal so that it fulfils the various safety functions assigned to it.

In the absence of a specific study site, the study of the repository design cannot draw on the description of a particular granite massif. The design principles adopted by Andra are therefore based first and foremost on the properties common to all granites (cf. § 2.2.1).

However, French granites present specific characteristics which the design studies must integrate in order to propose relevant options. Andra has therefore produced a reference knowledge document on French granites which has made it possible to identify which characteristics of the granites could influence the design of a repository (cf. § 2.2.2).

2.2.1 The properties of granite suited to the disposal of radioactive waste

For the study of a repository, the word "granite" designates both a rock and a geological formation. Granite as a geological formation is generally structured in massifs[7]**.** Therefore the possibility of constructing a repository in a granite medium depends on the properties of the rock and the characteristics and geological context of the massif studied.

2.2.1.1 Granite rock: a hard, strong rock

The common perception of granite as a stone long used for perennial ornamentation purposes is that of a hard rock with very low porosity and permeability (see Figure 2.2-1).

The mechanical strength of the rock is a naturally interesting element for the construction of structures at depth. It means that rock can be excavated without the need for significant ground supports over volumes compatible with the dimensions and depth of a repository. This mechanical strength is explained by the texture of the rock consisting of quartz (crystallised silica) and feldspars (alumina silicates).

Quartz also contributes to the generally high thermal conductivity of the rock, making it a formation likely to dissipate easily the heat given off by radioactive waste.

Granite rock contains little water: its porosity to water is generally less than 0.5%. The permeability of the rock is very low and may be barely measurable in situ.

Such characteristics denote properties which, in theory, are advantageous for the disposal of radioactive waste.

---

[7] Sedimentary geological formations are generally found in superimposed layers (example: the Callovo-Oxfordian clay formation at the Meuse/Haute-Marne site). The geometry of magmatic formations is more often 3D than 2D. For granite the term "massif" is generic and is applicable to most formations that we are likely to come across.

Figure 2.2-1 *Unfinished obelisk in the Aswan granite quarry (left) - "Rock of Ages" quarry (Vermont, USA, right)*

2.2.1.2 A granite massif: a vast formation whose properties are explained by history

- **Granite massif**

A granite massif capable of hosting a repository is a geological formation, generally of vast size, extending downwards which, on account of the strength of the rock, offers tremendous flexibility for the architectural design of a repository.

However, on the scale of a massif, granite is not a homogeneous, monolithic geological formation. It is important to understand and model the structure with sufficient detail in order to study how the design of a repository in granite could be adapted there or not.

This understanding is based on a fine characterisation of the massif studied by a series of methods implemented over the successive stages of site exploration. Due to the complementarity of existing methods, a gradual exploratory approach suited to the studied site can be defined.

The interpretation of the gathered data is largely supported by the reconstruction of the geological history of the granite massif. Establishing the geological history of a granite massif means understanding the phenomena that produced and structured it over time; it also means integrating the various components of a massif into a common, consistent historical logic.

Inset 2 *Origin of a granite massif*

A granite massif originates from magma produced at depth in connection with the movements and collisions of the "plates" structuring the lithosphere of the globe. This original magma is emplaced and solidifies at a depth of several kilometres or more. The emplacement conditions determine the structure of the granite and, in part, its fracturing mode.

The granite massif then becomes a constituent of the earth's crust and follows its evolution through the geological periods. It can therefore be affected by new deformations and fracturing. It can be "altered" by the circulation of hydrothermal fluids capable of changing the composition of the rock and mineralising the fractures. Finally, the rising of the earth's crust and the erosion phenomena can cause the granite massif to "outcrop" at the surface.

The massif retains the traces of the various stages of its history: enclaves of the surrounding formations traversed by the magma, local distinctions between rocks of different grain or mineralogy on crystallisation, "alterations" to the original mineralogy, nature of the minerals filling the faults or cracks created on fracturing, etc. All of these clues help reconstruct the sometimes complex history of the granite massif studied. This history thus determines the properties of the granite massif, and also of the elements of its environment (see Figure 2.2-2).

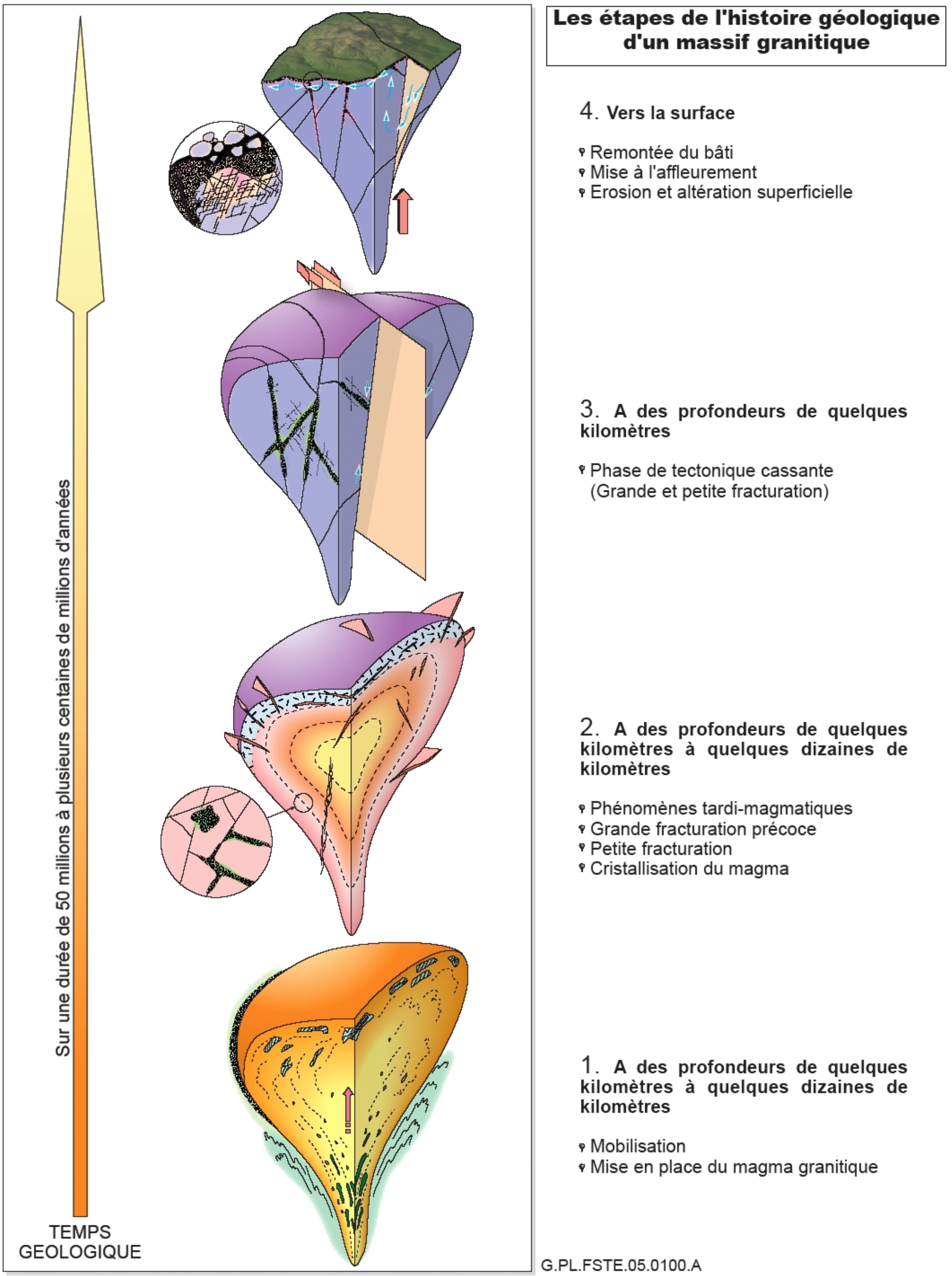


Figure 2.2-2 Stages in the geological history of a granite massif

On the surface, the pattern of the fractures in the rock is highlighted by the effects of alteration, decompression and surface erosion. Between the surface and a depth of a hundred metres or so, the effects of this surface alteration fade: the rock is "sound" and the variations in composition are the result of the original geological history of the granite.

The lithological nature and the mineralogical composition of the rock may also evolve from one point of the massif to another, depending on how the granite was emplaced. As a general rule, these variations do not significantly change the mechanical and permeability properties of the rock.

More important is the organisation of the fracturing of the granite as a result of its geological history. These fractures lead to more or less pronounced discontinuities in the properties of the rock which must be taken into account for the design of a repository.

- **Major, medium and minor fracturing**

A granite massif is traversed by fractures of various sizes (see Figure 2.2-3). Their number depends on their size. Minor fractures, measured on a scale of one to tens of metres, are far more numerous than major fractures, extending from one to several kilometres. Medium fracturing, ranging from tens to hundreds of metres, forms the transition.

Minor fractures can affect the permeability of the rock where the repository structures would be located. The permeability of the rock therefore depends on the properties of each of the minor fractures, their density and their length. Minor fractures, which may be more or less connected, generally conduct very little water. Therefore the permeability of granite, away from major and medium fractures, is low or very low in theory, and limits water circulation significantly.

Major fractures, or faults, are the principal vectors of water circulation in granite, although this does not mean that they contain large quantities of water. Most of the water in granite is stored in the largest faults, where they are not clogged by clay minerals.

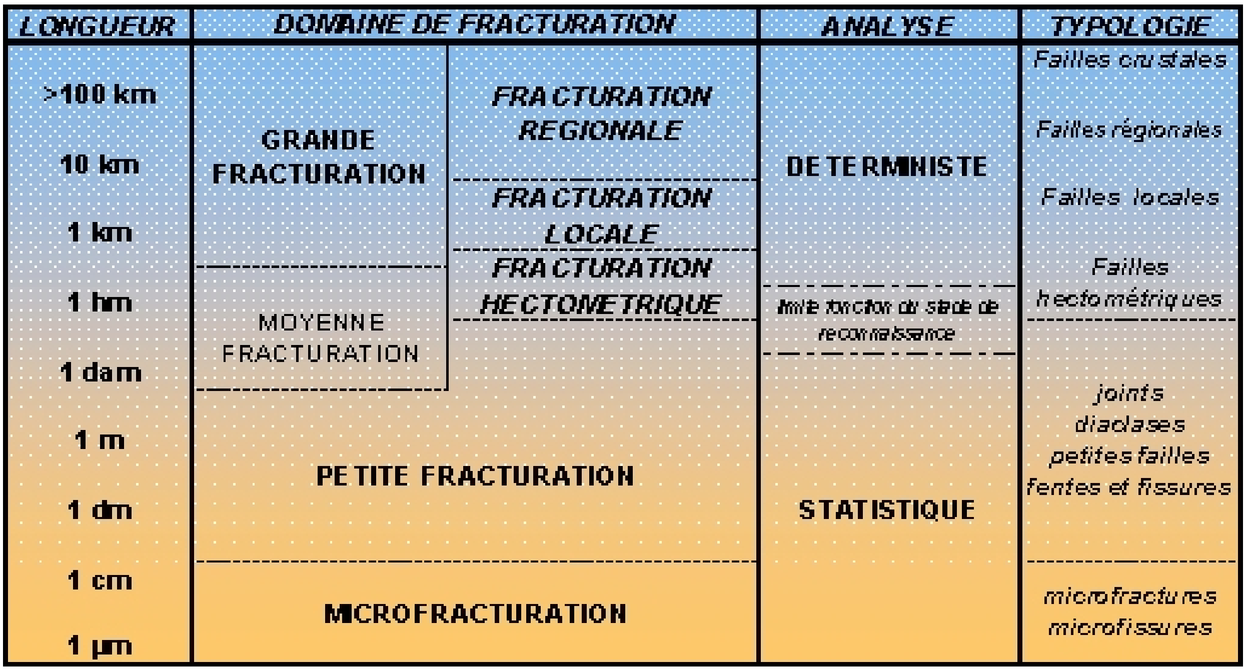


Figure 2.2-3 Fracturing scales

Major fracturing and minor fracturing are also differentiated by the way in which they are treated – deterministically or statistically:

- *minor fractures* result in part from thermal “shrinkage” which takes place on solidification of the magma and more generally during subsequent deformation phases. The characterisation and modelling of minor fracturing are based on a *statistical approach.* Fracturing models are based on systematic geological surveys which give rise to the laws of distribution of their main characteristics in the granite: size, orientation and dip;
- *major fractures, or faults,* always result from significant deformations in the granite massif during the tectonic phases. Figure 2.2-4). The mode and intensity of the fracture can vary from one point of a massif to another and from one massif to another, which leads to sundry fracture models dividing the granite massif up into “blocks” of various shapes. In order to detect and model them, large fractures and faults are based on a *deterministic approach* during the exploratory phases: they are generally large enough to be identified one by one and form the structuring elements of granite modelling.

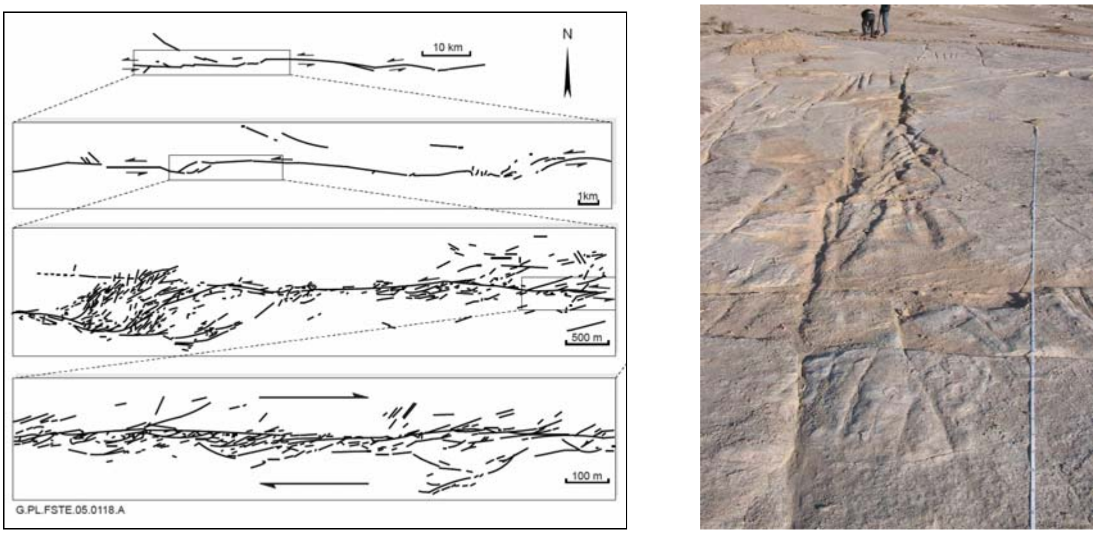


Figure 2.2-4 Diagram of a major fault composed of lower-order fractures

2.2.1.3 Slow hydrogeological flows at depth

The water present in the faults in the granite moves very slowly at depth. The driving force of the movements is a *hydraulic gradient,* linked to the topography. Schematically, the more the topography is contrasting, the more the gradients tend to be steep (see Figure 2.2-5). However, this driving force which tends to set the water in the massif in motion is opposed by major head losses in the granite fractures. Irregularities in the intimate geometry of a fracture network form an obstacle to water movement.

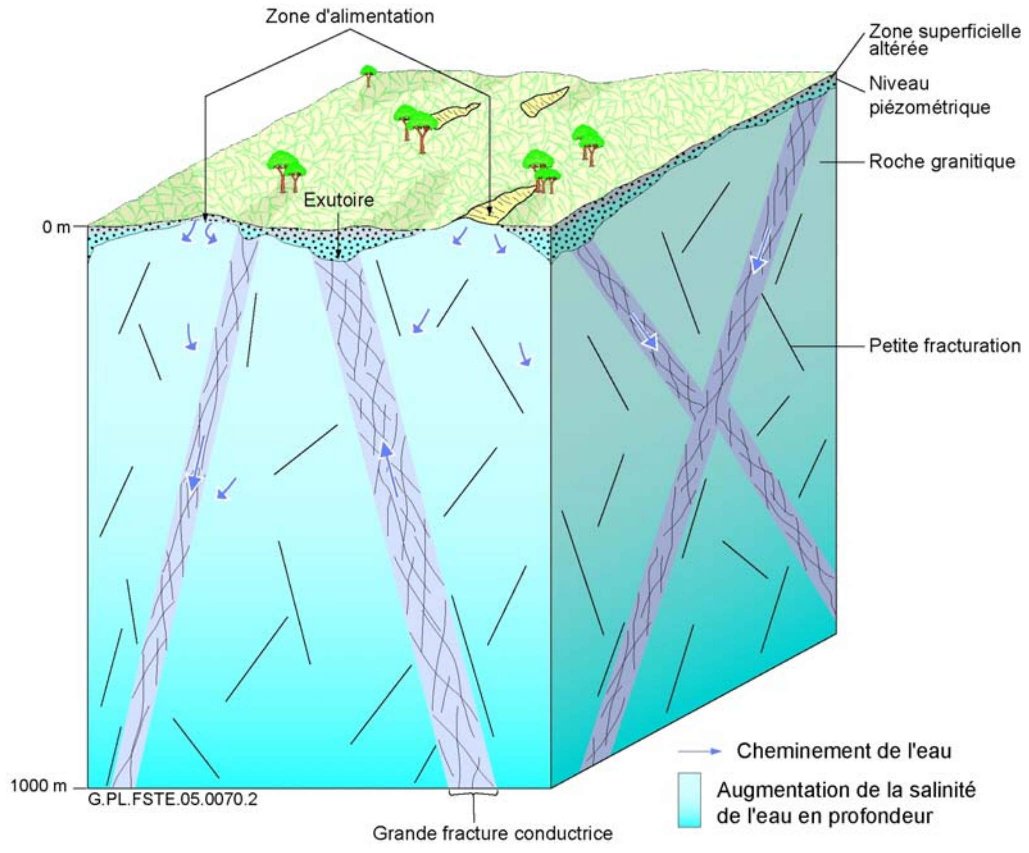


Figure 2.2-5 *Hydrogeology of a granite massif*

2.2.1.4 A deep chemical environment favourable to a repository

In deep granite, the chemical composition of water is generally in equilibrium with the rock or the minerals in the fractures containing the water. The chemical environment is therefore generally reducing. The pH of water in equilibrium with a granite rock is generally close to neutral or slightly basic[8]. These conditions are favourable firstly to the durability of the materials that could be used in a repository and, secondly, to the immobilisation of the majority of radionuclides.

2.2.1.5 Faults and fractures: a capacity to delay the migration of radionuclides

If the fractures in granite are potentially the location of water circulation and a possible vector for the transfer of radionuclides that may be released by a repository, they are also the home of phenomena liable to immobilise or delay this migration. A great deal of work has been carried out internationally on this major aspect.

In particular, experiments performed *in situ,* notably at the Äspö underground laboratory in Sweden, have identified the various phenomena delaying the migration of radionuclides in the fractures and helped to fully understand the nature of this [x].

In particular, by determining relations between these phenomena and the geological and mineralogical characteristics of the fractures, the experimental results can be extrapolated to various types of granite according to their specific characteristics.

8 In certain particular contexts, the pH may be slightly acidic.

2.2.2 The variability of granites in the French geological context

The favourable properties of a granite medium liable to be called upon in the design of a repository differ from one type of granite to another: they depend on its geological history in a given environment.

Consequently, if the design of a repository in granite is based on the generic properties of the granite medium, it also incorporates the specific features of a particular massif. In the absence of a specific study site, Andra carried out a typological analysis in order to gather data on the variability of French granites [vii]. By understanding the differences between granites, they can be taken on board in the design and safety analyses in order to ensure that the design options proposed are fit to fulfill the various safety functions expected, in a generic context.

The variability in the properties of French granites was therefore captured by a typological analysis, based on a broad inventory.Figure 2.2-6).

Figure 2.2-6 Granite massifs in France

Excluded from the inventory are granite zones which clearly would not meet the main criteria of RFS III.2.f. In particular, the zones considered have a surface area in excess of 20 km² and are located away from major faults[9]. Seventy-eight granite zones were therefore taken into consideration.

Initially, the analysis consisted in describing the characteristics of the granites by evaluating their actual or potential variations, and determining how these variations could impact repository design choices. As the granites have rarely been explored directly at depth with the exception of a few specific mining areas, they were described from maps of their outcropping surfaces. The geometric or fracturing characteristics were extrapolated to depths using geological arguments.

Thermal and hydrogeological characteristics were defined on the basis of modelling and extrapolations. These were based on available borehole data to characterise hydrothermal "alterations", determine thermal flows or measure hydraulic transmissivity. Sensitivity analyses corresponding to the level of uncertainty observed were also performed by modelling.

Once the main characteristics had been adequately completed in the inventory, the second phase consisted in performing a statistical analysis of their variability within the French geological context. This analysis enabled the granites to be classified into various "types" for each property studied and, as a result, the distribution of variations in the properties of French granites could be assessed.

Finally, the properties of the massifs studied were compared with those of granites in other countries. This comparison contributed to the validation of the use of data gathered from foreign underground laboratories.

2.2.2.1 Main lessons from the typological analysis

Typological analysis reveals the following points:

the mechanical strength of the granite rock differs from one type of granite to another, particularly according to their hydrothermal alteration. However, these variations are generally unlikely to cause significant differences in the response of a granite to the excavation of structures. Therefore, the differences for the architectural design of a repository are minor and only affect the detailed dimensioning of the engineered structures underground;

- variability in the thermal properties of French granites is considerable enough to play a role in the study of spent fuel disposal. The difference with Fennoscandian granites, where the subsurface temperature is approximately 10°C lower, is significant on this point. For C waste, and especially B5 and B1 waste, which are only slightly exothermic, the differences between granites are not considerable enough to affect the dimensioning of a repository significantly (see Inset 3) ;

Inset 3 *Thermal and dimensioning properties of a repository*

The temperature of deep granite and the conductivity of the rock are important dimensional design parameters for the architecture of a repository. The design of a repository must indeed control the evacuation of the heat given off by C waste and, if applicable, spent fuel. For better distribution of the heat sources and in order to control the evolution of the temperature in the repository, it is possible to juggle with the spacing of the packages in the rock.

The initial temperature of the granite at depth is not directly accessible without boring. Models have therefore been based on maps of thermal flows in France and the thermal conductivity of the rocks according to their quartz content. Estimates made for a depth of 500 metres lead to uncertainties regarding the initial temperatures by plus or minus 3°C to 3.5°C depending on the massifs.

The graph below (see Figure 2.2-7) illustrates the relatively contrasting situations of the various granites in France on this point. Initial temperatures at a depth of 500 metres vary from 17 to 30°C. Rock conductivity values range from 2.4 to 3.8 W/m/K.

The dimensioning of a spent fuel repository is sensitive to such differences: the footprint of the repository for spent MOX-type fuel may be increased or decreased by 30%.

9 A buffer zone of undisturbed rock of between 1.5 to 3 km is adopted according to the size of the fault.

For C waste, the differences have a less marked effect on the dimensioning of a repository for the design options proposed. Indeed, mechanical dimensioning considerations limit the possible "gains" for the most advantageous granites from a thermal aspect; furthermore, the positioning of a thick clay buffer between the packages and the rock in the concepts studied cushions the effects of the differences in thermal conductivity between granites.

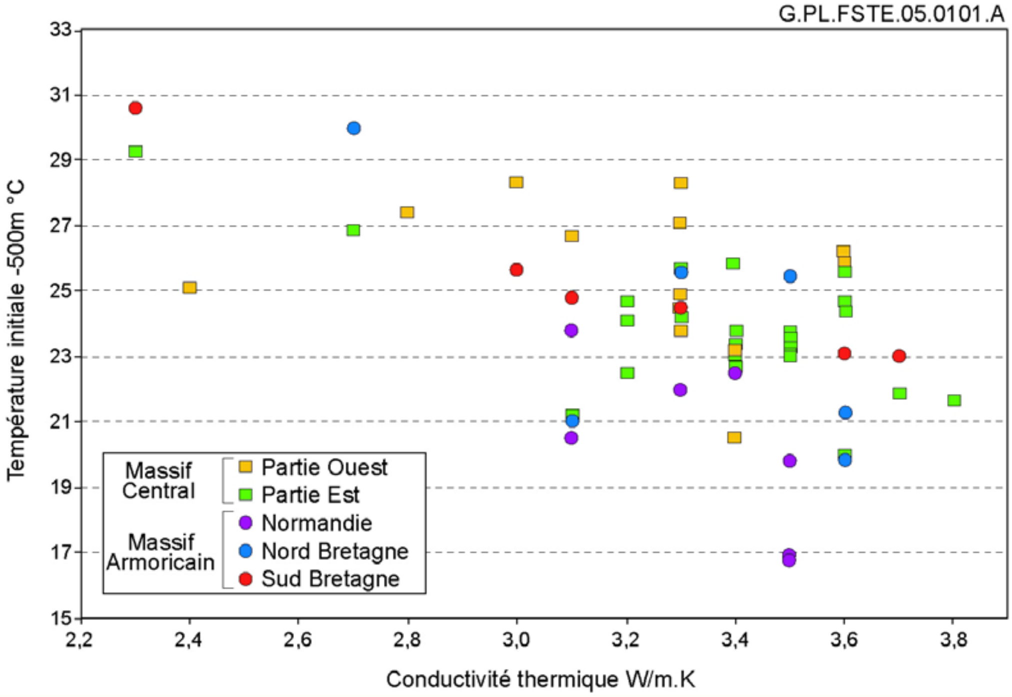


Figure 2.2-7 Estimated temperature of French granite massifs at a depth of 500 m in various regions

- major fracturing of granites is an important element in the architectural design of a repository. At a particular site, the organisation of a repository depends on the geometry of major fracturing. Depending on the tectonic history of the massif, it is set out more or less evenly and the separation of the massif is more or less marked. The analysis of a large number of French granites shows however that if major fracturing layouts vary from one massif to another, the distribution of the granite "blocks" where a repository could be located complies with rules which are relatively common to the French massifs studied;
- minor fracturing of a granite also plays an important role in the design of a repository. The rock's capacity to delay and reduce radionuclide migration will largely depend on the characteristics of the minor fractures (see chapter 6). The hydraulic conductivity of minor fractures is generally low or very low (less than 10^{-9} m/s). In this area of low permeability, values can however vary considerably from one type of granite to another and according to the types of fracture. They depend on their geometry, orientation and possible natural clogging by minerals (see Figure 2.2-8). The same applies to the properties of radionuclide retention by the fractures.

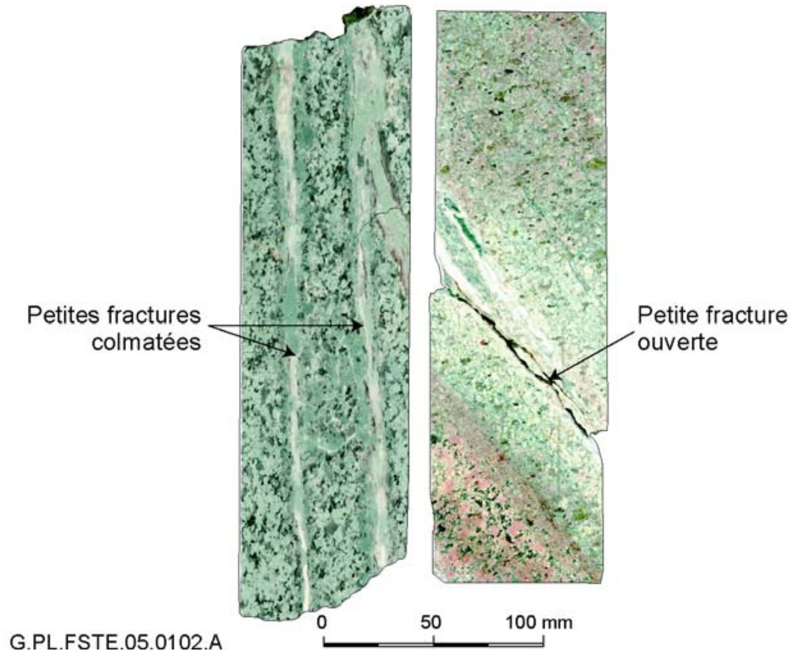


Figure 2.2-8 Clogged fractures and open fracture viewed on core samples

- the morpho-structural context of a granite has also been analysed. The topography and morphology of a site determine the hydraulic gradients, the prime movers of deep water flows. The differences between French massifs are significant (see Figure 2.2-9) Typological analysis has revealed three main morpho-structural configurations for granite, taken into account in the safety analyses:« granite massifs in a topographic depression in relation to the surrounding geological formations, dome massifs and sloping massifs. Each one of these configurations can correspond to various types of topography.

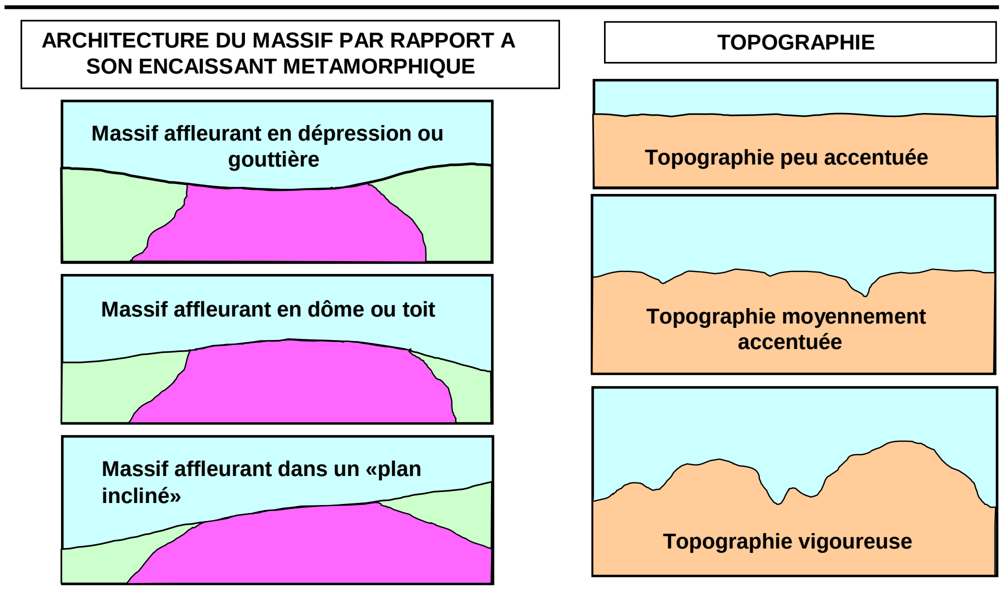


Figure 2.2-9 Granite massif morphologies and topography

- from the hydrogeochemical point of view, the inventory and analysis of the chemical compositions of granite waters available in France show that various types of water may be encountered here (see Table 2.1-1): so-called alkaline waters and carbonated waters. Carbonated waters are present in the Massif Central and can be linked to the geodynamic context and the more or less ancient volcanic activity. Alkaline waters correspond to compositions close to equilibrium with the granite medium. These differences in composition are unlikely to affect the design options proposed in terms of their principles. However, for certain granite massifs, they could lead to the formulations of the engineered barriers being adjusted to the chemical composition of the water.

Water type		pH	Na	K	Ca	Mg	Li	SiO_2	Cl	SO_4	HCO_3
Alkaline	Site 1	8,80	4 400	142	128	1	78	1 790	227	1 010	1 800
	Site 2	8,86	5 430	116	850	3,3	55	850	4 200	1 350	2 750
Carbonated	Site 3	6,64	95 500	4 360	1 570	1 940	1 420	1 700	105 100	6 500	29 500
	Site 4	6,80	41 700	635	3 550	3 500	590	770	1 200	220	57 100
	Site 5	6,67	39 100	2 420	15 100	16 100	770	2 100	56 600	2 600	40 200

Table 2.2-1 Examples of chemical compositions of water in various French contexts (contents in mg/l)

• *From the point of view of the long-term geological evolution of a site,* typological analysis of the granite massifs studied confirms that a large part of them are located away from active geodynamic zones, in other words, they are sheltered from significant long-term changes to their geological layout, and especially their deep fracturing. Climatic changes and erosion can also modify the hydrogeological and topographic context of a site over the long term. Variations exist between massifs due notably to the differences in morpho-structural context. The analysis has thus identified the main layouts encountered in the French context and the phenomena that could occur on a 10 000, 100 000 and 1 000 000-year timescale. It is observed that on a 100 000-year timescale, the models do not identify any significant differences in evolution between the massifs. Over 100 000 years, the situation of each massif would need to be specifically considered in the scope of a particular site study.

2.3 General structure of the repository architecture

The description of the repository architectures, and their justification in view of the expected safety functions, is covered in chapter 3. Here we shall simply introduce some elements of general description of the architectures and vocabulary which will be useful later. The reader will refer to [viii] for a more comprehensive description of the architecture.

A repository installation would consist of cells (underground caverns), excavated in the granite massif, containing *disposal packages* (see Figure 2.3-1). These packages consist of primary packages, as conditioned by waste producers, supplemented by overpacking according to repository requirements.

The architecture studied contains disposal cells for various categories of waste within specific repository zones. The repository zones for B waste, C waste and, if applicable, spent fuel are therefore physically distinct from each other. The repository zones are separated into repository modules, which reduces the quantity of waste that would be affected in a repository failure situation.

For cell construction, waste emplacement and reversible management of the installations, access is gained via *shafts* or a *ramp* between the surface and the repository level, and then via *connecting drifts* between these engineered access structures and the repository modules.

During the operating phase, *surface facilities* take delivery of the waste, prepare the disposal packages and provide support to the operation of the underground facilities.

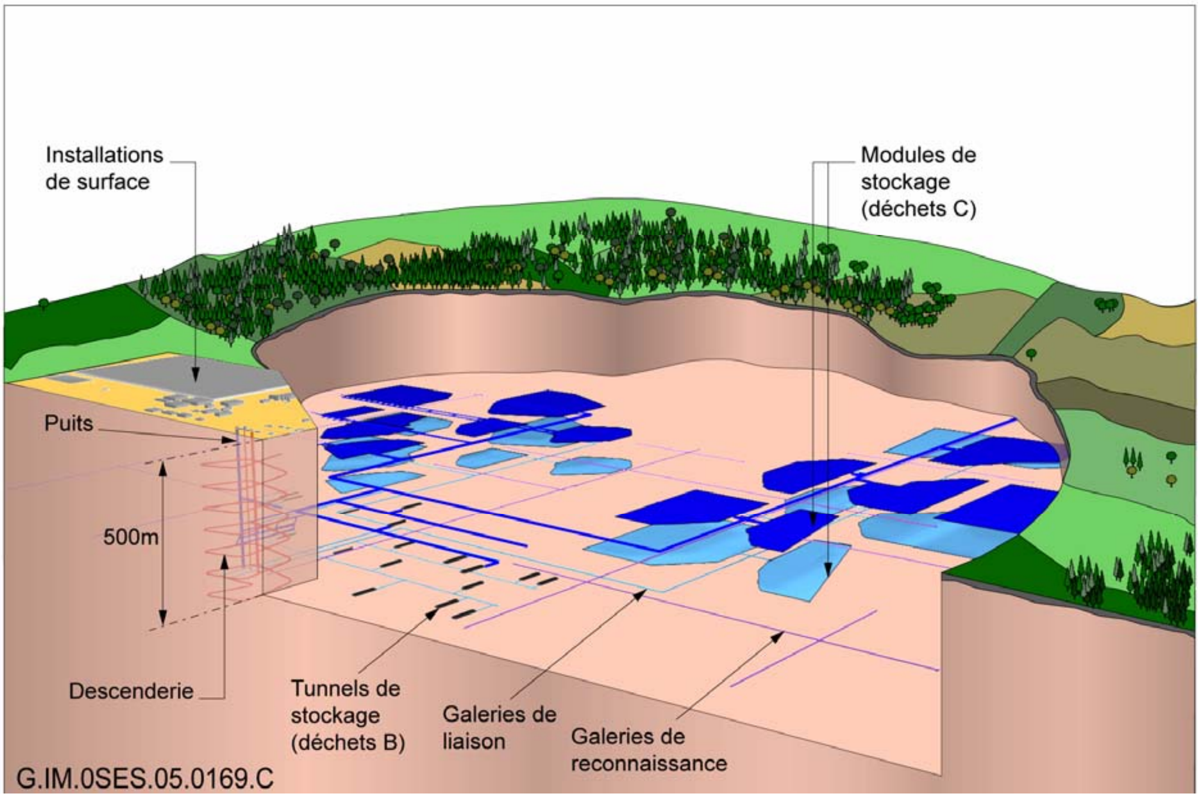


Figure 2.3-1 *Lay-out of a repository in a granite medium: surface facilities and two-level underground facilities*

3 Repository Safety and Design Functions

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3.2 Long-term repository safety functions 61
3.3 Design measures adopted to fulfil the functions..67

The purpose of this chapter is to set out the guidelines used in defining the repository architectures proposed for granite, in a generic framework. The main functions expected of the repository in the post-closure phase, which guide the definition of the architectures, are set out first of all. Then the architectures proposed are broken down according to this functions logic. As a result of the measures anticipated in the design, both of a material nature (provision of engineered structures) and an immaterial nature (general repository organisation, separation, etc.), the favourable properties of granite can be used to best advantage to protect the environment from harmful effects due to the radioactive waste. The engineered structures may also complement or provide a degree of redundancy to the functions fulfilled by the medium. They may also aim to minimise the disturbance caused by the repository to the rock, which Basic Safety Rule III.2.f sets out as an objective in itself.

3.1 General context

The control of repository safety in design through the allocation of safety functions is implemented by Andra, and some of its foreign counterparts, as a complementary method to the so-called "multi-barrier" approach. This approach, implemented in the scope of nuclear reactor safety, consists of building several confinement systems, mutually independent insofar as is possible, between the radioactive material and the environment. It has seen a number of developments that have enhanced and improved it without calling its fundamentals into question. They have led in particular to the development of the notion of defence-in-depth, which adds the concept of "lines of defence" to that of "barriers", supplementing the physical confinement systems with a range of material and organisational measures to safeguard against, or reduce or manage the consequences of an accident.

The notion of multiple safety functions constitutes a generalisation of the notion of multiple barriers. It consists of fulfilling safety objectives by implementing actions of various kinds, all of which contribute to the safety of the repository. These actions are accomplished by the repository components, the operators or the organisational arrangements put into place. The functions may be redundant, in other words have the same effect and be capable of substituting each other, but they are mostly complementary, which means that they contribute jointly to the achievement of safety objectives: the loss of a function then leads to the safety level being downgraded, but this loss may be acceptable due to the fact that the other functions are maintained.

This approach is now recommended on an international level [iv]. In practice, we begin by identifying which function the repository must accomplish with regard to its environment in order to be safe. This function is deduced from the objective set for the repository by RFS III.2.f. [i] and consists of "protecting people and the environment against harmful effects linked to the radioactive waste" (see chapter 1). This primary-level function can then be broken down into safety functions, in other words into actions accomplished by a part of the repository system and contributing to this principal function. Each safety function can itself be broken down into sub-functions, and so on until a level of detail is reached that the designer considers adequate with regard to his needs in order to characterise and specify the repository components. These needs depend themselves on the state of progress of the project.

3.2 Long-term repository safety functions

The primary function of a repository in the operating phase is linked to an industrial objective: taking charge of the packages. In this context, safety is absolutely vital but is taken on board in a conventional way, as for all nuclear facilities, through the analysis of the risks caused by industrial process (excavation, package emplacement and closure). It is a major design factor for the choice of items of equipment, their technical characteristics and their degree of redundancy but it does not condition the entire design process. It is in fact the very aim of the repository in the post-monitoring phase to ensure the protection of people and the environment.

It therefore appears important in the first place, at the point where the safety functions that the repository must fulfil in all phases are approached, to begin by a succinct description of the main functions expected in the post-closure phase.

The basic safety function is the protection of people and the environment from harmful effects linked to the radioactive waste. Harmful effects come in various guises. Firstly those linked to the radioactive nature of the waste: risk of external exposure to radiation and risk of ingestion or inhalation of contaminants. Others are more conventional and refer to the chemical toxicity of the waste: soluble substances or gases as applicable. Other risks may also be considered: some waste such as drums of bitumen-embedded materials release potentially explosive gases by radiolysis. It is, however, radioactivity which constitutes the particular feature of this waste, and the repository concept is therefore defined around this characteristic. This concept is above all designed to protect people and the environment from radioactivity. Other aspects are not neglected however, but it is possible to check with hindsight that the arrangements made with regard to the radioactive risk also cover the chemical risk, for example, or to amend the repository concepts in order to adapt them to specific risks.

The concentrate and retain strategy, as defined in Publication n°81 of the International Commission on Radiological Protection [xi], drives the design of the repository. This involves grouping the waste at a single site and preventing the harmful elements contained in the waste from reaching the population on a sustained basis.

The repository system is made up of all of the components which contribute to this concentrate and retain strategy: the host formation and the engineered structures added by humans to the repository. Insofar as it pre-dates the design of the repository, the waste as such is not, in the strictest sense, part of the repository system. However, the matrix of some waste helps to contain the radioactivity, generally because it was chosen for this purpose by the waste producer, either with a view to long-term management (the vitrified waste matrix, for example) or due to operational safety issues. By extension therefore, we include waste matrices within the repository system.

3.2.1 Isolating waste from surface phenomena and human intrusion

It will be noted first of all that deep disposal keeps waste away from erosion phenomena and ordinary human activities which only affect a superficial thickness of terrain on the hundred thousand year timescale. The selection criteria which would be defined if the approach involved choosing a site would restrict the choice to zones located away from volcanic influences or any other far-reaching phenomenon liable to affect the terrain in depth.

This function of protecting waste against erosion, other natural phenomena and human actions is one of the reasons behind choosing a deep repository.

In this context, the waste can only come back up to the surface by a deliberate human action, in the form of a borehole. This risk would be limited, in a site selection context, by the restrictions imposed by RFS III.2.f: the absence of exceptional resources in the repository environment discourages underground prospecting with a view to working an ore deposit or a water source.

It will be noted that, in the scope of a defence-in-depth approach, and in order to take account of all risks of an intrusive borehole intercepting the repository zone, even if such boreholes are not motivated by the search for a natural resource, this type of event is nevertheless considered in the design. It is also taken into account in the safety analysis (see chapter 5).

3.2.2 Preserving records of the repository

Beyond the repository operating and monitoring phase, the functions are passive which means that they require no human intervention. Basic safety rule III-2f states that human protection must be guaranteed "without depending on an institutional control which cannot definitely be relied on beyond a limited period (...) (500 years)". This falls in line with the aim of preserving records of the site for as long as possible. "Preserving records of the repository" is a safety function in itself; its only particularity is that possible loss must not impinge on the safety of the repository.

Repository records help initially to limit the risk of losing technical control over all or part of the technologies used for the disposal of radioactive waste. Subsequently, once the waste packages have been emplaced, the records constitute an element of defence-in-depth making it possible to understand phenomena possibly observed from the surface – if, for example, measurements on the state of the subsurface are carried out – and avoid the risks of intrusion within the repository or at least ensure that this is done with full knowledge of the situation.

As soon as the disposal facility is created, as early as the initial research period, records of the facility are gradually built up by preserving knowledge and data linked to the design of the facility, the justification of this design and the associated safety demonstration, the building and operation of the facility, as well as the control of the waste packages from their creation to their final emplacement within the repository and, last but not least, the gradual closure of the repository structures up to the final closure of the disposal facility. The minimum timescale for all of these phases will be around 100 years or even much longer depending on the reversibility stages of the repository.

Beyond this first major phase, of the order of a century, the preservation of the records can be illustrated by using feedback from the La Manche waste disposal facility, where the last packages were emplaced in the repository in 1994, as a basis. This example does not prejudge the arrangements that will be implemented for deep disposal. The preservation of the records for the La Manche waste disposal facility relies on four methods: the production of a "detailed record" (see Figure 3.2-1) and a "summary record" of the disposal facility, selection of documents with various levels of detail adapted to various uses, encumbrances deposited at the land registry, and permanent information systems, at least on a local level.

Figure 3.2-1 "Detailed record" produced on archival paper kept in the contemporary archives section of the French National Archives Centre

3.2.3 Main safety functions aimed at protecting people and the environment from radionuclide release

The functions aimed at isolating the surface environment from damaging effects linked to waste are designed to control the transfer channels which, in the long term, can conduct harmful elements towards humans and the environment. In principle, these elements consist of radionuclides and toxic chemicals, but at this preliminary design stage, it is considered that the arrangements made to safeguard against harmful effects relating to radionuclides cover de facto the issue of other toxic materials.

Potentially, there are numerous channels through which this can occur:

- the aqueous channel, as the elements are capable of being transformed into a solution (including in a colloidal form) and working up to the surface;
- the gaseous channel, as certain radionuclides can exist in this form;
- the solid channel only in cases where a particular event would force a part of the waste up to the surface (this scenario was discussed earlier).

As far as the gaseous channel is concerned in the long term, the majority of gaseous radionuclides are short-lived in comparison with transfer time in the formation (krypton 85, tritium, etc.). Of the long-lived elements, only iodine-129, chlorine-36 and carbon-14 are capable of significant migration. The first two have properties such that, under the conditions prevailing in the repository, they will be rapidly placed in solution. Carbon-14 could maintain a gaseous phase longer.

At this stage, it can however be stated that in view of the small radiological inventory concerned and the low probability of the gases finally being dissolved within the repository, the gaseous channel does not appear to be a significant channel for radionuclide transfer. It does not require any special design measures. This does not preclude other kinds of arrangements aimed at controlling the inactive gases which could build up in the repository (corrosion gases, radiolysis gases, etc.).

The design is therefore oriented towards the control of radionuclide transfer by water. This can occur by convection, in other words by a phenomenon of radionuclide entrainment accompanying a water flow, or by diffusion, in other words by the Brownian motion of the elements in water.

The diffusion phenomenon necessarily occurs when the disposal packages no longer provide sufficient isolation. On the other hand, it is possible to limit convection, or water circulation, considerably within the repository. This is a favourable element firstly with regard to the protection of waste packages in the initial phase following repository closure. Limiting the water circulation ensures that the kinetics of the degradation process are slow and controlled better. Limiting advection is also favourable in the longer term, with regard to radionuclide transfer as this is generally much slower by diffusion than by advection

The first safety function is therefore “preventing water circulation” (see Figure 3.2-2).

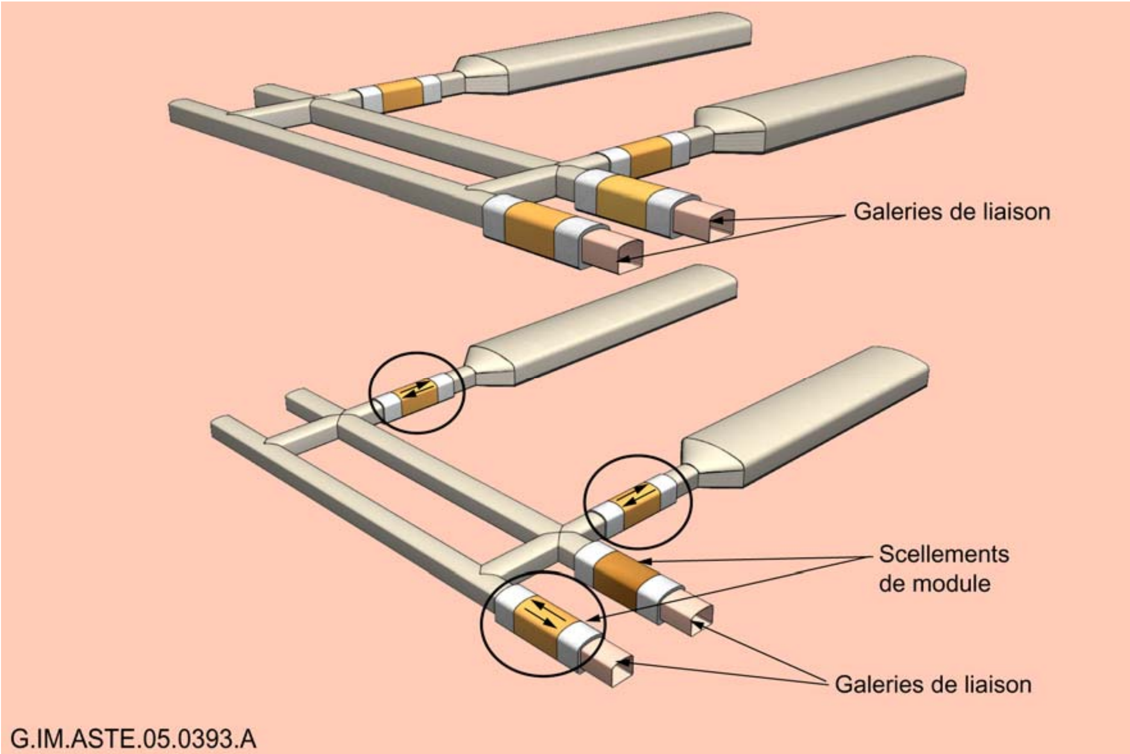

Figure 3.2-2 Preventing water circulation

The next step is to immobilise the radionuclides and toxic chemicals, in other words prevent them from dissolving, and where this occurs, encourage their precipitation as well as the barely mobile chemical forms of the solutes. This function is generally effective in the near field, close up to the packages, where the physical and chemical conditions prevail for this to be accomplished. This objective is compatible with more general practices in terms of safety, risk management "as close to the source as possible". This general function is maintained throughout the lifetime of the repository, but takes on different forms according to timescales (see next section). This function, at least during the initial phases, uses the favourable properties of the waste packages. It is notably based on the need to protect the waste packages and to place them under favourable physical and chemical conditions.

The second safety function is therefore "restricting the release of radionuclides and immobilising them in the repository" (see Figure 3.2-3).

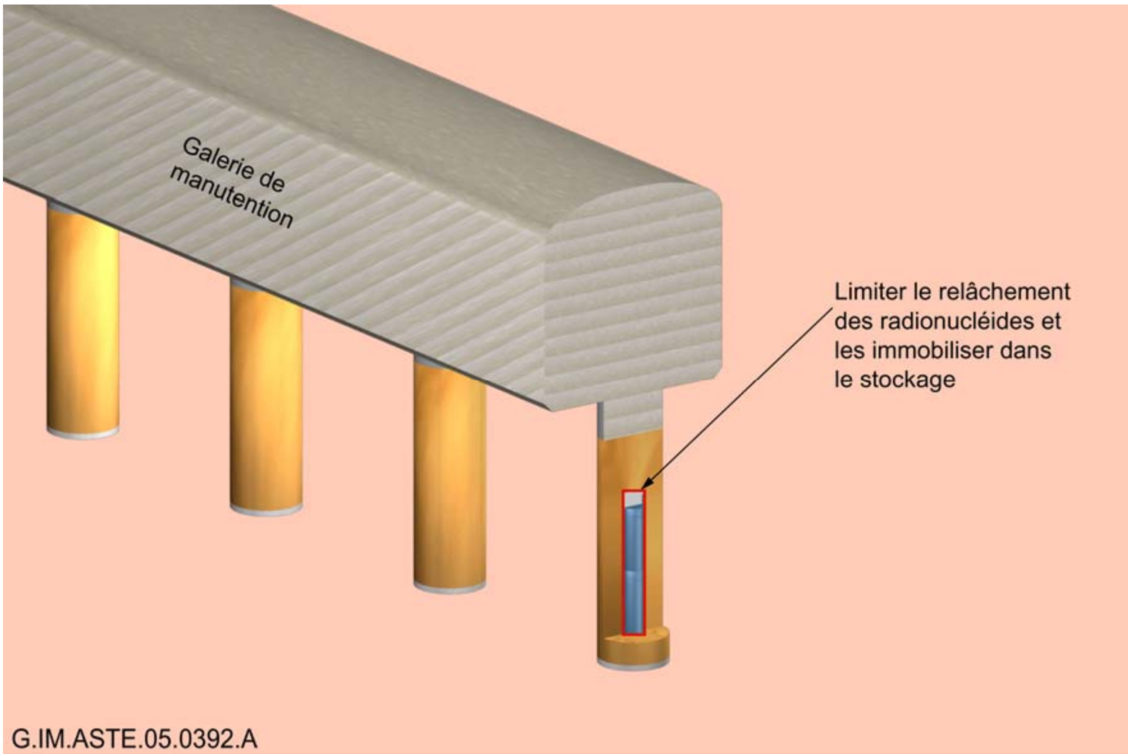

Figure 3.2-3 Restricting the release of radionuclides and immobilising them in the repository

Once the radionuclides are in solution, the aim is to delay and restrict their migration. By delaying and therefore increasing the time required for transfer to the surface, the impact is reduced due to radioactive decay. Restricting the flow of radionuclides at all points of the system equates to spreading the flow both in terms of time and space. For a given average transfer time and at constant quantity, a radionuclide flow will be all the less harmful if it is extended over a longer distance and its release into the biosphere is spread over a longer period. This function only comes into play when the "immobilising the radioactive elements" function is completely effective (as migration has still to occur) but begins to act from the very first releases. It is also available in the case of premature loss of containment as a result of an incident. For this reason, it is said to be "latent" in the initial phases of the repository (refer to this notion in section 3.3).

The third safety function is therefore "delaying and reducing the migration of radionuclides" (see Figure 3.2-4).

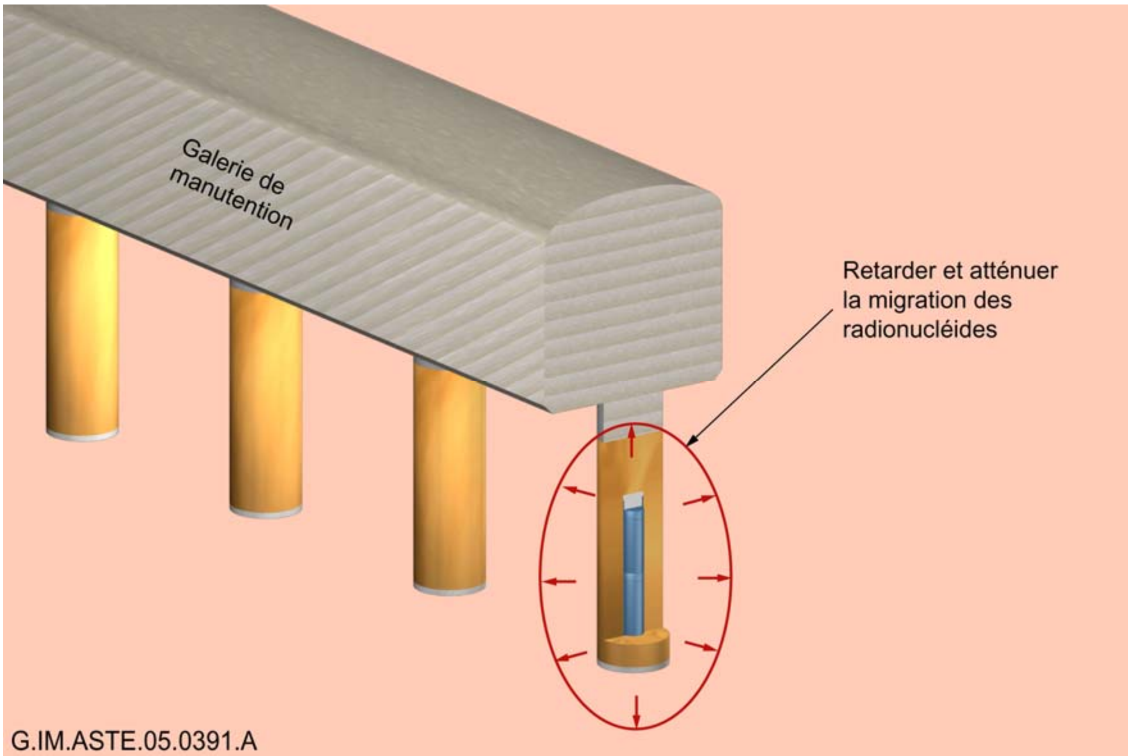

Figure 3.2-4 Delaying and reducing the migration of radionuclides

These three functions are limited to the repository system, consisting of waste packages, engineered structures (engineered barriers, seals and plugs, access structures, etc.) and the host granite formation. No function is allocated to the surface environment which is totally undetermined in a generic context. Furthermore, it may be subjected to significant evolutions over time (particularly surface erosion) making it difficult to allocate sustainable functions.

3.3 Design measures adopted to fulfil the functions

The scope of this section is to present the result of the internal functional analysis, consisting of allocating functions to the various components of the repository in order to achieve the three safety functions referred to above. It also describes the architectures adopted in a logic linked to the functions expected. A more linear, detailed description of the architectures is available in the volume focusing specifically on this subject [viii].

3.3.1 General methodology of the functional analysis

The breaking down of functions into sub-functions is not unique in principle. It reflects the designer's choice. This choice is based firstly on knowledge of the French granite context which enables characteristics to be derived which are suitable for it to be allocated functions in a generic manner.

Functional choices are also based on the state of knowledge acquired on the behaviour of the engineered components of the repository, which breeds confidence in the capacity of certain components to accomplish the functions. Also for this point, knowledge of the granite medium is not indifferent, under at least two aspects. Firstly, the performance of the functions allocated to the engineered structures may depend on their interaction with the granite medium (interactions linked to geochemistry for example). Secondly, and more specifically to granite, the arrangement and nature of the fracturing of the granite places a constraint on the repository architectures and the functions allocated to them. For example, the performance of the seals placed in the access routes to the repository shall be in relation with the need to guarantee transfer by diffusion and thereby delay radionuclide access to the fracture network. The dimensions of the cells and their arrangement in space are also restricted by the geometry of the fracturing to ensure that they are located in granite rock with very few fractures and therefore with very low permeability.

As a minimum, each function is characterised by:

- a level of performance, in other words a quantification of the level of efficiency of the expected action. It is not necessarily relevant to set a level of performance in principle. This is only meaningful if it is used to dimension the components which must accomplish the function. If the function must be fulfilled by at least one component which escapes the designer's action (the geological medium, for instance) or if the link between dimensioning and performance depends on the operation of the entire system (for example, the efficiency of the function limiting the water flow by the permeability of the seals), it is hardly useful to set a level of performance in principle;
- a period during which the function has to be available;
- one or more components that must fulfil the function, and the physical phenomenon or phenomena enabling these components to fulfil the function (for example, to maintain the leaktightness of a container, its corrosion resistance is invoked). By principle, only the host formation, waste packages and engineered structures added by people (seals, containers, backfill, etc.) are adopted as components. The other elements present in the repository due to operating conditions or its natural evolution (functional gaps within the disposal cells, etc.) cannot fulfil a function as their action is less predictable.

Depending on the case, a function may:

- be available, albeit in a downgraded form, beyond the period taken into account by the designer. We then talk about a "reserve function", the duration of this reserve not always being quantifiable. However, identifying reserves breeds confidence in the fact that the system has a greater level of safety than what is strictly planned and quantified by the designer;
- be available with a performance level better than that taken into account by the designer. We refer in this case to a "margin" on the performance, which means that the designer does not use all of the performances that could be expected to best advantage. The existence of margins also boosts confidence. The existence of a phenomenon favourable to safety but not counted as a function may be considered both as a reserve or as a margin, depending on how it is viewed;
- finally a function may be latent, in other words it does not act due to the existence of another function. For example, the containment provided by a waste matrix is latent as long as it is not subjected to the action of water, in other words as long as the container manages to isolate the matrix from water. The accidental loss of functions can therefore be managed due to the existence of latent functions (for example, in this case, a leak on the container). Latent functions are a particular form of redundancy.

This is illustrated by Figure 3.3-1.

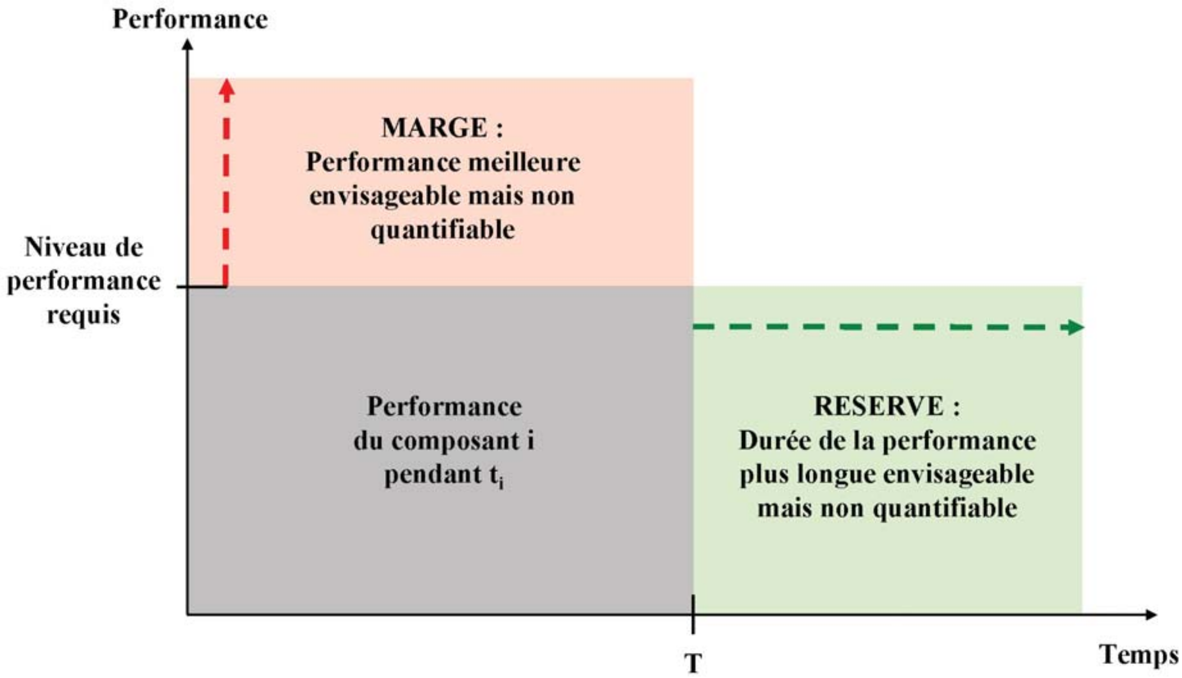

Figure 3.3-1 Illustration of margins and reserve functions

When the functions are presented, the design measures and physical and chemical phenomena enabling the safety functions to be accomplished are identified. In this case, it is not a question of systematically surveying the phenomenology of the repository, which is the scope of the PARS [xii], but of insisting on the most important phenomena, identified to guide the design work. These phenomena may be favourable (in which case the safety functions have to use them to best advantage) or unfavourable (in which case the functions must prepare to counter their effects). Checking that the system, once designed, is robust in the face of a broader set of disturbances and particular phenomena, without necessarily prejudging whether they are favourable or not, falls within the scope of later safety analyses (qualitative safety analysis and assessment of the scenarios).

