Report Series

Dossier 2005 Argile

Tome

Safety evaluation of a geological repository

December 2005

ANDRA
Agence nationale
pour la gestion des déchets radioactifs

The present English version is a translation of the original "*Dossier 2005 Argile*" 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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1.1 Introduction

The Law of 30 December 1991 [1] confers to Andra the mission of assessing the feasibility of a repository of high-level and long-lived (HLLL) waste in a deep geological formation.

The present volume describes the result of all the studies which were conducted within Andra to assess this feasibility from the viewpoint of safety. As such, it complements the volumes entitled « architectures and management of the geological repository »and « phenomenological evolution of the geological repository » in that it uses the results obtained from studies given in these volumes in a safety-oriented perspective. Therefore, it reviews generally in an abridged fashion with respect to the developments contained in the two other volumes the main knowledge acquired from the engineering studies and the research work. It describes also part of the work not dealt with in the other volumes (particularly, the results obtained from the performance assessments).

Since safety is not an independent topic, but is closely related to the other work conducted on geological repository, it seems indispensable to recall the results already presented in other documents, explaining how the search for the best level of safety led to the retained options and to certain research rather than others. The two other volumes are based on a more subject oriented approach of their respective fields. Consequently, the two other volumes are not a prerequisite to read this volume. However, the reader interested in a more detailed view of the result of the studies or in more complete justifications about some points should refer to the other volumes or to documents which offer much more detail. In fact, it was not considered possible or advisable to try to account for all the work in the present volume because it would have compromised readability.

Chapter 1 of this volume presents the basic approach adopted to assess feasibility from a safety viewpoint within the context of a clay site based on the characteristics of the Meuse / Haute-Marne (MHM) laboratory site. It is an introduction to the rest of the analysis, which is intended to establish the framework for safety assessments, present the adopted approach and globally serve as a reader’s guide for the remainder of the present volume.

1.2 Context and global objectives

1.2.1 The fundamentals of the safety approach

The feasibility study amounts to assessing the conditions under which a repository could be built, operated, observed and closed, while receiving all of the HLLL wastes existing in today's inventory or foreseeable in the short or medium term, and without the safety of workers, the public or the protection of the environment being jeopardized at any moment. The Basic Safety Rule III.2.f [2] formulates this protection objective as follows : « the protection of people and the environment in the short and long term is the basic objective assigned to a waste repository in a deep geological formation ».

Another requirement is stated by the RFS III.2.f. in these terms : the long-term safety of the repository must not « depend on an institutional control on which we cannot absolutely rely beyond a limited period ». Beyond a site's monitoring phase, the site must be able to evolve and remain safe without any human intervention being necessary and without being forced to maintain a « record » of the repository and monitoring. Obviously, such a record and such a monitoring may still be maintained as long as it seems possible.

The notion of feasibility, therefore, refers to an acquired proven belief regarding a specific site. The belief is that :

- There are technologies which allow carrying out all the repository's life phases ;
- These technologies are easy to implement (in particular, the implementation does not necessitate any prohibitive development needs or costs) ;
- These technologies allow the repository to be completed, and then keep it open or closed and let it evolve under safe conditions ;
- The short- and long-term safety of the repository concepts can be assessed with sufficient confidence.

Feasibility, therefore, appeals to the notion of confidence in the assessment over long time scales (up to several hundreds of thousands of years). The notion of « confidence » was particularly developed within the framework of the reflective thinking made by the Nuclear Energy Agency (OECD/NEA) [3, 4]. This confidence is based on the quality of the technical argumentation in all its aspects : confidence not only in the repository concept, in the data, the models, the analyses, but also in the approach itself, which organises all these technical data into a consistent set, that is, the « safety case » or « safety dossier ».

Confidence does not arise simply from the analyses presented by the organisation responsible for the development of the repository It also depends on the context in which these analyses were made. Resorting to predefined standards, clear safety objectives and shared methodologies up to a certain point internationally harmonised contributes in making a more robust dossier. Credibility of the presented analyses must also be given great importance ; credibility goes beyond the field of technical debate, and refers back to the dossier's establishment terms and conditions. With the organisation setup, the procedures which ensure that the data were acquired and processed under the right conditions to guarantee transparent and reliable data processing constitute an integral part of the « safety case ». As an illustration, the basic principles which guided the approach are mentioned in subsection 1.3 ; the acceptability criteria (objectives with respect to the impact of the repository) set by Andra and the references to the national and international texts which they are based on are mentioned in subsection 1.4. The organisation set up to configure and control the dossier's production is mentioned in subsection 1.5.

1.2.2 The repository's iterative development process and the place of uncertainties

A knowledge of the phenomena governing on the evolution of the waste packages, the repository structures, the geological medium and its environment, as well as the control[1] of these evolutions by design provisions adapted to all the repository's life phases, allow establishing on sound scientific foundations a judgement on the feasibility or non-feasibility of the repository with respect to its objectives, particularly safety.

The acquisition of knowledge is a progressive process which goes beyond the feasibility phase. Making a judgement about the repository's feasibility entails having previously acquired a large corpus of scientific knowledge and developed sufficiently detailed architectures. Nonetheless, neither knowledge acquisition nor the design activity stop at the feasibility phase. The progressive integration of feedback from the studies and then, if necessary, the sinking, operating and monitoring phases will gradually allow learning the physical characteristics and the behaviour in a finer and finer way during the life of a possible repository in order to optimise the implemented systems over time.

At all the phases of the knowledge acquisition programme, the limits of this programme lead to uncertainties which are both qualitative (incomplete control of the development of a physical phenomenon) and quantitative (uncertain assessment of quantified characteristics of the said phenomenon). The feasibility phase is obviously not an exception. These uncertainties may lead under the « worst case » conditions to an unexpected evolution of the repository different from that anticipated by the designer (no control of the repository's evolution over time). No decision should be made on the feasibility of a radioactive waste repository before being completely aware of the residual uncertainties and before having found the means as upstream as possible to prevent an undesirable evolution of the repository or reduce the consequences to an acceptable level. In view of this, it is not possible to define ex nihilo a corpus of knowledge which would be necessary and sufficient to decide on the repository's feasibility. Therefore, a justified appraisal of this matter can only be formulated after the acquired data have been shown and the residual uncertainties controlled.

This task is rendered more delicate for the repository's post-monitoring phase by the long duration involved. The assessment is carried out for a million years. All arrangements must be made to create the conditions which allow such an assessment (choice of a very stable geological medium, implementation of materials with a foreseeable behaviour). However, the ability to observe over long periods is limited to the use of natural or archaeological analogies. The uncertainties linked to long duration are a specificity of the dossier. This is why a special section is devoted to the analysis and the processing of the uncertainties.

Feasibility requires, on the one hand, that the designer has a good knowledge of the behaviour of all the repository's constituent parts (inventory, geological medium, structured elements, etc.) and, on the other hand, that the residual uncertainties are as much as possible identified, and that their potential effects are relatively well controlled.

One of the ways to ensure the control of uncertainties is to integrate safety already in the phases the farthest upstream from the design phase in order to direct the choices toward solutions offering the greatest robustness, that is, those less sensitive to the effect of external factors or the lack of knowledge. An upstream integration of safety in the design allows also taking into account the other requirements of a project (cost, constructibility, etc.) without them conflicting with the repository's global safety.

[1] The term « control » here refers to the fact of remaining wherever possible under conditions allowing an evolution of the repository's components that cannot be unfavourable to safety. This entails both adequately sizing the structured components and preserving the favourable characteristics of the natural medium (see chapter 3).

Research and design work is, therefore, by nature an interactive activity between the engineers in charge of engineering, the research programmes, and those in charge of safety assessments. The organisation set up within Andra and, in particular, the coordination authority between different units working together on the project, is a guarantee of this constant interaction. The repository architectures proposed within the framework of the Dossier 2005 are the result of these exchanges and take particularly into account what was learned from the previous safety assessments (especially the Dossier 2001 [5]). The 2002-2005 research programme was also supported by these same results. Although the research and design work of the deep repository should extend beyond 2005, the technical issues focused on in the present dossier would serve as guidelines for the work to be done.

1.2.3 The dossier's objectives

The objective of the present volume is to describe the result of the repository's feasibility studies under safety assessment. The issue is examined from three viewpoints, which are successively treated in the following pages :

- Focus on how safety was taken into account as far upstream as possible in the design choices in order to reach concrete technical solutions and to define intrinsically robust architectures adapted to the geological context ;
- Assess the repository's feasibility from the viewpoint of safety by defining the criteria allowing the repository's level of safety to be assessed over all the time phases, and make sure that the proposed technical solutions satisfy these criteria ;
- Ensure that the uncertainties in the knowledge and design of the repository are identified and sufficiently controlled in order not to weaken the assessment.

1.3 Basic principles used as guidelines for the safety approach

Safety assessment is based on concepts which are described to a level of details adapted to the assessment's needs.

These concepts must allow working under operating conditions which are completely safe to the public and to the workers, and then ensuring a good control of the long-term evolution of the repository. In particular, their safety in the post-closure phase must be guaranteed without requiring any particular intervention. It is qualified as « passive safety ». The concepts are defined with respect to two principles.

- **Robustness**

The notion of robustness means that the characteristics of the elements comprising the repository must be such that they can guarantee maintaining their functionalities against reasonably imaginable disturbances despite residual uncertainties.

Generally, the concepts retain the solutions which are as robust as possible against external disturbances and uncertainties. Safety analysis allows looking globally at the robustness of the proposed concepts.

In its reflective thinking, the OECD/NEA envisions two different types of robustness [3] :

- « Engineered robustness » refers to the intentional design provisions, which allow protecting oneself against known phenomena, uncertainties or guarding against the consequences of undetermined events.
- « Intrinsic robustness » refers to the site or design choice process and consists of avoiding harmful phenomena and sources of uncertainty by means of simple provisions for which good feedback is available and whose evolution is governed by well-known processes. For instance, the choice of a site whose stability can be certified based on its geological history is an example of an « intrinsically robust » choice.

Intrinsic robustness introduces the notion of demonstrability, which is another principle retained in the safety approach of the Dossier 2005.

- **Demonstrability**

In operation as well as in the long term, the repository concepts are selected so that their safeness can be checked as easily as possible and without calling for complex demonstrations subject to caution. It should be noted that demonstrability is a relative notion and that the simplicity of a check is not a goal in itself. The Dossier 2005 strives to make the best possible usage of multiple argumentation lines (safety assessment by calculation, by qualitative reasoning, by calling on analogies, by scientific experimentation or by technological demonstrators).

1.4 Objectives versus repository impact

1.4.1 Impacts considered

The objectives which must orient the design are supplied by the RFS III.2.f. At the most general level, the following two principles are involved :

- protection of people and the environment against possible aggressions related to radioactive waste ;
- limitation of an eventual radiological impact level as low as reasonably possible.

The protection of people and the environment is understood, above all, as a protection against the specific risk due to radioactive waste, that is, radioactivity and its induced effects. The protection against other aggressions due to wastes and, in particular, chemical risk, is not negligible, but does not represent a specific aspect of the wastes considered. As for chemical risk, because the problems are of the same kind (prevent and delay the transfer of chemical elements to the environment), the solutions to the problem raised by the radioactive elements cover, as a general rule, a great part of the difficulties raised by chemical toxics. At the feasibility stage, the chemical impact of the repository is studied in general and concentrates for the quantitative assessments on a few selected toxics in order to cover those which are potentially the most penalising.

Other aggressions to the environment than the marking or contamination by toxic, radioactive or chemical substances are a priori imaginable. In particular, the degradation of natural resources other than by contamination (access prohibited to a mineral deposit underlying the repository, heating of underground water) could be a problem ; this problem can be avoided as requested by the RFS III.2.f. by eliminating a site with such an exceptional resource.

Finally, the building and operating of a repository, has an impact on the environment like any industrial activity (impact on the landscape, noise, etc.). These problems are not in any way specific to the repository. They do not constitute a particular topic for a study in a feasibility phase since the ways to reduce this type of impact are known and frequently implemented. They would be taken into account in the development phases of a possible repository installation in case such a decision should be made.

The Dossier 2005 concentrates, first of all, on the radiological risk, without overlooking the other potential impacts of a radiological waste repository.

1.4.2 Impact indicators

Right from the inception of the analysis, it is of utmost importance to set the indicators, as well as the associated objectives and criteria, which will allow judging the impact of the repository. The existence of such criteria represents an objective appraisal element on the safety level reached. Nonetheless, these criteria must not conceal the more qualitative appraisal which should be made on the dossier. Besides the numerical demonstration that quantified objectives have been achieved, it is just as important that the report be based on a clear and systematic argumentation, on recognised international references and that its contents be subjected to a sufficient number of independent reviews. These aspects are examined in subsection 1.5 and the subsequent subsections.

1.4.2.1 Impact indicators during operation

The radiological impact is measured in a conventional way in agreement with the International Commission Radiological Protection (ICRP) publication 81 [6] and the RFS III.2.f, by calculating the individual exposure dose applied to the critical group. This critical group is defined by the ICRP as a representative group of individuals liable to receive the highest dose coming from a source or a group of sources. The definition of such a critical group for the study of the impact of an installation in operation today or to be in the near future is based on the study of the lifestyles of the populations living close to the site – this group is called then the « reference group ».

For the operating situations, the requirements are those of Andra for all its installations :

- The maximum allowable dose for a critical group which represents the public is 0.25 mSv/year, consistent with the objective imposed by the RFS III.2.f. for the long term (see below). This requirement is more severe than the regulatory limit (1 mSv/year) and fits into the framework of a prudent approach ;
- For workers not exposed to radiation (workers taking part in the building of the repository's drifts, for example), the requirement is set in consistency with the public at 0.25 mSv/year. In some situations, this limit can be very restrictive ; if necessary, for this kind of personnel informed about the risks and monitored, this limit can be set to the regulatory limit of 1 mSv/year within the framework of the installations design studies ;
- For workers « exposed to radiation », that is, working in the repository's nuclear zones, the requirement is set at 5 mSv/year, that is, a quarter of the regulatory value.

These constraints are understood to be the levels not to be exceeded, the objective being to reduce in any case the exposures as much as reasonably possible. However, a complete radiological protection optimisation approach is not presented in this dossier because it is premature at this stage. In addition, the constraints apply to a normal situation ; for the accident studies, for the public as well as the workers, the impact is judged acceptable for each specific case according to the likelihood of the situation considered. An impact is generally judged acceptable up to levels of a few millisieverts. As much as necessary in accidental situations and provided these situations seem to be very unlikely, an impact can refer to the threshold of 10 mSv/year, below which the ICRP considers that an intervention[2] for the populations is not justified. In any case, the appraisal of the acceptability of the consequences remains qualitative ; it is argued for each type of accident.

1.4.2.2 Long-term impact indicators

For the long term, the main safety indicator remains the exposure dose at the outlet within the context of a predefined biosphere and a predefined critical group. A dose of 0.25 mSv/year at most in a normal situation set by the RFS III.2.f is retained by Andra. The choice of the same constraint of 0.25 mSv/year for the repository's operating and closure situations refers more broadly to the notion of equity between the generations : we do not accept for future generations detriments which would not be accepted for present-day populations. For situations considered as altered, the calculated impact is assessed according to the likelihood of the situation, the chronic or timely character of the exposures, the degree of pessimism of the calculation assumptions.

The calculation of this dose for periods close to our own is a well-known exercise in the field of impact studies. The fact that it is frequently carried out must not hide the uncertainties weighing down on both the calculation parameters (the dose factors) and the dose – effect relationship.

This type of assessment presents specific problems when it is carried out over a million years. At this scale, it is illusory to pretend to have an assessment of the lifestyles of the beings that will inhabit the studied sector. The environment conditions and, in particular, the climatic conditions will also be subjected to major variations ; their main characteristics are predictable, but it will be difficult to accurately consider them in a local context. The definition of the critical group is necessarily partly conventional since the living habits of populations are hard to predict beyond a few generations. This point is further developed in subsection 5.3.2.6.

More generally, the models used for the impact calculation do not pretend to have a predictive character with respect to the transfer times of the radionuclides to the biosphere. They are intended only to provide a view of the impact as large as possible.

For all these reasons, the long-term calculated dose is indeed an indicator of the impact and not a prediction of the latter.

[2] Term by which the ICRP designates the activities aimed at reducing exposure after an accident by eliminating the sources, by changing the transfer channels or by reducing the number of exposed persons.

Other indicators can be proposed which show more clearly the repository's intrinsic performances without requiring any assumptions on the surface environment and the biosphere. In particular, radionuclide concentration flows assessed at relevant emplacements with respect to the safety analysis of the repository (typically at the host formation outlet) allow refining the judgement on safety and overcoming some of the uncertainties. They allow comparing different situations or different design provisions in order to see which one is the most favourable with respect to the limitation of the radionuclide transfers, but they cannot be compared to thresholds.

Finally, the repository's incidence on the environment can be likened in an initial analysis to the impact on man. It would be possible to study the « marking » of the environment by radionuclides using the biosphere model to assess the radionuclide concentrations induced by the repository in the post-closure phase in the environment ; but such indicators might be compared to thresholds except for a few radionuclides present in the natural state for which the contents generally encountered in the environment can be used as a reference. Therefore, we can consider as is generally the case for this type of assessment that the impact on the environment be treated through the impact on man. It should be noted that in any case the means to be implemented to preserve man and to preserve the environment from aggression due to the radionuclides are the same (confinement of the radioactivity, limitation of the water flows reaching the biosphere) and, therefore, the choice of indicators specific to either of the forms of impact does not lead to any difference in the repository's design.

In the case of the impact of chemical toxics, the retained methodology consists of calculating the concentrations of toxic substances at the outlet and/or the doses liable to be received by the reference group and comparing them according to what seems most relevant :

- in priority, to the reference toxicological values according to the methodology recommended for the impact studies of the classified installations [7, 8] when these values are available. These values allow characterising from the ingested or inhaled dose the impact of chronic exposure – over a period of 70 years – to a chemical toxic. They allow calculating the excess of individual risk (E.R.I.) for the toxics with a cancer-producing effect and the hazard factor (QD) for the toxics with a threshold effect. A value judged acceptable in a normal situation is 10^{-5} for an ERI and 1 for the QD ;
- otherwise, when the case arises, to the maximum concentrations in the environment set by the regulations on the quality of waters or to reference values recognised internationally, such as those defined by the World Health Organisation. However, this approach was not used in Dossier 2005.

1.5 Control of the dossier's production

It was pointed out that the confidence in the safety analyses was not based simply on the intrinsic quality of the supplied data. A good control of the use of these data, the methods employed, and the existence of independent reviews of the results offer major guarantees for any person who has to analysis or use the dossier's data.

Andra was granted an organisation and its own procedures to ensure the quality of the Dossier 2005, with quality being defined according to the ISO 9001 standard [9] as « the ability to satisfy the requirements ». These requirements can be those of the dossier requesters (the authorities), those imposed by regulations or the state of the art, or those selected by Andra to be applied by it (for example, the ISO 9001 standard's requirements themselves). In a wider sense, these requirements include also the requests by some other recipients : Agency evaluators, external re-readers.

These provisions are a true element of the control of the repository's safety in that they refer to the specific choices to ensure not only that the safety analyses have indeed taken into account all the relevant input data, but also that these data and analysis results are managed within a system which allows the traceability of the choices and the results. The organisation is also a means to guarantee that the various activities contributing to the feasibility assessment dossier (engineering, research, safety assessment) are well integrated.

1.5.1 The Andra organisation

1.5.1.1 Hierarchical organisation

The Agency's organisation is formalised in a manual. It has progressively evolved from 1991 to 2005, to benefit from the feedback from the studies. The Agency's structure itself has thus been adapted iteratively to dossier production.

One of the main axes of these evolutions was the reinforcement of the integration of the teams : closer links between the engineering and safety teams, the creation of a « scientific integration » department responsible for ensuring the synthesis of knowledge for its usage in the safety analyses. In parallel, the teams structuralisation evolved from an organisation initially based on the various academic disciplines required to study the repository (geology, geomechanics, geochemistry, etc.) to a more transversal approach centred on the major phenomena controling the repository's evolution (behaviour of the materials, transfers, etc.).

Particular care was devoted to the quality of the interfaces between units. In addition to the coordination at the Agency's steering committee, dialoguing was encouraged at all the organisation's levels and, in particular, at the most critical interfaces with respect to the project : between the scientific management and the Meuse / Haute-Marne laboratory teams, between the safety engineers and the computer engineers, between the safety studies and scientific integration.

1.5.1.2 Quality organisation

Andra has been certified by the Veritas International Bureau of Quality (BVQI) for the ISO 9001 and ISO 14001 standards [10] since 2001. The environment – quality management system imposes the structure for all the Agency's activities therefore in particular for the activities related to the repository in a geological formation.

- **Organisation by processes**

The hierarchical structure is crossed with an organisation by processes according to the principles defined in the ISO 9001 standard. The processes are a regrouping of activities within the Agency which contribute to the same finality and are oriented toward a customer's satisfaction. The definition of a process allows transversally looking at the units' activities and defining the actions of improvement related to the relevance, effectiveness and efficiency of the process with respect to its objectives. The performance of the processes is reported through performance indicators.

The processes are assessed in one or two annual reviews during which the results obtained are examined. They are linked to the notion of « continuous progress », which is essential in the quality field. A progress action does not necessarily indicate an insufficiency in the process, but rather an opportunity to improve its operation.

This organisation allowed inciting engineers in charge of the studies to identify possible ways of improvement. They involved especially the management of the project's configuration and the control of the scientific data (see subsection 1.5.3, data control),.

- **Andra's procedural system**

Another aspect of the Andra quality system is the procedural system. Although the latest version of the ISO 9001 standard is less centred on requirements in terms of written procedures than earlier versions, Andra has maintained a structured system of procedures, operating modes and guides, which allow harmonising working methods and to manage risks of a lack of quality in the produced documents.

Some procedures are general procedures applied by all in Andra : a general document management procedure, procedures related to project management (on the establishment of management plans, controlling reviews, etc.). Other procedures could be specifically applied to the clay HLLL waste project and are aimed at either the control of the interfaces between units, or the control of the production of data, or the document formats.

1.5.2 Project management

The « HLLL argile » project management responsible for the repository's feasibility study in the Callovo-Oxfordian formation was inspired by the most currently recommended management principles in this field.

From an organisational viewpoint, the project was globally structured by a R&D plan (PDD), respectively, of level 1 for the project in general and of level 2 for the more detailed topics : design and operating safety, geology, behaviour of the materials, long-term safety, etc., and a project management plan (PM) indicating the objectives, the breakdown of the project into major topics and the organisation set up within the teams to respond to these objectives. The project is then carried out and timed by the production of objective sheets (FO) issued by the team of project engineers based on the PDDs and intended for the scientific, safety or engineering engineers in charge of the studies. These latter engineers respond with a technical proposal sheet (FPT) which describes the tasks to be done, the time schedules, the dedicated budget. This FO/FPT system has reinforced the rigour necessary to carry out the project.

A HLLL project steering committee meeting generally held once a month brings together all the top managerial personnel (unit managements, project manager, engineers in charge of studies, management controllers, quality engineer, etc.) in order to make sure that actions are progressing correctly, to identify problems and to propose corrective actions.

For data acquisition, the Agency wanted to formalise, in addition, a scientific programme for the period 2002-2005, identifying the research priorities based on information acquired from the previous dossier (Dossier 2001). This programme allowed improved dialoguing internally as well as with the evaluators by clarifying the Agency's research priorities.

The project's main decision-making milestones were also worked out by a formal review organised according to the project management's principles : organising a review group responsible for examining all the documents and raising questions prior to the meeting, providing information on the questions by a group made up of engineers who contributed to the writing of the documents, in-session discussion and elaborating a decision proposal submitted to the top managerial personnel. Such reviews were thus organised for the choice of the architecture for the Dossier 2005, for the options related to the observation and monitoring of the repository, for the choice of scenarios to be processed, for the choice of safety calculation models and parameters.

The construction of the underground laboratory, surface area investigations, follow-up of the sinking of the structures and the experimentations were also organised in a project. Andra is responsible as contractor / owner for the construction and scientific follow-up of the shaft sinking operations and as scientific architect for the experimental activities in the drifts and bore-holes. A project management plan details the organisation principles. The different kinds of work are managed by a specific documentary structure, which allows notably dialoguing with the scientific management on experiments from the most general specification of the objectives to a detailed definition of their accomplishment.

1.5.3 Data control

1.5.3.1 Generic elements

Generally, the HLLL waste project studies are based on scientific and technical data acquired by Andra itself or through scientific partnerships or subcontracting contracts. In the context of an iterative design approach, the studies can also be based on the results from previous studies. Other data of a non-technical character (for example, analysis of the regulations applicable to an underground medium or in a basic nuclear installation) can also be used.

This vast data set requires rigorous management. The documents, whether produced by the Agency or ordered by it, are submitted for formal approval by a person in charge before they can be used in the project. In the actual writing of the documents, the principle of a « writer / checker » team is applied.

These principles, which are generically implemented within the framework of the quality system, ensure a good traceability of the data and guarantee that they have been systematically checked. Additional arrangements were set up to strengthen the quality of the data used and ensure that they will not be used in contexts for which they are not valid.

In the case of repository architectures, the management of valid data is based on the notion of configuration management, that is, the follow-up over time of an up-to-date state of all the repository system's characteristics and the control of evolutions. Particular attention is paid to the traceability and the consistency of the configuration evolutions.

The control of the scientific data used for the models and the calculations is based on a formalised process set up under the phenomenological analysis of repository situations (PARS), which strictly forces each space and time scale to be taken into account in the evolution of the repository and each type of interaction (thermal, hydraulic, mechanical, chemical and radiological). In parallel, an overall consolidation can be reached by writing reference documents on precise topics (materials, source terms, site geology, inventory model).

The typology of the models and parameters proposed by the scientific engineers was set early in the development of the Dossier 2005 by classifying their opinions with respect to the uncertainties (depending on whether « phenomenological », « conservative », « pessimistic » or other data are involved, see definitions in subsection 5.2.2). This typology allows the safety engineers to use the data while recognising their degree of pessimism versus the assessment of the impact.

1.5.3.2 Control of the suppliers and interactions with the scientific partners

The work contributing to the production of the Dossier 2005 is not simply derived from the actual work done by Andra's engineers. In its programme agency role, Andra ensures the coordination of many partnerships on varied research topics. In addition, it relies on subcontracting contracts for dedicated services (particularly in engineering).

The partnerships are defined in agreement with the Agency's scientific policy according to three principles : the scientific relevance in the field considered, the partition of common interest topics and the creation of multi-year programmes. The organisation is adapted to the entities concerned : with the CNRS it takes the form of research consortiums (FORPRO, PRACTIS, MOMAS). With the CEA it falls into the framework of topic-oriented technical committees.

As for subcontracting itself, it is controlled by the Agency's procedures in terms of the cost and quality of the supplies. The follow-up of the services complies with the requirements of the ISO 9001 standard in terms of the follow-up of contracts (kick-off meeting, definition of points of validation with the service providers, formalisation of the final acceptance of the services). The requirements, especially in terms of quality, applicable to subcontractors are stipulated in a specification. An annual supplier audit programme is defined and implemented.

1.5.3.3 Control of the data on the waste packages

Upstream from the project, Andra wrote a specification intended for waste producers describing the information which it would need to control the studies and the formalised framework in which it required to receive the information. The waste producers therefore communicated « Descriptive catalogues » which were checked by the Agency. On account of the variety of wastes and packages listed, classifications were made by « waste package families » in order to obtain homogeneous sets. The data obtained in this respect is described in the reference knowledge document and inventory model of high-level, long-lived waste packages (see section 2.1).

The initial versions of the knowledge files were created by the producers in 1998-1999, covering almost all of the waste package families already produced or being produced. Starting in 1999, details and additional information were given concerning the description of the chemical and radiological contents of the waste packages.

For the waste package families already produced or being produced, Andra addressed specific requests to the producers on the qualification of the conditioning processes and the quality control of the production. All the technical documents created by the producers in response to these requests comprise the waste package family conformity reference document ; this document includes, in particular, a « Process Description (DP) » and an « Activity Assessment Description (DEA) ».

After the producers had submitted the « Descriptive catalogues », Andra drafted a technical instruction. Beyond an appraisal of conformity with the package knowledge specification, it had to check the consistency of the technical data : comparison of the conformity reference document with the documents' data, consistency between the chemical inventory and the radiological inventory, cross-checking with the observations made during the monitoring actions on the production sites. After exchanges with the producers, and, where applicable, a file revision, the finalisation of the process for each waste package family was formalised by Andra's « acceptance » of the descriptive catalogue.

For the waste families already produced or being produced, the acceptance can be accompanied by a « level 1 approval ». This signifies that the quality assurance provisions which « govern » the production of the wastes (for the actual packages) and/or the descriptive data available on the packages already produced (for the old packages) provide sufficient confidence in the representativeness of the knowledge files' contents. It does not prejudice the acceptance of the packages in a possible future repository.

The acceptance and level 1 approval process was closed at the year-end in December 2004. At this date, 55 of the 62 knowledge files had been accepted and 27 of the 31 families submitted for approval had been effectively approved.

The unaccepted dossiers correspond to the old waste families (PIVER and SICRAL glasses of Marcoule, old source packages, diverse radiferous wastes, research spent fuels) for which it seems globally that the specification of the « Descriptive catalogues » was too demanding for the knowledge that the producers had about this type of waste. This unacceptance of the dossiers did not have, however, any consequence on the quality of the studies (on this point see chapter 6, subsection 6.2.4).

In parallel with the control of data on the packages, Andra carried out supervision actions on the production of conditioned wastes. Originally, these actions were intended to check the conformity of the produced packages with the production specifications. Since the writing of the « Descriptive catalogues », an additional objective has been added, which consists of ensuring the durability of the description supplied in these files because of the evolutions in the production process. Thus, these actions allow following the possible evolutions in the packages' characteristics. Storage conditions are also watched. Andra makes sure that information which might be eventually important (package arrangement, ventilation, etc.) is available. The traceability of the information on the packages, the checks made and the data archiving conditions also belong to the supervision field.

Supervision includes document analyses, as well as audits and inspections carried out on the production sites ; it consists of assessing the production and control processes, as well as the corresponding quality provisions, in order to regularly follow the evolution. Andra tracks also the production of packages not conforming to the reference document and qualified as « discrepancies » in order to complete knowledge about the packages. There are hardly any such discrepancies.

This entire supervision setup is managed by procedures under the Andra quality management system. It was regularly inspected (once a year) by the Nuclear Safety Authority.

It should be noted that the control of package knowledge is based on the producers' quality systems applied to the waste conditioning processes. These systems were set up or reinforced, accordingly, subsequent to the Decree of 10 August 1984, on the safety of the basic nuclear installations, which formalised the requirements in terms of quality.

1.5.3.4 Control of the data acquired on the Meuse / Haute-Marne site

Data acquisition on the Meuse / Haute-Marne site (data acquired in a bore-hole during the follow-up of the sinking, during experiments) is part of the general context of the control of scientific data described in the previous subsections. In particular, a scientific control by a qualified engineer, generally an Andra engineer, is systematically performed in order to make sure that the data are acquired according to the Agency's procedures and that they satisfy the expected quality standards (frequency of the samplings, usability of the acquired data, etc.).

From an organisation viewpoint, databases were defined (for the laboratory's geoscientific data, for the collection and usage of the data obtained from the experiments and for the hydrogeological data). They allow centralising and partition information between the engineers of the scientific management and the engineers on the laboratory site.

1.5.3.5 Control of the computer codes

The computer codes used in the establishment of the Dossier 2005 were verified by inter-comparative exercises and rigorous testing and validation procedures organised within the « Alliances » platform common to the CEA and Andra, which was developed to accommodate them in a unified framework.

Particular attention was paid to the « Alliances » platform qualification phase. This phase included three test levels :

- Unit tests, to validate the development of the functions ;
- The physical / numerical tests, to explore the various aspects of the validity domain of the Alliances modules, which simulate the physical phenomena to be represented. The results obtained were compared to analytical, numerical or experimental reference solutions ;
- The application tests, representatives of problems to be solved. In particular, they can require the implementation of a concatenation of several modules. The results obtained are subjected to comparative exercises between codes or an expert's recommendation.

Thus, more than six hundred unit tests were run.

The version 1 qualification plan used for the calculations of the present dossier comprised twenty-two physical / numerical tests and six application tests. For each of these tests, a sensitivity study was conducted including :

- Component sensitivity (computer code) ;
- Sensitivity to the different meshing sizes ;
- Sensitivity to the various discretionary methods ;
- Sensitivity to the resolution algorithms.

The qualification phase was essentially carried out by physicists capable of performing a fine analysis of the results and assessing the platform's usage ergonomics. For the most complex cases, the team relied on experts to validate the modelling choices and the numerical choices used to implement the test cases. Readers with a special interest in calculation architectures will find a description in chapter 5, section 5.4.

1.5.4 Taking relevant references into account

1.5.4.1 The Basic Safety Rule III.2.f

Andra took into account the recommendations of the Basic Safety Rule (RFS III.2.f), which defines a certain number of trends in terms of design. The BSR is not a regulatory text, but it makes up a basis for common discussion and understanding between the safety authority and the repository's designer. It defines a certain number of principles related to safety in the post-closure phase, indicates the generic role of the main components and sets objectives in terms of radiological protection.

A detailed analysis comparing the approach of the Dossier 2005 with the recommendations of the RFS III.2.f. is given elsewhere [11] ; we will restrict ourselves here to the review of a few basic elements. It should be noted that the BSR applies only in a strict sense to the post-closure phase.

The BSR objectives are taken into account at the most upstream design stage, which is the stage where the functions that the repository must fulfil are analysed. Among them the function called the « protection function » is in fact identified (protect people and the environment from the dispersal of radioactive elements). This function does not exclude other functions identified in the repository's functional analysis, such as the function of receiving the repository's waste packages, which is the reason for the existence of the repository.

This basic protection function is then split into safety functions, which can in turn be split into subfunctions (see chapter 3) down to more elementary functions, which can be fulfilled by one or several components of the repository.

The splitting up of the main function into clearly separate special functions offers a basis for a multi-function approach to safety. Several components of the repository can thus contribute to fulfilling the same function (notion of complementarity) or allow maintaining the function in case of a failure of one of them (notion of redundancy).

The BSR defines design bases linked to safety, which are taken into account in the assigning of safety functions to the repository's various components. In particular, it identifies special components called « barriers », which have the dual role of protecting the wastes against the circulation of water and intrusive human actions, and limiting and delaying the transfers of radionuclides. It identifies as « barriers » the waste packages, the engineered barriers and the geological medium. The way in which Andra has reapplied this notion of « barrier » and extended it to that of a « safety function » will be indicated in chapter 3.

In addition, the BSR defines three « aspects » which must be covered by the safety assessment :

- justify the favourable character of the performances of each of the containment barriers and the concept of repository versus the global safety of the repository ;
- assess the disturbances caused by the creation of the repository and confirm that these disturbances will remain acceptable with respect to the level of quality selected for each of the barriers and, in particular, the geological barrier ;
- assess the repository's future behaviour and check that individual exposures are acceptable.

This corresponds to checking the repository's performances versus the safety objectives identified as one of the goals of the Dossier 2005. Checks are carried out not only globally but also on the main components acting as a « barrier ». On the other hand, justifying performances is based on the level of understanding of the physical / chemical phenomena underlying these performances and, therefore, on the research programme and the way in which it applies to the analysis of safety. For details on the notion of « performances » of the functions and how it must be taken into account, the reader should refer to the chapter 3 which describes this topic more thoroughly.

The approach by safety functions, together with checking that these functions have a good level of performance is common to operating safety.

1.5.4.2 International references

The texts related to safety issued by international organisations (IAEA « requirements », OECD leaflets, ICRP recommendations) were considered as a reference for the creation of the present dossier. These texts determine the principles which allow dialoguing with the international community by establishing references common to all.

In particular and without prejudice to the application of other texts, Andra referred to ICRP 81 [6] for questions related to the radiological protection of the public within the framework of the management of long-lived waste. The main question raised is that of real or potential long-term exposures.

From a strictly technical viewpoint, ICRP believes that the repository's acceptability is based essentially on the principle of optimization « forced » under constraint, taking into account the economic and social factors and according to a mainly qualitative approach. Such optimisation is progressive and should be placed within the framework of an iterative repository development approach. It must rely on the best available technologies, « good engineering practices[3] » and quality management.

The commission recommends the following principle : the level of protection of future generations must be a minima on the same order as the actual level of protection. For the time intervals considered, the commission insists on the fact that the exposures of the critical groups are indicators of the level of protection reached and must not be considered as a forecast of the impact on health.

Both categories of situations to be considered for the application of the radiological protection criteria are the natural phenomena and accidental human intrusions. The commission considers the radiological protection to be satisfactory if the requirements related to the natural phenomena are respected or measures are taken to reduce the likelihood of human intrusions.

Beyond the questions simply related to impact, recent developments in international reflective thinking (IAEA « safety requirement » project no. DS 154 [12], « Post-closure safety case for geological repositories » of OECD/NEA [13]) insist on the notion of « safety case » in addition to « safety assessment » alone. Beyond the risk analyses and impact assessments intended to check regulatory conformity, the « safety case » takes the form of a synthesis of arguments of a multiple nature contributing in establishing confidence in the repository's safety.

Without pretending to be exhaustive, we can cite a few main recommendations made by NEA to establish a « safety case ».

As a first step, a « safety strategy » should be defined including the repository's design project management, the site choice and the design process. The strategy retained by Andra is necessarily adapted to a feasibility study phase. The way in which the processes of data acquisition and modelling, on the one hand, and definition of the repository architectures, on the other hand, and finally safety assessment are organised in an iterative process such as described in this dossier gives the main line to the retained strategy.

The NEA recommends carefully defining « the assessment basis », that is, the scientific and technical knowledge supporting the safety assessment, the modelling tools, the databases, discussing quality, credibility, and determining uncertainties. In the sense given here, the « assessment basis » is, first of all and without prejudice to the rest of the documentary tree, made up of the contents of the three synthesis volumes of the Dossier 2005 (« architecture and management of a reversible repository », phenomenological evolution of the repository″ and « safety assessment of the repository ») which describe, respectively :

- The retained concepts and the reasons for the choices ;
- The acquired knowledge base and questions still open ;
- These elements in perspective and their discussion to ensure the safety assessment.

[3] « good practices » or « state of the art ».

The NEA recommends the use of « multiple lines of evidence », that is, to not base the analysis on simply performance calculations, but also to highlight qualitative arguments, the use of varied indicators, to increase the credibility and the solidity of the analyses. The ever growing weight placed on qualitative analyses and the credibility of reasoning versus a view based exclusively on impact calculations and a comparison with regulatory thresholds is a major concern confronting both designers and controllers at the international level. The Dossier 2005 fits in this perspective. Although a large part of the dossier is devoted to impact assessments (chapters 5 and 7), we have tried to get as much information as possible out of these assessments beyond the calculation alone. The chapters treating the repository's design (chapter 3), the management of uncertainties (chapter 6) or the teachings from safety analysis (chapter 8) are also given equal importance.

Finally, the following qualities are expected from a « safety case » :

- « transparency », that is, clarity and intelligibility, and a concern for its adaptability to the various targeted readers ;
- « traceability », that is, ability to backtrack to the source of any assertion, data, assumption, by a clear presentation and the use of references ;
- « openness », that is, accounting and discussion of the uncertainties, open questions, or any element forcing reconsideration of the repository's safety ;
- organisation of internal and external peer reviews.

The Dossier 2005, whose structure and write-up were partly completed before the publication of the most recent documents formalising the NEA's reflective thinking, is however inspired by the « safety case » model such as it has been defined. To formalize a complete « safety case », in perfect coherence with the NEA's recommendations, is a long-term effort, towards which Dossier 2005 is only a step.

1.5.4.3 Other texts

Current regulations on nuclear installations and more general regulations on work or underground installations were also taken into account if they had a potential influence on the choices of technical solutions. Regulatory conformity is not an objective at this stage of the dossier and, therefore, no formal demonstration is given. Concepts, however, are developed in a way to be compatible with regulations.

In particular, the Decree of 10 August 1984 [14] is taken as a reference in terms of its principle, that is, the studies contributing to the Dossier 2005 were conducted according to strict quality assurance rules. On the other hand, neither « important elements for safety » nor « activities concerned by quality » are defined in the dossier, since the dossier is not a safety report. The definition of such notions may only take place at a later stage in the dossier, when the operating safety studies will have been carried out up to a more advanced stage.

1.5.5 Work assessment

The quality of the results produced within the framework of the Dossier 2005 is also based on an assessment system provided by law and ordered by Andra's trustee ministries or implemented by the initiative of the Agency itself. The notes and recommendations derived from these various proceedings covering the spectrum of activities of the HLLL waste project (engineering, scientific data, safety) are taken into account in order to contribute to the continuous improvement of the project.

1.5.5.1 National Assessment Commission

The National Assessment Commission was created by the Law of 30 December 1991 and assesses the quality of the scientific work performed by Andra. Since its inception and up to the production of the Dossier 2005, it has regularly listened to the Agency's scientists on all kinds of topics related to knowledge acquisition, modelling and engineering. The commission publishes an annual report in which it expresses its opinion on the work carried out by Andra.

This opinion and the related recommendations are input data for refining the research programme's priorities. In addition to the sessions devoted to special topics, an assessment of the studies and work published annually by the Agency is presented to the CNE (National Assessment Commission).

1.5.5.2 Nuclear Safety Authority

The Nuclear Safety Authority examines how the Andra programme is progressing. To do this, it can turn to the permanent group staffed by experts on waste issues to give an opinion on the documents produced. This group decides on the basis of a technical analysis made by the Institute of Radiological Protection and Nuclear Safety. Generally, opinions are made on the safety of the concepts proposed by Andra for the disposal of high- and medium-level long-lived wastes and the quality of all the data underlying the safety assessments.

The review process was conducted through an analysis of intermediary dossiers submitted by the Agency and, in particular, Dossier 2001, which was covered by five topic examinations (four in a permanent group on hydrogeology, geomechanics, geochemistry and source terms and finally on the safety approach in its entirety ; one based on an IRSN (Institution of Radiological protection and Nuclear Safety) instruction on the inventory model). The opinions and recommendations were formulated according to the conclusions of these analyses and were taken into account by Andra.

In addition, the safety authority makes also « surveillance visits» either to the Agency's head office or to the Meuse / Haute-Marne laboratory during which it gives an appraisal on the quality of the work underway. The following are among the topics regularly treated (annually as a general rule) : the control of the knowledge on the waste packages, the project's organisation, the quality of the work conducted in the underground laboratory.

1.5.5.3 Assessment by peers

At the request of Andra's trustee ministries, a peer review was organised between October 2002 and February 2003 to assess the Andra programme with respect to international practices. The review was organised by the Nuclear Energy Agency of the OECD ; it included international experts from either Andra counterparts or safety authorities' research or technical support organisations.

The review examined Andra's dossier 2001 and turned to the presentations made by the Agency's engineers and question / answer exchanges. It was based on a specification defined by the trustee ministries. The review issued a general opinion, in particular, on the quality of the documentation and the way in which the research programme compared to international standards ; it issued recommendations on a certain number of specific technical points. The report is available to the public [15].

The recommendations were taken into account in the definition of the research programme's priorities. They nurtured also some reflective thinking about the organisation of the Dossier 2005's documentation, which was directly inspired by the experts' remarks.

1.5.5.4 Andra Scientific Council

The Andra Scientific Council was created by Decree [16], and is responsible for « giving opinions and recommendations on the priorities [of the research programme] » and assessing the results. It is staffed by experts appointed by the Agency's trustee ministries. As such, it issues opinions which are taken into account in the definition of the research programmes. It examined the work conducted under Dossier 2005 as it was produced.

1.5.5.5 Other assessments

In addition to this, Andra committed itself to submitting the results of its work to an independent assessment as soon as it seemed relevant, which entailed :

- The creation of work groups staffed by experts who had contributed to review the Dossier 2005 documents (the volumes and the reference documents) as they were produced ;
- The presentation of the Andra work in a large number of international congresses and conferences in order to gather reactions on the presentations.

1.6 Specificity of the safety approach of the Dossier 2005 argile

After having introduced the major principles on which the Dossier 2005 is based and before turning to the contents itself of the « safety assessment » volume, it seemed useful to indicate to the reader what could be the specificities of the adopted safety approach with respect to the more conventional approaches implemented for the safety reports of the basic nuclear installations.

1.6.1 Feasibility's place in the safety assessment

The safety approach of the Dossier 2005 takes into account the fact that the studies are at the feasibility stage. This induces specificities compared to a more conventional safety dossier. The purpose of feasibility is to focus on the existence of technical solutions, but not to irrevocably freeze them. In particular, the concepts may evolve as the design phases advance, which could lead up to the opening of a repository. In a certain number of cases, several technological solutions exist as an answer to a given problem and, in particular, in operation. The feasibility assessment is thus based on one of the best controlled technologies, but it is still possible to implement another one.

As a result, and although safety was a major criterion in the definition of the architectures and their development, their optimisation was not necessarily accomplished to the end. Depending on the progress made in knowledge, ways of improvement are possible and could be developed in later phases of the project. An « ALARA » approach leading to the optimisation of the radiological protection of workers and the public was not developed to its fullest extent. Nevertheless, the proposed concepts already offer a high level of safety and form a solid basis for the execution of future work.

1.6.2 Place and role of the Meuse / Haute-Marne site

The peculiarity of the « Dossier 2005 argile » is that it is based on the observations and the results from experiments carried out on a real site, namely, on the Meuse / Haute-Marne laboratory site. This particular context allows establishing the feasibility and, therefore, the safety assessments on site data. It should not, however, lead to confusion in the sense of the safety approach of the Dossier 2005. Its objective is not to position the repository on a particular site in order to apply for licensing. The decision to take such an initiative is not Andra's responsibility. But it is the government officials elected by the people who must decide whether a repository should be built and under what conditions. If this decision would be made, an instruction process would be triggered to obtain a building permit and then an operating license. This process and its phases have not been defined to date, but we can imagine that it would imply a large public consultation and a detailed instruction of dossiers related to the repository's safety by the competent authorities. The Dossier 2005 is not a dossier of this kind.

The purpose of the Dossier 2005 is to assess the feasibility of the repository in a specific geological formation, the Callovo-Oxfordian. As such, it obviously deals with the repository's safety, an integral part of feasibility. It is based on the data collected on the Meuse / Haute-Marne laboratory site. Because of its very nature, this site underwent the most detailed exploration. However, Andra did ensure that the main characteristics observed locally of the stratum could be extrapolated to a larger zone called a « transposition zone », whose definition is given in [17]. This transposability is proof that the results obtained are not dependent on the specificities of a small-scale zone.

On the other hand, it is too early in this context to tackle the question of where the repository will be exactly located. We will see that for the needs of certain safety assessments (chapter 5, chapter 7) it is necessary to « localise » the repository in the zone. In such a case, it was generally decided to position the repository conventionally where the laboratory is located (choice made for the engineering studies, and for the impact studies). However, the possibility that the rock properties can be a little different at other locations within the zone is taken into account in the assessments and, in particular, in the management of uncertainties. The necessity of making this type of choice must not create confusion in the sense of the approach : the repository's position is absolutely not being optimised according to safety criteria or other criteria.

1.6.3 Operational safety and long-term safety

The above considerations lead to underlining the two-fold specificity of the Dossier 2005's safety approach :

- For the operational safety part, it is globally related to a conventional approach. It concentrates on the problems specific to the repository, and does not treat in detail all the safety provisions since they are already well known in another context. This is particularly the case for surface installations. Nor does it fit into an application for the creation of an installation ;
- For the long-term safety part, the emphasis is placed on the control of scientific knowledge and uncertainties.

Therefore, Andra developed two additional sequences : one intended for operational safety, the other for safety assessment in the post-closure phase. Each must take into account the other's constraints.

In fact, the operating conditions broadly speaking are constraints imposed by long-term safety needs. A technological solution (for example, for the sinking of the structures or for the handling of the waste packages) which could appear to be the best suited from the viewpoint of operating conditions, but which would lead to a non-control of long-term evolution conditions (damage of the rock to an unknown extent, risk with respect to the rock's long-term stability, etc.) is prohibited.

It should be noted that long-term safety is also imposed to a certain extent by the operation. On the one hand, the concepts should not correspond only to an optimum vision of long-term safety, but they should also be exploitable under safe conditions and without excessive technical complexity. During the definition of the concepts, we make sure that they can be effectively realised under good conditions and that the proposed solutions are realistic.

On the other hand, it may be necessary in the long-term safety analysis to explicitly take into account the conditions under which the repository architectures will have been built and operated : How were the structures sunk and how does this influence their state after closure ? How long were the structures left without backfill ? What are the empty spaces possibly present between the packages ?... The conditions of the evolution of the operating phase in a broad sense (construction, incorporation of the packages, observation and closure) define partly the initial state of the repository for its evolution in the post-closure phase.

The principle of intergenerational equity prohibits favouring one time scale over another. The operating and long-term safety conditions are, therefore, both taken into account in the choice of the concepts.

In the Dossier 2005, the description of the concepts and the explanation of the choices made were necessarily linked to the operational safety and the long-term safety. But consistent with the principle mentioned above, after this description the operating and long-term safety assessments are separated but have similar approaches.

Although the safety approach of the Dossier 2005 goes back to the concepts and the general spirit of a « conventional » nuclear installation safety approach, it differs finally by a few general aspects :

- The necessity of approaching in a coordinated way the different life phases (operation, post-closure) ;
- The taking into account of time scales which extend beyond human experience ;
- The strong relationship between design, knowledge acquisition and safety assessment for the feasibility assessment ;
- The key importance given to the notion of uncertainties control and, in particular, for the post-closure phase.

These peculiarities result as much from the studied object's specificity (the repository in a deep geological formation) as from the question raised (that of feasibility). It requires calling on many disciplines (mining and nuclear engineering, earth sciences, material sciences, safety) and implementing specific methods at the interface between these disciplines.

1.7 Overview of the approach – introduction to the following chapters

This last subsection of chapter 1 is an introduction and a reader's guide for the rest of the dossier. It explains how Andra determined the principles of its safety approach during the studies, and how the result of this work was reported in the Dossier 2005. It describes the main elements and their breakdown. The reader should refer to the references cited throughout the dossier in order to learn more about the topics of interest to him.

The repository design, scientific research and safety assessment approach is an iterative approach. It advances with the information learned from the previous phases (namely, the phases which preceded the production of the Dossier 2005 : definition of the initial design options and then first assessment with an essentially methodological intention in the Dossier 2001). The Dossier 2005 presents, however, for clarity the information acquired from the research work, without systematically making a distinction between what could be derived from the previous iterations and what may be obtained more specifically from the work carried out during the period 2002-2005. In particular, the repository architectures are presented and justified based on the knowledge updated in 2005. Nonetheless, their definition results from work carried out in successive steps and may still evolve beyond 2006.

1.7.1 Presentation of the input data

Any repository development project must start with an analysis of the technical input data on which it is based. The number and variety of these data increase with each iteration of the repository's definition process. The input data and the fruit of the analyses conducted under the Dossier 2005 are structured into a documentary architecture which is shown in Figure 1.7-1. The way in which the main documents are organised according to the safety approach is described below in this subsection.

At the start of the project, only the inventory of the packages to be disposed is known. It is examined at first generically in order to understand its major characteristics and then progressively in more details as the safety studies allow understanding what are the important characteristics.

The input data on the packages are described in the design inventory model [18]. In this volume, a review of the main characteristics of the packages essentially to determine a few important elements for the rest of the analysis is given in chapter 2.

A site selection process is then conducted in parallel with an initial requirements definition of the architectures. Once one or several site(s) are selected (in the case of the Dossier 2005, the Meuse / Haute-Marne laboratory site was retained for the feasibility studies), they become the input data for the analysis. They undergo a more and more detailed characterisation more and more focused on the major questions for the repository's engineering and safety.

The state of knowledge on the Meuse / Haute-Marne laboratory site is presented in the site reference document [17]. A presentation of a few key results to facilitate the understanding of this volume but without being exhaustive is given in chapter 2.

Once the study site is retained, it is possible to conduct studies on the main types of materials which will be used in the repository and on the behaviour inside the rock, including the waste packages themselves. The question of the transfer of the radionuclides and the chemical toxics into the rock is also a topic which is studied very early, because it allows characterising the confinement functions which can be expected from the geological medium. This knowledge constitutes as the project progressively develops an important input data set for the designer, who defines the architectures based on what he knows about the behaviour of the materials disposed.

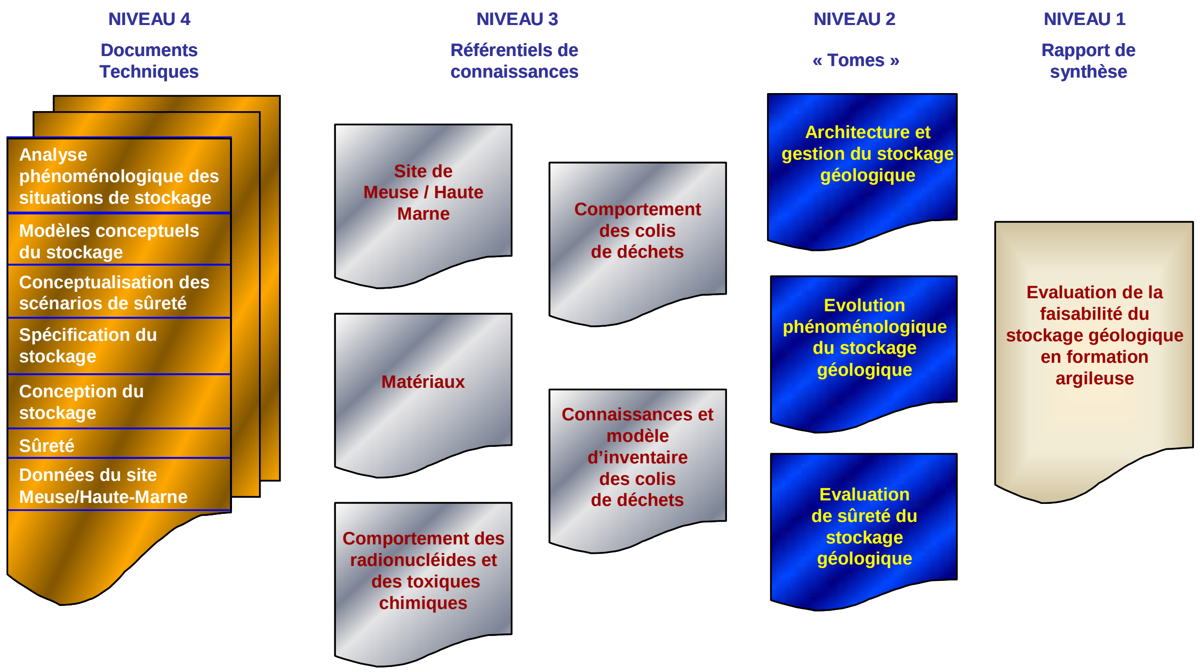


Figure 1.7-1 Organisation of the documentary structure of the Dossier 2005.

This knowledge is presented in the « materials » reference document, the « behaviour of the packages » reference document and the « behaviour of the radionuclides and the chemical toxics » reference document [19, 20, 21]. The corresponding elements are only referred to in the present volume when they explain an element of the safety analysis.

As the architectures are being developed, the technical options are progressively frozen. At the feasibility stage, the choices are still open. However, the major determining factors are considered to be sufficiently stable from both the viewpoint of the repository's organisation (the overall architecture and the way in which the repository zones, access structures, etc. are distributed) and how it would be designed and operated (with the requirement of reversibility as an example).

These concept description elements are very briefly mentioned in chapter 2 to facilitate the reading of the rest of the volume. **For more detailed information on the definition of the repository architectures, the reader can refer to the volume dedicated to this topic** [22].

1.7.2 Definition of the safety functions

The repository's design approach is based on an analysis of the expected functions : what do we expect from the repository ? **This functional analysis called an « external » functional analysis is a prerequisite for the definition of the architectures** [23]

Then, the designer defines the way he intends to respond to these functions : what are the technical solutions he wants to implement ; what are the subfunctions he is going to assign to each of the repository's components ? He is guided in his approach by a set of constraints (regulatory, technical, economic, other) and by the state of the knowledge. Particular care is given to determining the functions which are to ensure the long-term safety of the repository, which are the most structurally based for the definition of the repository architectures. **The fruit of this analysis and the way in which the repository was designed with respect to this approach by function are presented in chapter 3 of the present volume,** as well as in the volume titled « architecture and management of a reversible repository ». For more details, the reader can **refer to the internal functional analyses** [24, 25].

These analyses are a guide for the entire design approach. In particular, they set the engineering requirements **in the requirements technical specification** [26]. The engineering studies are covered in the volume titled « architecture and management of a reversible repository » and the associated design documents [22]. In this volume, only the part related to operational safety (safety of workers and the public during the construction, the operation and the closure of the repository) is presented through a risk analysis and a more detailed study of a few risks identified as top priority. This part presents the normal operations of the facilities, associated with a first estimate of doses so as to verify the completion of safety objectives, and a risk analysis identifying accidental situation (fire, fall of a package) that, due to their specifity or their influence on the design of the facilities and the equipment, require a detailed study. **These analyses are presented in chapter 4.**

1.7.3 Definition and description of the normal evolution domain

In parallel with the repository definition approach and in strong interaction with it, a detailed process of description of its evolution over time is carried out. This work is based on a breakdown of the repository into situations, with each of these situations corresponding to a space and time interval within which a few major phenomena dominate the evolution of the components. **This description is the object of the phenomenological analysis of repository situations (PARS) in a normal evolution situation.** [27, 28]. Thermal, hydraulic, mechanical, chemical and radiological phenomena are recorded in this context.

The description is not univocal : because of the space and time scales considered, uncertainties exist over the time frame of the phenomena, their spatial extension, and possibly even their nature. Therefore, it is not a matter of presenting a sure evolution of the repository, but rather a set of possible evolutions. These evolutions belong to the normal evolution domain, which combines all the likely evolutions, as well as possibly other less likely evolutions, whose consequences have no impact on safety. For example, if a container was sized to last ten thousand years, it is possible that its service life will be longer if it is placed under favourable conditions : all the service lives greater than ten thousand years belong to the normal evolution domain. On the other hand, a shorter service life which could jeopardise the repository's safety does not belong to this domain.

The definition of the normal evolution domain is progressive and is made interactively with the repository's design studies. It allows specifying the performances which can be expected from the functions.

Once this domain is defined, the objective is to check through a performance assessment first component by component and then globally that the normal operation domain complies with the set safety objectives. To do this, the behaviour of the repository's various constituents and its environment is represented by models. This is the **conceptualisation of the repository, whose results are presented in the dedicated documents** (see the complete list in the volume titled the « phenomenological evolution of the geological repository » [7]). This conceptualisation is itself « tainted » by uncertainties which are described in these documents. In order to proceed with a global assessment, the models are selected and concatenated to form a global safety model, which represents the normal evolution scenario. This latter can have variants and separate calculation cases in order to cover the normal evolution domain. **The definition of the scenario and the results of the performance calculation are given in chapter 5**. The results of the conceptualisation and the performance calculation are used to confirm the safety objectives are being met, as well as to feed back the design and the knowledge acquisition approach.

1.7.4 Uncertainties management and repository robustness assessment

The management of uncertainties is at the centre of safety analysis. It directs the design, participates in the definition of the normal evolution domain, and lays the foundation for risk analyses. It may therefore appear in all the chapters of the present document ; however, we decided to dedicate a chapter to it for better clarity.

Chapter 6 presents the results of the qualitative analysis of safety, which consists of identifying the uncertainties of knowledge and studying their influence on the repository's behaviour. It allows not only characterising more completely the bounds of the normal evolution domain, but also identifying the situations that are not included in the normal evolution domain. It also allows setting up an initial hierarchy of the uncertainties according to their importance with respect to safety.

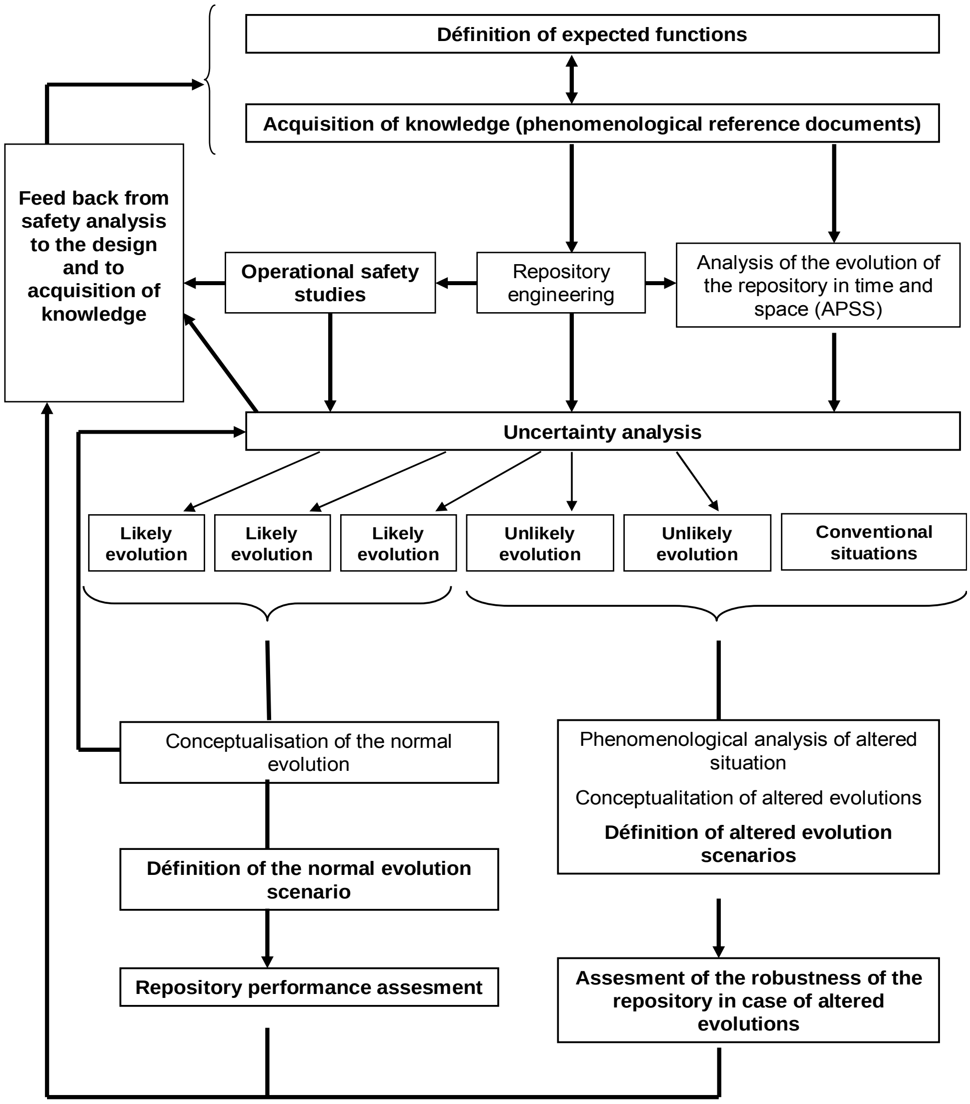
Some uncertainties can lead to an evolution of the repository which is not desired and no longer satisfies the expected safety functions. Such evolutions must be highly unlikely. It is advisable if this serves the objectives of the safety analysis to add other evolutions defined purely for safety reasons, which have no likelihood of occurrence and are studied simply to learn about the repository's behaviour faced with an unexpected influence. **The definition of these situations qualified as « altered » is presented in chapter 6.** It should be noted that beyond the simple definition of the altered evolutions the analyses of uncertainties are useful to finalise the design and to make it more robust to uncertainties, or to give priorities to the research programme.

Altered – reference situations, in this case a failure of the repository's seal devices, a failure of packaging elements, as well as an intrusive bore-hole intercepting the repository and left abandoned, were defined in advance based on the feedback from previous dossiers (of Andra and its counterparts). **Their phenomenology is described in the PARS called « PARS of altered evolutions** » [29, 30, 31], **which allow learning how the processes controlling normal evolution can be modified**. The situations of altered evolutions derived from the recording of uncertainties are attached to these major reference situations in order to form the altered evolution scenarios, which call for a special performance calculation. The objective is to check whether the repository remains safe under these worst-case conditions and to obtain additional information on the behaviour of the repository's components. **The altered evolution scenarios and their results are presented in chapter 7 of the present volume**.

1.7.5 Conclusion

The conclusion of the safety analysis (**presented in chapter 8** draws on all these studies, completes the hierarchy of uncertainties, and tries to define the paths of progress in terms of engineering or research learned from the safety analysis. Without anticipating the decisions to come, they can contribute to defining a post-2005 study programme under a new iteration of the repository design. They allow also strengthening the overall judgement on the repository's feasibility.

A schematic representation of the sequence of the various phases of the safety analysis is given in Block Diagram 1-1.



Block Diagram 1-1 Representation of the concatenation of the various phases of the safety analysis

2 General description

This chapter gives an overview of the architecture and the management of a possible repository installation. At this stage, the geological formation, the repository architectures and the management modes are only briefly presented without any special justification : the objective is simply to introduce the essential notions to allow understanding the rest of the document.

First of all, the inventory of the wastes to be considered is recalled (subsection 2.1) and the geological context for which its feasibility is studied (subsection 2.2). Subsection 2.3 then introduces the technical options in terms of architecture and packaging which Andra retained for the studies.

2.1 High-level and long-lived wastes

This subsection is a summary of the sections related to the HLLL wastes of chapters 2 and 3 of the volume titled « architecture and management of a repository » [22]. It is included here only to facilitate the reading of the rest of the present volume. The reader especially interested by waste package inventory questions should refer above all to the volume cited as a reference.

The high-level and long-lived wastes (HLLL contain both short-lived radionuclides generally in large quantity (high activity) and long-lived radionuclides in average to very large quantities.

Their content in long-lived radionuclides[4] confers on them a long radio-toxicity duration associated with the risk of ingestion, leading to an exposure of living tissues to α radiation ; the radioactive half-life[5] of some isotopes can exceed hundreds of thousands of years.

A large part of the HLLL wastes present also a high activity in γ radiation, which implies protecting man from an external radio-exposure.

The β-γ activity present in the HLLL wastes decreases relatively rapidly over time : thus, after a few tens of years, the nuclear fuels contain no more than a few percent of the radioactivity they had when the reactor was unloaded.

The energy produced by radioactivity is converted mainly into heat : the radiation is absorbed in the matter itself comprising the waste packages and to a lesser extent in the matter located in the immediate vicinity. After the β-γ radiation has greatly dropped (after a few centuries), the residual radioactive energy associated with the long half-life isotopes is very low and the produced heat becomes then insignificant.

The decrease in β-γ activity over time can require an interim waiting period for the more active wastepackages between their production and their disposal. This period takes place in the storage installations. It allows reducing the heat produced by the wastes, which is concomitant with the design of the disposal installations and their extension in the host formation.

2.1.1 Nature and origin of the HLLL wastes

The HLLL wastes originate from the electronuclear industry, as well as from research activities and national defence. In order to conduct the study of repository possibilities, Andra created an « inventory model » consolidating the data and assumptions on the HLLL wastes [18]. It takes into account the wastes already produced and presently stored on their production sites, as well as the future wastes.

The wastes of the electronuclear industry originate essentially from the spent fuels unloaded from the electricity producing reactors. Today, these fuels are recycled by « Cogema » in its plants of « La Hague »; the residues are separated from the uranium and the plutonium - they are fission products and minor actinides – as well as the mechanical structures of the fuel clad (duct sections, end cap pieces).

4 The long-lived isotopes include (i) the products of fission or activation resulting, respectively, from the division of heavy atoms such as uranium and plutonium during fission reactions in a reactor and the absorption of neutrons by the materials present in the reactor (mainly metals) (ii) the actinides, compounds of uranium and heavier atoms formed from uranium by the capture of neutrons.

5 The radioactive half-life of an isotope is the time required for 50% of the quantity present of this isotope to disintegrate (i.e. transform spontaneously into another element, radioactive or stable). An isotope is said to be « long-lived »of its half-life is strickly superior to 30 years.

For the study of the repository, we consider all the wastes engaged by the presently installed electronuclear base based on the assumption of an average reactor operating period of forty years.

Various production scenarios were considered in the study. They were selected in order to provide access to a wide range of waste types, even hypothetical, and thus allow dealing with the various problematics for the study of their repository. The first production scenario family considers the continuation of the recycling of spent fuels unloaded from reactors. The second scenario family assumes a suspension of recycling. The objective of these scenarios is not to prefigure an overall industrial model, but to examine how a repository architecture can take into account various inventories and possible management modes downstream from the electronuclear cycle. These scenarios were created in relation with the waste producers (EDF, CEA, Cogema).

The imagined scenarios allow taking into account the spent fuels, which are materials not considered at this stage as wastes. If we assume that they would not be recycled, they could make up a study object for waste management techniques. It would then involve MOX spent fuels (mixtures of uranium and plutonium oxides) obtained from a recycling of plutonium, or enriched uranium fuels.

To the spent fuels or the residues from their recycling we can add the nuclear reactor operating wastes (command or control rods), as well as the operating and maintenance wastes of the recycling plants (so-called « technological » wastes - replaced or obsolete parts contaminated by the materials and recycled radioactive wastes ... - ; liquid effluents ...).

It should be noted that the reprocessing wastes also come from the Marcoule plant shut down today, which recycled the fuels from the former « Natural Uranium Graphite Gas » (UNGG) line.

The HLLL wastes coming from activities other than electronuclear production (research, defence) are generally technological wastes.

It should also be noted that there is a small quantity of spent fuels originating from research or military reactors. No assumption is made about their possible reprocessing. The possibility of their direct disposal is explored.

2.1.2 HLLL waste conditioning

HLLL waste conditioning consists of (i) solidifying, immobilising the wastes which have been produced in a dispersible form – namely liquid - (ii) placing the wastes in a container facilitating handling and storage in the industrial installations.

The inventory of the HLLL wastes takes into account two kinds of wastes :

- Wastes already produced, which are stored under conditioning or not on the production sites ;
- Wastes to be produced, either with a conditioning in the continuity of the operation of the existing nuclear installations, or with adaptations which are still not known exactly and which will depend, in particular, on the strategy retained for the production of energy and for the fuel cycle.

To compile the inventory, it was decided[6] to refer to wastes under conditioning. This implies knowing or formulating assumptions on the nature and the conditioning and packaging modes of the existing wastes still not conditioned and future wastes, as well as the number and volumes of the so-called « primary » waste packages to be considered ; the primary waste packages are the objects which would be delivered to a possible repository site.

The identification of the various wastes and the definition of their conditioning mode (whether it exists today or whether it is retained as a reference assumption) were made in connection with the waste producers. The result is a relatively large variety of primary waste package families, which differ in terms of the radiological content, the release of heat due to the presence of certain radionuclides, the physical / chemical nature of the waste or the conditioning materials, and the dimensions.

6 However, the option of direct on-site conditioning is studied in the case of spent fuel assemblies.

A distinction is conventionally made between the following categories of HLLL waste, which have their own problematics :

- The so-called B wastes are characterised by low or medium β-γ activity and, as a result, by a low release of heat. They represent the largest number of packages, as well as the greatest variety of conditioning types. Their total inventory in long-lived radioisotopes, relatively lower than that of the other categories hereafter, is distributed over the large volume which they represent.
- The C wastes are made up of the fission products and minor actinides separated during the recycling of the fuels. Their high β-γ activity produces a large released quantity of heat, which decreases over time, mainly with the decrease in radioactivity of the fission products with a medium half-life (cesium 137, strontium 90). The conditioning of these wastes consists of incorporating them in a glass matrix ; the confinement capability of this material is particularly high and durable if under favourable physical / chemical environmental conditions.
- In addition, the spent fuels (marked by the letters « CU »), although they are not strictly speaking waste, but recyclable material, also have a high activity and, as a result, release a significant amount of heat. This release of heat is due to their content in fission products with a medium half-life, plutonium and americium (mainly obtained from the disintegration of plutonium) ; these last two elements lead to a slower decrease in heat over time. Other specificities are the large dimensions of fuel unloaded from the electronuclear reactors, if it were decided to dispose of them as is, as well as a larger content in fission materials associated with the question of a criticality risk.

Within each of the categories[7] introduced above, the various HLLL waste package families were grouped into a more limited number of representative « Reference packages » ; the purpose of these groupings was (i) to carry out more thoroughly the studies by limiting the number of cases to be specifically treated, but without neglecting the diversity of the waste packages, (ii) to propose a standardisation wherever possible of the structures and the resources which would be implemented in a repository installation. This approach allowed studying a possible repository solution for each of the inventoried waste packages independently from the fact that the other waste categories be the object of a repository or not.

2.1.3 Imagined production scenarios

The presently installed set of pressurised water electricity producing reactors comprises 58 reactors, which went into service between 1977 and 1999. The tonnage of nuclear fuels which would be unloaded from these reactors over their total operating life is estimated at 45 000 metric tons of « heavy metal » (tML). This estimation is based on a combination of assumptions on (i) the average service life of the sections (forty years), (ii) the energy production (16 000 terawatt-hours of cumulative production), (iii) the progressive increase in the « burnup » rates of the fuels in reactors[8].

The fuel types which were considered and their average burnup rates are as follows :

- Three generations of uranium oxide fuels, UOx1, UOx2, UOx3, irradiated, respectively, with 33 gigawatt-days per ton of fuel (GWd/t), 45 GWd/t, 55 GWd/t, on the average ;
- Fuels containing recycled uranium (URE) irradiated on the average at 45 GWd/t ;
- Mixed fuels with uranium oxide and recycled plutonium oxide (MOX) irradiated at 48 GWd/t on the average.

Based on these figures, four nuclear fuel management scenarios were retained to carry out the studies. The principle of these scenarios consists of « supporting » various possible industrial strategies without favouring one over another. This approach provides access to a very wide range of waste types and allows a technical instruction on the questions associated with the various waste packages.

7 In this document, the term « waste » recovers these three categories (B, C, CU).

8 The burnup rates of a nuclear fuel assembly translates the energy produced in a reactor by the fission material which it contains (uranium oxide or a mix of uranium and plutonium oxides).

The first three scenarios, called S1a, S1b and S1c, correspond to the continuation of the recycling of spent fuels unloaded from the reactors. Scenario S1a assumes the recycling of all these fuels (UOx, URE and MOX). It is associated with the assumption of an incorporation in some waste of mixes of fission products and minor actinides originating from the UOx and MOX fuels ; furthermore, a very small part of the plutonium originating from the recycled UOx fuels is assumed to be incorporated in the glass. Thus, this scenario covers the varied typologies of the vitrified C waste packages. In scenarios S1b and S1c, the MOX fuels are not treated ; consequently, the feasibility of the direct disposal of these fuels is assessed. A distinction was made between the scenarios S1b and S1c in order to study in scenario S1b the possibility of increasing the concentration of wastes in the glasses with respect to the packages presently produced ; this greater concentration would be translated by a slightly higher release of heat by the packages. Finally, the fourth scenario, called S2, which assumes the suspension of recycling, is the basis for the exploratory study of a direct disposal of the UOx and URE fuels, as well as that of the MOX fuels, as in scenarios S1b and S1c. This scenario assumes that the fuels are considered to be wastes, which remember is not the case today.

To allow a quantitative estimation of the produced wastes, scenarios S1a, S1b and S1c are based on the following breakdown of the various fuel types unloaded from the installed reactor base : 8 000 tML of UOx1 (33 GWd/t), 20 500 tML of UOx2 (45 GWd/t), 13 000 tML of UOx3 (55 GWd/t), 800 tML of URE (45 GWd/t) and 2 700 tML of MOX (48 GWd/t). In scenarios S1b and S1c, the direct disposal study covers the totality of the 2 700 tML of MOX spent fuels.

Scenario S2 considers hypothetically the continuation of the recycling of a part of the UOx fuels up to 2010 (that is, 8 000 tML of UOx1 and 8 000 tML of UOx2), and then a suspension of this recycling. The suspension of the recycling of uranium and plutonium changes the total breakdown of the various fuel types unloaded from the reactors. The study of the direct disposal of fuels not recycled concerns 29 000 tML, that is, 12 500 tML of UOx2, 14 000 tML of UOx3, 500 tML of URE and 2 000 tML of MOX.

As already mentioned in the introduction, the studies refer to conditioned wastes. To ensure this, conditioning modes were defined for the existing unconditioned wastes, as well as for future wastes. The adopted assumptions referred to the industrial processes presently implemented by the producers : vitrification, compacting, cementing, bituminisation.

The various scenarios considered for the repository study allow also having a robust approach with respect to the various possible evolutions in terms of downstream management of the cycle.

Aside from these scenarios, the management of spent fuels originating from other reactors than the PWR of EDF (research and defence reactors, notably) has been considered. Their reprocessing would only produce a marginal quantity of waste compared to those originating form EDF’s. Their direst disposal has been studied on an exploratory basis.

2.1.4 Description of the reference packages

The following subsections describe the wastes which are grouped into reference packages within the MID (waste inventory model).

2.1.4.1 B waste primary reference packages

- **B1 reference package**

The B1 reference package originates directly from the electricity producing reactors ; they are the operating wastes from the installed set of pressurised water moderated electricity producing reactors (REP), as well as some activated wastes originating from the SUPERPHENIX fast neutron reactor.

The REP control rod groups and moderators represent more than eighty percent of the total mass of activated waste. They all have twenty-four rods suspended in a « manifold » holding system which are inserted into the slots left free for this purpose in the fuel assemblies.

Some rods in the groups contain neutron-absorbing materials : boron in the form of PYREX glass for the rods of the moderator groups, boron carbide (B_4C) and/or a silver, indium and cadmium alloy (AIC) for the rods of the control groups. The number of rods containing these materials varies according to the reactors.

Other activated wastes originating from the REP reactors are the corresponding metal wastes, for the most part « blind » tubes called Core Instrumentation Network (RIC) glove fingers equipping the reactor tank (they are located under the tank). These tubes allow passing neutron probes which are used to control the nuclear reaction. They are replaced, if necessary, after a certain usage period and are then included in the waste.

The conditioning assumption considered in the study is a compacting of the wastes placed in claddings[9], and then put into stainless steel containers called « Standard Containers for Compacted Waste or CSD-C » of a small size (see Figure 2.1-1).

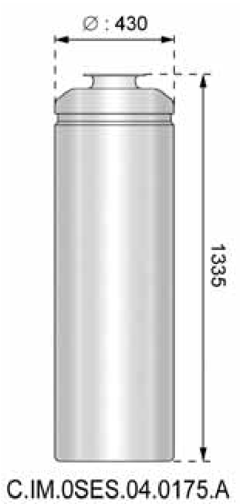


Figure 2.1-1 Standard Compacted Waste Container (CSD-C)

It should be noted that these packages do not contain organic substances and are not liable to generate gas (hydrogen) by radiolysis.

- **B2 reference package**

The B2 reference package describes waste made of bitumen originating from radioactive liquid effluents linked to the usage of fuel recycling installations. The effluents considered here are produced at various stages in the fuel recycling process and during the operations performed on the materials and the installations (decontaminations, rinsings). These effluents are collected in treatment stations where they are decontaminated by chemical processes before rejection. The residual waste is then recovered in the form of sludges.

In the STEL effluent treatment stations of Marcoule and the STE3 of « La Hague » started, respectively, in 1966 and 1989, these sludges were conditioned in an embedded and dried in bitumen form and poured into steel drums. On the other hand, the sludges coming from the produced effluents and chemically treated in the STE2 (No. 2 Effluent Treatment Station) of « La Hague » from 1966 to 1990, were progressively stored in the workshop's tanks and silos for subsequent conditioning. The imagined conditioning mode for these sludges is also an embedding in a bitumen matrix.

The peculiarity of these packages is essentially the chemical nature of the conditioned wastes, which contain a large load in salts and organic matter. Also, the radiolysis of bitumen generates gases, mainly hydrogen, as well as traces of carbon monoxide, carbon dioxide and methane (for hydrogen 1 to 2 litres at atmospheres pressure per year for STE3 and STEL packages, 9 to 10 litres for STE2 packages).

9 Rod groups and large RIC glove fingers are cut into small sections before being put into a compacting cladding.

In addition, the packages embedded and dried in bitumen do not all have the same geometry. The first waste package group (B2.1), representing 45 % of the inventoried packages, corresponds to 238-litre stainless steel primary drums (STE3/STE2) and 245-litre stainless steel primary drums (STEL starting from October 1996). These packages are shown in Figure 2.1-2.

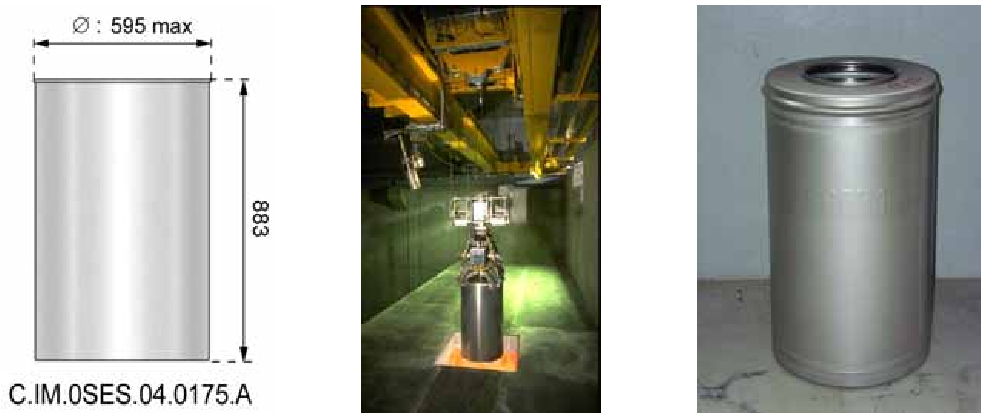


Figure 2.1-2 STE3/STE2 and STEL stainless steel drums

The second package group (B2.2 reference package, 55 % of the inventoried packages embedded and dried in bitumen) corresponds to 428-litre stainless steel drums. These drums (see Figure 2.1-3) are used for the reconditioning of the primary drums made of unalloyed steel produced on the STEL between 1966 and October 1996.

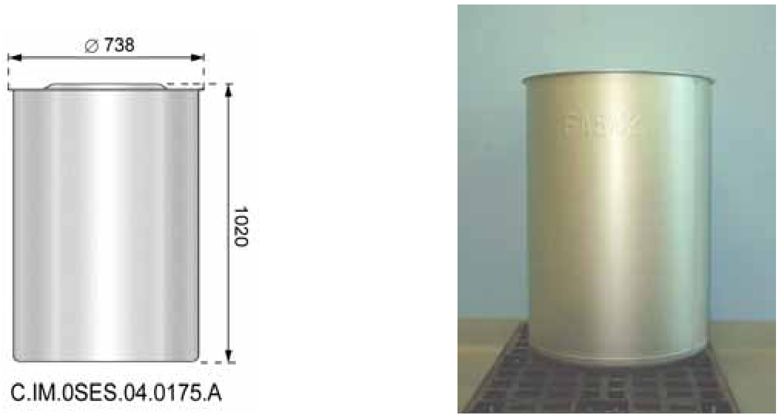


Figure 2.1-3 Stainless steel drum used for the reconditioning of the primary drums made of unalloyed steel

The assumption retained for the studies is a complete refilling of the interstitial empty space between the two drums by an incompressible material, such as mortar, to control the mechanical deformations in the repository.

- **B3 reference package**

The B3 reference package corresponds to the technological wastes resulting from the operation and maintenance of the nuclear installations operated by COGEMA and the CEA. They are made up to a great extent not only of solid wastes of various kinds (miscellaneous metals, organic substances), but also filtering sludges and evaporation concentrates. This mass includes also various wastes produced at Marcoule, such as graphite, ion exchanging resins and zeolites. The radiological activity of the wastes and, in particular, that of the technological wastes is most often due to the presence of a contamination at the surface of the wastes by the fission products and/or the activation products and/or the actinides.

Depending on their origin of production and/or their nature, these wastes are conditioned in different ways. The problematics raised by these waste packages are, therefore, for the most part linked to the diversity (i) of their chemical contents, in connection with the waste natures and conditioning matrices used, and (ii) the containers' materials and shapes. Due to their chemical nature, some of the packages are also liable to produce gases, mainly hydrogen, by radiolysis. These packages do not produce heat.

For the requirements of the waste package inventory model, the packages are defined on two levels taking into account the waste natures, their conditioning modes and the containers (see Table 2.1-1).. At level 2, the classification of the packages was defined based on the materials used for the containers, and the homogeneous or heterogeneous character of the conditioned waste :

- B3.1 : heterogeneous wastes contained in concrete containers ;
- B3.2 : homogeneous wastes contained in concrete containers ;
- B3.3 : heterogeneous wastes contained in metal containers.

The classification in level 3 reference packages corresponds to taking into account the chemical natures of the wastes, the risk of hydrogen being generated and the package sizes (the level 3 reference packages respectively associated with the level 2 reference packages : B3.1, B3.2 and B3.3 are classified in an ascending order of size) :

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

	Waste Package subclass	Container material	Embedding or immobilisation matrix	Presence of metal materials	Presence of organic materials
B3.1	B3.1.1	Concrete	Cement-bitumen	None	x
	B3.1.2	Fibred concrete or asbestos-cement	Cement	x	x
	B3.1.3	Concrete or unalloyed steel	Cement-bitumen ou mortar	x	x
B3.2	B3.2.1	Concrete	Cement	None	x
	B3.2.2	Fibred concrete	Mortar	x	x
B3.3	B3.2.2	Stainless steel	None	x	x
	B3.3.2	Stainless steel	Cement	None	x
	B3.3.3	Unalloyed steel or stainless steel	Cement-bitumen or cementitious material	x	x
	B3.3.4	Unalloyed steel	Cement-bitumen or cementitious material	x	x

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

- **B4 and B5 reference packages**

These reference packages of B wastes originate from the recycling of the spent fuels in the COGEMA plants ; they correspond to the elements forming the metal frame of the fuel assemblies. These wastes are separated from the recyclable nuclear materials (uranium, plutonium) and the fission products and minor actinides at the start of the recycling process during the fuel cutting and dissolution operation.

In the case of the fuel assemblies of the pressurised water reactors, these wastes are usually designated by the term « hulls and end caps ». The hulls correspond to the fuel rod tubes recovered in the form of sections about three centimetres long in which the nuclear material was extracted by dissolution in acid. The end caps correspond to parts located at the two extremities of the fuel assembly.

The cladding wastes considered here originate from the recycling performed in the COGEMA plants at « La Hague ». They include (i) the wastes produced during previous recycling of UNGG and REP fuels, stored today in silos and pools, (ii) the wastes resulting from the current and future recycling of the various typologies of fuels, REP UOx and MOX, defined in the study scenarios given in subsection 2.1.3.

There are several kinds of cladding waste materials : magnesium-zirconium and magnesium-manganese alloys for the UNGG fuels ; a zirconium-tin alloy (zircaloy 4) or a zirconium-niobium alloy (M5 alloy), stainless steels and nickel alloys for the REP fuels. Following the conditioning assumptions indicated below, some packages contain also technological wastes of a metal nature only (unalloyed and stainless steels) or a mixed metal and organic nature. These technological wastes represent as a whole about ten percent of the total mass of wastes conditioned per package.

The initial conditioning mode of the cladding wastes of the REP fuel assemblies consisted of a cementing in large stainless steel drums (B4 reference package, see Figure 2.1-4). This process was applied between 1990 and 1995, and then replaced by a compacting of the wastes implemented in the Hull Compacting Workshop (ACC) of « La Hague » starting from 2002 (B5 reference package).

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

The compacting process is applied to the cladding wastes originating from the UNGG and REP fuels previously recycled and stored today in silos and pools, as well as the cladding wastes coming from the recycling of fuels unloaded now and in the future from the reactors of the installed REP base. As mentioned earlier, some packages contain in addition compacted technological wastes coming from the operation and/or the maintenance of the site's workshops. Because of the diversity of the considered waste flows and their characteristics, four subclasses of compacted cladding waste packages (CSD-C) can be distinguished. They are summarized in Table 2.1-2.

Reference packages	Cladding waste materials	Presence of technological wastes	Presence of organic substances	Generation of gas by radiolysis (H_2)	Thermicity, level of irradiation at the packages production date
B5.1	Zirconium-tin or zirconium-niobium alloys, stainless steels, nickel alloy	x	x	x	High
B5.2	Idem subclass 1	x	None	None	High
B5.3	Zirconium-tin alloy, stainless steels, nickel alloy	None	None	None	Waste packages not thermal, medium irradiant
B5.4	Magnesium-zirconium or magnesium-manganese alloys	None	None	None	Waste packages not thermal, low irradiant

Table 2.1-2 Summary of the main characteristics of the subclasses of compacted cladding waste packages

- **B6 reference package**

This reference package groups wastes produced on the COGEMA site at Marcoule and stored today, except for bituminised wastes and cemented process wastes described above. It includes (i) the operating wastes of the Marcoule vitrification workshop, (ii) the cladding wastes originating from the fuels recycled in the UP1 plant, (iii) the technological wastes coming from the operation and maintenance of the installations on the Marcoule site.

Reference packages B6.1 contains the technological wastes resulting from the operation of the Marcoule vitrification workshop (AVM). The wastes consisting of fittings, tools, and various parts made up of steel are put into a stainless steel container having a geometry similar to that of the AVM vitrified waste containers. The packages have an average weight of 160 kilograms, but can attain 320 kilograms (excluding waste immobilisation materials). The radiological activity corresponds to a contamination at the surface of the wastes. These packages are not thermal packages.

Reference packages B6.2 and B6.3 contain the fuels cladding wastes. The previsional conditioning mode is to put the wastes in stainless steel drums called EIP drums. The packages are made up of either aluminium and stainless steel cladding wastes (B6.2) or magnesium alloy cladding wastes (B6.3). The packages have an average weight less than 300 kilograms (excluding waste immobilisation materials). The packages containing the aluminium and steel cladding wastes have a heat rating on the order of 10 watts ; it will be at most 0.5 watts by around 2025. The packages containing the magnesium alloy cladding wastes are not thermal packages.

The other two reference packages contains the technological wastes made up of either a mix of metal materials and organic substances (B6.4) or only metal materials (B6.5). The previsional conditioning mode is also to put the wastes in EIP drums.

The waste packages containing the metal and organic wastes have an average weight of 90 kilograms (excluding waste immobilisation materials). They are neither thermal nor irradiant packages. A release of hydrogen by the radiolysis of the organic substances is to be taken into account.

The waste packages containing only technological metal wastes are not thermal packages and do not generate gas.

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

Package subclass	Presence of metal wastes	Presence of organic wastes	Production of gas by radiolysis (H_2)	Thermicity, level of irradiation at the packages production date
B6.1	x	None	None	Packages not thermal, low irradiant
B6.2 B6.3	x	None	None	Thermicity zero or medium, level of irradiation medium or high depending on the packages
B6.4 B6.5	x	x	x	Packages not thermal and not irradiant

Table 2.1-3 Summary of the characteristics of the cladding and technological wastes put into drums

- **B7 reference packages**

Reference package B7 groups the REP source rods, as well as sealed sources for industrial usage.

The sealed sources for industrial usage contain radioactive materials of very different natures, activities and half-lives. Several thousand sources underwent between 1972 and 1985, a conditioning in concrete containers, the latter then being reconditioned in metal containers. These so-called « source block » packages are presently stored at the CEA on the Cadarache site (B7.1 reference packages, see Figure 2.1-5). These packages are large packages ; they have a weight varying between 6 and 9.2 tons.

The source rods (B7.2 reference packages) are the operating wastes of the REP reactors, like the various activated metal wastes described by the B1 reference package. As a part of the rods contained in the primary and secondary source groups, they are used to raise the flow level to a threshold detectable by the neutron counting devices during the start-up of the reactors. The primary source rods which contain a capsule of californium are unloaded at the end of the first cycle, while the secondary source rods made up of a mix of antimony-beryllium are subjected to several irradiation cycles before they are discarded. The primary source groups unloaded from the 900 MW reactors underwent a recycling to recover the capsules of californium and, therefore, are not considered to be wastes (as a result, they are not listed in the inventory). The total weight of the wastes to be conditioned is less than two tons.

Figure 2.1-5 Source blocks

For the study, the retained conditioning assumption is, like for the other REP activated wastes, a cutting and then a compacting of the source rods before they are put into a CSD-C reference container. It should be noted that the conditioning of the source rods will amount to at most the production of four CSD-C containers.

Furthermore, several other thousands of sealed sources are today stored in various installations. They cover a very wide range of radioactive isotopes, activities and variable half-lives. For the study, all the sources with half-lives greater than or equal to that of cesium 137 (half-life equal to 30 years) were to be taken into account, according to the domain of wastes accepted in a surface repository at the Aube centre. The conditioning assumption imagined at the present stage is a cementing of the sources in EIP drums (B7.3 reference packages).

- **B8 reference package**

The B8 reference package various types of wastes including radiferous lead sulfates, objects containing radium for medical usage (ORUM) and lightning rods. Whether these wastes will be taken into account in the HLLL inventory remains, however, exploratory.

The radiferous lead sulfates (B8.1 reference packages) originate from the recycling of uranium materials in the Bouchet plant. The wastes were initially put in metal drums ; these latter underwent successive reconditioning for their storage. For the studies, the retained assumption is to retake the primary drums of radiferous lead sulfates for a conditioning in EIP drums. It should be noted that the limitation conditions of the residual empty volumes inside the primary packages were not defined at this stage.

The lightning rods (B8.2 reference packages) are the objects containing either radium or americium. The retained conditioning already applied to a few lightning rods containing radium is a compacting, and then a cementing of the lightning rod heads in 870-litre unalloyed steel containers. The packages contain on the average about 200 radium or americium lightning rod heads and have an activity on the order of 10 gigabecquerels (GBq).

The ORUMs (B8.3 reference packages) are the needles and metal tubes of very small size, with each containing a few milligrams of radium. The radium is incorporated in a solid and insoluble, yet powdery chemical form (sulfate or chloride). The history of the radium industry shows that about a hundred grams of radium were extracted, with about fifty grams being used for the manufacture of ORUM. It should be noted that the ORUMs (a total of 5000 objects) can all be conditioned in a single EIP drum.

2.1.4.2 C waste (vitrified) primary reference packages

The vitrified wastes originate from the recycling of spent fuels. They are for the most part the fission products and the minor actinides (neptunium, americium and curium) created by a nuclear reaction and contained in the spent fuels, which are separated from the uranium and the plutonium during the recycling process. They are calcined and incorporated in a glass matrix. The worked glass is poured at the appropriate temperature into a stainless steel container. The radiological activity is homogeneously distributed in the mass of the vitrified waste.

In France, vitrification was developed in several pilot installations operated by the CEA – its PIVER pilot installation is shut down today – and industrially implemented in three workshops operated by COGEMA : The Marcoule Vitrification Workshop (AVM), started operating in 1978, the R7 and T7 vitrification workshops of « La Hague » entered service, respectively, in 1989 and 1992.

The characteristics of the vitrified wastes and, in particular, their activity and their heat rating depend on several parameters which are : (i) the initial characteristics of the solutions of fission products and minor actinides originating from the fuels recycled in these installations, (ii) the more or less high concentration of the fission products in the glass, (iii) the age of the wastes.

Thus, it was decided to make a distinction between several subclasses of C waste (vitrified) packages grouping, respectively, (i) the productions of older glasses, (ii) the productions of current glasses or glasses planned in the short-term, (iii) the productions of imaginable glasses for the future, including the UOx/MOX glasses and the UOx glasses with plutonium (reference package C3 and C4 respecting).

- **C0 reference packages**

This subclass of packages consists of (i) the glass packages containing the solutions of fission products originating from the recycling of fuels from the « natural uranium graphite gas » reactors (Siceral type of UNGG fuels) and fuels from the Phenix fast neutron reactor in the PIVER installation, (ii) the glass packages containing the solutions of fission products called UMo originating from the UNGG fuels previously recycled on the COGEMA site at « La Hague » and stored today, (iii) the glass packages produced since 1978 in the COGEMA Marcoule Vitrification Workshop (AVM glasses) containing the fission products and actinides originating for the most part from the recycling of UNGG fuels. These latter packages predominate in number.

A distinction is made between these packages according to their chemical contents, the composition of the glass matrix used, their radiological contents, and consecutively their heat rating and the geometry of the containers.

The PIVER waste packages (C0.1 Reference packages) produced between 1969 and 1981, are made up essentially of UNGG glasses. The vitrified waste is conditioned in stainless steel containers having the same diameter but different heights.

The UMo glass packages (C0.2 Reference packages) correspond to the future conditioning of existing solutions of fission products ; these latter originate from the UNGG fuels recycled in the COGEMA's UP2-400 plant of « La Hague ». The chemical nature of the solutions requires the development of a special glass formulation and adaptation of the process's equipment, particularly for the vitrification oven. Following the retained assumptions, the average weight of conditioned waste is 400 kilograms per waste package. These wastes will be conditioned in a stainless steel container identical to the one used today in the COGEMA's R7 and T7 vitrification workshops at « La Hague ». This container called the Standard Vitrified Waste Container (CSD-V) is shown in Figure 2.1-6.

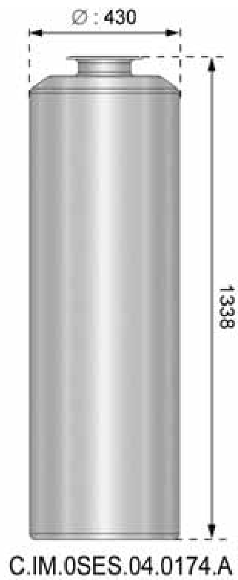


Figure 2.1-6 Standard Vitrified Waste Container (CSD-V)

The AVM glass packages (C0.3 Reference packages) group all the vitrified wastes produced since 1978 in COGEMA's Marcoule Vitrification Workshop. As indicated above, the vitrified solutions originate essentially from the UNGG fuels recycled in the UP1 plant on the site. It should be noted that there are four different glass formulations implemented for one or several vitrification runs. The vitrified waste is conditioned in a stainless steel container.

- **C1 and C2 reference packages**

These packages contain the solutions of fission products originating from the recycling of the REP UOx/URE fuels in the COGEMA plants at « La Hague » conditioned in the form of glass in a CSD-V container (see Figure 2.1-6). The production and the conditioning of the wastes are assumed to take place after an average storage period for the fuels of 8 years after being unloaded from the reactors. The conditioned wastes have on the average a weight of 400 kilograms per package.

The C1 reference package corresponds from a thermal viewpoint to the actual industrial productions. According to the retained assumptions, the vitrified waste consists of a mix of solutions of fission products originating from the fuels UOx1 (average burnup rates of 33 GWd/t), UOx2/URE (average burnup rates of 45GWd/t) and UOx3 (average burnup rates of 55 GWd/t).

The C2 reference package corresponds to packages with a slightly higher heat rating. The vitrified waste consists of a mix of solutions of fission products originating from UOx2/URE and UOx3 fuels, whose average burnup rates are as previously indicated 45 GWd/t and 55 GWd/t, respectively.

- **C3 and C4 reference packages**

These packages correspond to eventually imagined glass productions on the COGEMA site at « La Hague » . In the scenarios retained for the study, these packages were defined under the assumption that the production and the conditioning of the wastes would take place as for the previous glasses after an average storage period of the fuels of 8 years after being unloaded from the reactors. It should be noted that other possibilities may be envisioned.

A first subclass of packages (reference package C3) describes the glasses resulting from the conditioning of solutions of fission products originating from UOx and MOX fuels. They are defined to be a mix of 15 % MOX and 85 % UOx2.

A second subclass of packages (reference package C4) describes the vitrified wastes originating from the recycling of UOx fuels and containing a low additional load of plutonium. The incorporation ratio of the plutonium in the glass is set at one percent per weight, that is, approximately 4 kilograms per package. The incorporated plutonium originates from UOx2 fuels.

2.1.5 The case of spent fuels

The spent fuels considered in the study are the fuels originating from past reactor technologies, research reactors (UNGG, EL4), fuels from national defence activities, as well as fuels from the installed base of PWR reactors. Remember that they are taken into account in the study if they would be considered as wastes assuming a shutdown of the recycling process, which is presently not the strategy retained in France.

Within the scope of scenario S2, the REP fuels represent the largest number of assemblies. The fuel typologies considered (see subsection 2.1.3) are : UOx2 and URE (45 GWd/t), UOx3 (55 GWd/t) and MOX (48 GWd/t).

A conditioning of the assemblies was studied (see subsection 4.3 of « architecture and management of a geological repository » [22]). This conditioning takes care of the bare assemblies or those cladded.

- **CU1 and CU2 reference packages**

The reference fuel assembly corresponds to an « advanced 2nd generation » FRAGEMA design assembly with thickened tube guides and cladding made of a zirconium alloy. It is shown in Figure 2.1-7. It is designated AFA-2GE for the REP 900 MWe reactors and AFA-2LE for the REP 1300 MWe and 1450 MWe reactors.

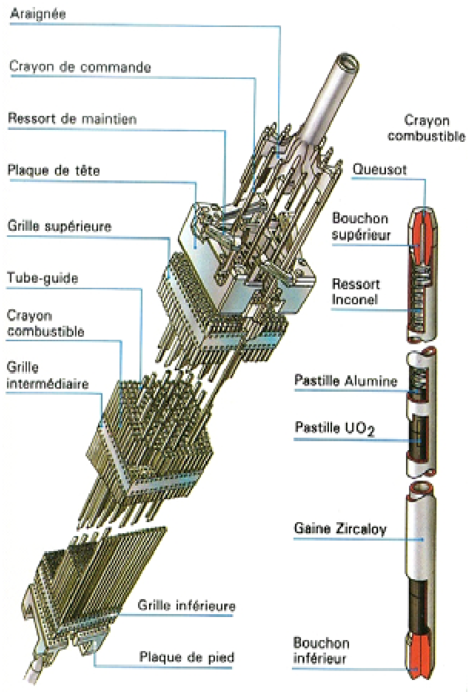


Figure 2.1-7 REP fuel assembly

The assembly consists of a rigid metal structure (skeleton) which supports 264 identically shaped rods distributed in a square network of 289 slots.

The fuel rod is made up of :

- A cylindrical metal cladding closed in its upper and lower parts by two welded plugs (the upper plug has a tube to empty the rod's atmosphere which is then closed after the rod is filled with helium) ;
- A stack of fuel capsules over approximately 95 % of the rod's height ;
- A coil spring in the upper part of the fuel capsule stack to ensure the stack is held axially during handling operations.

The CU1 reference packages correspond to the enriched uranium oxide fuels (UOx) or to the enriched recycled uranium oxide fuels (URE). The CU2 reference packages correspond to the mixed fuels consisting of uranium and plutonium oxides (MOX).

- **CU3 reference package**

These fuels have rather varied natures. They consist of (i) fuels originating from UNGG reactors, (ii) fuels from the EL4 heavy-water moderated and cooled reactor, (iii) fuel elements from the Célestin reactors installed at Marcoule and (iv) fuels from the reactors of nuclear propulsion systems installed on the ground or aboard ships.

The UNGG fuels correspond to a residual tonnage of unrecycled fuels of approximately 15 tons. They are conditioned in cylindrical claddings having a diameter of 88 mm or 130 mm and a height equal to 655.5 mm.

The EL4 fuels represent approximately 50 tons of heavy metal. The fuel element comes in the form of a group consisting of 19 rods enclosed in a structure made up of ATR (zirconium alloy with copper and molybdenum).

These first two fuel types comprise the CU3.1 reference package.

The Célestin fuel elements (CU3.2 reference package) consist of metal plates containing enriched uranium mounted on a metal structure. They are conditioned in stainless steel claddings having a diameter of approximately 340 mm and a length of 1100 mm.

The nuclear propulsion system fuels (CU3.3 reference packages divided into CU3.3.1 to CU3.3.6) consist of (i) oxide fuels based on calcined uranium oxide plates, (ii) metal fuels based on highly enriched metal uranium. The latter fuels are no longer used.

In both cases, the fuel comes in the form of an assembly consisting of several bundles. The latter are separated from the assembly and conditioned in claddings of the same diameter (approximately 340 mm like the claddings containing the Célestin fuel elements) but of variable lengths adapted to the bundle dimensions.

2.1.6 Number of primary packages and their volumes taken into account

Within the context given by the scenarios previously presented (see subsection 2.1.3), the quantification of the number of reference packages relies on the inventories and the waste production previsions established by the producers and assessed by Andra on the basis of the data produced.

For the wastes to be produced, excluding the recycling of spent fuels, sizing margins were added by Andra in order to take into account uncertainties. It should be noted that the eventual possibilities of disposal of certain waste packages under other disposal solutions were not considered to ensure the availability of prudent estimations.

For the past productions, the inventories are based on the data established by the producers. As for the recycling wastes, their inventories are deducted from the electricity producing assumption of the installed base.

The number of packages and their volumes taken into account in the studies for the B wastes are given in Table 2.1-4 and Table 2.1-5.

The number of C waste reference packages and their volumes are given, respectively, in Table 2.1-6.

The quantitative data related to the spent fuels for PWR reactors (CU1 and CU2) are given in Table 2.1-7. The CU3 reference packages total 5 810 claddings.

The volumes indicated in these tables correspond to the volumes of conditioned wastes according to the assumptions 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 of B2		**104 990**	**36 060**	**104 990**	**36 060**	**104 990**	**36 060**	**104 990**	**36 060**
B3	CEA	15 060	13 370	15 060	13 370	15 060	13 370	15 060	13 370
	COGEMA « La Hague »	9 890	10 470	9 890	10 470	9 890	10 470	7 340	7 750
	COGEMA Marcoule	7990	3420	7990	3420	7990	3420	7990	3420
Total of B3		**32 940**	**27 260**	**32 940**	**27 260**	**32 940**	**27 260**	**30 390**	**24 540**
B4	**COGEMA « La Hague »**	**1 520**	**2 730**	**1 520**	**2 730**	**1 520**	**2 730**	**1 520**	**2 730**
B5	**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 Global quantitative data in numbers and volumes of packages for the B waste reference packages

Reference packages	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 Details on the numbers and volumes of the reference packages B2, B3, B5, B6, B7 and B8

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)
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 Global quantitative data in numbers and volumes of packages for the C waste reference packages

	Production sites	Number of REP fuel assemblies			
		Scenario S1a	Scenario S1b	Scenario S1c	Scenario S2
CU1 reference « short » AFA-2GE UOx assembly	EDF	0	0	0	27 200
CU1 reference « long » AFA-2GE UOx assembly		0	0	0	26 800
Total of CU1 reference UOx assemblies		**0**	**0**	**0**	**54 000**
CU2 reference « short » AFA-2GE MOX assembly	EDF	0	5 400	5 400	4 000
Total of CU2 reference MOX assemblies		**0**	**5 400**	**5 400**	**4 000**

Table 2.1-7 Number of REP fuel assemblies

2.1.7 Radiological inventory

The radiological inventory of packages concerns a large list of radionuclides, including fission products (PF), activation products (PA) and actinides. It is described in [18]. Those radionuclides divide into three classes :

- 44 short-lived radioluclides (31 %), with a half life inferior to 6 years,
- 16 intermediate-lived radionuclides (11 %), with a half life between 7 and 30 years,
- 84 long-lived radionuclides (58 %), with a half life[10] superior to 31 years.

10 Nickel 63 is a particular case due to its half life (100 years). It is an activation product present in a lot of packages. Its activity during the first centuries is especially high, or even dominating in B waste (B1, B4, B5 except B5.4 and B6.2)

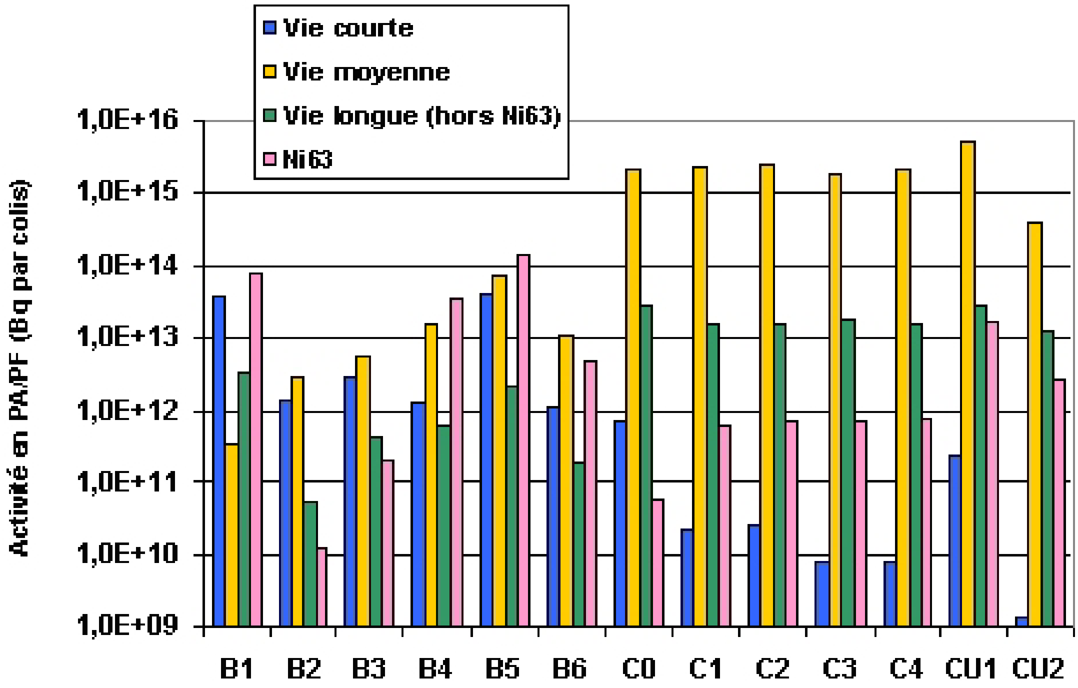


Figure 2.1-8 AP (activation products)/FP (fission products) activity of the various reference packages

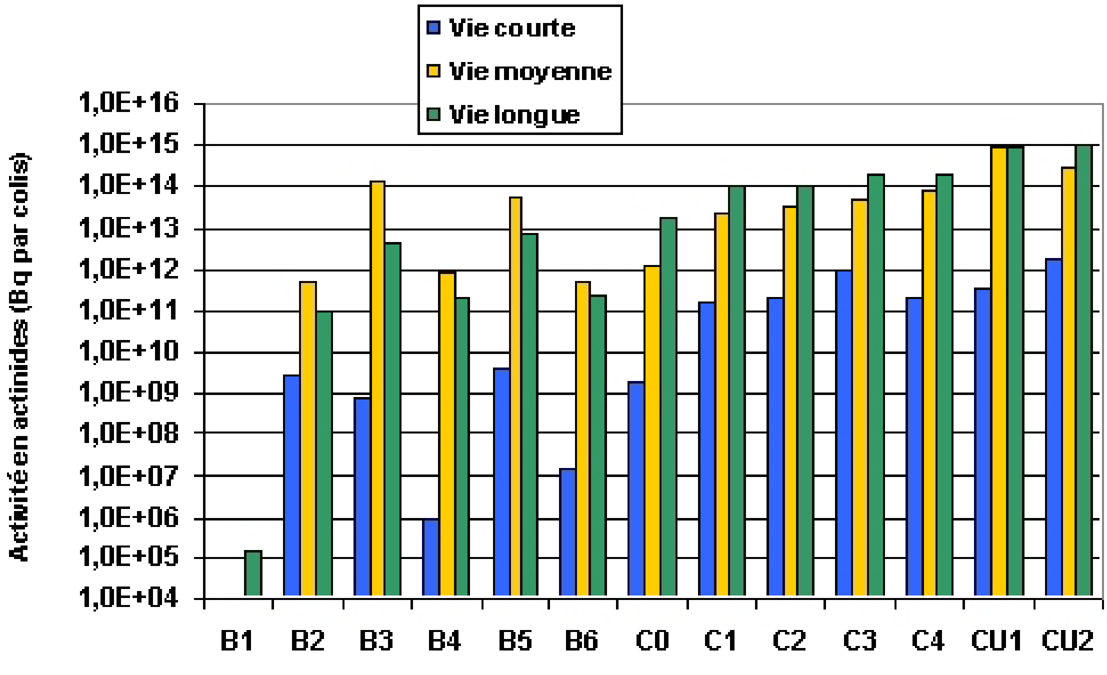


Figure 2.1-9 Actinide activity of the various reference packages

2.2 The geological context of the Meuse / Haute Marne site

This subsection is a summary of elements contained in the volume titled « phenomenological evolution of the geological repository » [7]. The reader looking for a more detailed description of the geological context should refer to this volumes or to the site reference document [17]. This section introduces synthetically the geological context of the Meuse / Haute-Marne site.

2.2.1 Overview of the studied sector

Geologically speaking, the Meuse / Haute Marne site is a part of the eastern border of the Paris Basin ; this basin forms a depression whose low point corresponds to the Paris region (Ile de France). In the studied zone, the Paris Basin is composed of an alternation of sedimentary strata with a clay dominance and limestone strata ; these strata were deposited between 250 million and 135 million years ago.

In the detailed figure (see Figure 2.2-1), the sedimentary series which this study is most particularly interested in is composed of in a bottom-up scheme (and from east to west in the outcroppings) :

- The limestone formation of the Dogger laying on the marls and binding clays,
- The clay formation of the Callovo-Oxfordian,
- The limestone formation of the middle to upper Oxfordian,
- The Marly Kimmeridgian,
- The outcropping at the site, the Tithonian limestones (called the Barrois limestones),
- A few cretaceous clayey-sandy superficial deposits of little thickness, crowning the highest points of the topography.

These sedimentary strata have a simple, monoclinal structure with a low regular dip from 1 to 1.5 degrees toward the northwest.

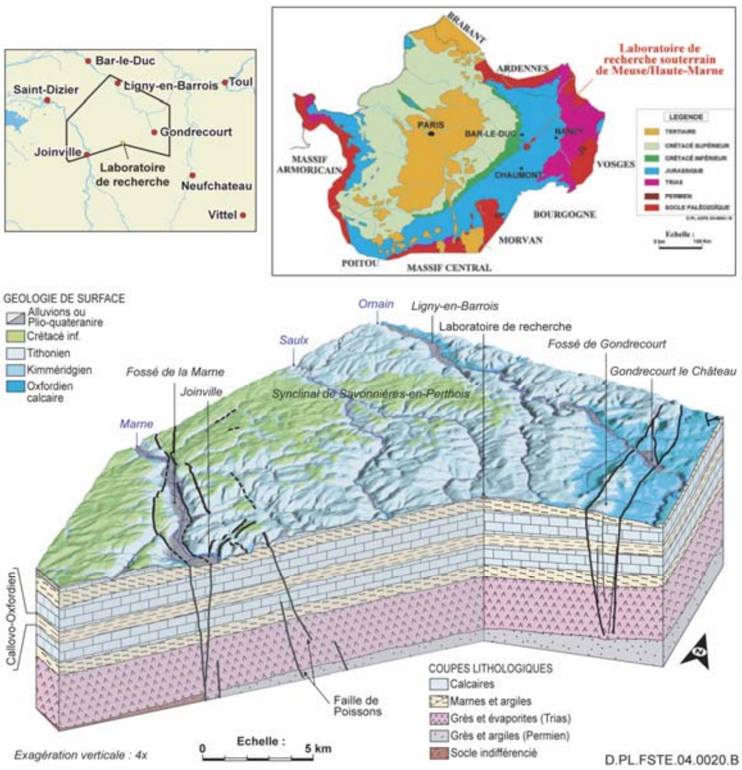


Figure 2.2-1 3D geological block diagram of the Meuse / Haute-Marne site

2.2.2 The host formation

Within the sedimentary series, the clay formation studied is that of the argillites of the Callovo-Oxfordian. This stratum was formed by the sedimentation of detrital materials (clay minerals up to 60 % in weight, fine quartz) and carbonates in a relatively calm sea. It has a wide geographic extension.

In a zone of approximately 200 square kilometres defined from the underground research laboratory, the Callovo-Oxfordian formation has physical and chemical characteristics tranposable or similar to those observed on the laboratory site ; this zone is called a transposition zone.

In this zone, the Callovo-Oxfordian is a homogeneous, hardly permeable stratum with its top located at a depth varying from 420 metres (corresponding to the laboratory site) to more than 600 metres in the direction of the dip and its thickness also progressively varying from 130 metres to the South to 160 metres to the North of the zone.

Vertically, the proportions of the main mineralogical phases vary and are structured into three sedimentary sequences[11]. The upper sequence is characterised by a higher carbonate content.

The Callovo-Oxfordian is surrounded by two large limestone formations, the underlying Dogger and the overlying calcareous Oxfordian, within which are located the porous sedimentary horizons where water flows. However, these horizons have a low permeability and are not aquifers in the sense of hydrogeology (they do not represent a water resource).

The transposition zone is located apart from the regional faults like the fault of the Marl to the Southwest. In particular, no fault was observed in the Callovo-Oxfordian and the overlying horizons.

2.2.2.1 Evolution of the formation since its deposit

Between the end of its deposit and the end of the Jurassic (150 to 135 million years ago), the argillite stratum was covered by at least 500 metres of sediments. It then underwent various successive processes called diagenetic processes. The first of these processes is the progressive compaction of the

[11] These sequences disclose low cyclic variations in the level of the sea at the moment when the stratum was deposited.

sediments with an expulsion of water under the weight of the subsequent deposits. Then physical and chemical changes (dissolutions and mineral precipitations) took place driven by pressure and temperature increases related to the burying, as well as by the circulation of fluids. These processes had hardly any impact on the stratum during the period because of the depths reached at the end of the Jurassic (5 to 600 metres).

Thereafter, the major events of geologic history from the end of the Cretaceous period (withdrawal of the sea) and then the Tertiary period (pyrenean phase, oligocene distension and then alpine phase) had hardly any effects on the formation. Its present characteristics were acquired for the most part during the Cretaceous period as suggests the age dating of the last diagenetic episode (earlier than 100 million years ago). This episode of low intensity concerned some exchanges between clays and carbonates in very small rock volumes. The initial phases and, in particular, the clay minerals were not affected by the major changes of their crystallochemistry and their texture.

The Callovo-Oxfordian appears, therefore, to be a geological formation formed 155 million years ago in a stable context. The mineralogical transformations posterior to the deposition of the stratum are space limited and precocious processes – anterior to 100 million years ago – affecting essentially the carbonated cements. The moderate thermal evolution of the stratum during its history did not allow mineralogical transformations of the clay phases to initiate (illitisation, in particular).

2.2.2.2 Textural properties and porosity

The porous volume is conditioned by the arrangement of the three main mineral phases :

- The clay phases (majority – at most 60 %) come in the form of clay mineral aggregates (more or less regularly shaped plates) having a length from several tens to one hundred micrometres and a thickness of several tens of nanometres. These aggregates act like a matrix and are roughly organised according to the stratification ;
- The quartzous detrital elements of small sizes (5 to 10 micrometres) are homogeneously distributed in this matrix ;
- The calcareous phase comes in the form of variable sized elements, which are sometimes rather large (up to millimetres), corresponding to the bioclasts and automorphic crystals resulting from secondary neoformations subsequent to diagenetic phenomena.

Because of the relative proportions of the clay phases and the other elements, the latter are not contiguous and appear dispersed within the matrix.

This arrangement determines two types of porosities in the rock :

- The internal porosity of the clay matrix (around the aggregates and the clay particles), which concerns pores less than one micrometre, corresponding to meso-porosity and microporosity (90 % of the porous volume) ;
- The porosity at the interface between the clay matrix and the quartzous particles and the bioclasts, which concerns pores greater than one micrometre and pores less than one micrometre for the neoformed carbonates, corresponding to macroporosity (10 % of the porous volume).

The thus-defined total porous volume is on the order of 18 % of the rock's total volume. This porosity induces a major tortuosity linked to the inter-arrangement of the clay aggregates.

A schematic representation of the Callovo-Oxfordian's texture and porosity is given in Figure 2.2-2.

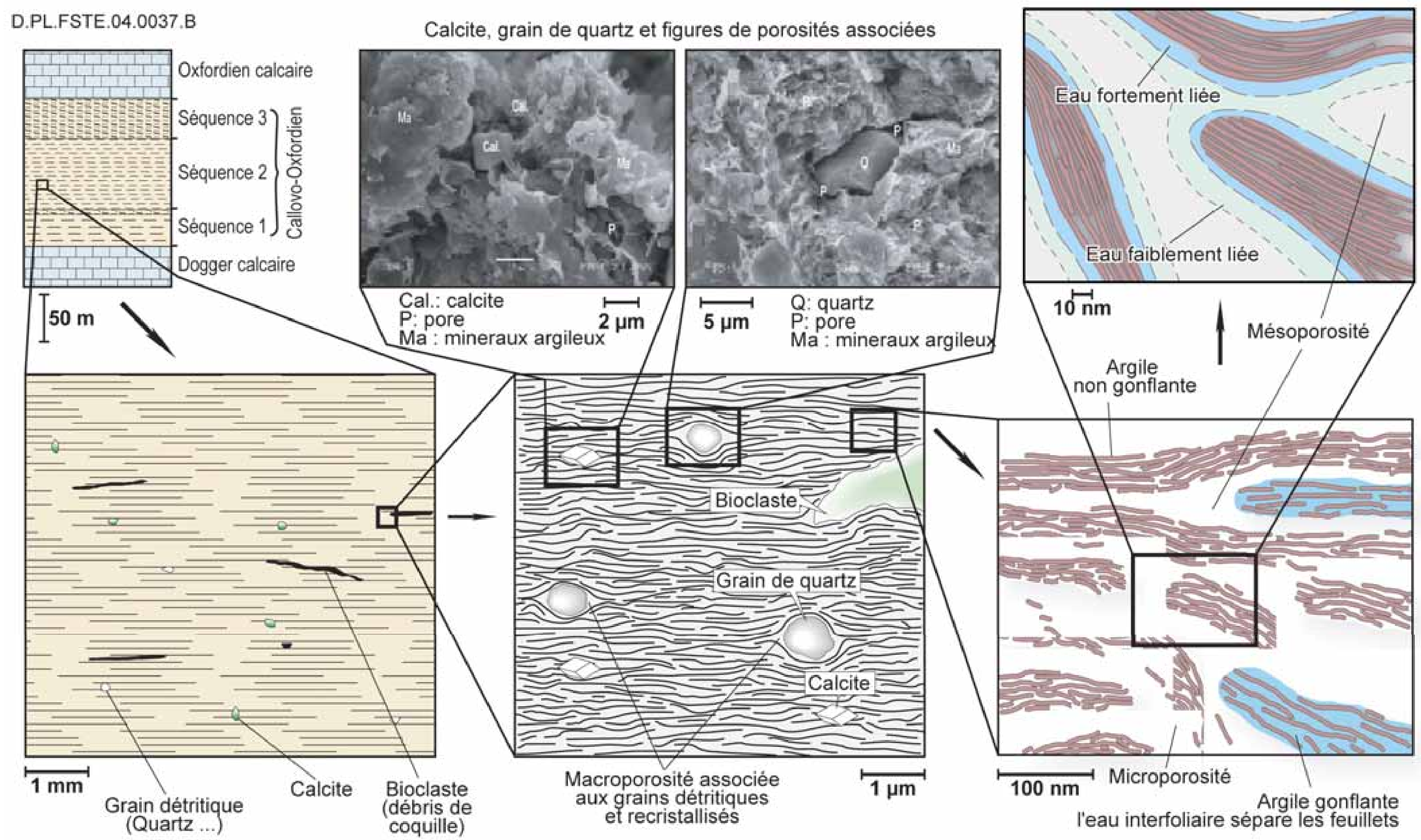


Figure 2.2-2 Schematisation of the Callovo-Oxfordian's texture and porosity

2.2.2.3 The status of water and water flows in the Callovo-Oxfordian

Generally, water is structured (i.e. is organised in a network) in the vicinity of mineral surfaces and, in particular, nearby those with a residual electric charge. In a porous medium, water is defined either as a « bound » water and therefore not movable or a « free » water not affected by this effect.

The clay particles carry a permanent electric charge and therefore have a major effect on the structuration of the water in their vicinity, contrary to quartz or the carbonates, which carry lower permanent surface charges.

Around the clay particles a bound water film forms, which can stop up some pores (« electrochemical bottleneck »). For a sediment such as the Callovo-Oxfordian argillites, the pores clogged by this « electrochemical bottleneck » have a size on the order of 10 nanometres. Since the rock's porosity is mainly distributed in the clay matrix, the proportion of bound water in contact with the particles is high. It corresponds to approximately 50 % of the water contained in the whole porosity. This bound water does not participate in the water flows within the rock.

These textural properties confer to the Callovo-Oxfordian argillites a low permeability over its entire thickness.

Water flow rates are quite different when looked at closely because of the pore structure of the medium. Similarly, some pores are not connected and cannot participate in this water flow. Also, a so-called kinematic porosity is defined, which is a fraction of the total porosity, allowing macroscopically to calculate an average rate for the water in the direction of the charge gradient. For the Callovo-Oxfordian, this kinematic porosity corresponds to the fraction of free water in the rock, that is, approximately 9 %, corresponding to half of the total porosity.

2.2.2.4 Chemical composition of the interstitial water in the Callovo-Oxfordian.

The formation's low water content, a weighted 7 to 8 %, as well as its low permeability, do not allow for direct analysis of the composition of the pore waters. This composition was determined from analyses on Callovo-Oxfordian samples and geochemical modellings. This composition is considered homogeneous over the formation's entire thickness, in consistency with its global lithological homogeneity. The compositions of the waters thus assessed have a neutral pH (slightly higher than 7). They are reducers as the presence of pyrite and organic substances confirms. Chemically speaking, the waters have a calcareous calco-soda character. Analyses of the water's stable isotopes confirmed by those carried out on the dissolved gases (H_2O and CO_2) show that the water is clearly of a meteoric origin. This meteoric water, therefore, replaced the initial sedimentation water during very old processes. On the other hand, the data on the stable carbon isotopes demonstrate the chemical and isotopic equilibrium of the interstitial waters with the calcareous phases. The waters are also sulphated and chlorinated (the chlorinity being determined analytically). The other elements (major cations and silicon) are checked by the various mineral phases.

The concentrations of some mobile specimens (chlorine, in particular) vary over the stratum's thickness (between 1000 and 500 mg/l from bottom to top for Cl plumb with the site), materialised by an exchange of these specimens between the Callovo-Oxfordian and the surrounding formations. At the sector scale, these profiles show :

- A control of the salinity in the Callovo-Oxfordian by the Dogger's salinity (significantly much higher to the Northeast than to the Southwest) ;
- Low salinities in the calcareous Oxfordian compared to those of the Callovo-Oxfordian and the Dogger.

2.2.3 The surrounding formations of the Callovo-Oxfordian

The two surrounding calcareous formations immediately underlying and overlying the Callovo-Oxfordian, the Dogger and the calcareous Oxfordian are limestone structures which impose conditions at the limits, and particularly hydraulic and chemical conditions, of the stratum of the Callovo-Oxfordian.

2.2.3.1 The Dogger

The Dogger (see Figure 2.2-3) laying on the Lias clays is a 300 metre thick formation. It corresponds in the Paris Basin scale to the development of large calcareous platforms under a very thin outcrop of water in a tropical environment. These conditions are responsible for the wide variety of facies observed (shelly debris, calcareous sands, reefs, etc.), which are deposited at the mercy of the streams and the evolutions of the water depth. These rapid variations are found again laterally and vertically. A more clayey interval found everywhere in the East of France (« Longwy Marls ») and separating the two limestone structures of Bajocian and Bathonian mark a relative deepening of the deposition conditions. At the top of the formation, the nacreous Slab Dalle which is very rich in shelly debris marks the passage between the Dogger and the Callovo-Oxfordian.

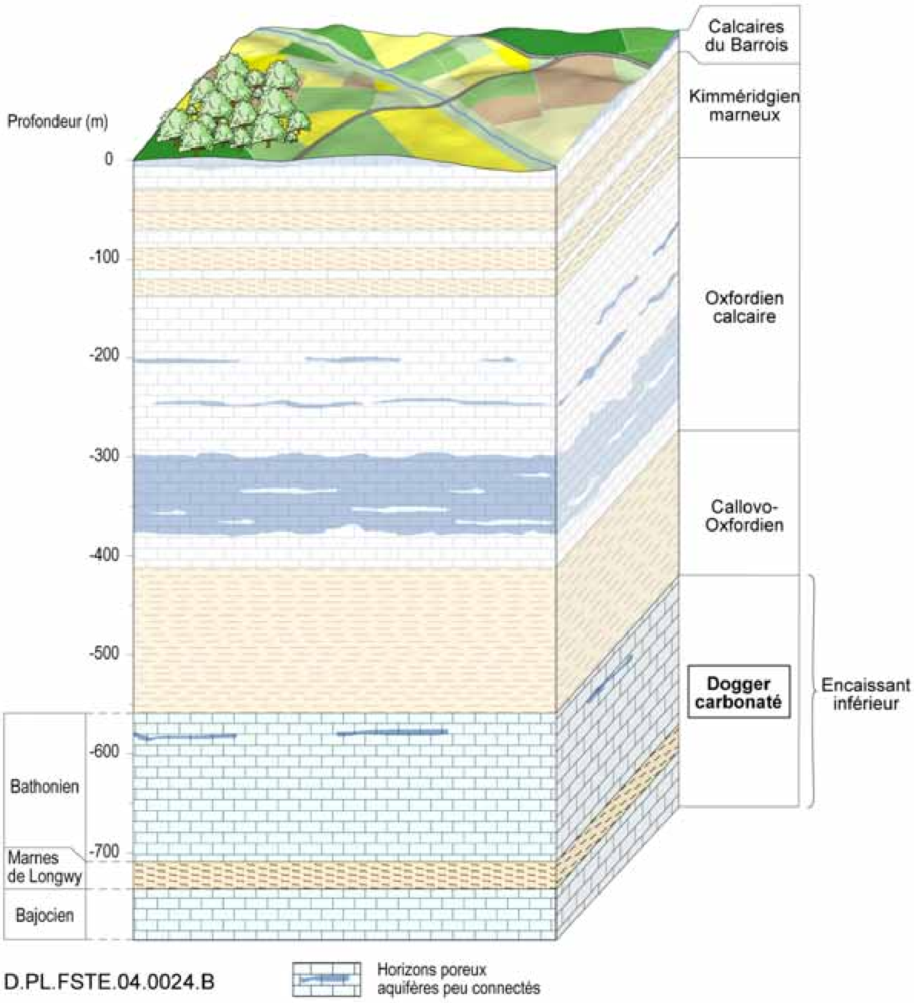


Figure 2.2-3 Schematic representation of the lithologies and the porous horizons in the calcareous Dogger on the Meuse / Haute-Marne site

Just after its deposition, the Dogger underwent a number of diagenetic processes which checked the successive phases of dissolutions and carbonate recrystallisations. These conventional processes for the calcareous formations took place in a continuous evolution :

- The first phases were precocious phases and occurred during and just after the sediment deposition ;
- The subsequent phases were related to the burying of the series ;
- Finally, the last phases involved the denudation of the sedimentary series and their return to the outcropping.

The Dogger has low porosities (6 to 10 %) and low global permeabilities (from 10^{-10} to 10^{-9} m/s). Only a few multi-kilometric extension areas isolated in the compact limestones and independent from the facies have porosities and permeabilities significantly higher (respectively, 15 % and 10^{-8} m/s). The productivities of these porous horizons are low, that is, on the order of a litre per minute, and even less. Deep water circulations are mainly found within these latter.

The Dogger waters are plumb with the site in chemical equilibrium with the formation. These waters are highly mineralised waters, that is, chlorinated – sulphated – soda containing waters, with a neutral pH (slightly higher than 7) and are reducers ; they are chemically close to those observed in the same formation elsewhere in the basin. These waters of a meteoric origin (see above) reveal long residence times (on the order of a million years in the bore-hole MSE101), as well as indicate the approaches made from rare gases.

At the Paris Basin scale, the successive re-clearances of the regional faults (Bray fault, in particular) seem to have facilitated the vertical transfers between aquifers, as well as suggest various approaches made starting from a modelling and geochemical and isotopic tracers.

At the scale of the Paris Basin's eastern aureole, the approaches made, in particular, from the helium isotopes reveal that the Lias clays located under the Dogger and separating it from the Triassic effectively isolate both of these formations. In the modellings, this means not taking into account the formations underlying the Dogger. In this context, the Triassic isotopic signatures refound in the filling calcites of fractures in the vicinity of the accidents do appear to be a very localised phenomenon and corresponding to very limited communications over time. This is illustrated by the existing salinity differences between the aquifers, that is, from about a hundred g/l for the Triassic to a few g/l for the Dogger. The Dogger salinities remain, nonetheless, high for the meteoric waters : they seem to be acquired by a diffusion of Triassic salt through the Lias clay. Therefore, these exchanges took place over long time scales of several million years.

2.2.3.2 The calcareous Oxfordian

Above the Callovo-Oxfordian, the calcareous Oxfordian (see Figure 2.2-4) reveals deposition conditions quite similar to those of the Dogger, that is, the superposition of various calcareous platforms over a thickness of approximately 300 metres. However, the deposit environments are significantly more contrasted than for the Dogger. Instead of a very widespread platform at the water level, clearly distinct environments can be distinguished for each of the platforms of the calcareous Oxfordian. Thus, the largest facies corresponding to small outcroppings of water at the moment of the deposition (clay sands, lagoon facies, reefs, etc.) are globally located in the Northeast of the sector (Commercy region), while in the Southwest beyond Joinville more clayey basin facies are developed.

All took place under a cover by a large-scale diagenesis reflecting that affecting the Dogger. Like for the latter, this diagenesis was responsible for a more intense recrystallisation at the sector level than in the other locations of the Paris Basin and, in particular, in the granular facies. Hence, the global permeabilities over the entire formation are on the order of 10^{-9} m/s, that is, two orders of magnitude less than those measured closer to the centre of the basin. However, some levels of the calcareous Oxfordian corresponding for the most part to fine facies reveal higher porosities and permeabilities than in the rest of the formation. These so-called « porous » horizons are defined plumb with the underground laboratory and identified by a numbering from 1 to 7, that is, Hp1 to Hp7. They correspond mainly to some well-defined homogeneous facies (calcareous lagoon sludges, inter-reef facies, etc.) for which a secondary « chalky » diagenesis transformed the primary facies into a homogeneous fine micrite revealing a high porosity with respect to the rest of the formation. The organisation of these porous levels is therefore directly linked to the initial sediments deposition conditions, which explains their global geometry and their correlation with the stratification, as shown by the increase in 3D earthquake data recorded on the site. At this scale, these data illustrate clearly that the distribution of the most porous zones can, however, become irregular, which explains the problems of making detail correlations without significantly changing the global hydrodynamic characteristics of the formation.

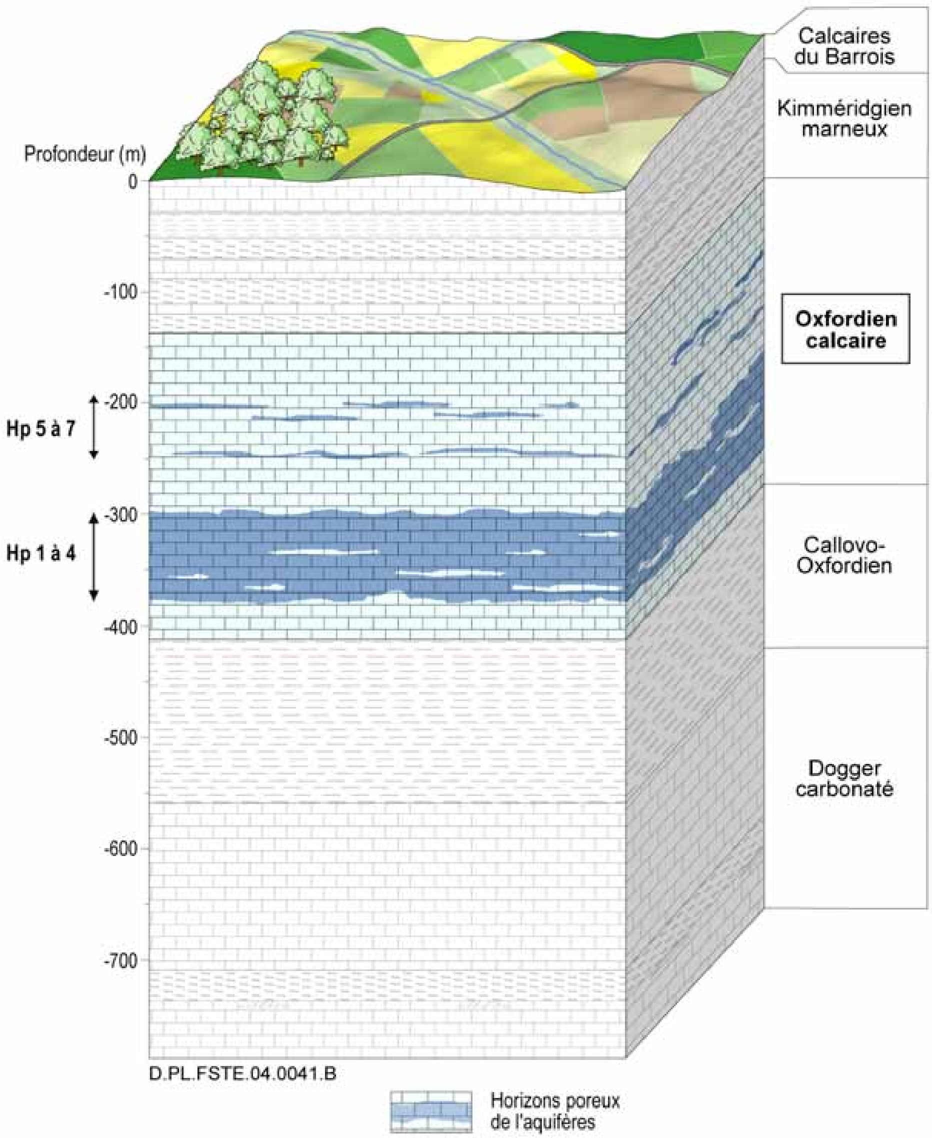


Figure 2.2-4 Representation of the porous levels in the calcareous Oxfordian at the present state

We can see above the facies corresponding to the emplacement of the first calcareous platform, the following horizons with very low permeables present over the entire sector (facies C3a/C3b : permeability on the order of 1 x 10^{-11} m/s) :

- A group of porous horizons (Hp1 to Hp4) recognised in the whole Northwestern part of the sector and corresponding to a great extent to the lagoon facies. These horizons reveal porosities on the order of 20 % and permeabilities of a few 10^{-8} m/s. They are individualized within the more compact limestones (porosities of 15 % and permeabilities of 1 x 10^{-9} m/s). A few kilometres to the South and West of the site the results of the bore-holes EST321 and EST351 reveal that the transition of these lagoon facies to recrystallised barrier facies and then above to the basin clay facies (Joinville region) is evidenced by the disappearance of these porous horizons ;

- In the Upper Oxfordian, porous horizons (Hp5 to Hp7) thinner than Hp1 to Hp4 are systematically present, although they are not continuous at the sector's scale.

It is mainly within the porous horizons that deep water circulation takes place. Like for the Dogger, the regional water flows are globally directed from the supply zones, which the plateau zones to the South and East of the site make up and on which the formation is karstified, to the centre of the basin. However, at right angles to the site, because of the proximity of the outcroppings, a part of these water flows are directed toward a local low point in the stream of the Marne to the Southwest. This low point represents for the trajectories passing plumb with the Meuse / Haute-Marne site the only present outlet of the Oxfordian limestone on the sector.

One of the consequences of the water flow within the porous horizons clearly differentiated in the calcareous Oxfordian is a sharp chemical individualisation plumb with the site of the waters between the lower porous horizons and the upper porous horizons :

- The chemical facies of the waters of the lower horizons is carbonated soda ; they are reducers with a neutral pH (7.5) ;
- The facies of the waters of the upper horizons is carbonated magnesia with also a pH of 7.5, but they reveal a less reducing and even slightly oxidising character.

These differences are linked to the slightly different geologic history of these two assemblages and, in particular, with respect to the deposition environments, the palaeocirculations and the diagenetic processes.

2.2.3.3 The Kimmeridgian marls and the Barrois limestones

Above the calcareous Oxfordian, the Kimmeridgian marls mark a new deepening of the deposition conditions : the sediments become again more clayey. This 110 metre thick formation plumb with the site is made up of marls with interstratifications of clayey limestones.

The deposition conditions reflecting the Callovo-Oxfordian explain the great constancy of the facies and the thickness of this stratum at the regional scale. When it is at the outcropping, its permeability initially low (on the order of 10^{-11} to 10^{-12} m/s) increases due to the surface alterations affecting specifically the calcareous intervals.

The interstitial waters of the Kimmeridgian under cover reveal a facies rather close to that of the upper horizons of the calcareous Oxfordian : reducing waters of the magnesia bicarbonated facies with a pH of 7.5. At the outcropping (or under cover for the part in contact with the aquifer of the Barrois limestones), the formation is subjected to an oxidising environment which deeply changes this facies (in particular, pyrite oxidation, with sulfates being produced and an increased iron concentration in the solution) and the waters naturally become oxidising waters.

The Barrois limestones, which are at the outcropping over a large part of the sector, correspond to the scale of the Paris Basin with a return from the large calcareous platforms to limestone sludges. The formation is more than 200 metres thick when complete. The initial nature of the limestone (clayey limestones more or less vesicular) and its evolution at the outcropping under the action of the meteoric waters separates this formation into two units with clearly distinct behaviours :

- A base unit (representative of the laboratory's surroundings) with a thirty-some metre thickness in which karstic developments are limited ;
- An upper unit in which the karsts are frequent. They are that much more developed when the argiloarenaceous Cretaceous is preserved above the formation.

Regardless of the unit considered, the runoffs in this surface aquifer fluctuate rapidly with rainfall. The transfers globally in the Northwest direction are rapid (several hundreds to thousands of metres a day). Because of the rapid reloading of the groundwater by the meteoric waters, the formation waters are oxidising and reveal a calcium bicarbonated facies more or less chlorinated.

2.2.4 Geo-prospective evolution

The long-term evolution of the geological medium like its past evolution at the geological time scale result, on the one hand, from the ground surface climate and, on the other hand, the internal geo-dynamic evolution of the plates forming the earth's crust.

2.2.4.1 Climatic evolution

Since the beginning of the quaternary period, climatic cycles have succeeded in a fluctuating fashion according to astronomical parameters with an alternation of glacial and interglacial ages. Periodically the surface soils are durably frozen down to a significant depth (permafrost) on the Meuse / Haute-Marne site (40 to 50 % of the time during the last 130 000 years). The frost penetrates to a depth of about a hundred metres. The deeper Callovo-Oxfordian formation is not directly affected by the frost. Notwithstanding the influence of the greenhouse effect, which could slow down this evolution, a permafrost could appear in about a hundred thousand years.

These climatic cycles have as a consequence a periodic resumption of surface erosion. The main erosive phenomena are the cutting of valleys and the raising of limestone plateaus, which change surface water runoffs by the evolution of the karstic networks and possible captures of rivers.

These phenomena have left their marks in the countryside (alluvial terraces, for example), which can be used to estimate their rate. The laboratory site is located on a plateau zone separated from major valleys located at the head of a secondary hydrographic network, where erosion is less rapid. A progressive disappearance of the Barrois limestones is possible beyond 500 000 years.

2.2.4.2 Long-term geodynamic stability

The only imagined tectonic movements are limited to the regional faults (the Marne's faults to the West, the Gondrecourt-le-Château trough to the Southeast). Except for these zones, no deformation of the geological strata seems imaginable. The great geodynamic stability of the region explains the practically no-earthquake character of the sector at the scale of historic times.

The actual convergence between the African plate and the European plate is low, only a few millimetres to a centimetre a year. The major part of this movement is absorbed by the deformation being produced in the mountainous zones (Maghreb and Alps at the latitude of France). The rest of the movements are absorbed by landslides along the major faults of the plate : this is particularly the case of the West European rift valley. Except for these zones, possible movements are very low and concern only the regional faults affecting the base (Marne and Vittel faults, for example). On these faults, the possible landslide is extremely slow and is estimated at between 0.001 and 0.0001 mm/year according to the fault's size.

These considerations explain why the Paris Basin is a region with very low earthquake activity except for the active zones located on its edges (Rhine trough, in particular).

In this intra-plate domain, the Meuse / Haute-Marne sector appears to be a remarkable zone of the West European plate. The lithosphere has here a uniform thickness compared to the more eastern regions and, in particular, the Rhine trough (see Figure 2.2-5). This uniformity is characteristic of stable and hardly deformed zones. The zone appears to have had no earthquake activity in terms of a historical earthquake record, as well as recent geological periods : in fact, there are no quaternary signs of tectonic activity on the major faults surrounding the sector. Thus, the sector corresponds to a very stable zone from the viewpoint of earthquake-tectonic activity. The maximum possible earthquakes in this context are on the order of 6.1 + - 0.4 on the regional faults surrounding the sector (Gondrecourt fault, Marne's faults), with return periods between 100,000 and 1,000,000 years.

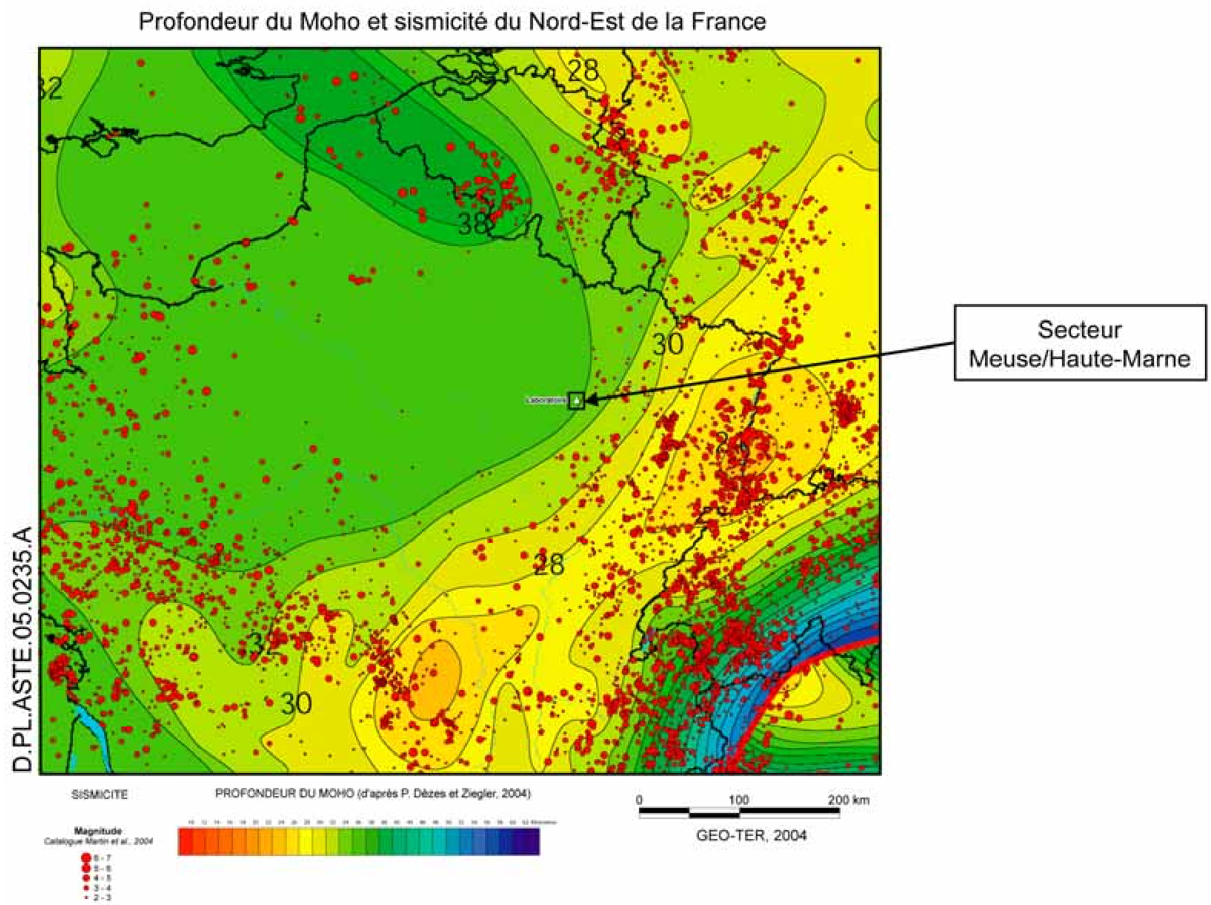


Figure 2.2-5 Geodynamic context of Eastern France : connection between earthquake and earth's crust thickness

For the design of structures in compliance with regulations, the « safety margin computed earthquake » (SMS) can be defined for the repository's operating period. The magnitude of this earthquake (magnitude of 6 at 25 km from the site) is determined on a « conservative » basis by adding to it the strongest historically known earthquake in the zone and by moving the centre from its place of occurrence as close as possible to the site. This earthquake is characterised by a maximum acceleration to the repository's depth of approximately 0.15 g (that is, 1.5 m/s^2) and on the surface of approximately 0.2 g (that is, 2 m/s^2), a difference due to the amplification of the movement during the propagation of the earthquake waves on the surface.

To assess the impact of a very long-term earthquake, we determine what would be the maximum physically possible earthquake (SMPP) for the configuration of the faults closest to the site by assuming them active (length, rooting depth, segmentation). We retained for this earthquake a magnitude of 6.1 at 6 kilometres from the laboratory site and at a depth of approximately 12 kilometres. Such an earthquake would induce a maximum acceleration at a repository's depth of approximately 0.3 g.

Although the characteristics of the SMS and SMPP are established and considered to be overevaluated, they remain of a moderate amplitude by comparison with those of the earthquakes actually occurring around the periphery of the Mediterranean sea.

2.3 Descriptive elements of the repository's architecture

The description of the repository architectures and their justification regarding the expected safety functions is covered in chapter 3. We will merely give here a general description of the architectures in order to introduce elements to facilitate understanding and a useful vocabulary for what follows without any particular justification. This subsection is a summary of a part of chapter 2 of the volume titled « architecture and management of a geological repository » [22].

A repository installation would be made up of cells (of underground cavities) sunk in the argillite formation and containing the waste disposal packages. These latter consist of primary waste packages such as conditioned by the waste producers completed by an over-pack according to the disposal needs.

The studied architecture groups the disposal cells of the various waste categories within specific repository zones (see Figure 2.3-1). The repository zones of B waste, C waste and, if necessary, spent fuels are thus physically separated from one another. This arrangement is imagined in order to offer an independence in terms of (i) management of the various wastes (ii) and behaviour of each zone in terms of the specific characteristics of the wastes which it contains.

For the construction of the cells, the disposal of the wastes and the reversible management of the installations, access is carried out by vertical shafts between the surface and the repository level, and then by connecting drifts between the shafts and the repository zones. By limiting the number of these structures, four vertical shafts ensure all the functions presented related to operation ; these shafts can be common to all the wastes in the repository.

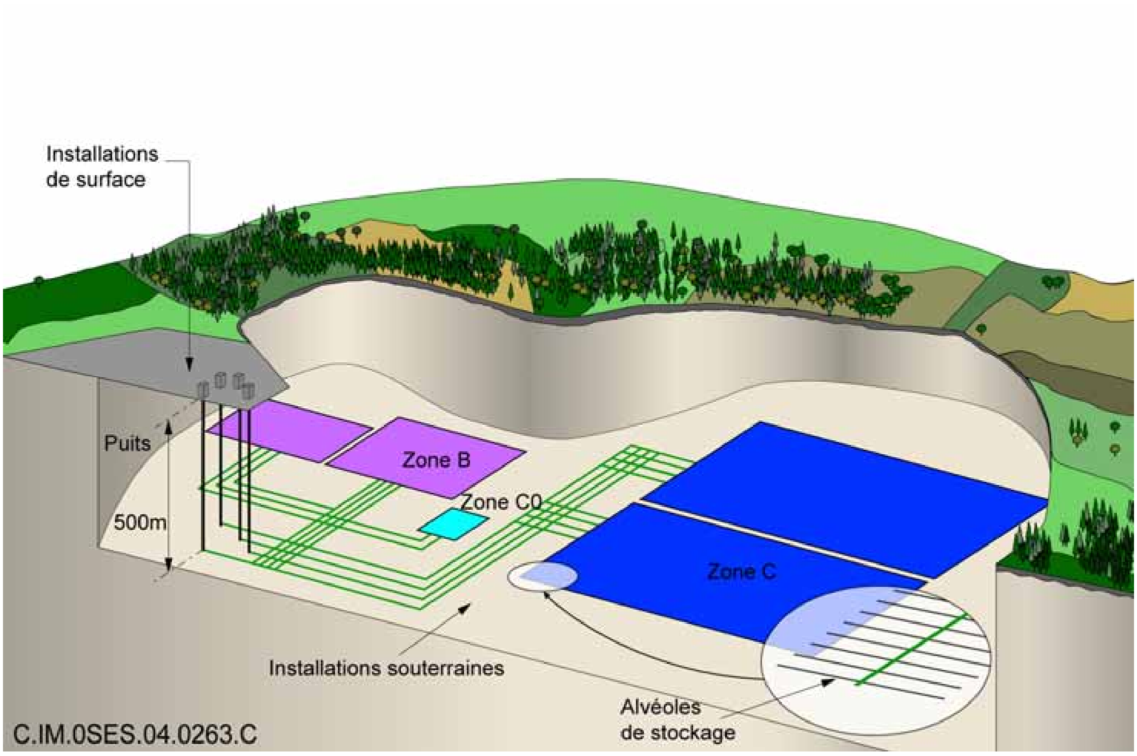


Figure 2.3-1 Stylistic view of an in-operation repository architecture

During the operating phase, the surface installations ensure the reception of the wastes and the preparation of the waste disposal packages. They comprise also a repository zone of the sinking broken rock, as well as the workshops and offices supporting the underground work and operation. In order to accommodate all the wastes previously described (B waste, C waste and spent fuels), the surface footprint of these installations would be at most about a few hundred hectares. The largest part would correspond to the broken rock repository.

The design of the surface installations is similar to the existing industrial installations.

The underground installations and the disposal packages are the most specific elements. The following sections describe summarily the options proposed for the cells and the disposal packages of each waste category and then a possible arrangement of the repository zones and the shafts and connecting drifts.

2.3.1 The shafts and connecting drifts

A set of four shafts ensures the connection between the surface installations and the repository level in the geological formation studied :

- a personnel transfer shaft, dedicated to the transfers of personnel and small equipment, as well as the introduction of fresh air in the underground installations ;
- a construction shaft, dedicated to the transfers of spoils, backfills, other materials and equipment ; this shaft is also an air entry ;
- a shaft specifically assigned to the transfer of disposal waste packages (emplaced in the protection « casks » against radiation) and empty casks ;
- an air return shaft of the air extracted from the underground installations.

As a variant, a ramp is imaginable to ensure the service, emergency or package transfer functions.

The repository modules are accessible from the shafts by the connecting drifts, which form a hierarchised unit. In order to allow a completely safe co-existence of the waste package disposal and new module construction activities, the connecting drifts are specialised by functions :

- some connecting drifts serve the construction worksites (they are equipped with rails or lanes for tyred machines). They are used to transfer worksite personnel, mining equipment, spoils and building materials ;
- other drifts are used to transfer the disposal waste packages ;
- finally, drifts are specifically dedicated to the return of air from the ventilation.

When, if necessary, their closure is decided, the connecting drifts and shafts will be sealed.

2.3.2 The repository zones

The design of the repository zones results essentially from the willingness to give a modular character to the repository. In fact, it is used to build and operate the various repository zones concerned in a simple and progressive way. The creation of the repository modules made up of one or several cells is subtended by this principle. The repository zones are also designed according to safety considerations and, in particular, ventilation in case of fire. Finally, geoetechnical considerations led to separating the structures apart from one another in order to guarantee their mechanical stability and their independence.

When closure is decided, the various components of the repository zone (disposal cells, cell access drifts, connecting drifts) would be sealed by plugs of a low permeability, swelling clay base and backfilled with sinking spoils of the argillite formation. This process would be implemented progressively in successive steps.

In the case of a B waste repository zone, a repository module consists of a single cell served by an access drift oriented in the axis of the cell. The cells dedicated to the disposal of packages containing organic substances are separated apart from the other cells (see explanations in chapter 3).

Figure 2.3-2 shows the possibility of separating the repository zone into two subzones according to their content in organic substances.

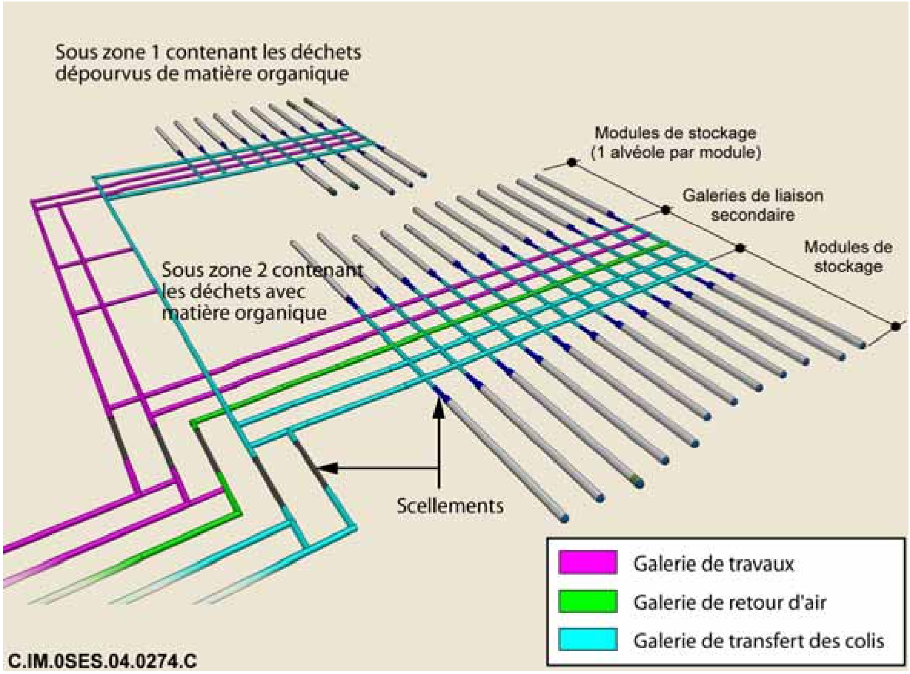


Figure 2.3-2 Organisation of the B waste repository zone

In the case of a vitirifed C waste (and spent fuels) repository zone, a repository module consists of several tens of cells. The cells are served by access drifts oriented orthogonally to the cells (see Figure 2.3-3). The spacing between the cells results essentially from taking into account the heat transfer phenomena and allows for a sufficient dissipation of the heat.

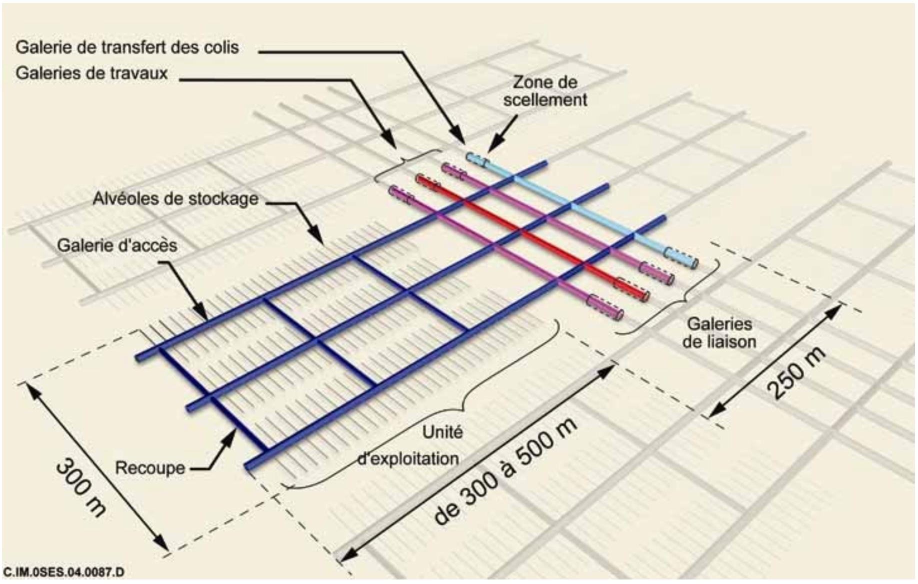


Figure 2.3-3 C waste repository modules

Figure 2.3-4 represents a simplified block diagram of the industrial operations of reception, preparation and transfer of the packages and then closure, illustrating the role of certain components in these operations.

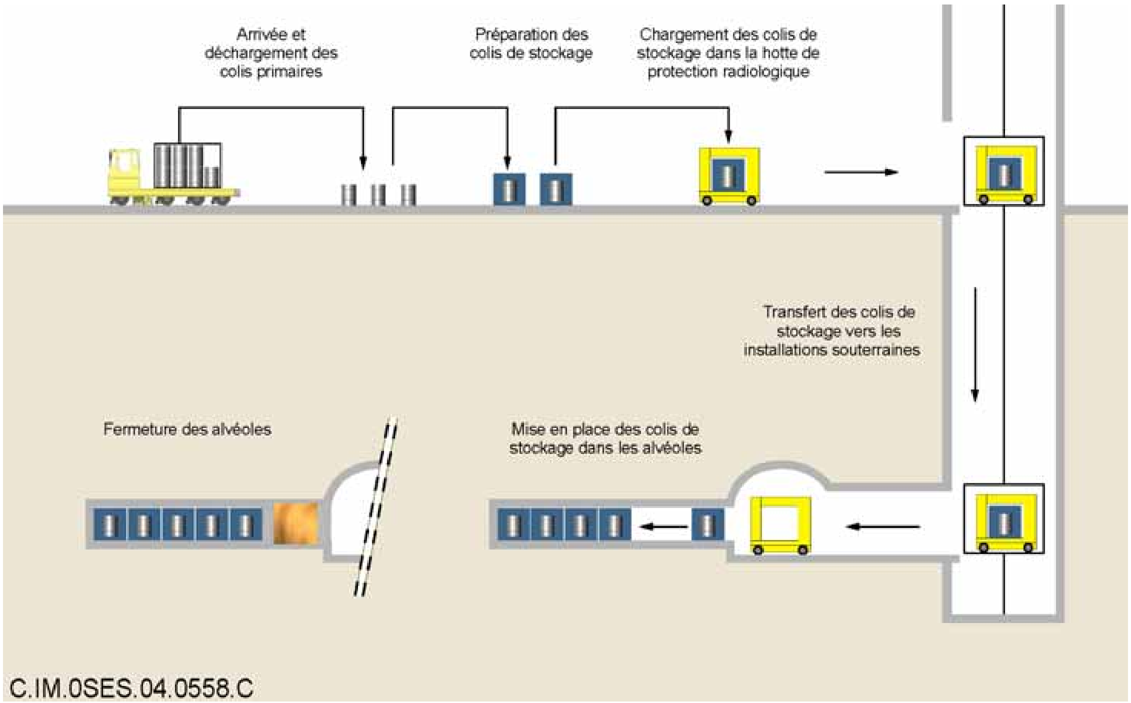
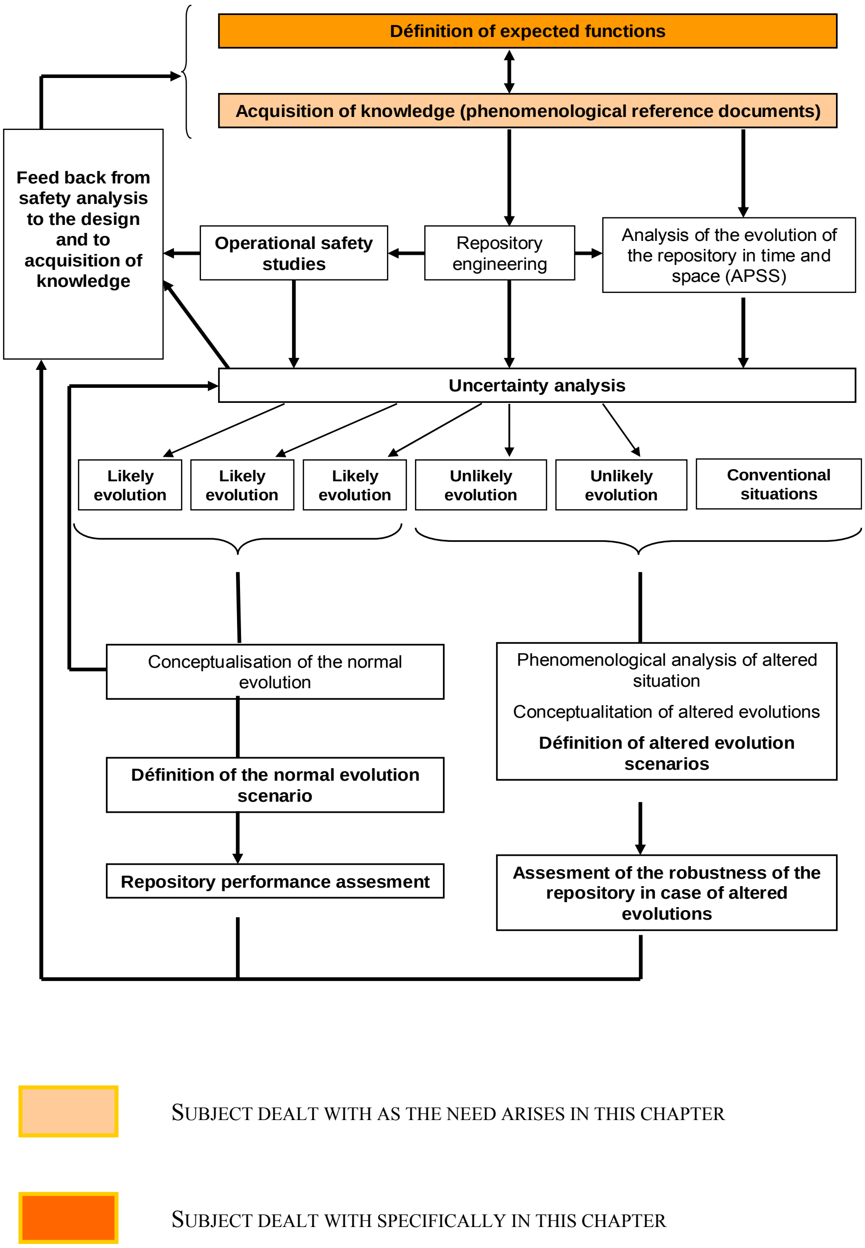


Figure 2.3-4 Block diagram of operation and closure

3 Safety functions and repository design



Block diagram 3-1 Representation of the sequence of the various stages of the analysis (see Block Diagram 1-1) Theme : repository safety and design functions

The purpose of this chapter is to describe the major guidelines followed in the design of the repository It describes the repository architecture proposed in the Dossier 2005, by explaining how safety guided the design choices. It thus forms both a more detailed description of the architectures than the general principles laid down in chapter 2 and an explanation of the way in which safety concerns were incorporated into the design.

3.1 Introduction to repository design

The repository's design follows a conventional approach, which consists of identifying the functions to be fulfilled by the installation and matching them to technical solutions. This matching is guided by various objectives, certain of which are well known in fields other than that of radioactive waste disposal. However, others are specific to it.

Amongst the former are the problems of constructibility, cost control and personnel, material and waste flow management. This document will concentrate more particularly on personnel and public safety during the repository construction and operational phases, which involves problems already encountered in the context of underground works or installations handling high-level waste. Only the combination of these two aspects within a single installation constitutes, as such, a specific feature of the document.

Two problems were however entirely specific to the design of a repository in a deep geological formation. They were, on the one hand, the requirement for the disposal procedures to be reversible (see insert below) A specific document is dedicated to the analysis of the various levels of reversibility [32] and to the provisions to be made to achieve them. These aspects will be discussed here but they are not the main purpose of this chapter. Essentially, the aim is to show that this requirement and safety are not incompatible (see chapter 6).

Insert 1 Disposal reversibility

Note : this insert gives a very broad-brush summary of the way in which ANDRA took the problem of reversibility into consideration in the studies. Anyone specifically interested by these questions should refer to the volume entitled « Architecture and management of a geological repository » in the Dossier 2005 [22], which deals with them in greater depth.

The reversibility of a repository's operation can be defined as its ability to be controlled in a progressive and flexible manner in such a way as to leave freedom of choice to future generations. With this aim in mind, the disposal process can be broken down into successive stages which, from the construction of the first modules to the eventual closing of a repository module or zone, provide the possibility of implementing a waiting or observation period before moving on to the next stage or reversing the disposal process. Moving on to a subsequent stage is not an irreversible final decision but a reasoned choice based on a full understanding of the scientific, technical, economic, social and environmental parameters and the consequences of moving from one stage to the next. Consequently, no phase has a pre-defined, mandatory duration. Reversibility implies leaving the future operator of the repository a certain amount of freedom in deciding the length of the various successive stages.

Reversibility thus requires a flexible approach to be developed, with periods whose length can be adapted and which is best understood in terms of levels. In order to propose such an approach, ANDRA's studies and research consisted of :

- analysing the principal phases of a repository's lifetime and the associated timescales, in order to determine the key stages requiring human intervention ;

- contriving an adaptive disposal staging management system, with decision-

> making milestones. Moving on from one stage to the next would make it possible to make the repository progressively more passive whilst only gradually reducing the level of reversibility and, consequently, the monitoring and maintenance loads.ANDRA took these objectives into consideration in the proposed repository design options, notably through modular architecture, in-depth research into the simplification of operations, dimensioning and choice of durable materials. However, reversibility in no way represents a compromise as far as the safety objectives are concerned : no technical provision that can significantly disturb a safety function has been made for reasons of reversibility.
>
> Reversibility is also made possible by knowledge of the evolution of the condition of structures and the definition of capabilities to be provided over a minimum timescale of a hundred years, or longer if considered necessary : that has led to the study of operational provisions for repository management, notably the recovery of packages and monitoring facilities which could be established within structures.

Furthermore, the designer must consider the long-term evolution of the repository, which must remain safe without human intervention, beyond a necessarily limited systematic surveillance period. That requires that any evolution of the waste, the host rock and the materials forming the repository allows for safe conditions to be maintained. For that, three complementary aspects have to be considered :

- As far as possible, the waste disposed must have characteristics that are conducive to safety. The processing and packaging of waste currently produced are the responsibility of its producers, who are in constant contact with ANDRA as part of the repository feasibility studies. Certain types of packaging have been designed from the start, with a view to long-term management (for example, vitrification of fission products) ; in other cases, notably for spent fuels which are not considered as waste, the physico-chemical form which they take can have characteristics that are conducive to repository safety, even if they were not designed with this end in mind. This document considers the objects which may potentially be disposed, waste and spent fuels, in the form in which it is currently envisaged that they will be delivered by the producer. It has been possible to make provision for additional packaging but not to make any changes to the primary waste packages themselves ;
- The chosen site must have favourable characteristics. Two sources enable these relevant characteristics to be identified : firstly the basic safety rule RFS III.2.f. [2] recommends a certain number of « site selection criteria », which are generically suited to the construction of a repository. Secondly, as the studies develop, the designer has to explain in detail what he expects of the geological medium and what functions must be fulfilled. That can lead to the selection criteria being detailed and supplemented. This document is based on the assumption that the repository will be built in the Callovo-Oxfordian formation. The site selection process in the strictest sense is not part of this document's remit. It does, however, attempt to identify the characteristics expected from the formation and how the Callovo-Oxfordian meets them, both in its initial state and taking into account its projected long-term evolution ;
- The engineered structures planned by the designer with a view to building the repository, operating it, ensuring that it is reversible and closing it in stages, must either fulfil safety functions in addition to those provided by the host formation or, as a minimum, not disturb the functions fulfilled by it. They must be selected and dimensioned so that their intrinsic characteristics, the processes governing their long-term future and their mutual interactions cannot lead to a situation in which personnel and environmental protection are no longer provided.

However, the repository's design cannot directly address all the characteristics and processes that govern the evolution of the waste, the host formation and the engineered components, and immediately propose a design that offers an appropriate answer to all the phenomena that can occur over the long timescales involved. The repository's design is part of an iterative approach making it possible, at each stage, to identify the important phenomena and to place them in order of importance, define the timescales and physical spaces involved, and gradually define the design to reflect this analysis.

The aim of this chapter is therefore not to give a detailed report of the repository's phenomenology covering all timescales and all possible physical spaces involved. Current knowledge in this field is discussed in the phenomenological analysis of repository situations [27, 28] and is discussed from the point of view of their effect on the repository and the functions to be fulfilled in the safety analyses [33]. A summary of this analysis is given in chapter 6. Only the most important phenomena, from the point of view of their effects on the overall evolution of the repository, considered in the design and around which the repository's architecture is constructed, are shown here.

The repository's design and the understanding of the phenomena that govern it during the operational phase and, more so, in the long-term, are inextricably linked. It seemed important to establish, at a very early stage of the presentation, the orders of magnitude of the timescales and physical extent of these major post-closing phenomena (section 3.2). These are important references in the understanding of the repository's design ; at a later stage, in finer detail, they form the support for the safety analysis.

Once this framework has been established, and after an introduction to the methodology implemented (section 3.3), the main repository safety functions are described, without reference at this stage to the technical solutions which will be adopted in order to meet them. The aim is simply to explain what the main principles were in designing the repository (section 3.4).

The design approach, the breaking down of these general functions into specific sub-functions and the proposed means of fulfilling them, are the subject of sections 3.5, 3.6 and 3.7.

3.2 General context : timescales and physical extent

Traditionally, the question of timescales in repository safety is approached using impact calculations : Up until what date is it necessary or reasonable to carry out long-term impact calculations? What level of accuracy should we expect from models depending on the timescales involved? These are the subjects on which the basic safety rules RFS III.2.f [2], in particular, gives general directions.

However, the problem of timescales and physical extent opens up other perspectives for reflection [34]. Determining the typical timescales over which the major repository components evolve (natural medium, waste, exogenic elements introduced by the repository) allows the designer, in particular, to structure his thought processes in time and space. It allows him to determine the technical solutions appropriate to each phase of the repository's life. The designer defines the safety functions to be fulfilled for each component, for each timescale. In order to do that, he takes into consideration the predictable behaviour of the components of his system. That allows him both to determine if it is realistic to assign such a function (« at that time, will the component still be in a physico-chemical condition that allows it to fulfil the expected function? ») and if it is necessary to add new functions applicable to the problems of each period (« at that time, will the extent of such a disturbance be of an order of magnitude such that it will be necessary to make provision for limiting its effects by means of some special arrangement or device? »).

From the time of its construction, and over a very long term period, the repository is subjected to changes, which can be grouped together into several major phenomenological categories :

- Thermal phenomena, essentially associated with the heat given off by certain types of waste, which decreases over time ;
- Hydraulic phenomena, associated on the one hand with the disturbances caused by natural water flows within the site, for which the repository is responsible and, on the other hand associated with the production of gases – gases produced by radiolysis but, above all, gases produced by corrosion – or, where applicable, with the site's geodynamic evolution (erosion, changes to water courses, etc.) ;
- Mechanical phenomena, associated with the changes of the site's natural stresses induced by the excavation of the structures ;
- Chemical phenomena, which include all the transformations undergone by exogenic materials introduced by the repository (waste, waste packages, construction materials, etc.) due to the effect of chemical processes - or biological processes in the event that micro-organisms present in the repository may have an effect. Within this category, we also include the chemical disturbances in the natural medium which these transformations involve ;
- Radiological phenomena, the most important of which is the radioactive decay of waste and all the disturbances which radiation can induce (radiolysis, effects associated with possible criticality accidents, etc.).

These phenomena determine the timescales and physical extent data used to support the safety analysis. One more timescale must be taken into account, that associated with the transport of radioactive nuclides and chemical toxins within the repository that are inevitable in the very long term.

The typical durations of all these phenomena are not always independent of the design adopted. In a certain number of cases, managing this duration is the very purpose of the repository's design : thus, the durability of certain waste packaging matrices has a positive effect on safety (glass being a prime example) and the repository must form a favourable medium for the slow evolutionary kinetics of the chemical breakdown of these matrices. Another notion is that of the physical durability of certain materials, which greatly depends on their formulation and on the manner in which they are used. No precise estimation of this can be given without reference to a given concept.

Insofar as in this section the aim is to introduce the repository's design, and not present its result, we limit ourselves here to quoting either typical timescales of processes that are beyond human control (radioactive decay for example) or orders of magnitude, the only purpose of which is to enable a general context to be established for further reflection. We have opted for the description that appears the most natural, starting with the major geodynamic changes, radioactive decay, then the behaviour of exogenic elements from the point of view of thermal, then hydraulic, then mechanical and, finally, chemical changes.

3.2.1 Timescales and geological medium

The Callovo-Oxfordian formation appears to be remarkably stable in the sector studied over a timescale of several tens of millions of years. Tectonic, seismic, climatic and erosion activity appear to have had little influence on it (see chapter 2).

The same does not apply to the surrounding overlying strata and the surface environment, which have undergone significant modifications in the last million years, due principally to climatic changes. ANDRA participated in the international BIOCLIM exercise, which made it possible to define climatic change scenarios over the million years. These highlight a chain of glacial / interglacial climatic cycles, controlled by changes to the earth's orbit around the sun, the periodicity of which is approximately 100 000 years. Depending on the effects of disturbances associated with human activity and induced global warming, the timing of this cycle may change. This alternate cycle of glacial and temperate climates will cause significant surface environmental changes, though no major event (of the type that could induce the presence of a glacier) is expected under any of the established scenarios. The carving out of valleys and the disappearance of the superficial layers of terrain could induce changes both in the lifestyles of the populations living on the site and in the regional hydrographic conditions. The depth at which surface climatic events could change the configuration of the terrain is estimated at around fifty metres.

These two factors, the long-term stability of the host rock and the greater possibility of changes to the surface layers and surrounding formations, require an initial important design choice. Given that the lifetime of the waste is long and that it may be transported within the host formation for more than 100,000 years (see next section), we will rely particularly on the properties of the Callovo-Oxfordian layer in order to confine the radioactivity. The over and underlying formation loyers will have a different role within the analysis (see section 3.3).

3.2.2 Timescales, radioactive decay and transport of radioactive nuclides

The radioactive nuclides taken into consideration in the HLLL waste radiological inventory have periods that vary greatly, from less than a few months up to 10 thousand million years, their mass radioactivity diminishing the longer the period progresses. Of the 144 radioactive nuclides considered by ANDRA (i.e. all those with a period greater than 6 months) only a few have a period greater than 100,000 years (see Table 3.2-1). Andra uses the OECD/NEA base called JEFF [65] as the reference nuclear database for radioactive half-lives. The half-lives presented below are extracted from JEFF version 2.2. It will be noted that the May 2005 update shows that the half-life of 79Se has been increased by a factor of approximately 17 to 1.1 million years.

Isotopes	Periods (years)
^{10}Be	1 600 000
^{I26}A	720 000
^{36}Cl	302 000
^{40}K	1 280 000 000
^{41}Ca	103 000
^{53}Mn	3 700 000
^{60}Fe	7 510 000
^{81}Kr	210 000
^{87}Rb	48 000 000 000
^{93}Zr	1 530 000
^{92}Nb	35 000 000
^{97}Tc	2 600 000
^{98}Tc	4 200 000
^{99}Tc	213 000
^{107}Pd	6 500 000
^{126}Sn	100 000
^{129}I	15 700 000
^{135}Cs	2 300 000
^{146}Sm	103 000 000

Isotopes	Periods (years)
^{150}Gd	1 790 000
^{154}Dy	2 850 000
^{176}Lu	36 100 000 000
^{182}Hf	8 990 000
^{186m}Re	200 000
^{205}Pb	15 2000 000
^{208}Bi	368 000
^{210m}Bi	3 000 000
^{232}Th	14 100 000 000
^{233}U	159 000
^{234}U	246 000
^{235}U	704 000 000
^{236}U	23 400 000
^{238}U	4 470 000 000
^{236}Np	152 000
^{237}Np	2 140 000
^{242}Pu	374 000
^{244}Pu	80 000 000
^{247}Cm	16 000 000
^{248}Cm	340 000

Table 3.2-1 Radioactive nuclides with a period greater than 100,000 years

Some of these very long life radioactive nuclides are only present in the waste at trace level, others, due to their mobility or their radiotoxicity are of little importance for the safety of the repository. Given all the radioactive nuclides initially present in spent fuel, we can say that after a million years (10 times a 100,000 year period), the radiological inventory has decreased considerably, and that this timescale, though conventional, is relevant in view of the decreased inventory [12].

Another order of magnitude applies to the transfer of these same radioactive nuclides through the host formation. This can occur by diffusion, i.e. due to the effects of the spontaneous movement of particles, or by advection, following the water circulating within the site. If we take the case of a radioactive nuclide that is not held in the host formation, a rough assessment of its range of travel (L) within the rock during a time (t) can be given by comparing its characteristic diffusion time and its apparent diffusivity, using the relationship : $L = \sqrt{(2.D.t)}$.

Using an apparent diffusion coefficient of 10^{-10} m²/s, representative of a rock such as the Callovo-Oxfordian, the following results are obtained. These are only rough figures, for several reasons : on the one hand, the diffusivity depends on the type of element considered, on the other the formula applied corresponds to transport and dimension in a homogeneous, infinite medium.

1 year	100 years	10 000 years	1 000 000 years
0.07 m	0.77 m	7.7 m	77 m

[12] It should be noted that this residual inventory beyond a million years is not directly comparable with a representative inventory naturally present in the ground as certain of the elements in it have no equivalent in the natural environment. It cannot therefore truly be called a « return to natural radioactivity ».

Similarly, a rough figure can be assessed for advection times, based on a mean permeability of the host formation (which for simplicity's sake we assess as $K=5.10^{-14}$ m/s) and a gradient fixed at gradH=0.2 m/m. The distance travelled (L) by a advective flow is expressed as a function of time by the relationship : L = K.gradH.t, which gives :

1 year	100 years	10 000 years	1 000 000 years
negligible	negligible	0.003 m	0.3 m

This order of magnitude calculation, which clearly is only an estimate, tends to show that transfers by advection will be dominated by transfers by diffusion in sound rock and that, for the latter, the million-year period is that in which we would expect to see solutes travel a distance equal to the thickness of the rock buffer zone (a distance of fifty to one hundred metres).

The million-year period thus appears relevant for the designer and for the safety analysis. In fact :

- It corresponds to a period during which it is possible to use the properties of the host rock as observed today, without having to envisage that they may be significantly changed ;
- It is a period which corresponds to a very significant decrease in the radiological inventory and, thus, of the « danger » that it ultimately poses, even if this argument alone is insufficient reason to opt for a 1,000,000 year period ;
- It is, in an initial rough approach, the period during which the host formation can be counted on to retain the radioactive elements. It thus constitutes a good period for analysis purposes, insofar as it must make it possible to assess the transfer of the elements and their impact on the biosphere.

Clearly, beyond the million-year objective, the repository is designed to be safe for as long as possible. However, over greater timescales, it becomes more and more difficult to predict how the geological medium will evolve, even in the most general terms, the analogy with past changes becoming subject to far greater reserve. Furthermore, to extend the analysis would be of no further particular benefit, from the viewpoint of radioactive decay. Similarly, in a totally different field, the calculation of performances (chapter 5) is stopped at a million years, unless it appears that this « million years » deadline may stop calculation before maximum of impact is reached. In such a case, one has to make sure qualitatively that maximum of dose will remain moderate.

3.2.3 Timescales, physical extent and thermal decay

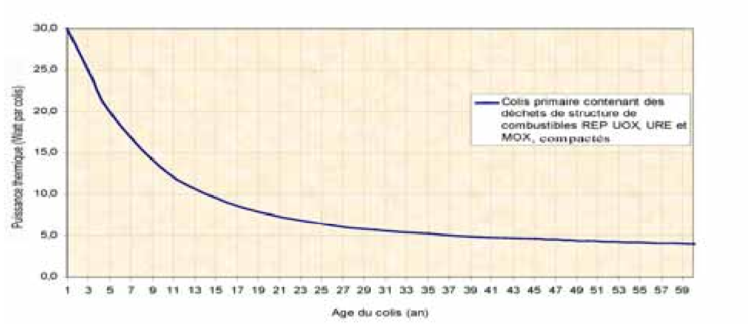
Due to their radiological content, vitrified waste and spent fuels release significant quantities of heat from the viewpoint of their possible impact on the host formation and transport phenomena. The power output of a vitrified waste package is several hundred Watts on production and that of spent fuel several thousand Watts. Type B waste produces a lower power output, with a maximum of a few tens of Watts for structural waste. The phenomenon of the thermal decay of waste packages is illustrated by the graphs in Figure 3.2-1).

Thermal decay is first controlled, over the first few hundred years, by short and medium life fission products. Beyond that point, it is the alpha emitters that take over, without the heat output necessarily being negligible : it still exceeds one hundred Watts for spent fuel after 1000 years. In absolute terms, the period beyond which the thermal power of the waste can be considered as of no importance to the repository cannot be fixed : it depends on the layout of the repository and the rock's ability to evacuate the heat flow. However it is considered that, for spent fuel, the waste's heat output is a phenomenon which occurs over a thousand to a few tens of thousands of years. In parallel with what was examined for radioactive decay, it is also interesting to carry out a rapid calculation of the natural medium's ability to transfer heat. Using a thermal diffusivity for the Callovo-Oxfordian, such as has been possible to assess, of around 10^{-6} m²/s, the order of magnitude of the heat progression in the medium can be assessed :

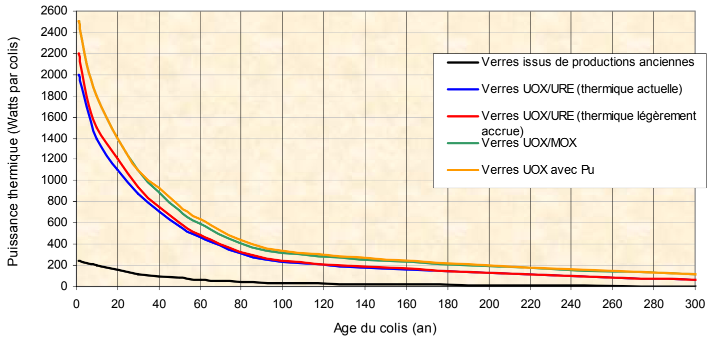
1 year	10 years	100 years	1000 years
7.7 m	24 m	77 m	245 m

As an initial estimate, and remaining conscious that this is only a rough calculation, the thermal influence of the waste is felt at less than 100 metres for 100 years, and at 250 metres for 1000 years. After such periods, thermal decay comes into play and the area of thermal influence expands no further.

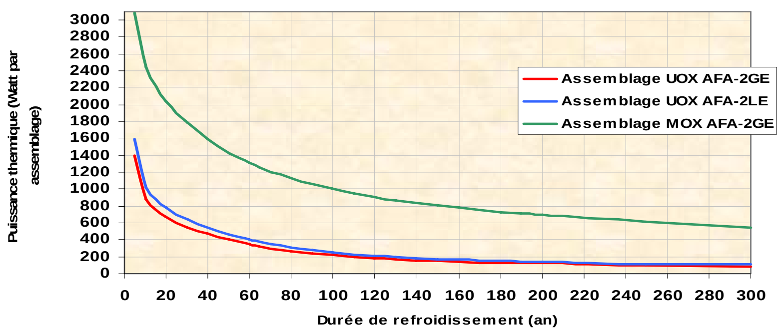
From this rapid analysis, it is clear that heat is a phenomenon that must be taken into consideration for a period whose duration depends on the type of waste, of up to 10 000 years. Based on a rough estimate, the waste's thermal influence within the formation does not extend beyond a hundred metres or so.



Thermal power of REP UOx, URE and MOX fuel structural waste packages



Thermal power of vitrified waste packages



Thermal power of UOx and MOX fuel assemblies

Figure 3.2-1 Thermal power of waste packages

3.2.4 Timescales of thermal phenomena

The natural medium is initially in a state of hydraulic equilibrium, the flows within the different formations having reached a stationary state. The excavation of the repository structures disturbs this situation (see detailed description in [35]). Here, we simply note that the various formations crossed undergo a hydraulic discharge, which is transmitted within each layer or between layers, depending on contrasts in their permeability (see Figure 3.2-2).

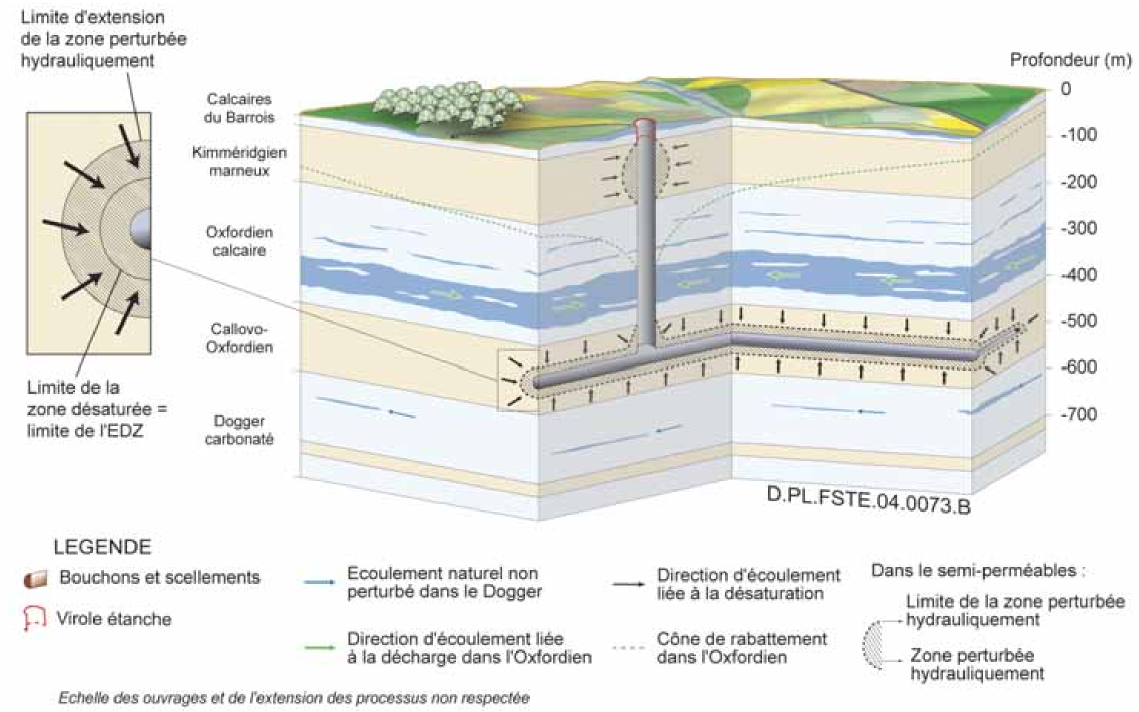


Figure 3.2-2   Hydraulic transient during the operational phase

At the same time, the excavation and operation of the structures, particularly the ventilation for those fitted with it (essentially, as will be shown, type B waste cells) de-saturate the wall rock. These phenomena last for as long as the repository is operational, i.e. over a timescale of roughly a hundred years. Based on the medium's mean permeability, it can be estimated that, in the host formation, the de-saturation cannot extend beyond the immediate environment of the structures during the roughly hundred-year operating period (see Figure 3.2-3) ; in principle, it remains limited to the area damaged by excavation (roughly a metre or a few metres). In the event that the operational period is extended over several centuries, for example within a reversibility context, this de-saturation propagation rate would remain unchanged. Its kinetics within a medium that has been only slightly mechanically disturbed is, in fact, much slower.

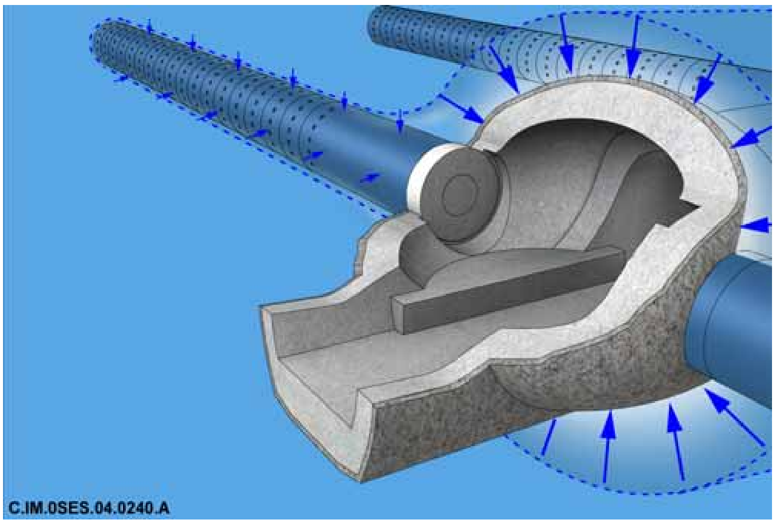


Figure 3.2-3   Diagram of the convergent flows (→) and the extent of the de-saturated area around the cell walls (----)

Once the repository has been re-closed, the medium re-establishes the « natural » loadings, i.e. the initial loadings that may have been modified depending on geodynamic changes. The duration of the re-saturation of both the rock and the structures themselves is a complex phenomena, which depends on both the permeability of the media encountered and the void fraction and the influence of chemical and thermal processes that occur simultaneously. It can be altered significantly by gases released by the corrosion of metal components. At this stage, it should be noted that it cannot be assessed by means of a simple calculation, since it depends more directly on the definition of the design of the repository.

However, we can note in advance that, generally speaking, structures with small diameters (type C and spent fuel disposal cells) re-saturate together (in a period of less than a hundred years). The hydrogen produced by anoxic corrosion is sufficient to delay the re-saturation of porous media (drifts, type B cells) for up to a hundred thousand years. The estimated re-saturation times or times taken to return to a state of hydraulic equilibrium, taking into account the influence of gases, are given in Table 3.2-2.

Constituent	Mean time required to reach 90 % re-saturation
Spent fuel cell (including plug)	≈ a hundred years
Type C cells (including plug)	≈ a few decades
Type B cells	≈ hundred thousand years
Spent fuel drifts	≈ hundred thousand years
Type B cell seals	≈ thousands of years
Connecting drift and shaft seals	≈ thousands of years

Table 3.2-2 Estimated re-saturation times or times taken to return to hydraulic equilibrium taking the influence of gases into consideration

In summary, the hydraulic phenomena do not extend their influence over large distances (a few metres at the most). The characteristic durations are very dependent on the concepts implemented ; they are important for the designer but cannot, at this stage of the development of the architectures, be taken at first sight as data for use in the analysis. They must be repeatedly assessed.

3.2.5 Timescales and physical extent of mechanical and chemical phenomena

It is not intended here to give a detailed description of the mechanical and chemical processes within the repository but simply to extract the main elements associated with the problems of their timescales and physical extent. It is again within a relatively generic context, without going into the specifics of the concepts, other than assuming that the exogenic materials placed in the repository (other than waste) are essentially made of metal, cement and clay.

During the excavation of the structures, and later during their operation, the host formation is subjected to mechanical damage and surface oxidation. These two processes, whose importance from the point of safety differ, both extend over a relatively small area within the repository, of around a meter. The damage should not get worse during the repository's subsequent operational period, the creep of the rock contributing to the closing of any fissures caused by excavation (see more detailed discussion of this point in chapter 6). More generally speaking, the disturbance of the site's stress field tends to return to a state of equilibrium as the rock and the surrounding medium gradually assert their mechanical load. The creep within the Callovo-Oxfordian layer being very slow, it can take several hundred thousand years to return to this state of equilibrium.

However, due to the high reducing potential of the water in the Callovo-Oxfordian layer and to the latter's mineral structure (particularly the presence of pyrites), the return to reducing conditions occurs very rapidly (a maximum of about a hundred years).

Furthermore, the rock is liable to be subjected to chemical disturbances other than oxidation, caused by metal and cement-based elements. Iron-argillite and alkaline disturbances will be assessed more fully in chapter 6 but, at this stage, it can be said that in general terms, they develop over a period of around a million years but do not extend beyond a distance of a meter.

The different types of waste have varying lifetimes, depending on the environmental conditions, sometimes varying drastically (glass for example being highly temperature-sensitive). It is not possible to establish a general rule concerning their retention performances, these will be proposed in chapter 5 in the context of behavioural model selection. At this stage, it is simply reasonable to envisage that glasses, bitumens, spent fuels and metal waste will retain their performances over periods of from a thousand to several hundred thousand years, or even longer under favourable conditions.

Cement-based materials essentially change due to the effects of hydrolysis, carbonation (action of carbon dioxide on the constituents of the cement matrix) and sulphate attack. Thus, they gradually lose their mechanical performance ; ANDRA has defined models to describe the effects of these disturbances, particularly the various phases of concrete due to hydrolysis (sound, modified, degraded and detrital concrete) whose duration depends on the formulation of the concrete itself and on the environment. They will be discussed in more detail in chapter 6. Generally speaking, it is noted that the concrete's mechanical cohesion depends on the hydric conditions and mechanical stresses to which it is subjected. Periods of several thousand years, or even significantly longer subject to favourable conditions, can be expected. The concrete can be expected to retain its chemical performance (its role as a base buffer) over much longer periods, depending on the environment in which it is placed.

Metal materials are subjected to corrosion, depending on whether they undergo oxidising or anoxic conditions. The rate and type of corrosion vary according to the environmental conditions. Generally the corrosion rates are less than a micron per year under anoxic conditions and a few microns per year under oxidising conditions. Based on these assumptions, and on condition that the environment is controlled, metal components around a centimetre thick have a life expectancy of several tens of thousands of years or more.

Finally, materials made of swelling clay are only very slightly susceptible to interact with the medium, itself being clay-based. Under repository conditions, given that water movement is very slow, they are only very slightly susceptible to ageing and erosion. They can, however, be affected by the same disturbances as the host rock : by iron, by cement and perhaps by other species. As an initial estimate, given that these disturbances affect only around a meter over a million-year timescale, it would appear possible for the designer to expect that components made of swelling clay, several metres thick, maintain the mechanical and chemical stability of a significant part of their dimensions for around a million years.

In summary, from this rapid analysis of the timescales and physical extents involved, it would appear that :

- The million-year timescale appears to be a realistic objective, both for the design and for the safety analysis ;
- Thermal disturbance is the one that is most likely to spread over approximately one hundred metres The other forms of disturbance are more likely to be limited to around a meter ;
- In an initial estimate, questions associated with heat only apply to the first phase of the repository's life (whose duration depends on the nature of the waste). It appears realistic, for the design, to distinguish between a thermal phase and a post-thermal phase, the design provisions being able to be different for managing the former and the latter.
- Hydraulic, mechanical and chemical disturbances are either more difficult to assess and depend on the design, or liable to persist throughout the million-year period. They have to be taken into account in the design but, for the purposes of the repository safety functions, separate « hydraulic », « mechanical » and « chemical » phases will not be defined. The functions to be fulfilled within the repository will be the same in all cases. Of course, that does not mean that the corresponding transients and their effect on the safety functions are not to be taken into account ;

- As an initial approach and, of course, subject to a more detailed analysis, we can define orders of magnitude for the lifetimes of cement-based, metal and clay-based compounds. Within the context of an iterative design approach, that makes it possible to know what type of function can be assigned to them. In particular, only the mechanical functions of clay-based compounds and the chemical functions of cement-based compounds can be permanent over the million-year period [13].

All these elements therefore establish preliminary timescales and physical extents that allow the designer, or the scientist or safety engineer, to establish the basis of their respective approaches. It goes without saying that these broad-brush elements are insufficient to define a framework for more detailed analysis. This would only result from a detailed consideration of the phenomena [36].

Once this general framework is established, it is possible to deal with the question of the design of the repository itself.

[13] Implicit in this case is that it has been chosen not to use other materials.

3.3 Functional approach to safety

ANDRA, like many of its foreign counterparts, has implemented a system of controlling the safety of the repository by assigning safety functions [37] as a method that complements the so-called « multi-barrier » approach. The latter, used in nuclear reactor safety, consists of placing several confinement barriers between the radioactive materials and the environment, as far as possible independent of each other. The development of this approach led to the establishment of the notion of defence in depth, which complements the « barrier » concept with that of « lines of defence », adding to the simple physical confinement barriers all the material and organizational provisions enabling accidents to be prevented or their consequences reduced and managed.

The functional approach to safety is another development of the « multi-barrier » strategy. Today, this approach is recommended at international level [12]. It consists of meeting the safety requirement by asking oneself what are the objectives to be sought. Safety does not necessarily simply involve placing successive physical barriers between humans and radioactivity. In certain situations, particularly for a repository, such an approach is inappropriate.

Certainly, some barriers can have lifetimes that are extremely long in absolute terms but which remain limited when compared to the radioactive period of the waste (the packages used in establishing the design of a deep repository in a clay environment, for example made of steel or concrete, cannot hope to have a level of durability similar to that of the half-life of radioactive nuclides, which can exceed hundreds of thousands of years). They cannot fulfil their barrier role throughout the period required to demonstrate safety.

However, the deterioration of a barrier does not mean that the repository's safety is reduced. In fact :

- If a vitrified waste container lasts sufficiently long to allow the essential decay in the radioactivity of the waste that it contains.
- If once the container has lost its sealing integrity, the waste itself releases its activity sufficiently slowly for an increase in the radioactive decay to occur.
- If, finally, the geochemical conditions within the cell are such that the radioactive elements released tend to be only very slightly mobile within the medium or, for certain of them, even precipitate,

then a single function, consisting of immobilising the radioactive nuclides within the waste cell, is carried out continuously with adequate performance levels, even if the barriers gradually deteriorate. Conversely, the functional approach can justify the importance for safety of a component that does not have a barrier role. Thus, for example, we do not expect the back-fill material in the access structures to contribute to limiting the migration of radioactive nuclides, in the concepts developed for the clay medium. Back-fill is not, therefore, a « barrier », at least as far as the transfer of radioactive nuclides by water is concerned. It nevertheless fulfils functions that may be important under certain circumstances : mechanical support of the rock to prevent damage developing in the very long-term, obstacle to any attempts at undesired access to the repository after closure.

The notion of multiple safety functions constitutes a generalisation of the notion of multiple barriers. It consists of meeting the safety objectives by implementing different types of action that all contribute to the safety of the repository. These actions are accomplished by the repository's components, the operators or the organisational provisions implemented. Functions may be redundant, i.e. have the same effect and be able to replace each other, but most of them are complementary and contribute jointly to achieving the safety objectives. The loss of a function then leads to a deterioration in the safety level, but this loss can be acceptable if the other functions are maintained.

3.4 Principal safety functions of the repository

3.4.1 Fundamental objective of the repository

The fundamental objective of the repository with respect to safety is expressed in the basic safety rules RFS III.2.f. It consists of protecting personnel and the environment against the hazards associated with the dissemination of radioactive substances. This objective is formally restated by ANDRA in functional form : « to protect humans and the environment from the dispersal of radioactive nuclides » chemical toxins contained in waste being implicitly associated with radioactive nuclides.

It should be noted that this is not a repository's sole function. It also meets the industrial objectives of taking responsibility for waste, for example. A more complete list of the repository's functions has been established [23]. From the post-closing phase onwards, repository safety is the sole remaining objective.

There are several ways in which humans and the environment can be affected. Firstly, there are those associated with the radioactive nature of the waste : risk of external exposure to radiation, risk of ingestion or inhalation of contaminants. Others are more conventional and refer to the chemical toxicity of the waste – soluble substances or gases where applicable. Other hazards can also be taken into account if considered relevant : thus certain types of waste, such as drums embedded in bitumen, release potentially explosive gases by radiolysis. It is, however, radioactivity which constitutes the main problem with this type of waste, and it is around this characteristic that the repository's design is defined (see chapter 1). It is designed, first and foremost, to protect humans and the environment against radioactivity. However, the other aspects are not ignored, but it is acceptable to check later that the provisions with respect to the radioactive hazard also cover the chemical hazard, for example, or to amend the repository design in order to adapt them to particular hazards.

The repository's design is driven by the strategy of concentration and confinement, as defined for example in publication N°81 of the International Commission on Radiological Protection [6] and recommended by the International Atomic Energy Agency [38]. It consists of grouping waste in a single site, and stopping the noxious agents that they contain from reaching humans in the long-term. It should be noted that the packaging of waste in stable matrices with favourable properties is also an application of this principle. This is not directly within the remit of this document, this aspect being the responsibility of the waste producer, before it is forwarded to a repository centre.

3.4.2 Principal safety functions during the operational period

During the period in which the repository is active, which can be in excess of a hundred years, the repository is the subject of various operations :

- Construction processes, for establishing the access structures, disposal cells and the necessary utilities required for its operation ;
- Operational processes, aimed at the placing of waste packages and, in the event that it occurs, their removal as part of the reversibility concept ;
- Observation and monitoring, designed to acquire parameter data concerning the short-term evolution of the repository[14] ;
- Cell, module and repository zone closing operations in a staged process.

[14] By the term observation we mean all the measures taken with the aim of better understanding the evolution of the repository, whether they be to confirm the models used in situ (thermal and mechanical observation etc.), to assess safety or to acquire feedback on the first modules, with a view to optimising the construction of subsequent repository modules. The term « monitoring » covers all the conventional measures designed to ensure that, in the context of defence in depth, the repository remains within its designed functional range (radiological protection measurements, gas content in the air, wall damage measurements etc.) or within the validity range of predictive models of its long-term behaviour. The observation and monitoring programmes can be partially over-lapping.

During the operational period, the safety functions are deduced from the potential dispersal channels and principal hazards posed by radioactive waste [24]. They are valid for all phases during which operators are liable to be present in the repository and to handle radioactive waste packages, whether in order to place them in cells, monitor them or remove them within the context of reversibility. These functions are also very similar to those considered in conventional basic nuclear installations. They are :

- To confine the radioactivity, so as to prevent the risk of dispersal. This function also makes it possible to confine any chemical toxins that may be present ;
- To protect persons from irradiation ;
- To provide criticality hazard safety, which is considered to be an aim in itself given the risks associated with the handling of objects such as spent fuels. It should, however, be noted that, in the strictest sense, the objective of this function is very similar to the « personnel irradiation protection » and « radioactivity confinement » functions.
- To evacuate the residual thermal power from waste in order to protect personnel from the hazards associated with the release of heat from waste and provide a satisfactory working atmosphere ;
- To evacuate gases produced by radiolysis, in order to manage both the explosive hazards and their potential toxicity.

It should be noted that these are nuclear safety functions. Others, associated more with the excavation phases or with the prevention and management of conventional working hazards, could be added. Personnel must be protected from problems associated with the mining or, more generally, the industrial context of the repository : for example, protection against dust, falling blocks and cave-ins and electrical hazards. In particular, the basic safety rule RFS III.2.f. requires that the host formation allows the structures to be constructed without intervention for re-gauging during their filling, which is indirectly associated with the safety of the operational conditions. The Callovo-Oxfordian layer meets this criterion thanks to its very slow creep characteristics.

Finally, there are number of other environmental protection objectives, not directly linked to personnel or public safety :

- The protection of underground water courses (in the water-bearing strata crossed) against the various possible forms of pollution ;
- The protection of the surface environment against impacts caused by the work site (protection of surface water, the air, the soil, fauna, flora, the surrounding area and the landscape).

Note should also be taken of the protection and control of nuclear materials contained within the installation.

3.4.3 Principal long-term safety functions

The so-called long-term safety functions, i.e. during the post-closing phase, constitute the repository's true specific character. Here, we limit ourselves to those at the highest level, very largely independent of the repository architectures eventually selected. We will discuss the manner in which they are implemented, if necessary breaking them down into safety sub-functions, in section 3.7.

3.4.3.1 Isolating waste from surface phenomena and human intrusion

This is the primary function of repository safety, which forms one of the principles of an installation in a deep geological formation. It consists of giving priority to a management solution in which the waste is kept out of reach of populations, in order to prevent them from being exposed to radioactivity (exposure to radiation the waste emits or risk of ingestion / inhalation), for periods linked to the decay of the radioactivity. We therefore seek to avoid surface phenomena, such as climatic events, erosion, floods and « day-to-day » human activity, leading to such situations. In that sense, a pre-disposal storage installation may carry out the same function but for periods not considered to be permanent. The geological disposal principle is to carry out this function passively, i.e. without surveillance being required beyond a defined period.

That is a general objective, not aimed at surface phenomena or those associated with specific human activity. However, it is not possible to truly determine whether waste is protected from such hazards until a safety analysis has been conducted which makes it possible to generally identify the various possible actions, their likelihood in the particular context considered and their potential effects. Certain types of action, such as accidental human intrusion following deep drilling, can never be totally excluded, though the likelihood of their occurring and their potential consequences can be reduced.

3.4.3.2 Preserving the repository record

The basic safety rule RFS III-2.f states that personnel protection must be provided « without relying on any institutional control on which it is impossible to count for certain beyond a limited period (...) (500 years) ». That does not contradict the desire to maintain the site record for as long as possible.

The problem of maintaining a record of the site begins during the operational phase when it is a question of maintaining the knowledge and technical skills required to manage the installations. Secondly, after placing waste packages in the repository, the record forms an element of the defence in depth making it possible, in particular, to prevent the risk of intrusion within the repository or to enter it knowingly. On this latter point, it is linked to the previous function but also covers a broader objective of defence in depth. Neverthelers, forgetting about the repository in the long-term, which cannot be totally excluded, should not have an adverse effect on safety.

3.4.3.3 Safety functions aimed at protecting humans and the environment against the release of toxic elements (radioactive nuclides and chemical toxins)

In the same way that waste must be protected against phenomena occurring on the surface, the surface environment must also be protected against any possible rising to the surface of the radioactive nuclides contained in the waste, or chemical toxins. The functions aimed at protecting the surface environment against the hazards associated with waste mostly tend to control the transfer channels which, in the long-term, can lead to noxious elements (radioactive nuclides and chemical toxins) from waste reaching humans.

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

- Aqueous channels, the elements being liable to form solutions and find their way up to the surface ;
- Gaseous channels, certain radioactive nuclides being able to exist and migrate in this form ;
- Solid channels, only in the case of a particular event causing some of the waste to rise to the surface, which would be the case in the event of most cases of human intrusion.

As far as the gaseous channel is concerned in the long-term, most gaseous radioactive nuclides have very short periods as far as transfer times in the formation are concerned (krypton 85, tritium etc.). Amongst the elements that have long periods, only iodine 129, chlorine 36 and carbon 14 are liable to migrate significantly. The first two have properties such that, under the conditions prevailing in the repository, they will be rapidly placed in solution. Only carbon 14 could remain in gaseous phase for longer periods, though in small quantities

This question will be discussed further in chapter 6, section 6.2.10. At this stage we can, however, note that given the small radiological inventory concerned and the high probability of the gases dissolving within the repository, the gaseous channel does not appear to be a very significant channel for the transfer of radioactive nuclides. It does not require any special design measures. That does not prejudge the requirement for other types of provision aimed at controlling inactive gases which may accumulate within the repository (corrosion gases, radiolysis gases, etc.). More generally, in the remainder of the document, we will use the term « hydraulic » to designate questions involving transfers either by water or in gaseous form, the consideration of these methods of transfer leading to similar design provisions. By default, it is water which is discussed ; the specificity of gases is stressed when required in the text.

The design is built around controlling the transfer of radioactive nuclides by hydraulic channels and, more specifically, by water. These can occur due to advection, i.e. by a phenomenon of radioactive nuclides being drawn along in a water flow or by diffusion, i.e. by a Brownian motion of elements in water.

- **« Resisting the circulation of water »**

Diffusion of elements is inevitable in the long-term when the waste packages no longer fulfil a sufficiently isolating role. However, it is possible to greatly limit advection, i.e. water circulation within the repository (see Figure 3.4-1). In section 3.2.2 we saw that the Callovo-Oxfordian layer had favourable properties in this respect.

Limiting advection contributes to the protection of the repository in the initial phase following closure, during which the rock, then the components of the repository itself re-saturate. The absence of advective motion contributes to slow, better control of the kinetics of the deterioration processes. Limiting advection is also favourable in the longer term, to radioactive nuclide transfer ; in a relatively permeable medium, it could become comparable with or faster than diffusion.

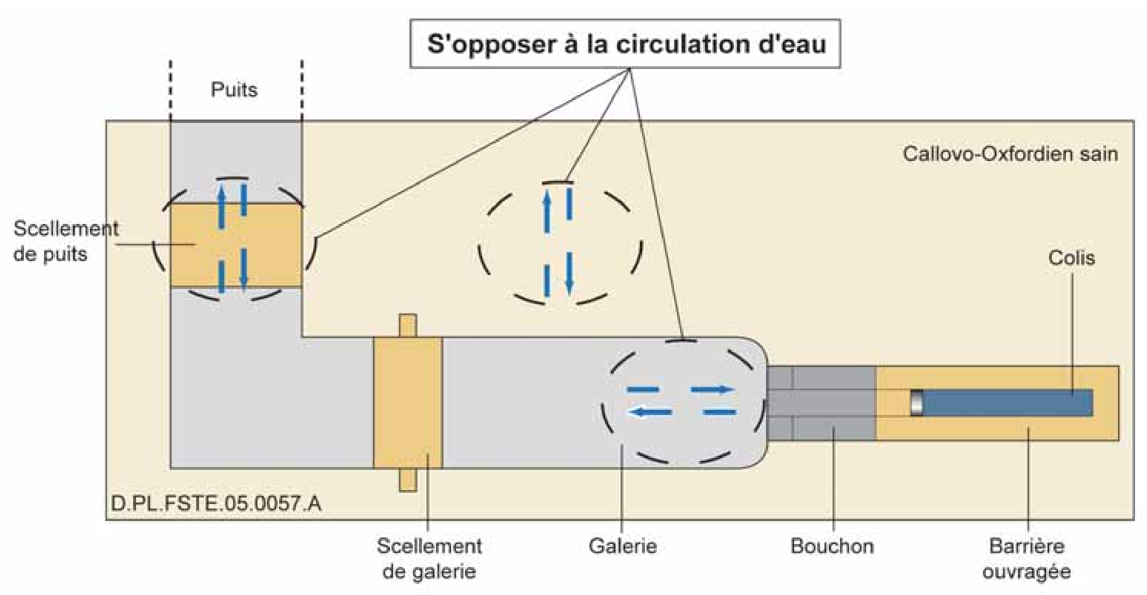


Figure 3.4-1 Resisting the circulation of water

- **« Limiting the release of radioactive nuclides and immobilising them within the repository »**

It is a question of limiting the release of radioactive nuclides and, by extension, of chemical toxins, i.e. preventing them from being placed in solution and, when it becomes inevitable, fostering their precipitation and less mobile chemical forms of the solutes (which we indicate using the term « immobilise »). This function is generally effective in the near-field, as close as possible to the waste packages, where the physico-chemical conditions are found for it to be fulfilled (see Figure 3.4-2). 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 life of the repository but takes various forms depending on the timescales, as mentioned in the example given in the introduction to section 3.3.

This function, at least during the initial phases, uses the favourable properties of the waste packages. In particular, therefore, it reflects the requirement to protect the waste packages from water and place them under favourable physico-chemical conditions.

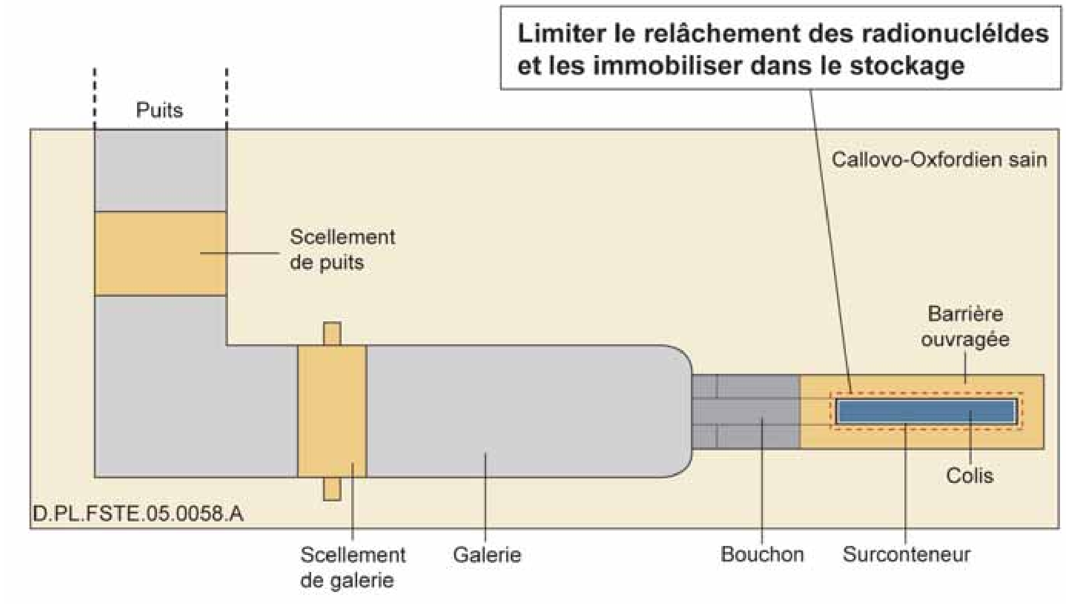


Figure 3.4-2 Limiting the release of radioactive nuclides and immobilising them in the repository

- **Delaying and reducing the migration of radioactive nuclides**

Once the radioactive nuclides are in solution, and on condition that the primary safety function is effective, the dominant radioactive nuclide and, by extension, chemical toxin transfer mechanism, must be diffusion. It is possible to delay and attenuate their transfer.

« Delay » means increasing the time taken to transfer them to the surface, which enables their impact to be reduced thanks to radioactive decay. By « reducing » their migration we mean reducing in the dual sense, in both time and physical extent. For a given mean migration time and a constant quantity, the greater the distance covered by a radioactive nuclide flow, the less noxious it becomes and the longer it takes to arrive in the biosphere (see Figure 3.4-3 and Figure 3.4-4).

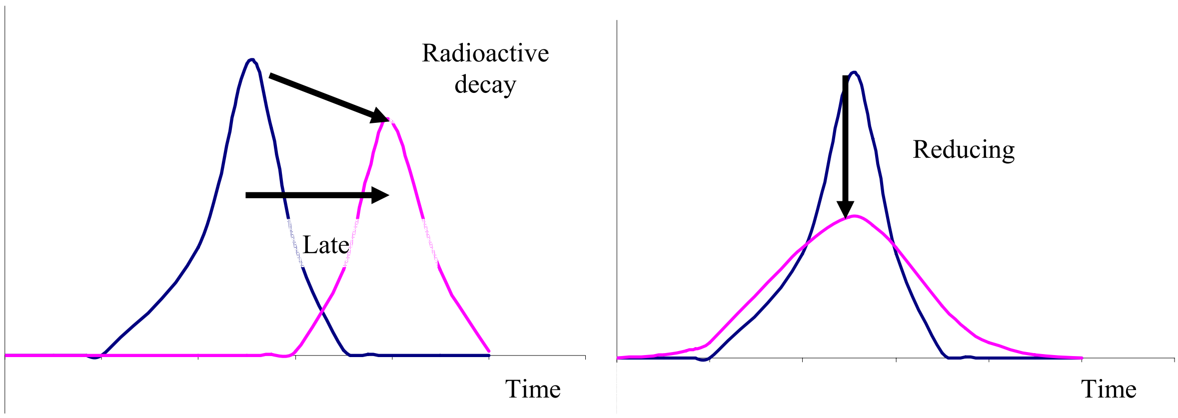


Figure 3.4-3 Illustrative diagram of the delay and reduction of a concentration of radioactive nuclides in solution over time

This function does not apply as long as the « limiting the release of radioactive nuclides and immobilising them in the repository » function is fully effective (as diffusion has not yet occurred) but begins to act as soon as the first releases occur. It is also available in the event of a premature loss of confinement due to an incident. For that reason, it is said to be « latent » in the early years of the repository's life (see this notion in the next section).

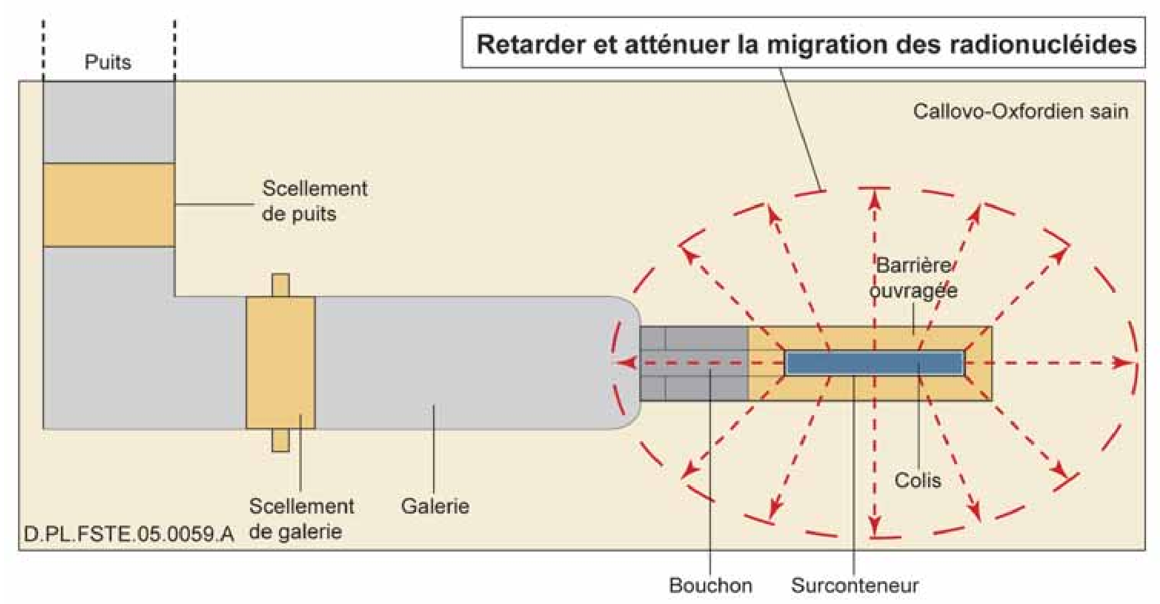


Figure 3.4-4 Limiting and reducing the migration of radioactive nuclides

3.5 Functional analysis methodology

The purpose of this section is to describe the procedure for conducting the internal functional analysis of the repository, i.e. the manner in which the functions at the most general level, as described above, are broken down according to timescales and physical extent into sub-functions accomplished by specific repository components. That makes it possible to describe repository architecture and explain the requirement that each component has to fulfil.

Each safety function can in fact be broken down into sub-functions and so on, to a level of detail that the designer considers sufficient with respect to these requirements, in order to characterise and specify the repository's components. The requirements themselves depend on the project's level of progress.

The functions are broken down into technical solutions using a defined « system », i.e. within the limits in which the designer proposes to act. Apart from the system (as previously defined and including all « engineered works »), one must take in account its environment, as made up of all elements whose characteristics and behaviour are taken « as is ».

Insert 2 Definition of the repository system

The repository system consists of all the components that contribute to the concentration – confinement strategy : the Callovo-Oxfordian host formation and the engineered structures introduced into the repository by humans. The host formation clearly cannot be « designed » as it pre-dates the repository project. However, it was chosen in preference to others for the feasibility study and we expect it to fulfil an important role with respect to safety functions.

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 certain types of waste contributes to confining radioactivity, generally because it has been chosen for this purpose by the waste producer, either with a view to long-term storage (for example the matrix of vitrified waste) or for operational safety reasons (for example spent fuel cladding). By extension therefore, we include waste matrices within the repository system.

However, the under and overlying formation layers are not part of the system. Even if some have favourable retention properties, they have not been selected as a priority for that purpose, retention being provided first and foremost by the repository and the host formation. Furthermore, their properties depend more on the unknown factors of the surface conditions (erosion, climatic events etc.). However, they are the subject of characterisation programmes.

The breaking down of functions into sub-functions is not, in principle, unique. It reflects the designer's choice. It is based on :

- The current knowledge of the behaviour of repository components, which provides confidence in their ability to fulfil their assigned functions.
- Experience feedback from earlier safety assessments, that have confirmed or not the benefits of certain safety functions compared to others and have, in particular, made it possible to identify external events or internal stresses which might endanger the correct operation of the repository, and against which it is possible to make constructive provisions.

The breakdown into functions therefore reflects the result of the designer's thought processes. It develops gradually as the design progresses. Once the overall functional context has been established, the design is revised and the fine detail added in order to enable the safety functions to be fulfilled. The research programme is aimed, in particular, at the phenomena that underlie the achievement of the functions (for example : corrosion for the container sealing function, the formation monitoring programme for its confinement properties, etc.).

As a minimum, each function is characterised by :

- A performance level, i.e. a quantification of the effectiveness of the action expected. However, it is not necessarily relevant in principle to fix a performance level. It is only valid if it can be used to establish the dimensions of the components that have to fulfil the function. If the function has to be fulfilled by at least one component that is beyond the designer's control (for example the geological medium) or if the link between dimensioning and performance depends on the functioning of the entire system (for example, the permeability of a given seal certainly influences the limitation of water flows but within a larger whole depending on other parameters), there is little point, in principle, in fixing a performance level ;
- A period during which the function has to be available ;
- One or more of the components that have to fulfil the function, and the physical phenomenon or phenomena that enable these components to fulfil it. In the particular case of safety during the post-closing phase, given the long timescales involved, only the host formation, waste packages and engineered components introduced by humans (seals, containers, back-fill etc.) are considered to be components with a safety function. The other elements present in the repository, due to the operational conditions or to its natural evolution (functional clearances within disposal cells, corrosion gases generated within the repository, etc.) cannot fulfil a function as there are too many unknown factors concerning their long-term evolution.

Depending on the case, a function may :

- Be available, possibly 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. But identifying reserves gives confidence in the fact that the system has a better level of safety than that which is strictly predicted and quantified ;
- Be available with a performance level better than that taken into account by the designer. We then talk about a performance « margin », i.e. the designer does not use all the performance capability which could be expected to be available to him. The existence of margins also improves confidence levels. The existence of a phenomenon that improves safety but which is not taken into account as a function can be considered as either a reserve or a margin, depending on how you see it ;
- Finally, a function can be latent, i.e. it does not act due to the existence of another function. For example, the confinement provided by the matrix of a waste is latent as long as it has not been subjected to the action of water, i.e. as long as the container protects it. The existence of latent functions makes it possible to manage accidental losses of functions (for example, in this case, a loss of the container's sealing integrity).

An illustration of margins and reserve functions is given in Figure 3.5-1.

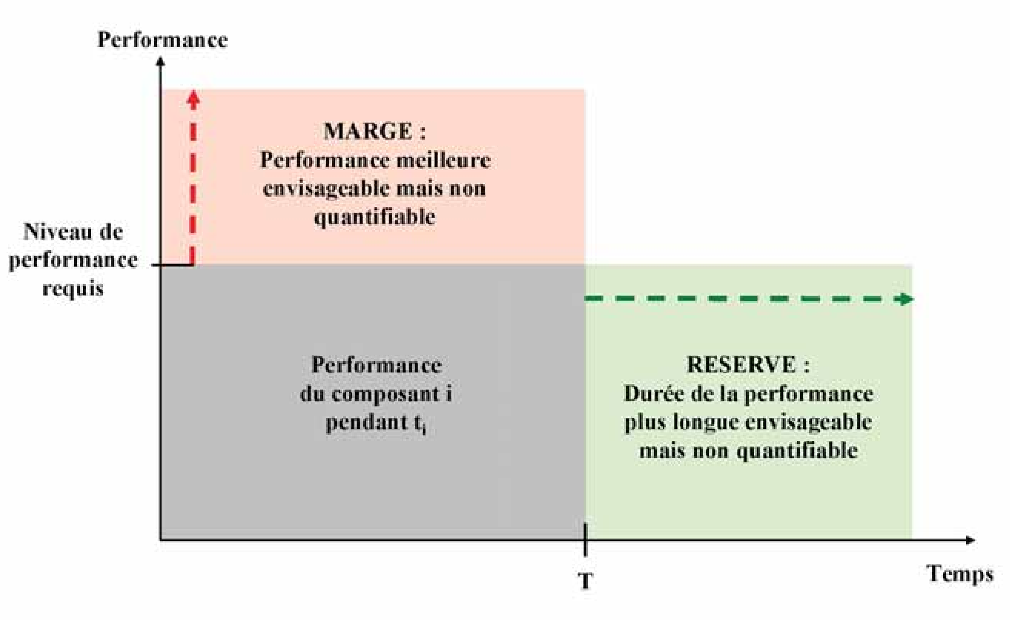


Figure 3.5-1 Illustration of margins and reserve function

In the remainder of this chapter, the current state of the design is described. At this stage, we will not seek to justify the fact that the design is able to meet the safety objectives nor check the performance level of each function ; that is the purpose of the later chapters on performance assessment. The aim here is to explain the range of safety functions proposed by the designer within the repository system, check that they are complementary to each other and identify the existence of redundancy, margins, reserves and latent functions. We also explain the safety strategy used by the designer to guide his choices throughout the design development process.

In the description of the function, we identify the design provisions and principal physico-chemical phenomena linked to each functions. 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). In some cases, they may be neither and simply have to be taken into consideration. Once designed, the check to ensure that the system can stand up to a wider range of disturbances and individual phenomena, without necessarily prejudging whether they are favourable or unfavourable, is the subject of later safety analyses (chapter 5 and subsequent chapters).

Insert 3 Functional analysis procedure implemented

This insert explains the procedure in accordance with which the safety function breakdown structure was established. It is not essential to the understanding of the results of the analysis but helps to understand what level of systemisation it provides.

The establishment of the safety functions is the result of an internal functional analysis, both during the operational phase and in the surveillance and post-closing phases [24, 25]. In the first case, the analysis is based on the experience feedback from installations required to manage high-level waste or spent fuel packages in order to define safety functions conventionally used in such cases.

During the post-closing phase, it was necessary to define a procedure that provided a systematic survey of the functions, within a context in which there is less experience feedback. ANDRA decided to apply a procedure based on the organisation of functions into a breakdown structure [39] and the identification of « flows ». A function being a component's effect on its environment, this effect can always be interpreted in terms of flow management. For example, a confinement function consists of slowing down or blocking a flow of radioactive nuclides. A heat dissipation function controls a thermal flow.

Identifying safety functions then comes down to identifying flows that have to be controlled. The flow of radioactive nuclides and chemical toxins within the repository is, of course, the most obvious flow but others also have to be taken into account :

- The flow of water within the repository, insofar as the concept of storing within clay is based on minimising water circulation ;
- The flow of thermal, chemical or mechanical stresses if they are likely to disturb the qualities of the components

Consequently, the procedure consists of following the major flows and ensuring that functions allow them to be controlled. This check is not sufficient to guarantee that the functional arrangements cover all eventualities, since by definition they cannot be complete : it reflects a designer's choice amongst all the possible ways of defining and arranging safety functions. However, it makes it possible to confirm that the analysis is coherent.

In order to illustrate the procedure we can, for example, explain how the three main functions allowing the risk of dispersal of radioactive nuclides by water to be managed (already discussed in section 3.4.3) have been derived.

The risk is that associated with the action of water. It is therefore a question, initially, of « harnessing this flow », i.e. controlling its onset. An initial function must therefore make it possible to ensure that any water circulation is under control and that the flows are limited (« resisting water circulation » function). This flow is then « transformed » : the water is liable to become charged with radioactive nuclides. The means of resisting this phenomenon must therefore be defined, i.e. of resisting radioactive nuclides from entering into solution and being transported by the water (« limiting the release of radioactive nuclides and immobilising them in the repository » function). Finally, all « incoming » flows must « exit » the system. This « exiting » must also be managed. A function makes it possible to ensure that the radioactive nuclides circulate as slowly as possible and that the flows are reduced (« delaying and reducing the migration of radioactive nuclides » function). The monitoring of flows using the « harness/transform/restore » method makes it possible to guarantee the systematic nature of the functional breakdown structure.

Furthermore, the design is « constrained » by elements that are beyond the designer's control. These include :

- The recommendations in the basic safety rule RFS III.2.f (limitation of water flows, protection of waste packages etc.) which guide the design by directing the main choices ;
- Certain objectives that are not derived from the repository safety objective but which the designer deems necessary to meet in addition. For example, this may involve resisting events occurring which, without directly contributing to accelerating or increasing the flows of radioactive nuclides, may endanger or complicate the safety analysis. For instance, preventing a long term criticality accident within the repository means that it is not necessary to study in detail the potential consequences (as heat, radiation, …) of the components performance in such a reaction configuration ;
- Requirements other than those concerning safety, for example those associated with reversibility

All of these elements are considered as being « constraints ». The constraints are mentioned in the functional analysis for reference purposes. In certain cases, they direct the breakdown of functions into sub-functions.

Other methods would probably have led to a different arrangement of the safety functions and a different expression of the constraints. However, insofar as the functional analysis establishes the current state of knowledge and the designers choices, the list of functions identified at the end of the analysis would have been similar.

3.6 Analysis of the safety functions during the construction, operational and observation phases

This section deals in general terms with the safety measures adopted during the construction, operational, observation and repository closure phase.

Construction operations are similar to conventional mining techniques, whose specificity essentially involves taking the safety constraints during the operational phases and in the long-term into account at this stage. One example is taking care to avoid damaging the walls of drifts and cells, so as to improve safety during the post-closing phase. These special provisions are discussed later in this chapter, in section 3.7.3.1, in the context of long-term safety.

It should also be noted that the safety assessment document is organised so as to bring together in a dedicated chapter, chapter 4, all problems relating to operational safety (preventive and protective measures, risk analysis, assessment of potential accidental situations). In this section, therefore, we remain at the generic stage. Only the major safety options adopted are discussed and their applicability to the functions to be fulfilled.

3.6.1 Confining radioactivity

As far as personnel protection against internal exposure is concerned, the basic safety rule RFS III.2.f. recommends that the waste is packaged in a non-dispersible form. The possibility of atmospheric release resulting from processing operations, the release of radioactive nuclides or the placing in suspension of labile contamination from packages must, however, be taken into consideration

The requirement to confine radioactivity in the repository leads, firstly, to the prevention of contamination, which can be caused either by contact with surfaces themselves contaminated, or by the emission of gases and radioactive aerosols.

Thus, we need to make provision for detecting any sources of contamination (ruptured spent fuel assemblies, surface contamination on means of transport, containers and primary packages), provide preventive measures to stop such contamination occurring (for example, by refusing to accept damaged unprotected assemblies) or take corrective measures (by decontaminating).

Primary waste packages liable to release radioactive nuclides in gaseous form are made as gas-proof as possible, except in cases where the emission of radiolysis gases would lead to over-pressures or inadmissible concentrations. In that case, neither the primary waste packages nor the additional packaging which can be added to them on entering the repository are sealed, and special attention must be paid to ventilation.

Generally speaking, the installation can be sub-divided into zones appropriate to the confinement hazard.

3.6.2 Protecting personnel from irradiation

Radiological exposure of personnel can be optimised as part of the detailed repository studies, taking into account technical and economic factors (ALARA principle).

For the general public, given the distance from surface nuclear installations, the only potentially significant exposure appears to be due to emissions via the chimney. The impact of radioactive gases released by packages has thus been the subject of an initial assessment (see chapter 4). We also need to consider radon, which could be emitted naturally by the soil, then drawn in by the ventilation, which might be installed during the development of an eventual repository project.

Personnel protection involves the dimensioning of radiation shields and the definition of the access and occupation conditions of the various spaces subjected to radiation. In chapter 4 there is a more detailed description of the provisions appropriate to each configuration, normal or accidental.

A classification system can be set up, sorting spaces according to their radiation hazard.

As far as mobile sources are concerned, particularly waste package transport casks, it is not intended to provide a permanent work station in the immediate proximity during transfer operations for these sources. The dose rates are thus limited.

3.6.3 Remaining sub-critical

All waste packages and installations remain sub-critical under normal operating conditions, by design and by deterministic modelling. Studies of accidental configurations are also being conducted, making it possible to check the sub-criticality of the full range of operating situations.

We are currently checking that waste packages and installations remain sub-critical using various test models (geometry of the layout of fissile materials, mass of fissile materials and, where applicable, the concentration if liquid phases are present). More detailed information concerning the conclusions of the criticality studies are given in chapter 4.

If even it is necessary to check that there is no criticality hazard for type B and C waste packages, the main hazard is that posed by spent fuels. It should be noted that, for reasons essentially associated with criticality, the concept is limited to four UOx assemblies per container, or one MOX assembly. The criticality studies (see chapter 4) demonstrate that such an arrangement ensures that container sub-criticality is maintained throughout the operational period. Only a few UOx fuels, if they had low combustion rates (representing a very small portion of assemblies), would be required to be disposed individually. An assembly combustion rate measuring installation would be built on the surface, designed to identify such situations.

3.6.4 Evacuating the residual thermal power from waste

The evacuation of thermal power is mostly associated with the requirements of the post-closure phase : in particular, it is a question of avoiding placing waste matrices or repository materials under temperature conditions that may be detrimental to their long-term evolution.

Personnel protection criteria will also be defined. In order to comply with them, ventilation may play a role in evacuating the residual thermal power.

3.6.5 Evacuating gases released by radiolysis

These gases are evacuated by not leaktight of waste packages or the associated additional packaging and by fitting an appropriate ventilation system in order to meet the criteria expressed in terms of the fraction of explosive concentration limits in the air, or limiting and mean toxic gas exposure limits.

In particular, if any type B waste package recovery operations were to be undertaken, it would be necessary to ensure beforehand that the atmosphere within the cell is conducive to human intervention and to make provision for evacuating the gases if necessary.

3.7 Analysis of safety functions during the post-closing phase

Here we again mention the safety functions during the post-closing phase, by introducing the architectural provisions made to fulfil them.

3.7.1 Isolating waste from surface phenomena and human intrusion

The decision to construct the repository at depth (in this case at least 500 metres) distances the waste from phenomena such as erosion and day-to-day human activity. On a timescale of hundreds of thousands of years, this only affects a shallow depth of surface ground. The choice of a zone located clear of volcanic influences prevents any waste from rising to the surface due to the action of such phenomena. The risks associated with major vertical movements within the repository due to internal geodynamics have also been studied [17] and allow us to exclude the possibility of any movement liable to significantly displace the waste. Glacial and sub-glacial phenomena, in the context of the Meuse / Haute-Marne site do not have any significant effect at depths greater than 150 metres. More generally, the basic safety rule RFS III.2.f. recommends that the site be stable with respect to tectonic and climatic phenomena liable to affect it for a period of at least 10 000 years. It would appear that the host formation has been stable for far longer periods [17], and there is no indication that this situation might change in the next million years.

In this context, waste can only rise up to the surface by deliberate human action, in the form of drilling. This risk is limited by the absence of any exceptional mineral resources in the vicinity of the repository, which makes any underground prospecting to exploit mineral deposits or water sources highly unlikely. From this point of view, the Meuse / Haute-Marne site and its region offers guarantees in accordance with the recommendations in RFS.III.2.f. : it is well away from the coal and oil resources of the Paris bassin. The exploitable coal resources begin further north of the sector, in the extension of the Saar-Lorraine bassin ; as far as oil is concerned, investigations undertaken notably by the « Institut Français du Pétrole » (French petroleum institute) in the 70's and 80's indicated that the areas of interest are located further west towards the centre of the bassin [17, chapter 8]

Elsewhere, as far as any geothermal resources are concerned, the most recent cartographic data obtained by the « Bureau de Recherche Géologique et Minière » (French geological and mining research bureau) show nothing of particular interest in the area of the Meuse / Haute-Marne site, which corroborates ANDRA's own observations by drilling. The formations underlying the Callovo-Oxfordian layer are insufficiently water-bearing to be of interest, the potential resources being located further east in the Nancy region. They also have a high to very high salinity level, making them unsuitable for geothermal exploitation [17, chapter 9].

It should be noted that, in the context of defence in depth and in order to take account of any risk of intrusive drilling intercepting the repository zone, including if this drilling is not motivated by searching for natural resources, this type of event is nevertheless taken into consideration in the design (see section 3.7.3.1), and an intrusive drilling scenario is also dealt with in the safety analysis (see chapter 7).

3.7.2 Preserving the repository's records

From the beginning of the repository project, starting at the initial research phase, the repository's records are established progressively, by conserving the knowledge and data linked to the design of the repository, the justification for the design, the associated safety demonstration, building of the repository then repository operation, with the management of the waste disposal packages placed in it, finally the progressive closure of the repository structures until final closing of the repository. The minimum overall duration for these phases will be about a hundred years, or much more, depending on the required duration for reversibility.

To manage this knowledge, the repository's designer, then the operator, could use a knowledge base and databases. This system could be used for the maintenance of the installations. To prevent any possible loss of the knowledge accumulated, several different methods could be used to protect the knowledge base records themselves as well as the tools and equipment necessary to exploit them. Redundant archiving on separate backup media stored in different places is one of the methods generally used to conserve records and guard against accidental loss. The data conservation should not pose a problem while the repository remains in its operational phase, because the operator will need to use the knowledge acquired and accumulated almost daily.

After this first main phase, which lasts for about a hundred years, we can illustrate the preservation of records by reference to feedback from the measures applied or planned at the Manche Repository, where disposal of the final packages dates back to 1994. This example will not, however, be used to dictate the procedures to be applied for deep disposal. The preservation of the records for the Manche Repository relies on four items : establishment of detailed records and a record summary for the repository, restrictions recorded in the Land Registry, and communication arrangements.

The detailed records of the repository consist of a selection of the accumulated knowledge managed until the end of disposal. This selection is done primarily in function of the analysis of the risks identified, to guarantee that no knowledge concerning the risks will be lost ; then to guarantee complete traceability of events connected to these risks, and finally for purely historical purposes. These detailed records, besides recording the existence of the repository, also allow the technical management of the repository and understanding of all the phenomena that may be observed – an issue that is more obvious for surface disposal than for deep disposal - and to apply appropriate solutions when necessary. The detailed records are conserved both at the site and in the National Archives, in the centre for contemporary archives. Use of special media, such as archival paper (see Figure 3.7-1) will permit conservation of the documents for from three to five centuries. Duplication of the records at regular intervals could also be considered.

Figure 3.7-1 Detailed records made on permanent paper and conserved in the French National Archives, in the centre for contemporary archives

The record summary is an additional element for long term protection proposed by the Turpin Commission in 1996 [40] for the assessment of the situation at the Manche Repository. It consists of briefly describing, in a single archive insert if possible, the history and topography of the repository, the complete inventory of the wastes disposed there, the precautions to be taken, the risks incurred… The purpose of this record summary, widely distributed to local, regional and national decision makers, is to provide them with easily accessible and easily understood information about the repository, in order to allow them to take any necessary decisions about the future of the repository in full knowledge of the facts, especially for any development work on the surface.

Restrictions could be recorded in the Land Registry, for example with the purpose of forbidding any drilling or boring within the repository's perimeter. Because of the importance of the Land Registry in land management, we can reasonably assume that these restrictions would be observed for several centuries, a duration similar to that planned for the record summary.

The Turpin Commission also proposed to extend a local information and surveillance committee after the operating phase, in order to maintain the dialogue between the operator of the repository or the body responsible for monitoring it, the local state representatives, the local elected representatives and representatives for the population and associations. This dialogue would, in addition to its other purposes, allow the memory of the existence of the repository and that of the two record sets described above to be conserved over successive generations. From this point of view, these provisions constitute part of defence in depth.

The transfer of these elements to the deep disposal case leads us to consider that a reasonable duration for the preservation of the records of the repository is about five centuries, as proposed by the basic Safety Rule III.2.f. Further into the future, it would be difficult to guarantee that the two sets of records would be regularly and correctly duplicated without loss of information. For the case that the record of the existence of the repository needs to be preserved for a longer period, international studies are currently underway to research the effectiveness of surface markers (for example megaliths or other structures) over a span of several thousand years. These studies come up against the risk of theft or deterioration of the surface markers, and, even if they were to be correctly preserved, the difficulty of maintaining the true signification given to them by our generation.

3.7.3 Resisting water circulation

Here we consider the control of water flow within the repository system. This depends mainly on the properties of the rock, especially on its low permeability. A set of design provisions allows these properties to be conserved and reconstituted where they may have been disturbed.

The deep geological formations are initially saturated with water. For the Callovo-Oxfordian argillites, this is interstitial water intimately bound to the material itself. The excavation of the repository structures partly desaturates the environment locally, and disturbs the hydraulic pressures. The return to equilibrium takes place progressively after the closure of the repository.

During a transitory period after the closure of the repository's underground installations, a hydraulic transient is established for a time span that varies according to the structure being considered (see section 3.2.4. Afterwards, the flow globally follows the natural hydrogeological gradient. Renewal of the water next to the waste is the main factor likely to deteriorate the package and permit the release of radionuclides within a repository. Circulation of this water is also a potential vector for advective transport of the radionuclides

Therefore, a primary set of sub-functions have the purpose of limiting the circulation of water within the repository ; they consist of :

- Limiting the water flow from the overlying geological formations penetrated by the access structures (See section 3.7.3.1) ;
- Limiting the water flow from the host formation of the Callovo-Oxfordian (See section 3.7.3.2) ;
- Limiting the velocity of the water flow between the repository and the overlying or underlying formations (See section 3.7.3.3).

3.7.3.1 Limiting the water flow from the overlying geological formations penetrated by the access structures

This function controls the water flow within the repository by limiting the progression of the water in the privileged pathways created by the backfilled access passages in the repository. These connect the overlying formations, more or less aquifers, with the repository itself. Here the aim is twofold : before resaturation, this function slows the arrival of water at the waste packages, and also makes sure that the saturation of the structures is controlled mainly by the water flow from the host formation, whose chemistry is better understood. After resaturation, it contributes to limiting the water flow within the repository.

In the first instance, grouping of the access shafts is an option retained at this stage of the project. The location of the grouped shafts, offset with respect to the repository zones, nearly cancels out the load difference between them within the formations crossed that are better aquifers and the associated component in the structures, linked to the horizontal gradient within the aquifer. Only the vertical gradient counts, rising or descending within the argillite.

- **Limiting the damaged zone**

Once this overall arrangement has been decided, the different pathways possible for the passage of water within the structures should be studied more thoroughly. In addition to the access passages themselves (backfill and linerlining left in position), the argillite zone damaged by the excavation, neighbouring the structure, also provides a potential privileged pathway.

The mechanical behaviour models distinguish three zones within the niche : the fractured zone, the microfissured zone, the influenced zone [7].

The fractured zone, which is closest to the structures, appears if the rupture threshold of the argillites, corresponding to the maximum strength of the rock, is exceeded ; it is characterised by the appearance of fractures that are more or less connected. These could cause an increase in the permeability of the rock.

The microfissured zone appears either behind the fractured zone, or directly in contact with the structures if the fractured zone does not exist. The mechanical load reduction linked to the excavation of the structures causes distortions that result in the form of diffused microfissuring, which is not very connected. The fact that the microfissures are not very connected limits the increase in permeability.

The influenced zone that extends out from the microfissured zone contains only limited modifications of the stress field, which have no effect on the properties of the rock (in particular its hydraulic properties). In practice, it is not distinguishable from sound rock.

The term « damaged zone » or the acronym « EDZ » are used to designate the group made up of the fractured and microfissured zones.

A comparison of this model with the observations made in the drifts of the Meuse/Haute Marne laboratory is covered in section 6.2.6.1.

The fractured zone can constitute a privileged pathway for water flow, in the same way as the backfill. This is less probable for the microfissured zone, because the damage is not very connected.

Two types of design provision are implemented to limit the water flow in the drifts. The first types aim to limit the formation of the damaged zone, particularly the fractured zone, or to prevent its extension after the excavation, in the short and long term. The others aim to intercept the access passages.

The excavation techniques to be used for building the repository have not been fixed at this stage of the project ; a variety of mining techniques exist that allow damage to be limited ; they have already been proven in other contexts. As an example, note that to create large diameter shafts, the traditional method consists in sinking the shaft from the surface by shattering the rock with explosives or with a road header, and putting into place the ground support and the shaft liner as the work progresses. This method is quite suitable for the shaft diameters planned, but could be combined with other methods if necessary.

For the drifts, the road header technique (see Figure 3.7-2) or excavation with explosives in solid seams, could be suitable methods, depending on circumstances.

Figure 3.7-2 Example of a high power road header (AC-Eickhoff type ET 380)

The orientation of the access drifts holding the seals and the cells in the direction of the main major stress is another provision that allows the extent of the damaged zone to be reduced. The placing in position of the ground support, then of the liner in the cell well, reduces delayed distortions of the rock, during the operating phase and long afterwards, until the concrete progressively loses its mechanical strength. This must, however, be carried out in such a way as to avoid causing additional damage, especially at the level of the future seals (see later).

In the shafts, the concrete ground support is bolted into place. To minimise damage to the rock in the seal zones, the excavation method could be adapted. Rather than bolts, pneumatically applied concrete and/or arches could be used. In the drifts, maximum rigidity of the liner is required in this zone, in order to minimise the extent of the damaged zone. Bolts are not planned to be used near the seals.

The damage could, a priori, be accentuated by the conditions prevailing during operation : oxidation of the rock, increased desaturation of the walls caused by ventilation of the structures, etc. These phenomena have little influence and do not modify the nature or the scale of the fractured zone. In the framework of reversibility, it appears prudent not to restrict for the time being the time span during which the drifts could be left open, in order to leave future generations with the greatest possible flexibility for the organisation of the management of the repository. The strategy is, therefore, because of the low impact of oxidation of the rock, to assess the influence of the oxidation on the rock without a priori taking any design measures to prevent it, other than the usual protection measures (lining in order to cover the bare rock).

During operation, shortly after closure, the damaged zone could evolve due to drilling of the rock. Basic Safety Rule.III.2.f. recommends the study of its mechanical properties, which are « important criteria ». The argillites are characterised by very slow creep, which is favourable [17]. The mechanical support function carried out first by the liner, then by the backfill and the seals in the very long term, limits delayed damage to the argillites. The backfill can be made up of concrete or of argillites from the site, recompacted and mixed if necessary with swelling clay, which by resaturating take the load caused by delayed creep of the rock, after deterioration of the liner.

At the level of the disposal cells, the function of protection of the rock is filled by minimising the spaces. The residual space around the disposal packages does not exceed 5 % of the total volume (see Figure 3.7-3). This is compatible with the handling of the packages inside the cell and with their retrievability. In the long term, when the disposal packages no longer perform their mechanical support function, the spaces to be taken into account include the clearances due to positioning the packages, together with the spaces of the materials inside the primary packages and the porosity.

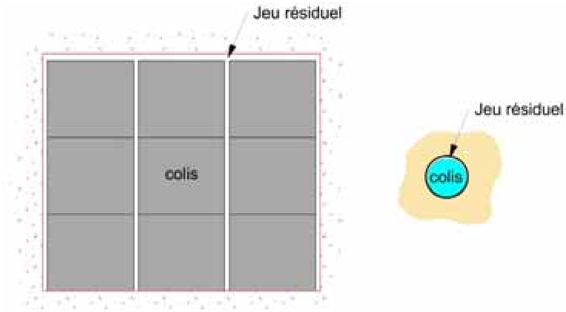


Figure 3.7-3 Residual space inside the cells (for B waste cells- left and C waste cells - right)

Once the fractured zone has been formed, measures should be taken to avoid its long term deterioration by any causes other than actions due strictly to rock movements : chemical interactions, mechanical interactions caused by gases, etc. One constraint that is considered is linked to possible thermo-mechanical distortions near the C waste or spent fuel cells. This is one of the reasons for limiting the heat emitted from the wastes and controlling the heat flow (see section 3.7.6).

- **Hydraulic control of the access passages**

It is necessary to restore low permeability within the access passages in order to limit the circulation of water. This function is carried out by specific devices - the seals – and not by the backfill, which could a priori also have been considered for this function.

The backfill used in the shafts and drifts could limit the water flow due its own permeability characteristics. At this stage of the project, however, the backfill is not considered to carry out this function. It would be possible to obtain a permeability of about 10^{-8} m/s but this is not a significant parameter for the project. However, the ground support left in place in the shafts and drifts constitute zones on the edges of the backfill that may have significantly higher permeability when they deteriorate. They could bypass the backfill, whatever permeability this had.

The hydraulic control function therefore depends on the different seals put into place in the structures : shaft and drift seals, B cell seals. This is also an auxiliary function of the C and spent fuel cell plugs, but is not their primary function, which will be described later. These structures are made of a substance that has very low permeability, swelling clay (bentonite, swelling clay « MX80 ») together with quartzose sand, which enhances mechanical strength. Once this has resaturated it :

- Interrupts the access passages themselves ;
- Makes a very low permeability contact with the rock wall, by creating pressure on the wall as it swells.

This pressure is enough to compress the fractured zone when it is of limited extent, and to restore good hydraulic properties.

The layout of redundant seals within the different access passages (shafts, drifts, modules) allows the repository to be separated into modules with a high degree of hydraulic independence. In a very general way, this layout guards against the propagation of an incident affecting a small part of the repository to the entire installation. A key example of this type of incident is drilling through one module of the repository : it is expected that its influence would only affect a limited number of modules. The effectiveness of this protective layout can therefore be tested in the framework of the « drilling » altered evolution scenario (see chapter 7).

The shaft seals are constructed in the upper part of the host formation, which contains more carbonate. This layer corresponds to a geotechnical horizon (horizon A) with more favourable mechanical characteristics. Current knowledge leads us to predict that the excavation of shafts will not create a fractured zone in this argillite horizon. For the construction of the shaft seals it is planned to remove the liner over sections of several metres in length, in order to place the swelling clay in direct contact with the argillite.

The drift seals will be located at the repository's main level, in the geotechnical horizon C, a median horizon containing more clay, with characteristics different from those of the horizon A and a depth of 500 metres at the level of the laboratory site (see Figure 3.7-4).

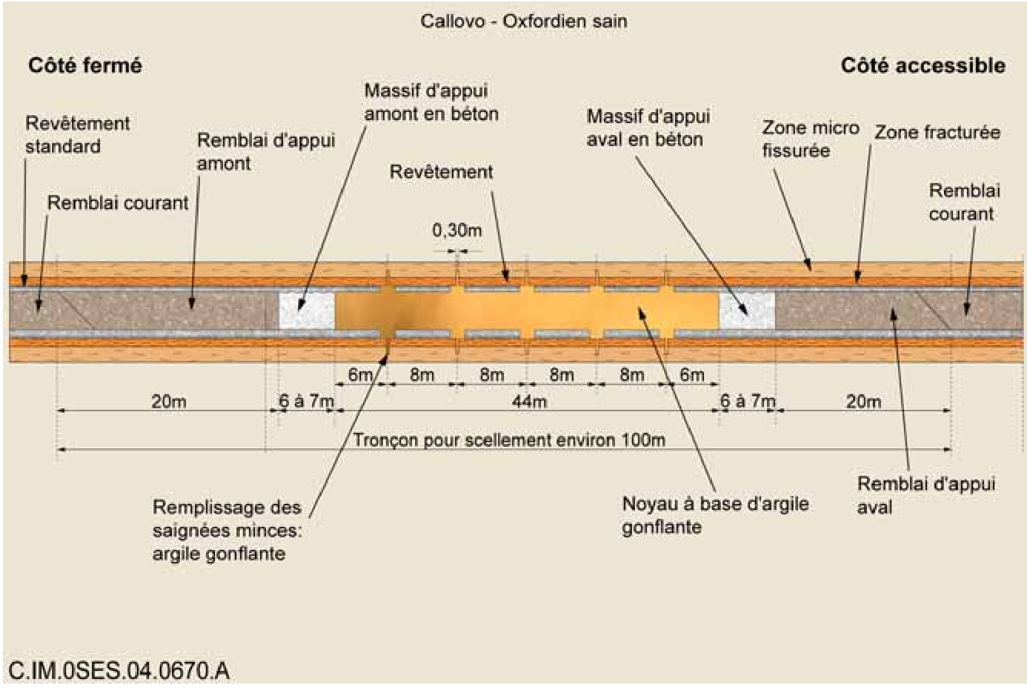


Figure 3.7-4 Schematic diagram of the drift seals

The diameters of the access passages (particularly the drifts) are such that the fractured zone that could develop, proportional to their radius, could become large compared to the seal body itself. It may then be necessary to block it with a hydraulic cutoff, integrated with the seal body.

The procedure for placing the drift and B cell seals in position consists of creating a groove in the rock and inserting a hydraulic cutoff made of bentonite. When this saturates, it applies enough pressure to the geological environment to create a discontinuity in the fractured zone and to locally clear the zone [41]. Methodology tests carried out at Mont-Terri have demonstrated the technological feasibility of placing this type of cutoff in position [42]. Counted from the excavation wall, its depth can reach that of the microfissured zone. Andra has tried to minimise the cross section of the structures to be sealed, in particular to limit the extent of the fractured zone.

The seals are built against a support base or concrete backfill. This aids pressurisation as they swell under the effects of resaturation. Around the concrete support bases in the drifts, the addition of quartz in the backfill raises its friction coefficient, again with the aim of promoting swelling of the seals in the radial direction.

The performance of the bentonite seals can be sensitive to perturbation caused by water that has percolated within the concrete (alkaline perturbation) or which has become loaded with iron (iron-clay interaction). These interactions are assessed and may lead to special protection arrangements for the seals : removal of ground support in the vicinity, oversizing the seal body length to obtain an extra mass of bentonite that serves as a buffer, use of « low pH » concrete near the seal if this is deemed necessary.

The list of measures taken at this stage of the project to perform the sub-function « limiting the water flow from the overlying geological formations » can be summarised as follows :

- Grouping of the access shafts ;
- Control of the excavation and ground support techniques ;

- Orienting the structures in the direction of the major constraint, as far as is possible (particularly for the structures in which the seals and plugs are placed) ;
- Putting low permeability seals into position ;
- The mechanical support function of the backfill and seals ;
- Minimising the empty space inside cells and disposal waste packages (except for the primary packages) ;
- Interrupting the fractured zone with hydraulic cutoffs near the seals, or further out, as far as is necessary ;
- Dimensioning the seals to take into account the alkaline or ferrous perturbations.

Note that these measures are for the most part the same as those that perform the function « limiting the velocity of water flow between the repository and the overlying or underlying formations ». The same measures serve both purposes : limiting both the arrival of water and the possibility for the water to leave the repository system.

Note that a large number of additional measures (with the purpose of controlling the EDZ or reducing the overall permeability of the access passages) also contribute to this function. Some redundancy is provided by the presence of systems of seals in series (in the shafts and the drift).

3.7.3.2 Limiting the water flow from the Callovo-Oxfordian's host formation

In parallel to control of the access passages, another sub-function has the purpose of limiting the circulation of water coming from the host formation itself. This function us provided by the choice of a rock, the Callovo-Oxfordian, which has very low permeability, is homogeneous enough, and which also has very low internal hydraulic gradients. These characteristics are compatible with the recommendations of The basic safety Rule III.2.f. They were the subject of an on-site research programme, by drilling and in situ inspection, with the aim of confirming and specifying these favourable properties : permeability measurements, checking for the absence of particular conductive structures.

This programme supports the conceptual models of the host formation, used particularly for the hydraulic calculations (see chapter 5).

In addition, the measures already described above (limiting the fractured zone, placing seals in position in the access structures) also prevent the repository from draining water from the Callovo-Oxfordian in the same way as they prevent water circulation coming from the overlying formations.

The list of measures taken at this stage of the project to perform the sub-function « limiting the water flow from the host formation » can be summarised as follows :

- Choice of a host formation with low permeability ;
- A survey programme of the host formation, from the surface, in the laboratory, and possibly during the construction of the repository ;
- Measures already described above (seals, limiting damage, etc).

3.7.3.3 Limiting the velocity of water flow between the repository and the overlying or underlying formations

Once the repository has been partially or totally resaturated, the aim is to reduce the velocity of water circulation out of the repository. This function has very similar effects to the first two. All the design provisions already described in sections 3.7.3.1 and 3.7.3.2 contribute to this. It is not therefore truly redundant with the first two, it complements them.

The function consists of slowing the water circulation, first from the zones resaturated initially towards less saturated zones, then in the direction of the natural flows within the formation. If this function is effective, the migration of radionuclides does not have any privileged direction and takes place in an isotropic way within the entire formation. The seals have an important role in this function.

Note that, in order to offer at least partial redundancy with the seals in case they become defective, Andra has also retained an architectural layout in the form of a hierarchy tree of dead-end components, whose principle is shown in Figure 3.7-5. At each level of the hierarchy tree, (from the repository zone to the cell), each component is reached from only a few access points that are grouped together. The flow drained towards the access structures in each dead-end component is thus limited to the flow that it can exchange with the argillite, and water flow rates are therefore reduced. The flows do not really group together until they are a long way from the cells, at the level of the drifts.

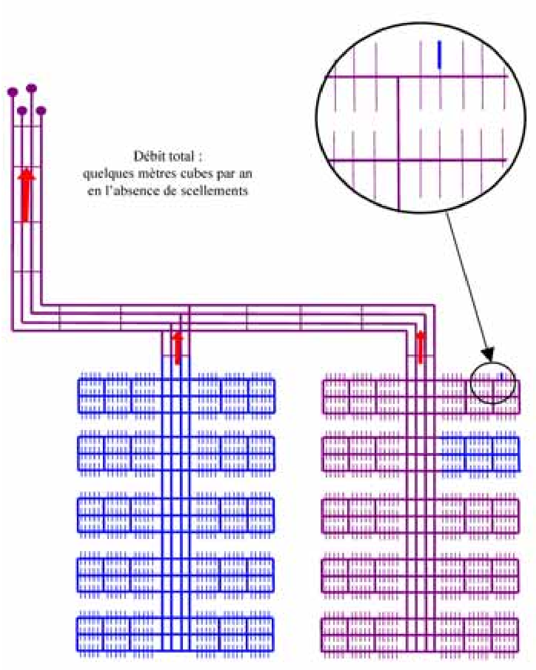


Figure 3.7-5 Dead-end component breakdown

The overall performances of the three sub-functions finally depend on a group of parameters :

- Some are linked to the site, such as the permeability of the rock and the hydraulic gradients at the repository's location ;
- Some are linked to the organisation of the repository : layout of the shafts, architecture of dead-end tunnels ;
- Others are technological, linked to the expected performance of the seals. An overall permeability equivalent to 10^{-10} m/s seems to be a minimal target that can be reached. As an example, measurements carried out at the Bonnet Lake laboratory in Canada, during the TSX test [43], demonstrated a permeability equivalent to 10^{-11} m/s for a bentonite seal ;
- Finally, others are linked to disturbance that could be caused by other components of the repository.

The final purpose of the hydraulic calculations within the repository is to assess the performance of these functions.

3.7.4 Limiting the release of radionuclides and immobilising them in the repository

The function described here is a near field function. It consists either of physically blocking the transfer of the radionuclides, or of converting them to a physico-chemical form that blocks their movement. It thus depends particularly on the notion of confinement, which is usual for safety functions in nuclear installations. In the case of deep disposal, confinement by the insertion of a physical barrier cannot last forever, due to the deterioration of the exogenous materials within the repository. The usual meaning of « confinement » is therefore extended to the notions of « limiting release », and « immobilisation », which respectively take into account the function of the waste matrices that retain the radionuclides and the capacity of the surrounding environment to encourage the reduction of the solubility of the radionuclides. Note, however, that the functions of slowing the transfer, which consists of slowing the transport of the radionuclides without really stopping it, are not included in the phenomena described here.

This definition of the function allows it to be maintained for the entire lifespan of the repository, though it changes, depending on the phases, in the way in which it is performed by the components providing the function (first by the conditioning of the wastes, then by the wastes themselves and the chemical conditions in the cell).

The conditioning of the wastes, as it exists, is taken as input data. The repository must conserve the favourable properties of the conditioning, or add defence lines to those provided by the initial conditioning. This is mainly done by adding container components that are specific for the repository.

This function is facilitated by measures adapted to each type of waste. The operation of the repository, and the conservation of a clearance thickness between waste modules of different types allows the physico-chemical conditions within each module to be optimised, without having to take into account any possible interference coming from other types of wastes. This consists, for example, of separating the bituminous wastes, that can be sensitive to heat, from the thermal influence of the C wastes and the spent fuel, and if necessary from some other B wastes (minimum distance 250 metres).

The list of general measures taken at this stage of the project to perform the function « limiting the release of the radionuclides and immobilising them in the repository » can be summarised as follows :

- Management of the repository by module according to the physico-chemical nature of the wastes ;
- Sufficient separation of different types of waste modules in order to guarantee their phenomenological independence.

The analysis is conducted per waste type.

3.7.4.1 In B waste disposal cells other than bitumens

Waste cells other than bituminous sludges are considered first. These are the B1, B3, B4, B5, B6, B7 and B8 waste cells (see chapter 2). The nature of these wastes is variable. The concept globally takes account of this variability. Some characteristics are particularly favourable for the protection of metallic wastes (cladding hulls and end cans, mass activation product wastes, etc.). These are therefore identified where necessary.

Several sub-functions are distinguished.

- **Protection of the wastes from corrosion.**

This sub-function is specific for metallic wastes. The toxic components and the radionuclides are either present in the form of contamination at the surface of the wastes or included in the mass. The waste's structure has, in the second case, the function of immobilising the components, and it is corrosion that controls their release. The first sub-function is therefore to maintain favourable conditions for limiting the corrosion within the cell. This is done by controlling the oxidation-reduction potential and the pH. As corrosion is inevitable, no a priori performance level is required for this function. The aim is to have the best possible conditions for reducing the corrosion rate and promoting the passivation of the steel ; an alkaline pH promotes this kind of passivation under reducing conditions. These measures are also valid for other metals presents within the wastes, such as zircon for example.

The standard concrete container, identical at this stage of the project for all the B wastes (see Figure 3.7-6), has been defined for ease of handling and for homogeneity. It also provides functions for the post-closure phase. In particular, it has an effect on pH regulation, together with the concrete assembly making up the B waste cells. Taking into account the masses concerned, an alkaline pH (from 10 to 12.5 in stable conditions) is imposed in the entire environment. The perturbation of this pH by the components released by some of the wastes, mainly the organic acids coming from the bitumens, is managed by placing the metallic wastes in separate cells assigned for them. The presence of organic wastes in the CSD-C packages can have an influence, but this is negligible due to the quantity concerned.

As a precaution, other waste packages containing organic material could be disposed in dedicated disposal cells as bitumens.

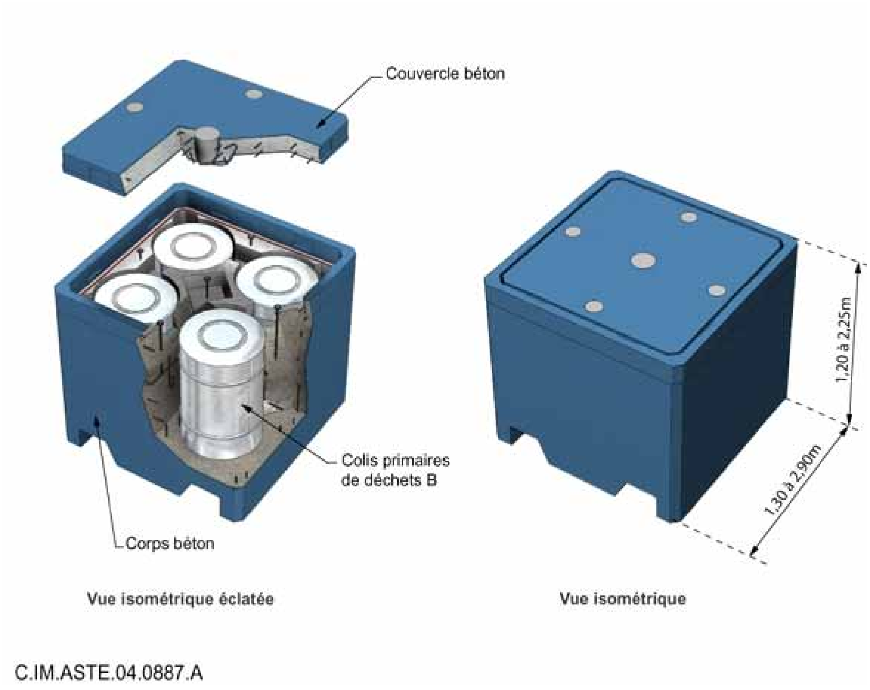


Figure 3.7-6 Schematic diagram of the B waste over-packing

The sealing of the B waste over-packing also contributes, as a complement, to the protection of the wastes against the oxidising phase associated with the operation of the ventilation – which, however, corresponds with relatively dry conditions. Once the cell has been closed, the environment becomes globally reducing, due to the absence of oxygen and the conditions imposed by the argillites.

The performances required for this function finally depend on the state of knowledge about the release models that can be associated with the different types of wastes. These models will be described in the framework of the performance calculations, in chapter 5.

The list of measures taken at this stage of the project to perform the sub-function « protection of the B metallic wastes from corrosion » can be summarised as follows :

- The choice of preferring concrete for the B waste packages and the cell liners ;
- Limiting the presence of organic components in the cells and segregating the waste.

- **Limiting the solution of the radionuclides**

Once the radionuclides (and the chemical toxic components) have become separated from the mass of the wastes, the aim is to limit their solution. The same conditions as those created in order to limit corrosion are required : an alkaline to neutral pH that globally favours limiting the solubility, and a reducing potential. Effectively, only a few elements (iodine, chlorine, and caesium) are not sensitive to these favourable chemical conditions. However, many elements are not very soluble in a cement-based environment [21].

Limiting the quantity of organic components in the B waste cell (both by the low quantities of organic wastes and by the absence of organic matter in the construction of the cells) allows the quantity of complexing substances that could form and facilitate the solution of the radionuclides to be minimised.

Note that this sub-function, like the preceding one, depends on the cement-based environment of the B cells, and the chemical conditions that this imposes. Due to the masses of concrete concerned, and the non-disruptive characteristics of the water from the Callovo-Oxfordian, which are close to neutral, these favourable conditions seem to be particularly robust.

The measures taken with respect to this sub-function are the same as for the preceding one.

- **Filtering the colloids**

The limited quantities of colloids present in the cells could increase the mobility of some toxic components in the packages. The term colloids includes a group of products that can have several origins and forms. The different types of potential colloids can be divided into :

- Intrinsic colloids : these are formed by the polynucleation and condensation of some of the actinides following the hydrolysis of the wastes. Their formation is closely linked to the environmental conditions (redox, pH…). Away from their zone of production, these colloids are generally found in chemical environmental conditions that impose thermodynamic instability ;
- Carrier colloids : these are the natural colloids possibly presents in the pore water of the argillites (fine mineral particles and organic molecules), the colloids released by the materials used for the construction of the structures (particularly the swelling clay and the cementitious material...) or created in the structures during the construction and operation phase (proliferation of bacteria, oxidation of the argillites…). There are also the colloids produced by more or less long term phenomena due to the presence of the packages (dissolution of glass, radiolysis…) and/or the cell materials (deterioration of the concretes, interaction of the alkaline front with the argillites, corrosion of the containers…).

As it is difficult, in absolute terms, to prevent their formation and release, the designer's choice is to create the conditions that allow colloids to be filtered in the near field. This is done by on interposing an environment with low porosity all around the cells. This is provided by the geological environment and the seal of the B waste cell. This seal, with a construction technique similar to that of the drift seals, comprises hydraulic cutoffs in the fractured zone at the entrance of the cell in order to close it (Figure 3.7-7). A seal thickness is provided on both sides of the anchor keys to buffer the alkaline perturbation coming from the concrete in the support base (also called the « concrete plug ») and the cells in order to preserve the structure's properties.

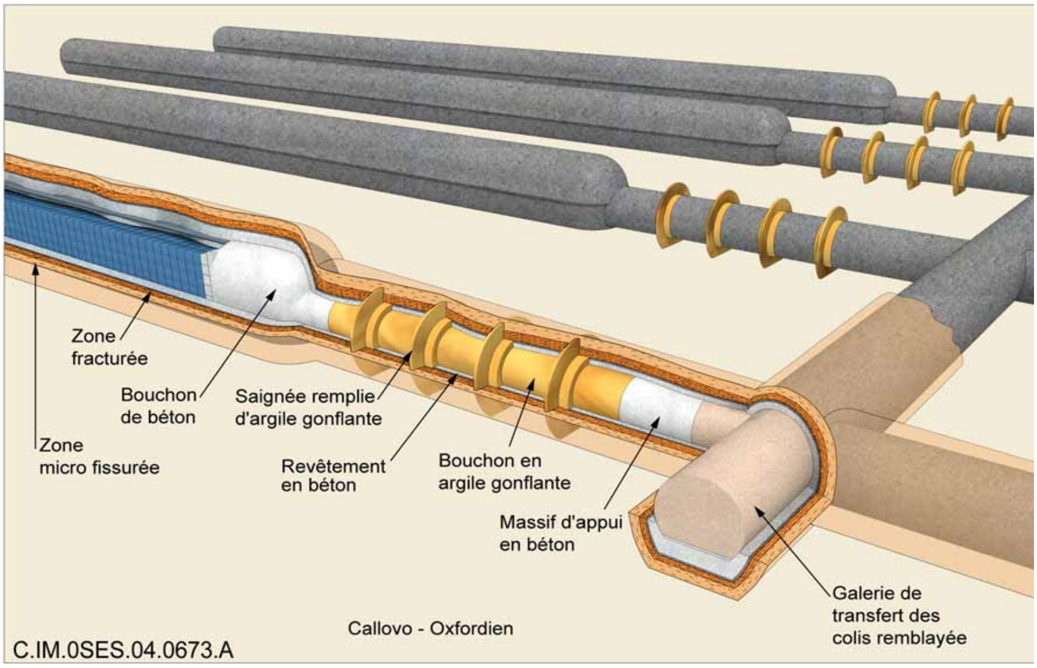


Figure 3.7-7 Schematic diagram of a B waste cell with a plug anchored in the micro-fissured zone

The list of measures taken at this stage of the project to perform the sub-function « filtering the colloids » can be summarised as follows :

- Choice of a geological formation with low porosity ;
- Closure of the cells by a seal interrupting the fractured zone.

3.7.4.2 In B2 waste cells (bitumen-embedded materials)

The B2 waste cells have the same types of function as the other B cells. Their particularity lies in the fact that extra design provisions have been made to favour the confinement capability potentially provided by the bituminous matrix. These are based on research carried out by the Atomic Energy Commission [20], which demonstrated some of the factors that allow this capability to be mobilised.

In order to create favourable conditions for control of release, it is first of all important to limit as far as possible the surface that the bituminous matrix presents to water penetration. Bitumen has a tendency to creep, and the release models are only valid in a field that has controlled geometry. The first sub-function retained is therefore to preserver the bitumen's dimensional stability. This is not provided only by the B disposal package, which is not designed specifically to maintain the geometry of the bitumen over its entire life span. It is the overall capability of the cell, its compactness and the minimising of the spaces, which ensure the preservation of the geometry of the bitumen-embedded materials. This function is not allocated a quantified aim, as perfect maintenance of the geometry of the bitumen is difficult to guarantee. The measures taken do, however, tend to favour this.

The evacuation of the radiolysis gases as they are produced, by the unsealed packaging, also contributes in preserving the matrix from distortions. Swelling of the bitumen itself, however, cannot be prevented. This is taken into account by the models.

Apart from controlling the geometry of the embedded waste, preservation of the retention capabilities of the bituminous matrix also demands that the homogeneity of the embedded waste is maintained. On the one hand, this involves avoiding the sedimentation of the salts (to which radioactivity is associated) while, on the other, enabling the embedded waste to retain deformability (by the salts taking in water) without fissuring. These two requirements are conveyed by a function consisting of maintaining a temperature of 20 to 30°C in the cells containing such waste. Note that the tolerance for this function is large - if the temperature criteria are slightly exceeded, for example during the surface storage phase, this does not have a significant effect.

Finally, compliance with a pH range of 7 to 12.5 is imposed to remain within an area where the behaviour of the embedded waste is known. In practice, a pH of 10 to 12.5 as for other B waste is imposed.

The separation of the bitumen cells from exothermic wastes, and the presence of large quantities of concrete in the cells, are all favourable factors with respect to these requirements. The concrete, in particular, buffers the effects of any organic acids released by the bitumens.

Finally, the functions already identified for the other B wastes, consisting of limiting the near field solubility and filtering the colloids, also come into play and have no specific aspect. Note, however, that the bituminous wastes originate, at their primary generator, from effluent sludges after treatment by solubility reduction. We can therefore expect that the radionuclides contained in them would not be very soluble. However, because of the difficulty of demonstrating the permanence of the speciation of the radionuclides within the matrix, then in the repository, this component is not taken into account as contributing to the immobilisation function. It adds a safety margin.

The degradation of the bitumen does not produce any organic species capable of imposing their complexing power in a cement-based medium [44]. The cement-based medium also limits the properties of the complexing agents incorporated into the matrix and their degradation products (legacy bituminised waste at Marcoule with a particular tributyl phosphate content, for example).

The list of measures taken at this stage of the project to perform the sub-function « limiting the releases from the B2 wastes » can be summarised as follows :

- Compactness of the B2 waste cells to favour the dimensional stability ;
- Evacuation of radiolysis gases ;
- Use of concrete for the packaging and the cell ground support ;
- Separation from exothermic waste cells.

3.7.4.3 In the C waste cells

For the vitrified C waste package, the first function is to prevent the arrival of water in contact with the glass during the period characterised by a relatively high temperature, which has a maximum length of about a thousand years. Release of the radionuclides should be prevented as long as the temperature does not allow us to reliably account for the behaviour of these radionuclides ; taking into account current limits of knowledge. The glass must also be protected against the risk of an increase in its deterioration in contact with the water at a temperature above about sixty degrees, in line with the currently considered behaviour models [20]. This function is performed by an overpack (see Figure 3.7-8) that isolates the glass package from the water, and whose life span covers the phase during which the temperature within the glass exceeds 50-60 °C. The lifetime of the overpack is estimated at 4000 years based on conservative assumptions (by assuming immediate resaturation, and by taking into account a phase of oxidising corrosion conditions before the changeover to reducing conditions.

Note that the current design of the overpack could in fact last longer, for about 15 000 years, by considering less pessimistic hypotheses and taking into account the current estimated resaturation kinetics and the presence of components that protect the container, for example the metal lining installed in the cell wall, which corrodes first. However, the main purposes for installing the lining are to facilitate operation and in order to favour easy recovery of the packages in the context of reversibility. The lining is not assigned for an overpack protection function, this is just a safety margin.

The life span of the overpack is controlled by its corrosion ; this is why the overpack is made from unalloyed or weakly alloyed steel, a material that has the simplest and most predictable behaviour under the effects of corrosion. Note that, for the same reason, no function is assigned to the stainless steel primary container, which therefore adds an extra line of defence, but whose effectiveness is, however, difficult to quantify. The primary container might not resist the radiolytic corrosion during the first few hundred years, or the hydrostatic and geostatic pressures applied. It therefore constitutes protection that is more of a qualitative nature.

Figure 3.7-8 Schematic diagram of a disposal package for vitrified C reference packages (glass, container and overpack)

The behaviour of the overpack is characterised by several phases :

- During the phase when it may undergo oxidising conditions (operation, and at the start of the post-closure phase until the oxygen is consumed by the surrounding environment), its rate of corrosion could potentially be rapid. The cell is therefore sealed as soon as possible by an operating plug and a protective cover, which strongly restrict renewal of the air inside the cell. This does not constitute an obstacle for reversibility, so long as these components can be easily removed, which is also a requirement in order to place the permanent cell plug in position (see later). During this phase, the aim is also to place the overpack in best possible physico-chemical conditions ; in particular, it is insulated as much as possible from the other metal components (by fitting a ceramic pad that separates it from the lining) ;
- At the same time, during the first few hundred years of the overpack's life span, it may be subjected to radiolytic corrosion caused by the package's internal radiation. The thickness of the metal available during this period is designed to protect it from this ;
- After the oxygen has been consumed, the conditions around the overpack become anoxic and reducing, even if the cell is not yet totally resaturated. In this case, the corrosion is uniform and at slower rate.

Extra thickness is provided to ensure the mechanical strength of the container for the required length of time. The container is designed to resist a pressure of 12 MPa, corresponding to the assessment of the weight of the overlying rocks and the pressure of the interstitial water.

The total design thickness of the overpack at this stage of the project is 55 mm.

Beyond the thermal phase, other sub-functions contribute to creating conditions around the glass that allow its behaviour to be controlled.

First of all, the mechanical cohesion of the glass must be conserved for as long as possible, both to avoid damage that would increase its internal fracturing and to promote any possible phenomena of self sealing of the internal fracturing, by silica precipitation when the glass is in contact with water. This function relies on the handling conditions for the vitrified waste packages, which must protect them from damage. Once they are inside the cell, this must provide the glass packages with an environment having the best possible mechanical stability.

In addition, limiting the heat release from the glass packages that enter the repository, and the temperature within the repository, motivated by the aim of protecting the geological environment (see section 3.7.6.1), also contributes to favouring the integrity of the packages by limiting the thermomechanical distortions in the cell. The quantity of packages per cell, and using inert separators between them also contribute to the dissipation of heat.

The deterioration of the glass is also controlled by the dissolution rate of the silica, which decreases as the silica concentration rises in the surrounding water. Measures should be taken to encourage the maintenance of a chemical equilibrium of the silica content between the water and the glass. This aim is obtained by limiting the transport of the species dissolved in the cell, particularly the silica, in order to encourage its immobilisation as close as possible to the surface attacked by the water. The measures retained to meet this function concern the materials located close to the wastes. In particular, note the existence of a continuous low permeability environment around the packages, with the Callovo-Oxfordian argillites and the bentonite plug placed at the cell entrance. Empty spaces are limited as much as possible. All this contributes to maintaining a diffusive regime inside the cell.

Assessments demonstrate that, due to the small diameter, the fractured zone that develops around the cell should be absent or very limited, so that the pressure applied by the swelling of the plug is enough to block it (without a hydraulic cutoff being required). In order to maintain an environment with very little fracturing, the cells are orientated in the direction of the major geostatic stress, and suitable excavation techniques are used.

The plug is made from bentonite. It is placed at the cell entrance, and it is the swelling of the plug during the resaturation phase that provides close contact with the rock (see Figure 3.7-9). A concrete support base (also termed a « concrete plug ») is placed upstream to confine the bentonite and encourage swelling.

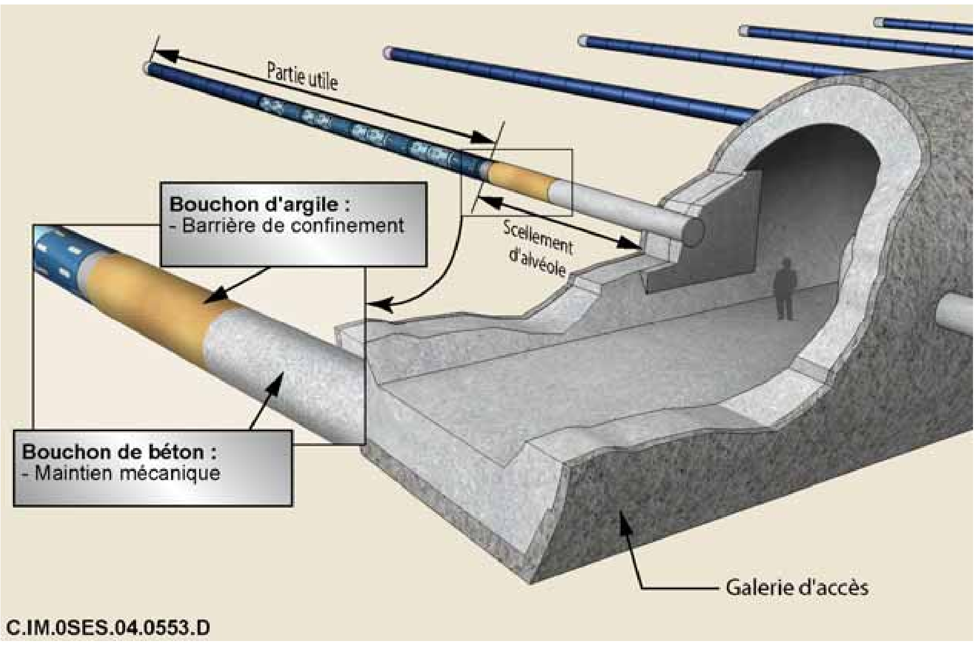


Figure 3.7-9 Schematic diagram of a closed C waste cell

In addition to the measures described above, components that could absorb the silica at the interface with the packages must also be limited. Note, however, that the corrosion products from the overpack could have this type of absorbing capability. Their presence is taken into account in the behaviour models [20].

In addition, the aim is also to control the pH (preferably between 7 and 9), as dissolution of the glass is potentially sensitive to this parameter. This is the reason for which the glass is kept away from any alkaline sources. The concrete in the support base at the head of the cell plug should not have any influence on the pH inside the cell. This requires good plug performance in terms of chemical insulation. The buffer capability of the bentonite, and its properties of low diffusion, come into play, and are designed according to the maximum extent of alkaline perturbation. Perturbation extent is of the maximum order of a metre.

Finally, as in the B waste cells, two general functions control releases out from the cell once the toxic components have left the package :

- Maintaining reducing conditions, in order to immobilise the radionuclides as much as possible. Solubilities in swelling clays (used for the cell plug) are globally low. Reducing conditions are ultimately imposed by the water from the Callovo-Oxfordian, once the repository has been closed ;
- Filtering any colloids produced. This sub-function is provided by the argillites, due to the absence of the fractured zone and the cell plug. Their low porosity prevents the complexing agents produced from progressing towards the drifts.

If necessary, when a fractured zone seems likely to develop, a swelling clay buffer (or clay engineered barrier) could be put into place in the C waste cells. This does not, however, appear necessary at the current stage of the design process.

The list of measures taken at this stage of the project to limit release from the vitrified wastes and to immobilise the radionuclides that they could release can be summarised as follows :

- Choice of an overpack made from unalloyed or weakly alloyed steel ;
- Limiting the oxidising period by protecting the cells with a removable cover, protecting the overpack against localised corrosion risks (particularly by placing a pad to insulate it from the lining to prevent the creation of a localised corrosion zone) and the design of the overpack to take into account an oxidising period lasting for several hundred years, radiolytic corrosion, then a uniform conservative corrosion rate, with built in safety margins ;
- Limiting empty spaces within the cell ;
- Orienting the cells in the direction of the major stress ;
- Control of the excavation techniques ;
- Placing a bentonite plug at the cell entrance ;
- Limiting the use of components that could absorb silica at the interface with the packages ;
- No use of concrete within the cells.

Note that CU3 spent fuels (Research or Defence fuels) would be placed in cells similar to those used for vitrified wastes if they were to be placed in the repository.

3.7.4.4 In the spent fuel cells

The spent fuel cells are designed on the same principle as the C waste cells (See Figure 3.7-10. Both are thermal wastes.

The functions are therefore of the same type, and are based on similar design provisions :

- Preventing the arrival of water on the spent fuel in order to avoid any release into the environment during the thermal phase. This is because the speciation of the radionuclides in water is less know under thermal conditions and because temperature can substantially accelerate the aqueous corrosion processes for the cladding and pellets. The measure retained is a container made from unalloyed or weakly alloyed steel which allows sealing for 10,000 years. This lifetime is estimated under pessimistic conditions, more realistic assessments carried out in conditions closer to the phenomenology (taking into account the unsaturated phases, less conservative corrosion rates) demonstrate that sealing periods of the order of 30 000 years can in fact be expected ;

- Control of pH, as the behaviour of spent fuel is not well known at pH values exceeding 10, by avoiding the presence of cement based materials in the cell and by assigning the plug the function of limiting the progression of the alkaline perturbation ;
- Maintaining a diffusive regime in the cell, to resist the transport of dissolved species. The aim here is not to immobilise the silica, but in a more general way to limit the dissolution of the fuel, by controlling the concentration of the uranium dissolved in the water. Note also that the diffusive regime renders concentration of the fissile material after the deterioration of the containers, and therefore any associated criticality accident, extremely unlikely. Diffusion effectively promotes the isotropic migration of the radionuclides ;
- Maintaining reducing conditions in the cell, to promote the formation of species with low solubility. The homogeneity of the reducing conditions also prevents the formation of geochemical « fronts » that could cause local concentrations of fissile material by precipitation ;
- Filtering the colloids at the level of the cell.

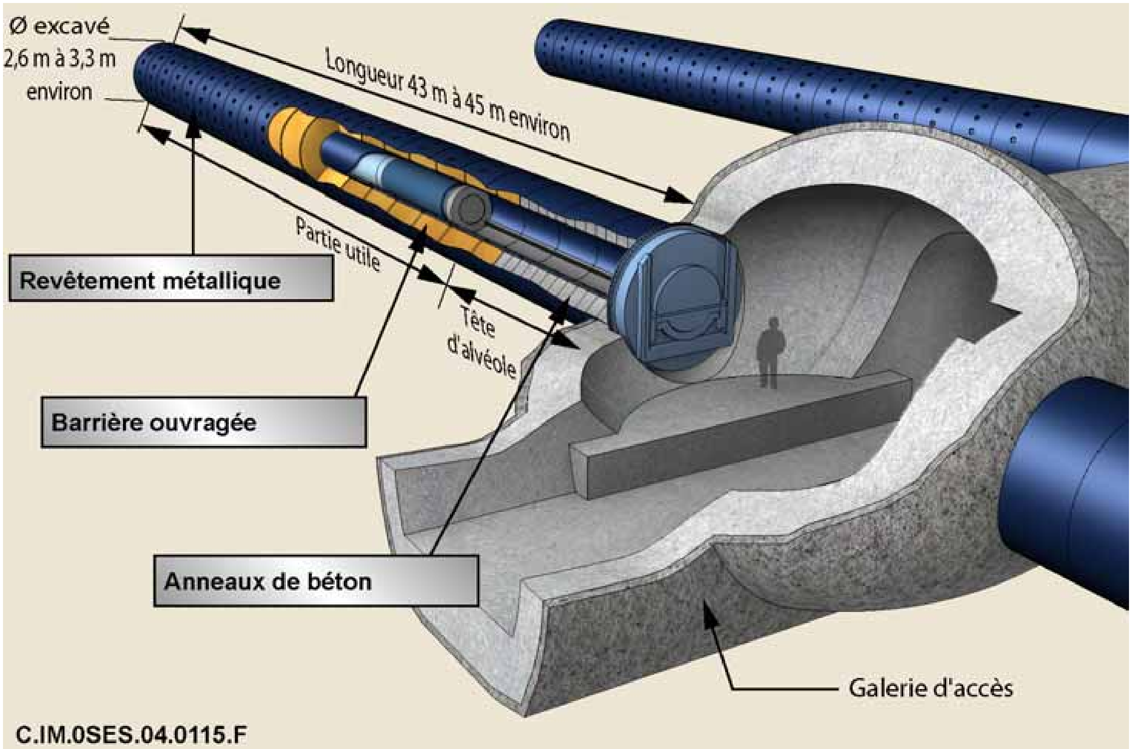


Figure 3.7-10 Main functions of components of the spent fuel cells

In addition, an engineered barrier is planned for the spent fuel. This type of waste has a thermal transient with a long duration. The effects on the argillites of a temperature that significantly exceeds the geothermal temperature over several thousand years are not currently well known. At this stage of knowledge, it seems important to retain extra protection for the rock, by means of a structure composed of bentonite and sand providing a thermal buffer. This also contributes to the maintenance of a diffusive regime in the cell. With the advances in terms of knowledge and the forthcoming observations from the underground laboratory, the benefit of such a buffer option could be reviewed

The spent fuel containers integrate management of the criticality risk.

In the four assembly UOx design, the cast iron insert integrated into the container maintains the package at a sub-critical level, even over the long term (see Figure 3.7-11). The mechanical behaviour of this component, including after the effects of corrosion, renders approach of the assemblies, which could cause a criticality accident, unlikely in the long term. The sub-criticality of the mono-assembly containers is therefore guaranteed, even in the long term.

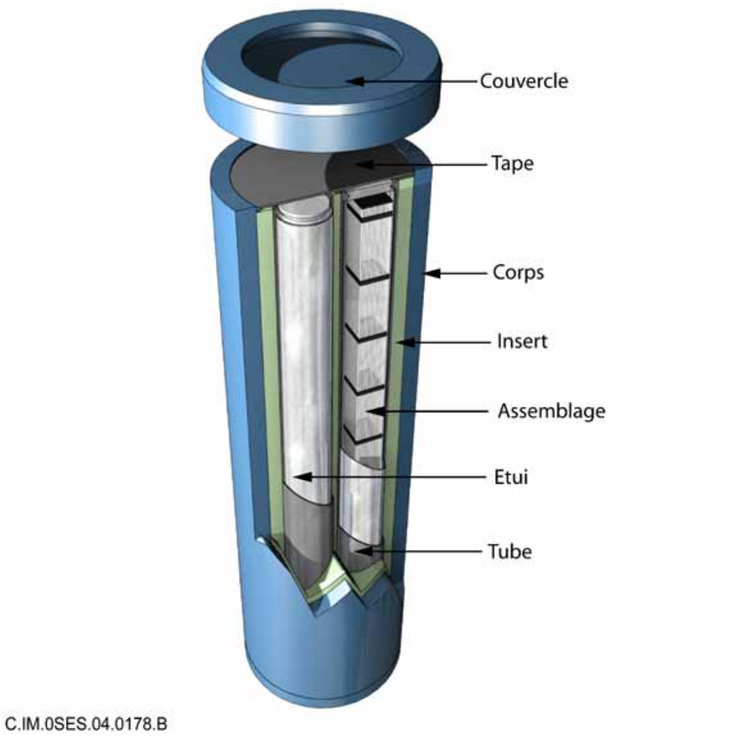


Figure 3.7-11 Schematic diagram of a spent fuel container (UOx)

3.7.5 Delaying and attenuating the migration of radionuclides

The third safety function consists in delaying and attenuating the flow of radionuclides finally released by the wastes to the surrounding geological formations in space and time. This is therefore latent during the first time phase while the radionuclides remain confined in the waste ; once it is active, it mainly mobilises the characteristics of the Callovo-Oxfordian argillites for their transport properties, as well as the performance of the artificial barriers, particularly the seals.

Three sub-functions have the aim of « delaying and attenuating the migration of radionuclides » (see Figure 3.4-3). They have the aim of :

- Delaying the migration of the radioactive elements by diffusion/retention in the host formation of the Callovo-Oxfordian ;
- Attenuating the flow of toxic components in the host formation of the Callovo-Oxfordian ;
- Preserving the capacity of natural dispersion in the repository's environment.

The first sub-function increases the transfer time and attenuates the rate of activity of the radionuclides from their release by the packages to the top or to the bottom of the Callovo-Oxfordian.

The characteristics of the Callovo-Oxfordian that contribute to satisfying this function are :

- A high capacity to limit the flow of radionuclides due to a diffusive transfer mode ;
- A high capacity to chemically retain most of the chemical elements by sorption phenomena (precipitation was already taken into account in the framework of the preceding safety function) ;

- A large clearance thickness of sound argillites between the repository and the surrounding formations. Over the area of the transposition zone, the thickness of the Callovo-Oxfordian layer is at least 130 metres, which guarantees a clearance thickness of at least 60 metres on both sides of the structures. Andra has retained underground installations that do not extend far vertically, organised on a single level, to preserve a considerable this thickness of sound argillite on either side.

These three characteristics contribute to guarantee a long transfer time in the Callovo-Oxfordian, slowing and spreading the flow of activity of elements leaving the Callovo-Oxfordian. The basic safety Rule III.2.f. identifies the geochemical characteristics of the host formation as « important criteria » because « *they control the retention phenomena of any radionuclides released* ». The choice of a suitable site is therefore the first way of meeting this safety function.

Note that diffusion accelerates with increasing temperature (see Figure 3.7-12) and that in a high temperature range it is difficult to reliably predict the chemical behaviour of the elements in solution. Therefore, for the exothermic waste zones, the sealed container or overpack, already mentioned, intervenes indirectly to provide the function of delay and attenuation in good conditions.

However, although it is less effective at higher temperatures, the capacity of the rock to delay and attenuate the radionuclides over time subsists during the thermal phase, and constitutes a latent function, which would come into play particularly in the event of failure of an exothermic waste container.

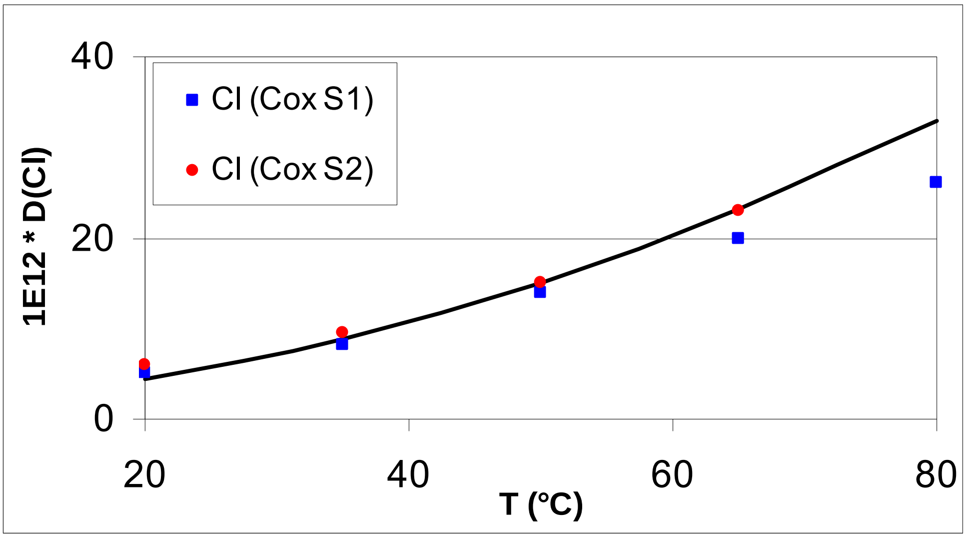


Figure 3.7-12 Experimental results for the evolution of the effective coefficient of diffusion for chlorides in function of temperature for two samples of argillite (Callovo-Oxfordian S1, Callovo-Oxfordian S2)

Note also that the migration of the radionuclides could be delayed inside some of the components of the repository (engineered barrier for the spent fuel cells, plugs, seal bodies). This would be extra performance added to that of the Callovo-Oxfordian, which remains quantitatively the most important under normal conditions[15].

A second sub-function concerns spatial spreading, this consists of preventing the radionuclides from diffusing in privileged directions. This relies on the homogeneity of the Callovo-Oxfordian, and the absence of significant heterogeneity with respect to diffusion within the formation. The function is latent in the first instance, so long as the radioactive elements and the chemical toxics have not reached the formation, then becomes permanent in the second instance.

[15] Therefore, another way to present the function « delaying and alternating the migration of radionucleides » is to distingwish between the role of the host formation and that of engineered components.

The last sub-function consists to protect the natural dilution capacity of the surrounding formations. The dilution itself is not a function provided by a repository component, and it is not an objective as such, because the basic principle of a repository is to privilege confinement. However, in the event that some of the radionuclides could reach the surrounding formations and be diluted within them, it could be useful not to prevent this dilution, as an extra feature and in order to limit the potential impacts even more. The aim is not, however, to promote this phenomenon.

Therefore this function will be limited to taking measures not to disturb the surrounding formations in the long term, as they are already inevitably subject to natural forces (erosion). This consists particularly :

- Making sure that settlement caused by the filling of the spaces within the repository cannot disturb the overlying surrounding formations. Note that the reason for limiting empty space is mainly governed by the requirement of protecting the host formation. The protection of the overlying surrounding formations is a secondary consequence ;
- By isolating the geological formations from one another, for example by not leaving unplugged drillings in place. This type of measure avoids privileged pathways that act as bypasses ;
- Making sure of the absence of significant thermomechanical effects on the surrounding formations caused by placing exothermic wastes in the repository. Again, the protection of the host formation against excessive heating covers this risk ;
- Putting seals and separations of the aquifer formations into place in the connecting structures passing through the surrounding formations to the surface, in addition to the backfill.

The list of design provisions for the function « delaying and attenuating the migration of radionuclides » can be summarised as :

- Choice of a formation having good diffusion and delaying properties ;
- Preserving a clearance thickness of clay formation between the repository and the surrounding formations ;
- Choice of repository architectures on a single level and with low height ;
- Placing backfill in the shafts and providing seals to isolate the aquifers in the surrounding formations ;
- Limiting the disturbance caused to the surrounding formations.

3.7.6 Generic arrangements aimed at protecting the host rock

The construction of functions linked to water transfer has shown the importance of the host formation as the performance of all functions rely upon it.

It is therefore important to protect the properties of the formation (permeability, porosity, mineralogy, geochemistry, etc.) There is no specific safety requirement, but it can be inferred from the necessity of fulfilling the safety function set out in the other sections of section 3.7[16]. The main arrangements made at the repository construction stage and during its operation are set out in this section.

3.7.6.1 Thermal protection

The first design provision is to dissipate heat produced by the radioactivity of the waste, by conduction in the geological medium. The objective at the current state of knowledge is to maintain the temperature in the installations and in the argillites below 90°C at all times and to return to a maximum temperature of 70°C before 1000 years, so as to prevent significant mineralogical transformations and to stay within the temperature ranges covered by current knowledge. This function concerns vitrified C type waste and spent fuels in particular. It must also be considered for certain B type waste, although their heat release is lower than that of highly active packages, and decreases more rapidly over time (the maximum repository temperature of the most active B type packages is less than 70°C and falls rapidly below 40°C). For these cells, the concretes must be protected from overheating, in the same way as for rock.

[16] In this context, the functions described below are sub-functions of the three safety functions presented previously.

This objective necessitates preliminary repository periods for the hottest packages (C1 to C4 glasses, spent fuels from for the existing pressurised water reactors). Moreover, the design also specifies that the most exothermal packages should have adequate spacing (vitrified waste, not including C0 waste, and spent fuel from EDF). To facilitate heat dissipation, spacing buffers are put in place. They are designed in such a way as to prevent further interactions in the cell environment (with the packages, with the engineered components, or with the geological environment). They must therefore be made of similar material to that used for the conditioning and packaging of the waste itself.

The heat dissipation capacity of the materials surrounding the thermal waste must also be carefully considered ; this concerns bentonite engineered barriers in particular.

3.7.6.2 Mechanical protection

Controlling mechanical deformation in the host formation is also important : opening underground cavities, in a medium subject to natural mechanical stresses relating to the weight of the ground, creates deformations in the argillites ; for highly active packages, an additional cause of deformation is the thermal expansion of materials, resulting from the heat given off by the waste. The aim is to limit the damage caused by mechanical deformations, and also their extension. This damage can bring about local increases in permeability, to a greater or lesser extent depending on the amount of damage (microfissuring, fracturing). This relates to the protection measures to be taken against heat flow rate.

In section 3.7.3.1 we have already set out the measures designed to limit mechanical damage of the rock during excavation and in the long term. Orienting the main works in the direction of the main stress, careful choice of excavation techniques, the fitting of ground supports, linings and back-fill according to the required mechanical characteristics.

We also mentioned the desirability of limiting empty spaces in the cells to a minimum. C waste cells and spent fuels are thus made as compact as possible. The voids in B waste cells are filled, except between the packages where filling would be difficult to accomplish. The stacking of packages in tightly packed networks ensures that the void rate remains lower than 5 %.

3.7.6.3 Other disturbances to be taken into account

Hydric (desaturation/resaturation) or chemical (action of cement, iron, or oxidant based products) based disturbances of the rock must also be taken into account.

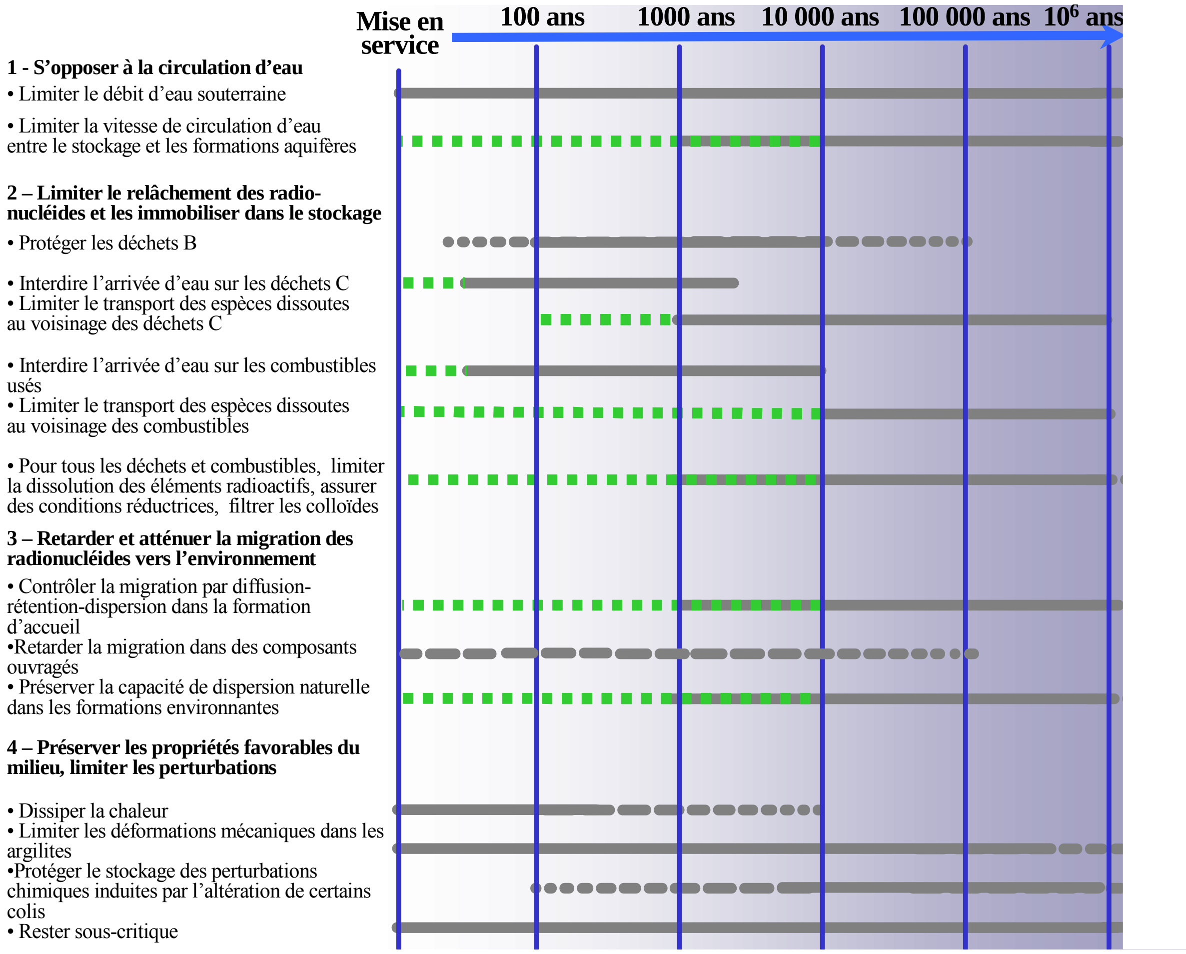
However, no design specifications directly related to these disturbances have been incorporated. The propagation of desaturation and oxidising disturbance is low enough for it not to be necessary to limit the operating period in relation to the possible extension of these phenomena. Also, the « limiting the water circulation » function, which leads to a diffusive regime for the radionuclides and toxic chemical elements, also leads to the same transfer regime for species that may disturb the argillites. Alkaline or iron-clay disturbance is thus managed by means of the same design measures.

These questions of chemical and hydraulic disturbances will be dealt with in greater depth in chapter 6 section 6.2.8.

Finally, the fissile material content of certain packages, requires us to ensure that the repository is kept in a sub-critical configuration, taking account of potential displacements of material and the long-term evolution of material. This obviates the need to assess disturbances in nominal operating circumstances, which would be brought about by nuclear chain reaction effects, on repository evolution and performance : reaction of the medium to thermal and mechanical energy generated, radiolytic effects, formation of additional fission products.

3.7.7 Summary

The diagram in Figure 3.7-13 shows the main long-term safety functions referred to above, and their corresponding time scales.



Legend :

Operating functions — Latent functions

Progressive need of a function — Progressive disappearance of the function

Figure 3.7-13 Safety functions over time

The description of the safety functions points up the existence of three complementary lines of defence which last throughout the analysis : one relies on advection control inside the repository, another limits the release of radionuclides and immobilises them in the repository in the near-field, and the last delays and attenuates flows.

These functions enable us to characterise the role of the components more accurately than would be possible using only the notion of a « barrier ».

One of the aims of the analysis of operational hazards (chapter 4) and of the qualitative safety analysis in post-closure (chapter 6) is to check whether there are causes of failure that can compromise the planned safety functions. The robustness of the system can nevertheless be affirmed at this point. It is based on :

- The different components : host formation, shaft seals, drift seals, cell plugs, over-pack and disposal containers, waste matrices,
- The different types of measures : control of the construction of structures, general organisation measures for repository, construction measures, natural properties of the site ;
- The redundancy of certain components, essentially seals installed in series ;

- The availability of reserve functions, for example the confinement capacities offered by metal containers, apparently greater than the reference capacity taken into account.

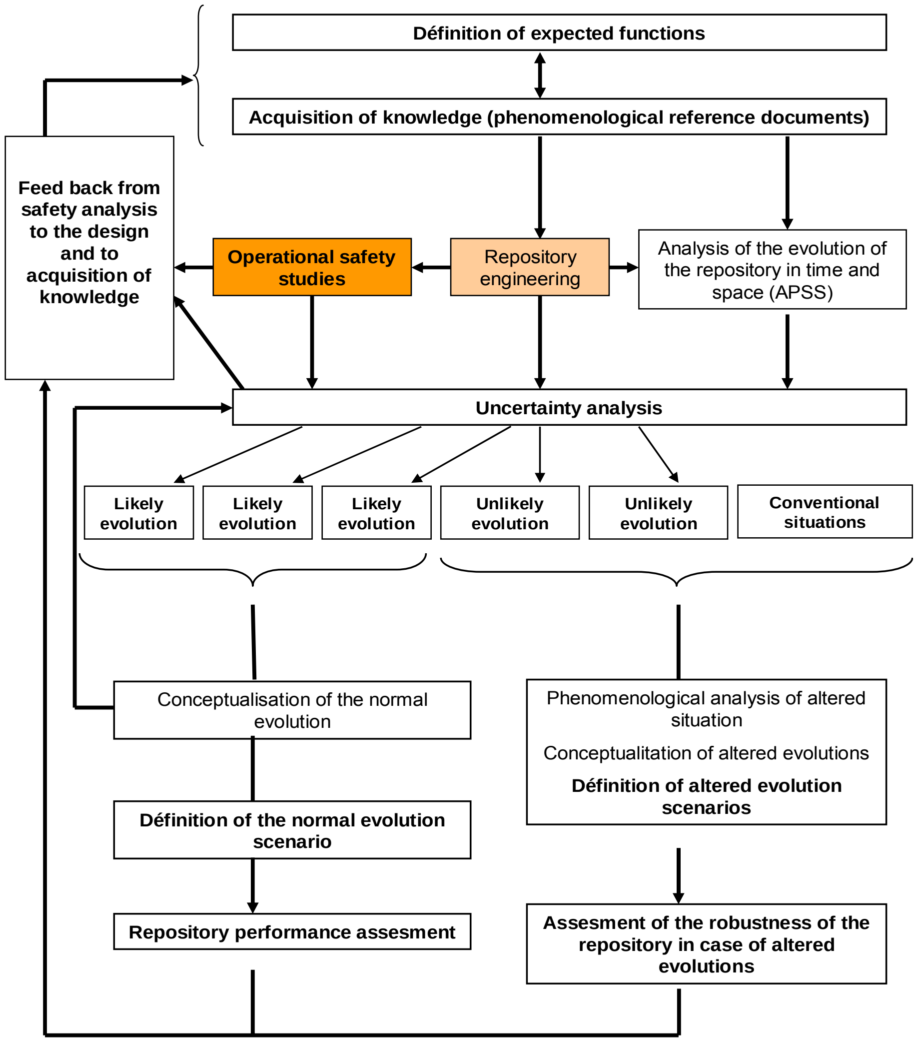
All these arrangements put in place to fulfil the safety functions make up a coherent process requiring a limited number of materials : clay, concrete, steel. Only the main design dimensioning components, directly dictated by safety, have been covered in this chapter. The reader will find a more detailed description, and explanations of the construction methods in the volume dedicated to this objective [22].

At this point it is necessary to verify that :

- The design measures selected make it possible to meet the safety objectives set by Andra. This will be the objective of chapters 4 (for operational safety) and 5 (for post-closure) ;
- In a more detailed manner, by consolidating all the components which make up the repository installation, interactions of all types (thermal, hydraulic, mechanic, chemical and radiological), cannot interfere with the operation of the safety functions. This will be discussed in chapter 6 ;
- Beyond the expected evolution, that the safety functions also make it possible to cope with situations of an incidental nature, whether it be during operation (chapter 4) or in the long term (chapter 7).

4 Operational safety

Note : The content of this chapter is identical to that of Chapter 11 of « architecture and management of a geological disposal system [22]

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Block diagram 4-1 Representation of the sequencing of the different stages of the analysis (see Block Diagram 1-1) Subject : operational safety study related to repository engineering

As for any other industrial facility, the various disposal activities (construction, operation and closure) can present a risk to personnel, members of the public and the environment.

This chapter presents a preliminary operational safety analysis on the basis of the current knowledge level of the disposal facilities and takes into account the experience of other comparable facilities (see block diagram 4-1). This analysis makes it possible to ensure that risks are brought under control through provisions in the design of the repository and operation in compliance with safety functions[17] defined by Andra.

Within the context of this analysis, a preliminary evaluation of the dosimetry with the installations in operation is presented.

A summary of the risk analysis is also set forth. It highlights certain particular risks for which additional studies have been carried out on account of their specific characteristics or their impact on the design of the repository and its equipment.

This chapter goes on to offer an analysis of the particular abovementioned risks, namely, the risk of explosion associated with the emission of gas by certain waste packages, the risk of fire breaking out in underground installations under construction and in operation, the risk of a cage falling down the waste package transfer shaft during transfer operations and the risk of a B waste package falling during its emplacement in a cell. Finally, a summary of the analysis is given by way of conclusion.

4.1 Evaluation of the dosimetry with the installations in operation

The operation of the facilities in which nuclear waste is received, conditioned, handled and disposed presents radiological risks to persons on account of the nature of the waste packages. There are several types of risk : the risk of external exposure (by irradiation) and the risk of internal exposure (by inhalation or ingestion).

This section presents a preliminary assessment of the dosimetry that takes into account the preventive measures envisaged to counter the radiological risks. These measures must ensure exposure of persons does not exceed the annual dose rate constraints set by Andra in the form of radiological protection targets of 5 mSv for personnel working in the nuclear zone and 0.25 mSv for members of the public outside the site [26].

This dosimetry evaluation informs the approach to the design of these installations from the point of view of radiological risk but is not, at this stage, part of an approach to optimise the doses received.

4.1.1 The nature of radiological risks and measures envisaged

The various radiological risks that affect the facilities are set forth in the following sections.

4.1.1.1 External exposure risk

Waste packages are sources of external exposure (associated with β, γ and neutron radiation) from the moment they are received in the surface installations until they are placed in the underground disposal facilities.

Transport casks containing the primary waste packages delivered to the disposal site have a radiological protection function and their structure is designed specifically for the radiological characteristics of the waste transported. Once the primary packages have been removed from transport casks, they are handled and placed in disposal packages inside cells which are not accessible to operators who work by remote control from behind radiological protection shields (walls, shielded windows).

[17] The safety functions defined (see section 3.6) are as follows :
- Protection of persons from radiation,
- Containment of radioactivity,
- Control of the criticality risk,
- Removal of residual heat of disposal packages,
- Removal of the radiolysisgases emitted by some packages.

During transfer operations and up until emplacement in the disposal cells, control of the risk of external exposure takes the form of the interposition of radiological protection shields[18] between sources of radioactivity and personnel to reduce radiation flux. Disposal package transfer casks, C and spent fuel cell access ports and B cell doors fulfill this role. The effectiveness of such protection would be inspected by means of irradiation measuring devices either in the form of stationary units or devices used by the monitoring teams.

Radiological protection or distance from the control station of the vehicles used to transfert or handle the packages in the cell would also contribute to the reduction of doses received by personnel.

During closure operations, definitive radiological protection shields are installed such as blocks of concrete to replace the doors of the B waste disposal cells and metal plugs of C and spent fuel cells (see Chapter 5 of the « architecture and management of a geological disposal system » [22]). These devices would facilitate any reverse operations (package retrieval) as decided.

4.1.1.2 Internal exposure risk due to the inhalation of radioactive gas emitted from disposal packages

Some B waste disposal packages (B2 and B5) emit small amounts of radioactive gas (tritium, carbon 14, etc.).

In surface installations, the limited number of waste packages present is such that the amount of gas emitted is negligible. In addition, most of the operations are carried out in cells that are not accessible to personnel.

During the course of disposal package transfers to the disposal cells in the underground installations, traces of radioactive gases are emitted from these packages in the transfer drifts and released via the ventilation circuit.

In the disposal cells where a large number of packages are disposed of ventilation with exhaust of the air by means of ducts to the ventilation shaft (see Section 6.4 of the « architecture and management of a geological disposal system » [22]) removes these gases preventing them from affecting personnel present in the underground installations.

4.1.1.3 Risk of internal exposure due to the inhalation of radon gas emitted from the rock in the underground repository installations

This risk, present from the beginning of construction activities, is associated with the level of radon naturally exhaled from the rock in which the underground installations would be situated. Given the argillite nature of the Callovo-Oxfordian formation, this risk is limited. It can be controlled by permanent ventilation of the underground drifts which would expel the radon and its descendants.

4.1.1.4 Internal exposure risk through ingestion of radioactive materials

In surface installations, this risk could be related to the dispersion of radioactive particles from transport casks, waste packages (primary packages, disposal packages) or transfer casks.

In surface installations, the management of this risk would depend on the organisation of the receiving and preparing facilities into containment systems[19] in order to prevent the dispersion of radionuclides towards areas in which personnel circulate or into the environment. These installations would also be equipped with filtering devices on their ventilation circuits, as is done in existing nuclear facilities of the same type[20]. Finally, it is important to mention that non contamination inspections[21] of transport casks, waste packages and transfer casks would be carried out systematically.

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[18] The nature of the material from which these screens are made depends on the type of radiation emitted by the radioactive source :
- In the case of γ radiation, heavy materials such as steel, concrete and lead glass are used.
- In the case of neutron radiation, specific materials (with boron or cadmium, etc.) or hydrogenated materials.
- α and β radiation do not need any particular type of screen as they are stopped by the package container.

[19] The principle of a containment system is to create a difference in air pressure between adjacent areas.

[20] These filtering devices are, however, justified when taking accident situations into account, in particular for the reception and conditioning of bare spent fuel whose surface would be contaminated by corrosive products deposited and activated during transfer from the fuel assembly in the reactor (see section. 4.2.2.2).

[21] The acceptance thresholds could be those established by transport regulations, that is to say, labile (not fixed) surface contamination restricted to 4 Bq/cm2 in β,γ emitters and 0.4 Bq/cm2 in α emitters [49].

4.1.1.5 Criticality risk

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

Criticality-safety analyses have shown that B waste and C waste packages do not contain enough fissile materials (critical mass) for this type of reaction. Spent fuel packages are the only type affected by this risk [45]

In surface nuclear installations, in the case of spent fuel[22], the absence of the inflow of water into the conditioning cells must be ensured to eliminate the risk of criticality as is practiced in similar existing storage facilities on waste production sites.

In underground installations, package transfer and emplacement in the repository is carried out dry. There is no criticality risk associated with this.

4.1.2 Dosimetric evaluation on site and around the periphery of the site

This preliminary evaluation, based on the radiological characteristics of the packages summarised hereafter takes into account the measures envisaged (radiological protection shields, non-contamination inspections, monitoring, etc.).

4.1.2.1 Data taken into account

- **External exposure**

The equivalent dose rate (EDR) values relating to primary packages are given for different types of package (see Table 4.1-1)

<table>
<tr><th colspan="2">Type of package</th><th>Nature of contents</th><th>Maximum EDR[23] in contact with the primary package (Sv.h⁻¹)</th></tr>
<tr><td rowspan="6">B</td><td>B1, B7.2</td><td>Compacted activation products</td><td>25</td></tr>
<tr><td>B2</td><td>Bituminised waste</td><td>2</td></tr>
<tr><td>B3, B7.1, B7.3, B8</td><td>Compacted or cemented technological waste</td><td>0.5</td></tr>
<tr><td>B4</td><td>Cemented cladding waste</td><td>0.5</td></tr>
<tr><td>B5</td><td>Compacted cladding waste with or without technological waste</td><td>15</td></tr>
<tr><td>B6</td><td>Cladding waste and technological waste in drums</td><td>2</td></tr>
<tr><td rowspan="2">C0</td><td>C0</td><td>Vitrified waste</td><td>150</td></tr>
<tr><td>Other C</td><td>Vitrified waste</td><td>250</td></tr>
<tr><td rowspan="3">SF</td><td>CU1 (UOx)</td><td>Spent fuel assembly</td><td>25</td></tr>
<tr><td>CU2 (MOx)</td><td>Spent fuel assembly</td><td>15</td></tr>
<tr><td>CU3</td><td>Spent fuel</td><td>150</td></tr>
</table>

Table 4.1-1 Equivalent dose rate in contact with primary packages [17]

- **Internal exposure**

Exposure through inhalation is associated, on the one hand, with radioactive gases and aerosols emitted by some B waste packages between their reception at surface installations and subsequent emplacement in the disposal cell, and on the other, with radon gas emitted naturally by the geological environment into the drifts of the underground installations.

Radioactive gases emitted by the packages

Some packages contain traces of radioactive gases. Measures to reduce the release of these gases were presented in Chapter 4 of the « architecture and management of a geological disposal system » [22].

[22] The presence of water, which attenuates the energy of the neutrons and slows them down, makes them more reactive to fissile materials and increases the reactivity of the system. In addition, the procedures used for conditioning the packages are dry procedures, no water is used.

[23] In the case of slightly or highly exothermic packages, the values given here correspond to a packages with an age of 5 years in the case of B1 packages, 10 years for B5 and SF3 packages, 20 years for C0, 60 to 70 years for CU1 (UOx) and 90 years for CU2 (MOx).

However, these measures do not apply to packages that also generate hydrogen due to radiolysis (B2 and B5.1 waste packages). For B5.1 type packages, the worst case, the release rates adopted (in becqeruels per hour and per primary package) would be in the region of 400 for tritium and carbon-14 and 7000 for krypton-85.

Radon gas emitted by the rock in the underground installations

The radon exhalation rate from the rock varies according to various factors such as : ventilation and nature of drift lining. At this stage, the value used corresponds to the average value observed in France.

4.1.2.2 Dosimetric evaluation

The dosimetric evaluation [46] is carried out for each type of work station identified on the disposal site and for the public around the periphery of the site.

- **Dosimetric evaluation for personnel operating in nuclear installations**

The results are presented by distinguishing between the different work stations in the surface and underground installations (see Table 4.1-2).

External exposure

The exposure of personnel to ionising radiation is a function of the nature and annual flow of the waste packages received as well as the operating process. The latter is defined by the mode of operation of the installations and equipment (local or remote operation or operation from a control room) as well as by the time required to carry out different operations related to the running of the facilities (see Chapter 9 of the « architecture and management of a geological disposal system » [22]).

Based on this data, the dosimetric evaluation is carried out as follows :

- The annual dose is evaluated for each production-related activity, taking into account the number of times this operation is repeated throughout the year. The average dose received by an operator is calculated based on the estimated total number of operators planned for each of these operations ;
- With respect to activities that are not directly related to production (control room, monitoring, etc.), the dosimetric evaluation is calculated directly for the station in question.

Internal exposure

Exposure through the ingestion of radioactive materials has been considered negligible on account of the preventive measures taken.

There are two component factors to exposure through the inhalation of radioactive gases :

- For B waste packages which emit radioactive gases, the scenario used is that of the transfer of a B5-1 waste disposal package in a transfer cask to a poorly ventilated drift. The exposure of personnel estimated on the basis of activity concentrations and dose coefficients related to inhalation for tritium and carbon 14 are negligible (of the order of 10^{-3} mSv per year) ;
- As far as doses related to radon emitted by the geological environment is concerned, the annual value used is 0.5 mSv for all operators working in the underground installations, both the nuclear zone and the construction zone.

Results

Activity	Type of package	Annual dose per operator (mSv/year/ operator)
Nuclear surface installations		
Reception of primary packages	B / C	2.1
	SF (UOx)	4
Control of primary packages	B / C	1
	SF (UOx)	1
Insertion of empty containers	B / C	1.3
	SF (UOx)	1
Preparation of disposal packages	B	1.5
	C	1.5
	SF (UOx)	1.4
Control of disposal packages	B	1
	C	0.9
	SF (UOx)	0.8
Placement of disposal packages in transfer casks and inspection of transfer casks at the surface	B	1.6
	C	0.9
	SF (UOx)	1
Surface control room	/	0.5
Surface monitoring	/	2.3
Surface maintenance	/	3.7
Underground nuclear installations		
Transfer and emplacement of the disposal packages in cells	B	1.2
	C	1.3
	SF (UOx)	4
Installation of cell plugs	C	0.6
	SF (UOx)	0.9
Transfer cask inspection at the bottom	B	0.9
	C	1.1
	SF (UOx)	0.8
Underground control room	/	0.5
Underground monitoring / maintenance	/	2.2

Table 4.1-2 Estimate of the annual dose received by the operators (mSv/year/person) operating in nuclear installations

It is clear from these results that the highest values, between 2 and 4 mSv/year, would be associated with the reception of primary packages, the transfer and emplacement in the cell of disposal packages and monitoring and maintenance of the installations. The values associated with other activities would be less than 2 mSv/year.

Doses received by personnel on site would therefore be lower than the limit set by Andra (5 mSv/year) and well below the statutory limit (20 mSv/year).

- **Dosimetric evaluation for members of the public around the periphery of the site**

External exposure is not taken into account for members of the public in this evaluation on account of the distance between them and the nuclear installations.

The internal exposure associated with the radioactive gases emitted by some B waste packages and released into the environment has been estimated on the basis of the assumption that the repository has completed operations, that is to say, with the entire inventory of B waste taken into account (see Chapter 3 of the « architecture and management of a geological disposal system » [22]). This preliminary estimate was made at the periphery of the repository site, taken as 500 m from the exhaust air chimney. The calculation, which takes into account the radiological activity released by the waste packages and a transfer factor in man that combines all three means of exposure shows that the annual dose for a member of the public of the order of 1 μSv, would be negligible.

Internal exposure associated with radon gas diffused by broken rock stored above ground or related to emissions from the underground installation ventilation system will depend on a number of factors : the radon exhalation rate from the argillites, ventilation flow in the underground installations and the height of the exhaust air chimney, length and type of open drifts (drifts under construction or in operation) and local atmospheric conditions. As in the preceding case, it can be considered negligible when the nature of the geological environment and experience in comparable facilities is taken into account.

Although the amount of radioactive gases given off from underground facilities may appear negligible, it would however be inspected with a measurement of air activity as is traditionally conducted in the scope of operational monitoring of a nuclear facility.

4.1.3 Conclusion

Given the measures employed to counter the radiological risk in the design of the installations and their mode of operation, the doses received by the personnel on site or by a member of the public at the periphery of the site should be below the annual limits fixed by Andra for radiolological protection, and well below statutory limits. It should also be remembered that these are preliminary estimates and do not take into account any subsequent optimisation approach.

4.2 Risk analysis

This section assesses the risks identified in the installations for all disposal activities (construction, operation and closure) and proposes associated reduction measures. It distinguishes between « internal » risks related to work carried out in the facilities and « external » risks, related to the environment (which are dealt with in a more generic fashion).

The risks examined in the context of the study are those that are liable to have an impact on people or the environment. However, the operational consequences (deterioration of work tools, drop in production) are not dealt with at this stage.

This assessment [46] can be used to highlight certain particular risks which have been the subject of additional studies (see sections 4.3 to 4.8) on account of their specificity or their impact on the design of the repository and its equipment.

4.2.1 Methodology

The analysis begins by identifying sources of danger associated with disposal activities. It has been carried out with the support of experts in the different technical fields concerned (nuclear installations, shaft transfert equipment, underground tunnels, etc.) that have used standard danger lists[24] and have brought their experience with comparable installations to bear.

This analysis, which has been structured around physical components (surface installations, access shafts, underground installations) and activities (construction, operation, closure), systematically takes account of compliance with operational safety functions. It offers the most exhaustive view possible at this stage in the studies of the risks likely to be encountered by personnel and the environment.

The risks are characterised by a source of danger and risk reduction measures. The latter comprise preventive measures to prevent or minimise the occurrence of risk, as well as protective measures that are taken to rule out or mitigate the effects (see Figure 4.2-1). Monitoring measures complete the risk reduction measures.

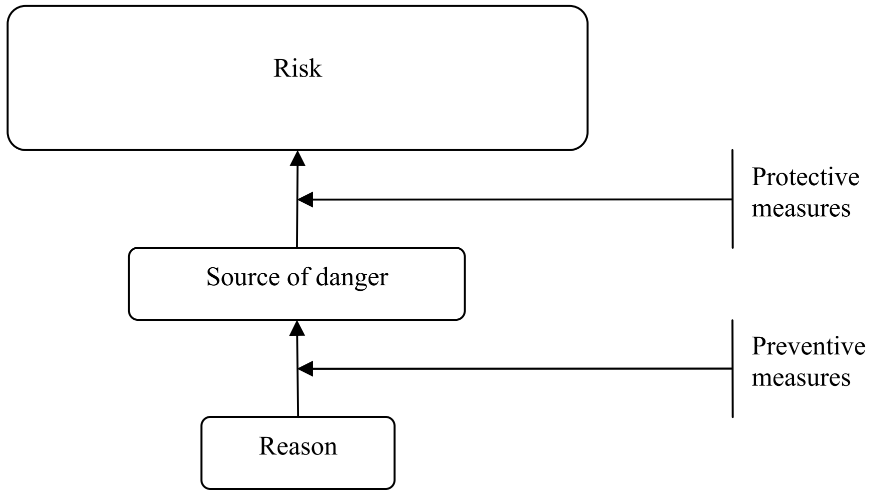

Figure 4.2-1 Risk characterisation

This characterisation enables a qualitative judgement to be made on the remaining degree of residual risk in spite of the risk reduction measures proposed. This expert appraisal is carried out as a function of the likelihood of the risk arising and the significance of its potential consequences for personnel, members of the public and the environment.

[24] Among the standard danger lists, the MADS (Methodology for the Analysis of the Malfunction of Systems) and the MOSAR (Method Organised for a Systematic Analysis of Risks) approaches are the most frequently used in the risk analysis of an industrial installation.

4.2.2 Internal risks associated with the disposal process

All internal risks identified are presented by risk type distinguishing successively between « conventional » risks, which are typically common to all industrial installations, and radiological risks, which are associated with the presence of nuclear waste packages.

4.2.2.1 Conventional risks

- **Listed risks**

The main conventional risks to be taken into account during the disposal process [46] are the risk to personnel of being crushed as a result of loads falling, falling blocks in drifts, objects falling down shafts, the risk of being crushed by equipment (crushed by a moving part during maintenance operations in shafts, etc.), the risk of being thrown from a vehicle, the risk of a collision between vehicles, the risk of a fall associated with work at heights (in the shafts in particular), the risk of electrocution and the risk of fire.

Risks inherent in the working environment (noise, dust, carbon dioxide and carbon monoxide gas emitted by motors….) which are different from the above risks on account of their fairly long term effect must also be monitored and controlled in underground installations in particular.

The other risks listed do not hold the same degree of importance. Among these, two types of risks related to a temporary loss of ventilation in the installations are noteworthy. These are the risk of explosion associated with the emission of small amounts of explosive gases from some B waste packages (B2, B5) which seems improbable given the ventilation throughputs planned for the installations which would ensure the dilution of these gases. However, a specific study has been carried out into this risk in support of this analysis. The risk associated with the presence of exothermic packages (C waste and spent fuel packages) should also be mentioned even if the temperatures of the metal envelope of the transfer casks in which they are transported do not exceed approximately thirty degrees and could not cause burns to personnel.

- **Risk reduction measures**

The risk reduction depends first of all on prevention, by selecting specially adapted, reliable, well maintained equipment fitted with all necessary safety systems, and on training personnel, raising awareness of the different types of risks encountered, compliance with procedures and on-site traffic regulations and the wearing of personal protective gear[25].

In the underground installations, the installation of physical protection systems (for work in shafts in particular), the use of equipment operated from control stations that are some distance from the work face, the equipping of safety networks on work sites (fire-fighting water network, communications network, etc.) as they progress also contribute to reduce the risks faced by personnel.

Of all the risks mentioned, the risk of fire is different from the others because it has collective consequences and requires specific provision for the evacuation of personnel.

In surface installations, the risks of fire are limited : the electric (or electronic) cabinets appear to be the main possible source of ignition. The measures to prevent, detect and limit the consequences of fire that would be used would comply with the Labour Code and the Basic Safety Rules (*Règles Fondamentales de Sûreté*) RFS I.4.a [47] and RFS II.2 [48]. Feedback on experience of nuclear installations with similar functions to those carried out in the repository would also be taken into account. The main provisions concern the choice of fire-retardant materials, the limitation of the calorific load of the installations (associated with the choice of handling by means of travelling crane or electrically driven vehicle), the sectorisation of areas that present a fire risk and the fire stability of structural components. The installation of fire detection systems, control of ventilation with a smoke extraction system and existence of evacuation routes (with clearances protected by overpressure, non-fumigation chambers, etc.) also play a role in the safety of personnel.

In underground installations, the principles adopted are those of a ventilation system with smoke extraction and connections between adjacent drifts to facilitate the evacuation of personnel. It is more

[25] This personal protective gear includes safety shoes, safety earmuffs, dust-masks and autonomous breathing apparatus (self rescuers) to be used in the presence of fumes.

difficult to make reference to other existing installations (road/railway tunnels, underground mines) as the solutions used are specifically related to the configuration of each installation.

- **Summary**

A summary of the analysis (see Tableau 4.2-1) is presented per type of activity (construction, operation, closure). Work associated with the possible retrieval of disposal packages has also been envisaged.

The following conclusions can be drawn from this table :

- The analysis shows no particular differences as regards the type of risks between the various construction, operation, closure and even package retrieval activities[26].
- The main risks listed that correspond to conventional risks for which preventive measures are known do not require specific studies to be undertaken at this stage in the project.
- Additional studies are, however, justified in the case of risk specifically attached to a disposal activity such as is the case for the risk of explosion associated with some B waste packages (see Section 4.3) or if it has a significant impact on the design of the facilities and equipment, as in the case of the risk of fire in underground installations (see Section 4.4).

Activities / Risks	Construction	Operations			Closure	Potential package retrieval
		Surface installations	Transfer surface / U/G installations	Underground installations		
Persons crushed associated with a falling object	X	X	X	X	X	X
Persons crushed by equipment or vehicle	X	X	X	X	X	X
Persons falling associated with work at heights	X		X	X	X	X
Electrocution	X	X	X	X	X	X
Fire (surface)	X	X			X	X
Fire (underground)	X		X	X	X	X
Pollution and nuisance in the work environment	X				X	
Explosion related to the release of explosive gases		X		X	X	X
Temperature rise related to heat released by packages						

X	Risk to be taken into account
(empty box)	Negligible risk
(grey box)	Not applicable
(thick-bordered box)	Risk is the subject of additional study

Tableau 4.2-1 Summary of the main conventional risks associated with the disposal process

26 The risks related to the potential retrieval of waste packages would not be distinguished from risks identified either as these techniques and the equipment used for this work would be very similar to those used during emplacement.

4.2.2.2 Radiological risks

Radiological risks (external exposure risks, internal exposure risk and potentially criticality risk) likely to be encountered during the course of the disposal process could be associated with radiological protection failures, interventions carried out close to a source of radioactivity or a fire or fall affecting waste packages.

Having taken account of their specific characteristics, these risks are analysed individually and emphasis is placed on measures to reduce the risks envisaged.

- **Failure of the radiological protection provided by the doors, vents and shielded windows of nuclear cells in surface installations, transfer casks, access ports of the C and spent fuel disposal cells and doors of the B disposal cells**

These events could be the result, for example, of clearance between moving parts that is not compliant with the initial dimensioning. Measures required to counter this risk would be a specific maintenance programme for moving parts associated with radiation detection monitoring of the nuclear cells, transfer casks and disposal cells.

- **Failure of radiological protection during the course of an intervention**

Equipment malfunction may lead to its immobilisation when being used to carry or handle a package (primary package or disposal package) and intervention by maintenance personnel may be required to repair the equipment in question. This situation would result in the external exposure of personnel if the latter had to operate near the source of radiation.

Preventive measures would be those conventionally used in nuclear facilities currently in operation, namely, appropriate maintenance of the equipment used and redundancy of certain component parts (motorisation, etc.).

Where intervention carried out on equipment in cells is concerned, the existence of emergency systems enabling the waste package to be put down and the equipment returned unloaded to its maintenance area would remove all risk of the exposure of personnel. In the latter case, it would, however, be necessary to ensure there is no malfunction of the closure system of the irradiating zone of the cell[27] by first monitoring the level of radiation in the maintenance area prior to any operation.

- **Fire in a nuclear cell of the surface installations**

A fire in the disposal package manufacturing cell would have a fairly insignificant heat rating given the low calorific load of the equipment installed.

Taking feedback from the experience of nuclear facilities into account in the design of installations, limiting the calorific loads present and possibly installing thermal protections around some equipment should be sufficient to limit the power of a fire and prevent it from spreading and affecting the packages resulting in radiological consequences.

- **Waste transportation or handling vehicle fire**

The fire of a transport cask vehicle (containing primary packages from waste producers), a transfer cask vehicle (containing waste disposal packages) or B waste emplacement equipment in the disposal cell could have radiological consequences in addition to the direct consequences of the fire (see Section 4.4).

On the surface, the transport cask vehicle leaves the waste reception area as soon as the transport cask has been set down which limits the presence of the main cause of the risk. Furthermore, the transport cask is designed to resist fire up to 800°C for 30 minutes in accordance with transport regulations [49].

[27] The system usually employed consists of locking in the opening of the maintenance area with the closure of the irradiating cell and, if possible, locating the maintenance area at some distance from a potential field of radiation from the sources of radioactivity present in the cell.

In underground facilities, there is no industrial reference directly applicable to the transfer vehicle as well as to the B waste emplacement equipment. It has therefore been necessary to make simulations on the basis of assumptions regarding the heat ratings involved and the nature of the exchanges in the course of the fire. These studies are presented in Sections 4.5 (Transfer cask transporter) and 4.6 (B Waste emplacement handling equipment).

- **Primary package falling in surface nuclear installations**

Primary packages could be dropped and damaged during handling. The foreseeable consequences of this could be a breach of one or more of the primary packages and the dispersion of radioactive material in the installations and subsequently into the environment via the ventilation. A fall could also result in the surface contamination of a package missed during other inspections becoming airborne (see Section 4.1.1.4).

Preventive measures are specific to each type of equipment. Equipment systems must be sized for loads greater than those envisaged, must ensure a degree of redundancy in some components and intrinsically safe devices must be designed for the possibility of malfunction (for example, keeping the grab closed in the event of a power failure). Training of personnel and maintenance are also very important in countering this risk. In addition, the mechanical procedure used must give priority to handling packages at a height lower than the height from which they are known to withstand being dropped.

As the protective measure to be used to take care of a potential release of radioactive materials, a filtered ventilation system could be installed on the ventilation circuit of the primary package reception and disposal package manufacturing cells.

This measure might be justified for some primary B waste packages, such as the bituminised sludge package, which is not totally protected from the possibility of the crimping of its lid failing and opening in the event of impact.

This measure would also apply to installations receiveing bare spent fuel assemblies which, if dropped during the handling operations that are necessary until they are placed in disposal packages, could result in the rupture of the cladding of the fuel rods. The main provisions for the management of this type of risk could be similar to those used for the dry unloading of this type of fuel in the reprocessing plant (Cogema T0 facility in La Hague).

- **B Waste disposal package drop during emplacement[28]**

Carring the waste disposal package in the low position over the whole length of the disposal cell limits the risk of it falling during the package lifting and emplacement. During this operation, an error in the positionning of the expected position of the package to be emplaced, or the malfunction of the lifting system, could result in a fall from a height of 4 to 6 m in the case of packages placed on the highest levels of the stacks.

Several provisions are envisaged to limit the occurrence of this risk and its consequences :

- The emplacement procedure with a row of waste disposal packages placed on the ground followed by a second layer and so forth limits the risk to a case in which a package might tip on its side ;
- Monitoring the package emplacement cycle (validated step by step by the operator, visual inspection using cameras) and checking that the emplacement position matches the cell map beforehand should prevent any error in positioning when the package is being set down ;
- The selection of design options of the equipment to ensure good stability (see Section 9.3 of the « architecture and management of a geological disposal system » [22]) with the load in all positions and, on the lifting system, provisions such as brakes and safety sensors, redundancy of various components of the lifting system, double electric power supply, device for lowering the load in the event an anomaly is detected, etc. should prevent the risk of a package being dropped. These various systems should be regularly inspected ;

[28] The procedures envisaged for emplacement in the cell of C and CU packages (see Section 9.3 of the « architecture and management of a geological disposal system » [22]) require a handling height of less than 2 meters. Because of this, these drop cases are not dealt with.

- In the event a package were to fall, the disposal packages provide protection from the primary packages contained.

To quantify the deformations of the B waste disposal packages in the event of a fall and to examine what the consequences would be, one initial approach has been to carry out a simulation study to verify the resistance of the package when dropped (see Section 4.8). These results are expected to be validated by full-scale B2 and B5 waste disposal package drop tests during 2005.

- **Uncontrolled cage displacement or drop when loaded with the B, C or spent fuel disposal package transfer cask in the shaft[29]**

Experience acquired in mines with this type of transport and the combined preventive measures and inspections are such that the likelihood of the cage falling is extremely low[30] [50].

The preventive measures that serve to counter this risk involve both the design of equipment (independent braking systems on the driving pulley, bundle of independent cage suspension cables [31], etc.) and maintenance, control and operating procedures. A cage anti-drop system could foreseeably be added to these measures to provide an additional safety system which would be independent of the cage's command and control system. The principle of this system would be to use cables suspended in the shaft as braking cables which would stop the cage in the event of excess velocities. In order to limit the stress to which these cables would be subjected in the event a load were to be exerted upon them, they would be connected to shock absorbers that would dissipate a large part of the kinetic energy associated with the movement of the cage.

Two kinds of measures would be employed to limit the mechanical consequences of the uncontrolled displacement of a cage or of a cage falling down the shaft :

- An end of travel braking system, similar to the kind installed by law in mines, some metres below the bottom station would stop a cage passing its stopping point if its velocity does not exceed approximately 10 m/s. This would deal with a case of cage displacement associated with the malfunction of the braking system when it reaches the stopping point.
- In the case of velocities in excess of 10 m/s, a shock absorber composed of « honeycomb » type material and installed at the bottom of the shaft would have the advantage of being a passive system capable of absorbing substantial amounts of energy. Laboratory tests have provided the characterization of the behaviour of this material for high velocity impacts (see Figure 4.2-2).

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29 Other drop cases have been envisaged involving the direct drop of the transfer cask transporter down the shaft (cage not in position) or the transfer cask falling through the floor of the cage (subsequent to impact or as a result of structural weakness). The identification of these two cases, which would appear even more improbable than the preceding one [50], serves, above all, to justify the reinforcement of preventive measures associated with access to the cage and the regular inspection of the condition of the shaft and the cage.

30 In Germany, a study carried out for the Gorleben radioactive waste repository project [50] estimates that for a comparable facility, the probability of a cage falling down the shaft is 5.10-7/year (for 5000 hours of operation per year).

31 The study envisages a system with 10 independent suspension cables which allows loads of around one hundred tonnes to be transported. Shutdown is triggered in the event a single cable breaks which makes the successive breakage of all multi-cables unlikely.

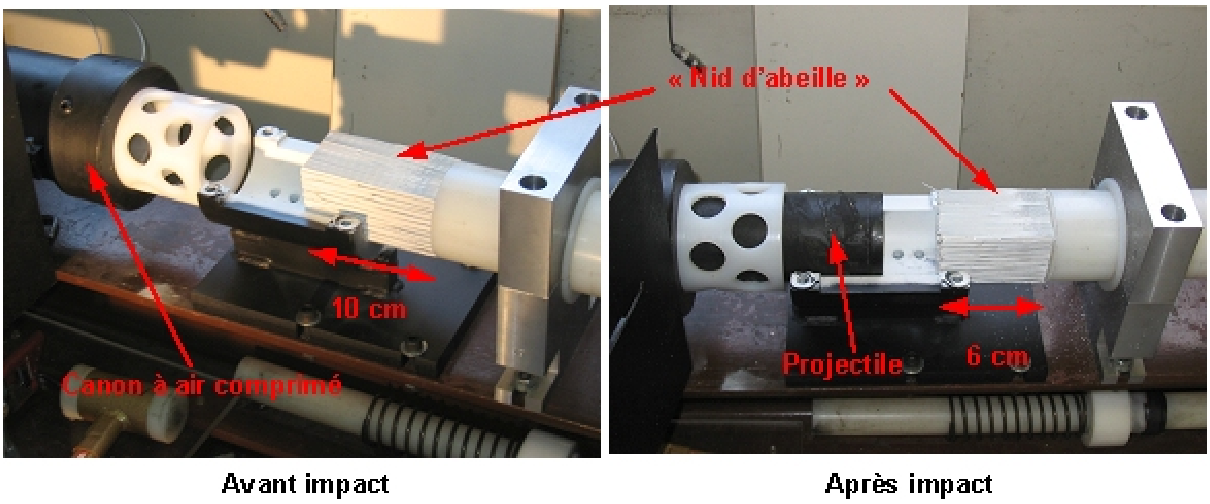

Figure 4.2-2   Honeycomb material crash test (Impact Velocity 200 km/h)

These different measures are illustrated in Figure 4.2-3.

Quantifying the effect of the shock absorber on the packages transported in the cage required simulation studies presented in Section 4.7 to be undertaken. They show that at the moment of impact, the shock absorber and the cage would absorb the largest part of the energy. The metal structure of the transfer cask would be misshapen but not split open. Primary packages should remain intact even if, in the case of B waste packages, the concrete container of the disposal package were damaged.

However, given the uncertainties associated with the sequence of events leading to a cage falling down a shaft that would not be covered by the aforementioned simulation studies, radioactive material release scenarios have also been envisaged in Section 4.7 in order to obtain an order of magnitude of the radiological consequences resulting from the drop and to ensure that the technical means exist, if required, to limit their impact to an acceptable level for the environment.

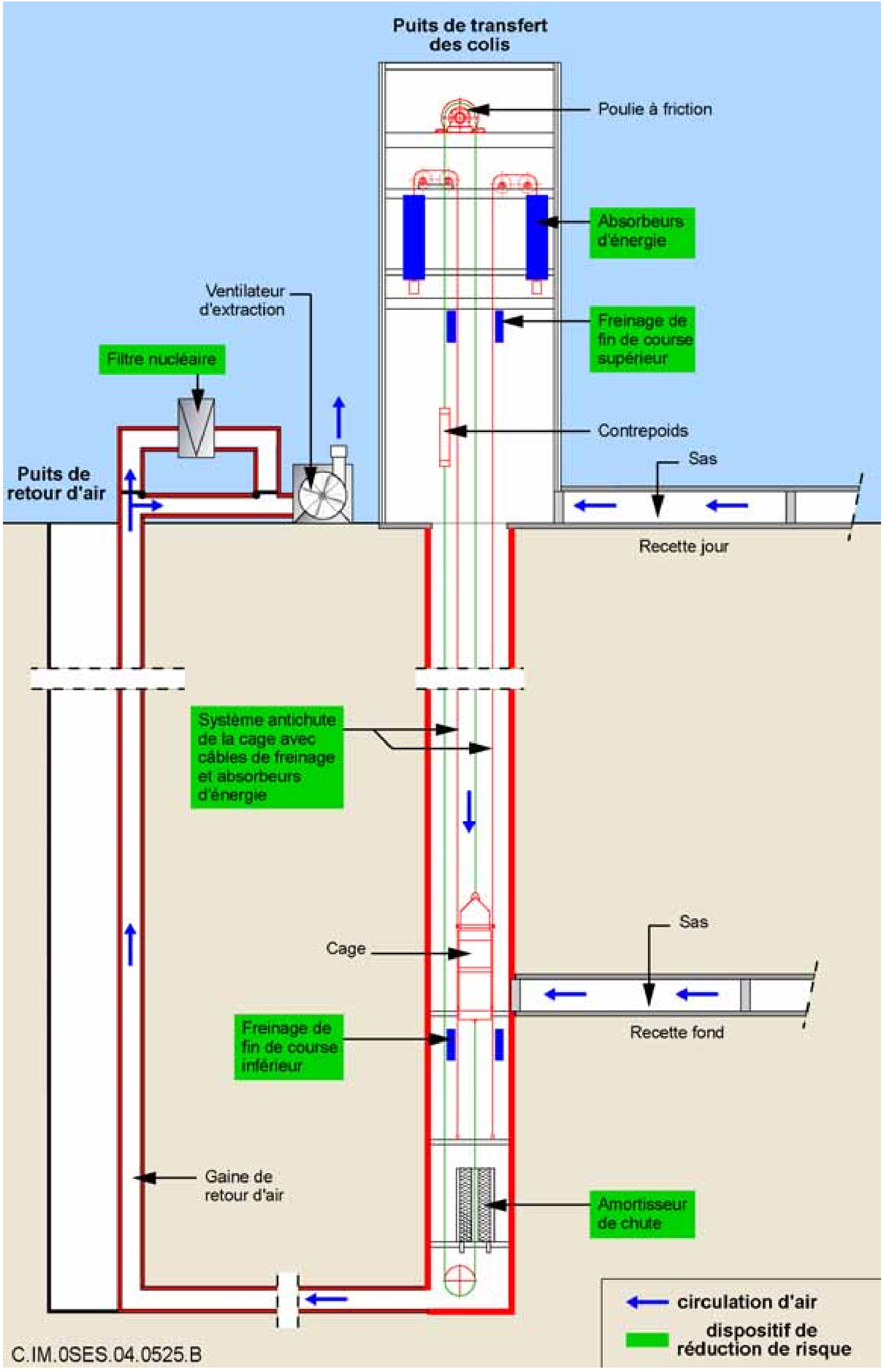

Figure 4.2-3 Schematic diagram of the devices envisaged to reduce risks during waste disposal package transfer in shaft

- **Risk of criticality associated with a cage loaded with a used fuel package transfer cask falling down a shaft**

A drop scenario involving a cage loaded with a transfer cask containing a spent fuel disposal package which would entail serious damage to said package (modification of its internal geometry, fracture and contact between spent fuel bundle elements) and an inflow of water could lead to a risk of criticality.

Given the measures envisaged to counter the risk of a cage falling down the shaft, such a level of damage to the package does not seem likely. However, an additional precaution would be to ensure the absence of water (or other hydrogenated fluid) in the shaft to rule out the possibility of this risk entirely. This would mean prohibiting the installation of pipes in the waste transfer shaft and also providing a water evacuation system at the bottom of the shaft.

The other dangerous situations envisaged [45] do not appear to bring about a risk of criticality.

- **Summary**

A summary of the analysis (see Table 4.2-2) is presented per type of activity (construction, operation, closure). The potential retrieval of disposal packages has also been envisaged.

Activities / Risks	Operations			Closure	Potentiel package retrieval
	Surface installations	Transfer surface / U/G installations	Underground installations		
Failure of radiological protection associated with equipment	x		x	x	x
Failure of radiological protection during the course of an operation	x		x	x	x
Radiological protection failure / loss of package containment associated with a fire in a cell of the surface installations	x				
Radiological protection failure / breach of package containment associated with a primary package falling in the surface installations	x (B and bare spent fuel)				
Radiological protection failure / loss of package containment associated with transporter or handling vehicle fire.	x		x		x
Radiological protection failure / breach of package containment associated with a B waste package falling during emplacement or removal.			x		x
Radiological protection failure / loss of package containment / criticality[32] associated with uncontrolled displacement or a cage falling down the shaft when transporting a transfer cask.		x			

x Risk to be taken into account — Negligible risk — Not applicable — Risk is the subject of an additional study

Table 4.2-2 Summary of radiological risks associated with the disposal process

[32] The risk of criticality only concerns spent fuel.

The following conclusions can be drawn from this table :

- The analysis shows no notable differences as regards the nature of the risks between package emplacement and removal. Operations associated with the possible removal of packages from the repository would be the reverse of operations carried out during emplacement ;
- Closure activities do not present risks associated with the handling and transfer of the packages. On the other hand, special attention will be necessary to counter the possible failure of radiological protection during operations carried out at the entrance of disposal cells ;
- The analysis lists conventional risks such as radiological protection failure for which the preventive measures and controls are known and do not require special studies to be undertaken at this stage in the project ;
- It may also be considered that the risks of a package falling and fire in surface installations do not justify additional studies at this point given the feedback on experience available from similar existing surface nuclear facilities ;
- The radiological risks associated with the transfer and emplacement of packages in the disposal cell, on the other hand, have called for further study. The risks induced by transport or handling vehicle fires must take into account their characteristics and the fact that the fire would occur in a semi-confined space. Similarly, package drop scenarios refer to drop heights that are different from the usual handling heights in surface installations, in particular during transfer cask transfer in shafts. These various risks will be dealt with respectively in Sections 4.5 and 4.6 for fire risks and Sections 4.7 and 4.8 for risks associated with drops.

4.2.3 Risks associated with the repository external environment

4.2.3.1 Earthquake

The sector covered by the study is a low seismic activity zone. The surface installations would have to be dimensioned in accordance with current aseismic regulations. In addition, measures would have to be taken to prevent any loss of safety function liable to have a radiological impact on operating personnel and the public.

The measures consist of dimensioning buildings (stability of all buildings) for a safety margin computed earthquake [51] in order to protect the sources inside these buildings and equipment which could, directly or indirectly, be a cause of dissemination of radioactive material.

Underground engineered structures withstand seismic loads better than surface installations due to attenuation with depth. It has been shown that the earthquake [52] would not have a significant impact deep underground.

4.2.3.2 Meteorological risks

The main risks only concern surface installations and have no impact on underground activities. They mainly concern rain and snow fall, extreme temperatures, lightning and wind.

4.2.3.3 Aircraft impingement

Independently of the direct physical consequences, the impingement of an aircraft on surface installations could result in the loss of safety functions leading to the exposure of personnel to radiation and its release into the environment.

This risk could be dealt with in accordance with the principles set out in Basic Safety Rules (*Règles Fondamentales de Sûreté*) RFS I.1.a [53] which recommends an evaluation of the probability of an impingement on « targets » within the facilities for which the loss of safety functions could have serious consequences. This evaluation is specific to the location of the site and takes different types of air activity into account : general aviation, commercial aviation and military aviation. The objective is to ensure that the probability of aircraft impingement leading to an unacceptable release of radioactivity is less than 10^{-7} per year. Above this value, the risk of impingement must be incorporated into the dimensioning of the installations in question.

In such a case, the measures employed to protect the installations, in particular those containing sources of radiation, would consist of dimensioning the concrete engineered structures to ensure that they are capable of withstanding aircraft impingement and marking out the highest obstacles (shaft superstructures) with beacons.

4.2.3.4 Risks associated with the loss of power and utilities

Even if this event is improbable (it could occur, for example, as a result of extreme weather conditions) and may not put personnel present in immediate danger, it could cause difficulties because of the large number of systems that would be stopped : ventilation, lighting, pumping and transfer cage. Preventive measures consist of redundant sources of power and emergency supplies (generators, batteries, etc.) for essential systems.

4.2.4 Summary

The operational safety analysis is based on the systematic analysis of risks supported by input from experts in the various technical domains concerned.

The construction of the various facilities is no different from the construction of other surface industrial installations or underground engineered structures (mines, tunnels, etc.). Because of this, the risks associated with this activity are the conventional risks (crushing, falls, etc.) listed in all construction work for this type of installation. No further studies are required at this stage but would, however, be taken into account during the detail design of structures and equipment.

Nuclear activity in the surface installations of the repository, which includes the reception, preparation and storage of waste packages, is comparable to activities carried out in the French nuclear facilities where the packages originate. Because of this, the analysis did not require specific studies of the repository installations themselves.

Nuclear activity in the underground installations, including the transfer of transfer casks (containing waste disposal packages) in the shafts and drifts and the emplacement of waste disposal packages in their cells, is carried out at the same time as drift and cell construction work. This is a specific issue, even though underground repositories exist all over the world[33] [54]and it is proposed that the design should ensure that these activities remain independent of each other by separating the respective traffic circuits and ventilation systems.

The closure activity does not entail any additional elements over and above those included in other activities. There is no particular difference between the closure of surface installations and that of a conventional dismantling site. The closure of underground installations, which takes the form of backfilling and sealing drifts and shafts, would be comparable to construction work in terms of site organisation and the type of equipment used.

The analysis has highlighted the risks that require particular attention on account of their specific characteristics or their impact on the design of the repository and its equipment.

These risks are the risk of explosion associated with the emission of gas from some waste packages (see Section 4.3), the risk of fire in underground installations (see Section 4.4) focusing on scenarios that would involve disposal packages (see Section 4.5 and 4.6) and the risks involved with the transfer of packages of radioactive material with the shaft drop scenario (see Section 4.7) and B waste disposal packages falling in cells (see Section 4.8).

This analysis is based on the current understanding of the main risks identified on the basis of current knowledge of the installations. It may evolve with the expansion of installation definition studies.

[33] These disposal facilities include the Waste Isolation Pilot Plant (WIPP) in New Mexico, USA, where transuranium waste packages (comparable to some B waste packages) are disposed of in underground installations access to which is via 650 m-deep shafts [54] and the SFR in Sweden where low- and intermediate-level waste packages are disposed of at a depth of between 60 and 100 m.

4.3 Study of the risk associated with the emission of explosive gases from some B waste packages

Most of the B waste packages (B2 and B5 in particular) emit gases between the time of their arrival at the surface installations and their emplacement in the underground disposal cells. These gases are caused by radiolysis which is associated with the effect of ionizing radiation (β, γ) emitted by radioactive materials on hydrogenated products present in the waste packages (organic materials, water in the conditioning matrix).

These radiolysis gases are mainly hydrogen (more than 90 % of the gaseous releases) and, to a lesser extent, methane[34]. The emission of these gases can cause an explosion if their concentration exceeds their lowest explosive limit[35].

The purpose of this section is to ensure that the emission of these gases by B waste packages is not liable to entail a risk of explosion [46].

4.3.1 Waste package characteristics

Waste packages affected by the release of explosive gases are mainly B2 and B5.1 type packages that contain organic matter.

Primary package	Nature and contents of the primary package	Emission rates of explosive gases (H_2, CH_4) (l/drum/year)
B2.1 **B2.2**	Metal drum with bituminised sludge	- 10 liters/drum/year (average value) - 57 litres/drum/year (maximum value corresponding to a minority of drums)
B5.1	Container with hulls and end caps and technological waste (including organic waste)	- 10 litres/drum/year (average value) - 500 litres/drum/year (maximum theoretical value for packages with a maximum organic compound content and a maximum activity value)

Table 4.3-1 Emission rates of explosive gases released by some B waste packages

Average values of gas emitted from the packages have been used for simulations in the surface installations and in disposal cells as they correspond to several hundred or several thousand packages ; on the other hand, the maximum theoretical value has been used for simulations relating to the phase in which a transfer cask loaded with a disposal package is transferred in the drift.

The concentration of gases in a given installation has been estimated as a function of its ventilation characteristics. The per hour air renewal rate is taken as being equal to 2 in installations in surface nuclear facilities. For the ventilation of B waste disposal cells, the data used are an airflow of 3 m^3/s which corresponds, for a 250m-long cell full of packages, to an air renewal rate per hour of 5. Hydrogen emitted by the waste disposal packages is diluted homogeneously in the free space above the disposal packages[36] given that ventilation remains in operation until the time when a cell is sealed.

Estimates were based on the pessimistic hypothesis of a constant outgassing rate over the duration of the operation and closure activity of the repository.

4.3.2 Analysis during the operational phase

This analysis is described in accordance with the logic of the cycle followed by the packages : storage of primary packages, storage of disposal packages, transfer of waste disposal packages to underground facilities, emplacement of waste disposal packages in disposal cells.

[34] Radiolysis is also the cause of the release of very small amounts of carbon dioxide and carbon monoxide. These gases are diluted by the ventilation of the installations as are those produced by vehicles with thermal engines.

[35] The lowest explosive limit is the minimum concentration of gas above which there is a risk of explosion in the presence of a source of ignition. The lowest explosive limit is 4 % for hydrogen and 5.3 % for methane.

[36] Hydrogen, being lighter than air, will tend to migrate above the packages towards the top of the cell.

4.3.2.1 Surface storage (primary and disposal packages)

A simulation, carried out based on the assumptions previously described, indicates that the hydrogen content in the atmosphere in the storage area is negligible under normal ventilation conditions (with the content varying from 10^{-6} to 10^{-7} in the various surface storage areas), and that the time taken to reach the explosive limit of 4.10^{-2} (4 %) in the event of a ventilation failure is several decades. There is therefore no risk of an explosion in these areas.

4.3.2.2 Transfer of waste disposal packages

Hydrogen is also produced inside the transfer cask during the transfer of the packages from surface installations to the disposal cell. The existence of the transfer cask door construction clearances and, where applicable, the presence of a vent, will allow the hydrogen produced by the packages to be diluted in the atmosphere of the spaces through which the transfer cask passes and there will be no risk of an explosion, given the low hydrogen emission rate compared to the ventilation flow rates in the various installations.

4.3.2.3 Disposal cells

The simulation was carried out with no ventilation in a 250 m long disposal cell filled with type B5.1 packages. Hydrogen would thus concentrate in the top 15 cm of the disposal cell, above the waste disposal packages.

Under these conditions, the time taken to reach the 4 % explosive limit is around 30 days. There is therefore no risk of explosion, even assuming a temporary ventilation failure.

4.3.3 Analysis during the closing phase

As far as the risk of an explosion is concerned, this phase can be sub-divided as follows.

4.3.3.1 Disposal cell sealing phase

The sealing process would begin by fitting a radiological protection shield consisting of concrete blocks fitted with pipes for maintaining the ventilation in the cell (see Section 5.1.6 of the « architecture and management of a geological disposal system » [22]). This stage is followed by the construction of a concrete retaining plug which fills the top of the cell and thus isolates the waste disposal packages from the access drift. Under these conditions, the release of hydrogen into the access drift would be very slight and could not be the source of an explosion, all the more so since this drift would be ventilated throughout the fitting of the seal swelling clay core.

4.3.3.2 Post disposal cell-sealing phase

The backfilling work may be carried out in two stages : backfilling of the type B waste repository zone infrastructure then, at a later date, backfilling of the remainder of the infrastructure before complete closing of the repository.

This backfilling work is isolated from the disposal cells by seals and cannot cause an explosion.

However, if it was decided to return to a cell after sealing, such an operation would only be possible after renewing the atmosphere in the disposal cell beforehand in order to vent away any hydrogen present (see chapter 10 of the « architecture and management of a geological disposal system » [22]). This would require special provisions to be made, of the sort practiced in fire-damp producing coal mines, to avoid any risk of ignition when the ventilation system is put back in place.

4.3.4 Conclusion

The risks associated with the emission of explosive gases (essentially H_2) by certain type B waste packages are controlled during the operational phase by ventilating the various installations, which dilutes their content. An interruption to the ventilation poses no real danger as a long period of time is available to carry out any repairs

The closure steps of the disposal process pose no explosive hazard, except in the event of a return to a cell after it has been sealed. In such cases, it will be necessary to re-establish the ventilation in order to extract the gases accumulated in the disposal cell, taking the necessary safety precautions during its installation.

4.4 Study into the fire hazard in underground installations

Fire remains one of the major preoccupations in an underground environment, since it develops in a semi-confined space, and the associated smoke and toxic gases may spread through the drifts into the installations, impede personnel evacuation and endanger a large number of persons.

Feedback from fires in underground structures indicates that the calorific potential of the drifts themselves being low, only the machinery and equipment can be the source of a major fire. According to a study of Swedish mines [55], the three main causes of machinery fires are electrical short-circuits (around 50 %), oil leaks on hot surfaces (around 25 %) and engine over-heating (approximately 10 % of cases). However, the risk of a collision causing a fire is low, given the slow velocities and low frequency of vehicles passing each other.

The essential fire prevention measures required in underground installations are as follows :

- preferred use of non-flammable materials that do not propagate fires and do not emit toxic smoke,
- control of inflammable products present (justification of product choice and use, maximum authorised quantities, transport conditions and utilisation procedures etc.),
- restrictions on the quantity of fuel for machinery with thermal engines, protection of sensitive components and choice, where possible of electrically-powered machinery with low heat load if the type of activity allows,
- implementation of machinery and equipment inspection, maintenance and operating procedures (driving licences for machinery, fire permits etc.) and personnel training in effective reactions in the event of an anomaly.

Fire detection equipment (with smoke, flame or temperature detectors) are also useful to allow rapid intervention in order to get the fire under control before it takes hold. These systems are preferably installed as construction work advances.

Fire protection measures consist, firstly, of providing personnel with first aid fire-fighting equipment, with extinguishers and a pressurised firemain fitted close to the work site. Fixed automatic extinguishing systems, fitted in hazardous areas (fuel or oil storage) and systems on board vehicles could also be effective means to be provided.

Over and above these fire precautions, underground installations should be fitted with a smoke extraction system enabling personnel to be evacuated under acceptable temperature, visibility and toxicity conditions. The installation of warning and alarm systems[37] and a centralised command post able to monitor the situation and direct operations would also contribute to the organisation of this evacuation under the safest possible conditions.

This section assumes that a fire has developed despite the preventative measure described above. Its purpose is to study the various representative fire scenarios, assess their consequences (temperature rise, smoke emission, toxic gas emission) and ensure that the personnel evacuation conditions are satisfactory. It does not deal with fires liable to have radiological consequences, which require further discussion and which are dealt with in sections 4.5 and 4.6.

4.4.1 Fire simulations

Simulation studies [46] have been carried in order to understand the potential consequences of the various fire scenarios envisaged, according to their characteristics.

4.4.1.1 Description of the various fire scenarios

The plant and machinery used to build the structures (diggers, rock bolters, transporters etc.) or conduct operational activities (transfer cask transporters etc) and a few items of special machinery

37 An interesting approach might be to provide personnel with an individual means of communication, linked to an internal network or to a PC. This would be used for work organisation during normal operations and, in the event of a fire, would be the best way to give the alert in real time.

(conveyer belts for carrying excavated material etc.) may cause potentially high-temperature fires[38]. This type of fire is characterised by its high thermal power, the quantity of toxic gases and the amount of smoke given off.

Determining its thermal power requires a realistic scenario to be defined. In the case of an item of plant machinery, the fire-producing event used in the scenario is usually a liquid fuel leak associated with a hot point (spark etc.). The liquid leakage rate considered determines the initial strength of the fire. The fire's duration depends on the machinery's calorific potential : the quantities of fuel or oil form a major part of this potential, immediately available in the event of a fire ; tyres contribute to increasing the duration of the fire and the amount of smoke produced. It is also assumed that the oxygen in the air is in sufficient quantity and that the seat of the fire is not extinguished by smothering.

The summary of the results for all the various items of machinery studied show that there are two major categories of machinery fire. Diesel or diesel-electrical machinery (dump trucks, loaders, excavators, bolting and drilling machines etc.) is that which produces fires which generate the most heat, approaching 25 MW, whilst electric transporters have a maximum thermal power of around 15 MW.

In order to overcome the problems associated with modelling of curves presenting transient phenomena, the fire simulations have used standard curves defined by the CETu (French tunnel studying centre), which cover the above in terms of power and duration. These curves, derived from feedback from tunnel fires, correspond to road vehicle fires with respective total powers of[39] 30 MW and 15 MW. Also associated with each type of fire are carbon monoxide emissions and smoke production.

Figure 4.4-1 which represents the evolution of the thermal power of a 30 MW fire over time, is an example of the standard curves which all have the same profile, with a start ramp, a levelling off and a descending ramp.

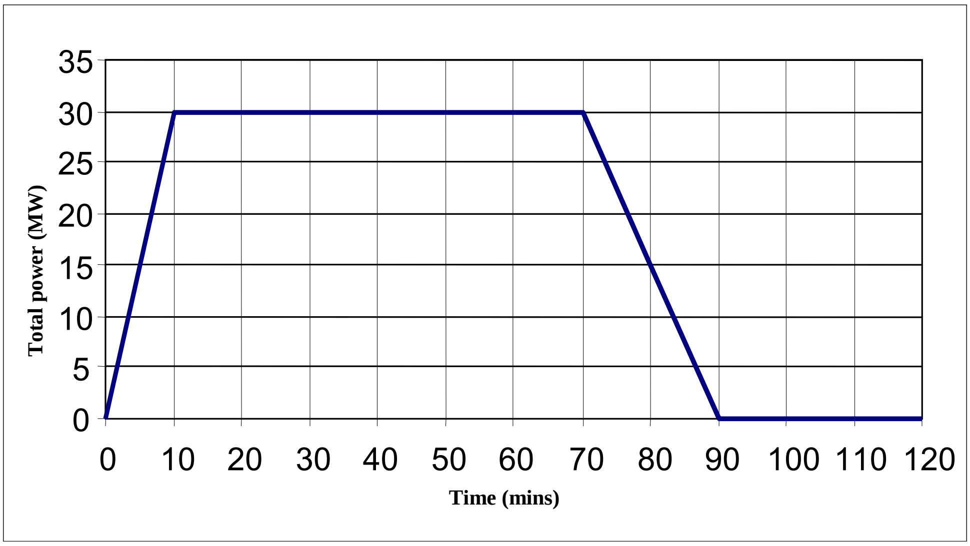

Figure 4.4-1 Standard total thermal power for a 30 MW fire

4.4.1.2 How fires develop

In an underground environment, fires emit smoke as they develop which spreads throughout the drift (fire with smoke de-layering) or forms localised pockets at the top of the drift (fire with smoke layering) depending on the ventilation conditions encountered.

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38 Another type of high-risk fire would be one in an inflammable product store, but this type of fire is not considered at this stage of the study, partly because its location and size are unknown and partly because this type of installation can more easily be the subject of special fixed fire-fighting facilities.

39 Generally speaking, for this type of fire, two thirds of the total power is dissipated by advection, the other third being dissipated by radiation.

- **Fires with smoke de-layering**

Fires with smoke de-layering give off smoke which gradually spreads through the drifts, turning the atmosphere opaque, and often hindering the evacuation of personnel. This type of fire, produced when the air velocity in the drift is greater than 1 m/s, is characterised by a uniform mixture of air and smoke throughout the drift's cross section at a given distance from the fire.

In the case of a repository, fires in type C waste and spent fuel disposal cell access drifts, ventilated by an inflow of fresh air throughout the drift's cross section, and exhausting the smoke via a smoke-clearance vent at the end of the operational unit, would be of this type.

For this type of fire (see Figure 4.4-2), personnel preferably escape by heading towards the connecting drift supplied with fresh air. If this is not the case, personnel downstream of the fire can head for the first interconnecting drifts in order to reach the adjacent access drift supplied with fresh air (with a maximum distance of 200 m to be covered thanks to the layout of the interconnecting drifts) then into the connecting drift.

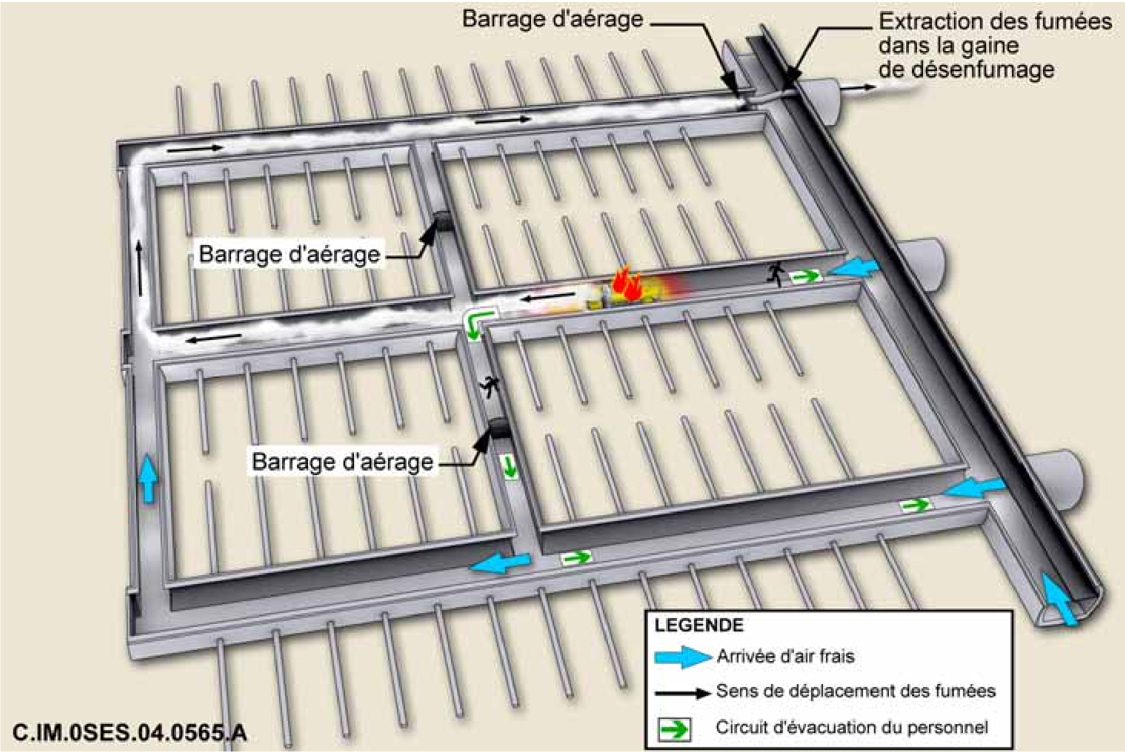

Figure 4.4-2 Smoke circulation and evacuation in the case of a fire in an operational type C (or CU) unit

- **Fires with smoke layering**

Fires with smoke layering lead to the smoke forming pockets at the top of the drifts (see Figure 4.4-3). It is then the temperature conditions associated with the radiation from this smoke which pose the main hazard to personnel. This type of fire occurs in the presence of a low airflow in the drift (under approximately 1 m/s) with smoke which spreads either side of the seat of the fire.

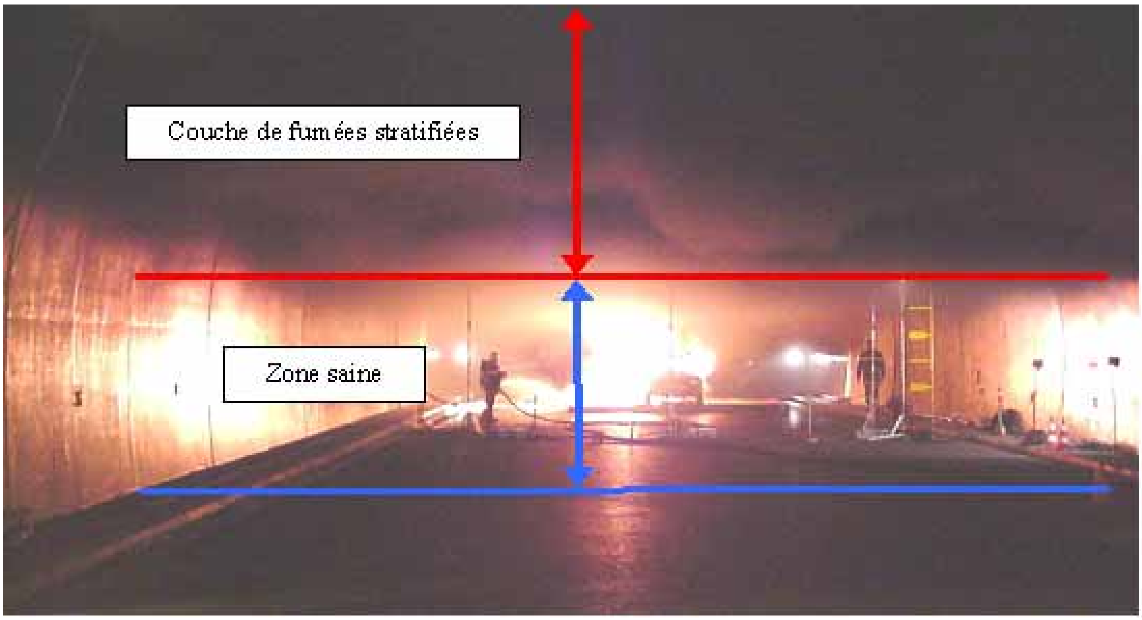

Figure 4.4-3 Qualification tests on the smoke control system in the Orelle tunnel (SFTRF and SETEC) – example of a fire with smoke stratification.

In a repository, fires in connecting drifts, with smoke extraction by vents fitted at regular intervals, would be of this type.

Depending on their position with respect to the fire, personnel can escape upstream or downstream of the fire, via the nearest interconnecting drift, in order to regain the adjacent connecting drift supplied with fresh air. The maximum distance to reach these interconnecting drifts varies from 100 m in secondary connecting drifts, inside repository zones (see Figure 4.4-4) to 400 m in main connecting drifts.

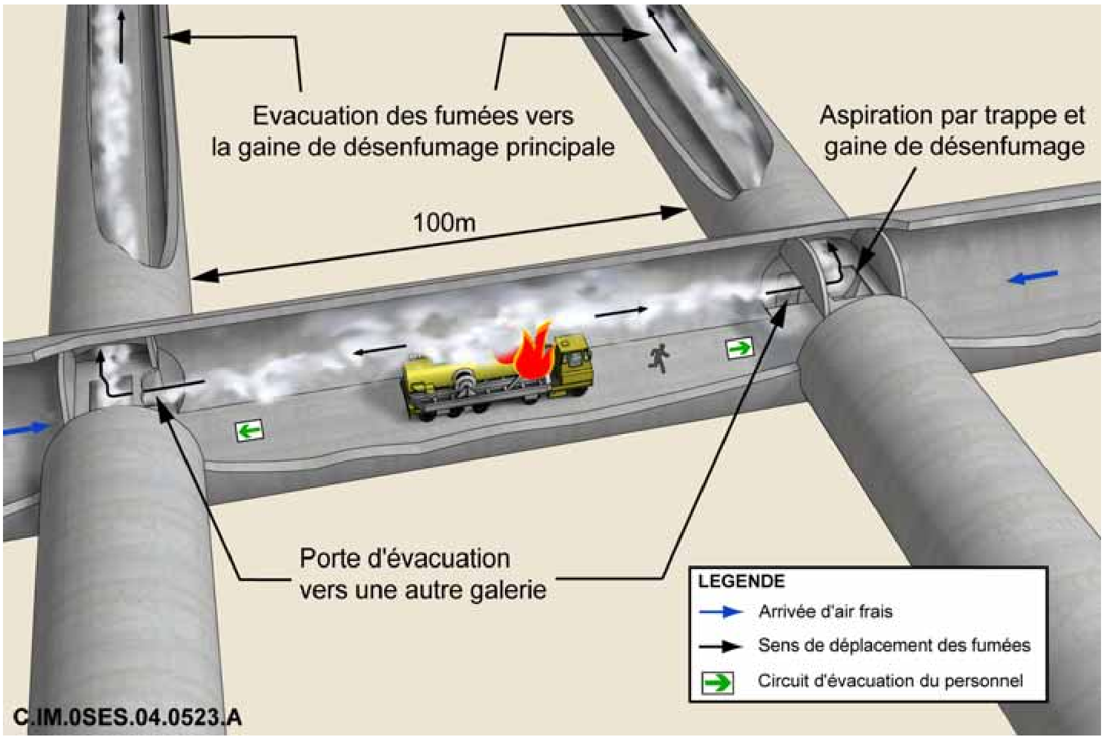

Figure 4.4-4 Case of a fire in a connecting drift : evacuation either side of the fire

- **Particular case of a fire in a dead end drift**

A fire in a dead-end drift is a special intermediate case between the two previous cases. On the one hand, it is similar to a fire with smoke de-layering as the smoke is only extracted via the site air extraction duct but, on the other hand, the short distance between the seat of the fire and this smoke extraction point limits the smoke encroachment into the drift.