The rest of this chapter is devoted to presenting the status of the design. In the absence of a precise site, functions can only be allocated to the geological medium on the basis of the knowledge of French granites as presented in the previous chapter. This allocation is therefore subject to the uncertainty inherent in the absence of a well-defined context. The functions proposed may be taken care of by the granites due to their actual nature, but the expected level of performance, or the allocation of possible additional functions, would depend on more accurate studies on a specific site. The role of the engineered structures as complements to the geological medium is particularly strengthened by this; this role could be reviewed in the scope of an actual site.

It is worth noting however that even in cases where a site is specifically studied, uncertainties would remain in the characterisation of the granite on a macroscopic scale, although evidently less than in a generic context. This is inherent in the characterisation of a site and a geological formation hosting a deep repository, irrespective of the formation concerned.

The absence of a site does however have an influence on the definition of the concept, insofar as design measures tending, in certain cases, to minimise the requirements relating to the geological medium by transferring them to engineered structures, have been studied. As a result, concepts adaptable to various contexts can be proposed within the limits of the variability presented by French granites. Such an approach should be reviewed in the scope of a study on a given site, in order to adjust the relative role of the geological medium and exogenous components.

Furthermore, in a generic context, it is neither useful nor relevant to proceed with accurate dimensioning of the repository components. The elements presented in this chapter, relating to the pre-dimensioning of the engineered structures or to the performance expected of the functions, are orders of magnitude designed to ensure that no measures are proposed which are impossible to be implement in practice. It is a starting point which could be reviewed and finalised if the work were to be pursued.

The functional analysis of disposal in a granite formation was essentially developed on the basis of the analysis already defined in the scope of feasibility studies on disposal in a clay formation. Owing to the generic character of this dossier, the functional breakdown was limited to a simpler level of detail than that of clay.

3.3.2 The "preventing water circulation" function

Excavation of the repository in granite leads to the drainage of water from the conducting fractures intersected by the structures and very localised desaturation of the rock in the vicinity of the structures. Once the various parts of the underground facilities have been closed, they become resaturated. In a granite context, the majority of the structures will be resaturated briefly compared with the periods considered in the analysis (resaturation periods of a century or less). A part of the cells located in the low-permeability granite rock may take longer to reach hydraulic equilibrium, over periods of the order of a thousand years. Beyond this, the entire repository will be resaturated; water circulations within the repository should then be avoided as much as possible, as they form transport vectors for chemical disturbances capable of harming the repository components and, in time, radionuclide transfer channels should the packages begin to leak.

In the granite context, water circulation can essentially occur within the fracture network. It is highly dependent on the hydraulic and transfer properties of these fractures and their connectivity. The layout of the various parts of the repository must therefore be adapted to the fracturing present, according to their specific characteristics. In the first instance, this implies knowing enough about the pattern and nature of the fractures within the massif to propose repository architectures adapted to a particular context, and then gradually defining the technical criteria to judge the ability of the rock, as it is characterised, to host disposal facilities. These criteria are necessarily the result of an iterative process between exploratory work on the fracturing and the safety analyses; they cannot be defined *ex ante*, out of context. They lead to the definition of various fracturing scales – major, medium or minor – in relation to which the various parts of the repository are positioned (access structures, modules, cells, etc. – see following sections).

Exploratory work on fracturing is therefore part of the wider issue of exploring the formation (which also includes thermal, geochemical and other properties), which is covered in the "Phenomenological evolution of a geological repository" [x]. This comprises several stages:

- surface exploration to develop the models required for an initial assessment of the site's suitability for the installation of a repository. The absence of insurmountable site characteristics is verified on the basis of design studies and safety analyses. A further aim of this stage is to specify the layout and research programme of an underground laboratory for in situ characterisation of the granite at depth;
- qualification through underground structures with the aim of assessing the suitability of the site for a repository. The work consists of specifying the geological, hydrogeological and geomechanical models of the granite site and models of radionuclide transfer and retention in the fractures. Site qualification is based on the design studies and safety analyses. This results in particular in the installation criteria for repository structures in the granite rock.
- In the course of disposal, the exact adjustment of the repository architecture to the site's characteristics, and especially the fracturing of the granite, requires exploratory and characterisation work to be carried out "on advancement". This work ensures that the layout of the repository structures match the criteria defined in the previous stage. This characterisation on advancement, which is aimed particularly at minor and medium fracturing, difficult to detect at distance, will remain an ongoing activity throughout the operation phase; indeed it is inconceivable to explore and characterise the entire volume of rock due to receive the waste before the start of operation. Furthermore, the means of investigation do not enable minor and medium

fracturing to be detected with certainty at distance, and therefore the blocks of containing granite and their characteristics cannot be defined with sufficient accuracy.

Several successive phases can be distinguished:

- ✓ exploration and characterisation of the repository zones from specific and non-specific exploratory drifts and boreholes from these drifts; this must confirm and detail the location of zones where sufficient modules may be located, and then the envelope of each module, on the basis of criteria defined in the previous stages and taking account of the characteristics of the fracturing and the geological medium;
- ✓ exploration and characterisation enabling the installation of access drifts to the cells within a module; this is done at the repository level, during construction, from drifts and/or from soundings made from the drifts;
- ✓ finally exploration and characterisation designed to specify the location of the cells, and to retain or reject all the positions planned for the various types of modules (see Figure 3.3-2).

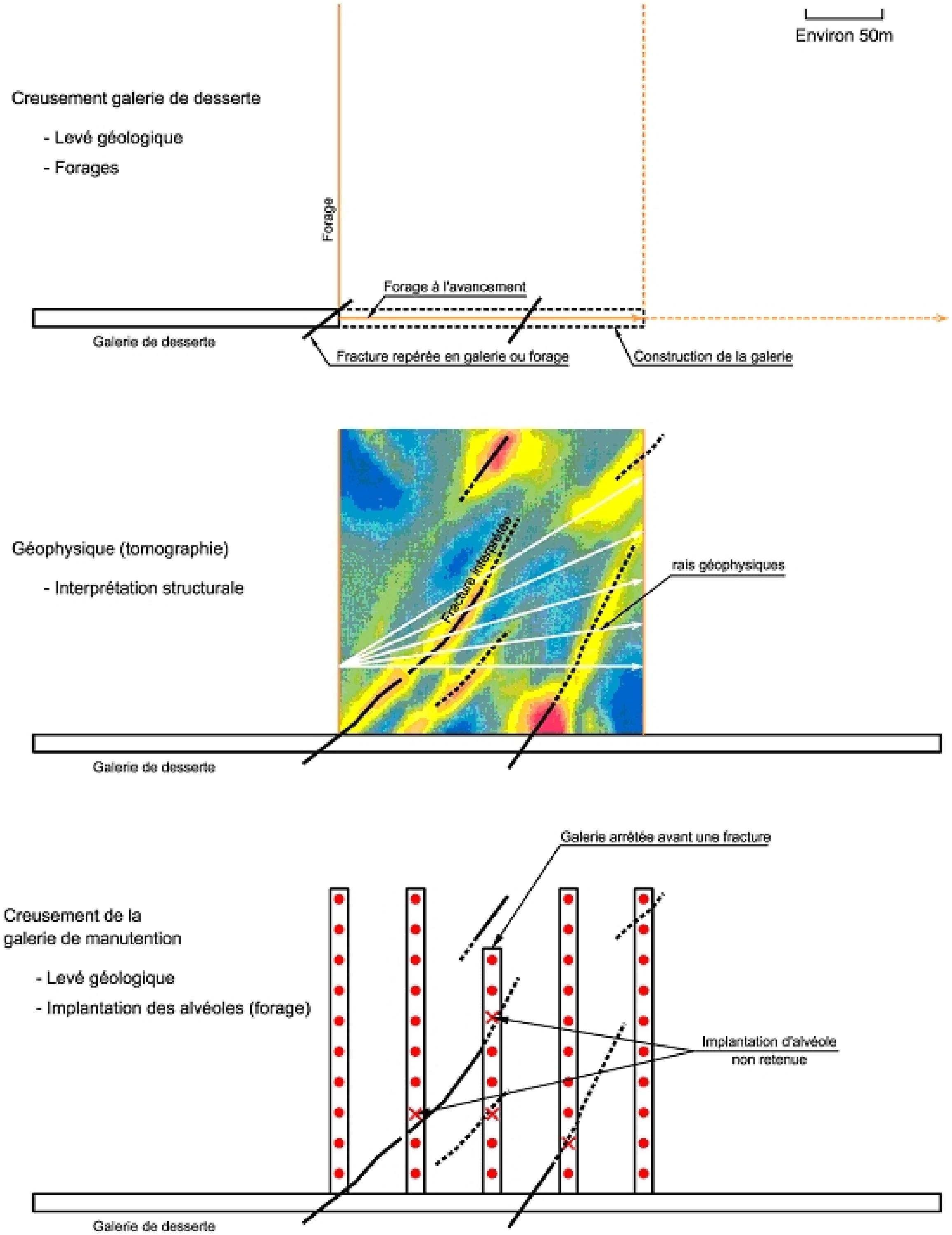

Figure 3.3-2 *Principle of "ongoing" exploration and characterisation in the course of repository construction (exploration on advancement)*

3.3.2.1 Limiting water flows between disposal modules and regional faults[10]

Regional faults are the most commonly used transfer channel on the scale of a granite massif and its environment.

Limiting the water flows which could originate from these faults therefore consists of safeguarding against possible hydraulic connections between these major faults and the fractures of lesser importance which will be intersected by the drifts of a repository. Connections are indeed possible either at depth or due to connections via the more permeable surface parts of the granite.

Several measures are planned to limit such connections. First of all, and in accordance with the recommendation of RFS III.2.f, the repository would be designed such that a buffer distance is maintained in relation to the regional faults. This design requirement does not seem to be unfeasible in the context of French granites; a distance of the order of 1.5 to 3 km appears reasonable.

Moreover, the joining of faults and fractures which could be intersected by the excavation of surface-bottom connecting structures (shafts and ramps) and access drifts to the disposal modules must also be limited. A second measure consists of backfilling structures with material of sufficiently low permeability to avoid creating preferential water circulation channels here. The performance level of the backfill must be in proportion to the overall permeability of the granite where these structures are located, in other words established according to the hydraulic conductivity of the fracturing intersected by the structures.

3.3.2.2 Limiting the water flow rate in the drifts of a disposal module

In the same way that a buffer distance is to be retained between the repository as a whole and the major regional fracturing, RFS III.2.f also recommends isolating the disposal modules from medium fracturing, liable to conduct water. The objective is to install these modules within blocks of granite free from such fracturing.

The large volume of rock generally available deep in a granite massif provides flexibility in terms of adapting the architecture of the modules to fracturing. Notably, it may lead to the design of general repository architecture on several levels. Such an arrangement reduces the footprint and facilitates "ongoing" exploration of the granite. This concerns not only the implementation of exploratory tools but also the interpretation of data and the geological modelling of the granite. In the majority of geological configurations for French granites, knowledge of a level can indeed be transposed to the next level, over a scale of around 100 metres. Therefore, the general repository architecture in the proposed design has two levels approximately 100 metres apart, and is capable of being adapted to the majority of configurations in the French geological context.

Furthermore, the type of fracturing excluded from disposal modules may not be the same, depending on the type of waste. Indeed, the footprint of the repository zones for the various categories of waste depends notably on the heat released by the waste. The footprint of a spent fuel repository is larger than that of C waste, which is in turn larger than that of B waste; it may therefore be necessary to adapt a spent fuel repository to a larger fracturing scale than for the other cells, for which stricter criteria could be defined.

A general design measure also consists of installing the access structures to the repository (shafts or ramps) according to the general hydrogeological context and hydraulic gradients, so that they cannot form drains from the repository to the surface.

Adapting the architecture of the modules to the fracturing is only totally effective if the internal drifts to the disposal modules are protected from the water liable to come from the conducting fractures "excluded" from the modules.

Two further measures are planned for this purpose in the design of a module: low-permeability backfill placed in all module drifts and very low-permeability seals at the most effective points of the drifts to limit water circulation in the modules.

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[10] Defined in chapter 2

The formulation of an adequate backfill can be based on the actual granite at the site, sufficiently crushed and ground to limit its expansion. Such backfill has been tested [xiii] ("Plug and backfill" experiment at the Äspö laboratory) and it seems possible to achieve permeability of the order of 10^{-10} m/s. The backfill is composed of ground granite and a proportion of swelling bentonite (approximately 15 to 20%) for this purpose. One of the technological difficulties would be to obtain sufficiently homogeneous emplacement to prevent the equivalent overall permeability from degrading, especially at the roof of the drifts. Techniques have been studied to favour this emplacement (see Figure 3.3-3).

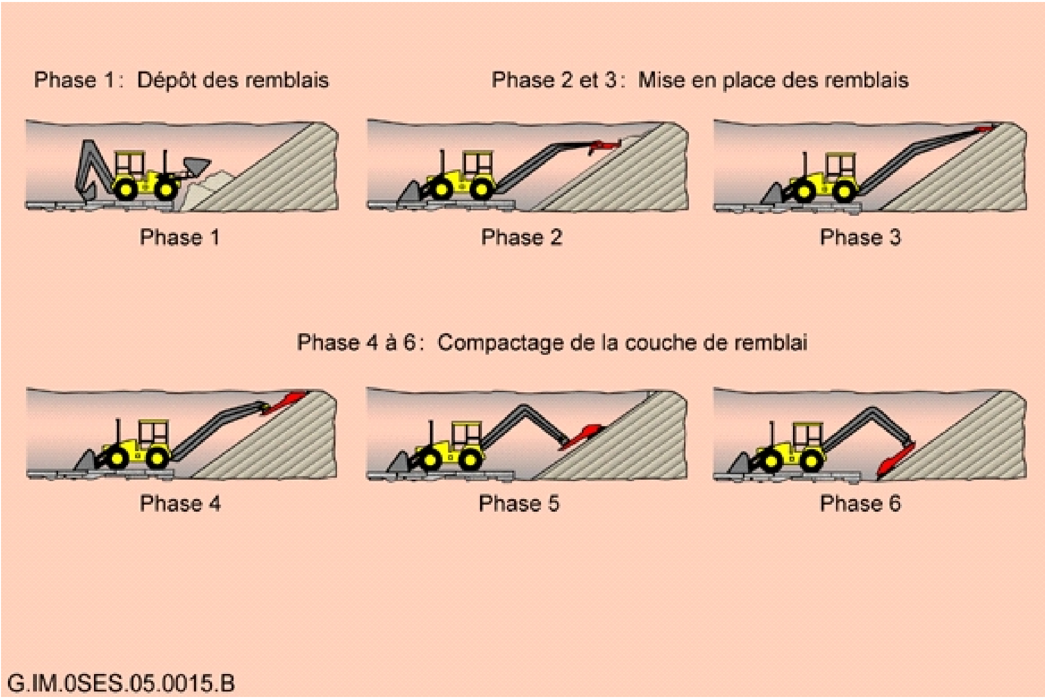

Figure 3.3-3 Principle of low-permeability backfill emplacement

Furthermore, to prevent the backfill from being short-circuited by the actual rock, it should be ensured that granite excavation will not have significantly damaged it on the wall. Any fractured or microfissured zones formed during the construction of the repository or by evolution of the granite subjected to the open air must have an overall permeability level in proportion to that of the backfill. In the granite context, due to the hardness and to the low initial permeability of the rock, wall damage to the engineered structures is essentially linked to the method of excavation. In cases where traditional blasting is used for excavation, rock damage to a depth of 20 to 30 cm is predictable. Overall permeability depends on the connectivity of the minor fractures created by excavation in the walls. This connectivity is generally mediocre between sections of drifts corresponding to two different "blastings". Permeability values of 10^{-9}m/s are generally considered realistic [x]. For excavations without blasting, using tunnel-boring or equivalent techniques, wall damage may be very slight (a few centimetres). The damaged zone does not therefore have any significant influence on water circulation in the drifts

Seals could also be used in addition to backfill; their number and arrangement being highly dependent on the nature of the site (see Figure 3.3-4). These seals could, on the one hand, isolate the drift backfill from the water liable to run from the medium fracturing intersected by the access structures and, on the other hand, interrupt a damaged zone if it was proved to be potentially damageable. The TSX test conducted in the Lac du Bonnet laboratory in Canada [xiv] has shown the ability of a swelling clay (bentonite) seal to form an intimate contact with the rock wall and recreate equivalent permeability measured at a value of 10^{-11} m/s. These seals would be emplaced in the form of bentonite "bricks", the mechanical stability of which would be ensured either by a concrete retaining plug or by an appropriate composition of the backfill in contact.

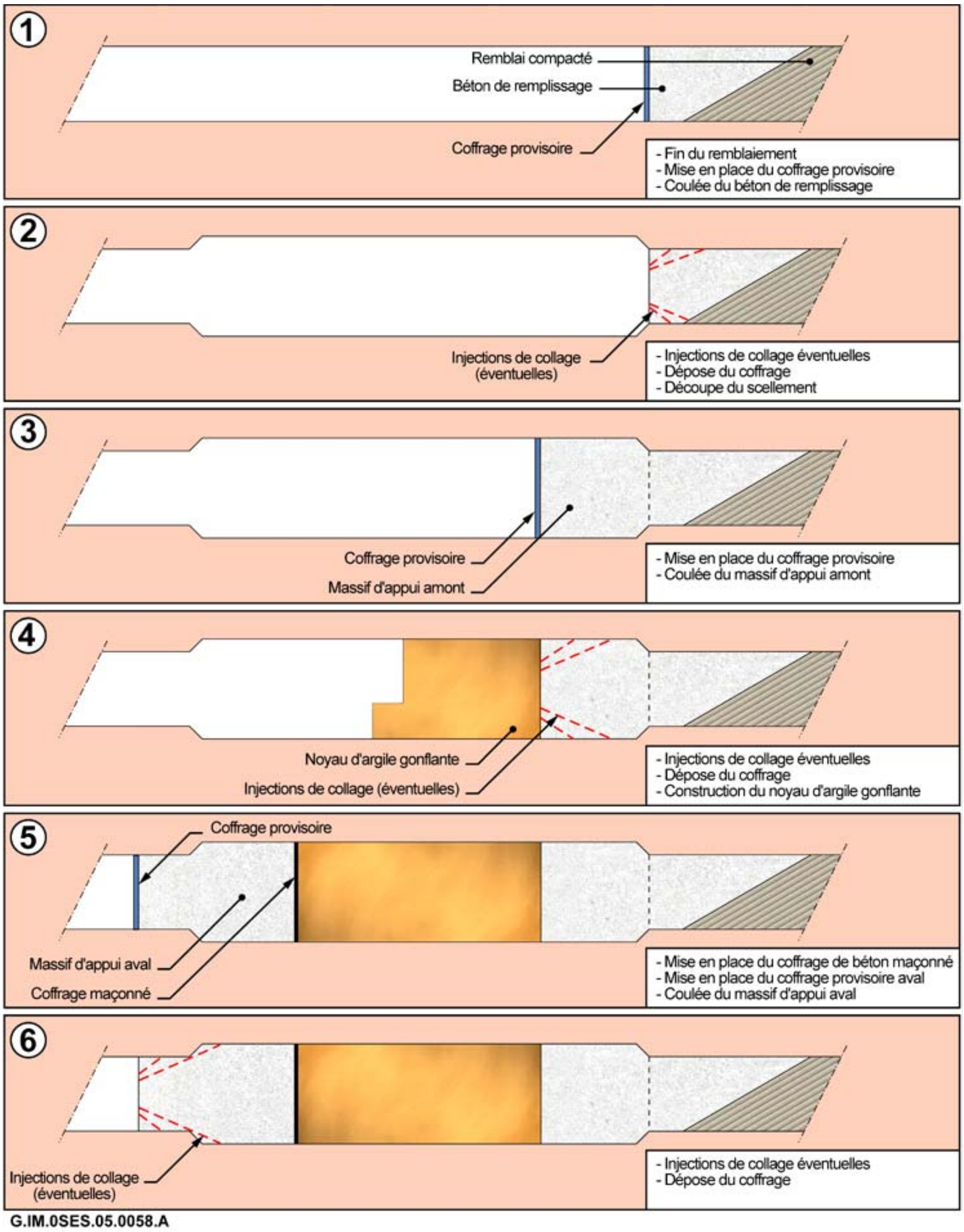

Figure 3.3-4 Principle of seal construction

3.3.2.3 Providing a diffusive transfer regime in the cells

The aim is to emplace the waste in the granite rock where it has no fractures or only minor fracturing conducting little or no water. As for the modules, cell layout is specified by a phase of characterisation "on advancement" during excavation of the repository. It should be mentioned that, on this scale, the dimensioning of the repository anticipates that a proportion of cell locations are liable to be rejected during this work "on advancement". For example, studies in Sweden anticipate a rejection rate of approximately 10% of the locations investigated for spent fuel disposal in a shaft.

In the case of B waste, the reduced number of modules (20 to 30 depending on the inventory model scenario considered) and their compactness (due to the heat released being largely non-penalising and to the mechanical strength of the rock) prove to be fairly unrestrictive with regard to the size of the blocks of granite to be identified, as several modules can be installed in a single block (typically between 1 and 4 according to the data from the typological analysis of French granites).At this stage of the project, Andra has selected horizontal cavern or "tunnel" concepts of moderate length (approximately 100 metres) in which the packages, conditioned in concrete overpacks (see below), would be stacked to a maximum of four to five high (see Figure 3.3-5). This aims in particular to safeguard against the risks of packages falling from great height (> 8 metres). In order to guarantee a predominantly diffusive regime within the B waste cells, these cells will be closed by a bentonite plug, of similar design to the drift seals.

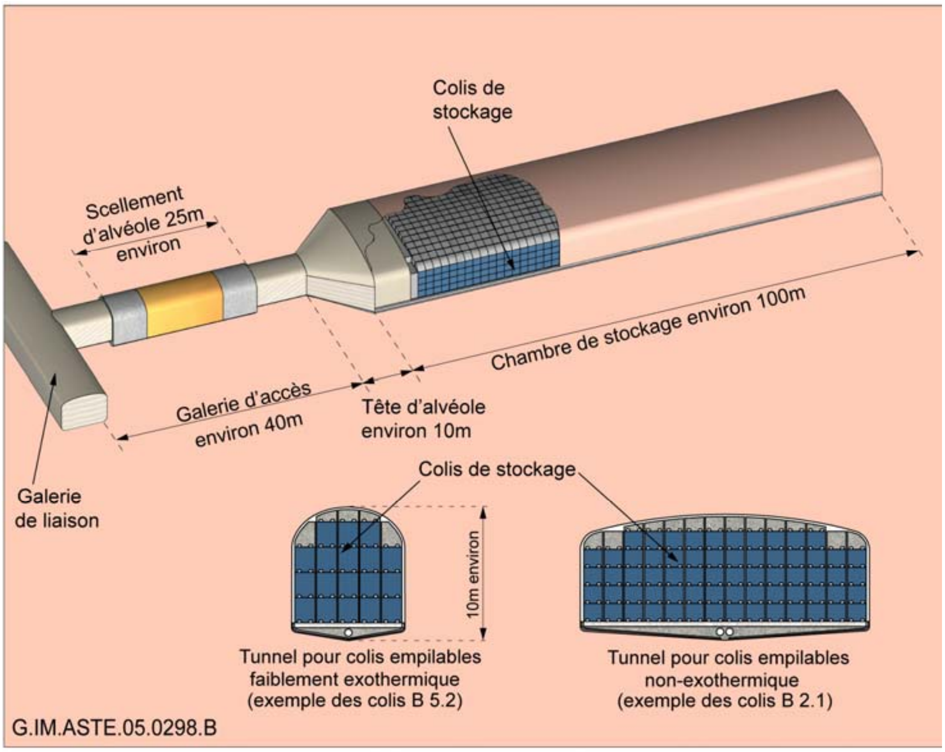

Figure 3.3-5 Disposal tunnel for stackable B waste packages

As far as heat-releasing waste (C waste and spent fuel) is concerned, the need to space out the packages to limit the increase in temperature leads to a larger footprint and therefore just one module per block is possible, each one containing several hundred small cells (see Figure 3.3-6). Fifteen to 40 modules are necessary depending on the inventory model scenario considered. On account of the dip of the fracturing in French granites, most often sub-vertical, preference is given to vertical concepts. The waste would therefore be emplaced in small vertical boreholes. For spent fuel, each cell contains one disposal package. For C0 waste, the least radioactive, each cell can contain up to five packages. For C1 to C4 waste, the reference is two packages per disposal borehole.

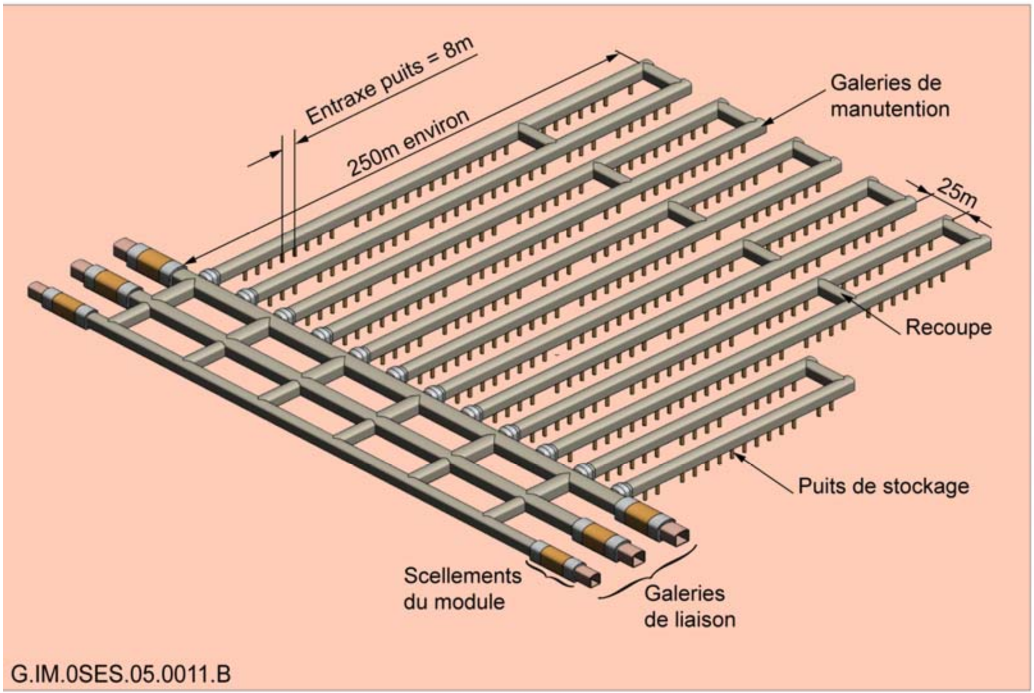

Figure 3.3-6 Representation of the reference architecture for a C waste type repository module

For this waste, it is necessary to isolate the disposal boreholes from the handling drifts within the module. It appears to be possible under these conditions to establish a diffusive regime within the cells, by means of a buffer surrounding the C waste and spent fuel packages. This could be formed by rings of bentonite. Apart from the isolating role with regard to the outside of the cell, essentially provided by the plug, the buffer provides isolation from all fractures, including the most minor, and ensures chemical homogeneity around the containers

3.3.3 Restricting the release of radionuclides and immobilising them in the repository

The “immobilising” function is a near-field function of the packages. It consists either of physically opposing the transfer of radionuclides or of converting them into a physical or chemical form which opposes their displacement. It thus depends particularly on the notion of containment, which is usual for safety functions in nuclear installations. In the case of a deep repository, containment by inserting a physical barrier cannot last indefinitely due to the degradation of exogenous materials within the repository. The usual notion of “containment” is therefore extended to a notion of “immobilisation”, which also takes account of the role of the waste matrices retaining the radionuclides, the capacity of the very low porosity medium to filter out the most voluminous molecules (such as colloids, for example) or more generally to reduce the solubility of the radionuclides. It will however be noted that the transfer delaying functions provided, for example, by the geological medium or by certain engineered components are not considered as “immobilisation”.

This acceptance of the function enables it to be maintained throughout the life of the repository.

In order to fulfil this function, the aim is to use the primary conditioning of the waste, as carried out by the producer, to best advantage and add further lines of defence to those offered by the initial conditioning as required. This is particularly done by adding container elements specific to the repository. In both cases, the repository must preserve the favourable properties of the container as much as possible.

This function is facilitated by specific measures adapted to each type of waste. The modularity of the repository, and the preservation of a buffer thickness between modules for different types of waste, optimises the physical and chemical conditions within each module without having to take account of possible interferences from other types of waste.

Under these conditions, the analysis is conducted waste type by waste type.

3.3.3.1 In B waste cells (other than bituminised waste)

First of all, the analysis deals with cells containing waste other than bituminised sludge. The waste here is diverse in nature, with the concept taking care of this variability in an overall manner. A few characteristics are particularly beneficial to the protection of metallic waste (hulls and end caps, core-activated waste, etc.) and help stimulate the favourable properties of this waste. They are therefore distinguished as and where required.

Radionuclides are either present in the form of waste surface contamination, or included in the core of the waste. In this second case, the structure of the waste has the function of immobilising the elements with their release being driven by corrosion in the case of metallic waste. The primary function is therefore to maintain conditions that promote the restriction of corrosion within the cell. This is accomplished by controlling the oxidation reduction potential and the pH. As corrosion is inevitable, there is in principle no required performance level for this function. The aim is to seek the best possible conditions to be able to reduce the corrosion rate and stimulate the passivation of the steel, with an alkaline pH encouraging passivation under reducing conditions

The concrete disposal container is similar in terms of its principle for all B waste (see Figure 3.3-7). As a result, handling conditions can be standardised and the container also fulfils functions for the post-closure phase. In particular, it has an effect on pH control, with all of the concrete forming the B waste cells. Given the masses involved, the latter tends to impose an alkaline pH (10 to 12.5) throughout the medium. For certain packages (B1 and B5.2), the container is also attributed performance levels with regard to the transfer of radionuclides (container said to offer reinforced containability – see section 3.3.4.1).

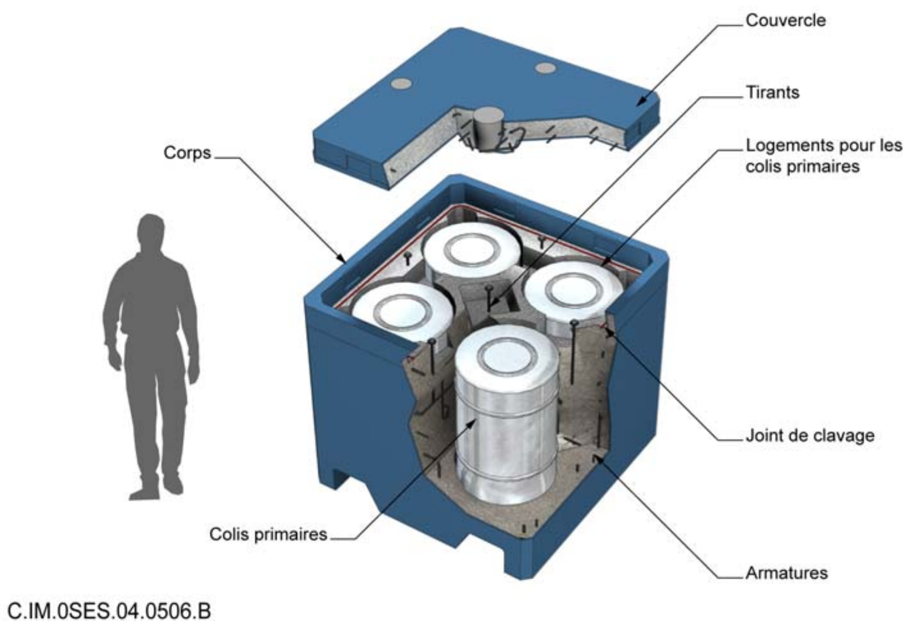

Figure 3.3-7 Illustration of a standard disposal container

Once the radionuclides are detached from the mass of waste, the aim is to limit their dissolution. The medium is globally reducing due to the absence of oxygen and the conditions imposed by the granite itself and the water contained in the granite. These conditions are favourable to the limitation of radionuclide dissolution and to the limitation of their mobility. Only a few radionuclides (iodine-129 and chlorine-36 in particular) are insensitive to these chemical conditions. Overall, this sub-function helps mobilise radionuclide retention properties in the medium in the near field (sorption and precipitation).

Moreover, by limiting the quantity of organic elements in the B waste cell (both through the low quantities of organic waste and by the fact that the construction of the cells and packages does not bring in any additional quantity of organic matter), their dissolution is minimised; limited quantities of colloids can however form as the packages are altered. However most of them only have low complexing power in a cement-based medium. The cell seal filters them in the event of migration to the drifts. Finally, in the architectural measures proposed, the packages containing organic waste (for example, cellulose) are emplaced in special cells in order to isolate the packages not containing any from potential complexing effects induced by the degradation of these materials.

3.3.3.2 In B2 waste cells (bitumen-embedded materials)

In order the control the release of bituminised packages, the studies conducted notably by the CEA highlight a number of favourable conditions [xv] determined both by the experiments and by the models associated with them:

- preservation as far as possible of the dimensional stability of the matrix,
- compliance with temperature criteria (with an optimum around 20 to 30°C),
- compliance with a pH range similar to that envisaged for other B waste (10 to 12.5).

These conditions may be fulfilled by the standard concrete package. It helps maintain the geometry of the bitumen by offering resistance to bitumen creep. The package must also be able to evacuate gases formed by radiolysis. It cannot therefore be allocated leaktight performance.

Keeping the bituminised waste cells away from exothermic waste, and the presence of high quantities of concrete in the cells, are favourable factors in the aim to limit release from the matrix.

Finally, the design measures already identified for other B waste, consisting of limiting near-field solubility and filtering the colloids, are also raised with no specific details being applicable to this case.

3.3.3.3 In C waste cells

For C waste packages, the first requirement is to prevent the inflow of water in contact with the glass during the period characterised by a relatively high temperature, lasting around a thousand years at most [xxxiv]. The aim is in fact to avoid the release of radionuclides for as long as the temperature prevents reliable identification of their behaviour, given the current limits of knowledge. This is also designed to protect the glass from the risk of its alterability in contact with water increasing with temperature, in connection with the behaviour models considered today [xv]. This function is fulfilled by an overpack which isolates the glass package from water and which has a lifetime designed to cover the phase during which the temperature of the glass is over 50-60°C, i.e. a period of the order of 1000 years. The period of leaktightness is driven by its corrosion; for this reason, the overpack is in non-alloy or low-alloy steel, a material whose behaviour under corrosion is the simplest and most predictable (see Figure 3.3-8). The presence of the buffer also enables the chemical conditions around the overpack to be homogenised and avoids possible localised corrosion by water from a minor fracture adjoining the wall. At this stage of the project, the thickness of the overpack is dimensioned at a total of 55 mm. This thickness guarantees leaktightness and its mechanical resistance over several thousand years (taking account of around 10 MPa of stress linked to water pressure and the swelling of the buffer).

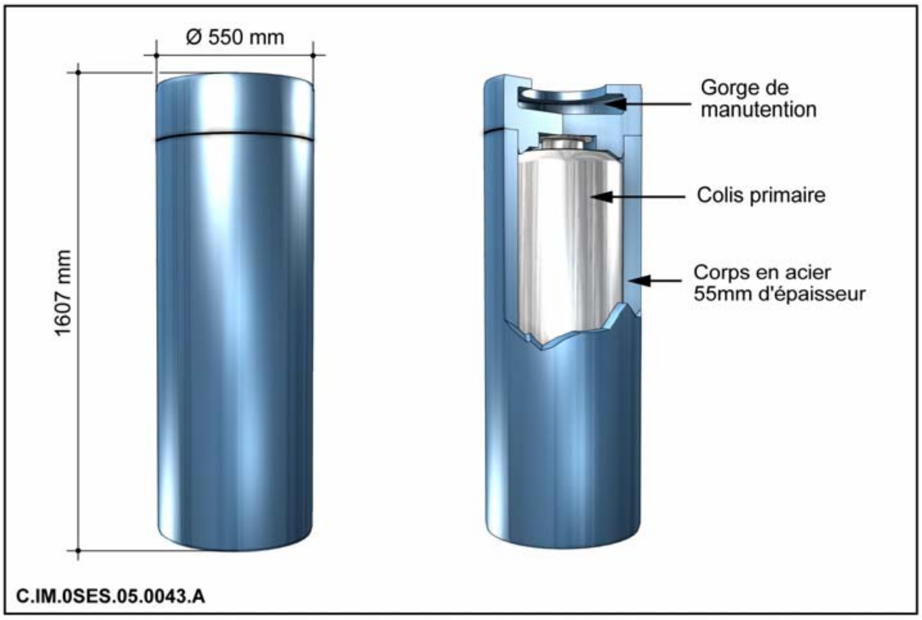

Figure 3.3-8 C waste disposal package (type R7/T7)

Beyond the thermal phase, other functions enable the glass to be placed within the cell under conditions which control its behaviour: the limitation of gaps ensures a certain degree of mechanical containment, to avoid overfracturing and activate the clogging of pre-existing fracturing as much as possible, while the distance from all sources of alkaline disturbance maintains a favourable pH (9 at most). Limiting the heat released from glass packages at the entrance to the repository and spacing the packages within the cells, motivated by the concern to protect the geological environment, also help to provide favourable conditions for the resistance of the packages by limiting the thermomechanical deformations in the cell.

The alteration of the glass is controlled by the silica dissolution rate, itself inversely correlated to the silica concentration in the surrounding water. Measures need to be taken to help maintain a chemical equilibrium between the water surrounding the glass and the glass itself in order to limit its alteration. The emplacement of a continuous medium of low porosity around the packages, formed by the buffer which controls transport conditions in the vicinity of the package and keeps it away from the influence of minor fracturing, helps maintain a high silica concentration at the water/glass interface (see Figure 3.3-9).

Finally, as in the B waste cells, releases outside the cell once the toxic elements have left the package are controlled by two generic functions:

- maintaining reducing conditions, in order to immobilise the radionuclides as much as possible;
- filtering the colloids; in the case of C waste, this is effected by the low porosity buffer.

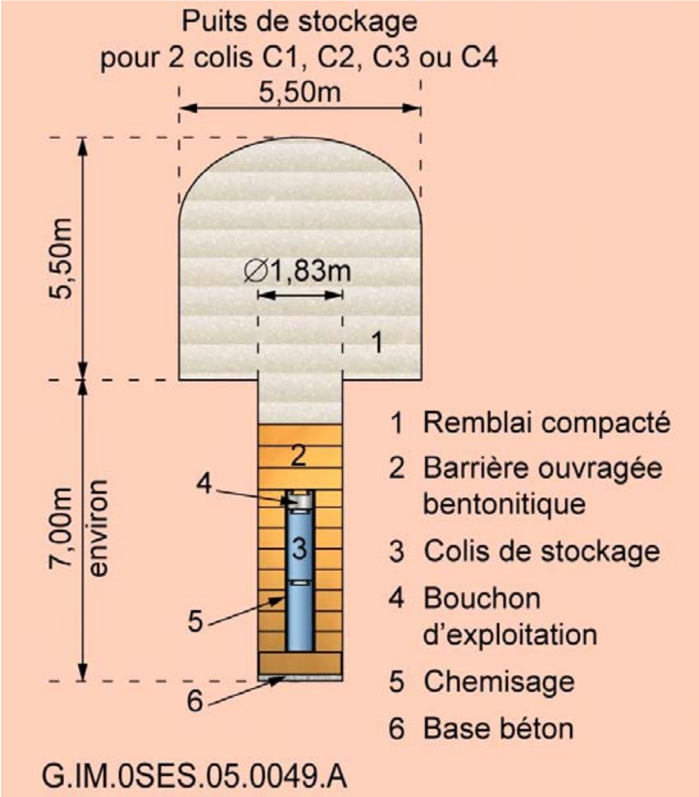

Figure 3.3-9 C waste disposal borehole

3.3.3.4 In spent fuel cells

Spent fuel cells are designed on the same principle as C waste cells and, overall, enable the same type of functions to be performed.

The main difference stems from the container chosen to isolate the spent fuel. The period of leaktightness of the containers must be at least equal to that of the thermal phase in order to control the release and migration of radionuclides during this phase of several thousand years. However, considering the possible footprint required for the disposal of spent fuel and the water-conducting faults liable to be intersected by the access drifts to the disposal cells, the leaktightness of the container will need to be maintained for a very long period (up to a million years) in order to safeguard against the failure of the seals emplaced in the repository and offer greater flexibility with regard to adaptation of the architecture, particularly in the absence of a defined site. This option - which could be reviewed according to the site studied - is the reference for this dossier. It is the one developed in Fenno-Scandinavia and is based on the "KBS-3" concept with a copper container (see Figure 3.3-10). This material is highly resistant to corrosion under reducing conditions.

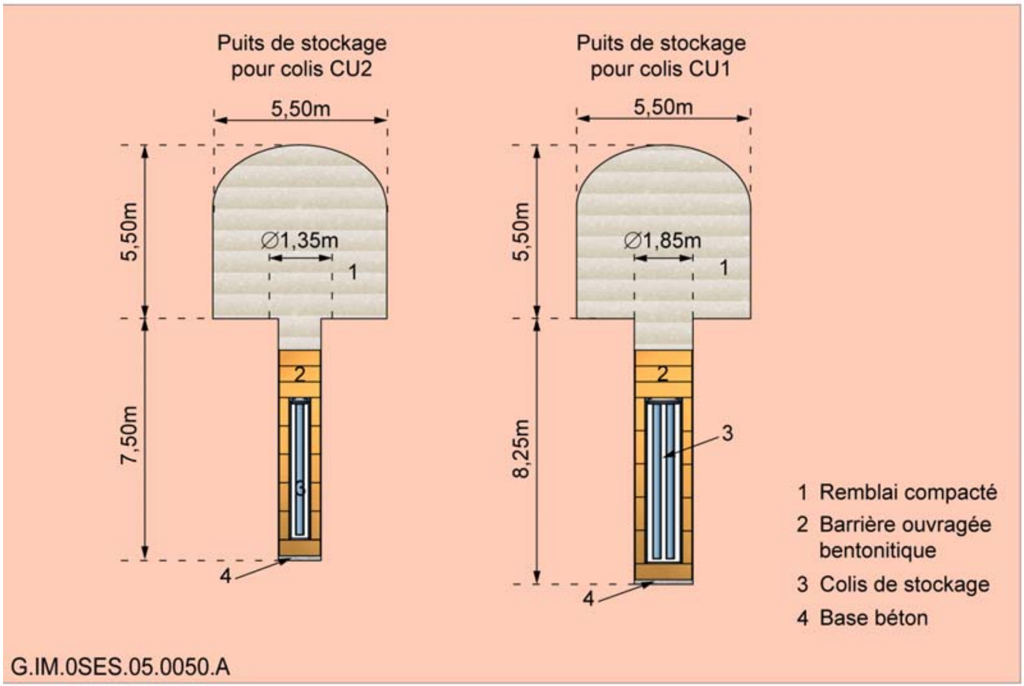

Figure 3.3-10 Spent fuel disposal cells

3.3.4 Delaying and reducing the migration of radionuclides to the environment

The third safety function consists of delaying and reducing, in space and time, the flow of radionuclides finally released by the waste to the environment. It is therefore latent during an initial time phase during which the radionuclides remain contained within the waste. In the case of spent fuel, it only comes into play in the event of a container failure

3.3.4.1 Controlling the migration of elements by diffusion/retention in the repository modules

Within the cells, toxic elements (radionuclides and chemicals) are essentially transported by diffusion. The transfer is delayed in this case by the sorption properties of the materials traversed, and by the low diffusion coefficients of these materials.

In the case of B waste cells, the additional concrete packaging also causes a delay. For waste packages not releasing any gas and containing a large proportion of the radioactivity, especially reference packages B1 and B5.2, the design of a durable disposal package, to which performance levels in terms of radionuclide transfer are allocated, is conceivable. This assumes that techniques aimed at preserving the mechanical resistance of the additional packaging over the long term are implemented: disposal package concepts in the form of monoliths, use of stainless steel reinforcing bars or fibres, etc. The dimensioning objective is 10 000-year durability for a package within a repository. As a result, in addition to the sorption capacity of the concrete, the factors of low permeability and a low diffusion coefficient, consistent with those of sound concrete, are mobilised over an equivalent period.

For other packages liable to give off significant quantities of gas (notably B2 packages – bituminised sludge), the container is not allocated any performance in terms of transport. In particular, it enables gas to be discharged outside the disposal package. Insofar as it does not form a sealed box, the delay to the transfer of radionuclides provided by this container over the long term is based solely on the chemical and sorption properties of the concrete.

In addition to the packaging, and in principle with no limit on duration, the cell plug also helps delay transfer.

In the case of C waste and spent fuel cells (if the container proved to be defective), the diffusion and spatial spread would be delayed by the bentonite of the buffer, including the plug.

3.3.4.2 Preserving the diffusion/retention capacity of the fractures in the granite

Elements having migrated may reach either the drifts or the geological medium itself. The very high compactness of granite limits transfer possibilities within the sound granite matrix to the extreme; it is therefore the fractures which can act as transfer channels. Moreover, if the "preventing water circulation" function within the access structures, including drifts, is sufficiently effective, the drifts do not form a preferential transfer channel compared with the fractures.

Fractures in the granite can present retention properties, at variable levels, linked to partial or total clogging by clay materials, or delay transfers by their tortuosity and the exchange surfaces that they present. Experiments performed in situ, notably at the Äspö underground laboratory in Sweden, have identified the phenomena delaying the migration of radionuclides in the fractures: resistance of the fractures to water circulation, radionuclide diffusion in the rock and radionuclide sorption at the surface of the minerals filling the fractures.

Due to the links made between these phenomena and the geological and mineralogical characteristics of the fractures, the experimental results from the Swedish laboratory can be extrapolated to various types of granite according to their own specific characteristics.

The preservation of the favourable properties of the geological medium, and the mobilisation of the characteristics of the fractures, assume that the rock has not been significantly disturbed by the excavation and operation of the repository structures.

- **Thermal and thermo-mechanical disturbances**

These disturbances are due to the dissipation of the heat generated by the radioactivity of the waste, by conduction in the medium. At this stage in knowledge, for C waste and spent fuel cells, design is based on a maximum temperature of 90°C at the hottest point of the buffer. This leads to maximum temperatures much lower than 100°C in the rock [x]. The aim is to avoid damaging the massif by a thermo-mechanical effect, and remain within temperature ranges covered by the state of knowledge and by the capacity to report phenomena and their coupled effects. It will be noted that this function is also to be considered for certain B waste, in the knowledge that the heat released remains much lower than that of the high-level packages, and decreases quicker over time (the maximum disposal temperature of the most active B packages will remain clearly below 80°C). In order to comply with the overall criterion, the sizing of the modules provides for sufficient space between the cells according to the thermal properties of the granites on the one hand and the storage time for C waste or spent fuel before disposal on the other.

- **Hydraulic and hydrogeochemical disturbances**

The excavation of underground facilities leads to the drainage of the water and disturbs the initial hydrogeology of the granite. As granite bears little water, the disturbances essentially affect the faults and fractures conducting most water. Therefore, during the excavation of the underground facilities, the phase during which disturbances are greatest, injection techniques are carried out on the faults and fractures conducting most water and intersected by the engineered structures, to limit the drainage of water from the granite and, at the same time, the quantities of water to be discharged.

Following the major disturbance phase linked to the excavation of underground facilities, an equilibrium between water drainage and supply is established in the granite.

Adequate phasing of the excavation, operation and closure of the various repository zones according to the context is thus a way of limiting hydrogeological and hydrogeochemical disturbances in the granite.

The characteristics of near-field fractures with regard to radionuclide retention are quite unlikely to be modified by these transient desaturation phases which essentially affect the significantly water-conducting fractures.

Due to their dimension, only a small fraction of the clay or carbonate filling of the water-conducting fractures can be modified. By limiting the importance of hydrogeological disturbances, these possible modifications are minimised.

Beyond the resaturation of the structures, the presence of metallic materials causes hydrogen to be produced by anoxic corrosion. The gas is evacuated from the structures via the medium by dissolving or diffusing. The risk to be managed is that of an excessive rise in pressure causing damage to the minor fracturing. This risk needs to be evaluated in a specific context. The gas is however expected to evacuate via the seal body to the drifts, without creating an irreversible opening which would form an ideal channel for the species in solution, or via the fracture network itself [x].

- **Mechanical disturbances linked to excavation**

The opening of underground caverns in a medium subjected to natural mechanical stresses generates a local redistribution of the stresses. As a general rule, the mechanical resistance of the granite rock and the stress system in the French context mean that such excavation will not create new fractures apart from the ones created by the breaking of the rock at the wall of the structures.

In the case of C waste and spent fuel, the small disposal boreholes could be excavated by merely boring, which would limit rock damage considerably. The properties of the minor fractures likely to exist in the wall are therefore only affected very superficially (over a depth of less than or of the order of a centimetre) with no consequences on their retention property.

For larger B waste disposal tunnels, the excavation method would use “softened” blasting techniques to minimise damage. The retention performance of the granite and minor fracturing is therefore modified over a depth of a few decimetres, which is taken into consideration in the buffer distance to be provided in relation to water-conducting fractures.

- **Chemical disturbances**

The input of cement-based or ferrous materials into the repository causes disturbances of a chemical nature which cannot influence the fracturing or the repository materials. This latter case is controlled by design by planning to keep the sensitive materials (such as bentonite) at a sufficient distance from the most harmful sources of disturbance (alkaline disturbance in this case), or to provide a “lost” thickness dedicated to buffer the disturbance (for example, for the seal bodies up against a concrete abutment). The case of the rock, and possible chemical interactions within the minor fracturing which could modify the initial properties of it, are still to be dealt with.

In the vitrified waste and spent fuel cells, the presence of the clay engineered barrier protects the rock which is consequently sheltered.

Due to their cement-based and metallic content, the B waste cells and, to a lesser extent, the drifts, can however be sources of disturbances in the medium; the one with the greatest extent, capable of affecting the fractures other than very superficially, being alkaline disturbance. The effect of this disturbance on the fractures could be marked by the precipitation of minerals favourable to clogging. Furthermore, the disturbance could not spread over significant distances in view of the extent of the fractures without being quickly cushioned.

3.3.4.3 Preserving the natural capacity of dispersion into the repository environment

Strictly speaking, dispersion is not a function of the repository insofar as there is no aim to optimise it by specific design measures. However, a granite massif possesses properties of dispersion of elements which would manage to reach the far field, in the very long term, through the actual dispersion of the fracture network. It is important that the repository preserves at least this dispersion capacity which complements the other functions.

To this end, efforts should be made to limit the creation of preferential transfer channels in the far field. Particular care should notably be taken during surface investigations by boring and during exploratory and characterisation work on advancement. This point is therefore integrated into the exploration strategy

3.3.5 Summary

The description of the safety functions reveals the existence of three complementary lines of defence which last throughout the period of analysis: one based on the control of convection within the repository, one enabling toxic elements to be immobilised in the near field, and one leading to the delay and spreading of the elements.

The design measures proposed to fulfil these functions are derived from techniques which either have equivalents in other industrial sectors, such as mining engineering, or which have been developed and demonstrated from a technological point of view in the scope of foreign programmes. At this stage, the definition of the concepts has not entered an excessive level of detail, preferring to remain sufficiently generic. It has notably aimed to offer a degree of adaptability which can be seen at two levels:

- adaptability with regard to the “siteless” character of the study, the design measures proposed calling on particular characteristics (chemistry, fracturing, etc.) of the granite medium as little as possible. This leads to the provision of engineered structures (engineered barriers, buffers, copper containers, effective backfill, etc.), the dimensioning or even the actual relevance of which could be reconsidered in a better-defined context;
- adaptability with regard to the variability of the characteristics of the granite massif and exploration uncertainties. By integrating this into every stage of the repository design process, from initial exploration of the site to the actual excavation of the structures, and by enabling the modules to be adapted to the fracturing effectively explored, it is possible to overcome the limits of exploration of the massif from the surface. Nevertheless, it remains that this issue of the characterisation of the fracture network is a central point of the safety analysis.

The qualitative safety analysis (chapter 5) and the repository performance calculations (chapter 6) aim to ascertain that the design measures proposed are effectively able to fulfil the safety functions required.

4 Operational Safety

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4.1 Elements relating to operational safety

As in the case of any industrial facility, the various repository activities (construction, operation, closure and, if applicable, withdrawal of the waste packages) may involve risks or hazards for people and the environment. The consideration of such hazards, beginning with the design of the facilities, is conveyed by the definition of operational safety functions and appropriate technical means to minimise the exposure of workers and the public to the hazards.

In the scope of a defence-in-depth approach, possible hazardous situations need to be identified along with the design of the facilities, even if they appear unlikely, and measures need to be proposed to prevent their occurrence and limit their effects. This analysis benefits from feedback from the nuclear and mining industries.

In the case of the study of disposal in a granite medium, hazard identification is conducted in a generic context while highlighting the main risks. A large part of the studies conducted during research into disposal in a clay medium can be transposed to the granite scenario[11]. The results obtained in the scope of studies relating to disposal in a clay medium are cited wherever necessary.

4.2 Protection of people

Operating activities relating to the repository lead to the adoption of safety functions which are specific to nuclear facilities. Moreover, measures consisting of protecting the operators against usual work hazards relating to surface and underground structures must also be taken into account.

4.2.1 Definition of safety functions

Nuclear safety functions are as follows:

- Confining radioactivity
- This involves containing radioactive materials to prevent their dissemination. In particular, releases of gaseous radionuclides liable to be given off by certain waste packages will be limited as far as possible and monitored ([xvi] and [xvii]). Systems monitoring the absence of surface contamination on packages also relate to this objective. Fulfilling this safety function limits the risks of inhalation and ingestion of radionuclides by workers and the public within the immediate environment of the disposal facilities.
- Protecting people against radiation
- This function consists of protecting operators and the public from radiation from the packages. This is done by interposing fixed or portable protection screens, keeping operators away from sources of radiation and managing their period of exposure.
- Controlling the criticality hazard
- This function consists of avoiding a criticality accident[12], the consequences of which could notably undermine the containment and radiation protection functions [xviii]. This objective is achieved by controlling reactivity which is obtained by monitoring fissile materials, the geometry of the packages, their distribution in the disposal facilities and, if applicable, the interposition of neutron-absorbing materials[13].
- Evacuating the residual thermal power

  This function consists of limiting the temperature levels in the facilities by dissipating the heat energy emitted by certain packages.
- Evacuating gases produced by radiolysis

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[11] The hazard analysis for disposal in a granite medium is comparable to the one conducted for disposal in a clay medium, with a few particularities linked to the geological formation: risk of falling blocks present not only in construction but also in operation (the drifts would generally not be concreted), water ingress (in the construction phase), etc.

[12] A criticality accident is an uncontrolled nuclear chain reaction (fission) between neutrons and fissile materials (uranium-235 and plutonium-239 and 241).

[13] The role of this type of material, capable of capturing neutrons, is to limit the nuclear reaction.

This involves evacuating the explosive gases linked notably to a phenomenon of radiolysis[14] specific to certain packages. Ventilation of the facilities contributes to this during repository operation.

4.2.2 Objectives relating to the protection of people

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

For the operating phase, the hazard reduction measures taken, consisting essentially of radiological protection screens and non-contamination inspection measures, should limit the exposure of people to below the restrictions set by Andra as radiological protection targets for people [xix]: 5 mSv per year for personnel working in a nuclear zone and 0.25 mSv per year for the public off site.

People must also be protected from other harmful effects potentially generated by the construction, operation and closure of the repository[xx] : including:

- physical hazards (falling, crushing, electrical injury, etc.), which exist throughout the various repository activities, and in particular during construction work;
- the risk of fire;
- the risk of explosion linked to radiolysis gases (hydrogen, etc.) emitted by certain B packages;
- hazards inherent in the working atmosphere (noise, dust, toxic gases given off by the transporters, heat released by exothermic packages, etc.).

Environmental protection restrictions (for the record: groundwater and surface water [xxi], air [xxii], landscape, the vicinity, soils, fauna and flora) would also need to be considered throughout the repository activities, including during the post-operation monitoring phase. This monitoring should also cover the parameters which are liable to occur on long-term safety functions.

4.3 Radiological hazards in operation

The operation of nuclear facilities involves radiological hazards linked to the nature of the waste packages. Design measures help reduce these hazards to a level below the set targets and ensure safe operation of the facilities.

4.3.1 External exposure hazard

The waste packages are sources of external exposure (linked to β-γ radiation and to neutrons) from the moment they are received at the surface facilities to their emplacement in the underground facilities.

The transport canisters containing primary packages delivered to the repository site have a radiological protection function and their structure is designed according to the radiological characteristics specific to the waste transported. Once extracted from their canisters, the packages are handled and conditioned inside cells inaccessible to operators who will work remotely behind radiological protection screens (walls and windows).

During transfer operations to the emplacement in the disposal cells, the external exposure hazard is controlled by the interposition of radiological protection screens[15] between the radioactive sources and personnel to reduce the radiation flux. The disposal package transfer casks, the C and CU disposal borehole operating plugs and the gates of the B waste disposal tunnels fulfil this role. The protection or remoteness of the control unit for the transporters used for the transfer or introduction of the packages into the disposal cell would also contribute to reducing the doses received by the personnel in order to meet the target set by Andra.

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[14] The phenomenon of radiolysis is linked to the effect of ionising radiation (β, γ) emitted by radioactive materials on the hydrogenated substances present in certain B waste packages (organic matter, water from the conditioning matrix and the concrete envelope of the package). It results mainly in the release of hydrogen and, to a lesser extent, methane.

[15] The material used for these screens depends on the type of radiation emitted by the radioactive source:
- for gamma radiation, heavy materials such as steel, concrete and lead glass are used;
- for neutron radiation, specific materials (with boron or cadmium, etc.) or hydrogenated substances are used;
- alpha and beta radiation do not require any particular kind of screen as they are stopped by the envelope of the packages.

An initial estimate of doses received at various work posts was conducted for the project relating to disposal in a clay medium (cf. chapter 4.1 of [xxiii]), which is highly comparable with regard to the operating processes. The external exposure dose is the main component of the radiological hazard. The maximum dose received by personnel would be of the order of 4 mSv per year. These doses could be optimised within the scope of an ALARA (As Low As Reasonably Achievable) approach in later phases of development of the disposal project.

4.3.2 Internal exposure hazard due to ingestion or inhalation of radioactive materials in aerosol form

This hazard can occur essentially in surface facilities. It could be linked to the dispersion of radioactive particles originating from transport canisters, packages (primary packages or disposal packages) or transfer casks.

The management of this hazard would depend on the organisation of the facilities receiving the primary packages and preparing the disposal packages in containment systems[16] in order to avoid the dispersion of radionuclides towards personnel traffic zones or into the environment. Furthermore, these facilities would be equipped with filtering devices on their ventilation system, following the example of what is done in other existing nuclear facilities of the same type[17]. Finally, it is important to mention that non-contamination inspections[18] will be carried out every time on transport canisters, packages and casks [xxiv].

4.3.3 Internal exposure hazard due to inhalation of radioactive gases emitted by disposal packages

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

In surface facilities, the limited number of waste packages present is such that the total quantity of gases released into the atmosphere is negligible.

In disposal tunnels where a large number of packages are emplaced, a ventilation system with ducted extraction of the air to the repository exhaust shaft evacuates these gases without affecting the personnel present in the underground facilities.

In the scope of studies conducted on an actual site, emissions into the atmosphere could be evaluated together with their potential impact. Considering the ventilation throughputs and the negligible quantity of radioactive gases released, the calculated doses would be very low.

4.3.4 Internal exposure hazard due to inhalation of radon gases emitted by the terrain in the underground repository facilities

This hazard, which is present from the beginning of the construction activity, is linked to the natural exhalation rate of the granite in which the underground facilities would be located. Data on mining operations in this type of environment [xxv] shows that, to control this risk, the underground drifts must be permanently ventilated in order to dilute and quickly evacuate the radon emitted into the outside atmosphere.

4.3.5 Summary

From the initial dosimetry estimates, it can be considered that the doses received via all channels of contamination together would be lower than the annual restrictions set by Andra for workers and the public.

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16 The principle of a containment system is to create a difference in air pressure between adjacent rooms.

17 These filtering devices are also justified by the consideration of accident situations, particularly for the receiving and preparation of bare spent fuel which may be contaminated by corrosion products deposited and activated while the fuel assembly is in the reactor.

18 The acceptance thresholds could be the ones adopted in the Transport Regulations, i.e. labile surface contamination (non-fixed) limited at 4 Bq/cm^2 in β,γ emitters and 0.4 Bq/cm^2 in α emitters.

4.4 Hazard analysis in an accident situation

This section presents the main hazards liable to occur during repository activities and the associated measures to reduce the hazards.

At this stage of the study, the hazard analysis focuses on the hazards inherent in the repository ("internal" hazards). These include "conventional" hazards which are encountered in any industrial facility and hazards linked to the presence of waste packages, which are essentially radiological hazards (external and internal exposure hazards, or even criticality hazards).

On the other hand, "external" hazards linked to the repository environment (seismic hazard, weather hazards, risk of an air crash, etc.) are not developed as the site has not been located. They would mainly involve the surface facilities which could be damaged by this type of event and could also lead to the degradation or loss of the safety functions ("radiological protection" function, "radioactivity containment" function, etc.). The consideration of such risks would therefore be based on rules usually applied to basic nuclear installations [xxvi][xxvii].

4.4.1 Conventional hazards

Hazards relating to the disposal process can be linked to the behaviour of the terrain: crushing hazard as the result of a rock fall in the drifts (especially during excavation work), risk of water ingress into the shafts or drifts, etc. They are also associated with the use of equipment (dropped loads during a handling operation, dropped objects in the shaft, falls during overhead work, crushing by moving equipment, etc.) or the use of transporters (collision with a person or another transporter). Equipment and transporters can also cause an electrical injury to operating personnel or a fire, the consequences of which may affect a large number of people, especially in underground facilities. Finally, hazards associated with the characteristics of the packages should be noted. The origin of these hazards would be a temporary loss of ventilation in the facilities. This would result in a risk of overheating associated with the presence of exothermic packages (C and CU packages) and the risk of explosion linked to the emission of small quantities of explosive gases[19] by certain B waste packages (B2 and B5).

Hazard reduction is firstly dependent on prevention with the use of suitable, reliable and well-maintained equipment and transporters fitted with all the safety features required, the training of personnel, raising awareness of the various types of risk incurred, and compliance with procedures and traffic rules. It also relies on the choice of fire-arresting materials and the limitation of the fire load present.

Furthermore, for underground facilities, explorations on advancement of the excavation face can forewarn of hazards linked to the poor condition of the terrain or the ingress of water. Emplacement of physical protection (especially for shaft work), use of machinery with control units back from the site face area, installation of heat shields (fire doors and fireproof materials) and extension of the sites' safety systems (fire smoke extractor system, fire-fighting water mains, communication network, etc.) as excavation work progresses also help reduce the risks facing personnel.

As far as package-related hazards are concerned, simulations made in the scope of the study of disposal in a clay medium (cf. chapter 4.3 of [xxiii]) show that the risk is controlled in the operating phase by the ventilation of the various disposal facilities, and that an interruption to the ventilation of the facilities does not present any real hazards as the available time to repair is long. These conclusions can be transposed to the scenario of disposal in a granite medium

[19] These gases (mainly hydrogen) originate from a radiolysis phenomenon which is linked to the effect of ionising radiation (beta, gamma) emitted by radioactive materials on the hydrogenated substances present in the waste packages (organic matter, water from the conditioning matrix and the concrete envelope of the package).

4.4.2 Radiological hazards

The results of the identification of the various radiological risks linked to hazardous situations in the course of repository operation are presented per hazard type, by highlighting the hazards that appear to be potentially the most harmful with regard to people or the environment. As in the case of the study conducted in clay [xxviii], this analysis benefits from industrial feedback from comparable nuclear facilities to repositories.

These hazards, and the main hazard reducing measures relating to them, are as follows

4.4.2.1 Hazards linked to failures in the radiological protection measures

An exposure hazard may be linked to a fault on the protection provided by the gates, hatches and windows of waste handling cells, transfer casks, C and CU disposal borehole plugs and gates of B waste disposal tunnels. This event could be due, for example, to play between moving parts which is not consistent with the initial design. The measures to be taken to overcome this risk would be the installation of irradiation detectors at the waste handling cells, transfer casks and disposal cells, and the carrying of appropriate dosimeters.

Accidental exposure during a maintenance operation in a room adjacent to an irradiating cell can also be envisaged. The maintenance operations would be preceded by a preliminary irradiation check in this room. It could also be anticipated to locate the maintenance area away from the field of potential irradiation linked to radioactive sources present in the cell.

A malfunction on equipment handling or transporting a package could immobilise the equipment and require the intervention of maintenance personnel. The preventive measures in this case would consist of adequate maintenance of the equipment used and the redundancy of certain constituent parts (motor system, etc.). A repair action would generate an exposure hazard for personnel working in proximity to the irradiating source. Means designed to put down the load and bring the equipment back empty to its maintenance room, minimising the duration of operator presence by precise planning of the repair action, and the dimensioning of the thicknesses of radiological protection would be the most effective measures to eliminate or limit the exposure of personnel

4.4.2.2 Hazards linked to the consequences of a fire

In the nuclear facilities of the repository, a fire could have radiological consequences if it involves a package of radioactive materials. The use of package transporters on the surface or underground would call on specific precautionary measures, insofar as a fire on the transporters could degrade the radiological protection in place, or even lead to a loss of containment of radioactive materials and their dispersion into the atmosphere.

As far as the receiving of packages is concerned, a fire on the transporter of packages placed in canisters capable of withstanding a fire at 800°C for 30 minutes [xxiv] should not have any radiological consequences.

For cask transporters, studies on this type of machinery conducted in the scope of the study of disposal in a clay medium (cf. chapter 4.5 of [xxiii]) show that temperature rises are liable to affect the metallic envelope of the casks but have no impact on the structure of the actual cask. The waste packages would not be subjected to a temperature rise liable to damage the packaging or degrade their matrix. It would however be necessary, following a fire, to check the level of radiological protection provided by the cask, in order to adopt appropriate measures, where applicable, to protect the intervention personnel (for example, portable radiological protection).

Similarly, simulations relating to fires on machinery used to emplace B waste packages in their disposal tunnel (cf. chapter 4.6 of [xxiii]) show that there would be no consequences on the waste packages insofar as the machinery is designed with a heat shield integrated between the motor system at the origin of the fire and the handling component carrying the package.

These various results are valid for disposal facilities in a granite medium due to the fact that the machinery and transporters used would be identical.

4.4.2.3 Hazards linked to the consequences of a dropped package

This is an event which is studied systematically in facilities where packages of radioactive waste are handled.

- **Dropped package hazards in surface facilities**

The handling of primary packages by overhead travelling cranes, transfer trucks, tippers, etc., could result in the packages being dropped and damaged in receiving and preparation facilities. The possible consequences would be a breach in the envelope of one or more primary package(s) and the dispersion of radioactive materials in the facilities, and then outside, into the atmosphere, via the air extraction systems.

The first preventive measure will be to limit package handling heights to their drop resistance height. Furthermore, at equipment level, prevention includes dimensioning of gripping components with margins, possible redundancies on certain components, systems to keep the grab in the closed position in the case of electrical power failure, etc. Personnel training and equipment maintenance are also very important in the control of this risk.

These various measures should reduce the risk of dropping packages and eliminate the risk of losing the containment function of the packages. However, following the example of what is done in nuclear facilities of the same type, the use of static and dynamic containment devices could be a measure which eliminates all risk of dispersion of radioactive materials into the atmosphere in the event of a package being dropped.

A disposal package could also be dropped during a handling and transfer operation.

The preventive measures are identical to those listed above. Considering that the maximum drop heights are of the order of two metres and that the envelope of the disposal package and the cask (during transfer to the shaft) protect the primary packages, the loss of containment would not be an issue.

- **Dropped package hazards during the transfer of disposal packages between the surface facilities and the underground facilities**

The transfer of casks containing the waste packages between the surface facilities and the underground facilities could be carried out either by a transporter circulating in an inclined drift (ramp), or by a cage in a vertical shaft. This drift or shaft would be assigned specifically to package transfer.

In the case of package transfer via a ramp, the associated risks would be limited. Should a cask be dropped due to the fasteners holding it on the transporter breaking, this would not have any radiological consequences for the same reasons as during transfer on the surface. If the impact on a cask resulted from the failure of the various braking systems on the transporter, the impact speed would be an aggravating factor. Protective measures such as the installation of an energy absorber on the transporter structure, limiting the length of drift straights and, possibly, escape mechanisms forming emergency braking areas, should enable this risk to be controlled and prevent the cask from breaking.

If the packages are transferred via a shaft, hazardous situations would involve uncontrolled drifting of the cage, or even a free fall in the highly unlikely event of the suspension ropes breaking20 [xxix].

The preventive measures for such situations (see Figure 4.4-1) are derived from the experience acquired in deep underground mines. They concern not only the design of the equipment (independent braking devices on the driving pulley, bundle of independent cage suspension ropes, etc.) but also maintenance, inspection and operating procedures.

The installation of a cage fall arresting system, triggered in the event of overspeed, could be added to these measures in order to have an extra safety system independent of the case inspection and control system.

These devices could be complemented by other protective measures:

[20] In Germany, a study conducted for the Gorleben radioactive waste disposal project estimated, for a facility comparable to the one planned for the Andra repository, that the probability of a cage falling down a shaft is 5.10^{-7} per year (for 5000 hours of annual operation).

- a limit-of-travel braking system, of a similar type to what is regulatory installed in mines, a few metres below the underground station, would limit the mechanical consequences of drifting;
- an energy absorber placed at the bottom of the shaft appears to be a solution to take into account cage free-fall. Simulations on such a system, conducted in the scope of the study of disposal in a clay medium (cf. chapter 4.7 of [xxiii]), show that the cage, the damper and the transfer cask would absorb practically all the kinetic energy generated. C and CU packages should withstand the impact without breaking, whereas a loss of containment cannot be ruled out for the more fragile B waste packages (such as the B2 bituminised sludge packages).
- These preliminary results would lead to risk reduction measures being envisaged, notably with the possibility of installing a filter on the shaft air extraction circuit (see Figure 4.4-1) to control releases in the atmosphere after a cage fall[21].

[21] It should be noted that this filtration unit, installed on the exhaust shaft, is only started up in the event of an incident.

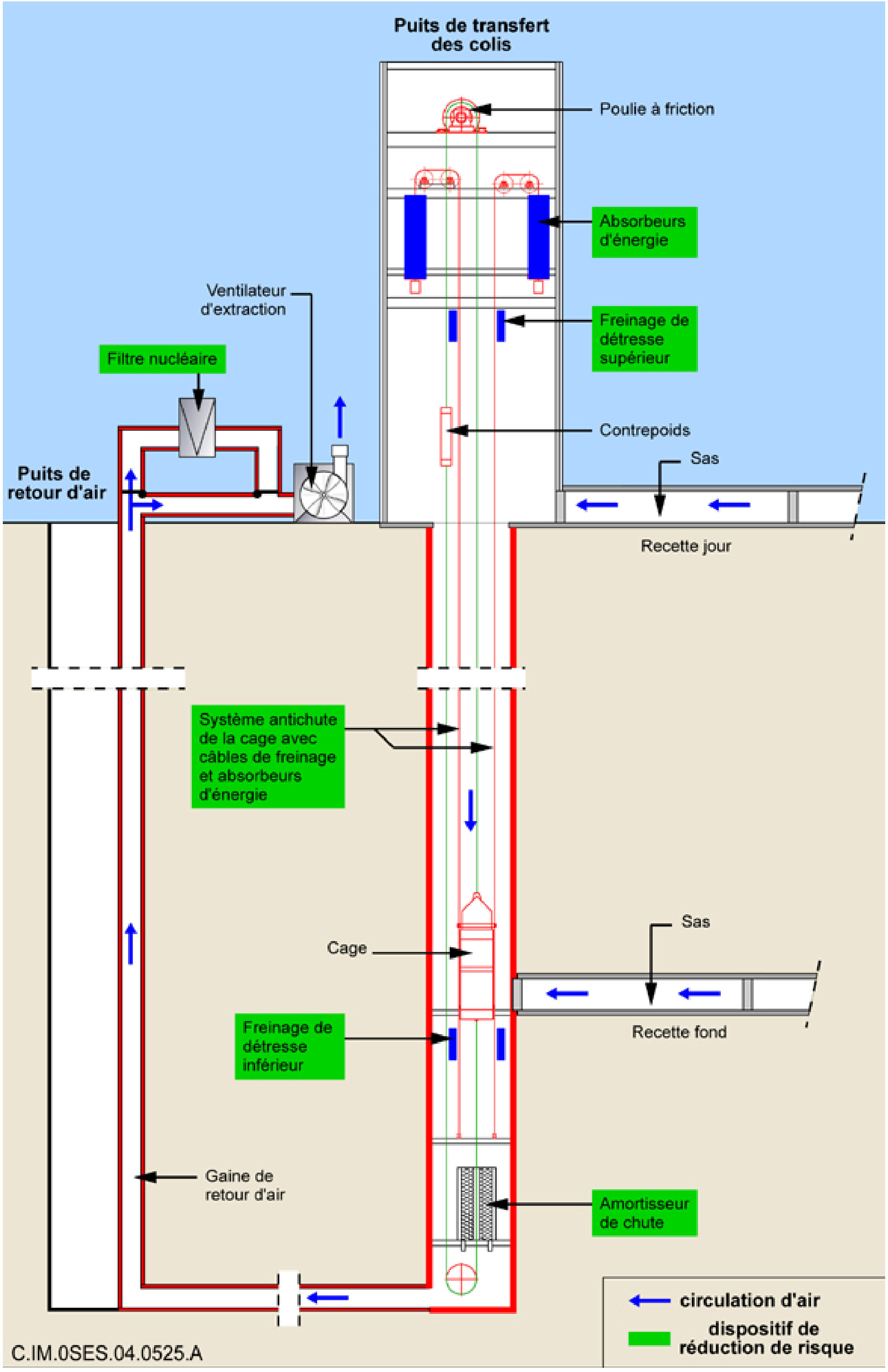

Figure 4.4-1 Diagram of risk reduction devices envisaged for package transfer in the shaft

- **Dropped package hazards in underground facilities**

As on the surface, if a cask containing the package is dropped while carried on a transporter, this should not have any radiological consequences due to its mechanical strength, the low potential drop height and the speed of transporter travel.

As far as the risk of dropping the package on emplacement into the disposal cell is concerned, the analysis depends on the type of package envisaged.

B waste disposal package

The handling (or retrieval) of B waste disposal packages with the need to stack (remove) them over several levels could result in the fall of one of the packages. The fall could be caused by a malfunction on the hoisting system or an error in identifying the expected position of the package to be emplaced.

The preventive measures relate to the choice of transporter which must be stable in all movements made with the load and be fitted with adequate safety equipment, with notably a device to lower the load if an anomaly is detected. Follow-up of the package emplacement cycle and visual monitoring by an operator, emplacement of the packages by completing one layer of packages before starting another (and not stack by stack) in order to limit the fall to the package tipping to one side, etc. Finally the disposal package protects the primary packages contained in it in the event of a fall.

Drop simulations have been conducted in the scope of the study of disposal in a clay medium (see chapter 4.8 of [xxiii]) for which the same handling mode has been adopted. They show that the concrete envelope of the disposal package would be damaged by the fall, but that the level of deformation of the primary packages would not cause their envelope to be breached.

C waste disposal package

The disposal of C waste packages is envisaged in a vertical borehole[22].

The dropping of a package on emplacement (or its possible retrieval), which may be caused by a malfunction on the grab used to handle the package, is not to be ruled out in spite of the safety systems designed into the handling equipment.

Specific simulations [xxx] have been conducted in a scenario involving the first package emplaced falling from the surface to the bottom of the disposal borehole. The 20-metre height adopted for the study comfortably covers the envisaged disposal borehole heights which are only around ten metres.

The results obtained illustrate that, even from this height, the primary C waste packages are practically undeformed, although the metallic envelope of their disposal package is damaged, but not broken, at the base of the package. All loss of containment of radioactive materials therefore appears to be eliminated.

Spent fuel (CU) disposal package

As for C packages, the dropping of a spent fuel package during emplacement (or retrieval) cannot be totally ruled out in spite of the safety measures taken on design.

A simulation has been conducted for a spent fuel package[23] by envisaging a drop of 7 metres (the height of the disposal borehole). It shows that plastic deformations could affect the base of the cast-iron package insert but that the spent fuel slots and the actual spent fuel assemblies would not be broken.

The integrity of the package should therefore be retained as a result of a fall down the disposal borehole and there would be no loss of containment of the radioactive materials.

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[22] This differs from disposal in a horizontal cell as envisaged in the case of disposal in a clay formation.

[23] The simulation study opted to study the case of a package with the more rigid structure of a metallic envelope which is less favourable than the case of a package with a copper outer envelope adopted in the case of disposal in a granite medium.

4.4.2.4 Criticality hazard

The criticality hazard corresponds to an uncontrolled nuclear chain reaction. This is initiated by an increase in 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) required for this type of reaction to occur. Spent fuel packages are the only ones concerned by this hazard [xxxi].

In the surface nuclear facilities for spent fuels, it should be ensured that there no water[24] can penetrate into the cells where spent fuel is received and disposal packages are prepared in order to eliminate the criticality hazard, following the example of what is done in similar existing storage facilities at waste production sites.

In underground installations, the transfer and emplacement of the packages must be undertaken in dry conditions. There is no criticality hazard in relation to this operation.

During the transfer of disposal packages through the transfer shaft between the surface facilities and the underground facilities, the hypothesis of a scenario combining serious damage to the spent fuel package (internal geometry modified, fuel bundles broken and brought closer together, etc.) and the ingress of water could lead to a criticality hazard.

Considering the measures envisaged with regard to the dropping hazard in the shaft (see Figure 4.4-1), such a level of damage to the package seems unlikely. However, an additional precaution would be to make sure that there is no water (or other hydrogenated fluid) in the shaft in order to eliminate this risk completely. This would consist of prohibiting mains from the package transit shaft while also fitting a water evacuation system from the shaft bottom.

The other hazardous situations envisaged [xxxi] do not appear to be able to cause a criticality hazard.

4.5 Conclusion

The analysis presented in this chapter has highlighted the main risks and hazards relating to the disposal process for people and the environment and, as a result, appropriate preventive and protective measures have been proposed. It is based on feedback from existing nuclear facilities and on lessons learnt from the studies developed in the scope of the “Dossier argile”. It does not intend to be exhaustive at this stage in the studies.

The analysis has differentiated between conventional hazards which are encountered in any industrial facility and hazards linked to the waste packages. The risks linked to the outside environment of the repository are not covered in the absence of a definite site.

In the surface facilities, conventional hazards have been identified at various levels throughout the various activities of the repository. These are essentially crushing hazards (dropped loads during handling, collision with a transporter, etc.), falling hazards relating to overhead work, risks of electrical injury and the risk of fire, etc. These hazards do not require any additional investigations at this stage. Nevertheless, they will need to be conducted in greater detail at later stages in the development of the studies.

In underground facilities, falling blocks of granite during the construction of the structures and, to a far lesser extent, water ingress are additional hazards to those mentioned above. At a later stage in the project, the risk of fire will have to be covered in a particularly study, on account of its influence on the design of the facilities, in order to ensure that the recommended solutions would enable people to be evacuated under satisfactory safety conditions.

[24] The presence of water, which reduces the energy of the neutrons and slow them down, makes them more reactive to fissile materials and, as a result, increases the reactivity of the system. Therefore, the processes adopted for the preparation of packages are dry processes with no added water.

The hazards relating to waste packages are essentially radiological hazards. They are inherent in the operational activity of the repository and, to a lesser extent, the closure activity. These hazards could be associated with exposure to radiation (defective radiological protection, operations in proximity to a radioactive source, etc.) in addition to a fire or a dropping hazard affecting the actual packages. The measures envisaged, which benefit from feedback from similar industrial facilities, must enable these hazards to be controlled.

Hazards relating to the repository environment (earthquakes, weather conditions, air crashes, etc.), which would be dealt with on the basis of the usual practices adopted by French nuclear facilities taking account of the local characteristics of the site chosen for the repository, do not appear to raise any specific difficulties.

The analysis conducted on operational safety has not revealed, at this stage, any elements disputing the technical feasibility of the construction, operation and closure of the repository and its reversible management in stages (with notably the possibility of reversing the process).

5 Qualitative safety analysis

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5.3 Conclusions of the qualitative analysis 138

The purpose of this chapter is to discuss the main results obtained from the qualitative safety analysis conducted on disposal concepts in granite. Within the context of the Dossier 2005, this analysis consisted of systematically reviewing the characteristics of the components, the physico-chemical processes and the external events which can affect the respository's evolution. This analysis allows specifying and justifying the representation of this evolution within the normal evolution scenario (SEN) by detailing the possible variations and the uncertainties which affect the models. By defining the limits of the phenomenology covered by the SEN, the analysis contributes also to identifying the situations not covered. These situations can be linked in such a case to an altered evolution scenario (SEA).

This chapter has therefore a triple purpose:

– present the calculation model of the normal evolution scenario, whose results are given in chapter 6;
– compare this model to the state of knowledge about granite and the behaviour of the repository's components in order to justify how this representation is influenced particularly by the uncertainties which persist at this stage. This means looking at the phenomenology without trying to be exhaustive but in a view point of safety; the reader interested in a more detailed description of the state of knowledge should refer to the volume "phenomenological evolution of the geological repository";
– compile an initial list of "altered" situations outside of the normal evolution field.

5.1 Methodology

In order to evaluate all the phenomena which could impact on the repository's evolution, it was decided to rely on international "Features, Events and Processes" (FEPs) databases. These bases constitute, in fact, a reference of events and processes affecting the intrinsic evolution of the components, the interactions between the repository's components, and the transfer phenomena of the radionuclides which should be taken into account in a safety analysis. Starting from internal FEP bases intended for diverse host formation types, a FEPs base adapted to the generic granite context was created [xxxii]. This FEPs base was derived particularly from the FEPs base of the OECD/AEN [xxxiii].

Once the list of FEPs a priori applicable was compiled, an initial analysis consisted of highlighting the processes (or events) concerned with the functions expected from the design components [xxxiv]. This initial stage allowed eliminating a number of FEPs considered irrelevant with respect to the execution of the safety functions. In fact, some FEPs can, for example, refer to processes which are not taken into account by Andra because they do not contribute to the safety functions, and can no longer impede the execution of these functions. The thus-compiled Andra/Granite FEPs base are upstream the qualitative safety analysis.

A second stage consisted of reviewing all the records created in the Andra/Granite FEPs base in order to extract, on the one hand, the FEPs presenting the components characteristics and, on the other hand, the events and processes describing their behaviour over time. Thus, the thermal, hydraulic, mechanical, chemical and radiological processes were examined to ensure their adequacy with respect to the safety functions. A mapping is made with the phenomenological analysis of repository situations (PARS) [xii] which describes the evolutions expected based on the specificities of the concepts studied and on the typology of the French granites.

The analysis of all the FEPs allows distinguishing those:

- whose occurrence is expected (“FEPs in normal evolution”);
- whose occurrence would disclose an abnormal condition (“FEPs in altered evolution”), either generically or in specific unfavourable contexts.

The detailed results obtained from this analysis are given in a separate document, which details each FEP and the way in which this FEP was taken into account or not [xxxv]. This chapter gives a summary of this analysis favouring a less analytical approach and, therefore, does not individually describe each FEP. The FEPs are mentioned when necessary but not systematically.

To represent the SEN, the phenomenology is simplified and the models available are selected according to the objectives of the calculation. By explaining why some phenomena do not have a significant influence on the safety functions, or why others can be represented in a simplified way according to their effects or their extension over time and/or space, the qualitative analysis contributes to defining and justifying the repository’s representation in the SEN.

The models and parameters adopted for the calculation are justified particularly with respect to the uncertainties of knowledge since the purpose of the SEN should be to cover the uncertainties by "pessimistic" choices. For the elements of the calculation model which depend only indirectly, or to a second degree, on the retained geological context – for example, the properties of the engineered components – analysis of the FEPs allows justifying the reason why the parameters and models having a more or less important degree of conservatism according to the uncertainties concerning their characteristics and the associated processes. For example, for a component subject to the influence of physico-chemical processes capable of causing its rapid degradation, it may prove to be necessary to adopt only minimum or short-term performances. The choice of parameters and models describing more or less pessimistically the behaviour of the components is therefore justified by means of the analysis of the FEPs. For the geological medium, if no defined site was identified, the same analysis cannot be conducted: the uncertainties on the massif’s evolution and its own specific characteristics are generally negligible or at most on the same order of magnitude as the uncertainty due to the absence of a site. The descriptive parameters adopted for the granite rock or to describe the hydrogeological context have a “test” value and are selected among the values considered representative. They do not pretend to cover the uncertainties about the FEPs.

If a FEP refers to an altered evolution, that is, that it is considered to be highly unlikely and that it may lead to an abnormal condition, the analysis consists of identifying whether it is taken into account in one of the altered evolution scenarios (SEA) given in the Dossier 2005 Granite. At this stage, we did not try to be exhaustive in the representation of abnormal condition situations because the SEAchoice focused on scenarios customarily taken into account in this type of exercise. Scenarios leading to the failure of each of the “barriers” identified by the Basic Safety Rule RFS III.2.f were adopted. The retained list is also inspired by the Basic Safety Rule’s recommendations. These scenarios emphasize the role of the packages in the repository’s safety (“package failure” scenario), the role of the seals (“seal or plug failure”), the role of the geological medium, and particularly the characterization of the average fracture (“fracturing characterization error” scenario).

5.2 Main results of the analysis of the FEPs base- Elaboration and justification of the scenarios.

5.2.1 Classification of the FEPs

The presentation is organised into major repository compartments, which are also, in fact, the compartments of the SEN's calculation model. For each compartment the corresponding FEPs are indicated without necessarily mentioning each process individually. A few phenomenological elements of understanding derived from the PARS are mentioned to support whether taking into acccount or not the FEPs in question in the SEN or the SEA.

5.2.2 Generic disposal principles

5.2.2.1 Inventory of the primary packages

Prior to safety analysis, the FEPs recommend specifying the nature of the waste, its qualitative and quantitative inventory and how to take the waste into account.

The nature of the waste is a key factor in the analyses to be performed. In fact, the analysis must take into account the period during which the radionuclides are liable to have an impact on health, this duration being related in turn to their radioactive decay. For this reason, the safety analysis is conducted over a million years; this period is in part conventional, but corresponds to a significant decay of all the short- and intermediate-lived elements. Although a large part of the radioactivity, particularly from the B and C wastes, decays during the first few centuries or millenia, the RFS specifies paying particular attention to the first 10 000 years of evolution.

The inventory of the waste to be disposed of includes waste of diverse nature and is based on hypotheses concerning the nuclear fuel reprocessing strategy in France. Various scenarios related to this downstream from the fuel's cycle are treated in order to cover a large-scale set of possible objects to be disposed of (vitrified waste of different thermal value, spent fuel not reprocessed). These scenarios are described in the inventory model [ix]

For vitrified waste, the inventory distinguishes 11 families distributed in five C waste reference packages (C0 to C4) characterised by their radiological inventory. A possible disposal of spent fuel is based particularly on scenario S2, which considers a separate inventory for the UOX fuel (CU1) and the MOX fuel (CU2). At this stage in the analysis, the CU3 fuel is not retained. The B waste inventory combines 46 B waste families. The variety of B waste led to defining eight reference packages characterised by their radiological inventory detailed in the MID: activated waste, bituminous waste, diverse technological waste, fuel structure waste cemented and compacted, miscellaneous waste, sources, waste containing radium [ix]. This inventory allows defining their radiological, chemical, thermal characteristics such as described in the MID. These characteristics are then taken into account during the definition of the disposal concepts.

Considering the objectives of the Dossier 2005 Granite, whose goal is not to exhaustively assess safety over the entire inventory within a specific site, only certain reference packages of each family, the CU2 packages, the C2 packages, and the B and B5.2 packages, are retained for the quantitative processing of the safety scenarios. This choice does not pretend to be exhaustive from all viewpoints, but is sufficient to deal with all the problematics linked to the disposal of the various wastes.

The C2 reference packages retained to carry out the quantitative safety assessments represent the most important C waste category. In addition, the radiological activity per package is greater than or on the same order of magnitude as that of the other C waste reference packages. Similarly, if the total radiological activity of the CU2 packages (MOX) is less than approximately an order of magnitude compared to that of the CU1 packages, the CU2 reference packages having a higher labile fraction are retained. The choice of two reference packages (B2 and B5.2.) with different volume and radiological inventory allows covering the main problematics linked to B waste.

5.2.2.2 Inventory of radionuclides and properties

The radiological waste inventory is identified on a "family by family" basis. Within each family the various physico-chemical sub-assemblies of the waste are described based on the producers statements, which give access not only to the contents in radionuclides, but also to the geometric and physical description, as well as the description of the chemical contents.

The nominal activity value retained for the assessments is generally the highest value among those encountered in the nominal inventories of the various families grouped inside reference packages. Because of calculation capacities, a selection has to be made among the 144 radionuclides having a half-life greater than 6 months accounted for in the inventory model. This selection is described in chapter 6. The characteristics of the radionuclides (half-life, radioactive relationship) are derived from the JEFF 2.2 database of the OECD/NEA [xxxvi].

A radionuclide's behaviour depends on the physico-chemical characteristics of the chemical element under which the radionuclide is an isotope (speciation in aqueous phase, solubility, precipitation/sorption capacity, etc.). Since no site is defined, the geochemical behaviour data derived from internationally conducted studies were retained. Specifically for each radionuclide, the transfer depends on the solubility of the radionuclides in the water from the granite and the possibilities of sorption on the clay materials (clay buffer, backfill), the cement materials (B waste containers) or the granite's fractures.

Some radionuclides may be present not only in a dissolved form, but also in a gaseous and volatile form. Among the gaseous elements coming from activation or fission products, it is estimated in an initial approach that only carbon 14 could form long-lived gaseous compounds whose chemical stability is sufficient for them to be capable of migrating in this form: it may, in particular, migrate in the form of methane. These gaseous radionuclides represent only low quantities and are not relevant to design a disposal concept. Therefore, no detailed study was conducted at this stage.

5.2.2.3 Host formation

The FEPs also require specifying as an introduction to safety analysis the context and the principles of a repository in the host rock.

The Dossier 2005 treats generic sites for which it identifies the relevant characteristics with respect to safety. Thus, it prepares for possible field reconnaissances so that the associated technical and scientific approaches can be controlled.

A reconnaissance principle stage by stage of the granite in which the repository is installed is retained. This approach includes characterization in the laboratory phase so that models can be created to allow understanding the arrangement of the structures in the massif and then during the disposal phase (see subsection 3.3.2).

Due to the lack of a site, the parameters retained for the repository performances study are not defined and can be retained among those describing the range of granites available in France, such as they were identified in the typological analysis. Even in the situation of a repository studied on a defined site, some parameters would remain uncertain due to the capacity limits of the characterization operations, like on any site. It is not possible at the stage of the Dossier 2005 to distinguish one source of uncertainty from another. An attempt was made to identify the main variations in the parameters describing the granite in order to define sets of values and the sensitivity studies allowing the main challenges to be determined; these variations tend to cover on account of their principle both the lack of a site and the limits of the characterization capacities.

In this context, the main uncertainty hinges on the variability of the geological, hydrogeological and hydrogeochemical context of the French granite massifs. The variability of the French granite massifs were taken into account by handling in the SEN three site morpho-structural configurations (inclined plane, dome and depression). These three models constitute the base of the quantitative assessments of the safety scenarios. In addition to the different morpho-structural arrangements, which lead to specific water flow modes (gradients, position of the low hydraulic points), the models differ with respect to the different fracturation organisation modes in the small fracturation in the vicinity of the disposal cells, as well as at the scale of the massif. The lithology and the mineralogy of the massifs, represented indirectly in the models through the hydraulic and geochemical properties, can also vary. These three site models, M1, M2, M3, are described in the introduction of chapter 6.

An important choice was to direct the analysis essentially on the "near field", that is, at the level of the repository modules. Thus, the host formation outside the modules, the surrounding formations, as well as the biosphere, are not in principle included for the elaboration of the FEP base. However, "external" events and processes, which can influence the conditions inside the modules, were included in the database. A number of "limit conditions" linked to the interface between the near field and the far field are set through the site models. These conditions would be specified within the scope of a site reconnaissance.

Considering the objectives of the dossier 2005, the biosphere is not considered in the studies. In fact, a dose calculation would not have any meaning in a generic type studies context.

5.2.2.4 General architecture of the repository

In the case of the granite medium, the design is based on the adaptation of the repository to the fractures of the medium. In accordance with the Basic Safety Rule (RFS.III.2.f), this principle is applied at the scale of the repository as a whole and then at the scale of the repository modules and the disposal cells: "*The repository should be located within a host block without any large-scale faults since these faults could form privileged hydraulic flow channels. The repository modules should be constructed away from the mean fracture of the granite, although the latter can still be traversed by access structures*". This architectural adaptation aims at forming around modules (that is, the fractured zones of the repository) confinement “volumes" in the form of barely fractured "granite blocks" limited by "average-size" fractures, which are possibly more water-conducting. It takes into account the various fracture scales (see Figure 5.2-1).

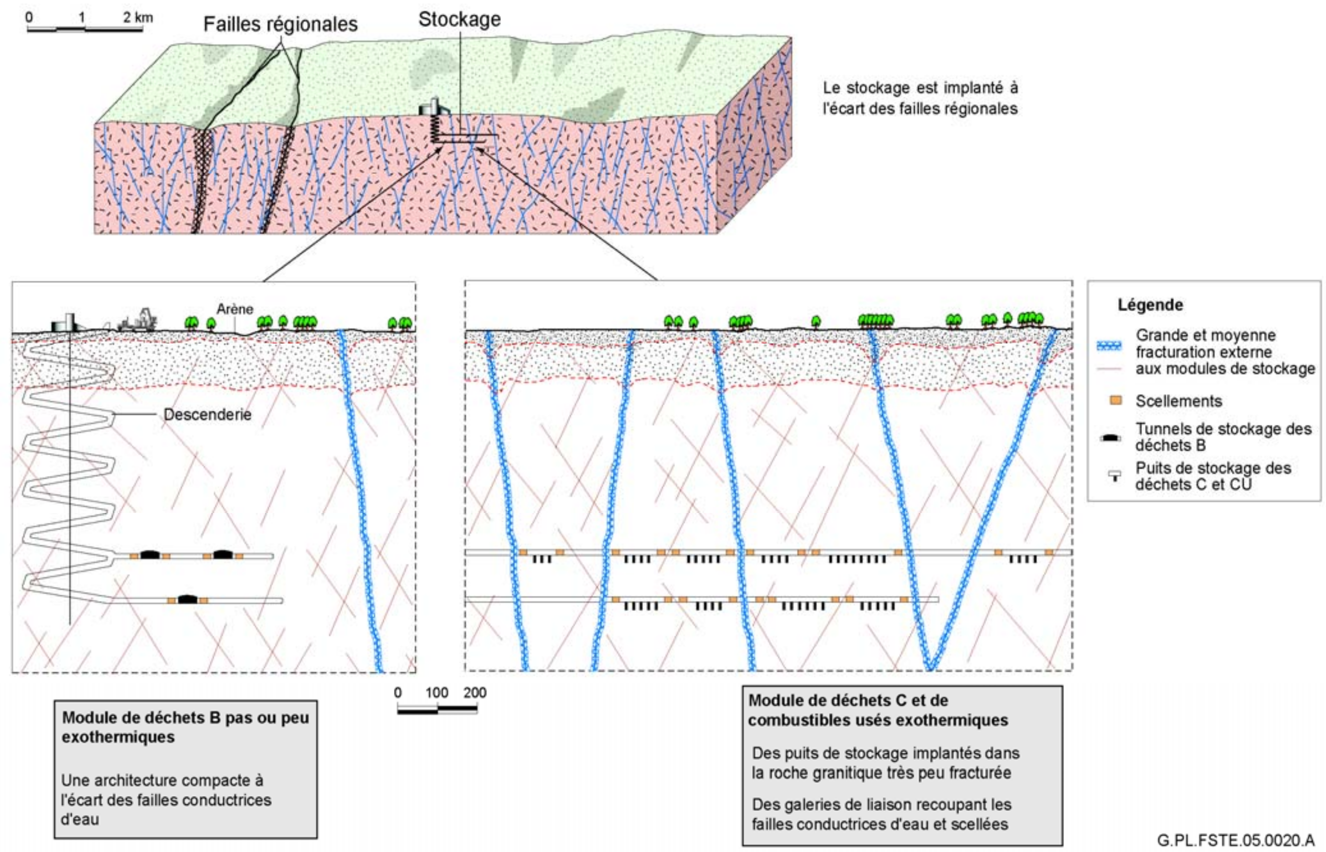

Figure 5.2-1 Adaptation of the repository's architecture to the granite's fracturation

The disposal concepts were presented in chapters 2 and 3. Here the modular design of the repository and its organisation into the compartmentalised repository zones, particularly according to the waste's thermicity, could be highlighted.

The scenarios consider for each type of waste (B, C and CU) a reference module assumed to be far enough away from the large-scale faults delimiting the layout of the repository and installed in blocks delimited by "intermediate" fractures and faults. The modules are assumed far enough away from one another to neglect chemical interactions between them.

In the normal evolution scenario, the components contributing to the various safety functions are represented, namely, the containers, the engineered barrier, the cells plugs, as well as the backfills and the seals installed in the drifts located inside the modules and the granite in the near field. These components are retained in the associated safety model specifying their representation and the transport mode of the radionuclides. It should be noted that the access shaft zone and the connecting drift system located outside these blocks above the conductive moderate fractures and faults, which could form a transfer path for the radionuclides to reach the surface and would require sealing, is not dealt with in the scenario.

This analysis and the altered evolution scenarios deal only with the repository's post-closure phase. Generally, the excavation, operation and reversibility phases can have an influence on certain phenomena to be considered with respect to the repository's long-term safety. This influence is to be taken into account when the models and the parameters are considered for selection for the safety calculation. The repository zones dedicated to each type of waste (B, C, CU) are designed according to a reversibility logic [xxxix] A long operation – observation period linked to a reversibility context, for example, may lead to an intensification of certain degradation processes. However, in a generic context, they are generally considered negligible. The operation period creates, in particular, a desaturation and oxidising conditions temporarily, but they are rapidly resorbed after the repository is closed. These aspects are developed below, component by component.

5.2.3 Spent fuel repository zone

5.2.3.1 Spent fuel cells and modules

The sizing of the spent fuel repository modules depends, first of all, on their thermal properties. Due to their number, in the scenario S2, the various repository modules must be installed in large volumes of granite which may be interconnected by small and medium granite fracturation ranging from metric to hectometric in scale. Considering the volumes and their various thermicity, the disposal of spent fuel is more demanding in terms of dimensions than that of C waste [viii].

Without a designated site, the small and medium fracturation of the near-field granite is not known. To process the scenarios, a reference module is taken. This module is installed in granite blocks with a low permeability delimited by water conductive fractures (see Figure 5.2-2). A buffer zone of undisturbed rock with respect to these fractures would have to be defined for an installation on a real site: it would take into account the uncertainties on the exact location of the faults, as well as on the extent of hydraulic connection between these faults and the small fracturation in the near field. Within the context of a generic study, a buffer zone of undisturbed rock of 100 metres was considered in an initial approach. The modules are installed according to the stress mode and hydraulic conditions of the site.

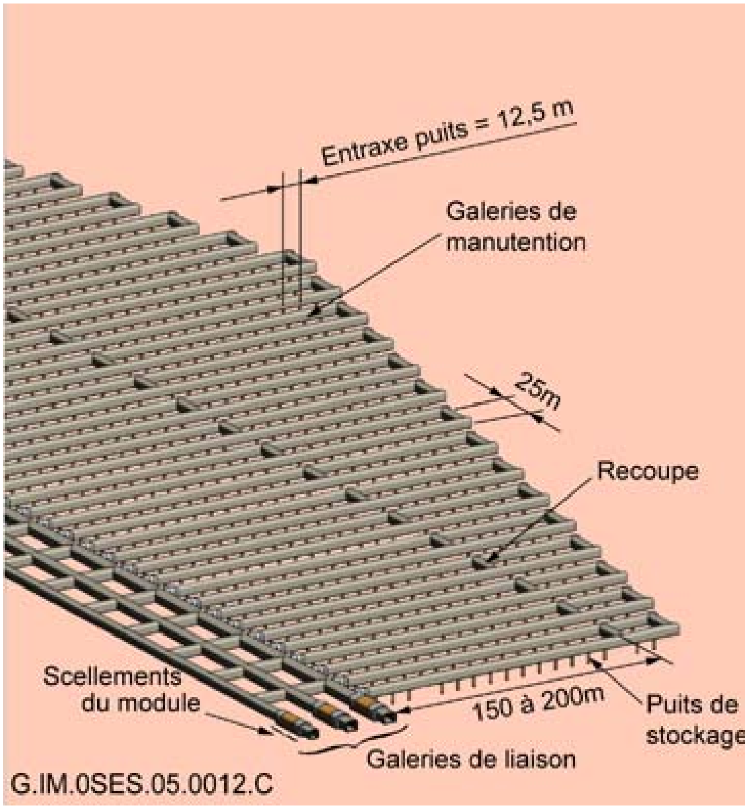

Figure 5.2-2 Schematic representation of a spent fuel repository module

The disposal cells are installed after determining by measurement the characteristics of the small and medium fracturation in the cell wall. It seemed worthwhile to make use of the calculation tools in the fractured medium ("DFN calculations"), which are able to generate various small and medium fracturation schemes in the near field in order to analyse the installation possibilities. The characteristics of this fracturation are treated in the form of a sensitivity analysis; chapter 6 gives a more detailed presentation of the tools used.

Considering the number of cells to be characterised, it is not possible to exclude a malfunction in the process characterising the properties of the small fracturation. This malfunction may be linked to a control failure either in the execution process of the hydraulic tests, or in that of the geological readings and geophysical measurements [x] This control failure can lead to an erroneous assessment of the fracture's characteristics and result in the choice of a cell which should have been rejected for the installation of a disposal borehole. The SEA"fracturing characterization error" covers this possibility. It considers that 10% of the disposal borehole installations which should have been rejected were not: according to SKB estimations, this corresponds to the percentage of cells which could be rejected in a normal characterization situation as the process advances. The location of these cells is hydraulically penalising inside the module. In addition, to evaluate the impact of this fracturing recognition error on the transfer of radionuclides, the scenario considers that a defective container (see next point) is located in one of the disposal shafts which should have been rejected.

The seals installed in the vicinity of the conductive fractures delimiting the blocks in which the modules are installed consist of a clay core of low permeability enclosed between two concrete retaining plugs. The retaining plugs which essentially ensure the mechanical strength of the clay core are represented by a conventional concrete.

Only the components of the cell which contribute to the various safety functions are represented in the models, that is, the containers, the fuel assemblies, the engineered barrier and the cell plugs.

5.2.3.2 Spent fuel container

The FEPs treat essentially the container's chemical evolution, and two types of FEPs can be distinguished: those related to the chemical processes affecting the CU container under normal conditions, and those related to the processes caused by the container's possible rupture. They are successively dealt with.

- **Characteristics of the containers**

The selected spent fuel container is made of copper (see Figure 5.2-3) and sized so that it guarantees a very long leak tightness duration up to several hundred thousand years [xxxiv]. The SKB choice of copper as the reference material is based on its resistance to corrosion [xxxvii]. Copper is a metal which is hardly affected by corrosion under the disposal conditions in a reducing medium; this implies, however, the absence of potentially aggressive compounds (chlorides, sulphur or nitrogen compounds).

Figure 5.2-3 Spent fuel container with copper envelope and insert (SKB source)

In addition, the choice of a copper container limits the formation of products liable to uncontrollably alter the transfer and retention properties of the engineered barriers and the granite's fractures [xii].

The geochemical conditions expected (pH, Eh, salinity) at the depth of the repository in a granite context are a priori favourable with respect to the copper's resistance to corrosion. Adding a swelling clay buffer around the containers is a factor which contributes to homogenising the chemical conditions and providing an additional robust element against possible local variations in the medium's chemistry.

The container's mechanical resistance is greater than 12 MPa, a value defined from the hydrostatic pressure and the swelling pressure of the clay buffer. It takes into account the addition of a cast iron insert inside the container to enhance rigidity. This value is much less than the elastic resistance (> 200 MPa) of the insert and the elastic resistance of the copper envelope (45 MPa). For this reason, the possible rupture of the containers due to a mechanical source is not retained in the safety analysis.

Quality control would be ensured during the manufacturing of the container (production, welding) [viii]. A possible quality control failure is, however, to be taken into account in the safety analysis.

In a normal situation, one defective container per module is envisioned (located in the vicinity of a transmissive fracture in order to guarantee the penalising character of the retained position) following an initial undetected manufacturing failure of the container; this failure is materialised by a hole of 5 mm^2 which subsequently evolves (see "chemistry after the container's rupture" below).

The SEA"package failure" covers a durable quality control failure leading to a greater number of defective containers (5 packages of a module). The defective package series is assumed to be located in a penalising fashion inside the module, that is, in handling drifts having the highest water flow rates leaving the cell.

- **Processes affecting spent fuel containers**

Heat transfer

The container's temperature under disposal conditions depends on the exothermicity of the fuel assemblies, the thermal conductivity of the bentonite and the granite, and the initial temperature of the rock. It can affect the chemical processes (corrosion) and the hydraulic processes (resaturation).

Thermal dimensioning limits by design the maximum temperature at the package's surface to temperatures less than 100°C in order to avoid the vaporisation-condensation processes which could cause corrosive salts to migrate to the copper container. The distance between the shafts takes into account this limit and the thermal properties of the granite and the containers (storage period of the spent fuel before disposal). The sizing calculations were carried out for three types of granites (of different heat transfer gradients) defined from the typological analysis of the heat transfer characteristics of the granite massifs ([vii] and [viii]). The maximum allowable temperature in the container wall was lowered to 90°C in order to take into account the uncertainties related to the heat transfer models (input parameters, saturation rate of the engineered barrier, equivalent conductivity of the functional clearance, uncertainties specific to the calculation code).

Considering these elements, it was decided not to represent in the calculation model the special effects from the heat transfer on the container.

Chemistry – Corrosion model

The copper container may be corroded a priori in the presence of oxidising fluids. It was estimated in Sweden that the pyrite contained in the MX-80 bentonite (approx. 0.3 % in mass) is sufficient in quantity to consume the oxygen initially present in the disposal shaft or the oxygen possibly present in oxidising waters. In a normal situation, the transient RedOx is considered for this reason to be of short duration compared to the times considered under long-term safety. The oxidising phase is not taken into account since the medium is considered to be a reducing medium buffered by granite waters.

The container is sized, on the other hand, to take into account aqueous corrosion in a reducing medium. Under anoxic conditions, the aqueous chemical compounds potentially corrosive for copper are the chlorides and sulphides. For a pH between 7 and 11 (domain embodying granite waters buffered by bentonite) and chloride contents less than 100 $g.l^{-1}$, there is no significant corrosive action due to the chlorides.

In the presence of sulphides, copper can corrode with the formation of products such as Cu_2S or CuS. Estimations were made in Sweden in order to evaluate the corrosion rate of copper under disposal conditions. They show that for high sulphide concentration ratios (5 mg/l) and large water flows (0.1 m^3/m^2/year), the corrosion rate is low enough so that even if the engineered barrier's role is not taken into account a 50 mm thick container will remain leak-tight for more than one million years [xxxvii].

An examination of the FEPs covering corrosion leads to imagining a number of other factors which could influence the corrosion rate:

– Bacteria which can reduce sulphates into sulphides, an aggressive form for the corrosion of a copper container. The microbial activity of sulphato-reducing bacteria was examined, but their influence is considered negligible because above a bentonite density of 2 g/cm^3 bacteria cannot survive. Therefore, the specific corrosive action of micro organisms is not retained;
– Localised corrosion, which is possible, namely, in connection with a heterogeneous corrosion due to local variations in physico-chemical conditions. Nevertheless, it would be limited to the oxidising phase;

- Corrosion under stress, which has no expected significant effect;
- Products resulting from radiolysis, thermal expansion and weakening due to hydrogen. Preliminary evaluations were carried out on these phenomena, which show that they are negligible. The systematic examination by SKB on the possible causes of a container's failure concludes that the differential thermal expansion of the container and the insert did not represent a risk important enough to be taken into account in safety analyses and that the radiolysis of the air and irradiation over long periods cannot cause mechanical damage [xxxviii].

It should also be noted that the copper container provides a radiological protection. Its corrosion under the effect of radiolytic phenomena on the outside surface is negligible.

In conclusion, the medium is very rapidly considered to be a reducing medium buffered by granite waters. The influence of aggressive chemical compounds is either negligible or taken into account in corrosion rates estimated to be very low. For this reason, the containers are represented leak-tight over the entire analysis period (that is, 1 million years), and the loss of leak tightness resulting from corrosion is not considered.

Localised failures considered in the SEN, as well as more frequent failures based on the SEA"package failure", allow taking into account uncertainties, namely on the manufacture and quality control of the containers, and all the phenomena mentioned above.

5.2.3.3 Container behaviour in case of rupture

In case of container rupture, the corrosion of the cast iron insert will lead to the formation of hydrogen and magnetite. It can lead to an accumulation of gas and impact on various processes, such as the dissolving of the matrix, the incoming water flows and the transfer of radionuclides. It contributes to imposing reducing conditions, in addition to the granite medium itself.

The impact of the corrosion gases on the transfer of radionuclides is treated in the section dedicated to the engineered barrier. The release model retained for the spent fuel assemblies does not take into account a possible inhibiting effect by hydrogen, which is conservative.

SKB has carried out several modellings indicating that the water flow arriving in the container depends on the form and extent of the leak-tightness failure [xxxviii]. This favourable effect remains nevertheless hard to quantify. In a defective container situation due to a quality control failure, a "partial" failure was retained at 150 years (after the exit of the assemblies from the reactors) which is expressed by a hole of 5mm^2 and initiates the release of radionuclides. Total failure is considered only after 20 000 years.

It should be noted that according to SKB evaluations, plastic deformation linked to the pressure increase caused by the formation of magnetite between the insert and the copper lid, in particular, leads to a damaging of the copper and an overall loss of the container's leak tightness after 200 000 to 500 000 years [xxxviii]. The adopted model therefore covers this effect.

Finally, the effects related to the possible retention by sorption/coprecipitation of the radionuclides by the container's corrosion products mentioned by the FEPs are not considered: this effect, a priori favourable, remains hardly quantifiable. In a conservative fashion, no role by sorption/co-precipitation is attributed to the corrosion products in the normal evolution scenario.

5.2.3.4 Assemblies behaviour in case of container rupture

In case the container fails, the spent fuel is liable to come in contact with water and thus initiate a degradation process which is described by resorting to the FEPs.

- **Assemblies characteristics**

The spent fuel assemblies are distinguished by a heterogeneous distribution of the radionuclides which is liable to influence the radionuclides release model. The spent fuel assembly is made up of zircaloy claddings and a UOx matrix. The UOx matrix contains the largest proportion of the radiological inventory. In addition to the rods (zircaloy claddings containing fuel pellets), the fuel assemblies include structure elements which can also contain radionuclides. Empty spaces and residual clearances persist in the containers. The empty spaces and residual clearances can be the place where gases accumulate, particularly the helium coming from the alpha decay of actinides, or a place where dissolution/precipitation is favoured after a rupture of the container. The release models take into account this heterogeneity of the fuel's structure by proposing models adapted to each sub-assembly [xv].

The FEPs evoke, in addition, various spent fuel quality control or handling failure situations: burnup fraction measurement failure, undetected presence of damaged assemblies, or an accident during a handling operation. Quality control procedures are required to ensure that the packages correspond to the repository's specifications. This type of measurement is currently applied in basic nuclear installations and does not raise any problems. No failure linked to a quality control failure of the assemblies themselves was retained at the generic stage of the Dossier 2005 Granite.

- **Processes affecting spent fuel assemblies**

The analysis of the evolution in the spent fuel disposal situation leads to distinguishing the FEPs related to the various thermal, hydraulic, mechanical, chemical and radiological processes. The release model of the radionuclides and their transfer are included in the chemical processes category.

Due to the fact that the data related to the packages hardly depend on the surrounding medium since the medium remains diffusive and within the controlled pH and redox potential ranges, the release model of the CU fuel assemblies is common with that retained for the studies on the repository in the clay medium [xv].

These models are specific with respect to the location of the radionuclides in the fuel. Thus, the alpha radiolytic dissolution model for the matrix, the labile release linked to accelerated diffusion by alpha self-irradiation (D3AI), the initial labile fraction and the release by corrosion of the metallic elements are successively described.

The release model of the radionuclides released by the fuel matrix retained in a normal situation is based on the α irradiation of the matrix. It takes into account the radiolytic dissolution of the matrix and the presence of radionuclides in the free volumes, the grain boundaries and the plutonium clusters of the MOX pellets in the form of a labile fraction. To this fraction of initial labile activity, the fraction corresponding to the release by D3AI in the grain boundaries should be added. This phenomenon results from alpha decay, which can lead to migration by diffusion of the radionuclides from the grains of the matrix to the grain boundaries of grains as soon as they leave the reactor. The release of other radionuclides located in the fuel is congruent with the matrix's dissolution.

Whether the antagonistic interaction between oxiding radiolysis and hydrogen generation (RedOx potential conditions) on the dissolution conditions of the fuel pellets should be taken into account or not has provoked discussions at the international level. If the effects from the accumulation of hydrogen are taken into account, then the phenomenon of radiolytic dissolution could be neglected. This favourable effect is not taken into account in the current model of the SEN.

The labile fraction retained in the reference calculation is an evaluated value for an average burnup fraction of the CU2 fuel. In order to cover the uncertainties related to the part of labile activity from spent fuel, a conservative approach consisted of assigning to the defective package at 150 years the fraction of labile activity associated with the part of activity calculated by the model after 10 000 years of accumulation in the boundaries after diffusion.

The release also depends on the separation condition of the spent fuel. The release model of the radionuclides released by the fuel matrix retained as the reference considers a separation rate based on observations made on the fuel leaving the reactor [xv]. The uncertainties on the evolution of this parameter under disposal conditions (namely, the impact of the generation of gases such as helium, the corrosion products of the container) are covered in a first approach by this value due to the fact that the small free space forces deformations and limits the possibilities of rupture. The sensitivity study on this point was not retained because the dossier 2005 gave priority to sensitivity studies on the characteristics of the geological medium.

Other processes evoked by the FEPs (precipitation of actinides in the empty spaces, influence of magnetite derived from corrosion, influence of microorganisms, etc.) are either favourable phenomena not taken into account, or negligible phenomena in the retained architectures and granite context. This is the case, for example, for microorganismes, which cannot develop in large quantities in the compacted bentonite or in the vicinity of the packages.

To sinn up, the treatment of the normal evolution scenario retains a dissolution model of the fuel matrix under the effect of radiolysis (the so-called radiolytic model) leading to a progressive release of the radionuclides located within the matrix. It integrates in a penalising fashion the effect of diffusion accelerated by alpha self-irradiation (D3AI). The dissolution model retained in the reference calculation is a conservative model which leads to releases in approximately 5 000 years, that is, several orders of magnitude below what is generally retained at the international level.

Incidentally, the arrival of water in the containers leads to the corrosion of the zircaloy claddings and the other metallic parts of the assemblies structures. The zircaloy claddings are subjected to uniform corrosion and possibly to localised corrosion responsible for the release of radionuclides. The corrosion of the irradiated zircaloy is estimated on the order of 3 to 5 $nm.year^{-1}$ at 90°C. The release of radionuclides is assumed congruent with corrosion. Radionuclides present within the zirconia are considered to be labile ([xii] and [xv]).

In a normal situation, the zircaloy claddings are not considered to provide a protection against the dissolving of the matrix, which is penalising.

Gaseous and volatile radionuclides which are located in free volumes (grain boundaries, rim's pores, gap, plenum and fractures) are labile. Some radionuclides can be released in gaseous form, particularly krypton 85 and carbon 14. For reasons already mentioned above, gaseous radionuclides are not taken into account.

- **Criticality**

The mechanical evolution within the container is limited in a normal situation, where it is isolated from water. On the other hand, if the container should fail, the corrosion of the insert and the assembly can cause mechanical deformations. An evaluation made under the research work carried out on the repository in the clay medium on a container having approximately the same geometric dimensions shows that the vast majority of the conceivable configurations are sub-critical and that those which correspond to critical states are highly improbable; they correspond to a crushing of the insert while the clearance between the rods within the assemblies is maintained [xxxix]. It should be noted that if the corrosion is greater in the insert than in the copper container, it should lead to, on the contrary, a spreading apart of the assemblies due to the expansion of corrosion products of the cast iron.

A post-closure criticality accident is, therefore, not retained at this stage in the studies either in the SEN or in the SEA.

5.2.3.5 The engineered barrier

The classification of the FEPs leads to distinguishing between those referring to the evolution of the engineered barrier subjected to heat transfer processes and to various interactions with the container or over-pack, and those refening to possible interactions with the concrete slab poured in the shaft bottom.

- **Characteristics of the engineered barrier**

The resaturation of the engineered barrier made of bentonite creates a swelling pressure on the rock and the container (see Figure 5.2-4). Maintaining this swelling pressure ensures a low hydraulic conductivity of the bentonite. The swelling pressure in the disposal situation should reach approximately 7 MPa, a reference value defined from the Swedish studies [xxxviii]. The low permeability of the engineered barrier limits the water flow in the vicinity of the container and thus guarantees the transport of the radionuclides by diffusion, and a better chemical homogeneity.

The important feedback coming from abroad on bentonite (FEBEX in Switzerland, Prototype Repository in Sweden, and the TBT experiment of Andra in Sweden) allows characterising its behaviour under disposal conditions and confirms that it presents the properties expected in terms of permeability and plasticity [x].

The modellings made in Sweden indicate that the resaturation duration can range from around ten years to several tens of years with variations according to the cell's situation with respect to the fractures in the massif and the architectures considered. The resaturation of the disposal shafts will take place for most of them during the operating phase of the repository or a little after.

The engineered barrier is represented as a continuous homogeneous porous medium This results from both its rapid resaturation, which fills the empty spaces between the bentonite bricks, and the limited extension of the chemical disturbances (which can extend over several centimetres for the iron-clay disturbance) – but do not place in doubt the plug's sizing.

The FEPs evoke various possibilities of error in the placement of the engineered barrier: erroneous supply, bad placement, contamination by organic substances. At the stage of the dossier 2005, the decision was taken to not treat this type of failure, which depends on the quality control performed during the installation. It should be noted that for spent fuel a concidence of the initial failure of the container occurring in the same cell is highly unlikely.

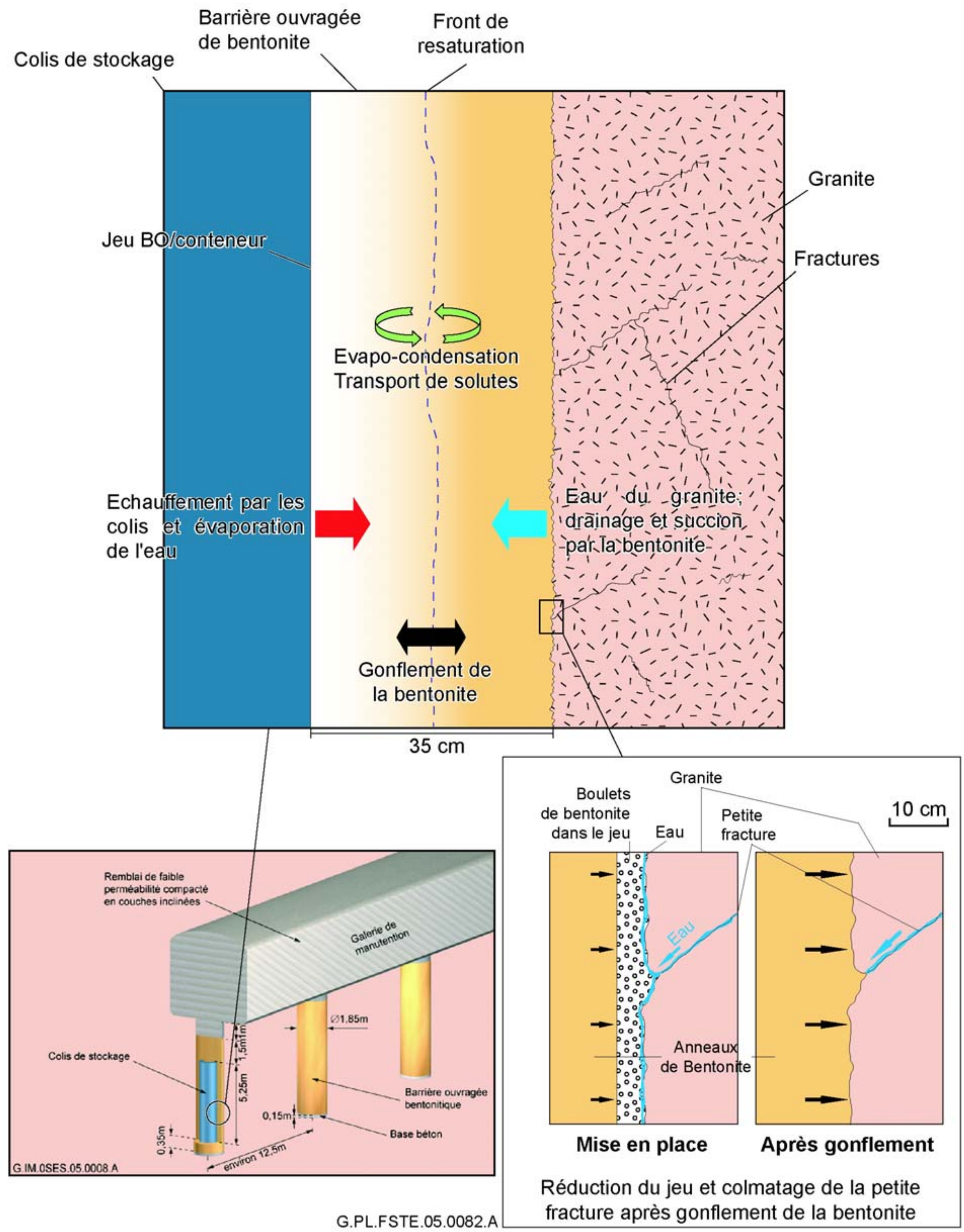

Figure 5.2-4 *Swelling phenomenology of the engineered barrier*

- **Processes affecting the engineered barrier**

Heat transfer

The heat transfer dimensioning of the repository is defined so that the maximum temperature within the engineered barrier is around 90°C. Considering the temperature criteria retained, heat flow does not cause a major mineralogical transformation on the bentonite, which preserves its properties (in particular, the precipitation of calcite, sulfates or silica is not taken into consideration by the modellings applied by SKB): therefore, this disturbance is not represented.