There are two possible situations (see Figure 4.4-5) : if personnel can escape upstream of the fire into the ventilation airflow, they head for the adjacent drift supplied with fresh air via the nearest interconnecting drift ; otherwise they head for the mobile refuge[40] to await rescue.

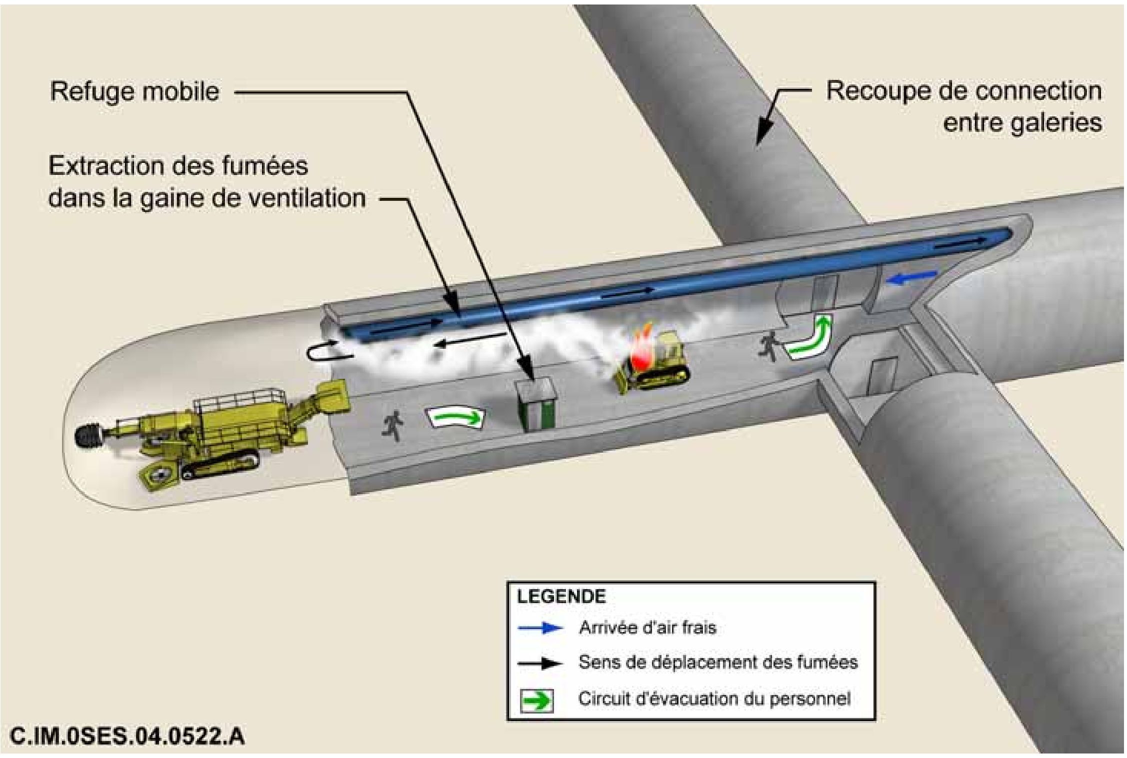

Figure 4.4-5 Fire in a dead-end drift : personnel evacuation via an interconnecting drift into an adjacent drift or sheltering in a mobile refuge

In all the above scenarios, personnel must then head via connecting galleries supplied with fresh air to the personnel transport shaft in order to return to the surface.

4.4.2 Simulation of personnel evacuation conditions

The results of these preliminary simulation studies are presented, as before, by type of fire as the personnel evacuation conditions are directly linked to whether or not smoke layering occurs.

4.4.2.1 Fire with smoke de-layering

The example given is a machinery fire with a thermal power of 15 MW in a type C cell access drift during nuclear operation. This power represents the maximum power of the machinery used.

The example used is that of a fire just after an intersection, which corresponds to a maximum evacuation distance before reaching the next interconnecting drift of almost 200 m.

[40] This mobile refuge, fitted with fire-resistant walls, has compressed air and water storage tanks.

The data obtained[41] (temperature, carbon monoxide concentration and air/smoke mixture opacity) are comparable with the permissible thresholds for survival and escape conditions for this type of fire. The temperature shall not exceed 80°C for more than 15 minutes. The carbon monoxide content shall not exceed figures within a range estimated at between 500 ppm for 60 minutes and 3000 ppm for 10 minutes. Finally, an opacity greater than 0.3 m^{-1} corresponding to a walking visibility of 7 m starts to hinder personnel evacuation ; it becomes difficult with an opacity greater than 1 m^{-1} (visibility less than 1.5 m).

The results of the simulations are presented, emphasising personnel evacuation conditions in accordance with two assumptions : the normal evacuation velocity is 1 m/s ; that of a group having to evacuate a casualty would be 0.5 m/s.

- **Temperature**

The diagram [downstream distance from the fire (m) / time since outbreak of fire (s) / temperature (°C)] highlights the influence of the smoke clearance system[42] the operation of which deflects the rise in temperature of the air in the drifts downstream of the fire ventilation airflow.

The simulation (see Figure 4.4-6) shows that the evacuation of personnel located downstream of the fire should be possible under acceptable conditions even for personnel close to the seat of the fire. Personnel escaping at normal speed reach the interconnecting drift in an atmosphere at a air temperature of below 30°C, and between 40°C and 50°C if they are slowed down by the presence of a casualty. These conditions remain acceptable with respect to the 80°C threshold mentioned previously.

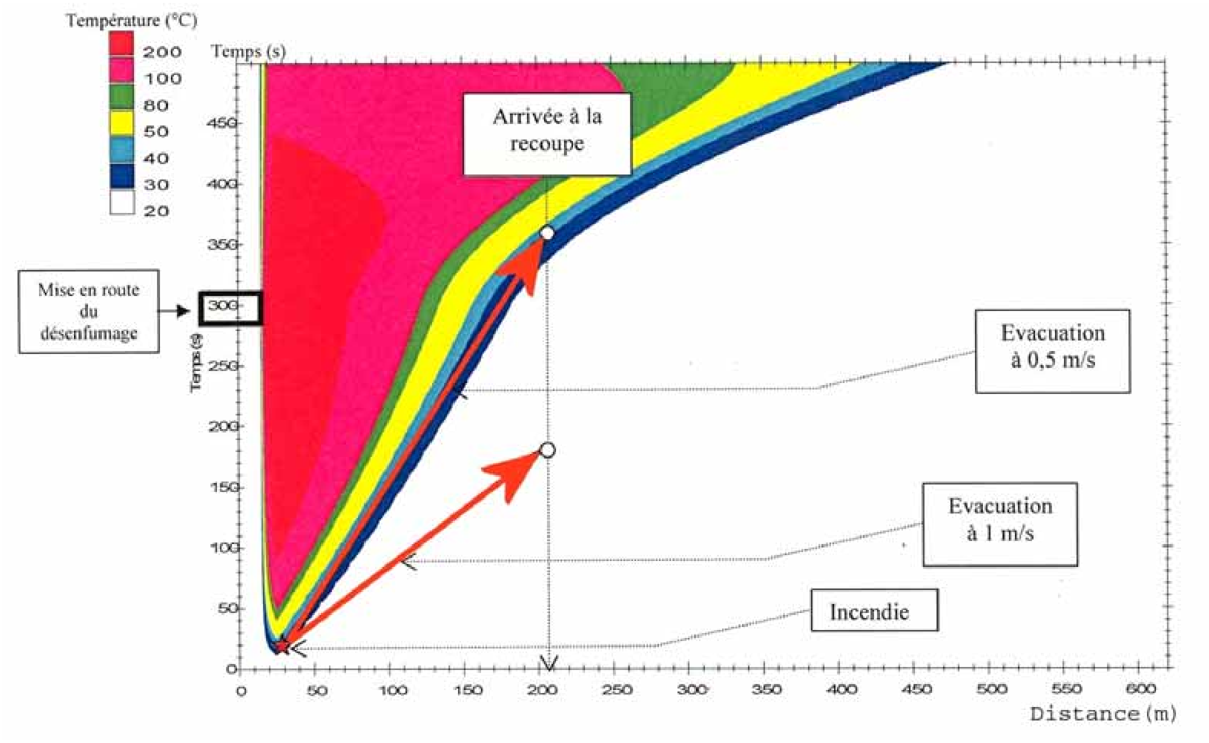

Figure 4.4-6 Spatio-temporal evolution of the air temperature (°C) in the case of a fire in a type C waste cell access drift during nuclear operation – Representation of the movement of personnel located downstream the fore ventilation airflow

41 The temperature changes have been calculated by applying the thermal advection laws in a turbulent air flow. The changes in opacity and the carbon monoxide concentration were digitally simulated, as for the temperature, using the Camatt (tunnel transient anisothermal monodimensional computation) software.

42 It was considered that that the smoke clearance installations would be started up 5 minutes (300 seconds) after the outbreak of the fire and that a stable state would be established after 8 minutes. These data come from experiments in underground structures

- **Carbon monoxide concentration**

Using the same approach as before, the simulation shows that persons located downstream of the fire ventilation, escaping at normal speed, can reach the interconnecting drift under healthy conditions. At an evacuation velocity reduced to 0.5 m/s, personnel arrive at the interconnecting drift in an atmosphere whose a carbon monoxide content (around 200 ppm) is still below the 500 ppm threshold.

- **Opacity**

The simulation shows that personnel escaping at 1 m/s can reach the interconnecting drift without being hindered by the opacity, which remains under 0.10 m^{-1}, corresponding to a visibility of around 10 metres.

However, with a slow evacuation velocity (0.5 m/s), personnel would quickly be caught up by smoke, with a mean opacity of around 0.6 m^{-1}, or a visibility distance reduced to 2 m. In this case, personnel may have to wear their individual breathing apparatus for the final few metres before reaching the interconnecting drift.

4.4.2.2 Fire with smoke layering

The example used is that of machinery with a thermal power of 30 MW in a connecting gallery. This case, which corresponds to a fire involving heavy construction machinery, is a bounding case compared with that of a package transporter which would have a lower thermal power (15 MW).

The propagation, height and temperature of the layer of smoke were determined analytically, using knowledge of stratification phenomena observed in the case of road tunnels. The effects associated with radiation emitted by the smoke from the fire are comparable with the permissible threshold for thermal effects on humans (2 kW/m² for persons not equipped with protection[43]).

The results of simulation (see Table 4.4-1) show that escaping personnel are subjected to a maximum heat radiation level of 0.9 kW/m² after 100 m in the case of an evacuation at the slower velocity (0.5 m/s), which corresponds to satisfactory conditions.

Heat radiation received by personnel (kW/m²)		
Speee of personnel movement / **Location with respect to seat of fire**	0.5 m/s	1 m/s
100 m	0.88 kW/m²	0.38 kW/m²
200 m	0.56 kW/m²	0.38 kW/m²
400 m	The smoke does not reach this distance due to the starting of the smoke clearance system	0.36 kW/m²

Table 4.4-1 Heat radiation received during evacuation in the case of a fire (30 MW) with layered smoke in a connecting drift

4.4.3 Conclusion

In underground repository installations, two types of fire can develop : fires with smoke layering (in connecting drifts) or smoke de-layering (in access drifts to type C or spent fuel waste disposal cells).

[43] This radiant rating corresponds to a maximum smoke temperature of around 200°C, wich would induce an air temperature lower than 80°C in the healthy zone (see Figure 4.4-3)

In both cases, simulations conducted tend to indicate that the design of the underground infrastructure, with clusters of parallel drifts connected at regular intervals by interconnecting drifts, enables personnel to escape from the location of the fire under satisfactory conditions, quickly reach a parallel drift supplied with fresh air (clear of the smoke circuit) then return to the surface under good conditions.

In a few cases (fire with smoke de-layering and slow evacuation speed) it cannot be excluded at this stage that the smoke might catch up with the escaping personnel and that they may have to use their personal breathing and eye protection.

A special case is that in which a fire develops during work in a dead-end drift. Personnel may find themselves between the fire and the end of the drift, unable to reach an interconnecting drift in order to escape. In this situation, they would have to take shelter in a mobile refuge (equipped with compressed air and water) which would be designed to be fire-resistant and smoke-proof. Once there, personnel would wait to be rescued by a rescue team which should arrive at the scene of the fire in order to act as quickly as possible.

Feedback from underground work sites indicates that particular effort must go into preventive measures and personnel training, with regular exercises in order to learn essential emergency reflex actions. If personnel are well trained and have adequate resources in order to intervene effectively in the event of a fire, they generally manage to extinguish the fire before it has time to develop. The use of specialised personnel, equipped with full fire fighting gear, would only be required in about 10 % of fires [55].

4.5 Study of the consequences of a fire in a vehicle transporting the transfer casks (B, C and CU)

The transfer casks for waste disposal packages are loaded in the nuclear surface installations on a transport vehicle. This vehicle brings the transfer casks to the package transfer shaft and places them on a metallic support in the cage, which lowers them to the underground installations. The transfer casks are then recovered by a vehicle similar to the one used on the surface and transported by this vehicle to the disposal cells.

In addition to the usual consequences resulting from a fire (temperature increase, smoke, …) treated in the previous section, if this vehicle is on fire, it could have radiological consequences if the fire impacts on the protection ensured by the transfer cask (protection against external exposure) and the primary package (maintaining the containment).

The purpose of this section is to estimate by means of simulation studies [46] whether radiological consequences may result and propose, if necessary, additional measures to prevent this risk or protect against it.

4.5.1 Assessment of the consequences of a fire in the transport vehicle on the transfer cask and its contents

This assessment is based on the assumption of a fire breaking out in a transfer cask transfer vehicle (B, C or spent fuel) despite recommended preventive measures (see Section 4.4). The nature of transfered packages, the evolution of their characteristics versus the temperature to which they are raised and the main characteristics of the transfer casks are referred to in Table 4.5-1.

Primary package (internal temperature)	**Specific data related to the temperature conditions**	**Disposal package**	**Transfer cask**	
			Materials	**Thickness (mm)**
B2.1[44] **bituminised sludge** (ambient temperature)	Bitumen-embedded waste softening ≥40°C flash point ≥230°C spontaneous combustion at approx ≥ 350°C	Concrete package (with 4 primary packages in vertical position)	Steel Thermal shield [45]	180 to 220 20
CSD-V vitrified wastes (64°C at waste core)	Crystallisation of the glass : 450°C	Metallic overpack (with a CSD-V package in horizontal position)	Internal steel Neutron absorhing External steel	200 to 230 140 to 150 20
CU1 spent fuels UOx (87°C at centre of assembly)	Embrittlement of the fuel clad above 500°C	Metallic container (with 4 spent fuels in horizontal position)	Internal steel Neutrophage External steel	70 120 20

Table 4.5-1 Main characteristics of the packages and transfer casks

4.5.1.1 Definition of a fire scenario

The vehicle used to transfer the transfer casks is a self-propelled electric transport vehicle on tyres. The reference fire studied is located in the underground installations. It was defined from the specific characteristics of the vehicle and the recommendations of the Road Tunnel Study Centre (CETu) ; it corresponds to a fire with a heat rating of 15 MW for a duration of one hour (see Section 4.4).

The fire breaks out and spreads in a connecting drift with a ventilation rate under normal operation of approximately 30 m^3/s, which increases to 50 m^3/s after the smoke removal system is started.

[44] The bituminised sludge package was retained among the various B waste packages because it is the most risky in terms of ignition.

[45] For some B waste packages (B4, B5), a neutron absorhing material whose characteristics would allow it to play the role of a thermal shield is to be used like for the C and CU waste packages.

4.5.1.2 Data related to the simulation studies

The simulation uses a finite elements method for an elaborated description of the temperature distribution taking into account the non linearity of heat exchanges.

The heat produced by the fire is transmitted from the source of the fire to all the components of the transport vehicle and the transfer cask by a advective flow and a radiant flow representing 2/3 and 1/3, respectively, of the total heat rating of the fire[46] :

- The advective rating appears in the fumes emitted by the fire source. They are mixed with the air circulating in the drift, they envelop the transfer cask and exchange heat with it,
- The radiant rating is related to the radiation generated by the fire source. It is applied differently depending on the transfer cask's geometric form. In the case of cylindrical transfer casks (C waste and spent fuel packages), it is uniformly applied to the vehicle's flat bed, the transfer casks' frame and their lower half cylinder. For parallelepiped transfer casks containing B waste packages, half of the rating is applied to the lower surface in contact with the vehicle's flat bed and half to the side surfaces.

4.5.1.3 Results related to a fire on the transfer cask, the disposal package and the waste package of bituminised sludge (B2)

Table 4.5-2 shows the results of the fire simulation with the maximum temperatures which would be reached at the external surfaces of the transfer cask, the disposal package and the primary packages. It also gives the duration required from the outbreak of the fire to reach this maximum temperature.

		Maximum temperature reached (°C)	Time to reach Tmax from the outbreak of the fire
Transfer cask	Lower surface	900	~ 1 h
	Side surfaces	455	~ 2 h
	Upper surface	127	~ 1 h
Disposal package	Depending on the surface	from 203 to 561	~ 3 h
Primary package	Container body	143 (in lower surface)	~ 16 h
	Wastes	< 120 (except in package base)	~ 16 h

Table 4.5-2 Temperatures estimated at the transfer cask, the disposal package and the primary package of bituminised sludge (B2) obtained by numerical simulations under fire conditions

The simulations show that there isn't any risk of igniting the bituminised sludge wastes because the integrity of the transfer cask, the disposal package and the container of the primary package is preserved, preventing any contact by a flame with the bituminised matrix. Spontaneous combustion of the bituminised products, which requires a temperature on the order of 350°C, is also impossible. The only consequence of the fire on the wastes would be a softening of the bituminised matrix.

The temperatures determined show that the transfer cask could be slightly affected from a mechanical viewpoint because of its large thickness.

Thanks to the protective effect of the thermal shield incorporated in the transfer cask's structure, the concrete overpack of the disposal package would be subjected to temperatures on the order of 500°C, which would not effect the waste package integrity.

The metallic container of the primary packages, which is not thermally affected by maximum temperatures on the order of 150°C (see Figure 4.5-1), would not be damaged by the fire.

Therefore, these results refute the assumption of a deterioration of the primary waste package and a loss of containment.

[46] This breakdown corresponds to the recommendations of the CETu (Road Tunnel Study Centre) for a fire in this type of vehicle.

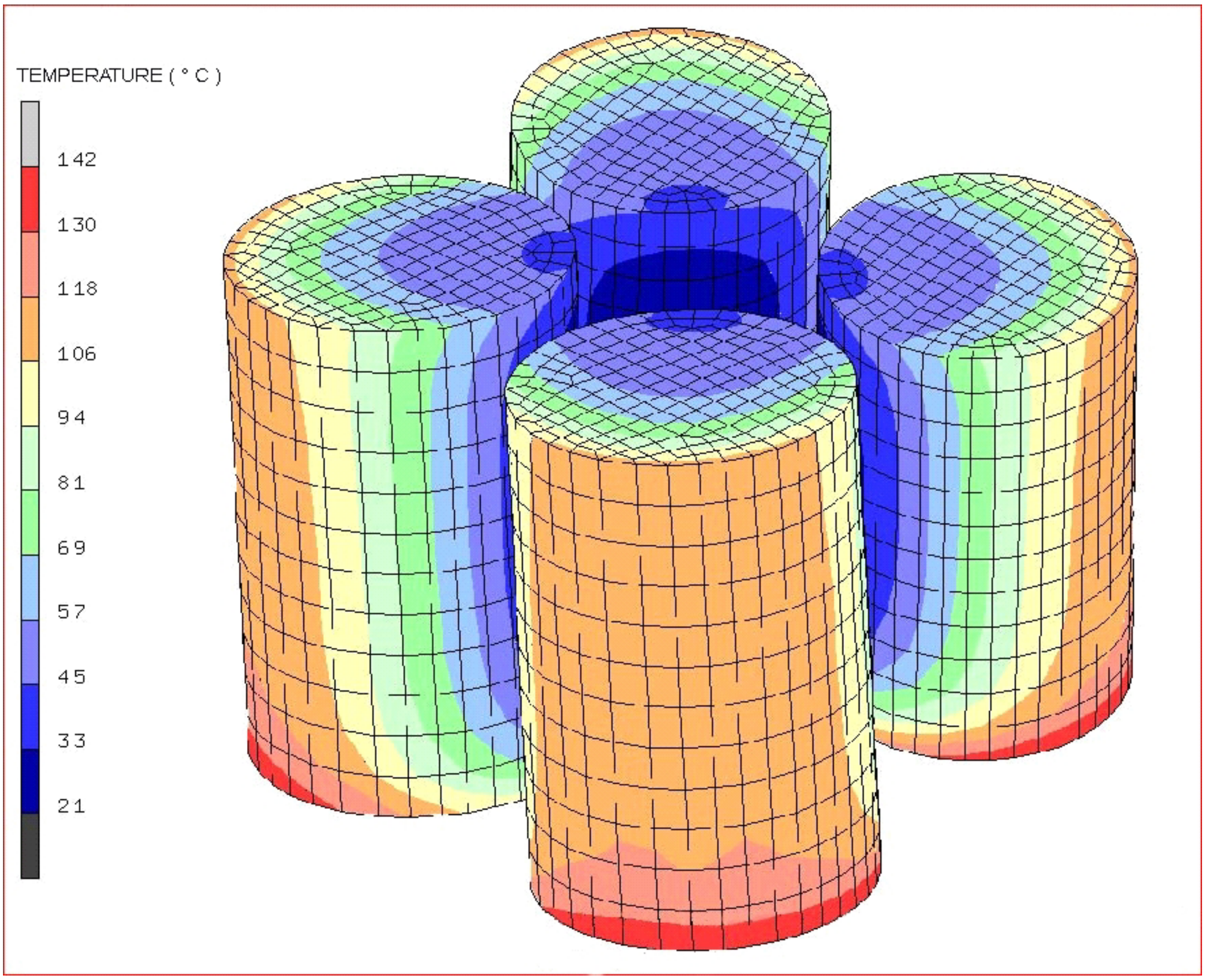

Figure 4.5-1 Temperature distribution in the primary packages of bituminised sludge (B2.1)[47]

[47] In this figure, the fire source is located under the packages, and the concrete overpack of the disposal package is not represented.

4.5.1.4 Results related to a fire on the transfer cask, the disposal package and the vitrified waste (C) package

In the case of C waste packages (see Table 4.5-3), the simulations show that the transfer cask's thickness and the presence of the PPB neutrophage material (Plaster Polyethylene Boron), which acts as a thermal shield, would allow limiting the temperature at the wastes to approximately 90°C.

This result appears to be a maximum value, given that the temperature on the external surface of the transfer cask is estimated at approximately 1 400°C, while experience shows that fires in an underground environment normally reach temperatures between 800 and 1200 °C [56], [57].

The integrity of the disposal package (over pack / container) therefore appears to be guaranteed and the temperature reached in the primary package would not be liable to change the characteristics of the vitreous matrix of the C wastes.

		Maximum temperature reached (°C)	Time to reach Tmax from the outbreak of the fire
Transfer cask	External surface	1410	~ 1 h
	Plaster polyethylene boron	1347	~ 1 h
	Internal surface	71	~ 4 h
Disposal package	According to the surface	70	~ 23 h
Primary package	Container body	71 to 83	from 23 h to 27 h
	Wastes	92	~ 30 h

Table 4.5-3 Temperatures estimated at the transfer cask, the disposal package and the primary package C obtained by digital simulations under fire conditions

Considering the fire's temperatures, the transfer cask may be affected by steel creep phenomena and PPB adhesion defects. This would necessitate checking the radiological protection level before performing any transfer cask recovery operation.

4.5.1.5 Results related to a fire on the transfer cask, the disposal package and the assemblies of spent fuels (CU1)

The results related to the fire on a vehicle transporting a transfer cask of spent fuel lead to the same comments as in the case of C waste packages (see Table 4.5-4).

The thermal stresses would not induce any mechanical consequences on the spent fuel assemblies. There would not be any risk of damage because they would only be subject to a rise of about 15 degrees with respect to their initial temperature.

As previously, the results obtained refute the assumption of a deterioration of the spent fuel packages provoking radiological consequences.

		Maximum temperature reached (°C)	Time to reach Tmax from the outbreak of the fire
Transfer cask	External surface	1280	~ 1 h
	Plaster polyethylene boron	1250	~1 h
	Internal surface	271	~2 h
Disposal package	Container	98	~4 h
	Insert	69	~8 h
	Cladding	67	~9 h
Assembly of spent fuels	Assembly centre	100	-

Table 4.5-4 Temperatures estimated at the transfer cask, the disposal package and the spent fuel (CU1) obtained by numerical simulations under fire conditions

4.5.2 Conclusion

During the transfer phase between the surface installations and the disposal cells, the fire on a transfer cask transport vehicle would not have any thermomechanical consequences capable of affecting the integrity of the transfered waste packages, which have a two-fold protection : the transfer cask, with one of its components serving as a thermal shield, and the disposal package.

Nevertheless, it would be necessary before any action required as a result of the accident is taken that the level of radiological protection ensured by the transfer cask be checked. In case this level is insufficient, it might be necessary, for example, to use movable radiological protections.

4.6 Study of the consequences of a fire on the emplacement vehicle (fork lift) of B waste disposal packages

In the surface installations, the B waste disposal packages[48] are placed in the transfer cask by an electric powered handling vehicle, or fork lift. These packages are then moved in this transfer cask up to the disposal cell where they are handled by the same type of vehicle to emplace them in a cell.

In addition to the usual consequences resulting from a fire (temperature increase, smoke, …), if this handling vehicle of the B waste disposal packages is on fire inside the disposal cell, it could have radiological consequences if the fire affects the function of maintaining the containment of the primary package.

The purpose of this section is to estimate by means of simulation studies [46] whether radiological consequences may result and propose, if necessary, additional measures to prevent this risk or protect against it.

4.6.1 Assessment of the thermomechanical consequences of a fire on the waste disposal package and the primary packages contained in it

This assessment is based on the assumption of a fire breaking out in a waste disposal package handling vehicle despite the preventive measures taken (see Section 4.4). The characteristics of the waste disposal package were previously described in Table 4.5.1.

4.6.1.1 Definition of the fire scenario

The retained scenario is the scenario which would take place while handling the type B2 waste packages because the bituminised matrix of the wastes for this type of package is the most sensitive to a temperature rise. On the other hand, the case of the fire on the underground vehicle was preferentially studied because this vehicle appears to have a higher heat potential than the surface vehicle.

The fork lift (emplacement vehicle) used to transfer the disposal packages from their transfer cask to their final emplacement in a disposal cell would be a self-propelled electric vehicle on rails. The fire considered would occur during the transfer of the disposal package between the cell's entrance and the already disposed packages. The fire would be electrically caused and would imply the vehicle's batteries and motors as the origin. The scenario takes into account the installation of a 2 cm thick thermal shield between the vehicle's motor section and the handling section.

The fire's characteristics were defined based on the vehicle's characteristics ; the fire corresponds to a heat rating of 3 MW for a half hour duration. This fire breaks out in a cell where the ventilation rate is on the order of 3 m^3/s.

4.6.1.2 Assumptions related to the simulation studies

The methodology used is similar to that indicated for the fire during a transfer of packages (see Section 4.5.1.2). The only difference is due to the fact that the disposal package is located laterally with respect to the fire source located at the vehicle batteries. Under these conditions, it is assumed that the radiant rating applied to the package side corresponds to half the total radiant rating and that 50 % of this rating is distributed over the side surface directly attacked by the fire and 50 % over the four adjacent sides.

[48] This illustrative case doesn't concern the C and CU waste packages directly emplaced from the transfer cask into their cells without requiring the use of an additional vehicle.

4.6.1.3 Thermal behaviour of the disposal package and the primary package of bituminised sludge B2 during the fire

Table 4.6-1 gives the results of the simulation with the estimations of temperatures which would be reached at the various surfaces of the disposal package and at the primary packages during the fire. It also indicates the time required from the outbreak of the fire to reach the maximum temperature.

		Maximum temperature reached (°C)	**Time to reach T max from the outbreak of the fire**
Disposal package	Side surface, from side of the fire source	521	~ 30 min
	Adjacent surfaces	319	~ 30 min
	Opposite side	134	~ 30 min
Primary packages located on the side of the fire source	Container body	98	~ 4 h
	Wastes	66 to 97	~ 3 h

Table 4.6-1 Temperatures estimated at the disposal package and the packages of bituminised sludge obtained by digital simulation of fire conditions

The simulations show that there isn't any risk of igniting the wastes because of the protection offered by the concrete overpack of the disposal package, which prevents any direct contact by a flame with the bituminised matrix. Spontaneous combustion of the bituminised matrix, which requires a temperature on the order of 350°C, is also impossible. The only consequence of the fire on the wastes would be a softening of the bituminised matrix related to its heating. Additional protection measures such as the shutdown or reduction of the ventilation flow in a cell (fire shut-off valve) could also be retained to isolate that cell and reduce the fire's intensity.

The primary packages would be subjected to temperature levels less than 100°C for the two most exposed packages on the fire side, which would not damage their metallic container and would not affect the contained bituminised wastes. Their average temperature would be around forty degrees (see Figure 4.6-1).

The concrete overpack of the disposal package would be subjected to temperatures on the order of 500°C, which should not affect its integrity. On the contrary, its mechanical properties may be altered and a scaling phenomenon is liable to occur on its surface[49].

In any case, the results obtained refute the assumption of a deterioration of the package of bituminised sludge provoking radiological consequences.

[49] Tests showed that fibre concrete packages of comparable thickness subjected to an average temperature of 800°C for 30 min could undergo superficial scaling phenomena on the surface in contact with the flame, but without the concrete bursting.

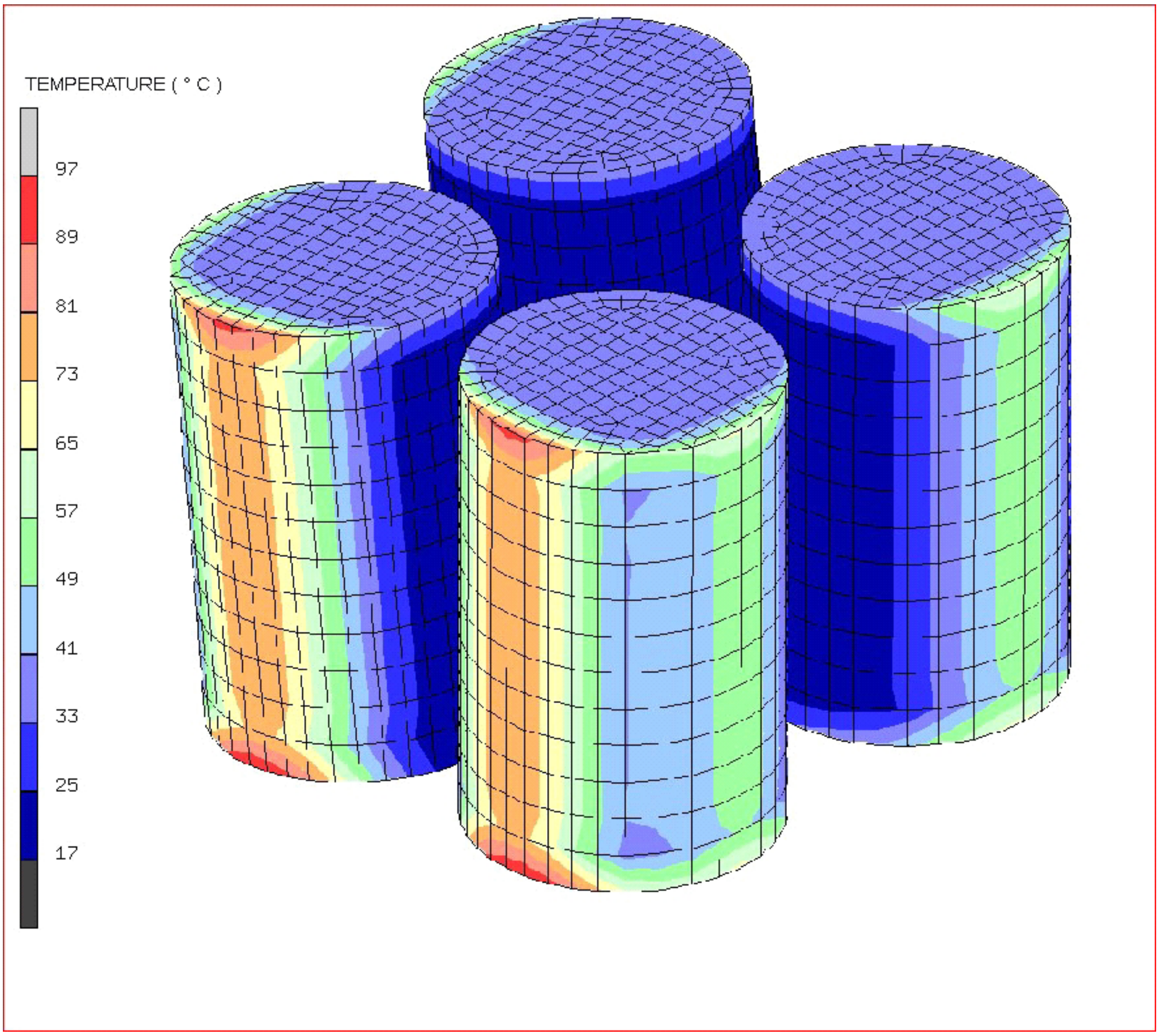

Figure 4.6-1 Temperature range at the primary packages of bituminised sludge (B2.1) in the case of a fire in a disposal cell[50]

4.6.2 Conclusion

Fire on the handling vehicle used to emplace the packages of bituminised sludge in the disposal cell should not have any radiological consequences because the concrete overpack of the disposal package and a thermal shield placed between the fire source and the package effectively protect the primary packages. On the other hand, the temperature which should be limited to about a hundred degrees is not of the kind to provoke the spontaneous combustion of the bitumen.

[50] The fire source is placed laterally to the left of the packages and the concrete overpack of the disposal package is not represented.

4.7 Study of the consequences of a cage fall during the transfer of the disposal package in a shaft

A free fall of the cage from the surface[51] is retained as the accidental scenario of falls in a shaft.

Within this context, the studies consist of, first of all, estimating the mechanical consequences of the fall on the various components of the moving body (cage, transfer cask, disposal package, primary packages) before assessing the radiological consequences of a potentiel loss of containment of the transfered waste packages and a release of radioactive materials [46].

They also include a risk analysis of the criticality [45] which may result from the cage fall during a transfer of spent fuel packages.

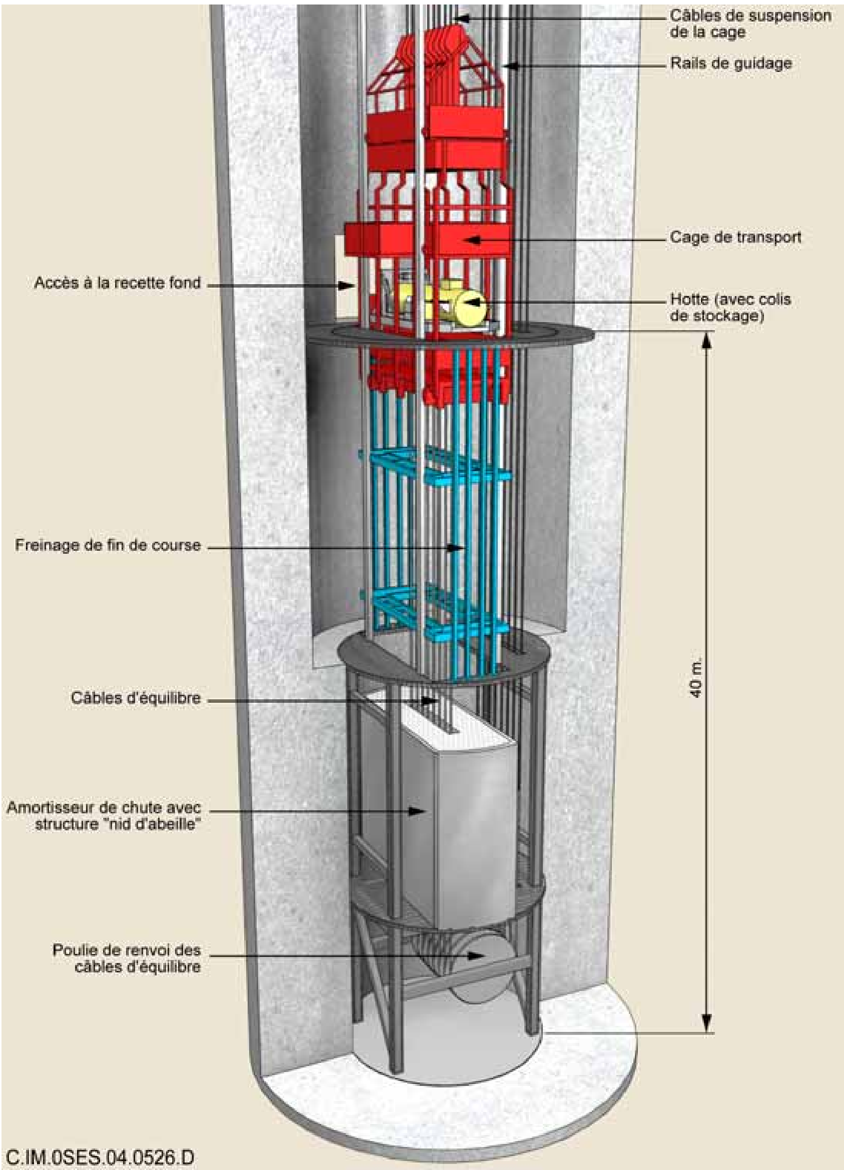

Figure 4.7-1 3D representation of the lower section of the shaft equipped with the fall shock absorber system

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51 The cage fall accidents related to a mechanical failure of the system are extremely rare. In France, the last serious accident known occurred at Reumaux (Lorraine) in 1925. It was due to a brake failure which caused a cage fall over approximately 600 m with the last 170 m in free fall after the cable broke (it was a drum winch). There was neither an anti-fall system nor a shock absorber in the shaft bottom. 55 miners were killed in this accident, while all of the 28 miners in the upper compartment of the cage were rescued alive.

4.7.1 Assessment of the mechanical consequences of a cage falling in a shaft

This assessment was made for the following B, C waste and spent fuel disposal packages (see Table 4.7-1).

Primary package	Disposal package	Transfer cask (load weight)
B2.1 : bituminised sludge [52]	Concrete package[53] (with 4 primary packages in vertical position)	Parallelopiped transfer cask (approx. 40 t)
CSD-V : vitrified waste	Metallic over-pack (with a CSD-V container in horizontal position)	Cylindrical transfer cask (approx. 50 t)
CU1[54] : spent fuel assemblies	Metallic container (with 4 spent fuel assemblies in horizontal position)	Cylindrical transfer cask (approx. 100 t)

Table 4.7-1 Main data related to the studied cases of a fall in a shaft

4.7.1.1 Fall scenario

The imagined scenario is that of the cage in free fall[55] over the entire height of the shaft, which would impact on a fall shock absorber system placed at the shaft bottom[56].

This scenario, which is overestimating for the fall height, doesn't take into account :

- An action of the cage anti-fall system,
- The possible braking effect due to a rubbing or a jamming of the cage's suspension cables during the fall,
- The effect of the end-of-travel braking stop at shaft bottom.

4.7.1.2 Data related to the simulation studies

The numerical simulation approach[57] is split into steps where the fall's consequences on the cage, the transfer cask, the disposal package and the primary package(s) transported are successively quantified. The condition of each element of these objects at the end of the fall is characterised by its plastic deformation, which corresponds to the cumulative total of all the types of deformations undergone (expansion, deformation, …). The corresponding coefficient expressed in % can be compared to the acceptable characteristic value (VCA) of the material making up this element. The risk of breakage is analysed versus the values of the deformation coefficient and the location, shape and extension of the deformed zone.

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52 The B2.1 disposal package was retained to represent the family of B waste packages because it corresponds to the most fragile primary package (metallic container of small thickness).

53 For the simulation, the characteristics of the fibre reinforced concrete package without rod reinforcement have been used, as this package is more vulnerable to the effect of a drop than the package made of rod reinforced concrete.

54 The CU1 packages (UOx), which contain several assemblies, appear mechanically more fragile with respect to the risk of a fall than the CU2 package (MOx), which contains only one assembly.

55 In this scenario, the breaking of the cage suspension cables has also as a consequence the fall of the counterweight over about fifty metres down to the shaft bottom.

56 Several technical elements condition the sizing of the shock absorber : the highest weight transported, the limitation of the deceleration of the transfer cask to a value not causing major mechanical damage, the limitation of the crushing of the shock absorber to half its height. In order to minimise the effect of the impact on the moving body, it is also planned to use a fall shock absorber with an upper stage having a crushing threshold less than that of the lower stage so that it operates in the most progressive possible way.

57 The fall simulation studies were carried out with the software Radioss, which allows studying by finite elements any strongly non linear behaviour of a structure subjected to forces from almost static up to rapid dynamic forces. Radioss temporally integrates the non linear dynamic equations by an explicit approach.

4.7.1.3 Consequences of a cage falling in a shaft

After an analysis of the fall's energy balance, the primary package's condition is examined to find out whether the damages caused by the fall on the shock absorber are or aren't liable to be evidenced in the end by a loss of containment and a release of radionuclides.

- **Energy balance**

The energy balance illustrated by the fall of the C waste transfer cask (see Figure 4.7-2) shows that the largest part of the incident kinetic energy was transformed during the impact into internal deformation energy in a very short time.

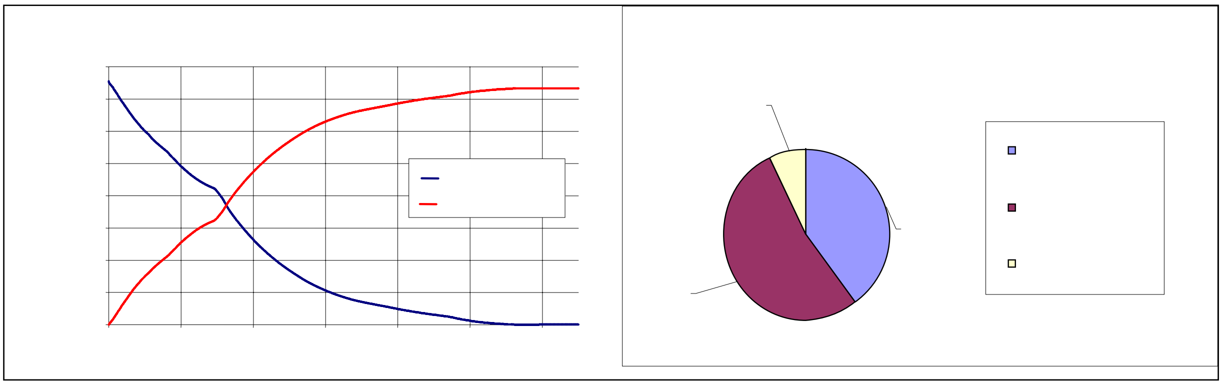

Figure 4.7-2 Energy balance of a fall in a shaft of a cage transporting a transfer cask with C wastes

More than 99 % of the shock's energy is absorbed by the shock absorber, the cage and the transfer cask. The disposal package absorbs less than 1 % of the energy. This distribution remains valid for the falls of B waste and spent fuel packages.

- **Mechanical consequences of the fall on the primary waste packages**

The estimation of the deformations observed on the primary packages can be used to understand the risk of a loss of containment of radioactive materials through the following two elements of estimation :

- The maximum value of the deformation coefficient of the primary package, which is only an indication (because a high deformation value limited to a few mesh elements is not significant),
- The surface of the primary package for which the deformation coefficient is greater than the acceptable characteristic value (ACV), which means a risk of breakage.

The results of the simulations (see Table 4.7-2) show that for all the B, C and spent fuel packages there isn't any risk of loss of containment of the radioactive materials since no deformation value is greater than the ACV of the materials making up the primary package. However, it should be noted that the B waste packages are the primary packages which undergo the greatest deformations and there persists an uncertainty particularly for the type B2 waste packages on the behaviour of their crimped lid to the shock.

Reference packages	Location	Maximum deformation (%)	ACV (%)	Surface for which the deformation exceeds ACV
B (B2.1)	Primary package	25.9	35	No over-extension is observed for the various types of packages. Their container would not be broken
C (CSD-V)	Primary package	0.3	35	
CU1 (bare assemblies)	Square compartment of the disposal package containing fuel assembly	1.2	26	
CU1 (assemblies in canister)	Cylindrical canister containing the fuel assembly	17.4	40	

Table 4.7-2 Estimations of the deformations undergone by the packages at the end of the fall on the shock absorber

The results obtained from the simulation studies underline the advantage of installing in the package transfer shaft a fall shock absorber. This absorber should prevent the breakage of the metallic envelope of the transfer cask in case the cage falls.

Inside this transfer cask, first estimations tend to show that the C waste primary packages, as well as the tubes or claddings containing spent fuel should resist the shock without breaking. Regarding the B waste primary packages, a damaging of some of the most fragile packages (such as the type B2 packages) can’t be completely excluded.

4.7.2 Assessment of the radiological consequences of a cage falling in a shaft

By taking into consideration the results obtained from the studies on the mechanical consequences of a cage fall, it could be considered that there is no loss of containment of radioactive materials during the transfer of packages in a shaft.

However, because of the uncertainities on how the fall in a shaft takes place and the definition of the moving body, scenarios of a release of radioactive materials were imagined in order to estimate the associated radiological risk [46].

4.7.2.1 Scenarios of a release of radioactive materials

The assumption is made of the transfer cask being opened, the disposal package being broken, and the primary package being broken, which would lead to a release of radioactive materials in suspension in the shaft atmosphere and rejected into the environment. The studied packages are those which have the most penalising radiological content within each category. Thus, the B5 package (B waste), the C3 package (C wastes) and the CU1 package (for spent fuels) are retained.

Various data sets were retained for the quantities of radionuclides liable to be freed and released into the shaft’s atmosphere. The data related to the B5 packages were defined by analogy with the observations made during the compacting operations (the energy has the same order of magnitude as during a fall and during a compacting). The data related to the C packages are linked to fall test results. Regarding the spent fuels, the parametric approach corresponds to assumptions on the extent of deterioration of the assembly, together with a more or less important breakage of the rods and crushing of the contained fuel pellets.

4.7.2.2 Data related to the simulation studies

The existence of doors at the shaft landing stations (see Figure 4.2-3) and the prohibition of access to the shaft zone during a package transfer operation are measures which should prevent any radiological risk to the persons present in the underground structures. Consequently, no estimation was made for the personnel assigned to the underground installations.

The radiological consequences of possible radioactive release into the environment at the outlet of an exhaust shaft were estimated taking into account all the potential exposure paths :

- The exposure related to passing releaseplume induces an external exposure and an internal exposure by inhalation,
- The exposure related to the deposits left by the plume induces an external exposure, an internal exposure by inhalation subsequent to the return in suspension and an internal exposure by the ingestion of foods.

4.7.2.3 Results

The simulation studies show that the phenomenon of an instantaneous suspension at the moment of the impact appears to be preponderant compared to the suspension phenomenon related to a sweeping of air circulating in the shaft. The released radionuclides would be completely delivered to the environment with potential consequences on people at the site limit, which is assumed to be 500 m from the reject outlet.

The preponderant ways of exposure would be for the three types of wastes the inhalation due to the passing by of the radionuclide plume and, to a lesser extent, food ingestion.

The preliminary results obtained from the simulations would lead to considering measures to reduce the risk, for example, by filtering the exhaust air of the underground installations. With this type of arrangement[58], which is commonly used in the existing nuclear surface installations, the total exposure dose would be on the order of 1 mSv[59] for a person at the site limit.

Another imaginable measure would consist of isolating the shaft zone (descent shaft of the packages and the air exhaust shaft) and then making an assessment of the state of events before taking an action adapted to the nature, scale and extension of the contaminated zone.

4.7.3 Analysis of the risk of criticality

On the basis of the disposal package concepts defined to date and for the movement of a single disposal package in the shaft, the accidental fall situations for the B and C waste disposal packages don't present a risk of criticality, regardless of the geometry of the disposal package after the fall.

For the spent fuel packages, the damaging of the assemblies (detachment of the rods, breakage of the clads, …) and the moving of them closer together after the cage fall could induce a phenomenon of criticality if, in addition, water ingress occurred within the assemblies [45].

Such a damage level doesn't seem probable as it appears from the fall simulation results. However, as a precaution, design measures aimed at preventing any water from being in the shaft would have to be retained (see Section 4.2.2.2) to completely exclude this risk.

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58 It may be interesting to subordinate the cage lowering movements to the start of the filtering unit to avoid continuous filtering with large air volumes. This type of operation was retained in the American transuranium waste repository of the WIPP, New Mexico [54], where the filtering unit installed on the air exhaust shaft is not started except if an incident occurs.

59 This value is to be compared with the fact that specific arrangements are regulatorily planned with respect to the public when accidental situations increase doses to more than 10 mSv.

4.7.4 Conclusion

In the transfer phase of packages in the shaft, the risk of the cage falling represents a very low probability of occurrence because of the recommended preventive measures. Despite these arrangements, should the moving body free fall in the shaft, the presence of a fall shock absorber in the shaft bottom would limit damage to the transfer cask, which would preserve its mechanical integrity according to the results obtained from the simulation studies.

However, due to the uncertainties of the fall simulations (definition of the moving body, fall conditions), the possibility of a release of radionuclides into the environment at the reject point was considered. In order to assess the radiological consequences of such an event, various illustrative cases were defined under a parametric approach. Thus, technical solutions could be proposed to limit the release of the radionuclides : one solution would consist of trapping the radionuclides by a filtration of the exhaust air at the exhaust shaft ; another would be to isolate the area comprising the waste transfer shaft and the point of release to prevent any risk of spreading. In both cases, the installations would have to be securised and control measures taken before proceeding, if necessary, with decontamination operations.

The risk of criticality associated with a fall of a spent fuel package is not imaginable because of all the arrangements proposed to limit package damaging and because of the absence of water in the shaft.

4.8 Study of the consequences of a B waste disposal package falling during its emplacement in disposal cells

B waste disposal packages are emplaced by a vehicle remotely controlled by an operator located outside the disposal cell. After the risk analysis, an accidental scenario was retained where a package would fall in a disposal cell while being stacked at the upper level of a package pile. This scenario is a bounding scenario with respect to the handling operations in all the facilities.

The purpose of the simulations [46] and full-scale drop test is to characterise the mechanical consequences of the fall on primary packages before estimating what the radiological consequences might be.

4.8.1 Assessment of the mechanical consequences of B waste packages falling in a disposal cell

The assessment of the mechanical consequences of a package falling was carried out for the two types of containers considered : standard container and container with reinforced retention capability. They are represented by a type B2.1 package and a type B5.2 package respectively. Their main characteristics are shown inTable 4.8-1.

Primary package Nature of the wastes	Number of primary packages per disposal package	Type of container	Demonstrator weight (t)
B2.1 : bituminised sludge	4	Standard container (the body and the prefabicated lid are in fiber and steel rod reinforced concrete)	6.1
B5.2 (CSD-C) : mix of hulls and end caps, and technological wastes	4	Container with reinforced retention capability (the prefabicated body and the poured in place individual lids are in fiber reinforced concrete)	12.3

Table 4.8-1 Main characteristics of the disposal packages studied [58]

4.8.1.1 Fall scenario

The retained fall scenario is based on the assumption of a flipping and turning over of the waste disposal package followed by a vertical fall of the package onto a corner of the cover on a rigid soil. The centre of gravity of the package is located vertical to the point of impact For the primary package located near the corner of impact, this configuration seems more severe than the in case of a « fall on a flat surface » or a « fall on an edge ».

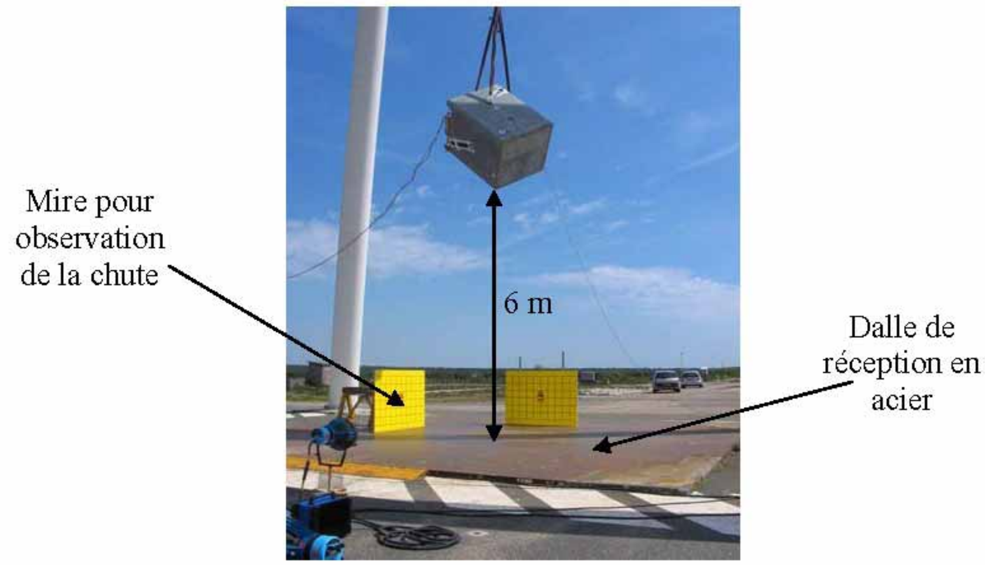


Figure 4.8-1 Installation of the drop-test demonstrator (package B5.2 - CEA/CESTA Test Centre - Gironde)

The fall height of 6 m[60] corresponds to the maximum handling height in of the disposal packages during emplacement in the disposal cells.

4.8.1.2 Data related to the simulation studies and drop tests

The simulation studies were conducted with the same approach as for the studies of a fall in a shaft (see Section 4.7.1.2). The software used takes into account the complexity of the occurring phenomena (materials subjected to plastic deformations and even breakage) and the variety of materials making up the disposal package.

The drop tests were carried out on a thick reinforced concrete slab, covered with anchored steel plates. The demonstrators used are identical in terms of dimensions and weight to the disposal packages as presented in section 4.1 of « Architecture and management of a geological repository » [22], except that the primary packages contain no radioactive material

4.8.1.3 Simulation assessment of the mechanical consequences of a fall of a waste disposal package

The purpose of the simulations is to estimate whether the damages caused to the primary packages by the fall of the disposal package would result in fine by a loss of containment and a release of radionuclides.

- **Energy balance**

The energy balance shows that the greatest part of the incident kinetic energy is transformed during the impact into an internal deformation energy. The concrete container of the disposal package absorbs 90 % of the incident kinetic energy. The primary packages receive the remaining 10 %.

- **Consequences of the fall on the primary packages**

The results (see Table 4.8-2) are given for the metallic canister of the primary package most exposed at the moment of the fall, which is the one in the immediate vicinity of the point of impact.

The criteria retained is the deformation of the canister of the primary package as compared with the acceptable characteristic value (ACV) of the material this canister is made of.

60 The fall height is defined as the distance at the start of the fall between the point of impact (lower corner of the cover) and the ground.

Reference package	Description of the fall	Maximum deformation of the canister of the primary package (%)	ACV (%)
B2.1	Fall on a corner of the lid	8	35
B5.2	Height of fall 6 m	9	35

Table 4.8-2 Estimated deformations observed on the metallic canisters of B waste primary packages subsequent to a fall in a disposal cell

The results indicate limited damage to the upper part of the envelope of the primary package with a maximum deformation value of almost 10 %, much lower than their permissible characteristic value (VCA). There is therefore no risk of envelope failure.

4.8.1.4 Lessons learnt from the drop tests on the demonstrators

The first drop tests on demonstrators B2.1 and B5.2 showed that the disposal packages are damaged but retain their mechanical integrity.

- Demonstrator B2.1 was crushed on the corner of the lid after impact (see Figure 4.8-2), which revealed some of its reinforcements. The scrapped lid absorbed the majority of the energy relating to the impact whereas the body of the demonstrator itself showed little visible damage. If the interface between the container body and the lid is fissured, the two components remain integral to each other (role of the tie-rods designed for this purpose).
- Demonstrator B5.2 (see Figure 4.8-2 absorbed the impact energy on the impacted corner and the side faces. Diagonal fissures originating from the impacted corner were visible. The lid presented thin fissures around the cast concrete plugs above the primary package slots.

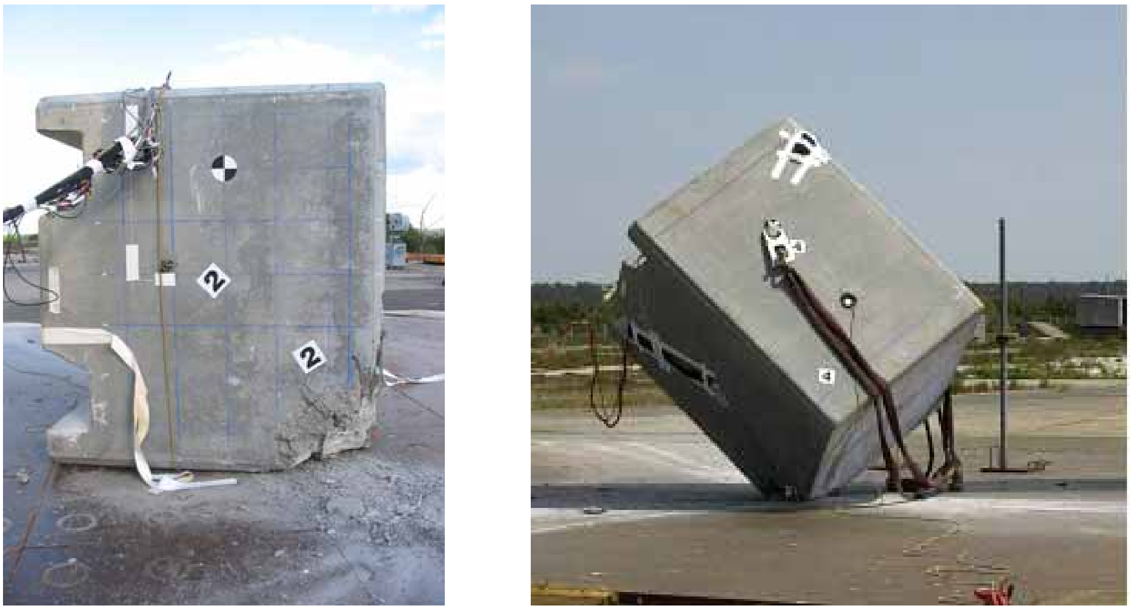


Figure 4.8-2 Appearance of the demonstrator following the drop test (left, demonstrator B2.1, right, demonstrator B5.2)

Coefficient de déformation en %

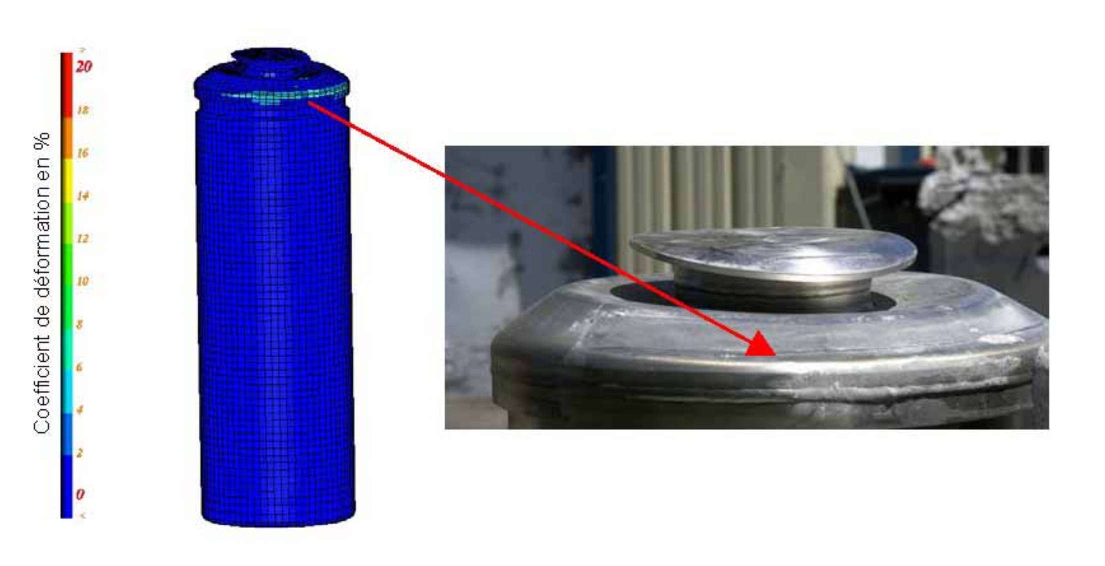


Figure 4.8-3 Comparison between simulation and drop testing – case of the primary package near to the impacted corner of demonstrator B5.2

In all cases, the primary packages removed from the demonstrators after drop testing had suffered slight deformations to the upper part of their envelopes (the part hitting the lid on impact), but showed no signs of failure. These observations match the results of the simulations (see Figure 4.8-3).

The studies and tests performed show that dropping a disposal package on emplacement in the cell is not likely to lead to a lack of confinement on a primary package level.

4.8.2 Assessment of the radiological consequences of a fall of B waste disposal packages in a disposal cell

Following analysis of the results of the studies and tests relating to the mechanical consequences of dropping a B package in the disposal cell, it is considered that no confinement of radioactive materials would be lost should such an event occur.

However, scenarios involving the release of radioactive materials in the case of B2 disposal packages have been envisaged, assuming that the package lid comes off at the moment of impact. This hypothesis was not considered in the case of B5 disposal packages given its different design comprising cast individual lids fully integral with the container body.

4.8.2.1 Retained assumptions

If the lid came off the disposal package following the impact, one or more primary packages could suffer varying degrees of damage, possibly leading to a release of radioactive materials which would be placed in suspension in the disposal cell atmosphere. The contaminated air would circulate in the return air circuit before being rejected into the atmosphere at the exhaust shaft.

Various assumptions were retained concerning the quantities of radionuclides liable to be released in the cell. In one case, one single package shows a local tear and, in another case, it is supposed that the lid of the four primary packages is detached. The nature of the matrix of bituminised sludges was also taken into consideration, because it has a good ability to immobilise the radioactive wastes.

4.8.2.2 Radiological consequences of a fall of B2 packages in a cell

The simulation studies show that the phenomenon of an instantaneous placement in suspension at the moment of the impact appears preponderant compared to the phenomenon of a continuous placement in suspension by the leaching of the radioactive materials through ventilationin the disposal cell.

The preponderant ways of exposing the public would be the inhalation due to the passing by of the radionuclide plume and, to a lesser extent, food ingestion.

Simulations show that the dose received at 500 m from the rejection outlet would be less than 0.001 mSv, regardless of the scenario retained. This negligible dose would not generate consequences to people or to the environment.

4.8.3 Conclusion

During the transfer and handling operations of the B waste packages in the disposal cells, no fall should occur because of the recommended preventive measures. Nevertheless, if it should happen that the package would flip over and fall on the cell floor in spite of these arrangements, this fall would cause mechanical deteriorations of the disposal package. Simulations show that the primary package should not suffer a loss of containment and should not release radioactive particles. However, for the packages of bituminised sludge (B2), scenarios were studied where particles are released in the disposal cell and then in the environment via the ventilation circuits. The radiological consequences were then estimated, and construed to be negligible for the public.

4.9 Synthesis of the analysis

The risk analysis performed covers the industrial activities of construction, operation and closure of the repository of long-lived radioactive wastes. Conducted together with the design studies, the purpose of this analysis at this stage in the project's study is to give technical orientations and propose tested measures to reduce risks guaranteeing reliable operation and allowing the operational safety functions defined by Andra to be satisfied. This systematically conducted analysis has benefited from feedback from existing industrial installations.

The assessment of the dosimetry with the installations in operation showed that the doses received by the personnel and the public would be less than the annual requirements set by Andra, that is, 5 mSv for the workers and 0.25 mSv for the public.

The risk analysis made a distinction between conventional risks, traditionally encountered in any industrial installation, the risks related to waste packages, and the risks related to the environment outside the repository.

In the surface installations, conventional risks exist at a more or less high degree throughout the various disposal activities. These are mainly crushing risks (fall of handled loads, being hit by a vehicle, …), fall risks related to work at heights, electrization risks, as well as the risk of fire... These risks do not justify additional studies at this stage, but will have to be carefully considered during the detailed design of the facilities and of their equipment.

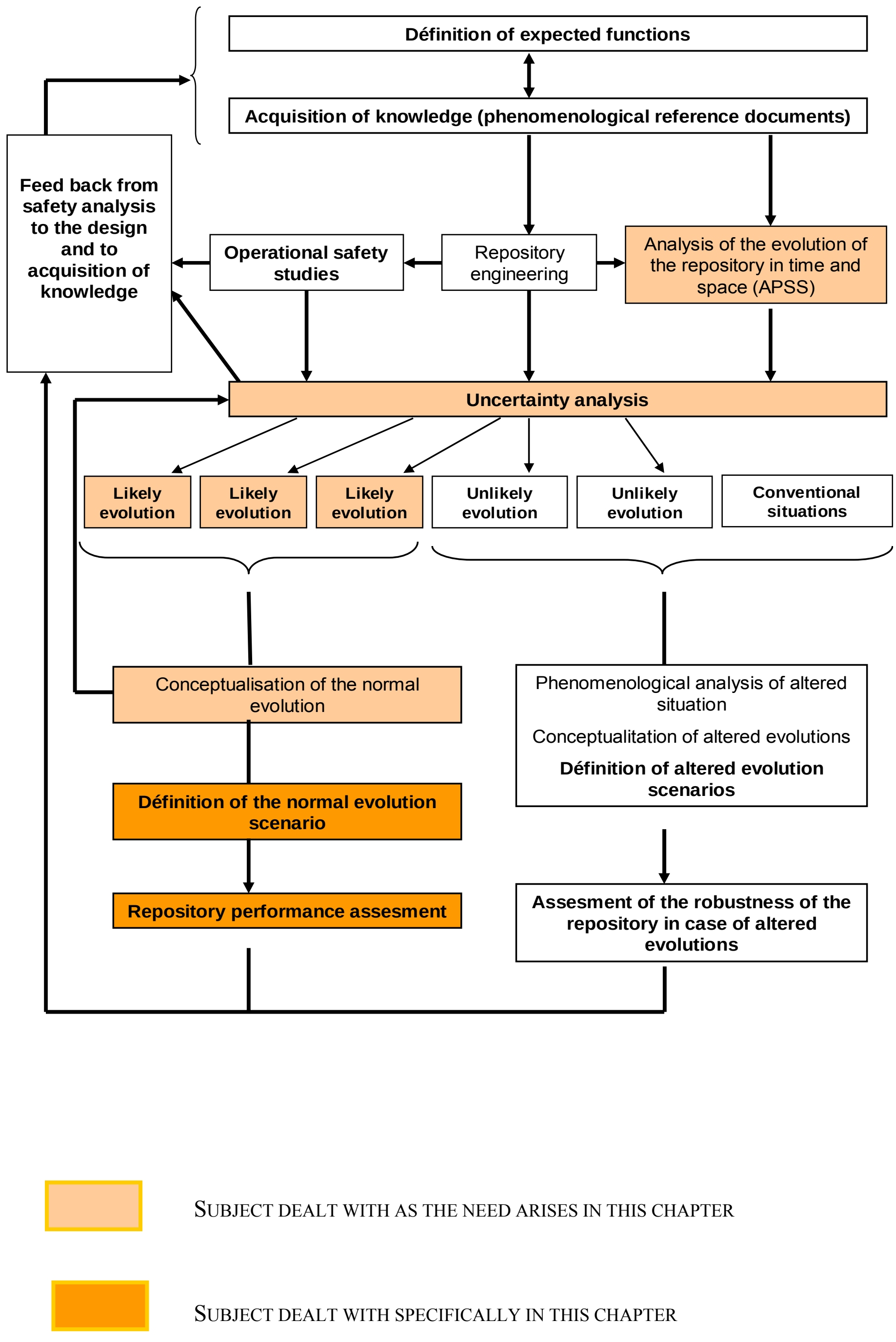
In the underground installations, these risks are also present. Among them, the risk of fire was covered by an additional study considering its influence on the design of the installations. This study ensured that the solutions recommended for the design of the infrastructures and their operation would allow evacuating people under satisfactory safety conditions.

The risks related to the waste packages are mainly radiological risks. They are present during the operating activity of the repository and to a lesser extent during the closure activity. These risks may be associated with radiological protection defects, with interventions performed in the vicinity of a radioactive source, as well as a fire or a fall affecting the packages themselves. The considered arrangements which benefit from feedback from similar industrial installations allow these risks to be controlled.

The risks related to the repository environment (earthquake, weather conditions, airplane crash, …), estimated based on the usual practices of French nuclear installations and taking into account the local characteristics of the site, do not raise any specific problems.

The operational safety analysis conducted did not reveal at this stage elements which could jeopardise the technical feasibility of the construction and operation, the closure of the repository and its reversible management by steps (with, in particular, the possibility of reversing the disposal process).

5 Assessment of the repository's long-term performance



Block diagram 5-1 Graphical representation of the sequence of analysis steps (see Block diagram 1-1) Topic : conceptualisation and definition of the normal evolution scenario and repository performance assessment

The purpose of this chapter is to evaluate the repository's impact in the normal evolution scenario (SEN), i.e. using simulations and calculations designed to cover all situations considered to be « certain or highly probable », according to the definition given by Basic Safety Rule RFS III.2.f. [2].

This step in the analysis validates the long-term safety of the proposed architectures. It will be supplemented in chapters 6 and 7 with an analysis of the robustness of the concepts with respect to residual uncertainties and any external events liable to affect the repository's evolution. This chapter draws on the description of the repository's architectures and the related post-closure functions, as described in chapter 3 as well as on the corpus of scientific knowledge (see in particular the volume « Phenomenological evolution of the geological repository » [7]) in order to formulate a representation of potential long-term radionuclide transfers. This chapter describes the various compartments of the calculation and the main arguments for the modelling choices, before presenting the results and the lessons to be drawn from them.

Note that, mainly to make chapter 5 easier to read, the uncertainties relating to the choice of situations, models and parameters in the SEN are not discussed in detail. Only the main choices dictating the basic calculation architecture are described here. Chapter 6 examines the uncertainties in greater detail. It offers a more systematic description of the way in which uncertainties are covered, either by the SEN and its sensitivity analyses or by the altered evolution scenarios. This two-stage presentation preserves reading clarity of chapter 5, while providing readers interested in a detailed analysis with the opportunity to refer to chapter 6 or the documents referenced therein.

5.1 Objectives of the normal evolution scenario

5.1.1 Definition of the normal evolution scenario

The performance analysis involves translating the functional choices and phenomenological understanding of the repository's evolution into a scenario representing the repository's life cycle. This scenario is known as the normal evolution scenario (SEN).

Because there are uncertainties relating to the repository's evolution over a one-million year period it is not possible to unequivocally define a single sequence of processes as being the reference evolution. There may be variants in the very nature of the physical and chemical interactions occurring inside the repository, and the length and spatial extent of the various phenomena are liable to vary. The concept of a « normal evolution domain » was introduced as a result of this uncertainty (see chapter 1, § 1.7.3) ; this domain represents the set of evolutions that appear probable enough to be treated as « normal ». The normal evolution scenario must represent these evolutions in a bounding manner, i.e. presenting a standard evolution having safety-related effects that are equivalent or unfavourable as compared to the situations in the normal evolution domain.

In addition to this definition, the SEN answers several distinct objectives. Its main aim is to verify that the repository, as designed and to the extent that its evolution over time is understood by contemporary science, fulfils the safety objectives assigned to it. This general objective can be broken down into several inter-related goals :

- Confirm that the performance achieved, as indicated by the chosen indicators, is consistent with the predefined threshold values. This safety objective implies the need to present a vision that exaggerates the repository's potential impact ;
- Provide an overall simulation of the repository's expected evolution, in order to assess the expected behaviour in global terms, in the form of a necessarily simplified and partially conventional representation that nevertheless aims to be as representative as possible. The aim is to assess the relative importance of the main phenomena and the performance of the safety functions. This understanding-oriented objective precludes the use of overly simplistic representations, which would make the models less representative.

- Provide a basis on which to judge the sensitivity of the level of safety to changes in the environment and the behaviour of repository components, and to use the sensitivity analyses as a tool for quantifying the repository's robustness.

The SEN is inextricably linked with a safety calculation model that is used to evaluate the SEN, yielding a quantified impact. The model is based on :

- The internal functional analysis of the repository (chapter 3), conducted in order to represent the components that perform safety functions. This representation is either direct (with the component and its characteristics being modelled directly), or via the safety function's effects (for example, a component whose function is to protect the host formation may not be represented but the characteristics of the host formation used for the calculation take the aforementioned protection into account). Certain components without a safety function may be modelled (in particular, components that act as a transfer path for radionuclides are modelled, even if they have no safety functions) ;
- The record of phenomena liable to occur inside the repository, from a chronological and geographical perspective, as described in the phenomenological analysis of disposal situations in normal evolution [28], which describes the reference phenomenology. Thanks to the systematic cataloguing of the phenomena at work, this analysis can be used to determine how the normal evolution scenario unfurls [59] ;
- The repository's detailed conceptualisation [60] which considers the initial definition of the phenomena and components to be included and proposes appropriate conceptual models (see insert 4) and representations. The models adopted for the purpose of calculating the SEN are a subset of the proposed conceptual models, chosen from a safety perspective (see section 5.2.1) ;
- An initial uncertainty analysis, performed continuously via the phenomenological analysis of repository situations (PARS) and the conceptual models, which allows them to be included in either the scenario description or the choice of sensitivity studies. This analysis, in the form of a discussion of the proposed phenomena, models and parameters, is not claimed to be comprehensive at this stage. The purpose of the qualitative safety analysis (see chapter 6) is to systematically run through the listed uncertainties and confirm that the SEN is part of a coherent whole.

Insert 4 Modelling levels

A model is generally a representation of a system or process expressed in a mathematical form. There are different types of model, according to their destined use :

- In research work, it may be important to 'test' an explanation of a phenomenon by comparing it with experimental results and observations. To do so, the hypothesis is translated into a model that allows the aforementioned observations to be reproduced, either empirically or by implementing the underlying phenomena. If they are not consistent with the experimental results, the model – and by extension the hypothesis – can be discarded. Conversely, if there is a good fit between the model's predictions and the experimental results, the model is said to have been 'validated'. Such models are developed, discussed and adopted or rejected by the scientific community on a regular basis ;
- For the purposes of the repository studies, Andra needs to represent the processes that take place inside the repository and govern its evolution. This representation in the form of « conceptual models » is a simplified but robust approach that reflects the influence of the key parameters. These models are used in particular in the preparatory calculations intended to gauge the impact of the various phenomena. Several conceptual models can be developed to cover uncertainties (see section 5.2.2.1) ;
- The « safety calculation model » (the generally-accepted term, although « safety calculation simulation » would be more accurate) is the mathematical representation of the repository and its environment that is used to evaluate radionuclide transfers. It is

based on the conceptual models, either by incorporating them directly into the model (for example, the conceptual waste matrix release models are included in the calculation by means of the equations that translate the model's result in mathematical terms), or by incorporating the model's results (for example, the conceptual model of the rock's mechanical behaviour is not taken into account into the simulation, but the extent of the damaged zone calculated using the model is included). Note that the safety calculation model cannot be subjected to a « validation » procedure. For one thing, it represents the repository's overall behaviour, which cannot be reproduced experimentally. Also, rather than faithfully reproducing a physical reality, the model provides a view of it that enhances its impact.

The SEN and its sensitivity studies form a non-dissociable whole. The scenario is made up of a series of calculation cases, as follows :

- A « **reference calculation** » that sets out Andra's current knowledge of the repository's foreseeable evolution, in an approach that considers both the fruits of scientific research and the safety strategy. The purpose of this calculation is to assess factors that would increase the impact of creating a repository. To this end, it includes a series of parameters and models, choosing those based on the best available scientific knowledge, and incorporating a degree of conservatism that varies according to the uncertainties, being less conservative where the parameters or models have been validated in detail, and more conservative where substantial questions remain outstanding ;
- A series of single- or multi-parameter **sensitivity analyses** that set out to rank the parameters and models by determining the ones that, if they were to vary, would have the greatest consequences for the overall assessment.

Note that the reference calculation itself has a variant, inasmuch as it uses two different hydrogeological models (see section 5.3.2.4). In broad terms, sensitivity studies can be treated as « variants » of the SEN if their configurations are deemed to be generally less representative of current knowledge than the reference calculation (whether because they are excessively conservative or because they include anticipated research results) while nevertheless relating to the « normal evolution domain ».

5.1.2 Protection objectives – the « critical group » concept

The protection objectives have already been described and discussed in chapter 1. They are merely restated below for information. Independently of any other indicators that may be used in the analysis in order to obtain information about the behaviour of the various safety functions (e.g. radionuclide flows at particular points in the repository system, and indicators such as the Péclet number that provides information about the general hydrological conditions), the main indicator generated by the SEN calculation is the individual dose at the outlet, which is evaluated for a critical group in a standard biosphere.

This group is located near the biosphere compartments liable to experience the greatest impacts (i.e. concentrations of radioactivity or chemical toxicity). The group's nutritional habits and lifestyle are determined on the basis of current knowledge relating to similar contexts.

This critical group lives in a surface environment – the « biosphere ». Predicting the evolution of the surface environment over long periods is an exercise fraught with uncertainty. Consequently, the concept of « standard biospheres » was introduced in Basic Safety Rule RFS III.2.f. These are defined on the basis of lifestyles as they are known today, without attempting to anticipate their evolution, as this cannot currently be reliably predicted. The major determinants of climate change and surface geodynamic evolution, to the extent that they can be predicted by models, are however taken into consideration when defining the model (for example, allowance is made for the possibility of cold periods and the natural evolution of the surface hydrographic system). The chosen biosphere is described in greater detail in section 5.3.2.6.

When considering the lifestyles of the critical group, the populations' current habits are assumed, and a safety margin is incorporated to increase the degree of self-sufficiency. The individual in the critical group represented in the calculation uses similar farming techniques to ourselves, but relies mainly on his own production to subsist. Although unrealistic in the context of the Meuse/Haute-Marne site, the influence of total self-sufficiency was also tested, as this assumption covers the uncertainties relating to future changes.

- **Impact assessment criteria**

The result of the SEN calculations is the set of adopted indicators, the values of which are compared with the criteria and thresholds specified in Basic Safety Rule RFS III.2.f. and by Andra (see chapter 1, section 1.4.2.2).

Note in particular that the dose is compared with the target of 0.25 mSv/yr for the first 10,000 years, as specified in Basic Safety Rule RFS III.2.f. This threshold is taken as a reference for the entire one million-year calculation period.

Two factors are used to assess the risk relating to the presence of toxic chemicals :

- The excess individual risk (ERI), expressed as a probability, which is used to assess the carcinogenic effects of toxins. The relevant limit has been set at 10^{-5} (i.e. 1 case per 100,000 individuals exposed for a 70 year-period) ;
- The danger quotient (QD), a dimensionless number used to assess the non-carcinogenic effects at the threshold. The associated limit is 1.

In addition, regulatory environmental concentration thresholds can be used if required.

5.2 From conceptual models to the safety calculation model

5.2.1 General principles

The normal evolution scenario represents the repository's operation while taking into account the most likely natural changes. The components play the roles assigned to them by the functional analysis, with a performance level equivalent to the best estimate of their behaviour in disposal conditions generally being used for the reference calculation. For example, given the degree of characterisation and highly consistent measurements obtained for the Callovo-Oxfordian layer, a model (expressed in terms of permeability, diffusion coefficient, porosity, etc.) that yields values representative of those regularly measured was chosen for the calculation. The characteristics and phenomena for which the research programme has yet to provide conclusive findings, or for which the results obtained reveal potentially substantial variability, are represented with a conservative vision.

The decision to represent the components performing the functions expected of them in the SEN should not cause any confusion as to the chosen strategy's purpose. As already stated in chapter 3, the functional choices made by Andra are consistent with current knowledge and supported by scientific arguments. They correspond to the repository components' most probable behaviour. Thus, although the SEN was devised as part of this approach in connection with the designer's choices, it complies with the definition of a « reference situation » in Basic Safety Rule RFS III.2.f., where it is described as « representing the predictable evolution of the repository with regard to certain or highly probable events ».

Once the SEN has been defined, it must be modelled to allow the calculation to be performed. The various components' behaviour models are based on the conceptual models. These models provide a mathematical representation of the physical phenomena. This representation can serve different purposes. For instance, some models faithfully reproduce a clearly-identified physical phenomenon, whereas others translate a number of experimental results (even if the underlying phenomena have not all been individually described), and certain models do not set out to reflect a reality at all, but serve some other purpose such as pessimistically representing a particular effect. A number of choices must be made in order to obtain a model suitable for use in assessing performance ; this model is referred to as the « safety calculation model » or « safety model ». These choices are intended to manage any uncertainties relating to the representation of reality by means of conceptualisation.

The following points should be noted :

- Several coexistent phenomenological models can be used to account for a given phenomenon, according to the state of progress of the studies, or the accuracy with which environmental conditions are taken into account.
- Models may depend on parameters fitting and adjustment. Such adjustments are based on available experimental data ; in numerical terms, this data may not be sufficiently representative to allow a mean and standard deviation to be calculated, which leaves a degree of leeway in the choice of the model's parameters ;
- In some cases, chaining the selected models together to form the overall calculation model can result in an exaggeratedly complex representation of the repository that causes prejudice to the good understanding of the fundamental mechanisms.

For all these reasons, certain choices must be made in order to position the « safety model », which forms the basis of the SEN assessment, in relation to the available conceptual models, as described for example in [60]. They must be made in such a way that they do not result in the repository's impact being underestimated.

Andra has sought to ensure that these choices are described as accurately as possible, and that the link between the conceptual models and the safety calculation model is as clear as possible, in keeping with the recommendations issued by the Dossier 2001's international review panel [15].

To this end, it is important to define standard terminology for qualifying the models and parameters proposed by scientists, to ensure that the « safety » choices are made on a standardised basis common to the science and safety engineers.

5.2.2 Terminology

This section introduces the terminology defined for the purpose of describing the conceptual models. This standardised 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).

5.2.2.1 Definitions relating to models

- **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 every parameter.

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.

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, one cannot define a « phenomenological » model, because the elements allowing to perform above described choice are not at ones disposal. For example, one cannot define a « phenomenological » release model 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.

5.2.2.2 Definitions relating to values

Four possible types of parameter are defined for a particular model.

- **« Phenomenological » (or « best estimate ») value**

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 anomalies ;
- In the absence of measurements, internationally-available data is used, provided it is sufficiently explicit in the literature.

« 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.

5.2.2.3 Value selection strategy for the safety calculation model

The strategy for defining the safety calculation model based on the conceptual models obeys the following principles :

- As already stated, 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 ;
- For reasons of calculation capabilities and analysis readability, not all of the phenomena that influence the repository's evolution are necessarily represented. This does not imply that they are overlooked or ignored in the analysis. They may have been studied separately or before the calculation was performed, with the conclusions of these preparatory studies leading to them either not being represented or being represented in a highly simplified form. This is the case with the hydraulic transient in the structures, which is not represented in the calculation despite being the subject of detailed studies (see section 5.3.1.3, or the damaged rock zone, which is directly represented as predicted by the mechanical models, although those models are not coupled with the rest of the calculation.

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 the documents that provide the framework for the modelling process (see [60] in particular).

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. 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.

In terms of method, models and parameters were chosen for the safety calculation and the sensitivity studies on the basis of knowledge acquired up to the end of 2003. An Andra internal review was conducted on these choices in order to validate them.

5.3 Description of the safety model

The conceptual models that describe the phenomena liable to occur in the repository are described and discussed in the volume « Phenomenological evolution of the repository » [7]. They are not re-discussed in this document. The rest of this chapter describes and justifies the safety calculation model directly.

The representational choices are described briefly, followed by descriptions of the various compartments of the calculation. A brief description of the main aspects of the phenomenological knowledge acquired by Andra during the research programme is given for each of these compartments, so that it can be clearly understood. An explanation of the manner in which each aspect is modelled is then provided.

The calculation architecture implemented to handle the scenarios is detailed in document [75]. A summary is made of the models selected and the methods used to guarantee the robustness of the results in section 5.4.

5.3.1 Input data and processing method

5.3.1.1 Definition of the transposition zone in the safety analysis context

Andra has determined a geographical domain around the Meuse/Haute-Marne laboratory, inside which the data relating to the geological environment determined by the laboratory are deemed to be transposable. This domain, known as the « transposition zone » has been defined using three categories of parameter, as described in volume 5 of the site reference document [17] :

- Layer thickness ; the chosen criterion is a minimum thickness of 130 m of Callovo-Oxfordian clay in the transposition zone ;
- Physical and chemical properties (mechanical, physical, chemical and formation containment properties) ; this category yielded two transposition zone definition criteria :
  - ✓ One criterion corresponding to the variation in facies, which provides the zone's boundary to the south of the flexure at Pertuis-en-Aulnois owing to a lack of information about the nature of the sedimentary deposits. Boreholes drilled in 2004 have provided new information about this part of the sector, offering the possibility of extending the transposition zone to the north. These recently-obtained results have not, however, been included in this dossier ;
  - ✓ A second criterion, corresponding to fracturing in the studied sector, which rules out the southern areas beyond the endpoints of the various faults, creates a 1.5 km « safety margin » zone relative to the basement faults that are not active (or at least not on the historical scale) to the west, and a 1 km zone relative to the Gondrecourt fault to the east ;
- The factors controlling the degree of disturbance around the underground structures that lead to the definition of a maximum depth criterion, thereby limiting the zone to the north-west.

The transposition zone can be defined by applying the criteria relating to the three categories of parameter. The zone occupies an area of approximately 250 square kilometres. Figure 5.3-1 shows the transposition zone boundaries.

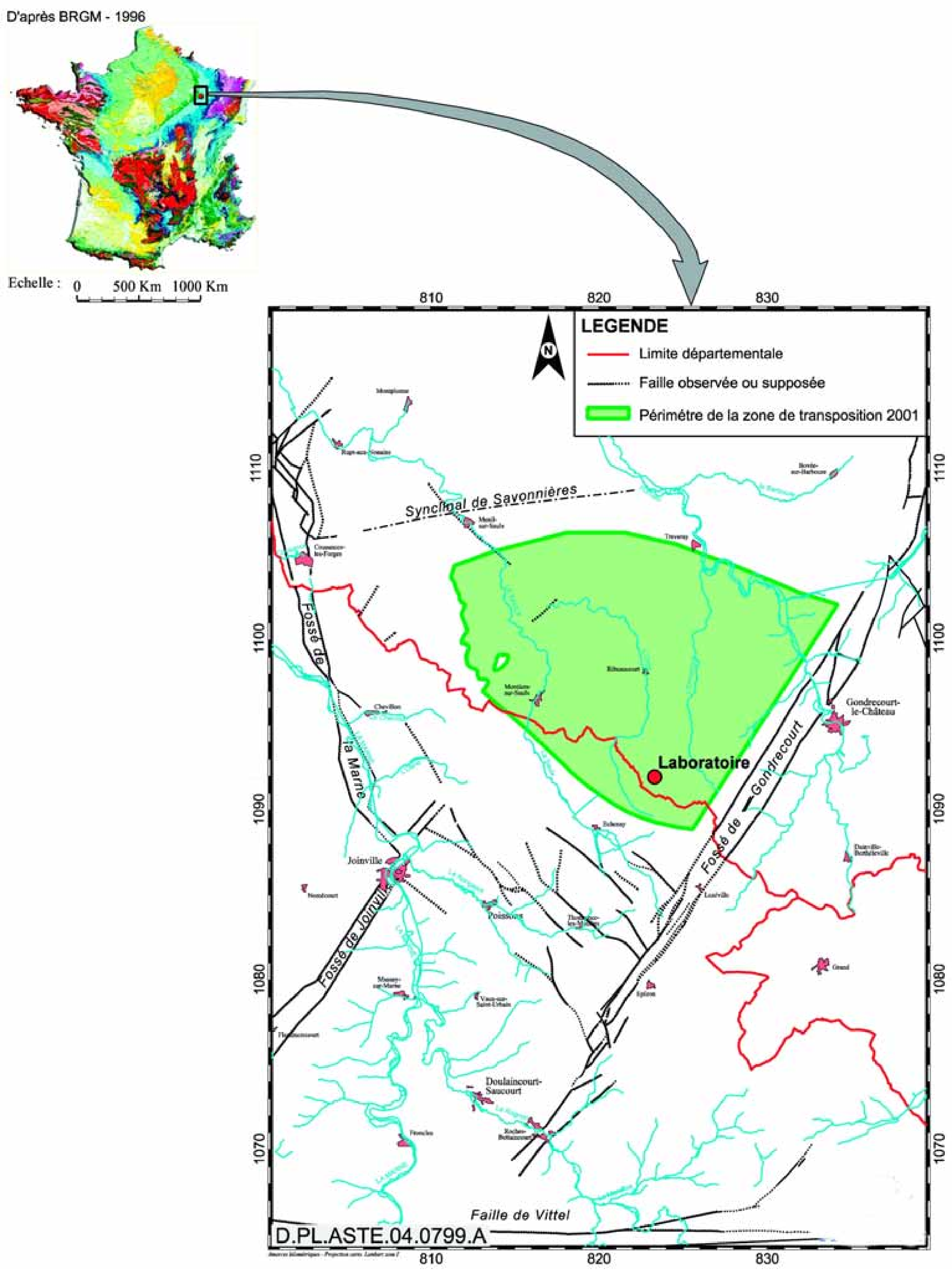


Figure 5.3-1 The Meuse/Haute Marne site – Transposition zone boundaries

The transposition-based approach raises two types of question, relating to the repository's location (« whereabouts in the transposition zone should it be located from a calculation perspective? »), as discussed in the next section, and to the performance assigned to the safety functions. Certain safety functions depend on the repository's position inside the zone. For example, the rock's geomechanical behaviour is more unfavourable at greater depths. The damaged zone may be more extensive in those areas. The performance of the « Resist water circulation » function could be impaired as a result.

The disposal concepts were defined for the depth of the Meuse/Haute-Marne laboratory, where the greatest volume of data is accessible and validated. However, to ensure that the conclusions of the calculation remain valid at the scale of the transposition zone, the performance of the disposal system are assessed in conditions that are unfavourable from a positioning perspective.

The most unfavourable location within the transposition zone, in terms of safety function performance, may vary between phenomena. Clearly, therefore, this approach can yield a global model that does not make allowance for a specific location of a repository in the transposition zone, as the global model is compiled by assembling models representative of phenomena that potentially occur in different locations.

In practice, two subjects are addressed : the thickness of the argillite layer, on which the performance of the « Delay and attenuate radionuclide migration » function is dependent, and the geomechanical behaviour of the rock, which notably influences the « Resist water circulation » function as the walls of structures are damaged. In the former case, the thinner the layer, the poorer the function's performance. In the latter case, the damaged zone is more extensive if the repository is located at a greater depth. There are therefore minimum thickness and maximum depth limits for the host formation in the transposition zone. Note that these two choices correspond to different positions of the repository within the zone : in the vicinity of the Meuse/Haute-Marne laboratory in the first case but further west in the second. The performance calculation does not make allowance for a particular location of the repository ; instead, it is doubly pessimistic in terms of safety function performance.

Such an approach ensures that the assessments are bounding with respect to the variability of the repository's possible positions within the transposition zone. If, during a subsequent step of the studies, it were decided to focus on a particular site in the zone, specific investigations into the proposed location would be necessary, in order to refine our knowledge of the local characteristics.

5.3.1.2 Repository location

In view of the choices made in the models in the Callovo-Oxfordian layer, specifying the repository's location within the transposition zone does not affect the calculation of transport through the structures and rock.

Although at this stage, pinpointing the repository's location is an entirely arbitrary exercise, it is essential in order to calculate how elements are transported through the surrounding formations to any potential outlets. For reference, a « default » location of the Meuse/Haute-Marne laboratory has been adopted by convention.

In view of the distances between repository zones (at least 250 m), the zones are thermally, hydraulically, chemically and mechanically independent, and are therefore treated separately. To eliminate the need for assumptions regarding the relative positions of the various zones, they are assumed to all be superimposed over each other in a single location.

When the calculations are performed, each repository zone or sub-zone is assumed to be centred on the « generic » location occupied by the research laboratory. This representation technique is illustrated in Figure 5.3-2. It is applied to all repository zones and sub-zones, considering a group of reference packages (see Section 5.3.2.1).

It is important to consider the impact of these choices on the calculation result. Inasmuch as it minimises the repository's extent, « superimposing » the repository zones concentrates the flows toward fewer outlets than if the different zones had been represented more extensively ; in this case, the current lines could take more contrasting courses. Furthermore, the migration of radionuclides from the B waste zone, for example, is not influenced by the presence of a superimposed mix of radionuclides from spent fuel zones, which could hamper their transport due to the limited solubility of the geological medium.

In defining the outlets (see Section 5.3.2.5), every effort was made to make the most pessimistic choices for a particular repository location. Grouping together all the repository zones in that location enhances the impact on the closest outlet.

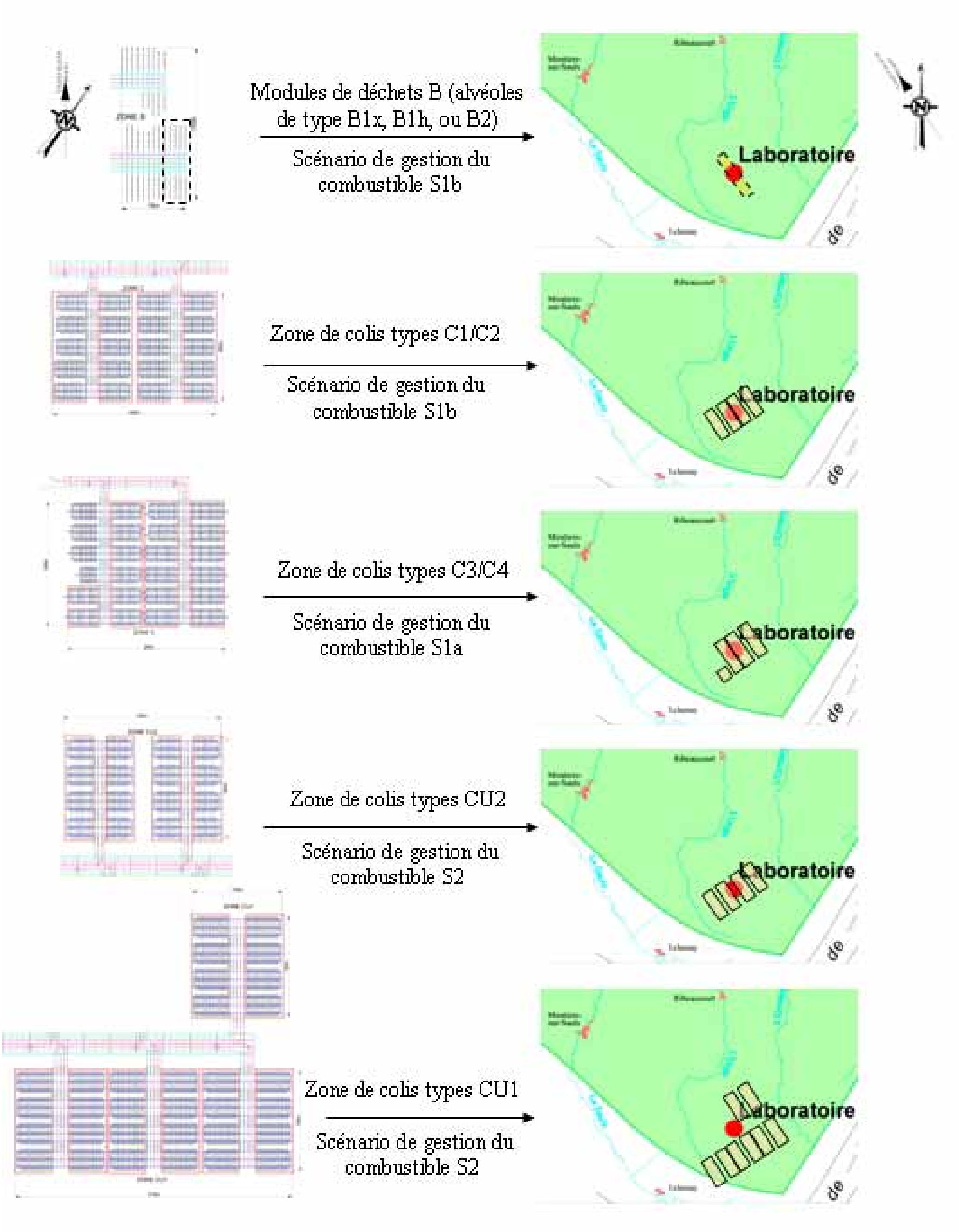


Figure 5.3-2 Position of the repository zones and spent fuel reference packages – example of B1x, B1h, B2 and C1/C2 packages for scenario S1b, C3/C4 reference packages for scenario S1a and CU1 and CU2 reference packages for scenario S2 reference package

5.3.1.3 Initial state of the repository and inclusion of transients

Variability relating to the conditions in which the repository is operated and closed can potentially lead to uncertainty regarding the repository's post-closure status. For example, the desaturation or oxidation of the argillite may vary according to the operating life of the cells, and in particular the ventilated cells. It has been decided to adopt an initial state that reflects a planned repository operating lifetime of from one to several centuries. The influence of the operating period on the calculation results is studied in greater detail in chapter 6, section 6.2.13.

It should also be noted that the thermal, hydraulic, mechanical and chemical transients occurring in the repository and during the post-closure phases are important factors that affect the evolution of the various components as well as the transfer of radionuclides and toxic chemicals. It is therefore important to specify the choices made in respect of this issue.

As a general rule, it will be remembered that the parameters which change in relation to time are represented by constant values (phenomenological, conservative or penalising, as applicable) derived from scientific studies or preparatory calculations, with the exception of a few particular cases (essentially temperature-dependent parameters – see below – and certain parameters of the waste release models).

- **Thermal transient**

The thermal transient is represented explicitly on the basis of prior calculations of the repository's thermal evolution. The temperature-dependency of the Callovo-Oxfordian clay's characteristics is taken into account for the most influential parameters (i.e. the diffusion coefficient and the partition coefficients that vary significantly with temperature, i.e. those of beryllium, calcium, and a caesium isotope (see section 5.3.2.3).

- **Hydraulic transient**

In order to determine whether the hydraulic transients must be included in the calculation model, it is import to describe its main effects first.

The host formation's initial hydraulic state is disturbed when the repository is excavated, leading to a pressure drop in the water in the pores of the argillite in the vicinity of the structures. This phenomenon is known as « head loss ». In ventilated areas (standard structures and B waste disposal cells), renewing the air causes desaturation affecting the concrete structures (rock support and lining) as well as the rock at the vicinity of the lining. Assuming a one-century operating period, the extent of this desaturation remains within the excavation damaged zone. Even if the repository operating period were longer, for example in a reversibility context, the undisturbed rock would not be significantly affected, owing to its low permeability [35]. The unventilated structures (vitrified waste and spent fuel cells) are largely unaffected by desaturation.

Under nominal conditions, the structures resaturate during the post-closure phase. The duration of this phenomenon depends not only on the preceding operating conditions, but also on the intrinsic physical characteristics of the components (i.e. permeability) and the fraction of residual voids in the structures.

The arrival of water triggers corrosion of the metal parts, in an environment that rapidly develops into a reducing medium. This anoxic corrosion produces hydrogen. In addition to this corrosion, other phenomena, such as radiolysis of water and organic materials and gas production by micro-organisms, contribute to a lesser extent to a build-up of gas in the repository. This gas is added to the air trapped in the repository after it is closed ; the oxygen, however, is quickly consumed by the return to reducing conditions.

Because corrosion occurs even at low saturation rates (of approximately 30 %) hydrogen production begins at a very early stage. Initially, the generated gases can escape into the geological medium by dilution/diffusion. In any event (B and C waste or spent fuel disposal cells), a gaseous phase ultimately becomes individualised. Gas can penetrate the various surrounding media provided it can overcome the capillary forces in the porosity (at which point it is said to have reached the « inhibition pressure » for a particular pore size in the geological medium in question). It occupies the porosity by driving out the free water, allowing it to migrate in a slow biphasic flow process. Above a certain pressure, which varies according to the geological medium, gas can also interact with the porosity, causing dilation and gaining easier passage. This process creates preferential pathways that can be looked upon as a form of micro-cracking. Experiments conducted by Andra and equivalent agencies abroad (notably Nagra), have shown that this process is reversible and does not over the long term compromise the porous medium's water permeability. Theoretically, at high pressures, the gas could irreversibly fracture the media through which it passes, but the evaluations conducted by Andra revealed that the required pressures (approximately 12 MPa in argillite) are never reached [61].

The duration of the gas pressure increase depends on the corrosion rate and the quantity of metal present in the repository (which in turn depends on the quantity of waste actually emplaced). Andra has conducted pessimistic assessments assuming maximum corrosion rates that do not take the degree of saturation into account (i.e. approximately 2 to 3 microns per year, although rates of less than one micron per year in an anoxic environment are expected). These assessments reveal that the pressure increases in the C waste and spent fuel cells for a period of 2,000 to 3,000 years, with the pressures respectively reaching 6 to 7 MPa in the C waste cells and approximately 9 MPa in the spent fuel cells. In the B waste cells, the pressure rises for a significantly shorter period (500 years) and peaks at around 7 MPa. The pressure subsequently decreases steadily as the gas escapes, until it has totally disappeared after 100,000 to 200,000 years.

The gas build-up can desaturate media containing free water that can be displaced by the gas – i.e. sufficiently porous media. The pressure increase partially desaturates the damaged zone in the argillite, together with the concrete and the empty spaces in the B waste cells. The gases have a negligible effect (at most a few percent) on the degree of saturation of the bentonite structures and the Callovo-Oxfordian clay. The cell plugs are therefore uninfluenced, and resaturate over a period of around one hundred years, with the drift and access shaft seals resaturating within a few thousand years.

The penetration of gas into the host formation only mobilises a fraction of the formation's porosity (3 % in the near field and less than 1 % in the far field). Nevertheless, that is enough to create significant additional hydraulic pressure in the very near field. This is a localised phenomenon, and the pressure rapidly subsides at distances of more than a few metres, although there is a residual influence to a distance of approximately forty metres. In addition, the phenomenon is transient, ceasing within the first 10,000 years of the post-closure phase (at the latest, at the perimeter of the spent fuel cells, and earlier in the case of the other cells). During this phase, divergent flows can develop in the rock, but the assessments show that the transfer regime remains diffusive in the host formation.

Over this period, the gas spreads rapidly through the backfill but its progress is impaired by the seals in the access structures. The pressure is lower at greater distances from the cells, and the drift corrosion source term makes only a minor contribution to the total volume of gas produced. There is sufficient expansion volume to ensure that the different zones remain independent.

During and after this phase, and up to the hundred thousand year term, the pressure in the B waste cells is high enough to considerably delay the resaturation process. Although cemented waste is saturated from the outset and retains a significant moisture content (80 %), other waste with a highly macroporous structure (such as B5) cannot become water-saturated. The clearances between packages are dry and fill with water only very slowly, by gravity flow. B waste disposal cells do not form a medium with a continuous water phase for several millennia, and they do not become totally resaturated until the hundred thousand year term.

In the vitrified waste and spent fuel cells, the pressure increase phase corresponds to corrosion of the metal sleeve, which is the dominant production term. The containers are spared by the corrosion during this phase. This « corrosion delay » is not included in the conservative container life assessments adopted for the safety calculation (4,000 and 10,000 years, respectively, for the C overpacks and the spent fuel containers). These assessments assume that corrosion commences immediately in a cell that remains totally saturated.

As of the phase during which the pressure subsides and gas gradually escapes, the flows converge toward the repository, helping it reach a totally resaturated state after 100,000 to 200,000 years. As the seals are saturated at that stage and suffered only minor disturbance caused by the gases, they instate a diffusive regime in the repository, and flow rates are very slow.

In summary, with the unfavourable assumption of rapid corrosion rates, the hydraulic transient can create relatively high pressures in the Callovo-Oxfordian layer during the first millennia. This phase is concomitant with desaturation of the B waste packages' environment and an absence of (or limited) corrosion of the C waste and spent fuel containers. As such, there is no related release of radionuclides. As the aim of the normal evolution scenario is to represent radionuclide transfers, the gas pressure phase occurring during the first ten thousand years is not represented in the scenario.

Considerations focus directly on a later situation, in which the seals are saturated. The flow regime is diffusive, and assuming the repository to be totally resaturated is a reasonable approximation with respect to radionuclide migration, and a pessimistic assumption with regard to the evolution of the various components (and in particular metal waste and containers) as it accelerates their corrosion.

To eliminate the uncertainties relating to the length of the hydraulic transient due to gas pressure (which depends on many factors), it is specified as zero by convention. Consequently, radioactive decay is not taken into consideration during this phase.

It emerges from the foregoing discussion that this assumption is :

- Pessimistic with regard to the pressure increase phase (early millennia), during which the inventory decays without any release of radionuclides ;
- Slightly conservative, but without any significant influence, on the subsequent resaturation phase, as the radionuclides migrate essentially by diffusion despite the presence of flows, as these are both broadly convergent and very slow.

Chapter 6, section 6.2.5, contains a more detailed discussion of the uncertainties liable to affect this reference scenario, as well as their influence on the calculation. Note that to cover these uncertainties, the SEN sensitivity studies include a study that represents the gas pressure transient and generates hydraulic loads in the model. This transient is assumed to be concomitant with release from the B waste packages and any failed spent fuel containers (this case covers an equivalent situation affecting the vitrified waste containers). This sensitivity study considers the simultaneous occurrence of physical phenomena that in reality are mutually exclusive.

- **Mechanical transient**

Mechanical transients are not explicitly represented but are indirectly taken into consideration in the calculations. The overall rationale, and notwithstanding the need for a more detailed component-by-component description, is as follows.

Throughout the calculation, the EDZ is represented in the state of equilibrium attained following the mechanical load relief caused by the excavation work. In subsequent phases it may be subject to actions caused by the effect of gases (if the pressure is high enough to cause the porosity to temporarily expand) or by the heat released by the packages, but these actions are reversible and correspond to phases with no radionuclide release [62, 63]. The EDZ is therefore represented in the state of equilibrium achieved once these transients have passed, and the subsequent healing phenomena are ignored.

The loss of mechanical strength in engineered components is taken into account at a date corresponding to a conservative assessment. After that date, the safety functions associated with mechanical strength are degraded : this is the case of spent fuel containers and vitrified waste overpacks, for example, whose mechanical strength cannot be guaranteed once a certain degree of corrosion has occurred.

- **Chemical processes**

Chemical phenomena can occur throughout the one million-year calculation timescale (see chapter 3). The corresponding interactions are included on a case-by-case basis when the models and parameters are chosen, depending on both their estimated extent, and their potential effects on safety functions.

- **Conclusion relating to the inclusion of transients**

The foregoing discussion shows that the SEN explicitly takes the thermal transient into account. The hydraulic transient, which would be difficult to represent in a global simulation, has been disregarded after confirming that such an assumption is indeed conservative. Nevertheless, sensitivity studies have been defined to investigate the potential for radionuclide transfers during the transient. These studies are described in the dedicated section. The mechanical and chemical transients are treated individually, without coupling them to the safety calculation.

5.3.1.4 Transport vectors

Two transport vectors can facilitate the dissemination of radionuclides and toxic chemicals contained in the waste : water and gas.

The normal evolution scenario only assumes water-borne transport, since the transport of elements in gaseous form appears negligible.

The only radionuclides liable to be present in gaseous form in the repository are radon 222, krypton 85, iodine 129, tritium, carbon 14, chlorine 36 and argon 39.

Radioactive rare gases (^{85}Kr, ^{39}Ar, ^{220}Rn) do not pose any problems in a normal evolution, either because they have short half-lives compared with the timescales relating to the leaktightness and transfer of containers in the repository (as is the case with ^{85}Kr, ^{39}Ar), or because they are descendants of elements that are strongly sorbed in the geological medium and do not contribute to the impact in the SEN (e.g. ^{222}Rn).

In the repository's environmental conditions, halogens (chlorine and iodine) solubilise readily and would not be able to migrate over significant distances in gaseous form.

On the other hand, in certain conditions, carbon could be found and remain in gaseous form (as methane). 98 % of the carbon 14 in the inventory is from fuel cladding (being present in zircon and zircaloy). Depending on the waste management scenario, the main source is either cladding hull and endcap packages (B4 and B5 packages) or spent fuel packages.

Carbon 14 can only enter gaseous form in the presence of micro-organisms. Compressed waste packages and spent fuel disposal packages are not a fertile environment for micro-organisms, owing to the lack of nutriments. Only B5.1 packages, which contain some organic waste, are liable to release radioactive methane. This type of waste accounts for between 6 % and 20 % of the total carbon 14 inventory, i.e. 20 to 75 moles, or an equivalent volume of 0.5 to 1.7 m^3 in normal temperature and pressure conditions, which is negligible compared with the quantities of corrosion gases present during the first 100,000 years. This release could not begin until the cladding has corroded, i.e. after the pressure increase phase (lasting several thousand years). It would then be congruent with the waste corrosion (which is likely to be a rapid process with zircon, but take approximately 100,000 years with zircaloy). During this period, the carbon 14, which has a half-life of 5,370 years will have decayed considerably. It is not therefore considered to be a major contributor to the radiological impact[61].

5.3.1.5 Radionuclides and toxic chemicals taken into account for the dose assessment

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 [18], those that contribute most to the radiological impact in the SEN. The approach and results are described below.

A preliminary calculation, the results of which are given in section 5.5.4.3, revealed that the high chemical retention of actinides in the Callovo-Oxfordian argillites delays the onset of the peak molar flow exiting the host formation until well after the study's million-year end-point. Over this period, the quantity of activity exiting the Callovo-Oxfordian layer is negligible, and any such activity remains

[61] Even in the most unfavourable cases (in the hypothesis of an early release of carbon, or if a greater-than-predicted inventory is mobilised), as a last line of defence, the methane would decay totally during the period required for the transfer to the surrounding formations. The transfer time is specified in the SEN as being at least 50,000 years.

almost entirely contained in the near field. Consequently, no actinides were selected for the SEN's global impact calculation [64].

For fission and activation products, a two-step selection process was adopted.

The first phase involved eliminating any radionuclides with a half-life of less than 1,000 years, which decay in the host formation. Based on the observation that the theoretical transfer times are in the region of a few hundred thousand years, or at least 100 times the half-life of such radionuclides, the radionuclides' initial activity decays by around 30 orders of magnitude at least, which would seem enough to render them negligible.

The second step involved considering the radionuclides with half-lives longer than 1,000 years and eliminating those that could not contribute to the impact. To this end, the remaining elements were organised into groups according to their class of geochemical behaviour, in order to group together the radionuclides that are expected to migrate together. Within each group, any elements with minimal activity were disregarded (for example, ^{99}Tc was selected rather than ^{98}Tc or ^{97}Tc, which are present in smaller quantities). A comprehensive calculation concerning all the radionuclides with half-lives longer than 1,000 years was performed in order to validate this approach [64] and confirm that it does not lead to the elimination of any important radionuclides. Many radionuclides are strongly sorbed in argillite, which further delays and reduces their contribution. This is particularly true of ^{137}La, ^{146}Sm, ^{150}Gd, ^{154}Dy, ^{176}Lu, ^{182}Hf, ^{186m}Re, ^{208}Bi and ^{210m}Bi.

After considering all the different types of waste package, this approach ultimately yielded the following selection of 15 radionuclides with half-lives in excess of 1,000 years for use in the SEN impact calculation, together with niobium 93m (a descendant of zirconium 93). They are listed in Table 5.3-1.These periods are consistent with those given by the database JEFF 2.2 [65] :

Isotopes	From anionic/cationic form	Radioactive half-life [years]
^{129}I	anionic form	15,700,000 years
^{107}Pd	cationic form	6,500,000 years
^{135}Cs	cationic form	2,300,000 years
^{10}Be	cationic form	1,600,000 years
^{93}Zr ($\rightarrow$ ^{93}Nb)	cationic form	1,530,000 years (TNb93m ≅ 16.4 years)
^{36}Cl	anionic form	302,000 years
^{99}Tc	cationic form	213,000 years
^{41}Ca	cationic form	103,000 years
^{126}Sn	cationic form	100,000 years
^{59}Ni	cationic form	75,000 years
^{79}Se	anionic form	65,000 years
^{94}Nb	anionic form	20,300 years
^{14}C	anionic form	5,730 years
^{93}Mo	anionic form	3,500 years
^{166}Ho	cationic form	1,200 years

Table 5.3-1 Radionuclides considered in a normal evolution scenario

Following the May 2005 update of the JEFF nuclear database, used as a reference by Andra for radioactive half-lives, the Andra assessed the consequences of such an evolution on radiological impact calculations (see inset 9).

The impact calculation was also performed for a few stable toxic chemicals (boron, selenium, nickel and antimony), in the same conditions as for the radionuclides. With these toxic chemicals, there is generally no radioactive decay, unless it is also a radionuclide or a descendant of a radionuclide. These toxic chemicals were selected from the list of elements that must generally be declared upon entry to disposal centres (class 1 technical burial facilities), plus a number of toxic chemicals more specific to the nuclear industry. The chemicals selected are those liable to be present in relatively large quantities in HL and MLLL waste.

5.3.2 Representation of the different calculation compartments

5.3.2.1 Representation of waste packages and initial inventories

Andra has developed four fuel management scenarios for the design inventory model (see chapter 2) in order to cover a range of possible industrial strategies. In the SEN, Andra applied the principle of selecting the most pessimistic, or one of the most pessimistic scenario as pertaining to quantitative inventory for each reference package. This working assumption meant that the approach was based on the S2 quantitative inventory for spent fuel, (which is a conservative hypothesis for CU1 fuels and results in a large amount of CU2[62] fuels), S1a for the C3+C4 reference packages and S1b for the other reference packages.

The reference calculation was performed by grouping together waste packages to keep the number of calculation cases manageable. It was performed for the following types of package :

- B waste containing no organic matter and not releasing gaseous hydrogen during the repository's operating phase. It was assumed for the purposes of the calculation that such waste could be disposed together. The corresponding group of cells are referred to as sub-zone B1x in the rest of this document ;
- B waste containing no organic matter but liable to release hydrogen during the operating phase. The corresponding group of cells are hereafter referred to as sub-zone B1h ;
- Bituminised sludge waste packages, which will probably be disposed of separately on account of their specific characteristics ;
- Other B waste (excluding B7 and B8) ;
- C0 glass ;
- C1 and C2 glasses, which are grouped together ;
- C3 and C4 glasses, which are grouped together ;
- CU1 spent fuel ;
- CU2 spent fuel ;
- CU3 spent fuel ;

In view of their very minor total contribution to the radiological inventory, B7 waste packages (sources) have been ignored in the calculation. Given its characteristics (containing radium, an element strongly sorbed in the natural environment), B8 waste is not expected to contribute to the impact, they are also neglected.

The package groups and related quantitative inventories are listed in Table 5.3-2. A mean radiological inventory based on the MID has been adopted for each group, except vitrified waste, for which the distinction is drawn between C2 and C1, and between C4 and C3 waste.

The entire inventory is not always considered in the sensitivity studies. The waste types most suited to requirements are selected on a case-by-case basis according to the lessons drawn from the reference calculation results.

62 It should be noted that scenario S1b produces slightly more CU2 (5 400 compared to 4 000) but it does not produce CU1 fuel.

Repository sub-zone containing reference packages or cell types covered in the calculations*		Management scenario	Number of packages
B1x*	Cell type containing non-organic waste packages that do not release gaseous hydrogen (B1/B5/B6 reference packages excluding B5.1 and B6.4)	S1b	44,670
B1h*	Cell type containing non-organic waste packages liable to release gaseous hydrogen (B3.1.1/B3.1.3/B3.2/B3.3.2/B4 reference packages)	S1b	16,770
B2BB*	Cell type containing organic bituminised sludge packages (B2 reference packages)	S1b	105,010
Other B waste	Cell type containing organic waste packages other than bituminised sludge (B3.1.2 / B3.3.1 / B3.3.3 / B3.3.4 / B5.1 / B6.4 reference packages)	S1b	26,290
C0	« Legacy » vitrified waste	S1b	4,120
C1 and C2	« Current thermal » or « future thermal » vitrified waste	S1b	(4,640 C1 and 27,460 C2) 32,100
C3 and C4	Vitrified waste produced by a mixture of dissolved fission products from UOx2 and MOX fuel with glass containing a small quantity of Pu	S1a	(13,320 C3 and 13,250 C4) 26,570
CU1	UOx spent fuel	S2	13,500
CU2	MOX spent fuel	S2	4,000
CU3	Research and Defence fuel	/	5,810
*Cell type containing package sub-types			

Table 5.3-2 Repository sub-zones used in the calculations, and total numbers of waste packages per sub-zone

Table 5.3-3, Table 5.3-4 and Table 5.3-5 show total initial inventories [63], for each major type of reference package as considered in the SEN (see Table 5.3-2) and for the 15 radionuclides selected. These tables have been built from the detailed inventories provided in the reference knowledge documents and the inventory model for high-level and long-lived waste [18]. Radiological inventories in the model are given for each type of reference package and for each of the physical-chemical subassemblies (SEPC) of the waste.

[63] The initial radiological inventories are defined with respect to the following reference time : 3 years after removal from the reactor for CU1 and CU2 spent fuels ; on the package production date (C1, C2, C3, C4, B5.1, B5.2) assumed to be within 8 years after the spent fuel was unloaded from the reactors ; in 2005 for reference packages C0 , B2, B3 and B4, B5.3, B5.4, B6.

Radio-nuclides	Number of reference packages / Half-life [years]	Total initial activity Zone « B1x » (Scenario S1b)	Total initial activity Zone « B1h » (Scenario S1b)	Total initial activity Reference package B2 (Scenario S1b)	Total initial activity Other B waste (Scenario S1b)
	Number of reference packages	*44,670*	*16,770*	*105,010*	*26,290*
	Half-life [years]	[Bq]	[Bq]	[Bq]	[Bq]
^{10}Be	1,600,000	$1.6x10^{9}$	$5.8x10^{8}$	nil	$1.7x10^{9}$
^{14}C	5730	$6.7x10^{14}$	$2.7x10^{13}$	$1.6x10^{10}$	$2.3x10^{14}$
^{36}Cl	302,000	$2.1x10^{13}$	$2.1x10^{12}$	$7.1x10^{11}$	$5.4x10^{12}$
^{41}Ca	103,000	$5.5x10^{12}$	$5.0x10^{9}$	$7.7x10^{10}$	$1.3x10^{10}$
^{59}Ni	75,000	$1.8x10^{16}$	$5.4x10^{14}$	$4.8x10^{12}$	$4.0x10^{15}$
^{79}Se	65,000	$2.1x10^{12}$	$7.6x10^{10}$	$2.0x10^{12}$	$4.5x10^{11}$
^{93}Zr	1,530,000	$2.6x10^{14}$	$1.6x10^{13}$	$8.8x10^{11}$	$6.1x10^{13}$
^{93m}Nb	16.4	$9.6x10^{13}$	$4.5x10^{13}$	$6.1x10^{11}$	$2.4x10^{13}$
^{94}Nb	20,300	$3.8x10^{15}$	$1.0x10^{14}$	$4.3x10^{9}$	$9.4x10^{14}$
^{93}Mo	3500	$9.3x10^{14}$	$9.8x10^{13}$	$1.3x10^{11}$	$2.0x10^{14}$
^{99}Tc	213,000	$2.4x10^{14}$	$2.0x10^{13}$	$2.2x10^{15}$	$3.0x10^{15}$
^{107}Pd	6,500,000	$9.5x10^{11}$	$5.1x10^{10}$	$1.9x10^{13}$	$2.4x10^{11}$
^{126}Sn	100,000	$5.7x10^{12}$	$2.8x10^{11}$	$4.0x10^{12}$	$1.3x10^{12}$
^{129}I	15,700,000	$1.9x10^{11}$	$8.6x10^{9}$	$5.8x10^{10}$	$4.8x10^{10}$
^{135}Cs	2,300,000	$4.2x10^{12}$	$8.8x10^{10}$	$1.4x10^{12}$	$8.8x10^{11}$
^{166m}Ho	1200	$7.0.10^{10}$	$8.5.10^{8}$	$1.5x10^{9}$	$1.9x10^{10}$

Table 5.3-3 Total initial activity in the various B waste zones considered

Radionuclides	Number of packages / Half-life [years]	Total activity Reference package C0 [Scenario S1b]	Total activity Reference packages C1/C2 [Scenario S1b]	Total activity Reference packages C3/C4 [Scenario S1a]
	Number of packages	4120	32,100	26,570
	Half-life [years]	[Bq]	[Bq]	[Bq]
^{10}Be	1,600,000	$0.0x10^{0}$	$4.6x10^{11}$	$4.1x10^{11}$
^{14}C	5,730	$4.3x10^{11}$	$1.2x10^{14}$	$1.1x10^{14}$
^{36}Cl	302,000	$3.2x10^{7}$	$2.4x10^{13}$	$1.9x10^{13}$
^{41}Ca	103,000	$2.0x10^{11}$	$2.5x10^{12}$	$2.0x10^{12}$
^{59}Ni	75,000	$1.0x10^{12}$	$2.6x10^{14}$	$2.3x10^{14}$
^{79}Se	65,000	$1.9x10^{13}$	$7.1x10^{14}$	$6.2x10^{14}$
^{93}Zr	1,530,000	$1.1x10^{14}$	$3.8x10^{15}$	$3.3x10^{15}$
^{93m}Nb	16.4	$5.4x10^{13}$	$1.3x10^{15}$	$1.1x10^{15}$
^{94}Nb	20,300	$1.2x10^{10}$	$5.5x10^{13}$	$5.0x10^{13}$
^{93}Mo	3500	$2.9x10^{13}$	$5.8x10^{12}$	$5.1x10^{12}$
^{99}Tc	213,000	$8.2x10^{14}$	$2.9x10^{16}$	$2.5x10^{16}$
^{107}Pd	6,500,000	$3.7x10^{12}$	$2.7x10^{14}$	$2.6x10^{14}$
^{126}Sn	100,000	$3.1x10^{13}$	$1.9x10^{15}$	$1.8x10^{15}$
^{129}I	15,700,000	$7.9x10^{9}$	$7.0x10^{11}$	$6.4x10^{11}$
^{135}Cs	2,300,000	$3.7x10^{13}$	$1.0x10^{15}$	$9.4x10^{14}$
^{166m}Ho	1200	$7.1x10^{9}$	$2.8x10^{13}$	$2.4x10^{13}$

Table 5.3-4 Total initial activity of C waste reference packages

Radionuclides	Number of packages / Half-life [years]	Total initial activity Reference package CU1 [Scenario S2]	Total initial activity Reference package CU2 [Scenario S2]	Total initial activity Reference package CU3 [No scenario]
	Number of packages	*13,500*	*4000*	*5810*
	Half-life [years]	[Bq]	[Bq]	[Bq]
^{10}Be	1,600,000	$3.2x10^{11}$	$2.0x10^{10}$	$2.6x10^{8}$
^{14}C	5,730	$1.1x10^{15}$	$5.0x10^{13}$	$1.8x10^{12}$
^{36}Cl	302,000	$3,0x10^{13}$	$6.3x10^{11}$	$2.5x10^{11}$
^{41}Ca	103,000	$2.1x10^{12}$	$1.7x10^{10}$	$1,1x10^{11}$
^{59}Ni	75,000	$2.2x10^{15}$	$1.5x10^{14}$	$6.6x10^{11}$
^{79}Se	65,000	$4,9x10^{14}$	$2.6x10^{13}$	$1.1x10^{12}$
^{93}Zr	1,530,000	$2.8x10^{15}$	$1.3x10^{14}$	$1.1x10^{13}$
^{93m}Nb	16.4	$6.5x10^{14}$	$2.4x10^{13}$	$5.2x10^{12}$
^{94}Nb	20,300	$4.7x10^{15}$	$1.7x10^{13}$	$7.2x10^{10}$
^{93}Mo	3500	$1.7x10^{13}$	$8.2x10^{11}$	$1.5x10^{11}$
^{99}Tc	213,000	$2.0x10^{16}$	$1.2x10^{15}$	$7.7x10^{13}$
^{107}Pd	6,500,000	$1.8x10^{14}$	$2.5x10^{13}$	$3.8x10^{11}$
^{126}Sn	100,000	$1.3x10^{15}$	$1.5x10^{14}$	$2.9x10^{12}$
^{129}I	15,700,000	$4.9x10^{13}$	$4.0x10^{12}$	$1.6x10^{11}$
^{135}Cs	2,300,000	$6.9x10^{14}$	$9.5x10^{13}$	$2.1x10^{12}$
^{166m}Ho	1200	$1.8x10^{13}$	$2.1x10^{12}$	$4.2x10^{9}$

Table 5.3-5 Total initial activity of spent fuels

- **B waste packages**

Release model for B primary waste packages

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 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.

Miscellaneous cemented or compacted technological waste (B3) and bulk technological and structural waste (B6) contain a source term that is immediately available upon contact with water (described as labile). This choice is primarily the result of a lack of knowledge that forces us to consider a pessimistic release model (as in the case of B6 packages), although it may also relate to the fact that contamination seems to be essentially a surface phenomenon (as with B3 reference packages).

The release models and related parameters for the other packages are chosen from the range of available models on the basis of :

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

For activation product waste (B1), cemented structural waste (B4) and compacted waste (B5), only the zircaloy cladding sections (B4/B5 only) 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[64] 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. A sensitivity study is conducted assuming a labile release of the activity contained in the cladding sections and metallic components. This sensitivity study comfortably covers all the uncertainties relating to the risks of localised corrosion by pitting, the variable characteristics of the different waste products or the possible presence of oxidising elements produced by water radiolysis. Table 5.3-6 and Table 5.3-7 detail the release rate values selected for B1, B4 and B5 reference packages.

	Reference calculation	Sensitivity calculation
Activation products present in nickel alloys	$\tau_R = 5.10^{-6}$ yr^{-1}	Totally labile activity
Activation products present in stainless steels	$\tau_R = 4.5.10^{-5}$ yr^{-1}	
Other locations	Labile activity	

Table 5.3-6 Release model adopted for B1 reference packages

	Reference calculation	Sensitivity calculation
Activation products present in zirconium alloy	$\tau_R = 10^{-5}$ yr^{-1}	Totally labile activity
Activation products present in inconel alloys	$\tau_R = 7.10^{-5}$ yr^{-1}	
Activation products present in stainless steels	$\tau_R = 1,5.10^{-5}$ yr^{-1}	
Other locations	Labile activity	

Table 5.3-7 Release model adopted for B4 and B5 reference packages

For bituminised sludge (B2), the release kinetics are represented by a model (Colonbo) developed on the basis of pheonomenology that has been experimentally validated to the extent permitted by laboratory reproduction capabilities[66]. 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 insolubilisation 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 [66]. 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.

64 Conservative estimate that notably allows the inclusion of any irradiation damage and the influence of water radiolysis on the corrosion process.

A sensitivity study has been conducted with a pessimistic model that covers the uncertainties relating to the packages' evolution (creep, cracking, etc.). This model does not take into account the water uptake by the bitumen ; the radionuclides are assumed to be associated with the soluble salts (pessimistic assumptions) and released as soon as they dissolve. This model is therefore dependent on the solubility of the aforementioned salts and the flows of water passing through the cells. The model predicts the gradual release (at a constant rate over time) of the entire mass within 1,000 years.

Table 5.3-8 details the release rates adopted for B2 reference packages.

	Reference calculation Colonbo 3 model	Sensitivity calculation
All radionuclides	$\tau_R = \Lambda \frac{4{,}5.10^{-3}}{\sqrt{t}} \Lambda \; an^{-1}$	$\tau_R = 10^{-3}$ yr^{-1}

Table 5.3-8 Release model adopted for bituminised sludge (B2) packages

Representation of disposal package concrete cladding functions

The package concrete degradation is an extremely slow process [67] that allows favourable geochemical conditions to be created for primary B waste packages (in particular in terms of their pH), in accordance with the function requirement, thereby limiting the degradation of the packages. No safety function requiring long-term mechanical strength has been selected for B waste overpacks. Consequently, the concrete used in the design of the overpack materials has no hydraulic role.

Disposal package overpacks are therefore represented for all B waste packages as a uniform chemical environment (undistinguished from that of the concrete cell) that limits radionuclide flows by precipitation and sorption. The values of the hydraulic, transport and chemical retention parameters are presented in the table below. The retardation coefficients (R) have been assessed on the basis of the distribution coefficient values (Kd) corresponding to the weaker performance of the first two states of the concrete (sound and degraded). The reference values are finally provided in the note [60].

	K = 1.10^{-6} m/s				
	Period [years]	$\omega_{Diffusion}$ [-]	De [m²/s]	R [-]	Csat [mol/m^3]
^{10}Be	1 600 000	0,3	6.10^{-10}	1	soluble
^{14}C	5 730	0,3	6.10^{-10}	3 500	1.10^{-2}
^{36}Cl	302 000	0,3	6.10^{-10}	1	soluble
^{41}Ca	103 000	0,3	6.10^{-10}	3 500	20
^{59}Ni	75 000	0,3	6.10^{-10}	14 000	$2,3.10^{-4}$
^{79}Se	65 000	0,3	6.10^{-10}	700	$1,3.10^{-2}$
^{93}Zr	1 530 000	0,3	6.10^{-10}	280 000	6.10^{-3}
^{93}Mo	3 500	0,3	6.10^{-10}	1	7.10^{-4}
^{93m}Nb	16,4	0,3	6.10^{-10}	70 000	$2,4.10^{-4}$
^{94}Nb	20 300	0,3	6.10^{-10}	70 000	$2,4.10^{-4}$
^{99}Tc	213 000	0,3	6.10^{-10}	1	Soluble
^{107}Pd	6 500 000	0,3	6.10^{-10}	4 200	1.10^{-2}
^{126}Sn	100 000	0,3	6.10^{-10}	70 000	3.10^{-5}
^{129}I	15 700 000	0,3	6.10^{-10}	8	soluble
^{166m}Ho	1 200	0,3	6.10^{-10}	700 000	2.10^{-3}
^{135}Cs	2 300 000	0,3	6.10^{-10}	71	soluble

Table 5.3-9 Parameter values for transport and chemical retention in concrete in B waste packages – reference calculation

The utility of attributing hydraulic properties to certain overpack materials over a period of 10,000 years is under consideration Such an initiative could apply to B waste packages that have a greater radiological content and release little hydrogen (B.5.2 and B1 waste in particular). This variant is being studied from an exploratory perspective as there are still uncertainties relating in particular to long-term mechanical strength. It is considered in a sensitivity study for the SEN. In this case, the overpack used with packages is represented as a uniform porous medium ; in addition to the geochemical performance already adopted for reference, the « delay and attenuate radionuclide migration » function is also provided, as transport by diffusion/advection occurs with a diffusion coefficient of 2.10^{-13} m²/s, a diffusion-accessible porosity of 10 % and a permeability of 10^{-13} m/s.

- **Vitrified C waste packages**

The overpack delays the arrival of water onto the glass by 4,000 years

The overpacks are designed to last for about a millenia, with an adequate margin to cover the chemical, mechanical and radiological uncertainties (see chapter 6). Assessments in conservative conditions consistent with the normal evolution scenario (immediate resaturation of the repository), taking into account the chosen design measures (in particular, protecting the container during the operating period by temporarily closing the cell) and assuming conservative corrosion rates, evaluate the overpacks leaktight duration to be an estimated 4,000 years [68]. Accordingly, there is no release from the matrix during that time.

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. The overpack's design is notable for its robustness with regard to possible manufacturing defects (see description of this aspect in chapter 6, section 6.2.12.2). In view of this, and inasmuch as the detailed studies into the manufacture of such objects are at an early stage, any attempt to define the possible flaws will be largely arbitrary.

However, in view of the inspections that it will be possible to introduce for the overpack manufacturing process, and considering their relatively simple design, one can assume that it is unlikely that a defect would become apparent immediately after being placed into the repository, and that only a few overpacks would be affected by random failures. The probability of this type of defect is generally assessed within the industry as being between 1 in 10,000 and 1 in 100,000. The exact number of overpacks in the repository depends on the spent fuel reprocessing management scenario, but there will be slightly over 50,000 packages. There should therefore be no more than a few defective packages.

In the reference calculation for the normal evolution scenario, there are assumed to be three defective overpacks, one in each package group. A defective overpack is conventionally represented by a total loss of leaktightness approximately one century after the repository is closed [65], this minimum period being a highly pessimistic assessment compared with the time that would be required for at least a little water to reach the waste via the primary container**.** As presented with the calculation results, the latter are only very slightly dependent on the exact number of defective overpacks, and are therefore relatively insensitive to the failure rate adopted.

In conclusion, the study assumes the premature release of the following waste, after 200 years of radioactive decay :

- One C0 overpack ;
- One C2 overpack in the repository sub-zone containing the C1/C2 packages ;
- One C4 overpack in the repository sub-zone containing the C3/C4 packages.

Taking into account defective packages implies a release and migration of radionuclides in an environment with a high thermal loading and steep temperature gradient. The effect of thermal considerations on the release and transport parameters and models is considered for all components.

For the reasons already stated, such early release is in principle incompatible with the existence of hydraulic pressure created by gas inside the repository. A sensitivity study is nevertheless conducted to evaluate a case in which the release occurs in the presence of steep hydraulic gradients caused by gases, that persist for 10,000 years. Although this sensitivity study is conducted for spent fuel, its results can be extrapolated for vitrified waste.

More drastic package failure hypotheses are also covered, in an altered evolution scenario (see chapter 7).

Release kinetics for vitrified C waste packages

Radionuclide release begins as soon as the overpack ceases to be leaktight. The release models and related parameters vary between reference packages ; they are chosen from the range of available models based on :

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

Packages may :

- Obey a model based on the glass's initial dissolution rate and exchange area (V_0.S). This conservative model has been adopted (with phenomenological parameters for the surface area in contact with water and the dissolution rate) for C0 glass ; it yields release periods of between a few hundred and a few thousand years, depending on the package ;
- Obey a two-phase phenomenological model (V_0. ➔ V_r). In the first phase, the model is based on the initial dissolution rate, until the surrounding medium becomes saturated with silica (V_0.S). In the second phase, the dissolution kinetic subsides until a residual rate (V_r) is reached [66]. The short-lived transient is not represented. This model yields release periods on the order of a few

[65] For calculation purposes, the situation is assessed in terms of the half-life of radionuclide decay. Taking the period of prior storage of the waste into account, a decay half-life of 200 years was assumed.

hundred thousand years. Furthermore, although the existence of the residual rate R_r is generally accepted, the related phenomenology is a subject of debate. Therefore, out of prudence, a sensitivity study was conducted in which only the V_0.S model was used.

- **Spent fuel packages**

The container delays the arrival of water onto fuel waste by 10,000 years

Assessments performed in the same conditions as for the vitrified waste overpacks (see above) revealed that spent fuel containers have a lifetime in excess of 10,000 years. For the purpose of the calculation, the value obtained for the leaktightness of spent fuel containers was rounded to 10,000 years without release. In the special case of research and defence fuel (CU3), which is assumed to be disposed in the same type of cell as vitrified waste, the container was assumed to remain leaktight for 4,000 years.

17,500 containers were included in the calculation. Adopting a failure rate of between 1 in 10,000 and 1 in 100,000, based on the same logic as for C waste, the normal evolution scenario reference calculation assumes that two defective containers (one CU1 container and one CU2 container) cease to be leaktight after 200 years of radionuclide decay. As explained later, the result of the studies shows that the impact is largely insensitive to the exact failure rate. The « Package defect » SEA (chapter 7) suggests that much more drastic rates would be acceptable.

The thermal conditions prevailing at the time of release are taken into account in the calculation. In sensitivity studies, it is also assumed that the release occurs while the repository is subject to overpressure caused by corrosion gases, and that this situation lasts for 10,000 years. As already described, this overpressure is theoretically incompatible with radionuclide release. Any uncertainties relating to this point are covered here by convention.

Spent fuel release kinetics

Radionuclides begin to be released as soon as the container ceases to be leaktight. At that point, the corrosion products from the container or the rest of the overpack itself cease to provide any containment and the disposal cells are assumed to be totally saturated.

In the case of CU1 and CU2 spent fuels, the release model depends on the location of the radionuclides in the waste's physical and chemical subgroups ; for those radionuclides contained in the matrix, this model also depends on the burnup factor of the fuel type in question (UOx2, UOx3 or URE). There are :

- 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 model representing the radiolytic dissolution of the fuel matrix. The model selected for the calculations pessimistically includes the effect of accelerated diffusion by alpha self-irradiation (D3AI). This model yields a gradual release of the radionuclides in the matrix over a period of approximately 50,000 years ;
- A labile fraction corresponding to :
  - ✓ The radionuclides contained in the pellet, in the grain joints, cracks and voids in the rods in the case of UOx fuel, or in the Pu clusters in MOX fuel. This fraction is essentially dependent on the burnup factor ;
  - ✓ Accelerated diffusion by alpha self-irradiation (D3AI) in the fuel matrix. When considered in conjunction with radiolytic dissolution, this phenomenon accounts for the possible effect of alpha decay. The calculations assume a labile release of 5 % of the activity in the fuel matrix. From a sensitivity perspective, accelerated diffusion by alpha self-irradiation (D3AI) is assumed to concern 15 % o the matrix's activity.
  - ✓ The radionuclides present in zircon.

A sensitivity study has been conducted to cover the residual uncertainties relating to :

- The radiolytic dissolution kinetics, and in particular the specific area available to the grains, the effect of temperature and the burnup factor. The study adopts a conservative burnup factor and multiplies the dissolution rates obtained with that burnup factor by ten. In this case, the radiolytic dissolution-related release periods are a few thousand years long ;
- The percentage of labile activity in the matrix. The conservative labile activity percentages specified in the following tables (Table 5.3-11 to Table 5.3-13) were adopted.

To consider the hypothesis that gaseous hydrogen has an inhibiting effect on radiolytic dissolution, a sensitivity study was conducted using an alternative model, based not on radiolytic dissolution but on the conventional dissolution of the spent fuel. The conventional dissolution model is governed by solubility of uranium (considered to be 7.10^{-7}mol/m^3).

For CU3 spent fuel, there is not currently enough data available to distribute the total inventory among the different physical and chemical subassemblies and specify a release model for each. Consequently, and in view of their relatively limited total inventory compared with other spent fuels, the pessimistic choice was made to treat all the activity as being immediately labile when the container is breached.

The parameter values chosen for each reference packages are detailed in the following tables (Table 5.3-10 to Table 5.3-13).

	Reference calculation
Activation products present in zirconium alloy (80 % of the cladding's activity)	$\tau_R = 5.10^{-5}$ yr^{-1}
Activation products present in zircon (20 % of the cladding's activity)	Labile activity
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 5.3-10 Release model for activation products present in metal components for CU1 and CU2 reference packages

		Reference calculation	Sensitivity calculation
Radionuclides in the gap, rim and grain joints (including accelerated diffusion by alpha self-irradiation) in the matrix		Labile activity	
Percentage of labile activity in the matrix	^{36}Cl, ^{79}Se, ^{126}Sn, $^{135,137}Cs$.	7.1 %	22.8 %*
	^{14}C	15 % (conservative)	25 %*
	Other radionuclides	6.8 %	20.4 %*
Remaining radionuclides contained in the matrix's grains, released in accordance with the radiolytic dissolution model		$\tau_{R\_ref.}$ = f(t ; BF)**	$\tau_{R\_Sensitivity}$ = $10*\tau_{R\_(BFmax)}$ ***
* : With defective packages, the fraction of labile activity correlated with accelerated diffusion by alpha self-irradation is 10 % lower			
** : See formula details in [67]			
*** : A sensitivity study considering a conventional dissolution model, with all else being equal, has also been conducted.			
BF : Burnup factor - 45 GWd/t for reference purposes and 50GWd/t for sensitivity studies			

Table 5.3-11 Release model for the radionuclides in spent fuel matrixes in CU1 (Uox2) reference packages

		Reference calculation	Sensitivity calculation
Radionuclides in the gap, rim and grain joints (including accelerated diffusion by alpha self-irradiation) in the matrix		Labile activity	
Percentage of labile activity in the matrix	^{36}Cl, ^{79}Se, ^{126}Sn, $^{135,137}Cs$.	12.3 %	30.1 %*
	^{14}C	15 % (conservative)	25 %*
	Other radionuclides	9.9 %	25.2%*
Remaining radionuclides contained in the matrix's grains, released in accordance with the radiolytic dissolution model		As for CU1 UOx2 except BF_{UOx3} = 55 GWd/t for ref. and BF_{UOx3} = 60 GWd/t for sensitivity studies	
* : with defective packages, the fraction of labile activity correlated with accelerated diffusion by alpha self-irradation is 10 % lower			

Table 5.3-12 Release model for the radionuclides in spent fuel matrixes in CU1 (Uox3 and URE) reference packages

		Reference calculation	Sensitivity calculation
Radionuclides in the gap, rim and grain joints (including accelerated diffusion by alpha self-irradiation) in the matrix		Labile activity	
Percentage of labile activity in the matrix	^{36}Cl, ^{79}Se, ^{126}Sn, $^{135,137}Cs$.	38.9 %	77.1 %*
	^{14}C	15 % (conservative)	25 %*
	Other radionuclides	35 %	65 %*
Remaining radionuclides contained in the matrix's grains, released in accordance with the radiolytic dissolution model		As for CU1 UOx2 except BF = 48GWd/t for reference & BF = 53 GWd/t for sensitivity studies	
* : with defective packages, the fraction of labile activity correlated with accelerated diffusion by alpha self-irradation is 10 % lower			

Table 5.3-13 Release model for the radionuclides in spent fuel matrixes in CU2 reference packages

5.3.2.2 Representation of the repository structures

- **General representation of disposal cells**

Note that the calculation is performed for each repository zone. Preparatory calculations revealed not only that a module has no hydraulic influence on other modules, owing to the efficiency of the seals, and also that there is negligible transport of radionuclides via the system of horizontal drifts. Therefore, although all the repository modules in each repository zone are represented, the connecting infrastructure between them (secondary connecting drifts) is not modelled.

- **Representation of the disturbed Callovo-Oxfordian clay in the vicinity of repository structures**

Excavating rock, building structures and placing waste packages into the repository create thermal, hydraulic, mechanical and chemical disturbances to the argillite in the repository structures' near field.

In certain cases, these disturbances can influence the Callovo-Oxfordian layer's properties with respect to the repository and the containment of radionuclides in the near field. In most cases, their effects are controlled by appropriate design measures. Where this is not the case (e.g. in the event of a container failure), their effects are represented conservatively.

From a thermal perspective, the temperature increase in the environment caused by placing exothermic waste into the repository should not, by design, generate a temperature exceeding 90°C in the Callovo-Oxfordian clay or repository structures. The temperature falls below 70°C after 1,000 years. At these temperature levels and timescales, the Callovo-Oxfordian would not undergo any significant irreversible mineralogical transformation. Only the transport and chemical retention parameters change as the temperature field evolves. This is reflected in the choice of values (see section 5.3.2.3).

From a mechanical perspective, models have been developed to describe the extent and geometry of the EDZ. These are proportional to the radius of the excavated diameter, and depend on the structure's depth and orientation.

In keeping with the approach described in section 5.3.1.2, the chosen safety model is the one evaluated at the most pessimistic position in the transposition zone, i.e. at a depth of 630 m, which is the maximum possible depth for a repository in the transposition zone.

The model is considered to be conservative in terms of both extent of the EDZ (being evaluated using the lowest mechanical strength values obtained with test specimens measuring a few centimetres in size) and hydraulic performance [17]. The manner in which it compares to the results of the initial observations made in a drift within the Meuse/Haute-Marne underground laboratory is described in section 6.2.6.1.

Figure 5.3-3 describes the extents calculated at the depth of the Meuse/Haute-Marne laboratory and at the maximum depth, for the various geomechanical horizons. Note that the shaft seal would be fitted in the geomechemical horizon A, and the shafts and disposal cells would be in horizon C.

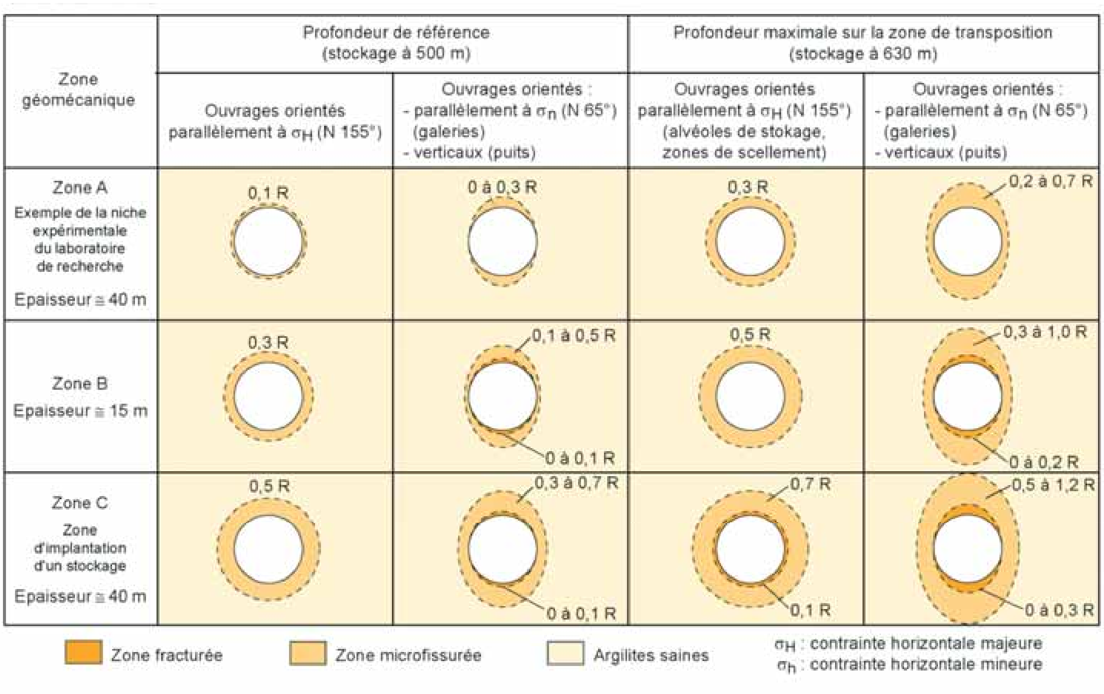


Figure 5.3-3 Extent of the EDZ according to the repository's depth and the individual structures orientation

The rationale consisted in considering those damaged zone extents corresponding to the position of the repository at a depth of 630 m, while taking the orientation of the structures into account, i.e. :

- For the cells and drift sections containing seals, an extent of 0.1 times the radius of the structure (0.1.R) for the fractured zone and 0.7.R for the micro-fractured zone. In the case of the C waste cells, the small diameter of the cells effectively means ignoring the presence of a fractured zone, the extent of which is insignificant ;
- For the main drift sections, an isotropic extent (for the sake of simplicity) corresponding to 0.3.R for the fractured zone and 1.2.R for the micro-fractured zone ;
- For the access shaft section that contains the seal, the fractured zone is ignored (in accordance with what the model is able to predict for horizon A and in keeping with initial observations made in the shaft at the Meuse/Haute-Marne laboratory). The micro-fractured zone predicted by the model measures between 0.2.R and 0.7.R. To ensure continuity with the structures in drifts and at the base of the shaft, it was decided to adopt an extent of 1.2.R. This is a conservative value because it decreases the shaft seal's performance.

From a chemical perspective, the main sources of disturbances are the cementitious materials and components that, when they become degraded (or corrode), can alter the mineralogy of the Callovo-Oxfordian clay and affect chemical, hydraulic and transport properties.

Alkaline disturbance is a process that organises itself in the argillite in two main fronts running from the walls of the repository structures. Theses fronts determine three zones : a highly-remineralised zone with interstitial water chemistry similar to that of concrete and a degree of permeability between those of backfill and the micro-fractured zone, which is relatively undisturbed and characterised by limited dissolution of the initial clayey phases and a water chemistry similar to that of argillite, and the sound argillite [69]. All of the assessments of the extent of alkaline disturbance that have been conducted to date using coupled chemistry and transport (PHREEQC) codes, on the basis of « conservative » or « pessimistic » assumptions, yield values on the order of tens of centimetres for the highly-remineralised zone rising to a few meters in the largely undisturbed zone, for periods of several hundred thousand years.

Furthermore, any disturbance of the Callovo-Oxfordian layer by corrosion products does not extend further than a few centimetres.

The extent of the chemical disturbances as a whole is limited compared with the total thickness of sound Callovo-Oxfordian clay (at least 50 m either side of the structures) ; in addition, any disturbances do not extend beyond the thickness of the fractured zone, which is intercepted by the hydraulic cutoffs at the seals. In the overall calculation, therefore, the chemically disturbed zone is not represented specifically, but assimilated to the fractured zone. The effect of chemical disturbances on the fractured zone is not well understood ; as a reference, it is assumed (in view of the already-conservative characteristics specified for the zone) that they are not aggravated by alkaline disturbance. A specific sensitivity study can be conducted to allow for more unfavourable situations (permeability of 10^{-6} m/s and no geochemical properties).

Table 5.3-14 shows the hydraulic, transport and retention parameters for the micro-fractured and fractured zones used in the reference calculation. As in the Callovo-Oxfordian layer (see section 5.3.2.3), the effect of temperature on the chemical retention of calcium and beryllium is allowed for by applying a correction factor of 0.1 to the value for these elements' partition coefficient while the increase in temperature in the medium is greater than 20°C. Hence the definition and description of two retardation coefficient values in Table 5.3-14 . Caesium sorption is also represented (see section 5.3.2.3) by a Langmuir isotherm.

	Half-life [yrs]	Micro-fractures zone $K = 5.10^{-11}$ m/s				Fractuerd zone $K = 5.10^{-9}$ m/s			
		$\omega_{Diffusion}$ [-]	De [m²/s]	R [-]	Csat [mol/m^3]	$\omega_{Diffusion}$ [-]	De [m²/s]	R [-]	Csat [mol/m^3]
^{10}Be	1 600 000	0,18	$2{,}5.10^{-10}$	31 900	10^{-2}	0,2	5.10^{-10}	29 000	10^{-2}
^{10}Be (deltaT>20)	1 600 000	0,18	$2{,}5.10^{-10}$	3 200	10^{-2}	0,2	5.10^{-10}	2 900	10^{-2}
^{14}C	5 730	0,05	5.10^{-12}	5,6	2,3	0,15	1.10^{-11}	2,5	2,3
^{36}Cl	302 000	0,05	5.10^{-12}	1	soluble	0,15	1.10^{-11}	1	soluble
^{41}Ca	103 000	0,18	$2{,}5.10^{-10}$	16	2,3	0,2	5.10^{-10}	15	2,3
^{41}Ca (deltaT>20)	103 000	0,18	$2{,}5.10^{-10}$	3	2,3	0,2	5.10^{-10}	2,4	2,3
^{59}Ni	75 000	0,18	$2{,}5.10^{-10}$	2 050	5.10^{-2}	0,2	5.10^{-10}	1 800	5.10^{-2}
^{79}Se	65 000	0,05	5.10^{-12}	1	5.10^{-7}	0,15	1.10^{-11}	1	5.10^{-7}
^{93}Zr	1 530 000	0,18	$2{,}5.10^{-10}$	12 800	2.10^{-5}	0,2	5.10^{-10}	12 000	2.10^{-5}
^{93}Mo	3 500	0,05	5.10^{-12}	139	1.10^{-5}	0,15	1.10^{-11}	47	1.10^{-5}
^{93m}Nb	16,4	0,05	5.10^{-12}	53 400	2.10^{-4}	0,15	1.10^{-11}	18 000	2.10^{-4}
^{94}Nb	20 300	0,05	5.10^{-12}	53 400	2.10^{-4}	0,15	1.10^{-11}	18 000	2.10^{-4}
^{99}Tc	213 000	0,18	$2{,}5.10^{-10}$	128 000	4.10^{-6}	0,2	5.10^{-10}	120 000	4.10^{-6}
^{107}Pd	6 500 000	0,18	$2{,}5.10^{-10}$	8 950	4.10^{-4}	0,2	5.10^{-10}	8 100	4.10^{-4}
^{126}Sn	100 000	0,18	$2{,}5.10^{-10}$	179 000	1.10^{-5}	0,2	5.10^{-10}	160 000	1.10^{-5}
^{129}I	15 700 000	0,05	5.10^{-12}	1	soluble	0,15	1.10^{-11}	1	soluble
^{166m}Ho	1 200	0,18	$2{,}5.10^{-10}$	639 000	1.10^{-4}	0,2	5.10^{-10}	580 000	1.10^{-4}
^{135}Cs	2 300 000	0,18	$2{,}5.10^{-10}$	Langmuir*	soluble	0,2	5.10^{-10}	Langmuir*	soluble
		*Langmuir : Kd = $(1{,}8462.10^{-7}/(4{,}7552.10^{-7}$+ Ceq) – Ceq : concentration in solution (mol/l)							

Table 5.3-14 Values of hydraulic, transport and chemical retention parameters in fractured and micro-fractured zones – reference calculation

- **B waste disposal cells**

Once released by the waste packages, radionuclides migrate through the concrete of the cell to reach the Callovo-Oxfordian layer or the clayey cell plug at the end of the repository drift.

Concrete cell body

The degradation of the concrete in the cells is a slow, radial process converging from the Callovo-Oxfordian clay toward the centre of the cell, owing to the effectiveness of the plugs, which limit the entry of water through the drift network [67]. Assuming pessimistic water flow conditions, the cell is only estimated to become totally chemically degraded after several tens of thousands of years. The temperature increase in the concrete caused by slightly exothermic compacted hull and end cap waste has a limited effect on concrete degradation (as the thermal phase coincides with a non-saturated or only very slightly saturated phase).

However, the cell's volume and residual voids, however small they may be, mean that low permeability and diffusion coefficient values cannot be guaranteed. In addition, there remain uncertainties as to the impact of corrosion on the metal reinforcement in the supports and the evolution of the mechanical loads imposed by the ground over the medium and long term. These two phenomena could cause cracks or localised fractures in the support or lining. In view of the current state of understanding of the cell's evolution, a prudent approach to the disposal cell's hydraulic and transport performance has been adopted.

The concrete in the cell is considered to be a uniform chemical environment that provides only chemical containment limiting toxic chemical flows generated by precipitation (« Limit the release of radionuclides and immobilise them in the repository ») and sorption (« Delay and attenuate radionuclide migration »). Permeability is high, at 10^{-6} m/s. The diffusion coefficient and diffusion-accessible porosity are 6.10^{-10} m²/s and 30 %, respectively. The chemical retention parameters are given in Table 5.3-9. These values are adopted for the concrete in all the B waste cells except those for reference packages liable to release products of cellulose degradation, which are included in the category « Other B waste » in Table 5.3-2. Pessimistic chemical retention values have been adopted for this category of waste, in order to allow for the uncertainties relating to the potentially harmful effect of degradation products on the concrete's retention capabilities.

A sensitivity study has been conducted with conservative parameters for chemical retention within the B waste cell's concrete body. This illustrative study was conducted for reference packages disposed in B1x cells ; it was combined with sensitivity to the parameters for transport and chemical retention in the host formation (see section 5.3.2.3).

Disposal cell seal

The cell seal is represented as a continuous uniform porous medium. This is due to its rapid resaturation, which quickly fills the gaps between the bentonite bricks, and the limited extent of chemical disturbance (which extend for between a few centimetres - in the case of iron-clay disturbance - and a few tens of centimetres – in the case of alkaline disturbance in the bentonite – but which do not affect the seal dimensioning). The reference calculation assumes that radionuclides migrate into the cell seal by diffusion/advection, with chemical containment afforded by sorption and precipitation (see Figure 5.3-4). In view of the level of scientific understanding attained, the values for the transport and chemical retention parameters are treated as phenomenological in the reference calculation. They are given in Table 5.3-15. As in the Callovo-Oxfordian layer (see section 5.3.2.3), the effect of temperature on the chemical retention of calcium and beryllium is allowed for by applying a correction factor of 0.1 to the value for these elements' partition coefficient while the increase in temperature in the medium is greater than 20°C [21]. Accordingly, two retardation coefficient values are included in Table 5.3-15.

A sensitivity study has been conducted with conservative parameters for transport and geochemical retention in the bentonite plug. This study is coupled with sensitivity to the parameters for transport and chemical retention in the concrete cell body and the host formation (see section 5.3.2.3).

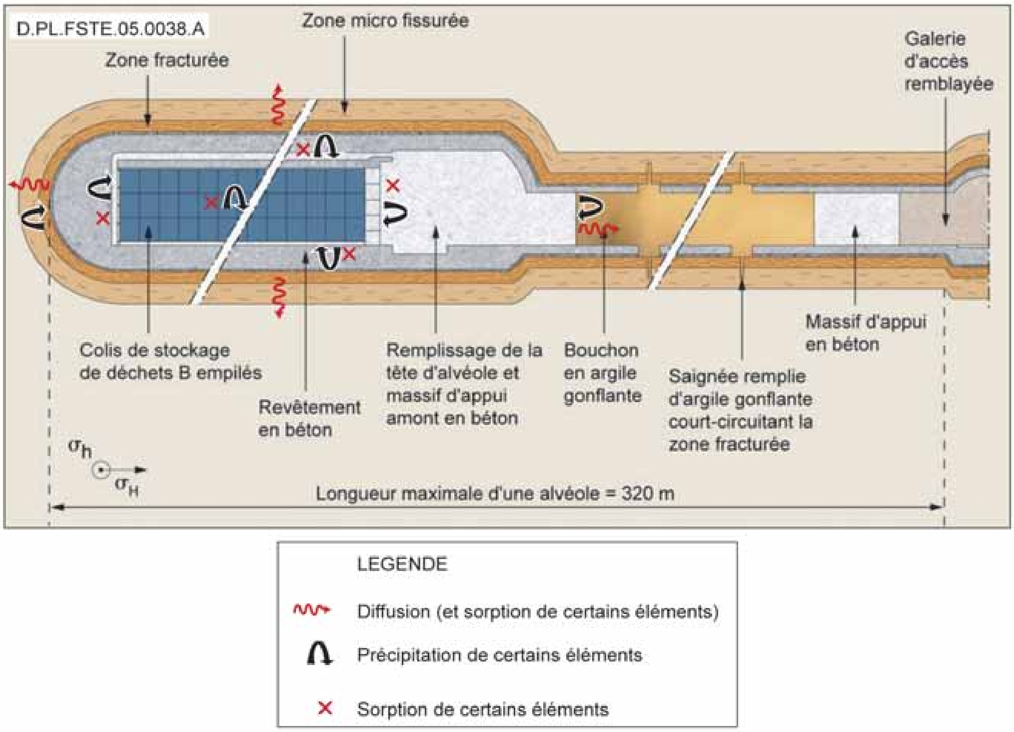


Figure 5.3-4 Representation of the B waste cells and containment characteristics

Regarding the permeability of bentonite, the adopted value represents the equivalent permeability of a clay-based seal calculated on the basis of the TSX full-scale test conducted in an underground laboratory in Canada [43]. The equivalent permeability calculated according to the water flows circulating in the drift and the imposed pressure gradients is 10^{-11} m/s. However, various tests indicate that the preferential path lies at the interface between the seal and the rock, and that the seal body actually has superior properties. Note that the values found in the literature or obtained experimentally at different densities are generally lower than 10^{-13} m/s.

For the sake of prudence, Andra adopted a value of 10^{-11} m/s for the permeability of the material forming the seal.. This value is conservative as it consists in treating the material's overall permeability in the same way as the equivalent permeability of the seal system including its interfaces. This representational choice yields a lower overall seal performance than what appears achievable in the light of feedback from the TSX given the material properties.

	K = 1.10^-11^ m/s				
	Half-life [years]	$\omega_{Diffusion}$ [-]	De [m²/s]	R [-]	Csat [mol/m^3]
^{10}Be	1 600 000	0,36	5.10^{-10}	973	10^{-2}
^{10}Be (deltaT>20)	1 600 000	0,36	5.10^{-10}	98	10^{-2}
^{14}C	5 730	0,05	5.10^{-12}	1	2,3
^{36}Cl	302 000	0,05	5.10^{-12}	1	soluble
^{41}Ca	103 000	0,36	5.10^{-10}	6	2,3
^{41}Ca (deltaT>20)	103 000	0,36	5.10^{-10}	1,5	2,3
^{59}Ni	75 000	0,36	5.10^{-10}	2 430	5.10^{-2}
^{79}Se	65 000	0,05	5.10^{-12}	1	5.10^{-7}
^{93}Zr	1 530 000	0,36	5.10^{-10}	486 000	2.10^{-5}
^{93}Mo	3 500	0,05	5.10^{-12}	1	1.10^{-5}
^{93m}Nb	16,4	0,05	5.10^{-12}	350 000	2.10^{-4}
^{94}Nb	20 300	0,05	5.10^{-12}	350 000	2.10^{-4}
^{99}Tc	213 000	0,36	5.10^{-10}	146 000	4.10^{-6}
^{107}Pd	6 500 000	0,36	5.10^{-10}	4 380	4.10^{-4}
^{126}Sn	100 000	0,36	5.10^{-10}	53 500	1.10^{-5}
^{129}I	15 700 000	0,05	5.10^{-12}	1	soluble
^{166m}Ho	1 200	0,36	5.10^{-10}	58 300	1.10^{-4}
^{135}Cs	2 300 000	0,36	5.10^{-10}	487	soluble
^{135}Cs (deltaT>20)	2 300 000	0,36	5.10^{-10}	50	soluble

Table 5.3-15 Values of hydraulic, transport and chemical retention parameters in swelling clay structures [70] – reference calculation

Furthermore, the seals are assumed to only intercept the fractured zone without extending into the micro-fractured zone. This representation is conservative with respect to the depth of the hydraulic cut-off grooves and the potential number of hydraulic cutoffs. In view of the difference in scale (30 cm trenches compared with drifts measuring hundreds of metres long), the seal has been modelled as a core with predefined transport parameters and an equivalent hydraulic cutoff represented over the entire length of the core as shown in Figure 5.3-5

Note that these conservative choices (high bentonite permeability and limited hydraulic cut-off lengths) ultimately result in seals that are much less effective in the model than in the dimensioning study. The dimensioning study targets an equivalent permeability of 10^{-10} m/s. The body of seal feasibility results obtained to date indicate that such an objective is realistic [41]. The equivalent hydraulic cut-off permeability represented in the calculation is only 10^{-9} m/s[66], which serves to emphasise the pessimistic character of the calculation. Additionally, the diffusion coefficient of the material with the highest value is used.

The rest of the seal (retaining plug and ground support) is represented like the backfill in standard drifts.

66 This figure represents the harmonic mean of the permeabilities of the media encountered by the water along its path, i.e. the micro-fractured zone, lining and the swelling clay placed in the grooves, weighted according to the distances travelled.

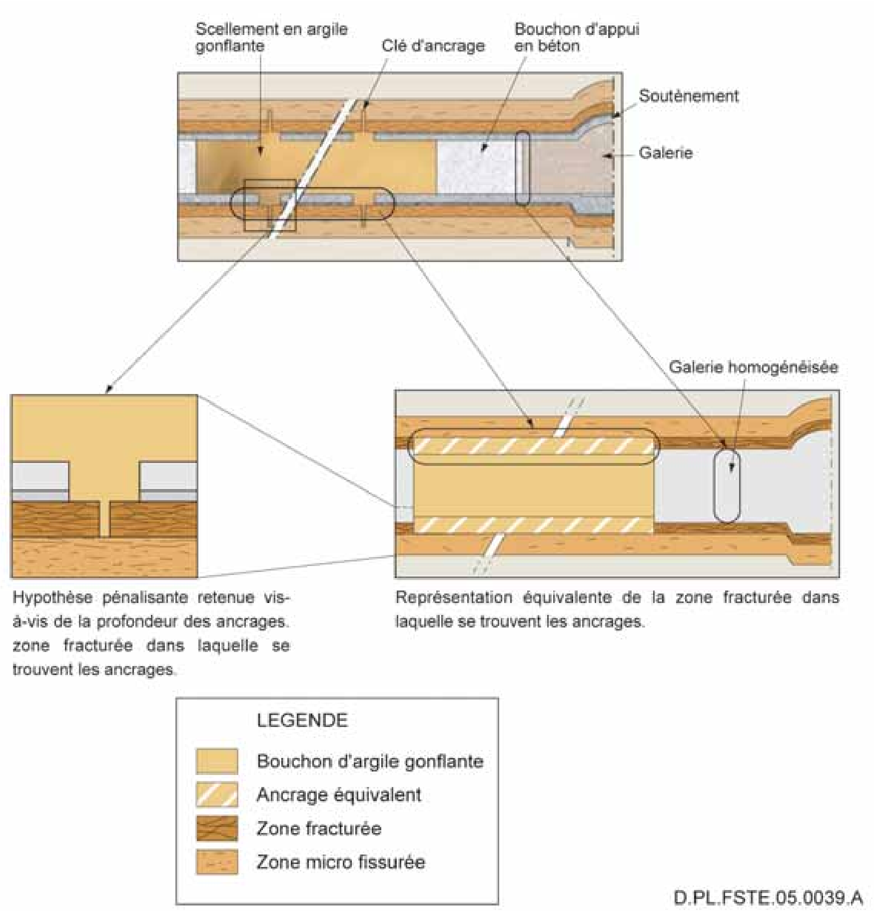


Figure 5.3-5 Equivalent representation of B waste disposal cell plugs and seals

- **C waste disposal cells**

Once released by the C waste packages, radionuclides migrate through the fractured zone around the packages or the clay plug and into the Callovo-Oxfordian layer or the repository drift adjacent to the plug.

C waste cell plugs are represented in a similar manner to B waste cell seals, except that they are not anchored, owing to the absence of a post-excavation fractured zone. The model yields a very small extent for the fractured zone, as cover no extension at all ; therefore it has not been represented.

The influence exerted by the evolution of the temperature field in the cells' near field on the transport and chemical retention parameters is taken into account inasmuch as it has a significant effect at cell level ; it is processed in a similar manner to that used for the Callovo-Oxfordian clay (see 5.3.2.3). The chosen hydraulic, transport and chemical retention parameters are similar to those of the swelling clay in the B waste cell seals ; they are described in Table 5.3-15.

A sensitivity study has been conducted with conservative parameters for transport and geochemical retention in the bentonite plug. This study is coupled with sensitivity to the parameters for transport and chemical retention in the host formation (see section 5.3.2.3).

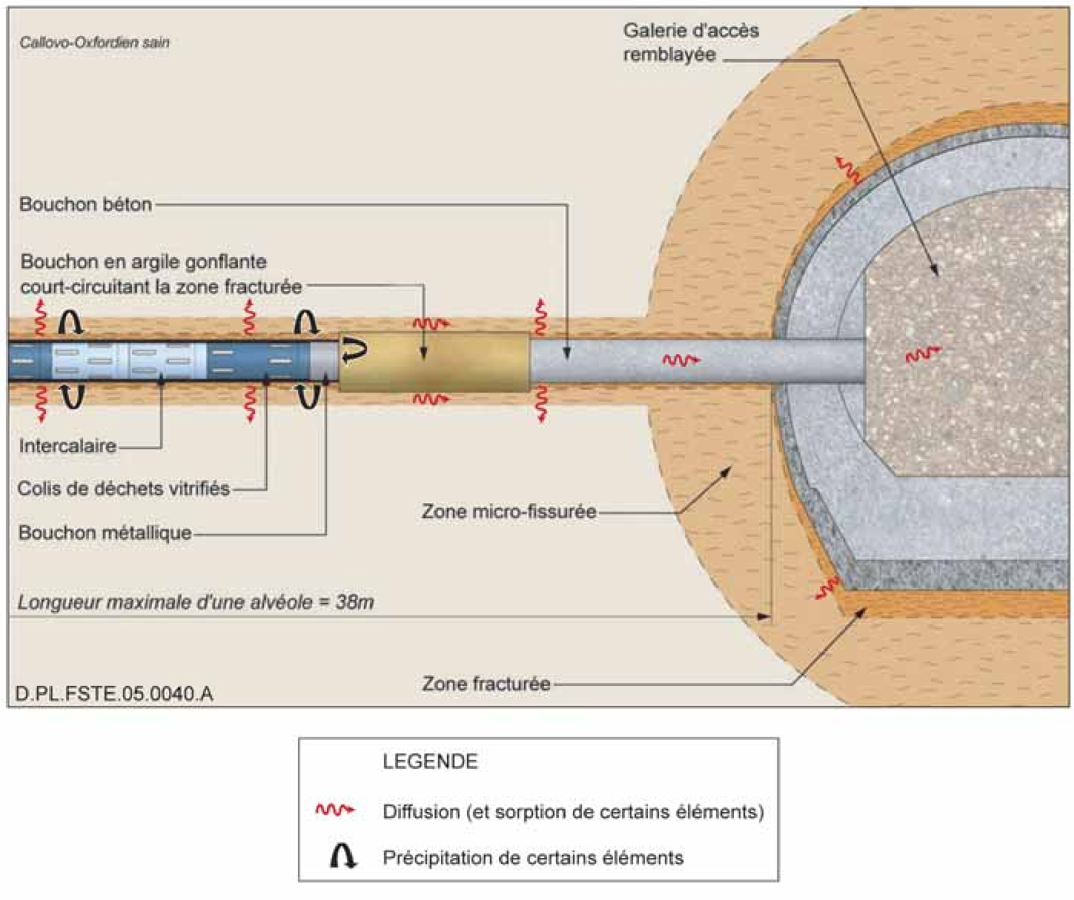


Figure 5.3-6 Representation of the C waste cells and containment characteristics

There are large numbers of C waste cell plugs. As with the packages, it is not possible to rule out the possibility of quality control problems leading to a failure to detect initial defects in the plugs, causing a problem at the plug/argillite interface. Following the same logic as for the initial failure of containers or overpacks, it was decided to study a cell plug failure case corresponding to a failure rate of approximately 1 in 1,000. This rate reflects an initial failure of :

- One C1 waste cell plug,
- Four C2 waste cell plugs,
- Two C3 waste cell plugs,
- Three C4 waste cell plugs.

The defective plugs are distributed evenly among the various modules, which amounts to considering the failed plugs in isolation. The aim is to verify that occasional failures have no influence on transport.

- **Spent fuel cells**

Once released by the spent fuel waste packages, radionuclides migrate through the core of the swelling clay buffer or the clay plug and into the Callovo-Oxfordian layer or the repository drift adjacent to the plug (see Figure 5.3-7).

Structures and internal transfers are represented in a similar way to the C waste cells. The temperature effects are included in exactly the same way.

In the light of current scientific understanding about the clay engineered barrier (i.e. body where applicable and plug), the parameter values are treated as being phenomenological values in the reference calculation, except in the case of permeability, for which the chosen value, judged to be conservative, is consistent with the value adopted for the B waste cell seals. The values are described in Table 5.3-15

A sensitivity study has been conducted using conservative parameters for transport and geochemical retention in the cell's bentonite plug and body. This study is coupled with sensitivity to the parameters for transport and chemical retention in the host formation (see section 5.3.2.3).

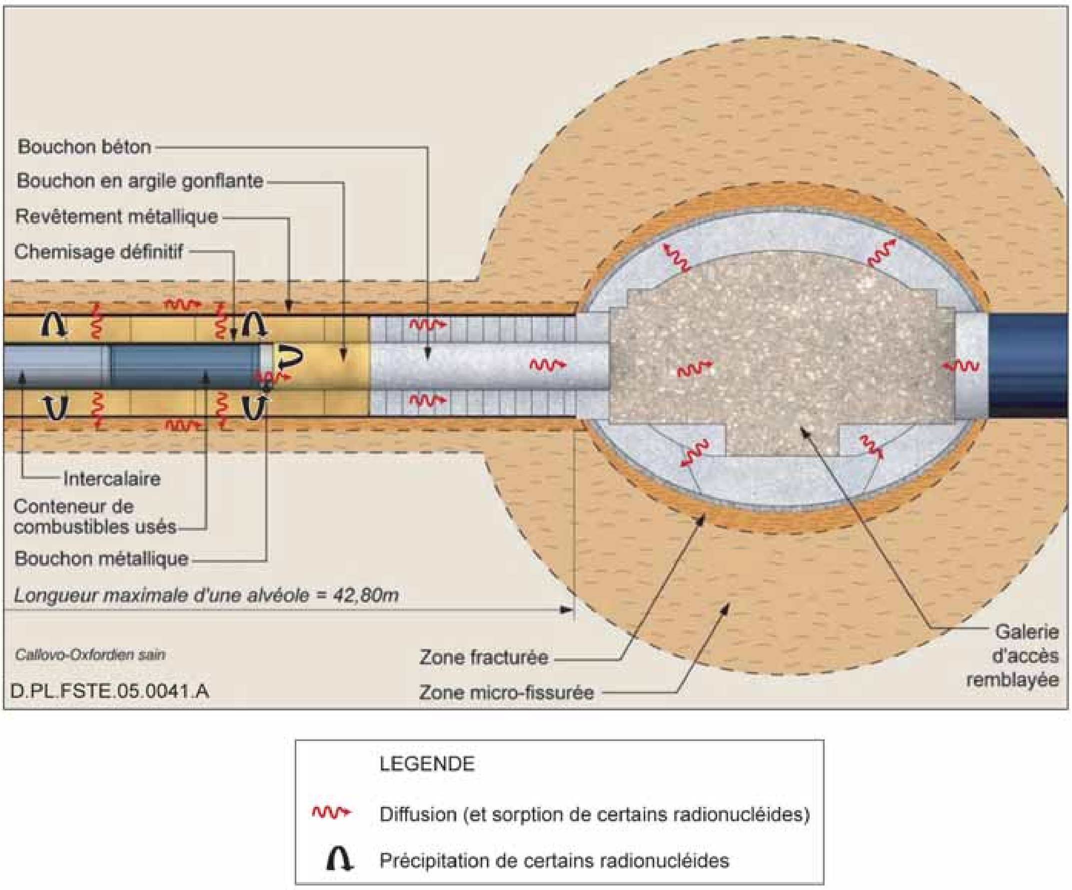


Figure 5.3-7 Representation of the spent fuel waste cells and containment characteristics

- **Seals and backfill**

After migrating through the cell plugs or the associated EDZ, a fraction of the radionuclides will reach the backfilled drifts and pass through them before migrating into the Callovo-Oxfordian layer or a drift seal.

Drift and shaft seals

Drift and shaft seals are represented in a similar manner to the B waste cell seals. Their chemical, mechanical and hydraulic environment is also comparable. In zones containing exothermic waste, the temperature increase does not appear to have a significant irreversible influence on mineralogy, thanks to the design measures that limit the temperature inside the repository to 90°C. In these conditions, the model used to represent the seal and the transport and chemical retention of toxic chemicals in it is similar to the model adopted for the B waste cell seals ; specifically, it assumes considerably poorer seal performance (in terms of hydraulic length, bentonite permeability, etc.) than what appears achievable in the light of the engineering studies.

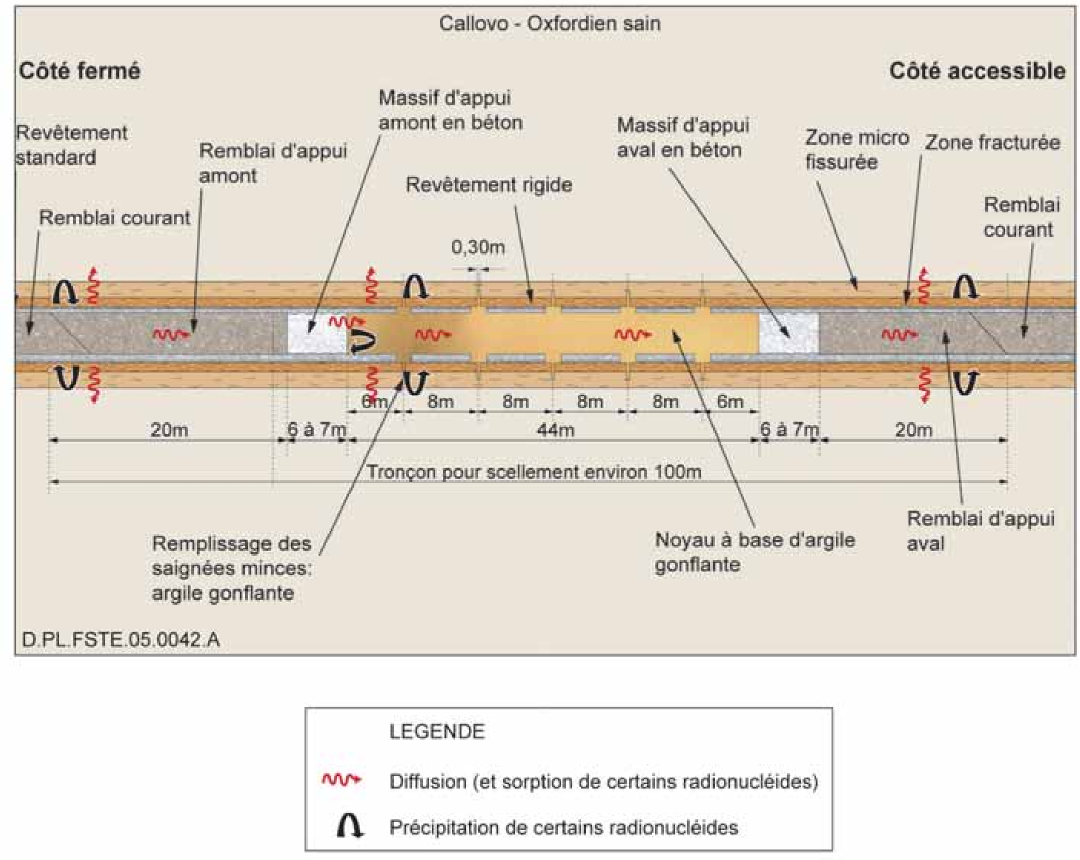


Figure 5.3-8 Representation of seals and containment characteristics

The transport and chemical retention models for the shaft seals are similar to those adopted for the drift seals. However, as the latter are located in geomechanical zone A, which has better mechanical strength properties, there is no fractured zone around the seal. The absence of a fractured zone eliminates the need to create hydraulic cutoffs.

Backfilled drifts

Backfilled drifts consist of the ground supports and linings that remain in situ following the repository's closure, together with backfill material, the composition of which has yet to be finalised, but whose primary function is to provide mechanical strength.

As with the B waste cells, there are still uncertainties regarding the impact that corrosion of the reinforcements in ground supports and linings has on their performance. Furthermore, conclusive experiments have been conducted to demonstrate the installation of low permeability backfill, but the technique remains subject to uncertainty at the scale of the complete repository. For the above reasons, a pessimistic overall approach has been adopted for the transport and hydraulic performances of the « support, lining and backfill » system. An equivalent permeability of 3.10^{-7} m/s was selected (see Figure 5.3-9). The diffusion coefficient is 10^{-9} m²/s (a pessimistic coefficient that corresponds to the diffusion of a water particle in water), the same value as adopted for the lining, support and backfill.

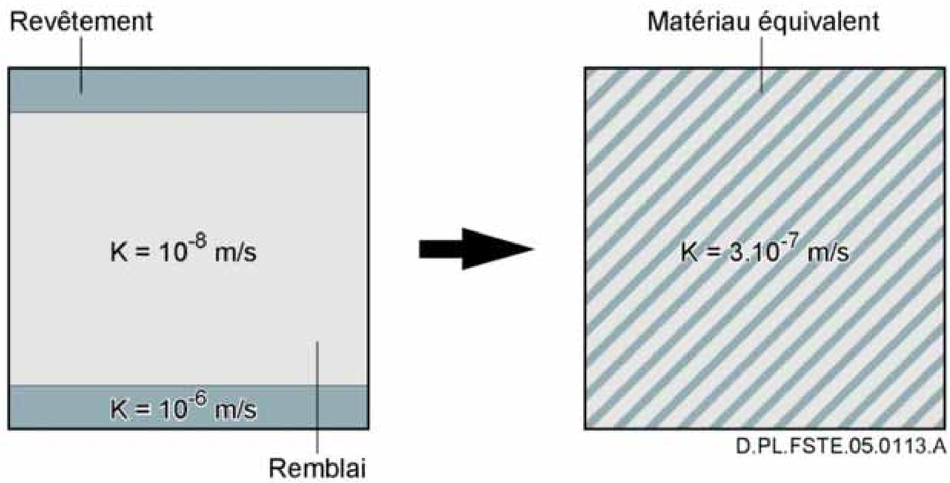


Figure 5.3-9 Equivalent representation of the drifts

In view of the fact that the backfill is not expected to contribute to delaying and reducing radionuclide migration (its function being to provide long-term mechanical support), it was pessimistically decided to ignore any geochemical retention properties in backfilled drifts.

5.3.2.3 Representation of the host formation

An especially sensitive evaluation element is the representation of the host formation model (see Figure 5.3-10). It is based on :

- The understanding of the history of the Paris Basin and the geological elements that may affect the formation at the regional scale. The understanding that we have from this history determines the general characteristics of the geological medium ;
- The measuring campaigns via bore-hole of 1999, by 2D and 3D seismic reflection carried out till 2000, which led to the realisation of a first hydrogeological model of the sector ;
- The reprocessing of 3D seismic, bore-hole campaigns of the sector especially meant for acquiring additional data for model fitting, that led to review the model for this Dossier ;
- Formation exploration boreholes campaigns conducted in 2003 – 2004.

The Callovo-Oxfordian characteristics are represented such that they best include their containment properties but without overestimating them. We integrate the simplifications of the phenomenological model of the host formation to include its properties at the sector's scale.

With regards to the data acquired on bore-hole core samples originating from different Callovo-Oxfordian lithofacies, it appears that the low vertical variability has no significant incidence on transportation of elements at the Callovo-Oxfordian scale. The latter is therefore represented by a porous homogenous medium of constant thickness, corresponding, in the logic given in section 5.3.1.1 to the minimum thickness encountered in the transposition zone (130 meters). Repository is deemed to be located at the middle of the layer (that is 65 meters under the top of the host formation).

The effect of increase in thickness of Callovo-Oxfordian to the North-West of the transposition zone is evaluated through a sensitivity study considering an approximate 10-metre increase in host formation thickness on both sides of the repository (the repository is then deemed to be located 75 meters under the host formation top).

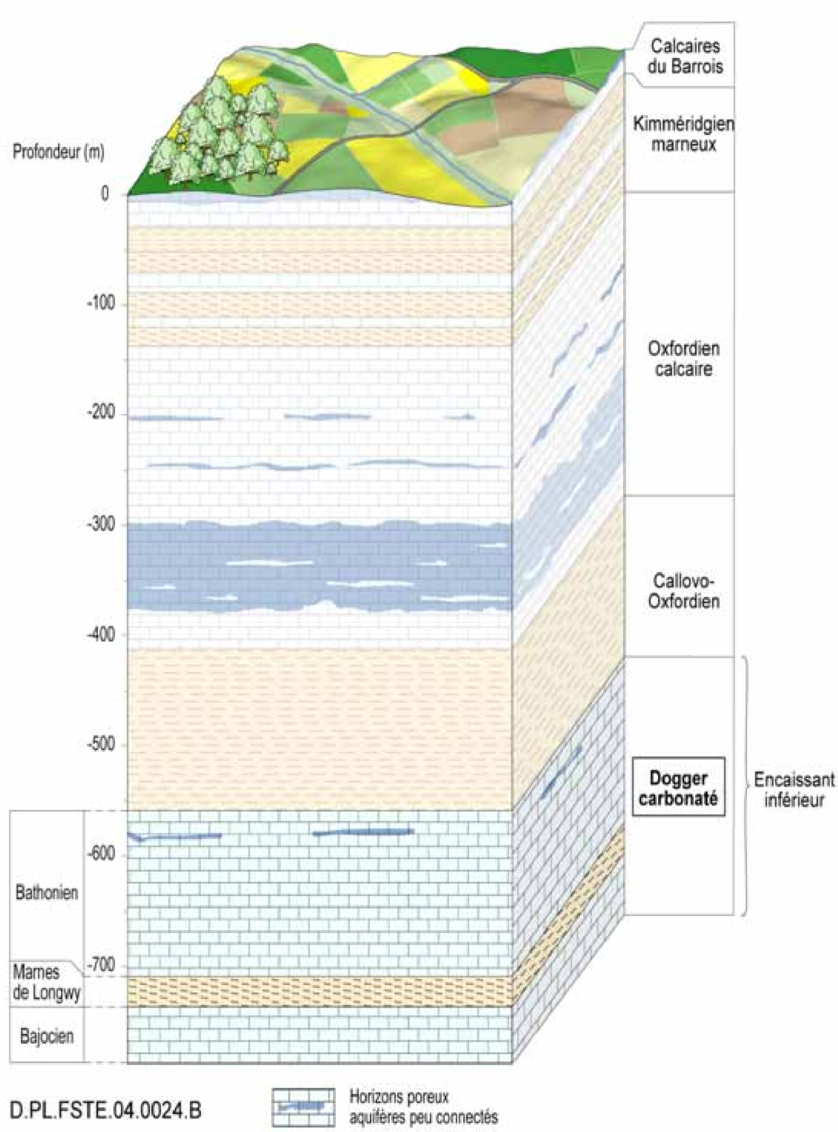


Figure 5.3-10 Geological formations at the level of the Meuse / Haute Marne site and position of the repository in the middle of the Callovo-Oxfordian

Behaviour of the elements in Callovo-Oxfordian is represented by (see Figure 5.3-11) :

- Diffusion/advection transport. The anionic exclusion phenomenon is used for diffusion. The likelihood of transporting elements in colloidal form (size of particles between 1 nm and 104 nm) was studied with the conclusion that these could be formed but small dimension of pores and different interaction phenomena of particles with the solid Argillite matrix should limit the potential flow to an insignificant value. Therefore, transportation in colloidal form has not been considered in SEN ; this point is discussed below in chapter 6, in section 6.2.10 ;
- Chemical retention of elements through sorption and precipitation phenomena. The sorption of elements is represented by a retardation coefficient. This is calculated from the partition coefficient evaluated experimentally or established through analogy with other elements[67].

[67] In the specific case of cesium, experimental results highlight that its retention level closely depends on its concentration in solution [71]. For this reason, the value of its partition coefficient (Kd) varies by several orders of magnitude. This behaviour could be explained by the existence of several types of exchange sites on clay surfaces. To take this phenomenon into account, sorption of cesium was represented by a law specifically depending on its concentration in solution, established through experimental results (called « Langmuir's isotherm »).

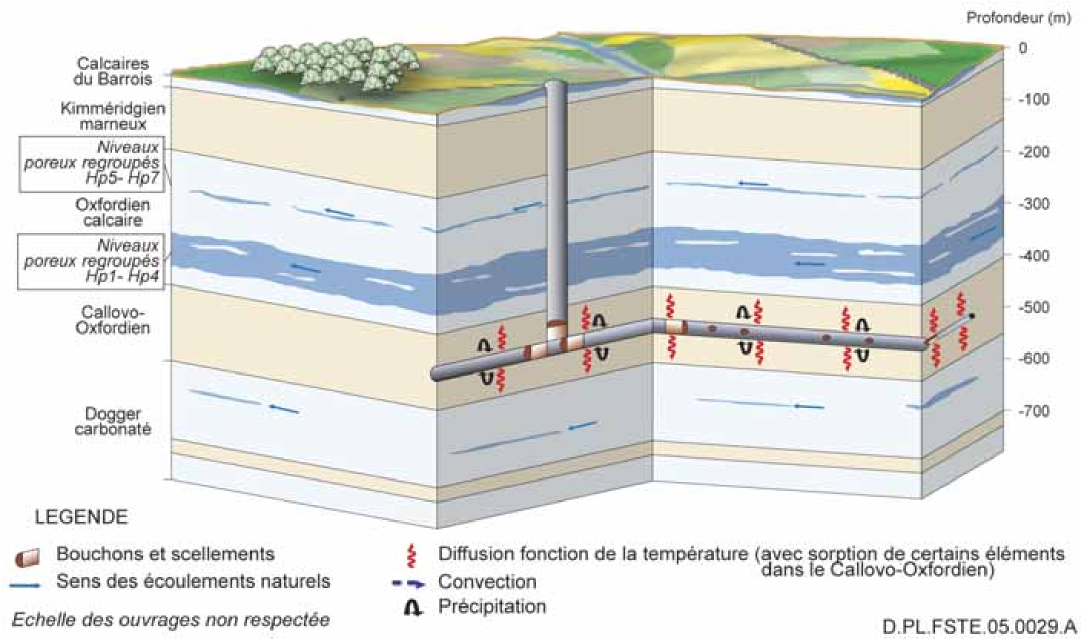


Figure 5.3-11 Transport and retention phenomena preponderant in the Callovo-Oxfordian considered in the SEN

- **Nominal parameter values**

The effective diffusion (De) and apparent porosity (ω_{app}) coefficient values were measured by performing diffusion tests. The values retained for reference calculations are « phenomenological values », the consistency of all the measurements was good.

- For anions, the retained value corresponds to the average of the effective diffusion and accessible porosity coefficient values on diffusion evaluated at an ionic force of 0.1M, that is, in absence of temperature effect, $De_{anions} = 5.10^{-12}$ m²/s and $\omega_{anions} = 5$ %.
- For cations, the retained value corresponds to the average of the effective diffusion coefficient values measured for cesium and accessible porosity on diffusion corresponding to the total porosity, that is, in absence of temperature effect, $De_{cations} = 2,5.10^{-10}$ m²/s and $\omega_{cations} = 18$ %.