Mechanical

The migration of the gases generated by the corrosion within the bentonite must be evaluated, because it can induce mechanical interactions, such as the formation of fissures and have an impact on the properties of the bentonite, particularly its permeability. This point hardly concerns the spent disposal fuel cells in which the production of gas is limited for the most part to the corrosion of the insert, but it is also treated by anticipation so as to create the model representing the C waste disposal cells.

The hydrogen generated by corrosion of the cast iron or the steels is accumulated, first, between the metal and the engineered barrier. The continuous generation of gas in a confined space raises the pressure until it penetrates by a two-phase flow into the bentonite. This process is a temporary process, which does not cause an irreversible damage on the bentonite due to its plasticity and its swelling, and slow pressure raising kinetics. Therefore, the mechanical damage on the engineered barrier is not represented.

As a result, the effect of gas on the bentonite's characteristics retained in the normal evolution scenario is not retained.

Chemistry

Bentonite's alteration may lead to a reduction of its swelling capacity and/or its plasticity. In the long term, bentonite's stability under the repository's temperature conditions can be affected by ionic exchanges with the water from the granite. Models made in Sweden allowed evaluating the kinetics of the ionic exchanges for the water compositions and the characteristic hydraulic properties of the Swedish geological context. In the most unfavourable case, the swelling pressure can decrease from 7-8 MPa to 4-5 MPa in 100 000 years, which would not lead to a loss in the mechanical or hydraulic functions of the engineered barrier [xxxviii]. In a normal situation, this disturbance is therefore not represented. In a generic context, the parameters considered in the Swedish modellings can be considered in a first approach as indicators of the phenomena to be taken into account in the French context, but particular factors such as CO_2 pressure should be taken into account.

Interactions between the bentonite and the metallic products can affect the swelling, the sorption and the hydraulic properties of the bentonite. The studies conducted within the research work on the repository in the clay medium show that the iron remains to a great extent blocked at the interface with the bentonite. However, the copper possibly released by the container or the Fe^{2+} ions possibly coming from the insert can diffuse over the bentonite's entire thickness and compete with the radionuclides for the sorption sites.

The influence of the concrete slab present at the disposal cell end is considered to be negligible due to the limited cement mass considered, which may not induce a considerable chemical impact on the bentonite. A slab made of a low pH concrete remains an option, as well as any other material inert with respect to the bentonite, if necessary. It is not represented in a normal situation.

- **Transfer of radionuclides within the engineered barrier**

Radionuclides are transferred within the bentonite by aqueous diffusion because of the very low permeability of bentonite. The transfer can be delayed by sorption phenomena. The uncertainties persisting on the thermal behaviour of the radionuclides in the bentonites prevent distinguishing their behaviour between the various failure cases during or after the thermal phase. The uncertainties concerning the values of Kd embody these indeterminations for moderate temperature fields [xl]. In addition, the influence of temperature on transport cannot be generically taken into account in the granite massif itself. Therefore, it was decided by similarity to neglect it, including in the bentonite.

The concentration of radionuclides in solution in the bentonite is limited by their solubility and a possible sorption/precipitation on the compounds resulting from the dissolution of the spent fuel and the iron's corrosion products. To represent these phenomena, parameters common with those used for the bentonite in the studies conducted on the repository in clay are exploited. The solubilities are, however, specific, because they are dominated by the composition of the granite waters; they come from representative data sets used by SKB.

Some phenomena are liable to accelerate the transport of radionuclides, particularly the formation of colloids coming from the corrosion products of the container and the insert. Not very many colloids should come from the copper container. The mobility of the colloids resulting from the corrosion products of the insert should be limited. The formation of intrinsic colloids based on polymerisation and condensation of the radionuclides cannot be excluded, but their thermodynamic stability outside their production zone is low. In a second line of defence and by design, the bentonite's density allows filtering the colloids coming from the primary waste packages and limiting their possible propagation outside of the CU disposal borehole. Therefore, this phenomenon is not taken into account to represent the transport in the engineered barrier.

In conclusion, after examining all the relevant FEPs, it is retained to represent the engineered barrier as a homogeneous, low permeability medium, and to neglect chemical disturbances.

5.2.3.6 Representation of the disposal shaft plugs

Cell plugs contribute to ensure a diffusive transport within the cell. The plugs are also designed to maintain chemical environmental conditions favourable for the packages in the cell (all phases), which means that they must limit the exchanges between the cell and the drift, particularly with respect to the species responsible for the disturbances (alkaline ions and species controlling the pH, or controlling the oxidoreduction conditions).

The clay core's low permeability is based on the development of a sufficient swelling pressure which allows a diffusive transport and a good clay/rock contact.

The plug is very similar in its evolution to the engineered barrier previously mentioned. Therefore, only the FEPs which are specific to the plug or which present a greater influence on the plug's safety function or on those of the engineered barrier are described in the rest of this subsection.

Insufficient mechanical pressure from the backfill (linked, for example, to a stacking or a bad placement) may prevent the backfill from providing sufficient mechanical support for the core to develop its swelling pressure.

Bad expansion linked to an insufficient swelling pressure of the plug's bentonite may cause a bad bentonite/granite contact. As a result, this bad interface has hydraulic properties greater than those expected and can thus constitute a preferential path for the transfer of radionuclides.

This situation is envisioned in the SEA concerning a C waste disposal cell plug failure (similar to that of the CU cells). The plug failure is represented by an imperfect interface between the bentonite and the granite with a centimetre thick zone (equivalent to that of the damaged zone) having a higher hydraulic conductivity (a value of 10^{-9} m . s^{-1} is retained based on ZEDEX feedback) [x].

In the particular case of spent fuel, it was decided to ignore plug failure, as radionuclide release could only occur in the event of container failure (double failure is unlikely to occur).

5.2.4 C waste repository zone

5.2.4.1 C waste cells and modules

From the viewpoint of safety analysis, the architectures retained for the disposal of the vitrified waste present hardly any difference with their equivalent for spent fuel; therefore, the elements already treated above are not rediscussed here.

Only the components which contribute to the various safety functions are represented in the normal evolution scenario, namely, for vitrified waste, the over-pack, the engineered barrier, and the plug of cells. The processing of the scenarios retained as reference comprises a shaft containing two C2 reference packages.

The lower thermicity of the C waste allows figurring out a greater flexibility in the excavation of the boreholes and adaptation to the fracturation than in the case of spent fuel. In a normal situation, a reference module installed in a block delimited by moderate conducting faults and fractures is considered. The module is installed at a distance far enough from these faults and fractures according to the site's hydrogeological context. The granite block has fractures of an order less than those delimiting it. The characteristics of this fracturation are treated as a sensitivity analysis.

Considering the number of cells to be characterised, a failure cannot be excluded from occurring in the characterization process of the properties of the small fracturation. This failure may be linked to an inspection failure either in the hydraulic test execution process or in that of the geological surveys and geophysical measurements [x]. The SEA"fracturing characterization error" covers the possibility of a failure in the characterization process of the properties of the small fracturation, which could lead to retaining cells which would have had to be rejected in order to install a disposal borehole. It considers only 10% of the disposal borehole characterised during the characterization process and which would have had to be rejected in a normal situation of characterization. The location of these cells is hydraulically penalising within the module.

5.2.4.2 C waste over-pack

The loss of confinement by the over-pack is liable to take place when the corrosion phenomena first in the oxiding medium and then in the reducing medium are cumulated over time causing either a mechanical rupture of the over-pack or a loss of local leak tightness by corrosion.

In addition, the over-pack's sizing confers to it a mechanical resistance of 12 MPa, allowing it to support the load due to the swelling of the engineered barrier.

The leak-tightness duration of the C waste enveloppe is retained arbitrarily penalising (1000 years). The FEPs envision that even after the over-pack ruptures the water flow coming in contact with the C waste and therefore the release of radionuclides are going to be limited by the size of the defects/holes/fissures. However, such an effect a priori favourable was not retained at this stage. It is assumed in a penalising fashion that the over-pack's rupture leads to a total loss of leak-tightness.

Quality control must be ensured during the manufacture of the container. These materials do not raise any major technological difficulties in their use, particularly concerning their weldability and the corrosion resistance of the welds. However, even though the manufacture and welding of the over-pack are inspected [viii] a possible quality control failure is taken into account. A failure of the containers as far as the weld is concerned is taken into account:

- In a normal situation, one defective over-pack is envisioned per module (located in the vicinity of a hydraulically pensalising position) subsequent to an initial defect of the container during its manufacture without the failure being detected;
- The SEA"package failure" covers a durable quality control failure leading to a larger number of defective containers (50 packages disposed of in neighbouring cells). The series of defective packages is assumed located in a hydraulically penalizing mode within the module.

- **Processes affecting the over-packs**

Heat transfer

Heat criteria at the repository entry and sizing allow beeing under favourable conditions. From the viewpoint of the safety analysis, it was not necessary to take into account special effects of heat on the vitrified waste over-packs for the same reasons as for the spent fuel containers.

Chemistry – Corrosion processes

In order to ensure leak tightness over 1 000 ans, the over-pack's sizing takes into account the FEPs related to the various corrosion processes imaginable from the moment it is emplaced in the borehole [xxxii].

The extent of the corrosion of the steel components depends essentially on the amount of oxygen available in the borehole during their closure : oxygen from the trapped air and oxygen dissolved in the resaturation waters in the case of waters coming from the most superficial parts of the granite. The generalised corrosion kinematics of carbon steel in an oxidising medium and in a humid atmosphere is around 100 $\mu m.year^{-1}$. On account of the heating and the phenomena of suction by the bentonite, the part of the disposal borehole closest to the package tends to be in a dry atmosphere, which is a corrosion limiting factor for the steels. In addition, the water passing through the engineered barrier before reaching the over-pack loses a major proportion of its oxidizing potential [xii]. Also, for the time periods envisioned for the operating phase, the corrosion of the various steel components is reduced. Dimensioning takes into account, however, the possibility of an oxidising phase at a rapid corrosion rate.

After the thermal phase, the chemical conditions in the disposal borehole become reductive again. The corrosion mechanism to be considered is essentially a generalised corrosion at a slower rate on the order of a few microns a year. Modellings indicate that the water flow will still be sufficient to supply the corrosion [xii].

The FEPs allow identifying several phenomena which may influence the corrosion processes:

- The microbial activity may catalyse the over-pack's corrosion. It contributes particularly to the reduction of sulphates into sulphides, which can react with the iron. The bacterial activity in the C waste disposal borehole should be limited, especially because of the presence of the engineered barrier, which fills the empty spaces and clearances and limits the inflow of nutritive substances;
- The localised corrosion of the unalloyed steel over-pack is possible under oxidising conditions, particularly in connection with the local variations of the physico-chemical conditions. Based on the knowledge acquired on unalloyed steels, the localised corrosion is not considered to be a determining mechanism as compared to generalised corrosion [xxxvii]. The localised corrosion process is not retained in the over-pack's corrosion model in the SEN. In case of the possibility – highly unlikely – of such a phenomenon, it would be covered by the premature rupture situation of an over-pack in the SEN or a defective series in the SEA"package defect";
- Corrosion of the over-pack under stress is judged negligible, including right above welds.

The granite's waters can be composed of species affecting corrosion (chlorides, etc.). However, the presence of the engineered barrier should allow buffering for any differences and placing the over-pack under favourable conditions.

In conclusion, the over-pack's sizing in terms of corrosion contains significant margins, which allow covering all the FEPs (oxidising phase, influence of microorganisms). Sizing may be reviewed depending on a given site's conditions in order to take into account the geochemical composition of the waters. The definition of a "package failure" scenario and a unit failure in the SEN allows in any case covering the residual uncertainties.

Chemistry – Influence of corrosion products

The over-pack's corrosion leads to the formation of corrosion products (hydrogen and magnetite), which can influence the RedOx potential conditions in the vicinity of the over-pack, the hydraulic flows, the corrosion and the mechanical resistance of the over-pack, as well as the transfer of radionuclides. This subsection mentions a few FEPs linked to these corrosion products, generally neglected in the SEN, explaining why it was decided to not retain them.

Magnetite, an expansive corrosion product, can induce additional stresses within the bentonite and on the over-pack. The steel's corrosion products having a lower density than the steel will tend to swell. The pressure in the disposal shafts, mainly linked to the hydrostatic pressure and the swelling pressure of the clay buffer can be thus increased. A premature rupture of the over-pack caused by the sleeve's corrosion products seems highly unlikely (at worst, the sleeve is covered by the package failure situation). The corrosion products can sorb the silica resulting from the degradation of the vitreous matrix and the radionuclides. The first effect of a kind to delay silica's equilibrium within the cell is taken into account in a penalising fashion in the models (see subsection 5.2.4.3); the second – favourable – is neglected.

Criticality

The risks of criticality of the C waste packages were analysed under the research work on the repository in the clay medium [xxxi]. They show that the criticality would be negligible. This phenomenon is not considered in a normal situation.

5.2.4.3 Behaviour of the primary packages after the over-pack's rupture.

- **Characteristics of the vitrified waste primary packages**

Since no confinement performance is attributed to the C waste primary container made of stainless steel, this container is not retained in the safety analysis.

The release models retained for the disposal of C waste in a granite rely on those developed for clay [**xv**]; only the vitreous matrix is the source of a release of radionuclides and is assumed congruent to the dissolution of the glass.

The glass's characteristics are assumed to comply with those which exist at the moment of the waste's production. The vitreous matrix can be fractured under the effect of cooling, gas production or subsequent to post-manufacture mechanical stresses, which implies in such a case an increase in the reactive surface during the dissolution of the matrix; this is, therefore, a parameter which can have an influence on the release. In a normal situation, a fracturation rate of 5 is retained (ratio obtained from experimental measurements). Sensitivity studies have not been carried out on this point which is not specific for granite. However, the glass is considered to be very fractured (ratio equal to 40) when its dissolution is initiated at the residual rate (see next subsection).

- **Processes affecting the C waste primary packages**

This subsection is dedicated to the radionuclides release models defined for C waste and to the transfer of radionuclides. The release of radionuclides is initiated as soon as the over-pack loses its leak tightness.

The radionuclides release model for the SEN is a model directly linked to the dissolution of the vitreous matrix. At this time, the corrosion products of the over-pack or the rest of the over-pack itself no longer ensure any confinement and the disposal cells are totally saturated.

The release models and the associated parameters differ a priori according to the reference packages; they are selected among the models available based on:

- their conformity with any experiments conducted in order to validate them;
- their applicability to the repository's environmental conditions;
- their ability to take into account potentially disruptive phenomena that might adversely affect the waste's resistance.

Waste packages may :

- obey a model based on the initial dissolution rate of the glass and the exchange surface offered by the latter (V_0.S model); this conservative model is particularly retained (with the phenomenological parameters for the surface offered to the water and the dissolution rate) for the C0 glasses;
- obey a two-phase phenomenological model (R_0.A $\rightarrow$ R_r). Initially, the model is based on the initial dissolution rate until saturation in silica of the surrounding medium (V_0.S). Then the dissolution kinetics decrease down to a residual rate (V_r) [xv] The transient is not represented;
- if only the C2 waste is treated, which corresponds to the second behavioural model, then this is the model which is retained. The rate Vr = 5.10^{-5} g.m-2.daj^{-1} measured at 50°C is retained in a conservative way.

A release at a higher temperature would only be due to a failure of the over-pack. For the defective packages, the same model V_0.S $\rightarrow$ V_r is retained, with a rate V_0 dependent on the temperature and the pH.

The model is applicable to a range of pHs between 7 and 9. Considering the distance from any alkaline disturbance source, the model retains in a conservative fashion a pH of 9. The role of the over-pack's corrosion products, which can sorb the silica, is taken into account in the model.

In conclusion, the glass's behavioural model is a "phenomenological" model, but used under conservative temperature and pH conditions, which reduces the expected lifetime of the matrix. Since the objective of the dossier 2005 leads to focus on the essential of the sensitivity studies on the granite's properties, it was not envisioned to carry out a sensitivity study to a "conservative" model (of the type, for example, of a model with a constant dissolution rate of the glass), considering the state of knowledge and the conservative choices already retained on the parameters.

5.2.4.4 Representation of the engineered barrier and the plug

Despite a few differences in the phenomena (shorter thermal phase, presence of ferrous compounds in larger quantity, etc.), the engineered barrier and the plug of the C waste cells experience practically the same evolution as that of the spent fuel cells. Therefore, this representation will not be rediscussed here.

5.2.5 B waste repository zone

First of all, the FEPs related to the characteristics of the modules and cells are presented. They allow specifying the retained design principles. Then each component is analysed, first its characteristics and then all the processes which can affect it.

5.2.5.1 B waste modules and cells

The repository tunnel is represented by:

- A disposal chamber where the packages are disposed. The cell body ensuring a chemical retention by the phenomena of sorption and precipitation is represented by a concrete whose properties are set according to the kind of waste.
- An access drift to the tunnel, closed and sealed on one side by a "plug". Since the sealing of B waste repository tunnels is positioned in the access drift to the cell (in the cell head), it is designed in a similar way as the service drifts.

Possible disturbances generated on the granite are taken into account. Similar to the damage liable to be occurring in the drift wall, a micro-fissure damaged zone of a few decimetres in thickness (see subsection 5.2.7) is considered to develop around the wall of the repository tunnel and the access drifts. This zone is interrupted by the seal at the tunnel’s entrance.

Like for the other wastes, the modules are installed in granite blocks with a low permeability delimited by water-conducting fractures (see Figure 5.2-5). Their installation takes into account the stress system and the hydraulic conditions of the site. The very low or moderate thermicity of the B waste allows designing compact disposal cells implying a small granite volume, resulting in a more demanding adjustment with respect to the fracturation [viii].

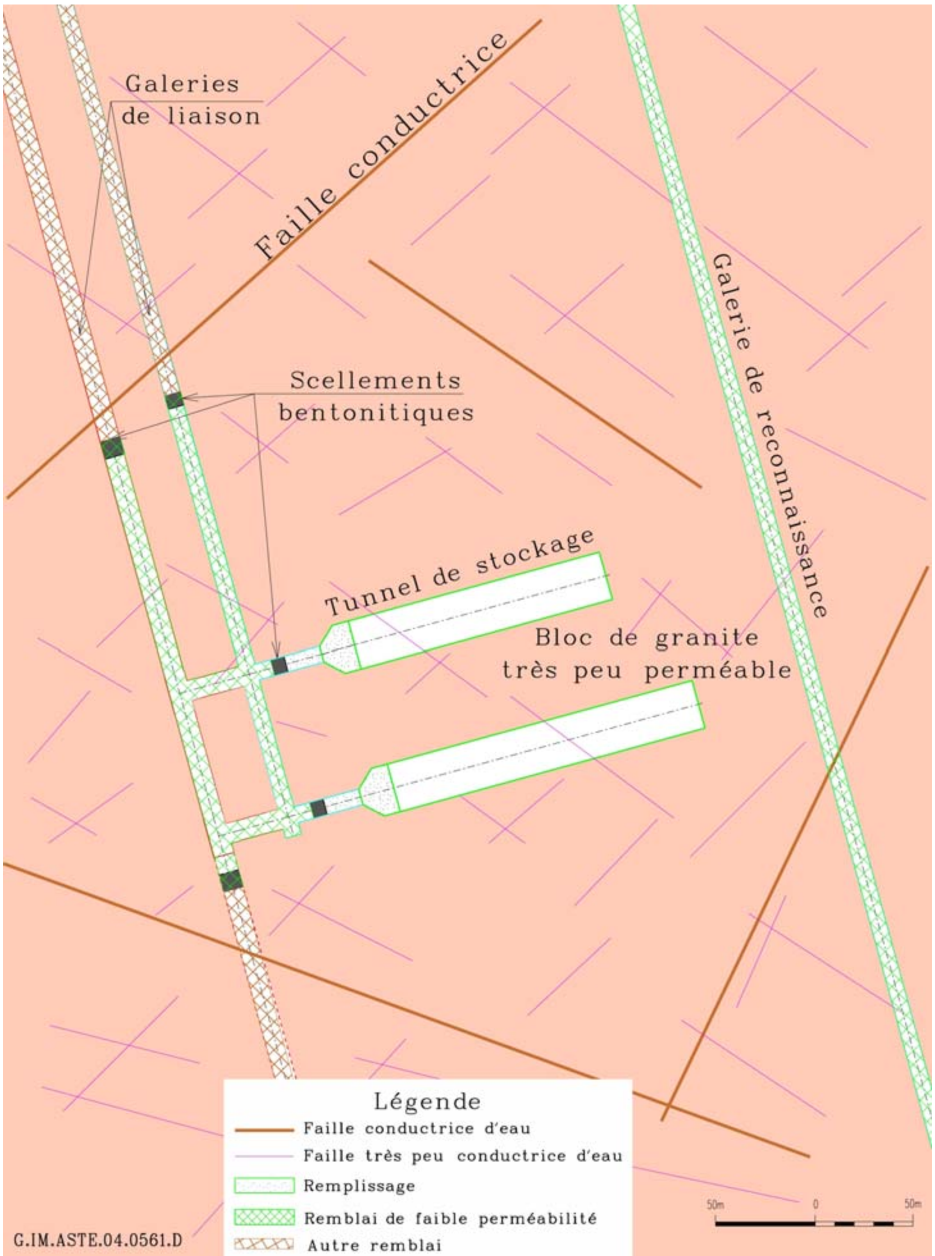

Figure 5.2-5 Installation of B waste repository modules in granite blocks with very low permeability

The characterization approach ahead of the granite blocks would allow specifying the architecture of the modules as a function of the fracturation. Considering the limited number of tunnels to be characterised, a failure in the characterization process of the properties of the small fracturation appears highly unlikely. It was not considered at this stage in the studies. It may be examined at a future possible stage, for example, by considering the presence of a more conductive fracture interconnecting the tunnel (transmissivity greater than the granite block's acceptance criterion).

5.2.5.2 B waste packaging

- **Characteristics of the concrete containers**

The degradation of the concrete of the packages is an extremely slow process [xli] which allows imposing geochemical conditions favourable to the B waste primary packages (particularly in terms of pH), thus limiting their degradation.

The representation of the packages is different for the two waste types (B2 and B5.2) retained for the calculations:

- For the B2 waste (bituminised sludge), the disposal package is assumed to not be water-tight as soon as the repository is closed. This hypothesis refers back both to the specific characteristics of the container, which is rendered permeable to the gases emitted by the bituminised sludge and to the uncertainties on its long-term evolution under the effect of the actions produced by the bituminous matrix (release of organic acids, mechanical stresses, etc.);
- For the B5.2 waste, a container with complementary confinement properties, namely, hydraulic properties slowing down the release of radionuclides for 10 000 years, was retained for the hydraulic calculations within the cells.

From the viewpoint of transfer, the packaging complements of the disposal waste packages are represented for all the B waste packages as a homogeneous chemical environment (not distinguished from that of the concrete cell) limiting the flow of toxics by precipitation and sorption phenomena. The parameters are adapted depending on whether or not the container is considered to be efficient over the first 10 000 years.

- **Processes affecting the containers**

Heat transfer

The exothermicity of the B waste can influence some chemical processes within the packaging (corrosion, concrete degradation, radiolysis). The heat transfer dimensioning of the repository guarantees a temperature less than 70°C throughout the cell. The concrete in the vicinity of the packages should not be subject to a higher temperature. The influence of temperature on physico-chemical processes liable to affect the container is not retained [xxxii].

Hydraulics

The resaturation of the cell depends on the water flows from the granite in the near-field. However, a resaturation by the structures is possible. The resaturation period depends mainly on the transmissivity of the fractures in the cell’s wall and the volume of empty space remaining in the tunnels after closure. Preliminary calculations indicate that the resaturation period of the tunnels can hence vary from several tens of years to a thousand years [xii].

In the scenarios, the B waste tunnels are assumed saturated at the closure of the repository, which is a reasonable hypothesis in a generic context; it covers the uncertainties linked to the resaturation period and it is penalising with respect to the durability of the concretes.

Chemistry – Physico-chemical conditions

The chemical evolution of the packaging takes into account the FEPs related to the evolution of the chemical conditions (pH, Eh) within the repository tunnels, the degradation of the packaging concrete and the production of gases.

The evolution of the pH within the repository tunnels is determined by the cement-based materials. The pH initially very alkaline should be stabilised very rapidly at 12.5, since the fluids are in equilibrium with the portland cement due to exchanges with the cement-based materials. Considering the quantity of concrete introduced in the cell, the resaturation fluids reaching the container are considered to be alkaline and do not induce a premature degradation of the container [xii].

The concrete is progressively degraded by contact with granite waters: this chemical degradation can lead to its fissuration and the loss of the retention properties of the concrete container. A conceptual model of the degradation of the concretes was developed to describe the behaviour of the concrete in the repository situation [xli]. The effect of carbonation due to the exposure of the containers to air or attack by sulphates present in the granite waters does not affect the container's behaviour. Nonetheless, various uncertainties persist, particularly on the relationship between these models and the mechanical behaviour of the concrete. These uncertainties would have to be studied in depth if the interest of a concrete container having performances at the multi thousand-year scale would be confirmed.

Considering the chemical variety of the B waste, disturbances from the reducing environment or the pH are liable to take place. In particular, the influence of radiolysis and the nitrates released by the B waste – which can disturb the RedOx potential due to their reduction by possible microorganisms - is to be taken into account, but remains a local and temporary effect which cannot have any influence at the scale of the overall module. In addition, it should be noted that the oxygen dissolved in the fluids would be rapidly reduced by the waste packages. In the long term, the RedOx potential is mainly determined by the granite waters.

No identified FEP tends to place in doubt the representation of the cells and the containers like a reducing medium in a stable alkaline pH.

Influence of gases

Corrosion of metallic components in a reducing medium leads to the production of hydrogen. In addition to the corrosion of metallic elements, the degradation of the organic substance of the B waste primary packages and the radiolytic phenomenon of bitumen, for example, causes the production of gases (H_2, CO_2, CH_4, CO), which will migrate outside the repository tunnels.

During the operating phase, ventilation ensures the dispersion of these gases outside the repository tunnels. By design, the concrete containers of the B2 waste are supposed to evacuate the hydrogen in order to prevent an overpressure in the waste packages related to the production of gases in the waste. Such an arrangement is not provided for in the B5.2 packages because they do contain neither organic compounds nor residual water [ix].

The estimation of the gases produced by the B2 packages indicates that this production has no influence on the phenomenological evolution of the tunnels during the resaturation phase and is not liable to modify the nature of the hydrolysis phenomena, which are the main degradation factors of the concrete.

The gases, therefore, have no effect in an initial approach on the representation of the B waste cells in the normal evolution scenario.

5.2.5.3 B waste primary packages

Except for the B2 waste embedded in bitumen, the other wastes are directly conditioned in the packages possibly compacted (wastes B1, B5, B6, B8) or in a cement-based matrix (wastes B3, B4, B7).

For quantitative assessments, a release model is retained which depends on the kind of waste and the conditioning mode of the waste. Thus, the model adopted for the B2 reference packages rests on a degradation model of the bituminised waste, although the model adopted for the B5 waste packages rests on a corrosion process [xv]. Recall that only these two types of waste are processed in the scenario.

- **Processes affecting the B waste primary packages**

Analysis of the evolution of the B waste led to distinguishing the FEPs related to the various processes, mainly chemical.

The release models are analysed with respect to the B2 and B5.2 waste packages. They take into account the kind of waste, that is:

- The release models of the radionuclides present in the metallic elements of the B5.2 waste are based on the progressive corrosion of the various metals [xv] according to a corrosion model taking into account the reducing and alkaline conditions imposed by the container's environment. It leads to a release in 100 000 years;
- For the B2 waste packages, the model adopted in a normal situation is the conservative model COLONBO 3, which describes the release of radionuclides subsequent to the intake of water by the soluble salts of the bituminised waste. This highly conservative model does not take into account a number of favourable phenomena (raming which the reduced solubility of the radionuclides within the matrix), but shows release rates much faster than those experimentally observed.

It is taken into account that there exists a fraction of the radionuclides which are liable to be instantaneously released as soon as water comes in contact with the waste. For the B5.2 waste packages, the release model considers that the radionuclides present in the zirconia, the fine elements and the surface of the waste (deposits, residues) are labile. The adopted model is conservative, which covers the uncertainties on the release kinetics of the radionuclides present in the zirconia, the deposits and the residues, and the uncertainty on the quantity of fine elements produced during the shearing of the metallic elements [xv].

In conclusion, the release models take into account most of the uncertainties. Considering the objectives of the dossier 2005 granite, it was decided to not carry out sensitivity studies on these models in order to cover a few uncertainties not formally covered by the models.

5.2.5.4 Filling of the B waste cells

It should be noted that the empty spaces between the package stacks and the granite in the tunnel dome are filled with concrete blocks which cannot have an unfavourable chemical interaction with the concrete containers. The backfill placed at the tunnel head after dismantling the operating dual-gate system is also of a cement nature.

The concrete of the blocks installed in the dome, as well as cement-based backfill, contribute to a high pH within the cell. This choice is based on the characteristics expected from the filling, particularly in terms of chemistry (retention favomed alkaline pH).

The filling is represented as an equivalent porous medium having chemical performances, but no hydraulic performances. In practice, within the models, the containers are not differentiated.

The FEPs describing its characteristics and its evolution are not rediscussed here.

5.2.5.5 Transfer of radionuclides in the B waste cell

The transfer of radionuclides in the B waste repository tunnels depends not only on the speciation of the radionuclides, but also on the sorption capacity of the cement-based materials. The sorption (and the solubility) of many radionuclides is favoured in alkaline waters having a pH around 12.5. This pH is liable to decrease over time according to the degradation condition of the concrete.

The limit values of solubility and distribution coefficients adopted for the cement-based materials are conservative, except for the B5.2 waste packages, which are of higher performance due to their mechanical resistance and for which "best estimate" parameters are considered for 10 000 years.

In particular, they do not take into account phenomena a priori favourable but hardly quantifiable, such as the co-precipitation of the radionuclides with calcite or with the corrosion products. In order to cover the uncertainties related to the degradation of the concrete (particularly by a decrease of the pH), penalising or conservative values are considered in the sensitivity.

The transfer of radionuclides can be influenced by complexing phenomena involving organic or inorganic molecules. The solubility of the radionuclides can be modified (increased) and the sorption can be decreased. Such compounds can be soluble. The complexing species can be natural (humic substances, etc.) or can come from the B waste, and even for the adjuvants used in the formulation of the concrete. Most of the complexing species coming from the B waste have hardly any influence in the cement-based medium on the radionuclides except for ISA (isosaccharinic acid), a degradation product of cellulose, mainly present in the B3 waste packages. Whereas no calculations were performed on transport in the B3 waste cells, which present a radiological inventory less than that of the B2 and B5.2 wastes, this phenomenon is not taken into account.

Gases may also a priori disturb transport in the cell. The possibilities of gas pressure increasing are limited by the evacuation through the medium and the structured components of the repository. In fact, the gas reaching the damaged zone can be dissolved or be transported in a two-phase mode. The way in which the gases flow through the fractures was studied by experiments conducted in underground laboratories. The results show that the gases tend to create their own flow paths through the fractures in a way distinct from the liquid phase and, therefore, do not interact with the transport of the aqueous solutions.

5.2.5.6 Seals of the B waste disposal cells

- **Characteristics**

The representation of the seals of the B waste disposal cells is similar to that of the other bentonite structures (plugs, engineered barrier); in particular, the same radionuclide retention capacity is considered. The hydraulic and transport phenomena taken into account in the calculations are of the same kind as the phenomena occurring in their case.

The seals of B disposal waste disposal cells are intended to interrupt the fractured zone in the B waste repository tunnels. It should be noted that the damaged zone is not discussed here since it is globally treated in subsection 5.2.7.

The TSX experimentation conducted in the Canadian underground laboratory of Lac du Bonnet [x] showed the feasibility of a granite seal. The permeability measured within the framework of this experimentation constitutes the reference adopted for the evaluations. It was measured at 10^{-11} m/s, a value assigned in the models to the permeability of the swelling clay. This choice represents a conservative option in that it underestimates the performances imaginable following the test: in fact, this test measured the overall performance of the seal (taking into account the permeability of the various media coming in contact with water, including the bentonite – rock interface). The radionuclides migrate in the seal mainly by diffusion.

- **Processes affecting the seals of B waste disposal cells**

Regarding the seals, analysis shows that performances can be reduced not only by the same processes as those described for the bentonite in the plugs of spent fuel or C waste shafts, but also by the processes specifically linked to the arrangement of the seals of B waste cells, which can be summariged as follows:

- Chemical disturbances, such as the alkaline disturbance linked to the retaining plugs, which may locally produce a bad interface;
- Mechanical degradation of the concrete plugs supporting the seals.

The alkaline disturbance produced by the fluids resulting from the degradation of the concretes can modify the mineralogical characteristics of the bentonite, with the smectites being replaced by non-swelling phases, such as CSH, CASH, or zeolites. Nevertheless, the evaluations made for a seal show that these remineralisations would be limited to approximately sixty centimetres for 100 000 years [xlii]. The thickness would be at most on the order of a metre for 1 million years, which would not affect the integrity of the clay core having a multi-decametre length. It should be noted, however, that beyond the remineralisations, the alkaline disturbance is a source of calcium which diffuses over longer distances. Calcium induces cationic exchanges, which can degrade the performances of the swelling clay, but in proportions which remain compatible with the dimensioning objectives of approximately 7 MPa in swelling pressure. The alkaline disturbance is not represented: the extension varying from a decimetre to multi-decimetres for this disturbance does not affect the integrity of the decametric clay core which preserves its properties. In addition, regarding the remineralised zone, the seal's length (5 m) taken into account in the calculations in an equivalent porous medium is conservative.

The concrete abutments must allow the clay core to develop its swelling pressure. The drift backfill can play this role over the long term after degradation of the retaining plugs or in the short term in case of mechanical rupture of the retaining plugs, but provided the backfill could swell. Bad support (linked, for example, to stacking or a bad placement) would not allow the backfill to provide sufficient mechanical support for the core to develop its swelling pressure. Similarly, a bad interface with the damaged zone may be envisioned if the bentonite does not sufficiently fill the damaged zone despite its thin thickness.

This situation is envisioned in the SEA"B waste cell sealing failure", where the failure is expressed by the existence of a damaged zone offering a preferential radionuclide transfer path 5 cm thick in the granite wall. The transmissivity or permeability characteristics are those of a damaged zone in the drift wall. They are expressed by an imperfect interface between the bentonite and the granite with a centimetre thick zone (equivalent to that of the damaged zone) having a higher hydraulic conductivity (a value of 10^{-9} $m.s^{-1}$ is adopted based on feedback from the ZEDEX experimentation).

5.2.6 Drift zones

5.2.6.1 Characteristics

In the normal evolution scenario, where the drifts within the module can constitute a potential radionuclide transfer path, those drifts are taken into account and represented. They are like a backfill composed of crushed granite and bentonite represented with a permeability of 1.10^{-10} m/s [xl]. The presence of bentonite allows reaching such permeability values, also adopted by SKB, particularly in view of measurements on samples taken after installing and compacting the backfills [xiii]. It is difficult to demonstrate at this stage that these hydraulic performances are uniformly reached at the scale of the entire repository ; therefore, an uncertainty is retained on the value of the backfill's permeability. Sensitivity studies were carried out in the SEN in order to evaluate whether this parameter is sensitive and, if so, from what value it becomes sensitive. These studies consisted of modifying permeability in a model representing the geological medium in the form of a homogeneous porous medium.

The hydraulic gradient of the drifts depends on the regional hydraulic gradient, the hydraulic conductivity within the modules, the architecture of the drifts and the processes inside the repository capable of influencing the pores pressure (exothermicity, gas production).

Without a designated laboratory site, the hydraulic conditions within the drifts and at the module level are not known. In the calculations in an equivalent porous medium, various hypotheses related to the granite's permeability and the hydraulic gradient are examined.

5.2.6.2 Transport in the drifts

The drift backfill can contribute to limiting the flow of water in the module's drift system. The backfill's properties can, however, be degraded by stacking, presence of empty spaces, emplacement clearances or even chemical degradation. These uncertainties are covered by the conservative values of the hydraulic and transport parameters, which were retained as the reference and the sensitivities to the hydraulic parameters envisioned in order to cover a larger range of values.

In the backfill, the SEN's model takes into account the transfer of the radionuclides by diffusion, advection and dispersion.

Radionuclides can be sorbed on the bentonite of the backfills. The backfill containing a certain percentage of bentonite contributes to the retention of the radionuclides. The approach adopted in a normal situation consists of considering that in the mixture of crushed granite and bentonite the granite waters should rapidly be in equilibrium with the crushed granite, a process estimated equivalent to the resaturation of the bentonite. Taking advantage of this similarity, the retention model adopted in the bentonite of the engineered barrier may be adopted for the bentonite of the backfills. In parallel, the sorption phenomena on the granite aggregates are neglected, which is conservative.

Backfill oxidation is neglected due to the capacity of the granite waters to impose a reducing environment rapidly after the closure of the repository.

5.2.7 Mechanically damaged granite zone

5.2.7.1 Characteristic

From a mechanical viewpoint, the granite rock is characterised by a high resistance, generally greater than 100 MPa in a uni-axial compression. Excavation of a structure can however induce mechanical damage in the granite wall, as a result of a local redistribution of the stresses. This damage depends particularly on the system of natural stresses and the excavation method. In any case, it is limited especially since the design calls for adapting the excavation methods according to the various structures to be excavated [viii].

In a penalising way, a damaged zone thickness of 5 cm is retained in the seal walls and a 50 cm thickness retained in the drift walls.

5.2.7.2 Processes affecting the damaged zone

- **Hydraulic**

Right above the seals, the swelling and the plasticity of the bentonite can fill the empty space at the damaged zone/bentonite interface and the fissures within the damaged zone. Thus, it limits the flow of water and the risk of transfer of radionuclides at the damaged zone/bentonite interface. Observations made on a shaft at a reduced scale confirmed the penetration of the bentonite into the fractures of the wall granite and thus the inner interaction between engineered barrier and granite during the resaturation of the shaft [x]. Thus, the bentonite ensures a hydraulic cutoff of the damaged zone. As a result, it can be considered that there is no hydraulic effect from the damaged zone in contact with the engineered barriers, the plugs and the clay cores of the seals.

For all these reasons, in the SEN, the damaged zone in the wall of the C and CU disposal boreholes, the plugs and the seal walls is not represented.

Only a situation in which the bentonite would not be sufficiently swollen may lead to an insufficient sealing of the damaged zone and result in a local ineffectiveness of the plugs and seals. Such a possible case is discussed in the SEA"seal failure".

The situation is a little different in the B waste tunnels. The damaged zone is characterised by a greater hydraulic conductivity than that of the host rock. Depending on the excavation technique retained, the damaged zone in the wall of these structures may develop over a thickness of a few decimetres; it was set at 0.50 m in the reference. The damaged zone is represented with a permeability of 10^{-9} $m.s^{-1}$ in the reference, which is very downgraded with respect to the very low permeability of the granite rock itself.

- **Chemistry**

In the vicinity of the retaining plugs or in the B waste tunnels, the damaged zone/concrete interactions, and particularly the characteristics of the cement-loaded waters (pH > 12, presence of carbonates, RedOx potential, salinity) can have two types of consequences on the granite wall of the tunnels : an alteration of the rocky matrix of the granite and a modification of the transfer properties of the small fractures of the wall granite with an extension of an alkaline disturbance in these fractures.

Because of the mineralogy of the granite rock (quartz and silicates), its alteration by the cement-loaded waters is linked to very slow diffusion phenomena. It can be neglected. The alteration of the concrete disposal packages can cause in the small wall fractures of the granites precipitations of minerals which reduce their hydraulic transmissivities. Considering the generic character of the analysis, this factor favourable to the limitation of the transport of radionuclides in the small fractures is not taken into account.

- **Influence of gases**

Gas migration within the damaged zone can lead both to the movement of the water in the pores and the transfer of radionuclides in the aqueous phase, as well as the transfer of radionuclides in the gaseous phase. From a phenomenological viewpoint, the transport of gas in the EDZ is based on the same mechanisms as in the fractures. Therefore, refer to the subsection below (see subsection 5.2.8).

5.2.8 Near-field

Water flows in the granite medium mainly through discontinuities (or fractures). These can vary in size, from small fractures likened to the matrix up to regional conductive faults. Depending on the adaptation of the repository to the fracturation of the massif, the drifts can be interconnected by fractures of "moderate" conductivity (C and CU waste modules). The cells and modules are installed away from fractures of significant conductivity. Such an adaptation of the installation to the fracturation relies on the prior reconnaissance programme, which will allow if a site is designated specifying the positioning of the modules and cells. The hydraulic properties of the near-field are characterised by the hydraulic conductivity and the kinematic porosity of the rock. They depend significantly on the fracturing network and its connectivity.

To message the uncertainty related to the absence of a designated site, the fracturation models retained respond to two different principles: semi-deterministic for the large faults, and stochastic for the small fractures; the calculation models are detailed in chapter 6. In a normal situation, the near-field represents the granite blocks in which the modules are installed and delimited by the water conductive fractures. It is represented as a fractured medium which includes the granite matrix and micro-fractures in which the water is free and in contact with the filling minerals (which can present sorption capacities).

- **Thermal characteristics**

Temperatures in the rock in contact with the CU disposal cells reach their maximum several tens of years after the emplacement of the waste packages and vary between 50 and 77°C for the French granite inventory. In France, the long-term temperature range in the granite massif will depend on the heat transfer characteristics of the site and the configuration of the repository's architecture. For an intermediary granite from the viewpoint of its thermal characteristics, the maximum temperature reached in the granite at the edge of a cell would be approximately 60 to 65°C. For the granite in the near-field of the C and CU disposal boreholes, the thermomechanical and thermohydraulic phenomena liable to be caused by a temperature increase in the massif are not of the kind to significantly modify the hydrogeological flow conditions of the site [xii]. Therefore, these phenomena are not represented in a normal situation.

An effect which may have to be taken into account, on the other hand, would be that of temperature on the hydraulic and transport parameters within the small fractures: temperature can modify the equilibria of sorption and even solubilities. It can influence the viscosity of water, and likewise the apparent permeability. In a generic context and in the absence of experiments dedicated to a particular site, it was not possible to derive variation laws for the various parameters (Kd, solubility, permeability) versus temperature. No effect as such is expected a priori to radically change the analysis' conclusions. Therefore, models were retained with temperature-independent parameters in an initial approach. However, this should necessitate carefully taking into consideration the informations obtained from the SEA "package failure" envisioning a heat transfer, which may only be partial.

- **Hydraulic**

The water flows in the near-field are located in the conductive fractures. They depend on the pressure gradients determined by the conductive fractures at the limit of a module and by the possible effects of temperature and density.

The hydraulic parameters – that is, transmissivity – of the small fracturation in the SEN are obtained from the review of data available from the typological analysis. For each site model, sensitivity studies are conducted to evaluate the sensitivity of the transfers of radionuclides to their characteristics. In fact, these studies also cover, at least in their basic principle, the characterization uncertainties and the physico-chemical processes which may lead to variations of this small fracturation in a given massif. It should be noted that another parameter, the connectivity of fractures, is conservatively treated: the intersections of fractures are systematically considered to be transmissive by the calculation models.

- **Effect from gases**

Gases can migrate in the near-field according to various processes in the gaseous phase or in dissolved form. The GAM experiment carried out in the Grimsel laboratory in Switzerland [xii] tends to show the independence of the paths followed by liquid water and the gaseous phase in the fractures. This phenomenon can thus contribute to a gas leak under overpressure in the tunnels toward the fractures in the granite wall. This hypothesis is the one retained in a normal situation; the gas is evacuated through the fractures within the near-field.

- **Mechanical**

The system of natural stresses can influence the evolution of the fracturation (opening/closure) of the granite and indeed affect the repository's components (container, seals, etc.). In order to analyse the sensitivity of the cavities' mechanical stability to the depth and anisotropy of the system of stresses, preliminary digital modellings were created. The calculations carried out for the case of decimetric sized structures (B waste cells) show that the stability of the excavations for the repository depths (approximately 500 m) is ensured in the French context for most of the stress configurations and for all the lithologies of the granite massifs [vii].

In addition, the geo-prospective studies completed indicate that a significant change in the systems of stresses with the possible creation of a new fracturation cannot take place at the safety evaluation time scales [vii] As a result, the influence of the closure and/or reactivation of fractures within the near-field is not taken into account.

- **Chemistry**

In a normal situation, the chemical disturbances induced by ventilation during the operating phase concern a priori rather the damaged zone and are judged negligible. There effects are not represented.

The alkaline disturbance concerns mainly the near-field of the repository tunnels of the B waste. Beside the B waste cells, only the retaining plugs of the seals can be source of an alkaline disturbance. It is hard to determine in a generic context how much an alkaline disturbance may extend. This disturbance can reach the granite matrix and the small fracturation in the vicinity of the retaining plugs and the B waste cells. The extension of an alkaline plume and its influence on the physico-chemical properties of the fractures are covered by an experimentation in the Grimsel laboratory in Switzerland. The results indicate a general tendency for the transmissivity of the fracture to decrease, which would be in relation with the precipitation phenomena [xii]. The alkaline disturbance is not represented in contact with the B waste cells.

The other chemical disturbances which may be induced by the compounds released by the waste (in particular, the B waste, which presents a more varied chemical nature) would probably be buffered inside the cell and would not reach the small fracturation.

- **Transfer of radionuclides**

The transfer path retained in the models in the near-field is the path followed through small fractures. The diffusion phenomena in the matrix outside the path in the fractures are considered to be negligible with respect to the transfer in the fractures. However, the model takes into account the possibility of a diffusion in the of the fractures lode walls.

In the fractures, the radionuclides are transported by advection (pressure gradient) and by diffusion (concentration gradient). The transfer of radionuclides in the fractures of the granite is however generally advective.

The sorption properties of the radionuclides in the granite in the near-field depend not only on the radionuclide, but also on the mineralogical composition of the rock and the fractures, as well as the chemistry of the waters (pH, content in carbonates, RedOx potential, salinity, presence of organic substances).

In a normal situation, the retention phenomena are taken into account in the lode walls of the fractures in the form of distribution factor (Kd). Without a designated site, the test values are those retained. For each of the radionuclides studied, these values correspond to realistic values in connection with the geochemical composition of the waters from the granite.

The solubility of the radionuclides and their possible co-precipitation with minerals in the water conductive fractures were not retained [xxxv]. This is due to the fact that the anticipated concentrations in radionuclides are low and that the co-precipitation phenomenon a priori favourable, is hardly quantifiable.

5.2.9 Far-field

The far-field intervenes in the model not only as a radionuclide transfer pathway, but also because it can influence in return the near-field (by imposing the gradient within the repository and in the small fracturation, for example).

A number of FEPs are common to the far-field and the near-field. Their analysis is not necessarily repeated here. Only a few specific points of the far-field are focused on.

The characteristics of water-conducting fractures (distribution, connection, hydraulic properties) are not known in a generic site context. In a normal evolution situation, only the part concerned by the transfer of radionuclides between the reference module and the limit of the geosphere is represented. It is modelled by a fractured medium in the form of radionuclide pathways in the faults ranging from hectometres to multi-kilometres (based on hydrogeological models ensuring the consistency of models at various fracturation scales (see subsection 6.1.3 below)). Sensitivities to the hydraulic parameters of the fractures are determined to cover a range of plausible values (test values). The connectivity of the fractures is treated in a conservative fashion by assuming it is effective.

The concentration of radionuclides can be reduced by dilution by means of the water flowing through the conducting fractures of the far-field or delimiting the reference modules. The dilution phenomenon is not a retained process. However, it leads to a hypothesis of a zero concentration of radionuclides as a limit condition, resulting in a maximisation of the diffusive component in the radionuclide transport equations.

Changes in the composition of the water flowing through the fractures delimiting the repository modules can influence the environmental conditions in the near-field (hydraulic, geochemical, etc.). The granite waters can have a variability in composition, particularly the salinity or their oxidising character. The capacity of the granite to make the waters reducing waters limits the oxidising potential to the surface waters. The salinity of the waters in the French granites (except for the coastal situation which is not very frequent) is considered to be generally less than that encountered in the coastal contexts of the granite sites in Sweden or Finland [vii] The effects linked, for example, to the presence of salty gradients are excluded to a great extent for French massifs, even though some close to the coastal line can present such a risk. In a generic context, it was decided not to take into account the occurrence of a salinity gradient in the scenarios.

In addition, characteristics which would result in the investigated massifs not clearly responding to the criteria of the RFSIII.2.f were not taken into account in the model [i]. The same holds for the massifs, whose presence might restrict the repository's design: strong heterogeneity of the massif, major hydraulic depression or overpressure zones. The results of the typological analysis of the granites show, in fact, that it would be possible in a site research context to select the massifs which do not present this type of restriction. In addition, the hydraulic depression or overpressure phenomena are unlikely in the outcropping granites.

5.2.10 Biosphere

Considering the objectives of the Dossier 2005, no dose calculations are envisioned. Therefore, the biosphere is not integrated in the analysis. The influence of the environment under the hydraulic conditions and the influence of the site's topography are nevertheless retained.

5.2.11 External events

Among the FEPs which record external events capable of influencing the repository system, the only exclusions were made on:

- Deliberately abusive human actions;
- Major natural disasters (such as a meteorite falling);
- Events having only an influence on the biosphere, since the biosphere is not being taken into account in the model.

Abusive acts are by nature unpredictable. However, it is possible to prevent them during the operation of the repository and initially following its closure by conventional site guarding means and access control. Thereafter, a closed repository would present due to its great distance from the surface and the conditioning of the radioactive materials which it contains a hard target of little interest for an act of sabotage. In addition, if it would be considered that a deep repository could nonetheless be of interest to possible offenders, it would still be possible to maintain surveillance around the site until its placement would be eventually forgotten. In this last possible case, the repository forgotten by all would no longer represent a potential target.

Accounting for major disasters leads back to natural accidents - of a very reduced probability – having consequences at the scale of a region or more. These events are not considered in the analyses due to the fact that their consequences would exceed by far the scope of the repository alone, and its presence would not represent an additional factor of risk, which is really significant. It should be noted that major natural events which are more localised and more frequent (volcanoes, earthquakes, floods), are, on the other hand, taken into account. Other major disasters are considered to be "out of scope", as they are for all industrial installations.

5.2.11.1 Climatic evolutions: Glaciation

The climatic scenarios defined in the BIOCLIM project [xliii] provide for an alternation of glacial and interglacial periods succeeding one another according to cycles of approximately 100 000 years. The existence of these cycles is supported namely by analyses of ice samples taken from Antarctica, giving an image of climatic evolutions over the last 400 000 years (see Figure 5.2-6).

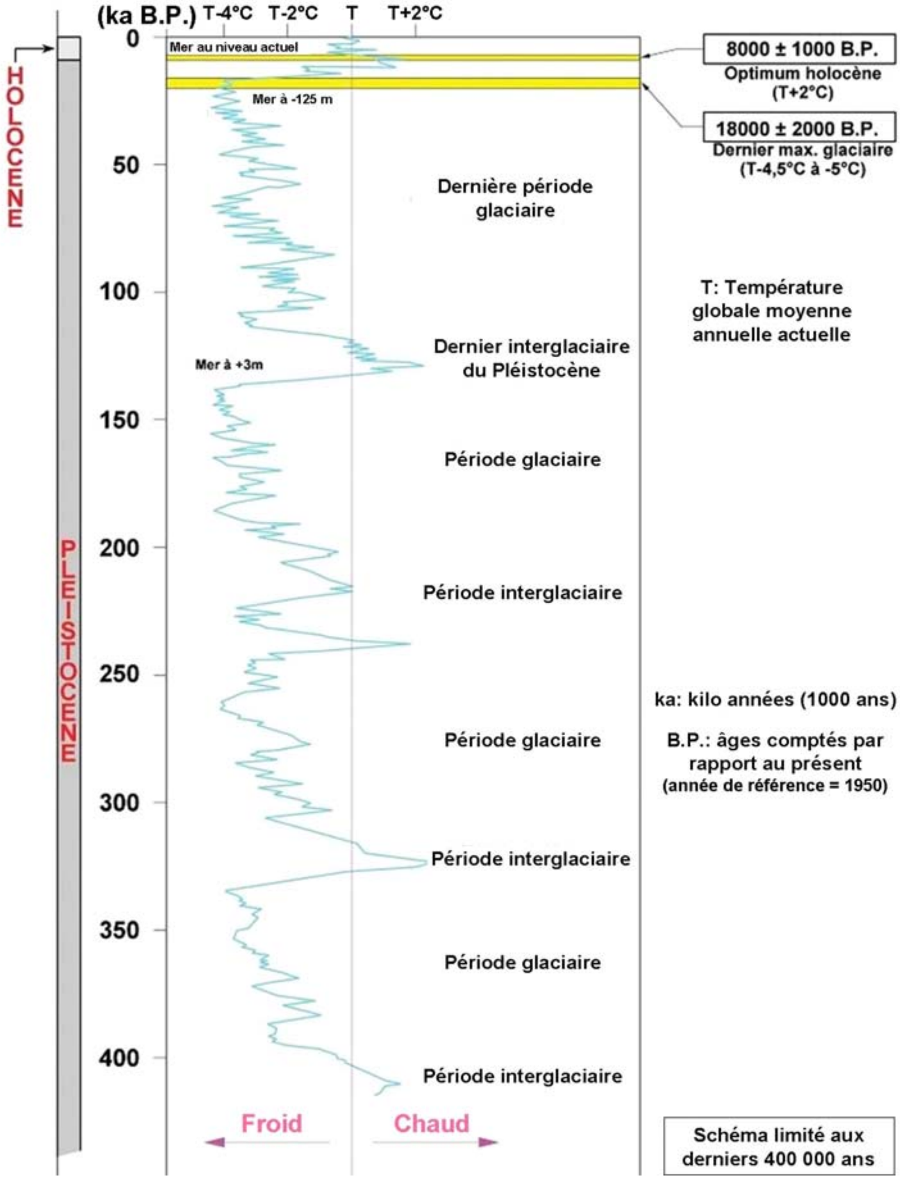

Figure 5.2-6 Temperature variations in Antarctica (Vostock) over the last 400 000 years

A major change due to a glaciation would be the appearance of a permafrost, which would modify the mechanics of the sub-soil and restrict the flows of underground water, the reappearances at the surface and, therefore, the transfers of radionuclides from the repository up to man. Geochemical conditions may also be changed at the surface. In most of the regions concerned by the granite massifs in France, the presence of a glacier is a priori excluded for the next 1 million years.

On a generic site, it is hard to define what would be the effect of a permafrost on the evaluation of the SEN. Considering the installation depth of the repository, it would be a priori moderate. It was decided to neglect it in the context of the Dossier 2005; under the hypothesis of an evaluation on a real site, this first approach may be reviewed, if necessary.

5.2.11.2 Climatic evolution: Interglacial periods

A interglacial period warmer or longer than expected in the reference – for example, by taking into account the disturbances caused by man on the climate – would not have a significant effect on the repository or the formation.

Such an episode would push off that much more the next glaciation.

5.2.11.3 Geodynamic evolutions

Horizontal and vertical tectonic movements may be take place at the regional level and modify the hydraulic conditions and the transfer paths within the repository. For an installation of the repository site far away the major regional accidents, the uncertainties related to changes in the hydrogeological, geomechanical and hydrogeochemical contexts are low at least for several hundreds of thousands of years. Beyond this time line, analysis must be carried out specifically on a particular granite massif [vii].

Erosion phenomena can cause significant changes to the site’s topography and hydraulic conditions. Erosion phenomena, and particularly the sinking of valleys, are liable to permanently modify, on the one hand, the underground hydrogeological gradients and, on the other hand, the trajectories of the hydrogeological flows toward the outlets modified at the surface. Generally, changes predictable on a time scale of a hundred or several hundreds of thousands of years are minor and endowed with relatively low uncertainties. Beyond several hundreds of thousands of years, the changes to be taken into account can be more important and the corresponding uncertainties higher, namely with respect to the underground flow trajectories [vii]. In a generic context, it was decided to neglect this aspect.

Movements along the conducting fractures intercepting the connecting drifts are possible and may have repercussions on the seals isolating the repository modules from these fractures. The repository’s design provides for a buffer zone of undisturbed rock with respect to these fractures, which prevents such a risk.

In addition, it does not seem necessary to take into account the occurrence of an earthquake of significant magnitude. The granite massifs are generally characterised by moderate earthquake activity, usually qualified as diffuse. On a defined site, a detailed study of past earthquake events would allow refining this observation; it allows, however, not taking into account an earthquake scenario in the context of a generic study.

5.2.11.4 Volcanism

Volcanic activity in the vicinity of a repository site would have a major impact on the repository’s performances. Under the hypothesis of a site selection, this risk would be treated by design, preventing the repository from being installed in identified volcanic regions (Puys chain, Ardèche volcanism). The appearance of new volcanic zones is, in addition, judged highly unlikely over the next 1 million years [vii]. In this context, the risk of volcanism is not retained.

5.2.11.5 Risks of human origin

Risks of human origin concern only unintentional intrusions. They would be due to a loss of memory of the repository.

It is extremely hard to predict the evolution of the society’s organisation and the technologies which will be accessible to future generations. In accordance with the recommendations of the RFS III.2.f., this type of uncertainty is treated by assuming that the future generations would have a technical level equivalent to ours. A very penalising additional hypothesis would be that they have lost knowledge of what is a radioactive waste, what is a repository in a geological formation and that such a repository might be found underground at a possibly undetermined location.

The borehole can aim at the search for exceptional resources. Some granites evoke particularly hydrothermal activity or the exploitation of uranium deposits; the RFS recommends to consider only sites which are free of this type of resource. Therefore, it is assumed that a repository would be located away from exceptional resources.

Despite being placed far from any natural resource of an exceptional character, a borehole can never be completely ruled out; it can always be motivated by the simple willingness to survey the massif without being aware of the repository's presence. Therefore, this risk is relevant for the definition of altered situations. Considering the preliminary character of the analysis, an altered evolution scenario is not presented, however, in chapter 6 dealing with the case of a bore hole. A borehole constitutes initially a hydraulic disturbance close to that caused by a fracture not identified in SEA"fracturing characterization error". In addition, it should be noted that the repository is partially protected from the borehole by its module separation, which would prevent the propagation of the effects of this bore hole beyond one or several seals. The practically impermeable backfill also contributes to the repository's robustness. The radionuclide immobilisation function also plays a role in replacing other safety functions of the repository, which can be more directly affected by the bore hole.

It should be noted that abandoning the repository before it is completely closed is not fully treated because of the preliminary and generic character of the dossier. The analysis of the consequences of this event would have to be conducted based on an architecture defined in a detailed fashion.

5.3 Conclusions of the qualitative analysis

Examination of the FEPs allowed gradually identifying the relevant characteristics, events and processes in order to better define the phenomenological domain covered by the SEN. In addition, it led to the proposal of several situations not within the framework of normal evolution to be treated by altered evolution scenarios.

It seems clear that one of the limits of this exercice is the absence of a site, which could lead to eliminating some phenomena corresponding to particular locations without a representative character (for example, the risk of intrusion by salty waters nearby the coast). Other processes, even if they are possibly not negligible, cannot be studied in a generic context (for example, the influence of temperature, or that of climates). Finally, the question of granite variability in France leads to favour "test values" representative of the various possible contexts combined with sensitivity studies. These sensitivity studies would also be useful even within a defined site to cover the residual uncertainties of characterization. The uncertainties on the thermal, chemical and mechanical phenomena, which could disturb the granite's characteristics in the near-field, may be covered in the same fashion. Therefore, the sensitivity analyses have both the mission of taking into account the site's indetermination and the uncertainties which would persist even for a given site.

The uncertainties on the structured components are not independent from those on the nature of the site. In particular, the composition of the granite waters can influence the behaviour or the characteristics of the repository's various components. These uncertainties are also investigated based on the test values obtained from international feedback.

With all these limits, qualitative analysis focuses on a number of favourable characteristics of disposal concepts in granite. Good complementarity between the chemical buffering role of the in-depth granites, the usage of clay or concrete engineered barriers, as well as the presence of packagings with adapted lifetime, allow working under conditions which lead to the treatment of most of the FEPs in the form of simple models or to neglect them. The FEPs relevant to situations considered as altered are relatively few in this context. They are grouped together in the form of a few well-defined scenarios.

The selection of the relevant FEPs takes into account the results obtained for the typology of the French granites, namely, concerning the representation of the near- and far-fields, or the phenomena linked to the overall characteristics of the geological medium (thermal conductivity, stress field, for example). The determining factors of the French granite do not lead to the retention of FEP which is particularly restrictive for the repository's safety, or which can be represented in the scenarios. It goes without saying that some FEPs excluded in a generic framework would have to be taken into account in a site selection context (for example, linked to the existence of special geochemical conditions).

At the completion of the analysis, the adaptation of the repository to the fracturation scheme seems, in addition, a key element for controlling the uncertainties and the phenomena to be taken into account. In particular, the control of the hydraulic system within the repository allows limiting the extension of chemical disturbances and favouring a slow degradation of the structured components and the waste.

In the special case of spent fuel, it was decided to not treat the plug failure because radionuclide releases may only occur if the container fails (unlikely double failure situation).

6 Evaluation of Repository Performance During the Post-Closure Phase

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This chapter contains the results of the last stage of the safety approach of the granite dossier 2005. It presents the result of the performance calculations carried out on generic granite sites.

These calculations are based on analyses presented earlier:

- The safety functions (Chapter 3) are represented by components that must provide them and by the parameters and models that best represent the underlying processes.
- The calculations also show the evolution of the different repository components through the PARS and the analysis of the FEP's (Chapter 5). These analyses do not aim at identifying a unique and certain mode of repository evolution but to broadly identify possible phenomena and associated uncertainties. Representation of the most probable evolution by means of what is known as the "normal evolution scenario" (SEN) takes account of this variability and aims to cover uncertainties by selecting conservative models and parameters each time this proves to be necessary. The normal evolution scenario therefore constitutes an envelope representation of the normal evolution domain.

As indicated above, FEP's listed as arising from altered evolutions are not dealt with systematically but the principles are covered by altered evolution scenarios dealt with independently of the normal evolution scenario.

These scenarios cover the main failures likely to affect each of the components: the package ("package failure" scenario), seals ("seal failure or plug failure" scenario), the geological medium and, in particular, the characterization of the average fracturing ("fracturing characterization error" scenario).

Note that the calculation is a performance calculation and not an impact calculation. With no particular study site, there would be no sense in evaluating the safety of a repository on the basis of impact calculated in terms of dose which would depend on specific site environmental conditions. In the context of generic studies of the granite medium, mobilized intermediate indicators used to understand the individual functions of each of the main repository components were preferred:

- Indicators relating to the quantities of water passing through various parts of the repository are used to evaluate repository performance concerning the "preventing water circulation" function.
- The quantity of radionuclides at certain key points of the repository at different phases of its evolution is used to evaluate the confinement performances of the different components and contributes to the identification of the elements of global robustness. More particularly, this indicator concerns the functions of "restricting the release of radionuclides and immobilizing them in the repository" and "delaying and reducing the migration of radionuclides.

6.1 Calculation models

6.1.1 Choice of generic representation

It is important that the choices maid to represent the normal evolution scenario both provide usable results and a certain degree of representativeness of the phenomena studied. This involves favouring simple models and parameters in order to avoid any ambiguity in their interpretation due to the complexity of the models. This also leads to ensure that the representation of the repository is not too conservative and that the performance levels of safety functions are sufficiently representative of the results of scientific studies for usable conclusions to be drawn from them. This is why it was decided to adopt models and parameters that were as close as possible to the phenomenology but which incorporated safety margins to take account of uncertainties. Sensitivity calculations could be performed to provide greater coverage of uncertainties related to knowledge. However, while as the dossier does not aim to be a comprehensive coverage of uncertainties but rather focuses on questions related to granite and the adaptation of architectures to the sites, analyses of the geological medium were given preference over others concerning engineered components.

The models take into account the wastes inventory per reference packages as supplied by the Design Inventory Model. To ensure a conservative approach, quantitative inventories have been increased per reference package for the different management scenarios which conduct to over-estimate the overall inventory (since the same radionuclides can be taken into account twice, in the vitrified waste and in spent fuel depending on whether or not a management scenario including processing is adopted). For this reason, and also in order not to prejudge the overall architecture of the repository, calculations were carried out per module of each type of waste (B, C and spent fuel).

Transfer calculations were carried out, where applicable, up to the limits of the granite massif and integrate the role of regional fracturing. On the other hand, possible surrounding formations and the surface environment have not been taken into account as transfer pathways and they are not explicitly represented by the calculation.

The geological models have been simplified at successive stages of hydrogeological models produced on different scales (regional, massif, repository and repository modules). As well was made the choice of hydraulic and transport parameters to be incorporated and the ranges of values to be taken into account both for the possible variability of these parameters for each of the geological site models and for uncertainties related in particular to techniques for surveying and characterising the fractured medium.

In the context of generic studies of the granite medium, it is not, however, possible to define the degree of conservativeness of the "test values" of parameters used in this way. Such a definition would depend on surveys and characterization work carried out on a specific site. However, a pair of values of the main hydraulic and transport parameters of the granite and fractures has been systematically tested in the normal evolution scenario. Thus, calculations made it possible to determine the sensitivity of the performance of a repository to the various parameters examined, and to obtain orders of magnitude as regards the performance of a repository for the different design options and geological site models in question.

From the temporal point of view, the evaluations are carried out for the case of a repository after closure on a time scale of one hundred years after the beginning of its construction. This does not make assumption regarding the duration of the reversibility phase which is by definition unknown (one or more centuries). This assumption arbitrarily sets a common reference for all calculations. Given the short duration of transient conditions, the repository is assumed to be resaturated and in reducing conditions at the beginning of the calculation.

As regards the geodynamic evolutions of a site in the very long term, the calculations do not take account of any possible modifications of the context of the granite massif that may arise from this (for example to hydrogeology). Besides the fact that they would be insignificant in the geological context of France over a time scale of several tens to several hundreds of thousands of years, such modifications would only be justified in the case of a specific site, which falls outside the generic scope of the studies.

6.1.2 The site geological models

As the research work is conducted on generic sites, it is important to state the conditions in which the granite massifs modelled in the calculation have been defined.

There are relatively few massifs in France that have been surveyed from the geological and hydrogeological point of view with sufficient characterization to enable a model to describe the properties of a specific site. Furthermore, a calculation for a specific site would run counter to the objectives of the dossier as it would remove its generic character that must be. The aim is not to qualify a massif from the point of view of evaluation criteria.

In order to take into account the variability of the properties of the French granite massifs, three geological site models (M1, M2 and M3) have been established based on the synthesis of the knowledge of French granites in the form of a typology. This typological analysis has made it possible to identify and establish a hierarchy of the granite characteristics the variability of which is the most likely to affect the design of a repository (see Chapter 2).

The geological site models were produced upstream of the hydrogeological models and of radionuclides transfer simulations. They have been described with sufficient accuracy to ensure they are geologically consistent and reflect configurations that are representative of the French geological context with a certain degree of realism. They are based on configurations present in the Massif Central and the Massif Armoricain in particular. The construction of the models does not aim to compare sites or claim to provide a comprehensive representation of configurations that may be found. This, based on realistic arrangements, makes it possible to examine the role of the different characteristics on hydrogeology and radionuclide transfer in safety analyses.

Thus, each model deals with:

- Different morphostructural arrangements, i.e. the relationship between the granite massif, surface topography and other terrains surrounding the granite massifs;
- Different fracturing arrangements both for minor fracturing affecting the scale of the repository engineered structures and large scale fracturing dealt with on the scale of the massif ;
- Different granite characteristics from the lithological and mineralogical point of view.

The calculations relate mainly to models M1 and M2.

6.1.2.1 Site model M1

- **Regional geology and morphological arrangement**

The granite massif, which forms the basis for the M1 site model, has a large surface area (several thousand km^2) and is situated in a region with relatively contrasted surface topography: The massif is divided into two main basins drained by rivers the altitude of which is several hundreds of meters below the crestline separating the two basins. This topography is common in the eastern parts of the Massif Central (see Figure 6.1-1).

From the morphostructural point of view, the massif has a dome configuration: Topographically, it overhangs the metamorphic geological formations that surround it. Among the three morpho-structural arrangements dealt with by the site models (dome, basin and sloping), the dome one is characterised by hydrological flows over long distances between underground and the surface outlets.

From the geological point of view, the M1 model aims at testing the influence of major characteristics of this type of massifs on repository performances (see Figure 6.1-1). These characteristics are the result of geological history that is to a large extent common to granites of this type:

- A rock with "average" mineralogical properties for French granite ones; this leads to “average” test values being adopted for hydraulic, diffusion or retention parameters among the range of French granite values;
- A fracturing arrangement that represents the Hercynian structure (approximately 300 Ma) common to the large majority of French granites and also provides evidence of the significant effects of more recent tectonic phases, in particular related to the Alpine and Pyrenean orogens. This provides fracturing patterns taking account of these different phases in the geological history of a granite massif in a way that is consistent with the French geological context.

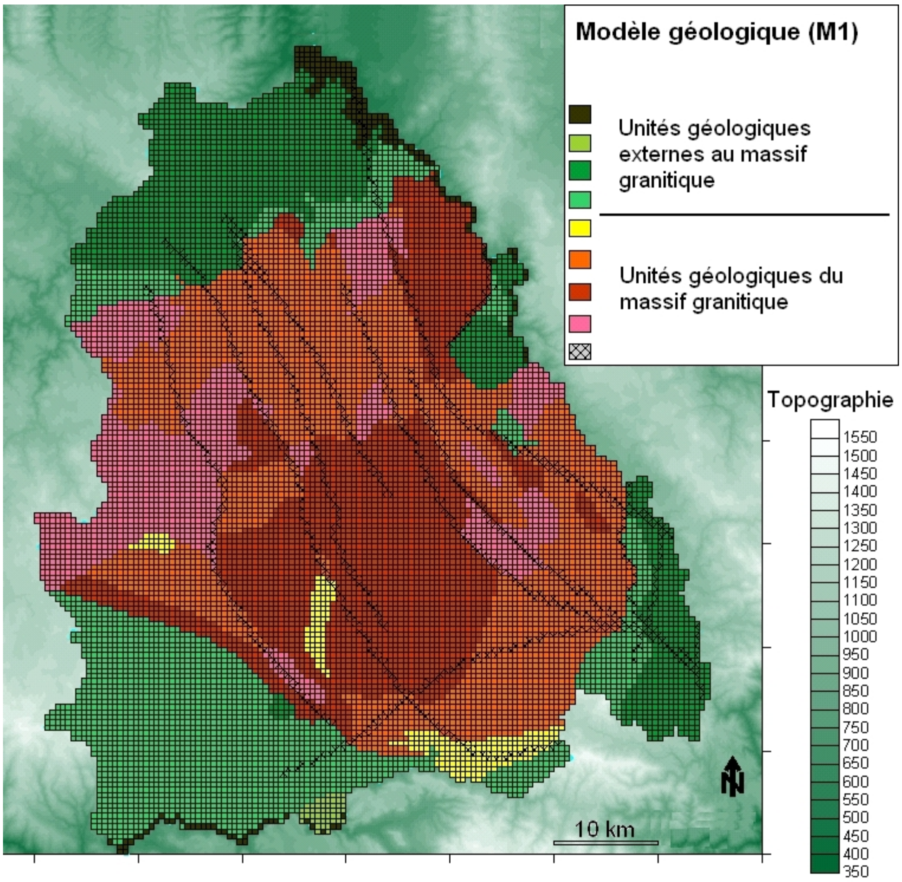


Figure 6.1-1 Geological and topographical context of granite massif M1

These characteristics are mostly found in the Eastern parts of the Massif Central.

The data taken as reference, regarding fracturing in particular, are mostly based on data taken from the field of mining.

The reference granite of the M1 site model is granite of "medium" mineralogical composition within the classification of granites: Monzonitic granite with facies precenting either a high (dark facies) or low (light facies) ferro-magnesium mineral content. Intrusions of a more differentiated type of granite (leucogranites) that is likely to be more permeable than the other units have also been individualised in the regional model.

The major fracturing of the massif is for the most part well arranged in the direction of large structures N150 -170° that may turn in the south of the massif to directions N130-140° or even N90-100°. It is assumed that the directions of horizontal stresses are, consistent with the tectonic context described, N140° in the case of the main stress and N55° in the case of the secondary stress.

- **Geology on the scale of the near-field and the far-field**

Granite rock

The mineralogical differences between the granite facies are not of a type that would modify the generally very low permeability of the rock. For the type of granite taken as a reference (monzonitic granite), it is assumed that any hydrothermal alteration phenomena that may be linked to Hercynian and subsequent tectonics did not significantly modify the diffusive properties of the original rock. The transport and retention values taken as a reference are those of "sound" rock.

Far-field fracturing model

The large and intermediate fracturing model (fractures with an extent in excess of 50/70 meters) has been established in deterministic fashion for an area of 8x11 km sized. It distinguishes four types of fractures that correspond to different dimensions and modes of fracturing. The model was established from geophysical data sets based on stronger or weaker electrical conductivity. The model incorporates the assumption of a link between electrical conductivity and the type of fracture.

Thus, the following are found (see Figure 6.1-2):

- Faults of the 1st order corresponding to the strongest electrical conductivity (approximately 800 faults),
- Faults of the 2nd order that are smaller in size and have weaker electrical conductivity (approximately 4000 faults), their organisation is connected with faults of the 1st order,
- Faults of the 3rd and 4th order whose range is smaller (structures of less than 100 meters) and whose conductivity is weaker.

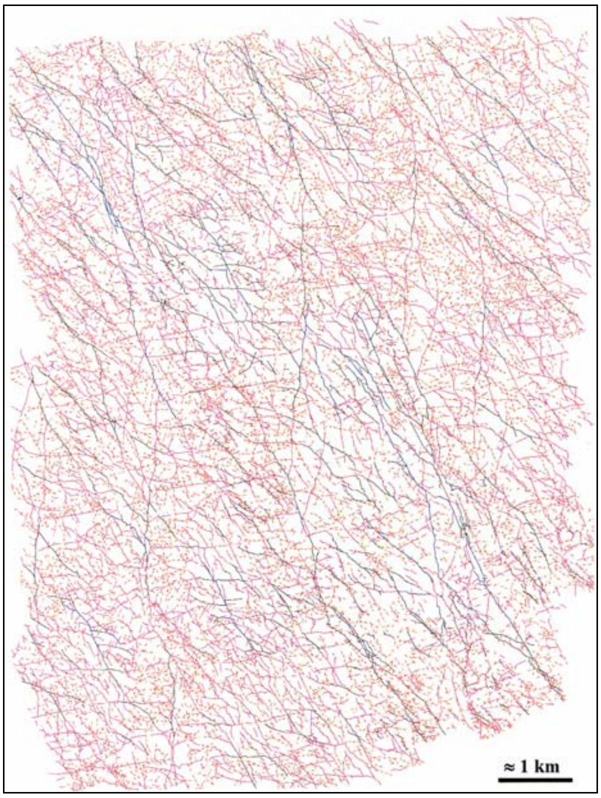


Figure 6.1-2 *2D model of far-field fracturing based on geophysical characteristics (electrical conductivity)*

The predominant direction of the fractures lies between N155°-160° which demonstrates the relatively strict arrangement of the fracturing. Another direction N110-120° has wider distribution. Two directions N20-30° and N70° are represented to a lesser extent.

Strike-slip deformations are the predominant cause of Hercynian fractures or their recurrence during more recent geological periods, meaning that the horizontal component of the deformation is predominant. The connected lens appearance of the 2D model of the fracturing reflects a tectonic style that is characteristic of such deformations (see Figure 6.1-3*)*. This has led to the incorporation of a large form ratio in fault models: meaning, a high ratio between the horizontal extent of the fault and its vertical extent. The fractures are represented in the model by ellipses extended in accordance with this form ratio.

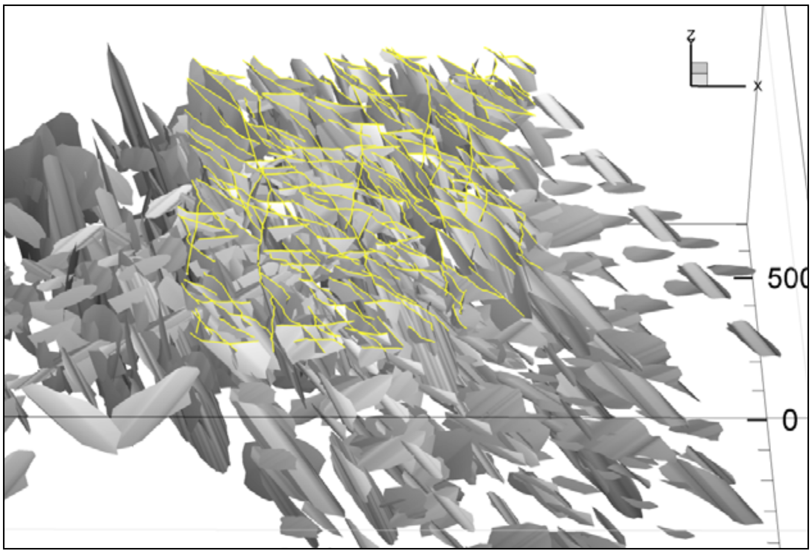


Figure 6.1-3 *Site model M1: 3D models of type 1 faults in a strike-slip style of deformation*

Taking account of the tectonic style, it is assumed that the fault dips tend to be vertical with the following distribution: 70-75°= 5%; 75-80°=15%; 80-85°=30%; 85-90°=50%.

Near-field fracturing model

The near-field fracturing model introduces small-sized fractures into the M1 site model (extent smaller than 50/70 m and greater than 5m). The statistical model is established using fracture generation software based on the distribution elements of minor fractures detected in this type of massif.

Four types of fractures are taken into account depending on their orientation and dip. From a geological point of view, they correspond to fractures originally created by thermal shrinkage of granite magma (mode 1) or during phases of tectonic deformation (mode 2). From a practical point of view, the model assumes that they are all of the same mode (mode 2) which is consistent with the post Hercynian orogenic recurrent phenomena taken into account in this model. Densities of fracturing are intermediate: The total volume density is estimated at 0.1 m² of fracture surface area per m^3 of rock.

As the definition of minor fractures is stochastic, several successive random sampling operations could have been performed in order to weight the effect of a specific random sample. It was decided, in the context of this analysis, to perform all calculations on the basis of a single restricted random sampling. This made it possible to examine the functioning of a given network in greater detail and facilitates comparison between the different scenarios calculated.

The fractures networks generation as described correctly reproduces the distribution of the dimensions, those of the orientations and the average densities supplied as input data by the geological model. The obtained spatial distribution of fractures is, however, probably more homogeneous than it would be in reality. The "homogeneous" generation used in this model doesn’t allow reproducing the "clustering" observed in certain massifs and which tends to increase the density of fractures in certain sectors and to diminish it in others with significant consequences for local connectivity.

6.1.2.2 Site model M2

- **Regional geology and morphological arrangement**

The granite massif, which forms the basis of the M2 site model, has a small surface area (200 km^2) and is situated in a region with a relatively uniform topographical surface: the differences in altitude between the highest points of the massif and the lowest points of the hydrographic network are approximately one hundred meters. This topography is common in the western part of the Massif Central and the southern part of the Massif Armoricain.

Geologically speaking, the massif is more than 3 km deep (see Figure 6.1-4 and Figure 6.1-5). It is made up of several types of granite rock formed during the first phases of the Hercynian orogene (between 355 and 340 Ma). The types of granite rock are varied and range from the point of view of mineralogical composition from the composition of diorite and gabbros (rocks with a low quartz and alkaline feldspar content) to tonalite rocks type (with a higher quartz content). This type of composition is characteristic of a group of granites from the Massif Central to which the Charroux-Civray Massif studied earlier by Andra belongs. As determined in the case of the Charroux-Civray Massif, the massif is assumed to have undergone hydrothermal alteration during the Hercynian orogene and tectonic phases of the secondary. This results in a clogging of the fracturing and, therefore, an overall reduction in the permeability of the granite.

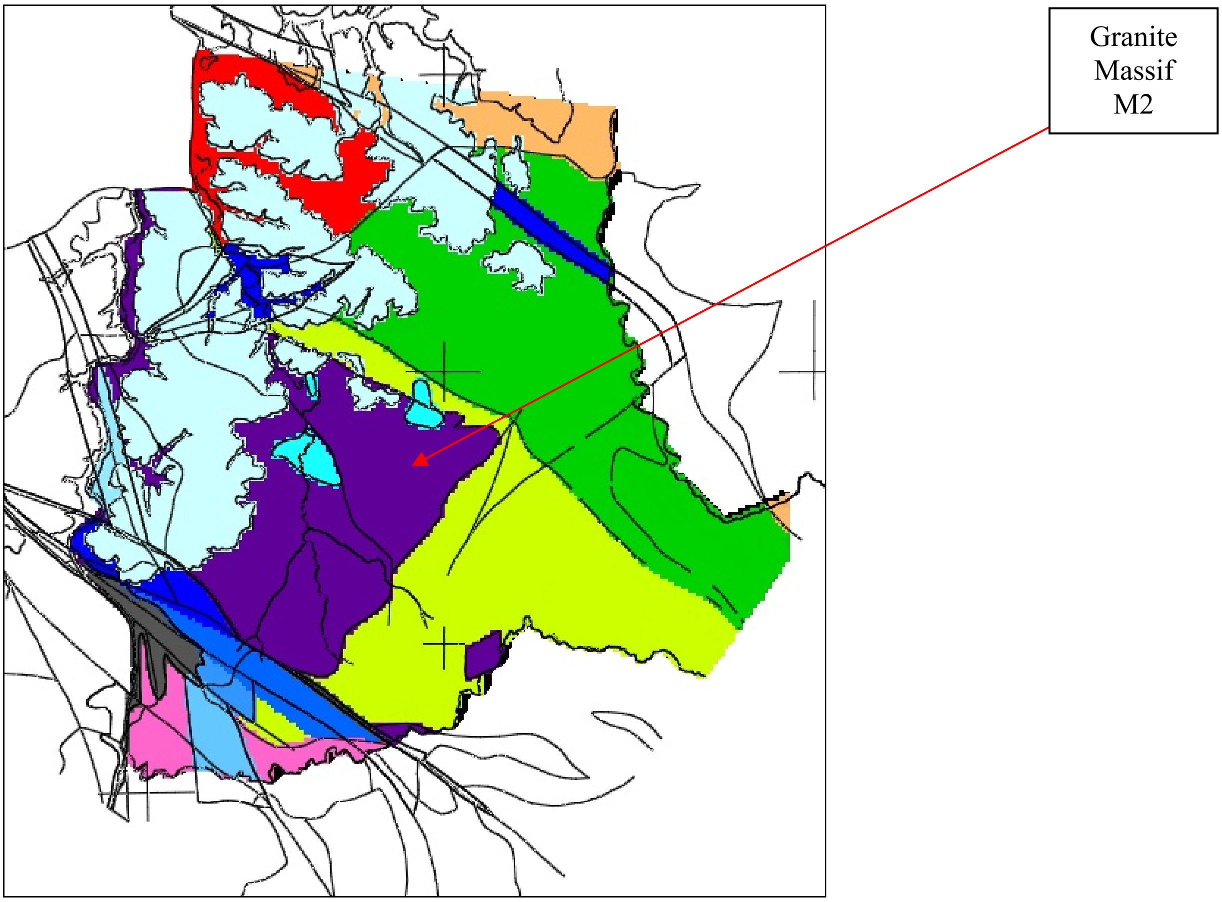


Figure 6.1-4 Geological context of granite massif M2

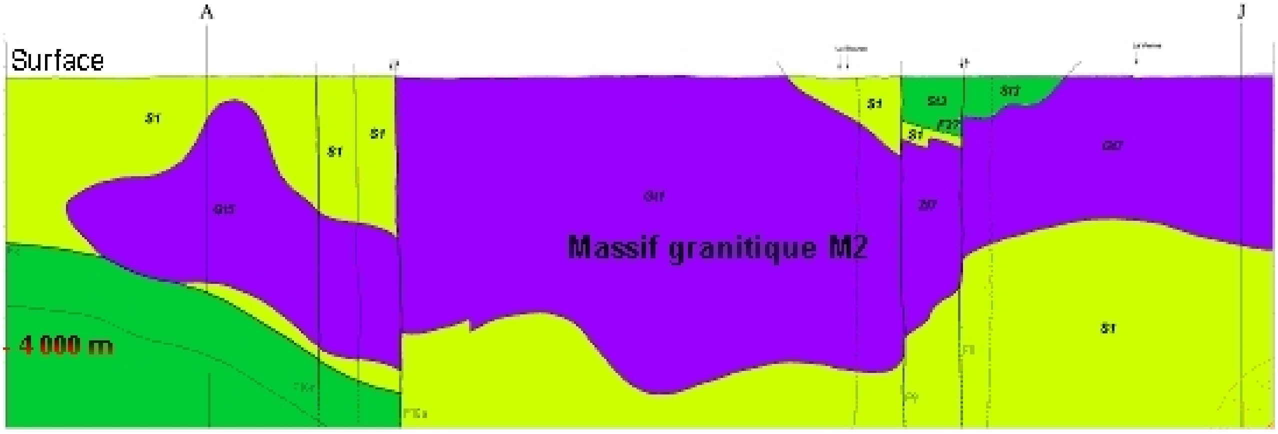


Figure 6.1-5 Vertical geological section of granite massif M2

From the morphostructural point of view, the massif is a “depression” compared with the surrounding geological formations, configuration which tends to hydrogeological flows over short distances between the depth and the surface outlets.

The major fracturing of the massif is made up of faults mainly in the directions N130-170° and N60-90° and a near vertical dip according to a pattern of faults inherited from the Hercynian orogen. The structures recurred during later geological periods. However, the intensity of these recurrences is less than in the case of M1 site model.

- **Geology on the scale of the near-field and the far-field**

Far-field fracturing model

The large and intermediate fracturing model (fractures with an extent greater than 80 meters) has been produced in two ways depending on the extent of the fractures.

Faults greater than 300 m long have been inventoried deterministically based on geophysical data and the transposition of the results obtained for the Charroux-Civray granite massif of the same type. The directions of these faults are essentially from Sector N130-140° and N160-170°. Directions N50° and N80-90° are secondary and are represented to a lesser extent. Their dips have been determined statistically within consistent ranges of variation.

Faults between 80 and 300 meters long have been modelled statistically based on data from boreholes drilled in the Charroux-Civray massif. They are processed in the same way as minor fractures incorporated in the near-field model (see below).

As in the case of the M1 model, the 3D model takes account of the dip of the faults and the tectonic style of the deformations that gave rise to them during the Hercynian or that led to their subsequent recurrence. The dips are 60 to 70° (towards the SW for the main directions). Deformation is mostly of the strike-slip type, meaning that a horizontal component of the deformation is predominant: However, unlike the M1model, the existence of other types of deformations (normal or inverse faults) has led to the adoption of a maximum form ratio of 3 for fracture representation (in the M2 model, the fractures are represented by rectangular planes).

Faults clogging with minerals of hydrothermal origin is one element characteristic of the M2 site model. It reduces the transmissivity of the fractures.

Near-field fracturing model

The near-field fracturing model introduces small-sized fractures into the M2 site model (extent smaller than 80m and greater than 5m). The model, statistical, is produced by generating fractures on the basis of data from boreholes drilled in the Charroux-Civray massif of the same type. However, in order to ensure that the distribution of minor fracturing is geologically consistent with that of large structures, the orientations of minor fracturing (and intermediate fracturing) have been distributed according to their proximity to the large regional faults. They also take into account existence of a particular facies of rock (coarse-grained porphyritic granite) in some part of the massif. The main directions of fractures are N35°, N15° and N155° in accordance with the parts of the massif. The dips of the fractures are generally 60° NW or SW. The assumed density of fracturing is the one determined through borehole surveys of the Charroux-Civray Massif. The total average fracturing density per volume incorporating all minor, intermediate and large fracturing varies from 0.3 m^{-1} to 0.6 m^{-1} depending on the sector.

6.1.2.3 Site model M3

In the case of this dossier, the site model has been dealt with from the geological point of view, which is presented below. It supplies the data necessary for the hydrogeological and transport modelling.

- **Regional geology and morphological arrangement**

The granite massif, which provides the basis for the M3 site model, has an intermediate surface area (approximately 600 km^{2}), and is situated in a region with a relatively uniform surface topography: the differences in altitude between the highest points of the massif and the lowest points of the hydrographic network are approximately one hundred meters. This topography is common in the western and northern parts of the Massif Armoricain.

From the geological point of view, this massif considers a type of granite whose geological history is, for a large part, older than the one taken into account in models M1 and M2. The granite massif took place during the Cadomian orogen (550 million years). It has undergone the effects of the Hercynian orogen at the origin of the granite massifs of models M1 and M2. However, like model M1, it is made up of a rock that is mineralogically close to the average in the classification of granites: granodiorite. Because of the way it set up during in the Cadomian orogen, it generated a contact metamorphism on the edge of the surrounding geological formations that resulted in a modification of the hydraulic properties of these formations.

From the morphostructural point of view, the massif slopes from east to west. It is the third case of morphostructural arrangement studied in the site models.

This type of massif is present in the northern part of the Massif Armoricain. The characteristics of the model are taken from the geological and geophysical data of a massif of this type.

At the scale of the massif, the fracturing is characterised by NW-SE fracturing paths resulting from the Cadomian orogen and recurrent faulting during the Hercynian orogene and subsequent tectonic phases. The extensive tertiary tectonics, in particular, contributed to the recurrence of these fracturing paths. Beyond these paths, the recurrences are of a lesser amplitude. These fracturing paths correspond to a densification of minor and intermediate fracturing without constituting continuous identifiable faults. Beyond these fracturing paths, faults with relatively variable directions intersect the granite (see Figure 6.1-6).

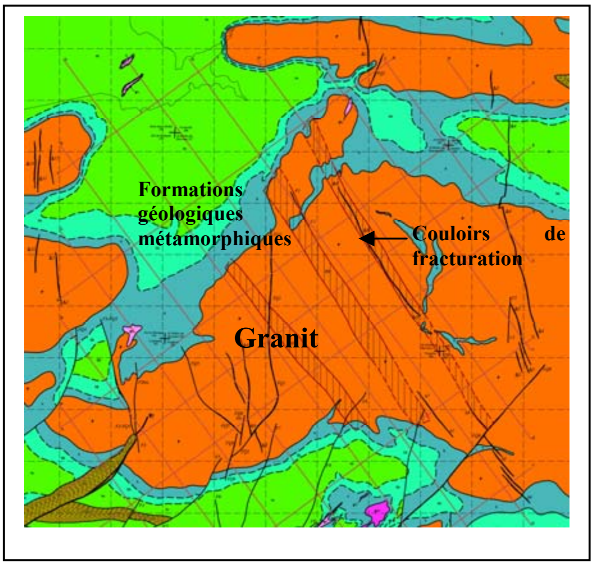


Figure 6.1-6 Geological context of granite massif M3: model at a depth of 500 meters

- **Geology on the scale of the near-field and the far-field**

Far-field fracturing model

The large and intermediate fracturing model comprises fractures over 70 meters long and has been produced in the same way as model M2.

Faults over 300 m long are represented in a deterministic fashion. Smaller intermediate faults are dealt with statistically.

A characteristic element of this type of granite is the existence of sufficiently large veins of quartz, dolerite and aplite to constitute significant hydraulic conductors. The main veins of these types of rocks have therefore also been identified in the model together with the faults and fracturing paths.

Near-field fracturing model

The near-field fracturing model takes minor fractures into account in M3 site model (extent smaller than 70m and greater than 5m). The model, statistical, was produced by generating fractures based on data from outcroppings on the granites. The main directions of fracturing are N145° to N165°. Besides the paths with a high density of minor fracturing, the density of this fracturing is intermediate.

The granite rock is assumed not to have undergone hydrothermal alterations that would have significantly modified its mineralogy and therefore its diffusive properties. The diffusion coefficient values used are those of "sound" rock.

6.1.3 Hydrogeological modelling

The hydrogeological modelling process of the sites includes two stages:

- *Regional hydrogeological modelling.* It is aimed to understand and represent the hydraulic functioning of the entire granite massif and its surroundings in a simplified manner. It makes it possible to propose repository positions as a reference for the calculations. From the methodological point of view, it sets the boundary conditions of the hydrogeological models in the repository near-field and far-field in the granite massif. Given the large volumes of rock to be taken into account, the regional modelling cannot explicitly take into account the detail of the fracturing of the granite massif, the approach adopted is a "continuous porous media" approach (named hereafter CPM[26]).

[26] CPM "continuous porous media"

- *Hydrogeological modelling of the granite in the repository near-field or far-field*. Its purpose is to represent the hydraulic paths that could convey radionuclides up to the surface via the network of granite fractures, and supply the necessary parameters to the transport and retention calculation. This phase employs embedded models that enable the granite fractures to be explicitly represented in the form of discrete fracture networks (hereafter referred to as "DFN[27]"). These DFN models are placed in a larger CPM model to enable paths to be followed up close to the surface.

In this dossier, the modelling approach has been applied to the two geological M1 and M2 site models. The presentation of the methodology of the following hydrogeological modellings is based on geological site model M2. Where applicable, the differences for the M1 model are stated, in particular as regards to the dimension of the models and the digital tools used. M3, for which there are only a few calculations, is not mentioned.

6.1.3.1 Regional hydrogeological model in continuous porous media (CPM)

The dimension of the regional model gives rise to a "continuous porous media" (CPM) processing. The influence of fracturing is integrated into the hydraulic parameters of the different geological units of the model.

In addition, in view of the long term transport simulations, the hydraulic calculation is performed in *steady-state conditions,* in accordance with an entirely saturated granite massif that undergoes no modification during the periods of time taken into account in the models (1 000 000 years). In the context of generic studies, the impact of geodynamic evolution of the site over the long term is not modelled.

- **The parameters of the model and their incorporation**

Extent of the model

The model is sufficiently large (approximately 30 km x 30 km in the case of M2) to cover the entire massif and its surroundings up to the most significant hydrogeological boundaries (crest lines and major rivers).

Hydraulic parameters

The hydrogeological data which allow parameterizing the various components of the geological model are as follows:

- *Equivalent permeabilities of the different types of rock at a depth of 500 m*. Permeabilities of the different granite units of the massif are assumed to be equivalent, with the exception of zones with more permeable leucogranites. The test values of model M2 permeability take account of the mineralogical characteristics of the granite and are based on measurements from the Charroux-Civray massif of the same type. The permeability of the surrounding metamorphic terrains is either equivalent to that of the granite units or to that of the leucogranite zones for the a priori most permeable formations. Permeability is given in the form of a permeability tensor, the major component of which is orientated in parallel to the main constraint, i.e. horizontally and at N160° in the case of massif M2.
- *Permeability of large fracturing:* High permeability which is considered equivalent for all structures described ($K = 10^{-7}$ m/s) is assigned to regional faults of ten or more kilometres.
- *Laws of evolution of the permeability as a function of depth*. Permeabilities of these different components of the regional model evolve with depth with an inflexion at a depth of 350 metres. The laws of evolution were established based on the references available on this subject on an international level (see Figure 6.1-7). In particular, it has been assumed that the permeability of large faults would not diminish further beyond a depth of 350 meters.
- *Kinematic Porosity*. Values equal to 1.10^{-4} for rocky volumes, and 5.10^{-4} in the case of faults zones, have been adopted.

[27] DFN "discrete fracture network"

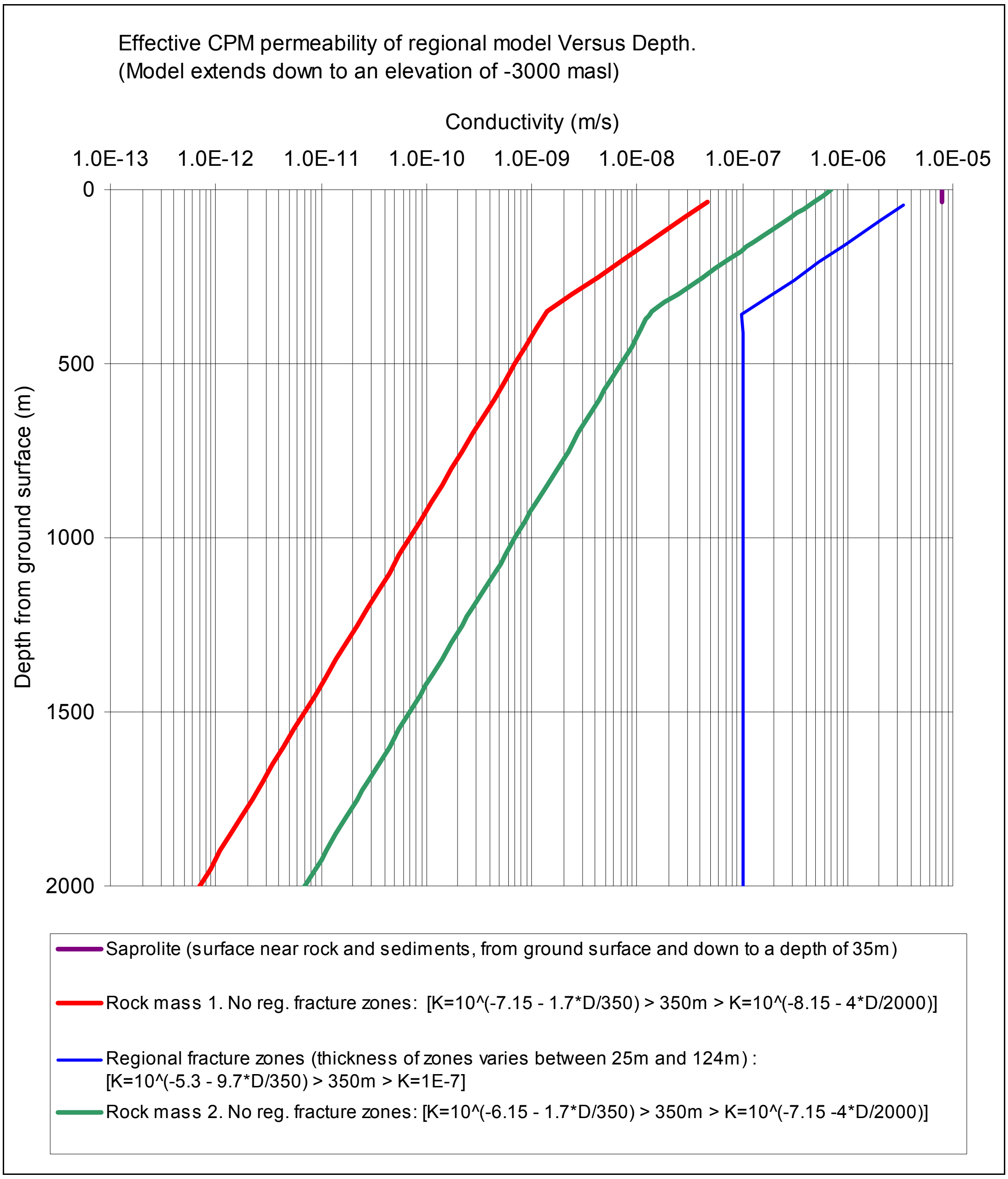


Figure 6.1-7 *Evolution of the permeability ("Conductivity") of the granite massif with the Depth ("Depth below ground level")*

The used calculation model determines an "equivalent permeability tensor" for each mesh taking into account laws of decrease with depth and the possible presence of faults (see Figure 6.1-8).

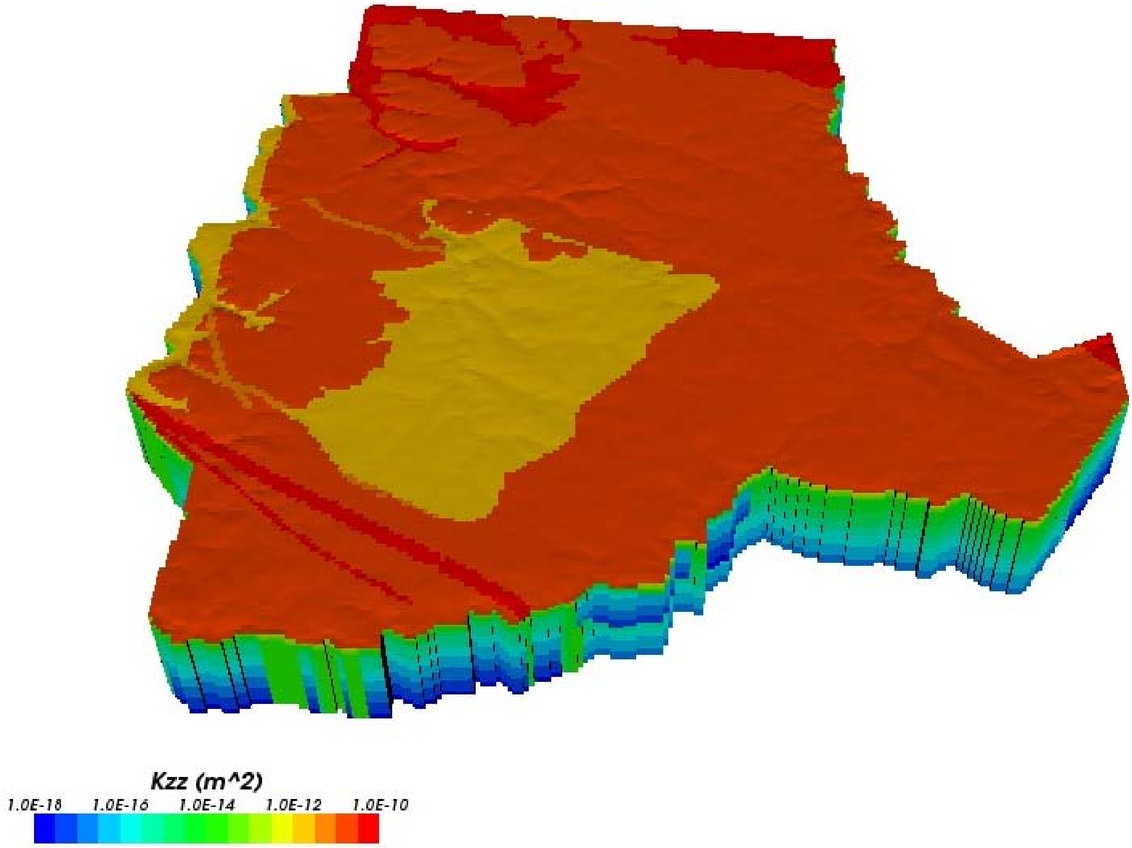


Figure 6.1-8 *Permeability (in m²) presented into the regional hydrogeological model showing the granite massif to be less permeable than its surroundings and the evolution of permeability versus depth*

Boundary conditions

The upper boundary condition follows a law of replenishment/drainage depending on the difference in level between the calculated piezometry and the meteorological input (there is drainage towards the surface when the difference is positive and replenishment of the massif when it is negative). The rainfall likely to percolate is set at 25% of 750 mm/year.

The lateral extent of the model is chosen in order to allow the use of no flow boundaries on the vertical faces: large water courses and groundwater dividing lines. The bottom of the model is assumed to be nil flow (z = - 3000 m).

- **Regional model and positioning of a repository**

This model provides a basis for the calculation of hydraulic paths between the repository and potential natural outlets for different possible locations. Thus, positions can be proposed for a repository depending on the criteria defined by the designer: it is, for example, entitled to favour positions leading to long hydraulic paths or slow transfer times or low water flows. The final selections also take account of the criteria established by Basic Safety Rule III.2.f such as observance of a minimum distance from large faults. In terms of methods, the calculations lead to the production of 3D models of the piezometry (see Figure 6.1-9) and Darcy velocities (see Figure 6.1-10).

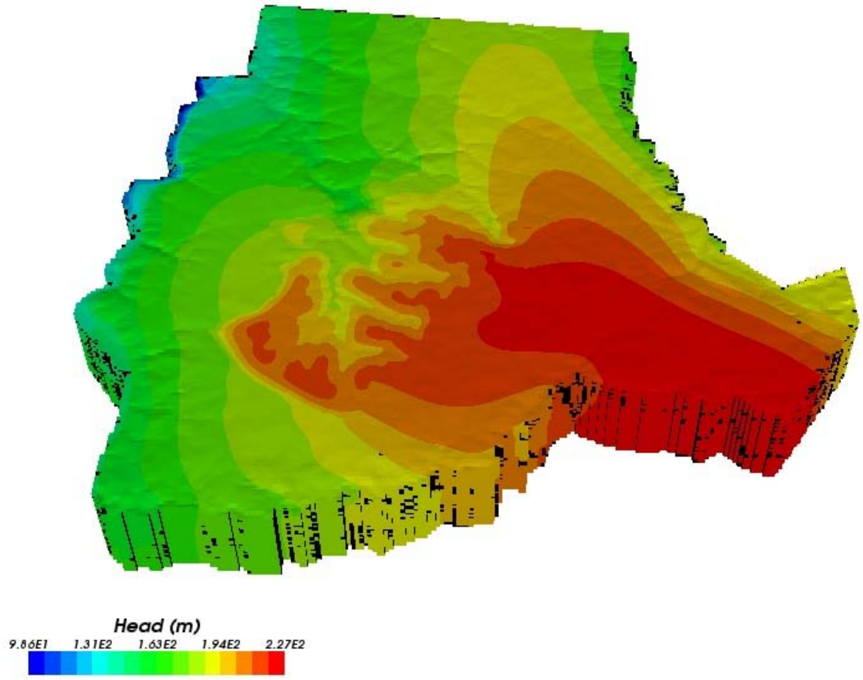


Figure 6.1-9 Calculated piezometric surface (Head in meters)

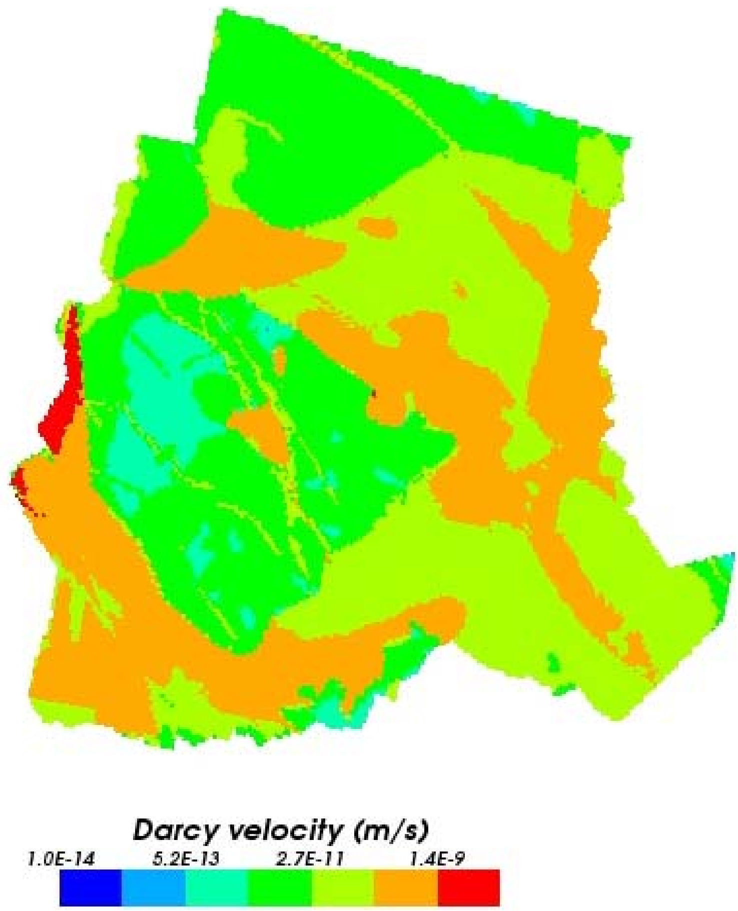


Figure 6.1-10 Absolute values of Darcy velocity at -500m

The technique of “particle tracking” from underground in the system allows to determine the distribution of hydraulic paths passing through a repository in the granite massif. In particular, this makes it possible to compare the repository positions.

Over 8000 particles were "released" at a depth of 500 m on a square grid network with 200 m sides.

“Travel time” (see Figure 6.1-11) and "path length" (Figure 6.1-12) bar charts have been established. A "favourable" location and an "average" location for the repository have been adopted on the basis of cartography of the travel times towards the surface along the paths from each release point.

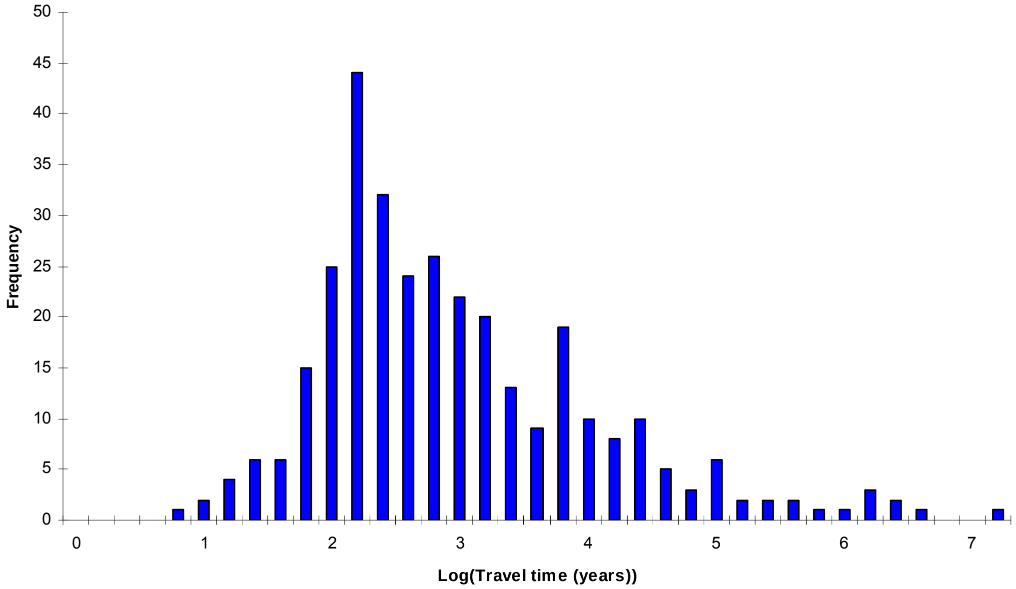


Figure 6.1-11 Frequency of travel time [325 particles distributed through the granite]

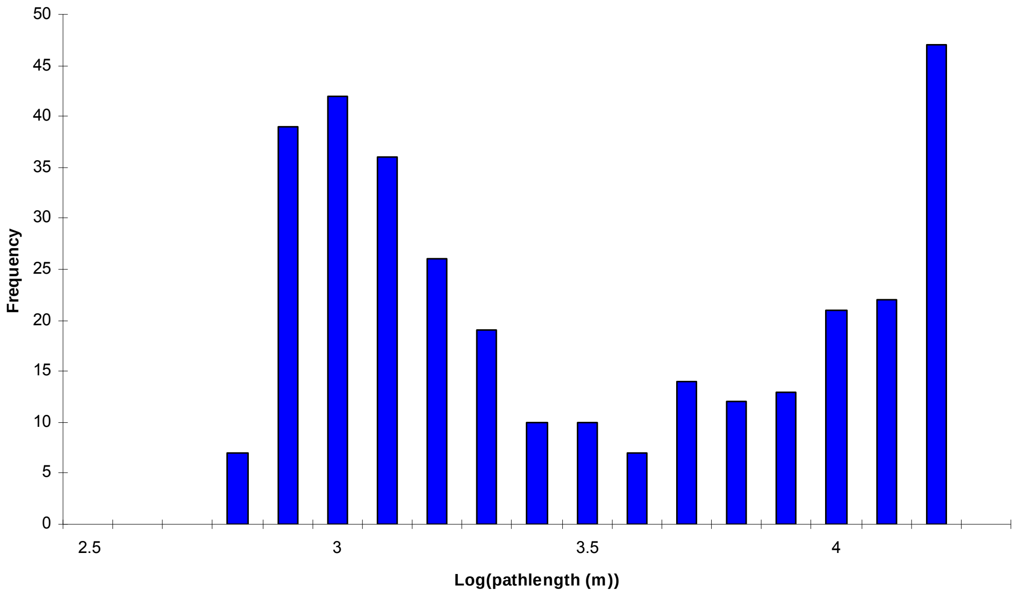


Figure 6.1-12 Frequency of path length [325 particles distributed through the granite]

- **Specific properties of the regional treatment of M1 site model**

The regional model of site M1 covers a larger surface area than M2 site model, (approximately 80 km x 60 km). In addition, given the higher and more contrasted topography, the thickness modelled ranges from approximately 3,500 to 4,500 m for a bottom at depth of - 3,000m.

The treatment used is very similar to that used in the case of M2 with two methodological differences:

- The treatment of the upper boundary condition involves a surface flow module depending on the local topography. This results in a non linear distribution of recharge and drainage areas, and adapted for this massif with a varied landform;
- The repository positions used afterwards for the calculations have been chosen on the basis of composite criteria of the path times (as in the case of M2), the path lengths and the hydraulic flow rates (see Figure 6.1-13).

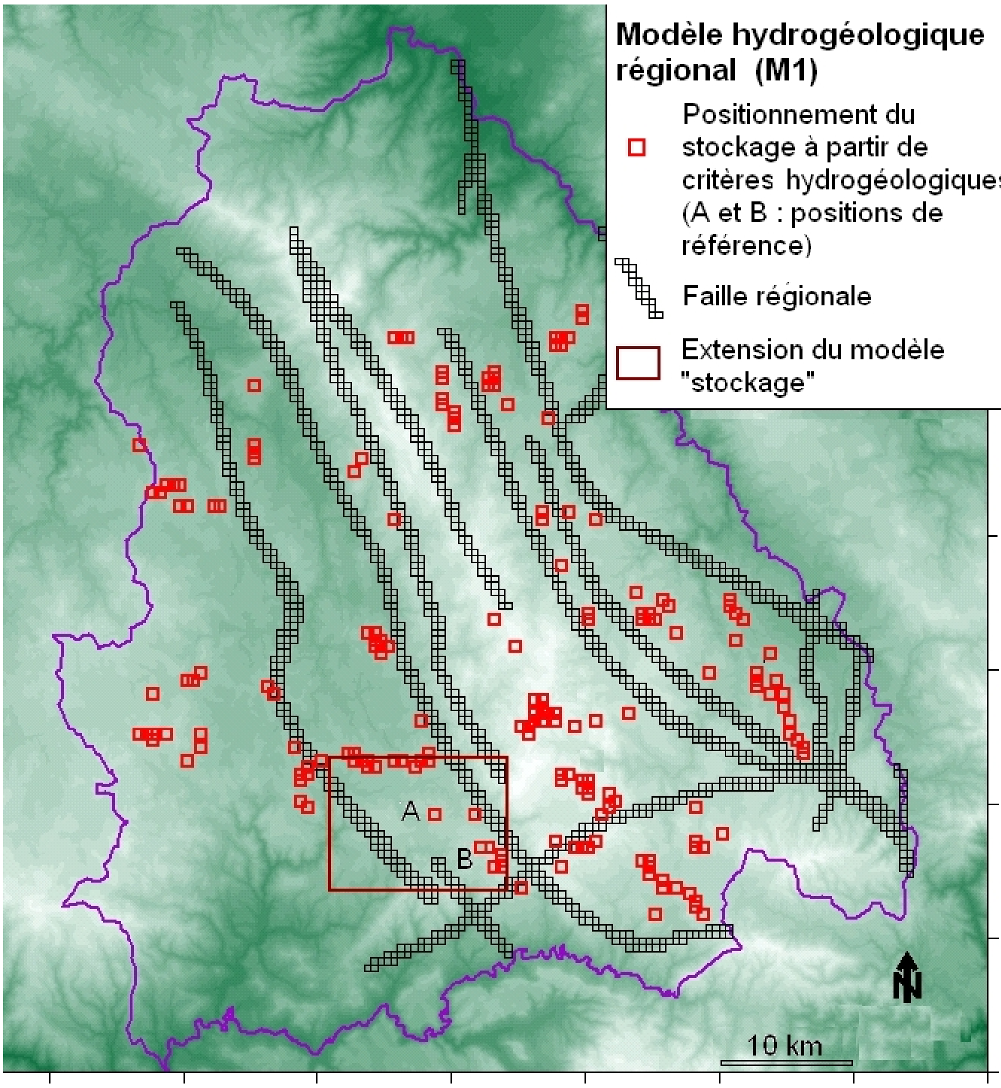


Figure 6.1-13 Regional hydrogeological model M1

6.1.3.2 Hydrogeological modelling in repository near-field or far-field: composite DFN and CPM model

The purpose of hydrogeological modellings on the scale of the repository and the granite massif are to establish the hydraulic paths between a repository and the surface, which forms the basis of transport calculations.

After positioning the repository from a regional hydrogeological model established by a CPM model, the hydrogeological models specify the paths by taking into account the distribution of fractures in the massif.

On the scale of the repository near-field (km^2 or fraction of km^2), the hydrogeological models in discrete fracture networks (DFN) explicitly take the hydraulic properties of the fractures into account. However, the representation of the fractures is simplified:

- The adopted pattern takes the form of two flat, homogeneous, and parallel, fracture walls that define the boundaries of a very narrow open space available for the advective movement of water. The parameter used is the transmissivity of fractures.
- The properties of the rock at the wall of the fractures are likely to give rise to transfer phenomena by diffusion between the open space and the rock: the parameters involved are diffusion coefficients and the porosity of the altered rock at the edge of the fractures. On the scale of the discrete fracture network, the connectivity between two fractures is always assumed to be perfect, which ensures a generally conservative approach in terms of the overall permeability of the massif.

In terms of method, the main constraint of the DFN approach arises from the large number of fractures to be taken into account individually and simultaneously in the hydraulic calculation. From the digital point of view, a compromise must be found between the extent of the volume modelled and the dimension of the smallest fractures taken into account that are also the most numerous. Thus, part of the "far field" ($decakilometres^2$) can be dealt with using a discrete fracture networks approach (DFN). The DFN model is embedded in a larger continuous porous media (CPM) model. The embedding of the different DFN models (near-field and far-field) and CPM (far field) is described below.

- **Hydrogeological parameterization of the faults and fractures of site model M2**

In a common manner for the different scales of fracturing, the model considers a ratio of 1/1000 between the thickness of the structure and its maximum extent. This ratio is a reference value, it is common one in generic modelling exercises and consistent with the observations made for the fracturing of granite. In the case of a particular site, this ratio should be specified especially according to the type of deformation of the granite in question. Working from this assumption, hydrogeological parameterization is performed for the different faults and fractures in the following manner:

- In the case of "deterministic" regional faults, fracture zones are considered to be sufficiently thick to be able to be treated as continuous porous media. A single permeability value is applied for CPM regional modelling (see Figure 6.1-14);
- The principle of the permeable fracture zone is the same for "semi-deterministic" large fracturing, but the permeability assigned depends in this case on the dip of the structure and its orientation relative to the main constraint. A specific permeability value selected in a range of $1.5 \cdot 10^{-7}$ m/s - $7.5 \cdot 10^{-9}$ m/s is therefore assigned to each structure of this type (see Figure 6.1-14);
- In the case of "stochastic" minor and intermediate fracturing, the hydraulic properties are no longer expressed in terms of permeability but in the form of transmissivity. The model envisages proportionality between the extent of the fracture and its thickness, and its transmissivity is deduced by applying a "cubic law" which governs hydraulic systems with perfect parallel plates (see Inset 4 and Figure 6.1-15).

Inset 4 *Equation of the flow rate of a fracture assuming the cubic law is valid*

$$Q = \frac{b^2 \cdot \rho_w \cdot g}{12\eta_w} \cdot l \cdot b \cdot i$$

where

Q : Flow rate (m^3/s)

b : Theoretical thickness of the fracture assumed to be flat and with parallel sides (m).

ρ_w : Density of water (kg/m^3)

g : Gravitational constant (m/s^2)

η_w : Dynamic viscosity (Pa.s)

l : Width of fracture in question (m)

i : Hydraulic head (m/m)

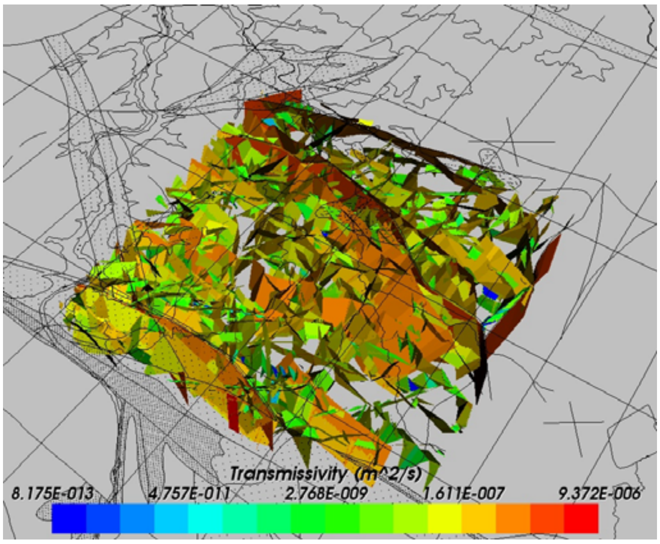


Figure 6.1-14 *"Deterministic" regional faults, "semi-deterministic" faults and large fractures*

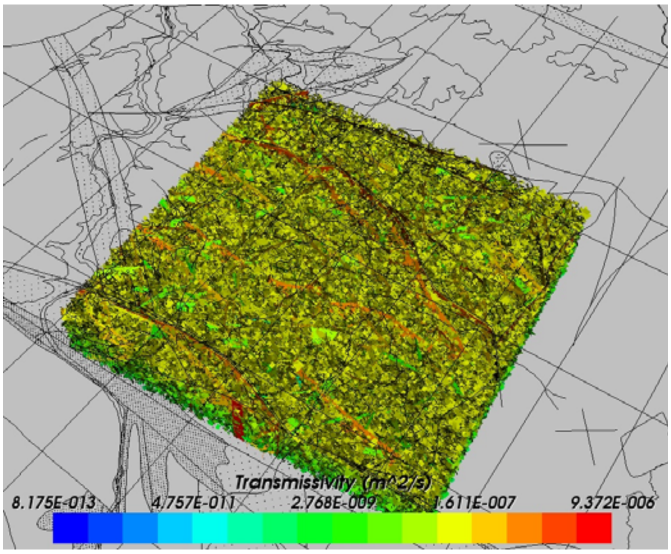


Figure 6.1-15 *Addition to the network of intermediate fractures with "stochastic" definition*

In the case of this model M2, the consistency of the transmissivity values of the fractures assigned with the measurements carried out on the Charroux-Civray massif of the same type has been verified. The results of the hydraulic tests carried out on this massif have been compared with simulations of hydraulic tests carried out on virtual boreholes placed in the M2 DFN models (Figure 6.1-16). This comparison allowed to demonstrate the validity of the method and to adjust the distribution of the fractures transmissivity in a consistent way with the geological context of model M2.