The permeability values interpreted in situ from the pressure measurements obtained over long periods of time as well as those measured on unaltered samples give us vertical permeability values between 10^{-14} m/s and 10^{-13} m/s. Horizontal bedding of Argillites may give rise to anisotropy in the permeability which could however not be highlighted by the different analysis that were carried out. For permeability values lying between the range of values given above, the anisotropy ratio shall stay less than 10. These different results give as reference values a value of 5.10^{-14} m/s for vertical permeability and 5.10^{-13} m/s for horizontal permeability. These values are « phenomenological », the anisotropy ratio was however probably overestimated in a negative sense for horizontal permeability. The kinematic porosity retained is 9 % for anions and cations.

Table 5.3-16 summarises the main transport and chemical retention data in reference calculations for the studied radionuclides.

	$K_h = 5.10^{-13}$ m/s $K_v = 5.10^{-14}$ m/s				
	Half-life [yrs]	$\omega_{Diffusion}$ [-]	De [m²/s]	R [-]	Csat [mol/m^3]
^{10}Be	1 600 000	0,18	$2,5.10^{-10}$	31 900	10^{-2}
^{10}Be (deltaT>20)	1 600 000	0,18	$2,5.10^{-10}$	3 200	10^{-2}
^{14}C	5 730	0,05	5.10^{-12}	5,6	2,3
^{36}Cl	302 000	0,05	5.10^{-12}	1	soluble
^{41}Ca	103 000	0,18	$2,5.10^{-10}$	16	2,3
^{41}Ca (deltaT>20)	103 000	0,18	$2,5.10^{-10}$	2,5	2,3
^{59}Ni	75 000	0,18	$2,5.10^{-10}$	2 050	5.10^{-2}
^{79}Se	65 000	0,05	5.10^{-12}	1	5.10^{-7}
^{93}Zr	1 530 000	0,18	$2,5.10^{-10}$	12 800	2.10^{-5}
^{93}Mo	3 500	0,05	5.10^{-12}	139	1.10^{-5}
^{93m}Nb	16,4	0,05	5.10^{-12}	53 400	2.10^{-4}
^{94}Nb	20 300	0,05	5.10-^{12}	53 400	2.10^{-4}
^{99}Tc	213 000	0,18	$2,5.10^{-10}$	128 000	4.10^{-6}
^{107}Pd	6 500 000	0,18	$2,5.10^{-10}$	8 950	4.10^{-4}
^{126}Sn	100 000	0,18	$2,5.10^{-10}$	179 000	1.10^{-5}
^{129}I	15 700 000	0,05	5.10^{-12}	1	soluble
^{166m}Ho	1 200	0,18	$2,5.10^{-10}$	639 000	1.10^{-4}
^{135}Cs	2 300 000	0,18	$2,5.10^{-10}$	Langmuir*	soluble
		*Langmuir : Kd = $(1,8462.10^{-7}/(4,7552.10^{-7}$+ Ceq) Ceq : concentration in solution (mol/l)			

Table 5.3-16 Hydraulic parameter, transport and chemical retention values in the Callovo-Oxfordian-reference calculations

- **Temperature effects taken into account**

At the exothermic waste level, the radionuclides, especially those released by failed packages (see section 5.3.2.1), can migrate in a medium with a high thermal load and high thermal gradient. Its transport and chemical retention models and parameters can be modified.

Under these conditions, transport is modelled under the effect of temperature gradient (soret effect) whenever it is high. For this, we have based ourselves on a temperature change model in the repository established on a two dimension conservative base, which under-estimates the capacity of the geological medium to remove the heat.

We also take into account the effect of temperature on the diffusion coefficient value. To do this, the Callovo-Oxfordian was vertically divided into three « thermal zones », the median zone itself being a sub-unit representing the engineered repository structures and the associated EDZ (see Figure 5.3-12). In each of these units, an effective corrective diffusion coefficient factor was evaluated at 20°C, as a function of time (the latter being dependant on temperature). Figure 5.3-12 illustrates the change in time of the corrective factor of the spent fuel CU1. The corrective factors are then presented for other reference packages in Figure 5.3-13.

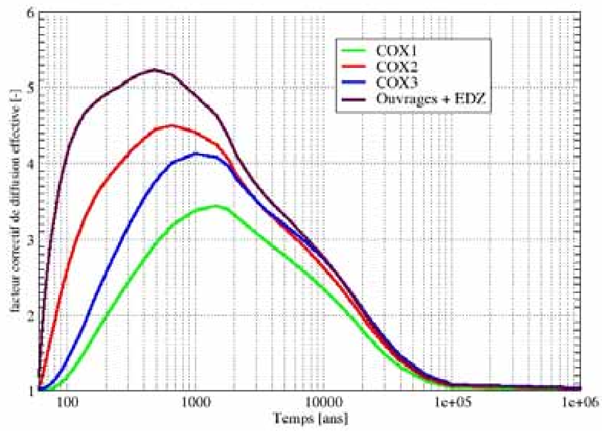
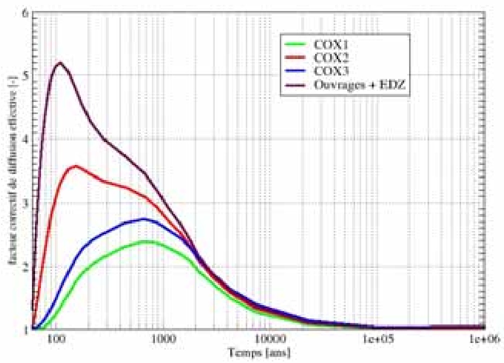
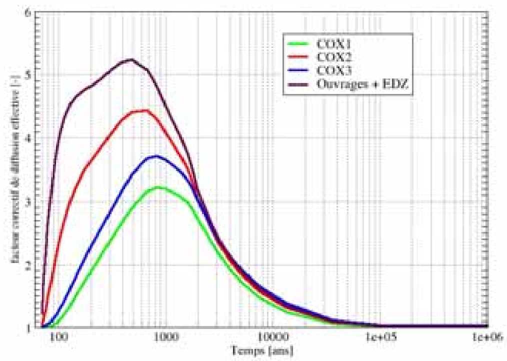


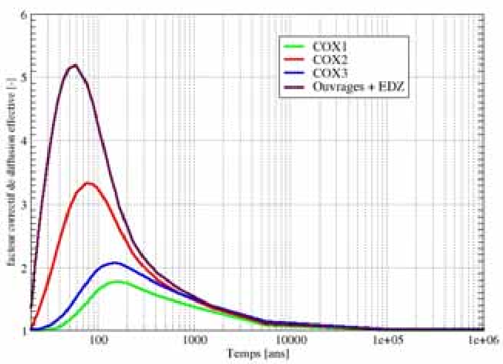
Figure 5.3-12 Corrective factor applied to the effective diffusion coefficient at 20°C for taking into account the temperature – spent fuel CU1 (COX = Callovo-Oxfordian)



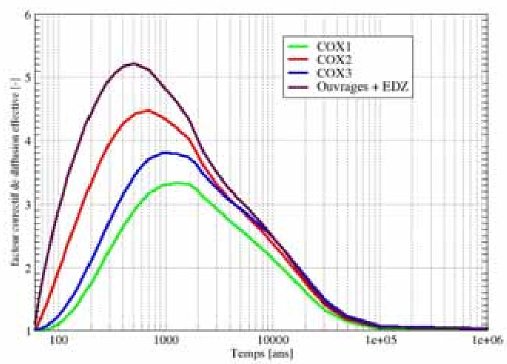
Reference packages C1 + C2



Reference packages C3 + C4



Reference package C0



Reference package CU2

Figure 5.3-13 Corrective factor applied to the effective diffusion coefficient at 20°C for taking the temperature into account– spent fuel CU2 and vitrified waste C

In order to simplify calculations and given that we expect that the transport be mainly diffusive, the influence of temperature on the permeability value has not been included in the overall calculation diagram of the normal evolution scenario.

The effect of temperature on the solvability of elements is difficult to apprehend ; it appears that it can have a favourable or unfavourable effect based on the elements being studied. Internationally, consistent and sufficient thermodynamic data sets do not exist to understand the influence of temperature for all elements studied. In addition, the speciation calculations highlight that the influence of temperature on the solubility of studied elements is generally negligible in face of sensitivity to water composition parameters (pH, pCO_2, Eh) [71]. Therefore, no corrective factor is retained to represent the effect of temperature on the solubility of elements.

Studies concerning adsorption of elements on clay-based materials as a function of temperature are also recent. They present difficulties associated on one hand with the representation of adsorption sites and their affinity vis-à-vis radionuclides present in very small quantities and on the other hand with understanding of the dissolution/precipitation balances interfering with the balances at the surfaces [71]. Similar to solubility, the effect of temperature, based on elements, can be favourable or unfavourable. It seems that the adsorption of cations and actinides increases with temperature. On the other hand, it tends to reduce for cesium. The approach therefore consisted of retaining a 0.1 reversible corrective factor of the partition coefficient for cesium and by analogy for beryllium as well as calcium, as long as the temperature rise of the medium is greater than 20°C. The durations for returning to a temperature differential less than 20°C with respect to the initial temperature, duration from which we no longer take the temperature effects into account, are presented in Table 5.3-17 As a reminder, they arise from a 2D conservative model. The breakdown of Callovo-Oxfordian into three zones, as is considered for calculating the corrective factor of the diffusion coefficient, is preserved ; however, after a certain date, the Callovo-Oxfordian temperature becomes quasi-homogenous and the dates for return to a temperature differential less than 20°C are identical for the three « thermal zones ».

Date when the temperature rise reverts to less than 20°C				
C2	C4	CU1	CU2	C0
3 500 years	5 000 years	15 000 years	15 000 years	1 000 years

Table 5.3-17 Date on which the temperature rise becomes less than 20°C – C waste and CU1 and CU2 spent fuel disposal cell (conservative thermal model)

- **Values retained in sensitivity studies**

In order to more broadly cover the uncertainties, conservative values were defined for those parameters to which « phenomenological » values had been attributed in the reference case. These conservation values are the most unfavourable amongst those measured, after possible aberrant values have been eliminated :

- A first sensitivity study was conducted with conservative horizontal and vertical permeability values, 5.10^{-12} m/s and 5.10^{-13} m/s , respectively.
- A second sensitivity study was conducted on transport and geochemical parameters in the host formation, especially on :
  - ✓ Effective diffusion coefficient and porosity values. The (De, ω) value pairs are (10^{-11} m²/s ; 4 %) and (5.10^{-10} m²/s ; 21 %) for anions and cations respectively ;
  - ✓ The conservative retardation coefficient and solubility limit values taking into account, among other things, uncertainties associated with the composition of pore water. In absence of conservative data for partition coefficient (Kd) of some radionuclides, an arbitrary approach is to retain the pessimistic values by resetting the low phenomenological values to zero or by applying a corrective factor of 0.1 to the phenomenological value of the retardation coefficient when it is significant but uncertain [71].

This study is combined with sensitivity to transport and chemical retention parameters in clay and concrete.

In addition, recent studies conducted on retention of radionuclides in the Callovo-Oxfordian showed that the iodine could present chemical retention leading to transport delay. The interpretation of these results and possible phenomena at the origin of this retention become all the more difficult since some results are contradictory. Thus, a null coefficient value of the partition coefficient (unit retardation coefficient) has been retained for reference calculations. A sensitivity study is nevertheless carried out to evaluate the possible gain on the activity flow out of the Callovo-Oxfordian associated with low sorption of iodine (Kd = 10^{-3} m^3/kg leading to a delay of 50 in the Callovo-Oxfordian).

Lastly, temperature calculations in the structures and the Callovo-Oxfordian that condition the diffusion coefficient and retardation coefficient values are conservative since the models used were two dimensional and therefore over estimate the thermal field and do not integrate energy dissipation in the repository. Thus a sensitivity study was conducted using more realistic thermal calculations carried out in 3D, and that include the different repository components in detail. In that case, the temperature rise calculated in the repository is less important.

These sensitivity studies allow you to take into account the residual uncertainties on the representation of the Callovo-Oxfordian formation in far-field. In chapter 6 we will find a more detailed discussion on these uncertainties, which will let us put the choices of modelling carried out and the sensitivity studies in perspective with regards to each of them.

5.3.2.4 Representation of surrounding formations

The « safety » model of surrounding formations is simplified with respect to the phenomenological knowledge acquired on their complete structure. This can be explained by the fact that :

- the surrounding formations are not part of the repository system, in the sense that they are not elements to which the designer attributes a safety function (see chapter 3). These formations nevertheless play a role in impact assessment, since they intervene in the path of radionuclides up to the biosphere, therefore in the conversion of containment properties of the host formation in dose at the outlet. The first objective of their representation in the safety model is therefore impact calculations, under conditions that do not under-estimate it. It should also not be grossly over-estimated ;
- formations, especially those overlying the host formation, present an overall structure that is more complex than that of the Callovo-Oxfordian and with a larger lateral variability. The « safety model » chooses to present its stylised vision, representative of flow directions at site scale.

In addition, the overlying formations are subject to erosion. Their evolutions are taken into account through a geoprospective model. Studies show that the overall evolution of surrounding formations over millions of years results on one hand in the disappearance of regional water flows in favour of local outlets and on the other by the coming closer of outlets towards the laboratory zone of Meuse / Haute-Marne.

The date to be selected to set the hydrogeological model of the surrounding formations for calculations is a delicate topic. It seems appropriate to retain a model corresponding to the piercing date of radionuclides outside the host formation. This solution is subject to several difficulties :

- this period could be different based on the considered elements, each of them not being similarly mobile within the Callovo-Oxfordian. It could also be different based on the considered repository zones ;
- the elements reach the surrounding formations with some dispersion in time, based on the delays encountered during their travel. However, a dynamic model of the formations coupled with their transport is too complex to realise. It could also prove to be less robust to uncertainties on transfer time ;
- the calculation does not expect to be predictive with regards to the emergence period of toxics. It is carried out under simplified conditions, where the aim is to arrive at increase in impact and not its prediction. The emergence times outside the host formation as calculated are good indicators of concerned time periods but do not have a predictive value.

Under these conditions and taking into account that the evolution from today to a million years is relatively regular, it was decided to test the million year model and the current model turn by turn, rather than fix an intermediary date that will always be subject to caution. Consideration of the two extreme cases provides a good frame for any intermediate situation. The comparison at the 0.25 mSv/year threshold is also carried out on the current model as on the prospective model.

The head gradients in the Callovo-Oxfordian are imposed by surrounding formations. In consistency with the selection made for these surrounding formations, a conservative and constant ascending vertical head gradient is considered over the entire transposition zone of 0.4 m/m for the 1 million years model and 0.2 m/m for the current mode, for the transport in the overlying formation. Far the transport to the Dogger, a vertical gradient of 0.2 m/m is considered.

The simplified representation of surrounding formations adopted for safety calculations required regrouping the layers presenting similar hydraulic and transport performances. The layers are modelled with constant thickness equal to those encountered at the level of the Meuse / Haute-Marne laboratory. No transport on geochemical property that could delay the radionuclides is attributed to these formations their diffusion coefficients are high (see Figure 5.3-14 [72]) :

- The Dogger, underlying formation, is represented as a low permeability layer with constant thickness (up to 150 meters). In this layer, a more permeable porous layer of minor thickness (around 5 meters) is represented around 20 meters under the Callovo-Oxfordian bottom to take into account the low water ingress observed at the Bathonian level (water ingress that does not however seem to be spatially organised on a scale of several kilometres) ;
- The calcareous Oxfordian is represented from the base to the top by :
  - ✓ A first low-permeability set of 60 meters (corresponding to C3a-C3b facies) ;
  - ✓ A porous level 50 meters thick, regrouping the four inferior porous horizons (Hp1 – Hp4) with higher permeability ;
  - ✓ A set of 140 meters representing inter-porous compact limestones ;
  - ✓ A second porous level 5 meters thick with high permeability regrouping the three superior porous horizons (Hp5 to Hp7) ; the latter are more distinct based on the verticality. They are represented grouped at Hp7 level ;
  - ✓ A set of 25 meters thickness representing compact limestones.
- The Kimmeridgian is represented by a single, 110 meters thick low-permeability layer.
- The Barrois limestones are represented in two layers :
  - ✓ A lower set of 30 meters in which no karstic manifestation is observed ;
  - ✓ An upper set whose thickness varies based on the topography, but which has a maximum thickness of hundred meters, representing the potentially karstic levels of the formation.

Geochemical properties of the surrounding formations (sorption, precipitation) are negligible, consistent with the fact that they do not have a safety function. We also consider that they do not limit the diffusion, by keeping a pessimistic coefficient (equivalent to that of a water particle diffusing in water).

These representation choices can increase diffusion in the surrounding formations. They are therefore pessimistic vis-à-vis an outlet that could be fed with radionuclides through diffusive transfer. We will see that such an outlet has been retained in the calculations at the Barrois limestones and Kimmeridgian level. Vis-à-vis other outlets fed by advective transfer, the influence of choice of representation is more uncertain. By favouring diffusion, we can slightly reduce the impact of radionuclides transiting through advection, which are « diverted » from their advective path by simplified diffusion. A sensitivity study is therefore conducted to ensure that this phenomenon was not quantitatively important. This study takes into account diffusion coefficient values weaker than those taken as reference in layers where such choice seems reasonable, that is, the least permeable, the C3a layer of the Oxfordian and the Kimmeridgian.

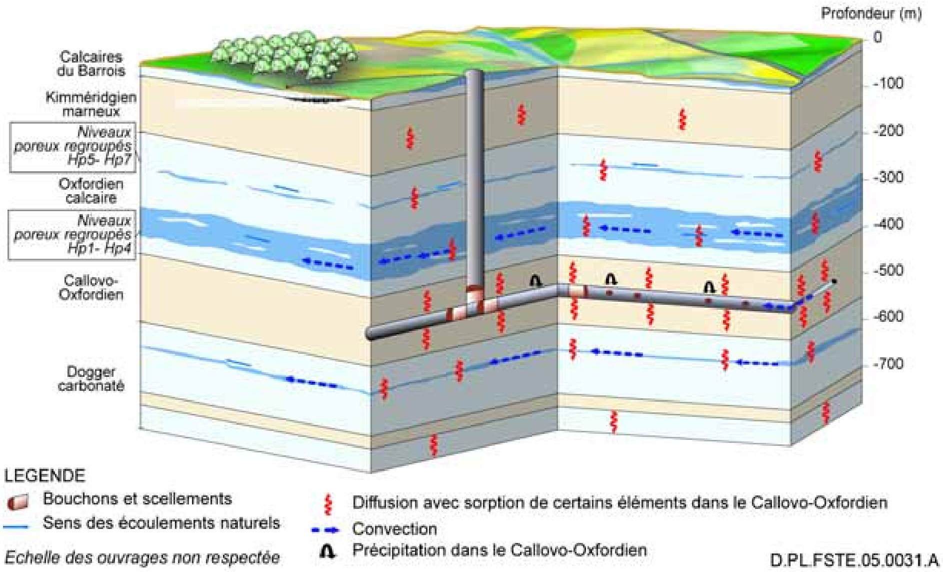


Figure 5.3-14 Representation of surrounding formations and properties of employed containment

5.3.2.5 Selection of outlets

- **Principle for selection of outlets**

Outlets are represented by water pumping operations at a level where an individual from a critical group collects radionuclide contaminated water for his drinking needs or agricultural operations. The selection of outlet locations within each hydrogeological model obeys a rationale comprising :

- preferring outlets closest to the laboratory site of Meuse / Haute-Marne (default repository location, adopted conventionally) in zones where the water flow is sufficient to plan water pumping operations even if they are improbable ;
- preferably choosing, at the level of each of these zones, pumping in the first aquifer or porous horizon encountered, even if it is relatively deep, that is :
  - ✓ primarily, the superficial aquifers at the level of outcrop layers (especially Barrois) or water courses, given that we do not take into account the dilution rate associated with their own water flow rate (we consider pumping close to the water courses and not in the water courses which would be more probable but less damaging) ;
  - ✓ or else, in the first porous horizon encountered ; in some cases, and mainly for the million-year hydraulic model where the valley bottoms were carved by erosion through the total thickness of the Kimmeridgian, the upper porous level of the calcareous Oxfordian (Hp5-Hp7) can be outcrop in some zones ;

This approach is consistent, even though pessimistic, with that of RFS.III.2.f that specifies that « the outlets will comprise water courses and shallow pumping for water supply ». Andra has nevertheless chosen not to retain those water courses where the radionuclide activity is more diluted.

- **Selection of outlets based on the Hydrogeological model of the sector**

This logic for selecting outlets is applied to each of the two Hydrogeological models that were prepared for the Dossier 2005, that for a million years and the current. We will start by the million-year model.

The « million-year » model

The hydraulic model prepared from million-year head fields reveals several outlet zones, for a repository arbitrarily located at the underground laboratory location (see Figure 5.3-15).

For the hydraulic model in overlying formations, we identify three potential outlet zones [72] :

- an outlet positioned in the Ornain valley. The outlet is assimilated to a source or a pumping in the porous interval representing Hp5-Hp7, showing favourable properties for extraction at this place (improvement of permeability linked to outcropping) ;
- an outlet at the level of Marne/Poissons faults, located slightly to the West of this zone (starting approximately at the valley of La Saulx) at the plume maximum. We have taken into account the results obtained on the EST 321 bore-hole, where water ingress was observed in the Oxfordian. At this place and along the entire direction parallel to the Marne fault, the model takes into account the existence of a « diffuse fracturing zone », which as of now is purely hypothetical and whose permeability is voluntarily considered as highly degraded. This zone is supposed to start exactly in the downstream of the EST 351 bore-hole, where the observed water ingress was very weak. It is located outside the transposition zone. An outlet is placed exactly at the entry of the diffuse fracturing zone so as not to include any pessimistic transfer beyond it. For numerical modelling reasons (see insert 5), we have adopted a transfer approach, which consists of « driving » the entire radionuclide mass circulating in the overlying formations and arriving towards the diffuse fracturing zone towards conventional water pumping ;
- an outlet by pumping water from the Barrois limestones. For the latter, the characteristics to be taken into account mainly depend on the exact position of the plume with respect to the position of the facies that may supply a water resource ;
- for the hydraulic model in the Dogger, no realistic outlet could be determined in the study zone. We have chosen to place an outlet at the same level as the Saulx outlet, and force flows in the direction of this outlet close to the site. The characteristics of the Dogger however strongly limit the expression of this outlet : we will see that only 0.03 % of the mass leaving the top of the Callovo-Oxfordian achieves it, the remaining migrates to the Dogger's bottom through diffusion.

We will note that the choice of retaining outlets close to the site limits the influence of the Hydrogeological model on impact calculations. Especially, the outlet in the diffuse fracturing zone hypothesis is at five kilometres from the site and two kilometres from the closest extremity of the repository and becomes independent of all the uncertainties concerning the closeness of the Marne faults.

« Current » model

For this model identical trajectories and outlet are chosen in the Dogger and prove to be as conventional as in the case of the 1 million-year model. On the other hand, the head fields induce different trajectories in the overlying formations. There do not appear to be any trajectories directed towards the valley of Ornain however, a part of the flow is directed towards North-North West. These trajectories are « regional » in part, that is, they do not encounter any zone where an outlet can be defined based on the previous rationale (no natural outlets, no identified fault zones). Another part joins the confluence of the Marne and the Rongeant, by traversing the diffuse fracturing zone. We therefore use the following outlets (see Figure 5.3-16) :

- an outlet at the « diffuse fracturing zone » level, close to the Saulx defined in the same manner as in the million-year model ;
- towards the North-West, a second outlet at the diffuse fracturing zone level, intercepting all the trajectories directed towards the Marne along the entire height of the surrounding formations (supposing the presence of pumping at this place). We will note that this outlet is conventional, since nothing would incite an individual to position a well at this place ;
- an outlet by pumping in the Barrois limestones or Kimmeridgian marls, at the outcropping consistent with the one used in the 1 million-year model.

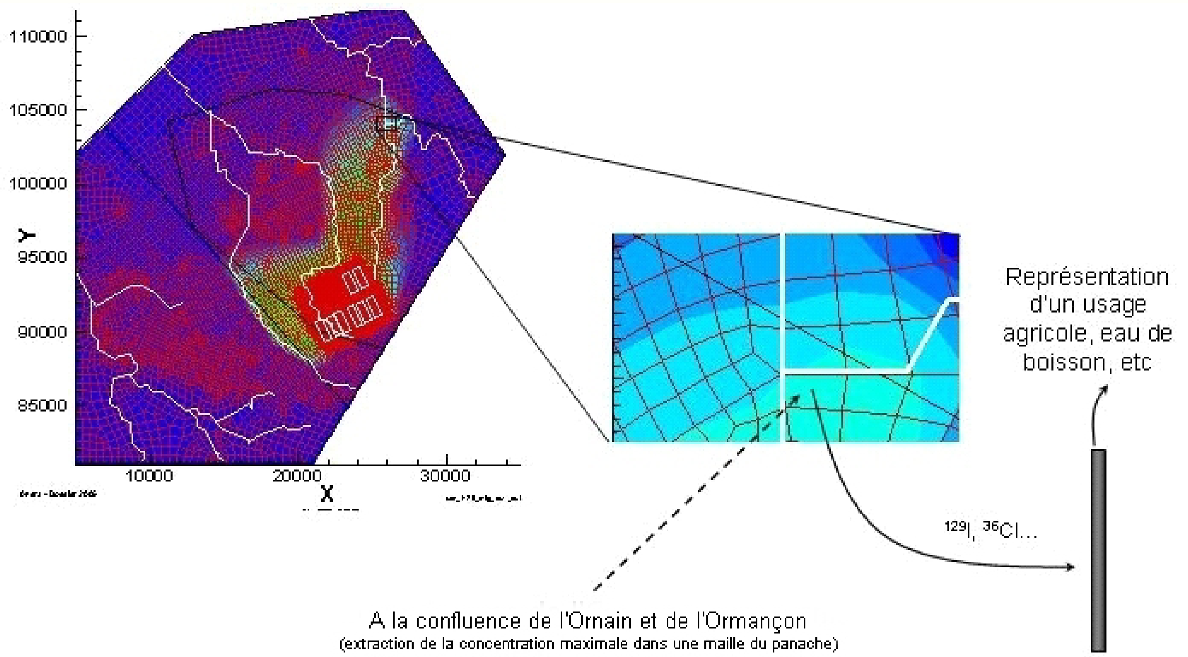
Insert 5 From the Hydrogeological model towards impact calculations

The Alliance platform represents the surrounding formations using a three dimensional mesh. Transport calculations let you determine the matter quantity flows (for each radionuclide) entering and leaving a mesh, water flows entering and leaving and the radionuclide concentration at the mesh level. The model mainly ensures conservation of matter, by ensuring that the incoming and outgoing flows for each mesh are equal.

For calculating transfers up to the critical individual, the biosphere model requires that the concentration of radionuclides in water that is used by the population for its domestic and agricultural use be known. This calculation mode is used to dispense a hypothesis over the size of the critical group : the total quantity of « available » radionuclides in water of the surrounding formations and the manner in which they are distributed between different potentially exposed individuals does not enter the picture. The individuals are all subject to uniform concentration, either directly in their drinking water or through contaminated food that they may consume.

The link between the two models is ensured by supposing that the radionuclide concentration of water used by the critical group of individuals is the concentration calculated in the surrounding formations, at the location where we decided to place an outlet. We select the maximum concentration mesh in a given zone (see figure below).

In such an approach, we neglect the influence that pumping may have on flow in the aquifer and the dilution of radioactivity that it might induce. The radionuclide concentration in pumped water is deemed to be influenced only by the Hydrogeological model and transport in the surrounding formations, and not by the pumping rate.



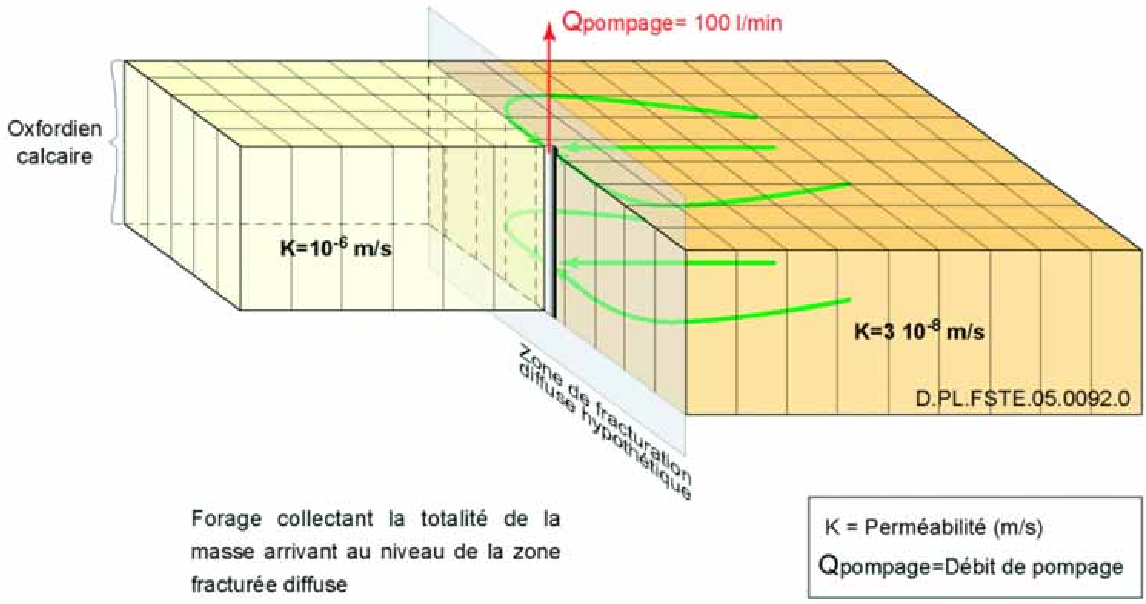
Radiological impact calculation mode at the Ornain outlet

This calculation mode however poses a specific difficulty in the case of the Saulx outlet in the « diffuse fracturing » zone. Pessimistically speaking, this zone was represented by a sudden change in permeability. When entering the zone from East, the model modifies the permeability, shifting from what was observed for the Oxfordian (3.10^{-8} m/s) to a very high value of 10^{-6} m/s. This choice induces a discontinuity in values of radionuclide concentrations. Upstream of the zone, the concentration is stronger, but it corresponds to a place where water flows are not suitable for pumping. Downstream of the zone, the concentration falls, since it is diluted in water circulations of the high permeability zone. The choice of locating the outlet at this place is questionable, since the evaluated dose here would be very weak due to dilution brought in by the model.

For these specific outlets, Andra has thus chosen a pessimistic model, that will give an over-estimation of the impact by another method. Since concentration is a numerically unstable parameter, we have rather decided to base ourselves on the flow of radionuclide quantity entering in the diffuse fracturing zone, which is conserved while traversing the permeability discontinuity. The complete flow, along the entire height of the surrounding formations, is assumed to be captured by an outlet positioned at any place in the neighbourhood of the zone. For calculations in the biosphere, to convert this quantity of matter into concentration, it must be divided by a pumping rate. We have used a flow rate of 100 litres per minute, perennial in time, which corresponds :

- to the observed flow rate range, over a short period, in the EST 321 bore-hole ;
- to a low value with regards to flows corresponding to pumping of potable water supply at such a depth. Existing pumping operations in similar formations were surveyed in the departments of Meuse and Haute-Marne using a sub-surface database from the Bureau de Recherches Géologiques et Minières and communal files to ensure that the flow used for calculations was a low value [59].

This approach is very pessimistic since it drives all the contamination circulating in the surrounding formations back to a well. In the Hydrogeological model, this is distributed over a zone more than a kilometre in length. No well, especially with such weak flow, seems capable of having such an influence on the flows.



Calculation mode of radiological impact for outlets located at the hypothetical diffuse fracturing zone

This calculation model is applied for the outlets in the Saulx valley and upstream of the Marne-Rongeant intersection.

Table 5.3-18 gives a summary of potential outlets used for 1 million-year and current hydraulic models. Figure 5.3-15 and Figure 5.3-16 give the trajectories of water flows and potential outlets at the site level.

<table>
<tr><th>The 1 million-year hydraulic model</th><th>Hydraulic model representative of current head fields</th></tr>
<tr><td colspan="2">« Saulx » outlet at the diffuse fracturing zone level</td></tr>
<tr><td colspan="2">Pumping in the Saulx valley intercepting the entire radionuclide flow arriving in the diffuse fracturing zone and associated with a pumping flow of 100 litres / minute</td></tr>
<tr><td>« Ornain » outlet</td><td rowspan="2">Not applicable</td></tr>
<tr><td>Pumping in the Ornain valley, in porous interval representing Hp5-7 at the place where the concentration is maximum</td></tr>
<tr><td rowspan="2">Not applicable</td><td>« Interception of trajectories towards Marne » outlet</td></tr>
<tr><td>Pumping, intercepting the entire radionuclide flow arriving in the diffuse fracturing zone and associated with a pumping flow of 100 litres / minute</td></tr>
<tr><td>« Barrois » or « Kimmeridgian » outlet
Pumping in the Barrois limestones or in the Kimmeridgian outcrop at the place where concentration is maximum</td><td>« Barrois » outlet
Pumping in the Barrois limestones at the place where concentration is maximum</td></tr>
</table>

Table 5.3-18 Summary of potential outlets used for « 1 million-year » and « current » hydraulic models.

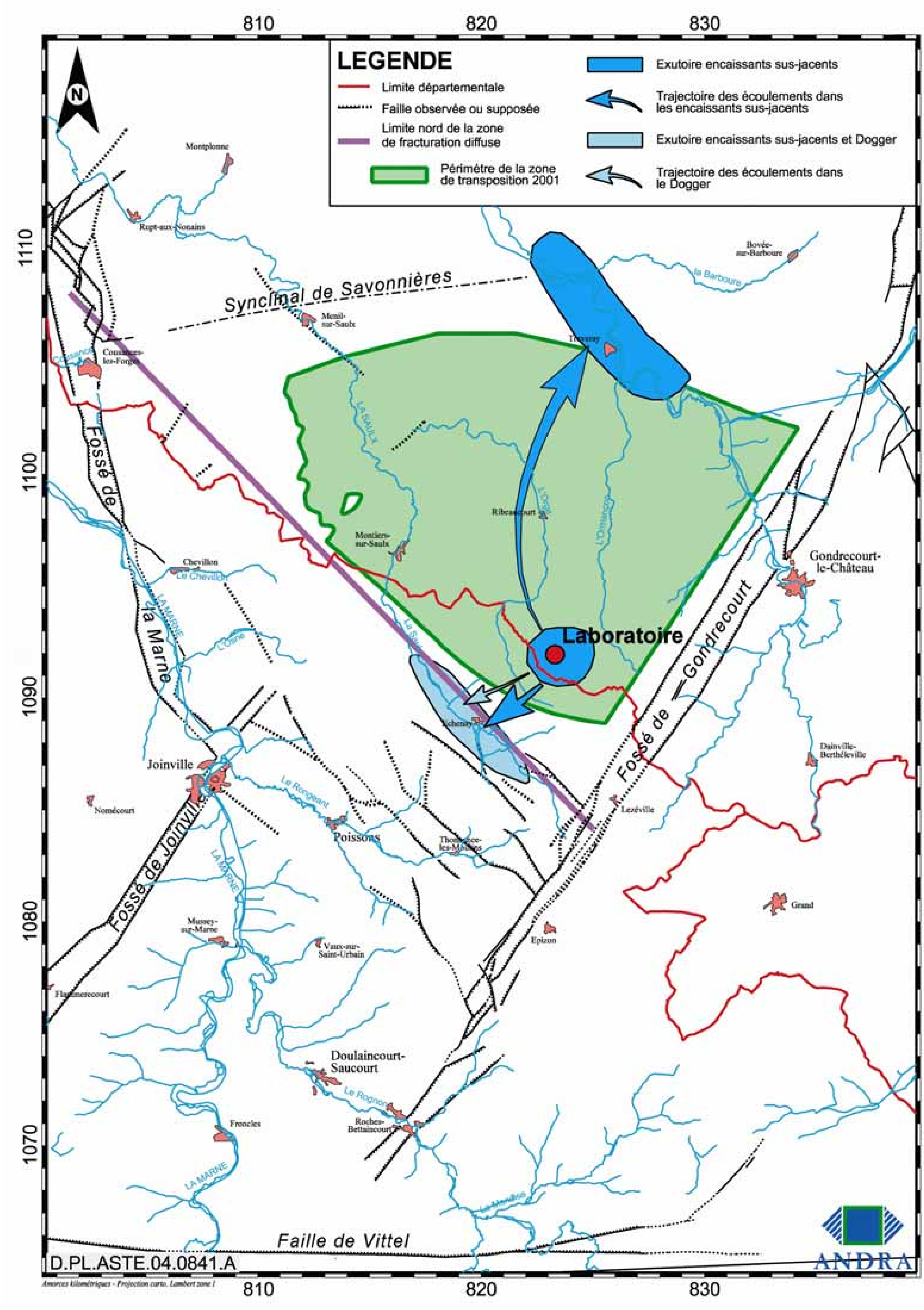


Figure 5.3-15 Trajectories of flows and potential outlets at the site - 1 million-year hydraulic model

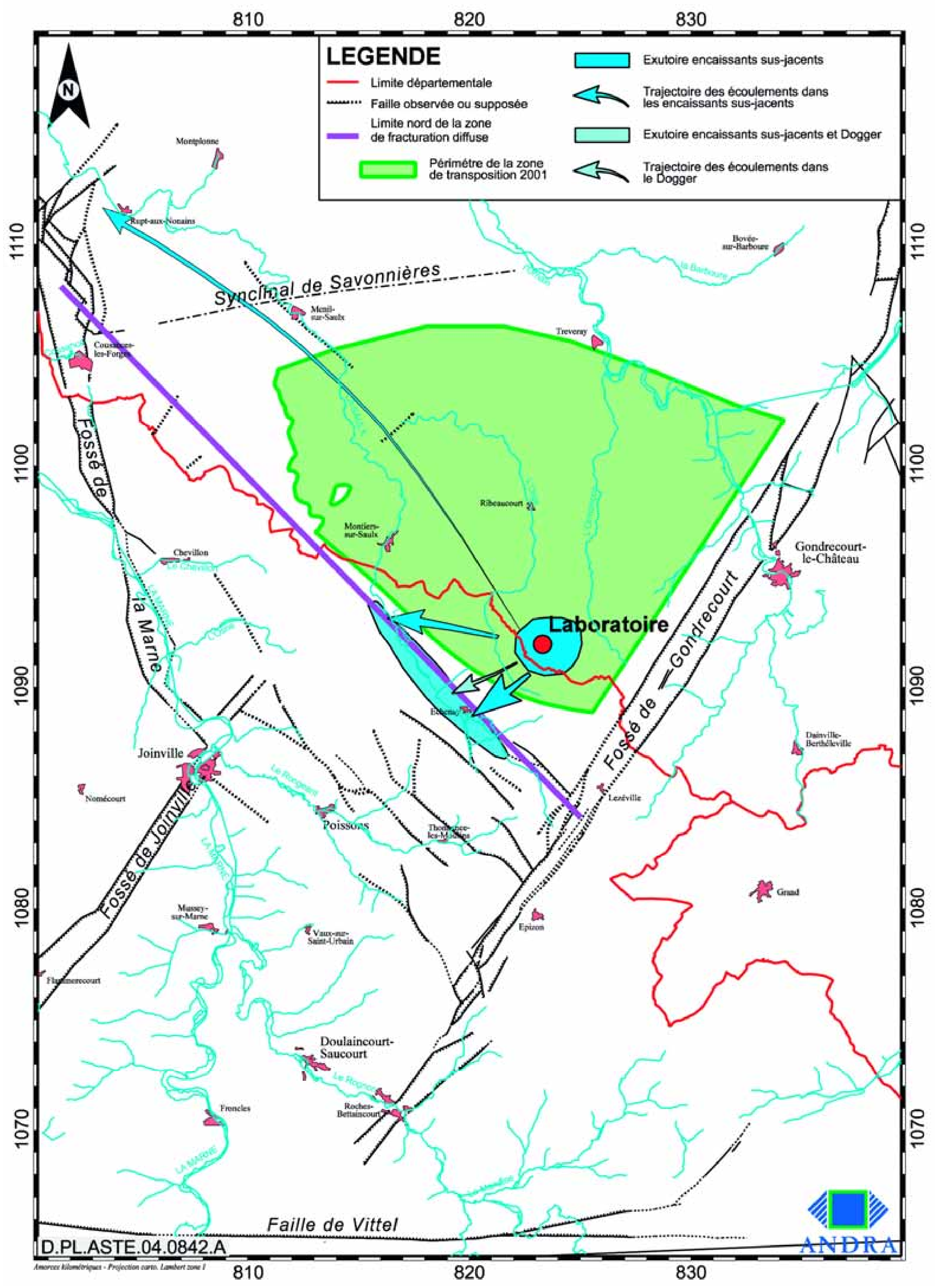


Figure 5.3-16 Trajectories of flows and potential outlets at the site – current hydraulic model

5.3.2.6 Representation of the biosphere

The biosphere is defined as the part of environment easily accessible to human activities and liable to be a transfer path of radionuclides (and possibly toxic chemical elements) up to the potentially exposed individuals. Its modelling is based on the knowledge of the behaviour of chemical elements in the environment, as well as on the daily habits of the population living in the concerned region. It allows calculation of doses in humans generated by the different exposure pathways (external exposure, inhalation, ingestion).

This approach aims to define a reference group whose daily habits, pertinent in the context of the site, expose it to contamination present in the aquifers. These groups are defined by their age, their lifestyle and their eating habits. Foreseeing lifestyles in several hundreds of thousands of years is not realistic. Therefore, in line with the ICRP 81 recommendations, RFS III.2.f recommends basing oneself on the current habits of the populations. Nevertheless, to cover uncertainties and variability of the possible lifestyles, we have chosen to use the group liable to be subjected to maximum exposure (notion of « critical group »). To do this, we have used a group of farmers living mainly from their own harvest and drinking water from their own wells : drinking water, irrigating a vegetable garden, watering and raising livestock from their own cereal harvest [73]. In fact, such a group is subject to of the greatest number of exposure pathways from radionuclides present in water. Food consumption arise from INSEE's surveys [74]. In chapter 6, we will find an analysis of main uncertainties pertaining to the definition of this critical group mainly envisaging other types of reference groups and other food habits including complete self-sufficiency. We will see that these uncertainties do not question the calculation of doses presented in this chapter.

The water resource that feeds the critical group comes from a well. If it were using river water (Saulx for example), possible flooding of the prairies could lead to an additional path of exposure in some scenarios consisting of livestock (by their feed contamination). On the other hand, radionuclides would be much more diluted in such an outlet. The selection of a « well » type outlet is therefore conservative for the radionuclides with an impact (mainly iodine 129 as will be seen later).

However, we note that the size of the critical group and the associated agricultural operations are not taken into account in the calculations. In fact, individuals are deemed to have used the contaminated well water for drinking, feeding the animals, irrigating their vegetable garden and cultures without ever using up the resource. The radionuclide concentration in water of the outlet is deemed constant, for any number of persons using the well, the radionuclide plume originating from the repository « re-supplies » the well as needed. This conservative approach means the size of the exposed group does not have to be dealt with.

We will note that the critical group is made up of adults. Andra also tested the possibility that children or infants would be exposed. Even though young individuals are more sensitive to radiation effects, their food consumption is less, and it appears that using an adult is a conservative choice [76].

The critical group lives in temperate climate, equivalent to that prevailing these days. We have seen earlier that the climatic changes that will happen will be marked by alternate temperate and cold or even glacial periods. The glacial periods should be less pessimistic from the view point of the critical group : cold climate incites the population to lead a nomadic life and leave the valleys to go and live on plateaus, thus increasing dilution and reducing periods of exposure. This will then distance the population of the Saulx or Ornain valley zone. As a precaution measure, Andra has studied groups living in transient manner in repository environment, in cold conditions [73]. Such groups seems less pessimistic in terms of dose than the group of sedentary farmers in boreal type cold conditions. This is therefore definitively a conservative choice, in cold or temperate climate.

5.4 Calculation architecture

The architecture implemented to evaluate the various safety scenarios is explained in document [75] which details in particular :

- the constituents of the Alliances platform through which the safety calculations have been made. This platform has already been briefly presented in section 1.5.3.5 of chapter 1 ;
- the numerical tools and methods used to guarantee the robustness of the calculations ;
- the numerical processing adopted for each safety scenario, by specifying the overall model and the various compartments of the calculation used to study the behaviour of the system with regard to hydraulics and the transport of radionuclides from the waste packages to the outlets.

Following a short presentation of the tools used in section 5.4.1, the approach adopted to reduce numerical errors and validate the results is discussed in greater detail in section 5.4.2. Finally, the numerical processing logic for safety scenarios is presented in section 5.4.3.

5.4.1 Alliances platform

In 2000, Andra entered into a discussion with a view to preparing the simulation means required to model the repository and conducting safety calculations as part of its assessment file. Thus the Alliances project was born, a project common to Andra and the CEA, and joined later by EDF, for the development of a simulation platform especially authorising :

- the selection and coupling of various numerical components,
- simulation of multi-physical (hydraulic, transport, chemical, mechanical ect) and multi-scale (package, disposal cell, repository, geological medium) phenomena,
- sensitivity analysis and uncertainty assessment,
- study management and traceability.

Thus an efficient, user-friendly simulation environment was built which, in addition, guarantees the modular, open-ended aspects of the software environment. Alliances enables the user to integrate computer codes from various origins and couple the various phases of calculation within a single environment.

The Alliances structure is centred on a common data model forming a pivot of communication between all of the calculation modules. This data model contains characteristic values (geometric, physical and numerical) used for modelling and simulation. It associates physical properties to the geometric characteristics of the investigated item.

For a single physical model (module), Alliances can take several computer codes (components) of various origins, combine them to concatenate various phases of calculation and produce couplings. This is done for a database independent of the components used.

The development approach is based on a description of the phenomena governing the behaviour of the repository constituents in the various phases of its life. These models reflect needs in terms of modelling and simulation. The models as a whole must cover all of the situations analysed.

The following are associated to a reference model :

- physical phenomena (hydraulic, thermal and mechanical) with their equations,
- couplings between the phenomena,
- spatial and temporal characteristics of the model,
- computer codes liable to meet modelling and simulation requirements,
- reference cases corresponding to implementation in the Alliances platform based on an objective (performance or safety).

By analysing the reference models, it is possible to :

- characterise and select computer codes,
- define and make couplings,
- define qualification and validation cases for the Alliances modules implemented.

For each model, several numerical components can be selected to reinforce the reliability of the studies conducted by Andra through comparative studies (benchmarking).

The modelling capabilities of Alliances version 1 are presented in the table below.

Modèles	***Composants***	***Exemples d'application***
Hydraulique - saturée - insaturée - transitoire	Castem Porflow Traces	- écoulement dans des milieux poreux - resaturation des alvéoles de stockage
Transport - simple - étendu	Castem Porflow Traces	- transport de contaminant (toxiques chimiques ou radio éléments)
Chimie - complexation - échange d'ions	Chess PhreeqC	- lixiviation - perturbation alcaline - perturbation oxydante - vieillissement - relâchement
Transport Chimie	Castem MT3D Traces ⇨ Chess PhreeqC	
Colis - verre - bitume	Prediver Colonbo	- dégradation des colis - relâchement
Sensibilité - échantillonnage - analyse	LHS Kalif Pastis	Sensibilités aux caractéristiques hydrauliques et de transport pour les évaluations de sûreté

The aim of Alliances version 1 within the Dossier 2005 was to provide design engineers with a tool capable of performing safety assessments of a possible repository in a deep geological layer. Its development included the main safety calculation characteristics :

- a large amount of data,
- complex calculation sequences involving different models,
- a large calculation volume (several thousand simulations),
- a requirement to control data and results.

From early 2004, Alliances has been used operationally by Andra and its partners for the purpose of these calculations and to conduct performance studies.

On the whole, this work represents several thousand calculation cases [75]. The safety calculations completed in 2004 for the 2005 dossier have made it possible to verify the quality and robustness of the tool as a minor number of incidents were noted during calculation.

The development of version 2 will enrich Alliances' modelling capabilities with the integration of modules dealing with thermal and mechanical aspects and their coupling with hydraulic matters.

In the future, the platform could be used with the purpose of a multi-physical simulation. It would then be a complete modelling, analysis and design tool for repository study which could provide support in a development or construction phase.

5.4.2 Result validation and numerical error reduction process

Quantification of numerical safety simulations as precisely as possible to guarantee the robustness of the calculation is based on a four-stage numerical process defined below.

5.4.2.1 Construction of the safety calculation architecture

The construction of the safety calculation architecture conveys the way in which the various calculation scenarios have been represented from a physical and numerical point of view.

This architecture, broken down into compartments of various scale arranged in sequence (see section 5.4.3), considers the characteristics of the safety scenarios (situations, indicators and objectives), the physical and geometric data and models of the various materials, the restrictions inherent in the codes (maximum mesh size, physical features, etc.) and a set of preparatory calculations justifying the physical and geometric simplifications. In particular, it can differ from one scenario to another according to the situations to be represented. Thus, for altered evolution scenarios (see chapter 7), the repository structures are generally represented more finely.

5.4.2.2 Choice of simulation tools

Prior to the calculation phase, the qualified and validated Alliances simulation tools were tested and compared on representative test cases of the 2005 dossier, based on physical and numerical criteria : numerical performance, availability of the physical features, choice and type of solvers, types of meshing, RAM required for the calculations, interface with post-processing tools, software support and experience.

This exercise led to the choice of codes offering the best performance with regard to the various criteria listed above.

5.4.2.3 Choice of numerical parameters

The choice of numerical parameters is designed to minimise numerical error and guarantee the robustness of the calculation. It consists of finding, for each calculation, the best compromise between spatial discretisation, temporal discretisation, the capacity of the solver and the accuracy target to obtain satisfactory convergence of the calculations.

Spatial discretisation (mesh size, shape and arrangement) takes account of several constraints : geometry of the domains to be modelled, kinetics of the physical phenomena (compliance with the Peclet number in a mesh to minimise numerical diffusion), contrasts in the physical parameters, mesh distortion and flattening to be respected and maximum number of meshes dictated by the computing environment (maximum of 750,000 meshes).

Temporal discretisation (time steps and time step progression) is adapted to the physics to be represented (nature and speed of transfer, release from the source). Sensitivity to the time step is systematically studied to ascertain the robustness of the result.

The matrix preconditioners and the solvers are then tested to identify the most efficient and accurate resolution methods.

Once the numerical data set has been established, a final meshing sensitivity test is conducted to ensure that the selected set is satisfactory.

For each calculation, the ratios between the numerical mass created and the input and output mass are systematically assessed per subdomain and in the global domain, while ensuring that a ratio of less than 0.001 is observed. Other consistency checks are conducted, for example by comparing the results of the simulations with simple analytical calculations in order to contribute to validating the numerical result of the calculation.

5.4.2.4 Subsequent intercomparison of codes

The Alliances codes (Porflow, Castem, Traces) are subsequently intercompared on calculation cases representative of the Dossier 2005 Argile. The various intercompared calculation cases have boosted confidence in the results obtained and justified the choice of codes.

5.4.3 Numerical processing logic for safety scenarios

The representation of all scales (near field and far field) within a single simulation is not accessible with current hardware. Compartments with interlocking geometric scales, arranged in sequence, have therefore been implemented. A calculation compartment takes account of the effects of the upstream and downstream compartments without representing them.

Each compartment has a specific geometric scale and modelling methodology. In each one, one or more steady-state hydraulic calculations are made, then transient-state transport calculations up to a million years. As a general rule, the compartments are sequenced in the opposite direction for hydraulics and for transport (see Figure 5.4-1).

- For hydraulics, we begin with « regional » modelling, the limits of which correspond to physical limits (Gondrecourt rift and the Marne and Ornain valleys), and then the calculations are carried out on increasingly restricted domains drawing their conditions at the limits (hydraulic head or flow rate) of the preceding model.
- On the other hand, for transport calculations, we begin with radionuclide release zones which are generally small, or even punctual, and end at modelling covering the entire geological medium, with the sequencing of the compartments managed by molar flow rate or concentration history.

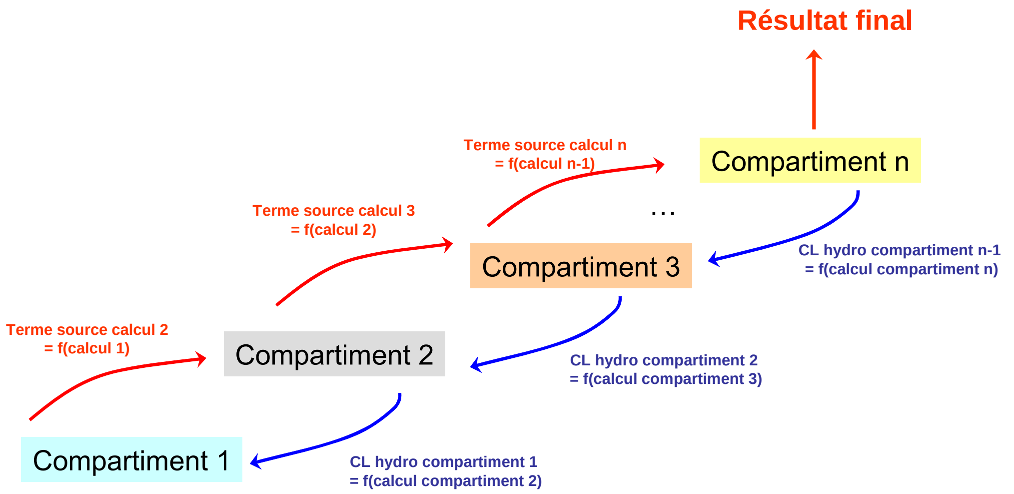


Figure 5.4-1 Sequencing of hydraulic and transport calculations between the compartments (CL : limit condition)

For each scenario, two transfer pathways are systematically quantified (see Figure 5.4-2) :

- transfer through the sound geological barrier[68] – in the SEN (reference calculation), this is in principle the preferential radionuclide transfer pathway as we shall see ;
- transfer through the engineered structures, from the repository zones to the access structures. This transfer pathway can either be :
  - ✓ calculated beforehand with the principal radionuclides to verify that it is still weak as compared to the transfer pathway through the geological barrier. This is notably the case for the SEN reference calculation,

[68] This term designates the host formation in the context of the performance calculation.

- ✓ or systematically quantified for all radionuclides in configurations which are more restrictive in relation to this transfer pathway (EDZ sensitivity of the normal evolution scenario and « sealing failure » and « very degraded operation » altered evolution scenarios – see chapter 7 for the latter two cases).

The models adopted for each of these transfer pathways are detailed in sections 5.4.3.1 and 5.4.3.2 respectively.

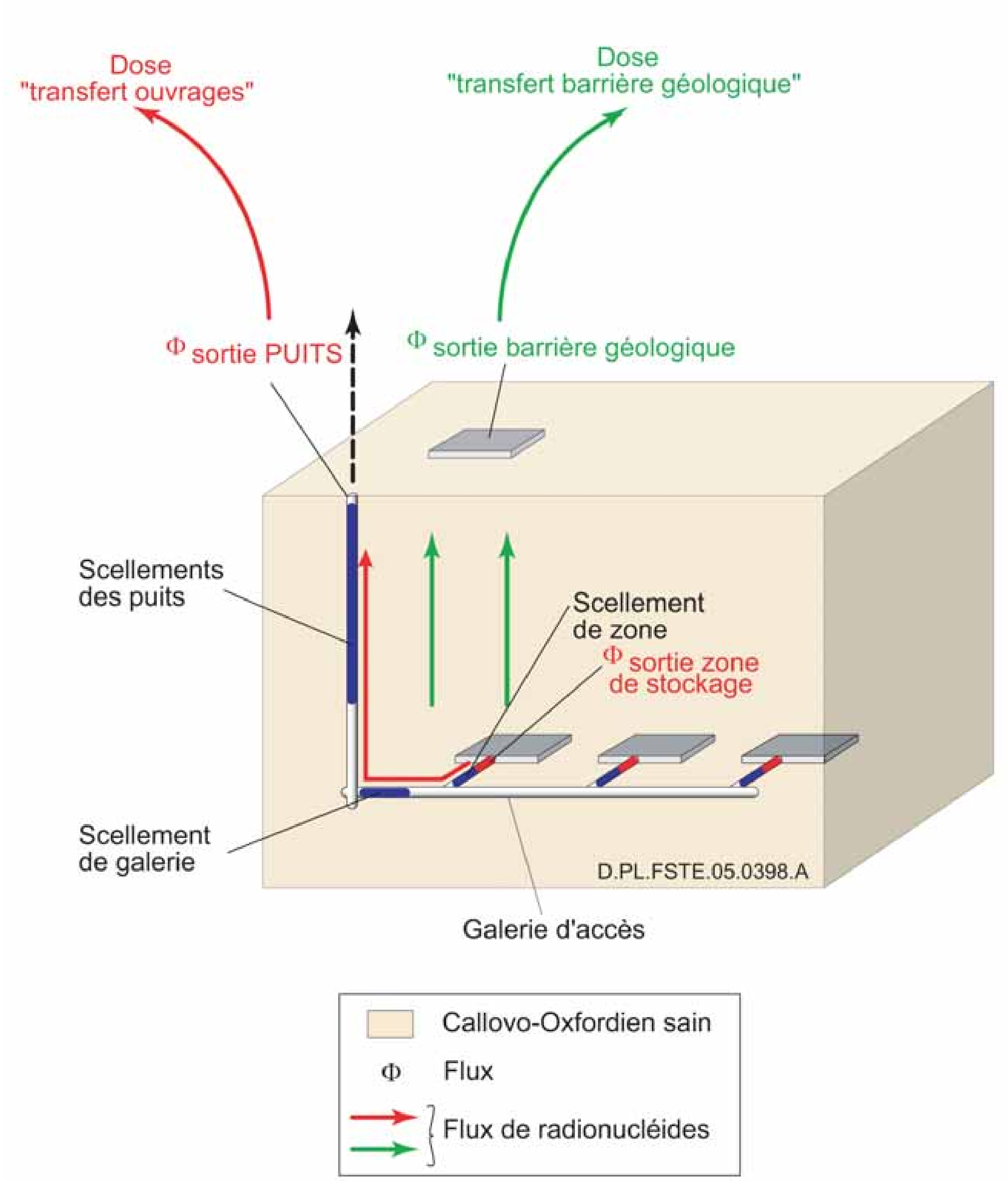


Figure 5.4-2 « Geological barrier » and « engineered structures » transfer pathways

5.4.3.1 Transfer pathway through the sound geological barrier to the biosphere

The numerical processing of this transfer pathway comprises five sequentially arranged calculation compartments (see Figure 5.4-3) relating to :

- the near field in which the « waste packages » compartment then the « disposal cells » compartment are distinguished,
- the far field in which the « repository » compartment, the « geological medium » compartment where transfer in the surroundings to the outlets (overlying and Dogger) is modelled, and finally the « biosphere » compartment are represented.

The compartments adopted for modelling the transfer through the geological barrier do not represent the access structures : we use a conservative configuration where the entire mass is forced through the geological barrier.

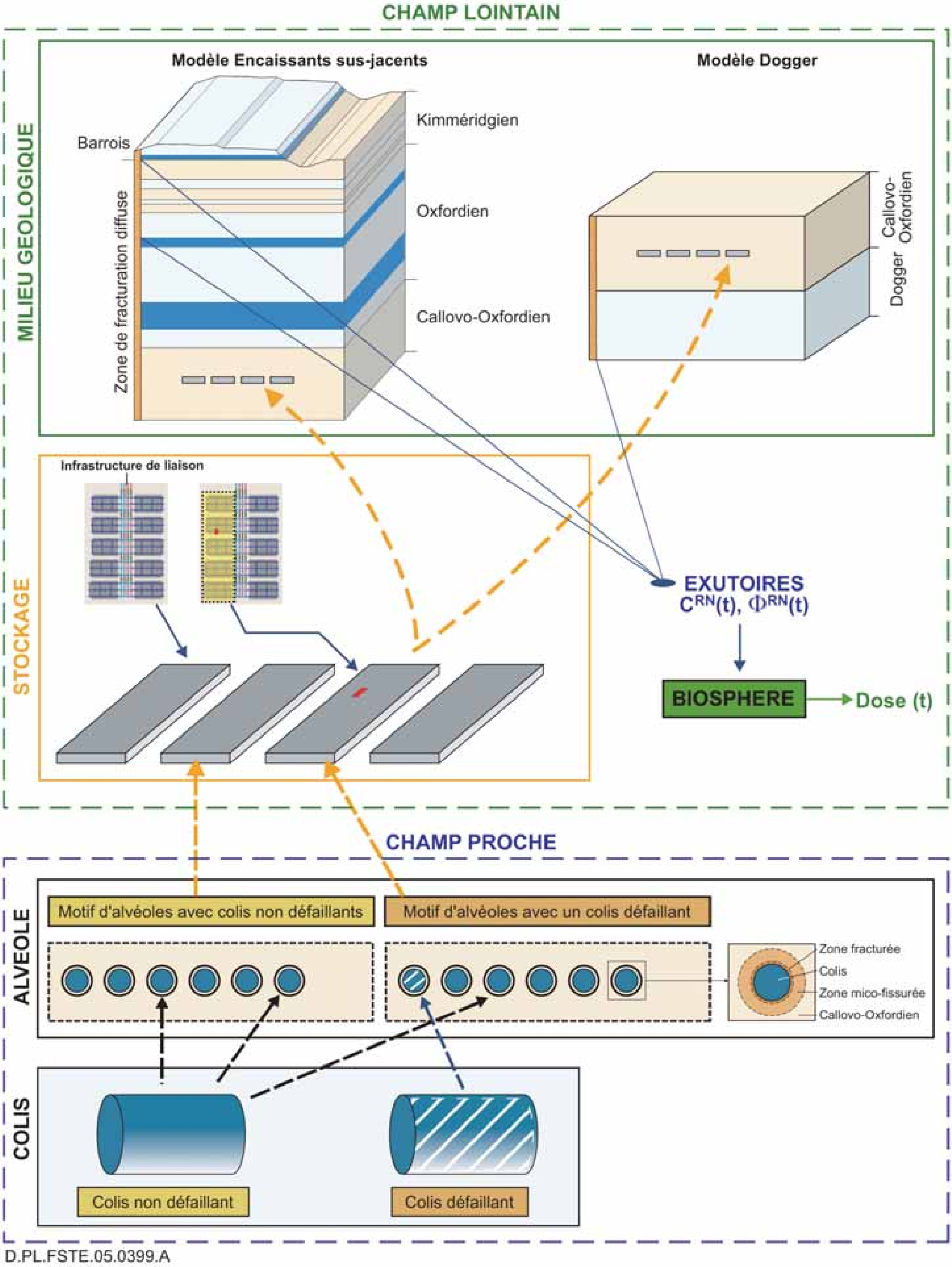


Figure 5.4-3 Calculation diagram for the transfer pathway through the sound geological barrier

The approach followed for each of the compartments is summarised below.

- **Near field**

« waste Packages » compartment

The « waste packages » compartment represents the models of radionuclide release by the disposal packages. For each reference package, the release of radionuclides is modelled by a release function (molar flow rate) of each physical and chemical subassembly inherent to :

- the alteration properties of the waste itself,
- radioactive decay and filiation,
- the time delay before water reaches the packages (consecutive to the time for the spent fuel containers and C waste over-packs to lose leaktightness).

« Disposal cells » compartment

The « disposal cells » compartment provides the molar flow rate entering the geological barrier for the compartment at the next scale up (repository). To model this compartment, a six-cell calculation pattern representative of all cells is adopted. The cell pattern not containing any defective packages is differentiated from the one containing one or more depending on the scenarios and the reference packages.

The cell is modelled by a series of equivalent porous media saturated with water from the first instant : stack of packages, plug and cell core. It is represented in its environment : the Callovo-Oxfordian, sound over its entire depth, and the EDZ (fractured zone and microfissured zone).

The molar flow rate out of the packages is the input indicator for the disposal cell model. The output indicator used as input data for the « far-field repository compartment » calculation is the molar flow rate out of the repository, which is represented by a parallelepiped approximately 15 metres high in the centre of which the cells are positioned. The same parallelepiped is then used for the far-field calculation, with an identical size.

- **Far field**

« Repository » compartment

For B waste, each cell is considered explicitly as a « calculation module ».

For C waste and spent fuels (see Figure 5.4-4), each « calculation module » is a grouping of repository half-modules at least fifty metres apart (adjacent along a dead-end side). It is represented by the envelope parallelepiped containing its constituent cells. One of these calculation modules comprises a zone representing a cell with a defective package in the centre of the calculation module. The adopted module containing the defective package is the nearest one to the outlet on which the impact is greatest.

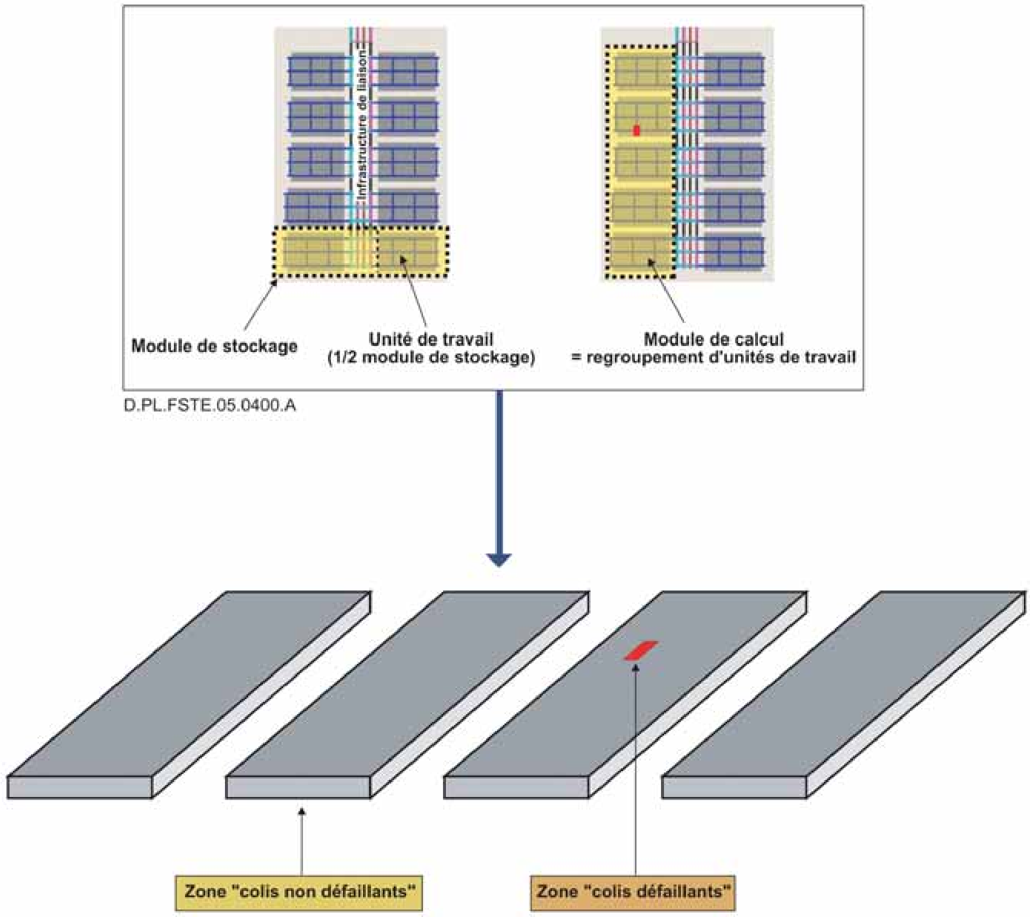


Figure 5.4-4 Principle of representation of « calculation modules » for the « repository » compartment

The molar radionuclide flow rate out of the repository (15-metre high parallelepiped collecting the releases from six disposal cells), calculated on the cell scale, is the input indicator for the far-field « repository » compartment. When multiplied by the number of cells contained in a calculation module, the molar flow rate out of the repository is reinjected into the volume of the « calculation module » (see Figure 5.4.3 and Figure 5.4-4) for far field. This method of representation avoids hydraulic discontinuities. In order to avoid counting the transfer twice in the first few metres of Callovo-Oxfordian (once in the « cell » compartment and once in the « repository » compartment), the transport parameters adopted in the calculation module of the near-field model are selected such that transfer times are minimised here. Therefore, a pessimistic water-in-water diffusion coefficient equal to 2.10^{-9} m²/s is adopted for radionuclides.

The calculation modules are far enough apart for them to be considered independent from the point of view of radionuclide transfer.

« Geological medium » compartment

The « geological medium » compartment is used to model radionuclide transfer from the limits of the formation to potential outlets.

In order to quantify the radiological impacts at the various outlets, numerical simulations are performed in a 3D representation in the geological medium. For the current hydrogeological model as well as for the geo-prospective model at one million years, two distinct calculation models have been processed (see Figure 5.4-5) :

- an « overlying formations », or « Oxfordian », model comprising the host layer and the overlying formations up to the Barrois limestone. In this model, an ascending vertical hydraulic head gradient of 0.4 m/m is applied for the one million-year model and 0.2 m/m for the current model ;

- a « Dogger » model comprising the host layer and the Dogger layer to the upper Bajocian formation, where a descending vertical hydraulic head gradient of 0.2 m/m is applied (a single model is used).

A total of three distinct models are therefore implemented.

As far as the surrounding formations are concerned, each model considers the so-called diffuse fracturing zone (see section 5.3.2.5), the Northern limit of which follows a line of north-westerly orientation running along the Saulx valley, to approximately five kilometres to the south-west of the Meuse/Haute-Marne laboratory. Inside this zone, the surrounding formations have a high level of permeability and are not differentiated.

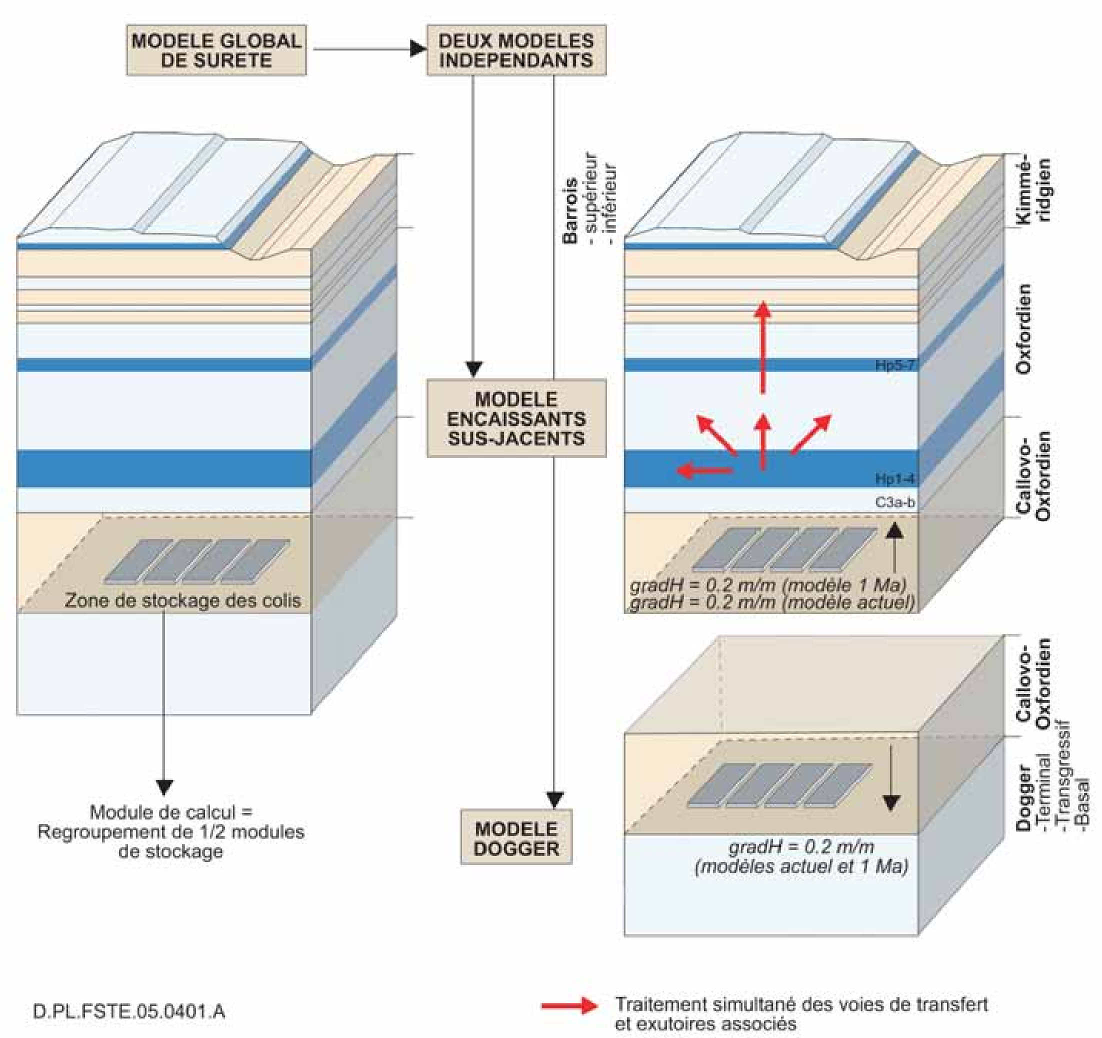


Figure 5.4-5 Breakdown of the global safety model into two sub-models

The logic presiding over the development of the hydraulic safety models is based on :

- the results of the more detailed phenomenological hydraulic models produced beforehand and presented in [72]) ;
- the characteristics of the surrounding formations ;
- safety choices aiming to minimise regional trajectories and reduce transfer times to the outlets (either by selecting artificial outlets close to the site, or by selecting layer characteristics with a view to increasing the flow rates to the outlets).

Therefore, for the Oxfordian limestone safety model, a representation with the closest approach to the flow trajectories of the phenomenological hydraulic model, but increasing the flow rate to the outlets compared with the latter, was selected.

The representation of the overlying formations consisted of grouping together layers which, in principle, have similar hydraulic and transport properties. The layers are represented with constant depths over all of the transposition zone. In particular, the most transmissive porous horizons (Hp1 to Hp4 and Hp5 to Hp7) are modelled with high permeability in two levels of constant depth over the transposition zone even if they may appear to be discontinuous on the sector scale. This choice tends to increase the flow speeds to potential outlets.

Considering the method of representing the diffuse fracturing zone and the continuity of the layers, the head fields in the overlying formations taken into account in the safety models are derived from a specific hydraulic calculation depending on conditions at the limits of the model.

A check was carried out in parallel to ensure that the flow trajectories obtained in the safety models were consistent with those of more detailed phenomenological models.

The Dogger is located on a piezometric plateau and, as a result, the flow trajectories are highly sensitive to the slightest head variations. The development logic of the Dogger hydraulic safety model therefore differs significantly to that of the overlying formations : the choice was made to force the flow directions towards the nearest « Saulx » outlet to the site in order to overcome the major uncertainties concerning the flow trajectories while observing the heads measured in various boreholes (MSE 101, EST 312, HTM 102, EST 322, EST 342 – see figure 5.6.4 of [60]). In this case, as the safety model was based on head values measured in various boreholes, the hydraulic head gradients and the characteristic flow times remain consistent with the data obtained on site.

« Biosphere » compartment

The transfer pathways up to the biosphere are represented in Figure 5.4-6. They take into account the various potential exposure paths. Calculations from the outlet up to the critical individual are carried out according to two different methods :

- for the majority of radionuclides, a so-called model « by compartments » [76] is used, which consists of defining transfer factors from one compartment to the next (for example, from well water to plants by irrigation, and from plants to humans by consumption) ;
- in particular cases, including chlorine-36, the fact that the stable element is present in the environment (in terms of weight) is taken into account. Due to constant exchanges of chlorine between different biosphere compartments, an equilibrium is established. An isotopic ratio between radioactive chlorine and the entire chlorine present is established from one compartment to another.

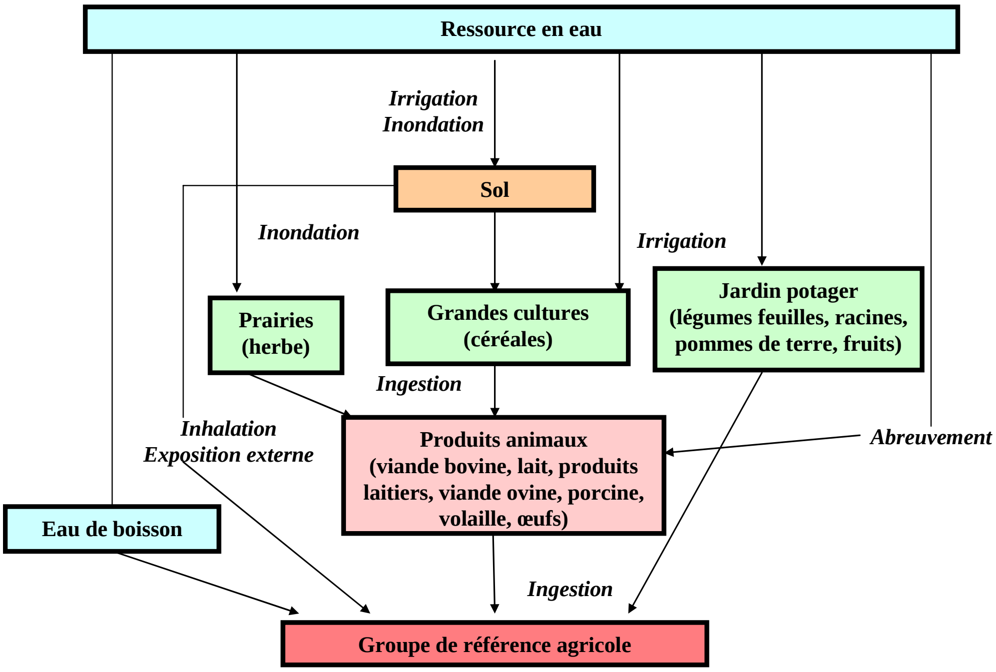


Figure 5.4-6 Transfer pathways taken into account in the biosphere model

Transfer calculations are done using the Aquabios computer code. This was tested and qualified, and compared to other codes, showing good convergence of results [77, 78, 79]. The definition of the transfer factors is derived from the methodology developed as part of the international BIOMASS exercise organised by the IAEA. The values used are regularly compared to the international level (BIOPROTA and EMRAS exercises [80]). These were subjected to sensitivity studies which enabled us to discern the magnitude of uncertainties for important parameters. These studies are presented in chapter 6, in section 6.2.3.

The conversion factors used for the safety calculations are presented in Table 5.4-1.

Radionuclides	Half-life [years]	Conversion factors « Farming community » (Sv/year)/(Bq/l)	Conversion factors « Drinking water » (Sv/year)/(Bq/l)
^{10}Be	1,600,000	$9.75.10^{-7}$	$4.84.10^{-7}$
^{14}C	5,730	$4.70.10^{-7}$	$2.55.10^{-7}$
^{41}Ca	103,000	$9.68.10^{-8}$	$8.36.10^{-8}$
^{36}Cl	302,000	$4.20.10^{-6}$	$4.09.10^{-7}$
^{59}Ni	75,000	$3.88.10^{-7}$	$2.77.10^{-8}$
^{135}Cs	2,300,000	$1.67.10^{-5}$	$8.80.10^{-7}$
^{93}Zr	1,530,000	$8.41.10^{-7}$	$4.84.10^{-7}$
^{93}Nb	16.4	$5.70.10^{-8}$	$5.28.10^{-8}$
^{93}Mo	3,500	$3.93.10^{-6}$	$1.36.10^{-6}$
^{107}Pd	6,500,000	$3.33.10^{-8}$	$1.63.10^{-8}$
^{166}Ho	1,200	$6.04.10^{-5}$	$8.80.10^{-7}$
^{129}I	15,700,000	$5.96.10^{-5}$	$4.84.10^{-5}$
^{94}Nb	20,300	$6.68.10^{-5}$	$5.28.10^{-8}$
^{79}Se	65,000	$1.04.10^{-5}$	$1.28.10^{-6}$
^{126}Sn	100,000	$5.53.10^{-5}$	$2.07.10^{-6}$
^{99}Tc	213,000	$3.53.10^{-7}$	$2.82.10^{-7}$

Table 5.4-1 SEN – Conversion factor values used for the farming community or for drinking water consumption – temperate biosphere

5.4.3.2 Transfer pathway through the engineered structures

The numerical processing of this transfer pathway comprises six calculation compartments (see Figure 5.4-7) arranged in sequence, as for the transfer pathway through the geological barrier, relating to :

- the near field (waste package and disposal cell),
- the far field (repository zone, repository and access structures, geological medium and biosphere).

The sequencing of the calculations at compartment level is summarised below. The following are carried out :

- at near-field level, a calculation representing the cell (« cell » calculation), its disturbed immediate environment and the repository drift liable to constitute a potential transfer pathway. The source term considered in the calculation is that of the « package » compartment which is identical to the one described in the preceding section. Following this calculation, the molar flow rate entering into and remaining in the sound geological barrier (ΦGB) and the molar flow entering into and remaining in the drift (Φdrift) were assessed :

- at far-field level :
  - ✓ a calculation representing a repository subzone (« repository subzone » calculation), considering all of the disposal cells in the subzone, the access and connecting drifts, their EDZ and the module seals. Insofar as advection may be the prevailing mode of transport for this transfer pathway, the surface of each calculation module (parallelepiped) corresponds to the « actual » exchange surface between the cells contained in a calculation module and the Callovo-Oxfordian. The input data is the molar flow rate into the drift (□drift) from the « cell » compartment. Following this calculation, the molar flow rate out of the repository zone, up to the zone seal (Φ repository zone output) is assessed ;
  - ✓ a calculation representing the repository zone, drifts and shafts (« repository zone and access structures » calculation). The source term considered is the result of the « repository zone » calculation. Following this calculation, the molar flow rate (Φshaft output) out of the shafts is assessed ;
  - ✓ two 3D calculations modelling the transfer in the surrounding formations to the outlets for the two hydrogeological models in question : the molar flow rate out of the access shaft is used as the input data for this calculation. The representation of the surrounding formations and the biosphere is the same as in the case of the transfer pathway through the geological barrier.

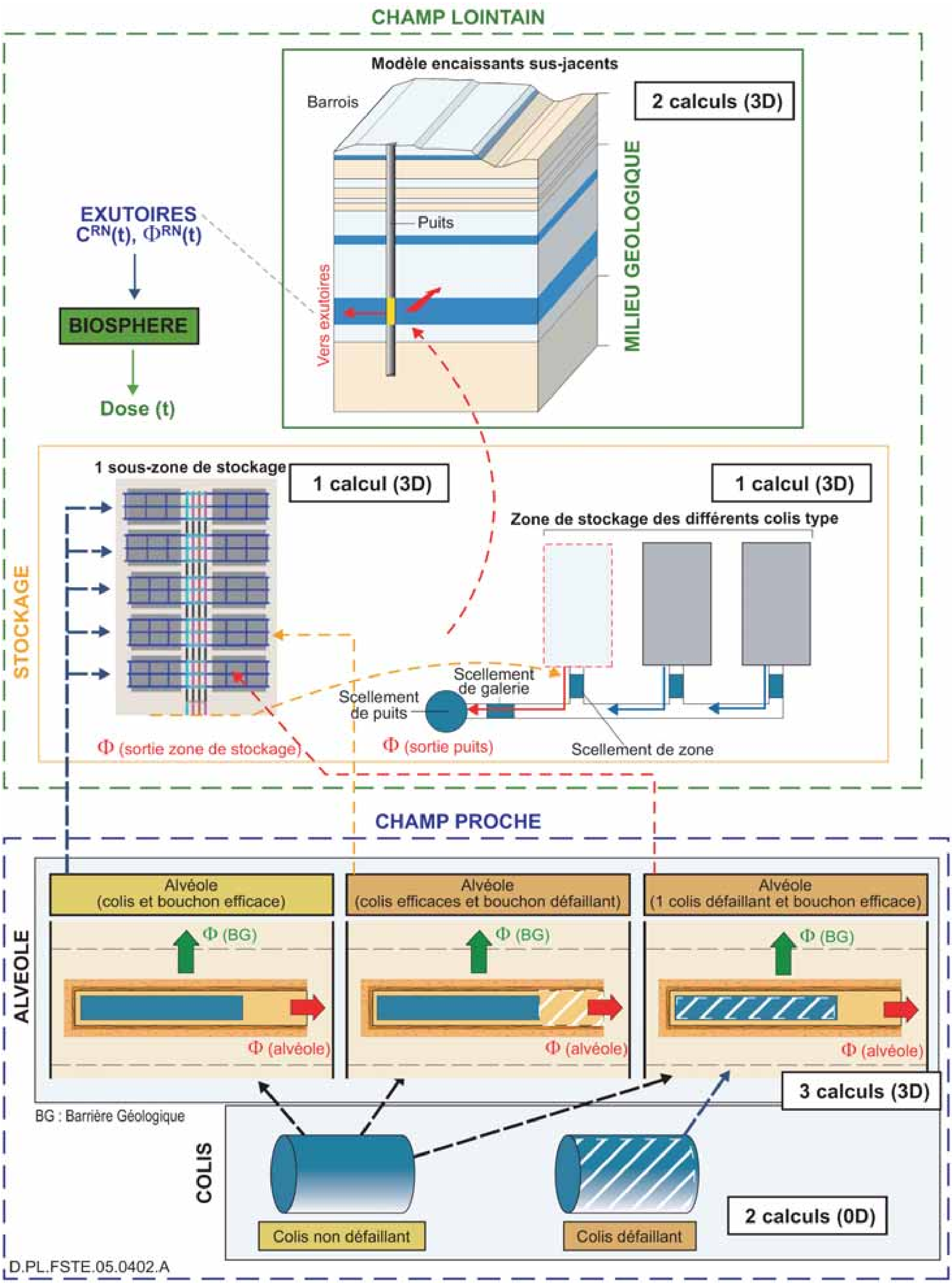


Figure 5.4-7 Calculation diagram for the transfer pathway through the engineered structures

For the normal evolution scenario, this transfer pathway is quantified as part of a preliminary calculation : the result showing that the dose associated with transfer through the engineered structures is insignificant, only the calculation diagram associated with transfer through the sound geological barrier is used.

For « borehole » and « sealing failure » altered evolution scenarios (see chapter 7), or for more penalising configurations (sensitivity study on the transfer properties of the EDZ in a SEN), the two calculation diagrams corresponding to the two transfer pathways are systematically taken into account.

5.5 Quantitative assessment of the normal evolution scenario

5.5.1 Context

The results analysis approach does not only consist of comparing the outlet dose to the 0.25 mSv/year constraint. It also relies on the analysis of several intermediary indicators that allow to :

- understand the individual functioning of each of the components as pertains to transfers and assess their performance with respect to functions assigned to them (especially what are their capabilities to delay and attenuate the migration of radionuclides) ;
- remove some uncertainties. For example, a « radionuclide flux at the exit of the host formation » indicator is, unlike the dose, independent of uncertainties on surrounding formations and the biosphere.

After presenting the results associated with safety function performances and with reference calculation impact, we will expose the results of the sensitivity study.

All these calculations [64] were conducted using the Alliances platform which was briefly presented in section 5.4. A more detailed description will be provided in the next two sections.

Two particular points are to be mentioned for a thorough understanding of the calculation results :

- to enhance readability, some curves are presented in thumbnail format to avoid breaking up the text too often. Readers wishing to study a particular curve in greater detail will find an overview in a larger format in the appendix to this volume
- in the presentation of the calculation results, long-lived radionuclides as defined in section 2.1.7 are grouped into two sub-categories :
  - ✓ radionuclides with a half-life of less than 10,000 years (generally called « medium-lived » in the presentation of calculation results below) ;

radionuclides with a half-life of over 10,000 years (generally called « long-lived » in the presentation of calculation results below).

5.5.2 Performances of safety functions : resisting water circulation

The performances of safety functions can be different based on the concerned waste zone. Moreover, for functions that let you control radionuclides, performances can also be differentiated based on the physical and chemical properties of the concerned element. Each time when required, we differentiate the performances calculated by type of waste and by radionuclide. When it concerns illustrating generic performance for a function, by default, specific case of CU1 fuel zone and 129 iodine zone is used which induce greatest impact.

5.5.2.1 A function quantifiable by the Peclet number, proportion of diffusive/advective flows and transfer paths

Other than design provisions that aim to limit the extension and intensity of the Callovo-Oxfordian damage, compliance with the function resisting water circulation is ensured by :

- characteristics of Callovo-Oxfordian, that conserve a diffusive system ;
- artificial barriers that restore host formation properties at the level of drifts and excavation damaged zone. The hydraulic performances of seals and plugs implemented in the drift network should allow for « resisting water circulation » and consequently preserving a dominant diffusive regime in the repository system. Control of seal performances is therefore an important element to meet this first safety function.

Different indicators quantify performances associated with the « resisting water circulation » function :

- the theoretical nondimensional Peclet (Pe) number (see insert), characterising the comparison of diffusive and advective transfer kinetics. For numbers greater than 2, advection becomes dominant ;

- advective and diffusive flow indicators, that provide a comparison of flow on exit from Argillites, around the repository ;
- distribution between the radionuclide mass transiting along and/or in the structures made up of drifts and shafts and the mass migrating by diffusion in the unaltered Callovo-Oxfordian, before reaching the top or the bottom of the formation. In fact one of the main objectives of this function is to avoid that the system of drifts and shafts does not constitute a preferential path for radionuclides up to the biosphere. Taking into account the geometry of the repository (important horizontal extension with respect to vertical extension), migration should mainly take place in the geological barrier based on the vertical direction. Radionuclide flow exiting the shafts after having transited in the structures should therefore be negligible in face of the radionuclide flow reaching the top of the Callovo-Oxfordian after having migrated in the geological barrier.

Insert 6 *The Peclet number*

The theoretical Peclet number is equal to the ratio of the characteristic diffusive migration time (Td) over the advective transfer time (Tc).

Pe = (Td/Tc) where :

$Td = L^2\omega_d/De$

$Tc = L\omega_c/(Kv_{geological\ barrier}.grad\ H_{vertical})$ if we are interested in vertical flow in the geological barrier

$= L\omega_c/(Kh_{structure\ EDZ}.grad\ H_{horizontal})$ if we are interested in horizontal flow in the structures

Where :

Td, characteristic migration time by diffusion [year],

Tc, characteristic migration time by advection [year],

L, migration distance [m],

ω_d : accessible diffusion porosity [-],

ω_c, : kinematic porosity [-],

De : effective diffusion coefficient [m²/s],

K : permeability (m/s),

Grad H : head gradient [m/m].

Figure 5.5-1 presents the possible displacement paths in the repository. A part of the flow is directed towards the Argillites where it mainly migrates by diffusion. This path is characterised by measuring the quantities traversing the geological barrier (Φ BG for the flow). Another part of the flow is liable to follow a path within repository structures towards the preferential drain that may constitute the shaft (Φ shaft) (even if seals are an obstacle).

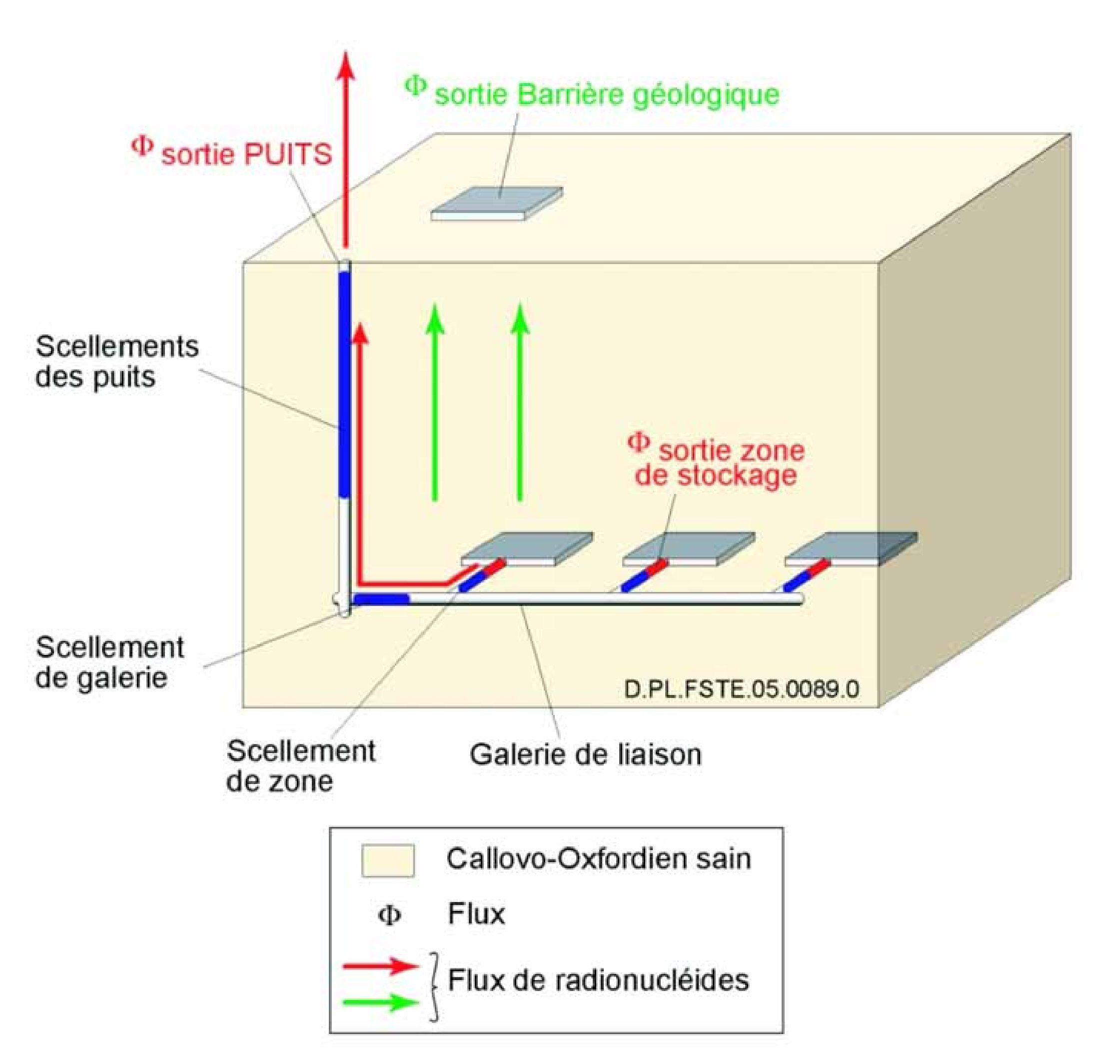


Figure 5.5-1 SEN - Quantification diagram of « structure and shaft » and « unaltered geological barrier » transfer paths

5.5.2.2 Results that confirm a dominant diffusive regime

- **In Callovo-Oxfordian**

In Callovo-Oxfordian, indicators highlight that diffusion is effectively the dominant transfer regime.

On one hand, the theoretical Peclet number is much lower than 2 both for anions and cations (see Table 5.5-1). In fact, calculations give the following results, by distinguishing anions from cations.

Pe (anions) = 0.13 Pe (cations) = 0.0096	where : L = thickness of unaltered Callovo-Oxfordian = 60 m ω_d = porosity accessible to diffusion = 0.05 (anions) = 0.18 (cations) ω_c = kinematic porosity = 0.09 De = effective diffusion coefficient = 5.10^{-12} m²/s (anions) = $2.5\ 10^{-10}$ m²/s (cations) Kv = vertical permeability = $5\ 10^{-14}$ m/s grad H = vertical rising head gradient = 0.4 m/m

Table 5.5-1 Calculation of Peclet number (Pe) in the Callovo-Oxfordian

On the other hand, the diffusive and advective flow calculations, taken from results of numerical simulations, confirms this dominance of diffusive flow, to almost two orders of magnitude (for example see Figure 5.5-2 and Figure 5.5-3 : ^{129}I anion of reference package CU1). It should be noted that this data is obtained with an ascending, maximum, vertical hydraulic gradient corresponding to a specific location in the transposition zone and to the one million-year Hydrogeological model (0.4 m/m).

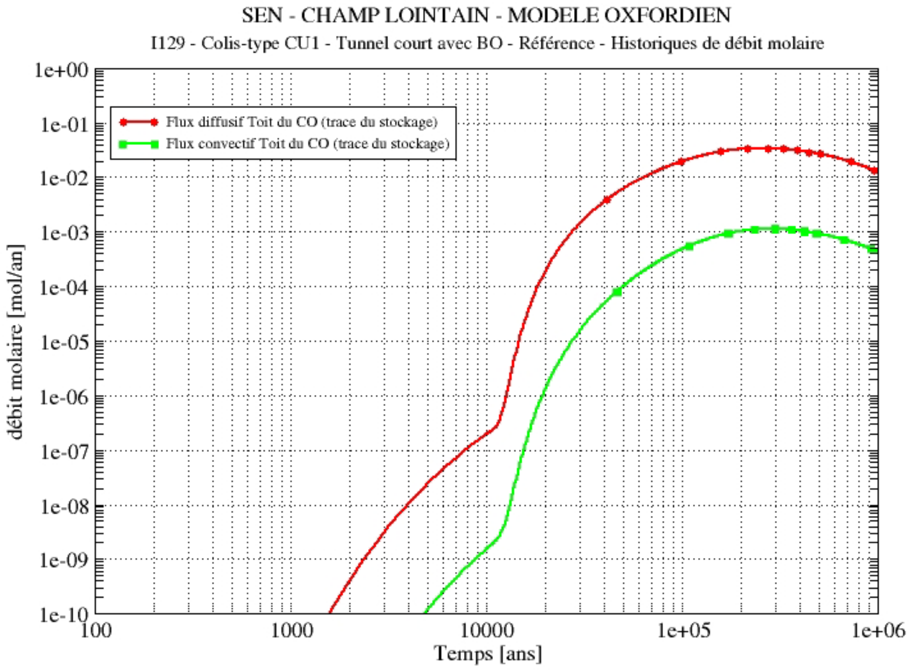


Figure 5.5-2 *SEN – History of the molar flow rate at the interface COX/overlying formation – example of iodine 129 of reference package CU1 (BO = swelling clay buffer, COX = Callovo-Oxfordian)*

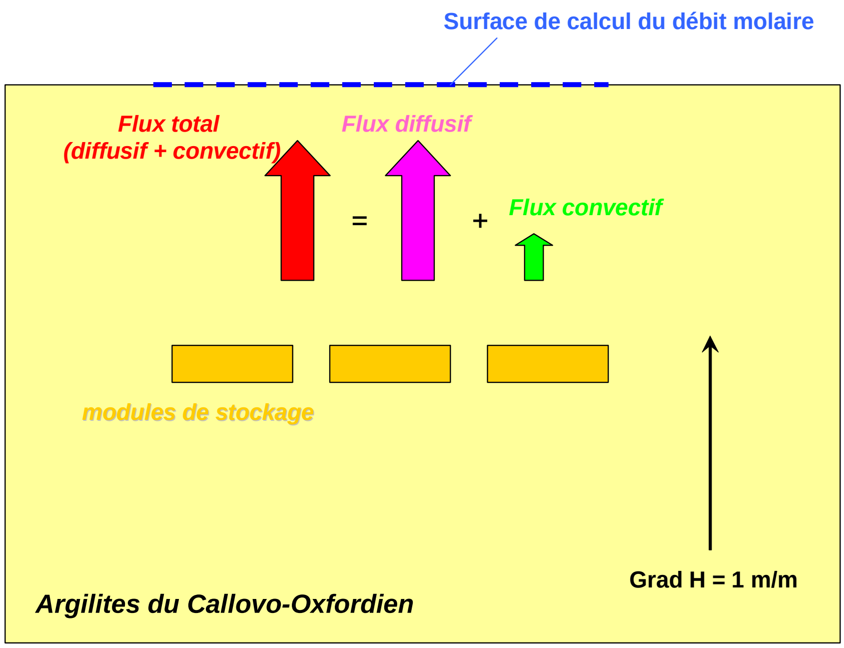


Figure 5.5-3 *SEN – Distribution of « advective » and « diffusive » transfer paths*

- **Structures /Callovo-Oxfordian comparison**

The quantitative assessment of distribution between the transfer paths through structures and unaltered Callovo-Oxfordian is based on the radionuclide (^{129}I). In fact, this soluble long-lived and not sorbed radionuclide illustrates the transfer effects due to water, since it is almost unsensitive to the medium's chemistry. We have used the case of CU1 as an example, since they present the strongest iodine 129 inventory. Calculation highlights that the majority of the mass finally takes the transfer path through unaltered Callovo-Oxfordian. In fact the next example concerning iodine of spent CU1 fuel gives the following conclusions (Figure 5.5-4 and Figure 5.5-5).

Nearly the entire released mass (99.999 %) exits by the top or the bottom of the Callovo-Oxfordian after having migrated by diffusion in unaltered Callovo-Oxfordian ; in fact :

- 41 % of the mass emitted by the packages reaches the Callovo-Oxfordian directly after having migrated in the disposal cell's body ;
- 59 % of the mass emitted by the packages reaches the drifts by diffusing through the structures located between the packages and the access drifts (especially the clay plug). This distribution corresponds to a pessimistic estimation of what can migrate in the access drifts. It results from limiting conditions (null concentration imposed around the plug) that induce high concentration gradient between the package and the access drift, thereby, at the disposal cell's scale, favouring horizontal transport through the plug and the damaged near-field of the Callovo-Oxfordian up to the access drifts.

In spite of this pessimistic hypothesis, of the 59 % reaching the disposal cells access drifts and internal connecting drifts in the repository zone, nearly all finally rejoin the Callovo-Oxfordian. In fact, flows in the drifts are very slow and limited. The quantity of radionuclides entering in the drift near the disposal cell therefore slowly migrates by diffusion (or advective/diffusive co-dominance near the boundaries of the zones). Taking this relative immobility into account and the transfer length (several hundreds of meters), nearly the entire mass present in the drift exits by diffusion in the geological barrier ; the molar flow exiting from the repository zone is extremely attenuated and corresponds to the immediate influence of some disposal cells, located near the entrance to the repository zone.

Consequently, a very small quantity of iodine 129 leaves by the shafts (approximately 3.10^{-5} % of the total injected mass) and the molar flow exiting the shafts is about 6 orders of magnitude less than what is transiting through the unaltered geological barrier up to the top and the bottom of the Callovo-Oxfordian. It is thus negligible.

The preponderant transfer paths is through the unaltered geological barrier.

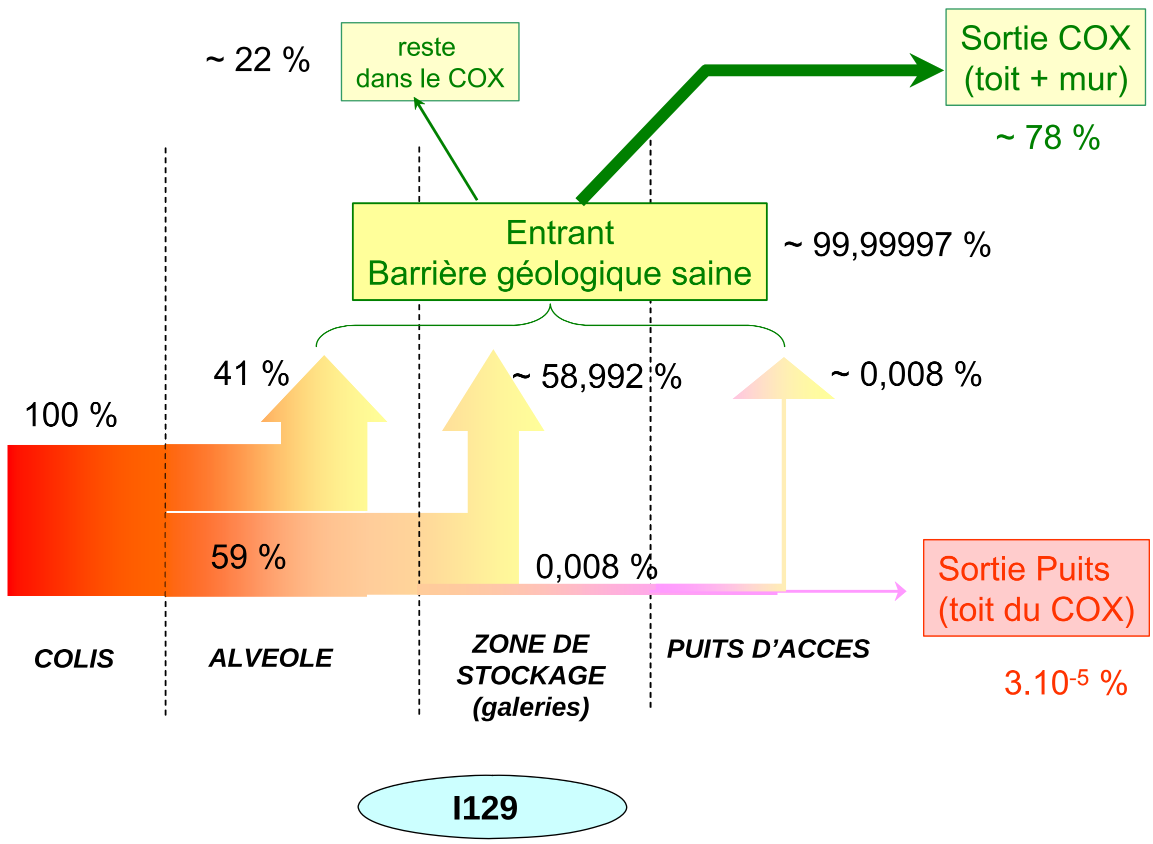


Figure 5.5-4 SEN – Distribution of mass through different calculation compartments (iodine 129 of CU1) (COX = Callovo-Oxfordian)

In addition, the length to be travelled through drifts from packages up to the top of the Callovo-Oxfordian is greater than the thickness of the host formation (several hundreds of metres against about 60 meters). As a consequence, the dates for appearance of activity flow maxima at the Callovo-Oxfordian top appear much later for the fraction of radionuclides that have transited in the drifts system than for the fraction that have migrated through the Callovo-Oxfordian.

To illustrate the last point, Figure 5.5-5 shows the evolution of the activity flow in different repository points (in the « package source term » release representation, we see two curves corresponding respectively to the failed packaging – beginning at 200 years and that due to the remaining inventory – beginning at 10 000 years) as a function of time for the entire UOx spent fuel zone. We observe that the flows on exiting the seal structures or shafts are clearly more delayed than those at the roof of the host formation, which themselves already appear late. The flows on shaft exit arrive at their maximum after 800 000 to 1 million years, against around 250 000 years for rock.

Here it directly concerns an effect of the « resisting water circulation » function. Since access paths are not preferential paths, as compared to the host rock, only the distance to travel counts, longer in their case.

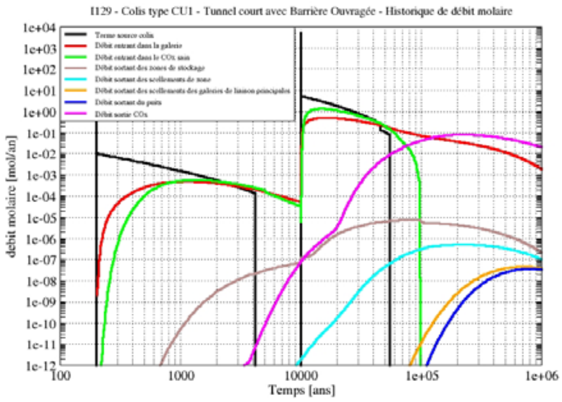


Figure 5.5-5 SEN – Evolution of molar flows through the repository – ^{129}I – CU1 (Cox = Callovo-Oxfordian, COX exit : exit through the top and botton of the formation)

This assessment made with a non-sorbed element in the geological barrier, also covers the sorbed elements in the host formation, such elements having indeed a tendency to preferentially migrate in rock and have greater affinity for it.

- **In disposal cells**

Assessments highlight that the regime within repository disposal cells remains dominantly diffusive for all the packages (for example see Figure 5.5-6. As an example, an assessment of the Peclet number in the structures was made for CU1 spent fuels. The Peclet number varies from 7.10^{-4} to 0.7 at the disposal cell head, the minimum value is located at the plug level, the maximum value at the fractured zone level where permeability is highest. With increasing distance from disposal cells, the advection increases in repository drifts due to the hydraulic flow collection effect. The Peclet number reaches 10 when exiting the spent fuel zone, upstream of zone seals. But the transfer however is very slow and limited (advective and diffusive transfer time of several hundreds of thousands of years along the secondary connecting drifts).

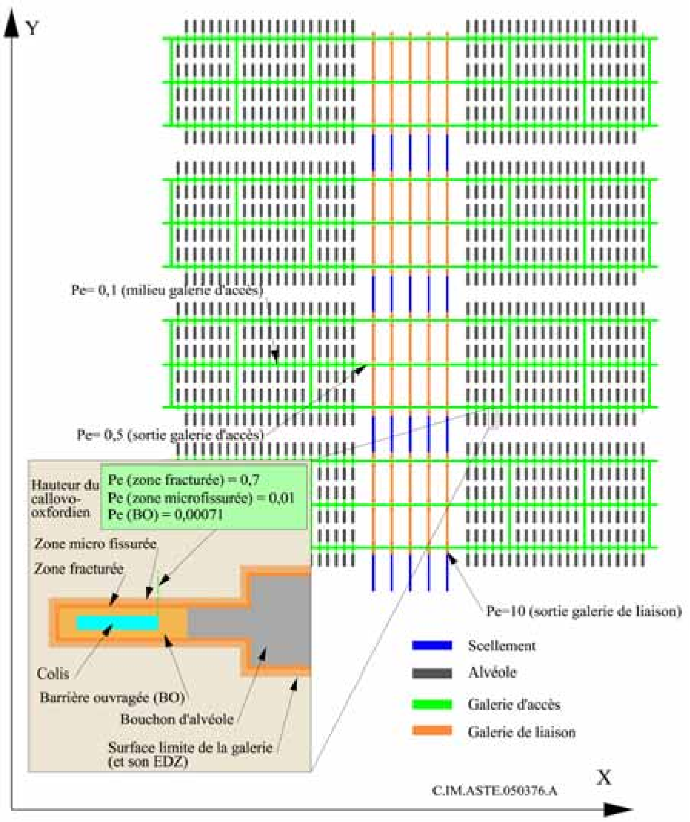


Figure 5.5-6 SEN – Assessment of the Peclet number in repository structures for CU1 reference packages

5.5.3 Performances of safety functions : limiting the release of radionuclides and immobilizing them in the repository

5.5.3.1 Possible indicators for analysing this function

Pertinent indicators for assessing the « limiting the release of radionuclides and immobilizing them in the repository »function are deduced from the objectives associated with this function. It consists of :

- prohibiting water ingress on waste (C, spent fuel) to avoid any release of radionuclides till the temperature of the waste or the surrounding medium is greater than the acceptable threshold (see chapter 3). Analysis of consequences associated with the premature release from C waste packages or spent fuel after initial failure of one or more containers allows us to asses, by difference, the interest of such provision at the scale of a package (the package failure alternate evolution scenario (SEA) will in addition highlight this aspect in chapter 7). It is in fact possible to compare the attenuation functions in constructed components located in the field near the packages, but also in the geological barrier, between prematurely released radionuclides and radionuclides released at the end of containers' sealing period ;
- resisting transport of dissolved species in the vicinity of glass and spent fuel ; this function is mainly ensured by the presence of mediums with low diffusion coefficient and permeability around waste (swelling clay buffer, if used, disturbed Callovo-Oxfordian (EDZ), plug…) that should induce diffusive transport in C waste disposal cells and spent fuel disposal cells. Analysis of the « resisting water circulation » function has highlighted that effectively we were in diffusive regime in the C waste and spent fuel disposal cells with the Peclet numbers much less than 1 in disposal cell head as well as at the plug level than in the micro-fissured zone. For reasons explained in section 5.3, the possibility of some C waste disposal cell plugs being defective was

included in the analysis. Corresponding calculations show that the hydraulic regime is not modified in the disposal cell, due to redundancy ensured by sealing of drifts ;

- limiting alteration of waste and consequently the release of radionuclides. Compliance with this objective is ensured by heat, water, mechanical and geochemical environmental conditions that are favourable and adapted to each waste type. The interest of this function can be assessed by difference when we conduct sensitivity analysis towards stronger release models for waste packages (see section 5.5.6.2) ;
- limiting the solubility of elements released by the packages. The performance of this part of the function is retranslated in the model by solubility limits applied to different radionuclides. This function plays a role for radionuclides such as actinides or selenium 79. The latter element illustrates the influence of the function especially well. In fact it is very similar to iodine, both representing non sorbed anions. The main difference between them is that in its reduced form, selenium has very low solubility in comparison with iodine. Comparing the masses of each of these radionuclides that migrate outside the disposal cell, with respect to the released masses, illustrates the precipitation effects well.

In the final analysis, this study allows to evaluate the added benefit of an overpack with respect to radionuclide transfer, by studying the results obtained with a failed overpack.

In addition, the mass of selenium 79 remaining in the field close to the disposal cells is also of interest.

5.5.3.2 A function providing efficient delay to migration of radionuclides

Early failure of one of the C waste and spent fuel packages as early as 200 years into the SEN leads to premature release of radionuclides into an environment with strong thermal load and strong thermal gradient for a multi-century period. For C waste and spent fuel, the consequences, during the period when the temperature is high with a strong gradient, are :

- premature release of radionuclides,
- faster diffusive transport during thermal phase (high diffusion coefficient),
- possibility of accelerated migration under the effect of temperature gradient.

We conduct analysis on three least sorbed radionuclides (^{129}I, ^{36}Cl, ^{79}Se).

When compared to migration beginning after thermal decrease, we note a very weak influence of transfer in thermal fields on iodine and chlorine when exiting the geological barrier. Examination of Figure 5.5-5 allows to again draw lessons in the case of iodine 129 from spent fuel CU1, but the results are identical for all the reference packages of thermal waste. We observe that the molar flow associated with failed package, clearly identifiable in the history of molar flow associated with the disposal cell content, is nearly confused with that due to the remaining inventory, on exiting the Callovo-Oxfordian or on exiting the repository zone. In fact, the lead taken by radionuclides migrating earlier and in a thermal environment is ultimately compensated by the faster diffusion of other radionuclides, for which concentration gradients rapidly grow stronger. Moreover, assessed flows are several times lower than those due to unaltered packages, which shows that even in the presence of several failed packages, the impact would be due to the « non failed » fraction of the inventory.

These results arise from diffusion coefficient and temperature delay values introduced in the calculations. It is important to remind that these values are based on experiments carried out on this day, which define value sets deemed to be representative. Nevertheless, temperature related transport is a domain that still requires to be studied for better understanding of the overlying mechanisms and to confirm the current assessments. The calculations presented here constitute an initial approach that allows us to make an initial judgment on the character, less pessimistic, of temperature related transfers. At this stage of knowledge, it is however desirable to remain cautious. These calculations, showing a modest influence of the « limiting the release of radionuclides » function during the thermal phase, do not invalidate the interest of this function, which stays an important element to manage uncertainties, vis-à-vis the short and medium term safety.

This initial assessment will however be completed and specified in view of other calculation cases, as part of the « package failures » altered evolution scenario, which allows you to generalise the results presented above.

5.5.3.3 A function allowing the immobilisation of radionuclides in near-field

Some radionuclides, such as actinides, are in part immobilised by their solubility limit inside the disposal cells. But actinides are also delayed by the sorption capability in the Argillite (« delaying and reducing the migration of radionuclides » function). The actinides therefore do not leave the near-field of the disposal cell, this effect being partly associated with solubility and partly with sorption. The details of results relating to actinides are presented in section 5.5.4.3.

The case of selenium 79 consequently provides the simplest illustration of solubility effects, in that this radionuclide is not sorbed in the argillite. The part of selenium 79 that remains confined in the disposal cells is therefore a direct quantification of the role of solubility.

If, for example, we place ourselves in the disposal cells of CU1 spent fuel and if we are interested in the total mass of radionuclides that have succeeded in leaving the disposal cell, we observe that the entire iodine 129 initially present in the fuel migrated into the Callovo-Oxfordian after a million years, whereas 99.4 % of the selenium 79 is still present in the disposal cell. With all other parameters being equal for both these radionuclides (which are both anions with a null Kd), this gap can be explained by the very low solubility limit of selenium 79, and thereby its « immobilisation ».

5.5.4 Performances of safety functions : delaying and reducing the migration of radionuclides

5.5.4.1 A function quantifiable by the mass attenuation function and the activity flows in the Callovo-Oxfordian

The aim of this function is delaying and reducing the migration of radionuclides emitted by the packages in the repository system. It is ensured (see chapter 3) :

- by the formation of the Callovo-Oxfordian, that presents very good capacities to chemically retain and limit (low diffusion coefficient) the flow of radionuclides. The large thickness of the undisturbed argillite layer on either side of the structures (minimum 60 metres) guarantees a good level of attenuation ;
- by structures (swelling clay buffer in spent fuel disposal cells, concrete in B waste disposal cells, seals in access structures), in partial redundancy with the host formation. These barriers are set up to comply with other functions (mainly to ensure a favorable chemical environment, and resist water circulation). But by design, they also contribute to the delay and mitigation of radionuclide flows. It is nevertheless expected that their contribution remains a minority with respect to that of the Callovo-Oxfordian.

Performance associated with the « delaying and reducing the migration of radionuclides » function in the repository are quantifiable with the help of three values associated with the molar flow of each radionuclide :

- the maximum molar flow (Φmax) ;
- the mass (m) integrally corresponding to the « molar flow » over the simulation duration (1 million years) ;
- the appearance time of maximum molar flow ($tmax = t(\Phi max)$),

These three values are illustrated in Figure 5.5-7 ; the curve is only presented as an example and does not aim to reflect any specific radionuclide behaviour, any molar flow history curve can be characterised by the three previously cited values.

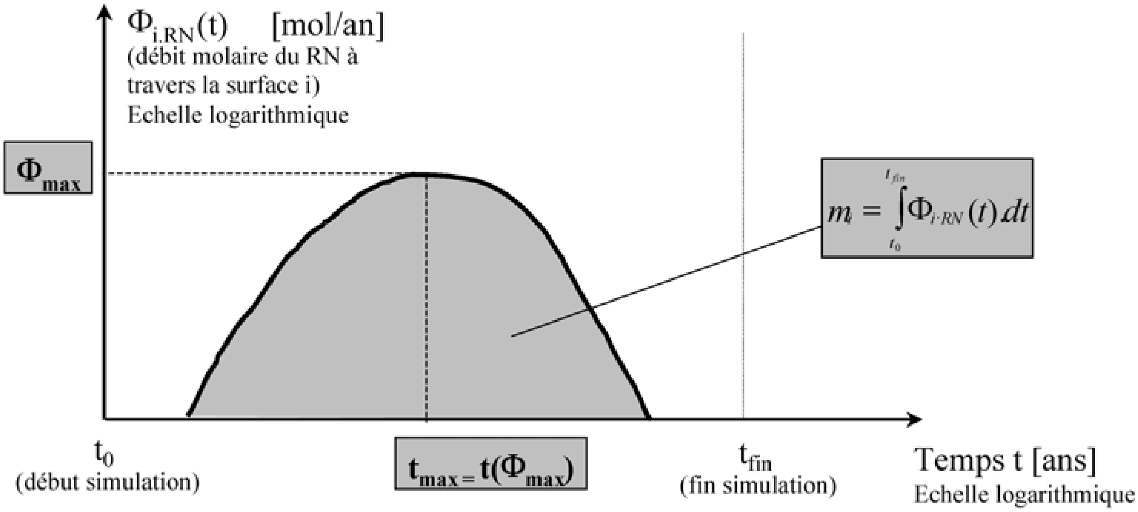


Figure 5.5-7 Values retained to assess the delay and attenuation

Comparison of values for each of these indicators, between two different surfaces (S_i and S_{i+1}), helps in assessing the confinement capability of barriers lying between these two surfaces. This is assessed as follows :

- attenuation of the radionuclide mass that corresponds to the fraction which does not exit the considered barrier(s) (especially the Callovo-Oxfordian) over the analysis duration. It thus integrates the radioactivity decay of radionuclides in the barriers resulting from more or less long migration time. The attenuation expression of the mass of each radionuclide is : $1- m_{i+1} / m_i$ [69] (or vice versa « m_{i+1} / m_i » if we refer to the « crossing », that is, not attenuated fraction). As a general rule, we take the mass initially present in the waste as reference. The attenuation is assessed in all the components between the last considered barrier and the package ;
- the delay, corresponding to the duration between the molar flow maxima entering and leaving a barrier ; this indicator is mainly pertinent for very long-lived radionuclides having low decay. The delay is expressed by $t_{max\ i+1} - t_{max\ i}$ [70] ;
- attenuation of the molar flow maxima which illustrates the « attenuation » aspect of the function. Expression of the indicator is $\Phi_{max\ i+1} / \Phi_{max\ i}$ [71] ; it lets you assess the order of magnitude of the maximum flow emitted by the packages for each radionuclide.

5.5.4.2 Results that confirm a good capacity for delaying and reducing the migration of radionuclides

- **In the near-field of the disposal cells (swelling clay buffer for spent fuel, concrete for B waste disposal cells).**

Before getting involved in the host formation, which is the main contributor of the « delaying and reducing the migration of radionuclides » function, we search to identify possible near-field effects, brought in by the constructed components of the disposal cells.

Concerning spent fuel, radionuclides having maximum activity flow emitted by the packages and most reduced in the swelling clay buffer, are :

- Elements strongly sorbed in the swelling clay buffer, mainly ^{93}Zr (➔ ^{93m}Nbm), ^{94}Nb, ^{166m}Ho, ^{99}Tc, ^{126}Sn (retardation coefficient at least equal to 60 000) ;
- Selenium 79 not sorbed, but precipitating in the swelling clay buffer.

69 Where m_i is the mass crossing the surface S_i during the entire calculation period.

70 $t_{max,i}$ is the date on which maximum flow crosses the surface S_i.

71 $\Phi_{max,i}$ is the maximum flow crossing the surface S_i

As an example, specifically in the case of Uox2 spent fuel, the maxima of activity flow of these radionuclides are reduced at least by five orders of magnitude in the clay engineered barrier.

For vitrified waste, the reference concept does not include a swelling clay buffer. Given that the geochemical characteristics that are very similar between the host rock and the swelling clay, we do not expect any significant difference, from the point of view of near-field transfer, between the reference concept and the variant with swelling clay buffer. Also, in the near-field of the disposal cells, radionuclides whose molar flow maximum emitted by the packages is most reduced are the same as for the spent fuel (^{93}Zr (➔ ^{93m}Nbm), ^{94}Nb, ^{166m}Ho, ^{99}Tc, ^{126}Sn) ; at about 7 metres in the Callovo-Oxfordian (distance conventionally chosen for ease of calculation), their molar flow is completely attenuated. As an example, for C2 reference packages and at the same distance, the most mobile radionuclides (^{129}I and ^{36}Cl) present the maxima of molar flow reduced by about two orders of magnitude.

Concerning the B waste, attenuations depend on waste release models that significantly vary from one reference package to another. We nevertheless note that like spent fuel, the most reduced radionuclides are the elements sorbed in concrete (^{93}Zr ➔ ^{93m}Nb, ^{94}Nb, ^{129}Sn, ^{59}Ni, ^{14}C, ^{41}Ca). As an example, maximum attenuation of molar flow of radionuclides previously cited between the package exit and disposal cell exit is at least five orders of magnitude for bituminised sludge whose release model is inversely proportional to square root of time.

- **In the geological barrier**

Performance assessment associated with the « resisting water circulation » function clearly shows the Callovo-Oxfordian as a diffusive barrier. The properties of the seals, of the Callovo-Oxfordian and the repository geometry jointly contribute to induce predominance of a vertical diffusive transport of radionuclides through the Callovo-Oxfordian, from the packages to the surrounding formations. The repository is deemed to be located in the middle of the Callovo-Oxfordian formation, the distribution between the ascending and descending flows is nearly equivalent.

Reduction of mass (with respect to initial mass) and the delay of radionuclides in the Callovo-Oxfordian are indicators adapted to assess the performances of « delaying and reducing » function. Analysis of results highlights that these indicators depend little on the studied concepts, the latter providing a gain in attenuation which remains limited as compared to the Callovo-Oxfordian performance. Their aim is indeed not to provide an additional delay and reduction of transfers through rock, but to participate in managing uncertainties (overpacks, for example, are used to prevent migration in a thermal environment).

For studied radionuclides, the Table 5.5-2 presents the main characteristics that condition their behaviour in the geological barrier : decay half-life, ionic form, solubility, retardation coefficient.

Radionuclides	Half-life of radioactive decay [years]	Ionic form	Solubility in Callovo-Oxfordian [mol/m^3]	Retardation coefficient in the Callovo-Oxfordian [-]	
				Delay at 25°C	Delay if ΔT > 20 °C*
^{36}Cl	3.02 E+05	Anion	Soluble	1	
^{79}Se	6.50 E+04	Anion	5.00 E-07	1	
^{129}I	1.57 E+07	Anion	Soluble	1	
^{14}C	5.73 E+03	Anion	2.30 E+00	6	
^{41}Ca	1.03 E+05	Cation	2.30 E+00	16	2.51
^{93}Mo	3.50 E+03	Anion	1.00 E-05	139	
^{59}Ni	7.50 E+04	Cation	5.00 E-02	2 050	
^{135}Cs	1.00 E+05	Cation	Soluble	Langmuir**	
^{107}Pd	6.50 E+06	Cation	4.00 E-04	8 950	
^{93}Zr	1.53 E+06	Cation	2.00 E-05	12 800	
^{10}Be	1.60 E+06	Cation	1.00 E-02	31 900	3.19
^{93m}Nb	1.64 E+01	Anion	2.00 E-04	53 400	
^{94}Nb	2.03 E+04	Anion	2.00 E-04	53 400	
^{99}Tc	2.13 E+05	Cation	4.00 E-06	128 000	
^{126}Sn	1.00 E+05	Cation	1.00 E-05	179 000	
^{166m}Ho	1.20 E+03	Cation	1.00 E-04	639 000	

On yellow background : radionuclide completely attenuated in the first tens of meters of Callovo-Oxfordian.
On green background : radionuclide completely attenuated on exiting from Callovo-Oxfordian.

* : Delay if Δt >20°C : for exothermic waste zones, the « temperature » specific retardation coefficient was assigned to the radionuclides wherein heat induces a significant influence on the value of retardation coefficient [62]. Thus, as soon as the temperature gap of the medium between the initial moment and the migration date of radionuclides (at each mesh level) is greater than 20°C, we define a specific retardation coefficient value.

** : Langmuir : $Kd = (1.85x10^{-7}/(4.76x10^{-7} + Ceq)$ where Ceq : solution concentration (mol/l)

Table 5.5-2 SEN – Main characteristics of radionuclides mobilised in the Callovo-Oxfordian

Assessment of attenuation arising from calculations highlights that :

- at about ten metres in the Callovo-Oxfordian around 50 % of the studied radionuclides are completely attenuated :
  - ✓ Elements strongly sorbed in the geological barrier : ^{10}Be, ^{94}Nb, ^{99}Tc, ^{126}Sn, ^{166m}Ho ;
  - ✓ Moderately long-lived and sufficiently sorbed radionuclides (^{93}Mo) ;
- at the top of the Callovo-Oxfordian, the majority of the radionuclides are completely attenuated. In addition to the radionuclides cited earlier, attenuation is complete for :
  - ✓ Long-lived, strongly sorbed cations : ^{59}Ni, ^{135}Cs, ^{107}Pd et ^{93}Zr ➔ ^{93m}Nb ;
  - ✓ ^{14}C, moderately long-lived and weakly sorbed, soluble anion that benefits from radioactive decay.

Note also that the actinides which formed the subject of a preliminary calculation, presented in section 5.5.4.3, have a totally attenuated molar flow rate at the limits of the formation.

Only the not sorbed anions (^{129}I, ^{36}Cl, ^{79}Se) and the weakly sorbed cation (^{41}Ca) present a non-negligible flow at the top of the Callovo-Oxfordian (see Table 5.5-3).

Radionuclides	Phenomena contributing to attenuation	Attenuation of the mass on exiting from Callovo-Oxfordian (top + bottom)	
		C waste / Spent Fuel	B waste
^{14}C	Diffusion	Complete attenuation (100 %)	
^{93}Mo	Diffusion and sorption		
^{10}Be	Sorption		
^{93m}Nb			
^{94}Nb			
^{99}Tc			
^{126}Sn			
^{166m}Ho			
^{135}Cs			
^{107}Pd			
^{59}Ni			
^{93}Zr			
^{79}Se	Diffusion and precipitation	> 99.95 %	
^{41}Ca	Diffusion and sorption	90 – 95 %	> 99 % (approx.)
^{36}Cl	Diffusion	65 – 75 %	> 70 % (approx.)
^{129}I		20 – 30 %	> 50 % (approx.)

On yellow background : Radionuclides **completely** attenuated in the first 10 meters of the Callovo-Oxfordian

On green background : Radionuclides **strongly** attenuated in the first 10 meters of the Callovo-Oxfordien

Table 5.5-3 SEN – Attenuation of radionuclides in the Callovo-Oxfordian.

However, for the four elements not totally attenuated (^{129}I, ^{36}Cl, ^{41}Ca, ^{79}Se), we note that :

- The attenuation is slightly higher for C waste than for spent fuel. This difference arises out of :
  - ✓ On one hand, effect of greater heat in case of spent fuel (temperature gradient and temperature ranges) than for C waste, which slightly accelerates the diffusion ;
  - ✓ On the other hand, the glass release model whose duration (app. 300 000 years) is significant in comparison with the diffusive transfer time.
- Attenuation of B waste is also higher than that of C waste and spent fuel. This difference results on one hand from the heat (absent for B waste) and on the other hand, for some radionuclides (especially ^{129}I and ^{41}Ca), from sorption in concrete that helps in delaying their transport as early as the disposal cell stage of B waste. Both these effects converge to induce a slightly higher attenuation for B waste concepts than for C waste and spent fuel concepts.

Amongst all the radionuclides present in the packages, only the four cited above in the previous table are not totally attenuated. We will next see that calcium 41 has a negligible impact. Below, as an example for the reference packages CU1, C1/C2 and B2, we will provide the exit dates from the Callovo-Oxfordian, corresponding to the maximum flow of ^{129}I, ^{36}Cl and ^{79}Se (see Table 5.5-4). Diffusion properties in the Callovo-Oxfordian are nevertheless good enough to strongly delay the appearance of maximum of molar flow at the formation's exit, at the approximate scale of 200 000 years. Chlorine 36 and iodine 129 have a similar behaviour (not sorbed soluble anions). Iodine 129 has a long half-life with regards to the diffusive transfer time (1.57 10^7 years), and decays only very little during its migration. The date when its maximum appears on exiting the host formation, 260 000 years for CU1 packages, is therefore the direct expression of its transfer time. On the other hand, chlorine 36, decays during its migration (its half-life is 300 000 years). The date when its maximum flow appears, 180 000 years, for CU1 packages is therefore earlier than that for iodine since it combines the effects due to migration and those due to radioactive decay.

Reference package	Radionuclides	Dates of molar flow maxima at the top of the Callovo-Oxfordian [years]
CU1	^{129}I	260 000
	^{36}Cl	180 000
	^{79}Se	400 000
C1/C2	^{129}I	460 000
	^{36}Cl	380 000
	^{79}Se	750 000
B2	^{129}I	465 000
	^{36}Cl	200 000
	^{79}Se	165 000

Table 5.5-4 SEN – Appearance dates of molar flow maxima on exiting the Callovo-Oxfordian for the three main impact contributors

5.5.4.3 Appendix : results relating to actinides in the SEN

A preliminary calculation was carried out on the actinides in order to show that they do not need to be considered in the overall impact calculation. This is due to the fact that, owing to their very high retention in the geological medium, they remained confined within the Callovo-Oxfordian and do not contribute to the radiological impact over the duration of the analysis (one million years). For these elements, the capacity of the medium to « restrict the release of radionuclides and immobilise them in the repository » and « delay and attenuate radionuclide migration » plays a major role. In addition to the elements revealed previously, this section presents the results associated with actinide transfer.

The calculation is simplified insofar as it only involves one CU1 cell (long UOx3 type). the conclusions remain nonetheless valid for all spent fuel packages and for other packages too (spent fuels being the most heavily charged with actinides).

The retention parameters taken into account for actinides are presented first of all, then the results of the calculation.

- **Data used**

The analysis of the chemical retention parameters (sorption and precipitation) is detailed in chapter 5 of the reference document on the behaviour of radionuclides and toxic chemicals [21]. It emerges that actinides have very low solubility and are strongly sorbed in the sound Callovo-Oxfordian argillites. The proposed distribution coefficient values (Kd) in the Callovo-Oxfordian are all over 0.9 m^3/kg[72], which leads to retardation coefficients (R) of at least 11,000. This data is :

- either derived from Kd measurements based on argillite samples (this is notably the case of uranium and plutonium),
- or established through analogy of chemical behaviour between elements based on results obtained for uranium and plutonium (this is notably the case of neptunium and thorium),
- or assessed through analogy of materials ; this is notably the case of trivalent actinides (actinium, americium and curium) for which it is considered that the sorption capacity of the argillites may be approached from the characteristics of an illite.

[72] Except for 210Pb, but this radionuclide has a short half-life (22.3 years).

The variability in environmental conditions during transitory periods (Redox and thermal transients in particular) and possible disturbances (alkaline disturbance and interaction with metallic components), as well as the associated uncertainties, are not liable to affect the very high actinide retention in argillites.

The values of the hydraulic, transport and chemical retention parameters in the Callovo-Oxfordian and in the engineered structures are presented in Table 5.5-5 and Table 5.5-6 below.

	Callovo-Oxfordian					
	$Kh = 5.10^{-13}$ m/s $Kv = 5.10^{-14}$ m/s	ρ = 2300 kg/m^3 ωkinematic = 0.09				
	Half-life [years]	ωDiffusion [-]	De [m²/s]	Kd [m^3/kg]	R [-]	Csat [mol/m^3]
^{244}Cm	18.1	0.18	$2.5.10^{-10}$	50	638,900	4.10^{-4}
^{240}Pu	6,560	0.18	$2.5.10^{-10}$	0.9	11,500	2.10^{-4}
^{236}U	23,400,000	0.18	$2.5.10^{-10}$	8	102,200	7.10^{-4}
^{232}Th	14,100,000,000	0.18	$2.5.10^{-10}$	8	102,200	6.10^{-4}
^{245}Cm	8,500	0.18	$2.5.10^{-10}$	50	638,900	4.10^{-4}
^{241}Pu	14.4	0.18	$2.5.10^{-10}$	0.9	11,500	2.10^{-4}
^{241}Am	433	0.18	$2.5.10^{-10}$	50	638,900	4.10^{-4}
^{237}Np	2,140,000	0.18	$2.5.10^{-10}$	0.9	11,500	4.10^{-6}
^{233}U	159,000	0.18	$2.5.10^{-10}$	8	102,200	7.10^{-4}
^{229}Th	7,340	0.18	$2.5.10^{-10}$	8	102,200	6.10^{-4}
^{246}Cm	4,730	0.18	$2.5.10^{-10}$	50	638,900	4.10^{-4}
^{242}Pu	374,000	0.18	$2.5.10^{-10}$	0.9	11,500	2.10^{-4}
^{238}U	4,470,000,000	0.18	$2.5.10^{-10}$	8	102,200	7.10^{-4}
^{234}U	246,000	0.18	$2.5.10^{-10}$	8	102,200	7.10^{-4}
^{230}Th	75,400	0.18	$2.5.10^{-10}$	8	102,200	6.10^{-4}
^{226}Ra	1,600	0.18	$2.5.10^{-10}$	1	12,800	1.10^{-4}
^{210}Pb	22.3	0.18	$2.5.10^{-10}$	0.16	2,050	4.10^{-3}
^{243}Am	7,370	0.18	$2.5.10^{-10}$	50	638,900	4.10^{-4}
^{239}Pu	24,100	0.18	$2.5.10^{-10}$	0.9	11,500	2.10^{-4}
^{235}U	704,000,000	0.18	$2.5.10^{-10}$	8	102,200	7.10^{-4}
^{231}Pa	32,800	0.18	$2.5.10^{-10}$	1	12,800	1.10^{-3}
^{227}Ac	21.8	0.18	$2.5.10^{-10}$	50	638,900	4.10^{-4}

Table 5.5-5 Preliminary SEN calculation – hydraulic, transport and chemical retention parameters in the Callovo-Oxfordian – actinides

	Concrete – reference calculation K = 10^{-6} m/s De = 6.10^{-10} m²/s ωDiffusion = 0.3			Bentonite engineered barrier, plugs and seals K = 10^{-11} m/s De = 5.10^{-10} m²/s Diffusion = 0.3		
	Kd [m3/kg]	R [-]	Csat [mol/m^3]	Kd [m^3/kg]	R [-]	Csat [mol/m^3]
Am, Cm, Ac	30	210,000	1.10^{-7}	12	58,300	4.10^{-4}
Np	20	140,000	5.10^{-6}	1	4,860	4.10^{-6}
Pb	1	4,200	soluble	3.1	15,100	4.10^{-3}
Pu	20	140,000	1.10^{-6}	1	4,860	2.10^{-4}
Pa	-	-	soluble	10	48,600	1.10^{-3}
Th	20	140,000	2.10^{-7}	3	14,600	6.10^{-4}
U	50	350,000	3.10^{-3}	100	486,000	7.10^{-4}

Table 5.5-6 Preliminary SEN calculation – hydraulic, transport and chemical retention parameters in the engineered structures – actinides

For illustration purposes, the inventory contained in a studied spent fuel cell (CU1 Uox3 long) is also provided for the long-lived isotopes of each of the four actinide chains (^{232}Th, ^{237}Np, ^{238}U, ^{235}U). This inventory may differ slightly from the average CU1 inventory which covers several types of fuel [18]. It should also be noted that it is to be used with precaution due to the contribution of parent elements to their descendants by filiation which can lead to a moderate increase in the mass of the latter over time.

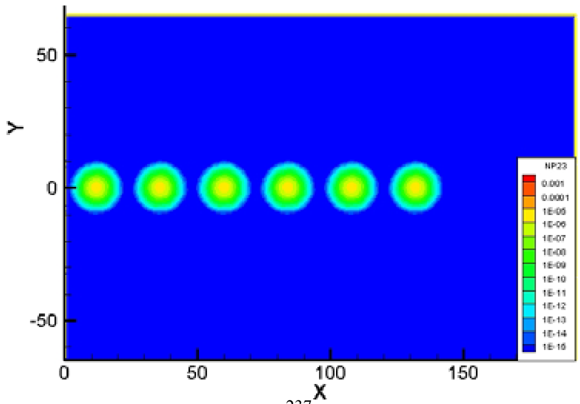
Isotope	Half-life [years]	Activity [Bq/ package]	Activity [Bq/cell]	Mass [mole/cell]	Mass [g/cell]
^{232}Th	14,100,000,000	nil	nil	nil	nil
^{237}Np	2,140,000	$4.5.10^{10}$	$1.3.10^{11}$	22	$5.1.10^{3}$
^{238}U	4,470,000,000	$2.4.10^{10}$	$7.3.10^{10}$	$2.5.10^{4}$	$5.9.10^{6}$
^{235}U	704,000,000	$1.2.10^{9}$	$3.7.10^{9}$	$2.0.10^{2}$	$4.7.10^{4}$

Table 5.5-7 Preliminary SEN calculation – Reference inventory (3 years after unloading the spent fuels from the reactor) of a CU1 spent fuel cell (UOx3 long) – ^{232}Th, ^{237}Np, ^{238}U, ^{235}U

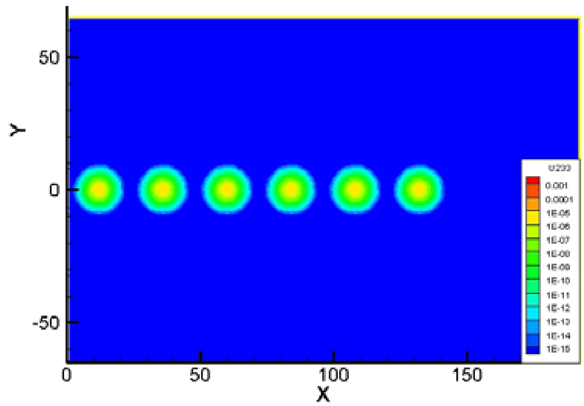
- **Results**

The results confirm that the Callovo-Oxfordian has a very good capacity to « restrict the release of actinides and immobilise them in the repository » and « delay and attenuate their migration » due in particular to their very high retention (sorption and precipitation) in argillites. This phenomenon leads to almost total confinement in the near-field of the cells over the next million years. It is noted in particular that :

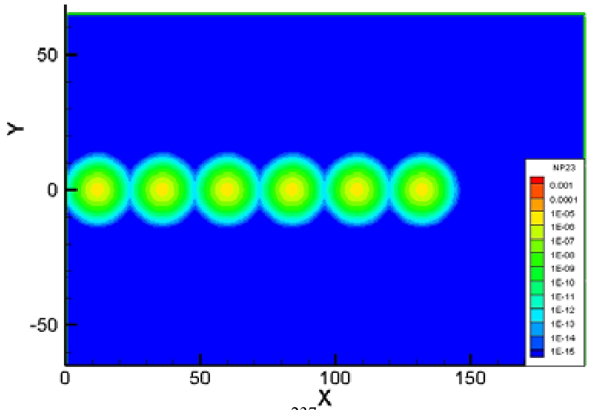
- After a few hundred thousand years, the actinides will only have covered a few metres in the geological barrier. Figure 5.5-8 below illustrates this strong confinement for ^{237}Np, which is one of the least-sorbed actinides, and ^{233}U (both belonging to the 4N+1 filiation chain).



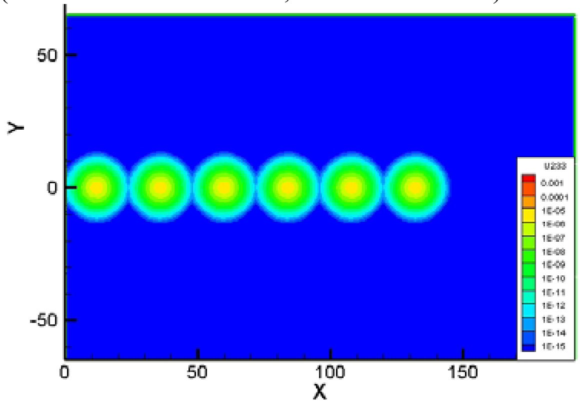
Concentration mapping of ^{237}Np at 200,000 years
(Concentration in mol/m3 ; X and Y in metres)



Concentration mapping of ^{233}U at 200,000 years
(Concentration in mol/m3 ; X and Y in metres)



Concentration mapping of ^{237}Np at 500,000 years
(Concentration in mol/m3 ; X and Y in metres)



Concentration mapping of 233U at 500,000 years
(Concentration in mol/m3 ; X and Y in metres)

Figure 5.5-8 Preparatory SEN calculation - Concentration mapping of ^{237}Np and ^{233}U – in the near field at 200,000 and 500,000 years

- Molar flow rate history in the Callovo-Oxfordian at 7 metres from the disposal cells, evaluated for the four actinide chains, confirm this very strong attenuation. Only ^{242}Pu, ^{239}Pu, ^{237}Np, ^{238}U, ^{235}U, ^{233}U ^{236}U and ^{229}Th in secular equilibrium with its ascendants present a molar flow rate in excess of 10^{-12} mol/year (see) for a three-package cell. The flow rates are nonetheless weak even for these isotopes. As an example, only 11 grams of ^{237}Np, 2.3 grams of ^{238}U and 0.0005 grams of ^{239}Pu per cell covered more than 7 metres of argillite over the total analysis period (one million years).

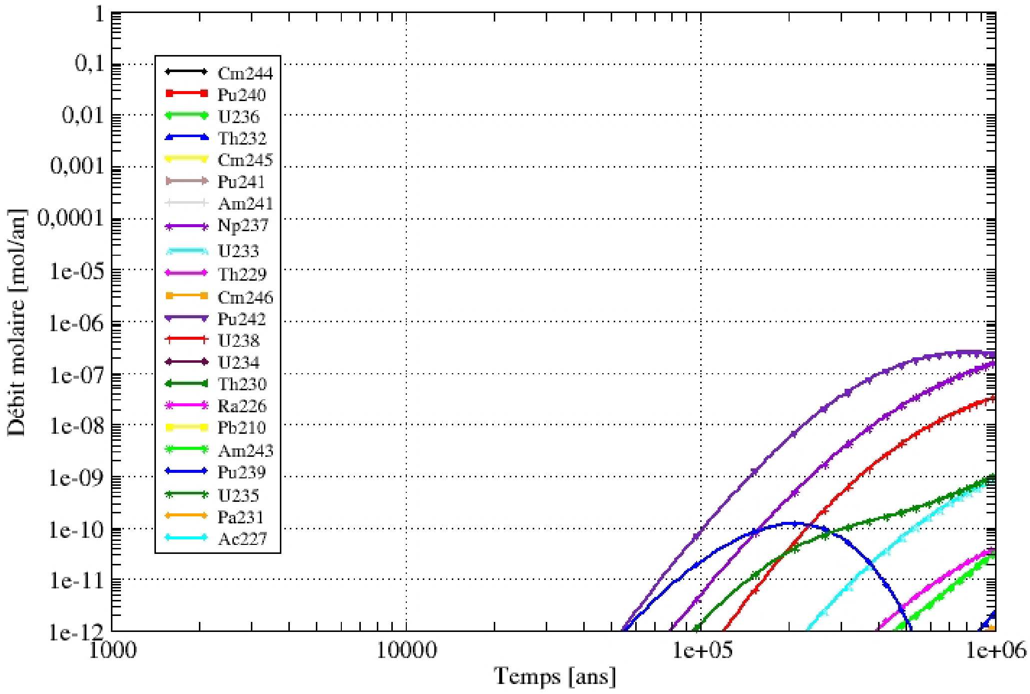


Figure 5.5-9 Historical molar flow rates out of the repository at 7 metres from the cells – Reference package CU1 Uox3

The attenuation in the actinide molar flow rates is total at the outlet of the Callovo-Oxfordian (see Figure 5.5-10). The actinide retention parameters lead indeed to theoretical transport times of several hundred million to billions of years, as indicated by the characteristic migration times[73] by diffusion, calculated for a depth of 60 metres of transport in the Callovo-Oxfordian :

- TD (neptunium, plutonium) = $9.5.10^{8}$ years
- TD (protactinium) = $1.0.10^{9}$ years
- TD (thorium, uranium) = $8.4.10^{9}$ years
- TD (actinium, americium, curium) = $5.3.10^{10}$ years

These very long periods have no physical significance, but do bear witness to the geological medium's very strong capacity to retain actinides.

With such transfer times, the quantity of actinides emerging from the Callovo-Oxfordian over a million years is nil.

[73] In the absence of radioactive decay

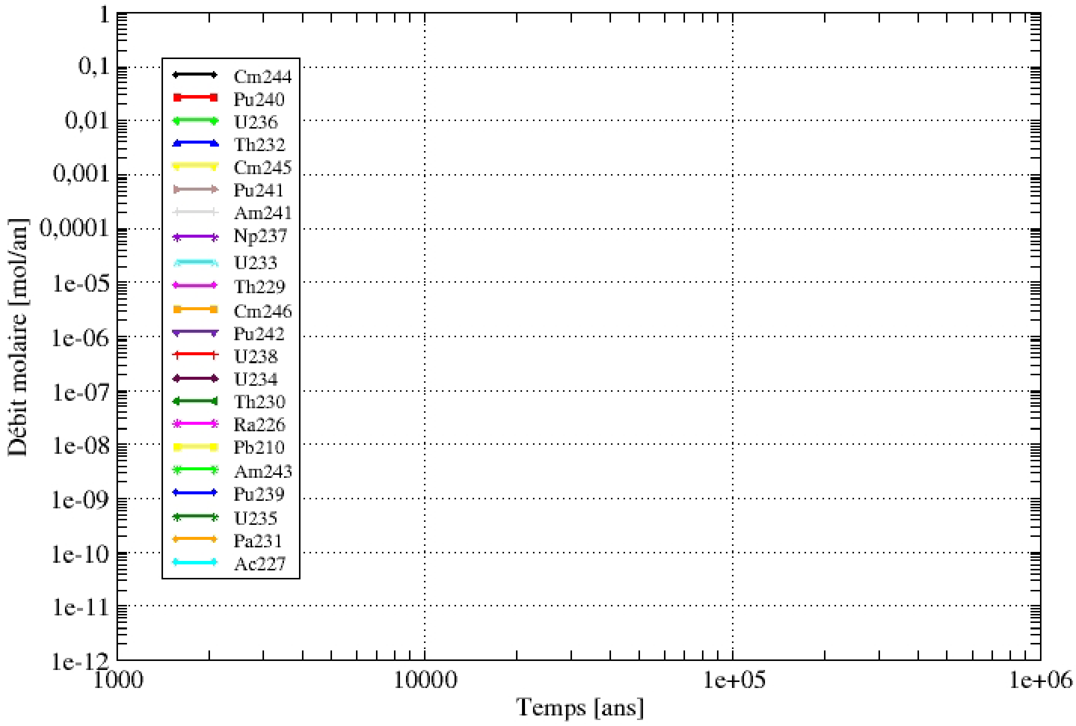


Figure 5.5-10 Historique des debits molaires sortant du Callovo-Oxfordien – Colis type CU1 Uox3

5.5.4.4 Summary of safety functions

At this stage of the analysis, it is possible to appreciate the performance of safety function. We observe that :

- The « resisting water circulation » function is efficiently ensured, since access paths to the repository are not the preferential migration paths. The sensitivity studies and the calculations of the altered evolution scenario allow appreciation of the robustness by testing the respective contribution of seals, host formation and the cul-de-sac architecture ;
- The « limiting the release of radionuclides and immobilizing them in the repository » function allows the retention of elements having weak solubility, in the C waste and spent fuel disposal cells. In addition, the interest of the function vis-à-vis management of heat transfer is important qualitatively (management of uncertainties in this field), but cannot be translated in terms of impact limitation with the data used at present ;
- The « delaying and reducing the migration of radionuclides » function highlights the preponderant importance of the host formation, which limits to four the radionuclides that can effectively exit at the end of over 150 000 years (in fact, mainly two : ^{129}I and ^{36}Cl). Due to its (weak) iodine sorbing capability, the B disposal cell concrete has a visible role. In normal situation, the clay engineered barrier of spent fuel does not contribute additional efficiency with respect to an equivalent thickness of Argillite.

The radionuclides that have exited from the repository system can induce an impact, as is stated in the next section.

5.5.5 Results associated with impact

5.5.5.1 Transport in surrounding formations

No safety function is attributed to the surrounding formations at the top and bottom of the Callovo-Oxfordian layer : they simply represent transfer paths to the potential outlets for any radionuclides that reach them.

We will start by presenting transport as part of the million-year geo-prospective model. In the second step, results corresponding to the model envisaging the current configuration of the surrounding formations will be exposed.

- **Transfer model in surrounding formations in a million years**

The Figure 5.5-11 and Figure 5.5-12 recall the position of trajectories and outlets as well as the advective transfer time up to the potential outlets, for upper or lower surrounding formations.

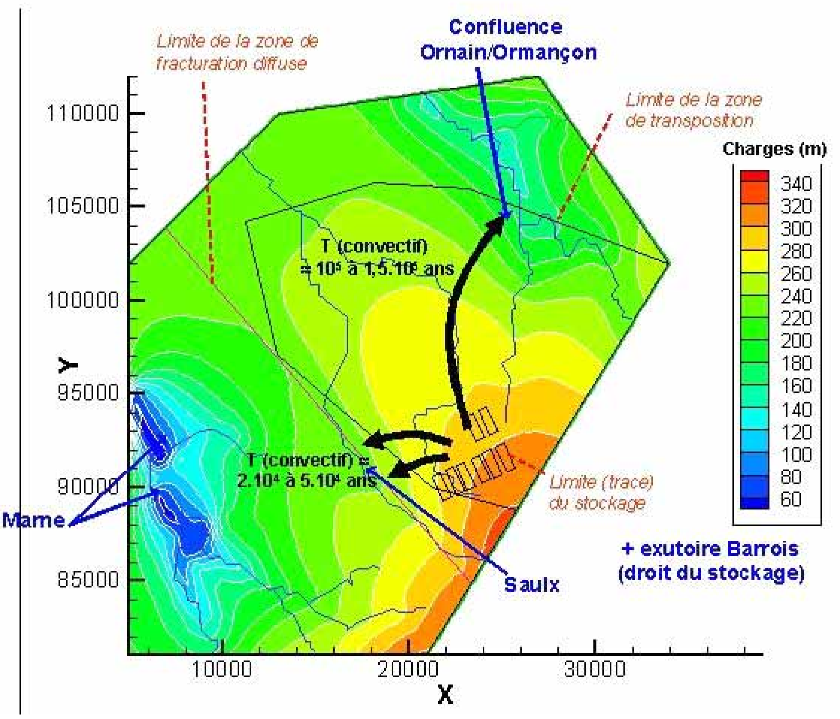


Figure 5.5-11 SEN – piezometry (hydraulic head in NGF[74] meters) and principle advective trajectories in the Oxfordian, and advective transfer time up to the outlets, given the million-year geo-prospective description

[74] NGF metres used in the calculation to quantify hydraulic loads, designate the altitude of surface of a water table that would present the same load as the point thtat is being considered.

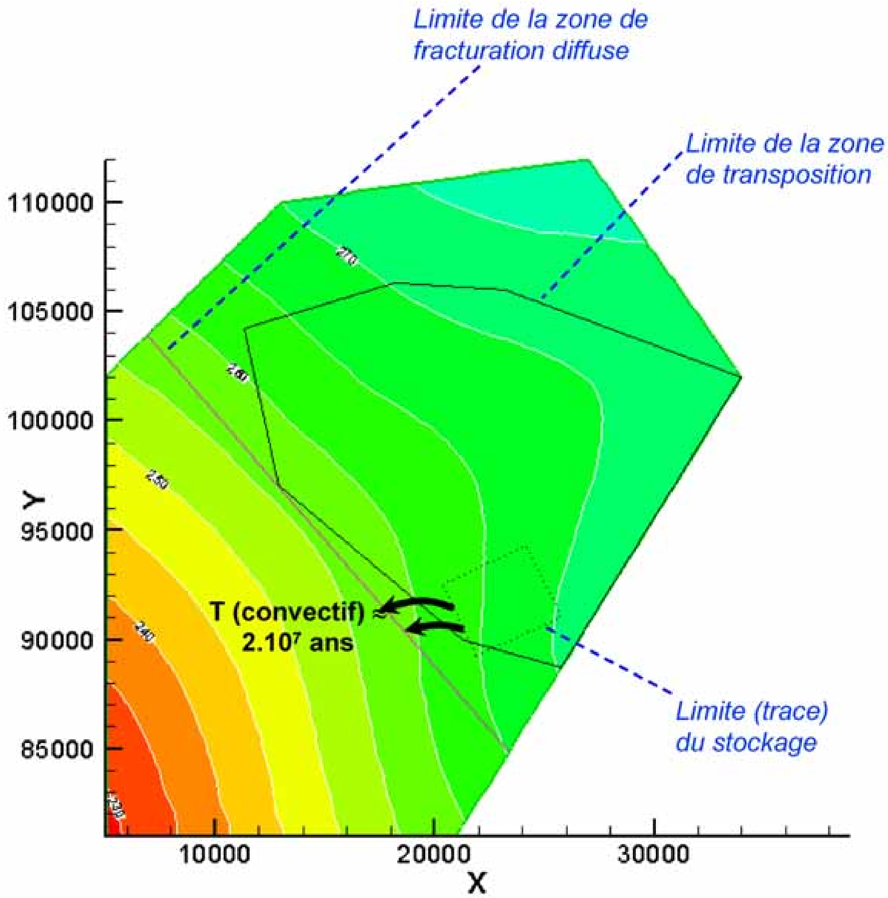


Figure 5.5-12 SEN – Advective trajectories in the Dogger and location of the outlet

The advective transfer time up to the Saulx and Ornain outlets is :

- 20 000 to 50 000 years for the Saulx outlet. These advective transfer times refer to the low distance from the repository to the outlet and are not of the type to engender large dilution and dispersion in the surrounding formations (as we know, the outlet was placed close to the repository, with precisely this objective) ;
- 100 000 years for the outlet positioned in the Ornain valley.

The Figure 5.5-13 illustrates the concentration plumes at different dates, in the porous horizons of the Oxfordian Hp1-Hp4 for iodine 129 of the CU1 spent fuel. In this horizontal section, it shows both transfer paths (one to the diffuse fracturing zone, other towards Ornain), as shown in Figure 5.5-11. The vertical transfer paths towards Barrois is not represented. In this figure we see relatively low concentration gradients in the Oxfordian, between the location of the repository and the outlets ; these results confirm the fact that the transfer in the surrounding formations up to the outlets does not induce notable additional dilution of the mass exiting from Callovo-Oxfordian around the repository.

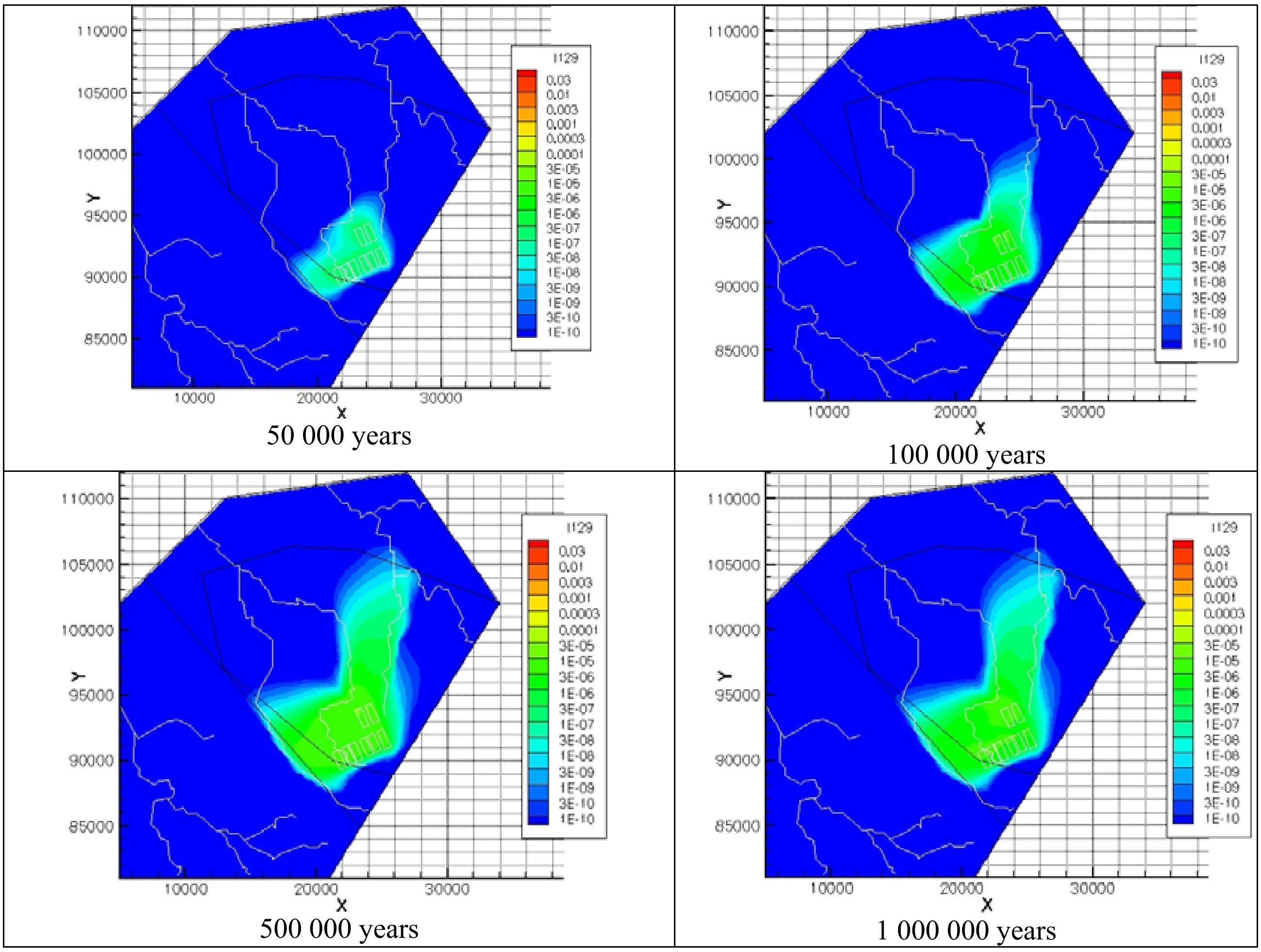


Figure 5.5-13 SEN – Molar concentration plumes of iodine 129 (mol/m^3) in the porous level representing Hp1-Hp4 (horizontal section) at different dates – CU1

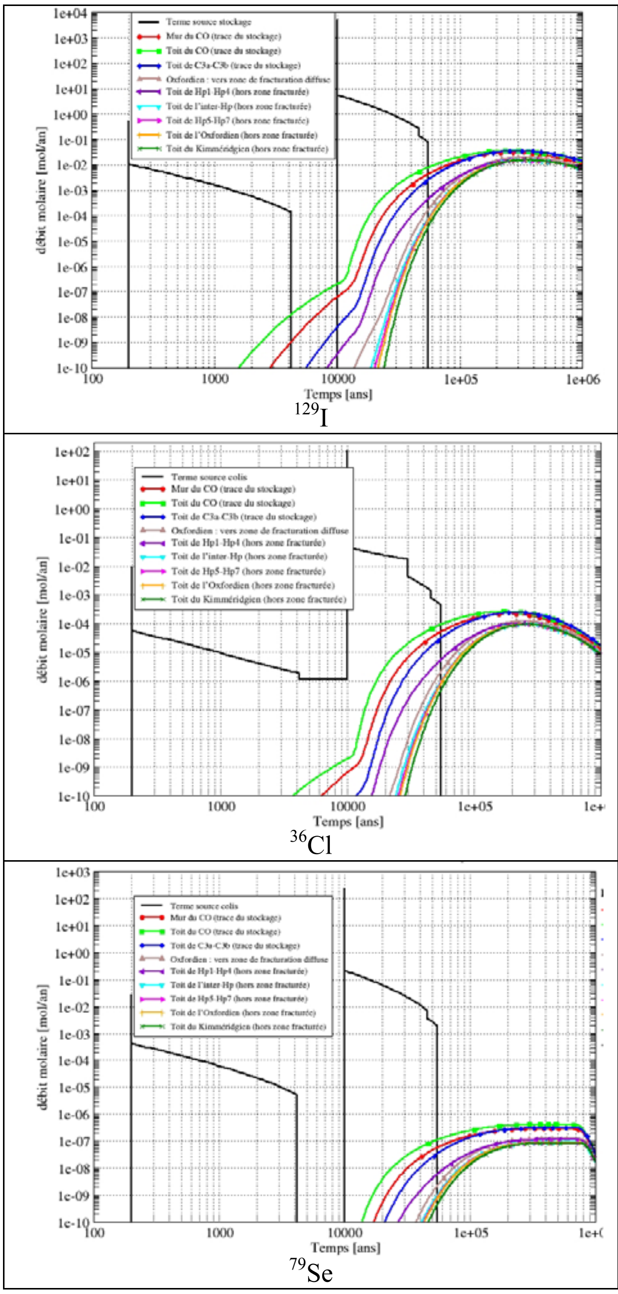
For this repository zone, the distribution between the mass reaching the Saulx and Ornain/Barrois outlets is 55 % and 45 % respectively. Taking into account the low transfer times and the low dilution for the Saulx outlet, the latter corresponds to the outlet that engenders the higher impact (at least one order of magnitude as compared to other outlets).

Distribution of mass between the two outlets depends on the considered waste zones. In fact, it is function of the location, but also of the geometric extension of the repository zone. All zones are considered to be located at the Meuse /Haute-Marne laboratory site level, only their size affects in the calculations. Result analysis shows that for repository zones having a low geometric extension (B and C0 wastes), no impact appears at the Ornain outlet since the plume extension in the Oxfordian is not sufficient to be intercepted by the hydraulic trajectories susceptible to reach that outlet. For other repository zones, with intermediate extension (other C wastes, CU1 or CU2 fuel), the distribution shows dominance of the Saulx outlet with respect to others.

The illustrations in Figure 5.5-14 give an example for the CU1 reference package :

- of the molar flows as a function of time for the three main radionuclides dominating the impact : ^{129}I, ^{36}Cl and ^{79}Se, as well as the release (« package source term ») (1 failed package and all others non-failed) ;
- of the standardised quantity (with respect to the initial quantity) of radionuclides having transited through the main components from the package up to the outlets, in a million years, used to measure the attenuation and distribution capabilities of the radiological activity on this date. For example, for iodine 129 of CU1 reference package, we observe that in a million years, nearly all the mass entering the first horizons of the C3a-C3b Oxfordian (to which we have not attributed geochemical properties in the calculations) is yielded in the Hp1-Hp4 porous horizons. A little

more than half of this mass (about 22 %) arrives close to the diffuse fracturing zone after transfer in the porous and inter-porous horizons (Saulx and Marne outlets), the balance (around 18 %) takes the transfer path towards the Ornain outlet and to the overlying formations (Barrois outlet).



	sortie colis	sortie BO	sortie COX	sortie C3a-C3b
I129	100,0%	100,0%	***40,3%***	39,5%
Nb94	36,0%	atténuation totale	sans objet	sans objet
Ni59	93,4%	57,6%	atténuation totale	sans objet
Cl36	96,7%	96,6%	***16,6%***	15,5%
Se79	84,9%	0,5%	***0,013%***	0,010%

Vers exutoires Ornain et Barrois

	sortie TOIT OXFORDIEN + ORNAIN
I129	17,0%
Nb94	sans objet
Ni59	sans objet
Cl36	6,2%
Se79	0,003%

Vers exutoire Saulx de l'Oxfordien

	sortie LIMITE ZONE DE FRACTURATION DIFFUSE
I129	21,5%
Nb94	sans objet
Ni59	sans objet
Cl36	7,9%
Se79	0,004%

Figure 5.5-14 SEN - CU1 reference package – History of molar flows through different surfaces of the geological medium and quantification of the attenuation of different formations for the million-year model

In the Dogger, (see example in Figure 5.5-15) most of the mass migrates by diffusion towards the bottom. Molar flow analysis show that 0.03 % of the mass transits in the layer with strongest permeability (10^{-8} m/s) up to the level of the conventional outlet. This transmissive layer presents to small a thickness (5 metres), as compared to its immediate surrounding formations (terminal and basal Dogger) having lower permeability (10^{-10} m/s), to constitute a preferential drain of radionuclides. Advective transfer times in the Dogger are very long, the molar flow maxima, already weak, only appear after the end of a million years.

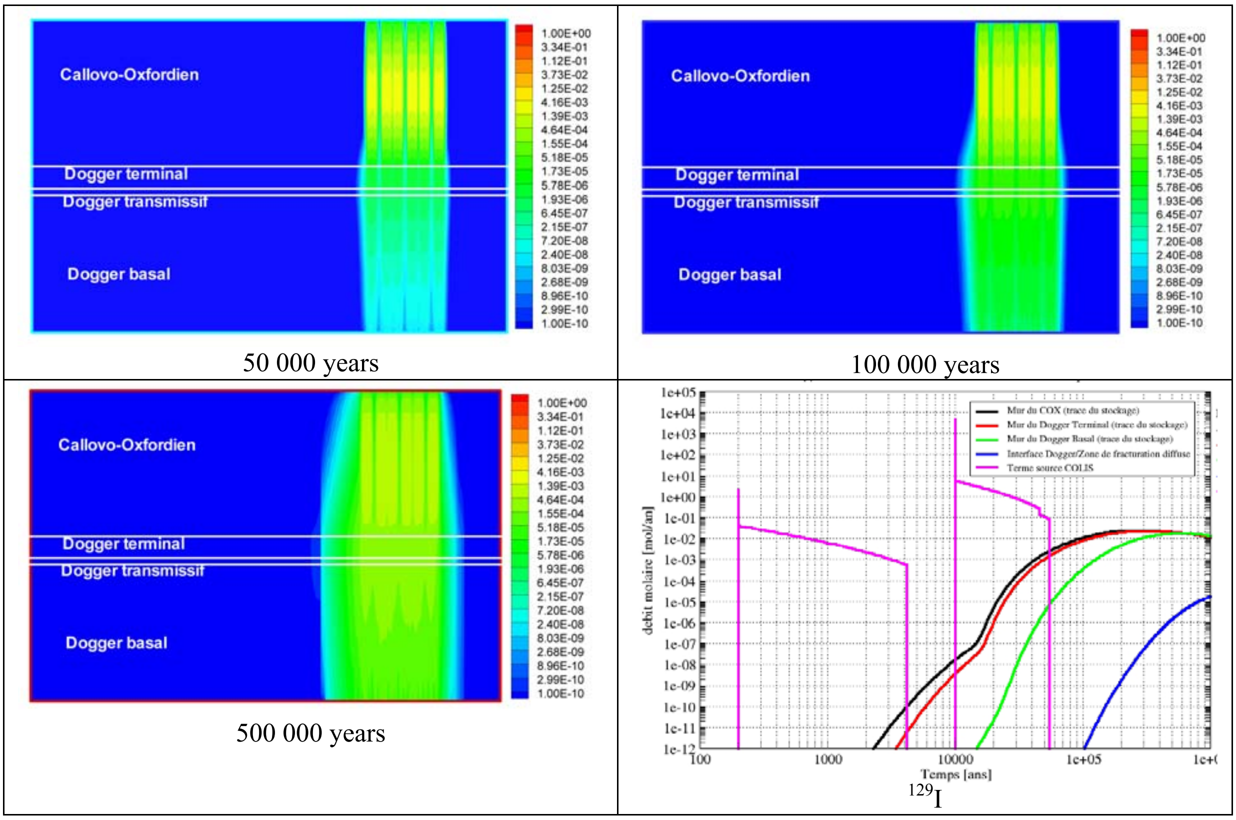


Figure 5.5-15 SEN - Concentration plumes in different Dogger horizons (vertical section) at different dates and history of the molar flow through the different surfaces - Dogger model– CU1 reference package – ^{129}I

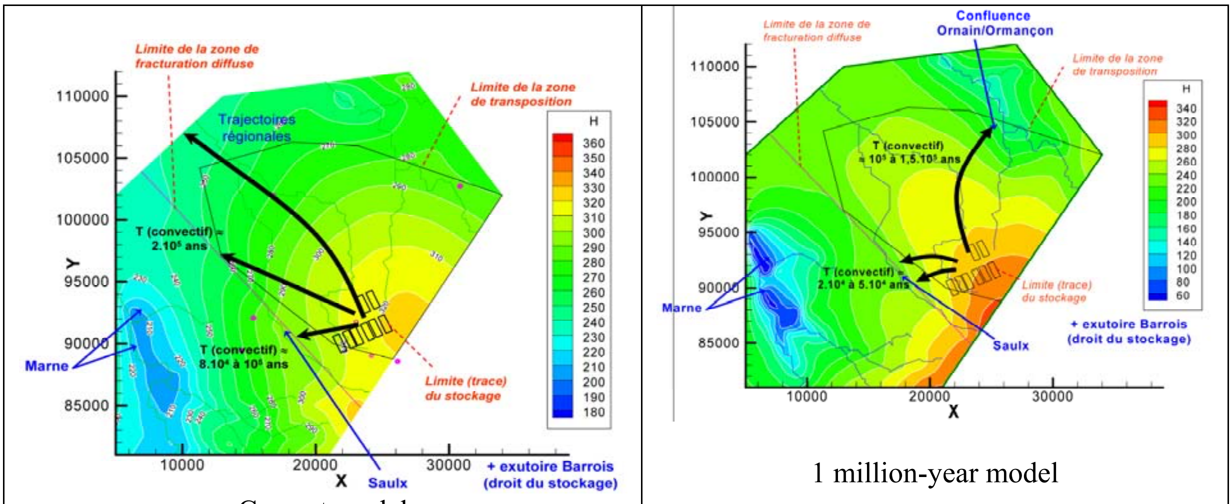
- **Transfer model in the « current » surrounding formations**

We now compare the results obtained prior to those that would result from the use of a « current » model. We observe that the modification of the vertical gradient in the Callovo-Oxfordian with the current model (0.2 m/m instead of 0.4 m/m) does not make any difference to the transfer up to the top of the host formation, the transfer is diffusive in both cases. We will see that the results are hardly sensitive to the retained Hydrogeological model.

If we consider the « current » flow model, we observe that, in comparison with the million year model :

- a low fraction of radionuclides (in case of iodine, 5 % of those that go towards the Saulx outlet) go towards the Marne with greater transfer times (up to 300 000 years if the trajectories go up to the Marne, but they are interrupted earlier in the model, in a conventional manner, at the diffuse fracturing zone level) ;
- a fraction of radionuclides (around 0.1 % of the total inventory in case of iodine) migrates in a regional direction, not associated to any outlet. It is pointed out that, even if these elements were conventionally driven towards the closest outlet (Saulx, which does not correspond to any physical reality), the orders of magnitude of the impact would not be modified ;
- the advective transfer time in the Oxfordian limestone from the top of the repository up to the Saulx outlet (only identical outlet for both models) is shorter in the million year model than in the current model (See Figure 5.5-16). This translates into the maxima of the molar flow being shifted by a little less than 100 000 years at the diffuse fracturing zone level (Saulx outlet) (see Figure 5.3-14),
- the Ornain outlet does not exist in the « current » model.

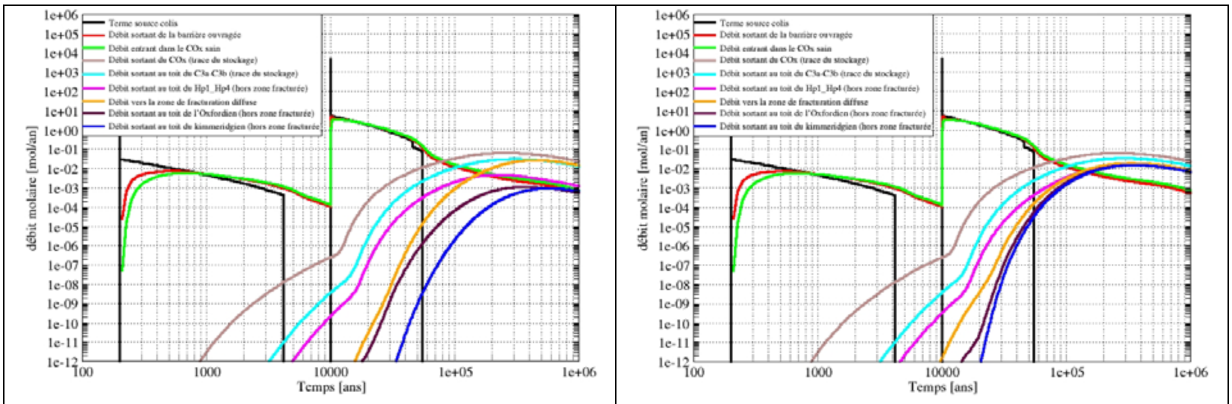
The differences between the two models ultimately seem of little significance in case the repository is positioned on the conventional laboratory site.



Current model

1 million-year model

Figure 5.5-16 SEN – Sensitivity to the Hydrogeological model – hydraulic cartography of heads (H = hydraulic head in NGF[74] meters) in Hp1-Hp4 and associated outlets



Current model

1 million-year model

Figure 5.5-17 SEN – CU1 reference package - ^{129}I – history of the molar flow in both configurations of the Hydrogeological model

5.5.5.2 Radiological impact on the critical group conforming to RFS III.2.f recommendations with late maxima dates

Dose associated with the radionuclides exiting from Callovo-Oxfordian is assessed at the outlets identified in the section 5.3.2.5 and for the critical group described in the section 5.3.2.6.

We observe for all the packages that, in the « million-year » Hydrogeological model (see Table 5.5-8 and Table 5.5-9) :

- the Saulx outlet is where we observe the highest doses. The outlets used in the Ornain and in Barrois give lower impacts ;
- the very low impact level in the Barrois is explained by the diffusive character of transport and the spread associated with it ;
- reference packages where the geometric extension of the repository is weak (B and C0 waste), do not engender impact at the Ornain outlet ;

- for radionuclides not totally attenuated in the Callovo-Oxfordian, both main contributors are Iodine 129 and Chlorine 36 and to a lesser degree Selenium 79, which benefits from radioactive decay. Calcium 41 presents a negligible dose for all the waste packages (less than 10^{-12} Sv/year), moreover, the latter has a very low conversion factor.

The results show that the dose maxima dates appear beyond 290 000 years. At the Saulx outlet, these dates directly retranslate the transfer time induced by the geological barrier. Characteristic migration times through advection in the Oxfordian limestone up to this outlet are effectively short in comparison with the migration time in the host formation, such that they do not bring about significant spread complementary to the appearance of maximum dose.

We must underline the fact that the addition of different impacts do not account for an actual situation since it counts the same inventory twice (we cannot have C1/C2 and C3/C4 glasses in maximum quantities at the same time in the inventory). Whatever be the case, we will see that even with this bias to dose constraints, global estimations will remain lower.

In the table below, we give the doses for all the outlets of the million-year model, with associated dates. We present the most pessimistic Saulx outlet separately. Other outlets are presented next, including for Ormain.

Reference packages	Maximum dose (mSv/year)	Date of maximum [years]	Contributing radionuclides
« Saulx » outlet (the most pessimistic)			
B1x - (non-organic package not giving off gaseous hydrogen)	0.00033	310 000	^{36}Cl (^{129}I to a lesser degree)
B1h - (non-organic package that can give off gaseous hydrogen)	0.000031	290 000	^{36}Cl
B2 – (bituminised sludges)	0.000021	370 000	^{129}I ; ^{36}Cl
Other B waste	0.00009	310 000	^{36}Cl (^{129}I to a lesser degree)
Total of B waste (scenario S1b)	around 0.00047	towards 300 000	^{36}Cl (^{129}I to a lesser degree)
Glasses C0	0.0000032	340 000	^{129}I
Glasses C1 and C2	0.00047	490 000	^{129}I ; ^{36}Cl
Glasses C3 and C4	0.00036	500 000	^{129}I ; ^{36}Cl
Total of C waste (scenario S1b for C0/C1/C2 and S1b for C3/C4)	around 0.00083	490 000	^{129}I ; ^{36}Cl
Spent fuel CU1	0.019	330 000	^{129}I
Spent fuel CU2	0.0017	340 000	^{129}I
Spent fuel CU3	0.000067	330 000	^{129}I
Total of spent fuel (scenario S2)	around 0.02	towards 330 000	^{129}I

Table 5.5-8 SEN – Total dose – date of maximum dose and main contributors at the Saulx outlet of the Oxfordian (most pessimistic case) – million-year model – all wastes

Other outlets :

Reference packages	Maximum dose (mSv/year)	Date of maximum [years]	Contributing radionuclides
« Ornain » outlet as a reference			
Total B waste (S1b)	No Ornain outlet		
Total C waste [C1/C2 (S1b) and C3/C4 (S1a)] (No C0 Ornain outlet)	Around 0.000002	660 000	^{129}I ; ^{36}Cl
Total spent fuel (without CU3) (S2)[75]	0.0006	towards 500 000	^{129}I
« Barrois » outlet			
Total B waste	around 0.000013	towards 300 000	^{36}Cl (^{129}I to a lesser degree)
Total C waste	0.0000055	500 000	^{129}I ; ^{36}Cl
Total spent fuel (without CU3)[75]	around 0.000099	towards 530 000	^{129}I
« Dogger » outlet			
Total B waste	All doses < 10^{-7} mSv/year		Not applicable
Total C waste	All doses < 10^{-7} mSv/year		Not applicable
Total spent fuel	0.000016	1 000 000	^{129}I

Table 5.5-9 SEN – Total dose – date of maximum dose and main contributors at the other outlets – 1 million-year model – all wastes

These results confirm that the impact respects the constraints of the individual dose limit of 0.25 mSv/year. In addition, we observe that all doses arise beyond 10 000 years for which RFS recommends strict compliance with the constraints. In addition we note that the maximum impact is associated with spent fuel, which is natural as far as it is driven by iodine 129, that is present there in larger quantities than in the glasses or in B waste.

Results associated with the « current » model (see Table 5.5-10 and Table 5.5-11) highlight that :

- only the Saulx outlets and to a lesser degree the Barrois and « interception towards the Marne », are expressed ;
- the main radiological impact is due to iodine 129. Impact is very similar to that observed at a million years, the molar flows of radionuclides going towards Saulx being similar.

[75] Since their impact is negligible in comparison with other spent fuel, the dose associated with CU3 is only calculated for the Saulx outlet

Reference packages	Maximum dose (mSv/year)	Date of maximum [years]	Contributing radionuclides
« Saulx » outlet (the most pessimistic)			
B1x - (non-organic package not giving off gaseous hydrogen)	0.00032	370 000	^{36}Cl (^{129}I to a lesser degree)
B1h - (non-organic package that can give off gaseous hydrogen)	0.00003	350 000	^{36}Cl
B2 – (bituminised sludges)	0.000022	440 000	^{129}I ; ^{36}Cl
Other B waste	0.000093	370 000	^{36}Cl (^{129}I to a lesser degree)
Total of B waste	around 0.00047	towards 370 000	^{36}Cl (^{129}I to a lesser degree)
Glasses C0	0.0000036	400 000	^{129}I
Glasses C1 and C2	0.00045	540 000	^{129}I ; ^{36}Cl
Glasses C3 and C4	0.00035	560 000	^{129}I ; ^{36}Cl
Total of C waste	around 0.0008	towards 550 000	^{129}I ; ^{36}Cl
Spent fuel CU1	0.02	410 000	^{129}I
Spent fuel CU2	0.0017	400 000	^{129}I
Spent fuel CU3	0.000073	400 000	^{129}I
Total of spent fuel	around 0.022	towards 400 000	^{129}I

Table 5.5-10 SEN – Total dose – date of maximum dose and main contributors at the Saulx outlet of the Oxfordian (most pessimistic case) – current model – all wastes

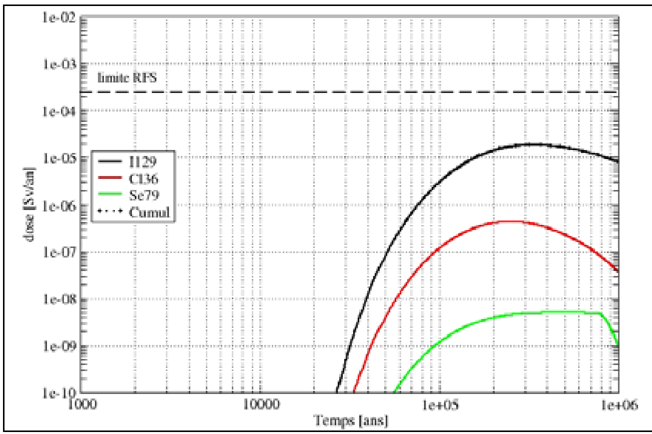
- Other outlets :

Reference packages	Maximum dose (mSv/year)	Date of maximum [years]	Contributing radionuclides
« Intercepting trajectories towards Marne » outlet			
Total B waste	0.0001	towards 400 000	^{36}Cl ; ^{129}I
Total C waste	0.00033	630 000	^{129}I ; ^{36}Cl
Total spent fuel	0.0073	500 000	^{129}I
« Barrois » outlet			
Total B waste	0.000044	towards 400 000	^{36}Cl (^{129}I to a lesser degree)
Total C waste	0.0002	600 000	^{129}I ; ^{36}Cl
Total spent fuel	0.0067	500 000	^{129}I
« Dogger » outlet			
Total B waste	All doses < 10^{-7} mSv/year		Not applicable
Total C waste	All doses < 10^{-7} mSv/year		Not applicable
Total spent fuel	0.000016	1 000 000	^{129}I

Table 5.5-11 SEN – Total dose – date of maximum dose and main contributors to the other outlets – current model – all wastes

The radiological impacts assessed at the Saulx outlet for different waste are presented below in the form of curves for different reference packages (for the one million-year model, the curves for the current model being nearly identical).