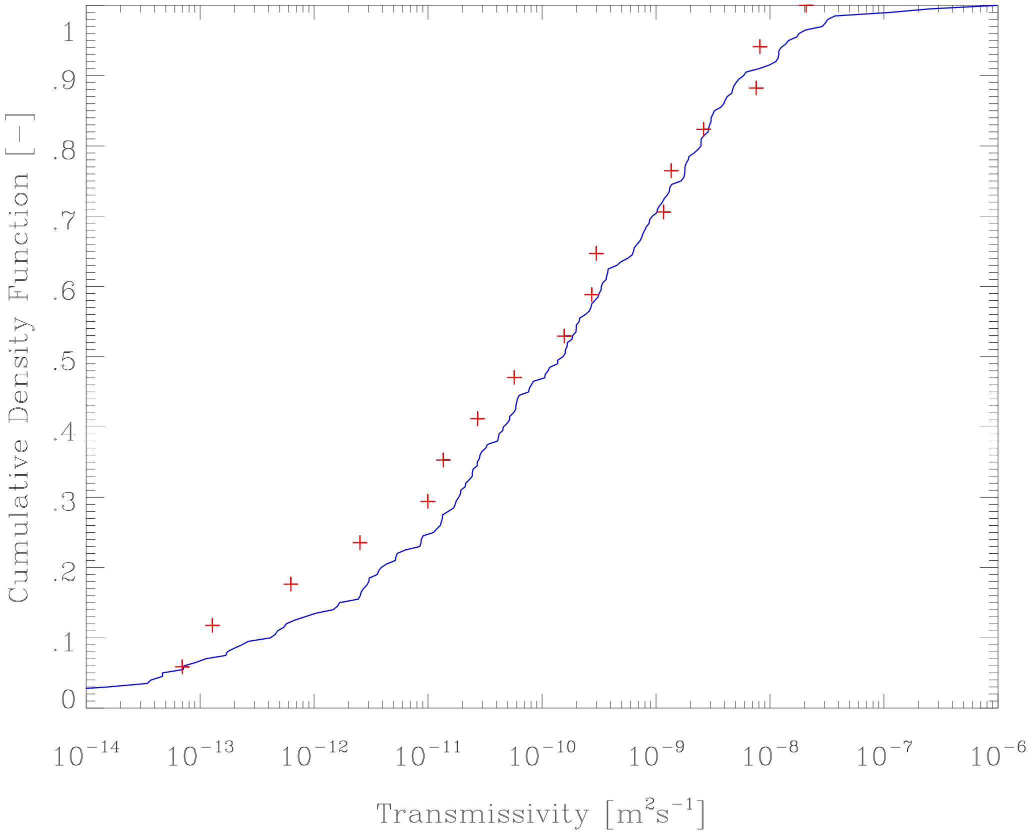


Figure 6.1-16 *Comparison of the cumulative density of transmissivities assigned to sections of the boreholes tested in vienne (in red) and transmissivities simulated for equivalent sections of virtual boreholes incorporated in model M2 (in blue)*

- **Determination of the hydraulic paths in massif M2**

Hydraulic paths, which form the basis of transport models, are determined from the repository cells up to the near surface by means of three embedded models: near-field model in DFN, far-field model for the DFN part and the far-field model for the CPM part.

Current techniques can be used to link these models: the potential fields and the flux balances are the main parameters that enable iteration and consistency between the models. Thus, the composite model successively resolves the potential field resolution for all fractures and for all mesh treated in "continuous porous media".

- In the near-field model, the components of the cells and the modules were treated in continuous porous volumes. The model takes into account both minor fracturing and intermediate fracturing. The dimensions of the model are 450 m x 350 m x 200 m in the case of B2 and B5.2 wastes (see Figure 6.1-17), and 750 m x 500 m x 200 m for C2 and CU2 waste;

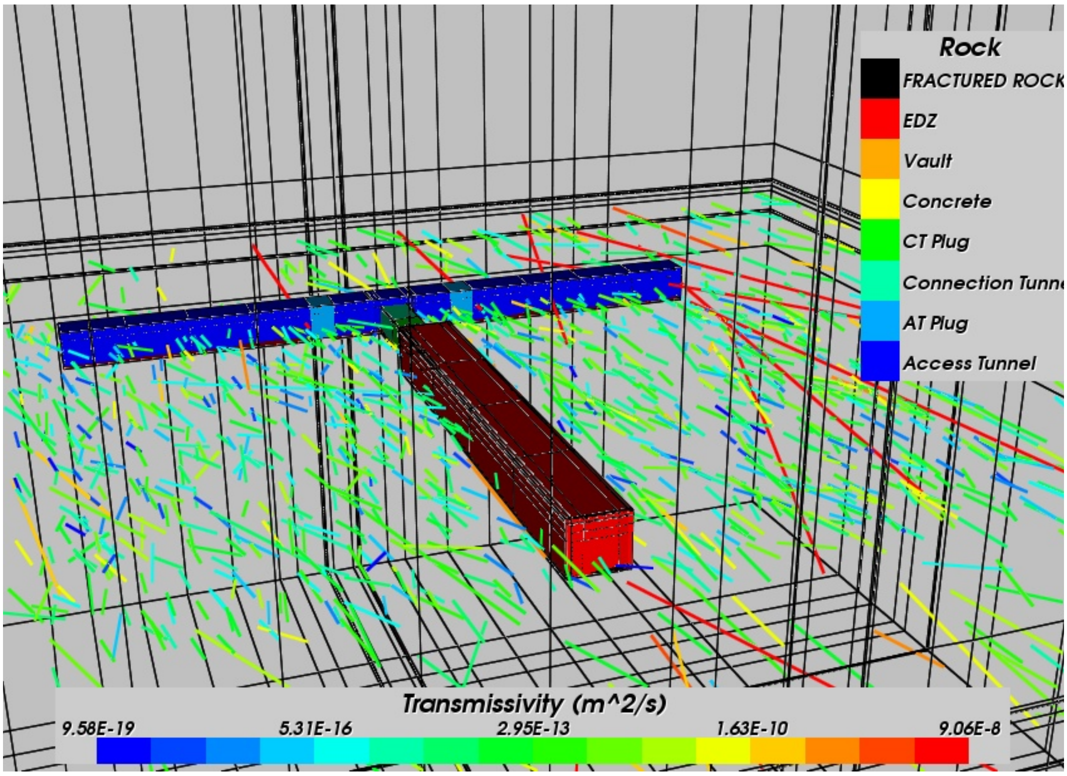


Figure 6.1-17 *Model on the scale of the module and "near-field granite". Constituent elements of the repository (module B5.2) dealt with in "continuous porous" volumes and arrangement of the fractures intercepting a horizontal plane.*

In the DFN models, the hydraulic gradient is determined for each fracture modelled. It can significantly change direction and value at each fracture intersection. A fracture with the same transmissivity may or may not be circulating depending on its direction (see Figure 6.1-18);

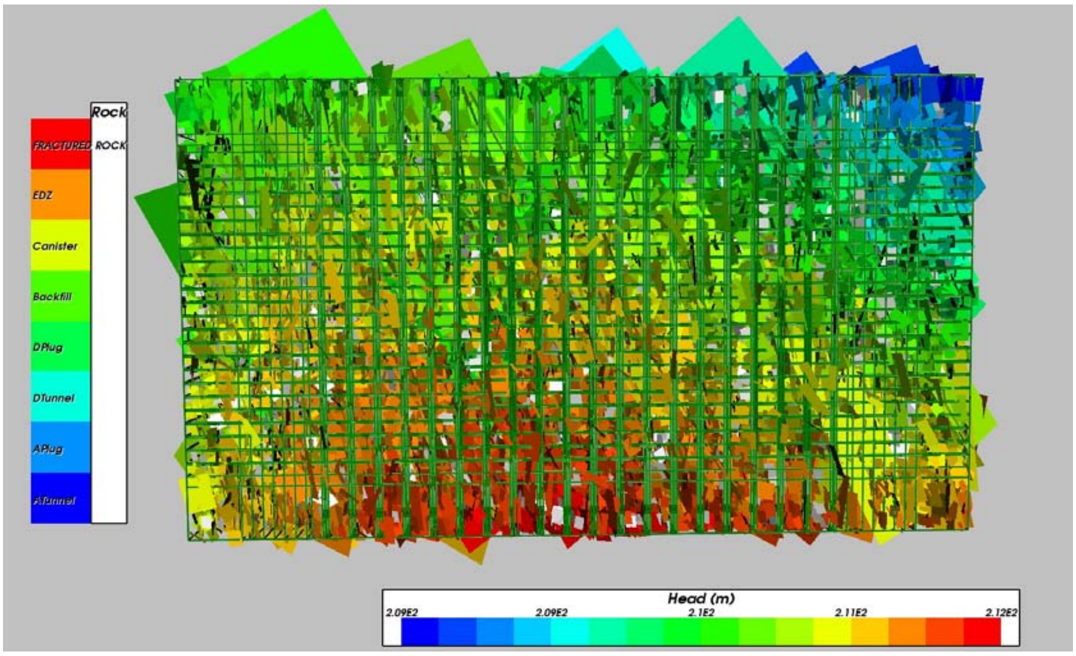


Figure 6.1-18 *Example of hydraulic resolution at the scale of a C2 module. Fractures are classed by hydraulic potential*

- In the far-field model for its part dealt with in DFN, most of the flows go through intermediate or large fracturing; minor fracturing is disregarded with no significant modification of the hydraulic of the system. The volume modelled is 5000 m x 3000 m x 2000 m (z = 0 to z = - 2000 m);
- The part of the far-field model dealt with in continuous porous media (CPM) is built around the two preceding models with a dual objective: To supply the DFN models with boundary conditions that are consistent and representative of the regional hydrogeological environment and to allow the prolonging of hydraulic paths that have reached the external faces of the far-field model in order to follow them up to the near surface.

  The model is first produced on a square surface 12 km x 12 km and over a thickness of 2000 m (z = 0 to z = -2000m).

  Because of its dimensions, this model must be treated in "continuous porous media". The permeabilities of the mesh are obtained using a technique known as "up-scaling" which allows to calculate the equivalent permeability of a discrete fracture network. Permeability values obtained in this way incorporate, on the scale of the massif, the effect of minor and intermediate stochastic fractures and that of large fractures and faults. Once the permeability values have been calculated and incorporated into the mesh, the lateral faces of the CPM model are sectioned following the hydrogeological boundaries of the site (see Figure 6.1-19 and Figure 6.1-20).

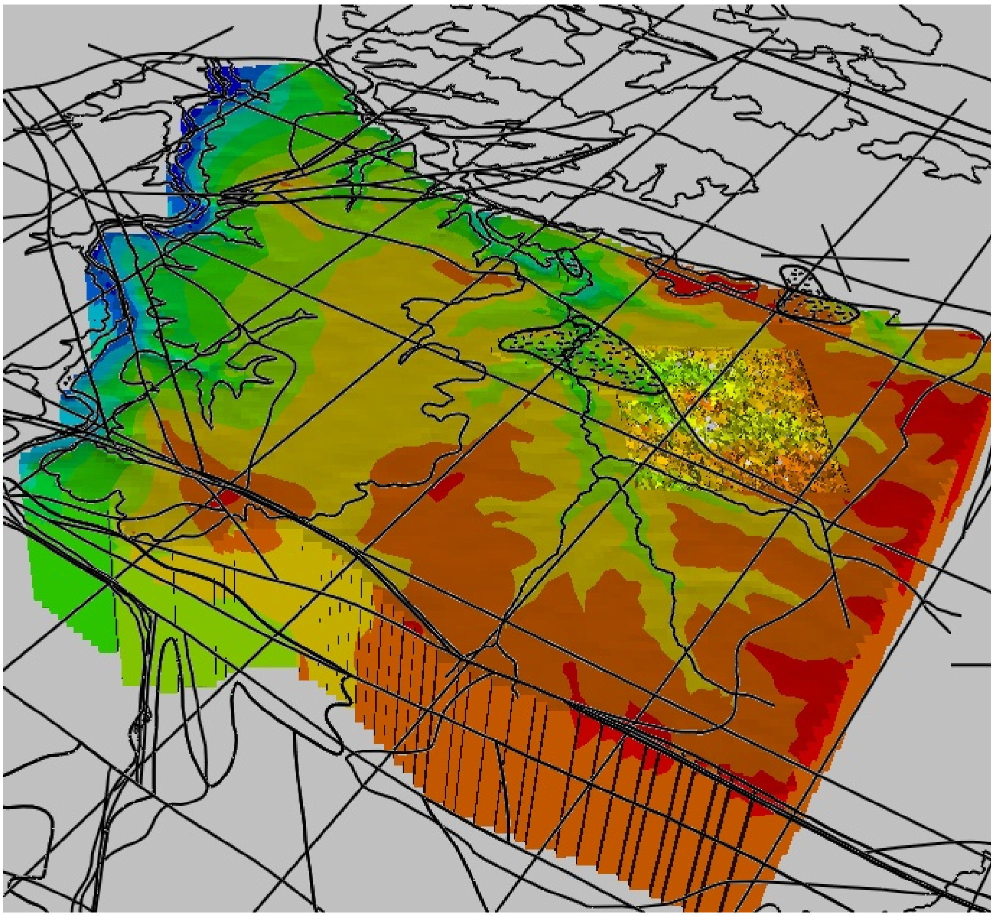


Figure 6.1-19 "Semi-regional up-scaled" CPM model and DFN "far-field" model included

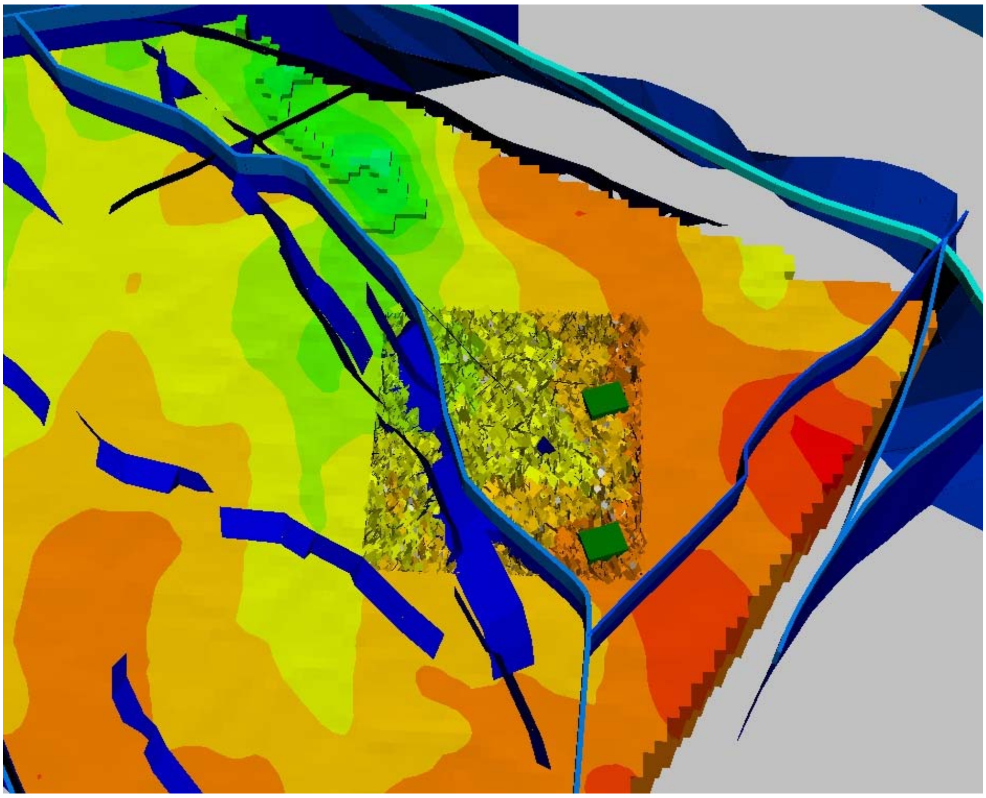


Figure 6.1-20 Position "A" (below) and "B" (above) of the "near field" DFN modes in the far-field DFN model

Once these embedded models have been produced, particles are released in the repository cells; they go through the engineered barriers, follow the potentials field and can penetrate the first fracture intercepted or continue for some time in the drifts. Each particle is followed up to the boundary of the "near-field" model where it is replaced in the closest "far field" fracture. The path is followed in the "far-field" and possibly followed in the extended CPM model if the boundaries of the model (close to the surface) have not yet been reached. As the upper boundary of the composite model is set at 150 meters under the topographical surface, the path of the particles in the last 150 meters of the geological barrier is not taken into account.

For each particle released, the calculation supplies the path's point of arrival, the distance developed inside the fractures, the advective travel time, the hydraulic flow and an integrating factor F used in the calculation of transport to take into account the part of the diffusion phenomena in the wall rock of the fractures. In the approach applied to M2, the path is not divided, even in the event of branching, the entire flow being directed along the path with the greatest flow.

- **Specific characteristics of the treatment of site model M1**

The methodology applied to M1 site model M1 is similar overall with regard to that used for M2 as regards the implementation of hydraulic models. The main differences in this domain are as follows:

- The petrographic characteristics of (monzonitic) granite rock give rise to the application to site model M1 of the average permeability, diffusion and retention test values that are comparable to those adopted abroad for equally undifferentiated granite [xxxviii]. In particular, these values are applied to the granite rock on the edge of the faults;

- The parameterizing of the hydraulic properties of the faults and fractures has no references of borehole data in France for granite of the same type. The transmissivity values have been set as a function of the extent of the fractures in accordance with a standard power law: $T = 3.5 \cdot 10^{-10}.\ R^{1.15}$ similar to that used at Aspö. A comparison was then made between the transmissivity assigned to the fractures of the model as a function of their dimension on one hand, and the same distribution established at the Aspö site on the basis of the hydraulic tests available on the other hand (see Figure 6.1-21). It shows that at an equal dimension, assigned transmissivities are from a factor 2 to 10 times smaller than that of the Aspö site, which is consistent with the differences in geological context;

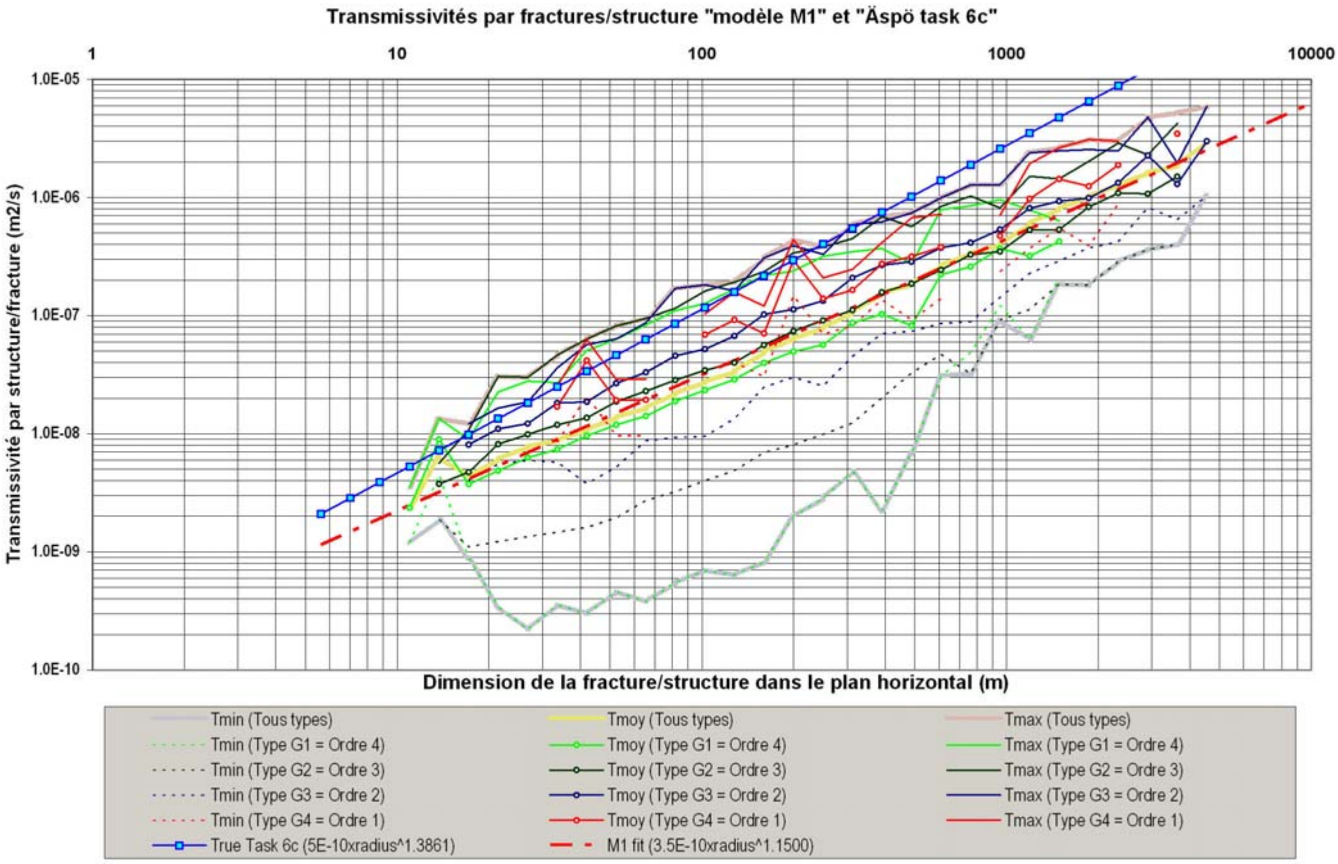


Figure 6.1-21 *Comparison of transmissivity as a function of the fracture dimension for model M1 and for the Aspö site*

- The repository components are integrated into the model in the form of hydraulically equivalent fractures. The hydraulic model takes into account a B waste repository tunnel and a fraction of a C2 and CU2 waste module (3 drifts of 33 cells);
- The dimensions of the DFN model are 1500 m x 1500 m x 500 and therefore a second DFN model in the far field (see Figure 6.1-22) is not necessary.

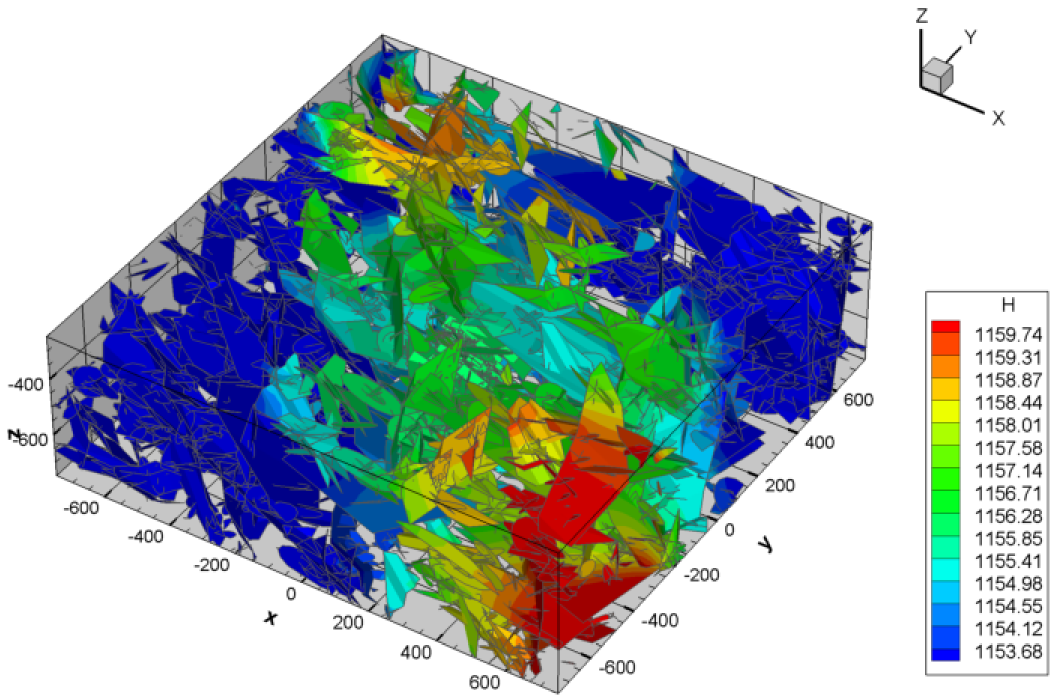


Figure 6.1-22 *DFN model of M1 site. The fractures are identified as a function of their hydraulic head (only a small part of the fractures is figured)*

The hydraulic paths of M1 are treated as a network with multiple branches if applicable, whereas those of M2 each constitute a series of successive tubes (see Figure 6.1-23). It is this network of tubes that is transmitted to the transport model and not simply the integrating parameters of each independent path.

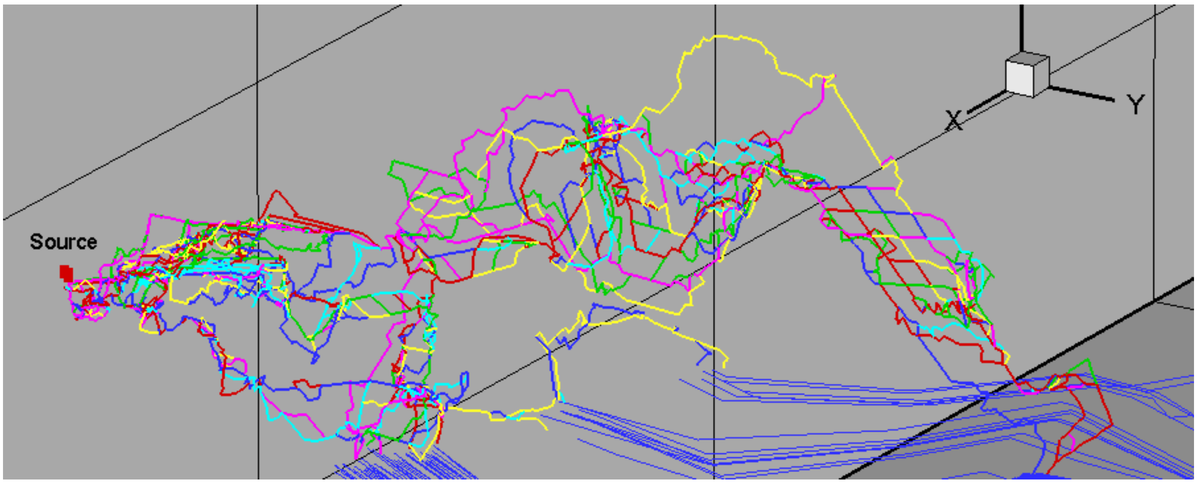


Figure 6.1-23 *Model M1: network of tubes representing the "near-field" paths from a B5.2 tunnel. The multicoloured lines are "near field" paths (DFN), they leave the CPM model via its lower surface. The blue lines represent the continuation of the paths in the "far field" (CPM) and their number is proportional to the hydraulic flow.*

6.1.4 The Representation of transport in granite

The representation of the radionuclides migration in granite is based on the determination of paths determined by the hydrogeological models at different scales (cf. Section 6.1.3 above).

The calculations of radionuclides migration along these paths involve:

- Advection (and associated dispersion) phenomena generated by the field of hydraulic pressures,
- Radionuclides diffusion phenomena in the rock on the fracture planes ("wall rock"),
- Radionuclides retention phenomena by sorption on the minerals of the fractures and the granite rock itself [x].

The representation of these phenomena in the complexity of a fractures network implies simplifications in the geometry of the paths and exchange surfaces between the water and the fractures planes. The modes of simplification adopted are based on the results of numerous experiments carried out in underground laboratories, in particular in the context of the "TRUE" programme and Aspö Task Force modelling exercises carried out within the framework of international cooperation over the last fifteen years or so. Transport parameters for the calculations are determined on the basis of the results of these experiments and take into account, via the different geological site models, the potential variability of the mineralogical characteristics of the fractures and the types of granite represented. Variability concerns more particularly the nature and intensity of natural clogging of the fractures by hydrothermal minerals as well as diffusive properties of the rock nearest to the fractures (see Table 6.1-1 to Table 6.1-4).

Granite M1 (matrix)	**Density: 2,650 kg/m^3**						
	Maximum thickness of diffusion [m]			**Effective diffusion De [m^2/s]**		**Diffusion porosity ω_d**	
	Near-Field Fracturing Reference	Intermediate Fracturing Reference	Far-Field Fracturing Reference	Reference	Sensitivity	Reference	Sensitivity
^{14}C	0.02	0.05	0.2	2.10^{-14}	2.10^{-15}	0.006	0.002
^{36}Cl	0.02	0.05	0.2	2.10^{-14}	2.10^{-15}	0.006	0.002
^{79}Se	0.02	0.05	0.2	2.10^{-14}	2.10^{-15}	0.006	0.002
^{93}Mo	0.02	0.05	0.2	2.10^{-14}	2.10^{-15}	0.006	0.002
^{99}Tc	0.02	0.05	0.2	2.10^{-13}	2.10^{-14}	0.02	0.006
^{126}Sn	0.02	0.05	0.2	2.10^{-13}	2.10^{-14}	0.02	0.006
^{129}I	0.02	0.05	0.2	2.10^{-14}	2.10^{-15}	0.006	0.002
^{135}Cs	0.02	0.05	0.2	2.10^{-13}	2.10^{-14}	0.02	0.006
^{229}Th	0.02	0.05	0.2	2.10^{-13}	2.10^{-14}	0.02	0.006
^{233}U	0.02	0.05	0.2	2.10^{-13}	2.10^{-14}	0.02	0.006
^{237}Np	0.02	0.05	0.2	2.10^{-13}	2.10^{-14}	0.02	0.006
^{241}Am	0.02	0.05	0.2	2.10^{-13}	2.10^{-14}	0.02	0.006
^{241}Pu	0.02	0.05	0.2	2.10^{-13}	2.10^{-14}	0.02	0.006
^{245}CM	0.02	0.05	0.2	2.10^{-13}	2.10^{-14}	0.02	0.006

De, $_d$: Test Values
Maximum thickness accessible to diffusion: Test value (expert)
Anionic exclusion is taken into account for Carbon 14, Chlorine 36, Selenium 79, Iodine 129 and Molybdenum 93.

Table 6.1-1 Values of the parameters adopted for the granite of M1 site model

Granite M1	**Partition coefficient [Kd in m³/kg]**		**Delay factor [R]**	
	Reference	Sensitivity	Reference	Sensitivity
^{14}C	0.001	0	440	1
^{36}Cl	0	0	1	1
^{79}Se	0.0005	0.0005	221	221
^{93}Mo	0	0	1	1
^{99}Tc	0.2	0	26000	1
^{126}Sn	0.001	0	131	1
^{129}I	0	0	1	1
^{135}Cs	0.1	0.005	13000	650
^{229}Th	0.5	0.01	64900	1300
^{233}U	1	0.01	130000	1300
^{237}Np	0.5	0.05	64900	6490
^{241}Am	0.5	0.04	64900	5200
^{241}Pu	2	0.5	260000	64900
^{245}CM	0.5	0.04	64900	5200
R = 1 + density Kd (1-ωd)/ωd where Kd (reference, sensitivity) = minimum values in non saline reducing conditions [xl] Reference: phenomenological; sensitivity: conservative				

Table 6.1-2 Chemical retention test values adopted for the granite of M1 site model

Granite M2 (matrix)	**Density: 2,650 kg/m³**						
	Maximum thickness accessible to diffusion [m]			**Effective diffusion De m²/s**		**Porosity accessible to diffusion ω_d**	
	Near-Field Fracturing Reference	Intermediate Fracturing Reference	Far-Field Fracturing Reference	Reference	Sensitivity	Reference	Sensitivity
^{14}C	0.1	1	10	1.10^{-13}	1.10^{-14}	0.015	0.004
^{36}Cl	0.1	1	10	1.10^{-13}	1.10^{-14}	0.015	0.004
^{79}Se	0.1	1	10	1.10^{-13}	1.10^{-14}	0.015	0.004
^{93}Mo	0.1	1	10	1.10^{-13}	1.10^{-14}	0.015	0.004
^{99}Tc	0.1	1	10	1.10^{-12}	1.10^{-13}	0.05	0.015
^{126}Sn	0.1	1	10	1.10^{-12}	1.10^{-13}	0.05	0.015
^{129}I	0.1	1	10	1.10^{-13}	1.10^{-14}	0.015	0.004
^{135}Cs	0.1	1	10	1.10^{-12}	1.10^{-13}	0.05	0.015
^{229}Th	0.1	1	10	1.10^{-12}	1.10^{-13}	0.05	0.015
^{233}U	0.1	1	10	1.10^{-12}	1.10^{-13}	0.05	0.015
^{237}Np	0.1	1	10	1.10^{-12}	1.10^{-13}	0.05	0.015
^{241}Am	0.1	1	10	1.10^{-12}	1.10^{-13}	0.05	0.015
^{241}Pu	0.1	1	10	1.10^{-12}	1.10^{-13}	0.05	0.015
^{245}CM	0.1	1	10	1.10^{-12}	1.10^{-13}	0.05	0.015

Table 6.1-3 Values of the parameters adopted for the granite of site model M2

Granite M2	Partition coefficient [Kd]		Delay factor [R]	
	Reference	Sensitivity	Reference	Sensitivity
^{14}C	0.001	0	175	1
^{36}Cl	0	0	1	1
^{79}Se	0.0005	0.0005	88	88
^{93}Mo	0	0	1	1
^{99}Tc	0.2	0	10100	1
^{126}Sn	0.001	0	51.4	1
^{129}I	0	0	1	1
^{135}Cs	0.1	0.005	5040	253
^{229}Th	0.5	0.01	25200	505
^{233}U	1	0.01	50400	505
^{237}Np	0.5	0.05	25200	2520
^{241}Am	0.5	0.04	25200	2020
^{241}Pu	2	0.5	101000	25200
^{245}CM	0.5	0.04	25200	2020
R = 1 + density Kd (1-ωd)/ωd where Kd (reference, sensitivity) = minimum values in non saline reducing conditions [xl]. Reference: phenomenological; sensitivity: conservative				

Table 6.1-4 Chemical retention test value adopted for the granite of site model M2

- **The path of the radionuclides in a network of fractures: the tube model**

The adopted "tube" model, generally, for radionuclides transfer simulations in a fracture network relies on the observation that irregularities in the geometry of a fracture and connections between fractures lead to water flows that are generally channelled in the fractures (see Figure 6.1-24). Tubes are a simplified representation of the channelling which meets the digital constraints of the calculations.

The entire water flow of the fractures passes through the tubes of the model. Given the low kinetics of the water movement, the radionuclides can migrate through diffusion in the altered rock at the edge of the fractures: the contact surface between the water and the granite ("wet surface") is a parameter that counts in the extent of the diffusion. The hydraulic paths in the fractures are transposed in the transport calculation in the form of a series of successive elementary 1D tubes, each of which being allocated an equivalent flow Q and flow wetted surface A. The A/Q ratio is designated by F (usually expressed in yr/m), it gives a measurement of the surface area available for interaction with the fracture edges and the time available for this interaction to take place, that is to say the equivalent of "resistance to transport". This parameter can also be integrated in such a way that several tubes or the entire path is represented. There are two values of F for each M2 path, one representing the near-field, the other representing the far-field.

The diffusion coefficients of the granite are generally higher at the edge of the fractures than in sound rock, which is a favourable element for radionuclides retention. The selected retention parameters for calculations take into account the mineralogy of the granite rock at the edge.

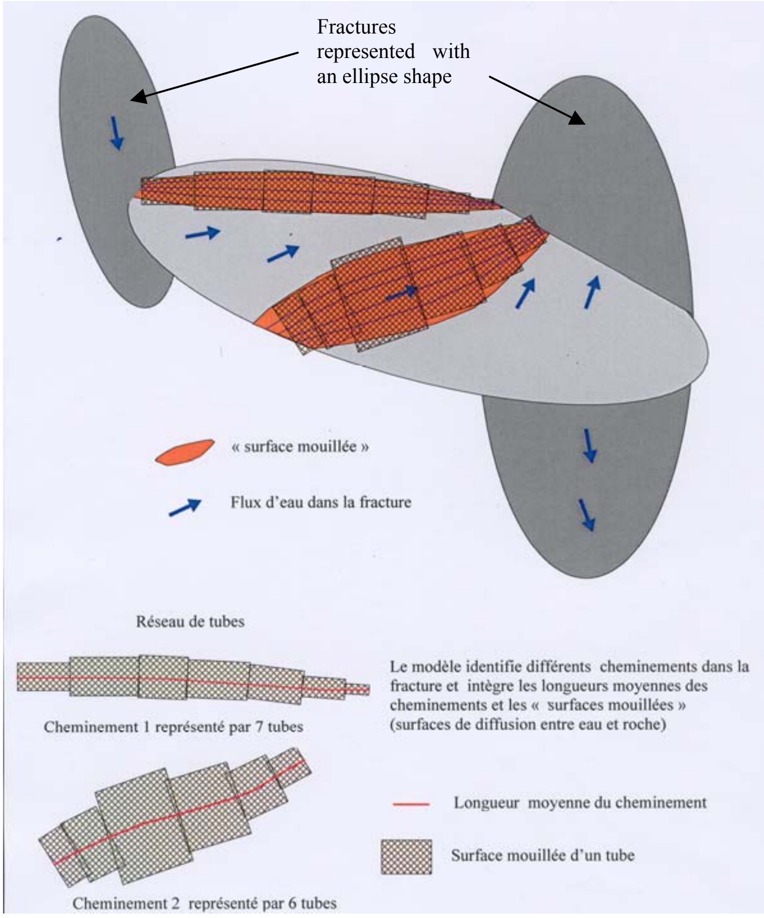


Figure 6.1-24 Model in water and radionuclide path tubes in a fracture

The radionuclides migration along the paths is represented up to near by granite surface. In terms of performance, the radionuclides inventory is evaluated at their entrance to the altered superficial part of the granite (which is approximately 100 meters under the surface). In generic evaluations, this choice makes it possible to dispense with site-specific uncertainties relating to the surface environment.

Transport models correspond to successive embedded volumes like hydraulic models. Close to the source of radionuclides, which corresponds to the beginning of paths, they take their parameters from the near-field DFN model. In the far field, and depending on the sites, the paths on which the transport calculations are based are determined either by a DFN model then a CPM (continuous porous media) model or directly by the CPM model (see Figure 6.1-25).

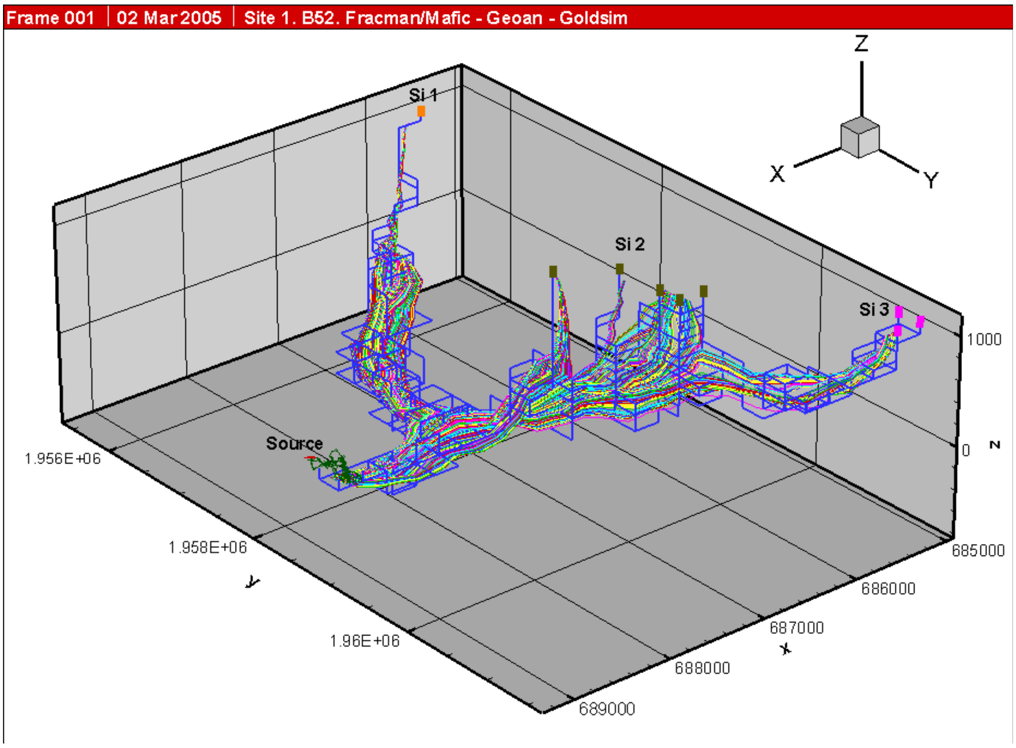


Figure 6.1-25 *Tubes network representing the far-field paths from a B5.2 tunnel on the "favourable" site .The green lines close to the source represent near-field paths (DFN), the multicoloured lines represent the far-field paths (CPM) leading towards several points on the subsurface.*

6.1.5 Architecture model

6.1.5.1 Selection of models and parameters

Each component is represented by its main characteristics as regards radionuclides transport. There are uncertainties regarding these characteristics, which may be minor or considerable depending on the knowledge acquired about the behaviour of the material in question and the influence that all the other repository components may have on it. It is important therefore to specify the nature of the choices of models and parameters that could be made in order to take uncertainties into account.

Andra's strategy consisted in proposing for each process or characteristic to be represented at least two models or parameters allowing for characterization of the possible variability. The parameters and models are qualified ("phenomenological", "conservative", "pessimistic") depending on the extent to which a conservative approach is adopted in respect of the uncertainties.

The terminology used for the models and parameters of repository components is explained in the following inset (see Inset 5).

Inset 5 *Models and parameters with regard to uncertainties for the repository components*

This inset introduces the terminology defined for the purpose of describing the conceptual models. This standardized vocabulary provides a framework for the safety calculation model selection process.

The term "impact" refers to the calculation's expected result (generally a dose expressed in sieverts, although the principle is applicable to any safety indicator).

Definitions relating to models

A model represents a physical reality, generally on the basis of experiment. Several possible types of models have been defined.

A "phenomenological" (or "best estimate" model)

This model can be defined as:

- Either, the model that is based on the most comprehensive understanding of the phenomenon to be modelled, and whose ability to account for direct or indirect measurements has been confirmed. This type of model may include all the relevant phenomena (in the simplest cases) at the most detailed level, or include the most influential environmental parameters (e.g. pressure, pH, temperature, etc.);
- Or, in comparison with the other available models, it might be the one offering the best match between the reality that it is supposed to represent and the numerical results that it generates in the impact calculation, within the parameter variability range adopted for the study.

Examples of the former include:

- Basic physical models (Coulomb's law, etc.);
- Mechanistic models representing Fick's law or Darcy's law, for example, which can be used as "phenomenological" models in certain environments where they account for the influence of the main parameters, even if they do not include all parameters.

Examples of the latter include all models subject to a broad-reaching experimental validation and/or a solid international consensus among experts in the field.

Note:

There cannot be a “phenomenological” model without a significant research program on the subject considered. In case of a poorly understood “phenomenon” or one not yet studied in detail, or one without corresponding data, a “phenomenological” model cannot be defined, because the elements required to perform the above described choice are not available. For example, a “phenomenological” release model cannot be defined for certain waste types, for which the waste matrix behaviour is poorly known (for example, wastes in a B3 type package).

A "conservative" model

This designates a model for which it is possible to demonstrate that its use, all things being equal otherwise, tends to overestimate the repository's impact, compared with the results that would be obtained by taking into consideration all the relevant phenomena in the chosen parameter variation range.

For example, selecting a transport model that ignores chemical retention could, in situations where retention has a potentially significant effect, be deemed "conservative".

A "pessimistic" model

This designates a model that is not based on phenomenological understanding, however empirical, but which is used exclusively to definitely overestimate the repository's impact.

For example, making an assumption that waste packages immediately release radionuclides is, except in special cases, a pessimistic choice.

An "alternative" model

This designates a model that is not considered to be closest to the "phenomenological", but is offered as an alternative, although it cannot be classified on a "phenomenological", "conservative" or "pessimistic" scale. Examples might include a model that if chosen does not have an unequivocal effect on the impact, or a model that appears more comprehensive than the selected reference model but has been less thoroughly validated.

Definitions relating to values

Four possible types of parameter are defined for a particular model.

"Phenomenological" value (also known as "best estimate")

A phenomenological value is one that is considered to offer the best match between the model's results and the measured results, all other parameters being equal. This choice is in principle made without reference to the impact; if the reasoning used to determine the phenomenological value leaves any margin for interpretation or an uncertainty interval, the value that maximises the impact calculation should be chosen.

The choice of phenomenological value must be supported by detailed arguments, which might include:

- A representative number of measurements, in which case the chosen value is based on appropriate statistical considerations (generally the most probable value), taking a safety margin into account where applicable;
- A physical reasoning that demonstrates that the chosen value is the most representative, based on reliable data (e.g. reference to measurements made for other research, with arguments supporting the choice of model used for the transposition);
- A judgement by recognised experts, unambiguously designating it as the most appropriate value for the study context.

"Conservative" value

This term refers to a value, chosen among those generated by the studies and measurements, which gives a calculated impact in a range of high values (all other parameters being equal).

In the simplest case, where the impact increases (or conversely, decreases) as the value of the parameter increases, a value in the highest (or lowest) range of available values. More specifically:

- This value might correspond to a sufficiently high confidence interval, if enough data is available to establish this type of statistic;
- If the available data is limited, the highest (or lowest) value is chosen after eliminating any values deemed to be abnormal ;
- In the absence of measurements, internationally-available data is used, provided it is sufficiently explicit in the literature.

Note:

"Conservative" values cannot be defined if the variations in impact are not monotonic with changes in the parameter.

"Pessimistic" value

A pessimistic value is one that is not based on a state of phenomenological understanding, but is chosen by convention as definitely yielding an impact greater than the impact that would be calculated using possible values. Such values can represent physical limits (e.g. assimilating the permeability of backfill to that of sand, based on the fact that it could not be degraded any further). A pessimistic value can also be equal to the conservative value plus (or minus, where applicable) an appropriate safety factor that places it significantly beyond the range of measured values. As with "conservative" values, a value cannot be described as "pessimistic" if the variation in impact in response to a variation in a parameter or a change of model cannot be characterised.

"Alternative" values

"Conservative" and "pessimistic" values cannot be defined in situations where the variation in impact cannot be characterised in relation to the change in the parameter. In order to explore the possible parameter variation ranges, one or more so-called "alternative" values can be suggested as a means of investigating the effect of contrasting values.

For example, in geochemistry, "alternative" values for the chemical composition of water might be tested in order to ascertain their effect on radionuclide transport, even without being able to describe the effect on the impact in advance.

The strategy for defining the safety calculation model based on the conceptual models obeys the following principles:

- If the degree of uncertainty relating to the models or parameters is low to moderate, the most scientifically-documented (i.e. "phenomenological") parameters or models are chosen for the reference calculation. Where it appears important to study the impact of a more unfavourable (i.e. "conservative" or "pessimistic") value, disregarded as being too improbable but not ruled out altogether, this parameter is processed in a sensitivity study.
- Where the degree of uncertainty is high, the conservative or pessimistic parameter value or model is selected, depending which models or values are available;
- A secondary consideration is to prefer simple, robust models over ones that are more complex or dependent on environmental conditions (chemical, thermal or mechanical conditions, etc.), provided such a choice does not cause the impact to be underestimated;

If the variation in impact dictated by the direction of variation of a parameter or a choice of model is not unequivocal, one possible approach is to test contrasting models and parameters as a means of evaluating the overall uncertainty.

The concept of "low, moderate or high" uncertainty is inevitably subjective to a degree, although it can be rendered objective in certain cases by including statistical considerations such as the dispersion of the experimental values or a confidence rating. The verdict on the uncertainty is discussed on a case-by-case basis by the experts proposing the values and models.

In addition to the actual reference scenario calculation, sensitivity calculations are performed to test different parameter sets and models other than those selected as being the most representative by Andra. These sensitivities are generally intended to cover a residual uncertainty (given that the main uncertainties are usually already included in the reference calculation). They therefore focus on more unfavourable values (from "phenomenological" to "conservative", or from "conservative" to "pessimistic"). They make it possible to assess the uncertainty's sensitivity in the context of the performance analysis.

Furthermore, certain sensitivity studies aim to "test" a more favourable value than one of the calculation's parameters, in order to assess the utility of studying the potential variation range more thoroughly at a later stage in the studies.

6.1.5.2 General assumptions and structure of the calculations

- **Waste packages**

Four fuel management scenarios have been developed by Andra in the context of the dimensioning inventory model (see Chapter 2) in order to cover different possible industrial strategies.

The perimeter of calculations has been defined to identify the main factors that determine repository performance. The most representative waste packages types of each of the package categories (B waste, C waste and spent fuel) were retained as reference for the impact calculations:

- *B wastes of the B2 type (bituminised sludge) and the B5.2 type (compacted hulls and end caps),* representative of the largest volume of B wastes, two types of waste with sufficiently distinct characteristics (thermicity, radioactivity, chemical composition) and two packaging modes;
- *C waste of the C2 type*, representative in the radiological inventory of the largest part of C wastes;
- Spent fuel of the *CU2* type (MOX) for which the labile activity released at the arrival of water is, in the case of a failure of the copper container, the highest among the various spent fuel types.
- In the context of the normal evolution scenario, Andra has retained, as an overall manner, the most pessimistic scenario per reference package in terms of quantitative inventory. This hypothesis has led to consider the quantitative inventory of S2 for the treatment of spent fuels, and S1b for the treatment of other reference packages.

The package groups and related quantitative inventories are summarised in Table 6.1-5.

Repository sub-zone containing reference packages or cell types covered in the calculations*		Management scenario	Number of packages
B2	Cell type containing organic packages of bituminised sludge	S1b	105 010
B5.2	Cell type containing non-organic packages that release no gaseous hydrogen.	S1b	29 600
C2	Vitrified waste with "increased heat transfer"	S1b	27 460
CU2	MOX spent fuel	S2	4 000

Table 6.1-5 Number of packages per reference packages used for the calculations

- **Radionuclides used for quantitative evaluation**

To limit calculation times and simplify the analysis, the performance assessment has been conducted using a limited number of radionuclides. To this end, the impact calculations were preceded by an exercise consisting in selecting, among the 144 radionuclides with half-lives long than 6 months that are liable to be present in the waste packages, and whose initial activity is given in [44], which are the most representative of the performance of a repository relative to the long-term safety. The method of selection applied to fission and activation products for each reference package taken into account comprises four stages:

- The first stage of selection consists in creating three groups of radionuclides in accordance with their half-lives. Radionuclides with a half-life of less than 30 years are not selected on account of the timescales studied. For the normal evolution scenario, only the selection of radionuclides with half-lives longer than 1,000 years is taken into account because of the theoretical transfer times in the geological barrier. Given the objectives of the dossier 2005 Granite, the same selection is adopted as a first approach for the altered evolution scenarios.
- The second stage consists in eliminating radionuclides with zero, very low or unknown radioactivity then organizing the remaining radionuclides within each selected reference package according to the "radiological inventory" criterion. The radiological inventories taken into account are those that correspond to approximately 150 years after leaving the reactor in the case of C2 and CU2 and the date of production in the case of B wastes (B2 and B5.2).

- The third stage consists in taking into account the capacity of radionuclides to sorb and/or precipitate in confining media. Application of this criterion leads to the classification of radionuclides that must be treated as a priority (see Inset 6).
- The fourth phase consists in classifying radionuclides within each selected reference package according to the radionuclide "toxicity" criterion (radiological toxicity or "radiotoxicity" in this case), on the basis of the public dose ingestion factor. In the context of the dossier 2005, this stage only consisted in classifying the radionuclides in greater detail.

Inset 6 *Specific behaviour of radionuclides in terms of solubility and retention*

The radionuclides are divided into three main categories in terms of solubility and retention in granite media (rock and fractures), in argillaceous media and in the bentonite of engineered structures: swelling clay-based backfill, cell plugs or engineered barriers, or cement-based media (B waste cells):

- "Mobile" elements whose solubility is high and retention low or nil, such as iodine or chlorine, for example,
- "Averagely mobile" elements, the solubility of which is high and retention strong such as caesium, for example,
- Elements that are "barely mobile", the solubility of which is low and retention strong, actinides for example, such as uranium or plutonium and lanthanides such as samarium or europium.

The radiological inventory of waste packages mostly comprises elements from the two last categories.

Given the objectives of the Dossier 2005 Granite, only five among the first radionuclides from each list are finally retained in the calculations. These are the long-lived and most mobile radionuclides such as for example chlorine 36, and averagely mobile elements on account of a high solubility or high retention such as for example caesium 135:

- *B2 type package*: iodine 129, chlorine 36, caesium 135, technetium 99 and selenium 79;
- *Package type B5.2*: iodine 129, chlorine 36, caesium 135, technetium 99 and molybdenum 93;
- *Package type C2*: iodine 129, caesium 135, carbon 14, tin 126 and selenium 79;
- *Package type CU2*: iodine 129, caesium 135, carbon 14, tin 126 and selenium 79.

These radionuclides are listed below; the half-life decay rates taken into account are consistent with those from the database JEF 2.2 [xxxvi]:

- ^{14}C (anionic form) Half-life T = 5 730 years
- ^{36}Cl (anionic form) Half-life T = 302 000 years
- ^{79}Se (anionic form) Half-life T = 65 000 years
- ^{93}Mo (anionic form) Half-life T = 3 500 years
- ^{99}Tc Half-life T = 213 000 years
- ^{126}Sn Half-life T = 100 000 years
- ^{129}I (anionic form) Half-life T = 15 700 000 years
- ^{135}Cs Half-life T = 2 300 000 years

In other respects, an actinide chain (thorium 229, uranium 233, neptunium 237, americium 241, plutonium 241, and curium 245) was treated for methodological purposes for part of the waste packages, in this case, packages B5-2 and packages C2:

- ^{229}Th Half-life T = 7 340 years
- ^{233}U Half-life T = 159 000 years
- ^{237}Np Half-life T = 2 140 000 years
- ^{241}Am Half-life T = 433 years
- ^{241}Pu Half-life T = 14.4 years
- ^{245}Cm Half-life T = 8 500 years

- **Overall architecture of the repository**

The calculations are based on the representation of a repository tunnel of B waste (B2 and B5.2 wastes) or a module of C2 waste or spent fuel CU2 (MOX). For this purpose, it was decided to keep in each of the three geological site models considered, the same emplacements for the tunnels (B waste) or repository modules (C waste and spent fuel) considered. This contributes to discriminating more simply in the analysis of the performance of a repository in granite factors related to package types from those related to the characteristics of the geological medium.

The choice of emplacements was note a result of a precise location of a repository as a function the characteristics of granite massifs of the three geological models. This would have been illusory given the generic character of the data used. As indicated above in Section 6.1.3, the emplacements have been fixed from geological and regional hydrogeological models established for each site model excluding the proximity of major fault zones and emplacements clearly non compliant from the hydrogeological point of view with the recommendations of Basic Safety Rule III.2.f (high hydraulic gradients, hydraulic head zones in line with the repository, etc.). Two theoretical emplacements have been retained upon site models which allow to analyse to which extent the location is a factor in the long term safety of the repository.

- **Segments of the calculation**

For each type of waste treated (B2, B5.2, C2 and CU2), the presentation of the calculation models introduces the wastes, the different cells components (disposal packages, engineered barrier, cell plugs/seals, drifts and access drifts) and granite.

6.1.5.3 Release models

- **B2 and B5.2 release models**

The repository is assumed to be saturated immediately upon closure and the B waste overpack is assumed not to be water-tight; based on this conventional assumption, the waste begins to release radionuclides and toxic chemicals as soon as the repository is closed. The release models and the values adopted for the assessment depend on the types of waste and the level of understanding available.

For compacted waste (B5), only the zircaloy cladding sections and structural waste have been described and understood well enough to consider a gradual release of the radionuclides that they contain. The release of activation products from cladding sections and metallic components is assumed to be congruent with the rate of corrosion, which is conservatively estimated[25] as being on the order of 10^{-5} $year^{-1}$ (from 5.10^{-6} to 7.10^{-5} $year^{-1}$ depending on the material and the waste concerned). The remaining radionuclides are assumed to be labile [xxiii]. The radioactive inventory contained in zircaloy cladding and cladding waste presents a release in contact with water that is directly linked to the rate of corrosion and which leads to a complete release over 100,000 years (see Table 6.1-6).

25 Conservative estimate that, in particular, allows the inclusion of any irradiation damage and the influence of water radiolysis on the corrosion process.

	Reference calculation
Activation Products present in Zircaloy	$\tau_R = 10^{-5}\ yr^{-1}$
Activation products present in stainless steel springs	$\tau_R = 7.10^{-5}\ yr^{-1}$
Activation products present in stainless steels	$\tau_R = 1.5 \cdot 10^{-5}\ yr^{-1}$
Other radionuclides:	Labile activity

Table 6.1-6 Release model adopted for B5.2 reference packages.

For bituminized sludge (B2), the release kinetics are represented by a model (Colonbo) developed on the basis of phenomenology that has been experimentally validated to the extent permitted by laboratory reproduction capabilities [44]. It is based on the water uptake by the bitumen and the behaviour of the radionuclides, which is assimilated to that of soluble salts in the embedded material. A number of sub-models may be defined, depending on the extent to which the relevant phenomena are included in the model. The case chosen for the reference calculation is conservative inasmuch as it neglects:

- The insolubilization process to which the radionuclides are subjected during treatment of the effluents that yielded the waste package contents. The radionuclides are assumed to be associated with the soluble salts;
- Radionuclide diffusion in the bitumen matrix's permeable zone (radionuclide release is congruent with the progression of the soluble salt dissolution front).

With the exception of the pessimistic assumption that the water in contact with the packages is significantly renewed, the model's parameters (apparent diffusion coefficient for the diffusion of water in the bitumen, actual diffusion coefficient in the permeable zone, etc.) are "phenomenological" values. They are the result of specific experiments to measure transfer properties for transfers through the permeable zone [44]. The proposed release rate is inversely proportional to the square root of time, yielding a release of approximately 90% of the original mass contained in the bitumen packages after 10,000 years. Table 6.1-7 details the release rates adopted for B2 reference packages.

	Reference calculation and SEA Colonbo model 3
All Radionuclides	$t_R = 4.5 \cdot 10^{-3}/t^{1/2}$

Table 6.1-7 Release model adopted for B2 reference packages

- **Release model of vitrified C waste**

Concerning the glass itself, the release of radionuclides begins when the over-pack loses leak-tightness. The release models and associated parameters differ depending on the reference package.

In the case of C2 waste, taken as reference in the calculations, the release rate obeys to a phenomenological model known as "$V_0.S \rightarrow Vr$" (see Chapter 5). This model results in the glass dissolution in approximately 300,000 years.

- **Spent fuel release model**

The release of radionuclides begins when the container loses leak-tightness in the form of a small hole of 5mm^2 in the case of hypothetical defective container(s). The release model is a function of the location of the radionuclides in the fuel's assemblies. The following can be distinguished:

- A model representing the gradual release of the radionuclides contained in metal components. The release, which is assumed to be congruent with the component corrosion rate, yields rates on the order of $5.10^{-5}\ yr^{-1}$ in the case of the radionuclides present in cladding (the radionuclides in the zircon at the surface of the cladding materials are considered to be labile) up to $2.10^{-3}\ yr^{-1}$ in the case of the radionuclides contained in inconel structural elements;

- A dissolution model of the spent fuel matrix under the effect of radiolysis (known as the radiolytic model) resulting in a progressive release of radionuclides located inside the matrix over approximately 5000 years, or several orders of magnitude below what is generally adopted internationally;
- A fraction conservatively assumed to be labile which corresponds to the radionuclides contained in the pellet at the grain boundaries, cracks and empty spaces; diffusion phenomena accelerated by alpha self-irradiation (D3AI) and radionuclides contained fluid in zircon.

The parameter values adopted are detailed in the following tables (see Table 6.1-8 and Table 6.1-9).

	Reference calculation
Activation products present in the zircaloy (80% of the cladding's activity)	$\tau_R = 5.10^{-5}$ yr^{-1}
Activation products present in the zircon (20% of the cladding's activity)	Labile activity 20% of the activity of the cladding
Activation products present in stainless steel structures	$\tau_R = 4.10^{-4}$ yr^{-1}
Activation products present in inconel structures	$\tau_R = 2.10^{-3}$ yr^{-1}

Table 6.1-8 Model of the release of activation products located in metal components for CU2 reference packages

<table>
<tr><th colspan="2"></th><th>Reference calculation</th></tr>
<tr><td colspan="2">Radionuclides contained in the gap, rim and grain boundaries (including accelerated diffusion by alpha self-irradiation, D3AI) in the matrix</td><td>Labile activity</td></tr>
<tr><td rowspan="3">Percentage of labile activity of the matrix</td><td>^{36}Cl, ^{79}Se, ^{126}Sn, ^{135}Cs.</td><td>38.9 %</td></tr>
<tr><td>^{14}C</td><td>15 % (conservative)</td></tr>
<tr><td>Other radionuclides</td><td>35 %</td></tr>
<tr><td colspan="2">Remaining radionuclides contained in the matrix's grains, released in accordance with the radiolytic dissolution model</td><td>TC = 48GWj/t</td></tr>
<tr><td colspan="3">** : Cf. Detail of the formula in [44]
TC : Burnup</td></tr>
</table>

Table 6.1-9 Model of radionuclide release from the spent fuel matrix -reference packages CU2.

6.1.5.4 Representation of the packages

- **B waste**

Representation of the functions of the concrete container of the disposal package

The disposal package's over-packs are represented for all B waste packages as a homogeneous chemical environment that limits the flow of toxins by precipitation and sorption phenomena. The values of chemical retention are similar to those used for the concrete of the B waste cells.

The representation of the packages is different depending on the two types of packages (B2 and B5-2) used for the calculations.

For B2 wastes (bituminised sludges), the disposal package is assumed not to be watertight on repository closure, which is a prudent hypothesis. The concrete container does, however, impose a chemical environment with a high pH (between 12.5 and 10) which limits the flow of radionuclides through the phenomena of precipitation and sorption.

For B5.2 wastes, additional properties of confinement of a container have been tested by calculatrice using hydraulic properties delaying the release of radionuclides for 10,000 years. Such a test of over-packs hydraulic properties would benefit for B waste packages of high radiological content but of low amount of hydrogen release (in particular B5.2 and B1 waste types). It is taken into account in the reference scenario; in this case, the wastes over-packs are represented by a homogeneous porous medium; in addition to the geochemical performances already adopted, the function of "delaying and reducing the migration of radionuclides" is also ensured by a transport by diffusion/advection with a diffusion coefficient of 2.10^{-13} m²/s, a porosity accessible to the diffusion of 0.1 and a permeability of $2\ 10^{-13}$ m/s.

The values of the hydraulic, transport and chemical retention parameters retained for the B2 and B5.2 waste concrete containers are respectively given in the following tables: Table 6.1-10 and Table 6.1-11.

B2 concrete		Permeability: K= 1.10^{-8} m/s Density: 3 000 kg/m^3 Diffusion coefficient De: 6.10^{-10} m^2/s Total porosity ω: 0,3			
Radio-nuclides	Partition coefficient Kd [m^3/kg]		Delay factor [R]		Solubility limit Csat [mol/m^3]
	Reference	Sensitivity	Reference	Sensitivity	Reference
^{36}Cl	0	0	1	1	Soluble
^{79}Se	0.1	$1.5.10^{-3}$	701	11.5	$1.34.10^{-2}$
^{99}Tc	0	0	1	1	Soluble
^{129}I	0.001	0	8	1	Soluble
^{135}Cs	0.01	5.10^{-4}	71	4.5	Soluble
ρgrain, K, De, ω: Conventional concrete [45] The porosity accessible to diffusion is assumed to be equal to kinematic porosity, i.e. total porosity.					

Table 6.1-10 *Values of hydraulic, transport and chemical retention parameters adopted in the concrete of the B2 waste packages.*

<table>
<tr><td colspan="3">Concrete B5.2</td><td colspan="5">Permeability: $K=2.10^{-13}$ m/s for t ≤ 10 000 years
$K=1.10^{-8}$ m/s for t > 10 000 years
Total porosity: ω=0.1 for t ≤ 10 000 years
ω=0.3 for t > 10 000 years
Diffusion coefficient De [m^2/s]: 2.10^{-13} for t ≤ 10 000 years
6.10^{-10} for t > 10 000 years
Density: 3 000 kg/m^3</td></tr>
<tr><td rowspan="3">Radionuclides</td><td colspan="2">Partition coefficient Kd [m^3/kg]</td><td>Solubility limit Csat [mol/m^3]</td><td colspan="4">Delay factor [R]</td></tr>
<tr><td rowspan="2">Reference</td><td rowspan="2">Sensitivity</td><td rowspan="2">Reference</td><td colspan="2">T 10 000 years</td><td colspan="2">T > 10 000 years</td></tr>
<tr><td>Reference</td><td>Sensitivity</td><td>Reference</td><td>Sensitivity</td></tr>
<tr><td>^{36}Cl</td><td>0</td><td>0</td><td>soluble</td><td>1</td><td>1</td><td>1</td><td>1</td></tr>
<tr><td>^{93}Mo</td><td>0</td><td>0</td><td>7 10^{-4}</td><td>1</td><td>1</td><td>1</td><td>1</td></tr>
<tr><td>^{99}Tc</td><td>0</td><td>0</td><td>soluble</td><td>1</td><td>1</td><td>1</td><td>1</td></tr>
<tr><td>^{129}I</td><td>0.001</td><td>0</td><td>soluble</td><td>28</td><td>1</td><td>8</td><td>1</td></tr>
<tr><td>^{135}Cs</td><td>0.01</td><td>5 10^{-4}</td><td>soluble</td><td>271</td><td>14.5</td><td>71</td><td>4.5</td></tr>
<tr><td>^{229}Th</td><td>20</td><td>3</td><td>2 10^{-7}</td><td>540 000</td><td>81 000</td><td>140 000</td><td>21 000</td></tr>
<tr><td>^{233}U</td><td>50</td><td>1.5</td><td>3 10^{-3}</td><td>1 350 000</td><td>40 500</td><td>350 000</td><td>10 500</td></tr>
<tr><td>^{237}Np</td><td>20</td><td>0.1</td><td>5 10^{-6}</td><td>540 000</td><td>2 700</td><td>140 000</td><td>701</td></tr>
<tr><td>^{241}Am</td><td>30</td><td>3</td><td>1 10^{-7}</td><td>810 000</td><td>81 000</td><td>210 000</td><td>21 000</td></tr>
<tr><td>^{241}Pu</td><td>20</td><td>3</td><td>1 10^{-6}</td><td>540 000</td><td>81 000</td><td>140 000</td><td>21 000</td></tr>
<tr><td>^{245}Cm</td><td>30</td><td>3</td><td>1 10^{-7}</td><td>810 000</td><td>81 000</td><td>210 000</td><td>21 000</td></tr>
</table>

Table 6.1-11 Hydraulic, transport and chemical retention parameter values adopted for B5.2 waste cells

- **Vitrified C waste packages**

The containers are dimensioned for an order of magnitude of thousand-year with margins to cover chemical, mechanical and radiological uncertainties (see Chapter 5). In a conservative approach, the calculations consider a loss of leak-tightness of the over-packs at 1,000 years which accordingly, assumes no release from the matrix during that time.

The overpack's design is notable for its robustness with regard to possible manufacturing defects (see Chapter 5). As it is not possible to rule out quality control problems where large numbers of C waste overpacks are produced, initial failures are assumed in the reference case. However, in a prudent approach, the possibility of such a failure has been taken into account in the calculations. At the current stage of the studies, and as detailed studies on the manufacture of such objects are premature, the definition of possible failures is necessarily arbitrary. However, given both the inspections that it would be possible to carry out on the spot during the manufacture of the over-packs and the relatively simple nature of the over-pack design, it can be considered that the expression of a failure immediately after placement in the repository is very unlikely and that only a few containers would be affected by random failure related to a quality control problem. In fact, the probability of this type of failure occurring is generally considered by the industry to range from one in thousand to one in a hundred thousand.

Practically, the calculations carried out for a repository module (900 packages) or a fraction of a repository module consider one defective package which corresponds to a higher proportion of packages failures than what can be envisaged in industrial references for this type of failure.

A defective over-pack is conventionally represented by a total loss of leak-tightness several decades after closure of the repository[30], this minimum duration being a very pessimistic evaluation of the time that would be required for at least a small amount of water to reach the waste by passing through the primary container.

[30] To carry out the calculations, we reason in terms of the time needed for radionuclides to decay. By taking into account the time needed for preliminary waste storage, we adopted 150 years of decay.

In conclusion, it is assumed that the failure will occur conventionally 150 years after leaving the reactor. In calculations, it is represented by a total loss of leak-tightness which is very pessimistic.

To consider defective packages leads to the release and migration of radionuclides in an environment with a high thermal load and high temperature gradient. At this stage in the studies, and taking into account of the possibilities of transport calculations in a fractured medium, the effect of heat transfer on the models and the transport parameters is not taken into account, as a first approach, for all components, except for the chemical retention of caesium in the engineered barrier for which knowledge from experience exists.

- **Spent fuel packages**

The copper container is designed to remain leak-tight for long periods of time. Without significant external modification of the geodynamic context which is unlikely in the French geology setting, leak-tightness is ensured for the period concerned by the calculations which is several hundred thousand years.

In a prudent approach, the case of an initial defective container is treated in the normal evolution scenario for a certain proportion of packages, which is arbitrary at this stage of the studies and fixed at one package for the CU2 inventory which is the base of the calculations. The failure is a 5 mm^2 sized hole at the weld of the container lid. The diffusion of radionuclides takes place from the fuel matrix towards the engineered barrier via this hole. Water that penetrates the container causes corrosion of the cast iron insert of the container and then the total rupture of the container after 20,000 years, date determined in view of the specific models produced by SKB in Sweden. These two periods of times for the loss of copper container leak-tightness constitute two successive phases of the model.

6.1.5.5 Representation of the cells

After being released by the waste packages, the radionuclides migrate *into the disposal cells* and reach fractures in the granite wall of the engineered structures or the plugs sealing the entrance.

- **B waste cells**

B2 and B5.2 disposal tunnels represented in the calculations correspond to 1/8 of the total B2 inventory, which is a tunnel length of approximately 95 m and width of approximately 23m, and 1/7 of the total B5.2 inventory, which is a tunnel length of approximately 90 m and a width of approximately 9m. The calculations are performed for a tunnel of each type. The total performance is estimated by multiplying the results by 8 or 7.

Phenomenology of the migration of the radionuclides

The radionuclides released by the packages migrate through the concrete of the packages up to the granite of the repository tunnels walls. The concrete limits the fluxes of some radionuclides by precipitation and sorption phenomena. Radionuclides that have not been retained in the concrete packages may reach a fracture in the granite of the tunnel wall. The thickness of the granite excavation damaged zone of the repository tunnels wall is 50 cm. It can be a radionuclide migration pathway in the tunnels towards fractures in the granite wall. It is interrupted by the seal plug at the tunnel entrance. The purpose of the seal is to delay the flow of water that could go through it. Therefore, it also, has properties that limit the radionuclides transfer. Part of the released radionuclides can migrate through the seal towards the access drifts leading to the disposal tunnel (see Figure 6.1-26).

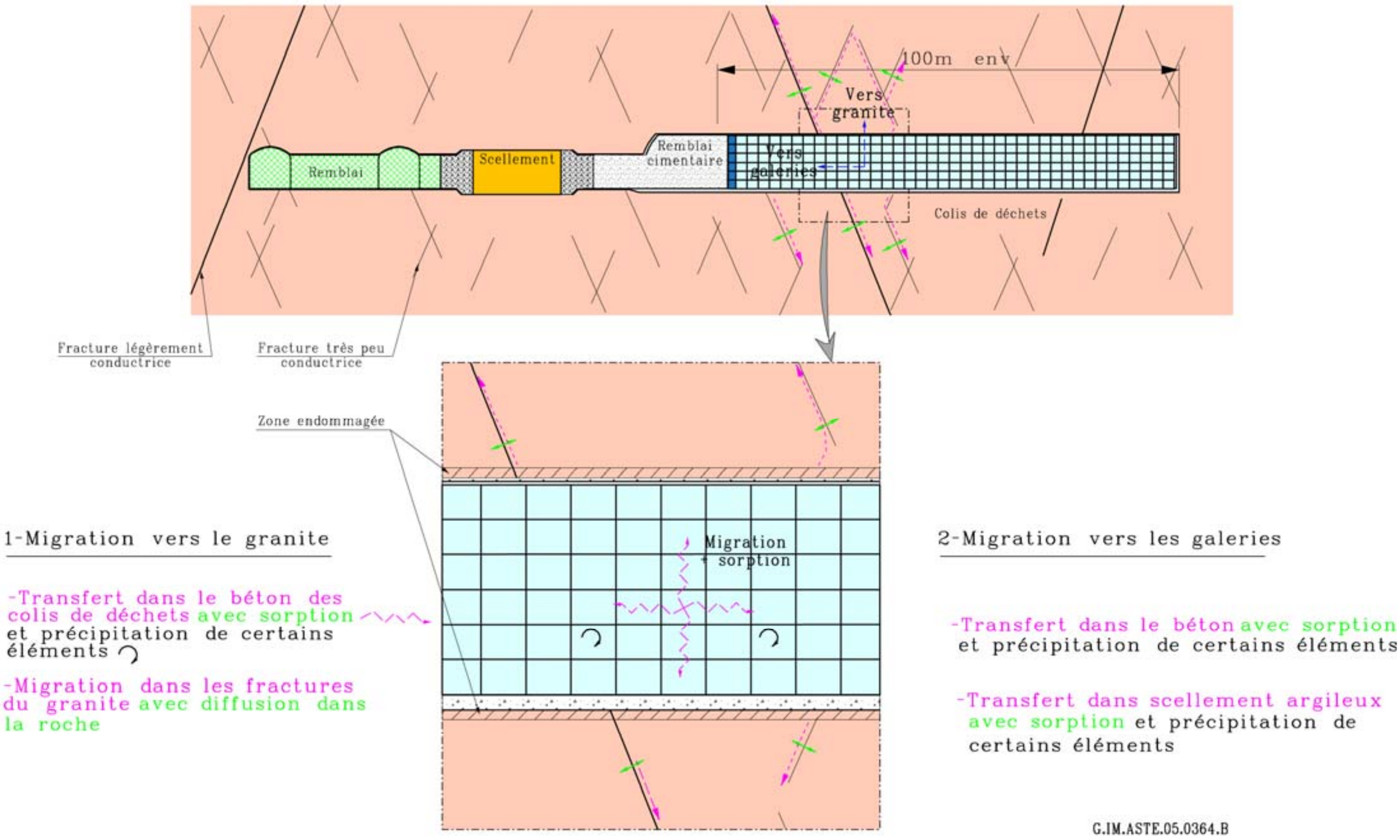


Figure 6.1-26 Phenomenology of the migration of radionuclides in a B waste tunnel

Representation of the B waste tunnels in the calculations

For the calculations, all the stacks of B waste packages are represented in the form of a single package with a 15 cm thick concrete envelope (see Figure 6.1-27). This "package" is assumed to be homogeneous.

The hydraulic parameters assigned to the concrete envelope depend on the waste type (see Table 6.1-10and Table 6.1-11).

The excavation damaged zone taken into account in the calculations has a thickness of 50 cm and a hydraulic conductivity of 10^{-9} m/s. This hydraulic conductivity is assumed to be continuing along the excavation damaged zone which is conservative, excavation by means of the drilling and blasting method results in discontinuities in the hydraulic conductivity [x].

The cement-based backfill put in place after closure of the tunnels in order to fill the empty space of the operating area is not represented. The radionuclide retention properties of the cement are not taken into account.

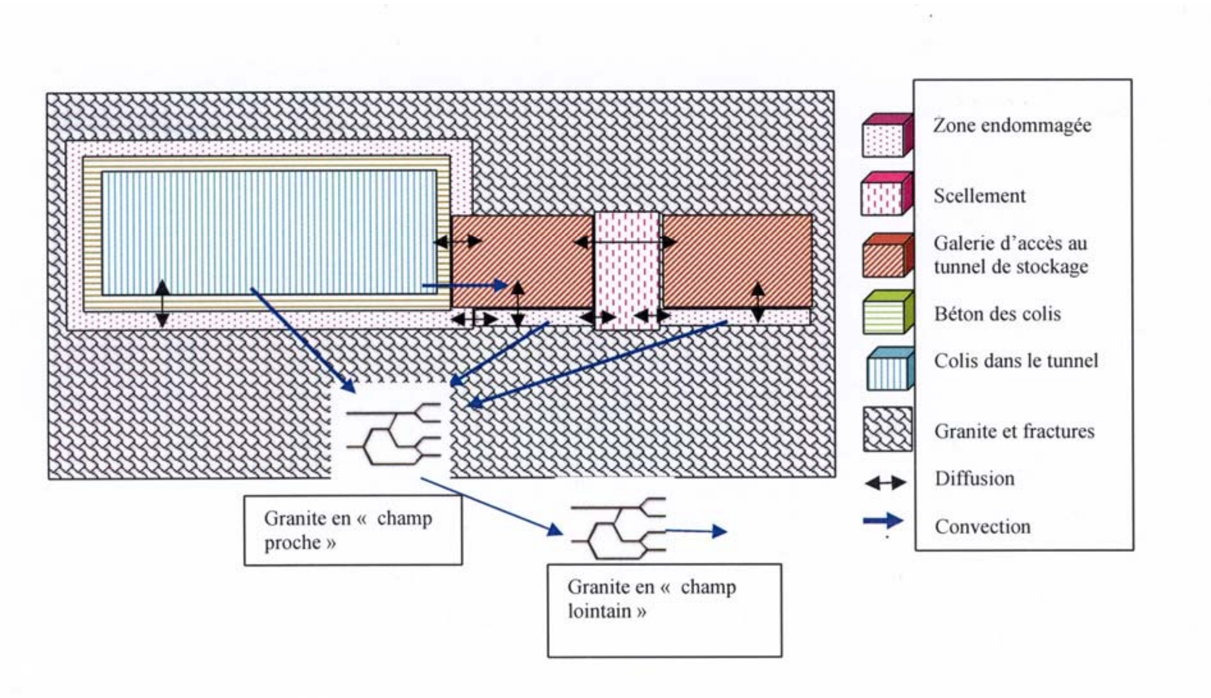


Figure 6.1-27 Representation of a B waste disposal cell

The plug of the repository tunnel is a seal of the same type as the drift seals. The values adopted for the hydraulic parameters are based on the equivalent permeability obtained during the test on scale 1 of a clay-based seal during the Tunnel Sealing Experiment (TSX) carried out at the underground laboratory in Canada, which is 10^{-11} m/s. This choice is prudent inasmuch as it understates the foreseeable performances following the test. The radionuclides migration in the seal mostly takes place by diffusion. Characteristics of the clay from which the seal is made also provide chemical retention performances.

Characteristics of the clay from which the plug is made also provide chemical retention performances via sorption and precipitation phenomena. The values of the parameters retained for the bentonite from which the body of the seal is made are the same as those adopted for the bentonite inside the C and CU disposal cells (see Table 6.1-17).

- **Access drift to the B waste disposal tunnel**

After possible transfer through the cell plugs, radionuclides migrate through the different drifts of the repository and from there may reach fractures in the granite wall.

The same properties are adopted for the excavation damaged zone as for the excavation damaged zone of the repository tunnels (see Table 6.1-12 to Table 6.1-14) in the reference scenario.

The backfill, made up of ground granite and bentonite, have low permeability (10^{-10} m/s).The properties of the backfill are such that it was also given retention properties. The retention values of parameters are determined as a function of a proportion of 15% of swelling clay (Table 6.1-15 and Table 6.1-16).

EDZ **Granite** All site models	Density: 2 600 kg/m^3	
	Thickness [m]	
	Reference drifts	Altered evolution scenario seal failure
All radionuclides	0.5	0.05

Table 6.1-12 Value of the thicknesses of the excavation damaged zone adopted at the wall of the engineered structure for the different situations studied

EDZ **Granite** All site models	**Permeability [m/s]**		**Effective diffusion De m^2/s**	**Diffusion porosity ω_d**	**Kinematic porosity ω_c**
	Reference	Sensitivity	Reference	Reference	Reference
All Radionuclides	1.10^{-9}	1.10^{-8}	2.10^{-10}	0.1	0.01
(K, De, ω_d, ω_c) : Conventional test values					

Table 6.1-13 Values of the hydraulic parameters adopted for the damaged zone of the granite

EDZ Granite M1	**Partition coefficient Kd [m^3/kg]**		**Delay factor [R]**		**Dispersivity [m]**	
Radionuclides	SEN	Sensitivity	SEN	Sensitivity	α_L	α_T
^{14}C	0.001	0	24.4	1	0.1	0.01
^{36}Cl	0	0	1	1	0.1	0.01
^{79}Se	0.0005	0.0005	12.7	12.7	0.1	0.01
^{93}Mo	0	0	1	1	0.1	0.01
^{99}Tc	0.2	0	4 680	1	0.1	0.01
^{126}Sn	0.001	0	24.4	1	0.1	0.01
^{129}I	0	0	1	1	0.1	0.01
^{135}Cs	0.1	0.005	2 340	118	0.1	0.01
^{229}Th	0.5	0.01	11 700	235	0.1	0.01
^{233}U	1	0.01	23 400	235	0.1	0.01
^{237}Np	0.5	0.05	11 700	1 170	0.1	0.01
^{241}Am	0.5	0.04	11 700	937	0.1	0.01
^{241}Pu	2	0.5	46 800	11 700	0.1	0.01
^{245}Cm	0.5	0.04	11 700	937	0.1	0.01
R = 1 + ρgrain Kd (1-ωd)/ ωd with Kd (EDZ) = Kd (granite matrix) Dispersivity defined in accordance with the length of the pathway						

Table 6.1-14 Values of the chemical retention parameters adopted for the damaged zone of the granite (Model M1)

Backfill of drift	Density: 2 000 kg/m^3				
	Permeability [m/s]	**Effective diffusion De [m^2/s]**	**Total porosity ω**	**Dispersivity [m]**	
	Reference	Reference	Reference	α_L	α_T
All Radionuclides	1.10^{-10}	1.10^{-10}	0.3	0.1	0.01
Permeability, porosity (ω_d, ω_c) and diffusion are test values taken from Report SR 97 The porosity accessible to diffusion = kinematic porosity, i.e. total porosity.					

Table 6.1-15 Values of the hydraulic parameters adopted for the backfill of drift.

Backfill of the drift	**Partition coefficient Kd [m^3/kg]**		**Delay factor [R]**	
Radionuclides	Reference	Sensitivity	Reference	Sensitivit y
^{14}C	0.00085	0	4.97	1
^{36}Cl	0	0	1	1
^{79}Se	0.000425	0.000425	2.98	2.98
^{93}Mo	0	0	1	1
^{99}Tc	4.67	4.5	21 800	21 000
^{126}Sn	1.65	1.65	7 700	7 700
^{129}I	0	0	1	1
^{135}Cs	0.1	0.0193	468	91.1
^{229}Th	0.875	0.459	4 080	2 140
^{233}U	15.9	15	74 200	70 000
^{237}Np	0.575	0.193	2 680	902
^{241}Am	2.23	1.83	10 400	8 540
^{241}Pu	1.85	0.575	8 630	2 680
^{245}Cm	2.23	1.83	10 400	8 540
R = 1 + density Kd (1-ωd)/ ωd where Kd is defined in accordance with the formulation of the backfill: 15% bentonite + 85% ground granite.				

Table 6.1-16 Values of the chemical retention parameters adopted for the backfill of drift

Transport regime and representation of the radionuclides path in the disposal tunnels

The migration of the radionuclides from the primary packages where they are released up to a fracture in the granite is represented by paths, the number of which depends on the fracturing of the granite in the tunnel walls. For B2 waste tunnels in M2 model, it is some ten of particles that are released in this way. The molar flux of radionuclides being transferred to the fractures are estimated by using an equivalent flow rate Q, which is a function of the hydraulic flow of the intercepted fractures, of the geometry of the contact between the envelope of the disposal packages and the granite as well as diffusion parameters of the concrete and the granite rock.

The radionuclide transfer regime in the concrete envelopes of the disposal packages is therefore a result of the hydraulic models taken directly into account by the radionuclide transport calculations. The models divide the disposal package envelope into a series of four compartments. The hydraulic model determines the regime (advection/diffusion) in each compartment.

Starting from the waste packages, the radionuclide paths go through the envelope of the disposal package and then follow the fractures network in the granite.

The logic of the calculation is the same for those radionuclides that pass through the plug/seal at the entrance to the tunnel and reach a fracture in the granite wall of the access drift; the different segments of the calculations correspond to the plug/seal, the backfill and the excavation damaged zone.

- **C waste cell and modules**

The C2 waste repository shafts , taken as reference in the calculations, contain two over-packs. In the case of M2 site model, the handling drifts include approximately 27 repository shafts. Each "average" module comprises 13 parallel drifts (M2 site model). The calculations are performed for a module containing approximately 450 repository shafts (which is 900 C2 waste packages). In the case of M1 site model, a fraction of module containing three disposal drifts constitutes the calculation unit. The variability of the hydraulic properties inside the module is integrated directly into the statistical distribution laws of the transport parameters.

Phenomenology of the radionuclides migration

After loss of the leak-tightness of the steel over-pack, the radionuclides migrate through the (60 cm thick) clay engineered barrier towards the granite wall of the repository cell. The transfer takes place by diffusion on account of the low permeability of the swelling engineered clay barrier. Radionuclides also migrate in part and following the same diffusive regime towards the overlying handling drifts through the cell plug made out of the same swelling clay (thickness 1.50m) (see Figure 6.1-28).

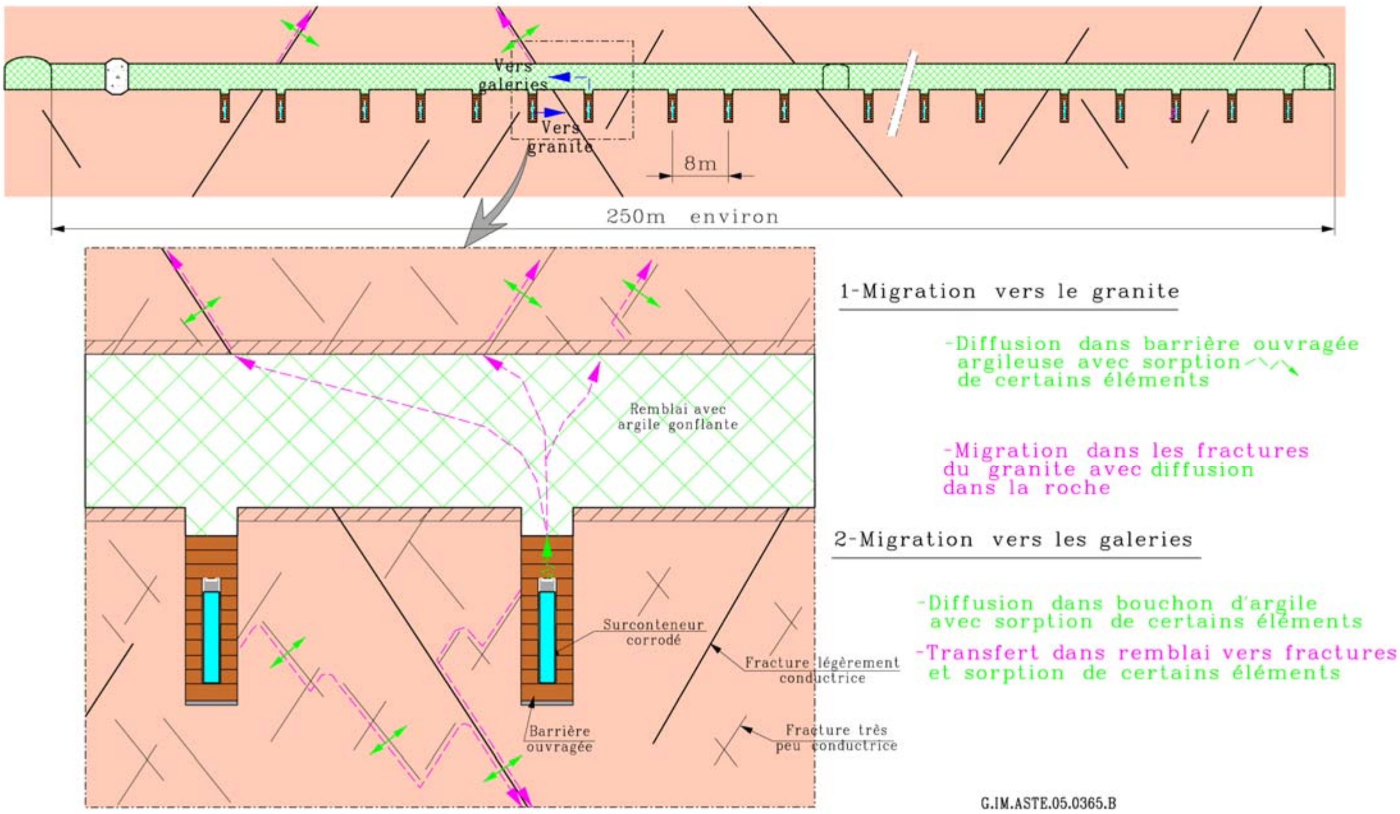


Figure 6.1-28 Phenomenology of the radionuclides migration in a C waste module

Representation of C waste modules

For the calculations, the two packages are represented as a single package that remains leak-tight for 1,000 years. After 1,000 years, the over-pack is no longer taken into account in the calculations (see Figure 6.1-29). The transfer of radionuclides can begin through the engineered barrier towards a conductive fracture intersected by the repository shaft, if such an intersection exists.

Part of the radionuclides migrates towards the handling drifts via the disposal cell plug, constituted, like the engineered barrier, from bentonite. The hydraulic and retention properties of the engineered barrier and plug bentonite taken as reference (or in sensitivity study) values in calculations are given in Table 6.1-17. Note that the anionic exclusion is taken into account in the bentonite for the anions carbon 14, selenium 79 and iodine 129.

Bentonite		K= 1.10^{-11} m/s Density: 2 700 kg/m^3 Kinematic porosity ω_c: 0.18						
Radionuclides	**Diffusion coefficient De [m^2/s]**	**Diffusion porosity ω_d**	**Partition coefficient Kd [m^3/kg]**		**Delay factor [R]**		**Solubility limit Csat [mol/m^3]**	
	Reference	Reference	Reference	Sensitivity	Reference	Sensitivity	Reference	Sensitivity
^{14}C	5.10^{-12}	0.05	0	0	1	1	soluble	soluble
^{79}Se	5.10^{-12}	0.05	0	0	1	1	3.10^{-6}	soluble
^{126}Sn	5.10^{-10}	0.36	11	3	52800	14400	5.10^{-6}	1.10^{-2}
^{129}I	5.10^{-12}	0.05	0	0	1	1	soluble	soluble
^{135}Cs	5.10^{-10}	0.36	0.1	0.06	481	289	soluble	soluble
^{229}Th	5.10^{-10}	0.36	3	3	14400	14400	1.10^{-6}	2.10^{-6}
^{233}U	5.10^{-10}	0.36	100	10	480000	48000	1.10^{-4}	2.10^{-4}
^{237}Np	5.10^{-10}	0.36	1	0.7	4800	3360	1.10^{-4}	2.10^{-4}
^{241}Am	5.10^{-10}	0.36	12	1.2	57600	5760	7.10^{-4}	7.10^{-3}
^{241}Pu	5.10^{-10}	0.36	1	0.7	4800	3360	7.10^{-6}	3.10^{-3}
^{245}Cm	5.10^{-10}	0.36	12	1.2	57600	5760	2.10^{-4}	2.10^{-3}

ρgrain, K, De, ωd, ωc: are those adopted for clay type MX 80 [45].
R = 1 + ρgrain Kd (1-ωd)/ ωd where Kd is adopted for clay type MX 80.
The solubility limits imposed in the bentonite (engineered barrier of the body and plug/seal) are test values taken from Report SR 97 = high values adopted (conservative) for bentonite and granite.

Table 6.1-17 Values of the hydraulic, transport and chemical retention parameters adopted for a swelling clay buffer type MX80

In the normal scenario, there is no excavation damaged zone in the granite wall of the shaft given its minor thickness (cm) and the swelling of the bentonite engineered barrier [x]. The plug failure scenario takes into account the case in which the contact between the bentonite and the rock is not continuous and allows for a 1 cm thick water conductive excavation damaged zone along the whole plug.

In the handling drifts, the backfill has the same properties as those of the access drifts to the B waste tunnel (see Table 6.1-15 and Table 6.1-16).

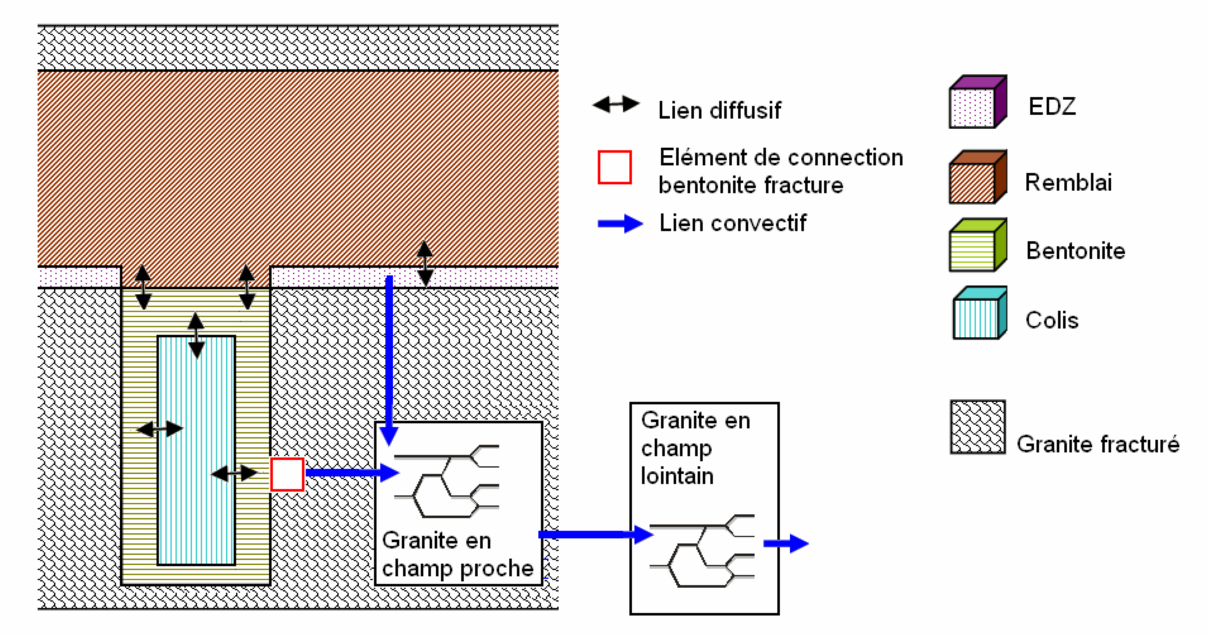


Figure 6.1-29 *Transfers modelling in a C waste disposal cell (DFN approach in the granite)*

Transport regime and representation of the radionuclide path in a C waste repository module

For each disposal cell, the calculation allows for a particles path from the vitreous matrix, then passing through the engineered barrier towards a fracture in the granite wall. For the bentonite engineered barrier, the calculation includes a series of 4 to 7 successive compartments to take into account the radionuclides concentration gradient between the package and the fracture in the granite precisely in a regime of transfer by diffusion. A molar flow of radionuclides thus constitutes the flux entering the fracture intersected by the cell. Part of the radionuclides is transferred towards the handling drifts, particularly in the case of repository cells that do not intercept any fractures. Particles tracking is carried out upright the repository cells in the handling drift backfill in order to take account of the "drift" transfer pathway. This determines a "drift" path. The calculation segments in the drift are: the excavation damaged zone of the granite wall and the backfill of the drifts. The calculation distributes the released radionuclides inventory of between the package path → engineered barrier → fracture, and the package path → repository cell plug→ handling drift backfill → fracture. The outflows leaving the models are obtained by summing the "fracture" path and the "drift" path.

However, in the calculations, the emplacement of the repository modules and the cells are not adjusted for fracturing as could be done in the real situation. It results in a "random" location that could be considered as being close to a situation of default of fracturing characterization (see Figure 6.1-30).

Thus, in the case of the normal evolution scenario, 10% of the most hydraulically pessimistic paths have been eliminated (this percentage corresponding to a conceivable release from the repository cells during "progressive" characterization work). In the case ranked as a characterization default, all paths are taken into account for the transport calculation.

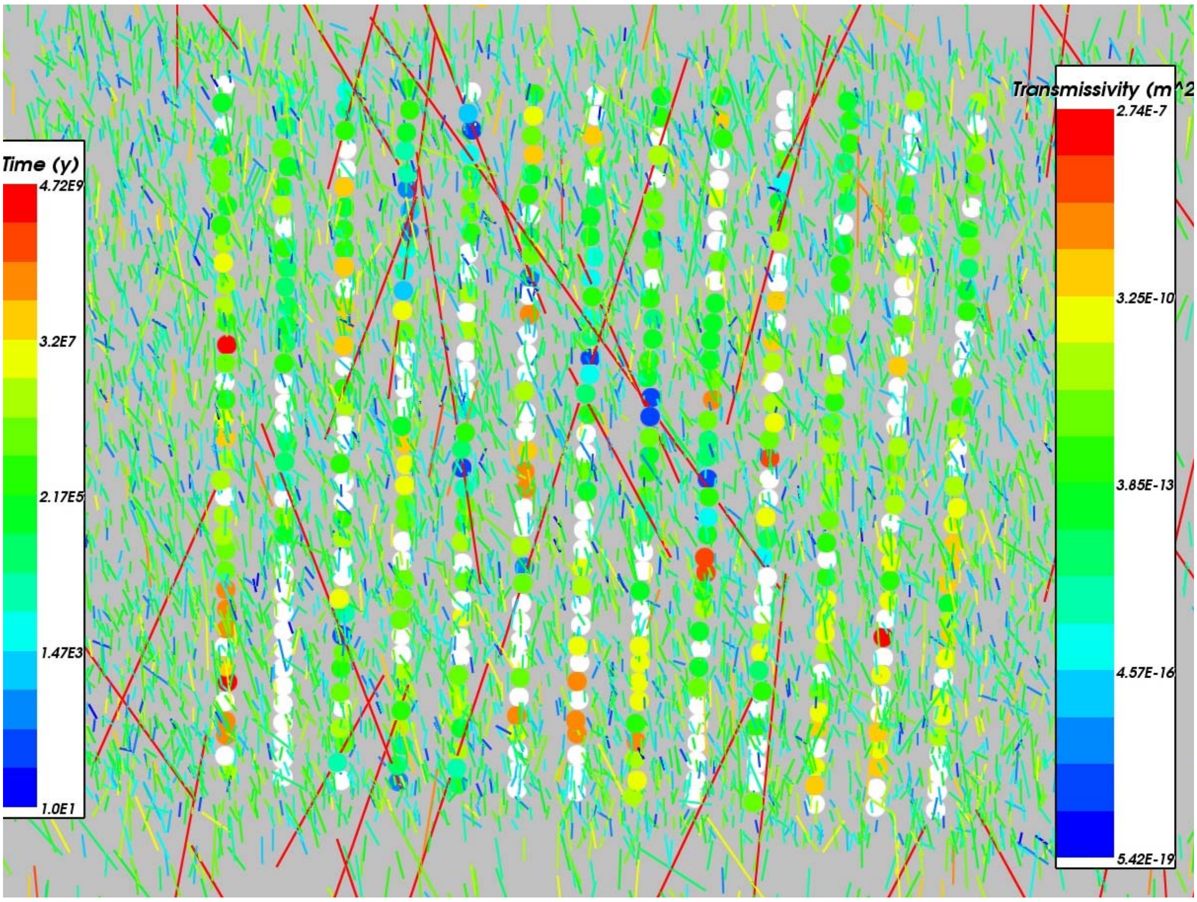


Figure 6.1-30 *Correlation between the transmissivity of the intercepted fractures and the advective transfer time of the corresponding "fracture" path. The white points correspond to cells with no direct connection to the fracture network. The dark blue points correspond to short transfer times that could be rejected by the characterization process.*

- **Spent fuel module and cell**

With regard to the calculations, the representation of repository cells and spent fuel modules is very close to that of C wastes. The difference lies in the fact that the release of radionuclides only affects one or 5 defective containers depending on the scenario, and it takes place over 20,000 years via the container failure: a 5 mm^2 hole (see Section 6.1.5.4 above). It is assumed that the failure is facing a conductive fracture in the repository cell: thus, the path concerns the transverse thickness of the engineered barrier (35 cm).

The transfer pathway through the handling drift is dealt with in the same way as for C wastes with particles paths through the excavation damaged zone of the handling drift and through the backfill up to a fracture leading to their entry into the fractures network of the granite (see Figure 6.1-31). The containers are placed in cells corresponding to pessimistic hydraulic paths.

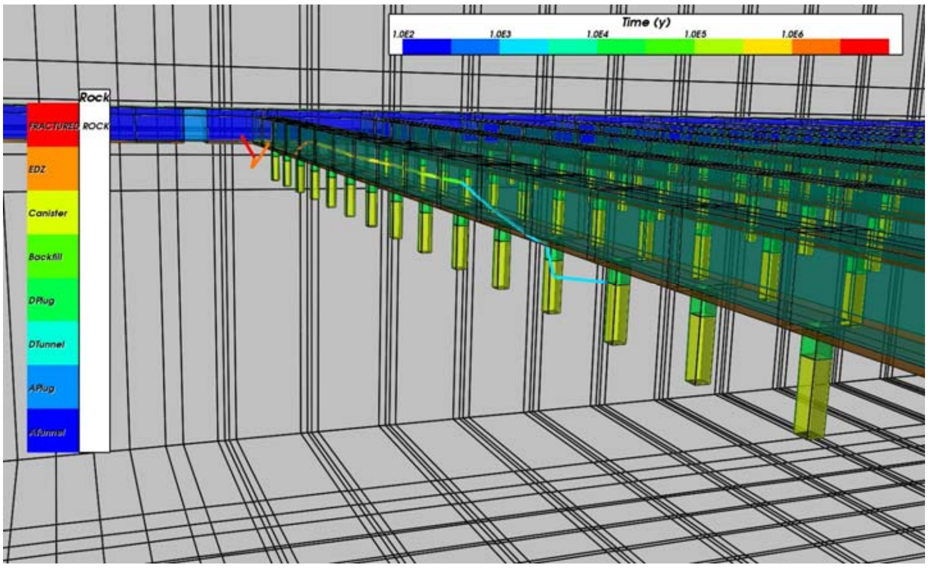


Figure 6.1-31 *the case of a particle emitted by a CU2 cell and taken up by the excavation damaged zone of the drift*

6.1.5.6 Representation of the engineered access structures

In the case of a repository in a granite medium, the radionuclide inventory that reaches the surface-bottom engineered access structures (shafts and ramps) by means of other repository drifts is negligible. In fact, the conductive faults intersected by the drifts joining the repository modules and the engineered access structures constitute hydraulic barriers to radionuclide migration. The calculations assume, for that respect, that all of the radionuclides released by the cells of a module migrate towards the granite in the area covered by the granite block in which the module is located.

6.2 Calculation tools used for modelling the transportation of the radionuclides

Two types of simulation tools were used for the calculations. In the case of simulations in "fracture networks", the tools are those used abroad for studies on repositories in geological formations. International co-operation programmes, in particular concerning the "Connectflow" project, deal with the development of embedded models in "discrete fracture networks" and in continuous porous media in order to ensure the continuity of different scales of modelling to be treated in transfer simulations. In the case of simulations in "continuous porous media", the tools used are of the same type as those used in clay media (see Table 6.2-1).

Models	Codes
Hydrogeology and "particle tracking" in continuous porous media	- Connectflow (NAMMU component, 3D modelling, finite elements). - Geoan (3D modelling, finite differences). - Porflow (3D modelling, finite differences).
Hydrogeology and "particle tracking" in discrete fracture networks.	- Connectflow (NAMMU component, 3D modelling, finite elements). - FracMan (generation of discrete fracture networks) and MAFIC (hydraulic resolution of the networks, 3D, finite elements).
Transport in continuous porous media.	- PROPER (COMP-23 component, modelling in segments of the engineered barrier, finite differences). - Goldsim (volume modelling of engineered barriers). - Porflow.
Transport in discrete fracture networks.	- PROPER (FARF-31 component, 1D modelling 1D stream tube concept). - PathPipe (conversion of networks of tubes for transport) and Goldsim (modelling in networks of 1D pipes).

Table 6.2-1 Simulation tools used for modelling the transportation of radionuclides

Before carrying out the performance calculations, Andra tested the different softwares to be used. These tests made it possible to ensure the feasibility of the calculations envisaged and to verify that the codes used had been validated.

6.3 Calculation results and main lessons drawn

The purpose of the calculations performed is not to evaluate repository impact on a particular site as the data required to proceed with this analysis is not accessible. Furthermore, such an evaluation would be of little value, the objective was not to carry out a selection of particular granite massifs.

The approach is not therefore to present the results of calculations in terms of dose but to focus on indicators that represent the safety functions performances. These indicators cannot be compared directly to regulatory references, standards or nuclear safety authority guidelines. Neither can they be used to compare one site model with another; this is not the purpose of the calculation. Actually, if one site has less favourable characteristics from the point of view of a given safety function, it is not necessarily disqualified from the point of view of the safety of the entire repository which combines different functions within one system.

However, from the analysis of the calculations for each site configuration and more generally, the following informations can be drawn:

- The characteristics of the granite massifs that have the greatest influence on safety functions performance, either in the absolute (i.e. on all site models) or in a specific morpho-structural context (one site model in particular),
- The way in which the engineered elements supplement or provide a redundancy in relation to the performances of the sole host formation.

In addition, the use of several types of methods (calculations in continuous porous media, calculations in discrete fracture networks - described hereinafter by the term "DFN calculation") and different softwares provide useful lessons on the type of information that would be accessible depending on the methods employed if an analysis was to be carried out on a real site. This point is highlighted in the conclusions relating to each safety function.

The calculations are presented in accordance with the three safety functions relating to transfer by water:

- Limiting water circulation
- Restricting the release of radionuclides and immobilizing them in the repository
- Delaying and reducing the migration of radionuclides.

Besides the normal evolution scenario (SEN) representing the performances of the safety functions in a context corresponding to the most probable phenomenology, scenarios described as "altered evolution scenarios" (SEA) correspond to accidental type situations. Thus, have been addressed cases of waste containers failure in series, seals failures and poor recognition of the average fracturing of the granite. Altered evolution scenarios provide additional information on safety functions. They provide more information than the normal evolution scenario on the importance of each component by revealing the consequences of a loss of functionality of each one of them. They also allow verifying that the repository remains robust in case of failures, even when improbable.

The results of altered evolution scenarios are therefore presented jointly with those of the normal evolution scenario, in order to supplement the information as it is read.

6.3.1 Lessons learnt relating to the function of "precenting the advection of water"

- **Positioning of a repository on the scale of the massif**

The purpose of the function is to limit the flow of water in the engineered structures which can both alter the repository materials and result in radionuclides migrating toward the environment.

The adaptation of the repository for different scales of fracturing makes it possible to meet this objective. In order to appreciate how a massif could accommodate a repository, a first stage would be to determine appropriate locations from the hydrogeological point of view. On the different site models for which calculation was performed, possible locations have therefore been determined on the basis of hydrogeological models produced on the regional scale (see Section 6.1.3.1 above).

This model provides a basis for the calculation of hydraulic trajectories between the repository and potential natural outlets for different possible locations. Thus, positions can be proposed for a repository according to the criteria defined by the designer: it is, for example, permissible to favour positions giving rise to long hydraulic paths, or slow transfer times, or low water flows. The final selections also take account of the Basic Safety Rule III.2.f criteria such as observance of a minimum distance from large faults.

These techniques could be employed to identify a possible zone for a repository within a massif.

- **Positioning of repository modules**

In the event of the construction of a repository in a granite massif, one approach of the survey consists of determining the implementation of the repository modules. In particular, this approach enables the positioning of the repository modules and cells to be fixed during the repository construction works on the basis of a precise characterization of the implementation sites. In the generic calculations performed, the *positioning of the repository modules* cannot be optimised as it would be in the context of a complete site survey. A statistical adjustment has been carried out on the basis of the extension of the fractures of the discrete fracture network (DFN) model: for B waste, the maximum fracture dimension intersected by the disposal tunnels is, by convention, 80m, and 300m in the case of C waste modules. This is a pessimistic approach in that, for some calculations (C waste in particular), it resembles a situation of characterization error, insofar as in a real situation the repository would gradually be adapted in order to make the best possible use of the available sound blocks of granite. The results presented below must therefore be weighted to take account of fact that repository module locations, distributed randomly, do not incorporate such an optimisation. The analysis is, however, indicative of matters that would underlie characterization works and surveys of a site prior to and during disposal.

- **Transport regime in the engineered structures of the repository**

An indicator used to provide a good indication of the hydraulic regime in the engineered structures of the repository is the adimensional Péclet's number (see Inset 7), the ratio between the characteristic times of diffusion and advection. In the case of low values (in particular values lower than 1), the hydraulic regime is dominated by diffusion. This indicator expresses the ratio between advection and diffusion but does not, however, judge the speed with which each type of transport takes place in the absolute. In order to do this, Darcy's velocities (used to assess the rapidity with which the transfers of water take place) allow assessing the rapidity of advective circulations.

Inset 7 Definition of the Péclet's number

$Pe = (Td/Tc)$
Where:
$Td = L^2\omega/De$
$Tc = L\omega/(K.gradH)$.
and:
Td, characteristic migration time by diffusion [year],
Tc, characteristic migration time by advection [year],
L, migration distance [m],
ω, the total porosity in the backfill of the drift [-],
De, the effective diffusion coefficient in the backfill [m²/yr],
K, permeability of the backfill [m/yr],
GradH, the hydraulic head [m/m] in the drift, calculated from simulations in homogeneous media.

In the drifts, the transfer regime is essentially determined by the permeability of the backfill, the transmissivity of the fractures of the granite walls and the gradient. Obtaining a diffusive regime in the repository drifts seems possible on the basis of the calculations performed in continuous porous media representing various possible sites (see Table 6.3-1), but depends on the backfill permeability. A failure in the backfill setting up and degrading its permeability by an order of magnitude (10^{-9} m/s instead of 10^{-10} m/s) in a section of drifts barely affects the transfer regime. On the other hand, if the permeability of the backfill was more significantly degraded (around 10^{-8} m/s) the hydraulic regime inside the repository could be affected. To put such a backfill in place seems therefore to be a useful design but for which the performances would have to be defined according the hydraulic regime of a given site.

	Near-field granite		**Backfilled drift**		
	Hydraulic conductivity (m/s)	**Hydraulic head (m/m)**	**Hydraulic conductivity (m/s)**	**Péclet's number ($Pe=T_d/T_c$)**	**Drift transfer regime**
B2 Waste (calculation carried out in 54 m of connecting drift between two seals)	10^{-11}	10^{-3}	10^{-10}	0.039	diffusive
			10^{-9}	0.16	diffusive
			10^{-8}	0.39	diffusive
	10^{-11}	10^{-2}	10^{-10}	0.39	diffusive
			10^{-9}	1.6	mixed diff/adv
			10^{-8}	3.9	advective
	10^{-10}	10^{-2}	10^{-10}	0.49	diffusive
			10^{-9}	3.8	advective
			10^{-8}	14	advective
C2 waste (calculation performed in 15 m of handling drift between the cell on the lateral margin and the seal)	10^{-11}	10^{-3}	10^{-10}	0.012	diffusive
			10^{-9}	0.099	diffusive
			10^{-8}	1.1	mixed diff/adv
	10^{-11}	10^{-2}	10^{-10}	0.11	diffusive
			10^{-9}	1.1	mixed diff/adv
			10^{-8}	12	advective
	10^{-10}	10^{-2}	10^{-10}	0.13	diffusive
			10^{-9}	1.1	mixed diff/adv
			10^{-8}	10	advective

Table 6.3-1 Estimate of Péclet's number in the backfilled drifts, ("continuous porous media" approach)

Robustness of the function of "preventing water circulation"

It is important to verify if performances of the function of "opposing water circulation" are robust in the event of a failure of any kind.

The "seal and cell plug failure" altered scenario envisages a situation consisting of not interrupting the continuity of the excavation damaged zone of the granite wall of the horizontal engineered structures over a thickness of approximately 5 centimetres. It shows the role of the backfill in that it is sufficiently efficient for the failure of the seal to not significantly modify the hydraulic transfer regime in the drifts. In a generic context, it is difficult to distinguish the role of the seals from that of the backfill. It should be noted that, in the context of implementation of a repository in a given massif, sound blocks would be first reserved for the location of cells; while the drifts leading to them could, however, intersect conductive structures at the edges of these blocks. In such a configuration, sealing the drift on either side of the fracture crossed could prove to be a more effective layout than backfill alone. In any case, in the context of a real site configuration, it would be possible to favour either the seals or the backfill depending on the objective pursued, and to distinguish more clearly the roles of one or the other than in the case of a generic configuration.

6.3.2 Lessons learnt relating to the function of “restricting the release of radionuclides and immobilizing them in the repository"

This function covers all physical and chemical phenomena that tend to prevent the release in solution of the radionuclides. This concerns all of the following:

- Flow conditions in the cells favouring the durability of the waste,
- The leak-tightness of metal containers that isolate the radionuclides from the water,
- Chemical conditions that favour the insolubility of chemical elements.

6.3.2.1 Transport in the disposal cells

The function of "restricting the release of radionuclides and immobilizing them in the repository" is based on the establishment of a diffusive regime inside the cells, especially in the case of vitrified wastes which are the most sensitive to transport conditions in their surroundings. In this respect, indicators such as the Péclet's number in the cell are not directly performance indicators of the function but make it possible to determine if the latter can act in favourable conditions. Diffusive conditions inside the cells and low renewal of the water allow considering with greater confidence the release models resulting in slow speeds.

In C waste and spent fuel cells, the Péclet number shows that the regime is diffusive in all cases (see Table 6.3-2). The most influential parameters contributing to maintaining this regime are on one hand the presence of the engineered barrier and on the other, the transmissivity of minor fractures around the cell. The "dead-end" architecture of the tunnels also limits water ingress.

Estimate of Péclet's number in the bentonite engineered barrier (C2 waste)	Near-field granite		Bentonite engineered barrier		
	Hydraulic conductivity (m/s)	Imposed hydraulic head (m/m)	Induced hydraulic head (m/m)	Péclet's number (anion) ($Pe=T_d/T_c$)	Transfer regime in bentonite engineered barrier
Horizontal gradient (calculation performed on 0.60 m of bentonite in the radial direction)	10^{-11}	10^{-3}	$1.1.10^{-3}$	~ 0.0004	diffusive
	10^{-10}	10^{-2}	$2.2.10^{-2}$	~ 0.008	diffusive
Vertical gradient (calculation performed on 5.30 m of bentonite in the vertical direction)	10^{-11}	10^{-3}	$1.1.10^{-3}$	~ 0.003	diffusive
	10^{-10}	10^{-2}	$1.1.10^{-2}$	~ 0.033	diffusive

Table 6.3-2 Estimate of Péclet's number in the bentonite of the C2 waste disposal cells ("Continuous porous media" approach)

Relative to the calculations in continuous porous media, which do not discriminate one cell from another, the calculations in discrete fracture networks (DFN) show that even if the regime remains diffusive inside the cell in all cases, the situations may vary locally depending on the fractures intersected by the cells.

Thus, use of the results obtained from the M1 site model (model for which the granite fractures have transmissivities in a range of relatively high values) show that out of ten statistically representative C waste repository drift locations:

- Three cell locations are not crossed by water flows as they do not intersect any fracture,
- Four locations have flows of less than one litre per year,
- Three locations have flows greater than one litre per year (ten litres up to several decalitres per year).

It is worth remembering that these results do not take into account an optimisation of the positioning of the cells as could be done by selecting the blocks of granite in which they would be constructed. As the fracturing is caused randomly in the model, this tends to amplify the role of the engineered barrier.

In the context of the installation of repository tunnels in a real massif, surveying minor fractures and selecting the soundest blocks of granite would reduce the size of the engineered barrier. In any event, the calculation performed shows that the latter constitutes an effective system if the massif were to have minor dense fracturing or in the case of a characterization error. In fact, the regime remains diffusive in all configurations tested.

In B waste cells, the flow of water passing through disposal tunnels depends mainly on the transmissivity of minor fractures in the granite wall. The location, in principle, of repository tunnels in barely fractured blocks of granite may result in cases in which the granite is practically impermeable: The transfer regime is diffusive between disposal tunnels and access drifts through the swelling clay seal. In the case of slightly conductive fractures, low flows of water pass through the repository tunnel between the fractures in the wall. The flows evaluated by the calculations in the discrete fracture network (DFN) are very low: tens to hundreds of litres per year for volumes of repository tunnels of the order of 10 000 or 20 000 m^3 (see Table 6.3-3).

In the case of concrete packages with the capacity for reinforced confinement (B5.2), the calculations take into account a degradation of the packages at 10 000 years represented by the loss of their initial hydraulic performance (low permeability – 1.10^{-13} m/s and low porosity – 10%). The results of the calculations show that water flows increase by a factor of 5 after degradation of the packages, indicating the role of hydraulic deceleration played by the packages in the repository tunnels (see Table 6.3-3).

B5.2 waste tunnel	Before 10,000 years	After 10,000 years
Flow of water in the tunnels	30 l/year	150 l/year

Table 6.3-3 Comparison of the water flows in B5.2 waste disposal tunnels before and after 10 000 years (Calculations in DFN for M1 site model, case of granite with slightly conductive minor fracturing)

6.3.2.2 Role of the containers

Another aspect of the function of "restricting the release of radionuclides and immobilizing them in the repository" is the leak-tightness provided by the metal containers. The first analyses carried within the performance calculations consider both efficient and failed containers (a limited number in normal evolution scenario and a greater number in the altered evolution scenario). The comparison of these two situations enables an initial assessment to be made of the value of this system.

In the case of *spent fuel*, the copper container has sufficient durability for the safety demonstration period to ensure that there is no release in a normal situation. In the event of package failure including water reaching the assemblies at the end of approximately a century, releases occur gradually as the matrix dissolves. This spans several thousand years (approximately 5 000 years) if a prudent dissolution model of radiolytic dissolution is considered. With a conventional dissolution model such as is adopted internationally, releases take place over duration in excess of one million years. In the case of a CU container failure, the radionuclides released, adsorbed (or low solubility elements in particular, do not immediately leave the cell as they are stopped or delayed by the engineered barrier (see Chapters 6.3.2.3 to 6.3.3.1).

In the case of C waste, the analysis concerns the comparison of two cases: the case of a repository module in which the over-packs "normally" lose their integrity at the end of 1000 years and the case of a repository module in which a fraction of the over-packs (5 %) fail at the end of approximately a century. The analysis is performed on the least adsorbed (or absorbed) radionuclides (^{129}I and ^{79}Se). Some lessons can be drawn from the case of iodine 129 on C2 waste through the examination of Figure 6.3-1. The history of molar flows shows, in the case of a fraction of failed packages, the more rapid arrival of radionuclides from the near-field in the granite during a period of several thousand years. After that, molar flows are similar in both cases dealt with. The role of the containers is therefore not very visible in this calculation.

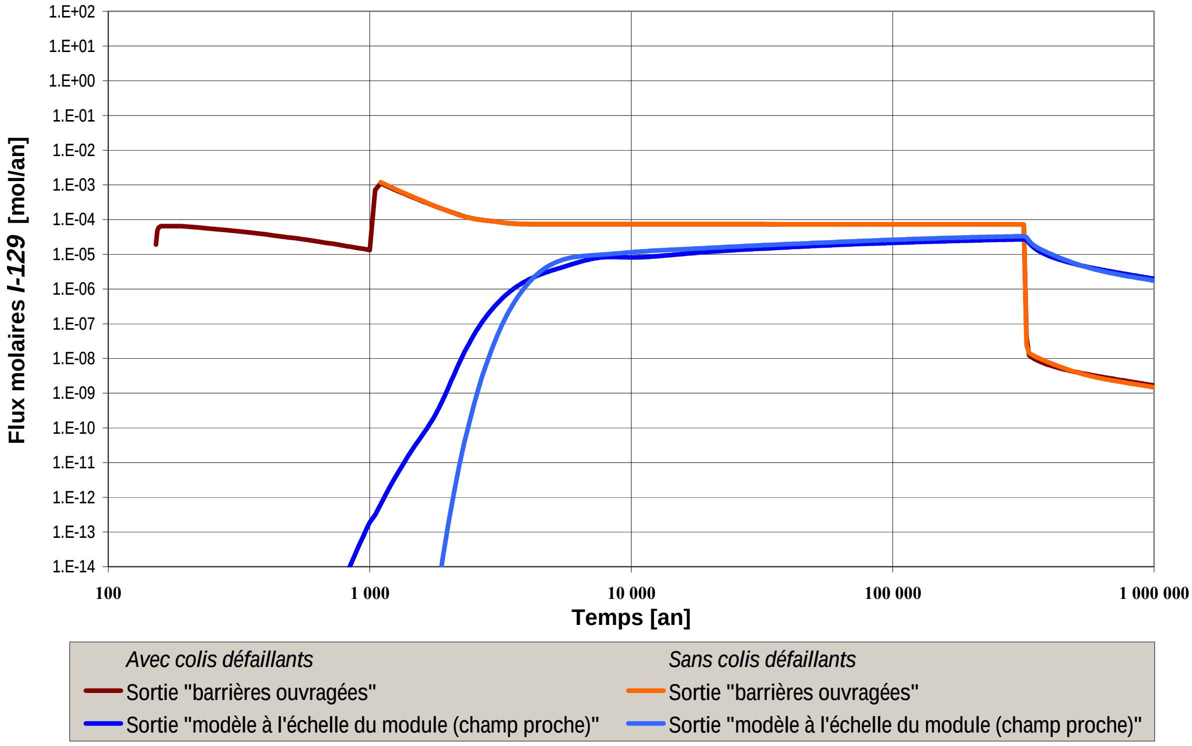


Figure 6.3-1 *Molar flows emitted by a C2 waste module - case of a scenario with no failed container and a scenario with a failed container (Site model M1 - DFN approach)*

It is also important to remember that, in a generic context and given the data available, it was not possible to explicitly represent the effects on the transport parameters that would be induced by significantly higher temperatures than natural ones, such as the one that could prevail when the radionuclides leave failed containers. Taking such effects into account could better highlight the role of the metal over-packs for C waste.

6.3.2.3 Precipitation in the disposal cells

The function of "restricting the release of radionuclides and immobilizing them in the repository" is also the result of restricting the dissolution of radionuclides into solution. Indirectly, the calculations allow measuring the effects of this function by identifying the radionuclides for which migration is controlled by reaching the solubility limit in the near-field. According to designs proposed by Andra, the sorption of elements in swelling clay (for radionuclides issued from C waste or spent fuel cells) and in concrete (in the case of B wastes) generally constitutes a limiting factor. For weakly sorbed radionuclides, the limit of solubility, however, controls the transfer. This is the case, in particular, of selenium whose flow is attenuated by several orders of magnitude by taking into account solubility in bentonite (see Figure 6.3-2).