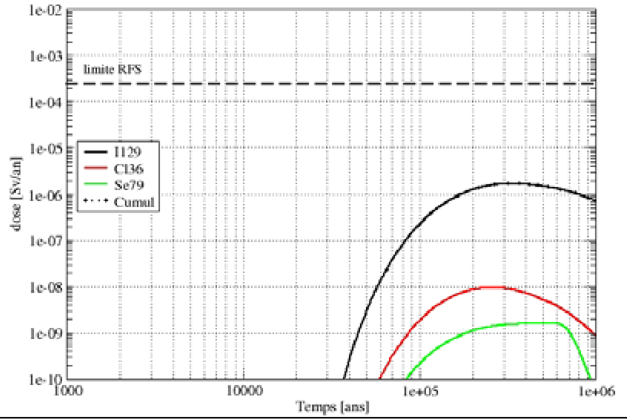
- **Spent fuel CU1**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.9E-05	3.3E+05
I129	1.8E-05	3.4E+05
Cl36	4.3E-07	2.5E+05
Se79	5.0E-09	4.7E+05

Figure 5.5-18 SEN – Reference calculation – million year model - Doses at the Saulx outlet of the Oxfordian – CU1 reference package

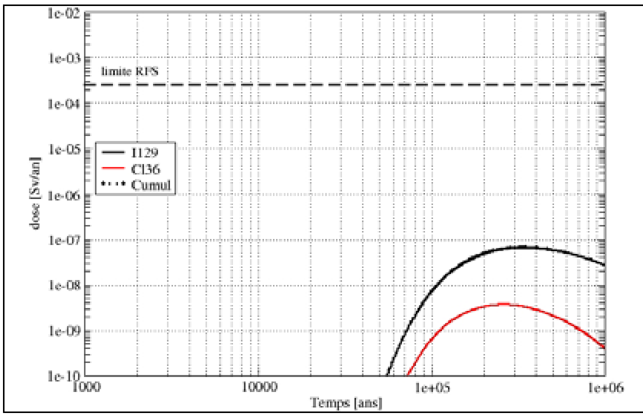
- **Spent fuel CU2**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.7E-06	3.4E+05
I129	1.7E-06	3.4E+05
Cl36	9.6E-09	2.6E+05
Se79	1.6E-09	5.6E+05

Figure 5.5-19 SEN – Reference calculation – million year model - Doses at the Saulx outlet of the Oxfordian – CU2 reference package

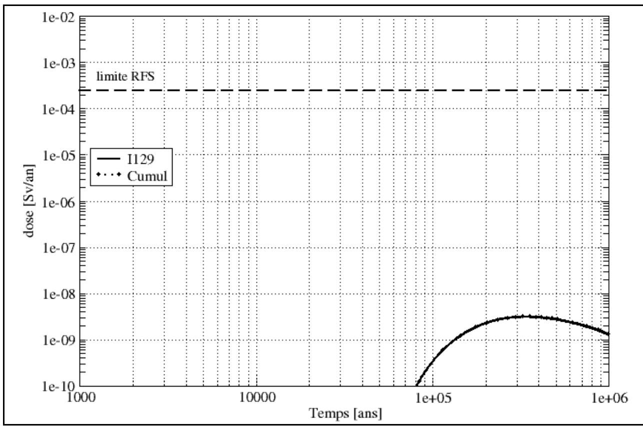
- **Spent fuel CU3**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cl36	3.7E-09	2.6E+05
Se79	dose < 1e-10	sans objet
I129	6.4E-08	3.4E+05
Cumul	6.7E-08	3.3E+05

Figure 5.5-20 SEN – Reference calculation – million year model - Doses at the Saulx outlet of the Oxfordian – CU3 reference package

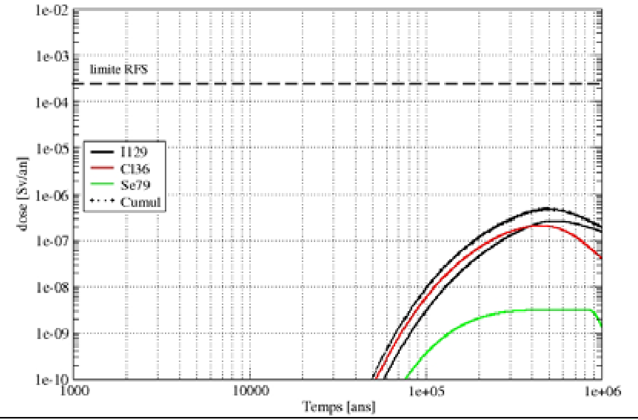
- **C0 vitrified waste**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	3.2E-09	3.4E+05
I129	3.1E-09	3.4E+05
Se79	dose < 1e-10	sans objet
Cl36	dose < 1e-10	sans objet

Figure 5.5-21 SEN – million year model - Doses at the Saulx outlet of the Oxfordian – C0 reference packages

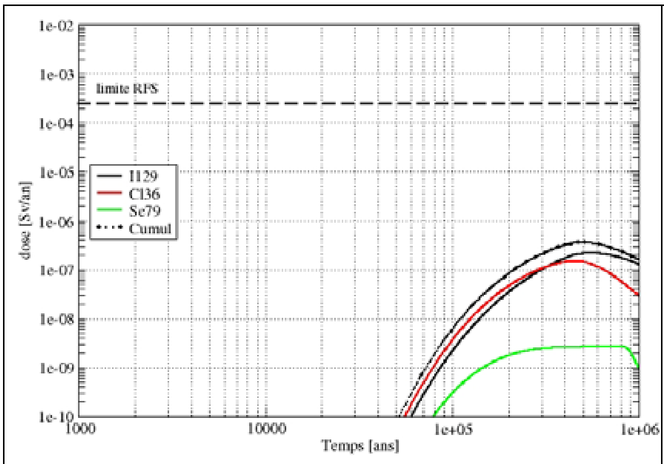
- **C1 + C2 vitrified waste**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	4.7E-07	4.9E+05
I129	2.6E-07	5.5E+05
Cl36	2.1E-07	4.5E+05
Se79	3.2E-09	8.0E+05

Figure 5.5-22 SEN – million year model - Doses at the Saulx outlet of the Oxfordian – Reference packages (C1+C2)

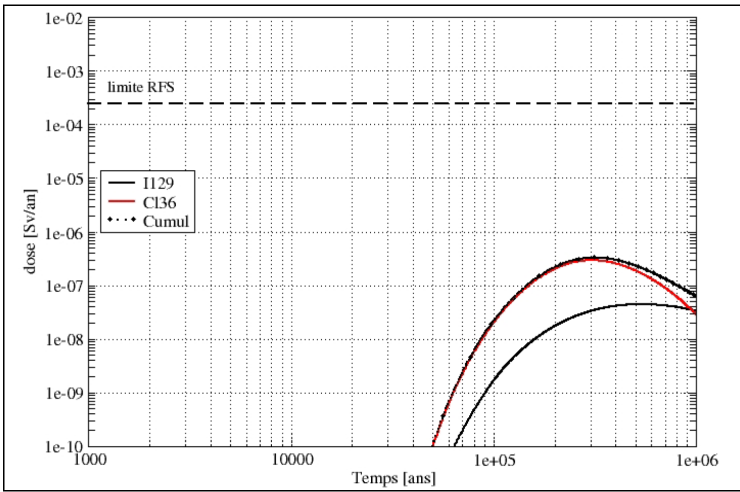
- **C3 + C4 vitrified waste**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	3.6E-07	5.0E+05
I129	2.2E-07	5.6E+05
Cl36	1.5E-07	4.6E+05
Se79	2.7E-09	7.4E+05

Figure 5.5-23 SEN – Reference calculation – million year model - Doses at the Saulx outlet of the Oxfordian – Reference packages (C3+C4)

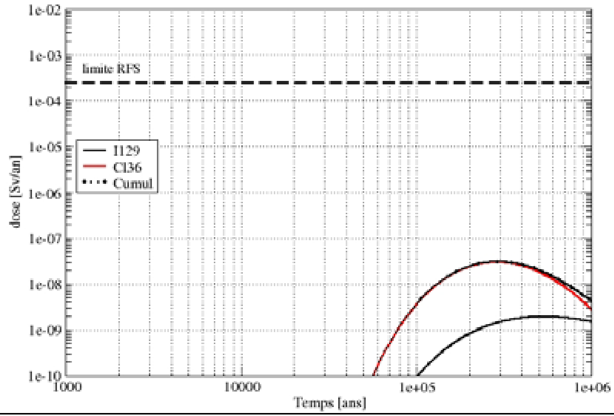
- **Reference packages B1x**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	3.3E-07	3.1E+05
Cl36	2.9E-07	3.1E+05
I129	4.4E-08	5.3E+05
Se79	dose < 1e-10	sans objet

Figure 5.5-24 SEN – Reference calculation – million year model - Doses at the Saulx outlet of the Oxfordian – Reference packages (B1x)

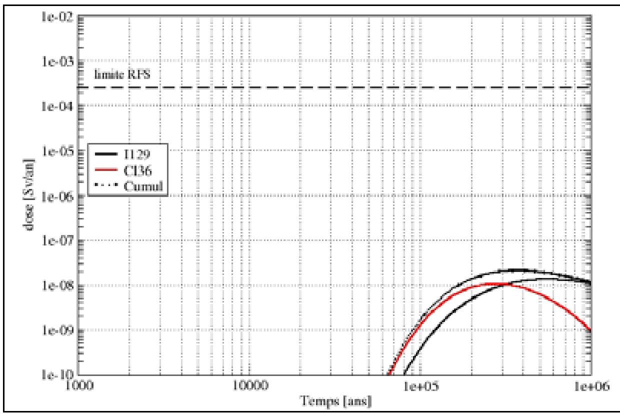
- **Reference packages B1h**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	3.1E-08	2.9E+05
Cl36	3.0E-08	2.9E+05
I129	1.9E-09	5.3E+05
Se79	dose < 1e-10	sans objet

Figure 5.5-25 SEN – Reference calculation – million year model - Doses at the Saulx outlet of the Oxfordian – Reference packages (B1h)

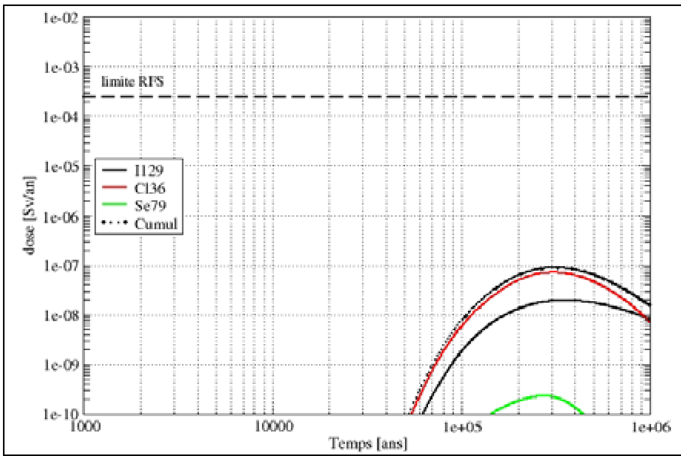
- **Reference package B2**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	2.1E-08	3.7E+05
I129	1.3E-08	5.5E+05
Cl36	1.0E-08	2.8E+05
Se79	dose < 1e-10	sans objet

Figure 5.5-26 SEN – Reference calculation – million year model - Doses at the Saulx outlet of the Oxfordian – B2 Reference package

- **Other B waste**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	9.0E-08	3.1E+05
Cl36	7.1E-08	3.1E+05
I129	1.9E-08	3.5E+05
Se79	2.3E-10	2.7E+05

Figure 5.5-27 SEN – Reference calculation – million year model - Doses at 1 million years at the Saulx outlet of the Oxfordian – other B reference packages

5.5.5.3 Impact of toxic chemical also controlled

- **Input data**

The chemical impact of the four toxic agents under study (B, Ni, Se, and Sb) was evaluated for C1/C2 waste packages that contain – at least for boron and selenium – the highest quantities of toxic agents.

Antimony is present in similar quantities in the vitrified waste packages, the technological waste, and the spent fuels.

We studied the case of vitrified waste packages, which is representative even if it is not completely bounding (the ratio of antimony inventories with respect to C waste packages is 1.6 for the spent fuels and 1.3 for the B waste packages). We shall see subsequently, however, that the retention characteristics (strong sorption in the argillites of the Callovo-Oxfordian) provide complete confinement of this toxic agent within the first meters of the Callovo-Oxfordian, and the conclusions of the calculation would not be modified if all the waste were taken into account.

Nickel, on the other hand, is contributed by the metal wastes, and mainly by the steels included in the repository (containers, sleeves, etc.). It is difficult at this stage of our study to define a sufficiently precise inventory of nickel quantities and to assign release kinetics to each potential source. In order to overcome these uncertainties, Andra has adopted a pessimistic approach, in which a saturation concentration of nickel is imposed in a continuous manner until the end of the simulation (1,000,000 years) throughout the entire repository. As a result of this choice, 13,600 metric tons of nickel are placed in solution in the C1/C2 repository modules that are included in the calculation.

The inventories considered for each of the chemical toxins studied are listed inTable 5.5-12.

	Total mass (metric tons)	Distribution for C1/C2 type waste packages	
		C1 mass (metric tons)	C2 mass (metric tons)
Boron (C1/C2)	526.90	76.56	450.34
Nickel (C1/C2)	13,600 m.t. in solution over 1,000,000 yrs.	Not applicable	Not applicable
Selenium (C1/C2)	3.18	0.46	2.72
Antimony (C1/C2)	0.57	0.08	0.49

Table 5.5-12 SEN – chemical toxin inventory considered for the evaluation of the chemical impact [64]

The selected transport and chemical retention parameter values for these elements are also presented in Table 5.5-13 (also included are the parameter values for the selenium and nickel that were already studied for their radiological impact).

	$K_h = 5.10^{-13}$ m/s $K_v = 5.10^{-14}$ m/s				
	Geological medium	ω□iffusion [-]	De [m²/s]	R [-]	Csat [mol/m^3]
Boron (Bo)	Fractured zone	0.20	$5,0.10^{-10}$	47	soluble
	Micro-fissured zone	0.18	$2,5.10^{-10}$	52	
	Unaltered Callovo-Oxfordian	0.18	$2,5.10^{-10}$	52	
Nickel (Ni)	Fractured zone	0.20	$5,0.10^{-10}$	1800	
	Micro-fissured zone	0.18	$2,5.10^{-10}$	2,050	
	Callovo-Oxfordian sound	0.18	$2,5.10^{-10}$	2,050	5.10^{-02}
Selenium (Se)	Fractured zone	0.15	$1,0.10^{-11}$	1	5.10^{-07}
	Micro-fissured zone	0.05	$5,0.10^{-12}$	1	
	Unaltered Callovo-Oxfordian	0.05	$5,0.10^{-12}$	1	
Antimony (Sb)	Fractured zone	0.20	$5,0.10^{-10}$	46,000	5.10^{-07}
	Micro-fissured zone	0.18	$2,5.10^{-10}$	51,000	
	Unaltered Callovo-Oxfordian	0.18	$2,5.10^{-10}$	51,000	

Table 5.5-13 SEN – value of hydraulic, transport and chemical retention parameters for the chemical toxins under study [64].

- **Transport of chemical toxins**

The results of the transport calculation reveal the following for the chemical toxins :

- the high sorption of the antimony in the Callovo-Oxfordian argillites totally attenuates its molar flow when leaving the host formation. It remains completely confined in the geological barrier for the entire duration of the analysis (1,000,000 years) ; Figure 5.5-28 shows that at 1 million years, the concentration front is located less than 10 meters from the waste packages. The molar flow exiting the Callovo-Oxfordian is nil. The impact on the biosphere is thus nil at 1,000,000 years. It should be noted that these results are obtained based on the geochemical behaviour of this element.

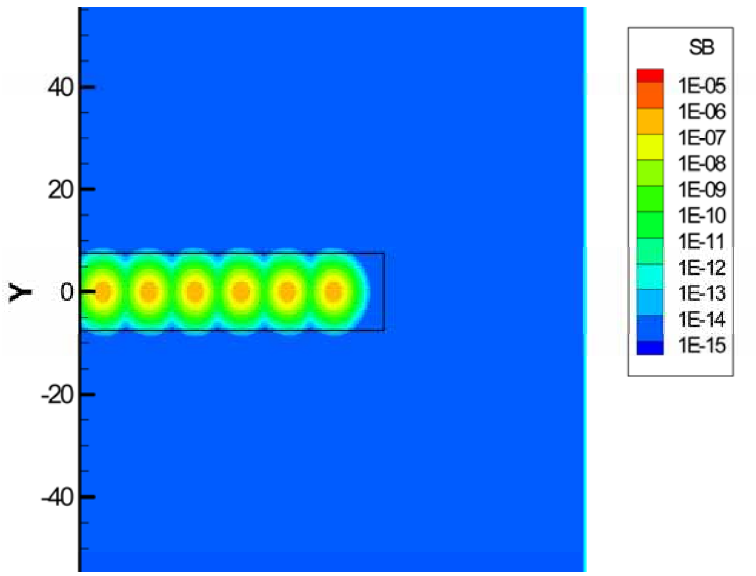


Figure 5.5-28 Vertical 2D cross-section – antimony concentration plumes at 10^6 years

- the relatively high sorption of nickel in the geological barrier significantly delays and attenuates its molar flow ; thus, its maximum molar flow is reached well beyond one million years. It may however reach the surrounding formations - at least in the pessimistic calculation hypotheses selected by Andra.
- selenium, which is limited by its solubility and does not benefit from radioactive decay (like its radioactive isotope – ^{79}Se) reaches its maximum molar flow at approximately 1 million years upon exiting the geological barrier
- the low sorption of boron in the Callovo-Oxfordian (retardation coefficient of about 50) is sufficient to delay its maximum molar debit beyond one million years.

- **Impact of chemical toxins**

The chemical impact results are only presented for the Saulx outlet within the million-year hydrogeological model, given that the results are very close for the so-called « current » model. It should be kept in mind that the calculations were only performed for the vitrified waste packages :

- they are representative for stable boron and selenium, since the glasses are the waste that contains the highest levels,
- they correspond to a significant overestimation for nickel, because the inventory taken into account for the C waste cells alone is in fact greater than the maximum quantity of nickel that may be found within the entire repository

As is the case for the radionuclides, the impact of chemical toxins is much lower in the other outlets.

The results are expressed as follows : as the maximum concentration in the outlet (a magnitude that is not directly indicative of health effects, but which is easier to understand) ; as the Excess of Individual Risk (ERP, for the stochastic effects on health – cancerigenic effects) ; and as the quotient of danger (QD, for risks with threshold effects – non-cancerigenous effects). It should be noted that a risk is considered to be acceptable for an ERP of less than 10^{-5} and a QD of less than 1. These indicators are evaluated with the same critical group as for the radionuclides.

Of the four elements in the study, only nickel is recognized as cancerigenous by inhalation (Table 5.5-14).

	Non cancerigenous effects (QD)		Cancerigenous effects (ERI)	
	Oral (ingestion)	Respiratory (inhalation)	Oral (ingestion)	Respiratory (inhalation)
Boron	Yes	No	No	No
Nickel	Yes	Yes	No	Yes
Selenium	Yes	No	No	No
Antimony	Yes	Yes	No	No

Table 5.5-14 Non-cancerigenous and cancerigenous effects of studied chemical toxins according to method of exposure

The danger quotients and the excess individual risks for each toxin are evaluated based on the following :

- levels of exposure (C) by ingestion or inhalation (which are calculated based on the concentrations of toxins at the outlet)
- danger quotients per unit (QD_{unit}) and excess individual risks per unit (ERI_{unit}), which correspond to the QD and ERI values resulting from the chronic[76] ingestion or inhalation of a concentration of toxins equalling 1µg/l (10^{-3} g/m^3)

[76] In the case of cancerigenous effects, teh ERI is evaluated based on a so-called « lifetime » exposure duration of 70 years.

The calculated impact is thus expressed as follows, based on the previously cited magnitudes :

$ERI = C * ERI_{unit}$ and $QD = C * QD_{unit}$

The danger quotient and excess individual risk values for each toxin are given for inhalation and ingestion in Table 5.5-15

	QD_{unit} [(µg/L)$^{-1}$] for chronic exposure (Non cancerigenous effects)		ERI_{unit} [(µg/L)$^{-1}$] for chronic exposure over one life time (70 years) (cancerigenous effects)	
	Oral (ingestion)	Respiratory (inhalation)	Oral (ingestion)	Respiratory (inhalation)
Boron	$4.6.10^{-5}$	Not applicable	Not applicable	Not applicable
Nickel	$5.6.10^{-2}$	$5.0.10^{-1}$	Not applicable	$2.4.\ 10^{-6}$
Selenium	$2.7.10^{-2}$	Not applicable	Not applicable	Not applicable
Antimony	$6.2.10^{-2}$	$2.5.10^{-3}$	Not applicable	Not applicable

Table 5.5-15 SEN – Values of excess of individual risks and quotients of danger per unit for chemical toxins studied

Table 5.5-16 provides the excess individual risk values and the danger quotient values for each of the toxins under study, for ingestion and inhalation at the Saulx outlet (model over 1 million years).

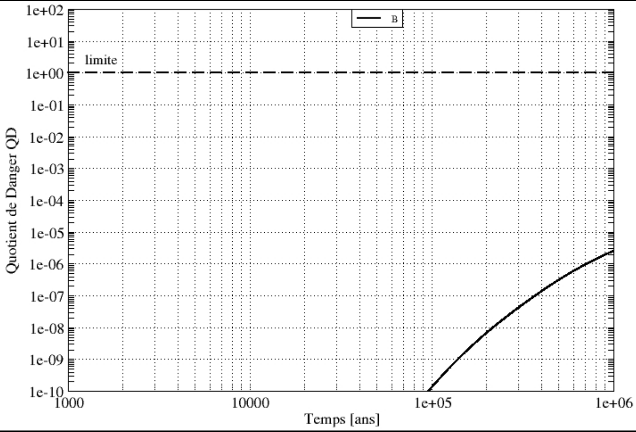
Chemical element (wastes concerned)	Maximum concentration [µg/L]	Date of maximum	QD - Non cancerigenous effects		EIR - Cancerigenous effects	
			Ingestion	Inhalation	Ingestion	Inhalation
Boron (C1/C2)	0.0543	> 10^6 yrs	$2.5.10^{-6}$ at 10^6 yrs	Not applicable	Not applicable	Not applicable
Nickel (C1/C2)	0.001	> 10^6 yrs	$5.8.10^{-5}$ at 10^6 yrs	$5.2.10^{-4}$ at 10^6 yrs	Not applicable	$2.5.10^{-9}$ at 10^6 yrs
Selenium (C1/C2)	0.0000174	Approx. 10^6 yrs	$4.7.10^{-7}$	Not applicable	Not applicable	Not applicable
Antimony (C1/C2)	0	Not applicable	0	0	Not applicable	Not applicable

Table 5.5-16 SEN – Quotient of danger and excess individual risk at Saulx outlet of the Oxfordian limestone for the 5 chemical toxins under study

The results for these toxins, which were chosen amongst those that have the most significant long-term mobilizable quantities inside a repository, show that the chemical impact is acceptable under normal evolution conditions. All of the toxins have an impact that is far below the defined thresholds.

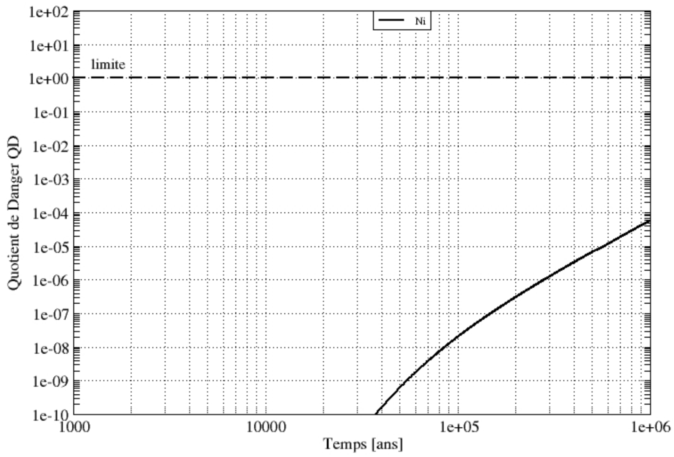
Note that the maximum values for nickel and boron have not been reached at one million years, which confirms the good confinement capacity of the repository. Considering the retardation coefficient of Boron, its maximum impact should be reached at between 1 and 10 million years, and should remain well below the limit (at 10^6 years, the impact of boron is five orders of magnitude below that limit). To evaluate the maximum impact of nickel, the calculations would have to be extended to dates that are so far off that they (the calculations) have no more meaning. Note that at one million years, this chemical element results in an impact that is three orders of magnitude below the limit.

From Figure 5.5-29 to Figure 5.5-33, the impact results of the toxins at the Saulx outlet for boron, nickel, and selenium are illustrated.



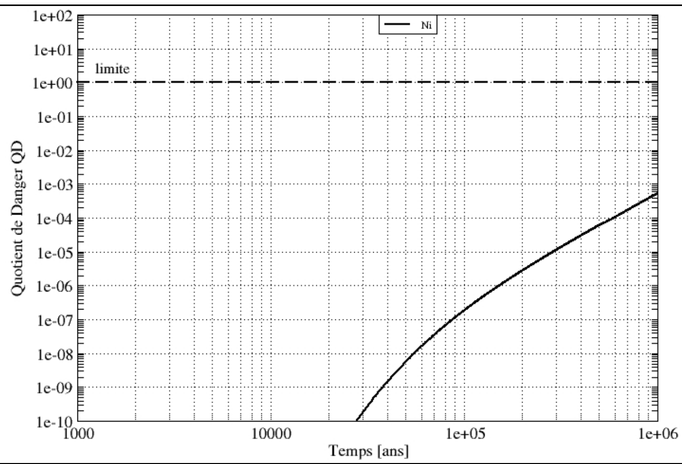
QD at 1 Ma	t(QD max)
$2.5.10^{-6}$	> 10^6 yrs

Figure 5.5-29 SEN – Reference calculation – model to 1 million years – Reference package C1/C2 – Boron – Chemical impact at Saulx outlet of Oxfordian linked to non-cancerigenous effects, for ingestion (QD)



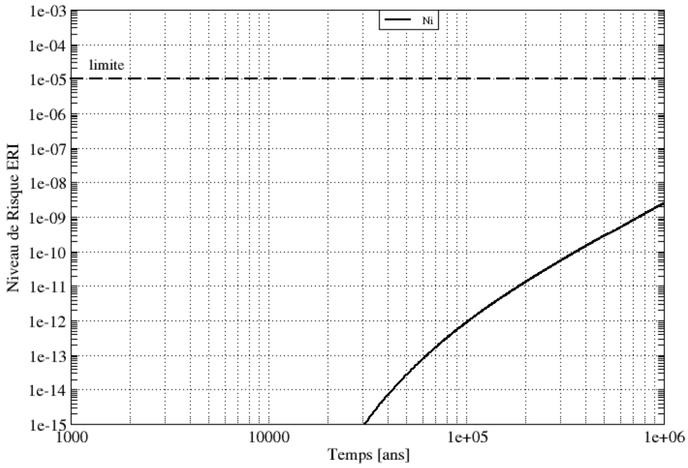
QD at 1 Ma	t(QD max)
$5.8.10^{-5}$	> 10^6 yrs

Figure 5.5-30 SEN – Reference calculation – model to 1 million years – Reference package C1/C2 Nickel – Chemical impact at Saulx outlet of Oxfordian linked to non-cancerigenous effects, for ingestion (QD)



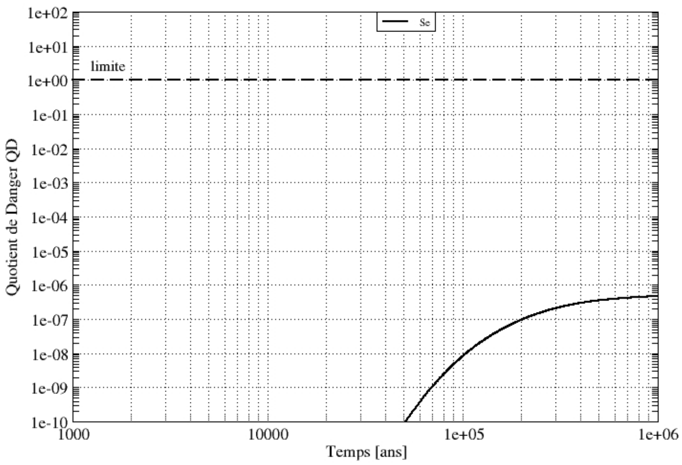
QD at 1 Ma	t(QD max)
$5.2.10^{-4}$	> 10^6 yrs

Figure 5.5-31 SEN – Reference calculation – model to 1 million years – Reference package C1/C2 Nickel – Chemical impact at Saulx outlet of Oxfordian linked to non-cancerigenous effects, for inhalation (QD)



QD at 1 Ma	t(QD max)
$2.5.10^{-9}$	> 10^6 yrs

Figure 5.5-32 SEN – Reference calculation – model to 1 million years – Reference package C1/C2 Nickel – Chemical impact at Saulx outlet of Oxfordian linked to non-cancerigenous effects, for inhalation (ERI)



QD at 1 Ma	t(QD max)
$4.7.10^{-7}$	> 10^6 yrs

Figure 5.5-33 SEN – Reference calculation – model to 1 million years – Reference package C1/C2 Selenium – Chemical impact at Saulx outlet of Oxfordian linked to non-cancerigenous effects, for ingestion (QD)

5.5.5.4 Summary

The calculation results show clearly that the main barrier for the confinement of the radionuclides is the Callovo-Oxfordian, which is the host formation of the repository. It attenuates and delays all the radionuclide flows. It only allows four radionuclides to exit over a time-scale of several hundred thousand years.

The Saulx outlet, which is the source of the highest dose, is located near the repository (approximately 2 kilometres from its boundary). Furthermore, the choices made for the modelling result in the majority of the trajectories being « redirected » towards this conventional outlet. Thus, every effort was made in the calculations to limit the delay and dilution effects of the surrounding formations. Notwithstanding, it was observed that the doses respect the constraint recommended by the RFS III.2.f, which is proof of the performances of the repository itself.

5.5.6 Sensitivity analyses

In addition to the reference calculation, a series of sensibility studies was performed in order to evaluate the influence of the parameter choices or of models that are different from those chosen for the reference calculation. The majority of the studies correspond to models or sets of parameters that are less favorable than those selected for the reference calculation. In this way, any remaining uncertainties about the values selected for the reference calculation, which were already cautious, will be covered.

Other sensitivity studies were also conducted with alternative models or with sets of parameters that are less pessimistic than those of the reference calculation. These studies are conducted either to establish predictions in order to evaluate a potential margin, or in order to integrate recent results that are less cautious than those of the reference calculation.

The sensitivity analyses also make it possible to classify the parameters and models according to their influence on the safety indicators (the impact, or any other intermediary indicator).

The presented sensitivity analyses thus provide useful information for chapter 6, which gives details about existing uncertainties concerning phenomena inside the repository.

The sensitivity studies have been divided into three main categories.

- **Sensitivity studies concerning parameters for the Callovo-Oxfordian, swelling clay, and concrete**

The majority of the considered parameter values are conservative, such as for the following :

- permeability in Callovo-Oxfordian
- the hydraulic, transfer, and retention parameters in the EDZ
- the transfer and retention parameters in the Callovo-Oxfordian, coupled with those of the swelling clay and concrete

Some studies were conducted with values that appear to be less pessimistic.

- **Sensitivity studies concerning the kinetics of release by waste packages**

Only spent fuels, C waste, bituminised sludge packages, and inorganic packages that do not release hydrogen (reference packages disposed in B1x type cells) are included in sensitivity studies, with parameters that are less favourable than those used in the reference calculations. Since the other reference packages were represented by a labile source term in the reference calculation, they do not require such studies.

Furthermore, a sensitivity test is performed for the spent fuels using a model based on the conventional dissolution of the matrix (and not the radiolytic dissolution used in the reference calculation) ; this model results in slower release kinetics.

And finally, as was stated in section 5.3.2.1, a sensitivity study is conducted to make predictions in order to evaluate the potential advantages of adopting durable concrete overpack for waste packages disposed of in B1x cells.

- **Sensitivity studies concerning the overall calculation model.**

This final set of sensitivity studies tests transfer methods for radionuclides other than those considered in the reference calculation. This category includes the following studies :

- Study of a radionuclide transfer under hydraulic transient influence, and thus of stresses caused by gas in particular. Studied for B1x wastes and CU1 spent fuels.
- Study using different properties for the overlying formations, in order to take into account a slower diffusion in the Kimmeridgian and in the C3a horizon of the Oxfordian. Studied for iodine-129 from CU1 spent fuels.
- Study testing the influence of the hydrogeological model of the overlying formations on the impact. Studied for iodine-129 from CU1 spent fuels.

The sensitivity studies are covered below based on the order shown above. They are summarised in Table 5.5-17.

Sensitivities	Section
1. Sensibility to parameters of Callovo-Oxfordian, bentonite, and concrete in cells	
1.1 – Conservative Callovo-Oxfordian permeability (factor of 10)	See page 305
1.2 - EDZ Pessimistic fractured zone : K = 10^{-6} m/s, diffusion coefficient of the water in water (Dp = 2.10^{-9} m²/s), no delay, no solubility Conservative micro-fissured zone : K = 5.10^{-9} m/s, (De = 1.10^{-11} m²/s ; ω = 0.04) ; degraded retention capacities	See page 306
1.3 – Conservative transfer and retention parameters (engineered barrier and Callovo-Oxfordian) : sorption, diffusion and solubility limit for the Callovo-Oxfordian, sorption and solubility limit for the bentonite and the concrete	See page 311
1.4 – Partition coefficient for Iodine of 10^{-3} m³/kg in the Callovo-Oxfordian	See page 317
1.5 – Callovo-Oxfordian thickness of 160 m	See page 318
1.6 – Phenomenological thermal evolution	See page 319
2 – Sensitivity to waste packages	
2.1 - B1x type packages : conservative release rate parameters	See page 320
2.2 - B2 type packages : bituminised sludges : rate = 10^{-3} /year	See page 321
2.3 - B1x type package : durable container used for B wastes	See page 321
2.4 – C waste packages : conservative parameters of model $V_0.S \rightarrow V_r$	See page 323
2.5 - C waste packages : pessimistic model $V_0.S$	See page 323
2.6 – Spent fuels : Conservative parameters (radiolytic dissolution)	See page 325
2.7 – Spent fuels : Control model via the solubility of uranium	See page 325
3 – Sensitivity concerning the overall calculation model	
3.1 – Transfers during hydraulic transient	See page 327
3.2 – Diffusion properties of semi-permeable layers of overlying formations	See page 335
3.3 – Pathways in the overlying formation model	See page 337

Table 5.5-17 SEN – Subjects of the different sensitivity studies

5.5.6.1 Sensibility to Callovo-Oxfordian Parameters

- **Sensibility to hydraulic parameters (conservative permeability)**

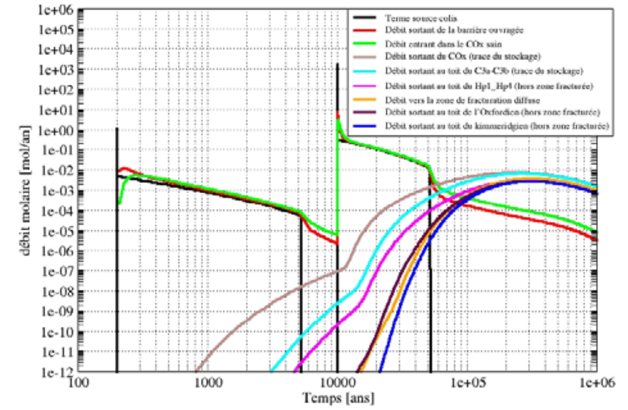
In this sensitivity study, only the permeability (vertical and horizontal) of the unaltered Callovo-Oxfordian is multiplied by 10. All of the other parameters retain their reference values (see Table 5.5-18). Note that no specific dose calculations were performed during this study which was limited to intermediate indicators. The indicator considered is this case was the « molar flow rate » out of the Callovo-Oxfordian.

	Callovo-Oxfordian Permeability	
	Reference value (phenomenological)	Sensitivity value (conservative)
Horizontal component (K_h)	5.10^{-13} m/s	5.10^{-12} m/s
Vertical component (K_v)	5.10^{-14} m/s	5.10^{-13} m/s

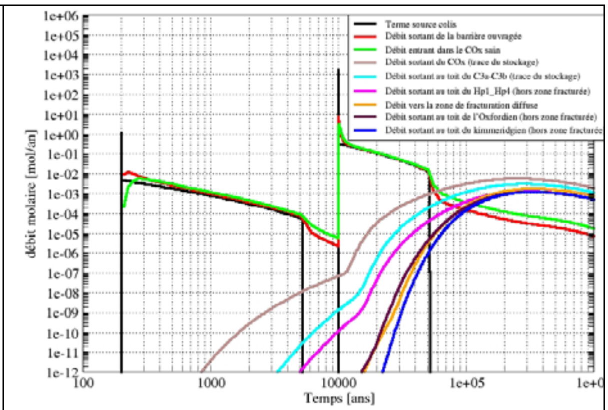
Table 5.5-18 SEN – Sensitivity to permeability of Callovo-Oxfordian.

Using these permeability values for the Callovo-Oxfordian, which is equivalent to uniformly applying some of the lowest recorded values for it, results in an increase of the total molar flow exiting the top of the Callovo-Oxfordian by a maximum factor of 2. The increase is moderate considering the extensive degradation of the permeability that is used in the study ; the result can be explained by the fact that the transport conditions remain diffusive or co-dominant diffusive - advective.

The study confirms that the dominant diffusion characteristic of the Callovo-Oxfordian is not drastically sensitive to the permeability of the host formation (see Figure 5.5-34), as long as it remains within values of less than 10^{-12} m/s.



Sensitivity to hydraulic parameters
CU2 reference package – ^{129}I



Reference calculation (REMINDER)
CU2 reference package – ^{129}I

Figure 5.5-34 SEN – Sensibility to hydraulic parameters : CU2 reference package – ^{129}I– molar flow history

- **Sensitivity to hydraulic, transfer, and retention parameters of the EDZ**

The purpose of this study is to make sure that the access structures and the accompanying EDZ do not constitute the preferential transfer path for the radionuclides, regardless of any eventual uncertainties about the hydraulic and transfer parameters within the EDZ (either due to the uncertainties resulting from changing the scale between the microscopic state of the EDZ and its « full size » properties, or from the influence of chemical perturbations). This in fact means making sure that the geological barrier effectively remains the primary transfer path for radionuclides despite EDZ characteristics that are less favourable than those used in the reference calculation and listed in Table 5.5-19.

	Reference calculation Phenomenological EDZ	Sensitivity calculation « degraded » EDZ
Fractured zone	$K = 5.10^{-9}$m/s <u>Anions</u> : $De_{Anions} = 1.10^{-11}$ m²/s $\omega_{Anions} = 0.15$ <u>Cations</u> : $De_{Cations} = 5.10^{-10}$ m²/s $\omega_{Cations} = 0.20$ Phenomenological geochemical retention	$K = 10^{-6}$ m/s Pessimistic coefficient ($Dp = De / \omega = 2.10^{-9}$ m²/s) No geochemical retention
Micro-fissured zone	$K = 5.10^{-11}$ m/s <u>Anions</u> : $De_{Anions} = 5.10^{-12}$ m²/s $\omega_{Anions} = 0.05$ <u>Cations</u> : $De_{Cations} = 2{,}5.10^{-10}$ m²/s $\omega_{Cations} = 0.18$ Phenomenological geochemical retention	$K = 5.10^{-9}$ m/s <u>Anions</u> : $De_{Anions} = 1.10^{-11}$ m²/s $\omega_{Anions} = 0.04$ <u>Cations</u> : $De_{Cations} = 5.10^{-10}$ m²/s $\omega_{Cations} = 0.21$ Conservative geochemical retention

Table 5.5-19 SEN - Sensitivity : Parameter values for transfer in the EDZ

This sensitivity study is very similar to a seal defect scenario, since the degradation of the EDZ properties is the same as partially bypassing the seal (its large-scale equivalent permeability is now only 10^{-7} m/s). It prefigures the results that will be presented in chapter 7.

As was done in the reference calculations, a complete calculation is performed from the waste packages to the outlets, taking into account the two transfer paths : one through the structures (drifts and access shafts), the other through the unaltered geological barrier

The following information was learned from this sensitivity study.

- Considering the relatively high permeability of the EDZ and the limited hydraulic effectiveness of the seals, the hydraulic head at the top of the Callovo-Oxfordian is transferred into the entire repository and to the drifts, with no significant head loss after crossing the different seals (see Figure 5.5-35). The flow rate at the shaft exit is increased (6.3 m^3/yr instead of 0.5 m^3/yr).

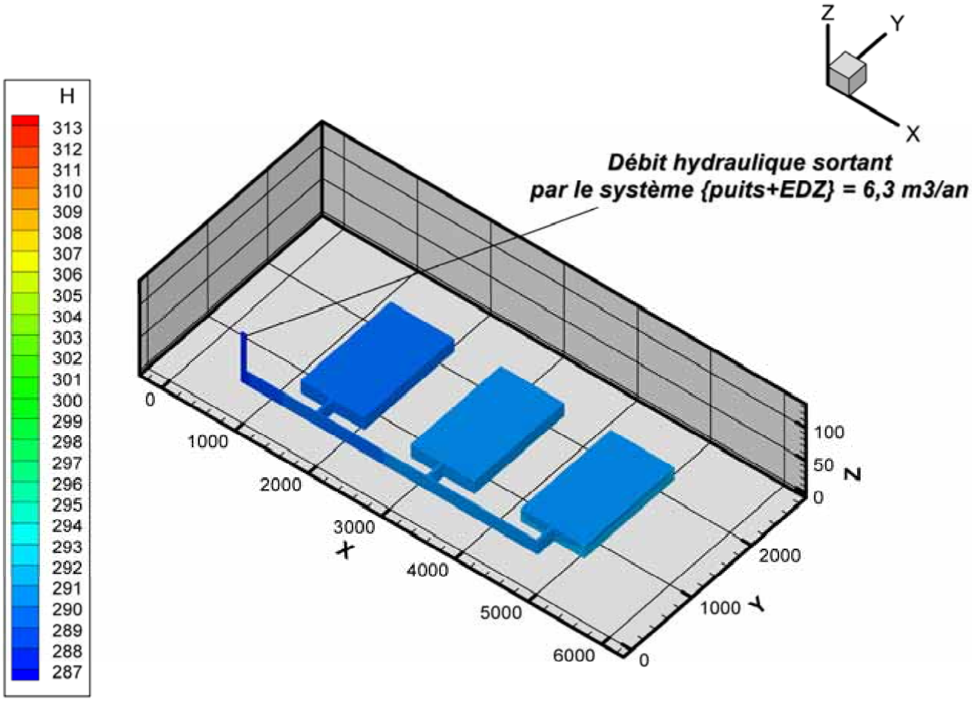


Figure 5.5-35 SEN - Propagation of hydraulic heads (heads in NGF[74] meters) in the repository access structures – « degraded » EDZ

- At the cells : using a highly degraded EDZ causes little change to the hydraulic regime in the field near the cell (see Figure 5.5-36). The Peclet number remains less than 1 at the spent fuel cell head, with a maximum value of 0.8 in the fractured zone (compared with 0.7 in the reference calculation) and a minimum value of 8.10^{-4} at the plug (compared with 7.10^{-4} in the reference calculation). As in the reference calculation, the transport is predominantly diffusive (or co-dominant diffusive-advective).

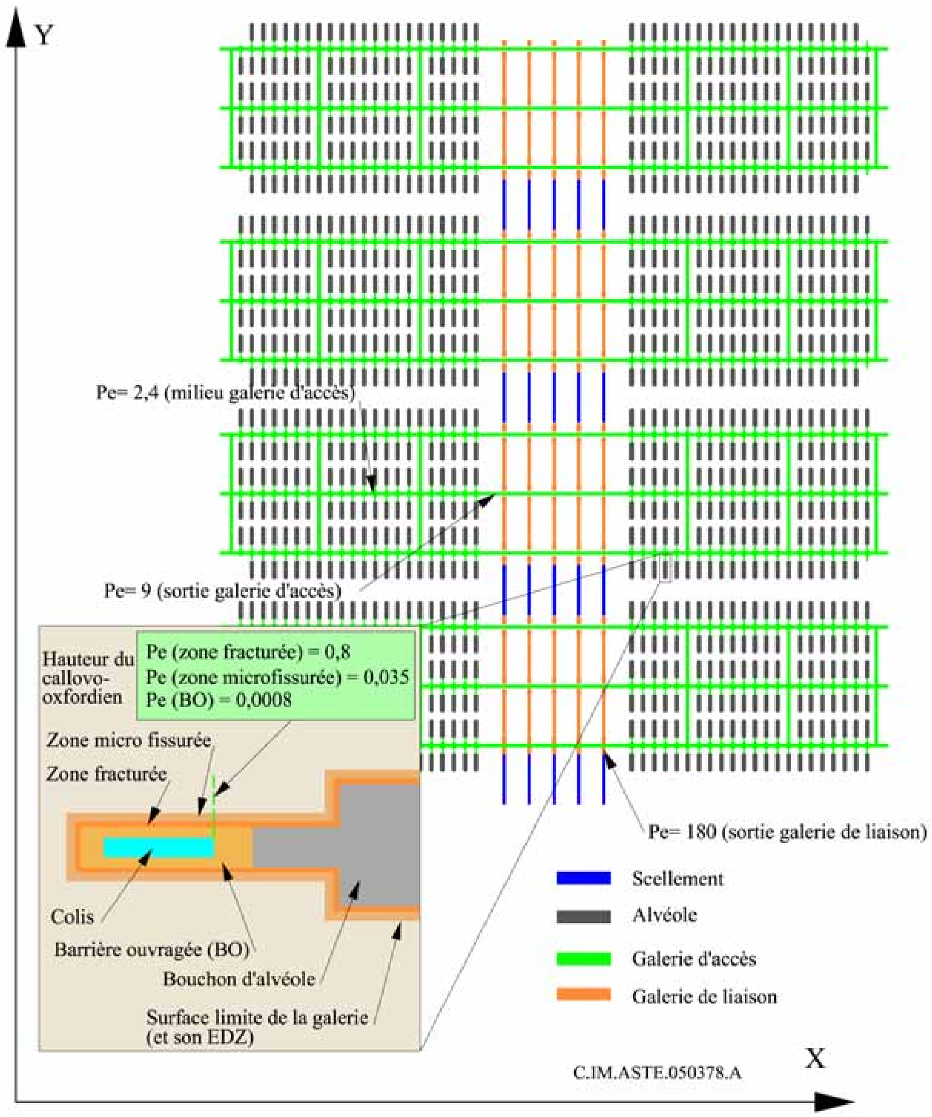


Figure 5.5-36 SEN - Sensitivity : « Degraded » EDZ (Pessimistic fractured zone, conservative micro-fissured zone) - CU1 reference package – ^{129}I - Peclet number values

Thus, the hydraulic draining caused by imposing a low load on the cell head (corresponding approximately to that of the top of the Callovo-Oxfordian) is not enough to impose dominant advective phenomena in the near field of the cell, as the advection is limited by the extremely low permeability of the geological barrier, thus limiting the water source. This result demonstrates the effectiveness of the dead-end architecture of the cells and repository modules by taking advantage of the rock's properties when establishing the hydraulic pattern of the repository.

Using a higher diffusion coefficient in the EDZ contributes to the 50 % increase in the maximum molar flow of iodine entering into the drift Figure 5.5-37), and to the 7 % increase in mass of Iodine-129 integrated over 1 million years that would enter the drift (66 % instead of 59 % in the reference calculation – see Figure 5.5-38).

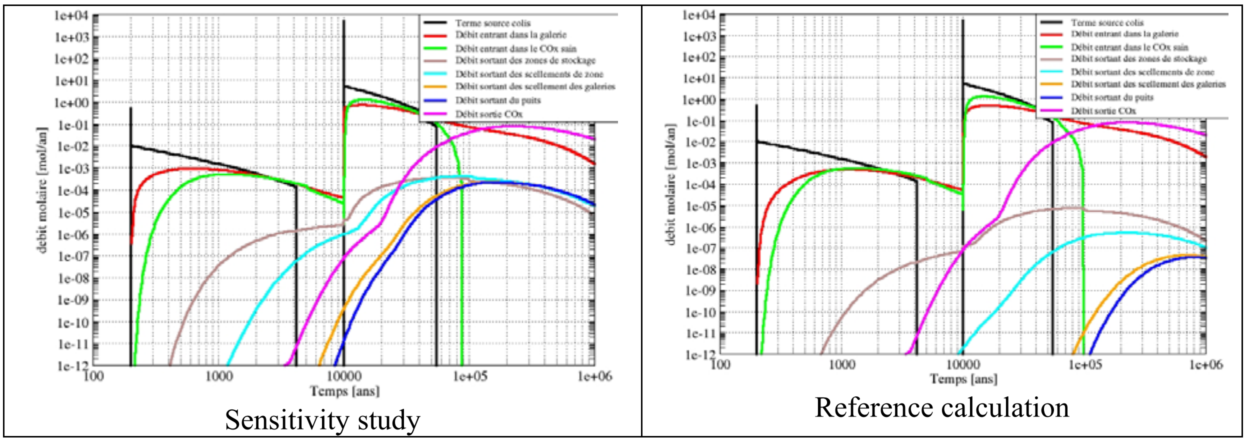


Figure 5.5-37 SEN - Sensitivity : « Degraded » EDZ (Pessimistic fractured zone, conservative micro-fissured zone) - CU1 reference package – ^{129}I - Molar flow history

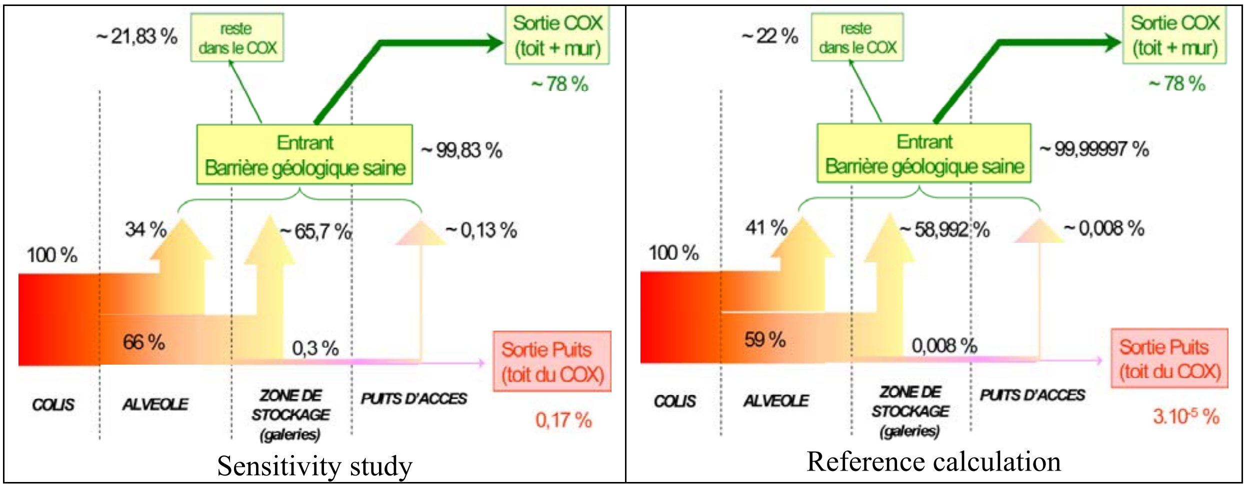


Figure 5.5-38 SEN - Sensitivity : « Degraded » EDZ (Pessimistic fractured zone, conservative micro-fissured zone) - CU1 reference package – ^{129}I - Distribution of transfer paths

For the repository zones, the flows remain limited in the access drifts, but are more significant (dominant advection) in the secondary connecting drifts. Exiting the repository zone, the transfers are extremely advective (the Peclet number is around 180). These phenomena all accelerate the transfer of radionuclides at the repository zone exit. For the CU1 spent fuels, the integrated quantity of iodine-129 exiting the repository zone over 1 million years is much higher than in the reference calculation (0.3 % versus 0.008 % in the reference calculation, or approximately a factor of 40). The maximum molar flow exiting the repository zone is also increased by a factor of approximately 70 ; since the transfer in this area is more rapid, the exchanges with the geological barrier are reduced here.

In the access structures, the phenomena observed for the secondary connecting drifts increase : in the primary connecting drifts, the regime is clearly advective, and the transfer times are much less than those evaluated in the reference calculation. The quantity of material exiting via the shafts is 5,000 times more than in the reference calculation (0.17 % versus 3.10^{-5} % in the reference calculation), with a maximum value reached earlier, at approximately 150,000 years instead of 800,000 years.

Thus, by using an EDZ with transport, hydraulic, and geochemical performances that are « degraded » in comparison with the reference calculation, the fraction of activity that migrates towards the access drifts increases. The fraction of activity that reaches the top of the Callovo-Oxfordian via the shafts is increased even more because the transport becomes advective in the primary connecting drifts, with faster kinetics.

However, the mass of the iodine at the shafts only represents 0.17 % of the total initial inventory of CU1 spent fuels, as the other fraction of the mass follows the transfer path through the unaltered geological barrier ; this result corroborates the observed variation between the maximum molar flow exiting from the Callovo-Oxfordian and the flow exiting from the shafts (factor of 400). What's more, it was noted that the molar flows exiting from the Callovo-Oxfordian are practically identical in the reference calculations and the sensitivity studies. Thus, we can conclude that the host formation remains the primary transfer path for the radionuclides, despite the use of « extremely degraded » properties for the damaged zone. The doses are thus similar to those in the reference calculation.

These different points are also confirmed for other radionuclides and other reference packages ; for the sake of convenience, only the results for the CU1 spent fuels are presented here.

- **Sensibility to transport and retention parameters (conservative values)**

In this study, the following values are considered :

- conservative values for transport parameters (De, ω) in the geological barrier and in the micro-fissured zone for the anions and cations (See section 5.3.2.3),
- conservative values for retention parameters (delay and precipitation) in the body (in concrete or clay) and the clay plug of the cells, the EDZ, and the Callovo-Oxfordian

A prior analysis of the apparent diffusion coefficients (Da), taking into account the changes in diffusion, porosity, and retardation coefficients, makes it possible to evaluate the characteristic migration time per diffusion Td [$Td = R.L^2.\omega/De = L^2/Da$] and the ratio between the characteristic migration times per diffusion obtained in the reference calculation and in the sensitivity study (fourth column of Table 5.5-20).

	Apparent diffusion of radionuclides Da = (De/ωR) in the unaltered argillites of the Callovo-Oxfordian (m²/s)		
	Reference	Sensitivity	Ratio {Rt} = Sensibility/Reference
^{36}Cl	$1{,}00.10^{-10}$	$2{,}50.10^{-10}$	2.50
^{79}Se	$1{,}00.10^{-10}$	$2{,}50.10^{-10}$	2.50
^{129}I	$1{,}00.10^{-10}$	$2{,}50.10^{-10}$	2.50
^{93m}Nb	$1{,}87.10^{-15}$	$5{,}80.10^{-15}$	3.09
^{94}Nb	$1{,}87.10^{-15}$	$5{,}80.10^{-15}$	3.09
^{59}Ni	$6{,}79.10^{-13}$	$2{,}17.10^{-12}$	3.20
^{41}Ca (deltaT>20)	$5{,}54.10^{-10}$	$2{,}38.10^{-09}$	4.30
^{126}Sn	$7{,}76.10^{-15}$	$3{,}62.10^{-14}$	4.67
^{107}Pd	$1{,}55.10^{-13}$	$1{,}36.10^{-12}$	8.75
^{99}Tc	$1{,}09.10^{-14}$	$1{,}09.10^{-13}$	10.00
^{14}C	$1{,}79.10^{-11}$	$2{,}50.10^{-10}$	14.00
^{93}Zr	$1{,}09.10^{-13}$	$2{,}17.10^{-12}$	19.98
^{166m}Ho	$2{,}17.10^{-15}$	$4{,}35.10^{-14}$	20.00
^{41}Ca	$8{,}64.10^{-11}$	$2{,}38.10^{-09}$	27.56
^{93}Mo	$7{,}19.10^{-13}$	$2{,}50.10^{-10}$	347.50
^{10}Be	$4{,}35.10^{-14}$	$2{,}38.10^{-09}$	54763.62
^{10}Be (deltaT>20)	$4{,}35.10^{-13}$	$2{,}38.10^{-09}$	5477.90

Table 5.5-20 SEN - Comparison of apparent diffusion coefficients (Reference/Sensitivity)

The table provides values for three different types of radionuclides :

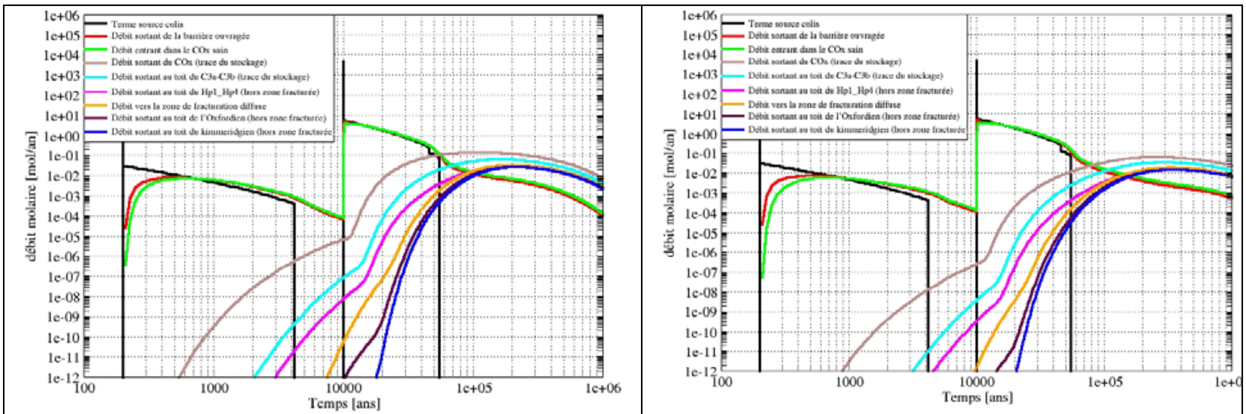
- non-sorbed anions, for which the sensitivity study is equivalent to accelerating their transfer by a factor of 2.5, due to a faster apparent diffusion,
- the sorbed anions, for which the lowering of the partition coefficient has an effect on the transport acceleration (up to a factor of 20 for ^{166m}Ho),
- the cations, for which the effects of using a more conservative geochemistry are more noticeable.

We also note, for selenium, the use of a much higher solubility, equal to 5.10^{-4} mol/m^3 compared with 5.10^{-7} mol/m^3 in the reference calculation. The transfer of this radionuclide is extremely limited by the solubility. Using a much higher solubility could cause a significant change in its results.

The following information was learned from these calculations.

The maximum molar flow of the iodine 129 and chlorine 36 from the CU1s and CU2s exiting the host formation is reached approximately 100,000 years earlier than in the reference case. In addition, the iodine-129 mass that remains confined in the Callovo-Oxfordian for the entire duration of the analysis is significantly lower than in the reference calculation (2 % in the sensitivity study versus 20 to 30 % in the reference calculation). The impact associated with the spent fuels only increases by a factor of 2.5 due to the spreading of the signal over time via diffusion (see Figure 5.5-39).

This sensitivity is a bit less significant for the vitrified C wastes because the duration of the waste package release, based on the model ($V_0.S$ ➔ V_r), is longer than that of the spent fuels, and is of the same order of magnitude as the transfers in the host formation.



Sensitivity study | Reference calculation

Figure 5.5-39 SEN – Sensitivity to transport and retention parameters : CU1 reference package – ^{129}I – History of molar flow

The use of less favorable transfer and solubility parameters for selenium 79 results in a very significant increase in the molar flow exiting the Callovo-Oxfordian (see Figure 5.5-40) and in the associated radiological impact. In fact, by multiplying by 1000 the solubility limit while increasing the apparent diffusion by a factor of 2.5, the selenium 79, which precipitated in the reference calculation, is rendered soluble, and the spreading by diffusion of the molar flow and its attenuation by radioactive decay are also reduced. As a consequence, the maximum molar flow exiting the Callovo-Oxfordian increases by a factor of 3,000 and the mass exiting the top and the bottom of the Callovo-Oxfordian over the total duration of the analysis (1 million years) is 600 times greater than in the reference calculation ; this factor of 600 takes into account both the increased solubility resulting in an increase by a factor of 150 of the molar quantity exiting the swelling clay buffer (the example of selenium 79 from the CU1s), and the transfer time, which is significantly shorter in the geological barrier and which increases the mass exiting the Callovo-Oxfordian by a factor of 4.

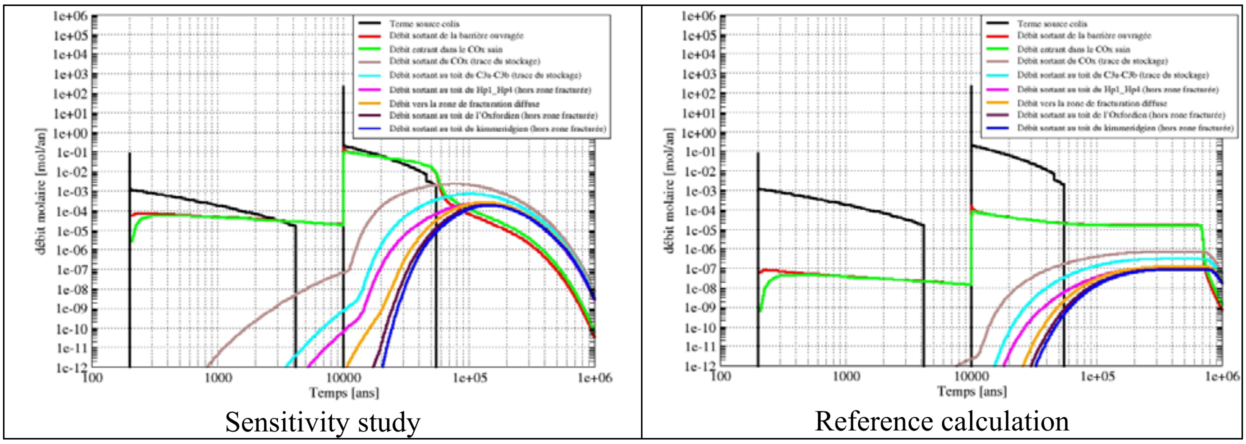


Figure 5.5-40 SEN – Sensitivity to transport and retention parameters : CU1 reference package – ^{79}Se– History of molar flow

For shorter-lived radionuclides (^{14}C and ^{93}Mo), the molar flow exiting the Callovo-Oxfordian, which was negligible for the reference calculations, becomes more significant in the sensitivity study, although still low compared with iodine (see Figure 5.5-41). In fact, their low radioactive half-life means they are very sensitive to transfer parameters. This effect is not expressed in the impact, because the total transfer time in the host formation and in the surrounding formations, although short in this last case, is sufficient to ensure the radioactive decay.

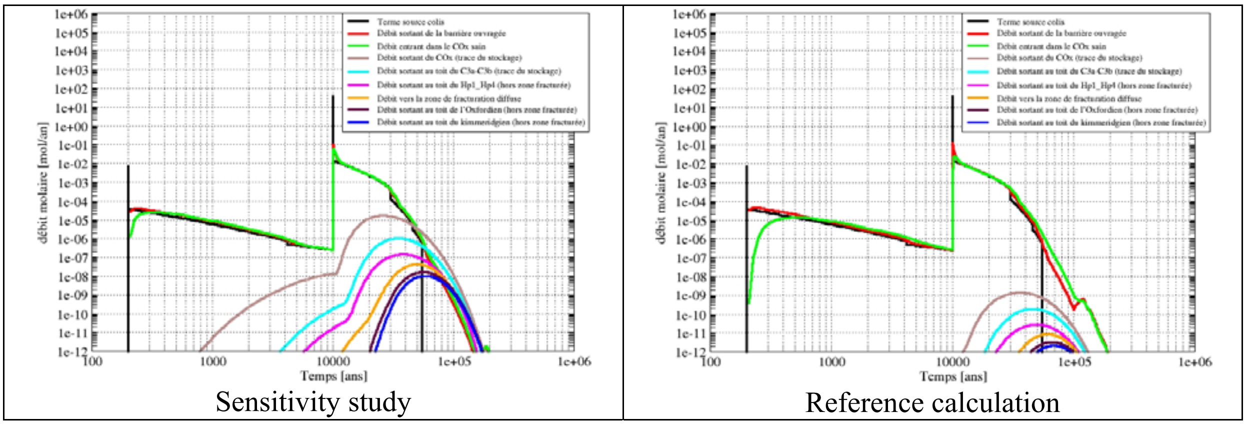


Figure 5.5-41 SEN – Sensitivity to transport and retention parameters : CU1 reference package – ^{14}C– History of molar flow

For the ^{10}Be, assigning a unit delay significantly increases the molar flow exiting the Callovo-Oxfordian, whereas it was negligible in the reference calculation. However, its low initial quantity does not imply that this radionuclide represents a significant contribution to the total dose (see Figure 5.5-42)

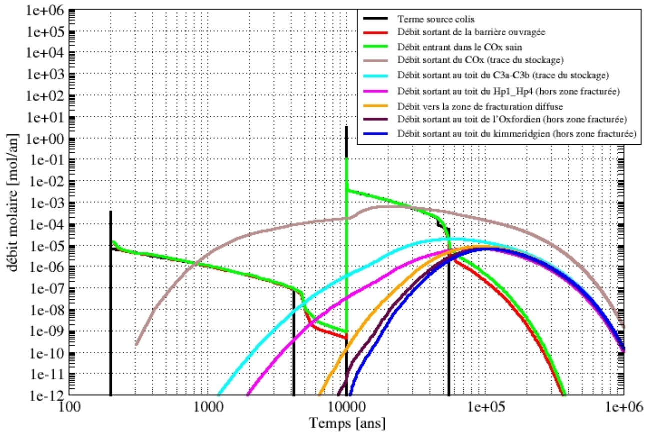


Figure 5.5-42 SEN – Sensitivity to transport and retention parameters : CU1 reference package – ^{10}Be– History of molar flow

For the ^{93}Zr, modifying by a factor of 20 the apparent diffusion coefficient results in a significant increase in the releases. However, considering its half-life, the theoretical maximum values of the molar flow are well beyond one million years. Even if this calculation were extended well beyond this time limit, the flows would not have any effect on the maximum dose (see Figure 5.5-43).

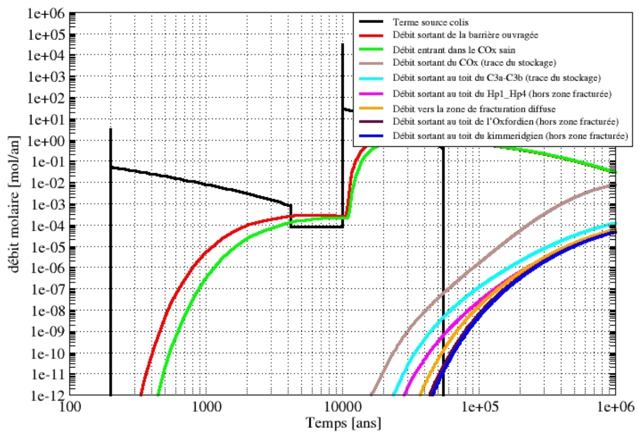
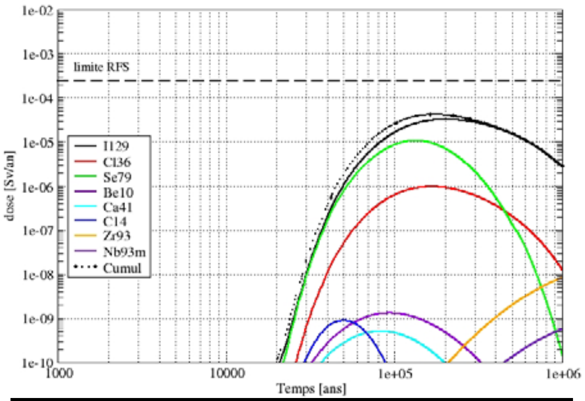


Figure 5.5-43 SEN – Sensitivity to transport and retention parameters : CU1 reference package – ^{93}Zr– History of molar flow

For the other long-lived radionuclides, the modification of the apparent diffusion coefficient does not significantly change the molar flow exiting the Callovo-Oxfordian : it remains very low, or negligible.

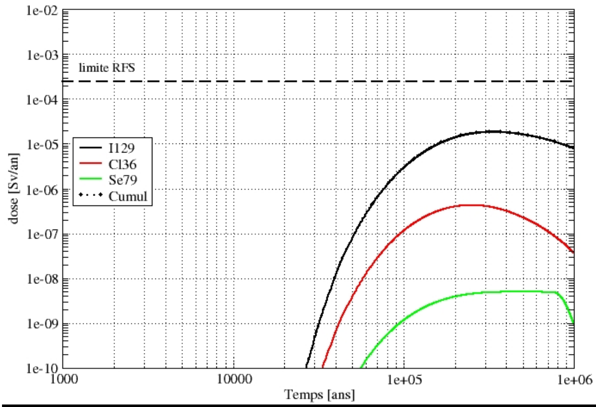
In conclusion, for the spent fuels, the sensitivity to the hydraulic, transport and retention parameters in the swelling clay buffer and in the geological barrier results in an impact that remains governed by iodine-129 and, to a lesser extent, selenium 79.

If we express the preceding calculations in terms of the dose at the outlet, for the spent fuels (CU1) which have the largest impact, the maximum dose at the Saulx outlet equals approximately 0.04 mSv/year at around 200,000 years, both in the 1 million year and in the current model (see Figure 5.5-44 and Figure 5.5-45). It remains well below the dose constraint recommended by the RFS III.2.f.



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	4.2E-05	1.7E+05
I129	3.3E-05	2.0E+05
Se79	1.1E-05	1.3E+05
Cl36	9.7E-07	1.7E+05
Zr93	8.4E-09	1.0E+06
Be10	1.3E-09	9.4E+04
C14	8.9E-10	5.0E+04
Nb93m	5.7E-10	1.0E+06
Ca41	5.1E-10	8.3E+04
Ni59	dose < 1e-10	sans objet
Pd107	dose < 1e-10	sans objet

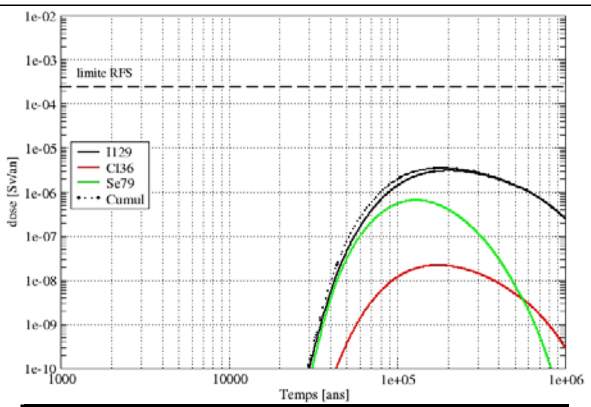
Sensitivity study



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.9E-05	3.3E+05
I129	1.8E-05	3.4E+05
Cl36	4.3E-07	2.5E+05
Se79	5.0E-09	4.7E+05

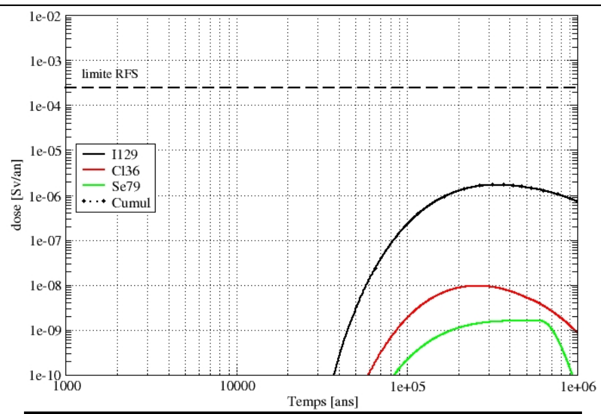
Reference calculation (REMINDER)

Figure 5.5-44 SEN – Sensitivity to transport and retention parameters (conservative values) - Dose at the Saulx outlet of the Oxfordian – CU1 reference package – 1 million year model



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	3.5E-06	1.8E+05
I129	3.0E-06	2.0E+05
Se79	6.5E-07	1.3E+05
Cl36	2.2E-08	1.8E+05

Sensitivity study

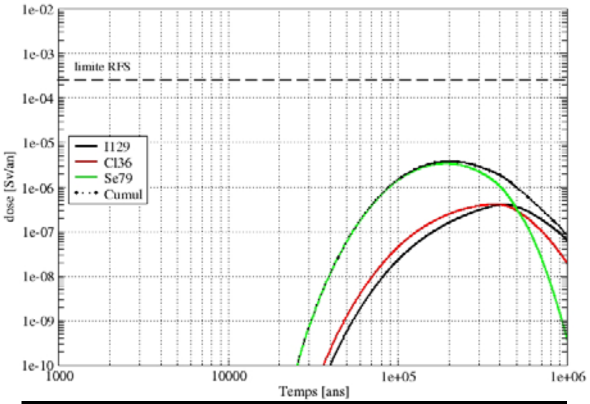


RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.7E-06	3.4E+05
I129	1.7E-06	3.4E+05
Cl36	9.6E-09	2.6E+05
Se79	1.6E-09	5.6E+05

Reference calculation (REMINDER)

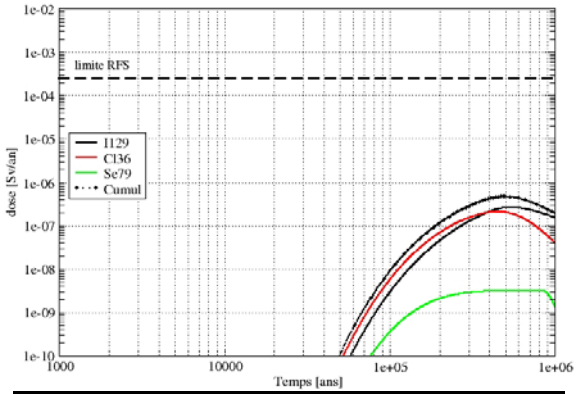
Figure 5.5-45 SEN – Sensitivity to transport and retention parameters (conservative values) - Dose at the Saulx outlet of the Oxfordian – CU2 reference package – 1 million year model

For the C wastes, selenium 79 becomes the dominant radionuclide up until 500,000 years, with a maximum dose of approximately 0.003 mSv/year at 200,000 years for the Saulx outlet. Beyond 500,000 years, the radiological impact of the C wastes is dominated by both the iodine 129 and the chlorine 36, below 0.0005 mSv/year (see Figure 5.5-46 and Figure 5.5-47).



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	3.7E-06	2.0E+05
Se79	3.3E-06	1.9E+05
Cl36	4.1E-07	3.8E+05
I129	4.0E-07	4.2E+05

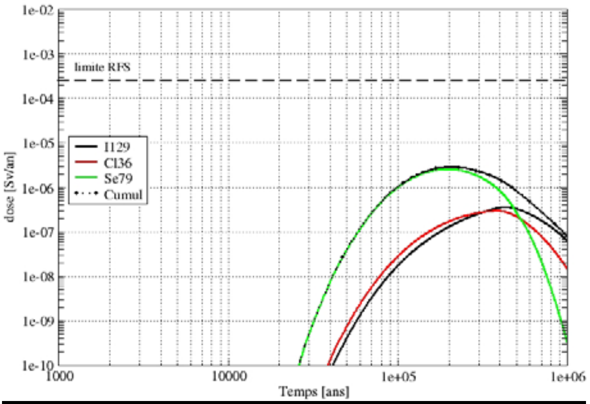
Sensitivity study



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	4.7E-07	4.9E+05
I129	2.6E-07	5.5E+05
Cl36	2.1E-07	4.5E+05
Se79	3.2E-09	8.0E+05

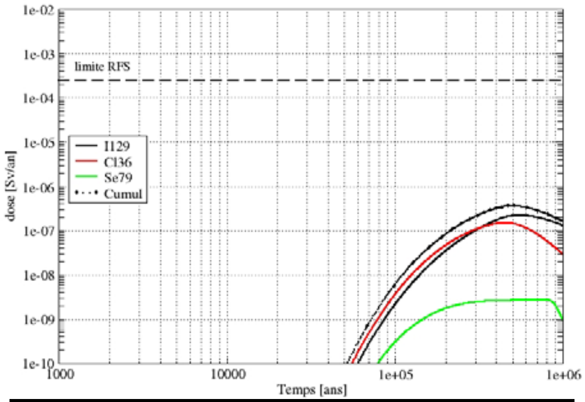
Reference calculation (REMINDER)

Figure 5.5-46 SEN – Sensitivity to transport and retention parameters (conservative values) - Dose at the Saulx outlet of the Oxfordian – Reference packages (C1+C2)



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	2.8E-06	2.1E+05
Se79	2.5E-06	2.0E+05
I129	3.5E-07	4.3E+05
Cl36	2.9E-07	3.9E+05

Sensitivity study

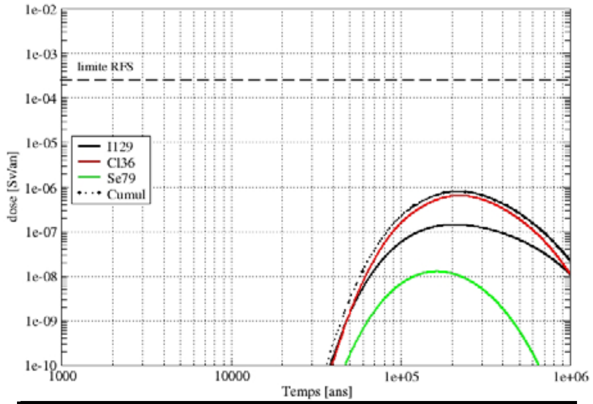


RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	3.6E-07	5.0E+05
I129	2.2E-07	5.6E+05
Cl36	1.5E-07	4.6E+05
Se79	2.7E-09	7.4E+05

Reference calculation (REMINDER)

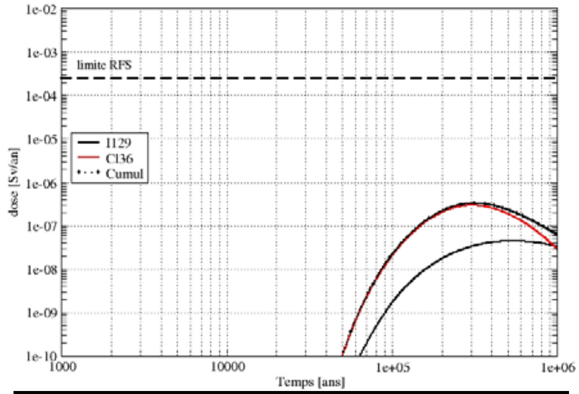
Figure 5.5-47 SEN – Sensitivity to transport and retention parameters (conservative values) - Dose at the Saulx outlet of the Oxfordian – Reference packages (C3+C4)

For the B1x wastes, the classification shown in the reference calculation remains valid for the sensitivity to transport and retention parameters. The iodine 129 and chlorine 36 are the primary contributors, with a total dose approximately twice as high, of 0.00079 mSv/year (compared to 0.00033 mSv/year in reference calculation) for the 1 million year model and the current model at approximately 200,000 years (see Figure 5.5-48).



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	7.9E-07	2.2E+05
Cl36	6.4E-07	2.2E+05
I129	1.4E-07	2.1E+05
Se79	1.3E-08	1.6E+05

Sensitivity study



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	3.3E-07	3.1E+05
Cl36	2.9E-07	3.1E+05
I129	4.4E-08	5.3E+05
Se79	dose < 1e-10	sans objet

Reference calculation (REMINDER)

Figure 5.5-48 SEN – Sensitivity to transport and retention parameters (conservative values) - Dose at the Saulx outlet of the Oxfordian – Reference waste packages B (B1x)

In conclusion, we note that by modifying the parameters governing the function « limit the release of radionuclides and immobilize them in the repository » (solubility) and the function « delay and attenuate the migration of radionuclides » (diffusion, delay), and in particular by modifying the parameters for the diffusion coefficient, the classification of the radionuclides contributing to the impact as well as the impact itself may be changed. However, the range of values obtained still respect the dose constraints of the RFS III.2.f. The effect is not the same for all of the radionuclides, and mainly concerns the following :

- The lightly sorbed, long-lived radionuclides such as chlorine 36 and iodine 129, which show an earlier maximum dose that is thus a little bit higher (owing to their less diffusive spread)
- The selenium, which becomes soluble in sensitivity studies whereas it is immobilised by its low solubility in the reference calculation
- A few radionuclides that diffuse much more rapidly in the sensitivity study because of the significantly lower retardation coefficients. Note that none of these radionuclides make a significant contribution to the impact. In particular, the molybdenum 93 and the carbon 14, which are very sensitive to a decrease in the transfer time due to their half-lives, do not contribute to the dose. Nor do berylium 10 and zirconium 93, which show very low doses due to the significant changes in the retardation coefficient.

For the other long-lived radionuclides (^{59}Ni, ^{94}Nb, ^{99}Tc, ^{107}Pd, ^{126}Sn, actinides, etc.) the change to more conservative transport and retention parameters do not result in any important modification of their contribution to the impact : for the most part, they remain confined in the geological barrier.

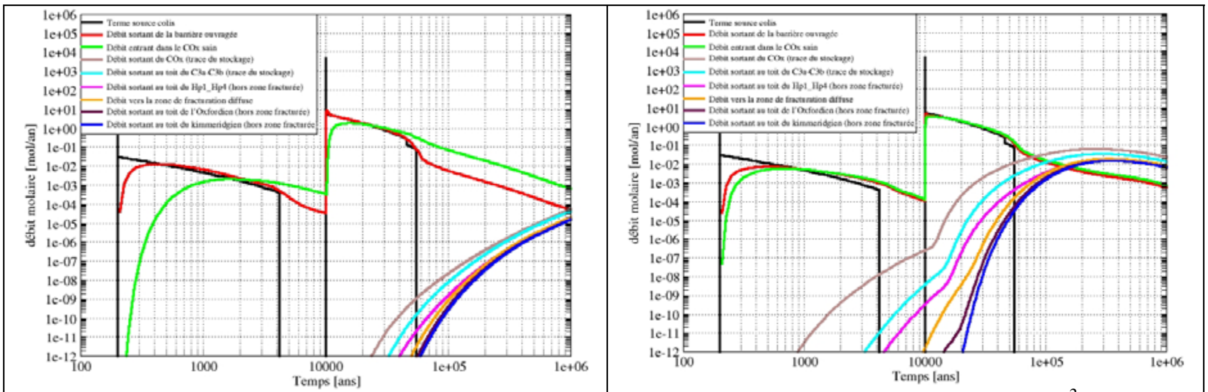
- **Sensitivity to the partition coefficient of iodine in the Callovo-Oxfordian (Kd = 10^{-3} m^3/kg)**

This study is conducted in order to take into account the experimental results of the sorption of iodine. It is only conducted to make predictions, in order to evaluate the potential gains from the low retention of iodine in the argillites of the Callovo-Oxfordian. Thus, all the parameters retain their reference value, except for the retardation coefficients of the iodine in the Callovo-Oxfordian and the EDZ. Their values are provided inTable 5.5-21.

	Retardation coefficient for iodine	
	Reference value	Sensitivity value
Unaltered Callovo-Oxfordian	1	47
Micro-fissured zone	1	47
Fractured zone	1	16

Table 5.5-21 SEN – Sensitivity of iodine retardation coefficient in the Callovo-Oxfordian.

The effect of an increase in the retardation coefficient is to significantly spread out the molar flows released upon exiting the Callovo-Oxfordian (see Figure 5.5-49) : the maximum values for the molar flow over the period of less than one million years are attenuated with respect to the reference calculation and delayed significantly. Thus they appear beyond one million years (theoretically at around 10 million years).



Sensitivity study (Kd = 10^{-3} m^3/kg) Reference calculation (Kd = 0 m^3/kg)

Figure 5.5-49 SEN – Sensitivity to the iodine retardation coefficient in the Callovo-Oxfordian : CU1 reference package – ^{129}I– History of molar flow

The difference between the maximum rate from the reference calculation and the flow at one million years in the sensitivity study is approximately three orders of magnitude. The mass that remains confined in the Callovo-Oxfordian for the duration of the analysis (1 million years) is much higher when the iodine is sorbed in the argillites of the Callovo-Oxfordian : for the iodine contained in the C2 reference packages, it grows from a bit less than 30 % in the reference calculation to 99.98 % in the sensitivity study (see Table 5.5-22).

	Reference	Sensitivity
Mass of iodine-129 remaining confined in the Callovo-Oxfordian for entire duration of analysis (1 million years) (in % of initial mass)	28 %	99.98 %

Table 5.5-22 Changes in mass of iodine-129 contained in the C2 reference packages

- **Sensitivity to thickness of Callovo-Oxfordian**

The sensitivity study is used to evaluate the advantages of a thicker Callovo-Oxfordian than that of the reference calculation, as encountered northwest of the transposition zone. The thickness of the Callovo-Oxfordian thus increases by 10 meters to each side of the repository, which remains in the middle of the layer.

In general, increasing the travel distance tends to favour the advection with respect to the diffusion, because the diffusive travel time increases with the distance squared, whereas the advection times are only proportional to it. But in this case, increasing the thickness of the layer has no influence on the migration regime, which remains mainly diffusive. The increase of approximately 15 % is in fact too low to significantly change the Peclet number. With respect to the reference case, the following was reported for iodine 129 :

- maximum flows approximately one third lower (see Figure 5.5-50),
- an appearance date of maximum values of approximately 115,000 years for the layer bottom, and of 140,000 years for the top,
- a decrease of one quarter in the total mass exiting the Callovo-Oxfordian, with no modification of the top/wall proportions.

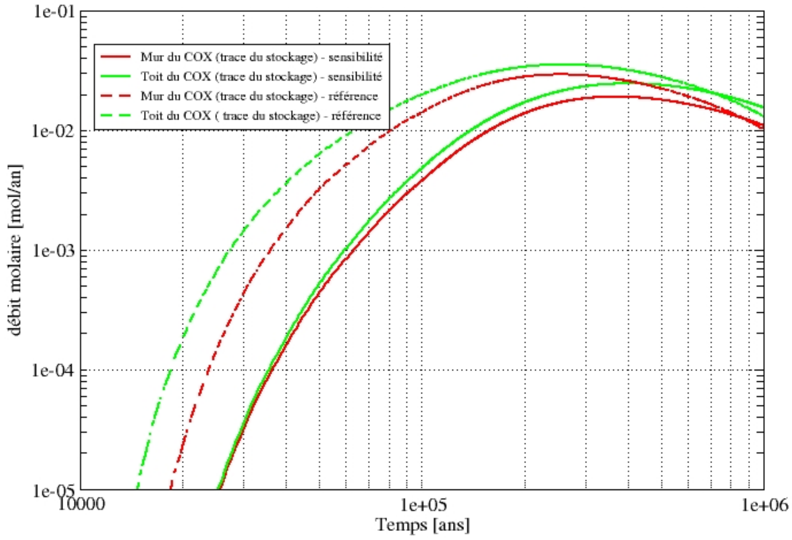


Figure 5.5-50 SEN – Sensitivity to the thickness of the Callovo-Oxfordian (150 meters instead of the 130 meters in the reference calculation) - CU1 reference packages – ^{129}I – History of molar flow. (COX = Callovo-Oxfordian)

The increased thickness has a more significant effect on the chlorine 36 due to its radioactive half-life of 300,000 years, which is less than the transfer time in the host layer.

- **Sensitivity to thermal effects**

The thermal effects considered in the reference calculations (and consequently their influence on the transfer and retention parameters) are pessimistic, because they are based on 2D calculations. This choice was made in order to improve the chances that the SEN calculation could withstand the uncertainties concerning the propagation of heat in the formation. A sensitivity study uses more realistic thermal calculations, performed in 3D, and in which the different repository components are represented in great detail.

By using a more realistic thermal model, the diffusion coefficients are reduced by a factor of three during the thermal phase (see Figure 5.5-51). However, the results show that the difference between phenomenological thermal effects and pessimistic thermal effects does not have a significant effect on the results of the radiological impact calculation ; in fact, the period during which the thermal effects modify the transfer and retention parameters is much shorter than the transfer duration in the unaltered Callovo-Oxfordian in both cases.

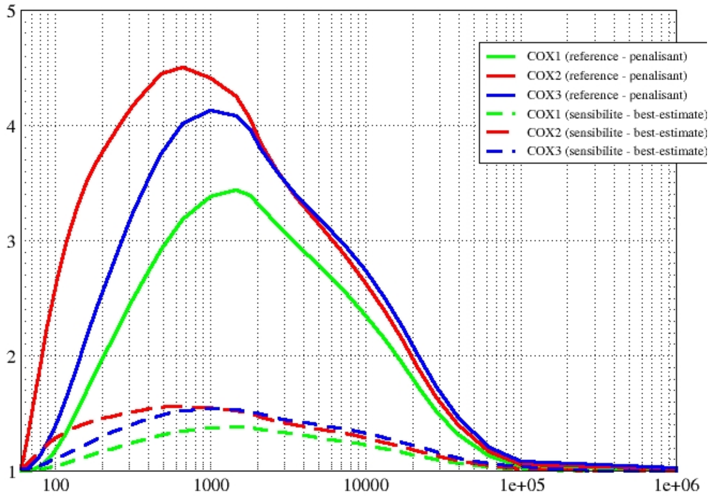


Figure 5.5-51 SEN - Comparison of diffusion correction factors (phenomenological and pessimistic thermal effects) – CU1 reference package

5.5.6.2 Sensitivity studies on release models

For the majority of the packages, the release periods of the radionuclides present in the wastes, that are used in the reference calculation, are short with respect to the time of transfer in the geological barrier. Thus, the molar flow exiting the geological barrier does not depend, in the majority of cases, on the flow released by the packages over time, but on the total mass released by the packages. Thus, using shorter release periods should not in principle cause a significant change in the molar flow exiting the geological barrier and thus the impact. The goal of the sensitivity studies is to verify this point.

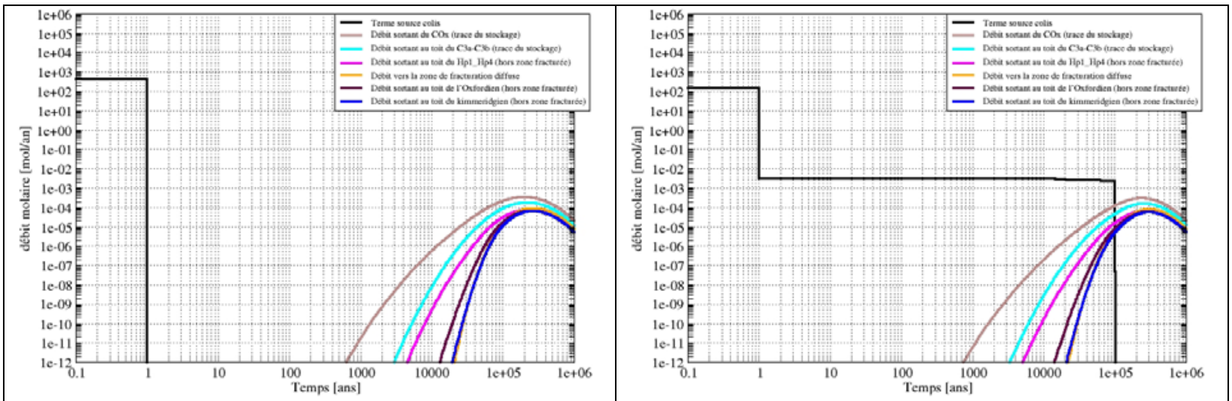
- **Sensitivity to reference model chosen for the inorganic waste packages that do not release hydrogen (pessimistic model)**

This sensitivity study is for metal waste packages that do not contain organic matter. These reference packages correspond to activation product wastes (B1 reference packages) ; compacted cladding hulls and compacted end caps that do not contain organic matter (B5 reference packages, except for B5.1) ; and to bulk cladding waste and technological waste (B6 reference package, except for B6.4).

The study considers that the activation products contained in the cladding and the metal components are released in a labile manner, whereas they are released progressively in the reference calculation (over periods of around 100,000 years). The study using a pessimistic release model is largely sufficient to cover any residual uncertainties concerning the variability of wastes or any effects of aggressive species resulting from the radiolysis of water or localised corrosion.

The analysis is only for chlorine 36, present in some of the cladding, since the iodine-129 and the selenium 79 are already considered to be released in a labile manner in the reference calculation.

The results show clearly that including a labile source term, resulting in a higher release over a very short period, does not, with respect to the reference calculation, result in a modification of the molar flows exiting the geological barrier (see Figure 5.5-52). In fact, the release periods remain short with respect to the transfer times in the geological barrier, whether it be in the reference calculation or the sensitivity study.



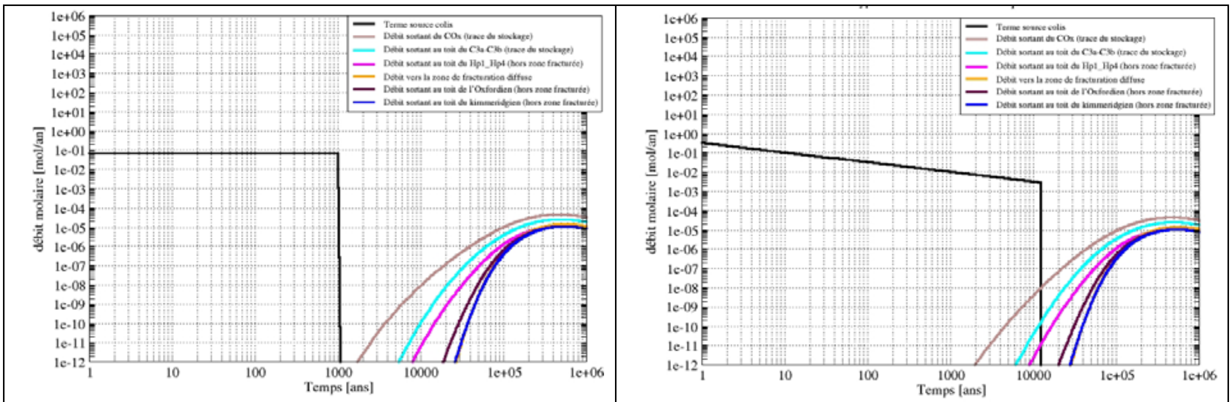
Sensitivity study | Reference calculation

Figure 5.5-52 SEN – Sensitivity to reference model used for inorganic packages that do not release hydrogen (pessimistic model) - B1x reference package – ^{36}Cl – history of molar flow

- **Sensitivity to parameters of reference source term for bitumen waste packages (pessimistic model)**

This sensitivity study is for the bitumen waste packages (B2 reference packages) ; it consists in using a pessimistic release model that covers any residual uncertainties stemming from changes in the packages (creep, fissures, etc.). The selected sensitivity study is for a release over 1000 years with a constant rate of 10^{-3}/year, whereas the selected reference model results in a slower release, with a release rate that is inversely proportional to the square root of the time [$\tau_R = 4{,}5.10^{-3}/\sqrt{(t)}$].

For the same reasons as for the B waste packages studied earlier, the results make it clear that using a pessimistic source term, which results in greater release over a short period, does not cause any modification with respect to the reference calculation (see Figure 5.5-53)



Sensitivity study | Reference calculation

Figure 5.5-53 SEN – Sensitivity to reference model used for the B2 reference package (pessimistic model) - B2 reference package – ^{129}I – history of molar flow

- **Sensitivity to containers of packages containing no organic matter and releasing no hydrogen (disposed in B1x type cells)**

This study is conducted to make predictions in order to evaluate the potential gains of using durable concrete for B waste package containers. In such a case, the container concrete has diffusive, hydraulic, and retention properties until 10,000 years. All the other parameters from the reference calculation remain the same.

The Table 5.5-23 shows the hydraulic, transport, and retention parameters used in the reference model and the sensitivity study.

		Reference calculation Structural concrete				Sensitivity calculation Durable container concrete (to 10,000 years)			
	Half-life [yrs]	$\omega_{Diffusion}$ [-]	De [m²/s]	R [-]	Csat [mol/m³]	$\omega_{Diffusion}$ [-]	De [m²/s]	R [-]	Csat [mol/m³]
^{36}Cl	302 000	0.3	6.10^{-10}	1	soluble	0.1	2.10^{-13}	1	soluble
^{129}I	15 700 000	0.3	6.10^{-10}	8	soluble	0.1	2.10^{-13}	28	soluble

Table 5.5-23 SEN – Reference and sensitivity values for transport parameters in the B1x waste package containers.

Using high performance concrete during the initial ten thousand years of disposal decreases a minimum of one order of magnitude the molar flows and delays their exit from the cell during the first 10,000 years - in particular for iodine-129 whose retardation coefficient changes from 8 to 28 (see Figure 5.5-54). However, the quantity of radionuclides emitted by the cell remains very low over this period (only a few %).

Beyond 10,000 years, the curves with or without containers overlap.

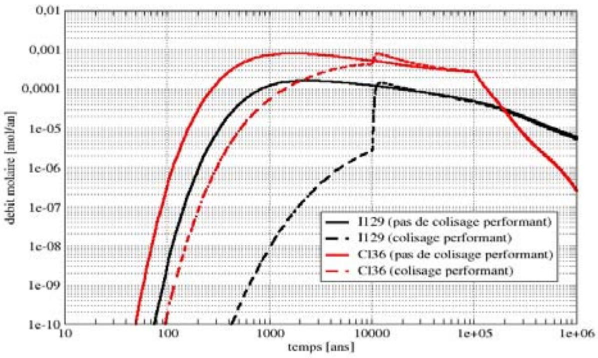


Figure 5.5-54 SEN – Changes in molar flow exiting B1x waste cells for ^{129}I and ^{36}Cl – comparison with and without effective containers

As the container durability is more than one order of magnitude below the diffusive transfer time in the Callovo-Oxfordian, the influence on the molar flow exiting the geological barrier is not significant (see Figure 5.5-55). However, the introduction of this type of container does provide additional safety during the initial time phases.

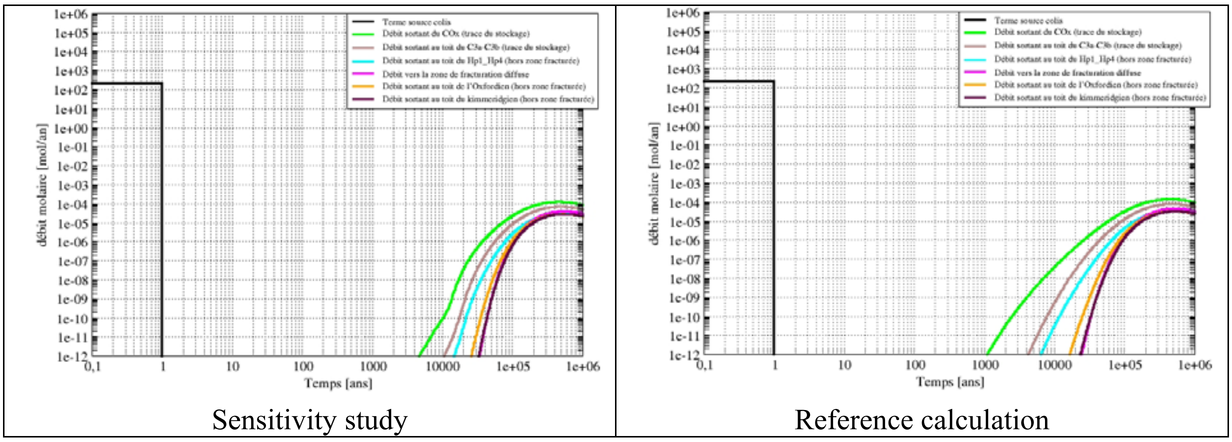


Figure 5.5-55 B1x reference package – ^{129}I – molar flow history

- **Sensitivity to parameters of reference source term for C waste packages (model V_0.S ➔ V_r)**

In this study, the release model for the vitrified waste packages was modified by adopting conservative values for the initial rate (using a higher rate of fracturing). The calculation results show no differences from the results of the reference calculation – see Figure 5.5-56.

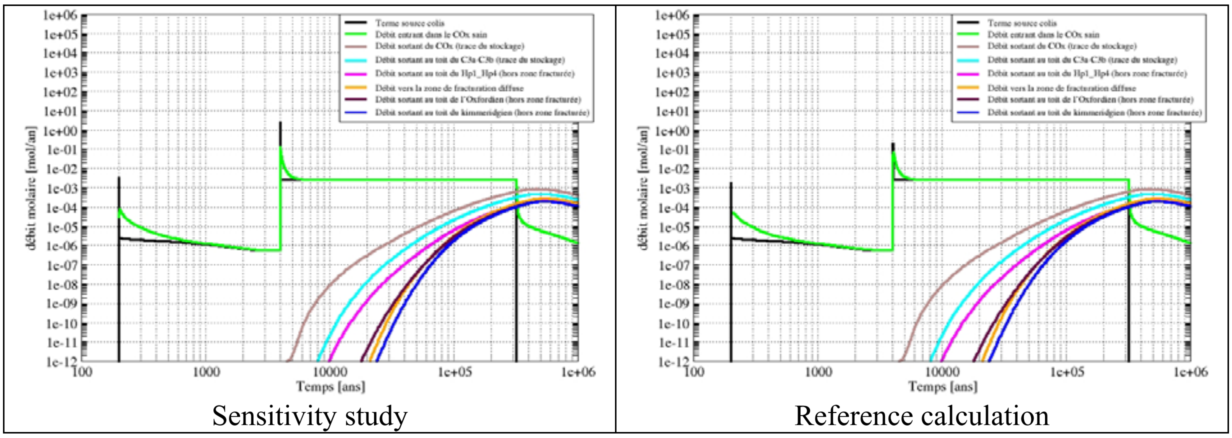


Figure 5.5-56 SEN – Sensitivity to parameters of reference source term for C waste packages (model V_0.S ➔ V_r) - C1 + C2 reference packages – ^{129}I – history of molar flow

- **Sensitivity to release model of C waste packages (model V_0S)**

A sensibility study covering the uncertainties associated with the residual rate model was performed. In this study, the effect of a more pessimistic release model was evaluated based only on the initial dissolution rate (model V_0.S). Examples of the maximum molar flows and the appearance dates for the maximum values are provided for the C2 reference packages in Table 5.5-24. The molar flows for C4 reference packages are slightly less.

	Maximum molar flow exiting Callovo-Oxfordian (mol/yr) and maximum dates (yrs.)	
	Reference	Sensitivity
^{129}I	$8.6.10^{-4}$ 460,000 yrs	$9.1.10^{-4}$ 250,000 yrs
^{36}Cl	$2.2.10^{-4}$ 380,000 yrs	$3.8.10^{-4}$ 190,000 yrs

Table 5.5-24 SEN - Attenuation ^{129}I and ^{36}Cl – C1+C2 – comparison between the models $V_0.S$ (sensitivity) and the model $V_0.S \rightarrow V_r$

The following information was learned from this sensitivity study.

- The appearance date of the maximum values for the molar flow comes earlier in the sensitivity study than in the reference calculation (maximum values appear approximately 200,000 years earlier). In fact, since the release period of model $V_0.S$ becomes shorter than the transfer time in the host formation, the influence of the release model is less visible upon exiting the geological barrier. The appearance dates of the maximum values for the molar flows then become comparable to those observed for the spent fuels (see Figure 5.5-57). The appearance date of the maximum values for the molar flow is later for the ^{129}I (250,000 years) than for the chlorine 36 (190,000 years), which benefits from radioactive decay. Of course, this time displacement also applies to the dose at the outlets.
- The maximum activity flow rate leaving the Callovo-Oxfordian is practically identical for the iodine-129 in the sensitivity study and the reference calculation, whereas it increases by a factor of 2 for chlorine 36 in the sensitivity study. In fact, since the mass of chlorine 36 is released more rapidly, the appearance date of the maximum values for the molar flow remains below the radioactive decay half-life of the chlorine 36 ($3.02.10^5$ years). In these conditions, it benefits less from the decay.

Thus, the model ($V_0.S \rightarrow V_r$) offers a significant advantage over the model ($V_0.S$) studied in the reference calculation ; it contributes to the attenuation of the chlorine 36 molar flow because of the radioactive decay, and it delays the appearance of the maximum molar flow by approximately 200,000 years.

However, since the impact of the vitrified waste packages is much lower than the dose constraint of 0.25 mSv/yr. (0.00063 mSv/yr. for C0, C1 and C2, and 0.00049 mSv/yr. for C3 and C4), adopting a constant dissolution rate model does not call into question the acceptability of the impact (see Figure 5.5-58).

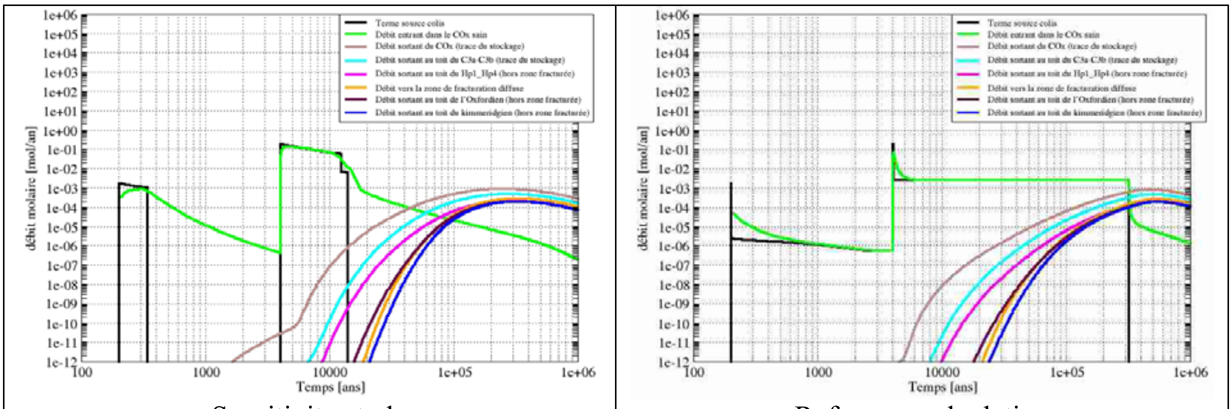
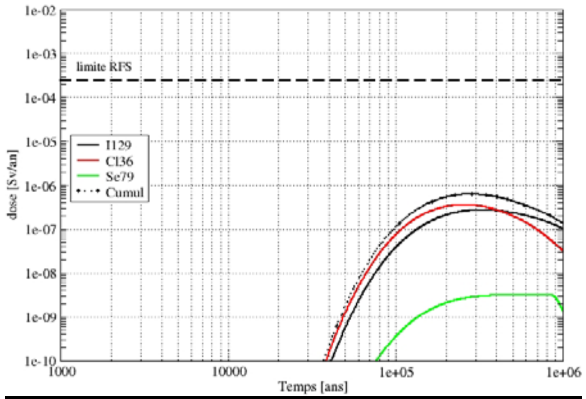
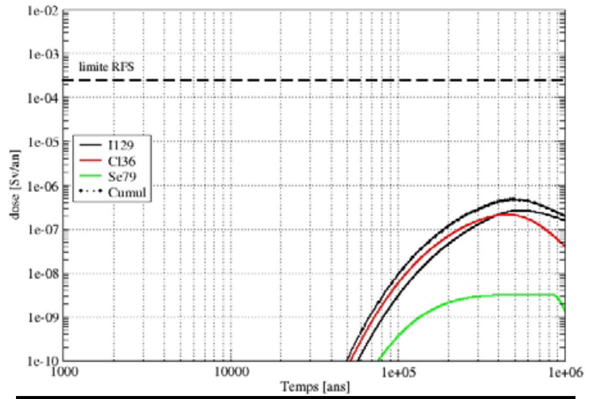


Figure 5.5-57 SEN – Sensitivity to release model of C waste (model $V_0.S$) - C1 + C2 reference package – ^{129}I – History of molar flow



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	6.3E-07	2.8E+05
Cl36	3.6E-07	2.6E+05
I129	2.7E-07	3.3E+05
Se79	3.2E-09	8.0E+05

Sensitivity study



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	4.7E-07	4.9E+05
I129	2.6E-07	5.5E+05
Cl36	2.1E-07	4.5E+05
Se79	3.2E-09	8.0E+05

Reference calculation

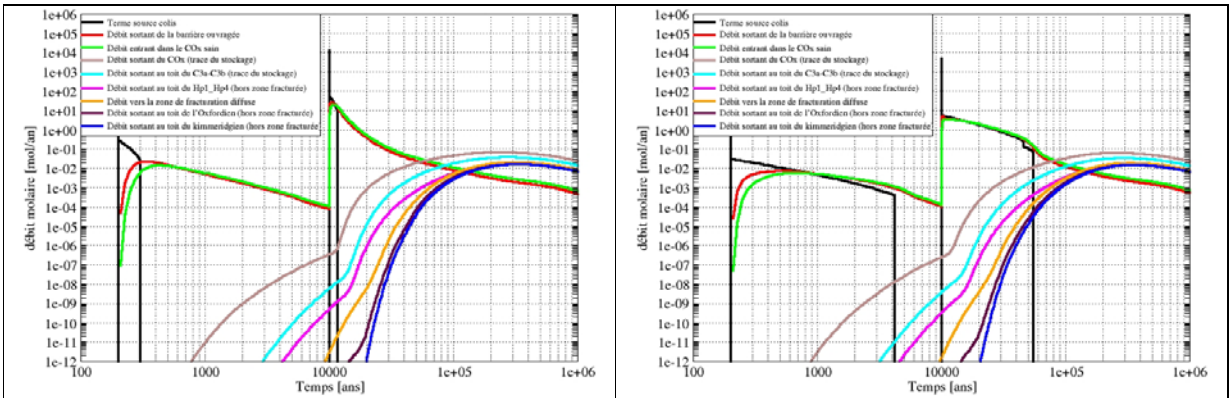
Figure 5.5-58 SEN – Sensitivity to release model of C waste (model V_0.S)– Dose at Saulx outlet of Oxfordian - C1 + C2 reference package

- **Sensitivity of spent fuels to parameters of reference release model (conservative values)**

In this study, conservative radiolytic dissolution values ten times higher were used, in order to cover any remaining uncertainties in the model - due for the most part to the influence of temperature.

The results of the study show no significant modification with respect to the reference calculation. In fact, the mass exiting the Callovo-Oxfordian over one million years remains practically identical to that of the reference calculation (only a 1 % increase in mass exiting the Callovo-Oxfordian with respect to the reference case for the iodine-129 of the CU1s). This has a correlation with the fact that the release period by the matrix is already significantly lower than the average transfer period of the radionuclides in the host formation for the reference calculation ; the flow exiting the geological barrier does not depend on the flow rate from the waste packages, but depends on the total weight released by the packages. Subsequently, this would be the case for an even shorter release period than the one in the reference calculation (see Figure 5.5-59).

Thus in the SEN, the influence of a source term for spent fuels that is less favourable than the one used in the reference calculation is not significant.



Sensitivity study Reference calculation

Figure 5.5-59 SEN – Sensitivity of spent fuel packages to parameters of the reference release model (conservative values) - CU1 reference package – ^{129}I - history of molar flow –

- **Sensitivity of spent fuel packages to release model (controlled by uranium solubility)**

In this study, an alternative model based on the classical dissolution of uranium in spent fuels is used ; the dissolution is controlled by uranium solubility (7.10^{-7} mol/m^3), and results in slower release kinetics. This model results in a much more progressive release than in the reference calculation activity. Apart from the conventional dissolution model being substituted for the radiolytic dissolution model, the characteristics remain unchanged (labile fraction, D3AI coefficient, release rate of activation products contained in the metal materials).

Table 5.5-25 provides the mass of the iodine-129 emitted by the waste package over 1 million years for the radiolytic dissolution and the classical dissolution. Note that at one million years, all the iodine-129 mass that was initially present in the packages is released in the radiolytic dissolution model, whereas this quantity is 10 % in the classical dissolution model. Furthermore, the majority of the 10 % mass comes from the labile fraction ; the fraction of activity released by the matrix only represents one percent of the total initial mass.

Mass of iodine-129 released by the spent fuels over analysis period (1 million years)			
Source term of reference calculation (radiolytic dissolution)		Source term in sensitivity study (controlled by Uranium solubility)	
Mass released by labile fraction	Mass released by the matrix	Mass released by labile fraction	Mass released by the matrix
8.4 %. M_0 [77]	91.6 %. M_0	8.4 %. M_0	**1 %** M_0
Equals **100 %** of the total initial mass of iodine-129		Equals **9.4 %** of the total initial mass of iodine-129	

Table 5.5-25 SEN – Total molar flow of source term at 1 million years for iodine-129 from CU1s

Thus, since at one million years only 10 % of the mass has been released by the packages and a large majority of this activity is labile, the molar flow exiting the geological barrier should be attenuated to the same degree (by a factor of 10) in the reference calculation with respect to the sensitivity study. The results corroborate this conclusion : they do make clear that the essential part of the radiological impact from iodine-129 is produced by the labile fraction, and that the molar flow exiting the Callovo-Oxfordian is approximately 10 times lower in the sensitivity study than in the reference calculation. The associated doses are also lower by approximately one order of magnitude (maximum dose of 0.0018 mSv/yr. for the Saulx outlet of the Oxfordian limestone versus 0.019 mSv/yr. at 330,000 yrs. for the reference calculation, see

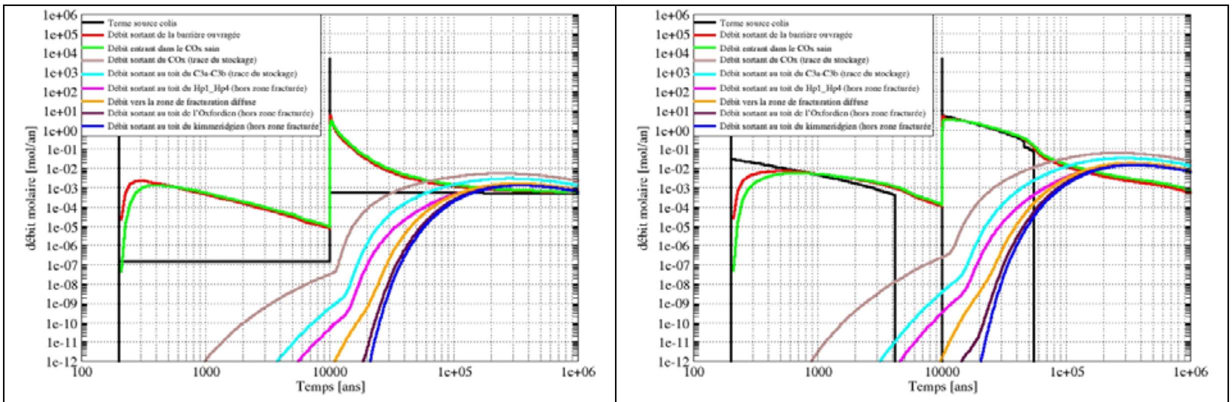
Figure 5.5-60 and

Figure 5.5-61).

For the CU2 spent fuels, the difference between the iodine-129 molar flows exiting the Callovo-Oxfordian in the reference calculation and in the sensitivity study is somewhat less significant (increase by approximately a factor of 3), because the labile fraction that is used for the sensitivity study is greater (by approximately 35 %). Thus, in the reference calculation, the total activity is released by the CU2 packages over the analysis period, whereas in the sensitivity study only 36 % of the mass was released (35 % of labile and approximately 1 % due to classical dissolution). The dose is thus approximately 3 times lower than for the reference calculation (0.00059 mSv/yr. versus 0.0017 mSv/yr.).

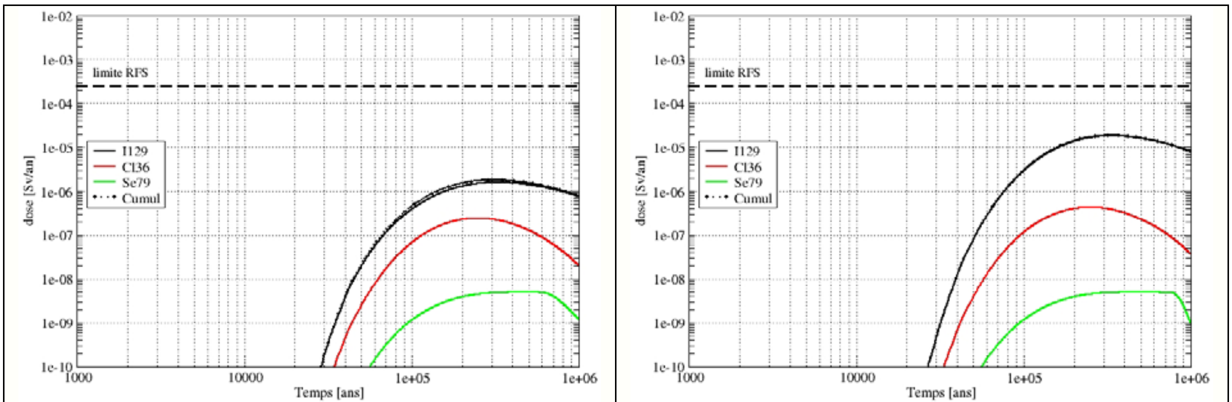
For chlorine-36, the gain is lower because more than half of the inventory is contained in the cladding, which has a release period of approximately 10,000 years. Considering the transfer time in the geological barrier, it can be considered the equivalent of a labile fraction from a physics point of view. It is not affected by a slower release model.

[77] M_o is the initial total mass.



Sensitivity study | Reference calculation

Figure 5.5-60 SEN – Sensitivity of spent fuel packages to release model (monitoring by uranium solubility) - CU1 reference package – ^{129}I – history of molar flow



Sensitivity study:

RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.8E-06	3.1E+05
I129	1.6E-06	3.2E+05
Cl36	2.4E-07	2.5E+05
Se79	5.0E-09	4.7E+05

Reference calculation:

RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.9E-05	3.3E+05
I129	1.8E-05	3.4E+05
Cl36	4.3E-07	2.5E+05
Se79	5.0E-09	4.7E+05

Figure 5.5-61 SEN – Sensitivity of spent fuel packages to release model (controlled by uranium solubility) - CU1 reference package – ^{129}I – Dose at Saulx outlet of Oxfordian

5.5.6.3 Sensitivity Study on the Overall Calculation Model

- **Sensitivity Regarding Transfers During the Hydraulic Transient**

Definition of the Sensitivity Study

As we have seen, the model of the normal evolution scenario ignores the hydraulic transient period and starts from a situation where the repository is completely and instantly resaturated. Section 5.3.1.3 explains to what extent this choice was conservative and did not lead to underestimating the impact. In particular, we saw that the hydraulic transient phases are not phases during which radionuclides are liable to be released.

Nevertheless, at this stage in the studies, no confinement function has been adopted for B overpacks. From the viewpoint of functional analysis, it cannot therefore be excluded that radionuclide releases may have taken place after closure of the repository, especially during the phase when gases affect water pressures and flows. However, from the phenomenological point of view, such a release is difficult or even impossible to imagine since it corresponds to a phase when B waste cells are highly desaturated. But in order to study its effect, it was decided that the sensitivity study should take into account a release by B waste cells at a stage when flows are not guided by natural gradients, but by transient phenomena.

Similarly, the possibility of a defect in a C over-pack or spent fuel container was not excluded. For the same reasons as those mentioned above, such a defect does not, in principle, cause a release during the hydraulic transient. However, a sensitivity study allows for the possibility that prematurely defective thermal waste packages may release their activity in a medium simultaneously subjected to the thermal and hydraulic transients.

The specific difficulty of such a simulation is the representation of flows in a medium in which there is a sufficiently large amount of gas to affect the degree of saturation of the materials. In such a configuration, the gas occupies all or part of the materials' porosity, and once it has removed the free water that its pressure has enabled it to dispel, it prevents water from circulating. Thus it reduces the porosity accessible to water. The migration of solutes by diffusion and advection in the various porous media is reduced due to this desaturation.

The various phenomena involved, their coupling and their sequencing cannot be represented in fine detail with current tools. Hence, Andra has performed a simplified, conservative transfer calculation from the disposal cells to the top of the host layer, assuming a transfer into the geological barrier and the various repository structures. A purely conventional, pessimistic representation was opted for, in which the gases place the water in the rock medium under pressure without desaturating it. The gas pressure thus creates a forced flow into the structures and into the host formation, and may generate a significant increase in the advective transfer of solutes. The calculations performed do not express a physical reality. The overpressure values from finer preparatory calculations (with the desaturation/resaturation phenomenon and gas pressure effect) at the level of disposal cells and drifts are taken pessimistically, imposing them in a system where all the materials are saturated (and whose transfer and retention properties are those of saturation). The calculation assumes a sequence of saturated hydraulic steady states, corresponding to different overpressure values applied per time interval. Representing such overpressures in a medium that remains at saturation would presuppose that water is continuously « created » in the zones undergoing gas pressure, so as to maintain the flow.

The calculation is applied to both transfer paths (the host formation and the access structures), and to B1x waste and spent fuel repository zones (bounding with respect to the impact from vitrified waste). As far as spent fuel is concerned, the cell containing the defective package is located close to the exit of the repository zone nearest the shafts ; this configuration both minimizes the distance between the defective package and the shafts, and maximizes the hydraulic influence of the repository drifts upstream.

Modelling Transients

The thermal transient is taken into account in the calculation performed on the CU1 in the same manner as for the reference calculation. In addition, insofar as the presence of gas may increase advective transport in the host formation and make it dominant with respect to transport by diffusion, the effect of temperature on permeability has also been taken into account in modelling transfer through the geological medium.

The hydraulic transient is taken into account in the model by applying the pressures resulting from calculations on gas production and accumulation [61]. These calculations are conservative, given that they assume that corrosion progresses at a rate that is high for an anoxic, reducing medium (greater than a micron per year) and that it is independent of the level of saturation.

For the transfer path via the geological medium, these water pressures are applied over the whole thickness of the Callovo-Oxfordian, by successive time increments. They are relatively high in the cells, but dissipate quite quickly after a few metres (see Figure 5.5-62). The pressure reaches its highest value at about 500 years for B waste and 2000 years for CU1 spent fuel. In the case of spent fuel, it is assumed that after 10,000 years the pressure has dropped sufficiently so that the gas imposed pressure in the Callovo-Oxfordian does not require simulation.

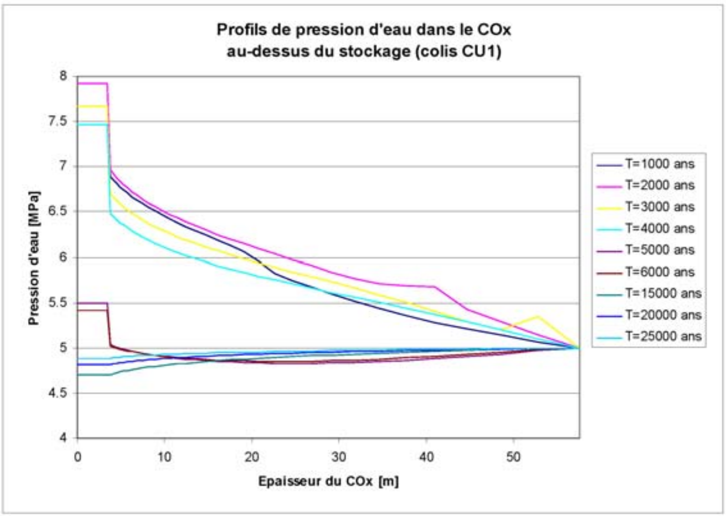


Figure 5.5-62 Water Pressure in the Callovo-Oxfordian (COX) above the Repository – Reference package CU1

For the transfer path via the structures, in B1x waste cells and for the cell with a defective CU1 package, the model adopts a pessimistic scheme so as to simplify the calculation : the hydraulic transient is represented over two time phases ; for each of these phases, a water pressure is assumed in the cell enclosure (actually at the level of the package) corresponding to the maximum gas pressure in the cell as shown in Figure 5.5-63 and Figure 5.5-64. However, the presence of gas within the cell access drifts and beyond is neglected. This maximizes the pressure gradient between the cell and the adjacent drift, and thus « forces » circulation in the structures. Each cell involved thus behaves like a pressurized water source supplying the repository.

In the case of spent fuel, the hydraulic overpressure imposed by the gas is :

- approximately 300 metres for 5000 years (corresponding to a gas pressure on the order of 9 MPa) ;
- then 50 metres for another 5000 years ;
- then a return to the state of equilibrium with the surrounding medium.

For B waste, the hydraulic overpressure applied is approximately :

- 200 metres for 1000 years ;
- then 100 metres over the next 2000 years ;
- then a return to the state of equilibrium with the surrounding medium.

It is important to emphasize the unrealistic nature of this model, which does not represent a physical reality, but ensures a conservative approach.

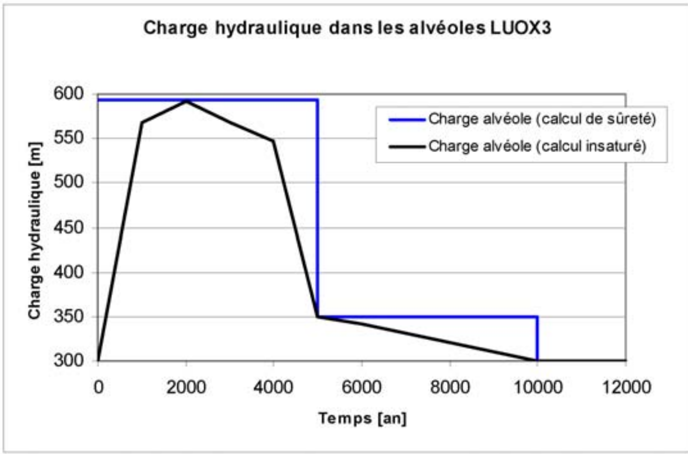


Figure 5.5-63 SEN – Sensitivity calculation on the influence of the hydraulic transient – Hydraulic head imposed during the overpressure phase – Reference package CU1

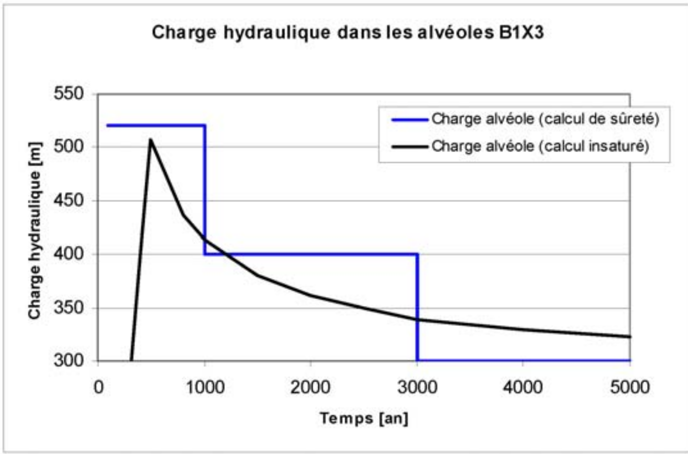


Figure 5.5-64 SEN – Sensitivity calculation on the influence of the hydraulic transient – Hydraulic head imposed during the overpressure phase – Reference package B1x

Calculation Results

Spent fuel

For the transfer path via the geological barrier, the results reveal that the molar flow rates out of the Callovo-Oxfordian relative to the defective package (with and without taking gas overpressures into account) are almost identical. The hydraulic transient period is sufficiently short with respect to the transfer time for its influence not to be visible at the exit from the Callovo-Oxfordian. It should further be noted that in the Normal Evolution Scenario, the impact of the defective package is masked by that of the non-defective packages (for which the overpressures have no effect). This means the conclusion can be reached that the influence of overpressures is negligible on the molar flow rate out of the Callovo-Oxfordian and especially on the impact via this path (see curves below).

As far as the transfer path via the structures is concerned,

Figure 5.5-65 illustrates the hydraulic head distributions in NGF[74] meters, in the structures for the three steady states considered (the two transient phases and the permanent regime , see above).

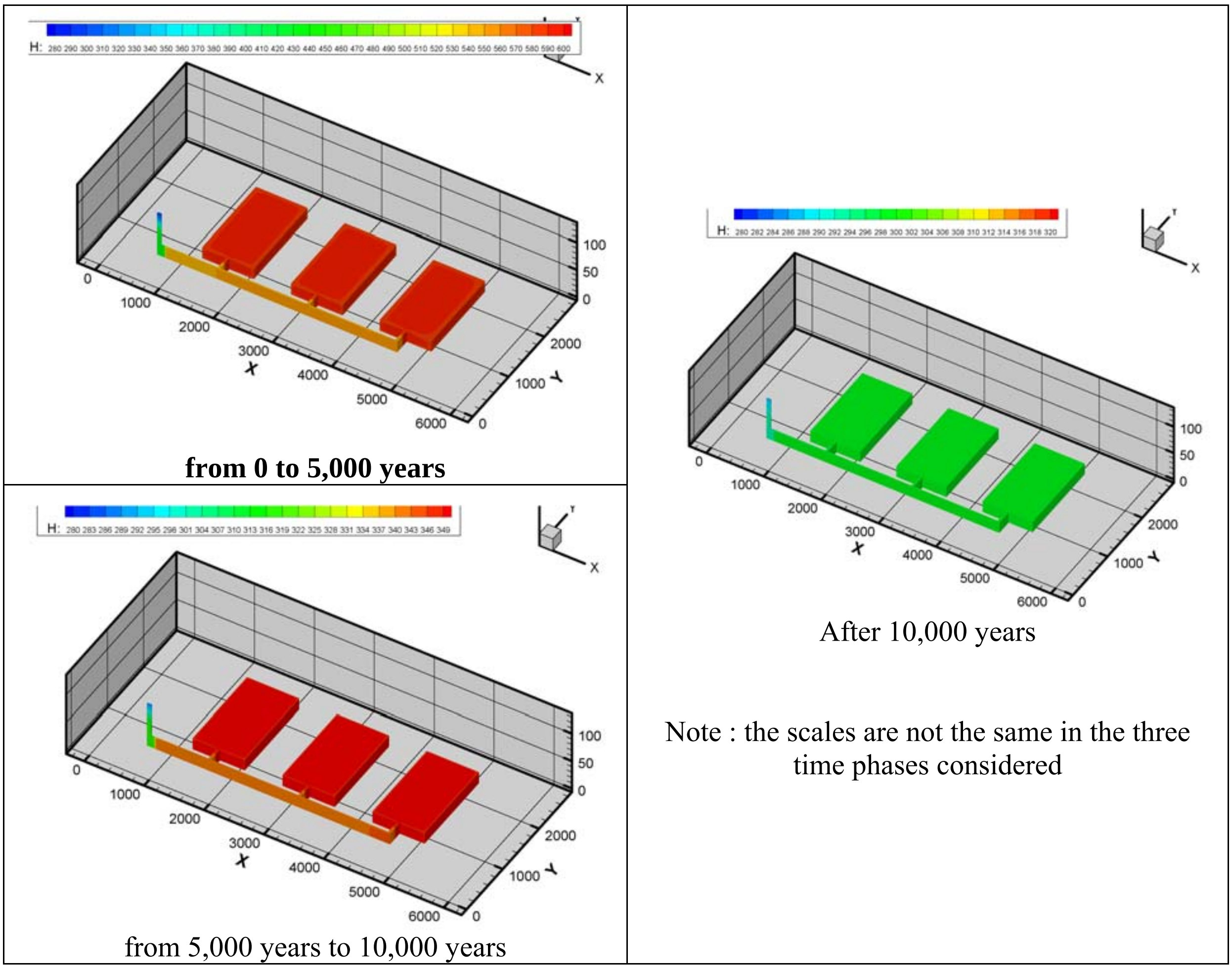


Figure 5.5-65 Normal Evolution Scenario sensitivity calculation on the influence of the hydraulic transient – mapping of hydraulic heads (NGF[74]), (m) for access structures and CU1 spent fuel repository zones

When the overpressures are high (0-5000 years), the head losses on either side of each seal are significant (several tens of metres). The same applies at the level of the shaft seals where the head at the base of the shafts is significantly higher that at the top (approximately 100 metres). These steep head gradients are liable to induce significant Darcy velocities[78] a non negligible molar flux appears at the level of the zone seals and liaison drifts after a few thousand years.

[78] As an example, during the first phase of the transient (0-5000 years), the Darcy velocity is of the order of 0.1 m/year in the main connecting drifts before sealing. Note that the Darcy velocities are variable over time (as a function of overpressures) and depend on the position of the drifts in the repository.

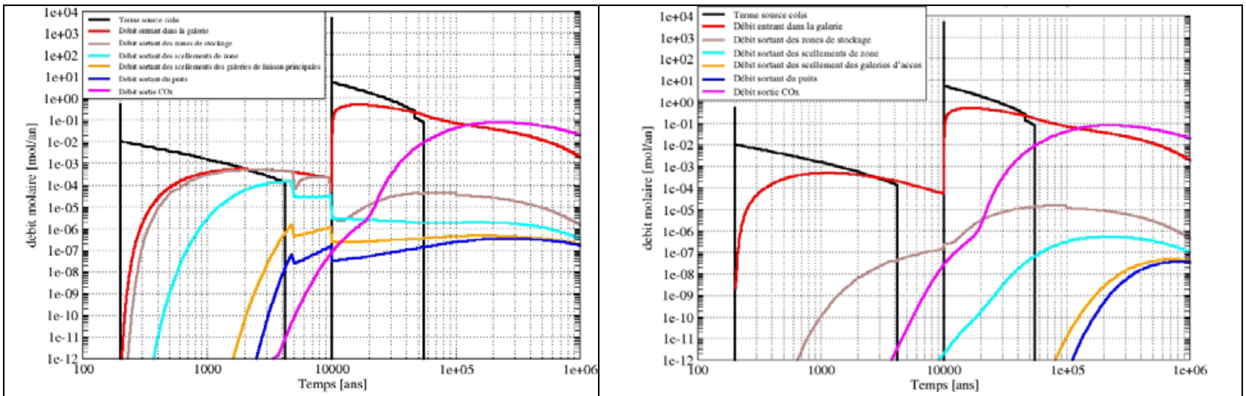
The molar flow rate historical curves shown in Figure 5.5-66 highlight the influence of gas pressures on the transfer path through the structures :

- under the adopted modelling conditions, the influence of gas causes an increase in the maximum molar flow rate of approximately one order of magnitude at the shaft exit and advances the date of the maximum by approximately 500,000 years (300,000 years instead of 800,000 years). Most of the iodine-129 coming out of the shafts originates from the defective package, even after the hydraulic overpressure transient : when, at 10,000 years, the system reverts to natural equilibrium, the radionuclide part originating from the defective package already present in the secondary connecting drifts presents relatively low transfer times to the shafts. Nevertheless, we should note that the amount of iodine-129 released by the shafts in a million years only corresponds to 5 % of the mass initially present in the defective package (that is, approximately 4.10^{-4} % of the total initial inventory against 3.10^{-5} % for the reference calculation) ;
- for the first 10,000 years the molar flow rate at the shaft exit is greater than that originating from transfer through sound argillites (in contrast to the reference calculation) ; after that, it becomes less. In any case, the molar flow rate from the shafts remains low over all the time phases and the maximum, reached at around 300,000 years, is very much less (approximately six orders of magnitude) than that from the sound argillites.

The molar flow rate historical curves also highlight the importance of the various seals during the hydraulic transient. At approximately 10,000 years, we note that :

- at the exit from the zone seal, the molar flow rate is attenuated by approximately one order of magnitude, thanks to the efficiency of the zone seal ;
- it is alternated by another order of magnitude at the level of the main gallery seals ;
- Transfer through the shaft seal diminishes the molar flow by yet another order of magnitude.

These results show that the efficiency of the seals contributes to attenuating the transfer path through the structures. The latter remains negligible compared with the transfer path via the geological barrier at the million-year scale.



Sensitivity Calculation | Reference Calculation

Figure 5.5-66 Normal Evolution Scenario – Sensitivity calculation on the influence of the hydraulic transient – molar flow rate history – ^{129}I – spent fuel CU1

In conclusion, taking into account a transient hydraulic overpressure due to gas generation has no significant influence on radiological impact, the levels and dates of occurrence of maximum doses remaining the same as those of the reference calculation.

B waste packages

It should be noted that the hydraulic disturbance over the B waste zone is shorter and less intense than that of spent fuel.

As far as the transfer path through the geological barrier is concerned, note that, as for spent fuel, the potential influence of overpressures is masked by the transfer time in the Callovo-Oxfordian layer, clearly longer than the hydraulic transient period. The curves of molar flow rate out of the Callovo-Oxfordian are combined with those of the reference calculation (see Figure 5.5-67).

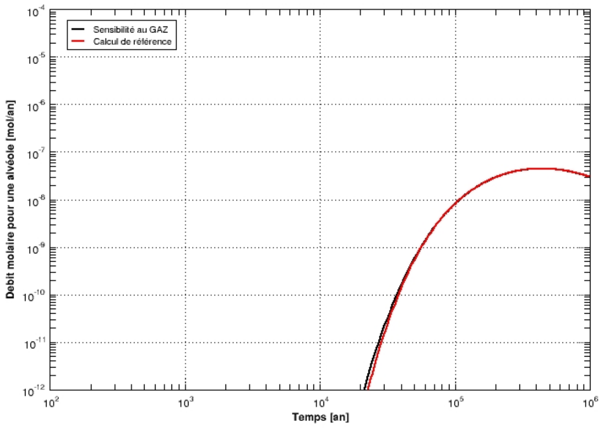
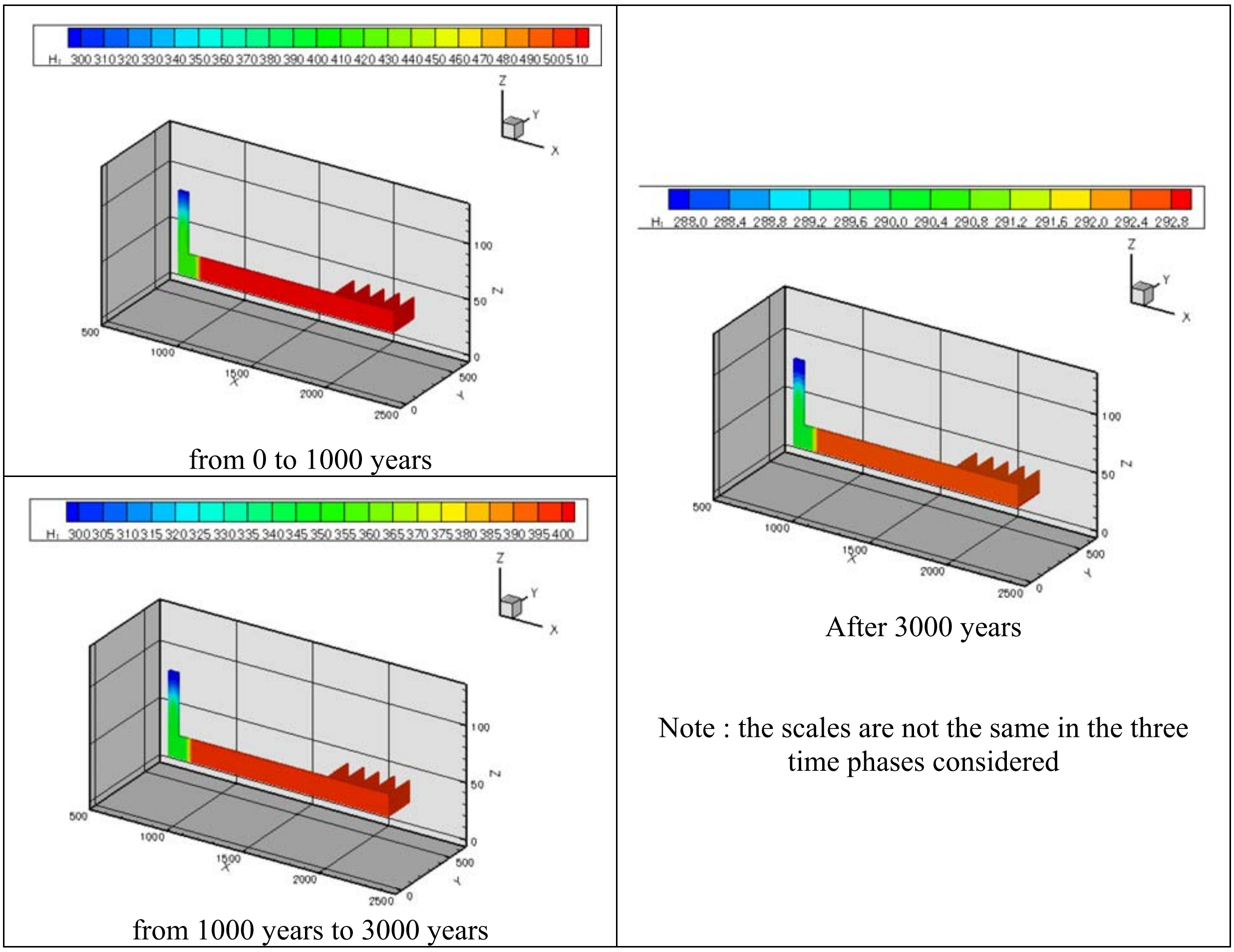


Figure 5.5-67 Normal Evolution Scenario – sensitivity calculation on the influence of the hydraulic transient – historical molar flow rates out of Callovo-Oxfordian for a B1x waste cell - comparison of cases with and without hydraulic overpressures – ^{129}I

As far as transfer via the structures is concerned, the results reveal that the molar flow rate out of shafts remains much lower than that from sound argillites of the Callovo-Oxfordian, and the influence of overpressures due to gas during the hydraulic transient may be regarded as negligible. The results with respect to hydraulics and transport are detailed below.

Figure 5.5-68 illustrates the distributions of hydraulic head in the structures for the three steady states considered.



from 0 to 1000 years

from 1000 years to 3000 years

After 3000 years

Note : the scales are not the same in the three time phases considered

Figure 5.5-68 Normal Evolution Scenario – sensitivity calculation on the influence of the hydraulic transient – mapping of hydraulic heads (NGF[74], m) for access structures and B1x package repository zones

The seals fulfil the same role as the one assigned in the calculation for the spent fuel zone. The head gradients are similar and likely to induce Darcy velocities capable of reaching 0.07 m/year.

From the transport point of view, given the length of the B waste cells and the sorption of iodine in concretes, the transfer times from the disposal cells to the adjoining drift are several tens of thousands of years. Hydraulic overpressure in the cell accelerates radionuclide transfer to the drift during the first 10,000 years (Figure 5.5-69). After that, the system returns to equilibrium and the release levels are the same as those of the reference calculation. Given the low hydraulic kinetics imposed by the B waste repository, which has only a limited drainage capability, the transfer times to the access shafts are significant, and the molar flow rate out of the shafts is very little increased compared with a situation disregarding the influence of overpressures.

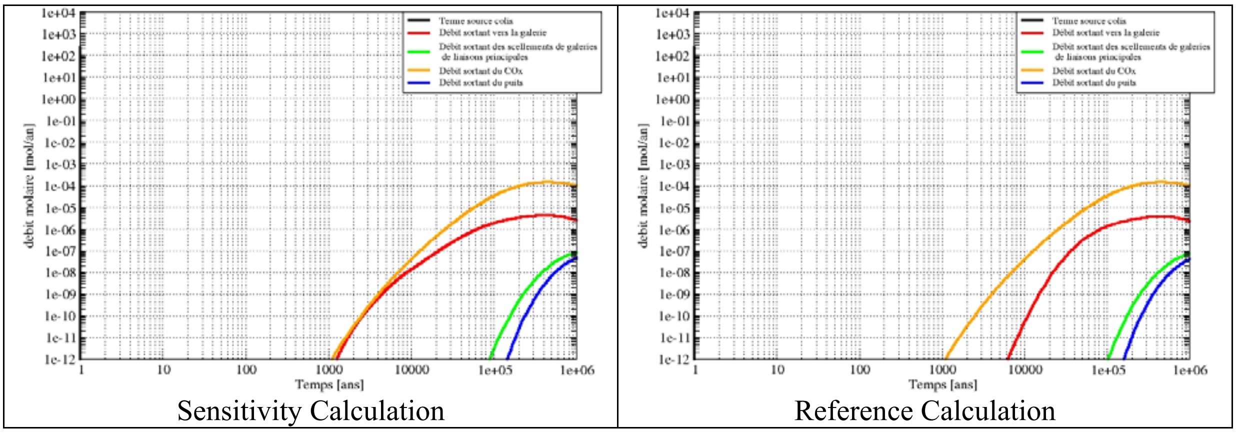


Figure 5.5-69 Normal Evolution Scenario – Sensitivity calculation on the influence of the hydraulic transient – molar flow rate history – ^{129}I – Non-organic reference packages not releasing gaseous hydrogen.

The results shown reveal that in the Normal Evolution Scenario, the hydraulic transient has no detrimental influence on the impact of the repository. Overpressures transiently degrade the performances of the « limiting water circulation » function since they cause an increase in the advective kinetics in the geological barrier and drifts during the first few thousand years. But the duration of this hydraulic transient is sufficiently short with respect to the transfer times in the geological barrier for its influence not to be visible on the molar flow rate out of the Callovo-Oxfordian.

The molar flow rates out of shafts may be significantly greater than for the reference calculations and the maxima occur earlier in the case of spent fuel. Nevertheless, they remain low compared with those out of the geological barrier.

Thus, it may be concluded that the hydraulic transient, even modelled in a very pessimistic way, has no significant effect on the Normal Evolution Scenario impact, the maximum dose levels being identical to those of the reference calculation. Note that it has been modelled in a pessimistic way, by reflecting the desaturation induced by gases.

- **Sensitivity Study on the Diffusive Properties of Semi-Permeable Layers in the Overlying Surrounding Rocks**

It has been seen that the choices of models for the surrounding rocks adopt high diffusion coefficient values, which in some situations could lead to underestimating the impact of radionuclides passing through by advection (see section 5.3.2.4). A sensitivity study has been conducted, adopting lower values for the diffusion coefficient and the porosity accessible to diffusion in the Kimmeridge so as to limit the diffusive spread. For consistency, the same logic is followed for the 20 metre-thick C3a layer which is also not very permeable.

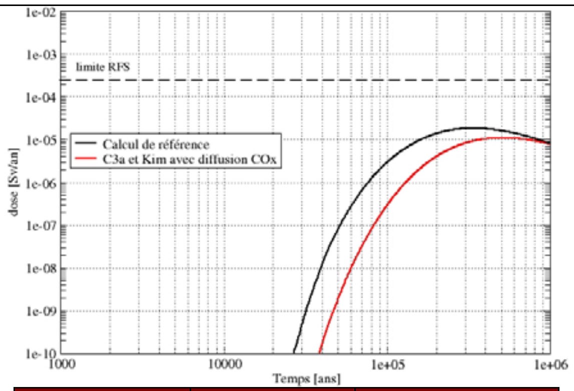
The diffusion coefficient and porosity values of the Callovo-Oxfordian are adopted for these two layers. These values seem realistic for the C3a layer, but are certainly overestimated for the Kimmeridge – but the aim of this calculation is precisely to overestimate its properties with regard to diffusion.

Since the calculation is performed for spent fuel iodine-129, which is the main contributor to the impact, an effective diffusion coefficient of 5.10^{-12} m²/s and a porosity accessible to diffusion of 5 % are assumed (values adopted for anions).

In this situation, no increase is observed in the dose at the Saulx and the Ornain ; the results actually reveal that :

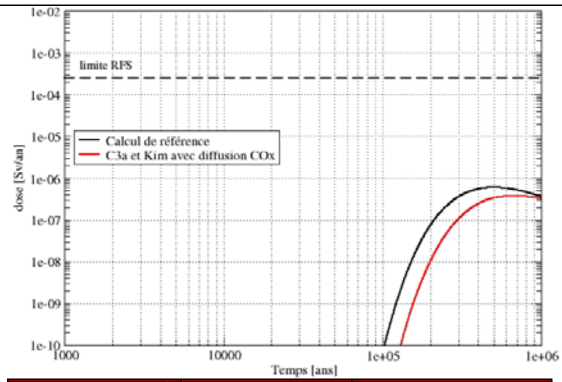
- taking into account phenomenological transport values in the 20 metres of the C3a layer at the top of the Callovo-Oxfordian causes an increase in diffusive transfer times up to the top of the C3a-C3b layers by a factor of approximately 1.7,

- taking into account a low diffusion coefficient value in the Kimmeridge does not lead to increasing the flow of radionuclides travelling by advection in the Oxfordian to the point where the impact is modified : the effects due to the performances of the C3a horizon remain dominant and overall lead to a weaker impact. The maximum dose at the Saulx and Ornain outlets drops by approximately 70 % and the occurrence of the dose maximum is delayed by about 170,000 additional years, as illustrated in Figure 5.5-70. Thus, the maximum dose of spent fuel CU1 iodine-129 is 0.011 mSv/year at 510,000 years at the Saulx outlet, while it is 0.018 mSv/year at 340,000 years in reference ;



	Max dose (Sv/year)	t(Max dose)
Reference	**1.8 x 10^{-5}**	340,000 years
Sensitivity	**1.1 x 10^{-5}**	510,000 years

Saulx Outlet

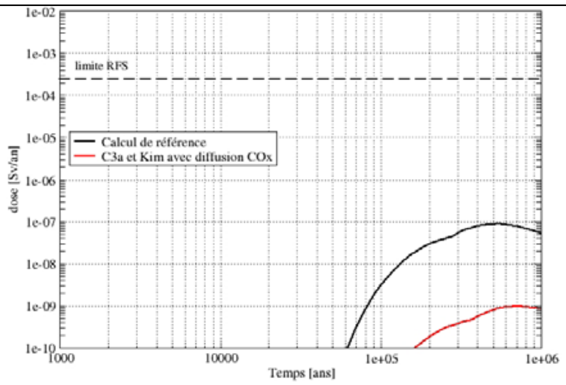


	Max dose (Sv/year)	t(Max dose)
Reference	**6 x 10^{-7}**	500,000 years
Sensitivity	**3.8 x 10^{-7}**	675,000 years

Ornain Outlet

Figure 5.5-70 Normal Evolution Scenario - Sensitivity to the diffusive properties of semi-permeable layers in the overlying surrounding rocks - Reference package CU1 – ^{129}I -Dose at the Saulx and Ornain outlet of the Oxfordian – Reference calculation comparison / Sensitivity study

- the more difficult diffusion in the Kimmeridge attenuates the maximum dose at the Barrois outlet by approximately two orders of magnitude as shown in Figure 5.5-71



	Max dose (Sv/year)	t(Max dose)
Reference	**8.8 x 10^{-8}**	530,000 years
Sensitivity	**9.7 x 10^{-10}**	700,000 years

Figure 5.5-71 Normal Evolution Scenario - Sensitivity to the diffusive properties of semi-permeable layers in the overlying surrounding rocks - Reference package CU1 – ^{129}I -Dose at the Barrois – Reference calculation comparison / Sensitivity study (H= hydraulic loads in NGF meters)

These results help confirm that allowing for less diffusion in the Kimmeridge and the lower horizons of the Oxfordian means reducing the impact at the outlets. The reference calculation assuming high diffusion coefficient values in the semi-permeable layers of the overlying surrounding rocks (diffusion coefficient of water in water – 10^{-9} m^2/s) is therefore well covered by situations taking into account the phenomenological transport parameter values in the surrounding rocks.

- **Sensitivity Study Regarding pathways in the overlying formations model**

The impact assessments were conducted in a conventional manner, assuming each repository zone to be centred on the location of the Meuse Haute-Marne laboratory. This location places the repository closest to the « Saulx » outlet in the 1 million year model. We have seen that, for small repository zones, all the advective paths in the Oxfordian are headed towards this outlet. In the case of extensive zones, such as the CU1 zone, the flows are partly headed towards the Ornain outlet. However, the CU1 zone could not be moved closer to the Saulx outlet. The reference location adopted already corresponds, among those near the Meuse Haute-Marne laboratory site, to a position on the transposition zone boundary.

However, it might be useful to « test » the reference of the trajectories in the hydraulic model on impact. That ca be done by calculating pathways standing from other locations of the repository in the transposition zone. The relevant sensitivity analysis consisted in « moving » the repository without altering the model in any other way to evaluation the role of the hydrological model alone.

The calculation was performed for the CU1 zone, for iodine-129, chlorine-36 and selenium-79.

Siting the CU1 zone at another location is just as arbitrary as the choice of reference location. The locations adopted must be consistent with the aim of the study : this is not a matter of looking for the « best » place to position a repository, but rather of checking that locations that are in principle unfavourable do not lead to increasing the impact very significantly. The aim is thus to check that the location of the Meuse Haute-Marne laboratory does not correspond to a singular position, which would be especially favourable.

Two locations have been chosen for testing, corresponding to the model at 1 million years ; it is here that the local outlets are most numerous. The repository zone was moved on either side of the crest line separating the flows to the west (towards the Saulx and Marne valleys) and those towards the Ornain (see Figure 5.5-72).

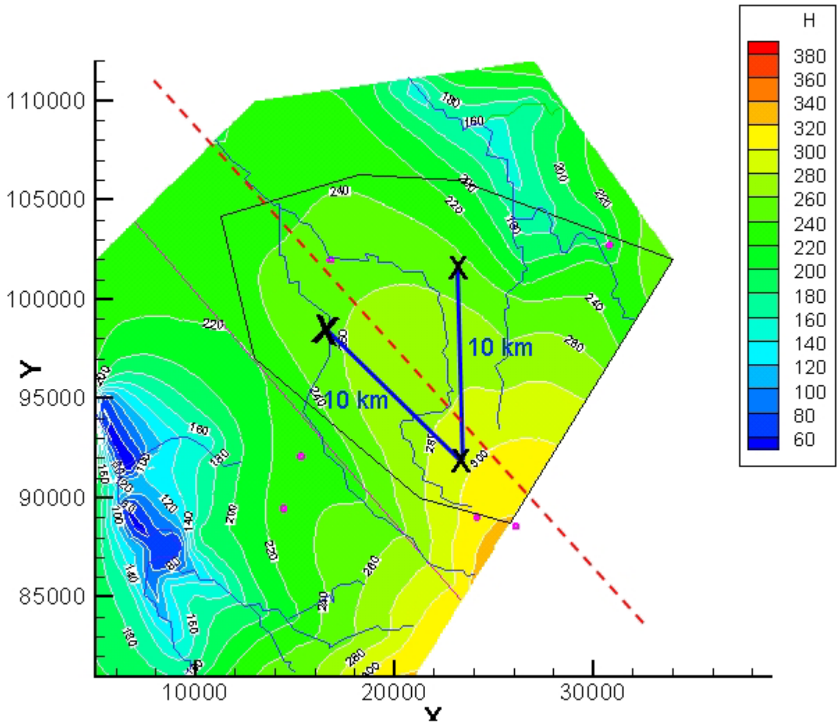


Figure 5.5-72 Normal Evolution Scenario - Repository Positions in Sensitivity Study (H= hydraulic loads)

The repository is modelled keeping the same argillite thickness with respect to the Oxfordian as in reference, i.e. the repository is positioned 65 metres from the top of the host formation, so that possible thickening of the Callovo-Oxfordian in the new positions tested does not mask the effects due to changes in hydraulic paths in the surrounding rocks. In accordance with the choices put forward for the reference calculation, the geomechanical behaviour adopted corresponds to the deepest position in the transposition zone, with the gradient rising and at its maximum (0.4 m/m) whatever the location.

By placing the repository ten kilometres from the laboratory site, to the north-west and north-east respectively, and tracing the paths from the hydrogeological model, we find that :

- for the north-west position, the hydraulic pathways to the Saulx are eliminated and replaced by paths to the confluence of the Marne and the Rongeant. The « interception of paths to the Marne » outlet, defined in the current hydrogeological model and which intercepts these paths at the level of the diffuse fracturing zone extension, then becomes a relevant outlet in the 1 million year model. Moreover, it is situated very close to the repository zone ; Figure 5.5-73 reveals that most of the paths are headed towards this outlet and that only a minute portion follow a regional path in the north-north-westerly direction. Calculating the dose at this outlet is the same as for the reference calculation : a pumping site at the start of the diffuse fracturing zone is assumed to capture all the radionuclides heading towards the Marne valley. This calculation can therefore be used to check whether or not the impact is sensitive to the position of the repository along the diffuse fracturing zone ;

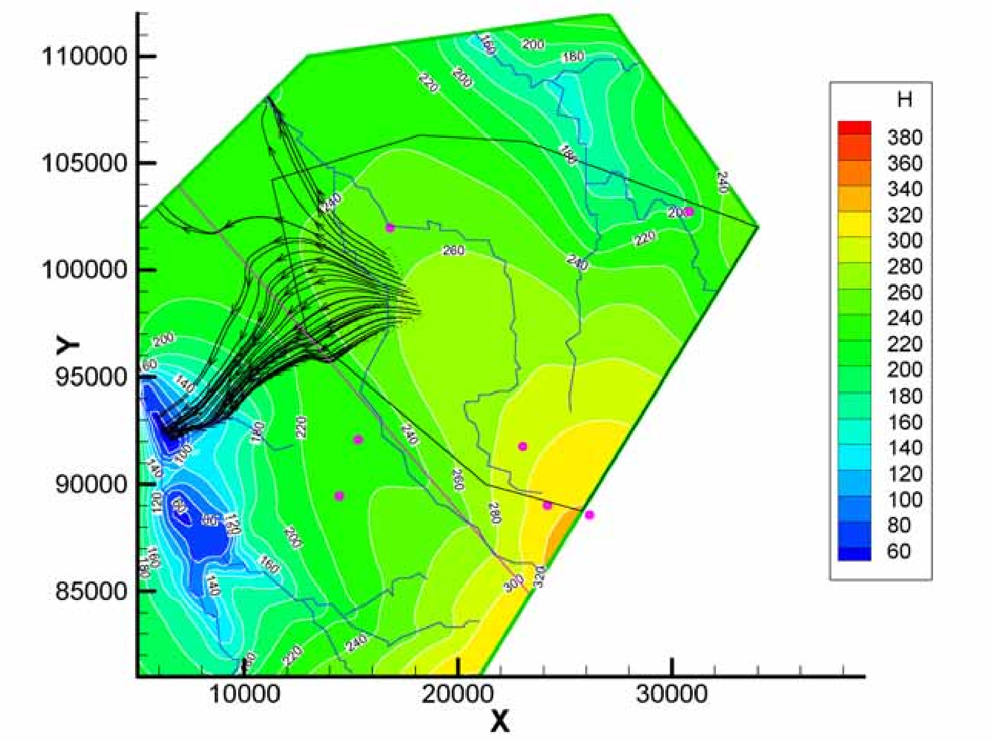


Figure 5.5-73 SEN – Sensitivity to the position of the repository - hydraulic pathways in the porous horizons Hp1-Hp4 – Position of the North repository (H= hydraulic loads in NGF[74] meters)

- in the case of the north-east position, the repository zone is located near the Ornain outlet, which favours the latter over the Saulx outlet, which no longer appears. Figure 5.5-74 shows that the dispersal of the paths from the repository is fairly restricted and that they all converge towards and along the Ornain valley. It should be noted, however, that other hydraulic outlets may emerge to the north of this valley. This position is particularly pessimistic, since it amounts to deliberately maximizing the dose at the Ornain.

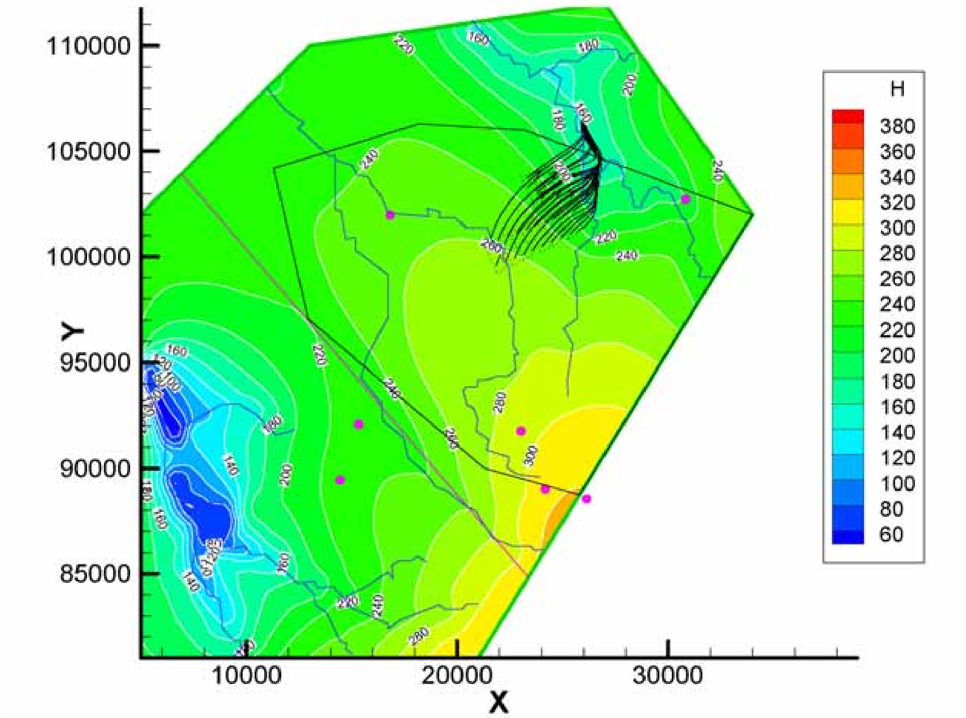


Figure 5.5-74 Normal Evolution Scenario – Sensitivity to the position of the repository - hydraulic pathways in the porous horizons Hp1-Hp4 – Position of the North repository

The results show that the main contributor to the impact is iodine-129. The dose calculations at the various outlets reveal the following results, which are compared with the reference calculation (see Table 5.5-26).

Outlet	Reference calculation (reminder)		North-west sensitivity		North-east sensitivity	
	Maximum dose (mSv/year)	Date of maximum [years]	Maximum dose (mSv/year)	Date of maximum [years]	Maximum dose (mSv/year)	Date of maximum [years]
Outlet upstream of the diffuse fracturing zone[79]	0.019	330,000	0.023	400,000	Not applicable	Not applicable
Ornain outlet	0.0006	500,000	Not applicable	Not applicable	0.19	390,000

Table 5.5-26 Normal Evolution Scenario – Sensitivity to the position of the repository - Impact of spent fuel CU1 in the 1 million year hydrogeological model, for different repository locations

[79] When calculating sensitivities, the outlet is the « interception of paths to the Marne » outlet, and in the reference calculation the « Saulx » outlet.

The results show little difference between the position on the Meuse Haute-Marne laboratory site and the north-west position, solely from the point of view of transfer in the surrounding rocks. This is the expected result, insofar as a choice of outlets close to the repository and intercepting all the radionuclide flows makes the impact little dependent on the detailed characteristics of the hydrogeological model. This result must, of course, be tempered by the fact that, at the laboratory level, the geomechanical behaviour is more favourable than what is shown in the model ; however, the host formation is thicker to the north-west than it is in the calculation.

The position to the north-west is, as could be expected, the most pessimistic, given that the repository is very near the outlet. In this case the dose increases by a factor of 10 approximately. The study of concentration plumes shows that dispersion is negligible between the repository and the outlet, and therefore that it would probably not be possible to find a more unfavourable configuration in the zone. Moreover it is particularly unrealistic due to the permeability of the formalisation this location, pumping flows of water are very weak and would lead the people to using the nearby river instead. The calculated dose is, in this respect, a pessimistic overestimation of possible doses, and relates to a location that is evidently not optimal from the sole point of view of impact reduction. Note that all other parameters of the models (the depth of the repository in particular) are set at conservative values not corresponding to what could be possible near the Orain zone.

This study does not cover all the possible locations of the repository within the transposition zone. In keeping with its objective, it shows that the impacts calculated in the reference position do not correspond to an isolated case, but that the same order of magnitude of impact can be obtained by moving the repository about within the zone. It also shows that the impact calculated at the « Saulx » outlet is not very sensitive to the hydrogeological model, and therefore that the calculated result is robust. In the most unfavourable position in principle, the increase is only of one order of magnitude.

5.5.6.4 Summary of the Impact of the Normal Evolution Scenario Sensitivity Studies

The maximum doses and the dose maxima dates of the sensitivity studies are summarized in Table 5.5-27. Only doses at the Saulx outlet of the Oxfordian limestone are presented, it being the outlet generating the greatest impact. Calculations leading to doses at other outlets, such as performed to study the sensitivity on the hydrogeological model, are not shown again here.

		Total spent fuel Scenario S2		Total waste C1+C2 Scenario S1b		Total waste C3+C4 Scenario S1a		Total B waste Scenario S1b	
		Maximum accumulat-ed dose (mSv/year)	Date of the maximum dose (years)	Maximum accumulat-ed dose (mSv/year)	Date of the maximum dose (years)	Maximum accumulat-ed dose (mSv/year)	Date of the maximum dose (years)	Maximum accumulat-ed dose (mSv/year)	Date of the maximum dose (years)
1. Sensitivity to the parameters of the Callovo-Oxfordian, bentonite and concrete of the cells									
1.1	Conservative permeability of the Callovo-Oxfordian (10 times the « phenomenological » values)	0.020 [80] (idem ref.)	330,000 (idem ref.)	0.00047[30] (idem ref.)	490,000 (idem ref.)	0.00036[30] (idem ref.)	500,000 (idem ref.)	0.00047[30] (idem ref.)	310,000 (idem ref.)
1.2	EDZ Pessimistic fractured zone Conservative microfissured zone	0.020 [30] (idem ref.)	330,000 (idem ref.)	0.00047[30] (idem ref.)	490,000 (idem ref.)	0.00036[30] (idem ref.)	500,000 (idem ref.)	0.00047[30] (idem ref.)	310,000 (idem ref.)
1.3	Conservative transfer and retention parameters in the Callovo-Oxfordian, the bentonite and the concrete of the cells	0.045	170,000	0.0037	200,000	0.0028	210,000	0.00079[81]	220,000
1.4	Iodine partition coefficient of 10^{-3} m^3/kg in the Callovo-Oxfordian	Iodine-129 dose less than that of the reference calculation Date of the maximum iodine dose after 1 million years							
1.5	Callovo-Oxfordian thickness of 160 m.	Maximum accumulated doses less than those of the reference calculation							
1.6	« Phenomenological » thermal	0.020 (idem ref.)	330,000 (idem ref.)	0.00047 (idem ref.)	490,000 (idem ref.)	0.00036 (idem ref.)	500,000 (idem ref.)	0.00047 (idem ref.)	310,000 (idem ref.)

80 Specific doses were not calculated for these studies, which were limited to intermediate indicators (in particular, radionuclide flux exiting the Callovo-Oxfordian formation). As the values of these indicators are very similar to those found in the reference situation, doses given are those of the reference calculation.

81 The maximum accumulated dose only corresponds to that of non-organic waste package types not releasing any hydrogen (Blx), since the study has been carried out for these. In the reference calculation, the maximum accumulated dose associated with these packages is 0.00033 mSv/year ; it occurs at 310,000 years.

		Total spent fuel Scenario S2		Total waste C1+C2 Scenario S1b		Total waste C3+C4 Scenario S1a		Total B waste Scenario S1b	
		Maximum accumulated dose (mSv/year)	Date of the maximum dose (years)	Maximum accumulated dose (mSv/year)	Date of the maximum dose (years)	Maximum accumulated dose (mSv/year)	Date of the maximum dose (years)	Maximum accumulated dose (mSv/year)	Date of the maximum dose (years)
2 - Sensitivity to the waste packages									
2.1	B waste : Labile release of reference packages B1, B4, B5	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable	0.00047 (idem ref.)	310,000 (idem ref.)
2.2	B waste : conservative release rate of 10^{-3} per year of bituminized sludge	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable	0.00047 (idem ref.)	310,000 (idem ref.)
2.3	B waste : taking into account durable containers lasting 10,000 years for non-organic packages not releasing gaseous hydrogen (B1x)	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable	0.00047 (idem ref.)	310,000 (idem ref.)
2.4	C waste : conservative parameters of the model V0.S$\rightarrow$ V_r	Not applicable	Not applicable	0.00047 (idem ref.)	490,000 (idem ref.)	0.00036 (idem ref.)	500,000 (idem ref.)	Not applicable	Not applicable
2.5	C waste : pessimistic model V0.S	Not applicable	Not applicable	0.00063	280,000	0.00049	290,000	Not applicable	Not applicable
2.6	Spent fuel : Conservative parameters (radiolytic dissolution)	0.020 (idem ref.)	330,000 (idem ref.)	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable
2.7	Spent fuel : model controlled by solubility of uranium	0.0024	310,000	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable	Not applicable

		Total spent fuel Scenario S2		Total waste C1+C2 Scenario S1b		Total waste C3+C4 Scenario S1a		Total B waste Scenario S1b	
		Maximum accumulated dose (mSv/year)	Date of the maximum dose (years)	Maximum accumulated dose (mSv/year)	Date of the maximum dose (years)	Maximum accumulated dose (mSv/year)	Date of the maximum dose (years)	Maximum accumulated dose (mSv/year)	Date of the maximum dose (years)
3 - Sensitivity on the Overall Calculation Model									
3.1	Transfers during the hydraulic transient	0.019[82] (idem ref.)	330,000 (idem ref.)	-	-	-	-	0.00033[76]	310 000
3.2	Diffusive properties of semi-permeable layers in the overlying surrounding rocks Calculation performed for CU1 (spent fuel) iodine-129	0.011[83]	510,000	Maximum accumulated doses less than those of the reference calculation					

Table 5.5-27 : Normal Evolution Scenario – Dose maxima dates for the sensitivity studies at the Saulx outlet of the Oxfordian limestone (most pessimistic case) – 1 million year models – All waste

[82] The dose shown corresponds to that of spent fuel CU1 iodine and non-organic packages not releasing any hydrogen (Blx), since the study has been carried out for these.

[83] The dose shown corresponds to that of spent fuel CU1 iodine. The calculation was only carried out for this waste package type insofar as the radiological impact is less than that of the reference calculation. The dose associated with spent fuel CU1 iodine-129 is 0.019 mSv/year and occurs at 330,000 years in the reference calculation.

5.6 Main Lessons from Performance Analysis

The repository performance studies highlight a significant number of results for safety analysis.

Safety functions are guaranteed a good level of performance, in both the reference calculation and in the sensitivity studies.

For the « resisting water circulation » function, the diffusive transport regime dominates in all configurations within the host rock, and in most of the structures. It should be noted that this is not solely due to the efficiency of the seals : even when this is degraded in the sensitivity study, the flows remain limited overall, since the water from the Callovo-Oxfordian is insufficient to supply them. The dead-end disposal cell architecture is also favourable to flow restriction. Taking a degraded EDZ into account around the structures accordingly has a limited influence on the impact and distribution of the transfer paths between the geological barrier and the structures, even when the seals are not very effective. The geological barrier remains the main transfer path for radionuclides.

For the function of « limiting the release of radionuclides and immobilizing them in the repository ». The low solubility of many radionuclides in the cells means that their impact is heavily restricted ; this is especially the case of Selenium-79. The containers and over-packs contribute an element of confinement, helping to delay the occurrence of dose maxima, but without strong influence on their magnitude. The properties of the Callovo-Oxfordian attenuate the flows even in the case of transfer in a thermal environment.

These assessments are based on the preliminary results obtained on the radionuclide transfer in function of temperature. Given the uncertainties regarding these parameters, the presence of durable containers and over-packs during the thermal phase is a cautious option.

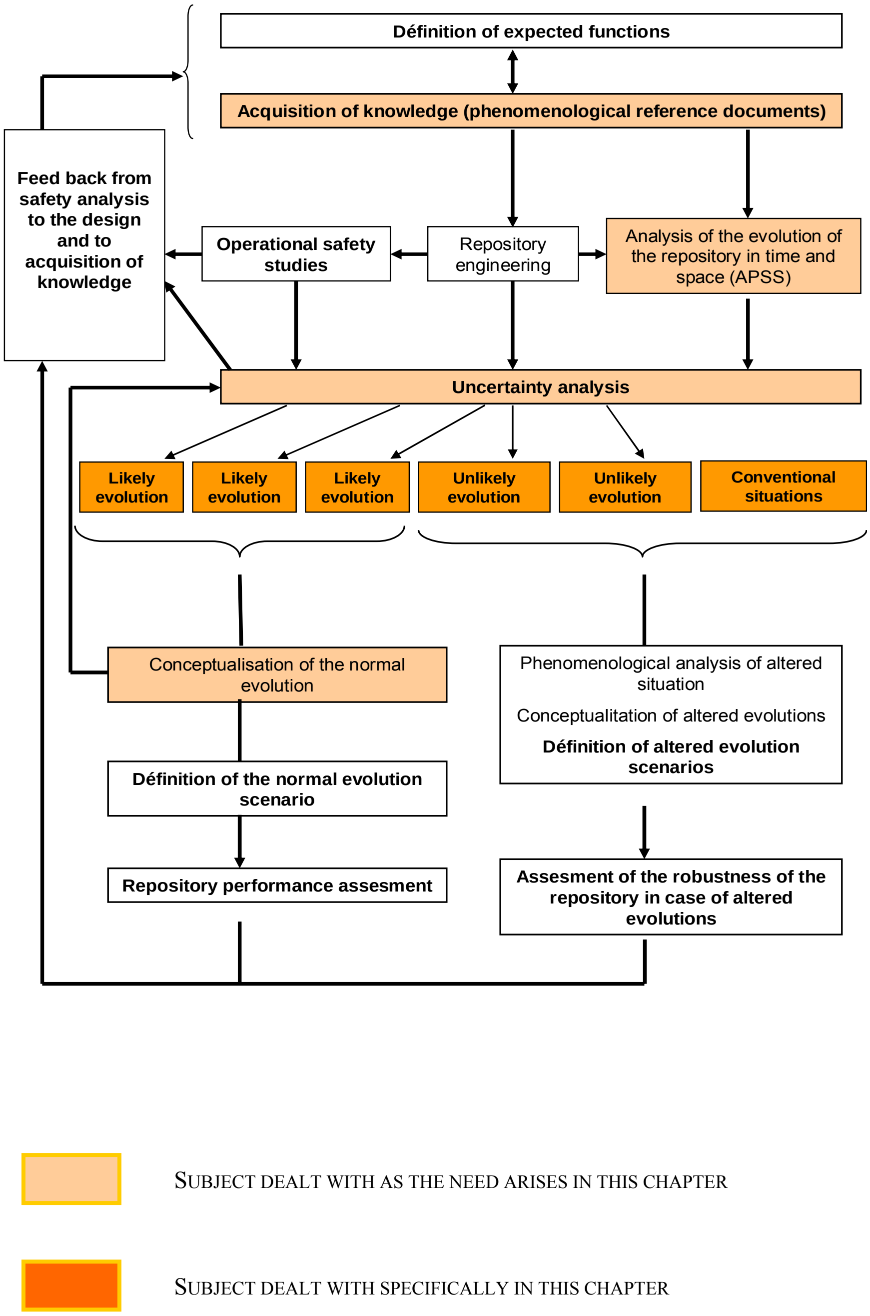
Taking conservative models and parameters into account for the source terms has little influence in the Normal Evolution Scenario where, in the reference calculation, the periods of radionuclide release by the packages are relatively short compared with the diffusive transfer time. It should be noted, however, that the glass release model, $V_0.S \rightarrow V_r$, has a visible influence on reducing the source terms. Whereas degradation of the spent fuel release model scarcely has. However, still for spent fuel, should the model based on classic dessolution ultimately prove to be the one recommended by the scientific community, it would significantly reduce the impacts associated with this reference package. The confinement role possibly provided by overpack B waste would not be significant with regard to impact. These conclusions regarding containers are only partial at this stage of the assessment volume : we should wait for the uncertainty analysis (chapter 6) and the assessment of altered scenarios (chapter 7) to draw conclusions revealing the full contribution of these packaging components.

For the function of « delaying and reducing the migration of radionuclides », the diffusion times are slow in the Callovo-Oxfordian and enable a decay of all the radionuclides that could contribute to the impact, except for iodine-129, chlorine-36 and selenium-79. The last two are, however, significantly reduced. The transport parameters prove sensitive in terms of the impact of these three radionuclides. In the argillites, the results reveal that the most influential factors are the diffusive transport parameters for the soluble, unsorbed elements like iodine and chlorine. Moreover, the influence of the conservative geochemical parameters (solubility and retention) adopted in the structures and in the geological barrier is considerable for selenium-79,. The prospective sensitivity studies show that a limited sorption of the iodine in the argillites causes a significant reduction in its associated impact by a factor of approximately 50. This also shifts the maximum dose accordingly, which then appears at dates in the neighbourhood of 10 million years.

This function analysis shows that the Callovo-Oxfordian is a particularly important component, whose characteristics ensure a good level of safety function performances, even in the event of mediocre operation of other components (defective containers, inefficient seals) or even of degraded properties of the geological medium itself.

From the impact point of view, that of B waste and vitrified waste zones is, in all calculation configurations, very much less than the dose constraint of 0.25 mSv/year (approximately 0.00047 mSv/year for B waste and approximately 0.0008 mSv/year for all C waste). In the current hydrogeological model, that linked to fuels reaches 0.022 mSv/year. In the sensitivity study, these impacts are at most doubled (assuming a conservative geochemistry in the host formation, the concrete and the clay).

6 Uncertainty Management



Block Diagram 6-1 Representation of the sequencing of the various stages of analysis (see Block Diagram 1-1) Subject : uncertainty management

Safety assessment and achieving safety aims are inseparable from the identification and handling of uncertainties. The chapter 3 highlighted the design measures for managing uncertainties, and the chapter 5 showed how the performance calculation models included margins of caution. This chapter refers to the material already partly described in the preceding chapters and completes it to present an overview of the main uncertainties regarding the post-closure evolution of the repository. It also describes in detail the procedure used to take these uncertainties into account.

6.1 Accounting for Uncertainties in the Dossier 2005

Before expanding on this, the concept of « uncertainties » must be set in the specific context of a deep geological waste repository.

6.1.1 Typology of Uncertainties

Generally, uncertainty arises from a lack of data or accurate knowledge about one of the elements of the system studied, or from the random nature of some phenomena. In the case of the safety analysis of the deep geological waste repository, this may cover gaps in parameter values, a lack of understanding of certain coupled mechanisms, or the inability to predict the occurrence of certain events.

For as clear as possible an approach to this question, it is important firstly to properly define the field covered. The overall concept of « uncertainties » covers a number of matters. Prior to their identification and treatment, it is necessary to draw up a general typology. The classification of uncertainties forms part of the basis of their management (see for example [81]).

We distinguish the following :

- uncertainties regarding repository project input data, i.e. in this instance the packages' inventory data and characteristics, independently of their behaviour in the repository (e.g. uncertainty about quantities or the radiological and chemical inventory) ;
- uncertainties regarding the intrinsic characteristics of a repository component. These are of several kinds :
  - ✓ they may be linked to inaccuracies in measuring techniques ;
  - ✓ they may also be tied to a number of variables that are not directly accessible to measurement, and for which reference is then made to data available in the bibliography, with an uncertainty concerning the relevance of their application, for example ;
  - ✓ they may be due to the variability of the component in space with regard to a necessarily limited sampling. In particular, in the case of the geological medium, the data used are acquired at different scales and from a limited number of measurements. The information acquired must then be extended to larger spaces or volumes, while managing the changes in scale. This applies to characterizing the rock from samples ;
  - ✓ they may be linked to the model underpinning the definition of a parameter that we seek to characterize. The characteristics of a repository component are only defined within the framework of a given model. Thus, the permeability of the geological medium, a medium whose structure is complex on the microscopic scale, corresponds to an overall property of this medium at the macroscopic scale with regard to water transfers. In some instances, if the model is too simplified to assess the physical reality, the associated variables may only be defined with a margin of error. This is one form of uncertainty regarding models, which links up with uncertainties regarding the processes themselves ;
- uncertainties regarding the processes governing repository evolution. Once the data is acquired concerning all the system components, and the phenomenological representation given in detail, it remains to understand and represent the way in which these various elements interrelate and act on the system's evolution. The complexity of the phenomena does not necessarily provide a detailed understanding of each interaction and necessitates adopting an overall representation of the medium that best describes the operation of the system. Representation by a model is subject to uncertainties since it simplifies a more detailed representation of the phenomena. There is also

some uncertainty here regarding the choices of models. This especially applies to coupled phenomena, which are generally harder to represent. Several kinds of uncertainties come into this category :

- ✓ those due to the necessity to predict long-term behaviour, sometimes up to a million years, based on observations that are generally made over much shorter periods ;
- ✓ those due to the validity limitations of the models or to the existence of several models characterizing the same set of empirical findings. An example of this are the uncertainties regarding the behaviour models of some waste packages, such as vitrified waste. In addition to measurement uncertainty regarding the model's parameters (the initial rate of dissolution, for example), there is uncertainty regarding the very nature of the phenomena governing the dissolution of the vitreous matrix ;

- technological uncertainties. At the feasibility stage, the technological measures to be implemented are not finalized, and there are choices between different solutions that may not have all the same consequences for the repository's long-term safety. Moreover, within the framework of repository management in stages, it is extremely unlikely that, even once the repository has been more fully defined, it should remain operated in the same way throughout its expected duration. It is therefore a matter of taking into account :
  - ✓ uncertainties due to the variability of possible repository operating conditions, either because different techniques exist (e.g. excavation), or because the operating scheme may vary (order and rate of delivery of packages, duration of observation and reversibility phases). This variability is only an uncertainty in respect of safety analysis if it leads to indetermination regarding the initial state of the repository in the post-closure phase. If, for example, the variability of possible excavation techniques or the operating period cause indetermination regarding the extent or nature of the damaged zone, it is important to take it into account ;
  - ✓ those due to limited knowledge about the conditions of implementing a particular technology in an underground context. Since the concepts proposed by Andra use well-known, proven technologies, these uncertainties are very limited and only relate to very specific operations, which do not have their exact equivalent in other industrial sectors ;
- external events. These form a special type of uncertainty regarding repository evolution. In general, a distinction is made between naturally occurring surface phenomena (climatic, tectonic events, etc.) which are, in principle, predictable but often subject to great uncertainty, and events due to human action (intrusion, anthropogenic effects) which are, in most cases, unpredictable after a reasonable lapse of time. These events are considered as to uncertainties, because of the disturbances that they cause. Partly conventional approaches are generally adopted to restrict the scope of the uncertainties to be taken into account. In accordance with the Basic Safety Rules III.2.f [2], it is assumed in particular that future human behaviour will be overall the same as today. However, it is possible to adopt a predictive approach, based on past evolution, for most natural phenomena. Even in this case, uncertainties regarding the far future should be taken into account.

6.1.2 Controlling Uncertainties in the Research Phase

In order to be able to assess these uncertainties, three additional approaches have been used in the research carried out :

- setting up reference knowledge documents [17, 18, 19, 20, 21] for obtaining as complete a view as possible at the time of writing, of the available information on the various components studied (geological medium, materials, waste packages, etc.). The purpose of these reference documents is to draw up a report on the state of available knowledge and correlatively highlight the gaps in this. Thus they help to identify the sources of uncertainty and to focus on the actions to be conducted in order to reduce them where necessary ;

- the implementation of methods as systematic as possible for analysing the system and its evolution. Such was the aim of the phenomenological analysis of repository situations (PARS) [27, 28]. The latter is an approach to the repository system's evolution through time and space. An inventory of the phenomena and associated models is drawn up for each of the repository situations identified in this context. This in return highlights the limits of knowledge or understanding and identifies the corresponding uncertainties. Furthermore, the very determination of repository situations with their breakdown in time and space can underline uncertainties regarding the occurrence of a phenomenon or its exact chronological limit. The systematic work achieved within the framework of the PARS has thus led to make an inventory of a series of uncertainties. Without claiming to be exhaustive, this work helps identify uncertainties in a relatively detailed way ;
- the care brought to the traceability of model and parameter choices, and to defining variation ranges for these. When creating conceptual models representing the broad processes within the repository, the various parameters, models or data used are systematically listed, and are described in terms of uncertainties (from « phenomenological », corresponding to the values best expressing the acquired data and reasonably covering possible variability of the data in space and time, to « conservative » or « pessimistic » taking all the uncertainties into account, see definition in Chapter 5, section 5.2.2).

6.1.3 Uncertainties and Time Scales

Following the operational phase, safety assessment must be conducted over a very long period of time (up to a million years). Uncertainties regarding the behaviour of the repository over such periods are significant. Feedback on the evolution of natural or artificial systems on time scales of hundreds of years is limited to archaeological analogues, or to natural analogues that in turn give access to periods representative of geological time scales. But this does not mean that these uncertainties cannot be mastered with a sufficient degree of confidence. They must be tackled in a very systematic way, their effects analysed and taken into account in assessments.

Uncertainties are not the same from one period to another, nor the components of the repository or its environment that are considered. Thus, by way of example :

- in the near field, i.e. in the immediate environment of the repository structures, uncertainties regarding the behaviour of the materials and the rock are going to decrease over time, when thermal, mechanical and hydrological processes due to disturbance of the repository dwindle or reach equilibrium. However, the time of attaining equilibrium and the exact nature of this equilibrium are subject to uncertainties ;
- uncertainty regarding the surface environment and the surface layers of the geosphere will increase overall, especially when major climatic changes such as periodic glaciations are included in the assessment.

In the particular instance of material behaviour in the broad sense (including rock), it is possible to obtain or produce samples representative of most of the repository components (waste matrix, bentonite, concrete, etc.). Generally, it is also possible to place these samples in experimental conditions representative of those expected in the repository (in terms of pH, Eh, etc.) given the relative homogeneity of the repository's environment. However, laboratory observations are necessarily limited to a few months, or perhaps years, and extrapolation to longer periods requires :

- either an understanding of the mechanisms of material degradation over a short period, which it is possible to extrapolate to the long term, on condition that no new phenomenon, latent in the short term, manifests itself over the period ;
- or transposition to the conditions of the repository of observations made in more unfavourable conditions, accelerating the rate of the phenomena (this is the case, for example, of iron-clay disturbance, too slow to be observed in the laboratory at the repository temperature, but which is observable at high temperature over short periods). This assumes the availability of experimental data for deducing the kinetics of repository phenomena based on that observed in laboratory conditions ;

- or extrapolation of observations made over short periods, under pessimistic environmental conditions, to long periods. This case differs from the previous one in that it is assumed that there is no transposition law between the experimental observations and the reality in the repository, but we are covered by the pessimistic experimental conditions against any possible long-term change in the phenomenon. This is the case, for example, of the « V_0.S » model, studying the alteration of glass over short periods under unfavourable conditions (no silica in the external medium, leaching by pure water) to deduce a conservative value from it for the rate of dissolution in the repository ;
- or by the study of natural cases, as for example for cement-based disturbance or the alteration of bituminous matrices, or archaeological analogues.

6.1.4 Uncertainties and spatial scales

Uncertainties related to spatial scales must be taken into consideration. They cover two types of difficulties that are very different in their nature.

Firstly, some problems are related to the modelling that involves representing phenomena, which on their own are well understood and simple at the microscopic scale, within more complex macroscopic environments (e.g. the evaluation of the permeability throughout the damaged zone from an understanding of its structure). These questions are dealt with by comparing theoretical models with laboratory experiments conducted on the surface or in situ in real environments at various scales. The reverse situation can also occur, when a phenomenon is correctly represented at the macroscopic scale without the mechanisms at the microscopic scale being completely understood. For example, the representation of the sorption by a macroscopic « distribution » coefficient, Kd.

Other problems are related to the use of observations that are necessarily local at this stage but are applied to a much large industrial scale, such as a repository. At the scale of a repository, the designer must expect to encounter geological conditions that show a certain degree of local variability. The design layout provisions, and in the first instance the very choice of a geological formation providing good physico-chemical uniformity in properties over a wide scale, helps to guarantee that the disposal packages environment will vary little from one cell to another, or within a given cell. Some residual variability could however remain and this must be taken into account in the safety analysis.

The safety analysis must take into account changes in temporal and spatial scales. This requires a spatio-temporal division that occurs in all the compartments of the analysis. We have seen this appear in the chapter 3 during the assignment of safety functions. It affects the performance assessment (chapter 5). It is also necessarily the basis of the analysis of uncertainties. The temporal division proposed by the phenomenological analysis of repository situations, itself based on the behaviour of components of disposal (thermal and radioactive decay, degradation of materials, returns to equilibrium), is conducted again in the safety analysis.

6.1.5 Management of risks and uncertainties

6.1.5.1 Methods used in the management of both « risks » and « uncertainties ».

The concepts of uncertainties and risks are clearly distinct in the case of operational safety : the risk concept refers to events occurring at a point in time with a low to moderate probability of occurrence (fire, criticality accident, etc.). Whereas the concept of uncertainty refers to insufficient control over technical parameters (e.g. uncertainties over the range of ventilation settings).

In the case of long-term safety, the two concepts widely overlap. Long-term risks refer to the occurrence of events whose probability is often difficult to quantify (e.g. the probability of an intrusion into a long-term repository) that can be considered to be a form of uncertainty about the future history of the repository. On the other hand, uncertainties about phenomenological knowledge can lead to a risk in the normal meaning of the term (e.g. uncertainty about the properties of the damaged zone could lead to the appearance of a short-circuit of the geological barrier in certain zones. This divergence from the normal situation is similar to a « risk »).

Risks and uncertainties are taken into account in a very similar manner in the Dossier 2005. There is therefore no systematic attempt to always distinguish them. Instead, we talk of « risks and uncertainties » or more simply of « uncertainties », without formally distinguishing them from one another.

6.1.5.2 The principles of uncertainty management

An uncertainty or a risk affects safety if it is likely to call into question the performance assessment of the repository, as described in chapter 5. This questioning can take various forms.

Firstly, the uncertainty can simply concern the quantitative assessment of the impact of the repository or an intermediate safety indicator. For example, it is possible that the radiological inventory of a package is not well known and could be higher than the reference value used. Such an uncertainty, that only involves data used in calculations, can easily be handled by purely numerical techniques — all that has to be done is to conduct a sensitivity study to determine whether this parameter is really important in terms of its impact. If it is, the value of this parameter that is selected must be the one that gives the highest value in the calculation. These evaluations have been described in chapter 5, or directly in the reference calculation, or in the sensitivity analyses.

Other uncertainties can involve the ability of one or more components to fulfil the expected function. This can concern the performance of the function (is the component as effective as envisaged?), the availability (can the function be maintained as long as needed?) or the reliability (does the function depend too much on environmental conditions?). For example, uncertainty about the corrosion rate of containers under storage conditions can have influence on the availability of the containers' sealing function.

A safety function can be affected in several ways :

- Either the performance (or the availability, or the reliability) is degraded, but the function itself is preserved. This would be the case for example of a local variability in the permeability of the host formation, which would not lead to any change in hydrological regime. This type of uncertainty can have an influence on the quantitative results of the safety assessments (in terms of an increase in the calculated impact) but does not lead to any divergent evolution in the repository compared to the situation considered to be nominal.
- Or the performance is degraded to such an extent that it is considered that the function is no longer correctly fulfilled. This would be the case for example of corrosion affecting a container to such an extent that its containment capacity can no longer be counted upon — arbitrarily it is assumed that there is no longer any containment. This type of assessment forms part of the qualitative judgement of the safety engineer, to the extent that there is a continuum of circumstances from the full performance of a function through to its total loss. To be prudent, a function is considered to have been lost as soon as its performance or availability are likely to be significantly different from what was planned by the designer.

The first case of uncertainty is similar to uncertainties of a purely numerical type, to the extent that it can be dealt with by a sensitivity analysis, and involves envisaging the worst-case scenario when calculating the performance, or directly including the uncertainty in the reference calculation in the form of a conservative model or parameter.

In contrast, the second situation is specific to the safety analysis in the strict sense. As soon as a function can be lost, the repository no longer behaves as planned in the reference design. The likelihood of such a situation occurring must therefore be evaluated.

If it appears to be sufficiently likely, then the function itself is unreliable and the repository design must be revised, as part of an iterative process between functional analysis and safety analysis. For example, at this stage of the research programme there is an uncertainty over the retention capability of the glass at temperatures over 50°C. An overpack has therefore been envisaged around the glass, that is capable of maintaining more reliable redundant containment than that of the glass during the period of temperature decrease in the glass.

If on the other hand, the loss of the function appears to be an unlikely occurrence, requiring for example a very rarely observed parameter value, or a sequence of events with a very low probability, the uncertainty in question can be tolerated provided that its consequences can be assessed. Such a situation of the inherently very unlikely loss of a function is qualified as an altered situation. In the framework of a defence-in-depth approach, it is also necessary to take precautions to reduce as far as possible the likelihood of an altered situation occurring, or to reduce the seriousness of its consequences. Although they are centred on controlling normal situations, the designer's actions must also take into account altered situations. For example, the division of the repository by separation seals would help limit the consequences of a bore-hole that accidentally pierced a drift or a module.

There is in fact a continuum of circumstances from the degradation of a function through to its total loss. Even if these circumstances do not all stem in principle from the same method of processing, the safety analysis does not differentiate between them in practice. It takes account of the effect of uncertainties on safety functions, irrespective of whether they force the system out of the domain of normal evolution or not.

The aim of the safety analysis is to carefully identify the importance of each uncertainty in the repository's overall behaviour.

The management of risks and uncertainties finally takes place using various means :

- The first, that should be favoured whenever possible, is risk management, i.e. that they should be taken into account in the very design of the repository, by the definition of design features that make the system robust in the face of the uncertainty in question (see chapter 1, the robustness principle).
- The second method involves including the risk or uncertainty in the « normal evolution » assessment of the repository's performance, i.e. by taking into account the possibility by choosing the worst-case scenario for normal evolution, either in the reference calculation, or in the sensitivity analysis. For example, if there are uncertainties about the containment capabilities provided by a particular waste matrix, it can be assumed in the assessment that release from the packages in question is labile. This choice does not intend to represent reality, but aims at protecting against any eventuality. This of course does not prevent characterisation works from continuing to better identify the behaviour of a given matrix, nor taking measures that would qualitatively favour containment.
- For uncertainties that cannot be dealt with using the two previous methods, the sequence of events leading to an altered situation, that is likely to be caused by the risk in question must be identified and its effects evaluated. We must then ensure that the likelihood of this situation occurring is very low. As it is impossible to individually assess all conceivable altered situations, they are grouped into categories of similar situations. Each category is attached to an SEA that describes a typical altered situation, possibly including variants and/or sensitivity studies, whose effects are similar to those of the situations that they intend to represent, and whose consequences are the most serious. The altered evolution scenario therefore represents a « bounding »[84] scenario of, in principle, diverse altered situations [2]. It thus reflects similar effects, while retaining those with the most serious consequences.

6.1.5.3 The place of uncertainty management in the dossier

In accordance with the general approach, the aim of this chapter is to give an overview of all the uncertainties and risks that could affect the « normal » evolution scenario that has been described and assessed in the previous chapters.

It should be noted that uncertainty management is an integral part of the very definition of the SEN. During the explanation of the choices that determined the safety calculation model (chapter 5), we saw that most of these choices are intended to cover uncertainties. The models adopted are « conservative » either in the reference or sensitivity analysis. Some potentially available margins or reservations are ignored. The SEN therefore constitutes, both in its reference version and after all the sensitivities have been included, a bounding scenario of a set of possible evolutions : the packages

84 See definition in applicable section of 5.1.1

could degrade slower than what has been represented, the releases could be slower, etc. In accordance with the basic safety rule RFS III.2.f, the SEN, if applicable, accompanied by its sensitivity studies, must be the bounding scenario of situations judged to be certain or very probable. It is not a « prediction » of the repository's evolution.