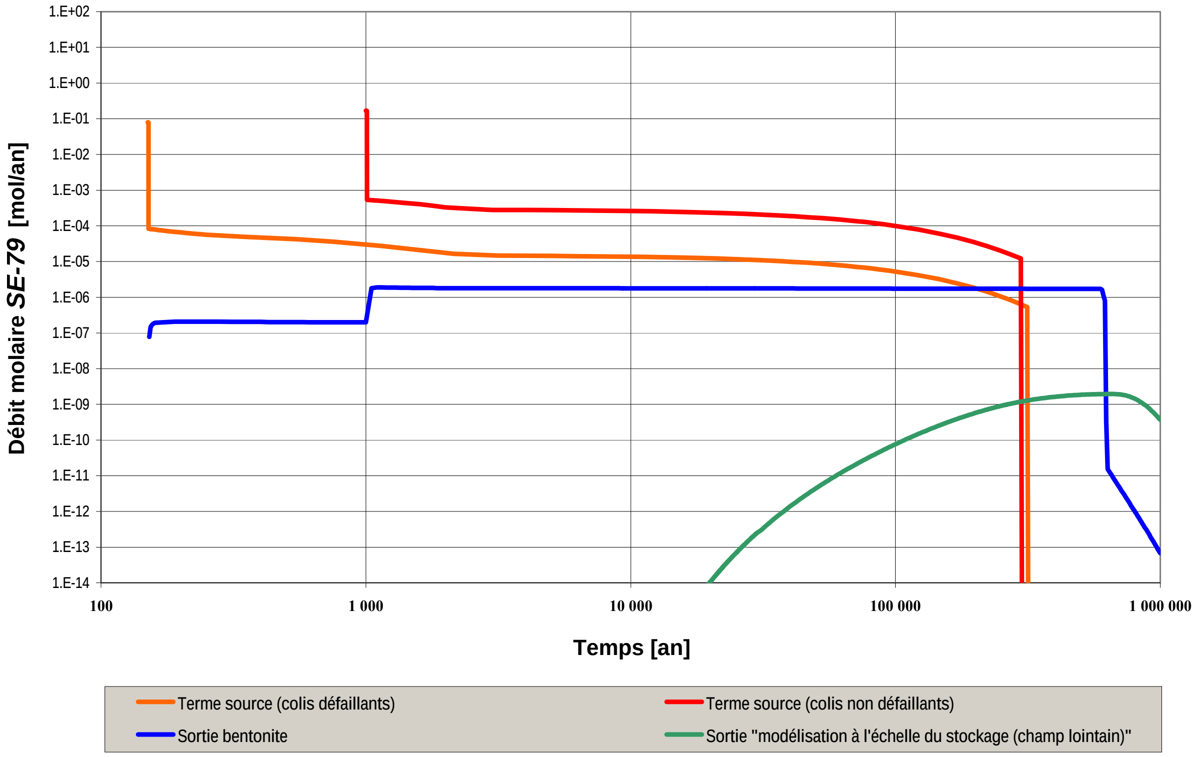


Figure 6.3-2 Limitation of molar flows by solubility - example of selenium in the case of a C2 waste module (massif M1 - DFN approach).

It is also notified, in the case of selenium 79 released by B2 wastes, element controlled by solubility and sorption in concrete, that the molar flow is also significantly attenuated by approximately three orders of magnitude when leaving the concrete packaging (see Table 6.3-9).

6.3.3 Lessons Learnt from the function of "delaying and reducing the migration of radionuclides"

The performances relative to this function are conveyed by attenuation indicators (ratio between the mass of radionuclides leaving one repository compartment compared with the mass that entered; ratio between the maximum flow leaving the a compartment compared with the maximum entering flow) and delay indicators (difference between the time required for the appearance of the maximum flow leaving and entering a compartment). These indicators can be evaluated in various locations:

- When leaving B waste cells, they show, by comparison with the wastes release’s record, the benefit of assigning hydraulic performances to B packaging, in particular by comparing the case in which these packages have hydraulic performances with the case in which these packages have only chemical performances associated with degraded concrete;
- When leaving the engineered barriers of C waste and spent fuel cells, they allow to assess the contribution of these argillaceous elements, notably when compared with that of the host formation;
- When leaving the near-field in the granite massif, they allow to assess the performance of the minor fracturing and the determining factors of these performances;

- When leaving the far-field in the granite massif, they can be used to assess the overall attenuation capacity of the massif and the transfer times within, as well as the characteristics that have the greatest influence on these performances.

6.3.3.1 Delay and attenuation in the disposal cells

The performances of the function in the cell are a direct reflection of the parameters of the model: partition coefficient Kd representing sorption in concrete (in the case of B waste tunnels) and swelling clay (in the case of other disposal cells).

In the case of B waste, a useful illustration of the role of the concrete container, which, as has already been seen, played a role in the protection of wastes from water flows, is the comparison of releases in the near-field by a non-degraded concrete container with hydraulic performances and the case of a container that has only chemical performances associated with degraded concrete. The calculations performed in continuous porous media indicate that the role of the container is clearer when the granite wall of the cells is more fractured. In all cases (standard container or reinforced confinement container) calculations in discrete fracture network (DFN) show that the concrete envelope of B waste clearly influences retention of highly sorbed radionuclides in the concrete (such as actinides). Comparison of the radionuclides molar flows for different values of partition coefficients Kd emphasises the role of the concrete in retaining selenium 79 (see Figure 6.3-3). In the case of very weakly sorbed radionuclides (such as iodine 129) the effect is negligible; it is not visible in the case of the most mobile or long-lived elements such as chlorine 36.

Table 6.3-4 shows, in the case of B2 waste cells, the percentages of radionuclides (by mass) that have left the cell and the granite during simulation. Thus, in the case of selenium 79, this indicator relative to the scenario of a concrete container shows that the percentage of mass transferred by the concrete is only approximately 15% at the end of a million years compared with their initial mass (DFN calculations), in view of the chemical performances of the concrete.

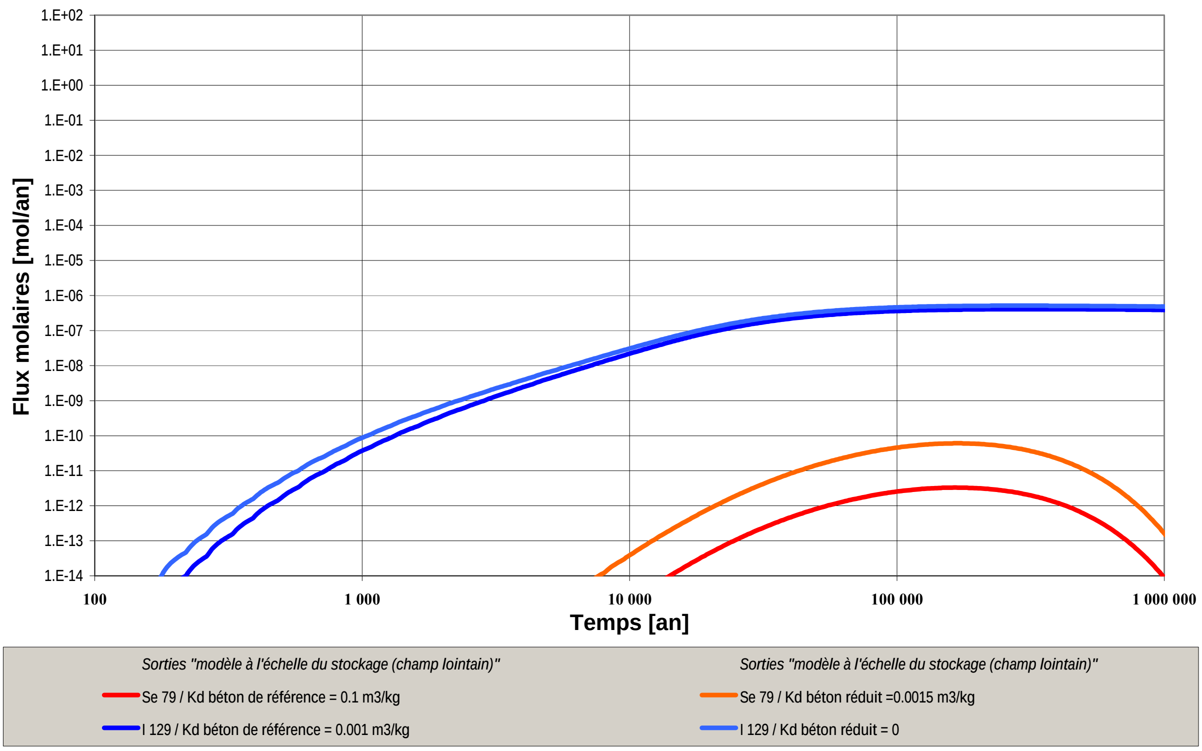


Figure 6.3-3 *Radionuclide retention by the concrete: role of the Kd sorption coefficient. Case of a B2 cell, (massif M2 - DFN approach)*

B2 waste – M1 site Ratio of masses that have passed through interfaces to the initial mass of the inventory							
Radionuclides	***Initial inventory (1 tunnel)***	***Source term***	***Packages***	***Concrete packaging***	***EDZ***	***Near-field***	***Far-field***
	Mass m_0 (mol)	*Mass m_i (mol) / Mass m_0 (mol)*					
^{36}Cl	2.02	100%	98%	98%	98%	97%	93%
^{79}Se	1.25	100%	91%	15%	13%	7%	4%
^{99}Tc	4410	100%	99%	99%	8%	0%	0%
^{129}I	8.69	100%	100%	100%	100%	100%	100%
^{135}Cs	30.84	100%	99%	99%	38%	0%	0%

Table 6.3-4 *Case of a B2 waste cell, M1 site model. Integrated masses leaving the interfaces of the cell and the granite during the simulation period, 1Myear, compared with the initial mass.*

In the case of C waste and spent fuel cells, the delay and attenuation performances in the cells are less dependent on the local fracturing on account of the presence of the bentonite engineered barrier. Non sorbed radionuclides diffuse through the latter and migrate outside the cell in one million years. The migration of radionuclides sorbed by the bentonite like actinides, tin 126 or to a lesser extent caesium 135, is significantly delayed. Table 6.3-5 shows, in the case of spent fuel cells, the percentages of radionuclides (by mass) that left the cell and the granite during the simulation. Thus, in the case of caesium 135, this indicator related to the scenario of a failing spent fuel container shows that the percentage of mass migrating outside the cell through the clay buffer is only approximately 8% at the end of 100,000 years and 15% at the end of one million years, compared with their initial mass (DFN calculations). This table also shows that iodine 129 has migrated outside the near-field granite close to 1 million years.

CU2 waste – M1 site Ratio of integrated masses that have passed through interfaces with the initial mass of the inventory							
Radionuclides	***Initial inventory (1 package releasing after 150 years)***	***Source term/ Initial inventory***	***Package/ Initial inventory***	***Bentonite/ Initial inventory***	***EDZ drift/ Initial inventory***	***Near-field/ Initial inventory***	***Far-field/ Initial inventory***
	Mass m_0 (mol)	*Mass m_I (mol) / Mass m_0 (mol)*					
^{14}C	$5.32.10^{-3}$	79%	9%	5%	0.3%	0%	0%
^{79}Se	$3.2.10^{-2}$	99%	2.5%	1.5%	0.5%	0%	0%
^{126}Sn	0.28	100%	85%	0.1%	0%	0%	0%
^{129}I	1.2	100%	100%	66%	38%	100%	100%
^{135}Cs	4.13	100%	100%	15%	8.5%	0%	0%

Table 6.3-5 *The case of a spent fuel cell, M1 site model. Integrated mass that passed through the interfaces of the cell and the granite during the simulation period, 1 Myear.*

The appearance time for the maximum flow and the maximum molar flow from the bentonite engineered barrier are provided in Table 6.3-6 and Table 6.3-7. These indicators illustrate the delay of several thousand years provided by the clay engineered barrier as well as the considerable attenuation of the migration of highly sorbed elements such as caesium 135 and tin 126 that it provides throughout the duration of the simulation.

CU2 waste Delay indicator "time required for appearance of maximum flow" leaving the compartments						
Radionuclides	***Source term (1 package releasing at 150 years over a 5mm²surface area)***	***Package/ Source term (Loss of leak-tightness)***	***Bentonite/ Package***	***Near-field/ Bentonite***	***Far-Field/ Near-Field***	***Far-field/ Source term***
	t_0 (yr)	*t_{I+1} (yr) - t_I (yr)*				*$t_4 - t_0$*
^{14}C	150	2.0 10^4	0	3.9 10^4	1.8 10^4	7,7 10^4
^{79}Se	150	2.0 10^4	9.5 10^4	4.4 10^5	4.2 10^4	5,9 10^5
^{126}Sn	150	2.0 10^4	9.2 10^4	7.6 10^5	1.3 10^5	1,0 10^6
^{129}I	150	2.0 10^4	0	6.6 10^3	5.5 10^3	3,2 10^4
^{135}Cs	150	2.0 10^4	1.4 10^3	9.8 10^4	Delay after 10^6 years	

Table 6.3-6 The case of a spent fuel cell, M1 site model. indicator of the time for appearance of the maximum flow from the cell and the granite

CU2 waste – site M1 Indicator of the range of the "maximum molar flow" through the interfaces						
Radionuclides	***Source term (1 Package releasing at 150 years)***	***Package/ Source term***	***Bentonite/ Package***	***Near-field/ Bentonite body***	***Far-field/ Near-field***	***Far-field/ Source term***
	ϕmax_0 (mol/yr)	*ϕmax_{I+1} (mol/yr) / max_I (mol/yr)*				*$\phi max_4/max_0$*
^{14}C	8.8 10^{-4}	7.6 10^{-4}	3.7 10^{-1}	8.8 10^{-10}	9,8 10^{-3}	2,4 10^{-15}
^{79}Se	1.2 10^{-2}	2.2 10^{-7}	3.8 10^{-1}	6.0 10^{-3}	4,0 10^{-1}	2,0 10^{-10}
^{126}Sn	1.0 10^{-1}	2.8 10^{-4}	4.7 10^{-5}	6.6 10^{-6}	2,6 10^{-1}	2,2 10^{-14}
^{129}I	4.0 10^{-1}	4.1 10^{-3}	3.7 10^{-1}	2.0 10^{-1}	7,8 10^{-1}	2,3 10^{-4}
^{135}Cs	1.5	8.6 10^{-3}	3.8 10^{-4}	Considerable spreading (flow < 10^{-16} mol/yr)		

Table 6.3-7 The case of a spent fuel cell, site model M1 indicator of the maximum flow rate from the cell and the granite.

For the C waste disposal cells, the retention capacity of the clay buffer appears in a similar manner for tin 126 and caesium 135. The tables providing the range of molar flows (see Table 6.3-13) and the appearance time for maximum flows (see Table 6.3-14) indiquate a considerable attenuation capacity of the clay engineered barrier, particularly for highly sorbed elements such as tin 126. In contrast, the non sorbed iodine 129 is entirely exited from the clay engineered barrier after one million years.

6.3.3.2 Delay and attenuation in the granite

Several parameters are involved in delaying the migration of radionuclides in the granite fractures and in attenuating flows:

- The hydraulic properties of the fractures and their connectivity,
- The retentive properties (sorption) of the fractures and of the granite rock on the edge of the fracture,
- The topographical and morpho-structural layout of the granite massifs in the far-field.

Because of this, the radionuclides can be distinguished according two large categories:

- The non sorbed elements (iodine 129, chlorine 36, etc.). The transfer time of the latter via the fractures is directly a reflection of the length of the hydraulic pathway leading to the surface. For these radionuclides, the function of "delaying and attenuating migration" is barely distinguished from the function of "preventing water circulation". Their migration is essentially dictated by the topographical and morpho-structural arrangement of the massif, which governs the gradient and pathway length, as well as by the transmissivity and the connectivity of the fractures;
- The elements that are sensitive to sorption in fracture walls, which may significantly be delayed; sorption may therefore give them the time to decay significantly in the fractures walls and this, from the beginning of the near-field if their half-life is short enough. It provokes a spreading of the signal emitted by the repository over time and therefore a reduction in the maximum flow of mass between entry and exit from the massif.

- **Delay and attenuation in the near-field granite**

The influence of the hydraulic properties of the fractures

The site models represent different and representative geological fracturing of geological configurations in the French context. Because of plugging by hydrothermal minerals, the transmissivity of fractures in site M2 is lower than that of the fractures in sites M1 and M3. In the case of generic studies, the intersections between transmissive fractures themselves are assumed to be systematically water conducting, which is pessimistic for many configurations of granite massif. However, the comparison of migration flow calculations for mobile radionuclides such as iodine 129 between the two site models M1 and M2 shows the important influence of the hydraulic properties of the fractures in the near-field.

The case of B waste is simple to illustrate on account of the fact that there are only few repository tunnels and they are constructed in blocks of granite with few fractures. The calculations are performed for the most disadvantageous path of iodine 129 between the tunnel and the limits of the model (see Figure 6.3-4).

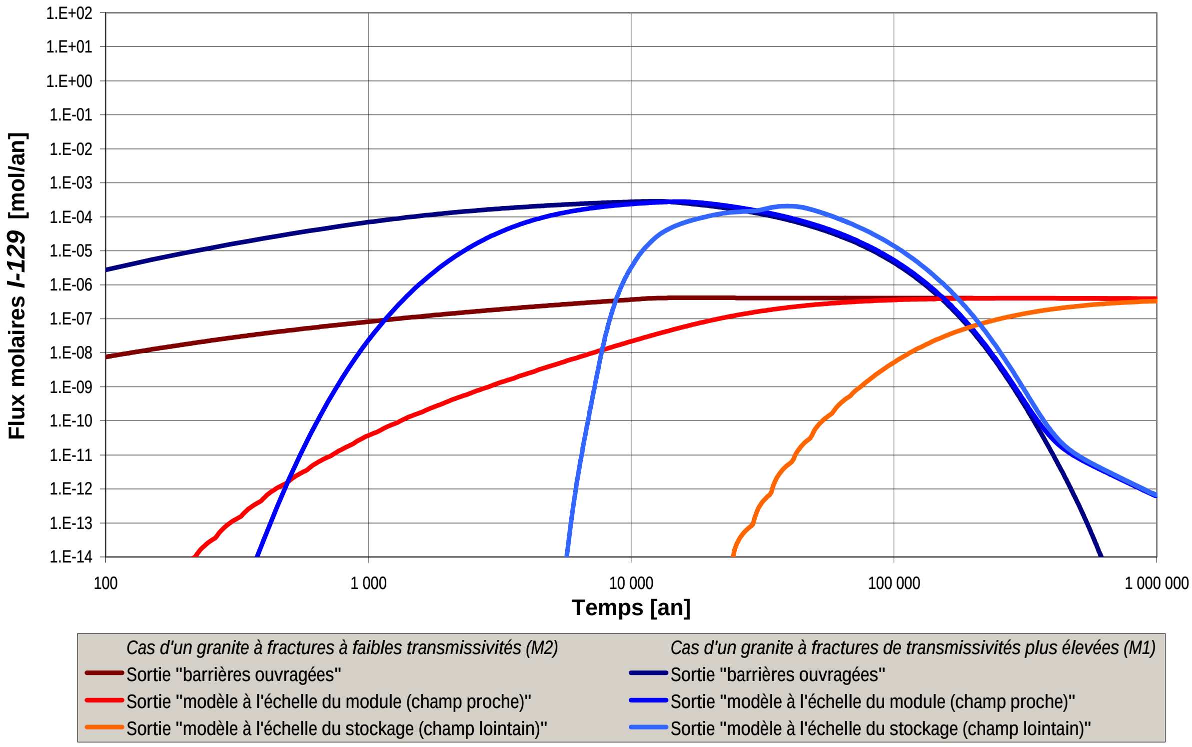


Figure 6.3-4 *Migration of iodine 129 in massifs M1 and M2 - molar flows for a B2 waste cell (DFN approach)*

In the case of a granite in which the fractures have very low transmissivity in the near field (site model M2), the maximum flows are reduced and iodine remains in the near-field granite for several hundred thousand years. In the case of a granite in which the fractures are more transmissive (site model M1), the flow rate is higher and the total inventory of iodine 129 reaches the limits of the model on the time scale of a hundred thousand years.

In terms of global balance, the place of the near-field appears to be important in the case of B wastes; the location of repository tunnels in blocks of granite with very low permeability is an important element of design. All of the following tables illustrate this fact.

Calculations dealing with the integrated mass having passed through the different interfaces of a B2 waste module show a significant attenuation in the near-field by sorption of technecium 99 and caesium 135 (see Table 6.3-4).

The molar flows leaving the near-field are very low (lower than 10^{-16} mol/yr for one tunnel, see Table 6.3-9, and Table 6.3-10) and are strongly delayed. The near field also plays an important role in time delaying of other elements (chlorine 36, selenium 79 and iodine 129) (Table 6.3-11 and Table 6.3-12).

B5.2 waste Ratio of integrated masses to have passed through interfaces to the initial mass of the inventory							
Radionuclides	*Initial inventory (1 tunnel)*	*Source term*	*Packages*	*Concrete packaging*	*EDZ*	*Near-field*	*Far-field*
	Mass m_0 (mol)	*Mass m_I (mol) / Mass m_0 (mol)*					
^{36}Cl	57.85	95%	90%	90%	90%	90%	80%
^{93}Mo	30.75	25%	3%	3%	2%	1%	0%
^{99}Tc	400.69	95%	93%	93%	7%	0%	0%
^{129}I	23.96	100%	100%	100%	100%	100%	100%
^{135}Cs	77.95	100%	99%	98%	34%	0%	0%

Table 6.3-8 *The case of a B5.2 waste cell, M1 site model. Integrated mass that has passed through the interfaces of the cell and the granite during the simulation, 1 Myr.*

B2 waste – site M1 Indicator of the range of the "maximum molar flow" leaving the compartments							
Radionuclides	*Source term (1 tunnel)*	*Packages/ Source term*	*Concrete packaging / Packages*	*EDZ/ Concrete packaging*	*Near-field/ EDZ*	*Far-field/ Near-field*	*Far-field/ Source term*
	Φmax_0 (mol/yr)	*Φmax_{I+1} (mol/yr) /max Φmax_I (mol/yr)*					*$\Phi max_5/\Phi max_0$*
^{36}Cl	$2.9\ 10^{-1}$	$5.0\ 10^{-4}$	$5.9\ 10^{-1}$	$9.3\ 10^{-1}$	$9.7\ 10^{-1}$	$6.6\ 10^{-1}$	$1.8\ 10^{-4}$
^{79}Se	$1.8\ 10^{-1}$	$5.5\ 10^{-3}$	$9.6\ 10^{-3}$	$2.0\ 10^{-1}$	$2.6\ 10^{-1}$	$5.9\ 10^{-1}$	$1.6\ 10^{-6}$
^{99}Tc	$6.3\ 10^{2}$	$4.4\ 10^{-3}$	$9.9\ 10^{-1}$	$4.6\ 10^{-4}$	$7.5\ 10^{-7}$	$8.4\ 10^{-7}$	$< 10^{-16}$
^{129}I	1.2	$2.0\ 10^{-3}$	$1.2\ 10^{-1}$	$9.8\ 10^{-1}$	$2.9\ 10^{-4}$	$7.4\ 10^{-1}$	$1.7\ 10^{-4}$
^{135}Cs	4.4	$5.1\ 10^{-3}$	$4.1\ 10^{-1}$	$1.9\ 10^{-3}$	$2.7\ 10^{-4}$	$1.8\ 10^{-4}$	$1.9\ 10^{-13}$

Table 6.3-9 *The case of a B2 waste cell, M1 site model. Indicator of the range of the molar flow leaving each of the components of a module.*

B5.2 waste – site M1 Indicator of the range of the "maximum molar flow" leaving the compartments							
Radionuclides	***Source term (1 tunnel)***	***Packages/ Source term***	***Concrete packaging /Packages***	***EDZ/ Concrete packaging***	***Near-field/ EDZ***	***Far-field/ Near-field***	***Far-field/ Source term***
	Φmax_0 (mol/yr)	*Φmax_{l+1} (mol/yr) /maxΦmax_l (mol/yr)*					*$\Phi max_5/\Phi max_0$*
^{36}Cl	14	$2.2\ 10^{-3}$	$9.2\ 10^{-1}$	$4.8\ 10^{-2}$	$8.5\ 10^{-1}$	$3.9\ 10^{-1}$	$3,3\ 10^{-5}$
^{93}Mo	5.9	$4.4\ 10^{-4}$	$9.2\ 10^{-1}$	$4.3\ 10^{-2}$	$4.3\ 10^{-1}$	$1.3\ 10^{-3}$	$1,0\ 10^{-8}$
^{99}Tc	200	$1.1\ 10^{-2}$	1.0	$3.6\ 10^{-5}$	$1.2\ 10^{-11}$	Considerable range (flow<10^{-16} mol/yr)	
^{129}I	24	$3.8\ 10^{-3}$	$2.8\ 10^{-1}$	$5.1\ 10^{-2}$	$8.6\ 10^{-1}$	$3.6\ 10^{-1}$	$1,7\ 10^{-5}$
^{135}Cs	78	$5.9\ 10^{-3}$	$3.9\ 10^{-1}$	$2.2\ 10^{-4}$	$8.1\ 10^{-8}$	Considerable range (flow<10^{-16} mol/yr)	

Table 6.3-10 The case of a B5.2 waste cell, M1 site model. Indicator of the spreading of the molar flow leaving each of the components of a module

B2 waste – site M1 Delay indicator "time required for appearance of maximum flow" leaving the compartments							
Radionuclides	***Source term (1 tunnel)***	***Packages/ Source term***	***Concrete packaging /Packages***	***EDZ/ Concrete packaging***	***Near-field/ EDZ***	***Far-field/ Near-field***	***Far-field/ Source term***
	t_0 (Yr)	*t_{l+1} (yr) - t_l (yr)*					*$t_5 - t_0$*
^{36}Cl	$1.00\ 10^{-3}$	$1.0\ 10^{4}$	0	$1.2\ 10^{4}$	$2.2\ 10^{3}$	$2.3\ 10^{4}$	$3.8\ 10^{4}$
^{79}Se	$1.00\ 10^{-3}$	$1.0\ 10^{4}$	$1.5\ 10^{2}$	$1.3\ 10^{4}$	$4.8\ 10^{4}$	$3.9\ 10^{4}$	$1.0\ 10^{5}$
^{99}Tc	$1.00\ 10^{-3}$	$1.0\ 10^{4}$	0	$1.4\ 10^{4}$	$9.9\ 10^{5}$	0	$1.0\ 10^{6}$
^{129}I	$1.00\ 10^{-3}$	$1.0\ 10^{4}$	$1.2\ 10^{4}$	$1.5\ 10^{2}$	$2.7\ 10^{3}$	$2.4\ 10^{4}$	$3.9\ 10^{4}$
^{135}Cs	$1.00\ 10^{-3}$	$1.0\ 10^{4}$	$2.0\ 10^{2}$	$1.5\ 10^{4}$	$9.8\ 10^{5}$	0	$1.0\ 10^{6}$

Table 6.3-11 The case of a B2 waste cell, M1 site model. Indicator of the time required for the appearance of the maximum flow rate of the components of a module.

B5.2 waste – site M1 Delay indicator "time required for appearance of maximum flow" leaving the compartments							
Radionuclides	***Source term (1 tunnel)***	***Packages/ Source term***	***Concrete packaging /Packages***	***EDZ/ Concrete packaging***	***Near-field/ EDZ***	***Far-field/ Near-field***	***Far-field/ Source term***
	t_0 (Yr)	*t_{l+1} (yr) - t_l (yr)*					*$t_5 - t_0$*
^{36}Cl	0	$1,0\ 10^{4}$	0	80	$4,2\ 10^{3}$	$5,1\ 10^{4}$	$6,5\ 10^{4}$
^{93}Mo	0	$1,0\ 10^{4}$	0	70	$3,0\ 10^{3}$	$2,6\ 10^{4}$	$3,9\ 10^{4}$
^{99}Tc	0	$1,0\ 10^{4}$	0	$8,7\ 10^{4}$	$9,0\ 10^{5}$	Delay after 10^{6} years	
^{129}I	0	$1,0\ 10^{4}$	0	$1,0\ 10^{2}$	$4,\ 10^{3}$	$3,2\ 10^{4}$	$4,6\ 10^{4}$
^{135}Cs	0	$1,0\ 10^{4}$	70	$2,0\ 10^{3}$	$9,9\ 10^{5}$	Delay after 10^{6} years	

Table 6.3-12 The case of a B5.2 waste cell, M1 site model. Indicator of the time required for the appearance of the maximum flow rate of the components of a module.

In of the case of C waste and spent fuel, a variation in the transmissivity of the fractures leads to modifications in radionuclide flow and migration time, as it does in the case of B waste. The multiplicity of radionuclide paths corresponding to the different cells of a module (450 cells) means that the overall total is, however, smoothed by the variability of the paths. Thus the influence of hydraulic parameters of the fractures must also be examined with the other properties of the granite, in particular their geometric organisation which determines the paths. Sensitivity to fracture transmissivity can, however, be illustrated by evaluating the flow rates at the limit of the model for transmissivity values that vary by an order of magnitude for all fractures (see Figure 6.3-5).

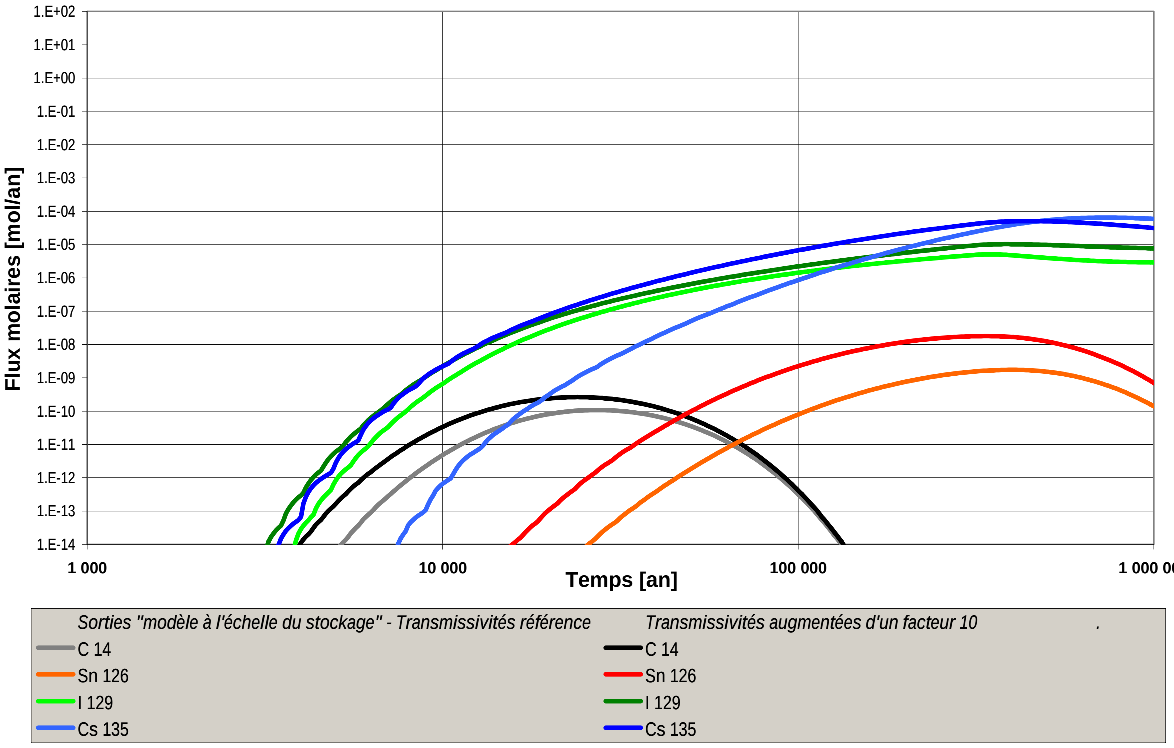


Figure 6.3-5 *Sensitivity of performances to fracture transmissivity - molar flows for a C2 waste module (massif M2 - DFN approach)*

In case of non sorbed radionuclides such as iodine 129, the highest transmissivities result in a slight overall increase in flow. In the case of sorbed radionuclides, the increase in flow speeds related to that of transmissivities also results in less intense diffusion and sorption phenomena in the fractures: For example, the molar flow of caesium is increased by several orders of magnitude at 10 000 years and 100 000 years; the maximum flow is not very different on the timescale of several hundred thousand years (see Figure 6.3-5).

The influence of retentive properties of the fractures in the case of sorbed radionuclides

Retention in fractures is the result of both the diffusion properties in the altered rock around the fractures and the sorption properties of the radionuclides.

The calculations show that molar flows are very sensitive to the sorption properties for radionuclides of granite fractures, these properties depend on the mineralogical nature of the rock. For example, the molar flows of caesium are significantly advanced in time when the retardation coefficient is reduced by a factor of 16 (see Figure 6.3-6). This underlines the importance of specific characterization of the radionuclide sorption properties for a site.

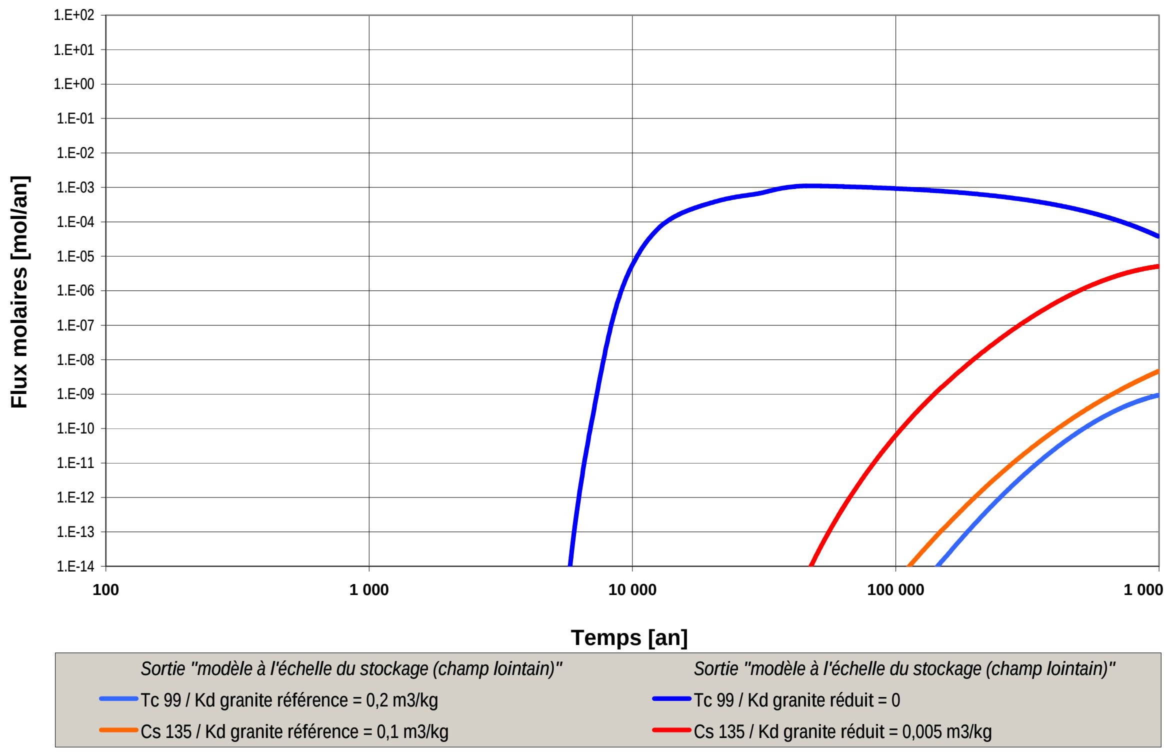


Figure 6.3-6 *Influence of sorption in fractures in the granite - molar flow in the case of a B2 waste cell (massif M1 - DFN approach)*

- **Delay and attenuation in the far field granite: the influence of topographical and morpho-structural arrangements**

In general, for the overall arrangements of the examined granite massifs, numerous radionuclides are sorbed during their migration through the fractures in the near or far field. Thus, the calculations show that actinides do not reach the limits of the model in the far field at one million years for any of the studied configuration of granite massif.

The topographical and morpho-structural arrangements of a granite massif determine pathway lengths between a deep repository and surface. The analysis of the lengths of the hydraulic pathways for the various geological site models considered confirms differences between the studied configurations. The analysis was carried out for the two most contrasting studied cases: M1 site model is representative of a "dome" arrangement and M2 site model is representative of a "depression" arrangement. It also shows that, in the case of M1 model, pathway lengths vary in the majority of them between 2 500 and 6 500 metres.

In the case of M2 site model, pathway lengths are shorter, on average, between 1 500 and 4 000 meters. These differences tend to be diminished in radionuclide transfer as the characteristics of the granite at the scale of the repository modules limits the migration of the majority of radionuclides in particular the one having a low mobility. In the case of mobile long-lived radionuclides, chlorine 36 and iodine 129 in particular, hydraulic transfer times directly determine migration times.

Figure 6.3-7 shows the radionuclides that reach the limits of the model in the case of M1 site model for C2 vitrified waste. Only radionuclides that are sorbed to a very small degree are present (note that chlorine 36, which is barely present in the glass, has not been taken into account in the calculation). In particular, caesium 135 does not reach the limits of the model on a timescale of 1 million years and remains in the massif. This is not the case for site model M2, direct translation of the shortest migration path times.

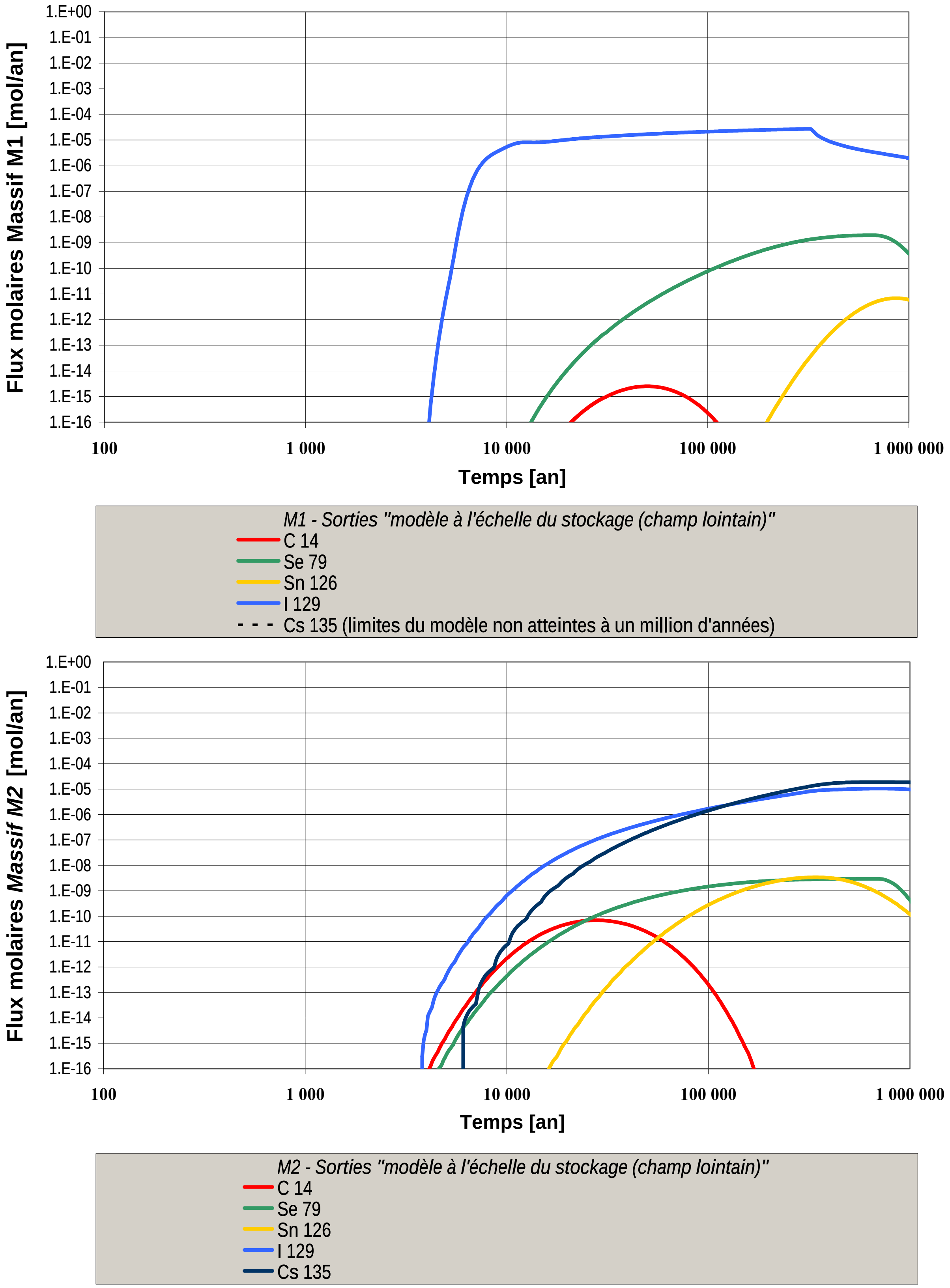


Figure 6.3-7 *Attenuation of the radionuclides flow in the case of M1 and M2 site models – molar flow for a C2 waste module (DFN approach)*

Calculations of the flattening of the molar flow and of the occurence time of maximum flow for 1 failed container at 150 years or for packages performing for up to 1000 years show a significant attenuation of the caesium in the near-field (see Table 6.3-13, Table 6.3-14, Table 6.3-15 and Table 6.3-16). The molar flows leaving the near-field are therefore very low (10^{-16} mol/yr for 10 cells) and significantly delayed for this element (after 10^6 years). Precipitation of selenium 79 significantly limits the molar inventory of this element leaving the cells.

C2 waste – site M1 **Indicator of the "maximum molar flow" range leaving the compartments**						
Radionuclides	***Source term (1 package releasing at 150 years + 19 packages releasing at 1,000 years)***	***Package/ Source term***	***Bentonite/ Package***	***Near-field/ Bentonite***	***Far-field / Near-field***	***Far-field / Source term***
	Φmax_0 (mol/yr)	*Φmax_{I+1} (mol/yr) /max Φmax_I (mol/yr)*				*$\Phi max_4/\Phi max_0$*
^{14}C	$5.2\ 10^{-5}$	$2.8\ 10^{-2}$	1.0	$8.9\ 10^{-12}$	$8{,}7\ 10^{-2}$	$2{,}2\ 10^{-14}$
^{79}Se	$3.8\ 10^{-3}$	$1.1\ 10^{-5}$	1.0	$1.2\ 10^{-3}$	$6{,}0\ 10^{-1}$	$7{,}9\ 10^{-9}$
^{126}Sn	$1.6\ 10^{-2}$	$1.4\ 10^{-2}$	$4.9\ 10^{-3}$	$3.7\ 10^{-7}$	$5{,}3\ 10^{-1}$	$1{,}3\ 10^{-11}$
^{129}I	$9.2\ 10^{-4}$	$2.8\ 10^{-2}$	1.0	$2.9\ 10^{-2}$	1,0	$8{,}0\ 10^{-4}$
^{135}Cs	$2.0\ 10^{-1}$	$3.8\ 10^{-3}$	$4.2\ 10^{-1}$	Considerable range (flow $< 10^{-16}$ mol/yr)		

Table 6.3-13 *The case of a C2 waste cell with a container failing at 150 years, M1 site model. Indicator of the range of the maximum molar flow leaving the various cell interfaces of C2 waste.*

C2 waste – site M1 **Delay indicator "time for maximum flow to appear" leaving the compartments**					
Radionuclides	***Source term (1 package releasing at 150 years + 19 packages releasing at 1,000 years)***	***Bentonite/ Package***	***Near-field/ Bentonite***	***Far-field/ Near-field***	***Far-field/ Source term***
	t_0 (an)	*t_{I+1} (yr) - t_I (yr)*			*$t_4 - t_0$*
^{14}C	1000	0.0	$5.5\ 10^4$	$1.3\ 10^4$	$6.8\ 10^4$
^{79}Se	1000	0.0	$6.6\ 10^5$	$1.5\ 10^4$	$6.8\ 10^5$
^{126}Sn	1000	$1.9\ 10^5$	$7.6\ 10^5$	$5.3\ 10^4$	$1.0\ 10^6$
^{129}I	1000	0.0	$3.2\ 10^5$	$7.5\ 10^3$	$3.2\ 10^5$
^{135}Cs	1000	$2.7\ 10^5$	$7.3\ 10^5$	Delay after 10^6 years	

Table 6.3-14 *The case of a C2 waste cell with a container failing at 150 years, M1 site model. Indicator of the range of the maximum molar flow leaving the various cell interfaces.*

C2 waste – site M1 Indicator of the range of the "maximum molar flow" leaving the compartments						
Radionuclides	***Source term (20 Package Releasing at 1,000 years)***	***Package/ Source term***	***Bentonite / Package***	***Near-field/ Bentonite***	***Far-field/ Near-field***	***Far-field/ Source term***
	Φmax_0 (mol/yr)	*Φmax_{l+1} (mol/yr) /max Φmax_l (mol/yr)*				*$\Phi max_4/\Phi max_0$*
^{14}C	$5.4\ 10^{-5}$	$2.8\ 10^{-2}$	1.0	$8.4\ 10^{-12}$	$8{,}7\ 10^{-2}$	$2{,}0\ 10^{-14}$
^{79}Se	$4.0\ 10^{-3}$	$1.1\ 10^{-5}$	1.0	$1.2\ 10^{-3}$	$6{,}0\ 10^{-1}$	$7{,}5\ 10^{-9}$
^{126}Sn	$1.6\ 10^{-2}$	$1.3\ 10^{-2}$	$4.8\ 10^{-3}$	$3.8\ 10^{-7}$	$5{,}3\ 10^{-1}$	$1{,}3\ 10^{-11}$
^{129}I	$9.6\ 10^{-4}$	$2.7\ 10^{-2}$	1.0	$2.8\ 10^{-2}$	1,0	$7{,}6\ 10^{-4}$
^{135}Cs	$2.1\ 10^{-1}$	$3.6\ 10^{-3}$	$4.1\ 10^{-1}$	Considerable range (flow < 10^{-16} mol/yr)		

Table 6.3-15 *The case of a C2 waste cell, site model M1. Indicator of the range of the molar flow leaving the various cell interfaces.*

C2 waste – site M1 Delay indicator "maximum flow appearance time" leaving the compartments					
Radionuclides	***Source term (20 Package Releasing at 1.000 years)***	***Bentonite/ Package***	***Near-field/ Bentonite***	***Far-field/ Near-field***	***Far-field/ Source term***
	t_0 (an)	*t_{I+1} (yr) - t_I (yr)*			*$t_4 - t_0$*
^{14}C	1000	0.0	$5.6\ 10^{4}$	$1.2\ 10^{4}$	$6.8\ 10^{4}$
^{79}Se	1000	0.0	$6.6\ 10^{5}$	$1.5\ 10^{4}$	$6.8\ 10^{5}$
^{126}Sn	1000	$1.8\ 10^{5}$	$7.6\ 10^{5}$	$5.3\ 10^{4}$	$1.0\ 10^{6}$
^{129}I	1000	0.0	$3.2\ 10^{5}$	$7.5\ 10^{3}$	$3.2\ 10^{5}$
^{135}Cs	1000	$2.7\ 10^{5}$	$7.3\ 10^{5}$	Delay after 10^{6} years	

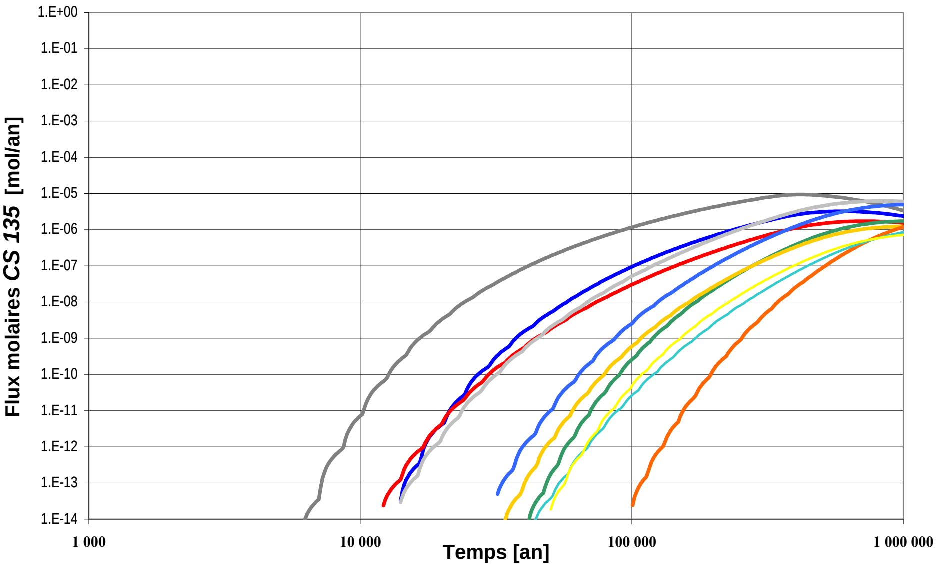
Table 6.3-16 *The case of a C2 waste cell, m1 site model. Indicator of the appearance time for the maximum flow from the cell and the granite.*

6.3.3.3 Robustness of the repository in the event of a possible characterization error

The altered evolution scenario "characterization error" illustrates the importance of a more of less detailed characterization of the fracturing. The adaptation of the architecture of a repository to the fracturing of the granite is subject to survey operations before various stages of site survey. A particularly important stage is the characterization of the fracturing carried out, *in situ*, in the repository prior to the excavation of the cells and the emplacement of the packages (see Chapter 4).

The consequence of a characterization error can also be analysed by comparing a normal evolution scenario in which this characterization is correctly carried out and a "characterization error" scenario which considers the failure to identify fractures that should have been avoided.

In the case of C waste, 10 % of the possible locations have not been taken into account in the case of a normal evolution scenario: these positions correspond to the most unfavourable hydraulic conditions in the near field. In contrast, total cell locations are taken into account in the case of the characterization error scenario. A transport calculation is performed independently for each of the paths of the module (2 packages per cell). Figure 6.3-8 shows the ten paths which produce the largest molar flow in both cases for caesium 135. The highest maxima appear to have increased by a factor of 10 in the case of a characterization error.



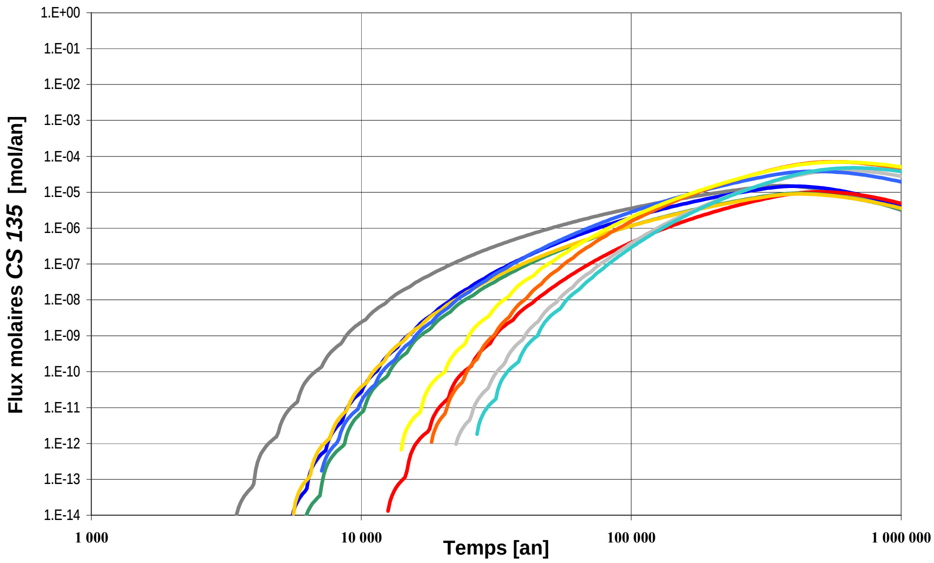


Figure 6.3-8 *Classification of the pathways leading from the cells of a C2 waste module (2 packages per cell) per their maximum molar flow. above, the ten pathways with the highest maximum molar flow rate of caesium 135 with a 10% rejection of cell positions. Below, the ten pathways with the highest maximum molar flow with no rejection beforehand (DFN approach)*

Radionuclide transfers for the entire C2 Module (450 cells) are calculated in both cases by distributing the corresponding inventory over the possible paths from the cells (10% rejection in the normal case and no rejection in the case of a characterization error). The comparison in terms of molar flow cam be found in Figure 6.3-9.

In the case of caesium 135, no rejection of 10% of cell locations corresponding to the most hydraulically pessimistic situations results in a maximum molar flow increased by a factor of 5 (after 400 000 years). The molar flow is increased by two orders of magnitude at 10 000 years and by about one order of magnitude at 100 000 years. This highlights the importance of the characterization and its impact on the overall performance of the system. In the case of one particular site, the influence of a selection in the implementation of the cell should be specified in more detail.

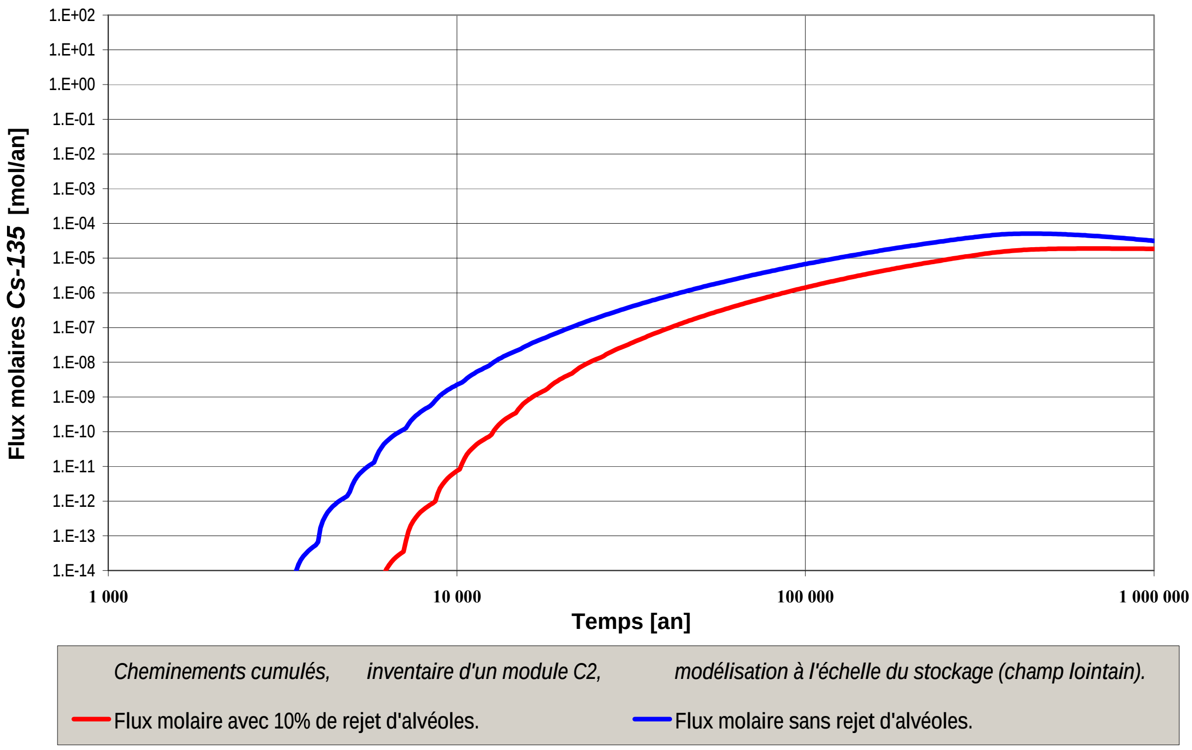


Figure 6.3-9 *Comparison of the molar flows with and without 10% rejection of the cells - molar flows of caesium 135 of a C2 waste module, massif M2 (DFN approach)*

In Case of B waste and B5.2 waste in particular, the impact of poor local characterization is relatively more sensitive in terms of the number of radionuclides released than in the case of C waste. This is due to the higher number of packages potentially affected. A local characterization error of minor fracturing concerned by the siting of a disposal tunnel would, however, not undermine the overall performance of a B5.2 waste repository. In fact a characterization error would only be sensitive in the event of a very serious characterization error involving the interconnecting drift in the disposal tunnels by a fracture of large dimensions (several hectometres) with significant transmissivity which is not realistic.

In total, even if the characterization of the fracturing and the adaptation of the repository in accordance with its hydraulic and transport characteristics appeared to be significant, the repository is, overall, not sensitive to an occasional error. In fact, because of module seperation (modules, cells), a characterization error would only affect a moderate part of the inventory from packages disposed of.

6.3.4 Conclusion of the calculations

In a generic context, the calculations performed cannot claim to be conclusive either as regards fulfilment of safety objectives or from the point of view of the performance of the safety functions provided by each component. However, performing such calculations provides a wealth of important information both from the methodological point of view and from that of major determining factors affecting the safety of the repository.

From the point of view of the methods, the use of calculation tools that are both "conventional" (such as those conducted in a homogeneous media) and more granite specific (calculations in a fractured media) highlight the complementarity of the two approaches. The first tools are used to determine in a simple way the influence of the main macroscopic parameters: hydraulic gradient, permeability of the rock and the engineered structures, Péclet number, etc. The calculations in fractured media provide a wealth of information allowing to link near-field performances to the structuring and distribution of minor fracturing or to establish a link between intermediate fracturing and permeability on a large scale. Such information should be used, in the hypothetical scenario of a site survey, to gradually determine pertinent criteria for the location and architecture of the repository as safety analyses are performed.

The survey of fracturing and correct characterization of its hydraulic properties, are important to control flows in and around the repository. On the scale of the repository modules, they result in locations that are free from intermediate fracturing where water conductivity is too high. At the scale of the cells, the adaptation of the location of the cells for minor fracturing provides an additional opportunity to limit flows and favour long radionuclide pathways. However, on account of their important fractioning of module, repository modules for vitrified waste and spent fuel are barely sensitive to occasional characterization error. B waste disposal tunnels, which are greater in size, may be more sensitive to the quality of the granite rock in which they are installed; a characterization error could affect repository performance in the case of failure to detect a fracture of large dimension and significant transmissivity. However, this seems unlikely in the framework of a survey proposed for characterization of a granite massif and given the limited number of tunnels.

From the point of view of the results, the calculations for a generic site show the good complementary between the properties of the engineered structures and those of the geological medium complementary. On the basis of current knowledge, the performance of the backfill is shown to be significant from the point of view of controlling flows inside the repository inasmuch as the site itself provides controlled permeability and low gradients. Engineered barriers provide a diffusive regime in the cells and immobilize low soluble radionuclides. The concrete B waste containers participate both in the limitation of water flows and the sorption of radionuclides. The C waste over-pack delays the release of radionuclides in the cells. Copper spent fuel containers provide durable confinement of radioactivity and flexibility at this generic stage as regards the siting of the repository. Retention in granite fractures significantly limits the flow of radionuclides that are sorbed and, in favourable configurations, prevents their transfer to the limits of the model.

7 General Conclusion

Safety analyses have been performed on the granite using an approach and methods that are consistent with those that would be used to evaluate the safety of a repository in a specific formation. In this respect, the Dossier 2005 is a methodological exercise in the application of these methods. This exercise shows that the qualitative (based on the examination and use of "FEP's" bases) and quantitative (based on the complementarities of approaches of calculation in homogeneous media and in fractured media) analytical methods have been mastered and could be implemented in respect of a given site. Clearly, the generic character of the analysis does not provide conclusions on repository impact or unequivocally determine the significance of the different repository components from the point of view of safety functions. A certain number of observations, however, can be drawn from the analyses presented in this tome.

The repository architecture proposed for granite media is based upon a strategy guided by the need to fulfil clearly identified functions (Chapter 3). Thus, the Dossier 2005 provides the basis for an approach that could lead to the design of a repository. Consideration of a specific site could lead to specifying or reviewing certain provisions envisaged in a generic context: the need for a container that remains leak-tight over a very long period of time in the case of spent fuel, the arrangement and number of cells, the respective roles of backfill and seals, etc. The evaluation of performances carried out in Chapter 6 shows that, even if the results cannot be compared to an impact standard, the components do considerably limit the flow of radionuclides from the repository. The results do not reveal any clear weakness in the repository system.

One element the importance of which has been highlighted throughout this volume is the adaptation of the repository for the fracturing of the massif. A strategy has been defined for positioning of the engineered structures in accordance with the characteristics of the fractures. In the event of installation in a real site, this strategy would be based on quantitative criteria that would be defined gradually in accordance with the result of safety evaluations. The evaluations carried out in Chapter 6 demonstrate the value of calculation methods in fractured media which makes it possible to understand the sensitivity of the positioning of a cell or a module in accordance with its position in respect of fractures. These calculation tools could provide the basis for an iterative approach between the evaluation of repository performances and site survey operations. These are also based on methods that are known and applicable in the context of French geology. The purpose of this tome was not to describe these methods but they are presented [x].

At this stage, there are still many unknown factors and no claim is made to the effect that these have all been taken into account. Some results presented in the dossier 2005 are generic in character and may not apply to specific massifs. The typological analysis of French granite, however, shows that in spite of the few massifs to have undergone real surveying, the characteristics taken into account for the evaluations are realistic. The variability of the massifs is sufficiently well identified to allow homogeneous treatment of the FEPs (see Chapter 5) once certain specific situations with clearly unfavourable site configurations have been ruled out. In this respect, French granite massifs are suitable for safety analysis and do not have characteristics that would make this analysis particularly difficult.

Qualitative and quantitative analyses cannot be conclusive at this stage. However, they do emphasize that methods to control the safety of a repository in a granite medium do exist, that designs can be specified which meet safety requirements and that the design performance evaluations carried out at this stage do not reveal any redhibitory elements.

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