The present chapter on the management of uncertainties could therefore have preceded that on the definition of the SEN. It seemed more convenient to discuss uncertainties once the reference that constitutes the SEN had been described, and when a preliminary ranking of important parameters had been made, so as to :

- More clearly present to readers which uncertainties are taken into account in the normal situation, possibly identify those that are not, and explain why.
- Judge, especially by means of the sensitivity analyses already conducted, the importance of the various parameters and models, so as to provide a better argued assessment on the ranking of uncertainties.
- Serve as an introduction to the definition of altered situations that could be envisaged, grouped into SEA, that will be dealt with in chapter 7.

A detailed inventory of uncertainties is produced by means of qualitative safety analysis [33]. This is used to demonstrate the effects of uncertainties on the safety functions of the repository's various components, and more generally on the repository's evolution.

A brief reminder of its methodology is given in the following section. The main results are then described. As the uncertainties are placed next to the expected functions and to normal evolution, this chapter provides a link with chapters 3 and 5 of the present document. It introduces the definition of the altered evolution scenario.

6.1.5.4 Methodology of qualitative safety analysis

The qualitative safety analysis (QSA) method, a tool for identifying and managing uncertainties, is described in a specific document [33]. Only a reminder of the aims and of the outline of the method is given here.

The aims of qualitative safety analysis, which was conducted in 2004 once the knowledge references had been established, are principally to produce an inventory of the risks and uncertainties.

All of the risks of external origin, i.e. those whose cause is a factor external to the repository (the risks of human intrusion, earthquakes, etc.) are identified on the basis of an inventory of possible interactions between the repository and the surrounding environment [82]. This examination is conducted using the standard method of the external functional analysis, and leads to the identification of a list of possible events, that are in principle unfavourable for the repository's safety. The results of the analysis are also compared with risks that basic safety rule RFS.III.2.f recommends should be taken into account, or which are normally considered in the approaches used in other countries and those in international databases [83, 84].

At the same time the analysis studies the uncertainties in the true sense of the term, on the basis of the knowledge references, the phenomenological analysis of disposal situations (PARS) and the description of the conceptual models ; these have been written by various experts, aware of the need to systematically identify the uncertainties. The PARS approach, and formal framework in which the conceptual models are established (with, in particular, the need to define the intervals of variation for the parameters) help make the identification of uncertainties more systematic. The analysis consists of comparing the state of knowledge with the expected safety functions. Some uncertainties can in fact have a direct influence on the confidence that can be had in a given safety function. For example, if the uncertainty about the permeability of the host formation is too great, this could call into question the performance of the function « prevent water circulation ».

Uncertainty is the subject of a systematic study that identifies :

- which component is concerned by this uncertainty, with if relevant the effects caused by one component on another by means of a perturbation ;
- which performance aspects of which safety function can become altered. A qualitative, but argued assessment, including the use of special calculations if relevant, is conducted on the risk of a significant reduction in the expected performances ;
- if applicable, and if such information is useful, the time period involved.

The first objective is to identify whether the uncertainties are correctly covered by the SEN, either in its reference version, or in the sensitivity studies considered. If some of the uncertainties are not, it must be confirmed that they would have little impact on the repository, or that they refer to very unlikely situations.

As a second stage, if the uncertainty is not covered by the SEN, the function(s) and component(s) that could be affected must be identified. A systematic component-by-component analysis is used in particular to identify the shared causes of the loss of several functions : for example, an incorrect assessment of the long-term behaviour of a material can affect all the components that contain it, even though these could have different functions. The qualitative safety analysis provides an assessment of the degree of independence of safety functions, by identifying the possible uncertainties affecting several functions.

The effect of taking each uncertainty into account is described (i.e. the behaviour of the repository if the worst-case value of the parameter in question was the actual value, or if the risk envisaged actually occurred), in terms of the repository's evolution. This is done on the basis of the functions that are likely to be lost. For example, if a series of uncertainties can call into question the function « regulate the pH in the vitrified wastes cells », the corresponding situation is described, i.e. the effects of an uncontrolled increase in pH. If the design can cancel this effect, or if this is taken into account in the SEN or in its sensitivity calculations, the analysis stops at this stage. If a safety function can be affected and the evolution of the repository could start to diverge from normal, with a possible impact on other components, this effect is then specifically identified.

A qualitative assessment of the likelihood of these effects is also made (is the risk likely?). This assessment can be based on quantified factors – for example, if a situation relies on worst-case values for parameters, each of which are rarely or never measured in the laboratory, it can be objectively qualified as being very unlikely. This type of assessment sometimes partly relies on a more qualitative reasoning, making reference for example to analogies with natural cases. Because of the time scales being considered, no attempt is made to quantify a precise probability for events of an uncertain nature, as this is too subject to error.

In addition to taking the uncertainties into account one-by-one, an attempt is also made to determine whether their combinations could lead to effects that deviate significantly from normal evolution. For example, it would seem likely that a series of uncertainties about the efficiency of a hydraulic cutoff at a seal and about the characteristics of the damaged zone that it intercepts could lead, when considered together, to significantly more unfavourable performances than would result when each uncertainty was considered individually. In certain cases combinations of uncertainties are excluded on the basis of a principle similar to that of « no double-failure », applied in normal nuclear installations. Combinations of failures with independent causes and occurring over the same limited time period are excluded on this basis. Combinations of failures arising from cumulative design errors were not considered for this reason. For example, although the possibility that one of the containers was badly made, or that the seals were poorly designed is envisaged, the possibility of the combination of these two malfunctions occurring simultaneously is not envisaged, provided of course that the analysis does not reveal a shared cause that could lead to the simultaneous loss of function of these two components (which could be the case, for example, if they were made of the same material, whose evolution was uncertain).

The analysis provides not only confirmation that all the sensitivity calculations for the SEN are appropriate, but also provides a model of altered situations that are judged to be possible, even if they must, in any event, be improbable to merit the term « altered ». Events that could occur in the situation envisaged are compared with those of each SEA, to check that the SEAs correctly « covers » the situation in one of their calculation cases. The bounding nature of these scenarios, with respect to current knowledge and the degree of development reached by the safety analysis, is therefore verified.

Four SEA have been adopted by Andra : the seals failure scenario, the package failure scenario, the bore-hole scenario and the very degraded functioning scenario. This choice is based on a prior selection of major types of situations to be dealt with, according to the logic described in the introduction to the chapter 7. This selection was therefore made before the qualitative safety analysis was conducted. This selection, by attaching the possible failure situations to one or other of these scenarios, and by defining the calculation case and the variants to be dealt with, has validated the relevance of the SEA that have been adopted.

Insert 7 Illustrative example of the application of the qualitative safety analysis methodology

To illustrate the method, an example is given below – simplified for the requirements of the description – of the way in which the qualitative analysis of the behaviour of a particular component is conducted, in this case the C-cell plug.

First the limits of the component to be covered are described : in the framework of the qualitative safety analysis, the « C-wastes cell plug » refers to the assembly consisting of the bentonite plug and the EDZ that can surround it.

The safety functions expected of this component are listed. On the basis of the functional analysis, the plug is identified as contributing to « limiting the release of radionuclides and immobilising them in the repository », by preserving favourable physico-chemical conditions in the cell : maintaining a diffusive regime, limiting the pH by preventing the progress of the alkaline plume coming from concrete components outside the cell. It also helps prevent any colloids forming inside the cell from migrating toward the drift. In addition it also slows and reduces the flow of radionuclides coming out of the cell, although its contribution to this function is negligible compared to that of the host formation.

A check is also made that this component cannot have a perturbing role on its environment. In this case, a mechanical action could be possible on the host formation if the plug were to swell excessively.

The plug's characteristics are then related to the safety functions to be fulfilled : permeability (for diffusive conditions), coefficient of diffusion and buffering capacity (for the pH), porosity (for the colloids), sorption capacities (slowing and attenuating). The plug's mechanical dimensioning also has to be considered.

These characteristics are reviewed and the uncertainties are systematically identified by examining the research results. For example, uncertainties could occur about the permeability throughout the plug, related to uncertainties of that of the EDZ that surrounds it.

As a second stage, the component's environment is examined : this consists of the concrete retaining plug, the metal operating plug and the contents of the cell (the wastes, the overpack and the lining). The thermal, hydraulic, mechanical chemical and radiological actions that these components can exert on the plug's characteristics are identified on the basis of the PARS. For example, the heat given off by the wastes can alter the permeability, the diffusion coefficient and the adsorption equilibria, etc. The gas derived from the corrosion of the cell's metal components can act on the plug, etc. A study is made on whether these phenomena, and the uncertainties to which they are subjected, can have an effect on the safety functions.

The analysis is finally completed by studying external events that could have an effect on the component. For example : is an earthquake likely to affect the plug's functions ?

The systematic identification of uncertainties therefore relies on this method that allows the environment of each component and the interactions of all types to be taken into account, in function of the state of concept development and knowledge acquisition.

This analysis therefore provides an inventory of uncertainties and enables a first ranking to be made, guided by the manner in which they could affect the safety functions. In conducting this ranking, account is taken in particular of the design provisions that exist for managing uncertainty. For example, the plug length can be adapted in relation to its capacity to buffer the adverse alkaline phenomena. This factor is taken into account for ranking this uncertainty by comparing it to possibly less controllable ones.

The analysis having dealt with the uncertainties on all the components, then identifies possible shared causes. For example, if the EDZ level with the plugs is identified as having a greater permeability than is expected, the same uncertainty would in principle affect the plugs of the spent fuel cells, or even the seals in the drifts. A check must therefore be made that such effects are well covered in the SEN, in its reference version or if applicable by sensitivity studies. If this is not the case, they must be covered by a SEA, provided that such effects appear to be sufficiently unlikely.

6.1.5.5 Verification of the qualitative safety analysis

The qualitative safety analysis was conducted by Andra engineers who were not involved in writing the scientific documents. In this way, the safety analysis is given a certain degree of independence, since the people in charge of analysing the uncertainties and the possible altered situations (the safety engineers) are not the same as those who established the phenomenological plan for normal evolution.

Andra did however want to back-up the qualitative safety analysis by comparing the results with analyses conducted internationally. This was one of the important recommendations derived from the peer review conducted by the OECD/ENA [15]. To do this, the Agency relied on the « features, events and processes » databases available internationally, in particular the FEP 2000 database of the OECD/NEA [84] and FEPCAT [83]. These databases were themselves supplemented by comparison with the OPALINUS database of NAGRA [85]. The Swiss safety analyses are in fact conducted in a geological environment similar to the Meuse / Haute-Marne context, and on repository concepts that are sufficiently similar for a comparison to be possible.

The FEPs databases list « features, events and processes » that are in principle important for safety analysis, which is a different approach from that of qualitative safety analysis which studies the uncertainties relating to these same features, processes and events. The qualitative safety analysis emphasizes the uncertainties, component-by-component and by function approach ; a FEP can therefore appear in several parties of the qualitative safety analysis. Establishing a link between each FEP and each part of the analysis requires going into detail of the qualitative safety analysis arguments, but did prove possible in practice, and useful for verifying and clarifying the qualitative safety analysis.

Furthermore, the FEPs are intended to cover all of the phenomenology that could be found in different safety analyses, conducted in different geological contexts, and some require to be adapted to become applicable to the Dossier 2005. This adaptation could be done without major difficulties, only a few FEPs concerning phenomena that could not occur in the particular context of the Meuse / Haute-Marne site could be identified in the databases, and were not included in the qualitative analysis.

The comparison between the FEPs databases and Andra's own analyses was an important exercise for the qualitative safety analysis, and provided supplementary information on several aspects, to finally end with consistency between the approaches. It proved to be very useful to safety engineers in ensuring that no fundamental characteristic of the components and no phenomenological process likely to have an influence on the repository had been forgotten. Apart from this aim of completeness, the comparison facilitated dialogue between engineers contributing to the safety analysis and engineers contributing to the development of scientific documents. Thanks to the FEPs, the safety engineers were provided with an integrated database offering them a different framework for the documents produced by the Scientific Directorate. The dialogue thus established enabled arguments to be developed and helped avoid errors of interpretation.

The analysis also led to a few FEPs being ignored, either because they were irrelevant or because they were premature in relation to progress on the project. FEPs relating to the presence of buffers in the C waste cells or FEPs relating to the conditioning of the waste are examples of this. In the scope of the safety analysis, it was considered that the waste complied with the descriptions supplied by the producers. This point was checked by Andra as part of the monitoring action that it conducts on HLLL packages. The comparison reached the conclusion that the qualitative safety analysis complied with respect to the reference constituted by the FEPs. This is a factor adding credibility to the Dossier 2005. The result of this comparison can be found in document [86].

6.2 Main results of the inventory of uncertainties

This section describes the results of the qualitative safety analysis. It lists the main uncertainties discovered during the analysis. They are briefly described and their possible effects and ways of being taking into account in the safety analysis are identified.

This chapter does not aim to describe all of the repository's phenomenological evolution, nor even to present the current state of scientific knowledge acquired up to now on the repository's evolution. Readers who wish further information on these aspects are invited to refer to the section on the « phenomenological evolution of the geological repository » [7] and to the knowledge reference documents, [17], [18], [19], [20], [21]. All that is presented here are the characteristics of the components, the physico-chemical phenomena and external events, which are subject to significant uncertainties and can influence the repository's normal evolution. The scientific results are only mentioned in order to make the text more understandable for readers. They are therefore only partially described.

Section 6.2 presents the main uncertainties, and describes how they are covered by the design, or how they are taken into account in the normal evolution scenario, whose results have been presented in the previous chapter. This section therefore helps in more completely defining the « normal evolution domain » that was only sketched out previously.

The identification of uncertainties is done according to the classification presented in section 6.1.1. To make the report clearer we have chosen a slightly different presentation, using a subject-based approach. The following will be described in succession :

- Uncertainties about physical knowledge of the environment : host formation, surrounding formations and surface environment. A distinction is made between those due to the natural variability in the properties of the geological environment at the scale of the repository's perimeter, those which are due to the limits of the characterisation programme, and those related to the extrapolation to long time scales (long-term geomorphological evolution).
- Uncertainties over the inventory and knowledge of waste packages, which are classified by major themes : inventory, radiological content and chemical content.
- Uncertainties about phenomena governing the repository's internal evolution, making a distinction between those that mostly refer to hydraulic, mechanical, chemical and radiological processes and those affecting the transport of radionuclides. The temperature, for which its influence on other phenomena is the main aspect, is dealt with last. Couplings between uncertainties are dealt with at the stage where this seems most natural.
- So-called « technological » uncertainties, i.e. those related to the incorrect use of techniques used to build or operate the repository (risk of poor quality control, risk of forgetting a critical procedure) or related to the effects of various building and operating techniques on the repository's evolution in the post-closure phase. In a separate section the problems related to the duration and methods used during the operating and observation phase are distinguished, to the extent that they could have an influence on safety in the post-closure phase.
- External risks, i.e. the possibility of earthquakes, or major climatic events, or the risks of human intrusion if the repository's whereabouts is ever lost from memory.

Each time the aim is to make an assessment of the potential criticality of each uncertainty, relying on the methods that that have been developed for warning of the effects (design provisions already widely discussed in chapter 3, past or current research programmes) and on its potential effect on the repository's evolution. These uncertainties are then ranked in order of criticality, giving priority to those that could jeopardise safety functions. The results of the calculations presented in chapter 5 have already enabled us to rank the major characteristics, especially of the geological environment, which have the highest potential impact on the repository. These results provide a useful scale for in turn classifying the uncertainties ; this is one the reasons for which the results of the SEN were given before the present chapter.

The list of uncertainties is derived from the qualitative safety analysis and presents those that have been chosen as being to various extents potentially important. Some uncertainties were excluded during the qualitative analysis, as they were judged to be of little quantitative or qualitative importance, or with no or very little effect on safety, in a normal or altered situation. They are not necessarily mentioned here.

Section 6.3 is a summary of and a perspective on those uncertainties that were identified in section 6.2 as requiring treatment by an SEA ; it serves as an introduction to chapter 7, which presents the quantification of the SEA.

6.2.1 Uncertainties about knowledge of the geological environment : the host formation

The Callovo-Oxfordian formation has been accessible to direct observation in the underground laboratory since 2004. All the measurements made previously had been done by a major programme of sampling by coring or by bore-hole measurements (see Figure 6.2-1).

Many precautions have been taken to minimise the possible bias in these measurements, such as mechanical containment of the sample and packaging in an inert atmosphere to prevent oxidation. An examination of the measurement protocols is conducted systematically by Andra, in order to reject experimental results that could have been too influenced by the sampling conditions.

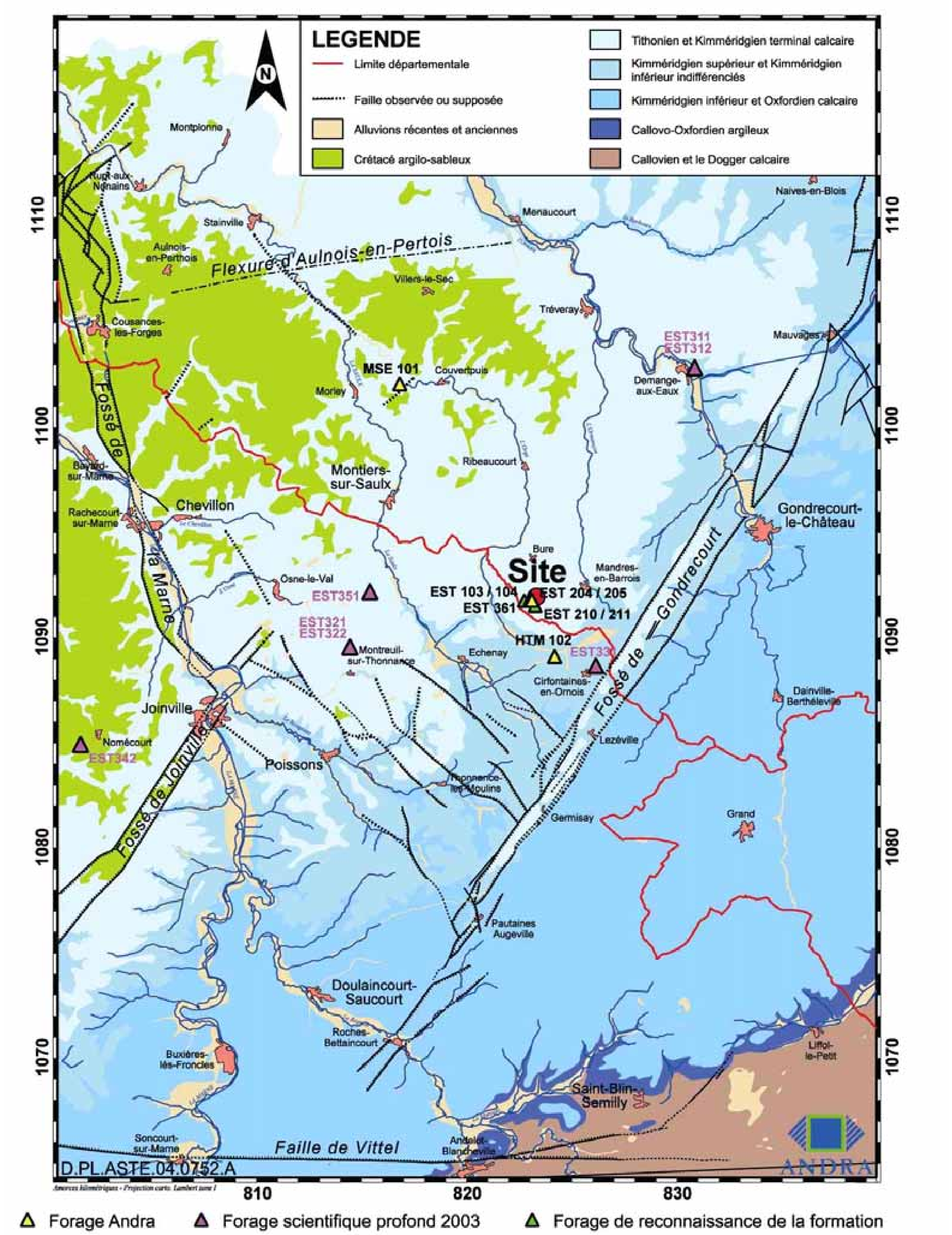


Figure 6.2-1 The Meuse / Haute-Marne site – Location of the main bore-holes

6.2.1.1 Lateral homogeneity of the formation

An intrinsic limit of the programme of characterising the formation is caused by the bore-hole sampling technique, which poses the question of whether there could locally be zones with either more or less favourable properties than those that have been observed, and which had escaped detection. These undetected zones are only of importance for the safety analysis if they could lead to a degradation of the safety performances expected from the host formation, at overall or local scale. The « expected » variability is covered by representing the host formation as a homogeneous and uniform medium, and by the definition of intervals of variation (definition of a « phenomenological » and a « conservative » value) for all the parameters included in the SEN. When a sufficient number of measurement exists, the « phenomenological » value is used as a reference and the « conservative » value is used in sensitivity analyses ; if not, the conservative value is taken into account (see section 6.2.1.4).

In these conditions, the uncertainty in the representation of the host formation as a homogeneous medium refers to the possible occurrence of unusual zones, deviating more or less strongly from the average properties of the Callovo-Oxfordian, and consisting of structures (fractures, fissures, etc.) that could have formed within the rock, but which until now have escaped detection.

At the current stage of the research programme (bore-hole surveys in the Meuse / Haute-Marne sector, 2D and 3D seismic surveys, reconnaissance bore-holes), no communicating fracture or fissure has been detected on the laboratory site. Furthermore, the deviated bore-holes conducted in 2004 showed any evidence for the existence of microfissures with geometries and volumes corresponding to early diagenetic phenomena. Furthermore, the hydraulic tests conducted vertical to these structures showed no evidence for any variation in permeability. Diaclases (micro-structures without any displacement of the crack edges) can be present in the host formation but are in general filled and do not affect either the geometry of the stratum, or its permeability [17]. Basic safety rule RFS.III.2.f, which deals with generic sites, mentions the possibility of lenses of sand occurring. Reconstitution of the conditions under which the Callovo-Oxfordian was formed make these very unlikely and they have never been observed.

It is nevertheless possible that, during the construction of a repository, a structure could escape detection by surface investigations, provided its properties did not differ greatly. In particular, its permeability would have to be low. During the environmental survey, as the repository construction advances, any structures cut into or simply detected could be managed by adapting the architecture if needed thanks to the repository's modular nature [33]. Depending on the size of the structure in question, whose hydraulic characteristics would be evaluated in situ, a choice could be made on whether or not to continue the tunnelling, and there would still be the option of sealing the structure if needed. The survey would not stop at the structures excavated, but if necessary would cover the proximity of the cells. The risk is therefore covered by the design.

The purpose of the qualitative safety analysis is to identify what type of structure could cause all these provisions to fail. The possibility that a measuring error could lead to an underestimate of the size of a fracture detected in advance and that no appropriate measures were taken to seal it has been excluded. In fact, the presence of a communicating structure in this formation, when none has ever been identified up till now, would constitute a rare event in the construction of the repository. It would be the subject of sustained attention by both the repository operator and the regulatory authorities. The possibility that its properties would be insufficiently characterised, seems to be very highly unlikely in such a setting.

The qualitative analysis therefore focused on structures that could escape the detection techniques currently used, both at the surface and at the bottom. It concluded that it could only be a minor structure of limited extent and throw. To be undetected from the surface, with the best 3D seismic techniques available, any possible structure would have to have a throw of less than 2 metres. To be unseen from underground it would have in addition to be at a great distance from the cell. It is estimated that there would be more than a metre of sound buffer argillite between this structure and the nearest cell.

Such a structure could have effects in terms of transport : it could constitute a favoured channel for the transfer of any radionuclides that reached it. In addition to a higher transmissivity and a lower retention capacity, it could also constitute a favoured channel, locally adversely affecting the function « delay and attenuate the migration of radionuclides ». Because it would have to have a sufficiently poor hydraulic conductivity to escape detection, it could not have a major hydraulic influence on the repository's evolution.

The risk of an undetected heterogeneity occurring near the repository would become evident by a locally increased permeability. Such an occurrence is indirectly envisaged in the sensitivity studies of the normal evolution scenario, which downgraded the permeability of the entire formation by a factor of ten, and reached the conclusion that such a change only had a slight influence on the overall hydraulics of the repository. In chapter 7 (in the « highly degraded operating scenario ») an even more worst-case calculation is made, envisaging a uniform permeability of 10^{-12} m/s. In such a case, where the calculation envisages a uniform degradation of the rock's permeability, there are potentially more serious consequences than in the case of local degradation.

Finally, it is noted that the purely hypothetical presence of an undetected structure, whose influence on the circulation in the Callovo-Oxfordian would be great, is covered by an SEA. In accordance with the basic safety rule RFS.III.2.f, the list of SEA includes a situation in which the geological barrier is crossed by an intrusive bore-hole. This has a very high permeability ; it is much more penalising than a structure, which if is undetected, would probably be partly filled, of small extent and at some distance from built structures.

6.2.1.2 Geochemical Characteristics of the Formation

Because the argillites contain only a small amount of water and have small pores, the composition of the pore water cannot be determined by direct analysis of the water, which can only be obtained from the core samples by pumping or leaching. The composition of this water is determined by modelling using a methodology tested at Mont Terri. This method makes it possible to constrain a certain number of the parameters of thermodynamic equilibrium, based on observations and measurements of the core samples (chlorides, CO_2 pressure, ion exchange capacity). This methodology largely relies on identifying elements and chemical compounds of the system (including the mineral phases). The number of measurements taken at regular intervals on core samples has given an understanding of the vertical variation of these properties in the argillite which affect the model of the composition of the pore water (water content, content and total concentration of chlorides, the mineral phases present and their morphological characteristics, the content of ion exchange sites, the types of sorbed cations, the activity of the compound H_2CO_3). The residual uncertainty essentially concerns the concentration in chlorides, where the method of leaching applied to the core samples can be affected by the drilling mud. Nevertheless, the measurement data from different bore holes shows very good agreement : the minimal and maximal values differ by only a factor of three.

The composition of the pore water is sensitive to possible mineralogical variations which can affect a parameter. The Callovo-Oxfordian has thus been divided vertically into three sub-units based on a detailed study of the mineralogy of the argillites, notably the interstratified illite-smectite ($R0_A$, $R0_B$ and R1) (see Figure 6.2-2). Considering the similarity in the mineral phases between the facies $R0_A$ and $R0_B$, the model of the composition of the pore waters deals only with a single facies, R0. A complete characterisation has been carried out using representative core samples of facies $R0_A$ and $R0_B$. The composition of a core sample of facies $R0_B$ has been calculated. As regards the facies R1, the effect of the change in the nature of the interstratified layer (R0 to R1) should only have a small influence, since the other hypotheses relating to Eh, pH, and exchange constants apply to this facies. Initial tests made on the site confirm this view.

(Profondeurs à l'aplomb du puits d'accès du laboratoire - Données géol. EST204 / 205 et 211 proj. - Minéralogie d'après EST207)

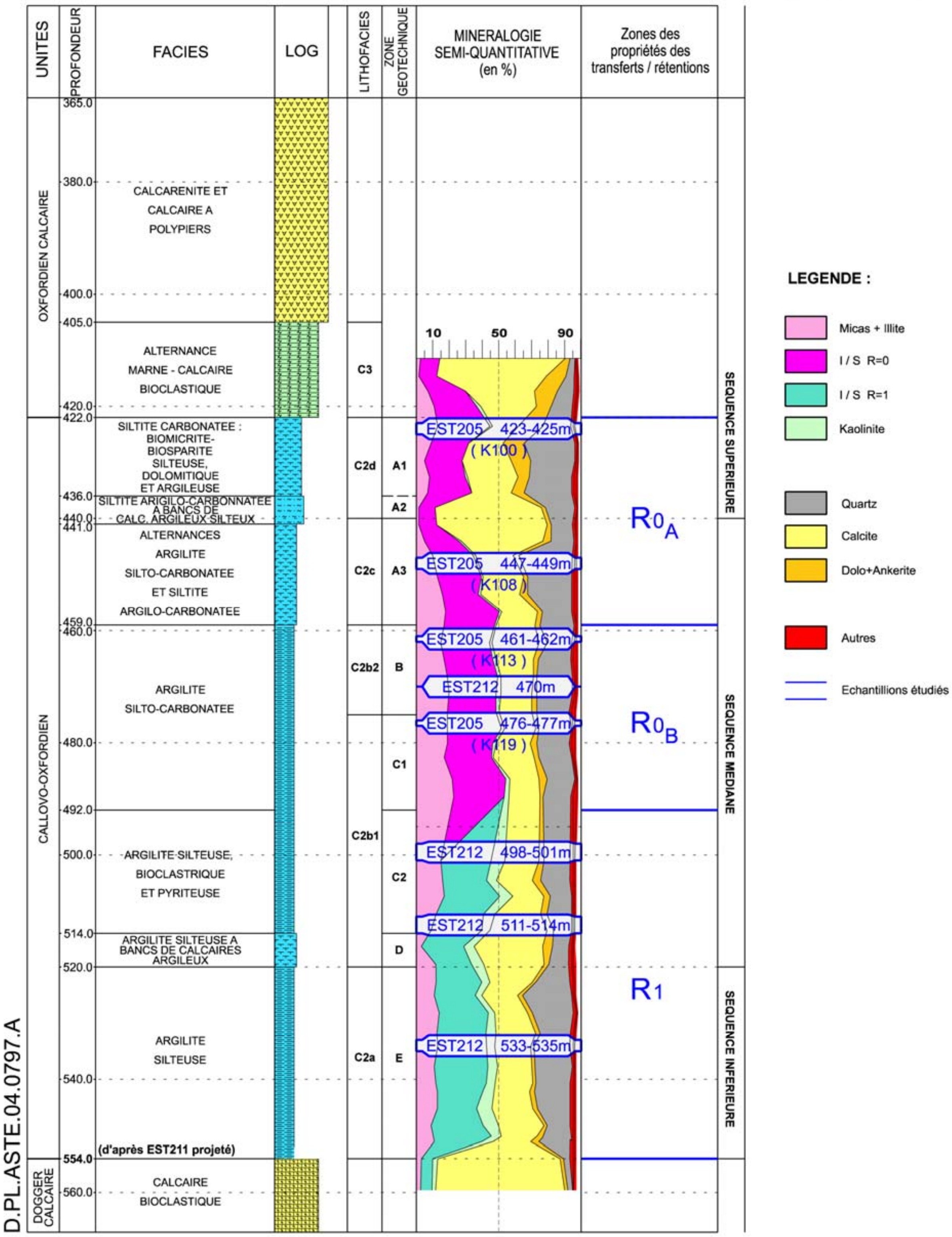


Figure 6.2-2 Vertical variation in Callovo-Oxfordian lithology and mineralology

The mineralogical data gathered from the set of bore-holes indicates that the phases considered in the thermodynamic models are present and unaltered throughout the transposition zone. Apart from the chloride content and the ionic strength, which may vary significantly within the transposition zone the main characteristics of the composition of the pore waters contained in zones R0 and R1 are expected to remain similar. As regards the parameters useful to the project, which are in order of priority, pH, the Eh potential, the ionic strength, the main cations (Na^+, K^+, Ca^{2+}, et Mg^{2+}), content of chlorides (for corrosion), the compounds of sulphur (particularly the sulfates, as concerns concrete), the compounds of dissolved inorganic carbon (HCO_3^-), dissolved iron and silica, ' extreme values[85] ' have been evaluated [87]. Taken together, they do not correspond to an equilibrium state that would be chemically possible, but they cover the range of uncertainty.

The goal of the « PAC experiments » (sampling for chemical and isotopic analysis) currently in progress in the underground Laboratory consists in collecting pore water from the natural flow in a bore-hole drilled with the least disturbance possible. This technique, tested at Mont Terri, should make possible direct measurement in the bore-hole of parameters sensitive to surrounding conditions (in particular to the partial pressure of CO_2) and the measurement the other parameters, such as pH and Eh on line and in a closed system. These fluids were collected using procedures which limited any contaminating factors, and were analysed. These analyses were then compared to the compositions determined by modelling. A second experiment is also being carried in the scope of PAC with the goal of putting a solution into equilibrium with the rock.

The results acquired at the level of the experimental drift at -445m corroborate the chemical composition of the water obtained by modelling. They already confirm the hypotheses adopted in the models regarding reactions governing the chemical composition of pore water and the main parameters of the model, especially neutral pH and reducing conditions. These *in situ* results also allow certain data, such as the exchange coefficient between sodium and potassium, to be specified.

The uncertainties concerning the geochemical characteristics of the formation are thus taken into consideration in the design studies, for the ions of interest, as a function of the 'extreme values' : the "extreme value" of the chloride concentration is taken into consideration in studies of corrosion, the "extreme value" of sulfate when studying the resistance of concrete, etc..

6.2.1.3 The Hydraulic Gradient of the Formation

The hydraulic load gradient within the Callovo-Oxfordian is determined by measuring the loads in the surrounding rock and the host formation. The measurements made on the site reveal a relatively weak load gradient between the over- and underlying formations (overall lying between –0.1 m/m and 0.3 m/m in the transposition zone). The measurements also uniformly reveal an overpressure in the Callovo-Oxfordian relative to the formations surrounding it (on the order of 20 meters water column equivalent) (see Figure 6.2-3).

[85] These are considered 'extreme' since the correspond to an « extreme value » – the minimum or the maximum as the case may be – of the possible interval of variation, and are not considered 'conservative', since in the absence of reference to a given phenomenon, it is not possible to determine if these values are in fact unfavourable from the point of view of safety. A 'maximal extreme' value of the concentration of chlorides is thus 'conservative' with respect to corrosion, for example.

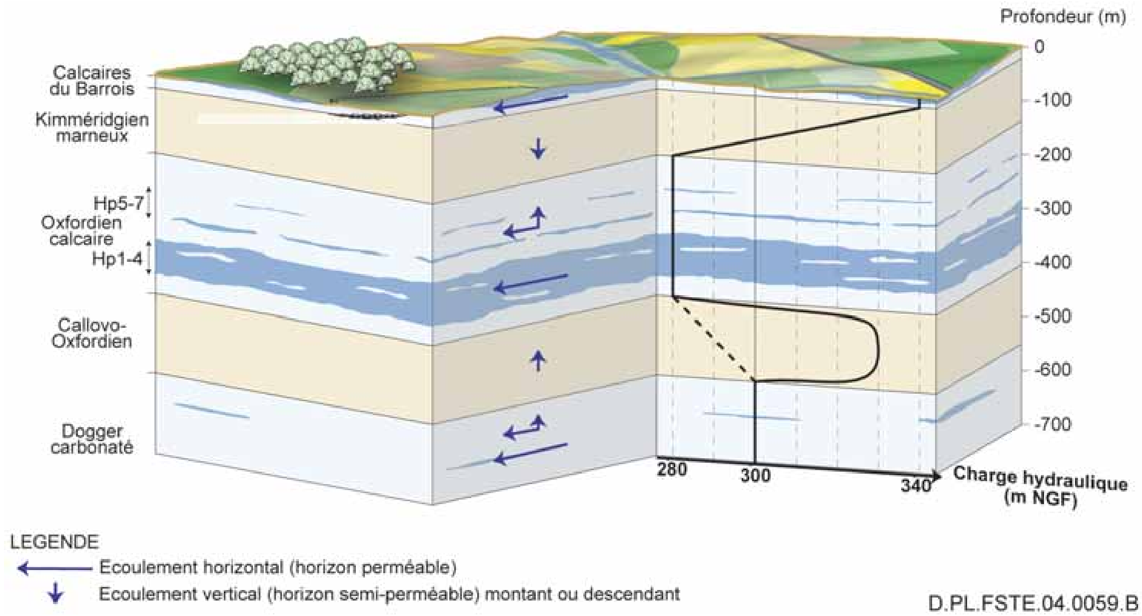


Figure 6.2-3 Schematic representation of the hydraulic load in the different layers

Such overpressures are frequently observed in clay-bearing formations. They may be explained in several ways, depending on the context [17]. A priori, the principle causes suggested in a formation such as the Callovo-Oxfordian are :

- a cause related to severe tectonic constraint because of the water has not had time to adjust to and come into equilibrium with the framework of the rock ;
- a transient change in the hydraulic loads in the surrounding formations which has not been brought into equilibrium by the flow of water within the rock ;
- the remnant of a mechanical load which was larger in the past, due to the weight of a sedimentary column or of ice, which has eroded or disappeared, and which the rock still feels some effect from.
- an artifact of the measurements, related, for example, to creep in the rock exerting an additional mechanical load, or any other error in the interpretation of results ;
- a cause related to the effects of the semi-permeable membrane of the argillites, which causes a reduction in the chemical activity of the water as a result of the reduced mobility of certain solutes. This phenomenon can be globally interpreted on the macroscopic scale as an osmotic phenomenon.

The Callovo-Oxfordian has never been covered by a glacier, which excludes a drastic change in its mechanical loading, and thus a change in it's hydraulic load as a result of the weight of such a covering.

Other phenomena can be discarded on the basis of their typical times of action with regard to that necessary to induce an overpressure, whether real or an artifact of measurement. Thus, the phenomena of erosion are too slow, with respect to the typical time for hydraulic loads to come into equilibrium, to result in the sort of overpressure measured. Similarly, on the typical time scale of measurement, a measurement of the creep of the rock shows that this cannot induce such pressures. An artifact of measurement can thus not explain the size of the measured overpressure.

The hypothesis of the persistence, within the rock, of a tectonic force which the pore water has not adapted requires a more detailed evaluation. On this hypothesis, under the continual loading of the solid skeleton of the rock, the pore water itself could not dissipate this load : while the rock can reorganize itself to dissipate the surplus load, the water cannot reduce its pressure without flowing from the rock. The estimated deformation rates, on the order of 10^{-17} s^{-1}, are negligible in terms of a source of overpressure.

In any case, the hypothesis of a persisting large tectonic force does not apply. In fact, since the typical time for the relaxation of a pressure disturbance in the formation is on the order of 10,000 to 100,000 years (the hydraulic diffusion rate being on the order of 10^{-8} m^2/s), it would be necessary to invoke a recent tectonic event to explain this phenomenon. However, no such event has taken place in the region since the Alpine uplift (23 million years ago). Such an overpressure in the rock cannot be explained entirely, nor even in large part, by a tectonic event so far in the past.

Only the phenomenon of osmosis seems adequate to explain the amplitude of the measured overpressures. A large number of studies have called on osmosis and the semi-permeable behaviour of clay-bearing media to explain the overpressures observed in the geological medium. At first sight, the overpressure might result from disequilibrium in the composition of solutes in the free water contained in the pores of the host formation, which would play the role of a semi-permeable membrane with respect to the surrounding formations. Nevertheless, a model of osmosis which accounts only for the quantities of solutes in the pore water (free water) predicts an osmotic overpressure which is very weak compared to that measured.

However, there are other processes and other families of solutes which could maintain concentrations of solutes in the Callovo-Oxfordian which are higher than in the surrounding formations [88]. In fact, the different studies carried out on the Callovo-Oxfordian pore water show that the composition of this solution (and so, ultimately, the activity of the pore water in the argillites) is determined by :

- compensation for an excess (or deficit) of electronic charge in the clayey phases by ions of opposite charge ;
- the solubility equilibria imposed by the local mineral suite ;
- the constant flux of mineral compounds whose content is not determined by the fluid-rock interactions.[87]

The various evaluations carried out [88] show that only a fraction of the compensating ions (those which are not directly adhered to the surface of the clays), on the order of 5 to 15 % of the total of these ions) does have an effect on the activity of the free water and thus on the interstitial pressures resulting from osmotic phenomena.

The total concentration of solutes in the pore water, if we include these compensating ions, thus attains a much higher level than that of the free water in the argillites. If we consider that 15 % of the compensating ions have an effect on the activity of the free water, the total activity of the water then seems much lower in the argillites than in the two surrounding formations. The osmotic overpressure profile calculated on this basis is in agreement with that measured.

Following this explanation, this overpressure became established very early in the history of the layer : from the time that the layer acquired the characteristics observed at the present time, that is, around 100 million years ago. Since then, the overpressure profile changed only slightly, along with chemical changes in the water in the surrounding formations, until around 30 million years ago. In the absence of major changes in the system it has remained very stable since then.

To the extent that the flow regime is fixed within the Callovo-Oxfordian, osmotic phenomena will not affect the flows since the sum of the chemical and mechanical potentials created by osmosis within the Callovo-Oxfordian is null. Perturbations induced by excavation of a waste repository, and following this by the different phenomena of re-equilibration (transient heat transfer, chemical and mechanical phenomena) do not induce major changes in the solutes and will thus not disturb that part of the equilibrium due to osmosis. In all cases, as a result, the flow regime remains uniquely determined by the hydrostatic load fields imposed by the surrounding formations. This flow can thus be represented by an application of Darcy’s law, based solely on the hydrostatic boundary conditions in the surrounding formations.

The hydrostatic load in the Oxfordian varies measurably over the transposition zone, but it is well defined and has been measured at a number of bore-holes. The charge within the Dogger has not been measured as thoroughly, but it shows little variation.

No source of uncertainty concerning the value of the gradient is sufficiently great to cause it to be called into question. In order to cover these uncertainties, we include in the set of evaluations a value which corresponds, at every point of the SEN model, to a range of gradients taken from the strongest estimates over the transposition zone, amounting to 0.2 m/m in the model for the current time (see Figure 6.2-4) and to 0.4 m/m when taking account of a perspective of a million years for the ascending transfer model, and -0,2m/m for the descending one.

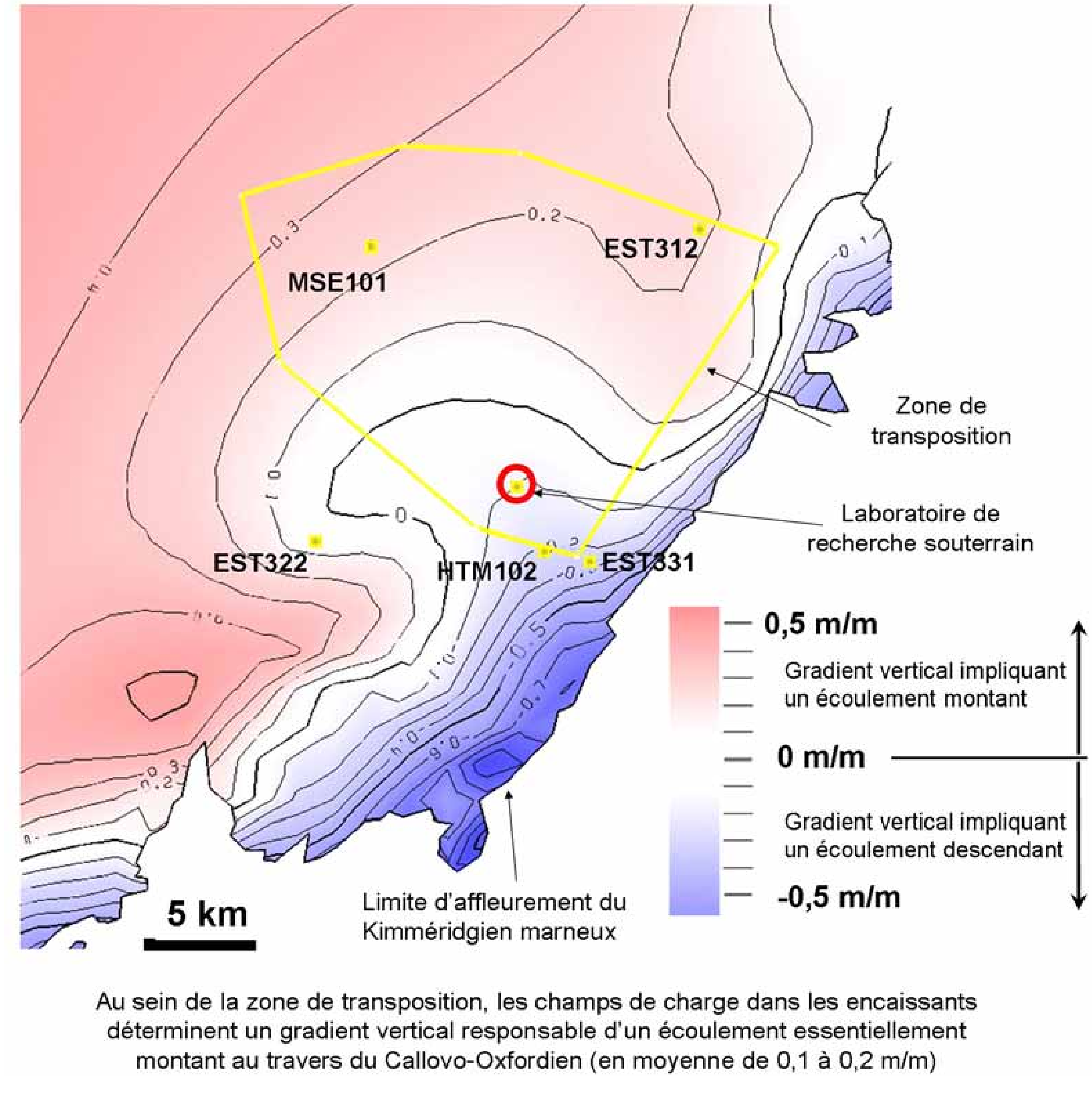


Figure 6.2-4 Evaluation of the vertical hydrostatic load in the Callovo-Oxfordian, derived from calculations based on the hydrostatic loads in the surrounding formations.

6.2.1.4 Transfer and hydraulic characteristics of the formation

- **Permeability**

Measurements of permeability fall within the range running from 10^{-12} to 10^{-15} m/s. Once those values which are of too uncertain a nature as a result of the measurement conditions have been eliminated, and working mainly based on measurements by « EPG probe » or the equivalent, which are the most representative, it has been possible to define with a good level of confidence the ‘phenomenological’ and ‘conservative’ values of the permeability of the Callovo-Oxfordian, showing agreement between the measurements based on core samples and the in situ measurements ($5x10^{-14}$ and $5x10^{-13}$m/s respectively). All measurements of permeability to water taken in core samples indicate that the ratio of anisotropy between the horizontal and vertical permeability cannot exceed 10, and this remains within margin of error in these measurements. The SEN takes this factor of anisotropy into account by increasing the horizontal permeability by a factor of 10.

Since the safety calculations were completed, permeability values have been confirmed by measurements taken from the experimental drift at -445m, as well as by borehole measurements taken in the experimental drifts at -490m. The permeability measurements are comparable regardless of whether they were obtained on pluricentimetric scale samples or in boreholes on a decametric scale, which supports the range of values proposed. These results also support the homogeneity of the layer and the constancy of its properties.

The uncertainty about permeability is covered in the SEN by a sensitivity study. As we have seen, this parameter is not very sensitive in the SEN, the transfer regime remaining diffusive in the host formation as long as the permeability does not approach 10^{-12} m/s. In view of the measurements already taken, it is possible to conclude that the existence of a diffusive regime, all else being equal (that is, on the assumption that the seals are effective) is assured even when the uncertainties concerning the Callovo-Oxfordian permeability are taken into account.

Apart from this, several joints have been found within the host formation. Measurements taken in a directed bore-hole have shown that these features have no impact on the permeability of the Callovo-Oxfordian. The few microstructures found in bore-holes are tension cracks, which have closed up. Considering the small vertical and horizontal extent of these structures and the fact that they have been closed (by creep or crystallization of calcite), their role is negligible with respect to permeability as a whole, and is covered in any case by the sensitivity study.

- **Coefficients of diffusion**

The values proposed at present for the coefficients of diffusion result from diffusion tests of tritium, of the anions (I^- et Cl^-) and the cations (Li^+, Na^+, K^+, Rb^+, Cs^+) carried out on samples of centimetric dimensions taken mainly from zone R0 (the upper part of the Callovo-Oxfordian).

The two main sources of uncertainty in the value of the diffusion coefficient of the anions and cations are [88] :

- the effect of spatial heterogeneity of the properties of the rock. The values for the diffusion coefficient measured in core samples coming from the same core can vary ; by way of illustration, a set of 13 measurements of the diffusion coefficient for tritium were taken in several samples coming from the same core. The mean value obtained is $2.5x10^{-11}$ m²/s with a standard deviation of $4.1x10^{-12}$ m²/s ;
- of purely experimental origin, observed when a set of measurements is repeated on the same sample.

Over all, the mean values of the coefficient of diffusion calculated from the different measurements taken on core samples show a degree of uncertainty on the order of 30 to 40 %.

Additionally :

- in the case of the anions, the ionic strength in the pore water, which is a matter of some uncertainty, seems to have an effect on the values of the effective diffusion coefficient of the anions (the diffusion coefficient being stronger the greater the ionic strength) ;
- in the case of the cations, it has been shown that there is a mechanism that induces an increase in the diffusion coefficient of the sorbed cations. Thus, diffusion coefficients for cesium (the cation most sorbed in argillites among the five studied) has been observed which is ten times greater than those of tritium.

Since the majority of measurements were made in zone R0 (the upper part of the Callovo-Oxfordian), there may be uncertainties in the extrapolation of data obtained in zone R0 to diffusion coefficient values in zone R1 (the lower part). Nevertheless, considering the relative homogeneity of the porosity of the mineralogical composition of the zones R1 and R0, there is no reason to think that the coefficients of diffusion will be significantly different from one zone to the other. The first results obtained in the Laboratory, those for anions in the basal part of the Callovo-Oxfordian give good confirmation of this. The diffusion coefficients de obtained for chloride in three core samples of facies R1 in bore-hole Est 212 fall in the same range of values as those obtained for facies $R0_A$ or $R0_B$. Similarly, the values obtained for tritiated water show no correlation with the mineralogical facies.

The general goal of the « DIR experimental » programme, planned for the laboratory in Meuse / Haute-Marne, is to verify the accuracy, for in situ conditions and in full spatial scale and over an extended time, the values for diffusion coefficients and chemical retention which were determined in the surface Laboratory on samples of small size. This goal should be met for three types of behaviour : the diffusion of inert solutes (HTO) ; the diffusion of solutes influenced by anionic exclusion (chloride, iodide, bromide) ; the diffusion of elements subject to significant chemical retention (sodium, cesium, lithium…). These tests will be conducted for three facies of the argillite : i) facies R0, ii) facies R1, and iii) phases with more carbonate.

The coefficient values used in the reference calculation of the SEN are phenomenological values corresponding to the mean values of the measurements obtained from laboratory samples of :

- the anions of a solution displaying an ionic strength of 0.1M (at the centre of the range of uncertainty) ;
- cesium showing the highest values for the diffusion of cations.

The values used, then, are $5x10^{-12}$ m²/s for the anions and $2.5x10^{-10}$ m²/s for the cations.

The tracking of the diffusion tests in situ by injecting tracers (in which the decrease in the tracer concentration of the injected solution is monitored in test chambers) has already enabled us to measure the diffusion rate of water (molecules containing tritium) and certain radionuclides present in the waste or of similar behaviour. For the DIR experimentation conducted at the experimental drift level at -445m, this concerns iodine-125, chlorine-36, sodium-22 and caesium-134. The interpretation of the tracking carried out for over six months for this test produces diffusion coefficient values which are consistent with the values obtained by measurement on argillite samples. The DIR results also reveal an additional retardation of cationic species such as caesium-134. In time (2006), a core sample of the area where the tracers have diffused will be able to be taken and these results will be able to be stabilised.

The diffusion measurements obtained at this level confirm the diffusive behaviour expected in the rock and confirm the diffusion parameters for the various tracers used. They show a high level of consistency, on the one hand with the various investigation scales (centimetric sample or diffusion chamber) and, on the other, with the levels of argillite of different mineralogy.

The effect of uncertainties in the values of the diffusion coefficient is evaluated in the SEN through sensitivity studies, in which the conservative values for anions and cations are considered to be twice the reference values. These values are higher to or equivalent to the maximal measured values of the diffusion coefficient and also cover the uncertainties related to ionic strength of the water (which has an effect on the diffusion coefficient of the anions). This study, part of a more general sensitivity analysis of Callovo-Oxfordian geochemistry and that of swelling clay, shows that a modification of the values has a moderate effect on the transfer time of the anions, but little influence on the level of the effect. This uncertainty is thus covered by the SEN.

- **Porosity**

It is appropriate to distinguish between accessible porosity, a parameter which affects the diffusion of cations and anions, respectively, and kinematic porosity, that is, the porosity associated with a hydrostatic load gradient (and so, with advection).

Accessible porosity was determined on the basis of various experiments on the diffusion of tritiated water, cations and anions. These values are subject to significant uncertainties, in particular those for the cations, which have two main sources :

- the effect of the spatial heterogeneity of the properties of the rock on the scale of the sample,
- a purely experimental origin.

The experimental results for the anions give values of accessible porosity which vary from 4 to 7 %. The result for the cations, taking into account the experimental uncertainties, is the value for the total porosity. These vary from 10 to 21 %.

In the SEN, the values used for accessible porosity correspond to a value of 5 % (the mean of the values obtained for the chlorides) and for the cations to 18 % (the mean value measured for the total porosity). The sensitivity study performed in the SEN on the geochemistry of the Callovo-Oxfordian considers lower values for the anions, to bias their velocity, since these are the ones which have the fastest transfer times. Since the cations are subject to strong sorption phenomena, their transfer time is less influenced by porosity. In their case a stronger value of sensitivity to porosity has been used, to augment their flow. As regards kinematic porosity (the porosity associated with water subject to being driven by a pressure gradient), the value used in the SEN assumes that 50 % of the water is bound, which is a pessimistic hypothesis for the argillites, but which covers the uncertainties.

The uncertainties in the porosity values for the argillites are largely covered in the SEN both in the reference calculations and the sensitivity studies.

- **Sorption and Solubility**

The values for the solubility of the radionuclides in the Callovo-Oxfordian have been derived using equilibrium calculations carried using the thermodynamic database ThermoChimie v5., using a sensitivity analysis procedure. These calculations consider the aqueous conditions judged most typical and the solid whose stability is best understood.

The conservative values are established by consideration of extreme aqueous conditions, or by using the product of solubility with the highest value for the reference solid, or by referring to the largest values stated in the literature or in research programmes carried out by Andra. These values have been accounted for in the SEN sensitivity studies.

The model of radioelement sorption used is based on a linear isotherm of adsorption model (Kd constant). The values for Kd are taken for 30 elements [71] They have been estimated :

- based on experiments on the core samples of the Callovo-Oxfordian. Uncertainties about the interstitial composition are taken into account by the matrix of experimental values defined for each radionuclide. This provides input relative to the sensitivity of the Kd to the variability of aqueous conditions and covers the set of possible conditions as a function of the composition of water in equilibrium with the undisturbed rock. In the SEN, these are the smallest values used for Kd when there was a difference between the Kd measured in two core samples ;
- by default, when no experimental data were available, based on analogy using mechanistic studies on pure clayey samples, as extremes (smectites and illites) or with Opalinus clay ;

The set of uncertainties in the Kd values relating as much to their measurement as part of the experimental program as to the notion of Kd itself, which represents, at the level of the rock, complex sorption mechanisms on a microscopic scale, is accounted for in the SEN by the 'phenomenological' values and by the sensitivity study using conservative Kd values, defined at the extremes of the experimentally obtained distribution of values, based on data from the Opalinus clays, or on arbitrarily chosen pessimistic values (null, or very low ones).

6.2.1.5 Summary of the uncertainties concerning the host formation

The state of knowledge of the geological medium has allowed the use of 'phenomenological' parameters for its representation in view of the calculation of performances. These values appear robust with respect to the residual uncertainties at this stage of the programme of investigations.

Uncertainties are, however, inevitable with this type of parameter. They have been covered by :

- a sensitivity study of the permeability of the Callovo-Oxfordian, augmented by a factor of 10.
- a sensitivity study on a very conservative set of parameters connected to the geochemistry : the diffusion coefficient, porosity, Kd, solubility.

This has made it possible to take into account, in particular, the indeterminancy related to :

- core sample measurement techniques ;
- the limits of the programme of characterisation studies on the rock, as far as generalization is concerned ;
- the representation of sorption phenomena by a bias in the Kd coefficient ;
- the hydraulic role of joint potentials ;
- the possible presence of undetected minor structures.

Note that the last uncertainty is also covered in the altered evolution scenario by a bore-hole within a disposal cell.

Moreover, the sensitivity studies of the SEN have shown that uncertainties relating to permeability cannot call into question the predominance of the diffusive regime, even with an ascending gradient as strong as 0.4 m/m. As an additional precaution, a study will concentrate on the influence of the hydraulic load gradient, considering a value which is higher than it has been possible to measure or to predict in the models, in situations where advection is in a position to become dominant in the works (altered evolution scenario ‘seal defect’ see chapter 7).

6.2.2 Uncertainties in the understanding of the geological medium : surrounding formations

Recall that the surrounding formations, over- and underlying, are not called on to provide a safety function. Uncertainties relating to them should be related to the stake they have in safety ; they do not have a direct effect on the performance of the disposal system, this stops with the host formation. The surrounding formations only come under study as part of the transfer path of radionuclides to the surface, it is firstly their overall representation in the SEN, together with the related transfer parameters (advective and diffusive) which are important, as well as their evolution in geodynamic time. None of their properties of geochemical retention is involved, so their geochemistry is not as important as that of the Callovo-Oxfordian, as far as safety is concerned.

Incidentally, we remark that their characteristics as aquifers are important for the choice of outlets in the SEN. The position of pumping stations supplying the hypothetical critical group, used in the dose calculations, is determined by the water sources in the surrounding formations.

6.2.2.1 Characterisation of faults/fractures in the surrounding formations

Three fractures on three main scales, depending on their extent and displacement, are liable to be present in the Meuse / Haute Marne sector (see Figure 6.2-5).

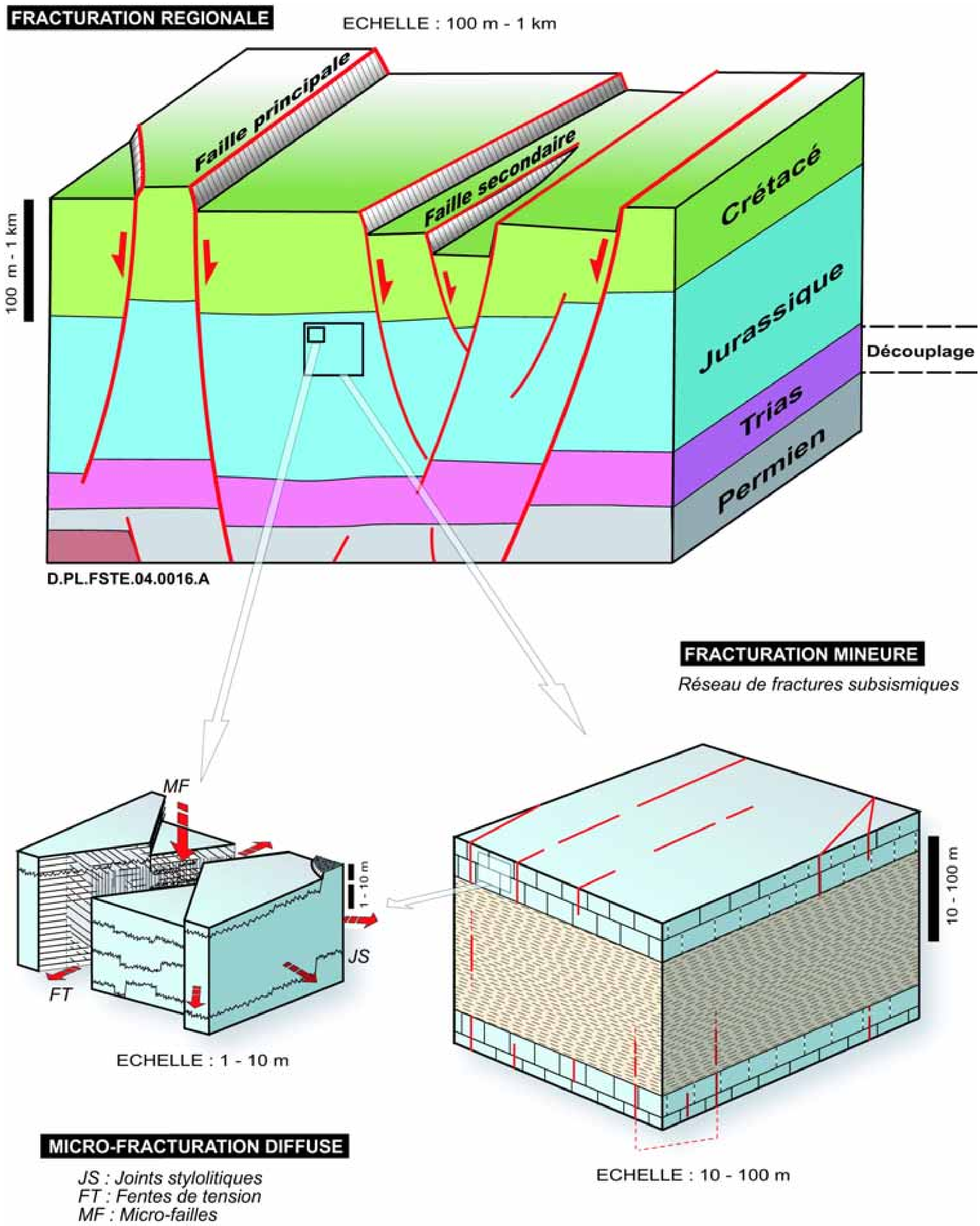


Figure 6.2-5 Different scales of fracture

- **Large scale fractures**

Large-scale fractures have been located based on geologic mapping and structural data as well as 2D seismic profiles (and a 3D block profile). These have made it possible to locate the faults in the X and Y axes with decametric precision. They reveal that, on the regional scale, there are two large families of structure (the Marne and Poissons structures (N140 to N160°E) and the Gondrecourt and Joinville trenches (N40°E)).

On the scale of the site of the Laboratory of Meuse-Haute-Marne, no major fault has been found in the overlying formations. There is not corresponding uncertainty.

To the south of the site a series of faults in the direction N140°-150°E with vertical decametric displacement have been found at intervals of around a kilometre.

To the north of this zone, and on the scale of several kilometres around the site, the cartographic and seismic geophysical data and the correlations with bore-holes make it difficult to foresee a fault with vertical displacement greater than five meters.

- **Minor fractures**

This term refers to possible structures with vertical displacement inferior or close to five meters, which could not be identified by classical 2D seismic studies.

The interpretation of the 3D seismic data obtained on the site in 1999, over an area of 4 square kilometres, repeated in 2003 and providing a 3 metre resolution for vertical displacement has not revealed any structure of this type, neither in the Callovo-Oxfordian strata nor in the overlying strata [17]. These determinations have been confirmed by bore-holes in the formation survey [89].

In the absence of detection of such a fracture, it is not possible to evaluate this type of fracturing and its consequences in a deterministic fashion. The approach taken is by analogy with structural elements of a different size : a minor structure which has possibly escaped detection would have a vertical displacement on the order of 2 meters at most, and would likely be found in the neighbourhood of a major fracture. It is this determination which has lead to the definition of the 'diffuse fracture zone' in the neighbourhood of faults in the Marne, to account for the increased permeability of the Oxfordian in this region (see following section).

Such structures would have in any case insufficient vertical displacement to cause a reconsideration of the hydraulic properties [72]. Various converging indices show that the presence of such structures would not be, a priori, incompatible with the collected data.

- **Microfracturing**

Small-scale tectonic objects are likely to be found in the set of formations composing the site : joints, microfaults, tension cracking, stylolithic tectonic joints [17].

The characterisation of the microstructures present in the sector rests on bore-hole studies and surface cartography, augmented by observations made during the course of excavation of the access shaft of the underground Laboratory. The most frequently observed global structures were joints. The principle families of sub-vertical joints are dominant in this sector and the area nearby : N°30°-50°E and N130°-150°E. These directions are in agreement with those of the major tectonic features. If these structures can be relatively well connected in an outcropping, they are generally blocked within the over-and underlying formations, when of sufficient depth. During sinking of the shaft for the Laboratory, only two structures in the Oxfordian were encountered showing seepage.

6.2.2.2 Modelling the surrounding formations

When the Oxfordian limestone is considered as a whole, its overall transmissivity changes little in the transposition zone : the main variations have to do with the distribution of production rates from one point to another. The model of the individual porous horizons (Hp1-Hp4 and Hp5-Hp7) corresponds to observations taken on the site of the Laboratory, but they cannot be correlated with the production levels in the set of other bore-holes [17]. Diagraphy of the 2003 boreholes shows that the porous horizons would likely develop in the northeast and central part of the sector [72].

The choice made in the normal evolution scenario is thus a conservative one, since it considers the case where these horizons are continuous through the sector from the point of view of sedimentology, while the diagraphs to date show their disappearance toward the west.

Similarly, the model of the Dogger shows more transmissive horizons, in order to take into account the observed variations within the upper Bathonian.

As regards the permeability of the surrounding formations taken as a whole, the data were obtained from measurements of various sections taken from different bore-holes, some bore-hole pumping tests were augmented by measurements in the shaft during excavations of the works for access to the Laboratory. The global permeability in the bore-holes is very consistent, both for the Dogger and the Oxfordian, and varies within narrow intervals. Only one bore-hole is an exception (EST 322), which shows an increased permeability. This corresponds to an open fracture which passes through the Oxfordian, which has had its permeability evaluated by means of a short pumping test, which provided an upper bound for it. This point of singularity is taken into account in the SEN by defining a hypothetical 'zone of diffuse fracturing' including this singular point in the immediate proximity of the Poissons, Joinville and Marne faults, outside the transposition zone.

Uncertainties concerning the permeability of the surrounding formations are thus entirely accounted for in the reference model of the normal evolution scenario.

Total porosity has been estimated based on measurements taken from bore-hole core samples as well as cuts from different bore-holes made in the Meuse / Haute-Marne sector. Kinematic porosity, that is, the porosity related to and corresponding to the amount of empty space in which water under a hydraulic load gradient can circulate, is evaluated as a fraction of total porosity. It has been determined on the basis of tracing experiments in the Oxfordian limestone, and it has been extrapolated to the overlying formations, for which no measurement exists. We remark that kinematic porosity has a direct influence on the transfer times of radionuclides from the host formation to the outlets. In fact, the mean velocity of the water, the vector for the radionuclides by chemical advection, is inversely proportional to kinematic porosity. To the extent that the surrounding formations do not have a safety function and are not part of the calculation of the committed dose at potential outlets, the choice was made to include from among the available data those values which tend to reduce the transfer time to the outlets. The management strategy for uncertainties has thus consisted in using the smaller values for total porosities, and thus a fortiori for kinematic porosity, in the lower ranges of the core samples available, in order to avoid overestimating the radionuclide transfer time in the surrounding formations, after they have left the Callovo-Oxfordian, until they reach potential outlets.

As regards diffusion, only a few diffusion coefficients exist resulting from testing of individual sites in the Oxfordian limestone. The diffusion coefficient values obtained with these measurements vary between 10^{-12} m²/s and 10^{-10} m²/s. There still remain some uncertainties regarding the extrapolation of this data to the Oxfordian limestone as a whole, because of the variability of its facies in the sector. Taking this into account, the values used in the scenario of normal evolution have been pessimistic with respect to diffusion (corresponding to the diffusion of one molecule of water in water). Sensitivity (see chapter 5) calculations have shown that the hypothesis consistent with using diffusion coefficients smaller than the basal Oxfordian and the Kimmeridgian resulted in reduced impact.

The same strategy (pessimistic coefficient for diffusion of water in water) has been adopted for the Dogger, for which no measurements have been taken at the site.

Regarding dispersivity, some data are presently available for the Oxfordian limestone formations. The longitudinal dispersivity is estimated at 200 meters and the transversal dispersivity at 20 meters. In the SEN, these were the values used. They are close to the values in the dispersivity reports mentioned in the bibliography. The value used for the other formations is based solely on the bibliographical data for this type of formation. Still, the effect of uncertainties associated with dispersivity in the surrounding formations is limited on account of the considerable extent of time and space covered before the activity enters the surrounding formations, and the short distance to the conventional outlet to the Saulx. This parameter seems to have little effect in the design of the normal evolution scenario.

6.2.2.3 Water flow at the sector scale

Flow at the sector scale in the Oxfordian limestone formations have been characterised on the basis of measurements of hydrostatic load taken in different bore-holes within the sector.

The construction of the hydrogeological model of the sector rests in the first place on a three dimensional view of the Paris Basin in its entirety and secondly on a view of the Meuse/Haute Marne sector. The basic hypothesis is that the current permeability of the formations depends on their facies (the nature and arrangement of the minerals according to manner of deposition) and on their history.

The model comprises a first step reconstructing the different facies as a function of their history. This stage leads to the generation of facies maps for each stratigraphic level (there are 147 levels) with their thickness, making it possible to estimate the variation of porosity as a function of the evolution of the facies, and then to estimate their porosity using correlations.

The second step consists in reducing the number of levels to facilitate the calculation of flow (a reduction from 147 to 27). The permeability values are averaged for each layer. The calculated results are verified by comparing them to data available for the Paris Basin (in this case, the three aquifers used in the zone near Paris). This test concluded that the model is consistent.

The model was then compared with the detailed studies of the Meuse / Haute-Marne sector. These studies comprise measurements taken in bore-holes and on samples. In order to be representative of the measurements taken on the site, the phenomenological values were the ones used.

As far as the Oxfordian limestone is concerned, the consistency of measurements on samples and in bore-holes revealed a small degree of uncertainty, which seems strictly related to the margin of error in the measurement of permeability and the correlation of flow with permeability. It is about one order of variable, somewhat reduced by the range of measured values, which is not large.

The model shows in the north of the sector a direction of flow to the north-northwest to northwest, and in the south of bore-holes EST 351 and EST 331 the flow is oriented more towards the west, in agreement with observations. From the hydrodynamic point of view, the model gives a good reproduction of the data measured in the sector.

The influence of regional fractures and their immediate neighbourhood is more difficult to determine, due to difficulties connected to numerical modelling. In the model, only the large faults (the Marne and Poissons faults) seem to have any influence.

In order to cover this uncertainty the hypothetical diffuse fracture zone has been defined upstream of the Marne faults, and the outlet used in modelling the SEN was sited just upstream of this zone. Dose calculation is made by 'drawing down' the radionuclides circulating in the Oxfordian to a pumping site, without taking the direction of flow into consideration.

Regarding the Dogger, at the sector scale, the direction of flow, oriented towards the west-southwest, has been described on the basis of the bore-holes EST 322, EST 342, EST 210 and two other bore-holes into the Bathonian (HTM 102 et MSE 101). The hydraulic load field present in the Dogger has been derived from various measurements taken in bore-holes around the transposition zone and on the site. The uncertainty in the measurements of the Dogger hydraulic load are estimated at from 1 metre up to 4 meters for artesian bore-holes. The hydraulic load gradient upstream of the site is estimated as being smaller (0,6 ‰ or less than 1 ‰) than that downstream of the site (0.2 %). Given the weak gradients measured on the site and upstream of it, and the number of measuring points, the main uncertainties bear on the actual direction of flow upstream. In the SEN, this uncertainty is handled by taking into consideration in the calculations the direction to the conventional outlet, and by directing the set of advective flows in that direction.

Overall, it seems that the uncertainties in the direction and velocity of flow in the surrounding formations are covered by :

- a model, in the overlying formations, in good agreement with measurements using trace elements and the definition of an hypothetical diffuse fracture zone, in which a pumping station is placed, which makes it possible to avoid consideration of downstream flows ;
- by 'forcing' the flows in the Dogger in the direction of the nearest outlet.

6.2.2.4 Presence of Water Resources

Where the Kimmeridgian just shows outcroppings in the neighbourhood of the laboratory site, about one third of the bore-holes penetrating this formation have either shown partial or total loss of water or have encountered small sources of water, notably in a limestone shelf situated at 15 to 20 meters beneath the top of the formation. The uncertainty of the origin of these water sources is dealt with by one of the outlets, which adds the possibility of long term pumping in the karstified Kimmeridgian (at the outcropping).

The strategy adopted for the Oxfordian was to use all water sources identified in the calculations, even if they were deep or not very productive, on condition that the use of their water did not seem too unreasonable. This lead to :

- systematic use of bore-holes as outlets, instead of directly using watercourses, which is however more likely. The bore-holes dilute radioactivity less and can be placed closer to the site than the water courses ;
- consideration of a water production bore-hole just upstream of the diffuse fracture zone, throughout the level of the Oxfordian, to cover the case where surface aquifers would not be considered fit for providing drinking water ;

The Dogger does not present a realistic outlet at the scale of the zone considered. Nevertheless, in order to break into the Dogger flows and so to consider them in the impact calculations, a bore-hole was included, as a matter of speculation, with the same positioning as that placed in the 'diffuse fracture zone' of the Oxfordian. This choice was largely conventional.

6.2.2.5 Summary of the uncertainties concerning the surrounding formations

The reference model used for the SEN, and the sensitivity studies used to test this model cover the uncertainties related to :

- characterisation of properties, as regards permeability, porosity and diffusion coefficient ;
- the direction of flow in the Dogger ;
- the lateral extent of the aquifers.

In addition, some uncertainties which have not been formally covered, due to the impossibility of giving an abstract definition of the conditions in which it might be shown that they are detrimental, have also been considered :

- as regards dispersivity, although it is difficult to determine conservative values, this parameter seems to be of second order with respect to transfer distances ;
- fitting of the hydrogeological model has resulted in transfer times consistent with the set of trace element measurements performed on the site. Directions of flow have been shown moreover to have little sensitivity to variations in the hypotheses.

Finally, in the larger perspective, uncertainties of any kind to do with transfer within the surrounding formations are covered by choosing outlets very near to the site, minimising the transfer times in the surrounding formations with respect to the time spent within the host formation (a factor of at least 5), and the choice of deep outlets which are more pessimistic than those suggested by the basic safety rule RFS.III.2.f.

6.2.3 Uncertainties related to modelling the biosphere

Chapter 5 describes how the biosphere has been defined. This rests on a number of hypotheses, of which several are standard practice, notably the choice of ways of life similar to our own for populations living in the distant future. This is not, strictly speaking, a source of uncertainty, insofar as it is the only reasonable choice to be made for the purpose of dose estimation. This is not a question of trying to precisely model all the possible types of human behaviour, rather, moderate hypotheses are made corresponding to a typical population considered stable over time. This typical population is however chosen in a manner which is conservative with respect to exposure, relative to those which might be chosen a priori.

Once this framework has been established, the uncertainties regarding the biosphere relate to :

- the choice of the critical group, from among the possible choices, taking into consideration its standardised character and the characteristics of the site, climatic ones, in particular ;
- quantification of the transfer paths of elements into the biosphere.

Recall that the dose calculation is used only to verify conformity to the standards for radiological protection, in order to provide a view of the overall safety of disposal. The approach to safety taken by Andra is one of optimisation under constraints, in accordance with the recommendations of ICRP, which seeks to minimise impact as far as possible, without considering an a priori threshold. Thus an uncertainty regarding the dose calculation, even one of a significant degree, is of no practical consequence as far as the size of the disposal is concerned, but can only affect the final appreciation of the impact. Whatever the result of the calculation, the most effective technical means appropriate are to be used to reduce the impact.

Only those uncertainties have been considered which bear on the calculation of the factor of the radionuclide dose which dominates the impact, in this case, iodine 129. Consideration has been also made when necessary of the uncertainties relative to chlorine 36 or selenium 79, in the knowledge that even a factor of 10 or more of uncertainty about their associated dose would not have an effect on the result of the safety analysis.

The parameters connected to the biosphere call for two types of data. Certain of these are related to the site (climate, soils, living practices) but are independent of the radionuclide or toxic chemical being considered. Others are dependent on the particular chemical in question : these characterise physical, chemical and biological phenomena which allow the transfer of elements.

6.2.3.1 Choice of the critical group and of the site parameters (climate, soils, living practices)

It has been mentioned that the biosphere chosen was a temperate one, stable over time. If a ‘cold’ climate prevailed in the site, though, the group of agriculturists defined in the context of a temperate climate would remain the most pessimistic critical group, although they would be less likely to be found in such a context. The behaviour patterns of semi-nomadic pastoralists, more realistic in a glacial climate, would result in lower doses [73].

Different reference groups have been considered by Andra. In theory the choice of a group which will result in the largest dose depends on the radionuclide being considered. The following have been considered for sensitivity studies with respect to the selected group :

- beef and dairy farmers
- sheep farmers
- pig farmers
- fowl breeders
- grain producers
- hunter/gatherers

The main difference between these groups is that they mainly consume the food product they specialise in producing, consuming very large quantities of it, and that they are self-sufficient for the product in question. Such groups are, by definition, unrealistic. In contrast, the group selected for the assessment is represented by an agriculturist living for the most part on his own products, whose dietary behaviour is characteristic of the Meuse/Haute-Marne region and based on INSEE surveys. This group has been used in the model since it combines the largest number of relevant opportunities for exposure and is highly representative.

The sensitivity studies and the evaluation of uncertainty show that a change in critical group, and particularly in eating habits, gives rise to a variation of 1.5, at most, in iodine 129, for example. It is 2.1 for selenium 79 and 3 for chlorine 36.

Additionally, if a condition of complete self-sufficiency of the agriculturist is considered, a condition which has never been observed but which puts an upper bound on the impact, the dose due to iodine 129 does not change, and that due to chlorine 36 is multiplied by a factor of 2.6 while that of selenium 79 is multiplied by a factor of 2.1.

This high degree of stability in the biosphere conversion factor of iodine 129 is due to the dominance of drinking water as a means of contamination. As a result, it is largely independent of other aspects of dietary behaviour.

The critical individual has been taken to be an adult, since children have been shown in other studies to be subject to a smaller or equivalent impact as a result of consuming less (see figure below) with regard to radionuclides with an impact in the normal evolution scenario : iodine 129 and chlorine 36.

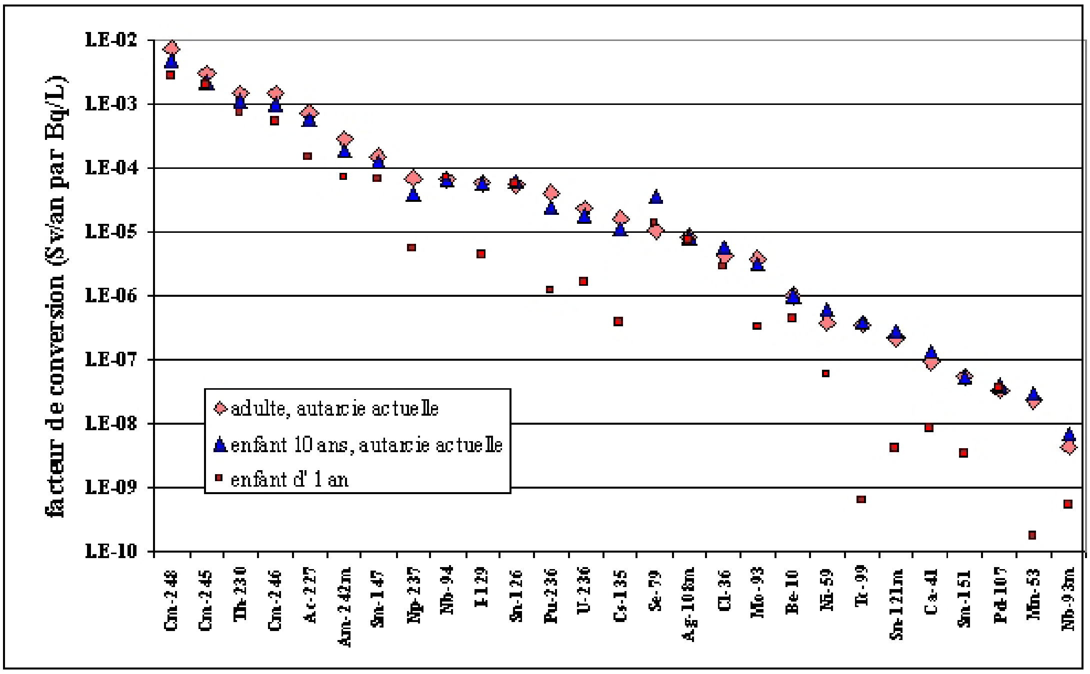


Figure 6.2-6 Variability of biosphere conversion factors according to the age of reference group members

6.2.3.2 Global uncertainties in the biosphere conversion factor

Sensitivity studies have been conducted to characterise the variation in impact as a function of transfer factor values between the different elements of the biosphere [76].

The results show that iodine is particularly insensitive to this type of parameter. This is a result of the simple chain of contamination (mainly drinking water), as was the case with its sensitivity to the critical group.

The maximal uncertainty, taking account of the entirety of possible variations (habits of the critical group, transfer parameters) and including an uncertainty dealing with water consumption, results in a global factor of uncertainty of at most 2.

The uncertainty in the biosphere conversion factor for chlorine 36 includes the uncertainties connected to the specific parameters of the model (for example, the concentration in stable chlorine in the neighbouring environment), of the site, and of consumption. The uncertainty is not very significant (a factor of 2.2) without exceeding the impact objective, even in the most pessimistic situations. The global uncertainty for selenium 79 related to transfer parameters, to the site and to consumption is a factor of 4.1, which does not call into question the conclusions of the impact calculation.

6.2.3.3 Conclusions on allowing for the uncertainties related to the biosphere

The biosphere is one of the areas of the calculation which can present a priori the greatest uncertainty because of the large number of parameters which characterise it and the multitude of its possible evolutions. These uncertainties are however bounded by making the standard choices, which are accepted at the international level, which restrict the domain of investigation. However, there does remain an uncertainty which reaches a factor of 10 for certain radionuclides. It is for this reason that the biosphere has only been considered when coming to the dose calculation. No safety function has been assigned to it.

A large and very detailed study of the uncertainties bearing on the radionuclides which have the largest contribution to the impact within the specific outline of the Dossier 2005 (essentially, iodine 129) shows that the uncertainty in the dose factors is, in their case, much more restricted. The comparison to international practice shows, moreover, that the values used by Andra are in accord with standard practice.

6.2.4 Uncertainties concerning package inventory

The waste packages are under the responsibility of their producer and/or the organization in possession. Apart from the occasional exception (high level waste resulting from past activities such as the production of objects containing radium), the inventory of high-level and long-lived waste is the product of the nuclear power industry, research activities associated to it, and activities of the national defence establishment. The sources used by Andra for questions relating to understanding the nature of the inventory are EDF, COGEMA and the CEA. The method Andra used in collecting from them the data necessary for the study and in ascertaining its reliability is described in section 1.5.3

6.2.4.1 Quantitative waste package inventory

The quantities of wastes now in existence are well understood, but uncertainties remain concerning waste to be produced in the future, the number and characteristics of which depend on the plan of operations adopted by the producer. We remark that, in other cases, the assignment of certain wastes to a class of Intermediate Level, Long-lived wastes is weighed against a different assignment, which creates a measure of uncertainty regarding the inventory to be considered.

To cover these uncertainties, Andra has decided to maintain a constant dialogue with the producers of wastes, and to adopt the best available hypotheses, or the bounding hypotheses, in order to define a model of the waste inventory (« MID »). In more detail, the strategy consists in :

- using hypothesis for future waste production consistent with those communicated to the government nuclear safety authority by the producers, in particular the 'fuel cycle report ' of the EDF [18] ;
- in the case of an uncertainty in the quantity of packages to be produced, use of the hypotheses providing the largest value. In particular, the MID has sought to incorporate the different possible strategies of the industry for dealing with the fuel cycle downstream. Two models (S1, with fuel reprocessing, and S2, with reprocessing halted after 2010) have been defined as a basis for quantitative forecasts. Model S1 maximises the number of glass packages to be produced, while model S2 maximises the packages of spent fuel. Model S1 considers different scenarios in order to cover the foreseeable variations in the manner of treatment of fission products and of plutonium downstream of the cycle. The normal evaluation scenario as a general rule uses the maximal inventory or an inventory very close to the maximum for the entire set of scenarios and for each type of waste. In this case, this is pessimistic, since a significant part of the inventory of fission products is potentially counted at least twice, once in the glass packages of scenario S1 and once in model S2 ;
- in the case of uncertainties concerning assignment to the Intermediate Level, Long-lived class or another class, use of the hypothesis resulting in the largest value for the group of Intermediate Level, Long-lived wastes ; this is the case, for example, with the old casks of bitumen-coated wastes at Marcoule, whose total inventory was included in the MID.

The MID thus offers various guarantees of its bounding character. However, there still remains the possibility of changes in the currently envisaged production processes leading to the production of wastes in quantities different from those estimated today. Moreover, it is possible that some older wastes which are now assigned to one group may in future have their classification changed and be considered as Intermediate Level, Long-lived waste. Thus, it cannot be concluded that the design of MID as an outer bound is perfect.

However, the MID does cover all the types of packaging (glass, bitumised, compacted, cemented, bitumo-cement, homogenous and heterogenous wastes, mixtures of organic and non-organic waste, sources…) known today. It also covers the hypotheses for concentrations of radioactivity in the packages (and thus of heat transfer, contribution to dose, auto-irradiation, etc.), hypotheses which are in accord with current practices but also take into account possible evolution in production : the glass packages with increased heat transfer (C2 reference package) are taken into account, as are the bitumised packages STE2 planned by COGEMA. The quantities are the result of estimations of maximal values, and any changes in the Intermediate Level, Long-lived or High Level inventories do not give any real cause to significantly question them.

The MID, and thus the inventory considered in the SEN in terms of reference packages, can thus be considered to provide an outer bound with a good margin of confidence.

6.2.4.2 Radiological characterisation of the packages

Characterisation of the radiological content of waste packages rests as a general rule on the mean values for activity per waste package, together with their margins of error. These values have been employed in the impact calculations which consider the entire inventory. For other purposes, (for example for operational studies which consider only one waste package, or a very small number of waste packages, at a time, during normal operation or during an accident), maximal radiological inventories have been defined.

Radiological inventories can be the result of :

- surveillance of production, particularly for packages which are subject to quality assurance standards, which is the case for the entire production since the mid 1990's. These can involve taking of samples prior to packaging, or more simply measurements of package contribution to dose, on the basis of which a radiological inventory can be arrived at using a typical spectrum, which itself has been validated against samples provided or by calculation ;
- calculation, which is the case especially in the case of wastes whose radiological inventory can be unambiguously linked to the conditions of fuel irradiation within a reactor. Spent fuel packages from PWR (pressurised water reactors) are covered in this case, as are vitrified waste packages, making use of factors correlating the content of fuel and of glass. It is useful in this case to know the irradiation history of the fuels, at least in general terms ;
- characterisation studies of particular cases, for older wastes which can have samples taken from them. These studies can be carried out in several phases. Thus, older wastes presently in temporary storage at the production site may have been the object of random sample analysis, but would be more systematically characterised when recovered for packaging ;
- knowledge of the production history, even in cases where this has not always been tracked according to quality assurance standards.

These different approaches can be tested against each other to reinforce the assessments. Thus, the inventory of vitrified wastes was established on the basis of measurements taken by COGEMA and compared to the results of irradiation calculations for the corresponding fuels.

The main uncertainty, in this context, rests both on the stock of older wastes which have not yet been the subject of a detailed characterisation, or whose characterisation, at the time, considered only particular radionuclides, as well on future packages whose characteristics have not been completely defined at the present time.

Older waste packages (in particular the sub-family B2.2 and some of the B-3 reference packages), some of which could in the future be the object of recovery and packaging operations, are known on the basis of sampling and the history of the production process. The representativity of these data has not yet been tested against the actual state of individual packages by a systematic characterisation study, something which is planned as part of recovery. Such studies should make it possible to refine the understanding of the packages.

The inventory of sources and similar objects (reference packages B7 and B8) considered for deep disposal is poorly known, both with respect to the number of packages in question and the diversity of the radionuclides they contain. This inventory is marginal, however, compared to that of the other packages, and this uncertainty does not seem to be a significant factor in the specifications of a repository.

Future waste packages, (in particular reference packages C2, C3, C4 and certain B2 waste packages) are the subject of a definition of their radiological content based on what can be projected at this stage concerning the process to be used. This definition can be more or less close to a current industrial practice : in the case of bitumen packages resulting from the treatment of sludge from station STE2, for example, whose radiological content is greater than that of current bitumens, use of this package will depend on approval by the nuclear safety Authority. Certain packages, such as the glass ones containing 1 % plutonium, are defined in a purely formal manner, with the sole aim of enlarging the range of reference packages under consideration.

For each reference package, the SEN assigns a radiological inventory which corresponds to a maximisation of the radioelement content of each wste package family or, if necessary, each type of waste constituting the referencel package. Within each waste package family, the inventory assigned corresponds to the mean inventory multiplied by the number of packages. This procedure guarantees a globally maximising character to the total inventory of radionuclides used in the calculation.

The level of knowledge of the radiological content of the packages is valued in direct proportion to its use in safety studies. The priority concerns the radionuclides which can contribute to the impact in normal conditions over the long term (chapter 5) or during operations (chapter 4), including here studies of criticality safety. At the same time, there is also an interest in those radionuclides which can cause an impact in the altered situation (see chapter 7). From this point of view, the most active packages (glass and spent fuel), those containing the largest load of mobile elements, or those most likely to cause a criticality accident (spent fuels of the PWR, UOx and MOX types) pose the most risk ; these are also those which are best understood from the radiological point of view.

In each case where available data were not considered sufficient, Andra has undertaken to define its own bounding inventory values by using calculations to complete the spectra (based on the calculated radionuclide content of the fuels, based on the ratios used in the standards set for similar ILLW packages).

The maximal impact is produced by spent fuels, whose inventory is determined on the basis of irradiation calculations. Other packages contribute significantly less to the impact. The uncertainties concerning the radiological inventory are thus not of a nature to call into question the dose estimated in chapter 5. Uncertainty concerning radiological content is thus reasonably well controlled.

6.2.4.3 Chemical characterisation of the packages

Chemical characterisation of the packages covers several problems :

- knowledge of the elements likely to be chemically toxic. These are defined on the basis of the toxic materials considered in standard regulations concerning toxic wastes which have been adapted to the specific processes employed in the nuclear industry. The materials considered are uranium, lead, mercury, antimony, cadmium, selenium, arsenic, nickel, boron, beryllium, total chrome, chrome VI, free and bound cyanide. The impact calculations considered only those present in the largest amounts ;
- knowledge of elements likely to form compounds facilitating the migration of radionuclides or toxic chemicals in disposal conditions. These may be organic compounds (for example, molecules resulting from the treatment process : tributyl-phosphate, etc) or inorganic ones (sulfates, etc) ;

- knowledge of elements likely to attack their surroundings (aluminium, nitrates…) ;
- materials which, as a result of degradation, might give rise to reactive compounds, or to release gas. This is mainly a matter of organic compounds and metals (as a result of corrosion).

The chemical characterisation of the packages is based on the declaration of their constituent materials which the producers supply in their descriptive catalogue. It is then a matter of determining the chemical elements contained in these materials. In particular, determining the details of the steel used in containers or for the manufacture of objects included in the waste (bits of shielding, technological waste, etc), the different types of cement-based materials employed, the supplier's specifications for bitumen, and the formulae for the vitrified matrices are needed for the calculation of an inventory. Knowledge of the history of the production processes also makes it possible to identify possible inclusion of particular products which might be found in the waste (for example mercury and tributyl-phosphate in the older bitumised packages). The inventory is subject to uncertainties connected to the nature of certain materials contained in the waste (particularly for waste resulting from packaging to items in bulk) as well as to possible variations in their composition.

In the current state of knowledge, the chemical inventories have not achieved the same level of reliability as the radiological inventories. Overall :

- the inventories for spent fuels and glass packages are well known, the chemical composition being a parameter that is kept track of during production. We remark that there are certain unknowns relative to the composition of certain fuels used by the Defence department, since the corresponding information is classified. But these uncertainties do not create any particular difficulty at this stage, considering the small quantity of material concerned ;
- the inventories of B type wastes produced from the early 1990's (B1, a large part of type B2, packages B4, B5, B6) are in general precise, due to quality control records ;
- the inventories for the oldest wastes (B2 and B3 packages) are less complete, but can be in large part reconstructed with the aid of representative mean values. Nevertheless, we remark that the oldest wastes among the B3 packages, which were produced during research, still pose certain difficulties related to the multiplicity of materials used during the period (glass, asbestos-cement, etc.) ;
- the chemical inventories for the B7 and B8 waste packages are at a very early stage, due to less full knowledge of these wastes of 'ancient' origin with respect to other waste types, and which additionally are of a more varied nature. However, these represent a small quantity relative to the remaining inventory of packages.

From the standpoint of safety, the inventory of elements which might attack the packaging must be considered ; at the feasibility stage, they do not seem to raise any specific problems.

The sorts of chemicals in packages of B waste are a possible source of colloids, as are the geological medium, bentonite and concrete. They are all handled together by the 'colloid filter function' which consists in establishing around the disposal cells a region of low porosity (the geological medium, the plug). The uncertainty is then a question of the capacity of the medium to control colloidal transport, which is brought up in section 6.2.10.

6.2.4.4 Summary of uncertainties relating to the inventory

The management of uncertainties relating to the inventory, shows different categories :

- uncertainties arising strictly from the inventory (in terms of number of packages, radionuclides or toxic chemical elements) which only affect the input data for the impact calculation. They are largely covered by the bounding choices made when dimensioning inventory model. However, it cannot be demonstrated that this model is based on the most pessimistic case. In all events, variations on any given family of packages do not seem to be sufficient to cast significant doubt over the calculations ;
- uncertainties that may, through the packaging modes, have an influence on thermal aspects, package irradiation rate or radiolysis gas emission rate. These are also widely covered by the inventory model hypotheses ;

- uncertainties concerning the role of certain substances that have either a complexing or an aggressive effect on their environment and may be present in families of the inventory model. These are covered in the first instance by the repository design - cementitious medium in the B cells, possible dimensioning of B waste packages to resist sulphate attack, for example - and by the « filter the colloids » function. They do not seem sufficient to cast doubt over the performance calculation. However, degraded geochemical properties have been considered in the normal evolution scenario for disposal cells containing cellulosic waste. Characterisation of waste whose specific chemical nature may pose a problem will have to be continued in order to further reduce the uncertainties.

After detailing uncertainties concerning the description of the host medium and the package inventory, the following sections will consider the processes governing repository evolution, including both the phenomena involved and the models describing them. The survey of uncertainties is based on the findings of the research work conducted by Andra and its international counterparts.

The aim is not to produce an exhaustive survey of the entire phenomenology as a more detailed analysis will be found in the AQS document [33]. Instead, this document concentrates on phenomena for which uncertainties with possible safety implications have been identified during the qualitative safety analysis process, on the basis of their potential influence on the safety functions and/or the representation of the SEN.

6.2.5 Uncertainty concerning near-field hydraulic phenomena

6.2.5.1 Uncertainties concerning the influence of resaturation on radionuclide transport

Uncertainties relating to near-field hydraulics concern primarily the duration of the repository structure resaturation phase, which depends on the phenomena affecting it. Before considering these uncertainties, a reminder of their importance for the safety analysis is required.

In the SEN, the repository is represented in a saturated state from the start. This is a calculation hypothesis used to simplify the representation of the repository. It only concerns the calculation itself. All analyses concerning repository evolution, and the behaviour and evolution of components, etc. take into account the evolution of the saturation level over time.

The objective is not therefore to evaluate precisely the resaturation durations in themselves, but to study whether the uncertainties capable of affecting these durations are likely to influence the phenomenology of the components to such an extent that the safety functions could fail or prove ineffective when they are required. It is also important to question the hypothesis adopted for the SEN, in which the repository is considered to be resaturated from the start, in order to verify whether it is pessimistic or overlooks phenomena that could encourage faster migration of radionuclides, and if this is the case, whether these phenomena are quantitatively significant.

Three questions can be asked :

- in itself can the phase during which the repository is not resaturated lead to phenomena capable of accelerating radionuclide transfer? It has been seen to have little influence on radionuclide transfer on the scale of the entire repository ;
- if parts of the repository are saturated more quickly than others, can the resulting hydraulic gradients cause movements of water contaminated by radionuclides?
- do the gases produced by corrosion or radiolysis after closing the repository, which influence the hydraulic transient, also influence safety functions and radionuclide transport?

With regard to the first question, it seems that desaturation of media surrounding the waste assists in slowing down or even stopping its migration. Uncertainties still exist concerning, especially, how much the diffusion coefficient depends on the level of saturation. However, from this point of view alone, it seems that the hypothesis of a repository resaturated from the moment of closure is a pessimistic hypothesis.

The second question refers to the possibility of hydraulic gradients forming between the disposal cells and the adjoining drifts because of their lower saturation level. B cells are the more extensive, contain more empty space and are exposed to stronger gas action than the drifts adjoining them. They therefore become resaturated more slowly and any hydraulic gradients would be oriented towards the inside of the cell.

The situation may be the reverse in C waste and spent fuel disposal cells, whose diameter is smaller than that of the adjoining drift. These cells are therefore likely to become resaturated more quickly. Over a short period of time, a hydraulic gradient may exist between the disposal cell and the drift. However, over a century, the plugs of C and spent fuel disposal cells become saturated and form an obstacle to advection. In a context where transport has become diffusive, horizontal gradients, which are weak in any case, can have no influence. In addition, this situation of a forced gradient outside spent fuel disposal cells has been addressed in a sensitivity study based on the SEN. Its purpose is to cover the effect of gases (see next section), but in doing so also covers a situation of differentiated saturation.

The question of the gas influence is addressed in Section 6.2.5.2.

6.2.5.2 Uncertainties concerning the kinetics of resaturation, gas production and transfer and coupling between the two

The resaturation phenomenon is only important for the safety analysis in so far as it can influence the repository safety functions either directly or through disturbance. The behaviour of some components may depend on their saturation levels. In this case therefore, the uncertainty in the behaviour of a component or its safety functions is related to the uncertainty concerning the evolution of its saturation level over time. However, it seems that the phenomenon that can have the most significant influence on resaturation is gas production inside the repository.

Most of the gases result from the production of hydrogen by corrosion under anoxic conditions, although other sources (radiolysis of water or organic matter, production by micro-organisms, air occluded in the repository at closure, etc.) may be added. The kinetics of gas production are therefore directly related to corrosion kinetics. In accounting for gas production, [61] these kinetics are considered under the following conditions :

- the rates adopted are high (2 to 3 microns a year) compared with those generally observed in an anoxic medium (usually less than one micron) ;
- the relationship between corrosion rate and water content of the medium is ignored, even though the influence is strong. The rates adopted correspond to media more than 80 % saturated ;
- any retroactive effect on the corrosion reaction due to the hydrogen itself is ignored. As this effect is theoretically negligible, this last point provides no significant safety margin.

The gases may then undergo various processes already discussed in Chapter 5 :

- in small quantities, they may dissolve in water and migrate with it by advection and/or diffusion. This process is not very efficient in formations with very low permeability, such as argillites ;
- in greater quantities, they may be released through biphasic flow, i.e. in a different gaseous phase through the pores of the materials. In such cases, they are likely to interact with the free water in these pores and to move it once they overcome capillary pressure. This phenomenon is called « suction ». It occurs at a pressure threshold that depends on the pore size of the medium, known as the « gas intake pressure »;
- at even greater levels the gases may act on the medium in which they flow and expand its pores, thereby increasing their flow. In such cases, the gas permeability of the medium increases but the water permeability drops as the gas occupies all available pore space. The opening of the pores, known as « microfissuring », despite being a different phenomenon from mechanical damage in the excavation damaged zone (EDZ), is considered to be reversible, a point that has been verified experimentally by Andra and international groups [7] ;

- in very high quantities, the gases may cause irreversible damage, comparable with fracturing. In this case, they are said to have reached « fracturing pressure ». In argillites this pressure has been evaluated experimentally (by boring) at 12 MPa.

The curve in Figure 6.2-7 shows the pressure evolution required to move water and the degree of desaturation caused by this pressure, for various materials according to their water content, on a logarithmic scale. It can be seen that the pressure in argillites with less than 90 % saturation is several orders of variable greater than the gas pressures that can develop in the repository. The curve is similar for bentonite. In these materials, the gases cannot cause major desaturation. However, suction pressure is much lower in concrete or highly porous media such as sand. This distinction plays an important role in the study of gas action inside the repository.

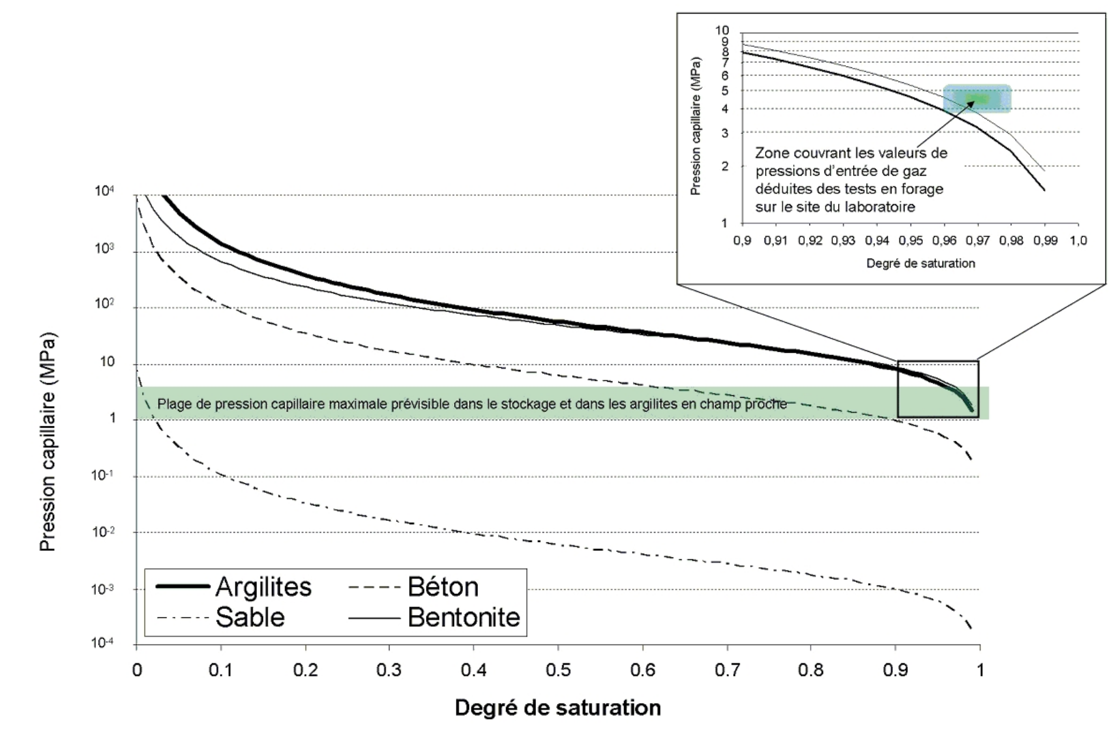


Figure 6.2-7 Suction in various porous media versus water saturation level

The components whose behaviour depends directly on their saturation level are :

- the host formation, whose properties, especially mechanical and thermal properties, are influenced by its water content. However, given that the gases only have a slight influence on its saturation level, the total resaturation hypothesis is a valid approximation ;
- fractured argillites, whose porosity may be greater and which may have lower gas intake pressures than sound or microfissured argillites. However, any desaturation of the fractured zone would tend to restrict its evolution and reduce its permeability, which are in all events positive effects in terms of safety. These uncertain effects are not included in the SEN ;
- bentonite seals, which can only have their full effect if their saturation level is such that they can develop sufficient swelling pressure to perform the « oppose water circulation » function ; these also become resaturated quickly (around one hundred years for C and spent fuel cell plugs, around one thousand years for other seals) and are not greatly influenced by the gases ;
- backfill, which depending on its level of compaction - directly influencing its porosity - may be partially desaturated by the gases ;

- concrete components, whose degradation depends on their water content. For these elements, the hypothesis of a repository resaturated from the start is a pessimistic hypothesis ;
- metal parts (ground support of thermal waste cells, linings, containers, etc.) which begin to corrode at low saturation levels (30 %), with maximum corrosion at around 80 % saturation. For the degradation of these components, the hypothesis of a repository resaturated from the start is pessimistic, since it is equivalent to accelerating corrosion and maximising the production rates of the associated gases.

Possible gas pressure, assessed according to the quantities of steel, corrosion rates and the ability of hydrogen to diffuse in the medium and the curve indicating the evolution in capillary pressure as a function of the rate of water saturation in argillites, may reach 7 MPa in B waste disposal cells, 6 or 7 MPa in C waste dispoqel cells and around 9 MPa in the spent fuel disposal cells. Sensitivity calculations show that, for a given gas production rate, uncertainties on these pressure values due to capillary pressure curve choices for argillites are of the order of 0.5 MPa.

The following sections deal with hydrogen behaviour in the various repository compartments.

- **At the level of C waste and spent fuel cells**

The gas circulation diagram takes into account the dissolvability and diffusibility of the gases in pore water. However, this release mode proves inadequate for managing all the gas produced in the vitrified waste and spent fuel cells, and a pressure rise seems inevitable. This rise stops when the gas can escape through the medium offering the least resistance.

Until the cell plugs become sufficiently resaturated, several gas transfer routes are possible : the interface between the plug and the rock, not yet blocked, the interstices in the plug itself, or even the microfissured zone. The evaluations show that the time before the plug becomes saturated (to a level greater than 97 %), is around one hundred years.

Once the plug has developed sufficient swelling pressure, the gas accumulates in the cell and desaturates the clearances around the packages, thus contributing to slowing down their corrosion. This effect is not taken into account in the SEN, which is pessimistic from this point of view.

The gas can move through the damaged zone, continue to escape through the plug (although the high gas intake pressure of the plug and its low surface area do not make this the most likely route), or escape radially through the host formation which, despite having low permeability, presents a large surface area (see Figure 6.2-8).

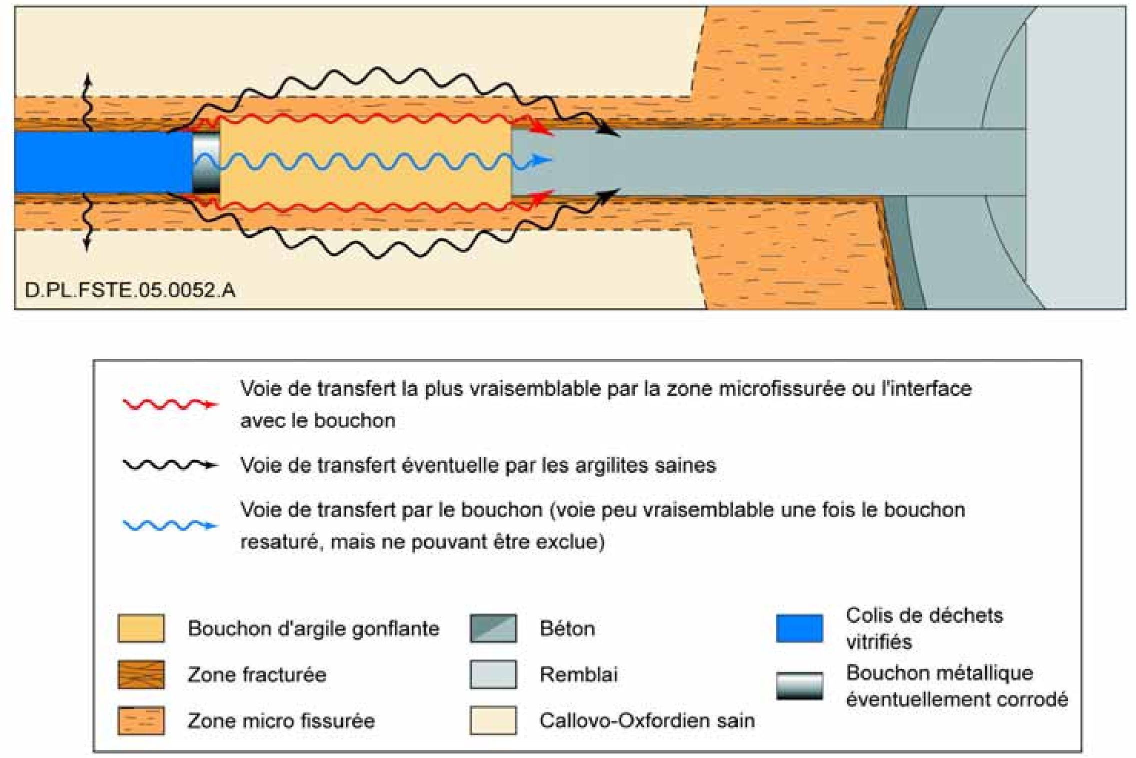


Figure 6.2-8 Diagram showing the three gas release modes in the waste disposal cells

In the spent fuel disposal cells, the gas produced by the perforated metal liner may follow the same route ; beyond this, the pressure of the gases produced by the corrosion of the lining and container rises to reach the gas intake pressure in the clay engineered barrier. The effect of the gases on the clay engineered barrier has been investigated in several international tests, which have shown its healing capability [90] and [91] ; the transient passage of gas does not jeopardise the functions required of it. The gas then escapes via the various routes considered above.

The cell therefore remains under gas pressure for a few thousand years. This pressure does not significantly desaturate the plugs or the rock. However, the gas may impose transient hydraulic gradients on the rock at a period when the containers are still leaktight (for spent fuels, the dimensioning imposes a minimum of 10,000 years ; for vitrified waste the period is 4,000 years in a totally resaturated medium, but significantly longer when the desaturation effect of the gases is taken into account).

The share of each transfer route depends on the relative intake pressures and surface area presented and therefore remains relatively uncertain. However, these uncertainties have no significant effect on the representation of the SEN, in so far as this scenario assumes that the cell is resaturated.

The SEN is based on the assumption that the C waste and spent fuel disposal cells are resaturated instantaneously and that the plugs are immediately effective. The preceding description confirms that this representation adequately accounts for the uncertainties relating to the effect of resaturation on the gases :

- it has no major effect on the plug, which is effective before 100 years and, because of its very low porosity, cannot then be desaturated by gas pressure ;
- it has no effect either on desaturation of the geological medium, which remains low ;
- it is pessimistic with respect to the cell interior, which may be significantly desaturated, but given the configuration, the lifetime of the containers and overpacks can only be greater that the lifetime evaluated under SEN.

The residual uncertainty concerns the occurrence of a possible seal defect on the spent fuel containers or C waste overpacks during the hydraulic transient. In principle, an event of this kind would have no consequences. As most of the gas production term comes from the lining (and also the metal liner around the clay engineered barrier in the case of spent fuel disposal cells), a defect during the pressure rise phase would occur at a time when the container was not yet corroded and was not in a saturated medium. The defect would not be likely to evolve and releases would therefore be prevented.

Nevertheless, a degree of uncertainty concerning this qualitative reasoning has been factored in by considering the sensitivity of the SEN to a container defect leading to release during the thermal and hydraulic transient. The pressure levels imposed in the near-field Callovo-Oxfordian are around 7 MPa at most in the C disposal cells and 9 MPa at most in the spent fuel disposal cells, for periods of less than 10,000 years. The modelling adopted does not take into account the favourable effect produced by gas occupying the pores, which prevents water migration. Despite these choices, it has been seen that this event is too short to have any consequence on the impact. This sensitivity study has been conducted in the « package defect » SEA.

However, it must be remembered that at this stage only uncertainties concerning resaturation time and transport are dealt with : the mechanical effect of gases on the repository components is dealt with in a specific section (see Section 6.2.6.2).

- **At the level of B waste disposal cells**

In current estimations, the gas (essentially corrosion-generated hydrogen, but radiolysis gases also contribute quantitatively to the B2 waste count) is released initially through the cell plug. The total saturation time of the cell plug is estimated at a few thousand years. However, after a few centuries the plug is efficient enough to oppose water and gas circulation. In a pessimistic evaluation of gas production, the models predict pressures of around 6 to 7 MPa after 500 years [61]. Under these conditions, not all the gas will be able to escape by diffusion ; some will migrate either through the damaged zone, by-passing the seals, or through the host formation.

Around the package itself, the high macroporosity of the medium will be rapidly occupied by the gases, preventing water ingress in the vicinity ; saturation therefore remains low throughout the gas production period. As for the waste itself, where it is isolated from the water by the package and the ambient gas, it remains at saturation levels close to the initial level : high for packages containing concrete, low for those made up of compacted wastes.

The point at which the drop in gas pressure occurs depends on the gas production kinetics and extends over roughly one hundred thousand years.

With regard to the representativeness of the SEN and the assumption of immediate resaturation, this hypothesis can be seen to be :

- pessimistic compared with some release models, especially those for CSD-C packages, which is unlikely to be exposed to a significant saturation level for the entire period during which gas pressure is maintained in the cell voids. For other waste, the hypothesis is not very sensitive ;
- pessimistic with regard to damage of concrete packages. In a dry atmosphere consisting predominantly of hydrogen, concrete is less likely to be exposed to chemical degradation processes ;
- very pessimistic in terms of transfer inside the disposal cell, which remains dry for very long periods. It is very difficult for the radionuclides to migrate in the waterless macroporous structure (especially the interstices between packages) ;
- without major influence on the host formation which undergoes little desaturation by the gas. It may be exposed to pressure gradients imposed by the gases over the first few metres, but these gradients reach their maximum level very soon (after 500 years in the models), before the radionuclides are able to migrate out of the packages. In addition, as with C waste and spent fuel cells, the low desaturation and occupation of a few percent of the rock porosity by the gases are in principle positive phenomena likely to oppose water circulation ;

- without major influence on the B disposal cell seal. During the first few centuries the seal will not have become sufficiently saturated to provide an effective barrier to the gas and will allow it to pass. After 500 years, when the pressure rise occurs, it is not fully resaturated in the strictest sense but is sufficiently effective from the hydraulic point of view.

Despite these favourable points, it was decided that sensitivity studies under the SEN should include an uncertainty element for all the points above, i.e. concerning the possibility of early migration of radionuclides inside the disposal cell and as far as the host formation. It must be remembered that the calculation represents a pessimistic hypothesis, since it assumes that the medium is both saturated and under gas pressure. These calculations concluded that the transient concerned is too short to produce any effect on radionuclide migration.

It should also be noted that only the interaction between the gases and resaturation is dealt with here. The mechanical effects of the gases in the B cell are addressed in the section 6.2.6.2.

● **In the access structures**

The evaluations of the gas source term show that the access structures, up to the shaft, may be the site of gas production, but at lower levels than in the disposal cells. The evaluations, based on simplified pessimistic conditions, show that the gas emitted by corrosion of the metal bolts should be totally dissolved [61] ; the reinforcements and metal liner plates can also contribute to gas production but are not sufficient to create pressure in the drifts.

The drift is therefore primarily an area of transit for the gases from the disposal cells. It includes relatively porous media (the liner once it is sufficiently degraded, the fractured zone) and the gas spreads quickly through it, but to continue further must cross the seals. It may become desaturated as it crosses the fractured zone and/or liner, without any impact on safety (strictly speaking, desaturation actually slows down hydrolysis of the concrete and stiffens the fractured zone). The progress of the gas into the backfill is more uncertain. If the backfill proved to be insufficiently compacted, it could be partially desaturated by the gas during the first few thousands of years. Taking a pessimistic view, it could be supposed that this desaturation would momentarily alter its mechanical properties. However, if the backfill proved to be desaturated, it is likely that the concrete would also be desaturated ; the drift wall liner would not therefore incur significant degradation and would continue to provide mechanical protection of the rock. This protective function would therefore be maintained at all times.

If ever an inadequate level of backfill saturation were not compensated by the liner, and the unsupported rock suffered irreversible degradation of its damaged zone, the situation would be covered by the sensitivity study under the normal evolution scenario which allows for a fractured zone and microfissured zone with degraded characteristics. It has been seen that this situation has a limited influence on radionuclide transport.

For the reasons set out above concerning the cells, it is improbable that radionuclides could have been transported to the drifts by the time that the gas pressure rise occurs. Any movement of free water by gas pressure in the backfill or EDZ would not concern contaminated water. Transport in the drifts under the effect of pressure from waste cell gases was nevertheless included with the uncertainties in the sensitivity study under the normal evolution scenario. This study showed that the induced radionuclide movement did not result in an increased impact.

● **Summary**

To summarise, at an overall level, the gas production and resaturation kinetics appear to involve uncertainties, linked essentially to the combination of these phenomena, particularly in media whose porosity exposes them to any influence (especially the damaged zone and backfill). Such phenomena cannot significantly desaturate either the sound rock or the seals, which will continue to provide their safety function correctly.

The evolution of the repository system during the resaturation phase, as represented through the SEN, stands up to these uncertainties, which do alter the pessimistic nature of the scenario.

Moreover, the gas production kinetics are linked to the probably overestimated corrosion rate. If the corrosion rate is less than a factor of 5, the calculation still shows hydraulic overpressure in the near-field argillites of the repository but this is more limited. In this case, the hydraulic pressure increases from 5 MPa to approximately 5.03 MPa in the immediate vicinity of the engineered structure. This represents a gradient of the same magnitude as the natural vertical gradient.

However, to manage the uncertainties correctly, and verify the robustness of the repository, we have identified a possible situation of radionuclide transfer during the hydraulic transient and under the effect of gas pressure, already processed in the normal evolution scenario but also to be considered in a « package defect » altered evolution scenario ;

6.2.6 Uncertainties concerning mechanical phenomena

6.2.6.1 Extent and characteristics of the damaged zone

The following sections, using the same principle as previously for the hydraulic phenomena, address the most significant uncertainties in the mechanical field, and also the uncertainties concerning the combined effects of hydraulics and mechanics. Mechanical behaviour is addressed for the repository as a whole and also more specifically for the host rock. For engineered structure components, mechanical behaviour is very dependent on chemical evolution and is only addressed briefly here. The interaction of chemical and mechanical phenomena is addressed in sections 6.2.8 and following, and thermomechanical interaction in section 6.2.11.3.

- **Initial damage**

For the Dossier 2005, the behaviour of argillites during excavation, and their immediate evolution, are evaluated partly by modelling and partly by observations made during excavation of shafts and in experimental drifts of the Meuse-Haute-Marne laboratory. Experiments are continuing in the underground laboratory structures to obtain additional data that will support the final stages of the evaluations. They will be used to adjust dimensioning of the ground supports and liner, and to evaluate more accurately the extent and permeability of zones that may develop fractures and microfissures during excavation, and their evolution over time. In particular, these evaluations will be used to confirm or fine-tune the models, and if necessary revise seal dimensioning according to the extent of the various zones. They will also identify the conditions for installing, and if necessary dismantling, the ground support.

Currently, the major mechanisms controlling the mechanical behaviour of the repository (initial damage to the rock, expansion of swelling clay materials and corrosion products, argillite creep, ultimate return to equilibrium) have been clearly identified. The uncertainties concern the amplitude and kinetics of each phenomenon, and therefore more especially the models used to predict them. A prime example is the evaluation of the extent of the EDZ, which largely depends on the choice of rheological model for the argillites. To this must be added possible scale effects that arise from transposing samples of a few centimetres. Possible effects accompanying the progress of the excavation may also be added.

From the in situ observations of the shaft and the first experimental drifts at levels -445 and -490m in the Meuse/Haute-Marne underground laboratory, certain phenomena and their amplitudes can already be specified, taking account of the shape of the engineered structures (horseshoe section) and the excavation techniques (use of the rock breaker in particular).

The observations of the first experimental drifts at level -490 m, the most clayey level, show the existence of a fractured zone by deconfinement, the maximum extent of which in relation to the radius is approximately 0.2 R, the same order of magnitude as that given by the 2D simulations (0.1 R). The observations of these first experimental drifts also show that shear fractures at an oblique angle to the drift centre line may appear at the front of the excavation face. Being intersected by the drift as it progresses, only the ends of these fractures persist at the drift walls, forming a « chevron » network extending to close to the microfissured zone (Figure 6.2-9). The conditions under which the shear fractures appear, and their geometry, seem to depend on the excavation conditions (speed and regularity of excavation and fitting of wall supports, and support of the excavation face). The formation of these fractures is well reproduced by 3D numerical simulations of the excavation of the drifts taking account of the excavation front and its progress. The water permeability levels of these shear fractures, at the drift wall, are of the same order of magnitude as those measured in the fractured zone of the argillite by deconfinement. These measurements match the permeability value adopted for the fractured argillites, 5.10^{-9} m/s. Beyond the fractured zone, the permeability values measured remain below 10^{-12} m/s which is less than the estimated permeability value for the microfissured argillite (5.10^{-11} m/s).

In the upper, more calcareous layers of the Callovo-Oxfordian argillite (geomechanical zone A), the observations made and measurements taken in the shafts of the underground research laboratory and in the drift at -445 m show that no fractures are created and that the thickness of the microfissured zone is of the order of 0.1 to 0.2 times the radius of the structure depending on its orientation. This extent conforms to that estimated by the 2D simulations. The permeability values in proximity to the shaft are modified very little and, in all cases, are less than 10^{-11} m/s, which is lower than the permeability estimated for microfissured argillite.

The initial EDZ observed on the first structures of the Meuse/Haute-Marne laboratory in geomechanical zone C is consistent with the conventional representation of the EDZ, and may therefore be assimilated to two successive zones as represented in Figure 6.2-9. This conceptual model is not undermined by a possible shear effect. In geomechanical zone A, the initial EDZ observed is consistent with the simulations and can therefore be assimilated to a single microfissured zone.

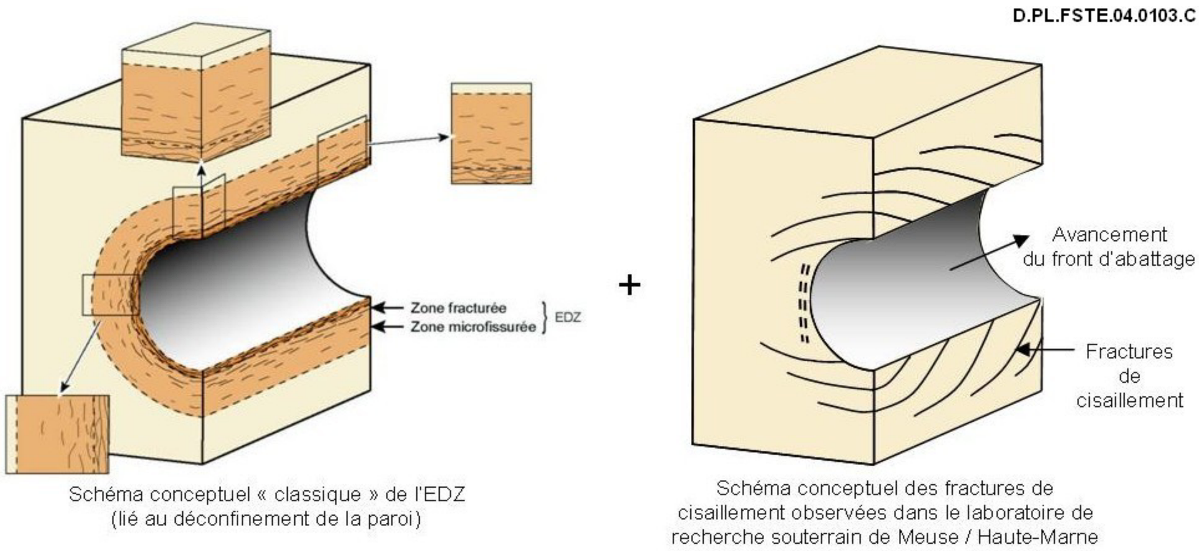


Figure 6.2-9 *Conceptual diagram of the EDZ linked to the deconfinement of the wall and conceptual diagram of the shear fractures observed in the Meuse/Haute-Marne underground laboratory*

These uncertainties are covered in the SEN by using pessimistic extension values, based on a rheological behaviour model from the international MODEX-REP programme [17]. This model is deemed to be the most « phenomenological » but is linked to pessimistic parameter sets.

It is important to note that to allow for the impossibility of determining the repository depth until its position on the transposition zone is known, the SEN evaluation uses the value of 630 metres, the maximum possible depth on the transposition zone, for the EDZ extent model (see chapter 5). This deliberately pessimistic choice (which is only representative of a limited part of the transposition zone) assists in covering all forms of uncertainty concerning the extent of the EDZ by basing the evaluation of fractured zone and microfissured zone extent on values representing the maximum expected values, which can only occur in a limited area.

- **Deferred behaviour of the damaged zone**

The long-term behaviour of the argillites is described in [17]. It may be marked by a creep process starting when the argillites become sufficiently saturated. The long-term mechanical evolution of the damaged zone is a process with very slow kinetics, which accompanies the evolution of materials in the vicinity. In the standard section of the drift this concerns the concrete liner and backfill. At the seal, the slow creep of the argillite is accompanied by saturation swelling of the bentonite.

The short-term evaluation model of the damaged zone, at mechanical unloading, does not incorporate support of the rock and is based on the hypothesis of total loss of confinement, ignoring the presence of the ground support and of the liner gradually installed during excavation. After resaturation and loss of the mechanical function of the liner, and depending on the swelling of the bentonite and very slow creep of the argillites, it is supposed that there is no further evolution of the damaged zone. This hypothesis implies that :

- the self-healing effects of the argillites are ignored, although they are possible, and even to be expected in the context. This is a pessimistic view of argillite evolution. It is assumed that the long-term damaged zone is the same as the zone developed during excavation ;
- the restricted evolution of the argillites is taken into account. In the standard drift section, they are supported by the liner for as long as it can provide a mechanical function. In the long term, the backfill takes over this role. At the seals, the presence of bentonite, swelling as it becomes resaturated, accompanies the evolution of the EDZ. In the B waste disposal cells, the EDZ is in contact with the liner concrete. In the C waste and spent fuel disposal cells, it is adjacent to either the clay engineered barrier or the metal lining (in the case of C disposal cells in the reference design).

These generic hypotheses included in the SEN, cover any uncertainties concerning :

- the scale of the self-healing phenomenon, which is very probable in argillites and has already been observed but not evaluated in detail ;
- the short-term evolution of the EDZ and the techniques implemented to support it.

On the other hand, the long-term evolution of the damaged zone supposes that the materials in contact with it provide support. Each uncertainty concerning the various possible situations will be now addressed in turn.

It should be noted that close to the C waste and spent fuel cells primarily, the EDZ is subjected to thermomechanical actions during the first few centuries of its evolution. These are addressed in the section 6.2.11.3, with all the thermomechanical effects.

Mechanical behaviour in contact with concrete structures

The evolution of the concrete is described as a succession of chemical states from « sound » concrete to « altered » concrete to « degraded » concrete (see section 6.2.8.2). In « degraded » state the concrete is conservatively considered to have lost its mechanical properties. Taking the conditions of the medium into account, the evaluations show a slow, concentric degradation of the liner and retaining plugs, from the outside of the block towards the inside. The loss of mechanical strength follows this pattern, with a gradual loss around the outside, while the core of the block keeps its properties (no collapse). This evolution is shown in Figure 6.2-10.

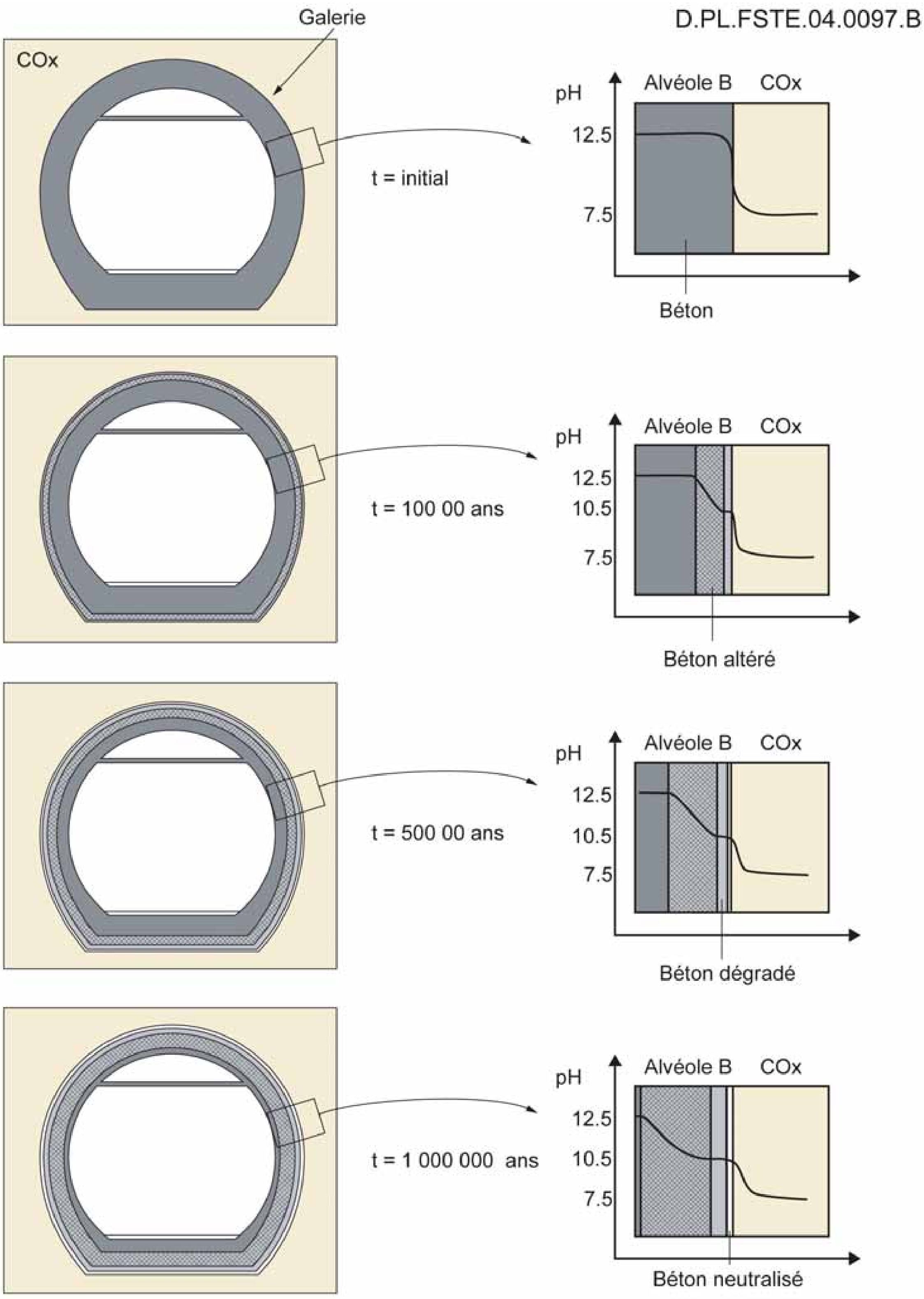


Figure 6.2-10 Diagrammatic representation of the progression of hydrolysis of cementitious materials on the drift wall.

Load transfer follows an identical pattern in the Callovo-Oxfordian, moving gradually to the core of the block as degradation increases. It begins from the start of the operating phase. However, the liner seems to be generously dimensioned for periods of several centuries or more [22]. Reduction of uncertainty on the rock creep rates would make it possible at any later phase of the project to specify conditions for evaluating this period and optimising liner dimensioning.

After degradation of the concrete, the evolution of the rock is restricted by the presence of backfill in the drifts and the degraded packages in the B waste disposal cells. In the first case, only unexpected behaviour of the backfill could lead to unpredicted evolution of the EDZ. In the second case, control of the voids inside the cell prevents secondary damage in the rock. These points depend on the quality of backfill and package emplacement, which will be addressed in the discussion of technological uncertainties (section 6.2.12).

If despite everything, the concrete is degraded more quickly than expected, without any compensation by the backfill, confinement of the EDZ could be temporarily lost and its permeability or extent could increase. This situation is improbable, given that it assumes both high saturation of the liner (sufficient to degrade it) and low saturation of the backfill. This also makes it difficult to quantify. It has been covered by a sensitivity study under the SEN, assuming degraded permeability for the fractured zone (equivalent to 10^{-6} m/s, i.e. the permeability of a sand, and 5.10^{-9} m/s for the microfissured zone less directly exposed).

Mechanical behaviour in contact with the bentonite structures

Rehydration of the bentonite in the seals causes it to swell. After resaturation, the clay core exerts pressure on the argillite. This radial swelling pressure helps to limit convergence. Loss of confinement in the rock could therefore only occur in a situation of extreme degradation of the clay core preventing it from developing swelling pressure. Given the swelling properties of bentonite, this situation is very improbable, even supposing that the seal is disturbed by unfavourable chemical conditions (this point will be addressed in section 6.2.8.1). However, it was decided to include poor bentonite swelling conditions among the situations that could lead to the « seal defect » SEA. In such cases, the support defect in the rock is represented by secondary damage around all the seals (taking into account unfavourable parameters for the fractured zone and damaged zone).

Although excessive seal swelling could damage the rock, the pressures required have been estimated as very high (greater than 12 or 13 MPa). The properties of bentonite prevent a phenomenon of this kind. The uncertainties are very low in this matter. Nevertheless, if such a situation did occur, it would result in a continuous fractured zone around the seals. Despite the improbability of a situation of this kind, with incorrect dimensioning of the seals, it was nevertheless retained for the definition of the « seal fault » SEA. It results in secondary damage to the EDZ around all the seals.

Contact with the metal elements (thermal waste disposal cells)

In the special case of C waste cells, the damaged zone is in direct contact with the metal lining, which in turn is close to the container. In spent fuel cells with clay engineered barrier, there is also a metal liner in contact with the argillite. The corrosion products of these materials are expansive and could develop pressure on the geological medium. The expected expansion coefficients for these types of product, and the residual space inside the cell, are in principle sufficient to prevent unfavourable mechanical action.

In all events, all unpredicted effects are covered by the sensitivity calculation carried out for the SEN with pessimistic values of fractured zone permeability in the standard cell section (10^{-6} m/s, giving the fractured zone the value of a sand). The metal lining close to the plug is designed to be removed before the plug is installed. In the worst case, the swelling pressure of the corrosion products could only damage the standard section and not the contact zone between the plug and the rock.

6.2.6.2 Uncertainties in accounting for the mechanical effect of gases

The question of what is going to happen for gases was addressed in section 6.2.5.2 with regard to resaturation kinetics. The question here concerns the mechanical action of the gases, mainly corrosion-generated hydrogen, on the behaviour of the various repository components.

As was seen in section 6.2.5.2, the media surrounding the repository cells (the cell itself, the plug, the adjoining drift, the geological medium) may become channels for spreading the gases produced in the repository. Each will now be considered in turn, taking type of material.

- **Inside Argillites and swelling clay**

For argillites the pressure level above which the gases are likely to interact with the rock has been evaluated by analogy with measurements made by Nagra on clays at Opalinus in Switzerland and also by analogy with bentonite. The value is around 9 MPa. Above this pressure, the gas can act on the porosity of the rock and expand it locally, creating the equivalent of microfissures which can only remain open while the pressure is maintained. By their very nature, these are reversible phenomena. On the other hand, fracturing by gases at greater pressures could lead to irreversible changes to the rock structure. Experiments have been conducted in the Meuse / Haute-Marne laboratory, in which gas was continuously injected into a bore-hole until reaching the pressure at which the geological medium is fractured. The value measured is 12 MPa. Injection lasted for several days. After the experiment, the rock regained its initial water permeability due to creep in the rock.

It seems therefore that the mechanical effects of gases on rock, including both microfissuring and fracturing, are reversible in terms of the hydraulic properties of the rock. In addition, gas production evaluations, even when conducted under pessimistic conditions, show that it is impossible to exceed 9 MPa, thus ruling out the possibility of fracturing of the medium by gases. With less pessimistic gas production kinetics, additional evaluations, still to be completed, may confirm that any form of microfissuring can be ruled out.

The behaviour of the bentonite under the effect of gases is similar to that described above for argillites, and the same conclusions can be drawn. The reversibility of the mechanical action of gases is also assisted by the swelling properties of this clay.

Argillites and bentonite are not therefore subject to any irreversible mechanical effect linked to gases. The SEN, which does not take such effects into account, is representative on this point. If however, for the purposes of managing residual uncertainties, it is supposed that this experimental finding can be called into question, the effects would be as follows :

- on the argillites, the zones that are theoretically the most fragile (fractured and microfissured zones) could undergo secondary damage. This results in degradation of their hydraulic properties. This situation has been considered in a sensitivity study under the SEN ;
- if even the sound argillite was affected, which seems improbable, the hypothesis that microfissuring is irreversible suggests that a continuous damaged zone would be created in the near field of the disposal cells, principally spent fuel disposal cells. The damage would cause a pressure drop which would prevent significant spread of gas in the rock. However, a local fractured zone would be created around the cell or could even form around the plug. This would be equivalent to bypassing the plug. Given that the pressures are lower, it is even more improbable that this kind of effect could spread as far as the drifts and access shafts. Nevertheless, a situation where all the seals are by-passed by a continuous fractured zone, represented for a drift seal by Figure 6.2-11 has been considered in a « seal failure » SEA and covers this situation ;
- an irreversible mechanical effect on the bentonite would open paths of least resistance inside the seal - and the clay engineered barrier of the spent fuel disposal cells - in the form of fissures. From the hydraulic point of view, it makes no difference if such a route is in the core of the clay massif or at the rock interface ; in either case the situation is equivalent to an interface defect between the seal and the argillites. If the seal is more severely damaged (which is highly improbable), its permeability would be degraded. A situation of this kind with by-passing of inefficient seals combined with degradation of their overall permeability is considered in the « seal failure » SEA.

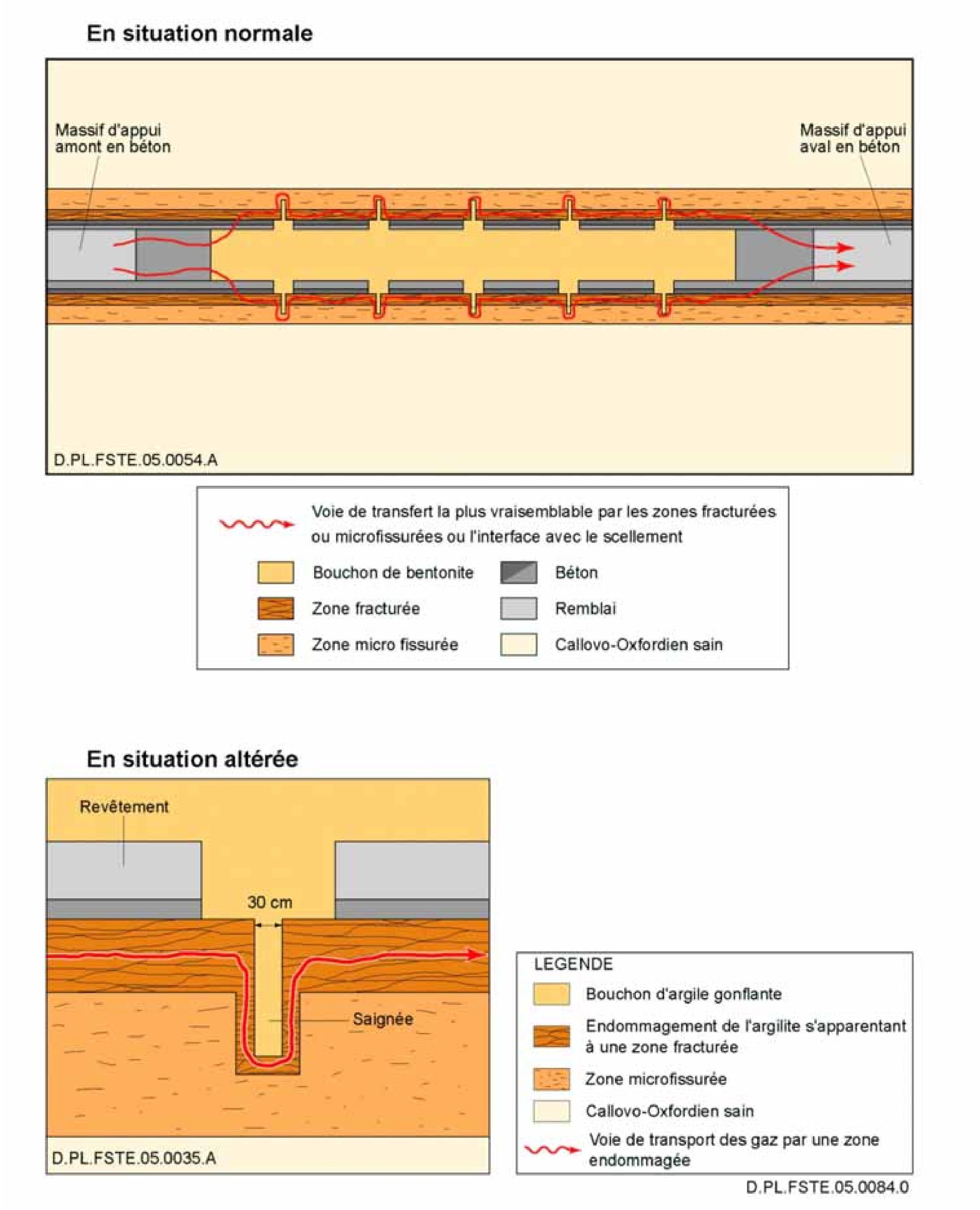


Figure 6.2-11 Hypothetical damage situation of a hydraulic cut-off of the seal by gases

- **Effects on the other disposal cell components**

The mechanical dimensioning of the vitrified waste and spent fuel containers is sufficient for them to withstand the predicted pressure ranges.

For the concrete overpacking of B waste, the pressure rise due to the production of gas inside the packages is managed operationally by using packages that are not hydrogen-tight for waste likely to release hydrogen (for example, those containing waste subject to radiolysis). Eventually, at the post-closure phase, a pressure rise could occur in metal waste containers through internal corrosion of the waste. This phenomenon would not necessary result in damage to the package before 10,000 years, when it is expected to lose its mechanical strength. However, this has not been investigated in detailed studies, and to cover all uncertainty relating to the mechanical strength of B waste overpacking, no hydraulic property linked to its mechanical strength is assigned to it in the SEN.

6.2.6.3 Summary of uncertainties relating to mechanics

Mechanics-related uncertainties concerning the short-term extent, characteristics and evolution over one million years of the damaged zone are covered in the SEN by choosing a « phenomenological » model and adopting conservative parameters, especially maximum depth for the repository. Any residual uncertainties concerning the extent of the initial damaged zone would be covered by the design and the possibility of interrupting the EDZ by a hydraulic cut-off at lengths greater than those adopted in the SEN or by specific treatment of the damaged surface in the shaft.

Uncertainties relating to the long-term evolution model are covered by the fact that the rock is permanently supported (by the liner, backfill and seals). Under these conditions, unfavourable evolution of the fractured zone would require re-examination of uncertainty in the support capability of one of these components. Section 6.2.5.2 for example, considered the case of inadequate support by desaturated backfill ; other cases will appear when chemistry-related uncertainties are analysed.

However, unfavourable evolution of the damaged zone in the standard section of a drift or shaft does not lead to a situation significantly different from that considered in the SEN, as sensitivity studies on damaged zone permeability have shown (see chapter 5). Only degradation of the microfissured zone at the seals, beyond the hydraulic cut-offs, can lead to short-circuiting them. This case is considered in sensitivity studies under the SEN and an additional study of a case of a continuous fractured zone along all the structures will be conducted under the « seal failure » SEA.

Finally, in principle the possible mechanical effect of gases on the damaged zone is reversible, and in all events is covered by the sensitivity study which supposes degradation of the permeability of the fractured zone and microfissured zone. Only in an unrealistic case, which corresponds to neither the experimental findings nor the expected pressure ranges, could there be overfracturing of the rock or seals, as covered in the « seal failure » SEA.

6.2.7 Uncertainties relating to waste matrix release models

The following sections address questions relating to chemical evolution inside the repository, in the wide sense : behaviour of the waste packages, behaviour of the materials, and disturbance of the host formation induced by the materials.

6.2.7.1 Release models and behaviour of B waste packages

It has been seen that, with no suitable model or findings available, the release models adopted in the SEN for B waste are in many cases equivalent to instantaneous release (chapter 5). These sections will only address packages with a non-labile release mode, the uncertainties relating to release by other types of waste being covered in the SEN.

- **Metallic waste**

To determine a release rate for metallic waste, the activated metal parts are assigned the lowest dimensions encountered for this type of waste (for example reactor-irradiated structures in B1 packages) and the most representative corrosion rate from those measured in an anoxic situation. These rates therefore correspond to packages that are isolated from oxidation for the period during which the cells are exposed to it (i.e. during the cell operating and ventilation phase). This is achieved through effective sealing of the packages.

As a precautionary measure, labile release was tested in sensitivity studies under the SEN to provide fuller cover of the uncertainties on this point.

- **Bituminous waste**

The ability of bitumen to immobilise radionuclides is expressed by a release model. Two models, COLONBO 2 and COLONBO 3, have been developed to describe radionuclide release following water take-up by soluble salts (the phenomenology underpinning them is shown in Figure 6.2-12). The

SEN takes the COLONBO 3 model as reference, and a pessimistic model based only on water take-up for sensitivity studies.

The COLONBO 3 model is based on the property of bitumen to control water take-up and takes into account the limiting of the transfer of water and ions from soluble salts in the permeable zone this creates, without taking the limiting of radionuclide transfer into account : these are assumed to be instantaneously solubilised and released outside the embedded waste.

The degradation process leads to a release proportional to the square root of time based on the activity of water saturated with the soluble salts of the bitumen and the coefficient of water diffusion in the permeable zone. The values adopted are taken from specific measurements.

COLONBO 3 is also based on a certain number of hypotheses, which are explained in detail in the reference baseline for release models [20], the most important being :

- the soluble salt content in the embedded material, which must be between 2 and 22 % ;
- the initial dimensions of the embedded material, later swelling due to water take-up being processed by the model ;
- the temperature of the bitumen throughout its lifetime, which must not significantly exceed 30°C, as the salt sedimentation rate remains negligible (less than 7 µm/year) beneath this temperature ;
- the pH, which must be between 7 and 12.5 : this validity range covers the pH range imposed by the concrete packaging ;
- the geometry of the bitumen. The models are only applicable if swelling by water take-up is unrestricted. However, this requirement cannot be met as the expected swelling for B2.1 waste packages is greater than the free volume available in the waste packages. The B2.2 waste package raises the same problem ; in this case it is planned to fill the entire interstitial void between the drum and the overdrum. According to a CEA study, still to be confirmed, the existence of a swelling constraint should limit the water take-up kinetics and hence the release of radionuclides.

The COLONBO 3 model, on the other hand, covers the possibility of fissuring in the bitumen.

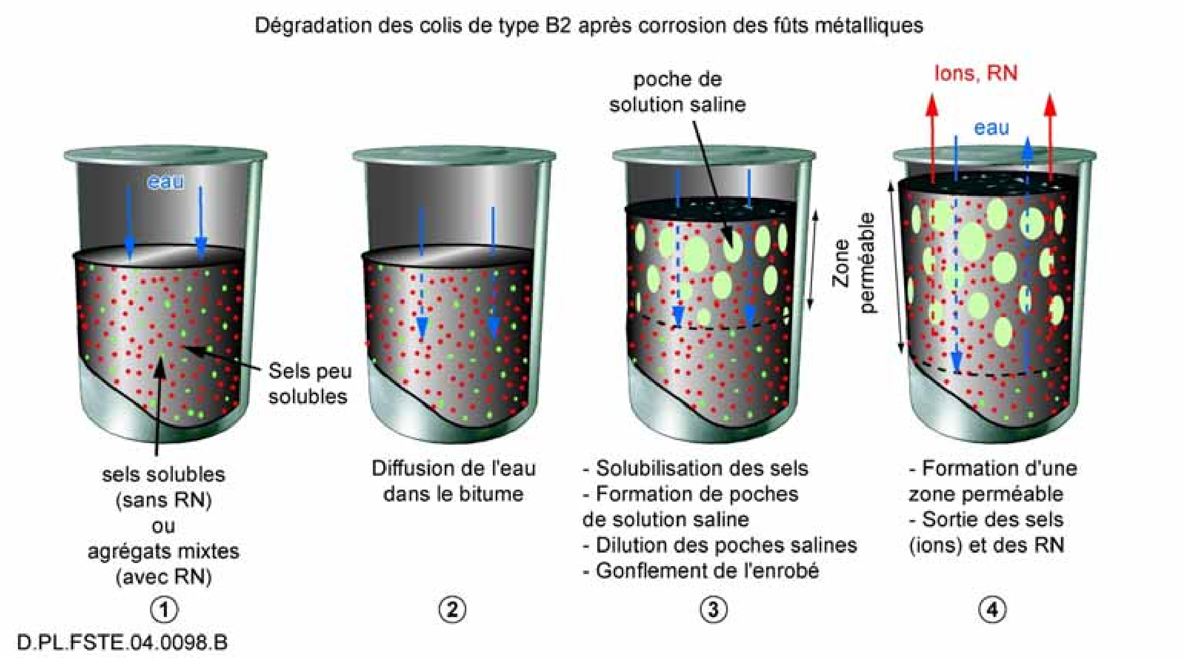


Figure 6.2-12 Schematic diagram of bituminised sludge release models

Given these hypotheses, COLONBO 3 is theoretically applicable :

- to all bitumen production from effluent treatment in the La Hague STE3 station, produced under quality assurance and having characteristics comparable to the application conditions of the model (more than 10,000 drums) ;
- to all future bitumen production resulting from effluent treatment at the STE2 station. The higher radioactivity concentration, compared with value adopted for STE3 bitumen, could induce a higher integrated dose and different mechanical behaviour, but the COLONBO 3 model is not greatly affected by these differences (40,000 drums) ;
- a large part of the old Marcoule bitumen, although a fraction could, because of its specific salt content or unfavourable storage conditions, be outside the model context.

The model can therefore account for most of the inventory (see section 2.1.4.1.) and is representative as a global performance calculation. It cannot account for release from some types of bituminised waste drums, but these uncertainties are covered by the sensitivity study using the « pessimistic » model based only on water take-up. This model is also unable to account for any deformation of the bitumen in the event of its geometry becoming difficult to control. The uncertainties concerning the long-term behaviour of bitumen are therefore well covered by the SEN.

Bitumens may also release complexing agents, mainly oxalates, but also tributyl phosphate, which may have been conditioned with the matrix (in the case of certain old bitumens from the STEL treatment station at Marcoule) or various other degradation products. Uncertainties exists as to the nature and quantities of complexing agents released over the long term, but for organic molecules, the effect has been shown to be limited in an alkaline context. In addition, the « filter the colloids » function (see chapter 3) prevents any complexes formed to leave the near-field. The uncertainties relating to transport by colloids are discussed in section 6.2.10

Bitumens may also release organic acids, sulphates and nitrates. The effects of induced sulphate attack on the concrete give rise to uncertainties and could have a long-term influence on the strength of the overpacking. Acids could have a similar effect, although in principle they are buffered by the surrounding medium. The packaging may also be subject to mechanical action due to bitumen swelling after water take-up. Primarily for these reasons, the overpacking must not be assigned any safety function requiring long-term mechanical resistance.

The nitrates, combined with micro-organisms, could locally modify the redox potential inside the disposal cell, thereby influencing the solubility of certain radionuclides. This phenomenon is likely to be very slight as the alkaline environment is unfavourable to micro-organism development and would remain confined to the immediate vicinity of the bitumen, the main source of nutrients. This local effect is negligible at disposal cell level, given the quantities of concrete installed in the disposal cell. The sensitivity study under the SEN investigating the geochemical parameters in the B2 waste cells (including concrete) provides very pessimistic cover of this uncertainty and has demonstrated that it has little effect on the impact[86].

6.2.7.2 Source term and long-term behaviour of vitrified waste

The models used to calculate the performance vary according to the vitrified waste families and were presented in chapter 5.

The long-term behaviour of type C1, C2, C3 and C4 glasses could be marked by a change in the long-term state owing to the gradual saturation of the medium with silica, which according to the CEA's observations induces a transition from conditions in which there is dissolution at the initial rate « V_0.S » to leaching at a residual rate « V_r ». The causes and conditions of the transition from one state to the next and the fact that this phenomenon is applicable to other types of glasses than those which were the subject of the bulk of the studies (the « C1 » glasses of the waste inventory model, that is the glasses currently produced in the R7/T7 workshops) are not as yet completely established. The observations do however agree that once the silica concentration reaches equilibrium between the glass and its immediate environment, and provided that the glass is placed in diffusive conditions, the

86 Note that this calculation was not applied to B2 waste cells, but to BIx cells, that are the major contributions to the impact. The conclusions remain however applicable to all types of B waste cells.

dissolution rate drops. The explanation chosen by the CEA involves the formation of a protective gel which is integral with the surface of the glass. Andra uses reference model « V_0.S → V_r » for the type C1 glasses and for those produced beyond C1 (for which it is reasonable to assume that their quality will be at least equivalent) ; in order to cover uncertainties in the model, SEN calculations adopted a « V_0.S » model as the reference for the other glasses and in sensitivity studies for the types C1, C2, C3 and C4.

In the C disposal cells, anoxic corrosion of the containers, liners and buffers in the immediate vicinity of the packages produces oxides or hydroxides (such as magnetite or even siderite) with no silica in their structure. They are thus unlikely to disrupt the silica equilibrium of the environment. However, in order to cover all uncertainties on this point, the « V_0.S →V_r » as applied in a SEN, assumes that the corrosion products have a high silica sorption capacity (equal to that of goethite), which delays the moment at which the rate « V_r » can apply. This choice covers the uncertainty regarding the chemical environment of the glass.

Furthermore, the initial dissolution rate, and its dependency on the glass fracturing rate, the temperature and the pH, can lead to uncertainty.

The value of « V_0 » is chosen from among the representative values measured by the CEA for the various types of existing waste, according to the experimental results. For C2 glasses, and possibly C3 and C4, long-term behaviour assessment programmes could be conducted once the actual packages are available.

The uncertainty over the pH dependency of dissolution is taken into account in the SEN, by conservatively assuming a pH of 9. Given the distance from any source of alkaline disturbance in a normal situation (owing to the buffer role played by the plug), this choice is conservative. It in particular takes account of the uncertainties over the composition of the Callovo-Oxfordian water as well as the, albeit minor, effects on the pH of the materials contained in the cell (metals, bentonite).

The fracturing rate cannot be directly measured on the package, as a result of which various uncertainty origins can be identified :

- the variability of the initial internal fracturing of the glass, which is hard to characterise ;
- the glass handling conditions during transport and emplacement, which can involve either shocks or changes to the internal stress field. It is unlikely that they would lead to damage of the vitreous matrix.

This uncertainty is covered in a SEN by an analysis of the sensitivity to the fracturing rate of C1 glasses, by adopting the highest value (40) measured by the CEA. This value corresponds to random glass samples and is a reasonable upper bound limit.

6.2.7.3 Source term and long-term behaviour of spent fuel

Standard UOx or MOX spent fuel from EDF's PWR nuclear power plant fleet must be differentiated from that from research and defence activities, which is more diverse.

- **Spent fuel from the PWR fleet**

The release models proposed are based on the behaviour of the various materials. Generically, we observe that the behavioural differences from one fuel to the next can be due :

- to the nature of the matrix and the fuel, which requires that UOx and MOX fuels be considered differently ;
- the burnup rate, which requires that this parameter be considered in the studies ;
- at a more secondary level, to the number of cycles in the reactor. The influence of this parameter is less well defined and at this stage in the studies it is covered by applying a uniform burnup fraction (the highest) to the release models for all the fuels.

The study rate of release of radioactivity contained in the spent fuels is based on two types of phenomena : those linked to corrosion of the metal parts of the fuel, which can be activated during the time in the reactor, and dissolution of the fuel matrix.

With regard to the metal parts, the uncertainties concern determining the rate of corrosion. Generally speaking, this falls within the context of control of corrosion within the disposal cells (estimation of the corrosion rate, chemical homogeneity within the disposal cells), in this particular situation of irradiating conditions. Conservative corrosion rates, which incorporate the experimental results, were proposed for the steel zircaloy and inconel parts. The transient effect of radiolysis owing to irradiation from the fuel is taken into account. These rates correspond to reducing conditions. The fuels are protected for at least 10,000 years, or possibly more depending on the effective lifespan of the container. When they are finally exposed to the chemical conditions in the disposal cells, the transition to the reducing state would have been completed a long time previously.

The distribution of activity on the metal parts is another source of uncertainty, particularly insofar as it depends on the irradiation conditions in the reactor's core, which can vary. The choices adopted by Andra consist in considering that the activity is uniformly distributed, which leads to no major bias insofar as the part dimensions are also minimised. When the material behaviour is harder to control (which is the case of the possible zircon layers formed on the cladding), the corresponding source term is assumed to be labile.

The behaviour of the fuel matrix is harder to model and involves a number of phenomena on which Andra follows an approach conforming to international practices, although always attempting to use an overestimation approach. In particular :

- dissolution of the matrix is assessed according to the effects of alpha radiolysis. This phenomenon is considered to be preponderant at this stage in the studies. The analyses conducted internationally would however tend to question this model, which assumes that the oxidising conditions in the immediate vicinity of the spent fuel are permanent, and prefer the use of a model piloted by conventional dissolution of the uranium matrix. This leads to far higher lifetimes ;
- the kinetics of oxidisation of the matrix under the effect of radiation are ignored. We assume that the reactions are instantaneous, which does not reflect experimental data [66] but which is more pessimistic ;
- the hardest parts of the matrix to model (the activity contained in the grain boundaries, the pores and the cracks) are assumed to be labile. This hypothesis corresponds to the best estimate in certain cases (for example for the gap), but is certainly pessimistic for others (for the rim grains in particular) ;
- accelerated dissolution by alpha auto-irradiation (D3AI) is taken into account through a coefficient corroborated by various models. No significant effect is expected from this phenomenon which only concerns a fraction of the light nuclei. To maximise its effects, we applied it to the entire fuel matrix, which led to a significant and pessimistic reduction in the lifetime of the fuel.

The model thus obtained therefore in the end depends primarily on the environmental conditions (pH, temperature). As our knowledge currently stands, in order to validate the model, the pH would have to be lower than 9, which is guaranteed by the concept and the remoteness from all sources of alkaline disturbance (absence of cement-based materials in the cell). The behaviour of the spent fuel with higher pH levels is currently little known ; it should not however be subjected to such conditions owing to the presence of the plug and the clay engineered barrier, which act as a chemical buffer.

After this analysis, it would seem that the release models adopted for the SEN cover the uncertainties through conservative choices. These in particular include using radiolytic dissolution, a hypothesis which would seem to be penalising given the research results. If account is taken of factors inhibiting this reaction (in particular the hydrogen present in the spent fuel environment, which would set reducing conditions) this could lead to adoption of a model based on « conventional » dissolution of the uranium matrix, driven by its solubility. A sensitivity study conducted in the SEN on this subject shows that adopting such a model would provide significant gains on the impact in the normal evolution scenario.

- **The other spent fuels**

The research and defence spent fuels are more varied than those from PWR reactors, both in their physicochemical form and their radiological inventory. In order to cover uncertainties regarding the release model that could be applied to them, they were at this stage of the studies considered to be labile.

6.2.7.4 Summary of uncertainties regarding the waste behaviour models

The release models used in the reference calculation SEN, easily cover the uncertainties for the waste for which the behaviour is uncertain, through voluntarily penalising choices : labile source terms once the chemical nature is inadequately understood, minimisation of the size of the metal waste, exclusion of the theoretically favourable properties of bitumen, conservative model applied to glasses for which the behaviour is least well-known, exclusion of phenomena which help limit releases from the spent fuel.

The validity of these release models only depends on residual uncertainties in certain specific cases. To mention only the main ones, we could say that generally speaking, the share of the inventory corresponding to the older waste raises the question of the applicability of the release models, owing to the lack of precise characterisation of their current state. In the particular case of bituminous waste, the model is not strictly speaking applicable to the entire production of old waste, even if this only represents a marginal fraction of the total inventory. By default, an overestimated « V_0.S » type model is applied to the older glasses.

The sensitivity studies for their part represent an intentionally highly pessimistic view of these same releases : labile releases for metal B waste, extremely simplified release model for bitumens, based only on their water absorption capacity to the exclusion of all other phenomena, application of constant and conservative rates to all of the vitrified waste.

The analysis does not directly reveal the cause of failure of the functions, nor any situations going beyond what is considered in the SEN. However, certain release models are only valid subject to control of the hydraulic and chemical conditions in the disposal cells. If, owing to another uncertainty, these conditions were to change, the source term would have to be adapted. In particular :

- in any situation which could lead to advection conditions in the C waste disposal cells, the source term would have to be adapted accordingly, using « V_0.S » conditions. For this reason, and even if the conditions in the disposal cells were to remain diffusive, this source term will, as a precautionary measure, be adopted for the cells affected, in a « seal failure » scenario and in a « bore-hole » scenario ;
- in situations in which the pH can rise beyond the validity scope of the studies (therefore for pH values higher than 9) - again in the C waste disposal cells, the source term is adapted, with adoption of a higher term.

6.2.8 Uncertainties over the long-term evolution of materials

Here we deal first with the evolution of exogenous materials once they are in place within the host rock. If this can have an influence on their evolution, the phenomenon in question is mentioned. The disturbances induced in return by the rock materials are mentioned afterwards.

6.2.8.1 Long-term behaviour of engineered bentonite structures

The mechanical behaviour of bentonite is characterised by its plasticity and its ability to swell when resaturated. It is this capacity that is used to anchor the seals and fill in any gaps between the bricks or rings of bentonite installed. There is no identified uncertainty over the strength of the bentonite elements as such, in that they comprise uniform assemblies of a simple, well-characterised material. The uncertainties could however concern the interaction with the water in the environment.

The variability in the composition of the Callovo-Oxfordian water must be taken into account when examining the behaviour of the seals. It can come from the natural variability of the site, or from modifications resulting from the operational phase of the repository prior to closure, more particularly the oxidising phase.

In particular the potassium contained in the pore water could encourage the transformation of smectite into illite, a non-swelling mineral that can appreciably reduce the swelling capacity of the seal. This process is unlikely and the geochemical composition assessments of the rock in all cases reveal low potassium concentrations in the Callovo-Oxfordian pore water. In the case of the shaft seal, the upper part of which could be resaturated, at least partially by Oxfordian water, the length of the bentonite core is enough to limit the influence of this water to an insignificant thickness.

The effects of the calcium are limited to modification of sodic smectite into calcic smectite. Although calcic smectite is still a swelling mineral, this cationic exchange can however reduce its swelling potential. We can imagine that during resaturation a significant part of the smectites could become calcic. The swelling pressure developed by the clay rock could then be slightly degraded, although not enough to alter the minimum hydraulic performance required [69].

However, to take account of these uncertainties, we can envisage two failure situations leading to the seals being ineffective :

- a first one which consists in envisaging that the influence of the Oxfordian water on the shaft seal is greater than expected. This unrealistic situation could lead to non-swelling of the shaft seal. In such a case, the seal would no longer provide contact with the rock of the shaft wall. In a « seal failure » SEA, we consider bypassing of the shaft seal by a fictitious fractured zone, for which we degrade the performance sensitivity. A calculation such as this covers the situation envisaged here ;
- a second situation could result from poor anticipation of the influence of the calcic water on the seals. In such a highly unlikely eventuality, all the repository seals could fail to swell adequately. This situation is covered in a « seal failure » SEA, which considers incorrect behaviour of these structures and their bypass by a continuous fractured zone.

Resaturation of the structures by Callovo-Oxfordian water can be disturbed over a period of time by possible couplings, with gradual degradation of the repository components. In particular, the fluids resaturating the bentonite in particular may have passed through a disturbed Callovo-Oxfordian zone, concrete and metal elements. These alkaline, ferrous or possibly silica-induced disturbances are also potentially accelerated by temperature for those seals close to vitrified waste and spent fuel disposal cells. These effects are mentioned jointly with the disturbances concerning argillites, insofar as they are of the same nature. Refer to sections 6.2.8.4 and 6.2.8.5

6.2.8.2 Long-term behaviour of concrete

In the B disposal cells, the concrete maintains an adequate pH range and reducing conditions which are favourable to slow corrosion of the metallic waste. There is little doubt of the concrete's ability to maintain a pH of higher than 10, given the quantities put in place and the relatively non-aggressive nature of the components surrounding it. The Callovo-Oxfordian has a pH close to neutral. Only certain B waste, such as bitumens, could release organic acids as it degrades. The quantities are small and do not compromise the overall pH within the disposal cell.

Concrete also has an interim function to provide the structures with mechanical support, in association with its chemical behaviour. Andra has developed a model of the successive states of the concrete, to describe its chemical and mechanical evolution [67]. This model takes account primarily of the hydrolysis of concrete, leading to successive dissolution of the various constituent minerals : first of all alkaline and alkaline earth oxides, then portlandite ($Ca(OH)_2$) then hydrated calcium silicates (CSH). Figure 6.2-13 shows the evolution of the pH with the various phases.

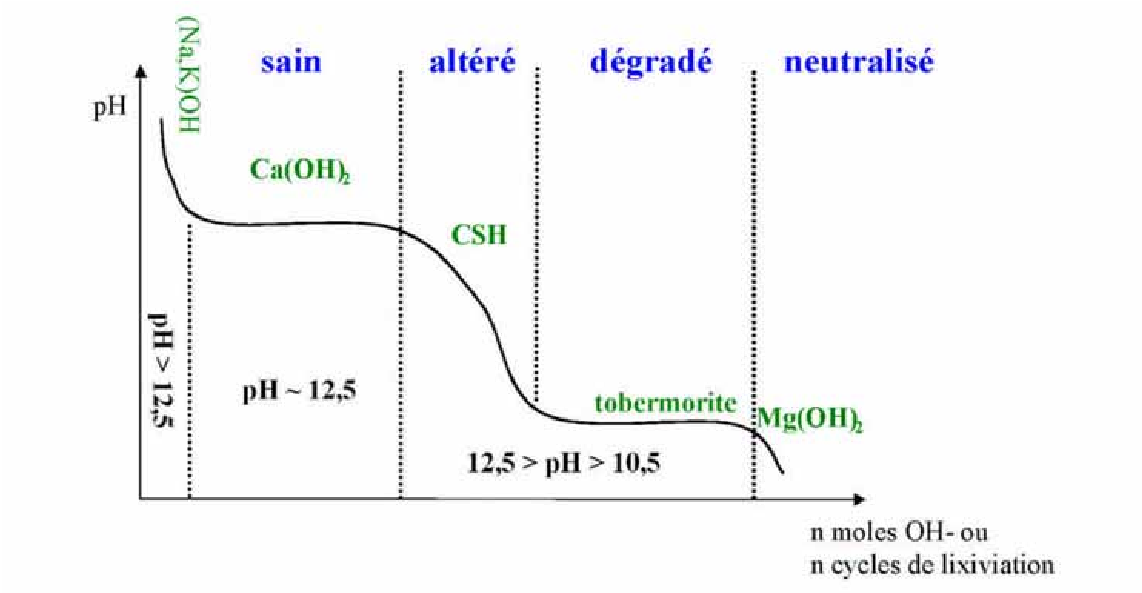


Figure 6.2-13 Evolution of the chemical state of cement-based materials

Uncertainties persist concerning the link between the mechanical states of the concrete and these various chemical states, owing to the difficulty with extrapolating the models produced on the basic of short-duration tests to long time-scales.

The concepts are not therefore based on the long-term mechanical strength of the concrete in those configurations in which it cannot be guaranteed and no reference function is assigned to the concrete which requires a particular strength capacity. It should however be noted that the mechanical strength of the concrete retaining plugs is useful in confining the seals while they are resaturating (no more than a few thousand years, which poses no problems).

Similarly, the strength of the concrete coating supports the rock before other components (backfill in the drifts, the filling and the underlying packages in the B waste cells) take over.

In both cases, the uncertainty then concerns the kinetics of concrete degradation, which must be slow enough with respect to the other processes, and the ability of the environment to redistribute and support the loads after loss of the mechanical function. In the particular case of B waste disposal cells, which is the most critical as it does not use swelling of another material to compensate for the loss of concrete, the assessments show that it is possible to rely on long-term ground support strength of about ten thousand years [63]. The slow convergence of the geological medium which would result from a coating failure would be stopped by the mass of materials and compensated for by rock creep, which would in principle tend to close rather than aggravate the cracks. Voids within the B waste disposal cells are therefore minimised.

The overpack for the B waste is designed to obtain the best possible durability, by ensuring a construction that is as stable as possible over time and the least sensitive to damage from external sources. This in particular guarantees the reversibility of the B waste repository for long periods, by guaranteeing the integrity of the package so that it can be removed. The techniques employed for the construction indicate that one could eventually for certain packages achieve highly significant mechanical durability, allowing package integrity for periods of up to 10,000 years (for those packages which could be made gas-tight). This possibility was included as a variant rather than the reference SEN.

In the SEN, concrete is also a transfer channel for radionuclides. The evolution of the transport parameters over a period of time is taken into account in the SEN, using conservative porosity, diffusion and permeability values in order to cover the uncertainties.

Finally, concrete is likely to undergo chemical attack from the geological medium and the waste, in addition to hydrolysis, which is the phenomenon included in the degradation model. A geochemistry that is prejudicial to the concrete could result from a strong but transient rise in the sulphate levels linked to oxidisation of the pyrite in the argillite ; this could accelerate degradation of the retaining plug. This process would have little impact as such concentrations would be transient. A similar effect could be caused by backfill pore solutions, the formulation of which contains a high proportion of reworked and therefore potentially « oxidised » argillite, after storage on the surface. However any loss of mechanical strength would not be expected to be rapid enough to compromise the strength of the concrete on a millennium scale. The aggressive compounds resulting from the waste (sulphates, acids, etc) would be consumed by the concrete placed in proximity to the waste, which is sufficient in quantity to avoid such reactions compromising the stability of the cell. The concrete specifications include resistance to these chemical hazards [67].

None of these uncertainties would be such as to compromise the chosen representation in a SEN. However, for more complete coverage of the uncertainties, one could envisage a situation in which the concrete degradation process is drastically accelerated, in order to completely cover the uncertainties in the evolution model. Such premature degradation could affect the components with a mechanical function :

- the concrete coating, which provide mechanical protection for the EDZ until mechanical equilibrium with the backfill occurs ;
- the seals retaining plug, which play the role of confining them while they resaturate.

Accelerated degradation of the concrete would therefore lead to an altered situation, which in the most pessimistic scenario would entail degradation of the poorly supported damaged zone and ineffective seals which would be poorly confined by a defective retaining plug. This situation is an extremely improbable case but it is however envisaged in the « seal failure » SEA in which all the seals fail to interrupt a damaged zone which is also degraded. This covers both the uncertainties concerning the link between the chemistry of the concrete and its mechanical strength and any chemical hazard problems (in particular from sulphates).

6.2.8.3 Long-term behaviour of metallic elements

The long-term behaviour of the metallic elements (lining, containers, various elements such as reinforcements, bolts, etc.) is dominated by the phenomenon of corrosion. In the short term, this could be influenced by the transient conditions reigning within the repository (redox transient, thermal transient, etc.). It also depends on the chemical homogeneity within the cells. We however begin by looking at the issue of corrosion in a homogeneous, anoxic medium, before looking at possible deviations from this situation.

- **Long-term corrosion models**

Here we mainly deal with spent fuel containers and vitrified waste overpacks plus, as necessary, other metal elements contained in the thermal waste cells.

The corrosion modalities (generalised, pitting, etc.) and the associated rates have been abundantly examined both in France and abroad, through study of radioactive waste disposal or other contexts [19]. In addition, this field contains a large body of experimental work. Archaeological comparisons exist highlighting very slow corrosion rates over periods of up to several thousand years, even in conditions that are generally oxidising.

Generally speaking, for specified physicochemical conditions, the corrosion rates entail few uncertainties. Andra has adopted values (a few microns maximum in an anoxic environment, a few tens of microns maximum in an oxidising environment) which encompass the rates observed.

However, the uncertainty concerns the conditions in which the metal components of the repository will corrode. The design of the repository comprises measures for on the one hand ensuring this homogeneity and on the other for encouraging anoxic conditions, which limit the corrosion rates. The care given to the construction of the cells, the presence of ceramic PARS (see Figure 6.2-14 isolating the containers from the lining (to prevent locally accelerating corrosion by the contact effect) are all favourable points. The buffers added within the cells to manage heat, would be made of an identical metal to that of the containers, to further encourage homogeneity. In addition, the planned arrangement - installation of a sealing cover at the entrance to the vitrified waste and spent fuel disposal cells during the time they are not yet closed as they are closed - limits air renewal in the cell and prevents oxidising conditions from persisting, even during operation. It also limits the renewal of the nitrogen in the air and therefore the formation of nitric acid through radiolysis (which can increase corrosion in the vicinity of the container). The very slight quantities of organic material in the C waste and spent fuel cells means that corrosion induced by micro-organisms can be considered to be of little importance.

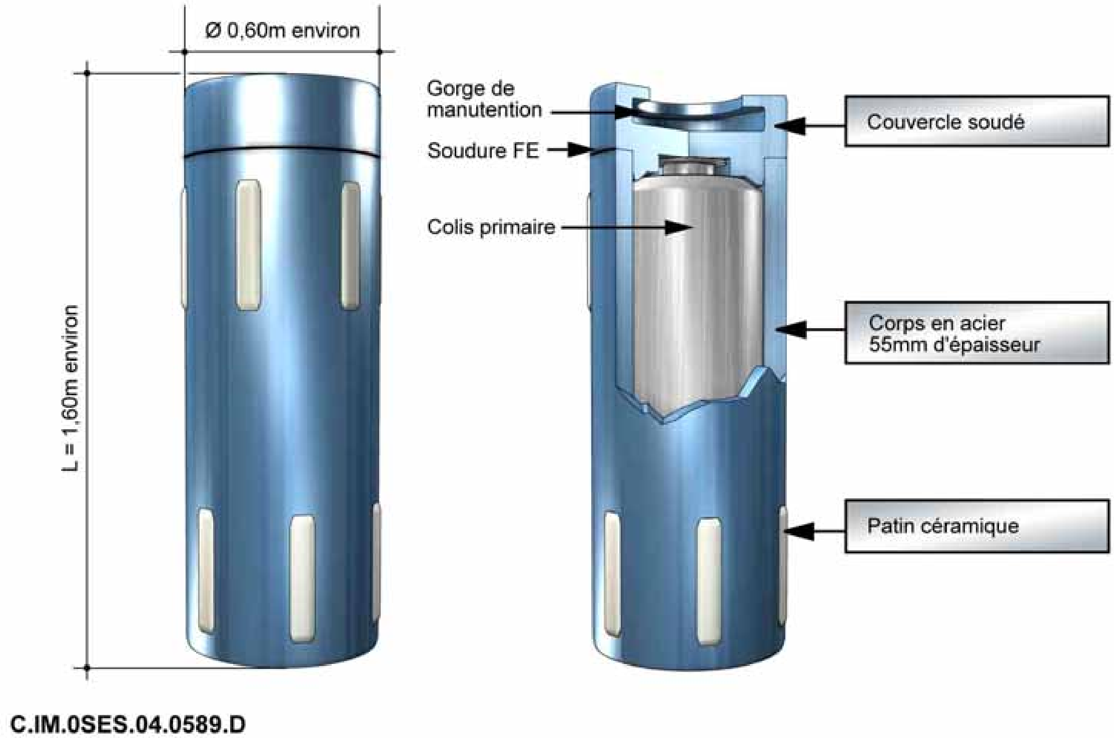


Figure 6.2-14 Diagram of vitrified waste container, showing the ceramic PARS

To take account of the residual uncertainties concerning the corrosion conditions, three corrosion models were defined by Andra [68].

The first, known as « phenomenological », because it is based on the best estimate of container evolution, assumes the following conditions :

- the containers and overpacks undergo a rapid transition to anoxic conditions, owing to the presence of the sealing cover ;
- during this transitional period, owing to the heat given off by the containers, the air is dry enough for corrosion to be extremely limited, and concerns only the lining ;
- if oxygen manages to enter the disposal cell, it is consumed by corrosion of the lining at the disposal cell entrance.

The corrosion rates adopted for the containers are consequently « phenomenological » and associated with anoxic conditions.

The second model, known as « conservative » assumes the following conditions :

- the possibility of corrosion in an oxidising medium and in the presence of water, including the container (about ten years for a C container, 70 years for a spent fuel container) ;

- consideration of local heterogeneousness in the chemical conditions of the medium (for example due to infiltration of water through the lining, the local presence of sulphides from the geological medium, etc.) and through application of a factor to correct the corrosion rates.

The third model, known as « pessimistic » arbitrarily assumes oxidising corrosion conditions in the containers for 300 years, which is the same as ignoring the role of the sealing cover and assuming the disposal cells to be open for a long period.

The lifetime of the containers and the overpacks according to these three models is as follows :

- about 15,000 years for vitrified waste containers and 30,000 years for spent fuel containers, using the « phenomenological » approach ;
- about 4000 years and ten thousand years respectively using the « conservative » model ;
- about 1000 and 10,000 years using the « pessimistic » model.

We have seen that the SEN calculation considers life-spans of 4000 and 10,000 years, adopted on the basis of the « conservative » model, that includes the uncertainties regarding the homogeneity of the chemical conditions.

Even if we apply the « pessimistic » model, the SEN calculation conditions would not be significantly changed. After 1000 years, the temperature has dropped sufficiently for the release rates from the glass and the transport conditions in the medium to be close to those involved in the SEN. The only area influenced would be the radioactive decay of a few medium-lived elements which would no longer benefit from the additional 3000 years of decay provided by the longer life of the containers in an SEN.

Given the considerable design margins of the containers, only highly improbable phenomena that are hard to identify beforehand could lead to life-spans significantly lower than those considered in the « pessimistic » model. The SEN however considers early seal losses for a small number of containers, as a precaution. To cover this uncertainty even more completely, Andra has defined a « package failure » SEA which considers very early loss of the functionalities of the metal containers on a series of containers and for the entire inventory. This extremely « what-if » scenario finally covers all forms of uncertainty concerning the corrosion conditions.

- **Mechanical behaviour of elements subject to corrosion**

The design of the components takes account of the conservatively estimated corrosion in order to define a thickness dedicated to chemical attack and a thickness dedicated to mechanical strength. In this way, the mechanical strength of a component is guaranteed for the entire time it is required to perform a function. This is the case with the containers.

The uncertainty regarding the mechanical behaviour may however no longer be covered by the design once the anticipated lifetime is exceeded. As of this period, no further function is expected of the component. However it may become a source of disturbance for its environment : mechanical action on the environment owing to its deformation, pressure due to uncontrolled expansion of the corrosion products, etc. The mechanical interactions between the lining or the container and the argillites were dealt with in section 6.2.6.1. We have seen that they are covered in the worst case by a sensitivity study of the SEN on the EDZ properties.

One can also wonder whether deformation of the vitrified waste overpack and the pressure of the corrosion products might, in the most extreme situation, damage the interior of the packages. The sensitivity study on the glass fracture rate in the SEN covers every type of damage.

The future of the metal components and in particular their impact on the chemical and mechanical evolution of the spent fuel containers was also taken into account in the long-term management of the criticality risk. This risk would appear to be under control regardless of the time frame.

In particular, for the spent fuel, corrosion of the assemblies and reduction of the voids due to overall mechanical compression of the repository will bring the assembly fuel rods closer together, guaranteeing that the package remains sub-critical owing to the absence of moderator as the space available for water diminishes. For UOx containers, a favourable factor would also be the expansion of corrosion products in the cast iron insert, which would tend to gradually increase the distance between the assemblies. A systematic study of the possible geometrical configurations for the various types of disposal packages was conducted, along with a verification of their non-criticality. Its main results are presented in section 6.2.9.

Apart from the containers and overpacks alone, deformation of the metal elements placed close to the rock (support reinforcements, lining in the cells) could in theory lead to secondary damage, although it would seem reasonable to suppose that the force exerted would not be very great. In any case, such damage would occur in the standard sections of the drifts or disposal cells. For questions linked to chemical disturbances (see following section), rock bolts are not used over the length destined to be sealed. Deformation due to these elements would not compromise the role of the seals in intercepting the fractured zone, and would not therefore compromise the representation of the SEN.

6.2.8.4 Alkaline disturbance of the argillites and bentonite

The design of the repository involves a limited number of material types, the physico-chemical behaviour of which has been the subject of numerous studies and for which there are comparable data of varying ages. This allows compilation of a body of knowledge of the behaviour of the materials in the repository, enabling this behaviour to be predicted and modelled [19].

The proposed design thus leads to controlled environmental conditions, limiting the possible variety of interactions. Thanks to the presence of the seals, the environment is on the whole diffusive, particularly after resaturation. The environment is reducing owing to the rapid consumption of oxygen [92]. The pH is maintained with the range of 10 to 12 by the cement-based materials in the B waste cells and kept by the medium at around neutral in the glass and spent fuel cells.

All these design measures significantly limit the chances of the geological medium being disturbed by chemical reactions resulting from exogenous materials. The main one (in terms of potential extension) is alkaline disturbance, linked to the action of cement-based fluids on the argillites.

Understanding the mechanisms of alkaline disturbance is based on experimental approaches. The thermodynamic models produced under the European ECOCLAY project are able to reproduce the geochemical processes observed. The programmes conducted in parallel on bentonite and argillite show that the reactivity of the argillites is consistent, confirming possible transposition of the results. Extrapolation of the conceptual model over time is supported by the observations made at Maqarin and Khushaym Matruk in Jordan, natural sites on which clay materials have been altered by alkaline fluids.

The extension of the alkaline plume was conservatively evaluated by 1D modelling over a period of 100,000 years at 25°C, considering an infinite concrete source and a diffusion coefficient of the same order of magnitue as the one for tritiated water, i.e. between the lowest values measured for anion diffusion and the highest values measured for cation diffusion, again a conservative choice. Using this approach, the consequences of the disturbance, according to the chosen scenarios, remain limited to a small thickness of argillite or bentonite within which it is possible to identify zones degraded to varying degrees. The first zone, in immediate contact with the cement but not exceeding 60 cm after one million years, is heavily degraded and corresponds to dissolution of the initial smectite which is then replaced by phases typical of the cement-based medium, the CSH and zeolites. It is followed by a second zone referred to as disturbed, as the smectite is preserved, and which is only affected by Na/Ca cationic exchanges. The extension of this zone can reach up to 1.8 metres from the interface with the cement in bentonite and up to 3 metres in argillite after one million years [69].

These results in principle depend on the diffusion coefficient in the medium considered. The sensitivity studies conducted however indicate that its increase does not modify the nature of the mineralogical transformations but leads to a widening out of the reaction face. A factor 3 increase in the high-temperature diffusion coefficient (at 90°C) does not change the order of magnitude of the alkaline disturbance extension assessed at 25°C.

In the light of these results, the choice of not explicitly representing the alkaline disturbance in the SEN can be justified in the following way :

- in the argillites, extension of the heavily remineralised zone will not exceed that of the fractured zone used in the model. The expected effect of the alkaline disturbance on the fractured zone is not fully determined. In principle, we expect clogging of the porosity and thus an on the whole favourable effect with respect to hydraulic properties. However, such an effect is not taken into account and as a precaution we consider the permeability in fact to be degraded. A sensitivity study on the characteristics of the fractured zone, was carried out in the SEN ;
- in the vicinity of the seals, the extension of the heavily remineralised zone is 60 centimetres in conservative assessment conditions. This thickness is incorporated into the design by a « lost » seal length that is at least equal to this. This length of bentonite is only intended to buffer the disturbance. Near the zones identified by the safety analysis as sensitive (that is primarily close to hydraulic cut-offs), we also avoid leaving cement-based and/or metal materials in place. At this stage, the plan is therefore to remove the ground support when excavating the hydraulic cut-off groove and installing the seal (see Figure 6.2-15). In these conditions, 2D models with transport-chemistry couplings show that the hydraulic cut-offs can undergo mineralisation modifications under the effect of cement-water but no remineralisation such as to compromise the swelling capacity of the bentonite, and thus its function (for example, no significant rise in its pH) [69]. As a variant, the possibility of a low-pH concrete ground support around these hydraulic cut-offs is under study ;

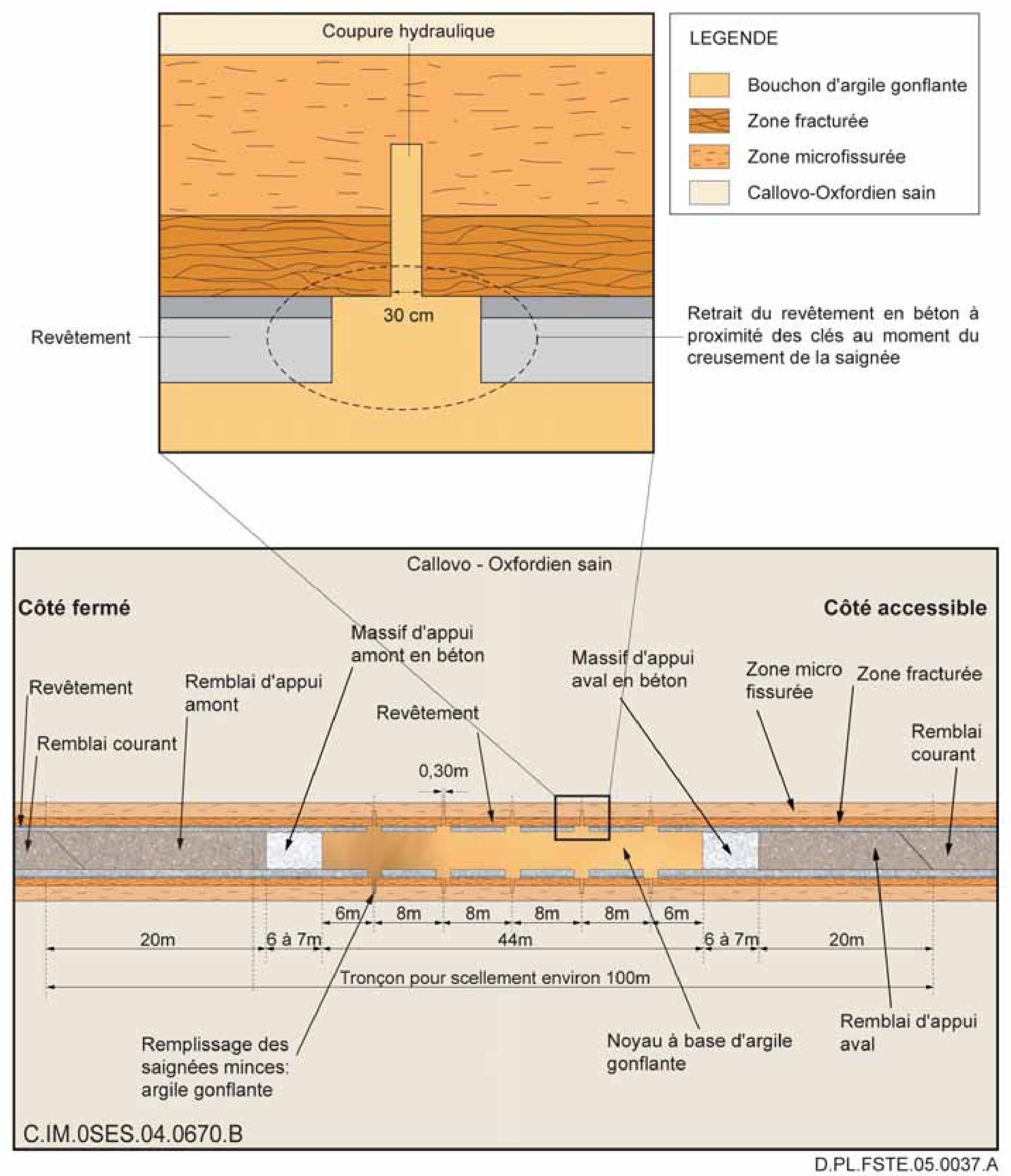


Figure 6.2-15 Diagram illustrating removal of the ground support close to the seal hydraulic cut-offs.

- the thermal waste cell plugs are for their part long enough to buffer the alkaline disturbance, which can in particular have no influence on the pH in the disposal cells (see Figure 6.2-16).

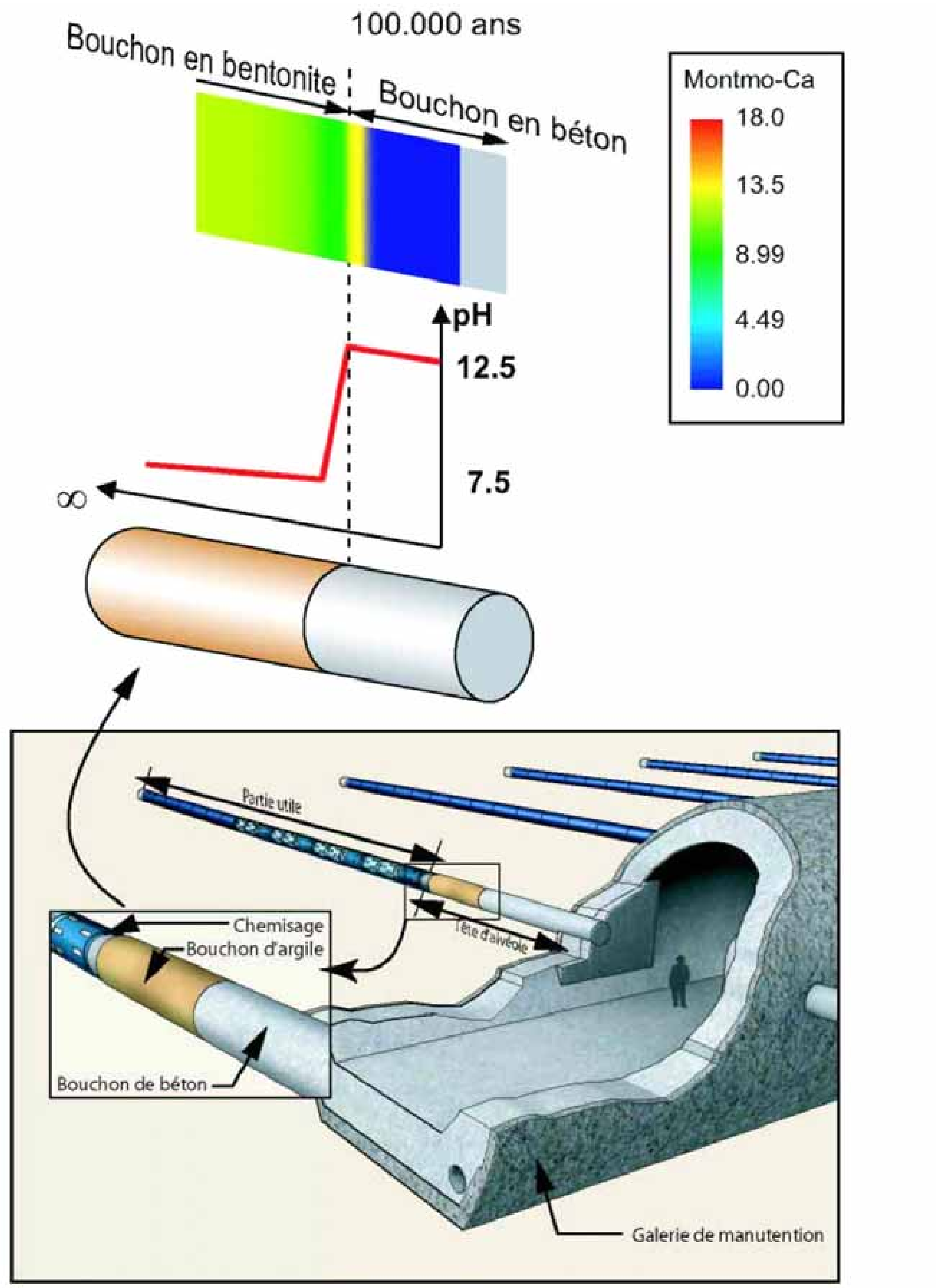


Figure 6.2-16 Representation of the alkaline disturbance extension on the C waste cell plugs

The residual uncertainties at this stage concern :

- the fact that the assessments assume a maximum pH for the cement waters of 12.5. For common concrete formulations, higher pH values can exist only temporarily, for juvenile water (see Figure 6.2-13) This uncertainty can also be managed by specific concrete formulations (with added silica) ;
- the fact that the assessments assume diffusive conditions. If, in an altered situation the hydraulic conditions were to become generally or locally advective, the assessments would no longer be valid. If the case arises during study of the SEA (see chapter 7), it would therefore be necessary to take account of the possible effect of propagation of the alkaline disturbance, in particular through the C cell plugs, if the « limit water circulation » function were to be lost ;

- uncertainties on installation of the bentonite elements (engineered structures, plugs, seals) and their homogeneity. If this is not done correctly, the interstices between bentonite rings or bricks could favour the propagation of the disturbance. This question is dealt with in the section devoted to the technological uncertainties, 6.2.12.

Excessive extension of the alkaline plume could then lead to degraded seal performance and, in a highly improbable situation involving significant extension of the pH to the vitrified waste and spent fuel cells loads, to a locally basic medium. To complete coverage of the residual uncertainties, we therefore envisage :

- the possibility of non-swelling of the bentonite under the effect of the alkaline disturbance. It is covered by a situation in which there is failure of all the seals in the « seal failure » SEA ;
- as a variant of the « seal failure » SEA, possible extension of the alkaline pH into the C waste disposal cell and a rise in the releases from the vitreous matrices under this effect. We cannot accurately describe the effect of a basic pH on the glass so we choose artificially to increase the rate of dissolution until the lifetime of the matrix becomes negligible given the transport processes. The influence of a high pH on the spent fuels is also very hard to describe, but the highly conservative release model used as the reference (leading to a lifetime of 50,000 years) would appear to be fast enough for there to be no point in accelerating it further.

6.2.8.5 Iron-argillites and iron-clay disturbances

The reaction mechanisms describing the iron-clay interaction are well known (see Figure 6.2-17) and raise few questions, as the chemical mechanisms in a uniform bentonite are well understood. The extension of the iron/bentonite interaction was assessed using two approaches, mass balance and coupled transport-chemistry modelling. The two approaches lead to a limited effect because chloritization of near field argillites or swelling clay consumes the source term. Using different diffusion coefficients, the modelling as a whole reveals a thickness of only a few centimetres for the heavily chloritized zone. Beyond this the iron exchanges with the cations in the clays [93].

In addition, the great affinity of clays for iron could potentially create competition between this cation and the cationic radionuclides, which could lead to a reduction in the sorption capacity, but over short distances. These effects are covered by the sensitivity study attributing conservative Kd values to the bentonite in a SEN.

With regard to the argillites, the diffusion of iron II resulting from degradation of the metal elements left in place (lining, concrete reinforcements, bolts, etc.) cannot be totally ruled out in the damaged zone close to the disposal cells. Extrapolation of the results obtained on bentonite leads us to consider that these processes could create modifications in the mineralogical composition of the argillites or the chemistry of the Callovo-Oxfordian interstitial waters, but only over short distances. In the same way as for bentonite, iron diffusion beyond the cell damaged zone can lead to competition between this cation and the cationic radionuclides, which can lead to a reduction in Kd [70], effects that are covered by sensitivity to degraded geochemical parameters in the SEN calculations.

Evaluating the propagation of an iron-clay disturbance also raises the question of installation of the bentonite elements (clay engineered barriers, plugs, seals) and their homogeneity. As in the case of alkaline disturbance, if this is not correctly done, the interstices between bentonite rings or bricks could encourage propagation of the disturbance, but over distances that remain short.

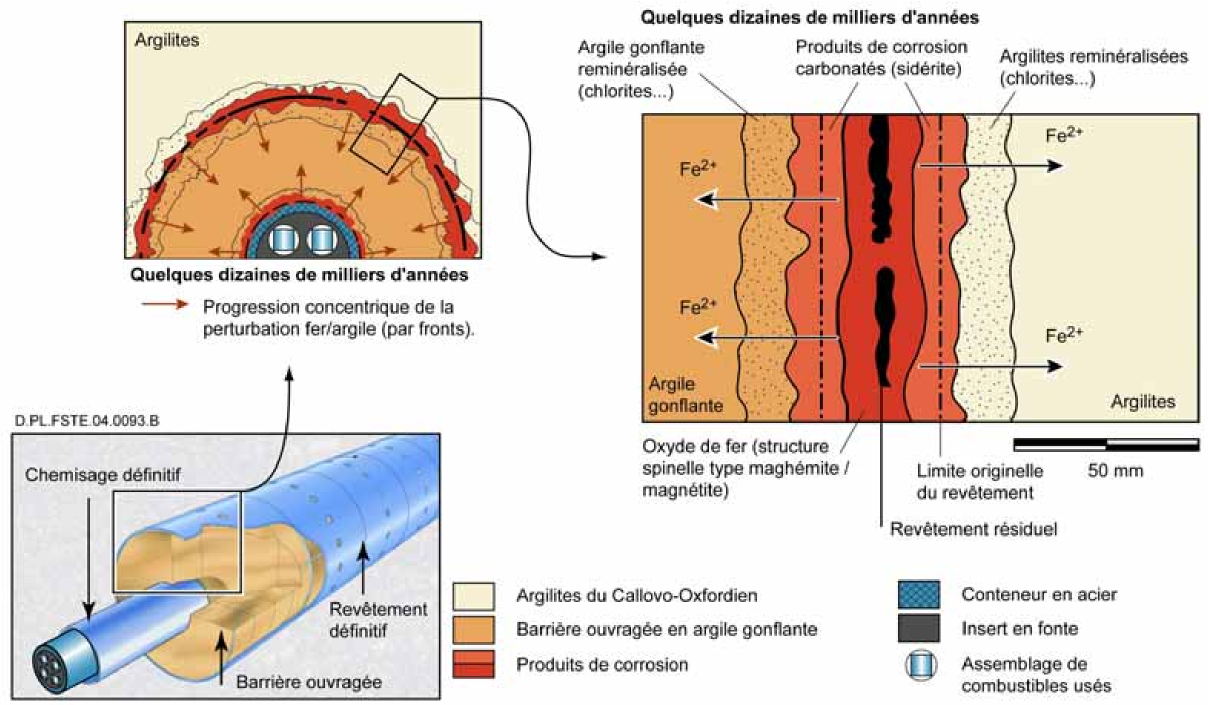


Figure 6.2-17 Schematic representation of an iron-clay disturbance

To conclude, we retain no uncertainty over extension of the iron-clay or iron-argillite disturbance which could be such as to compromise the SEN representation and its sensitivity studies.

6.2.8.6 Oxidising disturbance in the argillites

The chemistry of the Callovo-Oxfordian waters can also be altered close to the structures by disturbances induced by repository operation. Oxidisation may in particular locally modify the composition of the waters and increase the content of unfavourable elements (modification of the pH, increase in sulphates content, etc.). These effects are limited by the buffer capacity of the argillites. The possibility of a sulphate attack on certain components, or a pH deviating from neutral, cannot however be totally ruled out.

No particular effects are expected from a sulphate attack on the concrete (coating, additional packaging, etc.) given the high quantities of cement-based materials used and which act as a chemical buffer. The concrete formulations can be tailored to deal with a sulphate attack. However, if this were to prove more penalising than that envisaged in the reference, then the effects on the concrete (early loss of mechanical strength) would be of the same nature as in the situations mentioned in section 6.2.8.2, covered at worse by the « seal failure » SEA.

Oxidisation of the organic material present in the argillites may also lead to the formation of colloids. This phenomenon could not however concern large quantities and is negligible given the transport mechanisms on the scale of the repository.

6.2.8.7 Cumulative chemical disturbances

Cumulative chemical disturbances raise two different questions :

- when the same clay medium is subjected to simultaneous or successive disturbances from different chemical elements, is the total induced disturbance different from that which would be due to each interaction individually ?
- when different parts of a given component are subjected to different disturbances, is its overall design sufficient ?

The main accumulation to be considered, given the calculated extensions, is that of alkaline and iron-clay disturbances. There are other sources of chemical disturbance due to other compounds (for example silica - clay disturbance) near the C waste cells, but they prove to be negligible.

Given the slight extension of the iron-clay disturbance, it is reasonable to assume that it will not drastically modify the extension of the alkaline disturbance if it were added to it. Furthermore, the design measures already mentioned (removal of the ground support near the seals, seal design to take account of a « lost » thickness designed to buffer chemical interactions) also protect against cumulative disturbances.

Should it however be shown that cumulating the disturbances can lead to unexpected phenomena, then these effects would be covered by the seal failure and/or damaged zone degradation situations already presented.

6.2.8.8 Summary of chemical uncertainties

After the analysis it would seem that the main uncertainties identified concern extensions of the chemical disturbances. They are assessed in conservative conditions and, if not negligible, are handled by the design (either by eliminating the source of the disturbance, for example by removing the lining close to the drift seals - or by over-sizing the « lost » thicknesses, such as for the cell plugs). They are therefore no longer explicitly represented in the SEN, but given the fact that these are mechanisms running over the long-term and that uncertainties on the extensions can be associated with other uncertainties (on the transport conditions governing the transfer of aggressive chemical species, on the accumulation of different disturbing effects) the principle is to use an SEA to cover seal failure and/or EDZ degradation situations which can originate in chemical disturbances of excessive scale.

The long-term behaviour of the materials (concretes, steels) is well understood as a major research programme has been completed. The repository conditions (reducing, homogeneous medium) are favourable to long lifetimes being considered with a high degree of confidence. However, with respect to the residual uncertainties, the safety analysis systematically considers that either locally or overall, the lifetimes of the materials are shorter than planned, even using the most conservative models.

6.2.9 Radiological risks linked to a hypothetical criticality accident in the repository

The objective regarding criticality is first of all to ensure that such an accident is sufficiently improbable in the long term. The motivations behind this are different from those of the operational criticality studies. A criticality accident in the repository once closed would not expose any individual to the emitted radiation as the thickness of the geological formations provides adequate protection. However, the environmental conditions of the packages concerned are harder to describe and assess : modifications to the thermal and radiological environment and the radionuclides inventory.

We therefore aim to achieve conditions which rule out or at least minimise the occurrence of such a post-closure risk. The packages are designed to prevent a criticality accident in the repository in all operating situations. On this basis, the approach consisted in ensuring that these measures were sufficient also to cover the long-term risk, by taking account of the chemical and mechanical evolution of the packages. This check was conducted on each B, C and spent fuel repository zone [45].

In addition, we also envisaged criticality situations which could be induced by migration of radionuclides over the very long term and a possible reconcentration of fissile material at a point in the repository. The opinion of most international experts [94, 95] is that such a situation is highly unlikely. Nonetheless, this point was checked on the basis of a phenomenological assessment of the possibility of reconcentration of the fissile material [96].

6.2.9.1 Post-closure criticality risk due to the packages

The distances between the disposal cells are large enough for neutron uncoupling, so the criticality risk analysis is conducted on each cell considered separately.

- **Specific Risk for B waste packages**

For the B packages, the uncertainties concern their fissile material inventory, the mechanical evolution of the B waste cell and finally the chemical degradation of the overpack.

Uncertainty concerning the radiological characterisation of the packages, in particular the definition of the fissile inventory used as the input data for the criticality studies, is covered by calculations performed on the bounding reference package. In this case, this is a B5 type package, in other words the CSD-C package containing compacted hulls and end-pieces from reprocessing at La Hague of UOx and potentially MOX assemblies. Apart from checking sub-criticality over time on the basis of the initial inventory data, we also carry out the opposite calculation, consisting in evaluating the maximum mass of fissile material allowable per CSD-C in the repository cell. This evaluation was run in penalising conditions, assuming that the four CSD-C contained in the packages all comprised the maximum allowable amount of fissile material and were juxtaposed on each other without separating concrete. These calculation conditions cover a possible unfavourable mechanical evolution of the overpack with the primary packages inside moving closer together. The deduced fissile mass proved to be far higher than the maximum allowable mass per CSD-C currently defined by COGEMA for storage at La Hague.

Given the mechanical evolution of the B waste disposal cell, theoretical criticality can only be reached in the event of the primary packages contained in several different overpack coming together. The durability of the waste packages is a positive factor in limiting this type of situation, even if to date this is a qualitative margin that is not adopted for the studies (post-closure, the overpack only has a chemical retention function). Apart from the previously mentioned check, which consisted in moving the packages closer together four by four within the overpack, we also evaluated situations in which a large number of primary packages could find themselves in contact with each other. In the light of the thickness of the overpack concrete, and the minimal voids within the cells, this form of contact would be totally unrealistic. Neither the mechanical evolution of the packages nor an earthquake could lead to such a configuration. Given the results of the study, a super-critical situation is ruled out, all the more so as unlike the calculation hypotheses used, very few or possibly even none of the packages would be loaded with the maximum allowable mass.

With respect to the chemical uncertainties, we looked to ensure that the evolution of the concrete could not lead to the package stacks being made super-critical. Such a situation could in principle occur if the chemical evolutions were to increase the neutron reflection of the concrete. Chemical evolution of the concrete is however dependent on the arrival of water. In the overpack this water would act as a neutron absorber, making a significant contribution to controlling the criticality risk. The effect of any chemical evolution of the concrete would be covered by this favourable effect. However, to cover all uncertainty on this subject, we carried out an assessment assuming :

- unfavourable chemistry of the concrete, defined by increasing its silicon content to the detriment of more abundant but less reflective elements such as calcium ;
- a minimal water content.

It turns out that even in this highly penalising situation, a criticality accident is not possible.

- **Specific risk for C waste packages**

There are few radiological inventory uncertainties for the vitrified waste as the type C1 glasses produced or being produced meet quality assurance standards which guarantee a representative radiological inventory. For future or hypothetical packages, definition of their radiological content is based on what could be imagined at this stage of the process to be implemented. Their subcriticality will probably have to be re-assessed were they to enter production ; however, we will see that the studies contain considerable margins so this point should not raise any difficulties in the future. Although some of the C0 waste is old and predates production quality assurance procedures, it does not pose problems because the fissile material content is generally far lower (one order of magnitude) than in the other packages.

The criticality-safety assessment in the repository conditions conservatively considers that the packages are arranged horizontally in an infinite row. This is equivalent to eliminating the neutron decoupling role of the buffers which could separate each C1, C2, C3 or C4 type disposal package. This conservative approach guarantees criticality-safety for all subsequent mechanical evolution of a C disposal cell for whatever reason (increased stresses, earthquake, etc.). Only square or triangular pitch type arrays of packages heavily loaded with fissile material (type C4) could prove to be super-critical, but it is clear that no package movement could lead to such an arrangement.

With regard to chemical evolution in the C disposal cells, to cover the uncertainties linked to degradation of the nuclear glass, the subcriticality of the row of packages was checked by comparing the various components of the glass to silica alone. This represents more than 50 % by mass of the initial composition. Silicon, along with aluminium (but the Al_2O_3 oxide only represents 4 % of the mass of the glass), is a penalising element encouraging free movement of neutrons. The other elements present, such as sodium, calcium or zirconium are comparatively more neutrons- absorbent ; boron, which is present in significant quantities (B_2O_3 representing 18 % of the mass), is for its part a highly efficient neutron absorbent, and is consequently favourable to subcriticality. To provide a conservative representation over time of the other components of the C disposal cell and the geological medium, reflecting materials which are penalising with regard to their water content were modelled.

Given the above, the only uncertainty identified as theoretically prejudicial to controlling the criticality risk in the C disposal cells is the risk of glass fracture. The presence of water in the vitreous matrix will have the effect of better moderating the fissile material and thus increasing the neutron balance of the system. Parametric studies, which depend on the water content on the glass, were conducted assuming a reference fissile medium comprising silica, water and fissile material. This study demonstrated the safety of the C1, C2 and C3 waste package disposal cells (and implicitly C0 which contains ten times less fissile material) whatever the water content within the glass packages, in other words, whatever the extent of fracturing of the matrix. In the case of the C4 package, which it must be remembered is simply a concept and not an industrial reality, one need simply incorporate 1 % boron (of the total mass of the vitreous matrix) to rule out the risk of criticality.

The important margin represented by not using boron in the calculations could cover the uncertainty over the radiological characterisation of future packages, and in particular the definition of the fissile inventory. As with the B packages, it is possible on the basis of these data to rule out the possibility of a criticality accident due to the vitrified waste packages, in both a normal and altered situation.

- **Specific risk for the spent fuel packages**

Analysis of the criticality risk during the post-closure phase mainly concerns the spent fuel packages containing four fuel assemblies, as the case of UOx or MOX single-assembly packages can then be deduced. For UOx assemblies, this analysis takes account of fuel irradiation according to conservative methodology and hypotheses [45].

Unlike the B and C primary packages and despite minimising the voids in the cell and in particular in the disposal package, the residual void rate (space between rods, handling clearances, free space in the cans) can potentially initiate criticality. These voids are likely to be favourable towards geometrical evolution of the package and occurrence of criticality as the assemblies come closer together (maximising neutron interaction) or through modification of the pitch between the fuel rods (modifying moderation of the fissile material). The mechanical evolution will easily outweigh the chemical degradation of the assemblies and internal components of the package. For example, the formation of corrosion products, taking the place of the cast iron and steel of the container with fixed geometry, only changes the reactivity very little. The approach used primarily consists in checking that the foreseeable successive geometrical evolutions of the assemblies and packages, for the two container variants being studied (V1 and V2, see Figure 6.2-18), do not compromise the sub-criticality of the package in the disposal cell. It will be necessary each time to clearly differentiate between what is to happen to the rods within an assembly and to the assemblies with respect to each other.

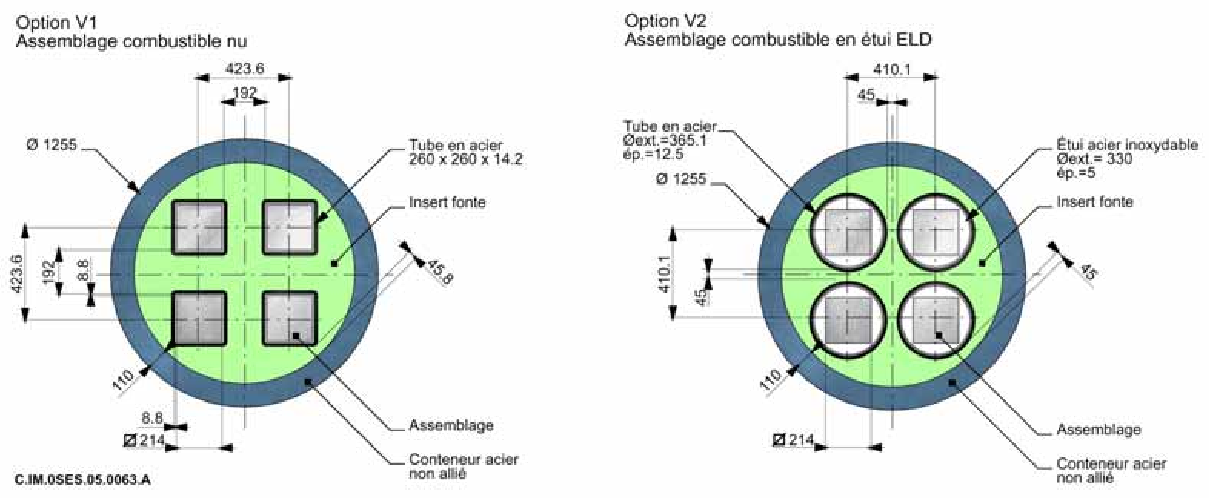


Figure 6.2-18 Cross-section of option V1 and option V2 variants of the CU1 disposal package containing 4 fuel assemblies

Evolution of rods within an assembly

Figure 6.2-19 shows four possible representations over time that can be compared to transient situations between the initial integral state which is subcritical by design, and a final state, which is subcritical owing to lack of moderation, corresponding to crushing of each assembly onto itself under the stresses induced by reduction of the voids. These representations were the subject of criticality studies for various assembly arrangements in the package and are defined as follows :

- conservation for a significant time of the initial geometry of the assembly potentially evolving towards a situation in which the fuel rods bend under the action of mechanical stresses or gravity (1) ;
- rupture of a significant fraction of the rods leading to loose fuel at the bottom of the housing, co-existing with an array of partially degraded rods. This rupture would take place dry through hydriding of the cladding or through internal over-pressure, or eventually under water owing to the mechanical stresses induced by corrosion (2) ;
- separation of the rods from the grids and end-pieces under their own weight or the action of corrosion, a situation which would lead to a rod bundle degraded to varying extents at the bottom of the housing (3) ;
- loose fuel at the bottom of the housing which could be a subsequent evolution of the previous situations (4).

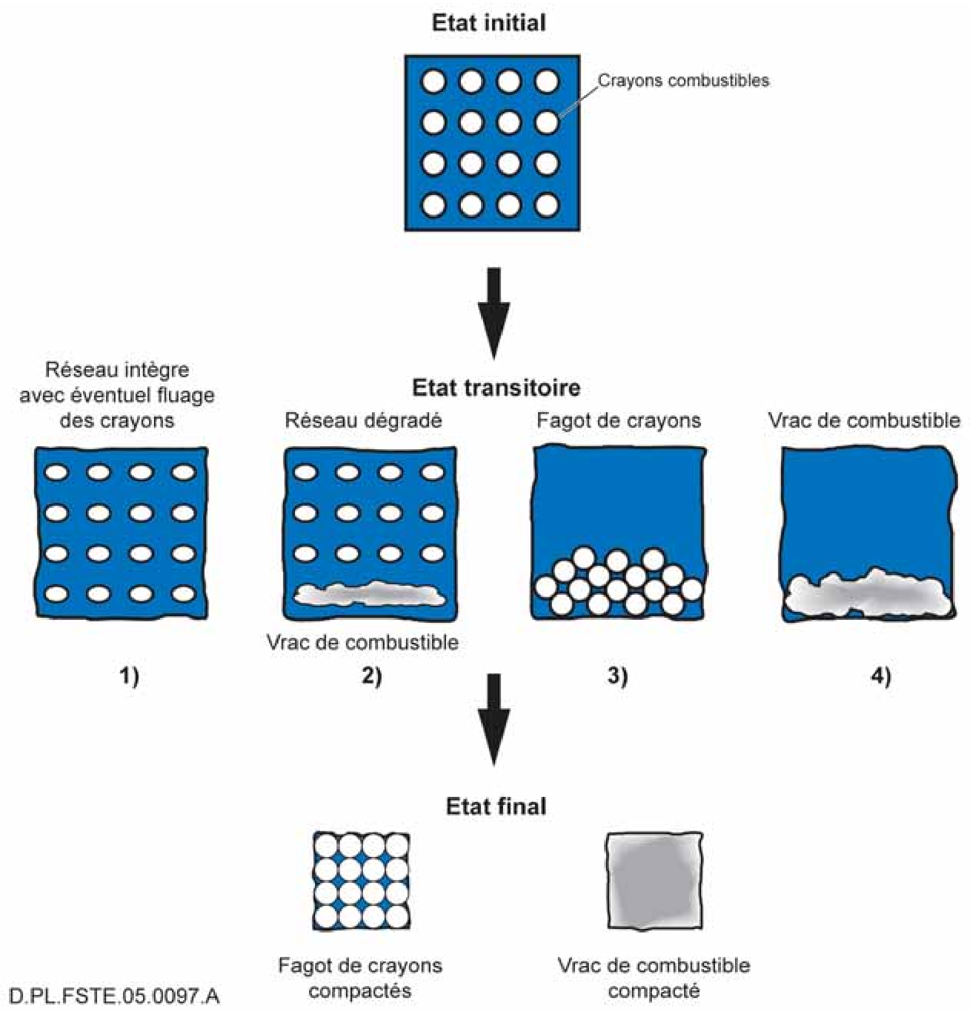


Figure 6.2-19 Evolution of spent fuel package assemblies following corrosion of the insert and expansion of the associated volume

Analysis of the criticality risk for all the transient situations was made for an arrangement of degraded rods (or loose rods) and for the level of moderation of the fissile material they contain. The criticality calculations showed that the initial integral situation and conservatively case (1) presupposing possible rod creep, proves to be the most reactive representation. The result of these studies shows that evolution of an assembly on its own cannot lead to it becoming supercritical, except in unrealistic configurations in which there would be a significant increase in the distance between the rods, in the V2 container option. Some of these configurations could be supercritical according to current initial estimates, but such a situation is not realistic as the rods would tend to fall as a result of gravity rather than move away from each other.

Evolution of assemblies within the packages

Within the package (Figure 6.2-20) we expect the expansion due to corrosion to move the assemblies apart (a), or at least preserve a sufficient thickness to guarantee subcriticality (b). A criticality event could only be possible in the highly theoretical situation of the assemblies coming together (c) and would be based on excessively conservative modelling, assuming both collapse of the internal components of the disposal package (the insert X-grid) and conservation of the initial pitch between the fuel rods favourable to moderation of the fissile material.

The estimates made on assemblies which have been little irradiated, which are the encompassing assemblies for the others, show that at the same time the rods would have to maintain their initial distance and the assemblies move until they are only seven centimetres apart. Even if the assemblies were to move closer with an array of rods that was on the whole undamaged, moderation would no longer be optimum owing to the necessary distribution of the corrosion products between the rods (d).

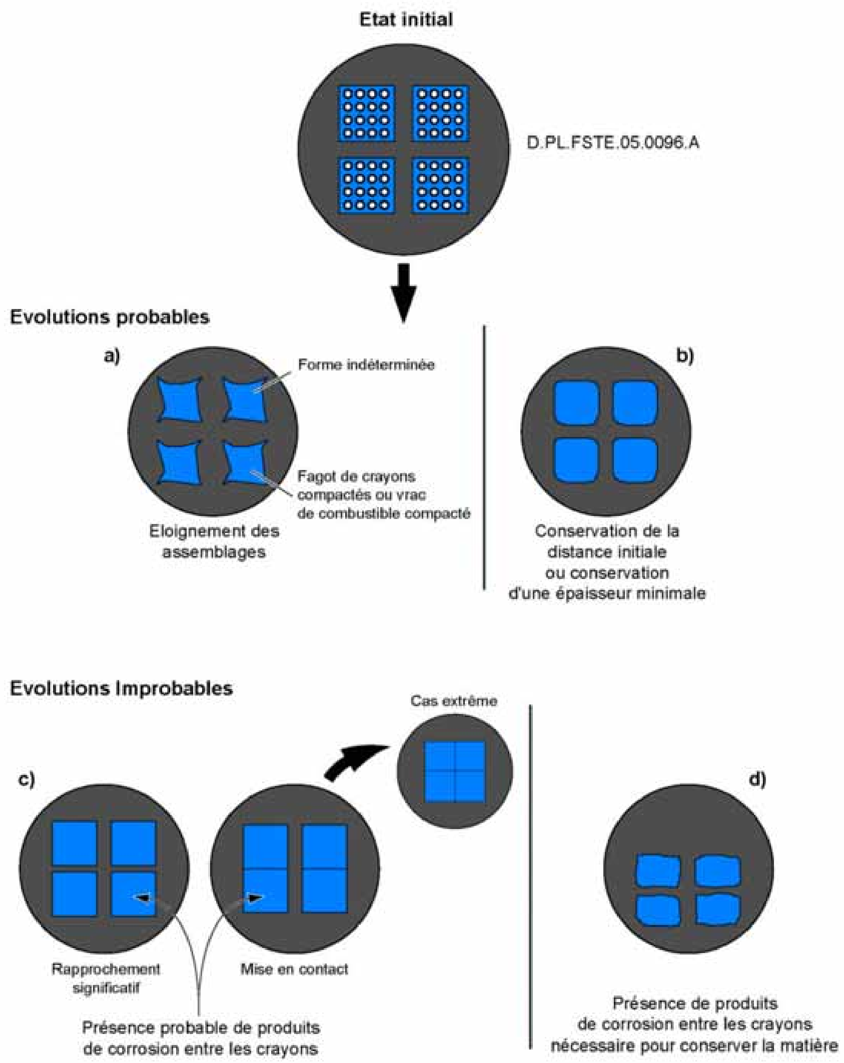


Figure 6.2-20 Evolution of spent fuel package following corrosion of the insert and the associated volume expansion

To conclude, the occurrence of a criticality accident owing to evolution of the spent fuel containers would seem to be highly improbable. For management of the residual uncertainties and in order to comply with the request from the Nuclear Safety Authority, the decision was nonetheless taken to conduct a deterministic evaluation of its consequences.

- **Evaluation of the consequences of a hypothetical criticality accident in a spent fuel cell**

Feedback from criticality accidents at nuclear facilities shows that the radiological impact (irradiation and contamination) is a key parameter, the objective being to assess the impact of the accident on the public and workers. The aim in such a situation is therefore not to provide an accurate description of the phenomenology of the accident, but to assess its radiological consequences in an arbitrarily conservative manner. In a hypothetical criticality situation in a disposal cell in the post-closure phase, the opposite holds true. The thermal and mechanical effects are the primary parameters and a phenomenologically representative situation must be examined without excessive conservatism.

It has been possible to judge the risk of criticality in spent fuel disposal cells containing four assemblies as sufficiently improbable in the long term. Nevertheless, for the requirements of this assessment and in order to define an accidental scenario, it was assumed that the hypothetical accident would occur in this package within a heterogeneous medium of fuel rods in the presence of water. The medium is presumed to be saturated as the accident would occur following degradation of the container, and thus well after ten thousand years. The reactivity of the medium to be considered was estimated and proves to be highly dependent on the initiating scenario and, in particular, the gap between the rods considered and the distance between the assemblies. The choice of scenario was that in which the assemblies would move closer together until they are five centimetres apart (versus twenty centimetres initially) without distribution of the corrosion products between the rods, which is a highly improbable situation. This scenario is also representative, in terms of inserted reactivity, of a hypothetical criticality further to an improbable increase in the gap between the rods.

Apart from the physical and chemical nature of the fissile medium, the burnup fraction of fuels and the potential reactivity of the system, the dynamics of a criticality accident is also governed by the rate of insertion of this reactivity, by the neutron counter-reactions (temperature effects, expansion effects and void effect) and by the environment (thermal exchanges, system confinement, etc.). A sensitivity study shows that evolution of the accident over time is comparable for reactivity introduction in one day, one month or one year, in other words, as long as we consider that the introduction of the reactivity is slow enough, which would seem reasonable given the kinetics of mechanical deformation in the cells.

By modelling the immediate environment of the package, the calculations highlight a slow rise in the power given off by the assemblies, which tends to stabilise, irrespective of the burnup fraction of the fuels contained in the package. Indeed, the rise in the temperature of the fuel and the water tends to limit the criticality reactions by phenomena qualified as neutron « counter-reactions ». This enables us to conclude that there is no violent reaction within the cell and no mechanical damage. Given the very low power level reached, the accident can persist and only finally stop with fuel exhaustion and poisoning.

The fuel temperature remains limited (about 65-70°C), which leads to a very slight rise in the buffer temperature and therefore in the temperature of the surrounding rock for the entire duration of the accident. Only the first three metres of Callovo-Oxfordian argillites see a significant temperature rise (5 to 20°C) ; beyond that, the temperature is close to that of the initial environment and equivalent at less than ten metres or so from the packages. With regard to the transfer of radionuclides leaving the spent fuel packages when the accident occurs, the situation is similar to the one considered for a defective package and corresponds, in an initial approach, to a transfer at temperature. The characteristics of the thermal field are however different in this case, both from the point of view of the duration of the phenomenon (longer in the case of a criticality accident) and its intensity and extent (lower).

On a millennium time-scale, the penalising two-dimensional thermal model used indicates a temperature of 77°C at the limit between the buffer and the argillites ; the temperature conditions linked to the criticality would therefore in principle be less penalising than those already considered for the thermal phase.

The duration of the phenomenous could be as high as several tens of thousands of years, or even a hundred thousand years, depending on the fuel exhaustion rate and the quantity of fissile nuclei available prior to the accident. The duration of the accident therefore depends on the level of irradiation in the reactor ; the presence of slightly irradiated fuels in the packages containing four assemblies being excluded by design, this duration should not exceed a few tens of thousands of years.

Mineralogical transformations linked to a temperature rise in the package's near-field environment (20 to 30°C at most) over a relatively long period would only be potentially damaging for the smectitic phases of the bentonite and of the argillites of the Callovo-Oxfordian. Iron/clay perturbation or illitisation phenomena would possibly lead to a reduction in their swelling and chemical retention properties. In any case, this evolution would only concern the immediate environment of the packages. Furthermore, owing to the limited temperature rise associated with the accident and following analysis of the experimental data and observations available, it may be considered that the rate of smectite to illite transformation is low for both bentonite and the Callovo-Oxfordian for durations of up to a hundred thousand years. It will also be noted that the temperature is insufficient, even in the event of a criticality accident, to envisage significant chloritisation of the smectite.

The accident adds new fission products to the inventory already present. The most abundant radionuclides formed are mainly very short-lived and cannot contribute to the impact. For long-lived and mobile radionuclides such as ^{36}Cl, ^{41}Ca, ^{79}Se and ^{129}I, the added inventory is negligible when compared with the initial inventory.

From the irradiation viewpoint, the particles emitted would above all be neutron and gamma radiation. The additional radiolysis induced by the criticality should not appreciably increase the spent fuel source term, as extremely penalising radiolytic dissolution scenarios had already been included in the reference. The effects of irradiation on the argillites are negligible in the face of the thermal environment created by the accident.

To conclude, the effects of a criticality accident are essentially similar, entailing a thermal ambience transfer, to those of a premature overpack failure, apart from the fact that the critical accident would occur later on in the life of the repository (after several tens of thousands of years) and that the high temperature phase could be longer but at a lower level. Continuing this comparison would require a more detailed assessment of the temperature evolution in the medium according to the dynamics of the accident, in principle a complex exercise, the expected benefits of which must be compared with the unrealistic nature of the envisaged configurations. As an initial approach, it is estimated overall that the transport of the radionuclides present in the near field when the accident occurs and over the duration of the accident would be perturbed over a limited distance.

In order to assess whether the distance separating the assemblies is a sensitive parameter, we envisaged a « what if » situation in which the four assemblies of the same container are positioned in contact with each other, while retaining a nominal spacing of their rods. This corresponds to the most penalising inserted reactivity. This situation was studied for various burnup fractions. Even in such a case, the power only rises slowly and there is no mechanical damage. However, the gradual temperature rise in the medium takes the system to boiling point and the phenomenology of the criticality accident becomes dependent on the burnup of the assemblies.

For envelope assembly burnup fractions acceptable in the package, calculations show that the rise in fuel and water temperature limits total reactivity ; this is insufficient however to compensate for the high inserted reactivity. This significant insertion of reactivity leads to fuel core temperatures of about 230°C (an approximately 150°C at the limit between the buffer and the argillites) and the accident persists in terms of duration (30,000 to 40,000 years) [87]. If we consider an accident with burnup fractions more representative of the spent fuel inventory to be emplaced in the repository (of the order of 55 GWd/t), the phenomenology and fuel core temperature should not evolve ; however, the duration of the accident could in such a case be brought down to around ten thousand years. However, such a situation remains purely hypothetical given the minimisation of the voids in the cell and the high metal/oxide expansion coefficient which should lead to a minimum thickness of metal or oxides being retained between the assemblies.

It is hard to describe the effects of this type of situation on the properties of the swelling clay and the rock, as they should remain limited to the near field without compromising the overall confinement capacity of the formation especially as such an accident, were it to occur, would only concern a small number of containers given its improbable nature. There would be no reason for them to be specifically grouped within the same module, unless we imagine the assemblies moving together as a result of a production run manufacturing defect.

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[87] It will be noted that, for even lower burnup (of the order of 18 GWd/t), the slow vaporisation of a part of the water stops the reaction and the system's temperature returns to 50°C in less than two years. The gradual return of water can eventually lead to resumption of the criticality accident. The phenomenology of this « what if » accident could thus be summarised as a series of temperature rise phases within the assemblies – albeit very short given the times characteristic of the thermal phenomena normally considered in the repository – separated by cooling phases. The maximum temperatures reached at each event should also decrease as the fuel is depleted and poisoned. In such a case, given the initial burnup and power level reached, the maximum cumulative duration of the chain reactions is about 40,000 years before they stop for good. Due to a design arrangement, fuels with very low burnup can only be conditioned singly in a dedicated package. A criticality accident in which the assemblies move closer together is therefore excluded in this case.

As an initial approach, we can therefore consider that the criticality accident due to the assemblies moving closer together or an increase in the gap between the rods does not vary much from the normal evolution scenario since the radionuclide transport conditions in the event of a long-term criticality accident differ little, due to the moderate temperature rise in the argillites around the incriminated package. Cases where the assemblies move together completely would require considering a phenomenology which it is more difficult to relate to a known scenario ; however, such cases refer to configurations which are totally unrealistic.

6.2.9.2 Criticality risks due to re-concentration of fissile material

In a normal situation, given the diffusive nature of radionuclide migration, they tend either to remain in place, or disperse into the immediate environment of the package, isotropically. This type of situation contributes to reducing the criticality risk. In a context such as this, the risk of supercriticality could only be associated with massive reconcentration of material in the vicinity of the packages, due to a sudden change in the physicochemical conditions (referred to as a « geochemical front »). A reconcentration such as this could also be encouraged by a hypothetical advective movement of material within the disposal cell. Even assuming that such a phenomenon could happen, for a criticality reaction to be initiated the mass of material would have to be sufficient, the geometry of the mass of fissile material concentrated would have to be appropriate for supercriticality and the surrounding medium would have to favour neutron reflection. A combination of these conditions would seem to be improbable enough for these situations to be rejected by the majority of experts who feel them to be too unrealistic. Nonetheless, to cover this risk more completely, Andra has opted to assess the possibility of fissile material reconcentration following migration over a short distance and to use the most unfavourable geometries possible (spherical), in a medium saturated with water. These latter representations give the mass values that would have to be mobilised to lead to criticality accidents and more fully prove the unrealistic nature of this type of situation given the fissile masses available in the packages.

In the light of the long times needed to envisage the build-up of radioactive materials in a geochemical front, only the long-lived fissile radionuclides present in significant quantities need to be considered. This concerns uranium 235, and to a lesser extent plutonium 239. The possibility for reconcentration of these two radionuclides within the repository were assessed. Two types of processes can generate localised reconcentration of radionuclides : retention in the broadest sense (adsorption, surface precipitation, incorporation/substitution) and solution precipitation (or even co-precipitation of uranium and plutonium). Re-concentration would only seem possible in the immediate vicinity of the waste, at a place where the physicochemical conditions can be heterogeneous. This mainly concerns the corroded layer of metal containers of exothermal waste (which can induce variations in redox conditions), and the immediate periphery of the B primary packages, where the cement is saturated with water (which can lead to a variation in the pH conditions).

The hypothetical criticality risk associated with the possibility of re-concentration was studied for each type of B, C and SF package assuming that the material concentrates in the form of spheres containing water and ^{239}Pu (for the B or C packages) or uranium and water for the fuel assemblies.

Neither did we take account of the elements potentially absorbing neutrons and poisoning the fissile medium, such as :

- the elements making up the host material (degraded concrete, cast iron or steel) ;
- the elements co-precipitating with the ^{239}Pu or the different isotopes of uranium owing to a similar chemical behaviour (lanthanides, zirconium, tin, etc.) ;
- radioactive elements (the other actinides or fission products) likely to provide negative reactivity and which are likely to precipitate with the uranium and plutonium.

The maximum mass over time likely to reconcentrate is compared with the critical mass of the reference medium considered, that is 510 g for plutonium and 260 kg for uranium with a maximum enrichment of 2 % over time (bounding value for UOx fuels).

- **Analysis for the B disposal cells**

The analysis of the maximum quantity of fissile material over time and available in each of the B primary packages shows that the fissile material contained in several packages would have to be concentrated in order to reach criticality. As the B disposal packages contain several primary packages possible overlapping of these hypothetical concentration zones was also studied : the criticality risk would be theoretically possible if the overlaps were perfect and spherical and if all the primary packages of the same disposal package contained the maximum fissile mass. These are completely unrealistic conditions and are included neither in the SEN, nor for defining altered situations.

- **Analysis for the C disposal cells**

A similar approach to the C packages shows that the fissile material available in the entire C0 disposal cell is insufficient to reach critical mass, even if we group all the fissile mass available in the packages emplaced in the cell. For the C1, C2, C3 and C4 packages, owing to the presence of buffers, we look at each individual package and rule out any overlapping.

For C1/C2 waste packages, the risk is excluded because the maximum quantity of fissile material present over time in each package is lower than the mass needed for criticality. For the C3 package, this risk can be considered highly improbable because the fissile material available is of the same order of variable as the critical mass.

For the C4 package, assumed to contain 2.3 kg of ^{239}Pu equivalent, the mass needed for criticality cannot be accumulated before 100,000 years, owing to the source term and the migration rate of the radionuclides. Beyond that time, the hypothetical fissile medium would then only consist of ^{235}U, the product of radioactive decay of ^{239}Pu. The critical mass of this new fissile medium is about 800 g or one third of the available fissile mass. A criticality situation would therefore appear to be unlikely given the numerous conservative assumptions associated with the definition of this mass (spherical geometry, no poisoning by the co-precipitating elements, etc.). In the light of these data, a criticality accident following radionuclide migration outside the vitrified waste packages, in both normal and altered situations, is ruled out.

- **Analysis for the spent fuel disposal cells**

For UOx type spent fuel assemblies, it would seem completely unrealistic for the physicochemical conditions to allow the concentration of 260 kg of « pure » uranium in the mineralised layer of the metal containers, all the more so as the voids in the cells are minimised and the quantity of uranium per assembly is about 500 kg ; it would therefore be necessary for half the uranium in an assembly to be concentrated in pure and spherical form. In the case of MOX fuels, in which the residual uranium 235 enrichment can rise to 4.6 % over time, the reconcentrated mass necessary to trigger a criticality accident is small (40 kg) but is significant enough to consider the risk as improbable, all the more so as it is based on the same penalising scenarios (spherical geometry, absence of co-precipitating elements or components of the matrix, etc.). Only a possible concentration of plutonium in the MOX fuels, independently of the uranium (and in particular the neutron-absorbing uranium 238) could theoretically enable critical mass to be achieved. The very similar precipitation conditions for plutonium and uranium (and the probable presence of other elements with similar behaviour) lead to this scenario being ruled out.

- **Summary**

To conclude, if a geochemical front were to be created in the repository, the assessments show that for all the packages, the risk of fissile material concentration in the repository cannot lead to a criticality accident.

6.2.10 Uncertainties concerning transport phenomena

The purpose of this section is to summarise the uncertainties concerning the transport scenarios adopted to describe the migration of radionuclides in the repository, in a SEN. The subject of solutes transport was already dealt with in the section covering hydraulic phenomena. Here we therefore look on the one hand at the possibility of radionuclide transport in gaseous form and on the other at colloidal transport.

6.2.10.1 Radionuclide transport in gaseous state

Certain waste packages are likely to release gaseous phase radionuclides. Others, although dissolved, could become gaseous in the right chemical environment.

We can first of all rule out gaseous radionuclides with very short half-lives or which are in too small a quantity to have any impact : this is for example the case with tritium, ^{39}Ar or ^{85}Kr. Iodine 129, in disposal conditions, dissolves very easily into a non-volatile and stable iodide. The same applies to chlorine 36, which is present in the form of chlorides.

Carbon 14 is a harder case to assess because it is likely to be present with two different valences (IV or VI). Only valence IV can be mobilised in gas form (CO_2 or CH_4). In this form, 98 % of it comes from spent fuel cladding (B4 waste, B5 waste and spent fuel packages). In the fuel matrices and in the glasses or other B waste, it is only found with valence VI.

Given the capacity of CO_2 to dissolve in water, only radioactive methane is likely to migrate in a non-dissolved form through the repository. The formation of $^{14}CH_4$ for its part requires bacterial activity, which is extremely limited in the spent fuels or the purely metal waste. It can therefore only be envisaged in compacted structural waste mixed with organic waste (reference package B5.1). The carbon 14 contained in these wastes represents a maximum conservative estimate of 75 moles. If this radioactive methane accompanied the migrating corrosion gases, up to the surrounding rocks or into the environment, its impact would be negligible, particularly as it would be diluted in far greater quantities of hydrogen.

The methane would therefore have decayed (the radioactive half life of carbon 14 being 5,730 years) before it could leave the repository. Even in altered situations, in which we could imagine it transiting extremely quickly - for example within a heavily degraded EDZ uninterrupted by seals - it could decay during transfer in the surrounding rocks (at least 50,000 years). Only direct boring into a cell could expose the person to radioactive methane. A conservative evaluation of the induced dose also shows that it cannot exceed 0.03 mSv, and is therefore negligible [97].

In short, this approach shows that all the scenarios, whether SEN or SEA, can ignore the transfer of gaseous radionuclides.

6.2.10.2 Colloidal transport

Colloidal phase transport, distinguished from the soluble and particulate phases by the size (from a few nanometres to a few tens of micrometers) should in principle be considered. It could concern a part of the actinides, as well as elements such as zirconium and niobium.

We will first of all examine the colloids forming in the packages and the clay engineered barrier.

The alkaline conditions in the B cells are such as to considerably limit the action of colloids. Only small concentrations of such molecules were in fact obtained by experiments with columns containing cement-based materials. These small concentrations are explained by the process of flocculation in cement waters with pH of 12.5 and an ionic strength of 0.07 M. The retention of a few radioelements was studied on the small proportion of colloids thus obtained. The result is that despite a high specific surface and high measured retention capacity, in particular for thorium, the low concentrations mean that colloids have little impact on the mobilisation of radionuclides in a cement-based material [21]. So as to take into account possible effects of cellulose degradation products that may have a complexation power, the transfer parameters within the B3 cells have been degraded in the SEN.

In the vitrified waste and spent fuel cells, organic colloids are unlikely to form, in particular owing to the small quantities of organic material. It is nonetheless possible for a small concentration of colloids to be produced as these wastes degrade.

Inorganic colloids can also form during the corrosion processes. The complexing role of the corrosion products is as yet little understood, but we expect them more to have the ability to sorb the radionuclides and retain them in the near field. This ability is not taken into account in the SEN.

Given the low porosity of the medium surrounding the cells, the colloids which are not retained in the near field, in particular by the plug, would be filtered by the sound argillites, characterised by their low porosity. This is the purpose of the « colloid filtering » function (see chapter 3).

As long as diffusive conditions predominate, the size of most of them will prevent them migrating significantly. Only small molecules (about a few nanometres) could diffuse through the bentonite and argillites, but owing to their high capacity for exchange with the ambient medium, they would not cover significant distances through the medium without undergoing chemical interactions immobilising them or separating them from the radionuclide they are transporting.

In these conditions, the uncertainties concern the possibility that this colloidal transport could take place in an unplugged fractured zone, or in bentonite that is not yet saturated. This is comparable to the possibility of migration in a medium with high porosity (for example a highly degraded EDZ) under the effect of advective transport conditions.

The « seal failure » and « bore-hole » SEA, envisage a sensitivity study in which transport within the fractured EDZ takes place without chemical retention properties (null sorption, infinite solubility) for the radionuclides which are covered by the calculation. In anticipation of the SEA results which will be presented in chapter 7, we can note that in such a case, the sound argillites act as an « absorbing barrier » sorbing the radionuclides which could in principle progress in the EDZ, thus preventing their transfer. A phenomenon such as this is representative of what would happen for transport by small size mineral colloids, which would end up entering the argillites and undergo considerable interactions owing to the medium's cationic exchange capacity. The larger colloids, such as clusters of actinides, cannot travel significant distances without breaking up.

6.2.11 Uncertainties over thermal coupling

In the following section we deal with uncertainties concerning the influence of heat on the various processes mentioned so far.

6.2.11.1 Reminder on management of thermal uncertainties

The heat-related coupling effects chiefly concern the cells of vitrified C waste and spent fuels, and their immediate environment. The disposal cells of B5 waste (compacted structural waste) undergo a more moderate and transitory temperature rise, which is unlikely to have a significant influence on the mechanical behaviour of the disposal cell or on transport.

It is important to remember that any uncertainties over the thermal effects are above all covered by the design. We protect against any irreversible mechanical or chemical effects on the rock :

- by limiting the temperature of the packages entering the repository ;
- by separating them with spacers [98].

We also limit the influence of temperature on the radionuclide specifications and migration modes in a thermal environment, by protecting the packages for the necessary period.

From the viewpoint of modelling the thermal field itself, few uncertainties remain. The thermal load assessments were carried out using two modelling approaches, the first two-dimensional and the other, more realistic, taking account of the cell geometry in three dimensions. The container design and the SEN calculations conducted are based on the more penalising 2D model. This gives considerable margins, covering all other uncertainties regarding evaluation of the temperature field.

6.2.11.2 Uncertainties over thermohydraulic effects

In the light of the design measures adopted, the effects of thermal and hydraulic coupling are minimal. Limiting the temperature of the packages entering the repository also limits the thermal gradients. Consequently, thermoadvection is negligible [99] ; the Soret effect (influence of temperature on diffusion) was assessed and taken into account in cases in which it can have an influence (on migration in a thermal environment around spent fuel cells) ;

It should also be noted that the heat given off in the vitrified waste and spent fuel cells is likely to lead to porewater being temporarily pressurised in their immediate environment. Although uncertain, the scale of this effect is in any case small and is secondary to the pressurisation under the effect of gases. It is therefore covered by the SEN sensitivity study envisaging hydraulic pressures imposed by the hydrogen during releases from a failed spent fuel container.

6.2.11.3 Uncertainties over thermomechanical effects

The impact of thermomechanical effects on the vitrified waste and spent fuel cells primarily concerns the argillites in the immediate vicinity of the cells. They can temporarily undergo deformations linked to thermal stresses, leading to a moderate and reversible increase in the EDZ around the edge of the cell. This damage is in the end compensated for by argillite creep. The effect is more appreciable close to the packages, where the temperature field is highest, and less so at the plug which is further away. So, any increase in the size of the EDZ, at a sufficient distance from the plug, has no significant impact on the disposal cells, insofar as diffusive conditions continue to be guaranteed by contact between the plug and the rock (for C waste disposal cells) or by the clay engineered barrier (for spent fuels). Even if the damage were to propagate as far as the plug, its swelling would make up for the EDZ.

For the spent fuel disposal cells, given that this damage would remain moderate, this does not change the representation of the SEN, which already envisages the presence of a lasting, unclosed fractured zone around the cells. In order to manage the uncertainties, we include the possibility that a fractured zone can be created in the C waste cells and not eventually taken up by the plug (for example by associating this uncertainty with failure a swelling clay swelling). An isolated defect of this type is envisaged in the SEN and has no influence on the hydraulics of the repository. Systematic failure of all the C cell plugs is included in the « seal failure[88] » SEA.

The thermomechanical load experienced by the containers and overpacks within the vitrified waste and spent fuel cells is covered by their mechanical sizing. It is not such as to compromise the functions performed by these components. The same applies to the cell coating (the liners in concepts without clay engineered barrier). Any deformations would be minimal and could not on their own generate additional forces on the environment of the containers and liners (and thus on the argillite itself).

6.2.11.4 Uncertainties on thermochemical couplings and coupling with transport

Discrepancies in the geochemical composition of the Callovo-Oxfordian waters and modifications in the transport properties can result from the effect of temperature on the environment of the thermal waste disposal cells. These effects were measured on rock samples [21] with thermal cycles of up to 80°C.

The effects of temperature on the diffusion coefficients are to be considered. The temperature field can in fact modify the phenomenological values of the diffusion parameters (De, ω_{acc}) representative of argillites that are not thermally disturbed. A specific experimental programme was set up to check whether the effect on the diffusion coefficients of tritium and chlorides in the argillites is modified by temperature. The results acquired show that for a temperature cycle of 20°C to 80°C, the effective diffusion coefficient of tritiated water increases (from $2.5\ 10^{-11}$ to $1.2\ 10^{-10}$ m^2/s) but that on returning to 20°C a value 20 to 30 % higher than the initial value was measured, which may indicate a partially irreversible effect but which is in any case negligible given the order of variable. A reversible dependency between the effective diffusion coefficient and temperature was included in the Callovo-

[88] This SEA takes into account a more pessimistic situation in which a continuous fractured zone is present around all the sealings. This calculation case encompasses the failure of the plugs alone.

Oxfordian volumes subjected to temperatures in excess of 22°C. This gives a correction function based on the diffusion coefficients curve for the chloride ion measured at different temperatures [88].

With regard to solubility, the effects of temperature are not explicitly taken into account. This is linked on the one hand to a lack of thermodynamic data for all the equilibria to be considered and on the other to the fact that the experimental assessments so far made show that the effects remain negligible. The decision was therefore taken to not explicitly take account of solubility dependency on temperature. This choice is the most representative in the light of current knowledge.

For the Kd, a specific experimental programme was run and temperature dependency was included for those radionuclides on which an effect was observed. For most of the radionuclides, the Kd is considered to be independent of temperature within the thermal variation ranges considered, given current knowledge.

These results tend to show that the thermal effects on the transport properties are limited, up to temperatures of 80°C. However, it should be stressed that the experiments concerned thermal cycles that were necessarily short when compared with those that will be experienced by the Callovo-Oxfordian. The parameters for transport at temperature are still relatively uncertain as things currently stand. The conclusions that can be drawn from the calculations made using defective containers and overpacks should be taken with a degree of prudence ; in particular they do not invalidate the relevance of the design measures taken to prevent transfers in a thermal environment.

In addition, from the hydraulic viewpoint, the intrinsic permeability only depends on the structure of the pore space. We do not therefore expect this to vary within the temperature range considered. However, the temperature fields produced by the waste have an impact on the dynamic viscosity coefficient [88]. The relative effect of temperature on the permeability was evaluated by modelling. This shows that the vertical component of the advective velocity of water through the Callovo-Oxfordian can be tripled (at 80°C) during the exothermal period.

This dependency of permeability on temperature was not used to represent the Callovo-Oxfordian in the SEN, insofar as the risk is predominantly diffusive with a good safety margin. However, the calculation does take account of this relationship once transport is likely to become advective (in particular in the drifts or in the EDZ).

6.2.11.5 Summary of thermal effects

The uncertainties over incorporating the potentially important couplings induced by the heat in the repository, are very significantly reduced by design measures.

These latter are associated with conservative assessments of the extension of the thermal field, ensuring that they meet the objective of a prudent safety assessment.

We do however accept that in the cell near field, and even though covered by the SEN, the thermomechanical interactions could - if they were larger than expected - lead to altered situations covered by the « seal failure » SEA.

Including the possibility of failed containers in the SEN however leads to having to envisage transports in the thermal phase. Within this context, the first data resulting from experiments in representative conditions were taken into account (diffusion coefficients, distribution factor, etc.). Taking account of a more generalised packaging failure scenario than that used in the SEN, and comparing its results with those of the reference scenario, enables the sensitivity of the impact to this type of data to be evaluated.

6.2.12 Technological uncertainties

Uncertainties as to the implementation of the techniques specified for the construction of repositories or the emplacement of packages are reduced by the strategy that was adopted, which consists in relying first of all on technologies which have proven themselves in mining or nuclear contexts.

In such a context, the uncertainties only stem from :

- certain techniques which, although already employed in the industry or already tested experimentally, are partly specific to the repository and are not frequently used in other industrial contexts. As part of a safety analysis looking to cover the uncertainties as broadly as possible, it would be wise to envisage the possibility that these technologies are, on isolated occasions during operation of the repository, incorrectly controlled ;

- problems of quality control in implementation of a technology, in other words because of random causes leading to an unexpected result, even though the technology itself was validated (for example a defect in the manufacture of a package).
- effects induced by accidents which may occur during the operating phase and which could have an influence on long-term safety (for example, fire by damaging the rock). The measures taken to prevent such incidents are mentioned in chapter 4. The measures to be taken into account to remedy the consequences of such accidents (replacement of certain components, on-site repairs, etc.) will need to be defined case by case on the basis of a detailed definition of the architectures. Such issues have not been specifically approached at this stage.

The first two situations are mentioned in turn below.

6.2.12.1 Uncertainty concerning the technologies implemented in the repository

A certain number of operations, carried out for different reasons, share the common goal of avoiding the creation of excessive voids, even though voids cannot be totally prevented within the repository. On the one hand they exist within the primary packages, for reasons linked to the packaging techniques or the safety of preliminary interim storage. The primary glass containers thus contain a void at the apex, as do the bitumen encapsulation drums. Clearances are also needed to allow easy insertion of the primary packages in their containers and the containers inside the cells. A minimum amount of clearance also facilitates reversibility.

Here we deal with the questions of emplacing the materials in the various repository compartments : in the cells, in the drifts, at the seals, in particular with respect to controlling voids.

- **In the disposal cells**

The voids have an influence on several types of phenomena, without it always being possible to give an opinion on whether or not they make a positive contribution to safety. For example, the presence of voids delays resaturation of the disposal cells – which is neither positive nor negative but which must be taken into account in the analyses - as well as the rise in the pressure of the gases produced by corrosion of the metal elements. They may however induce mechanical damage of the surrounding materials (absence of ground support for the structure, shear stresses on incorrectly stacked packages, etc.).

In general, we however look to minimise them, in order to control long-term mechanical deformations within the structures. A maximum void level of 5 % has been chosen, in particular for the B disposal cells in which the spaces between the packages are not filled in. The long-term behaviour of the materials promotes resorption of the voids, owing both to the expansive nature of the metal corrosion products and to the swelling of the clay. At those places where a significant void is likely to be left by operating conditions (at B waste disposal cell loads for example), it should be filled in. Care will also be paid to the compactness of the filling materials (backfill, bentonite) to avoid clearance developing allowing either excessive creep of the argillites, which is harmful in the very long term, or the creation of channels favourable to the transfer of radionuclides (at the interface between the seals and the rock for example).

The uncertainties over the quantity of voids left in place therefore stems from uncertainties :

- over the void within the waste primary packages, which is as a general rule a specified parameter for recently produced packages and can if necessary be measured on the older packages ;
- over the long-term behaviour of the materials (risks of failure of the bentonite to swell, expansion of corrosion products) already dealt with in section 6.2.8 ;
- over the technologies used for installing the repository materials, covered in this section.

In the B waste disposal cells, the interior of the overpacks minimises the installation gaps. However, there are no plans to fill in the voids between the B waste packages (Figure 6.2-21). An operation such as this would prove difficult in practice and would also interfere with the reversibility of the concept. The mechanical strength calculations within the cell take account of the clearances between packages. Installation of these concrete blocks in a regular stack using remote-operation is not a complex operation. There is considerable experience feedback and the necessary tools are available. However, perfect, flawless alignment is hard to guarantee and minor deviations in the positioning of the packages may occur, developing significant stresses on the concrete hulls. In order to cover this uncertainty in particular, Andra does not at this stage rely on the mechanical strength of the packages to guarantee the long-term safety functions, other than in a variant. If such a variant, assigning a mechanical durability role to the overpacks, were eventually to be adopted, more detailed studies on the strength of the package stacks and the acceptable installation defect tolerances would be necessary.

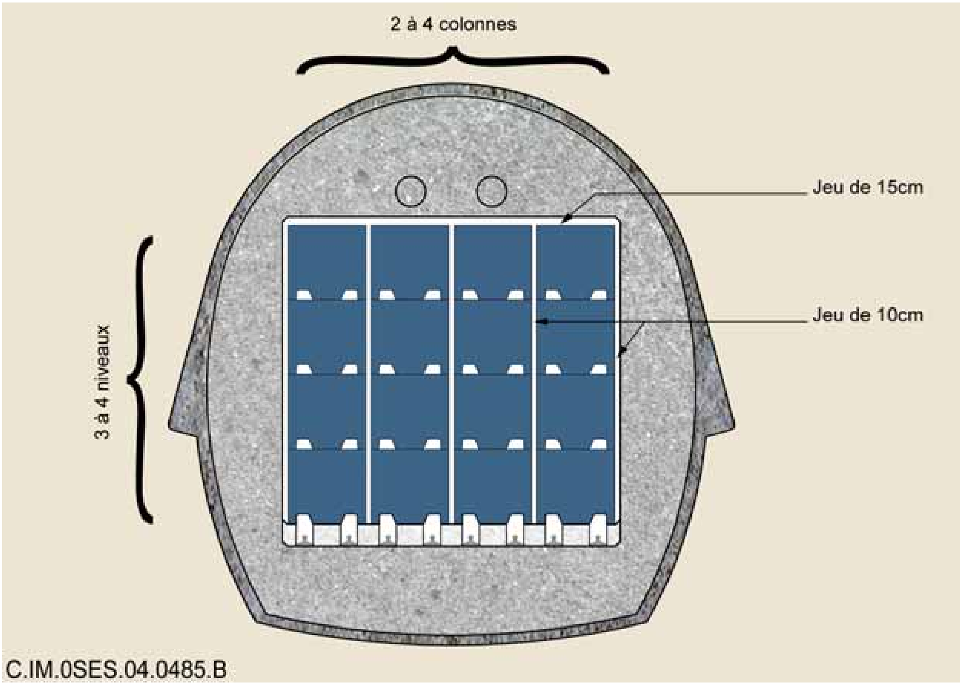

Figure 6.2-21 Residual installation gaps in the B waste disposal cells

In the C waste and spent fuel disposal cells, the voids are small and controllable. They are either inside the packages (within the CSD-V, between the elements of the spent fuel container) or between the liner and the container to allow emplacement (see Figure 6.2-22). In both cases, they would not appear penalising according to the safety analysis : the presence of air in the voids, which could oxidise the containers either directly, or through radiolysis by radiation from the packages, is designed-in. In terms of the long-term evolution, the slow creep of the geological medium, the swelling of the bentonite in the engineered barrier (for those concepts comprising it) and the expansion of the corrosion products, should eventually fill them in until overall mechanical equilibrium is reached within the cell. Closure via the plug maintains diffusive conditions within the cell. The presence of isolated voids poses no problems.

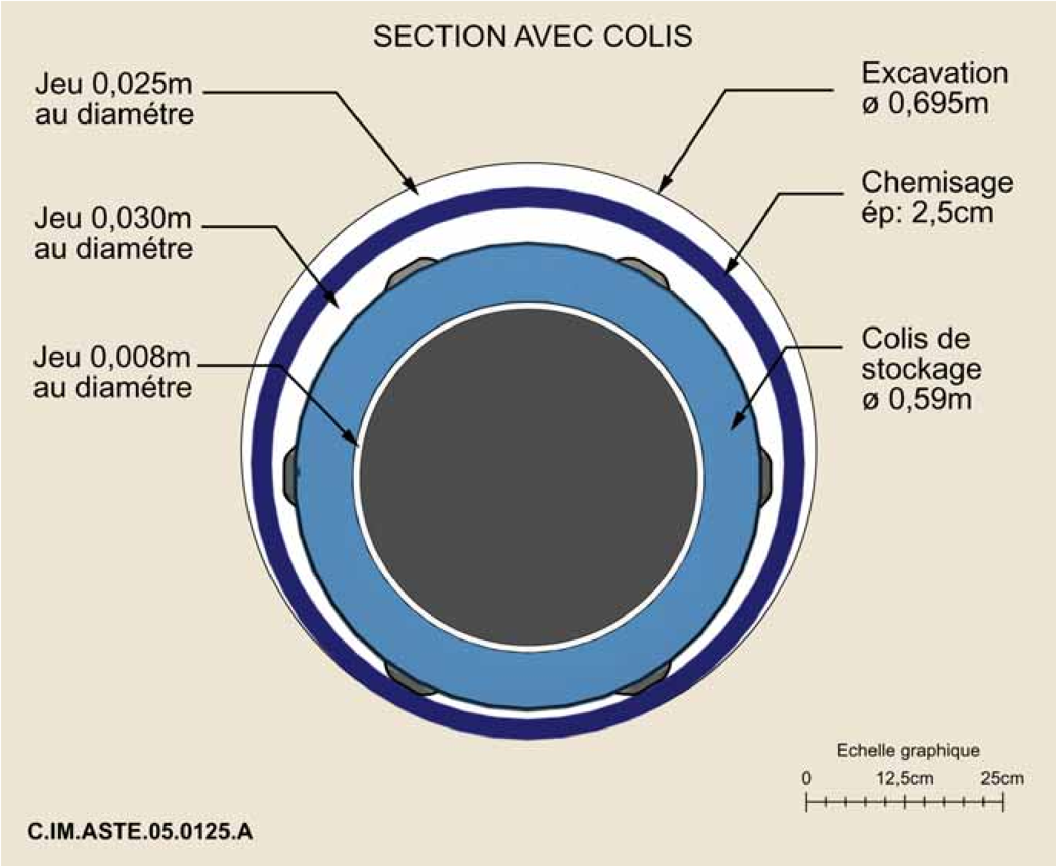

Figure 6.2-22 Residual installation gaps in the C waste disposal cells

- **In drifts**

The installation of backfill with good contact between the rock and the filler was tested in the Aspö laboratory [100], using a technique involving the positioning of successive angled layers offering uniform compaction up to the roof of the drift (see Figure 6.2-23). The support backfill in the vicinity of the drift seals has the role of confining the seal after degradation of the concrete retaining plug. Given the seal resaturation time, this is a line of defence offering additional safety, but which should not actually be used as seal swelling takes place in a maximum of a thousand years, compatible with the lifetime of the concrete. Failure of this « support backfill » would therefore in theory have no consequences.

By convention, one can however assume uncertainty over the specification and/or emplacement of the backfill, leading to a rock support failure and poor swelling of the seals. A situation such as this would be characterised by a loss of efficiency of all the seals fitted in the horizontal drift. If we associate this with an uncertainty on the evolution of rock damage in the long term, we can also imagine that the support failure leads to deterioration of the properties of the EDZ. Such a drift seals failure situation, for which the sensitivity analysis uses highly degraded properties of the fractured zone and the micro-fissured zone, is envisaged in the « seal failure » SEA and is in fact covered by a situation involving failure of all the seals.

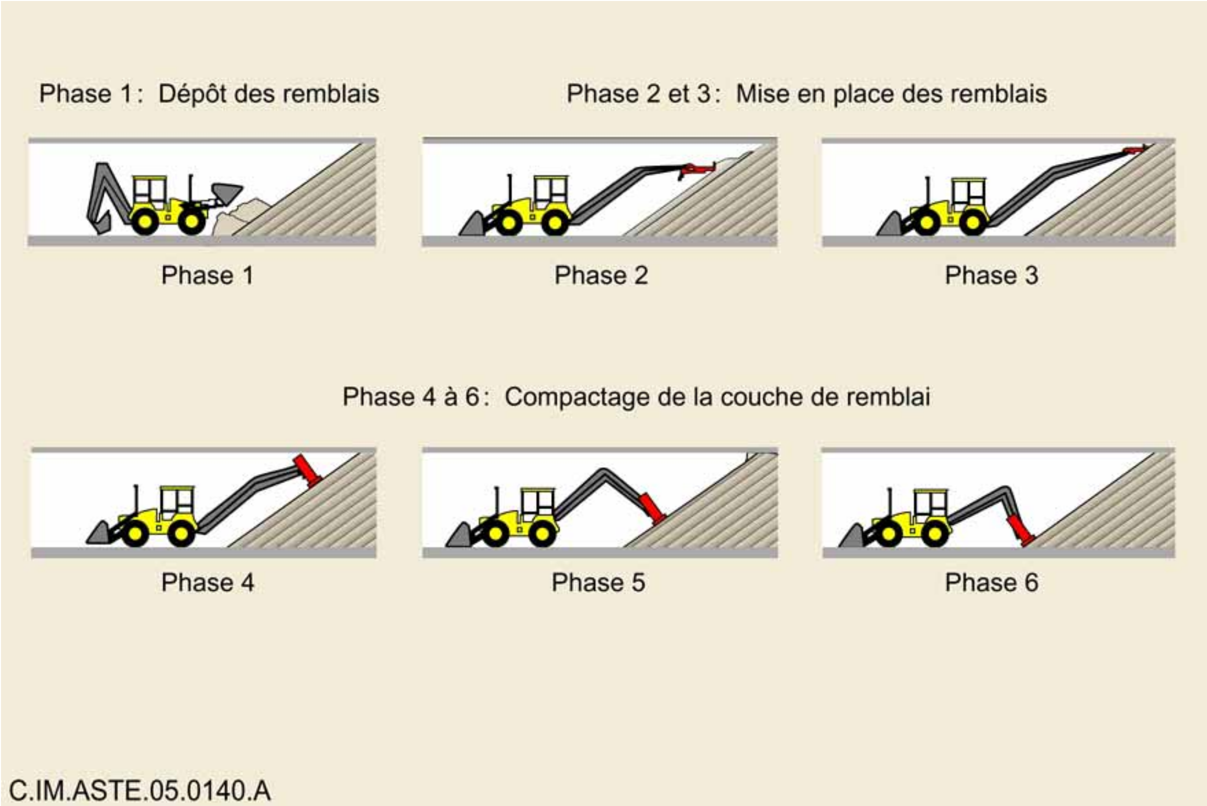

Figure 6.2-23 Schematic diagram of drift backfill installation

The same uncertainty can be envisaged for the concrete backfill in the shaft. Its mechanical performance, which is designed to support the weight of the retaining plug and the shaft seal above it, is not however particularly demanding. Failure of this backfill could lead to failure of the retaining plug and consequently to incorrect swelling or slippage of the shaft seal, which is included in the « seal failure » SEA which envisages failure of this component.

Installing the ground support and then the coating is a conventional mining engineering operation and only the requirements concerning durability for centuries and even longer are more specific to the repository. Shrinkage cracks, even if penetrating, would not lead to any mechanical stability problems over such a period. Only the permeability is increased (but it is not required as no safety function is assigned to the ground support or the coating with respect to water flow limitation).

- **Swelling clay seals and clay engineered barriers**

One final uncertainty concerns installation of the seals and clay engineered barriers. The problem arises differently in the drifts, in other words for the horizontal or sub-horizontal seals, than for the seals in the shafts.

Installation of the bentonite seals was experimentally validated in the Lac du Bonnet laboratory in Canada, as part of the TSX experiment (see Figure 6.2-24). This experiment [43] utilised and hydraulically tested drift seals. Although the test drift was excavated with explosives, thus creating significant damage, experience shows that the grooves interrupting the damaged zone can be created using mechanical means (abutting boreholes) and are effective. The swelling capacity of the bentonite reduces the effect of any installation heterogeneities.

Transposition of this test to a clay context is self-evident with respect to installation of the seal body, while there are specific aspects for construction of the hydraulic cut-off. Its feasibility was shown with Opalinus clay similar to the mechanical behaviour of the Bure argillites, in the Mont Terri laboratory. It is currently being verified as part of the KEY experiment in the underground laboratory. Uncertainties concerning deployment are thus at this stage slight and could be reduced even further with the results of the future experiments.

Figure 6.2-24 Experimental seal consisting of bentonite bricks - TSX test conducted in Canada (on an industrial scale, these bricks would be larger)

The first KEY tests were conducted in an experimental drift at level -490m, the most clayey level. Three grooves were cut with a saw specially designed for this test based on experience acquired at Mont Terri. In the course of this test, the feasibility of cutting two metre-deep grooves to a thickness of 30 cm and the good working order of the saw were demonstrated. This test also showed the good behaviour of the argillite during sawing operations, especially through the quality and stability of the vertical walls obtained (cf Figure 6.2-25). The first tests to verify system performance indicated that the grooves interrupt circulations in the damaged zone effectively. Tests are continuing with the emplacement of bentonite bricks

Figure 6.2-25 KEY experiment : Saw-cutting of a trench in the wall of the experimental drift, then with a view of the walls.

With regard to the clay engineered barriers, the technology using insertion of bentonite rings limits the risk of incorrect installation, which could be greater with a stack of bricks.

However for the seals and the clay engineered barrier, the safety analysis requires inclusion of the risk of imperfect installation of the swelling clay elements. The effect of these contact faults is attenuated by the swelling and plasticity of the bentonite. The final homogeneity of the seal, in hydraulic terms, depends on the possibility of filling in the voids during swelling. When the FEBEX experiment was dismantled, it revealed good healing of the bricks, as did dismantling of the OPHELIE mockup in the Mol laboratory in Belgium. One cannot however entirely rule out that permeability will locally be a little higher along imperfect contacts with respect to the core of the rings. If this were the case, it could encourage an extension of the chemical reaction front linked to disturbances along the heterogeneities. Situations such as this would seem to be unlikely. They are however taken into account in the definition of altered situations, representing the effect of bricks that are poorly compacted and poorly healed (for example owing to insufficient swelling pressure) owing to increased permeability of all the bentonite structures associated with loss of the protective role of the plug against the content of the cells. The « seal failure » SEA thus envisages a contact defect between the seals and the rock (from the calculation viewpoint, equivalent to a contact defect within the bentonite) coupled with high pH within the vitrified waste disposal cells and degradation of plug permeability leading to rapid release by the vitreous matrices being taken into account in the sensitivity analysis.

A non homogeneous installation or an heterogeneous swelling of the buffer could result in excessive constraints on the spent fuel container. Its mechanical dimensioning should be sufficient to bear them. In any case, the isolated package failure envisioned in the SEN encompasses such situation.

Installation of a seal hydraulic cut-off is a delicate operation which in particular entails removal of the ground supports, excavation of the rock in order to cut the grooves, then installation of the bentonite bricks inside these grooves with the minimum of gaps. The risks are that loss of confinement of the argillites, caused by removal of the ground support, or excavation of the cut-off, will generate secondary damage of the EDZ leading to an unanticipated extension of it. Given the small thicknesses concerned, and the limited duration of the operation, this risk is however slight.

The hydraulic cut-off as envisaged today on the basis of these evaluations also has the ability to intercept the entire damage, including that which may be created by temporary loss of confinement (see Figure 6.2-26).

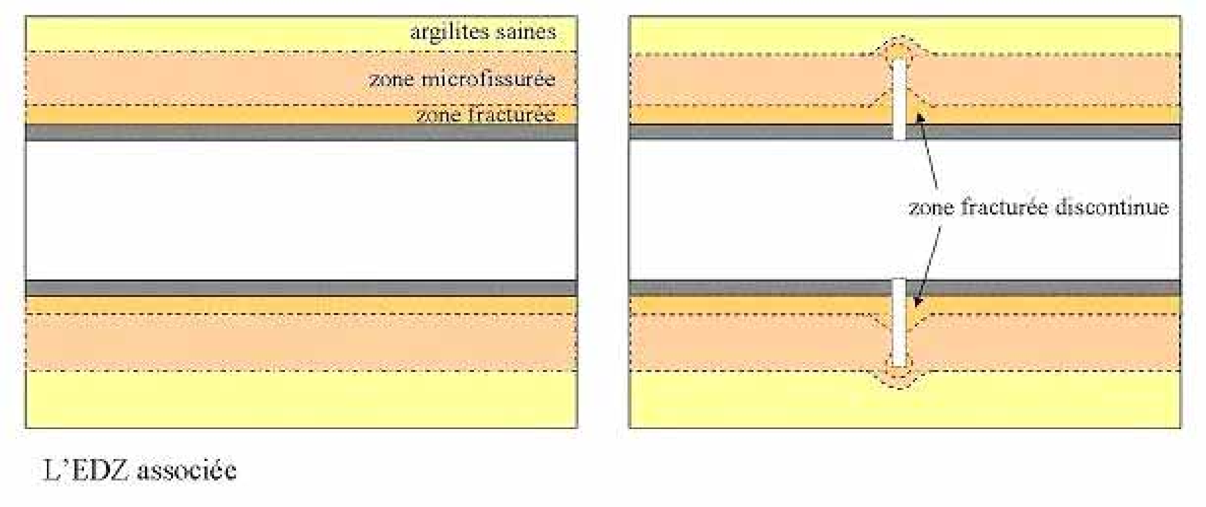

Figure 6.2-26 Conceptualisation of the fractured zone discontinuity at the grooves (located at maximum depth on the transposition zone, i.e. 630 m)

In these conditions, failure to control the conditions for installation of the seal hydraulic cut-off, leading to undetected secondary damage, would seem to be unlikely. A case such as this would be an altered situation, which we nonetheless include in the « seal failure » SEA, which envisages simultaneous failure by all the anchored seals.

6.2.12.2 Quality control errors

The importance of human factors, in other words errors that can be committed by the operators, is underlined as a major risk in industrial facilities. Particular attention is given to it. The experience feedback acquired by Andra on the Aube repository for example shows that the majority of non-conformities is due to an error by the personnel. The good level of personnel training and the regular checks on activities enable serious faults with potential safety consequences to be eliminated. Nonetheless, the simple fact that this type of phenomenon cannot be ruled out, even though its effects can be controlled, demands permanent vigilance.

At the feasibility study stage, it would be premature to assess the human factor risks on each workstation and the potential long-term safety consequences. We initially only identified the types of errors that could result from repetitive actions (operations carried out regularly during operation of the repository and which are « routine » in nature) concerning the manufacture or installation of elements with a long-term safety function.

The first important elements from this point of view are the disposal containers (B waste overpacks, vitrified waste overpacks, spent fuel containers).

For the B waste overpack, the manufacture of concrete containers can entail manufacturing defects, even if the concrete installation techniques are well known. From the point of view of homogeneity of the final material, the main problem lies in the shrinkage cracks due to removal of the water from the concrete once it is in place. On large scale structures, the cracks which form can lead to a reduction in the permeability of the structure and thus impair its durability. There are techniques to control this cracking, in particular the use of an appropriate concrete formulation (such as « high performance ») allows a highly significant improvement over conventional concrete. The cracks may be weak points with respect to subsequent evolution of the material. It should be noted that in the case of packages intended for geological disposal, the design limits the areas of fragility by optimising material continuity. The visual check on faults (shrinkage cracks, chipping from shocks) is easy and poses no particular technological problems. A quality control error is therefore unlikely. Nonetheless, if such an error were to occur on a random basis, it would not compromise the overpack safety functions, which do not rely on its hydraulic properties.

For the C waste overpacks and the spent fuel containers, their manufacture can entail chemical composition, metallurgical or welding faults. The composition of the steel selected entails few constraints and its systematic ladle and part spectrometry analysis is easy and poses no particular problems. Metallurgical consistency and electron beam welding (see Figure 6.2-27) could be automatically checked by ultrasounds. These technologies have been widely used for many years, in particular in the nuclear sector. If failure to detect a surface defect can be ruled out, that of a subsurface defect, even if unlikely, cannot. Given the design margins employed, a subsurface defect would have no consequences on the strength of the containers. Depending on the ambient conditions when generalised corrosion reaches this defect, an extremely momentary acceleration of the phenomenon could occur. A defect with consequences on the tightness of the container could in principle be ruled out at this stage in the studies but as a precaution we nonetheless retain the occurrence of such defects in the SEN but without specifying their origin.

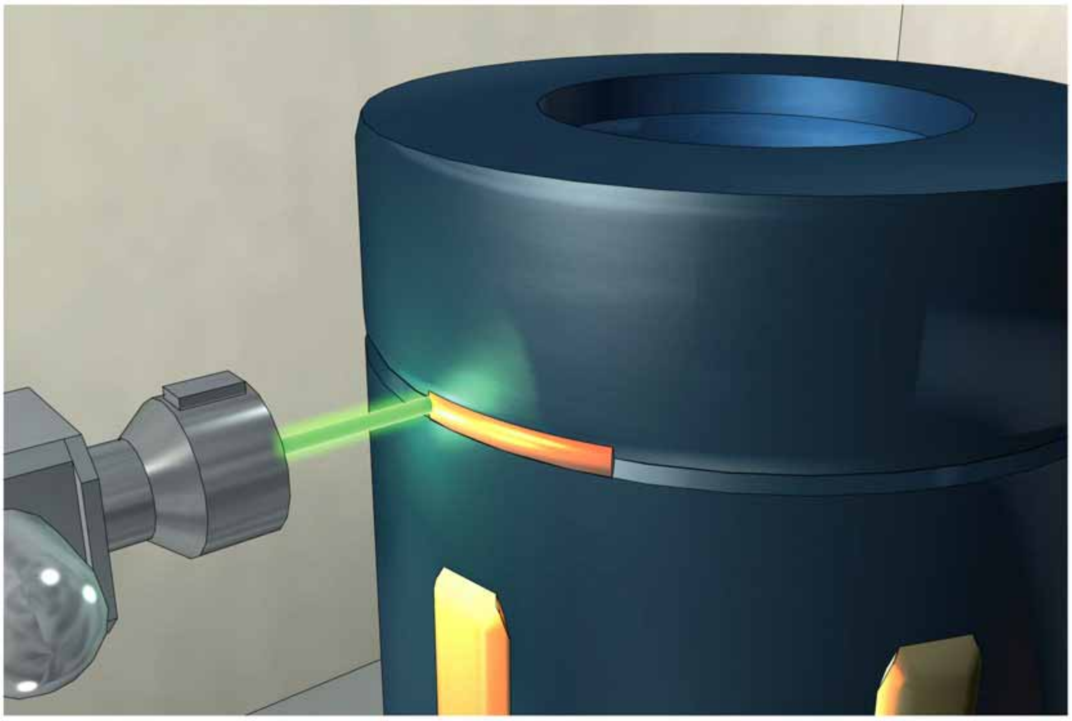

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Figure 6.2-27 Detailed representation of an overpack (or container) weld.

These possible defects, which would slip through all the inspection systems, would in principle only affect a very few containers dispersed around the repository, given the expected inventory (several thousand units) and the defect probabilities usually considered in this field (10^{-4} to 10^{-5}).

It should also be noted that the defective container adopted for the SEN is an extremely bounding item with respect to the nature of the defect. In the absence of any clearly identified cause it is in fact hard to postulate the defect geometry and evolution kinetics. Andra adopted a highly pessimistic approach which involves considering complete disappearance of the container's confinement function about one century after emplacement in the repository[89]. This leads to thermal phase releases.

[89] To be exact for the sake of calculation, 200 years after the radionuclides started to decay, i ;e. after their formation within the reactors.

Another type of quality control error can concern not the individual inspection of the manufactured packages, but the maintenance and calibration of the measuring devices themselves. In the case of an observed drift in the measuring instrument, standard practice involves back-tracking to the earliest date on which the drift could have occurred and checking the condition of all the manufactured objects that could have been inadequately inspected. This type of situation could take place in a container manufacturing facility, either because of drift in a measuring instrument, or because of insufficient training of an operator in charge of visual inspections. Typically, the redundancy of the inspection facilities for the more important items prevents this type of situation. If drift is however observed, the reversibility of the repository allowing easier removal of the packages, would make it simpler to extract and inspect them. A whole series of defective containers resulting from a quality control error is therefore in principle excluded. A situation such as this was however used as the « what-if » basis for the « package failure » SEA.

With regard to the cell plugs, incorrect installation or more generally poor contact with the Callovo-Oxfordian wall, would lead to an isolated loss of their safety functions, an event that has been included in the SEN in the form of a few isolated failures in the vitrified waste cells (where the plugs are more numerous and where the absence of clay engineered barrier means that incorrect installation cannot be compensated for). This isolated defect would seem to have no influence on the repository hydraulics in the SEN.

For the other repository elements (shaft seals, drift seals, etc.), their small numbers allows specific quality control of installation, which avoids the pitfalls of « routine ».

6.2.12.3 Summary of « technological » uncertainties

The technologies proposed in the Dossier 2005 are based both on practices commonly employed in similar industrial contexts, and on demonstration tests conducted internationally. Current programmes should provide a certain amount of clarification on the conditions in which the operations could be carried out.

Generally speaking, the SEN representation is one in which there is « minimal » sizing of the components : millennium duration for the vitrified waste overpacks, low performance of all seals, no retention function by the backfill.

Furthermore isolated construction faults cannot be excluded, no more so than in any industrial installation. A systematic approach consisted in :

- including in the SEN isolated faults for those elements present in large numbers (cell plugs and containers) ;
- including more systematic faults in altered situations.

6.2.13 Uncertainties concerning the duration and modalities of the operating and observation phase

The repository operating and observation phase is when physicochemical phenomena take place, some of which are similar to those which subsequently occur during the post-closure phase, while others are caused by the particular conditions of an open, ventilated repository. These phenomena may be important for the long-term safety of the repository insofar as they determine its condition at the beginning of the post-closure phase.

Consequently, the possible variability in operating conditions can lead to the initial state of repository evolution after closure being impossible to determine. The SEN used a repository operated during a century. It is therefore important to check whether, assuming that the repository were to be operated and left open for longer, the scenario would be modified or not. This in particular enables us to assess whether or not reversibility has a safety impact.

The main phenomena involved have already been covered, from another angle, in the previous sections. We will therefore here mainly recall the aspects already dealt with.

The main phenomena that can be influenced by the operating period are :

- desaturation of the rock and repository materials, the scale of which depends on the ventilation period of the structures. The desaturation rate however falls off with time. In particular, once it has reached the fractured zone of the cells, this phenomenon then only progresses slowly in the microfissured zone and the sound argillite for several centuries. Furthermore, the degree of saturation of the structures would not seem to be a key parameter in the SEN, which ignores the hydraulic transient ;
- oxidisation of the argillites leading to possible mineralogical transformations and in particular to the production of sulphates by oxidisation of pyrite. The sulphates could prove to be aggressive, in particular for the concrete. In the light of the masses of concrete involved, they would appear to be able to withstand such an attack. In addition, oxidisation can also cause transformation of the organic material contained in the argillites in very small quantities. This can lead to the formation of colloids, but at levels which are insignificant for radionuclide transfer ;
- carbonation of the concrete, caused by its prolonged exposure to air, leads to embrittlement of its structure. This phenomenon progresses on a front within the concrete and its effects (reduced porosity through precipitation of calcite) tend to slow down as it progresses. Its extension would not exceed a decimetre over a century of operation and would not appear to be a design factor thereafter. It would need to be observed during the course of coating and lining maintenance. It should be noted that it does not concern the retaining plugs which are installed at the moment of closure ;
- the risks of accelerating corrosion of the metal components of the repository – mainly the waste containers and overpacks - by maintaining oxidising conditions for a longer period of time. We have seen that they can be managed by design, closing the disposal cells with a removable cover limiting air renewal. Long oxidising phases can also be taken into account if necessary in the sizing of the steel thicknesses ;
- increasing the mechanical stresses on the repository structures, with gradual creep of the rock. As the containers are sized for millennia, the question above all concerns the concrete coating and lining of the cells. They are designed to resist for several centuries, or even longer. Reducing uncertainty concerning the rock creep rates would, in a possible subsequent phase of the project, allow clarification of the conditions for assessment of this duration and optimise the design and sizing of the coating. A specific problem lies in the removal of elements installed at construction and that have to be removed at closure, with the risk of loss of host formation confinement. We have seen that removal of the coating at the hydraulic cut-offs posed no particular problems (see section 6.2.12.1). Removal of the vitrified waste cell loads liner and installation of the final swelling clay plug could, if not carried out with the necessary care, lead to a loss of confinement of the rock which would be prejudicial if it occurs at a late stage (after several centuries, when the rock creep has already filled in the liner clearance). Such a situation would not appear to be probable ; and in the worst case would lead to a contact failure which could be compensated for by swelling of the plug. If a plug failure persisted locally, this would be taken into account in the SEN. A more generalised failure would be covered by the « seal failure » SEA.

More generally, the repository observation programme during its operation [101] will enable the damage to the structures to be assessed, to identify whether unanticipated phenomena are occurring and to intervene with preventive or corrective maintenance. If suitably prepared, a long operating period would not seem to be a handicap for the long-term safety of the repository.

6.2.14 External events

Here, we first of all recall the list of « external » events (in that they originate in an event occurring outside the repository) considered during the study. These events were identified from basic safety rule RFS.III.2.f. and international databases (mainly the OECD's FEP-CAT base [83], but also the FEP 2000 bases [84] and a base created for Andra from international references). These bases cover a broad range of events of diverse types and origins (natural, human).

The only exclusions concern intentionally malicious human acts and natural cataclysmic events (such as a falling meteorite for example).

Malicious acts are by their very nature unpredictable. However, during operation of the repository and in the initial period following closure it should be possible to provide protection against them using conventional site guarding and access control measures. Subsequently, owing to its distance from the surface and the packaging of the radioactive materials it contains, a closed repository would be a difficult target and one that would be of little interest to a potential saboteur. Furthermore, if it was felt that a deep repository could be of possible interest to anyone with malicious intent, it would still be possible to maintain surveillance around the site, until its location was possibly forgotten. In this latter case, the repository once completely forgotten, would then no longer be a potential target.

Consideration of cataclysmic events refers us to the field of extremely improbable natural accidents with consequences that would affect at least a region or possibly more. These events are not considered in the analyses because their consequences would go far beyond the context of the repository and its presence would not be a really significant additional risk factor. It should be noted that major but more localised or frequent natural events (volcano, earthquake, flood) are however taken into account, particularly through the choice of the site (see below)). The cataclysmic events are considered to be « beyond-design-basis » as indeed they are for all industrial installations.

We then have to take account of a number of events which can fall into three categories :

- gradual natural processes (climatic evolution, geodynamic evolution) ;
- isolated natural events (earthquake, volcanic activity) ;
- inadvertent human intrusion.

6.2.14.1 Climatic evolution : glaciation

The climatic scenarios defined in the BIOCLIM project [102] indicate an alternation between glacial and inter-glacial periods which follow each other with a cycle of about 100,000 years. The identification of these cycles is based primarily on analysis of core samples of Antarctic ice, which give a picture of climatic changes over the past 400,000 years (see Figure 6.2-28).

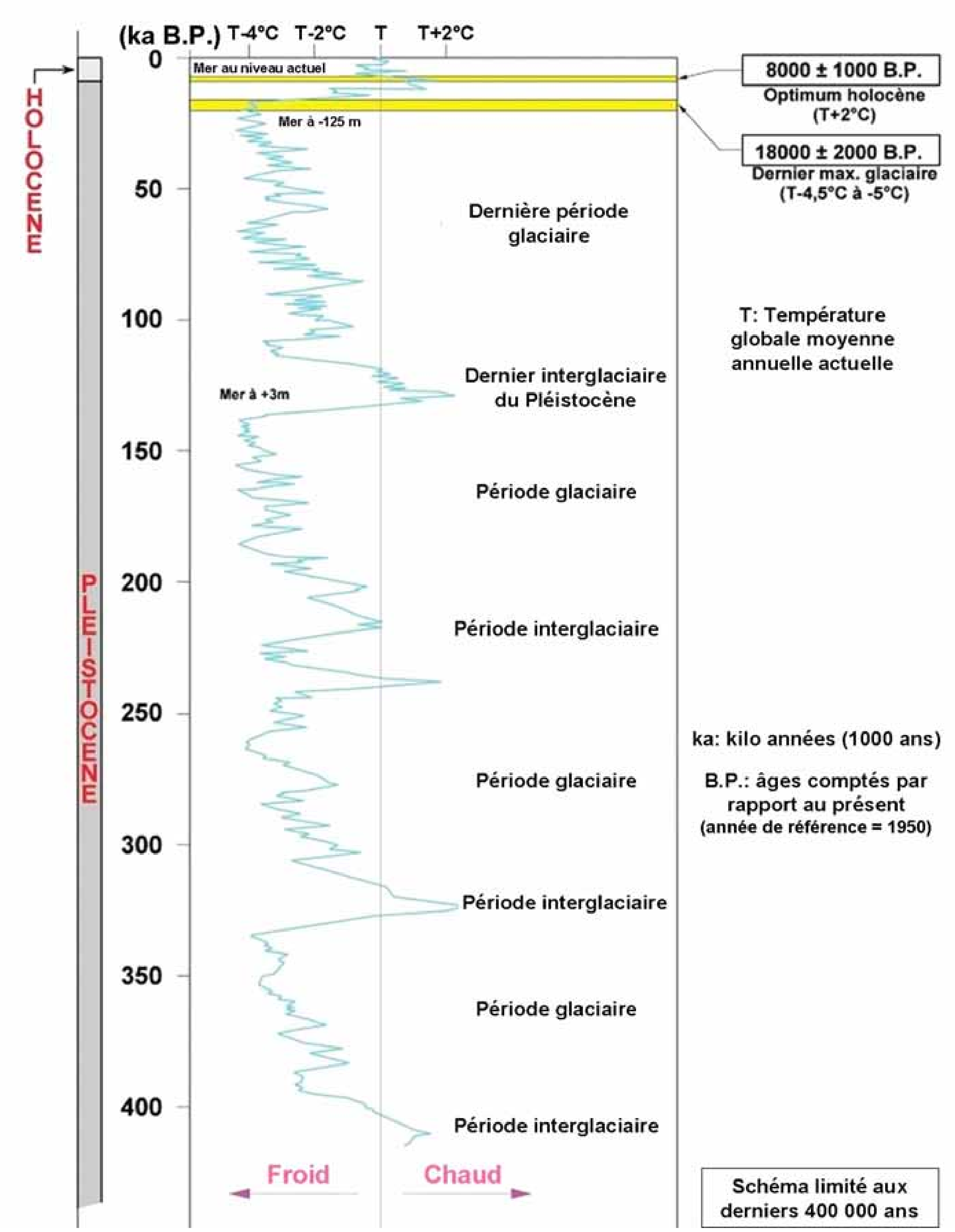


Figure 6.2-28 Variations in Antarctic temperature (Vostock) over the last 400,000 years

The range of possible future climate situations provided by BIOCLIM covers the climatic situations envisaged by the basic safety rule RFS III.2.f.

With regard to the biosphere, the SEN does not include a dynamic vision of climatic evolution, in order to avoid cumulating uncertainty over the dates of appearance of radionuclides in the biosphere with the anticipated climate change time-frame. We chose a « temperate » type biosphere after ensuring that a critical group living in a glacial climate would be less exposed than the critical group used in the reference. The choice of a temperate biosphere is therefore conservative.

We will also exclude continuous evolution of the surrounding rock formations under the effect of erosion, but two configurations were covered in the SEN, one representing the site in its current state and the other representing the site in a million years. These two configurations describe two contrasting situations and thus give an idea of the possible variations in the impact according to changes to the surface environment. The geodynamic model takes account of the greatest conceivable erosion predicted by the BIOCLIM scenarios and therefore covers the uncertainty on the evolution of the surrounding rock formations. In particular, the model amplifies topographical contrasts : major incision of the valleys and lower ablation rate of the limestone plateaux We would also recall that the surrounding rock formations play a minor role owing to the choice of outlets close to the repository (50,000 years passage time as against 300,000 in the host formation), which reduces the impact of any uncertainties on future evolutions.

A major modification resulting from glaciation would be the appearance of permafrost, which would modify the subsoil mechanics and lead to the circulation of underground water with resurgence on the surface, and thus the transfer of radionuclides from the repository to the human environment. The appearance of permafrost is possible (see Figure 6.2-29) :

- during the cooling phase, when the climate in the Meuse / Haute-Marne region becomes periglacial (creation of a discontinuous then continuous permafrost) ;
- during the warming phase, when the Meuse / Haute-Marne region experiences a boreal climate (possibility of residual permafrost).

Field work and observations in the shafts of the Meuse / Haute-Marne laborator show that the maximum possible depth of frost, identified through surface fractures, is about 100-120 metres on the site itself, while the maximum depth of the 0°C isotherm is 325 metres [72]. Flow disturbances are however possible down to a maximum of 300 m, according to the scenarios and physical processes (possible creation of gas hydrates under the segregation ice).

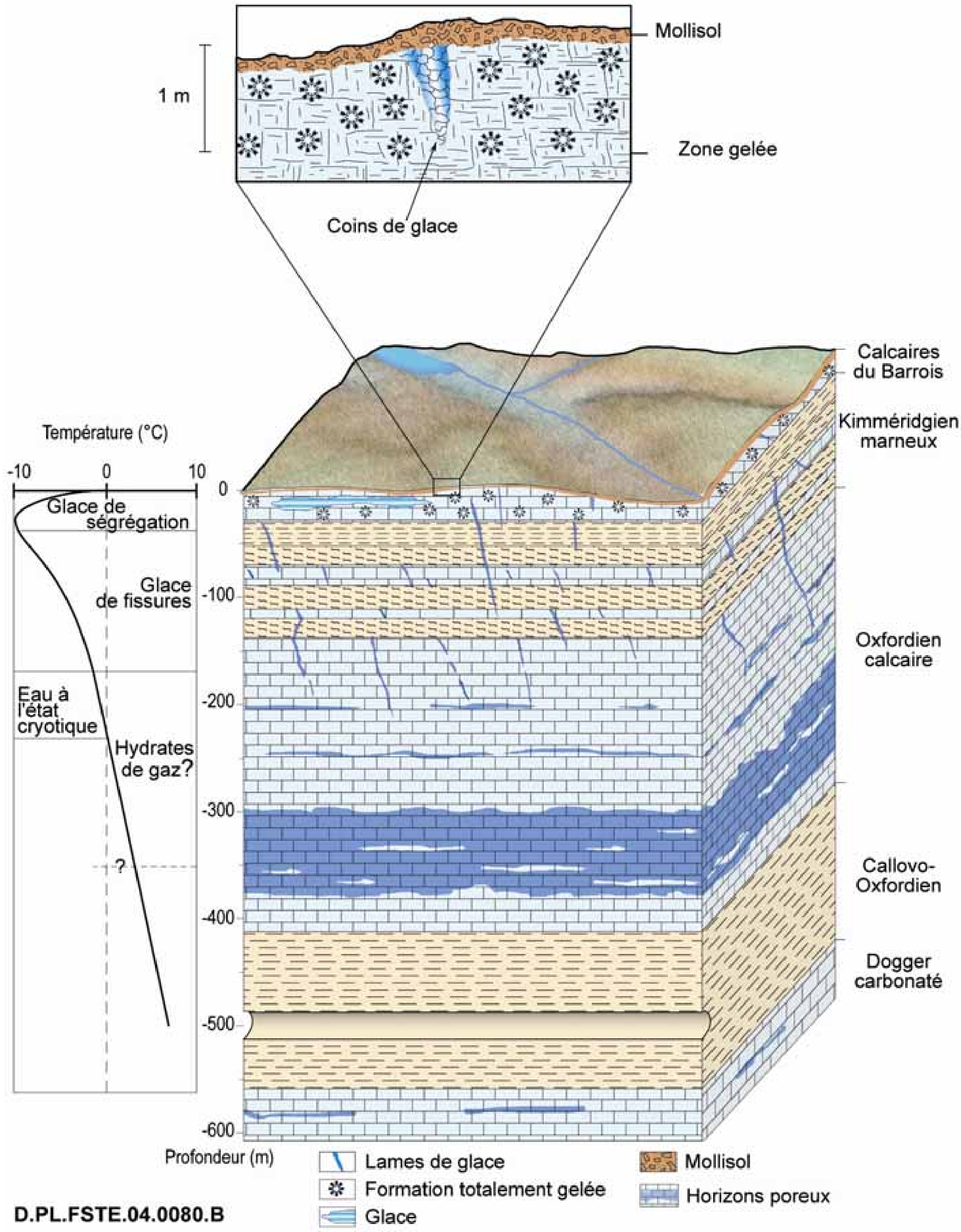


Figure 6.2-29 Extension of disturbances induced by the permafrost

In no case would the properties of the Callovo-Oxfordian be disturbed. If we consider glaciation at the extreme end of the possible range, only the seals located closest to the surface (aquifer isolation seals) could reach temperatures close to 0°C. The shaft seal would not be reached.

There are three possible effects of extreme glaciation on the upper aquifers :

- mechanical effects on the rock or the aquifer isolation seals, such as cracking created by freezing of water. They should be reversible in a clay material such as bentonite, and would thus have little influence, but they could locally damage limestone formations ;

- possible chemical effects linked to a modification of the state of the water (cryotic state) and the salinity (salt expelled by freezing). These effects could reach the porous levels ;
- water circulating in the upper surrounding rock could be partially blocked by the surface permafrost, which would prevent it recharging.

During the glacial period, all these phenomena thus entail a slow-down of water circulation through the soil, which is a favourable factor in limiting the migration of radionuclides to the biosphere. This effect - not included in the SEN – is maintained for the entire duration of surface glaciation. When the thaw occurs, restoration of water circulation could temporarily induce more contaminated waters to the surface, until equilibrium is restored. Specific calculations have shown that even in penalising configurations, the induced effect was not significant (in the worst case, the impact was increased by about 30 %) [64].

The return to temperate conditions, with disappearance of the permafrost, could leave an imprint in the upper surrounding rock, owing to the damage caused by the ice. Surface erosion is taken into account in the geodynamic model. In the most prejudicial situation, the possible ice damage to the first 100 to 120 metres of the surrounding rock could open up cracks or small fractures in the ground. The hydrogeological model is relatively insensitive to this type of structures, which would remain localised in the upper layers of the ground.

The possible modification of the chemistry of the surrounding rock also has no consequences on the radionuclide transport model, which includes no retention properties for these formations.

6.2.14.2 Climatic evolution : interglacial periods

An interglacial period that is warmer or longer than expected in the reference, for example taking account of man-made disruption of the climate, would have no significant effect on the structures or the host formation. The only effects would concern the biosphere and the aquifers : rapid impoverishment of the soil and disappearance of agricultural activities, over-exploitation of water courses and aquifers which could tend to dry up. Even in such a case, assessments made as part of the BIOCLIM project show that the rainfall associated with a possible « hot » climate in the Meuse/Haute-Marne region would remain around 600 to 1000 mm/year, which does not prevent the use in an initial approach of the temperate type of biosphere model adopted for the calculation

An episode such as this could postpone the following glaciation accordingly. Insofar as climatic evolution is not dynamically modelled in the SEN, this uncertainty over the date of occurrence of glaciation has no influence on the impact calculation.

6.2.14.3 Geodynamic evolutions (excluding seismic).

The geomorphological evolution scenarios for the site and the region (Paris basin) are used as the reference for analysing future hydrogeological flows, for the time-scale of the next million years, including the possibility of regional uplift.

The uplift rate taken into account for the next million years for the Meuse / Haute-Marne sector, in other words, the rate with which the geological formations rise under the effect of natural forces, is the maximum possible value estimated according to current knowledge. It is about one hundred metres over one million years. This uplift is progressive and continuous.

6.2.14.4 Effects of an earthquake

This uncertainty is initially strongly limited by the choice of site, in an area of very low seismicity.

The earthquake considered for the post-closure phase is different from that considered in the operating phase (see chapter 4). For a period of a million years, it is necessary to include the occurrence of seismic phenomena, even if highly improbable. Andra therefore based itself on the notion of the « maximum physically possible earthquake » (SMPP) as defined in the basic safety rule RFS.III.2.f. The methodology employed [52] allows consideration of uncertainties in knowledge (of seismic activity in the region) and uncertainties linked to the unpredictable nature of the phenomenon.

The phenomenological analysis of the effects of an SMPP on the repository in the post-closure phase, shows that the risk is limited to the interfaces between materials, primarily those with a void, which constitute obstacles to the progression of seismic waves.

The impact of the earthquake is therefore primarily an issue before disappearance of the voids in the repository and before the mechanical stresses in the structures are balanced with the rock. The probability of occurrence of an SMPP level earthquake is reduced accordingly over this period shorter than one million years.

The effects would be extremely attenuated with respect to those expected on the surface. If we identify the zones potentially containing the most voids, we can see that the effects could concern :

- the internal voids in certain primary packages (bitumen, glass, spent fuel). The damage to these primary packages would have no significant impact. Local damage to the packages would not modify the nature of the release models (glass is sensitive to this, but a fracturing rate equal to that of loose pieces of glass was employed in the SEN sensitivity studies) ;
- the voids in the B disposal cells (between disposal packages and between the packages and the filling/coating). In the least favourable case, package movements could lead to them being damaged. There is no safety function associated with their mechanical strength so this situation is not considered a failure ;
- the possible voids between the engineered components, consisting of the installation clearances, leading to local deformation. The sizing of the metal containers, liners, etc. is in principle adequate to withstand the seismic stresses, which in any case will remain slight ;
- at the seals and clay engineered barrier, we can imagine movement of the bentonite elements in relation to each other and in relation to the geological medium before resaturation. These voids would be compensated at the moment of bentonite swelling under the effect of the resaturation water, and the overall effect would not be appreciable ;
- the earthquake would not lead to significant stresses on the rock, nor the wall or the massif [52].

Generally speaking, the geological medium directly above the site, in particular the Callovo-Oxfordian, has been stable for several tens of thousands of years. The low-level earthquakes it is likely to experience in the next million years would not be different from those it has previously experienced over longer periods. No traces of past seismic events can be observed in the layer. The absence of physical discontinuities likely to be reactivated by an earthquake is also a favourable factor. The only action possible would be at the regional faults, in the light of their hypothetical tectonic activity. These transient or permanent modifications would have no impact on regional flows. In the sector, given the adopted scenarios (envisaging an earthquake on the Marne faults), the trajectories in the outlets in the upper surrounding rock are located upstream of the potentially disturbed zones. We therefore consider no disturbances due to an earthquake which could influence the path followed by the radionuclides.

The effects of an earthquake would therefore, on the whole, be negligible and are covered by the SEN and its sensitivity analyses.

6.2.14.5 Volcanic activity

This risk is designed-out by choosing a site free of volcanic activity.

6.2.14.6 Diagenesis

During its past history, the Callovo-Oxfordian has been affected by diagenetic processes, most of which are early. These determined the petrophysical characteristics of the clay sediment and in particular its porosity and permeability. In its current state, it offers no possibility of circulation liable to reactivate fluid-rock interactions capable of modifying these petrophysical characteristics [17].

A temperature rise leading to thermal diagenesis could only be linked to burial of the Callovo-Oxfordian under several thousand metres of sediment, a phenomenon which is ruled out in the context of the site.

6.2.14.7 Risks of human origin

The risks of human origin only concern inadvertent intrusions. These would be due to a loss of the memory of the repository. Before detailing the different types of intrusion envisaged, it is therefore worth mentioning a few important notions concerning this risk.

The possible means of maintaining a record of the repository, in principle with no time limit, were presented in chapter 3. It nonetheless remains highly likely that on a million year time-scale the memory of the repository, of its location and of its purpose, could be lost.

It is extremely hard to predict the social changes and the technologies that will be accessible to future generations. In accordance with the recommendations of the basic safety rule RFS.III.2.f., this type of uncertainty is dealt with by assuming that their technical level will be equivalent to our own. To this we add the highly pessimistic scenario that they will however have lost the notion of what radioactive waste is, what a deep geological repository is and the fact that such a repository could be located underground at a place which may be undetermined.

In these conditions, no particular reconnaissance technique would be employed to specifically locate the presence of the waste or the arrangement of the repository before the intrusion takes place.

This intrusion could have a variety of causes and consequently take various forms. The following typology is inspired by that proposed in basic safety rule RFS.III.2.f.

The intrusion could first of all be the result of detection of the fact that there is a particular object in the ground and the desire to find out what it is. The disposal could in theory represent a geophysical « anomaly » that can be detected from the surface : a magnetic anomaly owing to the presence of metal materials in the repository, a gravity anomaly owing to the materials density different from that of the geological medium. The studies performed by Andra show that only a 3D seismic investigation could detect the anomaly caused by the repository [82]. In terms of a gravity anomaly, the repository's visibility is very low as the densities of the various materials on the whole compensate for each other.

The repository may also have been detected from the surface : excavations bringing to light traces of former surface installations or old access structures. In such a case, an « archaeologist » might attempt a reconnaissance of the structure. Good practice would require that he begin by looking for the origin of his discoveries in the archives, where he could then identify that he had found a radioactive waste repository.

If, having acquired information and defined the site to be investigated, the archaeologist tries to enter the repository, he would probably contact mining archaeologists, who would take all the necessary precautions for accessing the site. Mining archaeologists as a general rule enter a site through an existing entrance. This consequently means that in the case of a closed repository, in which all the access and ventilation shafts have been completely plugged, the archaeologist would *in principle* be unable to bypass the geological barrier which could influence the subsequent evolution of the repository. It is however probable that the work involved in removing the backfill and the seals would be too great. In such a case, drilling could be used, even though this type of technique is not normally used in archaeology.

Finally, intrusion into the repository could be completely inadvertent, with the person totally unaware that such an installation exists on the site. The type of drilling imaginable at the depth of the repository would be geotechnical prospecting for natural resources (oil, gas, geothermal). Choosing a site with no such exceptional resources, in accordance with basic safety rule RFS.III.2.f, is protection against this eventuality. One could however imagine a drilling campaign aimed simply at reconnaissance of the formations.

This analysis as a whole indicates that a small diameter bore-hole passing through the rock or the repository at various points is possible, albeit improbable.

This risk is thus pertinent for definition of the altered situations and is the subject of a « borehole » SEA.

The repository is partly protected from drilling by its fractioning, with parts made hydraulically independent of the others, which would prevent propagation of the drilling effects beyond one or more seals. The radionuclide immobilisation function also plays a role in supplementing the other repository safety functions, which could be more directly affected by the drilling. This qualitative assessment needs however to be quantified and this will be the case in the « borehole » SEA (chapter 7).

We also consider that this borehole could be part of a larger-scale campaign comprising several boreholes. In such a case, the number of boreholes is of little importance : once they are close enough for them to be no longer isolated from each other by a sufficient number of seals, they are likely to cause circulation of water. This is the phenomenon that it is interesting to consider, envisaging the possibility of a two-boreholes system.

6.3 Definition of altered situations

On the basis of the uncertainties presented in section 6.2, altered situations were defined. We recall that these correspond to the loss of one or more safety functions on one or more components. It is also important to remember that these situations are by definition repository operation « limit » cases, felt to be unlikely but defined in order to cover the uncertainties as extensively as possible. Some situations presented may even be totally impossible and simply envisaged hypothetically to check what the effects would be of the loss of a broader range of functions than is actually conceivable. In this case, we talk of a « what if » situation.

The uncertainties and risks that can lead to altered situations are in principle of diverse natures (event outside the repository, design error or manufacturing defect, process running in an unexpected way, etc). However, in their modes, in other words in their effects, described independently of the causes that can trigger them, we have seen that these situations can be linked to a limited number of types, themselves linked to a small number of SEA. Insofar as a small number of repository components performs important safety functions, the altered situations correspond to cases involving malfunction of one or more of these components. Provided that representations that sufficiently envelop these malfunctions are chosen, we can then consider a limited number of cases. For example, in order to cover container failure of various origins (accelerated corrosion, manufacturing defect, etc.) with a single bounding situation, we can choose to represent a completely « what-if » situation, involving artificially and totally « eliminating » the confinement capacity of this component. Such a situation is in itself totally unreal, but in terms of the safety functions, it « covers » the possible malfunctions of the container. If such a procedure were to prove too coarse to be able to draw conclusions for the safety analysis, the definition of the bounding situation could be revised.

The altered situations, and their connection to the various SEA, were presented in parallel with the analysis. We summarise them here in a table. In addition, and to make reading easier, we did not in section 6.2 systematically deal with the possible combinations of uncertainties that could aggravate a given altered situation. The table gives an idea of the combinations that were considered.

It is constructed in the following way :

- a first column presents the description of the uncertainty ;
- a second column specifies which components are likely to be affected. As applicable, a given uncertainty can affect components differently, according to their particular configuration and the function expected of them ;
- a third column recalls in which section of chapter 6.2 the uncertainty was described, so that the reader can refer to it ;
- a fourth column presents uncertainties likely to amplify the effects of that being considered ;
- the corresponding SEA is recalled last, specifying the calculation case concerned.

A commentary gives any additional information.

If the reader wishes to look more closely at the reasons or such or such a combination of uncertainties was not given, he should refer to the Qualitative safety analysis itself which presents a more detailed version of the same table including the uncertainties covered in the SEN.

Description of the uncertainty	Components concerned	Section concerned	Associations with other uncertainties taken into account in the analysis[90]	Type of scenario	Comments
Uncertainties on the presence of minor structures in the Callovo-Oxfordian	Callovo-Oxfordian	6.2.1.1	None	SEN Borehole SEA	This uncertainty is mainly covered by a sensitivity study on the permeability of the rock. A borehole event would have similar effects (short-circuiting the geological barrier), but more damaging.
Uncertainty on possible insufficient swelling of the bentonite against the argillite (combined with uncertainty on deferred behaviour of EDZ)	All seals and plugs	6.2.6.1	Uncertainty on effect of calcic water on the bentonite, on the extent of alkaline disturbance (for drift seals and plugs) and on the geochemistry of water resaturating the engineered barriers (for shaft seal)	Seal failure SEA, all seals	Combined with uncertainties on the impact of chemical disturbance on the bentonite, this uncertainty suggests a situation where seals fail to swell sufficiently to be effective.
Uncertainty on scale of effect of bentonite recompression on argillite	Clay seal core and seal hydraulic cut-offs, engineered barriers of C waste and spent fuel	6.2.6.1	None	Seal failure SEA, all seals, with sensitivity to degradation of EDZ	This situation has the opposite (but highly unlikely) effect : excessive swelling of seals damages rock.
Uncertainty on composition of water resaturating engineered structures	Shaft seal	6.2.8.1	Uncertainty on effect of calcic water on swelling capacity.	Seal failure SEA, ineffective shaft seal	Here we imagine that, resaturated by water with unfavourable chemical composition coming from surrounding formations, the shaft seal fails to swell correctly. This would imply that the seal's buffer capacity has been poorly evaluated, otherwise such a major effect is hardly conceivable.

90 Which have similar effects or could aggravate the effects examined here

Description of the uncertainty	Components concerned	Section concerned	Associations with other uncertainties taken into account in the analysis[90]	Type of scenario	Comments
Uncertainty on the effect of calcic water on swelling capacity of bentonite	All seals	6.2.8.1	Uncertainty on extent of alkaline disturbance. Uncertainty on long-term evolution of the EDZ.	Seal failure SEA, all seals fail, sensitivity to the vitrified wask release model and degradation of EDZ.	If the bentonite does not swell sufficiently as it resaturates, the seals' hydraulic cut-offs could be partly ineffective. In that case they could be preferential pathways for disturbances, which could change the chemical conditions at the C waste disposal cell loads. An effect of this type in the spent fuel cells is ignored owing to the already very conservative source term chosen as reference. In an extreme situation, the bentonite does not accompany the deferred creep of the fractured zone, which becomes more severely damaged.
Uncertainty on the mechanical effect of gases in the argillite	All seals and plugs, clay engineered barrier	6.2.6.2	Uncertainty on deferred behaviour of the argillite, uncertainty on geochemical composition of the water (which may affect the bentonite's swelling capacity).	SEN Seal failure SEA, all seals with sensitivity to degradation of the EDZ	This scenario envisages the possibility of increased damage to the argillite if gases create a permanent, continuous fractured zone around the seals. The creation of permanent micro-fissures, although no more probable, is in fact covered by the SEN, which does not envisage healing of the micro-fissured zone created by excavation.
Uncertainty on the mechanical effect of gases in argillite and bentonite	All seals and plugs, clay engineered barrier	6.2.6.2	Uncertainty on swelling capacity of the bentonite, uncertainty on extent of alkaline disturbance	Seal failure SEA, all seals, with sensitivity to degradation of seal impermeability on a macroscopic scale.	Situation fairly similar to previous one. This time we imagine that the mechanical effect of the gases may cause irreversible damage to the bentonite (seal cores, clay engineered barrier). This is a highly conventional situation, given that the swelling pressure of the bentonite is amply sufficient to close any fissures if gases pass through them. It is therefore combined with uncertainties on the swelling capacity of the bentonite, or disturbance of the bentonite by an alkaline plume.

Description of the uncertainty	Components concerned	Section concerned	Associations with other uncertainties taken into account in the analysis[90]	Type of scenario	Comments
Uncertainties on the effect of gases on transport during hydraulic transient	B waste, C waste and spent fuel disposal cells	6.2.5.2	Any uncertainty leading to very early releasesinto drifts or to release of radionuclides in gaseous form	SEN Package failure SEA, with sensitivity to role of hydraulic transient.	This uncertainty concerns the role of gases in radionuclide transport during the pressure increase phase. It is covered in the SEN by a sensitivity study an transport during the hydraulic transient phasefor B1x and CU1 waste cells. Sensitivity is also taken into account in the package failure SEA.
Uncertainty over models of long-term degradation of concrete	Support bases of shaft and drift seals and cell plugs	6.2.8.2	Uncertainty on water geochemistry	Seal failure SEA, all seals	Here we conventionally envisage early degradation of seal retaining plugs, preventing seals from swelling under good conditions.
Uncertainty on the effects of sulphate attack on concrete over several centuries	All seal retaining plugs	6.2.8.2	Uncertainty on water geochemistry	Seal failure SEA, all seals	This uncertainty has similar effects to the previous one.

Description of the uncertainty	Components concerned	Section concerned	Associations with other uncertainties taken into account in the analysis[90]	Type of scenario	Comments
Uncertainty on extent of alkaline disturbance in the bentonite	All seals, C and spent fuel disposal cell plugs	6.2.8.4	Uncertainty on swelling capacity under effects of calcic water, seals installed with residual clearance	Seal failure SEA, all seals, and sensitivity to the vitrified waste release model	In the event of excessive spread of alkaline disturbance in all cell plugs and drift and shaft seals (highly improbable in the light of evaluations, even in the event of an unexpected effect of calcic water or badly installed seals), one could envisage weakened seal efficiency and an increase in pH at C waste and spent fuel cell loads. This effect is assumed to be negligible in the spent fuel cells given the very rapid release (50,000 years) in the model adopted. It is taken into account for C waste cells by accelerating release rates.
Uncertainty on pH of cement water	All seals	6.2.8.4	Uncertainties on assessment parameters for alkaline disturbance	Seal failure SEA, all seals and sensitivity to the vitrified waste release model	If excessive pH lasts longer than envisaged in the reference model, this could lead to more extensive spread of remineralisation associated with alkaline disturbance, and effects similar to the previous table entry.

Description of the uncertainty	Components concerned	Section concerned	Associations with other uncertainties taken into account in the analysis[90]	Type of scenario	Comments
Uncertainty on cumulated effects of chemical disturbance	C waste and spent fuel cell plugs	6.2.8.7		Seal failure SEA, all seals and sensitivity to the vitrified waste release model	Here we envisage an accumulation of alkaline and iron/clay disturbances, leading to their unexpected extension. The effects are similar to those described above. In principle the impact is limited to thermal cell plugs, but the SEA is more comprehensive, addressing the case of generalised seal failure.
Occurrence of criticality accident in the spent fuel cells (for the record)	spent fuel Cells	6.2.9.1	Uncertainty on the mechanical behaviour of corrosion products	SEN	A long-term criticality accident induces moderate heating of the argillite around the package. The transport conditions are similar to those of the SEN
Specifications for argillite backfill inadequate in the light of expected long-term mechanical performance	Support backfill	6.2.12.1	Uncertainty on effects of sulphate attack on concrete, on the model of long-term behaviour of the concrete	Seal failure SEA, drift seals[91], with sensitivity to degradation of EDZ	The support backfill is only there to ensure long-term confinement of the seal, as swelling of the seal is forced by the concrete retaining plug. There would have to be poor infill specifications coupled with unexpected behaviour of the concrete to create a situation where the drift seals, and possibly the rock as well, could not be properly supported.

[91] In fact the SEA envisages failure of all anchored seals, including those of the B waste cells.

Description of the uncertainty	Components concerned	Section concerned	Associations with other uncertainties taken into account in the analysis[90]	Type of scenario	Comments
Specifications for concrete backfill inadequate in the light of expected long-term mechanical performance	Access structure backfill (shaft)	6.2.12.1	Uncertainty similar to that concerning initial state of concrete (cracking) and model of long-term evolution of concrete (see below)	Seal failure SEA, shaft seal	If the shaft seal is not properly supported by the concrete backfill (highly unlikely situation, given the slight mechanical demand) the shaft seal might not swell correctly.
Uncertainty on thermal-mechanical effects in C waste cells	Around C waste disposal cell plugs	6.2.11.3	Uncertainty on the swelling capacity of the bentonite (calcic water), extent of alkaline disturbance.	Seal failure SEA, all seals fail, with sensitivity to accelerated release of vitrified waste	An unfavourable evolution of the damaged zone under thermal conditions, combined with poor support by the plug (which either does not swell sufficiently, or is poorly supported by its retaining plug) could lead to short-circuiting by a fractured zone. This situation assumes simultaneous poor behaviour of rock and plug. The SEA does not directly address this case, but it is covered by a more pessimistic situation in which a fractured zone bypasses all the seals. Combining this situation with a poor evaluation of the extent of the alkaline disturbance, we imagine plug failure leading to increased pH in the C waste cells and accelerated release.
Uncertainty on ability to emplace the bentonite rings without residual gaps	Clay engineered barriers of spent fuel disposal cells.	6.2.12.1		Seal failure SEA, all seals fail, with sensitivity study of permeability of clay engineered barrier	Poor emplacement would result in locally defective permeability of the engineered barrier, if, in addition, the bentonite failed to swell as expected. This is not expected to affect the engineered barrier in a normal situation, since its only role is to protect the cell and it has no role with regard to transfer. The seal failure SEA envisages failure of the buffer combined with seal failure.

Description of the uncertainty	Components concerned	Section concerned	Associations with other uncertainties taken into account in the analysis[90]	Type of scenario	Comments
Uncertainty on the ability to emplace bentonite bricks with no residual gaps	Seals	6.2.12.1	Uncertainty on the extent of alkaline disturbance	Seal failure SEA, all seals fail, with sensitivity to accelerated release for C waste.	Equivalent to previous case for spent fuel cells : poor compacting of bentonite bricks would degrade seal performance. For C waste, this is coupled with a possible excessive spread of the alkaline plume through the plug, and a resulting increase in release.
Technological uncertainty on emplacement of hydraulic cut-offs	Anchored seals (drifts, B cell plugs)	6.2.12.1	Uncertainty on the model of deferred behaviour of argillite, uncertainty on the effect of re-compression of bentonite, uncertainty on duration of operating phase	Seal failure SEA, all drift and B waste cell seals	The uncertainty concerns all seals supplied with a hydraulic cut-off (drifts, B waste cell plugs). It consists of assuming that excessive loss of rock confinement, after a long period of operation, is not compensated by swelling of the clay in the groove. The swelling is ineffective because bypassed by a fractured EDZ.
Technological uncertainty on removal of lining and emplacement of plug	C waste disposal cell plugs	6.2.13	Uncertainty on the model of deferred behaviour of the argillite, uncertainty on the duration of the operating phase, uncertainty on the extent of the alkaline plume	SEN Seal failure SEA, all seals fail, with sensitivity study of release by glass.	The uncertainty concerns the C waste cell plugs and the risk of excessive damage during plug emplacement. The SEN envisages punctual failure. There is no specific calculation case of a defective plug in the SEA, but the situation is covered by a calculation in which all seals fail. Coupling with an excessive spread of alkaline disturbance through the defective plug, situations are envisaged where radionuclide release by glass is accelerated.
Systematic error during container weld quality control, not detected during inspection	C waste overpack and spent fuel container	6.2.12.2	None	Package failure SEA	

Description of the uncertainty	Components concerned	Section concerned	Associations with other uncertainties taken into account in the analysis[90]	Type of scenario	Comments
Borehole	A priori, all components	6.2.14.7	None	Borehole SEA	

As the typology shows, the altered situations that emerge from the analysis can be divided into a limited number of scenarios, provided different calculation cases are defined within these scenarios, to cover the different configurations.

We do not consider altered situations for uncertainties on characterisation of the geological medium because most of them are covered by the SEN and the selected sensitivity analyses. The only exception would be an undetected heterogeneity involving very deteriorated, conductive permeability. However, no observation made at the site suggests that such a phenomenon might exist, and it could hardly remain undetected throughout the construction and operation of the repository.

The components involved in most failure situations are seals and the associated EDZ. This does not mean these types of component have inherent weaknesses ; it is mainly because these objects are specific to the repository, unlike the containers, for example, for which there is ample feedback available from industrial experience. So it is natural that the safety analysis should focus on components that have particular features and relatively less feedback from prior experience.

Once the SEA are decided on this basis and their calculation cases have been specified according to the qualitative analysis, it is possible to quantify them. Chapter 7 will give other calculation cases not directly derived from the risk analysis but intended to supply additional information for understanding the functioning of the repository system or its robustness in the face of variations in parameters. In particular, we will define a fourth SEA, called a « highly degraded evolution » scenario, which does not emerge from the qualitative safety analysis, but which covers several types of failure at once. Such a scenario can be classed as a « conventional » scenario.

6.4 Conclusions on uncertainty control

Taking account of uncertainties identified throughout the scientific studies it has been possible to verify that the SEN amply covers many of them, either in the reference scenario or through sensitivity studies. However, boundary cases involving particular uncertainties or combinations of uncertainties may reveal unlikely situations that are outside the scope of the SEN, because they involve a priori improbable failures of the safety function. At this point in the document it is not possible to say whether they would have any significant effect on radionuclide transfer, or would increase the impact, or whether such an increase might become unacceptable. It is the purpose of Chapter 7 to make these calculations.

Analysis of the most frequent combinations of failures reveals some points of importance for the safety analysis. At the stage reached in this chapter it is not possible to say how important the uncertainties are for safety. They can only be properly ranked once their impact on repository evolution has been quantified. This is the purpose of the next chapter. However, we can at this stage raise some major themes that emerge from the uncertainties inventory. These intermediate conclusions supplement those set out at the end of Chapter 5 ; we have therefore not covered issues already covered after the performance assessment.

- **Uncertainties concerning the geological medium appear to be under control**

On-site investigations, first from ground level, then from boreholes and then during construction of the underground laboratory, have provided the data to build a conceptual model of the host formation, the uncertainties of which are under control. The favourable characteristics of the medium (low permeability, low porosity, diffusive environment, high chemical buffering capacity) also limit the influence of other uncertainties affecting chemical or hydraulic processes inside the repository.

The host formation seems to vary little on the scale of the transposition zone ; the main parameters seem likely to vary within one order of magnitude at most. Significant variations of the layer on this scale are unlikely both because of the conditions under which the sediments were laid down and because of the formation's subsequent evolution. Further investigations on the scale of the transposition zone would refine this observation.

The surrounding formations play no part in repository safety and are represented with conservative hypotheses in the SEN. As a result, uncertainties that affect them have no direct influence on safety since they do not alter the conclusions of the calculations set out in Chapter 5. Nonetheless, with a more precise knowledge of some of their characteristics it would be possible to make a more accurate conceptual representation of them, reducing the margins allowed for in the calculations. Examples are the direction of flow in the Dogger, characterisation of the retention properties of the basal Oxfordian and the Kimmeridgian, and continuation of work on the hydraulic role of regional faults and their environment (particularly the « diffuse fracture zone »).

- **Better knowledge of transients is a way forward**

Representation of the hydraulic, thermal and mechanical transients is sufficient for assessing safety and ensuring that the performance assessments conducted are conservative. However, a better understanding and more detailed modelling of transients is an important way forward for the future, to give greater precision over repository safety assessment conditions and identify margins of conservatism. The issue of transients is inseparable from the issue of modelling couplings, with which it overlaps to a large extent.

The hydraulic transient reveals the preponderant role, during the first few thousand years, of gas production and evacuation and the associated mechanical couplings. At this research stage the safety assessment assumes considerable gas output in connection with rapid corrosion kinetics. If the corrosion rate taken into account is reduced as knowledge improves, this will reduce the estimates of pressure inside the repository and simplify the analysis. Additional studies of gas evacuation processes are needed, to make the overall picture of repository evolution more precise.

The thermal transient is controlled by design, by leaktight sealing of the containers in the cells during the thermal phase. However, the safety assessment needs to take into account the possibility of container failure, taking a defence-in-depth approach. This opens the question of transfers in thermal conditions. Data on this issue have been gathered experimentally and used for the evaluations.

- **The damaged argillite zone plays a part in the uncertainty analysis, but its importance for safety needs to be quantified**

Seal failure and possible transfer of water and gas by the damaged zone recur several times in the qualitative analysis. More generally, uncertainties on the characteristics of the EDZ and its evolution over time appear in the definition of several altered situations. The rock damage research Andra has conducted in the underground laboratory must therefore continue in order to improve knowledge of damaged zone characteristics and the parameters governing its long-term evolution.

Also in this case, it is important to stress that the analysis is only qualitative at this stage. Quantification of situations in which the EDZ may play a role (the seal failure SEA particularly) will provide a more complete understanding of this aspect, at the end of Chapter 7.

- **Qualitative analysis shows that repository operating conditions will not have a very marked effect on long-term safety**

The conditions under which the repository is built and operated will have an impact on the repository's future evolution. Chapter 3 showed, in particular, how the care taken in excavating the engineered structures contributes to controlling water circulation over the long term. However, detailed analysis of phenomena likely to take place during operation and affect the state of the repository at the beginning of the post-closure phase do not reveal any elements that might impose severe constraints on the future operator of the repository. The reference baseline was operation of the repository over a hundred years, but extending this to several hundred years poses no major difficulty. This point obviously favours reversibility.

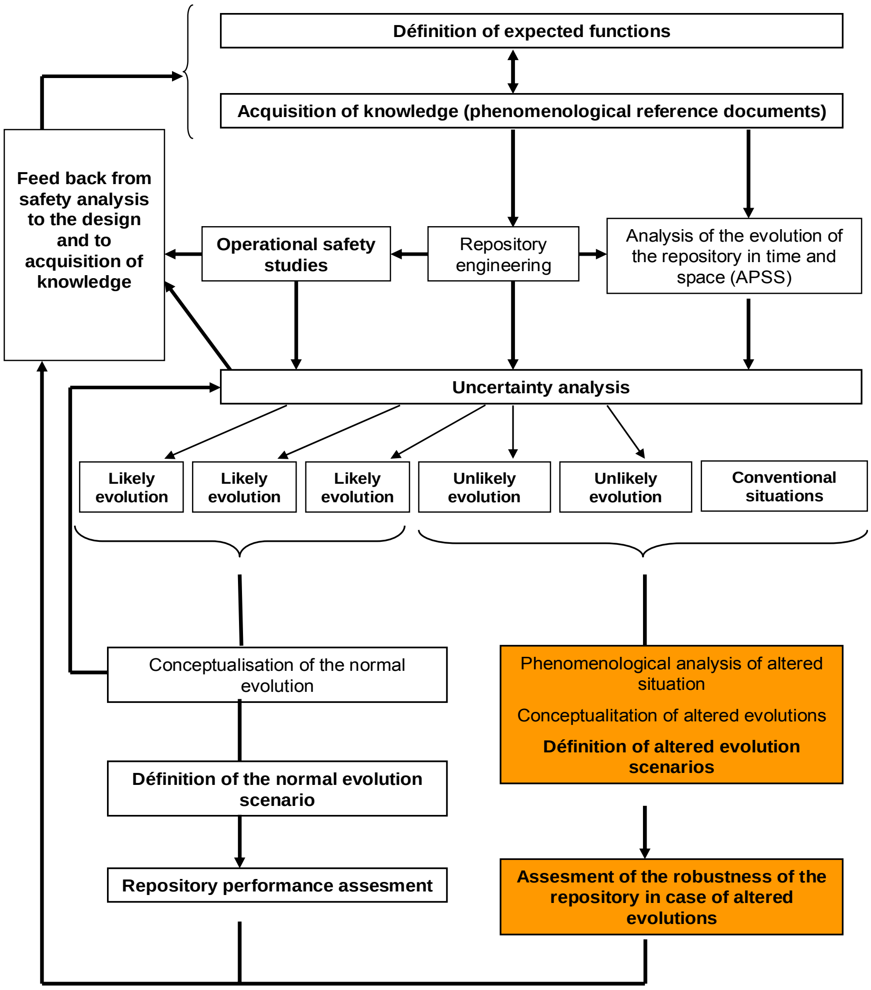
- **The only « external » event that appears to be really pessimistic is an intrusive borehole**

Should an intrusive borehole ever reach a disposal cell, this would endanger the very function of the repository, which is to isolate the waste from all human activity. Any waste management solution consisting of gathering the waste in one place exposes it to the risk of intrusion. This is a consequence of the concentration-and-confinement strategy. The human intrusion problem is being studied in all countries aiming for a deep disposal solution. It seems preferable to take this risk than to disperse the waste in the environment, which would create risks with much higher probability and not necessarily less serious effects.

In the nature of things it is simply not possible to prevent this risk entirely, or to annul its effects. On the other hand, effective steps can be taken to reduce its likelihood or the severity of its effects. These steps include choosing a site that has no particular natural resources and dividing the repository into separate zones. As we saw in Chapter 3, preserving the memory of the repository by various means will prevent involuntary intrusion, but we cannot rely on that memory lasting indefinitely. Quantification of borehole impact will provide a basis for assessing whether the protections measures taken are sufficient to reduce the impact to an acceptable level.

The choice of site and repository design mean that the impact of other potential events (volcanism, diagenesis, seismic events) can be disregarded or seriously limited.

7 Altered evolution scenarios



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Block diagram 7-1 Sequence of stages in the analysis (see Block Diagram 1-1) Theme : phenomenological analysis of altered situations, definition of altered evolution scenarios and consequent assessment of repository robustness

The purpose of this chapter is to describe hypothetical scenarios called « altered evolution scenarios » (SEAs) intended to supplement the normal evolution hypothesis described in Chapter 5. It also assesses, in the light of the quantification results, the ability of the repository to withstand the various external stresses and safety function failures that might arise.

7.1 Definition of altered evolution scenarios

Under the logic on which the Dossier 2005 is based, the altered evolution scenarios (SEAs) were first defined in principle, based on feedback from Andra's experience, analysis of situations taken into account internationally, and the recommendations of basic safety rule RFS III.2.f [2]. The main types of situation to be covered and the main calculation cases were established on the basis of this definition.

Only after completion of the qualitative safety analysis (results in Chapter 6), it is possible to ensure that the altered evolution scenarios identified cover all the situations Andra has identified as being beyond the scope of the normal evolution scenario and its sensitivity analyses.

Some sensitivity analyses may be induced by the will to evaluate the influence of a parameter, and have no direct connection with the AQS (see section 7.1.1).

Once the SEAs have been defined and their bounding characteristics verified by the QSA, they still have to be quantified.

To describe the corresponding sequences of events, sufficient knowledge of the evolution of the repository « outside the scope of normal evolution » is required. Phenomenological analyses have therefore been performed for altered situations corresponding to the scenarios [29, 30, 31]. The SEAs are assessed by a safety model derived from the SEN safety model but taking into account the particular features of the evolutions in question.

Sensitivity calculations may be performed on the SEAs in order to :

- cover variants of the situations envisaged in the main calculation, usually variants that constitute aggravating circumstances ;
- cover phenomenological uncertainties on the parameters.

The results of the SEA calculations (basic case and sensitivities) must be compared with thresholds. Basic safety rule RFS III.2.f. gives no such thresholds, since it seems difficult to define the acceptability of the results of SEAs generically. After all, as already stated, the SEAs are situations covering several altered evolutions due to various causes (e.g. a waste package failure scenario may be due either to a manufacturing defect or to the container corroding much faster than normal). The SEA represents these different situations in an « bounding » way, i.e. it provides a description that generally overestimates the different possible effects. In the example given, the SEA would imagine the total « disappearance » of the container after 200 years. While one can assess the plausibility of each altered situation, it is a more delicate matter to assess the plausibility of a scenario that may represent several such situations in the form of stylised hypotheses. In some cases, an altered evolution scenario may not represent any physically possible situation : in this case one speaks of a « conventional » or « what if » scenario.

The acceptability of the calculated impacts is assessed case by case. One simply bears in mind that :

- the criterion of 0.25 mSv/yr is taken as one benchmark among others, i.e. the calculation compares the results with this value, but it is not mandatory that the calculated dose comply with this limitation ;
- up to a few mSv/yr (10 at most), the impact can be regarded as acceptable on a case by case basis, provided the situation(s) described are sufficiently unlikely. In any event, even if the impact is considered acceptable, one seeks to reduce it by appropriate means, if any ;
- particular attention must be paid to any calculated impact of more than 10mSv/yr. The scenario may be too over estimating ; design methods that would prevent such a situation must be carefully examined. In the case of purely hypothetical situations, such an impact is not necessarily unacceptable as such, insofar as it does not refer to a situation that could actually arise.

7.1.1 Defining the major types of scenario

The initial consideration that led to the definition of the altered scenarios is based on a breakdown by function. The intention was to define an exemplary failure situation for each of the three main safety functions, regardless of the probability of the situation described.

For the function of « limiting water circulation », we saw in Chapter 3 the importance of the shaft, drift and module seals. It seemed natural to build a seal failure scenario for failure of combinations of seals [103], differentiating those equipped with a hydraulic cut-off. Although C cell plugs do not formally have a « limiting water circulation » function, they are also included as defective components in this SEA because they are of a similar nature to the other seals.

The function of « limiting the release of radionuclides and immobilising them in the repository » is fulfilled by different components at different time periods : containers at first (for vitrified waste and spent fuel), waste matrices, physical-chemical form of the elements released, chemical and hydraulic conditions in the disposal cells. It is difficult to define a scenario to cover the failure of all these components. A scenario involving failure of thermal waste containers [104] was chosen ; this would allow early release of radionuclides and their diffusion in a thermal environment, which in principle would accelerate migration beyond the near field.

The « delay and attenuate radionuclide migration » function mainly relies on the host formation, though the seal cores and disposal cells are also involved. The features involved are the predominantly diffusive conditions in the host formation and the spatial dispersal that these conditions allow, supplemented by measures to preserve the dispersal capacities of the surrounding formations (see Chapter 3). It therefore seemed useful to consider an intrusive borehole [97] intercepting the geological formations and the repository at various points, defined in such a way as to short-circuit all barriers including the aquiferous horizons. The aim was to disrupt the spatial dispersion and encourage advection.

These three particular situations were intended to illustrate cases of function failure, not to cover all possible situations in theory. The safety analysis tells us whether the causes envisaged are plausible, and whether other phenomena than those initially considered could cause the effects covered by the scenarios. This work was presented in Chapter 6.

It seemed useful, to complement the SEAs defined above, to define a fourth one that would take into account a generalised failure of all safety functions. This is based neither on feedback from scenarios defined by Andra's counterparts, nor on altered situations identified through the QSA. It is a « severely degraded evolution » scenario that consists of systematically reducing the performance of the safety functions to exceed the scope of the normal evolution scenario. The first three SEAs serve to test the degree of redundancy between the safety functions : the idea is to minimise or eliminate the contribution of one function, and then study whether the others are sufficient to comply with the safety objectives. The « severely degraded evolution » scenario assesses the complementary nature of these functions : by degrading all of them at once and comparing the results with the results of a normal evolution scenario, one can observe whether minimal performance levels, below what is normally expected, complement each other sufficiently well to control the impact.

7.1.2 Verification of consistency with RFS III.2.f

Basic safety rule RFS III.2.f. recommends considering the effects of the following situations :

- Major climatic changes as covered in chapter 6 (see section 6.2.14.1 and 6.2.14.2). No significant effects are expected due to the depth of the repository installation. Possible consequences departing from evolution situation might be similar, in the worst case, to significant modifications in the living and feeding habits of individuals residing on the site and in the characteristics of surrounding formations. These scenarios are covered :
  - ✓ in a preparatory study based on a critical group associated with a cold biosphere and demonstrating that said group is less pessimistic than the reference group chosen,
  - ✓ by selecting 'deep' outlets close to the site and not sensitive to the possible damage of surrounding formations near the surface,

- Exceptional vertical movements or earthquakes. The tectonic risks in the Meuse / Haute-Marne site are low. The effects of a possible earthquake were taken into account in the qualitative analysis (see section 6.2.14.4) and shown to be negligible in the engineered structures and in the rock ;
- Various possible forms of human intrusion, covered by the 'borehole' altered evolution scenario ;
- Geological barrier defects. The basic safety rule proposes considering sedimentary hiatuses in the form of sand lenses for the argillaceous sites. Such structures are excluded in the Callovo-Oxfordian argillites. The safety analysis conducted in chapter 6 (see section 6.2.1.1) indicates that the only undetected structures possibly present are minor structures with limited extent and release. The effects associated with such structures are theoretically very limited. They are covered by the borehole scenario as a last resort
- Seal failures, covered in an altered evolution scenario
- Waste package defects, at least for sensitivity analysis purposes. This possibility is covered in the normal evolution scenario and in a specific altered evolution scenario.

7.1.3 Generic treatment of the scenarios

The SEA calculation are based on the same model as the SEN, adapted according to the failure that is to be represented. The following sections give a detail of the adaptations, according to each scenario.

The selection of radionuclides for the calculation, presented in section 5.3.1.5, was applied to the SEA depending on expected transfer times, we may have added some more. As an example, the SEA borehole has all radionuclides with a more than 30 year half-life. The actinides were taken into account in the same way as the fission and activation products, in three of the four altered scenarios :

- the « seal failure » scenario,
- the « borehole » scenario,
- the « severely degraded evolution » scenario.

Indeed, these scenarios favour transfer pathways other than the host formation or degrade the properties of the host formation (the first scenario enables a greater transfer via the access structures, the second short-circuits the geological barrier and the last one envisages unfavourable hydraulic and transport properties for the Callovo-Oxfordian). In this context, and insofar as chapter 5 concluded that the properties of the Callovo-Oxfordian explained the low actinide mobility, it was considered to be useful to take them into account in these scenarios.

However, they were not taken into account the « package failure » scenario, the retention parameters of the actinides are indeed barely modified, if at all, by the higher temperature during the transitional period where the radionuclides are potentially released in the case of a package failure. Moreover, the duration of the thermal transient period was estimated to be too short in comparaison with the actinide thermal transfer times in the Callovo-Oxfordian (from several million to billions years) to influence significantly significant influence on the molar flow rates out of the geological barrier and therefore on the impact.

7.2 « Seal failure » scenario

7.2.1 Scenario definition

This scenario is intended to consider a failure in all or part of the seals so as to assess the robustness of the repository system with respect to various combinations of such defects in repository components (shafts, drifts, module separation) or cell plugs. This scenario also includes any failure possibly associated with the development of a damaged zone around the engineered structures, more significant than that considered in the normal evolution scenario and possibly constituting a radionuclide transfer pathway and/or influencing the long-term evolution of the repository, even in the event that this unfavourable evolution is not due to the seals themselves (for example, in case of EDZ damage by gases, as discussed in chapter 6).

The undesired outcome is the release of radionuclides from the shaft outlet at the roof of the Callovo-Oxfordian layer, at any given date and with a flow exceeding that considered in the normal evolution scenario during the post-closure phase.

The qualitative safety analysis has shown that the 'seal failure' scenario can cover various types of situations :

- Partial short-circuit of the host formation due to poor contact between the rock and swelling clay, or excessive extent of the EDZ. Radionuclides can then travel preferentially through the drifts and shafts towards the surface by a advective transport instead of a diffusive one
- In certain situations, a modification of the hydro-chemical conditions within the cells (beyond the range of values where the components are most capable of fulfilling their safety functions), which also results in a partial failure of the 'limitation of radionuclide release and immobilisation of radionuclides inside the repository' function ;
- In certain situations (very improbable), a general degradation of seal permeability.

The following seal structures are considered :

- C waste and spent fuel disposal cell plugs ;
- Seals of connecting drifts within the C waste and spent fuel repository zones (repository module and zone seals), B waste disposal cell seals and B waste repository zone seals, seals of connecting infrastructures near surface/underground access structures, seals of surface/underground connecting structures (shafts).

The uncertainties and failure modes covered by this « seal failure » scenario were listed in chapter 6, both with regards to of their presentation (see section 6.2) and the construction of altered situations (see section 6.3). The table below summarises these elements for the record.

Uncertainties considered	Other uncertainties taken into account in the analysis[92]	Components concerned	Comments
Insufficient swelling of the bentonite against the argillite (combined with uncertainty on deferred behaviour of EDZ) → See § 6.2.6.1	Effect of calcic water on bentonite Extent of alkaline disturbance (for drift seals and plugs) Geochemistry of water resaturating the engineered barriers range (for shaft seal)	All seals All plugs	Combined with uncertainties on the impact of chemical disturbances on the bentonite, this uncertainty suggests a situation where seals fail to swell sufficiently to be effective.
Scale of the effect of bentonite recompression on argillite → See § 6.2.6.1	None	Clay seal cores Seal anchors C waste and spent fuel engineered barriers	This situation has the opposite (but highly unlikely) effect : excessive swelling of seals damages rock.
Composition of water resaturating engineered structures → See § 6.2.8.1	Effect of calcic water on swelling capacity.	Shaft seals	Here we imagine that, resaturated by water with unfavourable chemical composition coming from surrounding formations, the shaft seal fails to swell correctly. This would imply that the seal's buffer capacity has been poorly evaluated, otherwise such a major effect is hardly conceivable.
Effect of calcic water on swelling capacity of bentonite → See § 6.2.8.1	Extent of alkaline disturbance Long-term evolution of the EDZ.	All seals	If the bentonite does not swell sufficiently as it resaturates, the seals' hydraulic cut-offs could be partly ineffective. In such cases, they could be preferential pathways for disturbances which could change the chemical conditions at the C cell head. An effect of this type in the spent fuel cells is ignored owing to the already very conservative source term chosen as reference. In an extreme situation, the bentonite does not accompany the deferred creep of the fractured zone, which becomes more severely damaged.
Mechanical effect of gases in the argillites → See § 6.2.6.2	Deferred behaviour of the argillite Geochemical composition of the water (which may affect the bentonite's swelling capacity)	All seals All plugs Engineered barrier	This scenario envisages the possibility of increased damage to the argillite if gases create a permanent, continuous fractured zone around the seals. The creation of permanent microfissures, although no more probable, is in fact covered by the SEN which does not envisage healing of the microfissured zone created by excavation.

[92] Having similar effects or being able to increase the considered effects

Uncertainties considered	Other uncertainties taken into account in the analysis[92]	Components concerned	Comments
Mechanical effect of gases in argillite and bentonite. → See § 6.2.6.2	Swelling capacity of the bentonite Extent of alkaline disturbance	All seals All plugs Engineered barrier	Situation fairly similar to previous one. This time, we imagine that the mechanical effect of the gases may cause irreversible damage to the bentonite (seal cores, engineered barrier). This is a highly conventional situation, given that the swelling pressure of the bentonite is amply sufficient to close any fissures if gases pass through them. It is therefore combined with uncertainties on the swelling capacity of the bentonite, or disturbance of the bentonite by an alkaline plume.
Models of long-term degradation of concrete → See § 6.2.8.2	Water geochemistry	Support bases of shaft of shaft and drift seals and cell plugs	Here we conventionally envisage early degradation of the seal retaining plugs, preventing the seals from swelling under good conditions.
Effect of sulphate attack on concrete over several centuries → See § 6.2.8.2	Water geochemistry	All seal retaining plugs	This uncertainty has similar effects to the previous one.
Extent of alkaline disturbance in the bentonite → See § 6.2.8.4	Swelling capacity under effects of calcic water	All seals C and spent fuel disposal cell plugs	In the event of excessive spread of alkaline disturbance in all cell plugs and drift and shaft seals (highly improbable in the light of evaluations, even in the event of an unexpected effect of calcic water or badly installed seals), weakened plug and seal efficiency and an increase in pH at C waste and spent fuel cell heads can be envisaged. This effect is assumed to be negligible in the spent fuel cells given the very rapid release (50,000 years) in the model adopted. It is taken into account for C waste cells by accelerating release rates.
pH of cement water → See § 6.2.8.4	Assessment parameters for alkaline disturbance	All seals	If excessive pH lasts longer than expected in the reference model, this could lead to more extensive spread of the remineralisation associated with alkaline disturbance, and effects similar to the previous table entry
Cumulated effects of chemical disturbance → See § 6.2.8.7	None	C waste and spent fuel cell plugs	We envisage an accumulation of alkaline and iron/clay disturbances, leading to their unexpected extension. The effects are similar to those described above. In principle, the impact is limited to thermal cell plugs, but the SEA is more comprehensive, addressing the case of generalised seal failure.

Uncertainties considered	Other uncertainties taken into account in the analysis[92]	Components concerned	Comments
Specifications for argillite backfill inadequate in the light of expected long-term mechanical performance → See § 6.2.12.1	Effects of sulphate attack on concrete Model of long-term behaviour of the concrete	Support backfill	The support backfill is only there to ensure long-term confinement of the seal, as swelling of the seal is forced by the concrete retaining plug. There would have to be poor backfill specifications coupled with unexpected behaviour of the concrete to create a situation where the drift seals, and possibly the rock as well, could not be supported properly.
Specifications for concrete backfill inadequate in the light of expected long-term mechanical performance → See § 6.2.12.1	Initial state of concrete (cracking) Model of long-term evolution of the concrete	Backfill of access structures (shafts)	If the shaft seal is not properly supported by the concrete backfill (highly unlikely situation given the slight mechanical stress), the shaft seal might not swell correctly.
Thermomechanical effects in C waste cells → See § 6.2.11.3	Swelling capacity of the bentonite (calcic water) Extent of alkaline disturbance	Around C waste cell plugs	An unfavourable evolution of the damaged zone under thermal conditions, combined with poor support by the plug (which either does not swell sufficiently or is poorly supported by its retaining plug) could lead to short-circuiting by a fractured zone. This situation assumes simultaneous poor behaviour of rock and plug. The SEA does not directly address this case, but it is covered by a more pessimistic situation in which a fractured zone bypasses all the seals. Combining this situation with a poor evaluation of the extent of the alkaline disturbance, we imagine plug failure leading to increased pH in the C waste cells and accelerated release.
Ability to emplace the bentonite rings with no residual gaps → See § 6.2.12.1	None	Spent fuel disposal cell buffer	Poor emplacement would result in locally defective permeability of the engineered barrier if, in addition, the bentonite failed to swell as expected. This is not expected to affect the engineered barrier in a normal situation since its only role is to protect the cell and not with regard to transfer. The seal failure SEA envisages failure of the buffer combined with seal failure.
Ability to emplace the bentonite bricks with no residual gaps → See § 6.2.12.1	Extent of alkaline disturbance	All seals	Equivalent to previous case for spent fuel cells : poor compacting of bentonite bricks would degrade seal performance. For C waste, this is coupled with a possible excessive spread of the alkaline plume through the plug, and a resulting increase in release.

Uncertainties considered	Other uncertainties taken into account in the analysis[92]	Components concerned	Comments
Emplacement technology of seal hydraulic cut-offs → See § 6.2.12.1	Model of deferred argillite behaviour Effect of bentonite recompression Duration of the operating phase	Anchored seals (drifts and B cell plugs)	The uncertainty concerns all seals supplied with a hydraulic cut-off (drifts and B waste cell plugs). It consists of assuming that excessive loss of rock confinement, for example after a long period of operation, is not compensated by swelling of the anchor. The swelling is ineffective as it is bypassed by a fractured EDZ.
Lining removal and plug emplacement → See § 6.2.13	Model of deferred argillite behaviour Duration of the operating phase Extent of the alkaline plume	C waste disposal cell plugs	The uncertainty concerns C waste cell plugs and the risk of excessive damage during plug emplacement. The SEN envisages ad hoc failure. There is no calculation case specific to a defective plug in an SEA, but the situation is covered by a calculation in which all seals fail. Coupling with an excessive spread of alkaline disturbance through the defective plug, situations are envisaged where radionuclide release by glass is accelerated.

Table 7.2-1 « Seal failure » altered evolution scenario – uncertainties

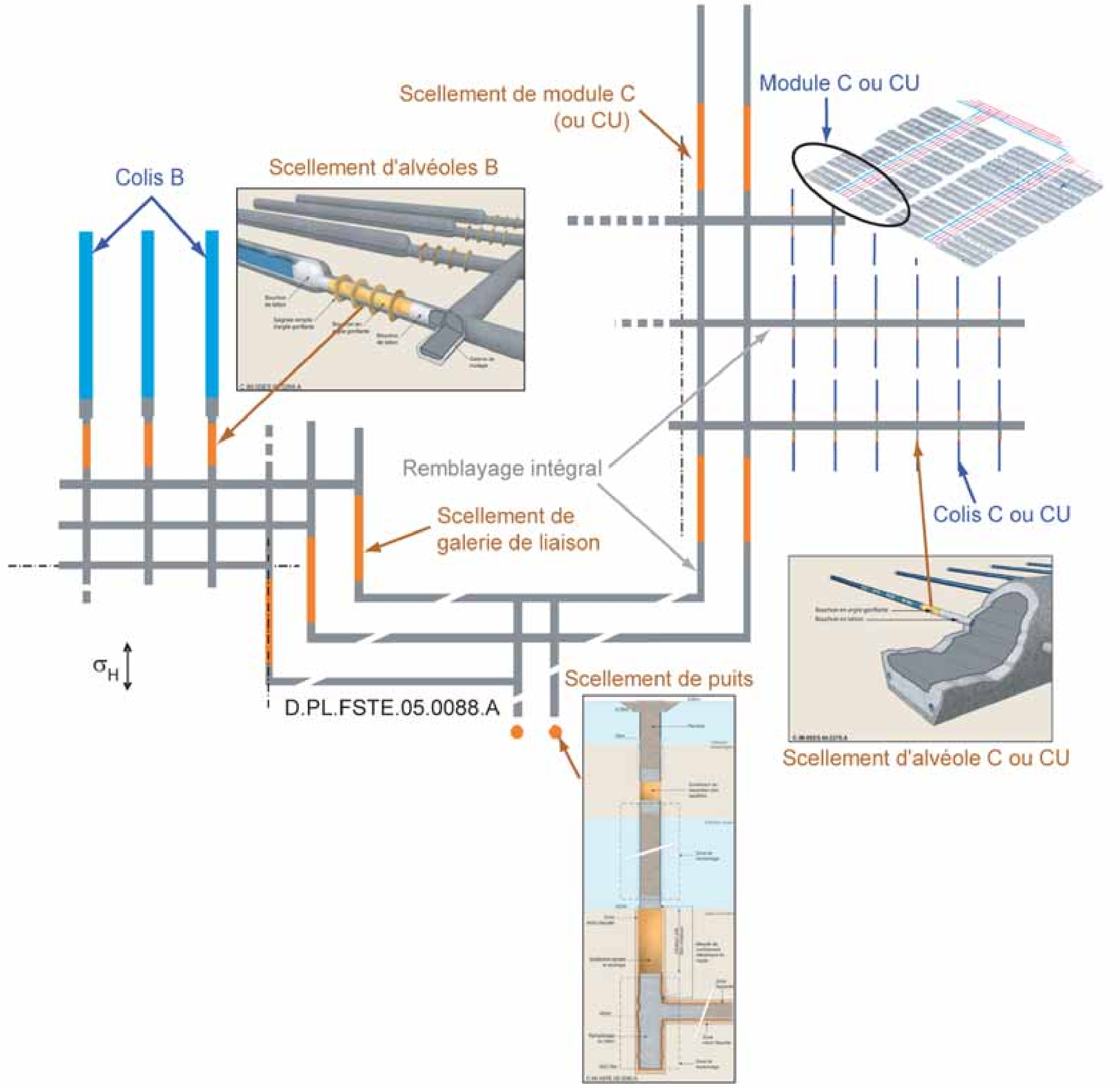


Figure 7.2-1 Schematic representation of the various seal structures

In chapter 6, a number of bounding situations to be considered were identified, corresponding to possible failures identified within the scope of the qualitative safety analysis. Three types of calculation cases have been performed, and a situation not directly coming from in the qualitative safety analysis has been added to complete the list (see Table 7.2-2) :

- Calculation case for failure of shaft seals ;
- Calculation case for failure of seals with hydraulic cutoffs[93] (B waste cell/module seals, repository zone seals and connecting drift seals) ;
- Calculation case for failure of all seals ;
- Calculation case for abandonment of repository without surface/underground access structure seals.

[93] This calculation case is hereinafter referred to as 'defective drift seals' for practicality's sake, although it also includes the B waste disposal cell seals.

	Cell plugs	Repository module and zone seals	Connecting drift seals	Access structure seals
Normal evolution scenario (reminder)	Isolated failure[94]	Efficient	Efficient	Efficient
Shaft seals defective	Isolated failure	Efficient	Efficient	Defective
Drift seals defective	Isolated failure	Defective	Defective	Efficient
All seals defective	Defective C and B waste disposal cell plugs	Defective	Defective	Defective
Abandonment of repository	Isolated failure	Efficient	Efficient	Not sealed

Table 7.2-2 « Seal failure » altered evolution scenario – List of calculation cases analysed

7.2.2 Scenario analysis

This section explains how seal failures are represented. All models and parameters not mentioned below are implicitly identical to those considered in the normal evolution scenario (release models, container lifetime, isolated container defect, etc.).

7.2.2.1 Generic data for all calculation cases

Seal failure is represented by a swelling clay – rock interface defect (suppression of hydraulic cutoff, or presence of fractured zone for non-anchored seals), with the seal core maintaining its hydraulic properties as in the normal evolution scenario, thereby allowing the assumption that the repository zones are sufficiently distant to be independent as regards radionuclide transfers. On the other hand, the degradation of the hydraulic function may make them hydraulically dependent. The hydraulic contribution of the repository zones is taken into account via boundary conditions determined on the basis of a preliminary calculation integrating all repository zones.

In order to limit the number of calculation cases, it was decided to only consider the repository zones that produce the lowest impact in the normal evolution scenario. The spent fuel zones (CU1 and CU2) and vitrified waste zones (C1/C2 and C3/C4) are therefore systematically considered [105]. B waste zones are only considered in the 'all seals defective' calculation case, and only for B1x type waste (see Chapter 5) and bituminised sludges.

In order to represent a possible effect of the establishment of a advective regime within the vitrified waste disposal cells (thereby disturbing the 'limitation of radionuclide release and immobilisation of radionuclides inside the repository' function, particularly in case of plug loss), a degraded glass release model is applied for the reference package considered (C1/C2 and C3/C4). The effect of a advective regime in the vitrified waste disposal cells is taken into account through a constant dissolution rate of the vitreous matrix, equal to the initial rate ('V_0.S' model).

Since relatively rapid transfers are anticipated in the engineered structures, a hydraulic gradient corresponding to the current hydrogeological model has been adopted for the calculation, i.e., 0.2 m/m (highest gradient in the zone). Even higher gradients are nevertheless considered in a sensitivity study to supplement the analysis.

Finally, in the event that this parameter has an influence on the quantities of water travelling through access structures subject to a seal failure, the explicit dependence on temperature of the permeability values for the Callovo-Oxfordian layer and the bentonite has been maintained. The effect of the temperature has been taken into account via a permeability correction factor, by considering various hydraulic states permanent over time. These hydraulic states are obtained by adapting the permeability according to the average temperature of the various materials. These temperatures are derived from the 2D thermal models and represented as constant for each time interval (Figure 7.2-2).

[94] Reminder : An isolated failure is analysed by considering the defective C waste disposal cell plug. Calculations have shown that this is negligible in the normal evolution scenario.

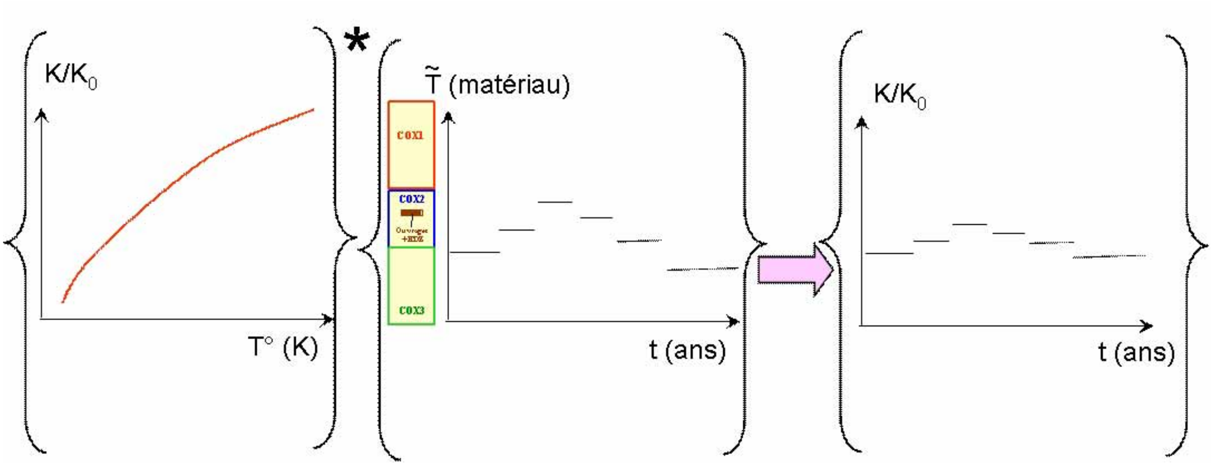


Figure 7.2-2 Seal failure' altered evolution scenario – Taking into account of variations in material permeability

The only radionuclide transfers considered are solute transfers. As discussed in chapter 6, the quantity of carbon 14 contained in the waste package and capable of migrating in gaseous form is very low in all cases and will decrease very significantly before reaching the outlets. The other gaseous radionuclides have a shorter radioactive half-life than carbon 14 or a greater affinity with water.

Note that the influence of hydraulic transients, due to corrosion gases, are not taken into account. Wherever a sealing is defective, it is permeable to gases ever more than to water, and pressures equilibrate on both sides of it. Gases do not induce overpressure gradients, with the possible exception of spent fuel cells, due to the presence of a continuous buffer. In such case, only a defective package would release radionuclides when pressures are still high. This case is a multiple failure and concerns a small part of the inventory, it is not represented.

7.2.2.2 Distinction according to calculation case

- **Calculation case 1 : Failure of all shaft seals**

This calculation case covers the failure of all shaft seals. Unlike the normal evolution scenario (see chapter 5), it takes into account a continuous fractured zone around the shaft throughout the entire height of the Callovo-Oxfordian layer. This fractured zone is assumed to extend to 0.1 R (where R is the shaft radius) and not compressed by the seal. Said extent is arbitrary in that the models do not predict a fractured zone in that region of the Callovo-Oxfordian layer (see Figure 7.2-3). The permeability values adopted for the seal core and for the fractured and microfissured regions of the EDZ are similar to those adopted in the normal evolution scenario and recalled in Table 7.2-3.

Permeability in the seal core	$K = 10^{-11}$ m/s
Permeability of the fractured zone	$K = 5.10^{-9}$ m/s
Permeability of the microfissured zone	$K = 5.10^{-11}$ m/s
Vertical permeability in the unaltered Callovo-Oxfordian	$K_V = 5.10^{-14}$ m/s
Horizontal permeability in the unaltered Callovo-Oxfordian	$K_H = 5.10^{-13}$ m/s

Table 7.2-3 « Seal failure » altered evolution scenario – Permeability value in the fractured zone, microfissured zone, seal core and unaltered geological barrier (reminder)

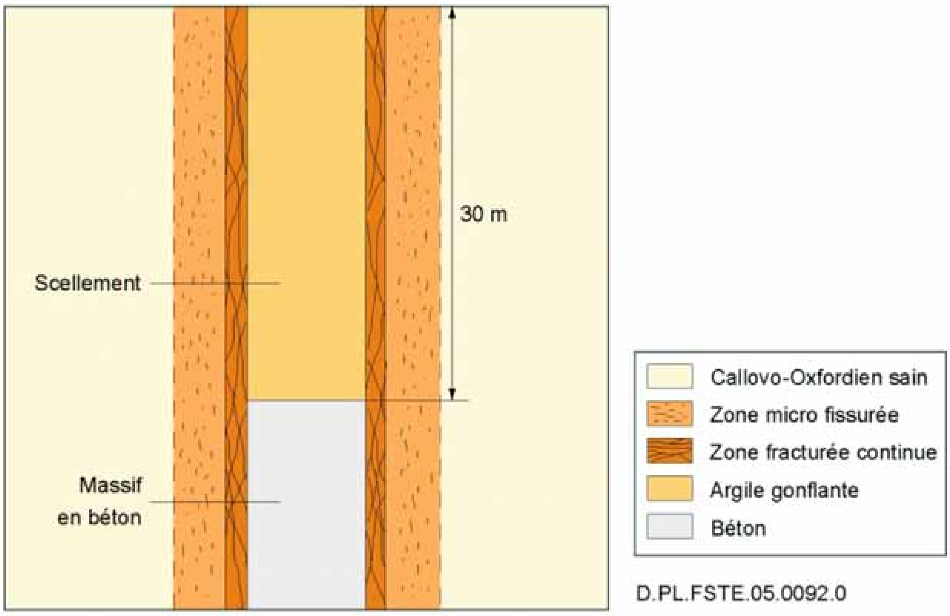


Figure 7.2-3 Schematic representation of a defective shaft seal

The failure start date is difficult to evaluate in that it depends on the process that causes the failure. For example, in case of insufficient bentonite swelling, the seal is never efficient. The seal therefore becomes 'defective' as from the moment that it is assumed to have swelled sufficiently to fulfil its function. On the other hand, if the causes of the failure are associated with the propagation of a chemical disturbance (such as alkaline disturbance), then the failure must involve prior degradation of the concrete engineered structures and sufficient penetration of the alkaline plume into the rock-bentonite interface, which requires several thousands of years.

The seal failure is definitively and conventionally considered as effective a few centuries[95] after repository closure.

This calculation case allows for determining the degree of redundancy offered by the drift and cell seals with respect to the shaft seals.

- **Calculation case 2 : Failure of all seals with hydraulic cutoff of infrastructures and repository zones**

Calculation case 2 covers the failure of all drift seals and B waste cell head seals with a common failure mode for hydraulic cutoffs ensuring the interruption of the fractured zone with the highest permeability in the engineered structure walls. As in the normal evolution scenario, the drift seals are represented with a seal core and equivalent anchoring. In case of drift seal failure, only the equivalent hydraulic cutoff parameters differ from those of the normal evolution scenario (which correspond to the fractured zone) (see Figure 7.2-4).

95 A date corresponding to 500 years after the start of radionuclide decay is adopted for calculation purposes. This date is purely arbitrary and obviously unrelated to the institutional monitoring period for the repository.

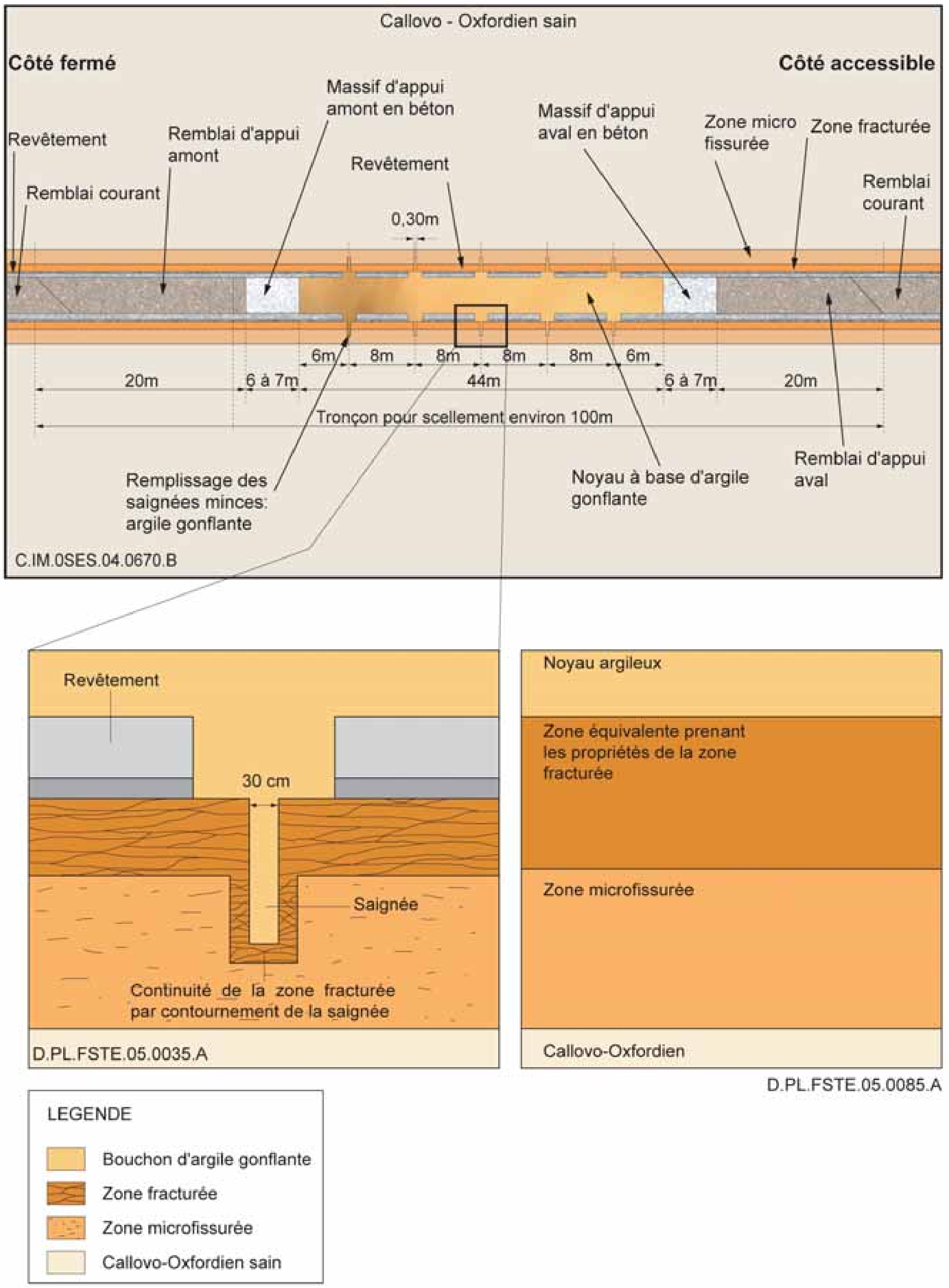


Figure 7.2-4 Schematic representation of drift seal failure via the hydraulic cut-offs

In this situation, the shaft seals completely fulfil their role of isolating the repository from the aquifers. This calculation case allows for determining whether these engineered structures offer a certain degree of redundancy with respect to the seals installed in the drifts.

- **Calculation case 3 : Failure of all the seals**

This situation requires the analysis of all repository zones. It covers the failure of all seal structures, from the cell plugs to the access shafts. It leads to considering the existence of a continuous fractured zone from the waste package to the surrounding formations. This type of situation may cover an unanticipated evolution of the EDZ or excess damage due to gases, for example.

Shaft seal failure is represented in the same manner as for the first case. Drift seal failure is modelled similarly to the second case. Calculation methods are similar to those used in the first and second cases.

- **Calculation case 4 : Abandonment of repository without shaft seals**

It is assumed that the repository is abandoned and the shafts are not sealed. Nevertheless, all other engineered structures (main and secondary connecting drifts, cell access drifts) are backfilled and sealed. All drift seals are efficient. This situation is analysed differently from the others because the disruptive event (absence of sealed shafts) occurs as from repository closure.

In this particular situation, the absence of shaft seals is modelled with a high permeability (10^{-6} m/s) at the location where the seal core should have been installed, as opposed to 10^{-11} m/s when it is present and efficient.

7.2.2.3 Sensitivity studies

The following sensitivity studies are conducted :

- In all the calculation cases, in order to take into account possible excess damage to the EDZ (for example, further to an unanticipated evolution of its properties), a sensitivity study has been conducted for a situation where the hydraulic, geochemical and transport properties of the fractured zone and microfissured zone are degraded to values more unfavourable than the reference values recalled in Table 7.2-4). In said study, possible damage to clay engineered barriers and seal cores is represented by degrading their overall permeability from 10^{-11} m/s to 10^{-9} m/s ;

	Reference calculation Phenomenological EDZ	Sensitivity calculation 'Degraded' EDZ
Fractured zone	K = 5.10^{-9} m/s Anions : $De_{Anions} = 1.10^{-11}$ m²/s $\omega_{Anions} = 0.15$ Cations : $De_{Cations} = 5.10^{-10}$ m²/s $\omega_{Cations} = 0.20$ Phenomenological geochemical retention	K = 10^{-6} m/s pessimistic coefficient ($Dp = De / \omega = 2.10^{-9}$ m²/s) No geochemical retention
Microfissured zone	K = 5.10^{-11} m/s Anions : $De_{Anions} = 5.10^{-12}$ m²/s $\omega_{Anions} = 0.05$ Cations : $De_{Cations} = 2{,}5.10^{-10}$ m²/s $\omega_{Cations} = 0.18$ Phenomenological geochemical retention	K = 5.10^{-9} m/s Anions : $De_{Anions} = 1.10^{-11}$ m²/s $\omega_{Anions} = 0.04$ Cations : $De_{Cations} = 5.10^{-10}$ m²/s $\omega_{Cations} = 0.21$ Conservative geochemical retention

Table 7.2-4 « Seal failure » altered evolution scenario – Sensitivity to EDZ hydraulic transfer and chemical retention parameter values

- In order to take into account the possibility of a propagation of the alkaline plume through the defective cell plug and up to the first vitrified waste package (highly improbable according to the evaluations conducted), we consider an accelerated release in an environment with a pH level greater than 9, at least for the package nearest to the plug. No model is available to directly represent such a release. It is integrated into a sensitivity study by applying a conservative dissolution rate and a fracturing rate of 40 (eight times greater than in the reference model) to all vitrified waste (which is highly pessimistic). This results in a complete dissolution of the glass after a few centuries, equivalent to a labile release (in terms of transfer times through the host formation or access structures). This sensitivity study is conducted with degraded EDZ parameter values and a high bentonite permeability value (10^{-9} m/s instead of the reference value of 10^{-11} m/s). It is conducted for the 'all seals defective' situation and, for illustrative purposes, for the 'shaft seals defective' situation ;

- In order to take into account a possible degradation of the 'delay and attenuation of radionuclide migration' function within the seal care or clay engineered barrier of cells equipped with such structures, due to a partial loss of their properties (for example, further to a heterogeneous chemical disturbance within the bentonite), we also consider a sensitivity study in which the bentonite has conservative chemical retention and transfer properties. The values of the chemical retention and transport parameters, already adopted as normal evolution scenario sensitivity values, are shown in Table 7.2-5 ;

	Reference Calculation					Sensitivity to geochemical parametres	
	Period [years]	$\omega_{Diffusion}$ [-]	De [m²/s]	R [-]	Csat [mol/m^3]	R [-]	Csat [mol/m^3]
^{10}Be	1 600 000	0,36	5.10^{-10}	973	10^{-2}	1	10
^{10}Be (deltaT>20)	1 600 000	0,36	5.10^{-10}	98	10^{-2}	1	10
^{14}C	5 730	0,05	5.10^{-12}	1	2,3	1	9
^{36}Cl	302 000	0,05	5.10^{-12}	1	soluble	1	soluble
^{41}Ca	103 000	0,36	5.10^{-10}	6	2,3	1	9
^{41}Ca (deltaT>20)	103 000	0,36	5.10^{-10}	1,5	2,3	1	9
^{59}Ni	75 000	0,36	5.10^{-10}	2 430	5.10^{-2}	487	1
^{79}Se	65 000	0,05	5.10^{-12}	1	5.10^{-7}	1	5.10^{-4}
^{93}Zr	1 530 000	0,36	5.10^{-10}	486 000	2.10^{-5}	48 600	3.10^{-3}
^{93}Mo	3 500	0,05	5.10^{-12}	1	1.10^{-5}	1	1.10^{-3}
^{93m}Nb	16,4	0,05	5.10^{-12}	350 000	2.10^{-4}	35 000	2.10^{-3}
^{94}Nb	20 300	0,05	5.10^{-12}	350 000	2.10^{-4}	35 000	2.10^{-3}
^{99}Tc	213 000	0,36	5.10^{-10}	146 000	4.10^{-6}	48 600	1.10^{-4}
^{107}Pd	6 500 000	0,36	5.10^{-10}	4 380	4.10^{-4}	4 380	1.10^{-2}
^{126}Sn	100 000	0,36	5.10^{-10}	53 500	1.10^{-5}	14 600	1.10^{-4}
^{129}I	15 700 000	0,05	5.10^{-12}	1	soluble	1	soluble
^{166m}Ho	1 200	0,36	5.10^{-10}	58 300	1.10^{-4}	5 830	1.10^{-3}
^{135}Cs	2 300 000	0,36	5.10^{-10}	487	soluble	290	soluble
^{135}Cs (deltaT>20)	2 300 000	0,36	5.10^{-10}	49	soluble	30	soluble

Table 7.2-5 « Seal failure » altered evolution scenario – Reference and sensitivity values of chemical retention parameters in the clay engineered barrier and in the seals

- In order to take into account the uncertainties associated with the value of the vertical head gradient (and its evolution with time in particular), a sensitivity study has been conducted by considering an upward gradient of 0.4 m/m (corresponding to a million years model) and a pessimistic upward gradient of 1 m/m (not corresponding to a physical reality, but used to test the influence of this specific parameter in a context where advection may possibly constitute an important migration mode).

Table 7.2-6 sums up the sensitivity studies conducted and the reference package for which they have been conducted.

	Situations	Reference package
Sensitivity to EDZ (degraded EDZ + Kcore = 10^{-9} m/s)	All seals defective	CU1, CU2
	Access structure seals defective	CU1, CU2
Sensitivity to EDZ (degraded EDZ + Kcore = 10^{-9} m/s) + penalising source term for glass (penalising V_0.S)	All seals defective	C1/C2, C3/C4
	Access structure seals defective	C1/C2, C3/C4
Sensitivity to bentonite geochemistry (conservative geochemistry)	All seals defective	CU2
Strong upward gradient	All seals defective	CU1

Table 7.2-6 « Seal failure » altered evolution scenario – List of sensitivity studies conducted

7.2.3 Effects on safety functions

As in the case of the normal evolution scenario, the results analysis approach is geared towards two objectives :

- The first objective is to understand the operation of the system in case of seal failure or non-closure. Intermediate indicators are used to verify the efficiency of the functions implemented ;
- The second objective is to evaluate the radiological impact associated with the various situations considered with regard to reference and sensitivity values.

As previously discussed, seal failure situations generate a degradation or modification of the performance characteristics of the three safety functions. Situations such as defined above directly or indirectly jeopardise the hydraulic properties associated with the seals. In these types of configurations, water circulation through the engineered structures is intensified, thereby degrading the performance characteristics of the « resist to the water circulation » function. In addition, a generalised seal failure induces a potential short-circuit of the confinement barriers between the waste package and humans, and therefore a deterioration of the performance characteristics associated with the 'delay and reduction of radionuclide migration' function. Finally, as in the normal evolution scenario, it is assumed that a few spent fuel containers or C waste overpacks are subject to premature failure as of 200 years after repository closure (see chapter 5). This premature failure leads to a decrease in the performance of the 'minimisation of radionuclide release and immobilisation of radionuclides in repository' function in that it causes radionuclide release and migration during the thermal transient.

First the effects of the various seal failure situations on the three safety functions will be described based on intermediate indicators identical to those used in the normal evolution scenario (see chapter 5.5), and then the impacts associated with the fraction of activity subject to an altered evolution and circulating through the access pathways will be evaluated. The impact associated with the rest of the activity follows a normal evolution and is not represented (cannot be distinguished from the impact associated with the normal evolution scenario).

Results may vary according to waste repository zone and radionuclide-specific analysis. They are differentiated according to waste type and radionuclide whenever necessary. To illustrate a result, the specific case of iodine 129 for the spent fuel zone CU1 is used as the default example (highest impact).

7.2.3.1 CU1 spent fuels

- **Effect on « resist water circulation » function**

Shaft seal failure or abandonment of repository without shaft seals (calculation cases 1 and 4)

These two situations are analysed together since they both jeopardise the shaft seal hydraulic function (due to the presence of a fractured zone around the shaft seal, or the absence of seals). They nevertheless differ in that the permeability considered for the fractured zone around the shaft in the first case ($K=5.10^{-9}$ m/s) is significantly lower than that considered in case of repository abandonment (10^{-6} m/s). It must be noted that in case of efficient sealing (no fractured zone), the highest permeability in the shafts is in the microfissured zone (5.10^{-11} m/s).

If the seals are efficient, the hydraulic influence of the overlying formations on the repository is low. Water flows in the repository remain slow and limited (water flow exiting the shaft : 0.5 m^3/year) and the 'minimisation of water circulation' function is effective. The hydraulic head gradients in the Callovo-Oxfordian layer are imposed in a pessimistic manner (upwards) throughout the transposition zone (and therefore in the shafts). As a result, when the equivalent permeability of the shaft seal increases, a head loss is observed in the shaft and tends to reach equilibrium with that of the overlying formations.

In the two situations considered, the permeability values in the shafts are sufficiently high to allow the hydraulic head in the overlying formations to be transmitted to the shaft base. This low hydraulic head then tends to be transmitted to the repository structures, but its influence is limited due to the efficiency of the drift seals (redundant with the shaft seals), whose equivalent permeability is in the order of 5.10^{-9} m/s. The head loss produced in the repository amounts to approximately 2 metres with respect to the reference situation, as illustrated by the piezometric cross-sections in Figure 7.2-5 and Figure 7.2-6.

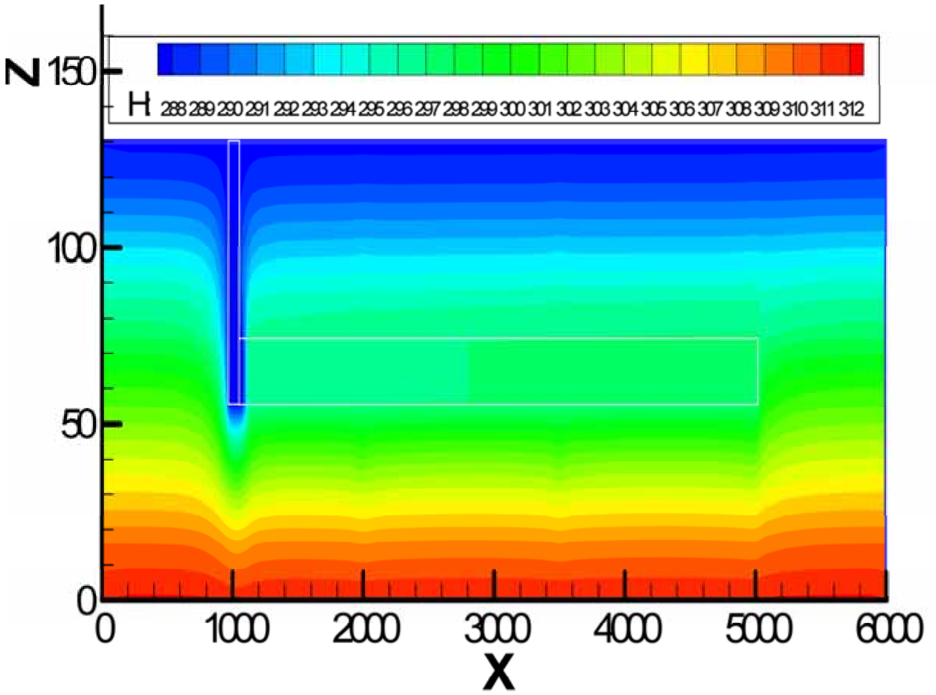


Figure 7.2-5 « Seal failure » altered evolution scenario - Vertical cross-section through the connecting structures and shafts – Piezometry in CU1 spent fuel repository zone – Shaft seals defective (H = NGF hydraulic head)

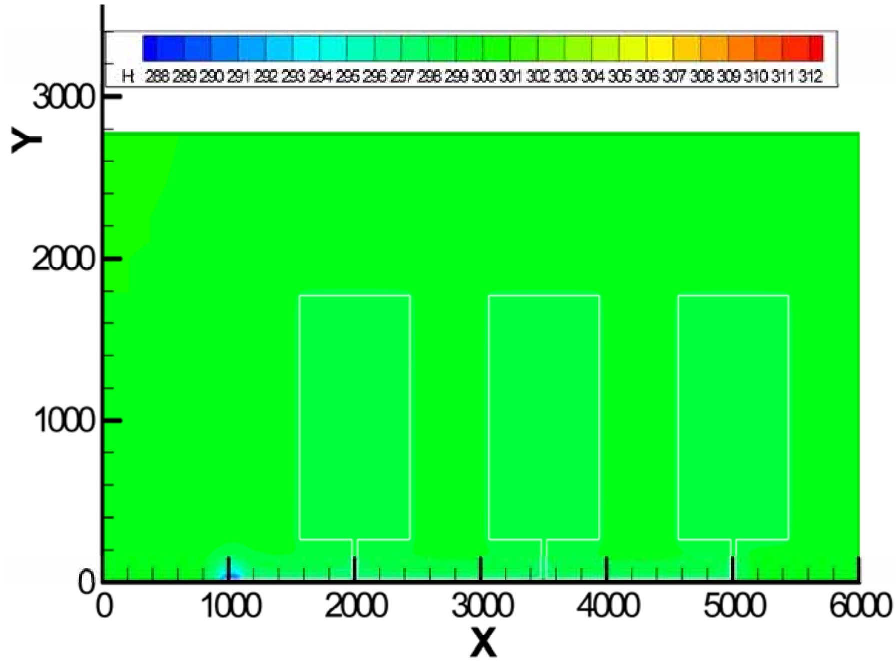


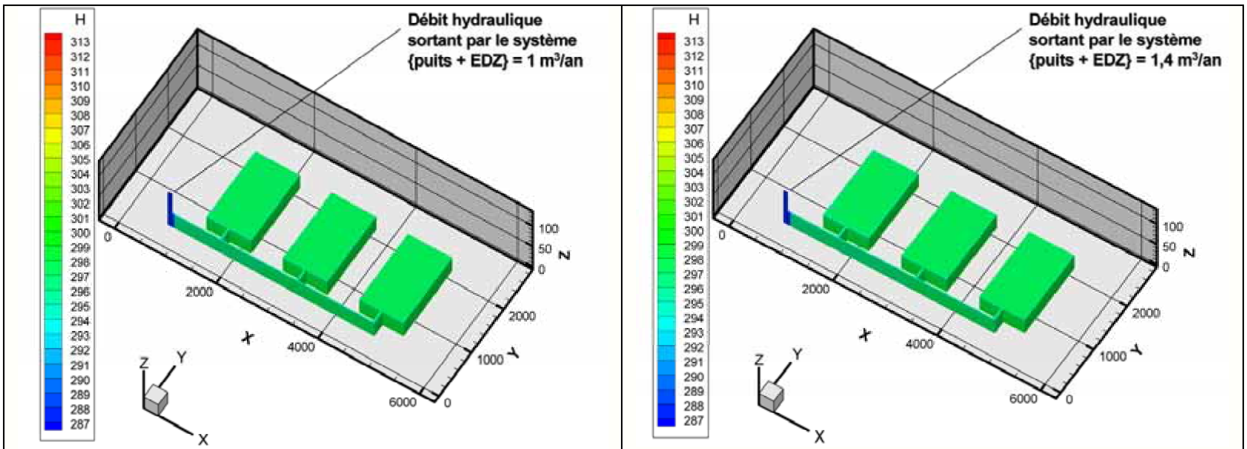
Figure 7.2-6 *Seal failure' altered evolution scenario – Horizontal cross-section through the repository – Piezometry in CU1 spent fuel repository zone – Shaft seals defective*

In addition, the horizontal head gradients between the repository structures and the shaft favour water flows towards the latter. In the shaft seal failure situation, the water flow exiting the shaft amounts to 1 m^3/year.

The system is advective at the repository zone entrance, with a Peclet number of approximately 20 at the secondary connecting drift entrance (as opposed to 10 in the normal evolution scenario).

In the shaft abandonment situation, the higher permeability value adopted in the shaft produces a slightly greater hydraulic disturbance in the repository structures. In particular, water flows of approximately 1.4 m^3/year are observed exiting the shaft.

It must nevertheless be noted that these flow rates remain low even though they are significantly higher than in the reference situation (0.5 m^3/year). Despite their pessimistic representation in the scenario (uninterrupted microfissured zone), the efficiency of the various drift seal hydraulic cutoffs is redundant with that of the shafts seals and significantly reduces the hydraulic disturbance in the repository (see Figure 7.2-7).



Situation de scellements de puits défaillants | Situation d'abandon du stockage

Figure 7.2-7 *Seal failure' altered evolution scenario – CU1 reference package – Hydraulic flow exiting the shafts – Shaft seal failure and repository abandonment*

In the access drifts and disposal cells, the system remains diffusive (Peclet number of approximately 0.2 in the middle of the access drifts). Under these conditions, the distribution of the transfer pathways inside the disposal cells (among the flows entering the geological barrier and the drift) is the same as in the normal evolution scenario.

Isolated plug defects have no visible influence on water flow, in this case or in any other case discussed below.

Failure of seals with hydraulic cutoff (calculation case 2)

This situation produces a smaller hydraulic disturbance in the repository than the situations previously considered, due to the fact that the shaft seals remain efficient. The head loss in the repository structures is more limited (approximately 1 metre).

This situation is close to the normal evolution scenario in that the hydraulic head at the shaft base is not disturbed by the hydraulic head in the overlying formations. Nevertheless, the drift seal failure, characterised by a fivefold increase in the equivalent permeability of the hydraulic cutoffs[96], produces water flows slightly higher than those considered in the normal evolution scenario, i.e., 0.75 m^3/year and 0.5 m^3/year at the shaft exit, respectively. In both cases, the flow rates remain extremely low.

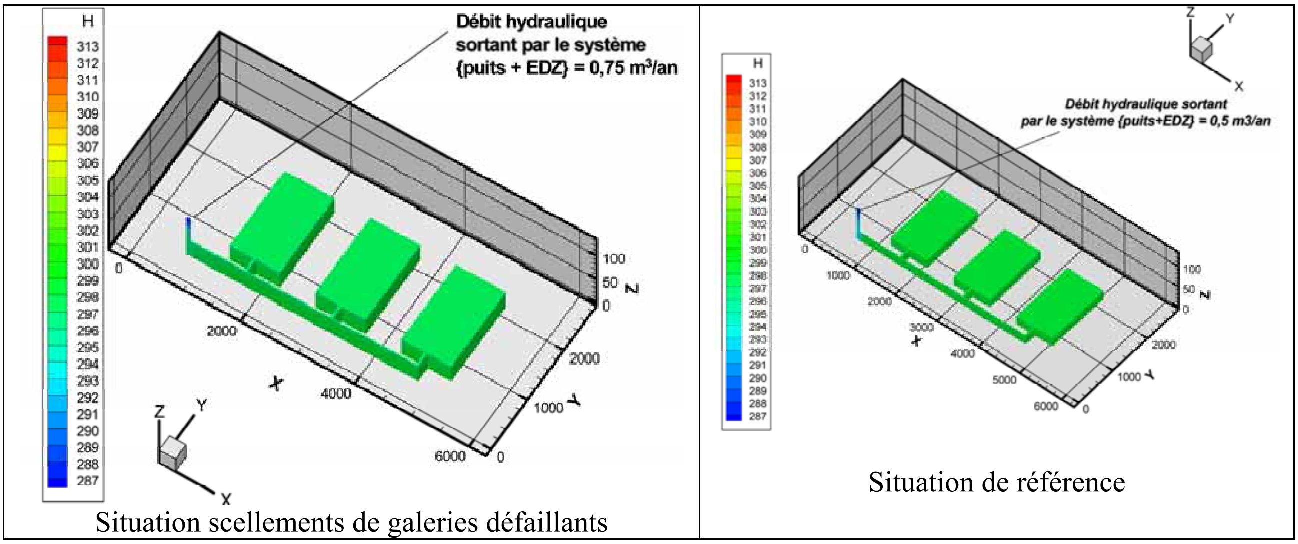


Figure 7.2-8 Seal failure' altered evolution scenario – CU1 reference package – Hydraulic flow exiting the shafts – Drift seal failure – Reference calculation

The advective phenomena in the repository zone are slightly greater than in the shaft seal failure situation, due to the inefficiency of the drift seals (Peclet number at zone exit in the order of 25, as opposed to 20 in the shaft seal failure situation and 10 in the normal evolution scenario).

However, the system remains diffusive (or codominantly diffusive and advective) in the repository modules and in the cells. As in the previous cases, the distribution of the transfer pathways inside the disposal cells is no different from that assessed in the normal evolution scenario.

All seals defective (calculation case 3)

This situation produces a greater hydraulic disturbance than the two previous ones because the seal systems contributing to the hydraulic resistance of the access structures (drifts and shafts) are defective. The hydraulic head in the overlying formations therefore propagates more easily from the roof of the Callovo-Oxfordian layer to the repository, generating a head loss (in the repository) of approximately 4 metres with respect to the reference situation, as shown in Figure 7.2-9 and Figure 7.2-10.

96 Global permeability of the system on account of the hydraulic cutoff, the concrete lining left in place between the anchor ing keys, and the surrounding fractured zone.

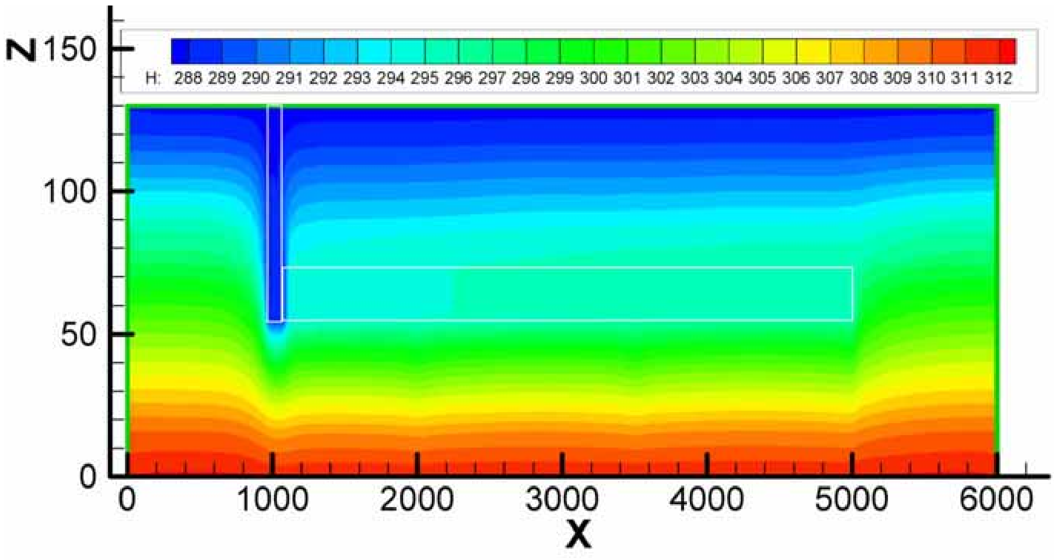


Figure 7.2-9 'Seal failure' altered evolution scenario - Vertical cross-section through the connecting structures and shafts – Piezometry in CU1 spent fuel repository zone – All seals defective

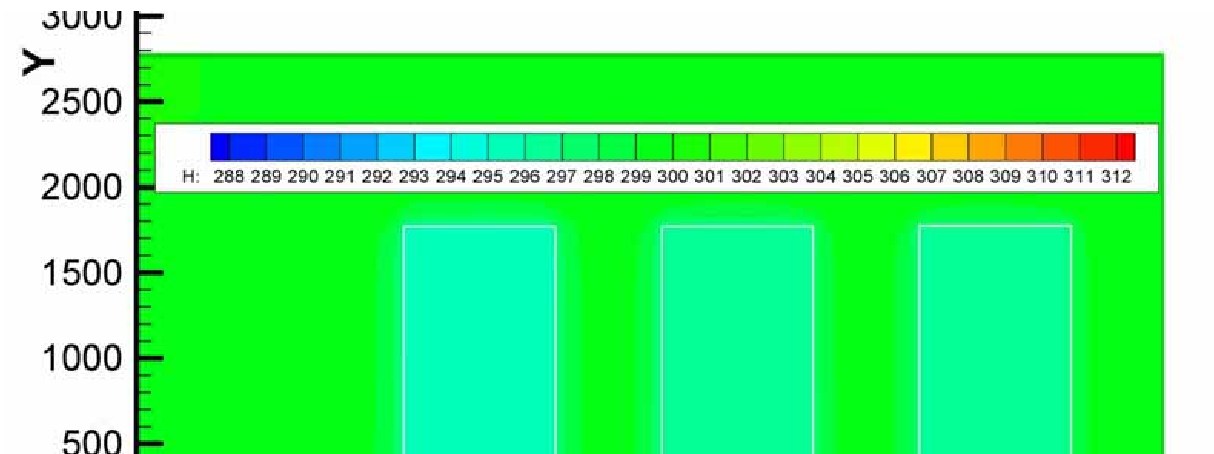


Figure 7.2-10 'Seal failure' altered evolution scenario – Horizontal cross-section through the repository – Piezometry in CU1 spent fuel repository zone – All seals defective

In addition, the head loss in the shaft seals and the permeability of the degraded hydraulic cutoffs both contribute to increasing the water circulation in the repository. The water flow exiting the shafts is also more important (approximately 2.4 m^3/year). In this configuration, the characteristic migration time via advection is approximately 250 000 years from the repository zone to the shafts (see Figure 7.2-11).

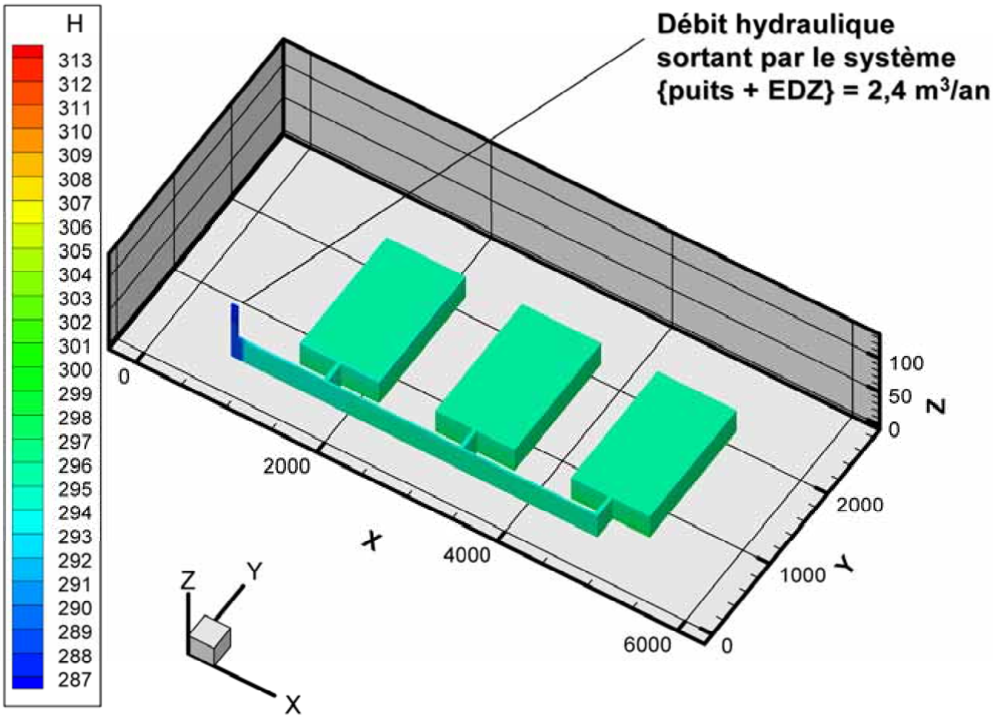


Figure 7.2-11 'Seal failure' altered evolution scenario – CU1 reference package – Hydraulic flow exiting the access structures – All seals defective – Reference calculation

- **Effect on 'delay and reduction of radionuclide migration' function**

The preceding analysis showed that, despite the seal failure, the system remains diffusive (or codominantly diffusive and advective) in the cells and in the cell access drifts. Nevertheless, in the rest of the repository, the seal failure considered for the various calculation cases generates an increase of the advective kinetics, which contributes to accelerating the radionuclide transfer in the structures, from the packages to the roof of the repository.

It therefore seems relevant to evaluate the effect of the various seal failure situations on radionuclide confinement, and more particularly on the capacity of the repository and geological environment to delay and attenuate radionuclide migration.

In case of seal failure, once released from the packages, radionuclides migrate into the cell body or plug. The activity level that finally reaches the shafts depends on the following :

- Initial activity level of the packages ;
- Distribution among the transfer pathways within the cell itself (see Figure 7.2-12). One fraction of the activity follows a horizontal trajectory (pathway 2) before reaching the access drifts, whereas the other follows a vertical trajectory (pathway 1) before reaching and remaining in the Callovo-Oxfordian layer. The distribution among these two transfer pathways depends on the degree of radionuclide sorption in the host formation and engineered structures. Since the transport regime remains diffusive (or codominantly diffusive and advective) within the cell, this distribution is comparable to that observed in the normal evolution scenario ;

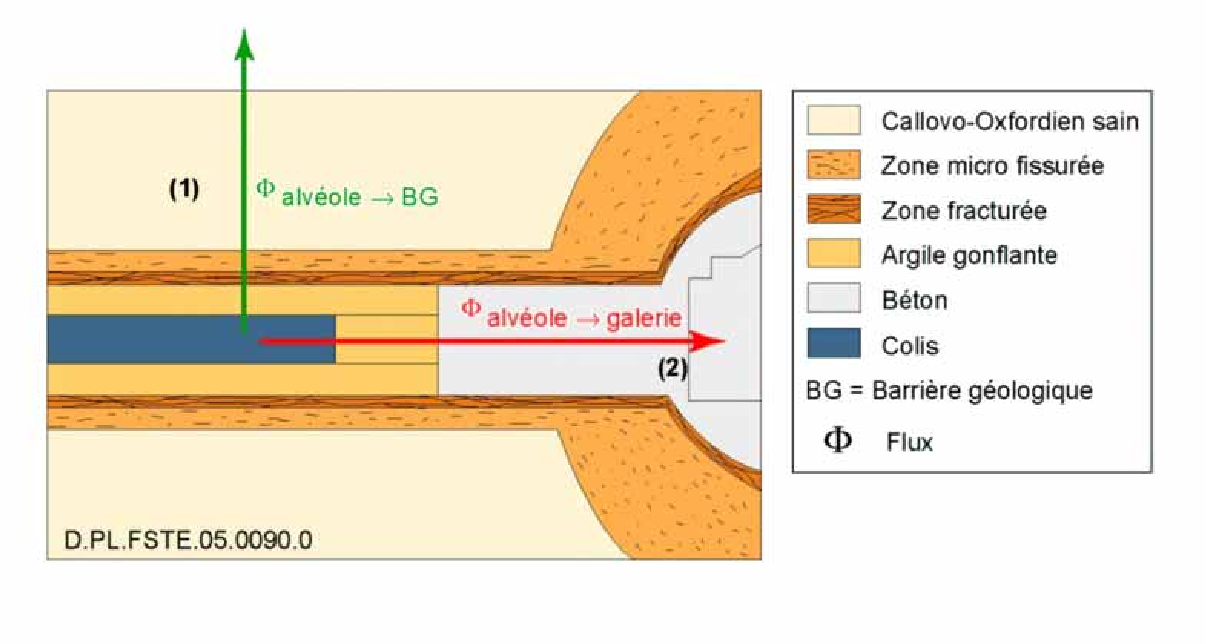


Figure 7.2-12 Seal failure' altered evolution scenario – Transfer pathways in case of C waste or spent fuel disposal cells

Ability of the radionuclides to attain and remain confined within the access drifts. Radionuclide migration in the drifts may be attenuated between the cell exit and the shaft due to a diffusive migration from the drifts to the Callovo-Oxfordian layer and/or the radioactive decay of the radionuclides (see Figure 7.2-13).

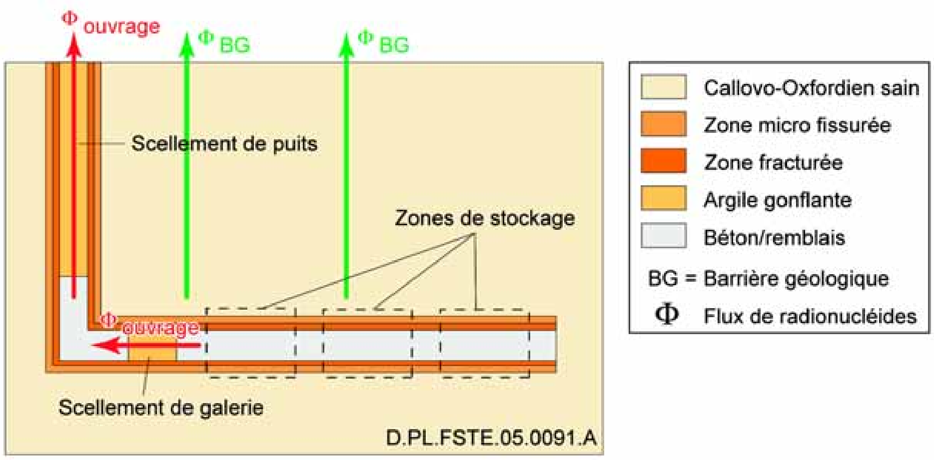
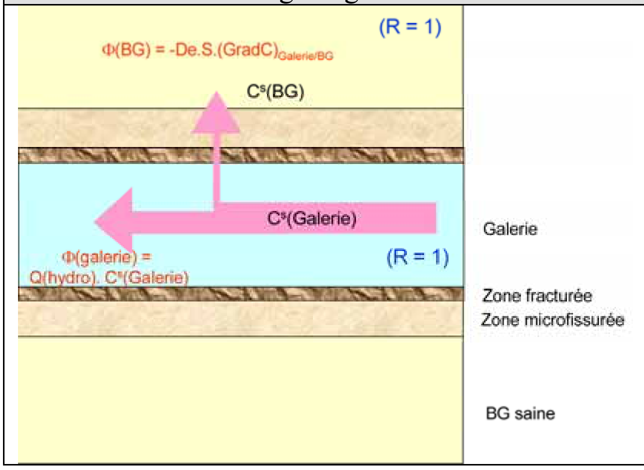


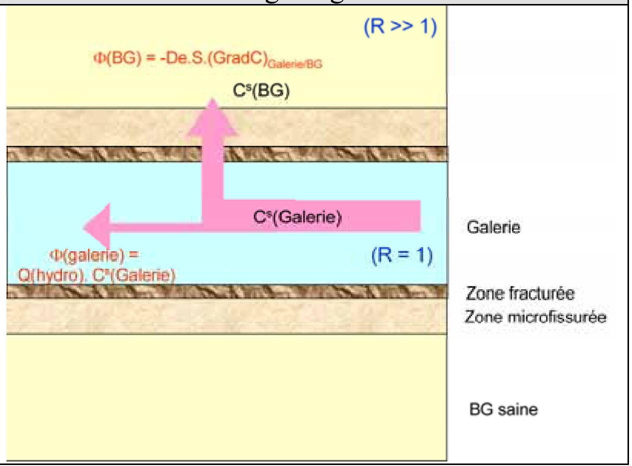
Figure 7.2-13 Seal failure' altered evolution scenario – Potential transfer pathways in case of seal failure

The evaluation conducted clearly shows that most of the radionuclides initially present in the packages benefit from radioactive decay in the drifts (due to the slow transfer kinetics) and/or migrate into the Callovo-Oxfordian layer before reaching the spent fuel repository zone exit. In all the situations considered, the long-lived radionuclides subject to significant sorption in the geological barrier (^{59}Ni, ^{107}Pd, ^{93}Zr, ^{99}Tc, ^{10}Be, ^{41}Ca, ^{135}Cs, etc.) display a very low (or even nil) molar flow rate at the repository zone exit. This result also applied to the actinides. The hundreds of metres of drifts within the zone, where advection is slow, favour exchanges with the geological barrier, which acts as an « absorbing barrier » (see Figure 7.2-14).

In addition, the molar flow rate of medium-lived radionuclides subject to little or no sorption in the geological barriers (^{93}Mo) is almost completely attenuated at the repository sub-zone exit due to radioactive decay. As a result, only ^{14}C, ^{36}Cl, ^{79}Se and ^{129}I have a non-negligible molar flow rate at the zone exit. The analysis that follows addresses these four radionuclides.

Exchanges {drifts → GB} in the case of a <u>radionuclide subject to little or no sorption</u> in the unaltered geological barrier	Exchanges {drifts → GB} in the case of a <u>radionuclide subject to significant sorption</u> in the unaltered geological barrier





Where : R = retardation coefficient (R = 1 in drift for all radionuclides), BG = geological barrier
De = diffusion coefficient in the unaltered geological barrier
$C^s(GB)$ = solution concentration in the geological barrier
$C^s(drift)$ = solution concentration in drift
Φ(GB) = radionuclide flux entering the unaltered geological barrier
Φ(drift) = flux travelling through backfilled drift
Q(hydro) = hydraulic flow through the drifts
S = exchange surface
$(GradC)_{drift/GB}$ = solution concentration gradient between the drift and the unaltered geological barrier

Figure 7.2-14 'Seal failure' altered evolution scenario – Exchanges from drift to geological barrier for radionuclides not subject to or strongly subject to sorption in the geological barrier

Distribution within the disposal cell

As previously discussed, the distribution of radionuclide mass between the vertical and horizontal trajectories is linked to the retention properties of the radionuclides in the host formation and, in the case of spent fuel disposal cells, in the clay engineered barrier.

The main intrinsic characteristics of the factors involved in the radionuclide transfer are stated in chapter 5. The sorption characteristics and radioactive decay half-lives of the four radionuclides considered are listed in Table 7.2-7).

Radionuclides	Radioactive decay half-life [years]	Geological barrier		Clay engineered barrier	
		Retardation coefficient [-]	Solubility [mol/m^3]	Retardation coefficient [-]	Solubility [mol/m^3]*
^{36}Cl	3.02 E+05	1	Soluble	1	Soluble
^{79}Se	6.50 E+04	1	5.10^{-7}	1	5.10^{-7}
^{129}I	1.57 E+07	1	Soluble	1	Soluble
^{14}C	5.73 E+03	6	2.3	1	2.3

Table 7.2-7 Seal failure' altered evolution scenario – Chemical retention characteristics of radionuclides mobilised in the Callovo-Oxfordian layer and in the swelling clay components (for radionuclides contributing to impact in the 'seal failure' altered evolution scenario)

The mass distribution emitted by the packages in the near field of the disposal cell over a million year is listed in Table 7.2-8, i.e. :

- mass percentage passing through the first metres of the geological barrier and remaining inside it, as compared to the total initial mass contained in the repository zone ;
- mass percentage entering the drift, as compared to the total initial mass contained in the repository sub-zone ;
- sum of the two masses.

Radionuclides	Mass exiting the near field of the disposal cell (mass entering and remaining in the host formation + mass entering the drift)	Distribution of radionuclide release from the disposal cells	
		Mass entering and remaining in the host formation (pathway 1 Figure 7.2-12)	Mass entering the drift *(pathway 2 – Figure 7.2-2)*
^{129}I	100.000 %	41 %	59 %
^{36}Cl	79 %	32 %	47 %
^{14}C	12 %	8 %	4 %
^{79}Se	0.29 %	0.17 %	0.12 %

Table 7.2-8 'Seal failure' altered evolution scenario – Distribution of transfer pathways in the near field of the disposal cell integrated to one million years – CU1 reference package

These results indicate the following :

- Iodine 129 and chlorine 36, long-lived soluble radionuclides not subject to or very slightly subject to sorption in the host formation or bentonite, completely (or almost completely) exit the disposal cell after one million years. In this case, the decrease in mass in the near field is very small, or even nil. The mass emitted by the packages preferentially reaches the drifts via diffusive migration in the engineered structures between the packages and the access drift (with 59 % entering the drift and 41 % entering the Callovo-Oxfordian layer in the case of iodine 129). It must be noted that the boundary conditions chosen (nil concentration above the plug) tend to favour horizontal migration for all radionuclides, so the quantity of iodine 129 entering the drift is probably overestimated ;
- Carbon 14, a medium-lived radionuclide subject to low sorption in the Callovo-Oxfordian layer, benefits from radioactive decay and remains confined for the most part (at least 80 %) in the near field of the disposal cell. This element is also slightly subject to sorption in the geological barrier and migrates preferentially into the latter.
- Selenium 79, not subject to sorption but precipitating in the near field, has a low activity rate strongly attenuated at the cell exit. This attenuation affects the mass exiting the near field of the cell over the analysis period (1 million years).

This distribution is valid for all situations involving CU1 spent fuels, because the system remains diffusive in the access drifts and in the disposal cells for all the situations considered.

Transfer to the shafts

The approach consists of evaluating the fraction of activity that reaches the shaft, as compared to the total inventory of the repository zone. As previously discussed, the quantity of activity that reaches the shaft is closely related to the following :

- Radionuclide transfer times in the engineered structures, from the packages to the shafts. The higher the transfer times in the engineered structures, the more the radionuclides benefit from radioactive decay before reaching the shaft. Moreover, the slower the mass travels through the engineered structures, the more it tends to diffusively migrate from the drifts into the Callovo-Oxfordian layer ;

- Sorption capacity of the radionuclides in the geological barrier.

Table 7.2-9 to Table 7.2-13 below show the distribution of the transfer pathways for the four failure situations considered and for the normal evolution scenario, from the disposal cells to the shafts over a million years.

	Cell		**Repository zone Efficiency of repository zone seals**		**Access structures Efficiency of shaft seals**	
	Entering the drifts	*Entering the unaltered Callovo-Oxfordian*	*Exiting the repository sub-zone (i.e., **before** the repository zone seals)*	*Exiting the repository zone seals*	*Exiting the main connecting drift seals*	*Exit {shaft + EDZ}*
^{14}C	4 %	8 %	0.000006 %	0.00000001 %	nil	nil
^{36}Cl	46 %	32 %	0.0092 %	0.0011 %	0.00017 %	0.00014 %
^{129}I	59 %	41 %	0.015 %	0.0022 %	0.00055 %	0.00048 %
^{79}Se	0.12 %	0.17 %	0.000013 %	0.0000008 %	0.000000033 %	0.000000023 %

Table 7.2-9 Seal failure' altered evolution scenario – Distribution and quantification of the various transfer pathways – Shaft seals defective – CU1 reference package

	Cell		**Repository zone Efficiency of repository zone seals**		**Access structures Efficiency of shaft seals**	
	Entering the drifts	*Entering the unaltered Callovo-Oxfordian*	*Exiting the repository sub-zone (i.e., before the repository zone seals)*	*Exiting the repository zone seals*	*Exiting the main connecting drift seals*	*Exit {shaft + EDZ}*
^{14}C	4 %	8 %	0.000007 %	0.00000000 %	nil	nil
^{36}Cl	46 %	32 %	0.0106 %	0.0010 %	0.00011 %	0.00008 %
^{129}I	59 %	41 %	0.017 %	0.0021 %	0.00038 %	0.00030 %
^{79}Se	0.12 %	0.17 %	0.000014 %	0.0000007 %	0.000000015 %	0.000000008 %

Table 7.2-10 Seal failure' altered evolution scenario – Distribution and quantification of the various transfer pathways – Drift seals defective – CU1 reference package

	Cell		**Repository zone Efficiency of repository zone seals**		**Access structures Efficiency of shaft seals**	
	Entering the drifts	*Entering the unaltered Callovo-Oxfordian*	*Exiting the repository sub-zone (i.e., before the repository zone seals)*	*Exiting the repository zone seals*	*Exiting the main connecting drift seals*	*Exit {shaft + EDZ}*
^{14}C	4 %	8 %	0.000032 %	0.00000053 %	nil	nil
^{36}Cl	47 %	32 %	0.033 %	0.011 %	0.005 %	0.004 %
^{129}I	59 %	41 %	0.056 %	0.022 %	0.012 %	0.011 %
^{79}Se	0.12 %	0.17 %	0.000042 %	0.00001 %	0.0000019 %	0.0000016 %

Table 7.2-11 'Seal failure' altered evolution scenario – Distribution and quantification of the various transfer pathways – All seals defective – CU1 reference package

	Cell		Repository zone Efficiency of repository zone seals		Access structures Efficiency of shaft seals	
	Entering the drifts	*Entering the unaltered Callovo-Oxfordian*	*Exiting the repository sub-zone (i.e., **before** the repository zone seals)*	*Exiting the repository zone seals*	*Exiting the main connecting drift seals*	*Exit {shaft + EDZ}*
^{14}C	4 %	8 %	0.000006 %	0.00000003 %	nil	nil
^{36}Cl	46 %	32 %	0.0092 %	0.0011 %	0.00019 %	0.00017 %
^{129}I	59 %	41 %	0.015 %	0.0024 %	0.00061 %	0.00055 %
^{79}Se	0.12 %	0.17 %	0.000013 %	0.0000009 %	0.000000039 %	0.000000028 %

Table 7.2-12 Seal failure' altered evolution scenario – Distribution and quantification of the various transfer pathways – Shaft abandonment – CU1 reference package

	Cell		Repository zone Efficiency of repository zone seals		Access structures Efficiency of shaft seals	
	Entering the drifts	*Entering the unaltered Callovo-Oxfordian*	*Exiting the repository sub-zone (i.e., **before** the repository zone seals)*	*Exiting the repository zone seals*	*Exiting the main connecting drift seals*	*Exit {shaft + EDZ}*
^{14}C	4 %	8 %	0.000003 %	nil	nil	nil
^{36}Cl	46 %	32 %	0.0045 %	0.0002 %	0.00001 %	0.00001 %
^{129}I	59 %	41 %	0.007 %	0.0005 %	0.00005 %	0.00003 %
^{79}Se	0.12 %	0.17 %	0.000006 %	0.00000014 %	0.000000001 %	nil

Table 7.2-13 'Seal failure' altered evolution scenario – For reference : Distribution and quantification of the various transfer pathways – All seals efficient (normal evolution scenario) – CU1 reference package

The following is observed in the repository sub-zones :

- The very slow advective kinetics in the drifts (approximately 3.10^{-3} m/s max. in secondary connecting drifts for 'all seals defective' situation) favour exchanges with the geological barrier. As a result, most of the mass exiting the sub-zone comes from the repository module nearest to the main drifts ;
- The molar flow rate exiting the repository zone is strongly attenuated for all radionuclides. In the most pessimistic situation, only 0.06 % of the iodine 129 inventory and 0.03 % of the chlorine 36 inventory exit the repository sub-zone through the shafts after one million years.

For radionuclides with a non-negligible flow rate at the shaft exit, the following is observed :

- Iodine 129 and chlorine 36 have the highest fraction of activity reaching the roof of he Callovo-Oxfordian layer. The fraction of iodine 129 activity reaching the shaft is more important than for ^{36}Cl because the latter benefits from radioactive decay. The mass percentage of iodine 129 reaching the shaft (as compared to the mass initially present in the repository zone) varies from 0.011 % when all seals are defective to 0.0003 % when only the drift seals are defective (see Figure 7.2-15). In the normal evolution scenario, this percentage is 10 times lower (0.00003 %). The efficiency of the hydraulic cutoffs decreases the quantity of iodine 129 exiting the shaft by one order of magnitude over 1 million years.

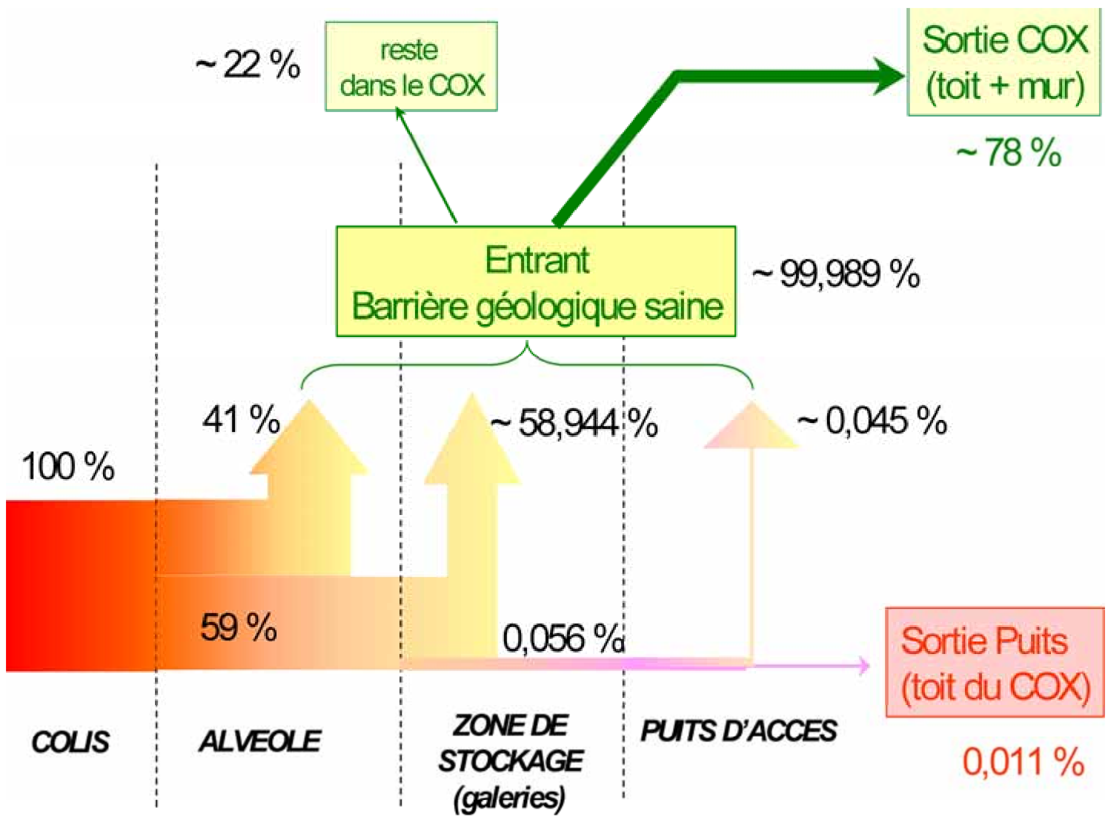


Figure 7.2-15 'Seal failure' altered evolution scenario – Distribution of transfer pathways integrated to one million years – All seals defective - ^{129}I- CU1 reference package

- Soluble carbon 14 has a strongly attenuated mass when exiting the repository zone for all situations. On the one hand, its sorption capacity in the host formation favours its migration from the drifts to the host formation. On the other hand, its radioactive decay half-life is sufficiently short (compared to the transfer times in the engineered structures) to allow it to benefit from radioactive decay. The carbon 14 mass is completely attenuated when exiting the main connecting drift seals for all situations.
- Selenium 79 is strongly limited by its solubility, which induces a small contribution at the shaft exit. In addition, its half-life is sufficiently short with respect to the transfer times to allow it to start benefiting from radioactive decay.

Figure 7.2-16 shows the molar flow rates at various locations of the repository, in the case of CU1 reference package and for the four calculation cases. As in the normal evolution scenario, we observe the influence of the defective package, very significant upon release in the disposal cell and decreasing progressively as we move away from the cell.

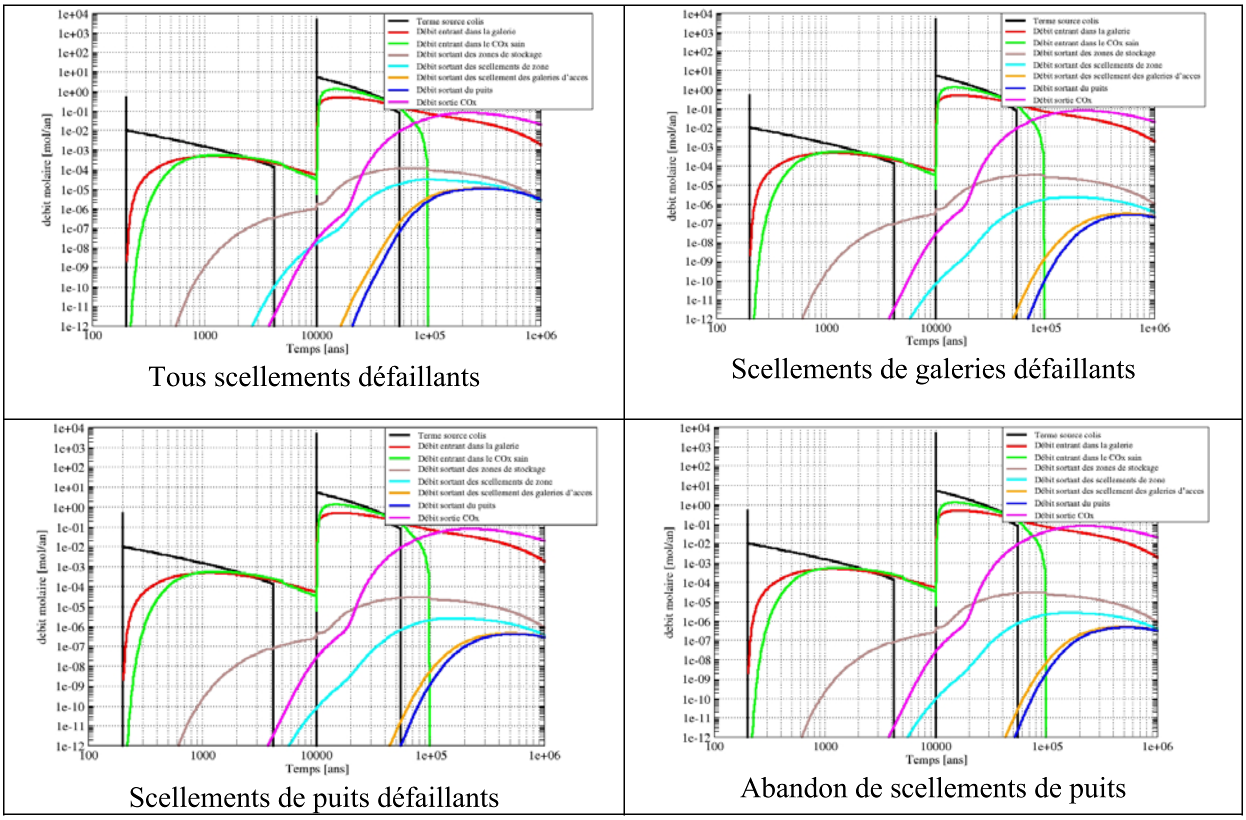


Figure 7.2-16 'Seal failure' altered evolution scenario – All failure situations – Molar flow rate history – Reference calculation - CU1 reference package - ^{129}I

- **Relative role of the various seals**

By comparing the iodine 129 appearance times at the repository zone exit and shaft exit (see Table 7.2-14) as well as the mass attenuations, we can evaluate the benefit associated with each type of seal in terms of transfer times and attenuated mass, i.e. :

- The failure or absence of the shaft seals make the maximum molar flow rates at the shaft exit appear approximately 300 000 years sooner than in the normal evolution scenario, and greater by a factor of 15 to 20. The hydraulic head in the overlying formations that is transmitted to the shaft base produces horizontal gradients that accelerate the advective kinetics. The two situations (absence or failure of shaft seals) are nevertheless relatively equivalent in terms of repository hydraulics and radionuclide migration.
- The loss of drift seals makes the maximum molar flow rates at the shaft exit appear approximately 200 000 years sooner than in the normal scenario, and greater by a factor of 10. This situation clearly shows that the efficiency of the drift seals allows significant attenuation and delay.

These two situations are similar because the shaft and drift seals are redundant with respect to one another. The failure of one type of seal is partially compensated by the efficiency of the other type.

The loss of all seals reduces the interval between the dates of appearance of the maximum molar flow rates at the shaft exit by approximately 500 000 years with respect to the normal evolution scenario, and the corresponding molar flow rates are greater by a factor of approximately 400.

	Appearance times of maximum molar flow rates at the repository zone exit [years]	Appearance times of maximum molar flow rates at the shaft exit [years]
All seals defective	80 000	320 000
Shaft seals defective	83 000	550 000
Abandonment of shaft	83 000	540 000
Drift seals defective	83 000	610 000
All seals efficient	100 000	800 000

Table 7.2-14 Seal failure altered evolution scenario – Appearance times of maximum molar flow rates at the repository zone exit and shaft exit for all failure situations and normal evolution scenario – CU1 spent fuels

It must nevertheless be noted that for all situations the molar flow rate of radionuclides exiting the shaft is smaller by several orders of magnitude (minimum factor of 10 000 for iodine 129 with all seals defective) than the molar flow rate in the host formation. In particular, given the hydraulic performance characteristics of the geological barrier and the associated EDZ, even in case of failure of all repository seals, the geological barrier remains the preponderant transfer pathway for all radionuclides.

7.2.3.2 CU2 spent fuels

The hydraulic characteristics in the case of CU2 spent fuels are identical to those considered for CU1 spent fuels. Comments regarding transport (attenuations, molar flow rates) are qualitatively similar to those applicable to CU1 reference package. Only the results of the impact are different, as discussed in section 7.2.4-2.

7.2.3.3 C waste

The results associated with C waste (see Figure 7.2-17) are qualitatively similar to those of CU1 spent fuels. However, the repository zones associated with spent fuel management scenario S1b (adopted for the analysis of C1/C2 reference package) and consequently the exchange surface with the Callovo-Oxfordian layer are more limited than those associated with CU1 spent fuels (analysed within the scope of management scenario S2), so the quantity of water that the repository receives from the Callovo-Oxfordian layer is also smaller. The head loss generated by the seal failure is then more important for C waste than for spent fuels, but the flow rates are lower.

For the most pessimistic situation (all seals defective), the head loss amounts to approximately 5 metres for C1/C2 reference package and 6 metres for C3/C4 reference package[97]. The water flows exiting the shafts are lower than those evaluated for CU1 spent fuels in the same situation.

[97] This small difference is an artifice associated with the repository configuration according to the inventory considered. Transfers in the C1/C2 waste repository zones are calculated on the basis of waste production scenario S1b of the dimensioning inventory model (scenario with the highest number of transfers). In this waste production scenario, MOX fuels are not reprocessed and a repository zone is specifically dedicated to corresponding spent fuels. The hydraulic influence of this zone contributes to decreasing the head loss. It is absent in the case of C3/C4 waste, which is supposedly stored within the scope of a waste management scenario leading to MOX reprocessing.

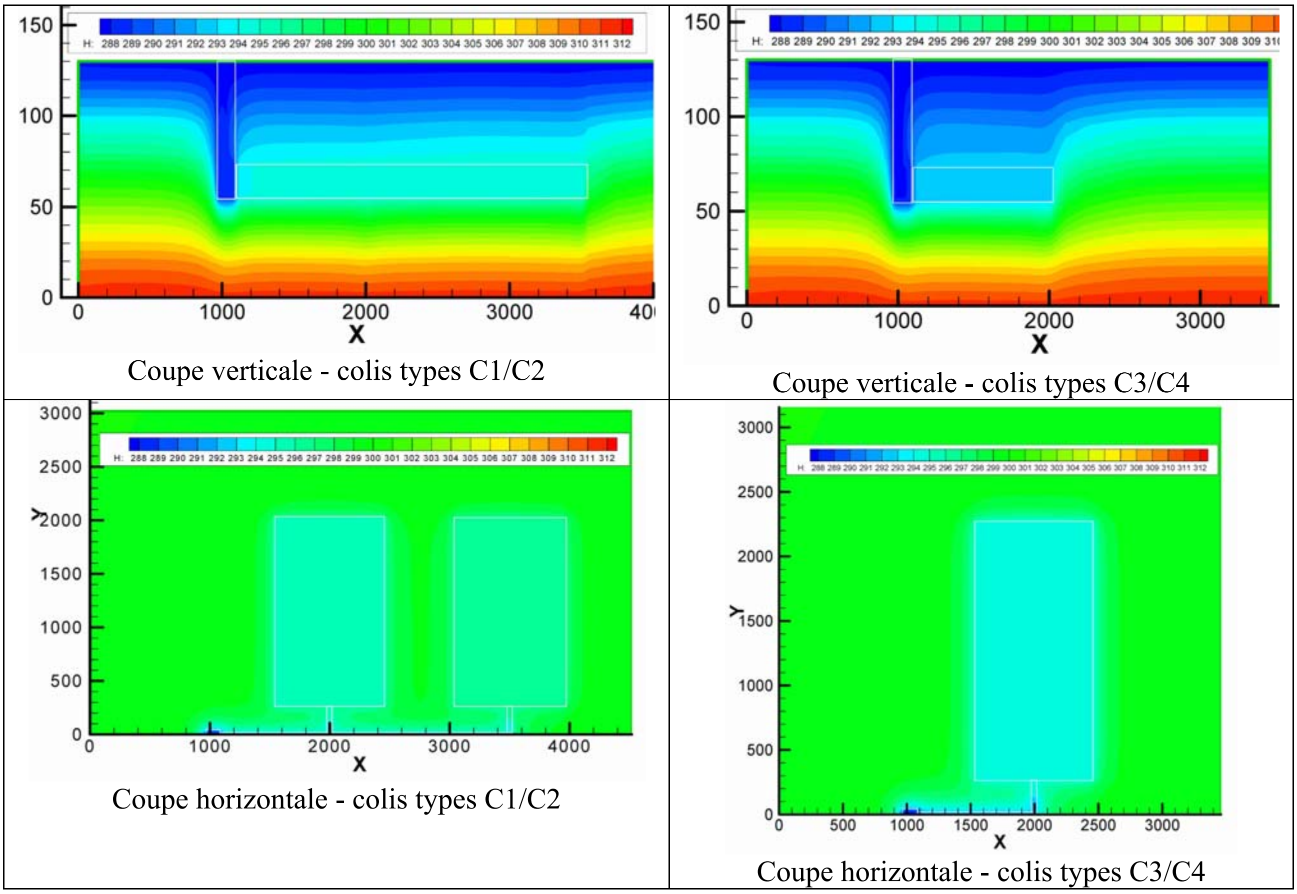


Figure 7.2-17 Seal failure altered evolution scenario – Vertical and horizontal cross-sections through the engineered structures and shaft – C1/C2 and C3/C4 reference package – All seals defective

In case of failure of all seals, the advective transfer times amount to approximately 150 000 years from the repository zone exit to the shafts for C1/C2 and C3/C4 reference package.

Figure 7.2-18 shows the molar flow rate history at various locations of the repository, in the case of C1/C2 reference package, for the 'all seals defective' and 'shaft seals defective' situations. As in the normal evolution scenario, we observe the influence of the defective package, very significant upon release in the disposal cell and decreasing progressively as we move away from the cell.

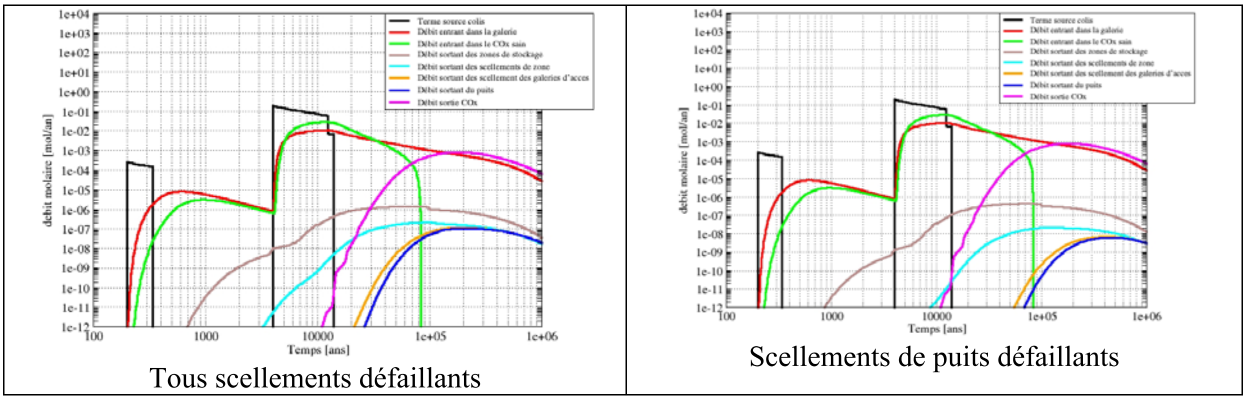


Figure 7.2-18 Seal failure altered evolution scenario – All seals defective' situation – Molar flow rate history – Reference calculation – C1/C2 reference package - ^{129}I

7.2.3.4 Non-organic B waste not releasing hydrogen (disposed of in B1x disposal cells)

It is reminded that B waste has only been covered within the scope of the « all seals defective » bounding scenario.

For B waste, the head loss generated by the failure of all seals is significant (approximately 13 metres). On the other hand, the quantity of water drained by a B waste repository is very low compared to the quantity mobilised for spent fuels. This is explained by the small extent of the corresponding repository zone. Water flows are very slow. As a result, transfer times are very significant up to the access shaft, thereby favouring exchanges with the geological barrier throughout the length of the main connecting drifts (see Figure 7.2-19).

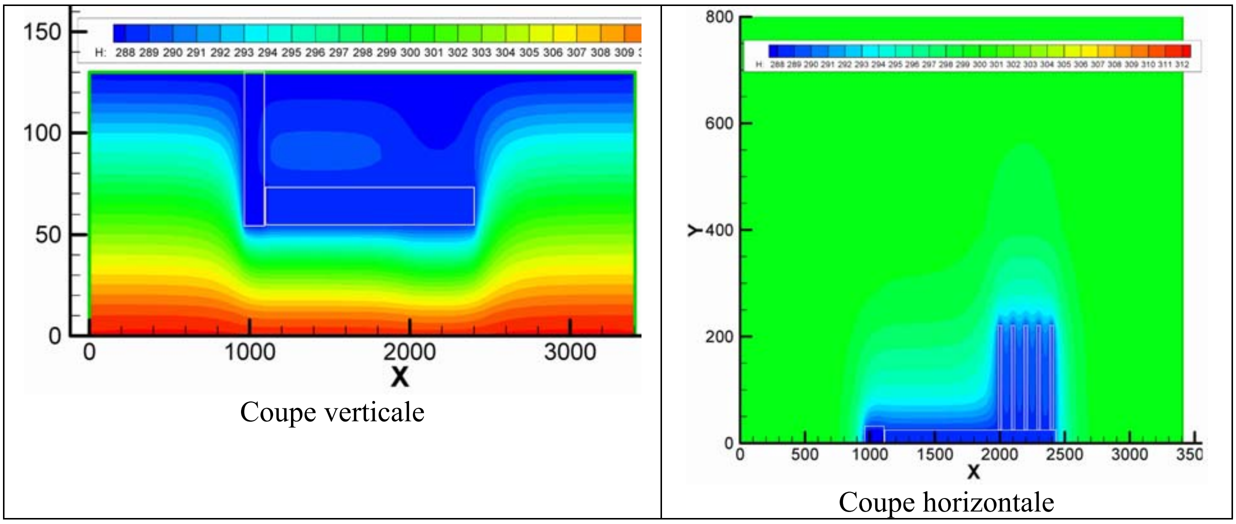


Figure 7.2-19 Seal failure altered evolution scenario – Vertical and horizontal cross-sections through the engineered structures – Piezometry in B waste repository modules containing non-organic packages not releasing hydrogen (B1x disposal cells) – All seals defective

For B waste, each disposal cell corresponds to a repository module, with transfers in the repository zone strictly limited to transfers within the cells, i.e., the mass exiting each cell through the engineered structures travels directly to the main connecting drifts. Figure 7.2-20 confirms the slowness of transfers to the access shafts. The appearance times of the maximum molar flow rates are indeed very remote, i.e., approximately one million years for iodine 129 and 600 000 years for chlorine 36 (which therefore begins to benefit from radioactive decay).

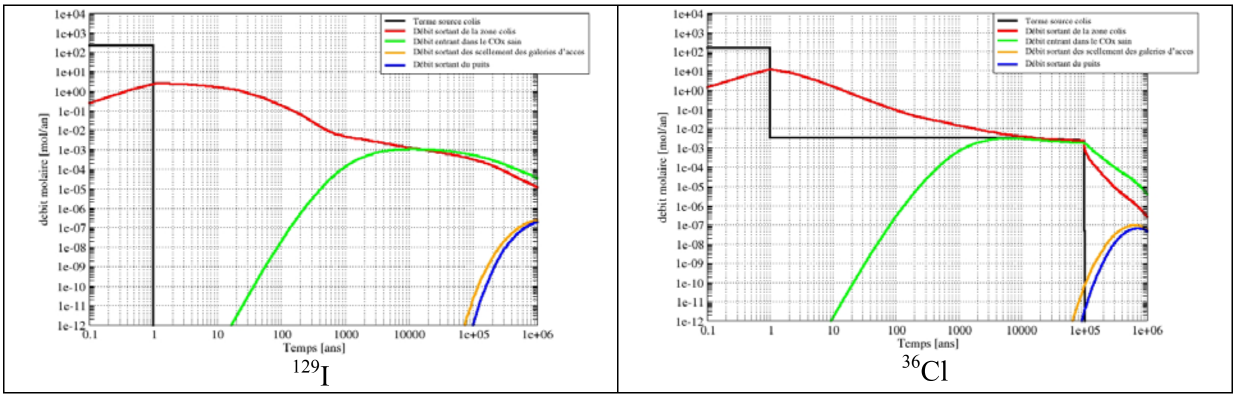


Figure 7.2-20 Seal failure altered evolution scenario – All seals defective - Reference calculation – Molar flow rate history – Non-organic packages not releasing hydrogen - ^{129}I and ^{36}Cl

7.2.4 Impact assessment

For simplification purposes and to ensure proper readability of the results, the impacts discussed below are those associated with the most pessimistic outlet, i.e., the 'Saulx' outlet of the normal evolution scenario. It is reminded that impacts result from two radionuclide transfer pathways :

- Impact due to radionuclides that travel through the access structures and defective seals and then migrated in the surrounding formations.
- Impact due to radionuclides that travel through the Callovo-Oxfordian layer. Since the host formation remains the transfer pathway for most radionuclides in all situations as described above, this impact is the same as in the normal evolution scenario (actually slightly smaller, but this difference is not identifiable in the calculation).

The two impacts occur in the same outlets, but their maximum magnitudes may be attained at different dates. As will be seen, the impact due to the fraction travelling through the access structures is in all cases negligible with respect to the fraction passing through the host formation.

7.2.4.1 CU1 spent fuels

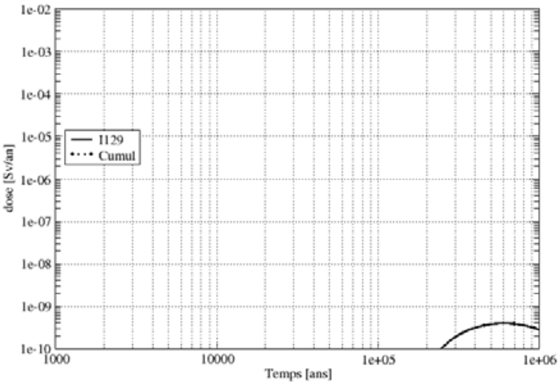
The dose rate at the Saulx outlet associated with the fraction of activity that exits the shaft after travelling through the engineered structures is dominated by iodine 129 in the four situations considered.

The most pessimistic situation corresponds to the failure of all seals, with a maximum dose rate of 10^{-5} mSv/year after 370 000 years (associated with iodine 129). The impact of chlorine 36 is significantly lower, with a dose rate of approximately 10^{-7} mSv/year. All the other radionuclides have a nil or negligible impact.

For the three other situations, the maximum dose rate is extremely low, comprised between 10^{-7} and 10^{-6} mSv/year after approximately 600 000 years.

For the most pessimistic situation (all seals defective), the dose rate due to the transfer pathway through the engineered structures remains significantly lower than that due to the transfer pathway for the iodine 129 contained in CU1 spent fuels (0.00001 mSv/year versus 0.02 mSv/year), with identical maximum value dates. The dose rates associated with the fraction of activity that migrates through the engineered structures and through the geological barrier (identical for all situations and equal to those considered in the normal evolution scenario) are indicated below (see Figure 7.2-21 to Figure 7.2-25).

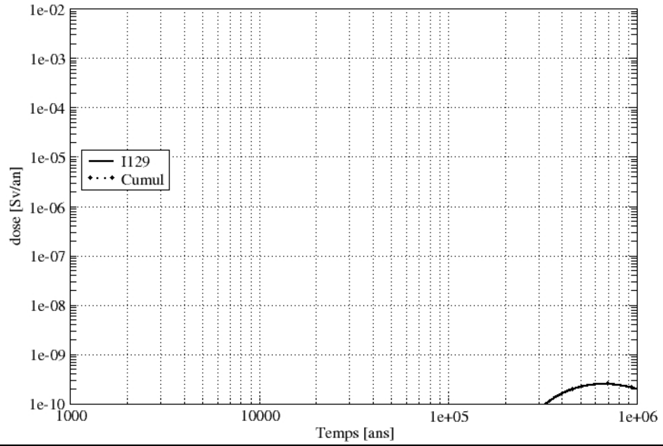
- **Impact assessment for the fraction of activity exiting the shafts : Shaft seals defective**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	3.9E-10	6.0E+05
I129	3.9E-10	6.1E+05

Figure 7.2-21 Seal failure altered evolution scenario – Shaft seals defective – Reference calculation – Dose rate history for the transfer pathway through the engineered structures – CU1 reference package

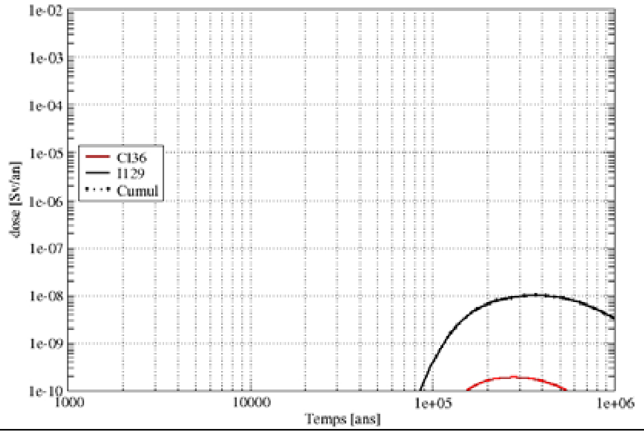
- **Impact assessment for the fraction of activity exiting the shafts : Drift seals defective**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	2.5E-10	6.7E+05
I129	2.5E-10	6.7E+05

Figure 7.2-22 Seal failure altered evolution scenario – Drift seals defective – Reference calculation – Dose rate history for the transfer pathway through the engineered structures – CU1 reference package

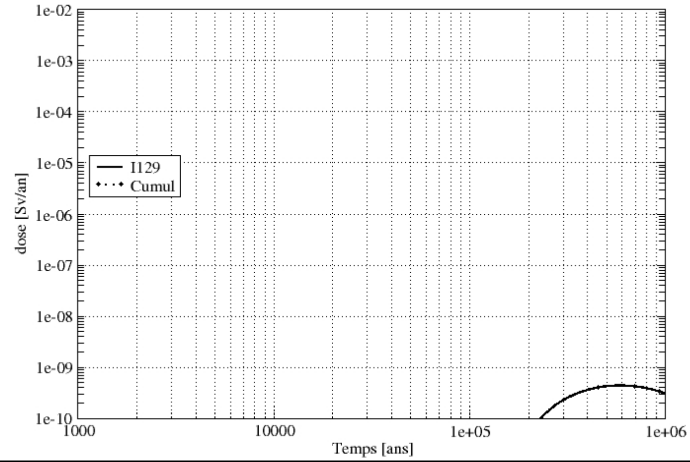
- **Impact assessment for the fraction of activity exiting the shafts : All seals defective**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	9.9E-09	3.7E+05
I129	9.7E-09	3.7E+05
Cl36	1.9E-10	2.8E+05

Figure 7.2-23 Seal failure altered evolution scenario – All seals defective – Reference calculation – Dose rate history for the transfer pathway through the engineered structures – CU1 reference package

- **Impact assessment for the fraction of activity exiting the shafts : Abandonment of shaft seal**

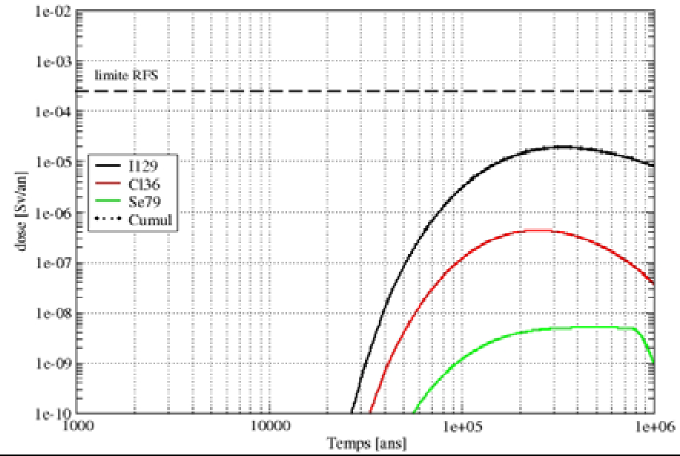


RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	4.4E-10	5.9E+05
I129	4.3E-10	5.9E+05

Figure 7.2-24 'Seal failure altered evolution scenario – Abandonment of shaft seal – Reference calculation – Dose rate history for the transfer pathway through the engineered structures – CU1 reference package

- **Impact assessment for the transfer pathway through the geological barrier : All calculation cases**

Since the fraction of radionuclides travelling through the engineered structures is extremely small regardless of the calculation case considered, the impact associated with the transfer pathway through the geological barrier is the same as in the normal evolution scenario. It is recalled here for reference. Since this impact is greater than that due to the transfer pathway through the engineered structures, the maximum dose associated with the 'seal failure' altered evolution scenario is actually the same as that associated with the normal evolution scenario.



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.9E-05	3.3E+05
I129	1.8E-05	3.4E+05
Cl36	4.3E-07	2.5E+05
Se79	5.0E-09	4.7E+05

Figure 7.2-25 Seal failure altered evolution scenario – All situations – Reference calculation – Dose rate history for the transfer pathway through the host formation – CU1 reference package

7.2.4.2 CU2 spent fuels

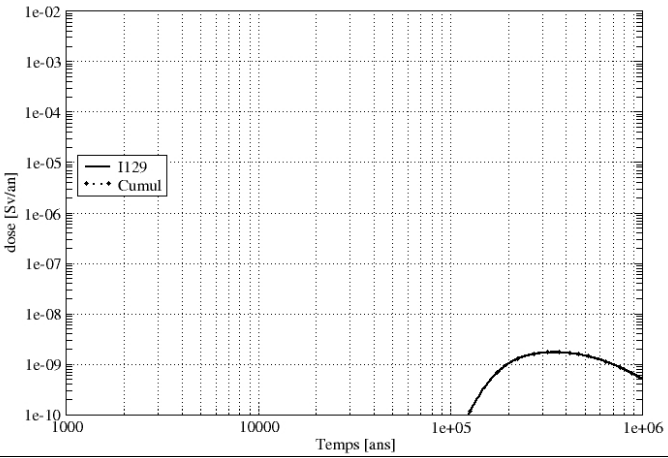
- **Impact assessment for the fraction of activity exiting the shafts**

The main contributor to this impact is also iodine 129. Nevertheless, the impact associated with CU2 spent fuels is significantly lower than for CU1 spent fuels due to the more limited iodine inventory (by a little over one order of magnitude).

For three of the four situations analysed, the dose rate at the Saulx outlet associated with the fraction of activity that exists the shafts after travelling through the engineered structures is less than or equal to 10^{-7} mSv/year.

For the 'all seals defective' situation, the dose rate due to the transfer pathway through the engineered structures remains significantly lower than that due to the transfer pathway through the unaltered geological barrier ($1.7.10^{-6}$ mSv/year versus 0.0017 mSv/year for iodine 129), with identical maximum value dates (approximately 350 000 years).

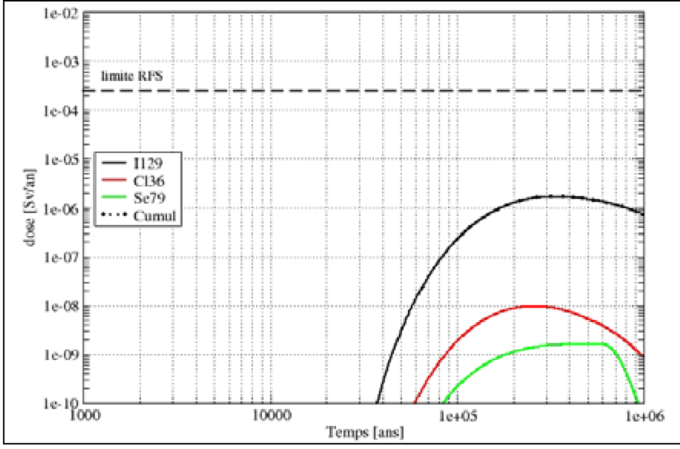
The dose rates associated with the fraction of activity that migrates through the engineered structures and through the geological barrier (identical for all situations and equal to those considered in the normal evolution scenario) are indicated below (see Figure 7.2-26 and Figure 7.2-27), with an impact exceeding 10^{-7} mSv/year (i.e., 10^{-10} Sv/year).



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.7E-09	3.5E+05
I129	1.7E-09	3.5E+05

Figure 7.2-26 Seal failure altered evolution scenario – All seals defective – Reference calculation – Dose rate history for the transfer pathway through the engineered structures – CU2 reference package

- **Impact assessment for the transfer pathway through the geological barrier : All calculation cases (for reference, identical to normal evolution scenario)**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.7E-06	3.4E+05
I129	1.7E-06	3.4E+05
Cl36	9.6E-09	2.6E+05
Se79	1.6E-09	5.6E+05

Figure 7.2-27 Seal failure altered evolution scenario – All situations – Reference calculation – Dose rate history for the transfer pathway through the host formation – CU2 reference package

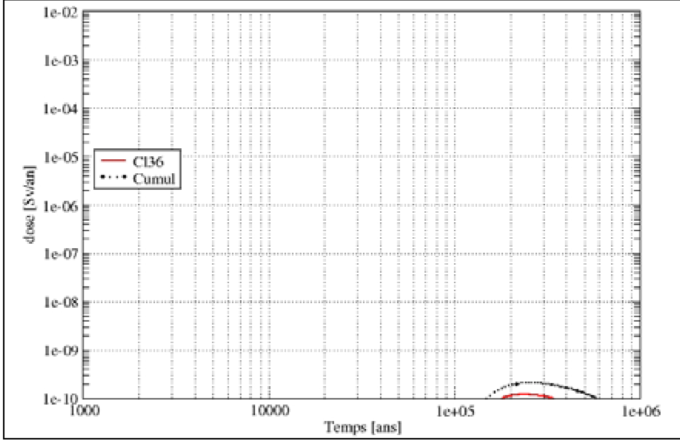
7.2.4.3 C1/C2 reference package

- **Impact assessment for the fraction of activity exiting the shafts.**

For C1/C2 reference package, only the 'all seals defective' and 'shaft seals defective' situations are considered, with the latter producing dose rates of less than 10^{-7} mSv/year for the fraction of activity that migrates through the engineered structures. The two main contributors to are chlorine 36 and iodine 129, which also contribute for glasses in a nearly identical manner (with the chlorine presenting a slightly higher impact).

As in the case of the spent fuels, the dose rate due to the transfer pathway through the engineered structures remains significantly lower than that due to the transfer pathway through the unaltered geological barrier, with a maximum dose rate 2000 times lower for transfers through the engineered structures than for those via the geological barrier ($2.1.10^{-7}$ mSv/year versus 0.00047 mSv/year). The dose rates associated with transfers via the engineered structures in the 'all seals defective' situation are indicated in Figure 7.2-28. The dose rates associated with the rest of the activity are indicated in Figure 7.2-29.

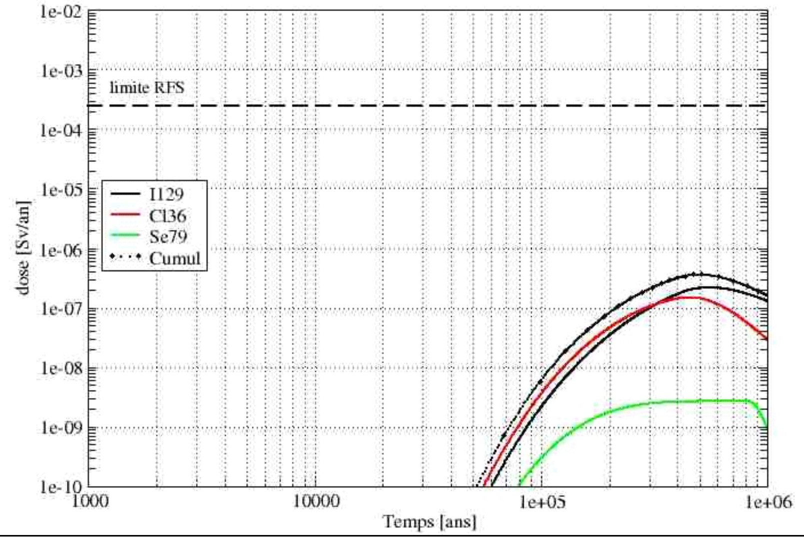
- **All seals defective**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	2.1E-10	2.5E+05
Cl36	1.2E-10	2.4E+05

Figure 7.2-28 Seal failure altered evolution scenario – All seals defective – Reference calculation – Dose rate history for the transfer pathway through the engineered structures – C1/C2 reference package

- **Impact assessment for the transfer pathway through the geological barrier : All calculation cases**

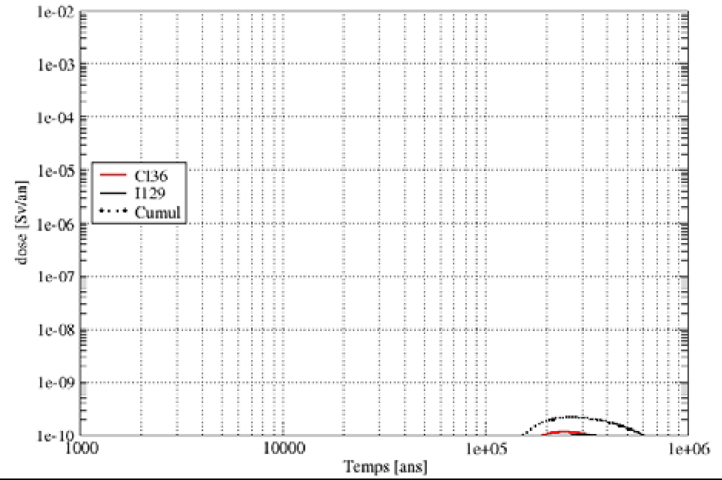


RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	4.7E-07	4.9E+05
I129	2.6E-07	5.5E+05
Cl36	2.1E-07	4.5E+05
Se79	3.2E-09	8.0E+05

Figure 7.2-29 Seal failure altered evolution scenario – All situations – Reference calculation – Dose rate history for the transfer pathway through the host formation – C1/C2 reference package

7.2.4.4 C3/C4 reference package

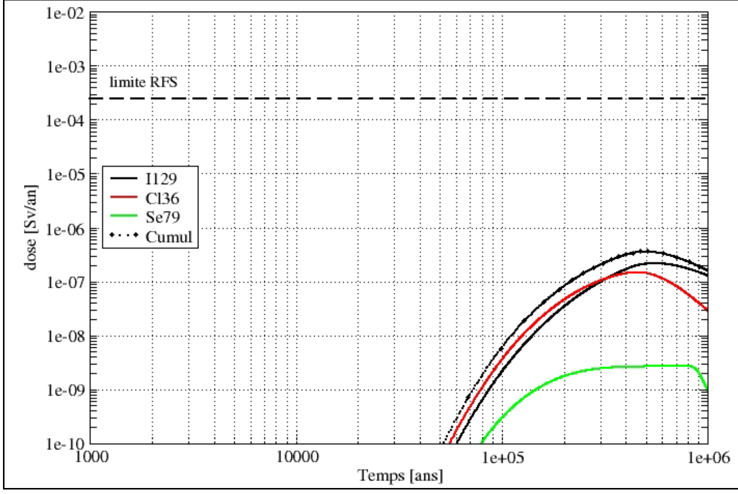
The conclusions are similar to those for the C1/C2 reference package, both qualitatively and quantitatively. The dose rates associated with transfers via the engineered structures for the fraction of activity that migrates through said structures in the 'all seals defective' situation are indicated in Figure 7.2.30. Doses associated with the remaining of the activity are indicated in figure 7.2.31.



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	2.2E-10	2.6E+05
Cl36	1.2E-10	2.4E+05
I129	1.0E-10	3.0E+05

Figure 7.2-30 Seal failure altered evolution scenario - all situations – reference calculation – history of transfer pathway doses through the structures – C3/C4 reference package

- **All situations – impact associated with the transfer pathway of the geological barrier (normal evolution)**



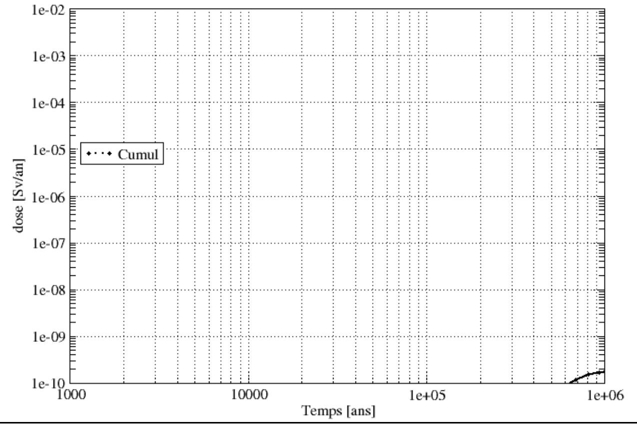
RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	3.6E-07	5.0E+05
I129	2.2E-07	5.6E+05
Cl36	1.5E-07	4.6E+05
Se79	2.7E-09	7.4E+05

Figure 7.2-31 Seal failure altered evolution scenario - all situations – reference calculation – history of transfer pathway doses through the formation host – C3/C4 reference package

7.2.4.5 Non-organic B reference package

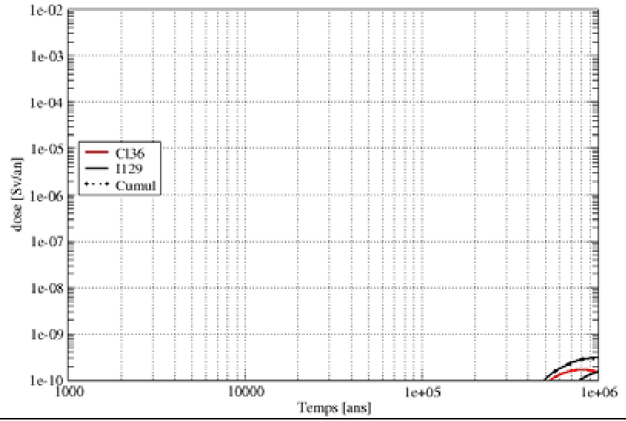
For these types of reference package (not degassing hydrogen), only the 'all seals defective' and 'shaft seals defective' situations are considered. These two situations generate extremely low and delayed doses, with a maximum cumulated dose rate in the order of 10^{-7} mSv/year after approximately one million years and iodine 129 and chlorine 36 as the main contributors.

As with the other reference package, the dose rate due to the transfer pathway through the engineered structures remains significantly lower than that due to the transfer pathway through the unaltered geological barrier (see Figure 7.2-32 to Figure 7.2-34).



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.7E-10	1.0E+06
I129	dose < 1e-10	sans objet
Cl36	dose < 1e-10	sans objet

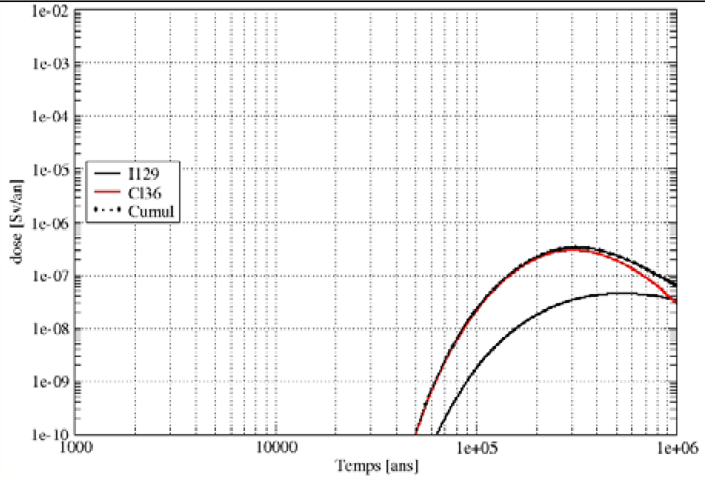
Figure 7.2-32 Seal failure altered evolution scenario – Shaft seals defective – Reference calculation – Dose rate history for the transfer pathway through the engineered structures – B1x reference package



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	3.0E-10	1.0E+06
Cl36	1.7E-10	8.1E+05
I129	1.5E-10	1.0E+06

Figure 7.2-33 Seal failure' altered evolution scenario – All seals defective – Reference calculation – Dose rate history for the transfer pathway through the engineered structures – B1x reference package

- **All situations – impact associated with the transfer through the geological barrier (normal evolution)**

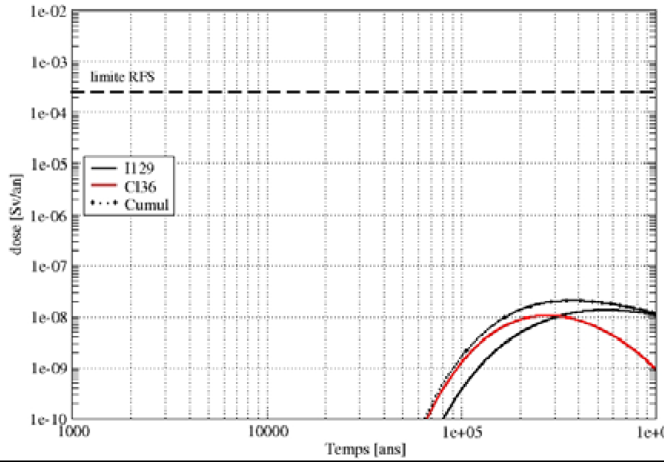


RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	3.3E-07	3.1E+05
Cl36	2.9E-07	3.1E+05
I129	4.4E-08	5.3E+05
Se79	dose < 1e-10	sans objet

Figure 7.2-34 Seal failure' altered evolution scenario – All situations – Reference calculation – Dose rate history for the transfer pathway through the host formation – B1x reference package

7.2.4.6 Bituminised sludge reference package (B2)

For bituminised sludge reference package, only the « all seals failed » and « shaft seals failed » situations have been studied (see Figure 7.2-35). These two situations lead to doses lower to 10^{-7} mSv/yr through the structures. The only impact in this situation is that of transfer via the geological barrier, equivalent to that chosen for the normal evolution scenario



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	2.1E-08	3.7E+05
I129	1.3E-08	5.5E+05
Cl36	1.0E-08	2.8E+05
Se79	dose < 1e-10	sans objet

Figure 7.2-35 Seal failure altered evolution scenario - All situations - Reference calculation - dose histories of the transfer pathway via the host formation - B2 reference package

7.2.5 Sensitivity analyses

Sensitivity analyses have been performed in such a manner as to take into account any potential aggravating effects for the performances of safety functions. Table 7.2-6 summarises the various sensitivity studies performed for each of the reference package, along with the different situations studied.

7.2.5.1 EDZ and seal core performance sensitivity

The purpose of this study is to assess the consequences, in terms of transfer pathways and impacts, associated with possible hydraulic parameter, transfer and EDZ retention-related uncertainties in seal failure situations, along with seal core performance. This study consisted in considering the following :

- a « degraded » EDZ, of which value and parameter details are provided in Table 7.2-4, with :
  - ✓ a « pessimistic » fractured zone, represented by a high degree of permeability (10^{-6} m/s), a diffusion coefficient equivalent to that of water particles in water and a lack of geochemical retention capability,
  - ✓ a microfissured zone with « conservative » hydraulic, transfer and retention parameters,
- a bentonite core permeability of 10^{-9} m/s instead of 10^{-11} m/s (« phenomenological » value). This assignment of permeability concerns all seal cores, including cell plugs.

- **« All seals failed » situation - CU1 spent fuel**

The head loss induced by the failure of all seals in the repository is of approximately 12 metres, as illustrated in Figure 7.2-36 (i.e. a head of 288 metres NGF in the repository, instead of 300 metres NGF[74] in the normal evolution scenario). This significant head loss is the result of the high hydraulic conductivity of the fractured zone (continuous at 10^{-6} m/s from the shafts to the repository), which transmits the head from the top of the Callovo-Oxfordian to the repository.

In the repository zone, advection is dominant, with Peclet numbers of approximately 200 at secondary connecting drifts exits, compared to approximately 10 under normal evolution. Consequently, the advective transfer times are much shorter : the average advective transfer time between the CU1 spent fuel repository sub-zone exit and the shaft is of approximately 60,000 years (i.e. some 3 times less than for a « degraded » EDZ situation in a normal evolution scenario). The high permeability of the continuous fractured zone along the structures constitutes an hydraulic sink and the flow rate of water leaving the shafts is significantly higher than for the same reference situation : 7.5 m^3/year compared to 2.5 m^3/year with a « phenomenological » EDZ.

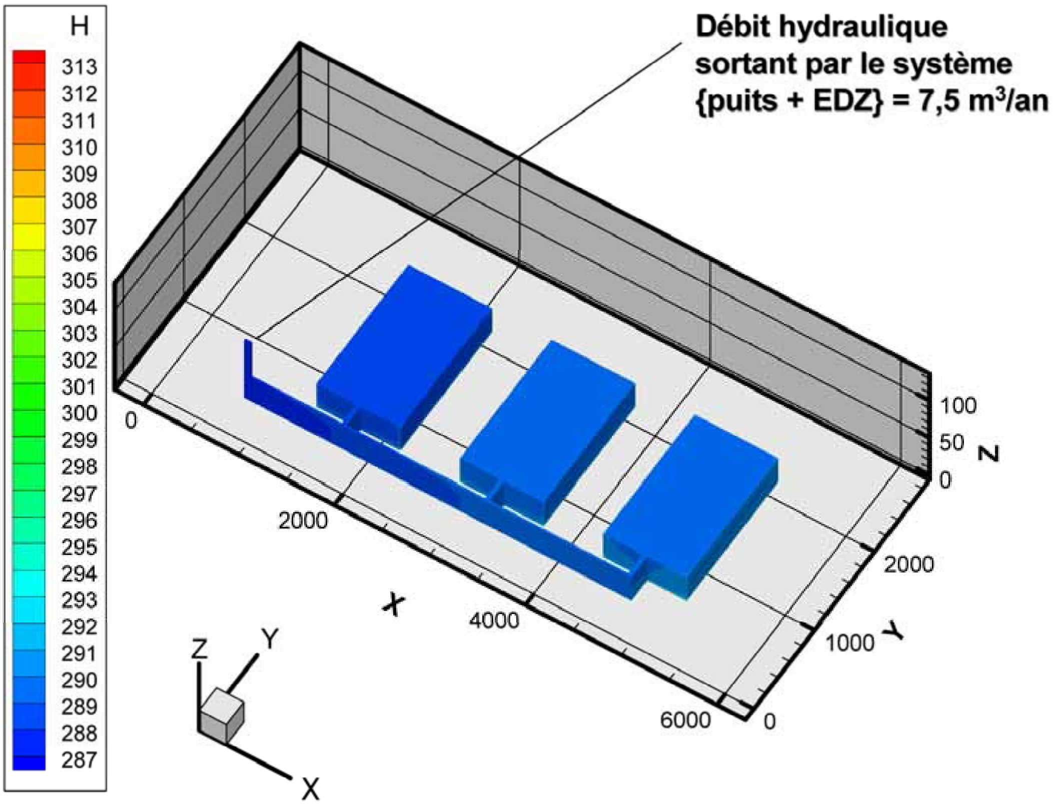


Figure 7.2-36 Seal failure altered evolution scenario - All seals failed - EDZ and bentonite core permeability sensitivity - CU1 reference package - hydraulic flow leaving the access structures

In the cell's near-field, the integration of a degraded EDZ causes an increase in the quantity of radionuclides entering the drift of approximately 10 % for chlorine 36 and iodine 129 (see Table 7.2-14 and Table 7.2-15). This relatively low increase is mainly due to the modification of the diffusion coefficient in the EDZ, transfer remaining dominantly diffusive (co-dominance at cell head in the fractured zone) in and around the cell. The hydraulic disturbance around the cell, even when associated with a degraded EDZ, is not sufficient to induce a dominant advective regime. For the other radionuclides and in particular the actinides for which a dedicated calculation has been done, results did not significantly change. The « absorbing barrier » role of the Callovo-Oxfordian would enable the actinides to sorb into the formation even if they arrived to to reach the drifts owing to a degraded EDZ.

	Cell		Repository zone Zone seal efficiency		Access structures Shaft seal efficiency	
	Entering the shafts	*Entering the unaltered Callovo-Oxfordian*	*Exiting the repository sub-zone, ie. « before zone seals »*	*Exiting the zone seals*	*Exiting the main connecting drift seals*	*Exit {shaft + EDZ}*
^{36}Cl	55 %	32 %	0.246 %	0.171 %	0.119 %	0.115 %
^{129}I	67 %	33 %	0.372 %	0.281 %	0.227 %	0.223 %
^{79}Se	0.39 %	0.38 %	0.00096 %	0.00057 %	0.00026 %	0.00024 %

Table 7.2-15 Seal failure altered evolution scenario - Distribution and quantification of the different transfer paths – sensitivity : all seals failed, degraded EDZ and core permeability equal to 10^{-9} m/s - CU1 reference package

In the repository zone, transfer is distinctly advective due to the hydraulic contribution of the different disposal cells and drifts. Transfers are thus faster and the exchanges between the repository drifts and the unaltered geological barrier less pronounced than in the « phenomenological » EDZ calculation case. The mass of iodine 129, leaving the repository zones via the shaft over one million years, is 6 times higher. It should furthermore be noted that this mass represents only 0.4 % of the initial inventory, which is still low. The maximum molar flow values appear around 70,000 years for radionuclides with little or no decay (^{129}I, ^{36}Cl).

In the main connecting drifts, transfer is governed by upstream hydraulics, coming from the repository zones. The maximum molar flow at the shaft exit is offset by approximately 60,000 years (i.e. a little after 100,000 years) compared to the repository zone exit, with limited transfer from the main connecting drifts to the geological barrier. Thus, only 0.2 % of the initial iodine 129 inventory has left the failed shafts over one million years. This fraction is even lower for chlorine 36 (0.1 %), which benefits from radioactive decay. This situation is summarised in Figure 7.2-37.

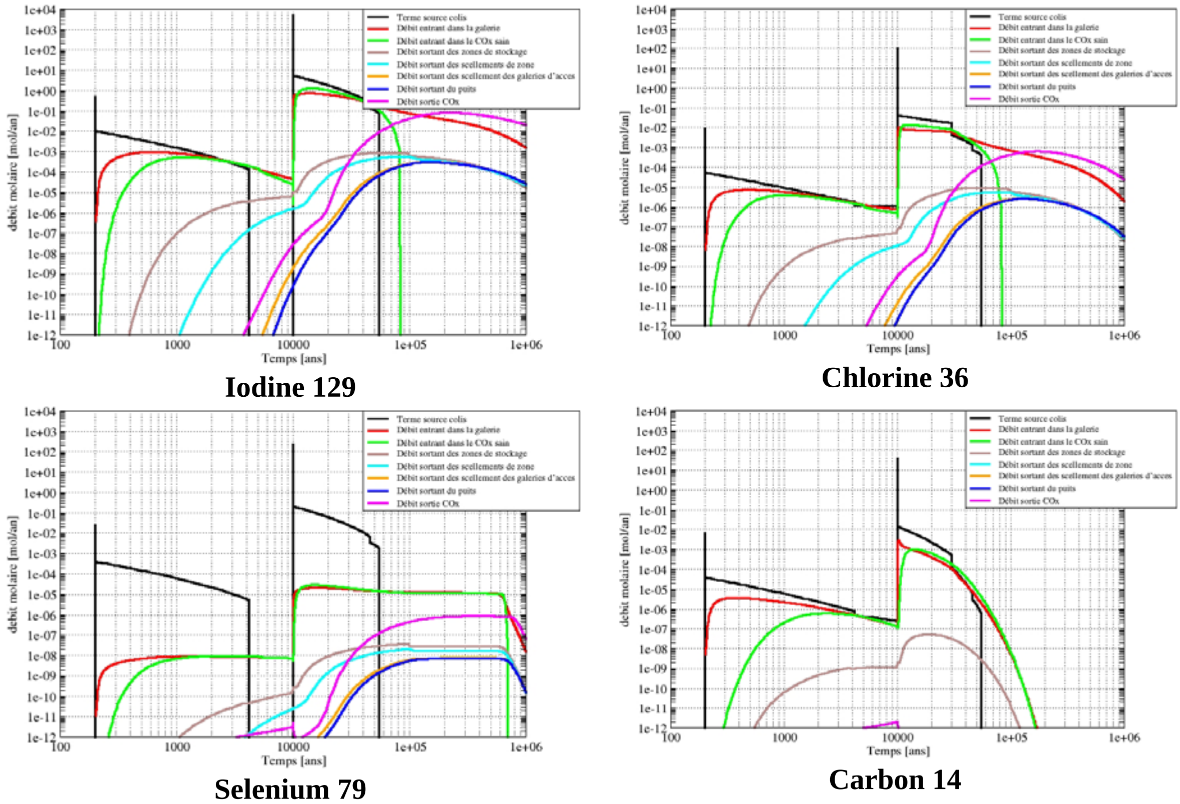
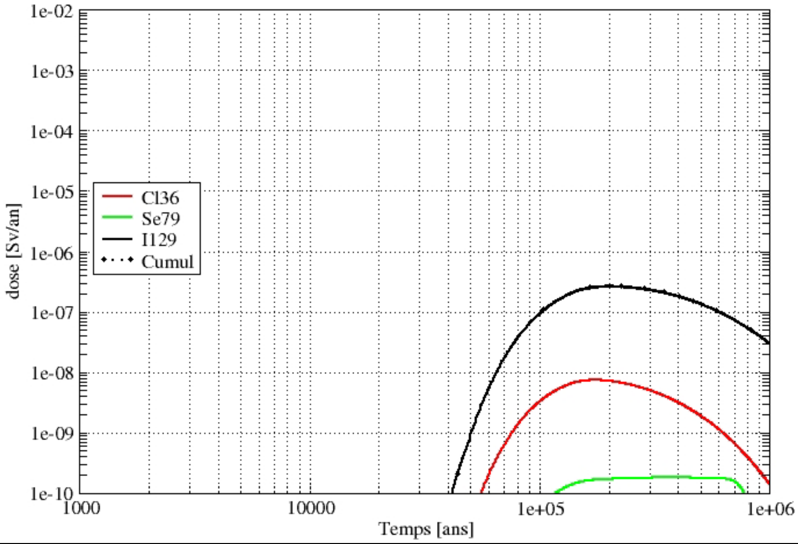


Figure 7.2-37 Seal failure altered evolution scenario - molar flow history - All seals failed : EDZ and seal core performance sensitivity : CU1 waste reference package - ^{129}I, ^{36}Cl, ^{79}Se, ^{14}C

The high degraded EDZ section around the drifts and bentonite core is sufficient to short-circuit the seals : their failure, in this case, has only limited impact. The results of this study are very similar to the sensitivity study performed for normal evolution with a degraded EDZ : the hydraulic flow leaving the shaft in the « all seals failed » study is of 7.5 m^3/year, compared to 6 m^3/year in the « all seals effective, but degraded EDZ » situation.

The dose associated with the mass passing through the structures and leaving via the shaft is dominated by iodine 129. The peak dose for this pathway is of around 0.0003 mSv/year at approximately 200,000 years (see Figure 7.2-38).

The dose associated with the mass passing through the geological barrier is identical to that of the normal evolution scenario (the fraction of the mass passing through the structures is negligible). The maximum dose is of around 0.02 mSv/year at approximately 300.000 years, that is to say a little less than 2 orders of magnitude more than the fraction associated with the activity fraction migrating through the structures.



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	2.7E-07	2.0E+05
I129	2.6E-07	2.0E+05
Cl36	7.4E-09	1.7E+05
Se79	1.8E-10	3.8E+05

Figure 7.2-38 Seal failure altered evolution scenario - All seals failed - EDZ and seal core performance sensitivity - dose histories of the transfer pathway through the structures - CU1 reference package

- **« Shaft seals failed » situation - CU1 spent fuel**

In a shaft seal failure situation, the results are very similar to the « all seals failed » configuration since the permeability of the fractured zone significantly reduces the efficiency of hydraulic cutoffs and seal cores. The water flow leaving the repository is of 7.1 m^3/year.

The « resist to water circulation » function appears to be sensitive to the parameters adopted for the EDZ, in particular the fractured zone. In a « degraded EDZ » case, the seal core and cutoffs, as represented (anchored only in the fractured zone) do not provide any significant gain. This drift seal short-circuiting by the fractured zone is partially related to modelling choices, and more specifically to the short hydraulic cutoffs taken into account. With cutoff lengths closer to those that engineering allows us to consider, these conclusions would be moderated.

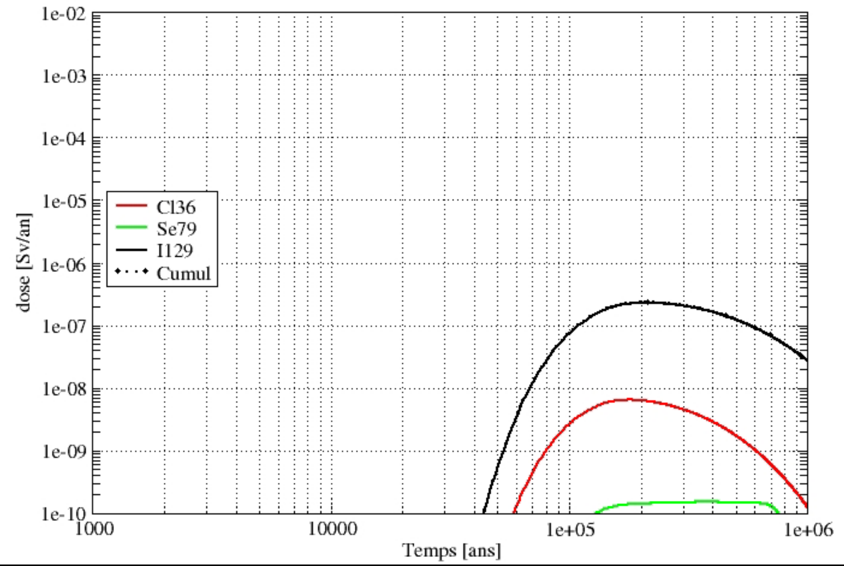
Under these circumstances, the efficiency, or lack thereof, of the seal cores and cutoff have little influence once the EDZ around the drifts is severely degraded. Thus, transport results are similar to those of the « all seals failed » situation : approximately 0.2 % of the mass of iodine 129 initially present in the spent fuel repository leaves through the shafts after having passed through the structures, as detailed in Table 7.2-16.

	Cell		Repository zone Zone seal efficiency		Access structures Shaft seal efficiency	
	Entering the shafts	*Entering the unaltered Callovo-Oxfordian*	*Exiting the repository sub-zone, ie. « before zone seals »*	*Exiting the zone seals*	*Exiting the main connecting drift seals*	*Exit {shaft + EDZ}*
^{36}Cl	55 %	32 %	0.228 %	0.155 %	0.106 %	0.102 %
^{129}I	67 %	41 %	0.346 %	0.256 %	0.204 %	0.200 %
^{79}Se	0.39 %	0.38 %	0.00087 %	0.00049 %	0.00021 %	0.00020 %

Table 7.2-16 Seal failure altered evolution scenario - Distribution and quantification of the different transfer pathways – sensitivity : shaft seals failed, degraded EDZ and core permeability equal to 10^{-9} m/s - CU1 reference package

The dose associated with the mass passing through the structures and leaving via the shaft is dominated by iodine 129. The maximum dose is of approximately 0.00023 mSv/year at around 210,000 years (see Figure 7.2-39), that is to say of the same order as that obtained in the « degraded EDZ - all seals failed » configuration.

The dose associated with the pass passing through the geological barrier is identical to that of the normal evolution scenario.



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	2.3E-07	2.1E+05
I129	2.3E-07	2.1E+05
Cl36	6.5E-09	1.8E+05
Se79	1.5E-10	3.8E+05

Figure 7.2-39 Seal failure altered evolution scenario - Shaft seals failed - EDZ and seal core performance sensitivity - dose histories of the transfer pathway through the structures - CU1 reference package

- **« All seals failed » and « shaft seals failed » situation - CU2 spent fuel**

These situations are strictly identical to those of CU1 spent fuel, also considered for the S2 spent fuel management scenario in terms of hydraulics. From a transport point of view, the results are similar.

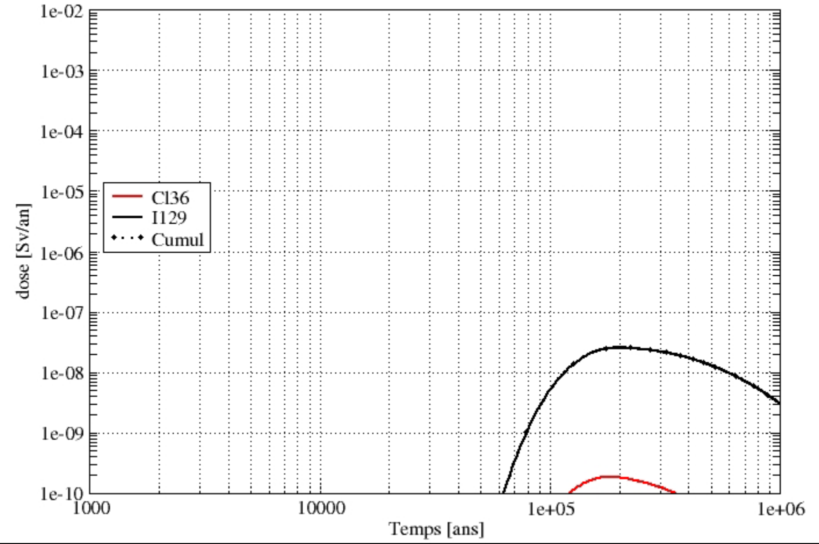
This section presents only those doses pertaining to the CU2 reference package, for the different situations studied.

For these two situations (see Figure 7.2-40 and Figure 7.2-41) and as for the CU1 spent fuel, the dose is dominated by iodine 129.

For the « all seals failed » situation, the maximum dose generated by the radionuclides passing through the structures is of 0.00003 mSv/year at 200,000 years.

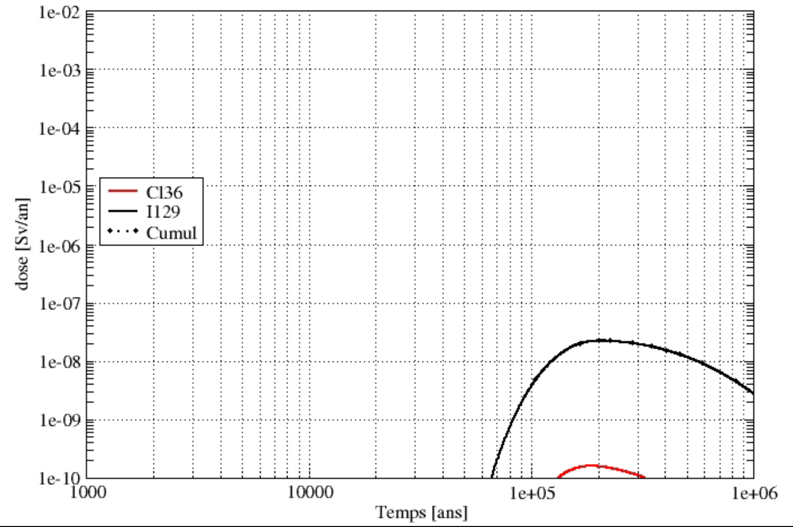
For the « shaft seals failed » situation, this same maximum dose is of 0.00002 mSv/year at 210,000 years. As for the CU1 spent fuel, the severely degraded EDZ masks the efficiency of the drift seals and the doses are similar to the « all seals failed » configuration.

The dose associated with the pass passing through the geological barrier is identical to that of the normal evolution scenario (the fraction of the mass passing through the structures is negligible). The maximum dose is of approximately 0.0017 mSv/year at around 300,000 years, that is to say 100 times more than the activity fraction that migrates through the structures.



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	2.6E-08	2.0E+05
I129	2.5E-08	2.0E+05
Cl36	1.8E-10	1.8E+05

Figure 7.2-40 Seal failure altered evolution scenario - All seals failed - EDZ and seal core performance sensitivity - dose histories of the transfer pathway through the structures - CU2 reference package



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	2.2E-08	2.1E+05
I129	2.2E-08	2.1E+05
Cl36	1.6E-10	1.9E+05

Figure 7.2-41 Seal failure altered evolution scenario - Shaft seals failed - EDZ and seal core performance sensitivity - dose histories of the transfer pathway through the structures - CU2 reference package

7.2.5.2 Seal core geochemistry sensitivity

The purpose of this study is to assess the consequences, in terms of transfer pathways and impact, associated with variations in seal core geochemistry that may result from chemical disturbances. These latter are considered with conservative chemical retention performance levels, whose delay and solubility coefficients are summarised in Table 7.2-5.

This study was performed for the « all seals failed » situation, with CU2 spent fuel. In hydraulic terms, the situation is identical to the results observed in the reference calculation. Furthermore, with respect to the radiological impact, no evolution was observed. Indeed :

- considering the slowness of flows through the structures, the majority of sorbed radionuclides diffuses through the geological barrier during transfer in the drifts. The host formation therefore plays an absorptive barrier role for radionuclides, thus masking any geochemical changes in the structures ;
- the release levels (dose greater than 10^{-10} Sv/year) remain governed by the soluble non-sorbed long-life radionuclides iodine 129 and chlorine 36 (and are hence independent upon structure geochemistry).

The only difference between this calculation case the previous ones concerns selenium 79 (not sorbed in the geological barrier). The release level of this radionuclides at the shaft exit is increased by three orders of magnitude. In the reference calculation, its low solubility in the seal contributed to limiting its migration through the drifts. Nevertheless, this radionuclide does not provide a significant-part to the impact.

7.2.5.3 EDZ, seal core permeability and source term sensitivity

This study consists in taking into account, in addition to the degraded EDZ and seal core performance levels, a number of degraded parameters for the vitrified waste release model in such a manner as to integrate possible chemical disturbances to which the waste may be subjected.

This study consists actually in assuming that the vitrified waste is quasi-labile. The impact is the same as when a V_0.S model is applied, with phenomenological parameters. Indeed, this model already assigns a short lifetime to vitrified waste with respect to transfer times. Modelling a quasi-labile or labile glass does not alter the calculation's conclusions.

7.2.5.4 Hydraulic head gradient sensitivity in the Callovo-Oxfordian

This study consists in assessing the influence of a greater hydraulic head gradient in the host formation. The following were adopted :

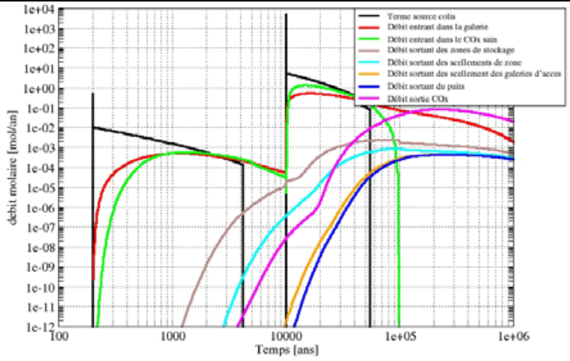
- a vertical ascending hydraulic head gradient of 0.4 m/m applied in a permanent manner,
- a vertical ascending hydraulic head gradient of 1 m/m corresponding to a pessimistic value, to more thoroughly assess the role of this specific parameter.

The calculation was performed for the iodine 129 of CU1 spent fuel in an all seals failed situation.

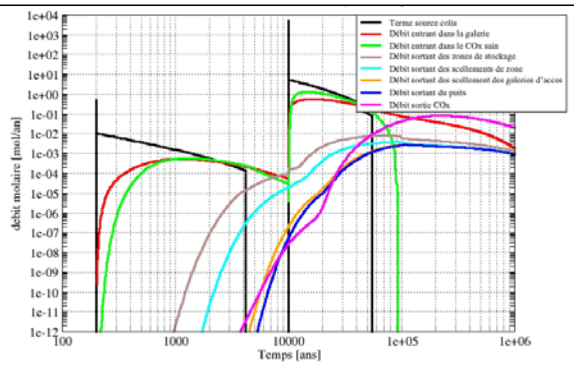
The main lessons drawn from this study were as follows (see Figure 7.2-42)

- the integration of a 5-fold greater ascending hydraulic head gradient in the Callovo-Oxfordian has no impact on the transfer pathway through the geological barrier, with transport remaining mainly diffusive ;
- the molar flow at the shaft exit is sensitive to variations in ascending hydraulic head gradient in the Callovo-Oxfordian : a two-fold higher hydraulic head gradient induces a forty-fold higher maximum release rate at the shaft exit. A five-fold higher hydraulic head gradient, on the other hand, leads to a three hundred-fold higher release rate at the shaft exit. It can be noted that there is no proportionality between the gradient and the mass that manages to leave the shaft : indeed, the exchanges between the drift and the Callovo-Oxfordian increase as transport slows down. For strong gradients, the Callovo-Oxfordian's « absorptive barrier » role is no longer as efficient.
- in the case where the ascending hydraulic head gradient in the Callovo-Oxfordian is equal to 0.4 m/m, the radiological contribution ascribable to the shaft alone is increased, but the total dose remains dominated by the sound argillite transfer pathway, by at least two orders of magnitude. The impact therefore remains identical to that of the reference calculation ;
- in the case where the ascending hydraulic head gradient in the Callovo-Oxfordian is equal to 1 m/m, then the molar flow at the shaft exit is of the same order of magnitude as that passing through the sound argillite up to 40,000 years, then becomes to one order of magnitude lower.

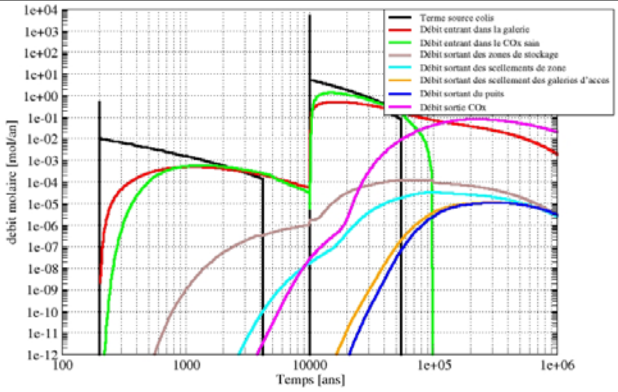
It can therefore be estimated that, obviously in the case of seal failure, the contribution of access structures approaches that of the host formation at values superior to 1 m/m. It should be reminded that such gradients are very significantly higher than those measured from the head differences between overlying and underlying rocks (see Figure 6.2.4, in chapter 6).



sensitivity 1 : 0.4 m/m hydraulic head gradient



sensitivity 2 : 1 m/m hydraulic head gradient



Reference calculation : 0.2 m/m hydraulic head gradient

Figure 7.2-42 Seal failure altered evolution scenario - hydraulic head gradient sensitivity study - molar flow histories - all seals failed situation - CU1 spent fuel - ^{129}I

In conclusion, the integration of a higher hydraulic head gradient significantly increases releases through the structure transfer pathway, but has no impact on transport through the geological barrier, the system remaining dominantly diffusive ; it is only from a vertical ascending hydraulic head gradient of 1 m/m and above, applied in a permanent manner in the Callovo-Oxfordian and considering all seals to have failed, that the release levels become close to those of the geological barrier. Nevertheless, even with such values, the level of the maximum dose still remains equivalent to that estimated in the normal evolution scenario.

7.2.6 Lessons drawn from the seal failure altered evolution scenario

The analysis of the seal failure scenario has shown that the repository system is robust with respect to a failure of all seals, and to chemical disturbances relating either to the seals, or to the vitrified waste cells.

Indeed, given the studied seal failure situations, even considering degraded parameters for the EDZ, the seal core and the release of C waste, impact associated with the activity that migrates through the structures is negligible when compared to that which diffuses through the geological barrier (as in the normal evolution scenario). The peak doses associated with the structure transfer pathway remain, in all situations, significantly lower than the activity fraction that migrates through the geological barrier (by a factor of at least 100).

More specifically, several points were highlighted from an hydraulic point of view :

- In cases where we consider a non-simultaneous failure of shaft and drift seals, the hydraulic disturbance of the repository remains limited because of the efficiency of the seals, which provide redundant cover for the failed seals. The head loss in the repository is of 4 metres at the most and the flows leaving the shaft do not exceed 1.4 m^3/year, that is to say three times the water flow leaving the shaft in the normal evolution scenario. These flows remain limited.

- In the event of simultaneous failure of shaft and drift seals, the hydraulic disturbance is stronger, but remains limited due in particular to the low permeability of the Callovo-Oxfordian (exploited by the « dead end » architecture) and EDZ, which limit water ingress.
- In the case of degraded EDZ performance, this zone constitutes a significant drainage for zones with greater heads to shaft zones. The flows increase and the water flow rates leaving the geological barrier are significantly higher than in the reference situation (approximately 7 m^3/year). Once the EDZ presents severely degraded characteristics, the properties of the seal core and hydraulic cutoffs, as pessimistically represented (anchored only in the fractured zone), no longer provide any significant gains. This conclusion would be tempered with the use of more deeply anchored seals.

Table 7.2-17 compares the water flows leaving the shaft under the different situations of the « seal failure » altered evolution scenario and in the normal evolution scenario (reference calculation and degraded EDZ sensitivity). The results confirm those previously highlighted points : the properties of the EDZ significantly influence repository hydraulics and in particular the flows leaving the shaft. However, even in the most pessimistic configuration, the flows remain very small.

Situations	**Hydraulic flow leaving the shafts (m^3/year) Disposal of spent fuel - S2 scenario**
All seals effective (normal evolution scenario)	0.5
Access structure seals failed	1.0
Shaft abandonment	1.4
Drift seals failed	0.75
All seals failed	2.4
All seals effective + degraded EDZ	6.3
All seals failed + degraded EDZ	7.5
Access structure seals failed + degraded EDZ	7.1

Table 7.2-17 Seal failure altered evolution scenario - comparison of shaft leaving flows for the different situations studied

From an impact point of view, it can be shown, for all situations studied, that :

- the dominant transfer pathway remains the geological barrier. The dose associated with this transfer pathway is identical to the results of the normal evolution scenario, considering the non-significant proportion that migrates through the structures ;
- the dose associated with the activity fraction that migrates through the structures is very low in comparison, by at least 3 orders of magnitude. The most significant impact is that associated with CU1 spent fuel for the all seals failed situation, with a peak dose of 0.00001 mSv/year at approximately 350,000 years. Most situations lead to doses inferior to 10^{-7} mSv/year.

Impact appears to be sensitive to EDZ properties. The integration of degraded properties increases the dose observed in the structure transfer pathways by an order of magnitude and brings it closer in time (around 200,000 years), with no consequences, however, on the global impact.

The integration of conservative vitrified waste release model parameters, simulating for example an alkaline disturbance in the cells, has no influence on the impact.

The integration of conservative seal core geochemical parameters has no influence :

- considering the slowness of flow through the structures, the geological barrier « traps » the radionuclides that may be sensitive to bentonite geochemistry ;
- the radionuclides contributing to the impact are not sensitive to the geochemical properties of the seal cores.

It should be noted that the seals also immobilise any selenium passing through the drifts.

In conclusion, the results validate Andra's interest in defining the properties of the damaged (fractured, micro-fissured) zone, along with the bulk hydraulic properties of the seals. The analysis of the influence of the damaged zone nevertheless underlines that, although this latter is significant in terms of seal efficiency, it remains limited with respect to the general operation of the disposal system.

It is worth noting that the seals are significant contributors to the « resist to water circulation » function, but that the geological medium (mobilised by the « dead end » tree architecture) provides a measure of redundant cover by limiting water ingress, even in the event of the failure of all seals. These components also contribute to immobilising certain radionuclides (such as selenium) that may be soluble in the drifts. Their contribution to the « delay and attenuate radionuclide migration » function, however, appears to be minor, as may have been expected.

Due to the efficiency of these systems, the impact of the « seal failure » altered evolution scenario does not appear to be greater than that of the normal evolution scenario.

7.3 The « waste package failure » scenario

RFS III.2.f states that « For an initial assessment, the case of a package conditioning failure may be taken into account in terms of the uncertainty assessment associated with the reference situation ». Conditioning failure is therefore recommended, to a first approximation, as a sensitivity calculation associated with the normal evolution scenario.

The normal evolution scenario (chapter 5) already takes into account, in the reference situation, an initial failure for the containers of each type of spent fuel (SF) and for three C waste overpacks (one associated with C0 glasses, one associated with C1/C2 glasses and the last with C3/C4 glasses), these failures resulting from a one-off quality assurance problem.

Amongst the scenario choices made by Andra for the Dossier 2005, a specific scenario, called « package failure » is designed to deal with a larger failure of the packaging elements added in view of disposal. The « package failure » scenario is, in principle, equivalent to the normal evolution scenario in terms of calculation : only a large inventory fraction than that used in the normal evolution scenario is released during the first centuries, the other variables remaining identical.

7.3.1 Scenario definition

Chapter 6 enabled to identify a certain number of failure situations that could be covered by the « package failure » scenario ; the aim here is not to reiterate the description of all situations, but rather to expose the calculation method itself. We remind simply that this scenario is, for the most part, conventional : it is based on a major and repeated failure of the packaging manufacturing quality control system. Considering the definition stage of the container studies and in order to guarantee the scenario's conservative nature, we opted to represent this defect as a total « disappearance » of the container, occurring at the scale of the century following disposal of waste to cells[98]. This one hundred year period pessimistically takes into account the time required for the water to reach the package, damage the primary container and start to dissolve the glass or spent fuel.

The uncertainties and failure modes covered by this « waste package failure » scenario were tackled in chapter 6, both in terms of their presentation (see section 6.2) and the construction of altered situations (see section 6.3). The table below summarises these elements for the record.

Uncertainties considered	Other uncertainties taken into account in the analysis99	Components concerned	Comments
Effect of gases on transport during the hydraulic transient → See § 6.2.5.2	All uncertainties leading to very early releases into the drifts or radionuclides releases in gaseous form	Disposal cells for B and C waste packages, and for spent fuel	This uncertainty relates to the role of gases in the transport of the radionuclides, during the early pressure increase phase. It is covered in the SEN by a sensitivity study on transport during the hydraulic transient phase for B1x waste and CU1 cells. Sensitivity is also taken into account in the package failure SEA.
Systematic error during the quality control on the container welds, not detected during the inspection → See § 6.2.12.2	None	C waste over-pack and spent fuel container	

Table 7.3-1 « Package failure » altered evolution scenario – uncertainties

98 As calculations required that a specific date be defined and adjusted to the decay of the radiological inventory, we chose a period of 200 years, from formation in the reactor, for the radionuclides comprising the waste.
99 Which have similar effects or which could aggravate the effects examined here

For type B waste, there is no expected performance for the overpacks with respect to confinement (see chapter 3). The overpack made from cement-based materials, guarantees a favourable chemical environment for the waste. For bituminous sludge waste, it contributes towards ensuring the dimensional stability of the bitumen in such a manner as to guarantee the proposed release model. Failures of these functions are covered by reference parameters and normal evolution scenario sensitivity studies : sensitivity study to highly pessimistic release models for all type B waste (labile for the most part, and governed only by water inflows into the cell for bitumen), sensitivity study for release during the hydraulic transient. The calculations have shown (chapter 5) that these different situations had no significant consequences on impact. Indeed, at the current stage, the radionuclide release times by the packages, along with the estimated transfer times through the concrete overpack, remain short compared to the transfer time through the host formation. B type waste is therefore no longer addressed in the remainder of this scenario.

For type C waste, the primary container has no specific performance. The overpack, on the other hand, possesses an expected water tightness performance during the post-closure phase (see chapter 3) in order to prevent water from entering into contact with the glass while this latter is at a temperature in excess of 50 to 60°C, in such a manner as to take advantage of lower dissolution rates. This function furthermore avoids any solute release into the medium while the temperature is significant, within which diffusion is faster and certain elements are likely to be sorbed to a lesser extent. Assuming container failure is therefore primarily equivalent to considering radionuclide release and transfer in a thermal environment.

For spent fuel (SF), the issue is the same as for C waste. The initiating event of the considered « package failure » scenario corresponds to the premature failure of the spent fuel containers. This scenario also initially covers the case of a criticality accident occurring in the repository following bringing of assemblies together in a highly improbable configuration.

In either case, the number of packages to take into consideration remains arbitrary. We decided to deal with two cases.

The first, corresponding for example to a quality control system failure situation, is equivalent to considering one month of defective container manufacture. With the currently selected waste flow hypotheses, this corresponds to :

- 35 defective overpacks for type C0 packages,
- 50 defective overpacks for type C2 packages, for type C1/C2 packages processed together,
- 50 defective overpacks for type C4 packages, for type C3/C4 packages processed together,
- 30 defective CU1 spent fuel containers (UOx2, UOx3, URE fuel),
- 12 defective CU2 spent fuel containers (MOX fuel).

The second case, covered as a sensitivity analysis, takes into account the complete failure of the whole inventory of vitrified waste and spent fuel containers. It corresponds, for example, to a poor assessment of container lifetime for the whole repository. In this respect, assuming failure after 200 years is a most pessimistic hypothesis.

Finally, in an approach identical to that governing the normal evolution scenario and its sensitivity studies, the possibility that radionuclide migration should occur in the presence of large amounts of corrosion gasses is also taken into account. This situation is, in principle, incompatible with release by vitrified waste and spent fuel packages. Nevertheless a sensitivity study is performed in which an hydraulic transient « forced » through gas is superimposed onto the thermal transient, during the first 10,000 years. This transient is assumed to correspond to the maximum accumulation of gas pressure.

The hydraulic transient is modelled according to the same terms as for the normal evolution scenario (see section 5.5.6.3) in the safety calculation, with respect to that generated by conceptual modelling. As it will be shown that it is the CU1 spent fuel package failure that has the greatest impact and that this impact is due to iodine 129, this calculation is only performed in this case, bounding of others. We remind that the adopted modelling is not realistic, since as it is equivalent to considering that water is « created » in zones under pressure in such a manner as to permanently maintain saturation.

Seal permeability may be an important parameter ; it is worth reminding that it is low in the reference calculation (the seal core has a permeability of 10^{-11} m/s and the structure is only anchored in the fractured zone, thus leading to a bulk seal permeability of 10^{-9} m/s). Thus, to assess the degree of conservatism of this hypothesis, a variant was selected in the calculation, two additional studies were performed, taking the following into account :

- a saturation-dependent permeability in the structures,
- an anchoring equivalent permeability of the seals of approximately 10^{-10} m/s instead of 10^{-9} m/s.

In a similar way to that modelled in the normal evolution scenario, the cells containing defective packages are close to the exit of the repository zone closest to the shafts ; this configuration on the one hand minimises the distance between the defective package zone and the shaft and, on the other hand, maximises the hydraulic influence of the upstream repository drifts.

To measure the combined effect of the hydraulic transient and of a larger number of defective packages, we also considered the highly unrealistic case in which all of the spent fuel packages were defective, which correspondants to the most pessimistic case.

7.3.2 Processing the scenario

The « package failure » scenario is only distinguished from the normal evolution scenario by the higher number of defective packages and consequently the higher level of prematurely released activity. [106].

The premature failure of the C waste and spent fuel containers affects two of the three safety functions :

- on the one hand, the « limit the release of radionuclides and immobilise them in the repository » function, since water reaching the waste during the transient may jeopardize release kinetics. This is particularly the case for C waste, in which the release kinetics increase significantly with temperature ;
- on the other hand, the « delay and attenuate radionuclide migration » function. Indeed, since the radionuclide transport or retention parameters in a high thermal load and temperature gradient environment are less favourable and/or less well understood, the parameters selected during the thermal transient are more unfavourable than those selected in the long term for diffusion and for the sorption of certain elements.

Initially, the analysis shall concern the performance of the safety functions that have been affected by the premature failure of the containers and overpacks. The impact results are then presented.

7.3.2.1 Description of defective packages and their environment

The number of defective packages aside, the description of packages, models for release, transport, and chemical retention of radionuclides are identical to those adopted in the SEN. The effect of different disturbances, particularly of the thermal disturbance, is also addressed in a similar manner. The defective packages are supposed to « disappear » after 200 years of radionuclide decay.

The premature failure of C waste overpacks leads to a loss of performances associated with glass releases. In theory, at the moment when dissolution begins a diffusive regime is established within the cell and the model founded on the residual rate of dissolution ($V_0.S \rightarrow V_r$) is applicable, at least for the glasses for which experimental observation was possible. In the SEA, however, in order to conservatively increase the release, we systematically adopt the highest source terms (corresponding to a $V_0.S$ model for all vitrified waste). A stronger initial rate takes the temperature's effect into consideration.

As a result of these choices, the activity release contained in defective packages occurs over a 200 year period for all types of packages, instead of over approximately 3 000 years for not-failed C0 reference package and around 100 000 years for not-failed C1/C2 and C3/C4 packages.

Similarly, to take premature release in a high temperature field into account, we adopt a loss of performance associated with the fuel matrices of defective containers. For example, the release of iodine-129 activity contained in defective packages occurs over approximately 4 000 years instead of over the 40 000 year period for not-failed reference package. The labile fractions are the same as in not-failed packages.

The temperature's effect is taken into account for diffusion in cases in which transfer is envisaged to an area with a strong thermal load. Averaging the diffusion coefficient in three thermal sub-units of the Callovo-Oxfordian formation and in one zone comprising the structures and EDZ, this is presented as a normal evolution scenario. In each of these zones a correction factor of the effective diffusion coefficient, evaluated at 20°C, is defined according to time. Figure 5.3-12 shows the evolution over time of the correction factor of CU1 spent fuels applied to the effective diffusion coefficient at 20°C, taking the temperature into account. The correction factors are then given for the other reference package in Figure 5.3-13.

The increase in temperature leads to a more significant increase in the diffusion coefficient than in permeability. When the temperature rises, then, we tend to increase in diffusion over advection. Transport already occurring by dominant diffusion in the geological barrier, an increase in permeability of the Callovo-Oxfordian formation during the thermal transient has, in theory, no influence on the molar flow rate leaving the host formation ; the system remains in a dominant diffusion regime. In order to simplify the calculations, the temperature's effect on permeability was not taken into consideration in evaluating radionuclide transport in the geological barrier.

As in normal evolution scenario, no correction factor is adopted to account for the effect of temperature on the solubility of elements, as no significant effect was observed under experimental conditions [21]. For adsorption, the process entailed retaining a reversible correction factor of the partition coefficient of 0.1 for cesium and, by analogy, for beryllium as well as calcium, so long as the temperature in the medium is above 20°C. As in the calculation of the correction factor of the diffusion coefficient, the division of the Callovo-Oxfordian formation and the structures into four sets is maintained ; the time taken to return to a differential temperature below 20°C with regard to the initial temperature was evaluated in each of these zones. The results are presented in Table 7.3-2.

Date at which the temperature level returns below 20°C				
C2	**C4**	**CU1**	**CU2**	**C0**
3 500 years	5 000 years	15 000 years	15 000 years	1 000 years

Table 7.3-2 Date at which the temperature level returns below 20°C - C waste cell, CU1 and CU2 spent fuels (in two-dimensional conservative thermal models)

7.3.3 Effects on safety functions

The effects of package failure on the « limiting the release of radionuclides and immobilising them in the repository » function are reflected by a reduction in the lifetime of packages which are, de facto, quasi-labile with regard to transfer times in the rest of the repository system. As such, here we examine only the performance indicators of the « delaying and attenuating radionuclide migration » function.

With regard to molar mass and flow rate leaving the Callovo-Oxfordian formation, the principal teachings of this study are illustrated by the case of failed CU1 containers. They are similar in the case of vitrified waste.

- Compared to the SEN, the premature failure of a larger number of overpacks and containers leads to a more significant activity release in a thermal environment in which the containment performances of the Callovo-Oxfordian formation are less favourable. Nonetheless, the thermal transient's life is sufficiently short and the number of defective packages relatively limited, compared to the number of not-failed packages, that the remaining mass confined in the Callovo-Oxfordian formation is identical in package failure SEA and in SEN. For the primary impact contributors :
  - ✓ 20 % to 30 % of the iodine-129 mass remains confined in the Callovo-Oxfordian formation,
  - ✓ 65 % to 75 % of the chlorine-36 mass remains confined in the Callovo-Oxfordian formation,

✓ 90 % to 95 % of the selenium-79 mass remains confined in the Callovo-Oxfordian formation.

- In the first 10 000 years, the molar flow rate leaving the Callovo-Oxfordian formation for the three primary impact contributors is greater in « package failure » SEA than in SEN (See Figure 7.3-1), as in both cases it is due to defective packages. There is a direct proportionality between the defective inventory and the molar flow rate, a factor 30 in the case of CU1 spent fuels.[100] Later, once the contribution associated with the fraction activity emitted by not-failed packages appears, the curves of the molar flow rate leaving the Callovo-Oxfordian formation blend together. These results are explained by two points :
  ✓ in both situations the number of defective packages remains vastly inferior to the total number of packages,
  ✓ the thermal disturbance (modification of the transfer and retention parameters), of relatively limited term when compared to the long times characteristic of diffusion in Callovo-Oxfordian argillites, has no significant influence on releases leaving the host formations.

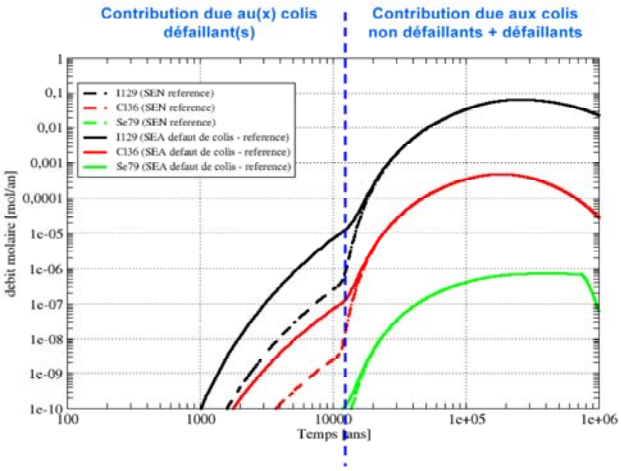


Figure 7.3-1 Package failure SEA – Molar flow curves leaving the Callovo-Oxfordian formation – comparison ^{129}I, ^{36}Cl et ^{79}Se between the SEN situation (1 defective package) and package failure SEA (reference – 30 defective packages) - CU1 spent fuels

7.3.4 Impact calculations

As in the SEN, the impact calculation is evaluated after transfer of radionuclides in the Dogger and overlying formations to the potential outlets for models of the present day and after one million years (See section 5.3.2.1).

Figure 7.3-2 gives a horizontal section of the Hp1-Hp4 (Oxfordian) horizons of CU1 iodine-129 concentrations. In accordance with the molar flow curves presented in Figure 7.3-1, the figure clearly shows a picture of the defective package zone, visible at very weak concentrations for up to 30 000 years before being concealed by the contribution of not-failed packages.

[100] We shall note that the curves make the molar flow rates apparent at the entrance of the host formation very early (from 1 000 years). One must not be wrongly induced by the logarithmic scale adopted. The diffusion equation is exponential, it allows for the calculation of an extremely weak rate leaving the host formation from the instant following the release of radionuclides. Such a calculation has no physical sense. Once the rates are more substantial, this calculation becomes representative.

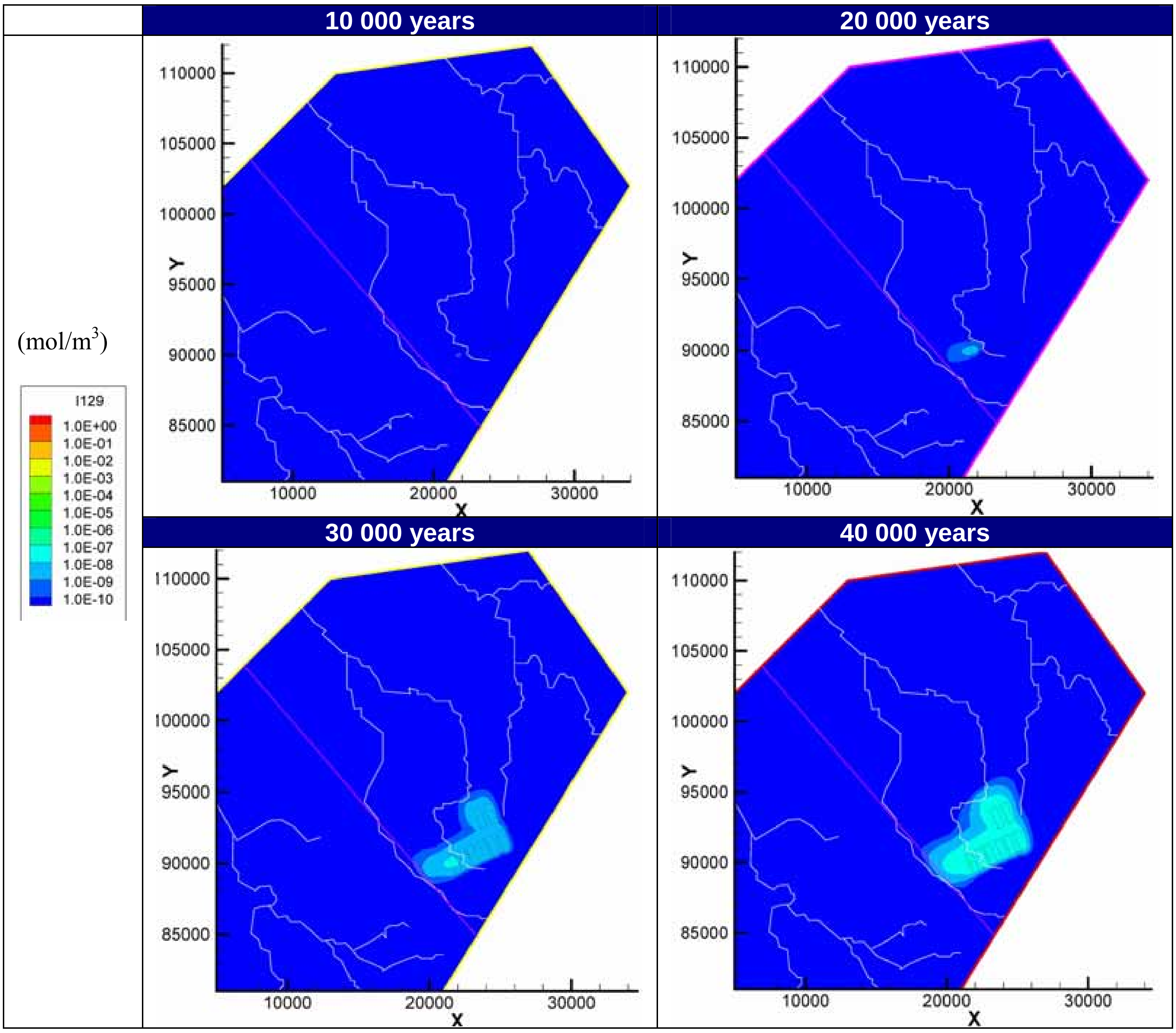


Figure 7.3-2 Package failure SEA – Horizontal section in Hp1-Hp4 (Oxfordian) of CU1 iodine-129 concentrations from 10 000 years to 40 000 years

Given its proximity to the repository and the nature of the current lines, the Oxfordian Saulx outlet presents the highest dose. The activity fraction associated with defective packages is concealed a few thousand years after not-failed overpacks and containers lose their water tightness. The dose associated with defective packages is negligible compared with that of performing packages and the results, in terms of total impact (see Table 7.3-3), are identical to those in the SEN presented in chapter 5.4. Here, we recall them for the model at one million years (keeping in mind that the doses are very similar in the present day model).

As with the SEN, let us also recall that the addition of different impacts does not report a real situation, as it ends up counting the same inventory twice (we cannot concurrently have a maximal quantity of C1/C2 and C3/C4 glasses). However that may be, we shall see that despite of this skew the total estimates remain weak.

Reference package	Maximum dose [mSv/an]	Date of maximum [years]	Contributing radionuclides
« Saulx » outlet (the most pessimistic)			
Type C1 and C2 glasses	0,00047	490 000	^{129}I ; ^{36}Cl
Type C3 and C4 glasses	0,00036	500 000	^{129}I ; ^{36}Cl
Total glass	approximately 0.00083	490 000	^{129}I ; ^{36}Cl
CU1 spent fuels	0,019	330 000	^{129}I
CU2 spent fuels	0,0017	340 000	^{129}I
Total of spent fuels	approximately 0.02	around 330 000	^{129}I

Table 7.3-3 Package failure SEA – Total dose – dates of dose maxima and principal contributors to the Oxfordian Saulx outlet (most pessimistic case) – model at 1 million years - all waste

7.3.5 Sensitivity analysis

7.3.5.1 Sensitivity to the failure of all packages

This sensitivity analysis is very pessimistic and covers all situations of premature failure. It is also purely conventional, because no cause identified in chapter 6 could provoke such a premature failure of all inventory (barely a century after having been placed in repository). A failure of all containers, resulting in the complete loss of water tightness in C waste over-packs or spent fuel containers is presumed at an extremely pessimistic date, identical to that considered in the reference calculation.

The main teachings of this study for spent fuels are the following :

- the iodine-129 mass (CU1 spent fuels) leaving the Callovo-Oxfordian formation during 1 million years is superior by about 4 % to the SEN, as illustrated in Figure 7.3-3. This small difference results from the fact that the duration of the thermal transient is short compared to the transfer time in the Callovo-Oxfordian formation. So, during this first period the average thickness traversed in the Callovo-Oxfordian formation is limited compared to the total thickness of the Callovo-Oxfordian formation ; beyond this, radionuclides migrate once again in the Callovo-Oxfordian according to normal evolution. There is little change in the total argillite release kinetics, as most of the mass emerges from the Callovo-Oxfordian formation beyond 100.000 years ;

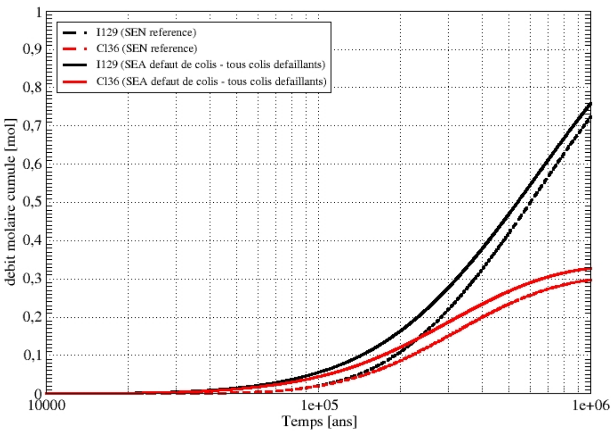


Figure 7.3-3 Package failure SEA - Evolution of cumulative molar flow rate leaving the Callovo-Oxfordian formation - comparison of « all defective packages » and SEN – CU1 spent fuels

- we note a slight increase in the molar flow rate leaving the Callovo-Oxfordian formation. As an example, at 100.000 years the iodine-129 molar flow rate leaving the Callovo-Oxfordian formation is twice as high and the mass twice as significant as in SEN ;

- the maximum molar flow rate leaving the Callovo-Oxfordian formation occurs 40 000 years earlier than in the SEN (see Figure 7.3-4).

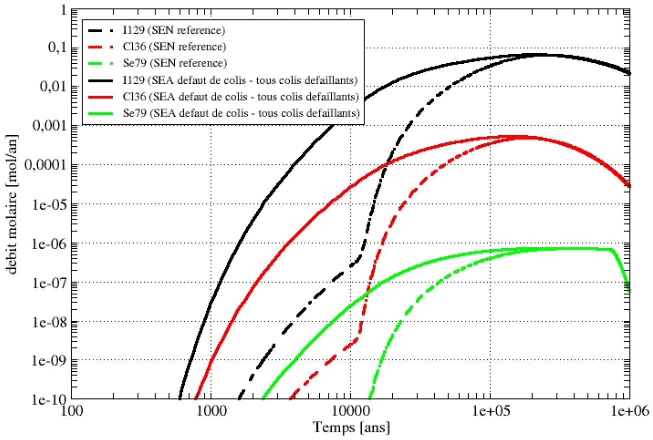


Figure 7.3-4 Package failure SEA - Evolution of molar flow rate leaving the Callovo-Oxfordian formation - comparison of « all defective packages » and SEN – CU1 spent fuels

Behaviour is different for vitrified C waste as the release kinetics of defective packages become short compared to diffusive transfer times. So, the quantity of iodine-129 (for C1 and C2 waste) released by the Callovo-Oxfordian formation over one million years is superior to the SEN by about 10 %. This difference, more significant than for CU1 spent fuels, is due to :

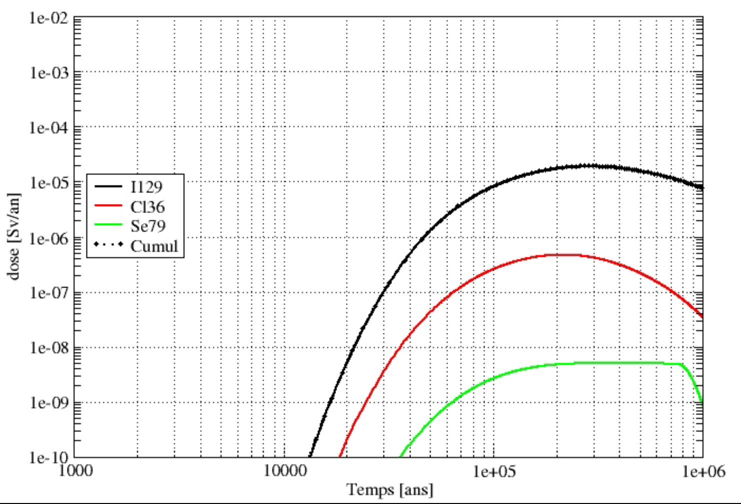
- the taking into account of a V_0.S model leading to a release over a few hundred years rather than over 100 000 years or more in SEN. The release kinetics of packages are concealed by the diffusive transfer time in the host formation ;
- the premature release of radionuclides in a hot environment which slightly modifies the total kinetic transfer in the argillites. This phenomenon increases the mass leaving the Callovo-Oxfordian formation over one million years by 5 % for iodine-129 and 6 % for chlorine-36.

The maximum molar flow rate leaving the Callovo-Oxfordian formation is pushed forward by approximately 230 000 years for the « all defective packages » scenario. The predominant effect is due to using a V_0.S model for all waste ; transfers occurring in a thermal environment account for only 30 000 years of the 230 000 years.

The doses are given in Table 7.3-4 The maximum doses remain generally unchanged with regard to the reference situation : nonetheless, the time at which the dose maxima appear are moved ahead by a few tens of thousands of years for CU1s and by about 200 000 years for vitrified C waste. We present the doses in the hydrogeological model at 1 million years and for the most pessimistic outlet, the Saulx. The doses would be similar for the present-day hydrogeological model (see Figure 7.3-5 to Figure 7.3-8).

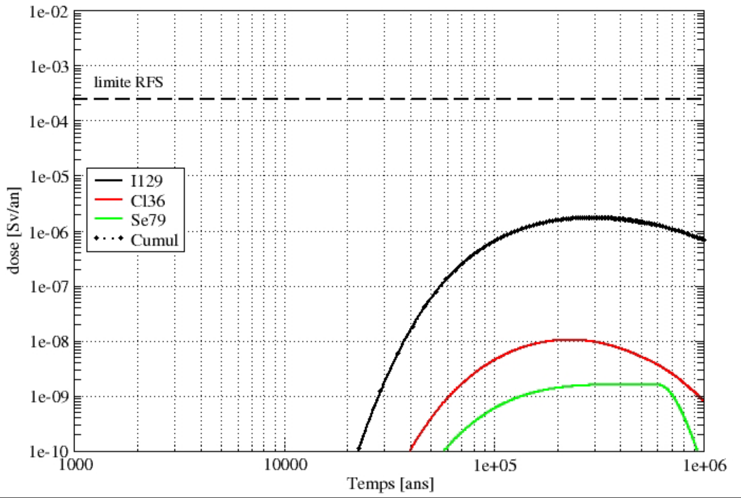
Reference package	Maximum dose [mSv/an]	Date of maximum [years]	Contributing radionuclides
« Saulx » outlet (the most pessimistic)			
Type C1 and C2 glasses (S1b scenario)	0,00066	270 000	^{36}Cl ; ^{129}I
Type C3 and C4 glasses (S1a scenario)	0,00049	270 000	^{36}Cl ; ^{129}I
Total glasses	0,0012	270 000	^{36}Cl ; ^{129}I
CU1 spent fuels	0,019	290 000	^{129}I
CU2 spent fuels	0,0017	290 000	^{129}I
Total of spent fuels (S2 scenario)	approximately 0.021	around 290 000	^{129}I

Table 7.3-4 « Package failure » SEA – sensitivity to all defective packages – Total dose – dates of dose maxima and principal contributors to the Oxfordian Saulx outlet (most pessimistic case) – model at 1 million years - C waste and spent fuels



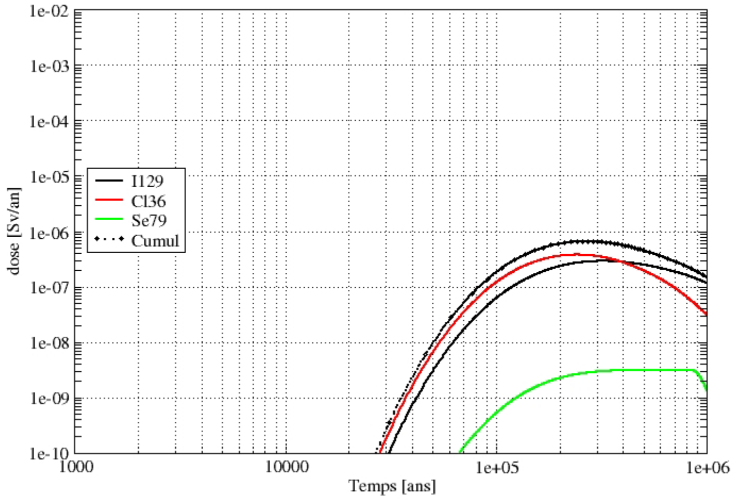
RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cl36	4.7E-07	2.2E+05
Se79	5.0E-09	3.4E+05
I129	1.9E-05	2.9E+05
Cumul	1.9E-05	2.9E+05

Figure 7.3-5 Package failure SEA – Sensitivity calculation for all defective packages – Dose at the Oxfordian Saulx outlet – CU1 reference package



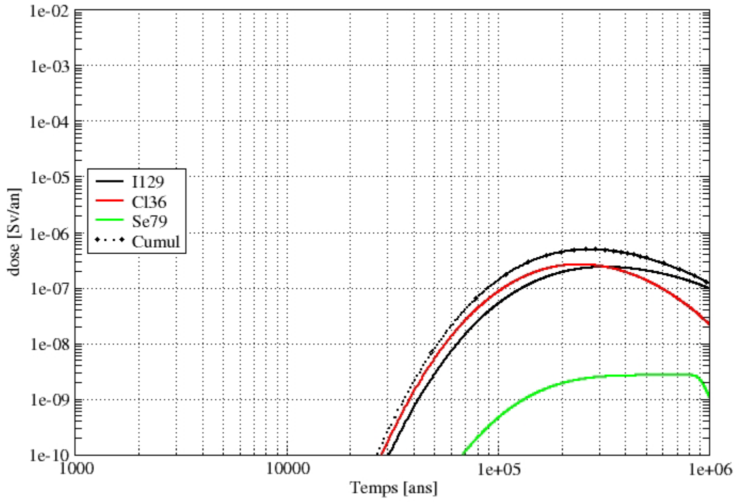
RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cl36	1.0E-08	2.3E+05
Se79	1.6E-09	4.9E+05
I129	1.7E-06	2.9E+05
Cumul	1.7E-06	2.9E+05

Figure 7.3-6 Package failure SEA – Sensitivity calculation for all defective packages – Dose at the Oxfordian Saulx outlet – CU2 reference package



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cl36	3.8E-07	2.4E+05
Se79	3.2E-09	8.0E+05
I129	2.9E-07	3.2E+05
Cumul	6.6E-07	2.7E+05

Figure 7.3-7 Package failure SEA – Sensitivity calculation for all defective packages – Dose at the Oxfordian Saulx outlet – C1/C2 reference package



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cl36	2.6E-07	2.4E+05
Se79	2.7E-09	6.9E+05
I129	2.4E-07	3.2E+05
Cumul	4.9E-07	2.7E+05

Figure 7.3-8 Package failure SEA – Sensitivity calculation for all defective packages – Dose at the Oxfordian Saulx outlet – C3/C4 reference package

7.3.5.2 Taking the hydraulic transient into account

We recall that this calculation consists of taking the possibility of a radionuclide migration due to gas pressure into account during the first ten thousand years, in a model which is not a physical reality but an overestimate of effects. The calculations were done on the iodine-129 from CU1 spent fuels, which presents the most significant impact. A number of cases were evaluated and are recalled below.

Cases addressed	Page number
30 containers of defective CU1 spent fuels Completely resaturated medium Equivalent hydraulic cutoff of drift seals : 10^{-9} m/s Calculation done for the « geological barrier » and « structures » transfer pathway	page 522
Sensitivity to the number of defective packages : all defective packages Calculation done for the « structures » transfer pathway	page 525
Sensitivity to the saturation rate of structures (influence of the saturation rate on the permeability of structures) Calculation done for the « structures » transfer pathway	page 525
Sensitivity to the permeability of equivalent anchorings of seals (10^{-10} m/s instead of 10^{-9} m/s) Calculation done for the « structures » transfer pathway	page 526

Table 7.3-5 Package failure SEA – Sensitivity study on hydraulic transient influence – synthesis of cases addressed

- **Basic case**

Transfers via the host formation

As in the SEN, the results reveal that overpressure caused by gas production during the hydraulic transient has no influence on the molar flow rate of iodine leaving the Callovo-Oxfordian formation. The transient's life is sufficiently short compared with transfer time in the geological barrier that it does not significantly increase the molar flow rate leaving the Callovo-Oxfordian formation (see Figure 7.3-9). The influence of the hydraulic transient is, *a fortiori*, negligible on the impact.

This justifies the decision to address only the transfer pathway via the structures in the remaining calculations.

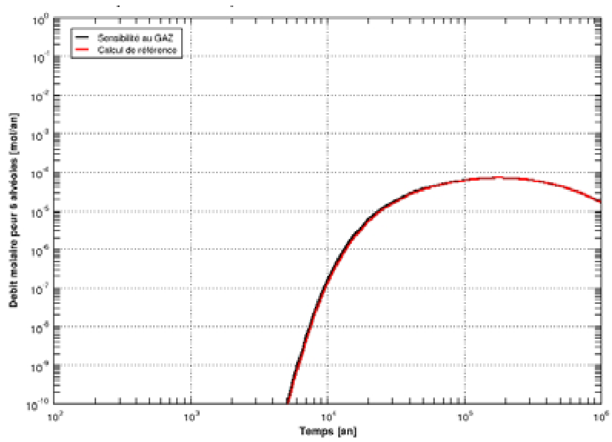


Figure 7.3-9 Package failure SEA – sensitivity calculation of hydraulic transient influence – records of the molar flow rate leaving the Callovo-Oxfordian formation due to 6 CU1 cells – comparison of cases with and without hydraulic overpressure (^{129}I)

Transfers via the drifts

The model being the same, the results relating to water flows are identical to those presented in the SEN for sensitivity to the hydraulic transient (see section 5.5.6.3).

Let us recall that the defective packages were placed near the edge of the zone closest to the shafts ; this configuration allows, on the one hand, to minimise advective transfer times in the connecting drifts to the seal zone and to the shaft, and, on the other hand, to favour the transfer pathway via the structures during the first 10 000 years in which gas pressure from all upstream repository is influential. The results reveal that most of the mass contained in defective packages uses the transfer pathway via the structures (roughly 80 % of the initial mass reaches the access drifts).

The molar flow curve records presented in Figure 7.3-10 demonstrate the influence of gas pressure on the transfer pathway via the structures ; indeed, we note that :

under the circumstances used in the model, the influence of gas leads to a factor 70 increase in the maximum molar flow rate leaving the shafts and to more forward by roughly 500 000 years the maximum date compared to a situation neglecting this transient. The majority of the iodine-129 comes from the fraction of defective packages, even beyond the hydraulic overpressure transient. Nevertheless, we must note that the quantity of iodine-129 released by the shafts during one million years is multiplied only by a factor of six compared to the SEN (while there are thirty times more defective packages in the SEA) ;

during the first 10 000 years in which hydraulic overpressure is active, the molar flow rate leaving the shafts is greater than that from the transfer through unaltered argillites (something not observed in the SEN) ; beyond this, it becomes inferior. All things considered, the molar flow rate leaving the shafts remains weak over all time phases and the maximum molar flow rate leaving the shafts, reached around 300 000 years, is approximately five orders of magnitude weaker than that leaving the Callovo-Oxfordian formation.

These results show that the importance of the transfer pathway via the structures remains inferior to that of the transfer pathway via the geological barrier. The mass leaving through the shafts over the total duration of the analysis remains weak compared to the total mass leaving through the geological barrier.

Nonetheless, we note that the molar flow rate leaving through the shafts may be slightly more significant than that leaving through the geological barrier during the first 10 000 years. The maximal dose levels are not dependent on this activity fraction and are driven by the mass leaving through unaltered argillites, for which the molar flow rate is not influenced by overpressure.

In the case of a premature packing failure of a fraction of a CU1 spent fuels package, alongside a set of pessimistic hypotheses, taking hydraulic transient overpressure due to gas generation into account has no significant influence on radiological impacts. The maximal dose levels remain identical to those in the reference calculation.

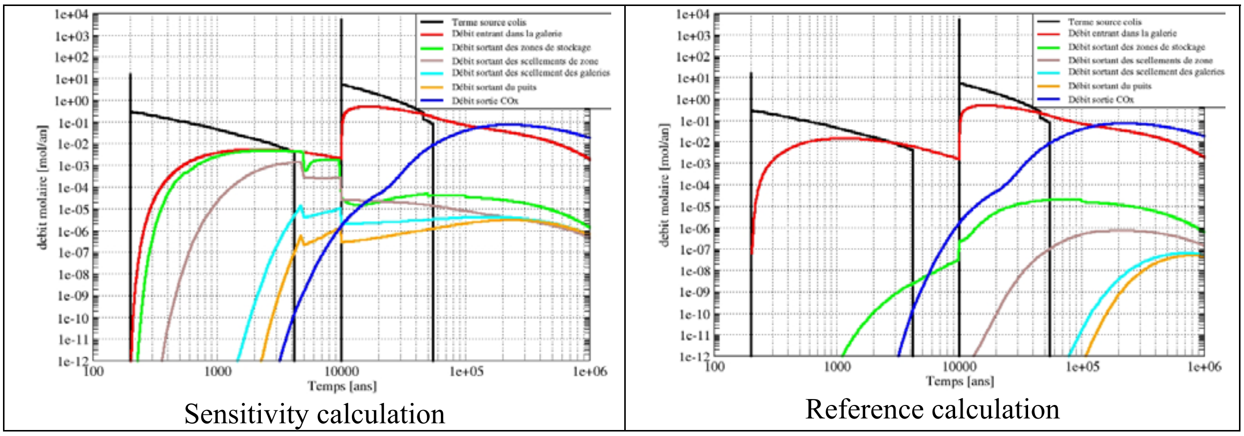


Figure 7.3-10 « Package failure » SEA – sensitivity calculation of hydraulic transient influence – record of molar flow rate – ^{129}I – CU1 spent fuels.

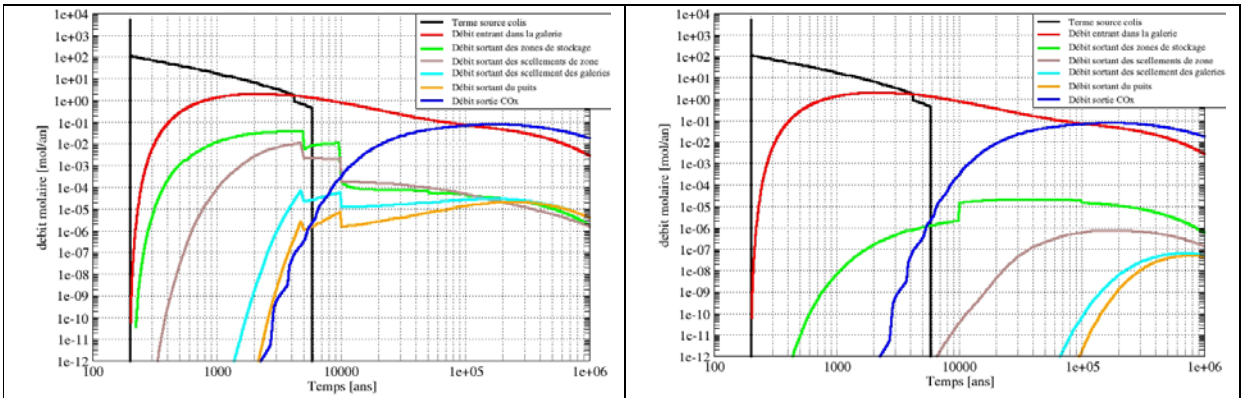
- **Sensitivity to the number of defective packages – Transfer pathway via the structures**

This conventional situation consists of considering a premature defect in all spent fuel containers after 200 years. Gas pressures could then be likely to lead to a favoured advancement, via the structures, of the radionuclides present in all of the packages.

The results of this hydraulic calculation are identical to those in the reference calculation.

With regard to transport over repository structures, the molar flow record curves presented in Figure 7.3-11 reveal that :

- When leaving the shaft through transfer in the structures, the maximal molar flow rate increases by only one order of magnitude between a sole defective fraction situation and all defective packages, while the potentially mobilisable mass is 450 times more significant. This small difference is due to the fact that the mass leaving the shaft comes essentially from cells in the first module of each repository zone. The other modules, further from the shafts, under weaker upstream hydraulic pressure and isolated from the shafts by one or more supplementary module seals, play a very limited role.
- During the phase in which overpressure is significant (roughly 5000 years), the two transfer pathways remain co-dominant ; nonetheless, the molar flow rates leaving the shafts and the Callovo-Oxfordian formation are weak (roughly factor 10 000) compared to the maximal molar flow rate leaving the Callovo-Oxfordian formation after approximately 200 000 years. We must note that the molar flow rate maxima leaving the shafts remains in all cases clearly inferior to that leaving the Callovo-Oxfordian formation.



Sensitivity calculation – all defective packages | Reference calculation – all defective packages

Figure 7.3-11 « Package failure » SEA – sensitivity calculation of hydraulic transient influence – record of molar flow rate – case of all defective packages- ^{129}I – CU1 spent fuels.

So, in the case of all CU1 containers being prematurely defective, taking transient hydraulic overpressure due to the generation of gas on solutes transfers into account has no strong influence on radiological impacts. The maximal dose levels remain identical to those in the reference case.

- **Sensitivity to the saturation rate of structures – Transfer pathway via the structures**

This study consists of evaluating the influence of the saturation rate on advective transport (in particular on permeability) in the repository structures. Let us recall that in the basic case the structures were presumed saturated by water from the initial point in time. This gives a pessimistic character to the calculation, in particular with regard to advective transfer in the drifts and their EDZ.

The effective saturation (Sr) (hereafter referred to as saturation rate) was evaluated with particular regard to the hydrogen production rate in the backfill, the bentonite in drift seals, and EDZ fractured and microfissured zones. Evaluations are based on the conceptual models which reveal the following saturation rates : around 70 % in the backfill, 95 % in the fractured EDZ, and 97 % in the microfissured EDZ and bentonite during the first 5000 years in which hydrogen production is relatively significant. Beyond this and up to 100 000 years, when hydrogen production is lower, saturation rates are as follows : 90 % in the backfill, 97 % in the fractured EDZ, and 100 % in the microfissured EDZ and bentonite. These values are also valid for the principal connecting drifts. After 100 000 years, all materials are 100 % saturated.

The permeability K(Sr) was then evaluated according to the saturation level in four previously referred to structures based on the Van Genuchten law [61]. Table 7.3-6 gives the correction coefficient applied to the permeability of each of the components following the previously referred to time breakdown.

	Backfill	Fractured EDZ	Microfissured EDZ	Bentonite
From 0 to 5 000 years	3.10^{-2}	0.3	0.4	0.4
From 5 000 to 100 000 years	0.2	0.4	1	1
Beyond 100 000 years	1	1	1	1

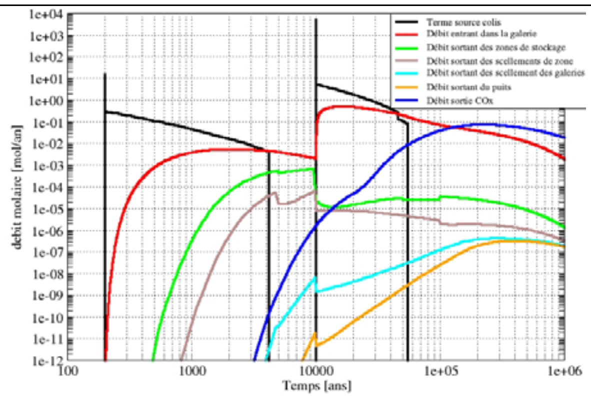
Table 7.3-6 Correction coefficient applied to permeability for the principal connecting drifts and drifts in repository zones for spent fuels

The records of molar flow rate provided in Figure 7.3-12 allow for a comparison of results with and without taking the influence of saturation levels in the structures on the permeability value into account. They demonstrate that :

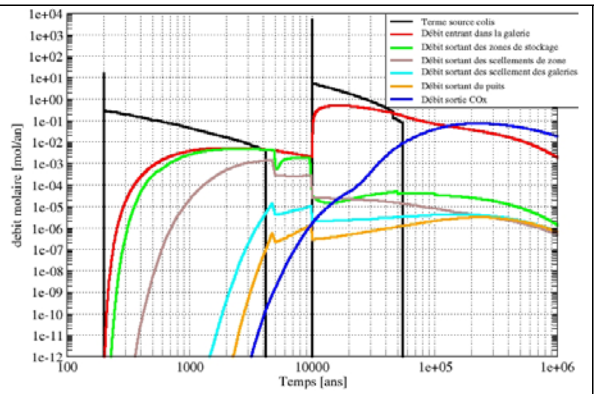
- in the first 10 000 years corresponding to the hydraulic transient and to a weak degree of saturation in the structures, the molar flow rate leaving the shafts is significantly weaker than in the basic case (roughly five orders of magnitude). The slow rate of transfer in the non-saturated drifts favours exchanges with the geological barrier ;

- beyond 10 000 years and up to 100 000 years, saturation is not complete and the molar flow rate leaving the shafts remains inferior to the « resaturated from the outset » calculation case ;
- the maximal molar flow rate occurs around 250 000 years. At this time saturation is complete but, as the activity fraction present in structures during the preceding phases was less than in the « resaturated from the outset » calculation case, the maximum molar flow rate leaving the shafts is weaker by one order of magnitude.
- This point allows us to quantify the extremely pessimistic nature of calculations that consider an area completely saturated during the hydraulic transient.

As with the basic case, the transfer pathway through the geological barrier remains particularly dominant during the hydraulic overpressure phase, with a molar flow rate leaving the Callovo-Oxfordian formation superior by roughly five orders of magnitude compared to the flow rate leaving the shafts. The maximum dose is therefore not modified.



Sensitivity calculation
Taking the degree of saturation into account
Fraction of defective packages



Sensitivity calculation (basic case)
Fraction of defective packages

Figure 7.3-12 « Package failure » SEA – sensitivity calculation of hydraulic transient influence – record of molar flow rate – taking the degree of saturation in the structures on the permeability value into account - ^{129}I – CU1 spent fuels.

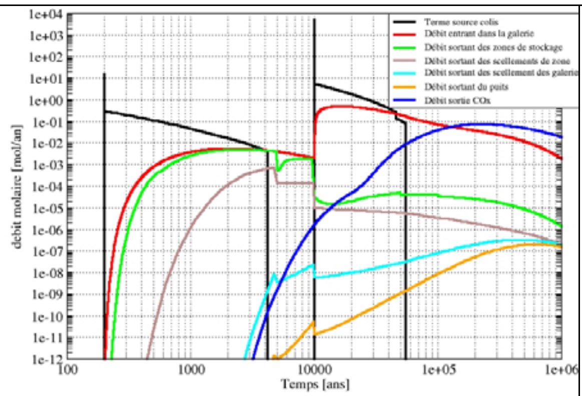
- **Sensitivity to the permeability of equivalent anchorings of seals – Transfer pathway via the structures**

This study consists in considering an equivalent permeability of anchoring drifts at 10^{-10} m/s instead of at 10^{-9} m/s (see chapter 5 section 5.3.2.2). We presume that all media are saturated.

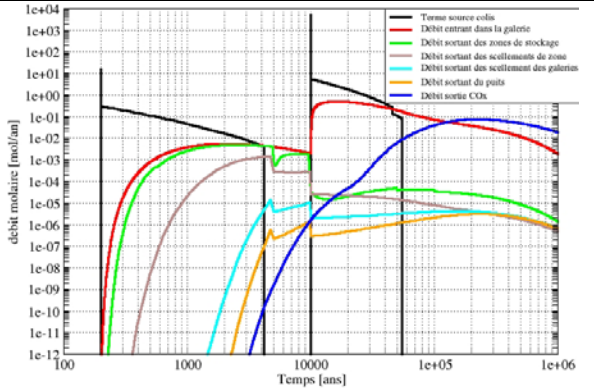
The records of molar flow rate provided in Figure 7.3-13 allows us to compare the influence of taking a weaker permeability coefficient into account for the equivalent anchorings drift seals (10^{-10} m/s rather than 10^{-9} m/s). They demonstrate that :

- During the first 10 000 years in which there is overpressure due to gas, taking the weaker permeability of equivalent anchorings into account allows for a reduction of the molar flow rate leaving the drift seals (at the base of the shafts) of roughly three orders of magnitude.
- Beyond this, the molar flow rate leaving the shafts remains weaker. The maximal molar flow rate occurs around 250 000 years and is reduced by roughly 1 order of magnitude compared to the basic case (considering a permeability of equivalent anchorings seals of 10^{-9} m/s).
- As with the other cases, transfer pathway through the geological barrier remains largely dominant during the hydraulic overpressure phase, with a molar flow rate leaving the Callovo-Oxfordian formation superior by roughly 4 to 5 orders of magnitude compared to the maximal flow rate leaving the shafts.

This calculation complements the preceding observations which demonstrate the role of seals in limiting the influence of a failed package coupled with an unfavourable hydraulic transient.



Sensitivity calculation
Taking into account the equivalent permeability of anchorings : 10^{-10} m/s
Fraction of defective packages



Sensitivity calculation (basic case)
Permeability of equivalent anchorings : 10^{-9} m/s
Fraction of defective packages

Figure 7.3-13 « Package failure » SEA – sensitivity calculation on the influence of the hydraulic transient – record of molar flow rate – taking into account the permeability of the equivalent anchoring of seals : 10^{-10} m/s - ^{129}I – CU1 spent fuels.

7.3.6 Conclusion

Analysing the package failure scenario demonstrates that the premature failure of a few containers or overpacks does not modify the total impact of repository. The effect of a premature release of radionuclides on the delaying and attenuating function is weak as the thermal transient's life is short compared to the total transfer time in the geological barrier. With the present stage of knowledge, effect of temperature on transport and retention parameters in the structures seems limited. Elements whose the partition coefficient diminishes when the temperature increases remain sufficiently sorbed in the structures and geological barrier so as not to contribute to the total impact.

The extremely pessimistic scenario in which all containers and overpacks are defective influences the mass and molar flow rate leaving the Callovo-Oxfordian formation ; this effect is visible more particularly with C waste for which, as opposed to SEN, the influence of the degraded release model leads to a more premature appearance of maximum molar flow rates. Even if all containers are defective, the geological barrier continues to play an important role. Despite the appearance of maximum molar flow rates leaving the Callovo-Oxfordian formation a few tens of thousands to hundreds of thousands of years earlier, this occurrence remains far off (beyond one hundred thousand years). The increase in the dose compared with the SEN is negligible.

The effect of the hydraulic transient was studied in sensitivity, even though, in reference, we do not expect that releases are possible while the cells are under gas pressure. The results reveal that, even when modelled in pessimistic manner, the gasses do not have a significant influence on the transfer pathway through the geological barrier. With the conservative hypotheses adopted, however, overpressure significantly increases the contribution of the transfer pathway via the structures. Nevertheless, this transfer pathway remains negligible in the end. The results of complementary studies have shown that a larger number of defective packages does not lead to a proportional increase in the impact ; packages closest to the edge of the zone contribute the most to impact. Calculations taking into account the dependence of permeability with the saturation of the medium reveal that the hypothesis of an initially resaturated repository is indeed pessimistic.

It shall be noted that the best available knowledge of thermal influence on rock transport was taken into account in these calculations. Yet this field remains far from mastered and large uncertainties weigh on diffusion coefficients and partition coefficients at high temperatures. Since non-sorbed radionuclides remain the primary contributors to impact and the diffusion coefficient experiences only a limited increase, the results of the scenario reveal that we do not expect a major effect from transport in a thermal environment. Nonetheless, the presence of containers and overpacks to manage uncertainties concerning transport in a thermal environment is a design provision adapted to present-day knowledge.

7.4 The « borehole » scenario

This section describes situations resulting from the creation of a borehole in the repository. Consistent with the Basic Safety Rule (RFS) III.2.f, two types of situations are examined :

- situations resulting from the resurfacing of cores, debris, or cuttings and contaminated rock from a borehole. In these situations the radiological impact is immediate. Waste first give rise to an exposure, by external irradiation, one or more of the workers causing the intrusion. Notably, the study of the impact of these situations allows us to address the RFS III.2.f which states that « the exploration borehole crossing the repository with core extraction gives rise to an external exposure that must be assessed » ;
- situations resulting from the abandonment of one or more boreholes. These boreholes, depending on their localisation with regard to the structures, create a partial or total short-circuit in the host formation. Such situations, in which the potential radiological impact is deferred, bring into play part of the radionuclides initially contained in waste and which have migrated from the package to the biosphere through the borehole. The persons potentially exposed in the medium or long term are assimilated with individuals belonging to a hypothetical critical group. The calculated dose is the individual dose.

The repository architecture is similar to that defined in the SEN (See chapter 5), but, for immediate impact situations, the analysis focuses on elements resurfaced through the borehole ; in deferred impact situations it focuses on a repository zone located in the hydraulic influence radius of the borehole. The remaining repository follows normal evolution and is not subject to evaluation as its impact is covered by the SEN findings. It shall suffice to recall this situation for comparison.

The characteristics of the borehole are based on the RFS III.2.f recommendation stating that « the level of technology used is the same as today » ; the borehole diameter corresponds to the order of magnitude of plausible exploration borehole diameters at repository depths, that is to say in the decimetric range.

The uncertainties and failure modes covered by this « borehole » scenario were tackled in chapter 6, both in terms of their presentation (see section 6.2) and the construction of altered situations (see section 6.3). The table below summarises these elements for the record.

Uncertainties considered	Other uncertainties taken into account in the analysis[101]	Components concerned	Comments
Presence of minor structures in the Callovo-Oxfordian → See § 6.2.1.1	None	Callovo-Oxfordian	This uncertainty is mainly covered by a sensitivity study on the permeability of the rock. Moreover, a borehole in an event that would have similar effects (short-circuit of the geological barrier) but more pessimistic.
Borehole → See § 6.2.14.7	None	All components in principle	

Table 7.4-1 « Boreholes » scenario – uncertainties

[101] Which have similar effects or which could worsen the considered effects

7.4.1 Immediate impact situations

This section deals with the definition and results of calculations associated with immediate impact situations.

7.4.1.1 Definition of immediate impact situations

In order to cover all situations, we have elaborated a scenario in which a driller is exposed to radiation from a core taken directly from a waste package.

It should be noted that the repository is not intended to protect an intruder who, voluntarily or accidentally, short-circuits all of the barriers in place, should such an intrusion occur. Such a risk is an inescapable consequence of choosing to concentrate large quantities of waste in a relatively restricted space ; the opposite option, of largely disseminating waste, would lead to the potential exposure of many more people, for individual doses that would not necessarily be any weaker. Also, numerous foreign regulations directly dismiss the protection of an intruder from the requirements applying to a high-level waste repository (see [107] for example).

The dose rate of the core depends on the date of the intrusion and also on the nature of intercepted waste. Search for the conservative case required us to perform numerous evaluations taking different reference package into account.

For spent fuels, calculation was carried out of the FSI reference package, with a higher specific activity than CU2 and CU3 packages with regard to radionuclides responsible for the long term dose rate (^{94}Nb, ^{126}Sn, ^{137}Cs…). Moreover, the 4-assemblies per package configuration makes it possible to intercept 2 assemblies in the same core, whereas the CU2 package does not allow for interception of more than a single assembly.

The package adopted for representing type C waste is the C2 reference package, whose radiological inventory is more pessimistic than the C1 package and equal to the C3 package with regard to radionuclides at the origin of the long-term dose rate. A preliminary study showed that the C4 reference package would not cause any noticeable difference in the results.

The package adopted for type B waste is the B5 reference package, likely to have the strongest dose rate in the very long-term due to its high levels of ^{94}Nb, a gamma emitter with a half live of 20 000 years. However, as it contains the silver-indium-cadmium control rods of powerful reactors, the B1 reference package is very high in ^{108m}Ag, also a gamma emitter radionuclide over a 418 year half life. At the intermediary dates (500 to 1000 years), this package presents a higher dose rate than the B5 package, another reason for which it was adopted.

Finally, the case of intercepting a package containing sources was also addressed. The B8.3 package was adopted. It is a unique package containing radium-based antique objects used for medical purposes until the 1950s and collected by OPRI and Andra. It must be noted that this choice is very pessimistic as the probability of this particular package being touched by the borehole is practically non-existent.

7.4.1.2 Hypotheses considered in calculating the hourly dose rate of cores

The dose rate calculation point is situated 40 cm from the surface of the core.

The cores are meant to be one metre long and have a diameter of 10 cm, a realistic value for a repository at this depth. The length of the active part depends on the volume of waste in the package used and on how the package is disposed of. The specific activity of the active parts comes from the waste and is deducted from the reference inventory of reference package.

Table 7.4-2 summarises the different configurations addressed. The red represents intercepted waste portions.

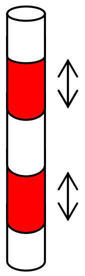
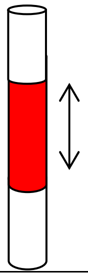
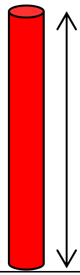
 **CU1 reference package Horizontal disposal h = 21.4 cm**	 **C2 reference package Horizontal disposal *h* = 40 cm**	 **B1, B5, and B8 reference package Vertical disposal h = 100 cm**

Table 7.4-2 Borehole in the repository with resurfacing of waste cores SEA – Calculation configuration modelled for the different reference package addressed

> *Insert 8 Radiation attenuation and transport code used to evaluate the dose received by the driller*
>
> The attenuation and transport code used is the Mercure 6 code, developed by the Atomic Energy Commission. Mercure is a straight line attenuation code that allows for the calculation of a dose rate on one or more points and for one or more gamma emission sources. This code has been extensively used and is widely recognised for its validity.

7.4.1.3 Results of dose rate calculations

The diagram in Figure 7.4-1 shows the results for different dates and different reference package successively addressed. Although the dates inferior to 500 years are not envisaged as they fall within the institutional surveillance phase [2], they were nonetheless dealt with to further demonstrate the marked effect of radioactive decay on the dose rate of waste. The short-lived radionuclides are the most irradiating and we can observe that the dose rate of waste diminishes by several orders of magnitude from the first hundreds of years onward.

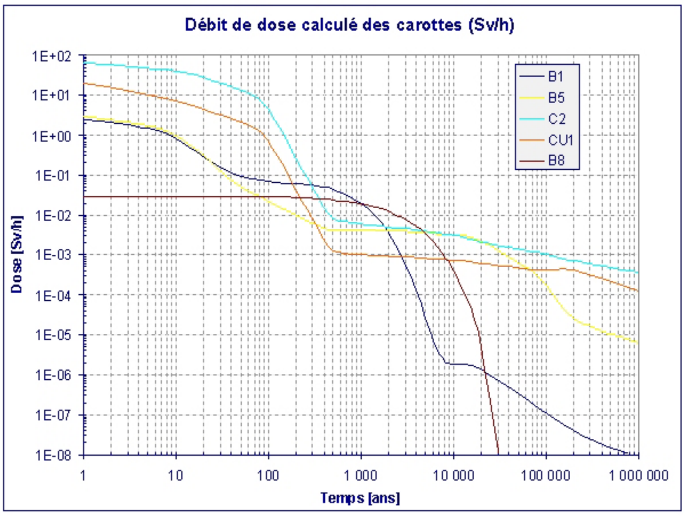


Figure 7.4-1 Borehole in the repository with resurfacing of core waste SEA – Dose rate of core waste – B1, B5, C2, CU1, B8 reference package

- **Calculation of the dose received by the driller**

To confer a conservative nature on the impact assessment, we presume that the intrusion occurs at 500 years, therefore immediately following the institutional surveillance period.

The exposure time considered was 10 minutes long, as the driller does not handle cores for longer than the time necessary for their extraction from the core barrel and for their conditioning. A geologist examining a core after extraction could potentially be exposed for a longer period, but in the spirit of the RFS III.2.f which presumes that « the technological level employed [by future generations] is the same as today », we can acknowledge that a scientist would quickly realise the danger surrounding unusual material resurfaced from a very deep borehole. Even if this were not the case, the results show that contemplating a longer exposure period would not radically change the conclusions of the impact assessment.

In the conditions described, the dose received by the driller for each intercepted package is given in Table 7.4-3 below.

Package intercepted	Hourly dose rate of the core	Dose received by the driller after 10 minutes exposure
B1 package	40 $mSv.h^{-1}$	7 mSv
B8.3 package	25 $mSv.\ h^{-1}$	5 mSv
C2 package	8 $mSv.h^{-1}$	1.3 mSv
B5 package	4 $mSv.h^{-1}$	0.7 mSv
CU1 package	1 $mSv.h^{-1}$	0.2 mSv

Table 7.4-3 Borehole in the repository with resurfacing of waste cores SEA – dose rate received by the driller

7.4.1.4 Conclusion

We note that the calculated impacts are limited to values inferior to ten millisieverts. The most pessimistic situation is extraction of a B1 package core, due to the presence of large quantities of silver-108m. The latter has a relatively short half life, if the intrusion occurred after one thousand years the observed doses for this reference package would be no different from those for C2 packages or spent fuels. Another pessimistic situation is the interception of the B8.3 package, which contains radium-based objects designed specifically to give strong doses. For the other waste the doses received remain moderate, even if exposure lasts more than 10 minutes and so long as direct contact does not last more than one or two hours.

We recall elsewhere (see chapter 6, section 6.2.10.1) that the driller's exposure to gases contained in the repository would expose him, at most, to a dose of 0.03 mSv, due to methane labelled by carbon-14. The possibility of gas overpressure is sufficiently unexpected in this geological context that it would draw the attention of persons in charge of the borehole.

7.4.2 Definition of deferred impact situations

Situations resulting from the abandonment of a borehole in the repository vary depending on hydraulic and geographical location (in the Callovo-Oxfordian formation, in the drifts, in the disposal cells, or in the packages ; in an area where the head gradients are ascending or descending in the Callovo-Oxfordian formation), and its depth, date of occurrence, and evolution over time.

The preliminary analysis of different possible cases led us to consider a limited number of pertinent situations, that are bounding with respect to potential borehole situations.

The instigating event is an exploration borehole completed after the loss of repository memory, meaning at the earliest 500 years after its closure. In order to mobilise the maximal radiological inventory we presumed that the event took place after exactly 500 years.

The borehole is assumed to be permanent (an improbable case, but bounding of impact which can result from several successive drillings in close zones). For each repository zone (B, C, SF) a « reference » place was defined for the borehole. We chose to place it near the packages, in a structure whose diameter was sufficiently large for realistic interception.

For the C waste cells and spent fuel cells, the boreholes used for reference touch an access drift near the cells (knowing that the case of a borehole intercepting a cell would be addressed in the sensitivity studies for FSI spent fuels). The reference situation is presented in Figure 7.4-2 for vitrified C waste. The spent fuels situation is identical to that for C waste, except that the reference concept has a swelling clay buffer.

This situation is quantified for all C waste package and spent fuels, meaning the module holding C0 reference package ; modules containing the C1/C2 reference package (processed together), the borehole is located next to the C2 reference package ; modules with the C3/C4 reference package (processed together), the borehole is located next to the C4 reference package ; CU1 and CU2 spent fuels modules.

For B waste, the borehole meets a repository cell in reference ; such an interception is less improbable as the cells have a larger extension. The calculation case is presented in Figure 7.4-3. The sensitivity study addresses the case of a borehole reaching an access drift. It also addresses cells hosting non-organic reference package not releasing hydrogen (type B1x cells) or bituminised sludge packages (B2 reference package), which are the cells likely to lead to the most pessimistic impact.

The borehole is meant to extend to the Dogger. To remain conservative, also took into account possible water flows from this formation that could flow through the borehole and dilute radionuclides.

The exploration borehole is abandoned and poorly sealed. This borehole can :

- be located near a drinking-water supply pump in the Barrois formation ; the pumping drawdown passes into the borehole. This scenario leads to creating a potential transfer pathway for elements from packages to the biosphere and humans by totally or partially short-circuiting the intermediate geological formations between the repository and the biosphere. This hypothesis is very pessimistic and implies a double accident as having a borehole and pumping at the same site would be a very rare coincidence ;
- lead to creating a potential transfer pathway for elements from the packages to the overlying formations. Once in the overlying formations the radionuclides in this case evolve similarly to the SEN case.

These two examples are presented as alternatives to each other.

It is presumed that only components in the hydraulic influence radius of the borehole have an altered evolution. The other components evolve as in normal evolution and the impact associated with these elements is not re-evaluated.

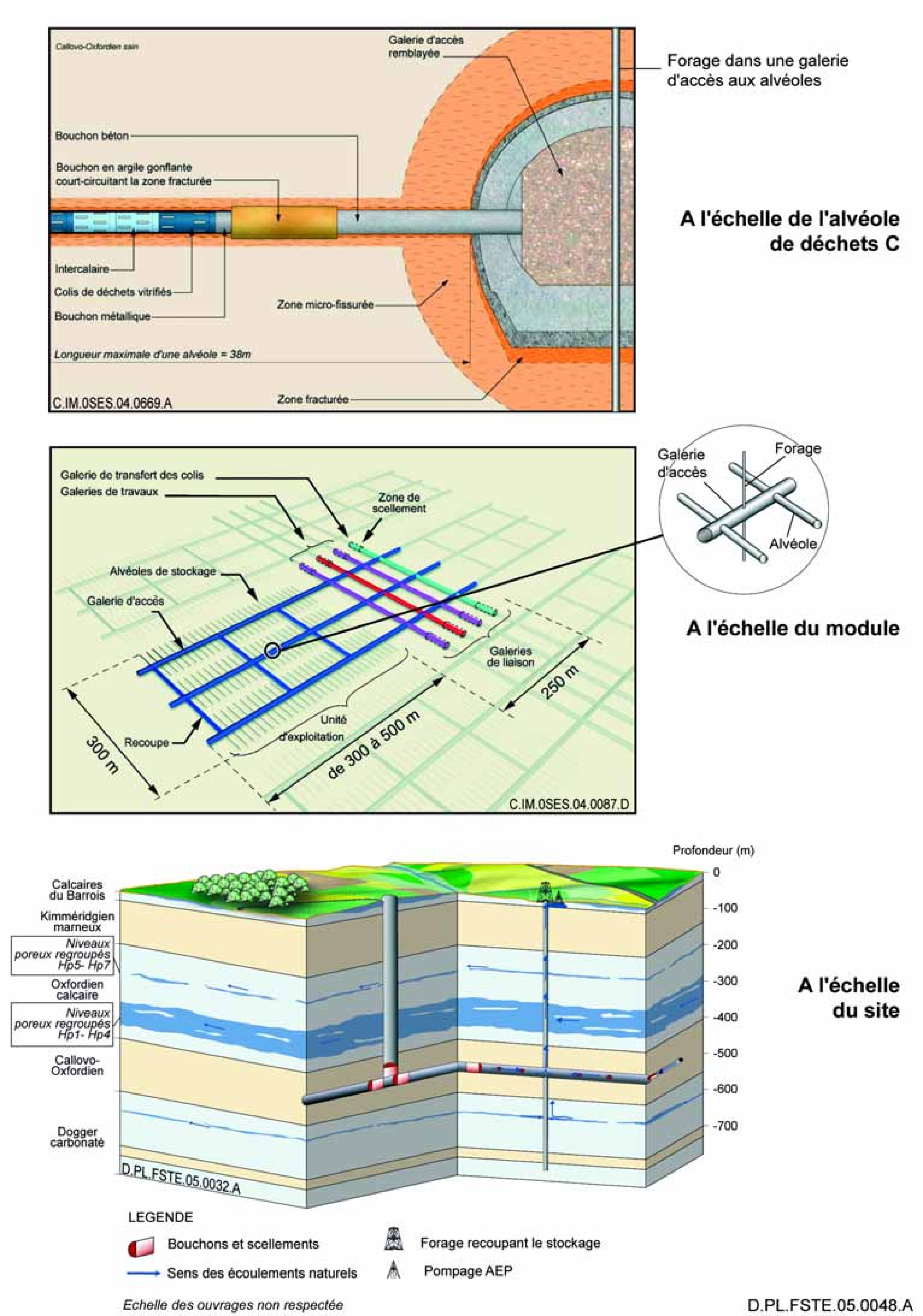


Figure 7.4-2 Abandoned borehole in the repository SEA – reference situation for vitrified C waste package – borehole in an access drift

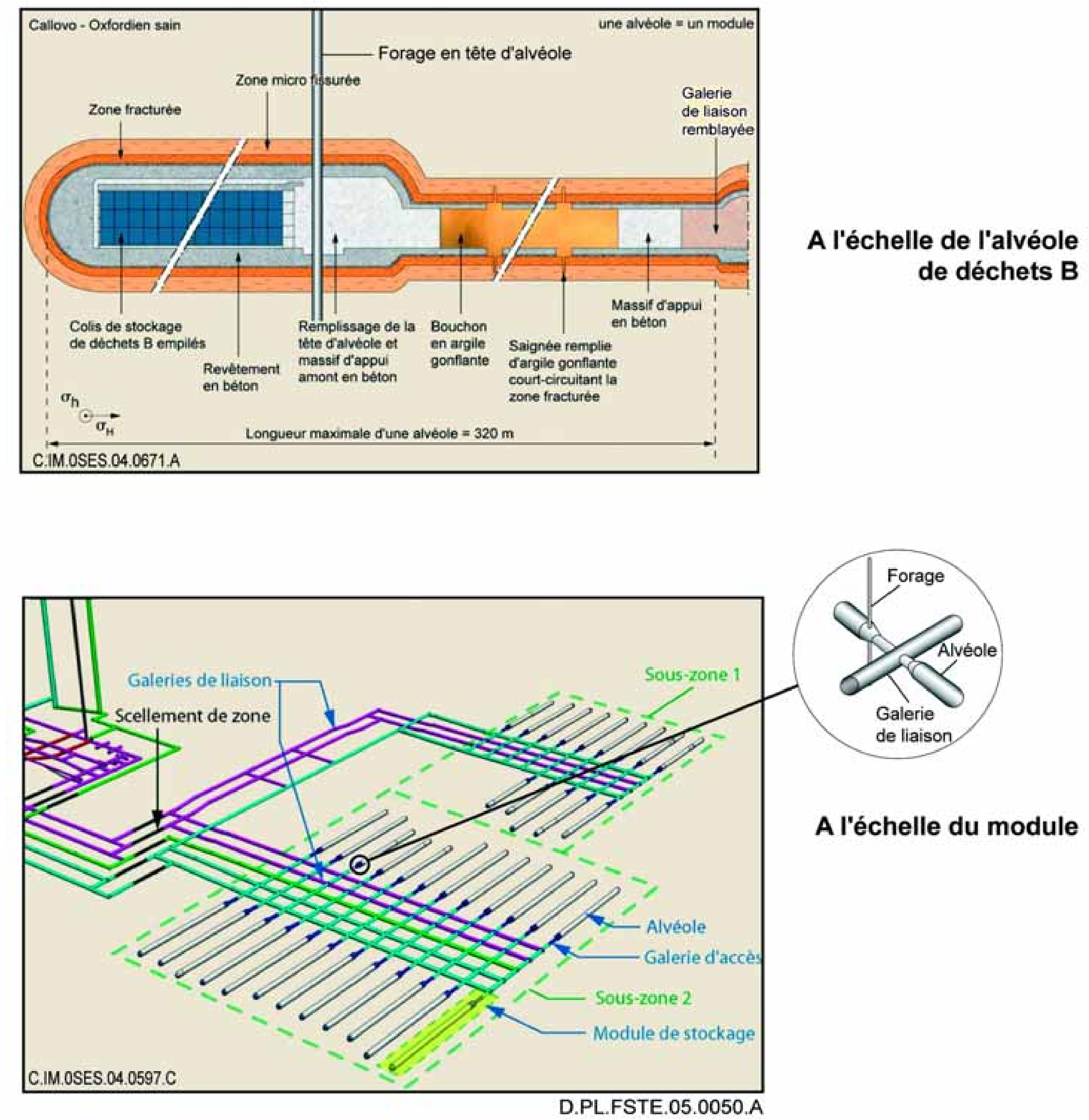


Figure 7.4-3 Abandoned borehole in the repository SEA – reference situation for B waste package – Borehole in a waste cell

7.4.2.1 Discarded situations

It must be noted that certain situations were eliminated as they were considered previously addressed by reference or alternative situations. More particularly :

- borehole situations in the Callovo-Oxfordian formation not intercepting the repository are not quantified as they are covered by borehole situations in the repository structures. Regarding borehole situations in the structures, such a situation would only lead to preserving the supplementary clay buffer between the packages and the borehole, limiting and delaying the radionuclide flow into the biosphere ;
- borehole situations in areas in which the head gradients are descending in the Callovo-Oxfordian formation (to the Dogger) are not addressed. As demonstrated in the SEN (See chapter 5), the adopted transfer model in overlying formations leads, if all is otherwise equal, to impacts significantly superior to the transfer model in the underlying formations. As a result, a descending gradient in the repository would lead to the entry of radionuclides into the Dogger and to a much weaker impact.

7.4.3 Processing the scenario

7.4.3.1 Detailed definition of the reference case

Here we describe in more detail how the borehole is modelled and its effect on the repository. [31], [108].

A cylindrical zone of about ten centimetres in diameter represents the borehole. It has a permeability of 10^{-6} m/s in the Callovo-Oxfordian formation and extends to the Dogger.

Radionuclides reaching the borehole transit through the borehole to the roof of the Callovo-Oxfordian formation. At this level we distinguish two cases :

- the case in which the outlet corresponds to a drinking-water supply pump near the borehole (see Figure 7.4-4. We « inject » all of the molar flow rate leaving the exploration borehole through the roof of the Callovo-Oxfordian formation into the Barrois formation, without taking transfer in the surrounding formations into account. Superficial pumping near the repository in the Barrois formation can only have a very weak flow. To the extent that we use an altered evolution scenario that is maximalist and corresponds to very selective exposures, we deliberately chose a pumping rate that could permanently supply a critical group. This rate is 10 litres per minute ;

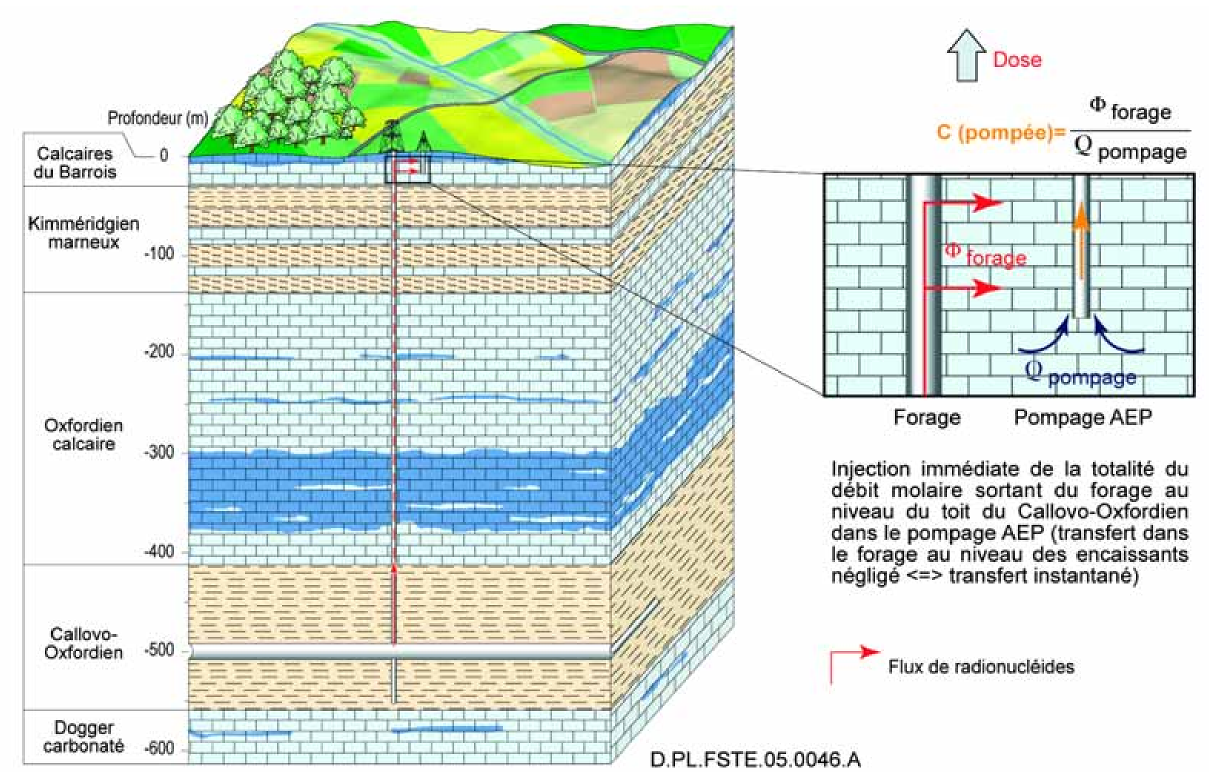


Figure 7.4-4 Abandoned borehole in the repository SEA – radiological impact calculation at the drinking-water supply pump outlet in the Barrois formation (reference outlet)

- the case in which the borehole is presumed abandoned (without a drinking-water supply pump nearby) and in an area in which Oxfordian limestone imposes its head (see Figure 7.4-5). The outlets are coherent with those considered in SEN. To simplify, we shall deal only with the most pessimistic exit, the Saulx outlet, by keeping it in the present day model. The radionuclides from the borehole will migrate quickly and, as such, reach the surroundings sooner ; the present day model should therefore be the most representative. In this case, the configuration used is pessimistic as it ignores the geometry of the repository and places the impact of the borehole closest to the « Saulx » outlet. The molar flow rate leaving the exploration borehole at the roof of the Callovo-Oxfordian formation is fully and immediately injected into the base of the inferior porous levels of the Oxfordian limestone, right of centre in the most western module (module nearest to the Saulx outlet).

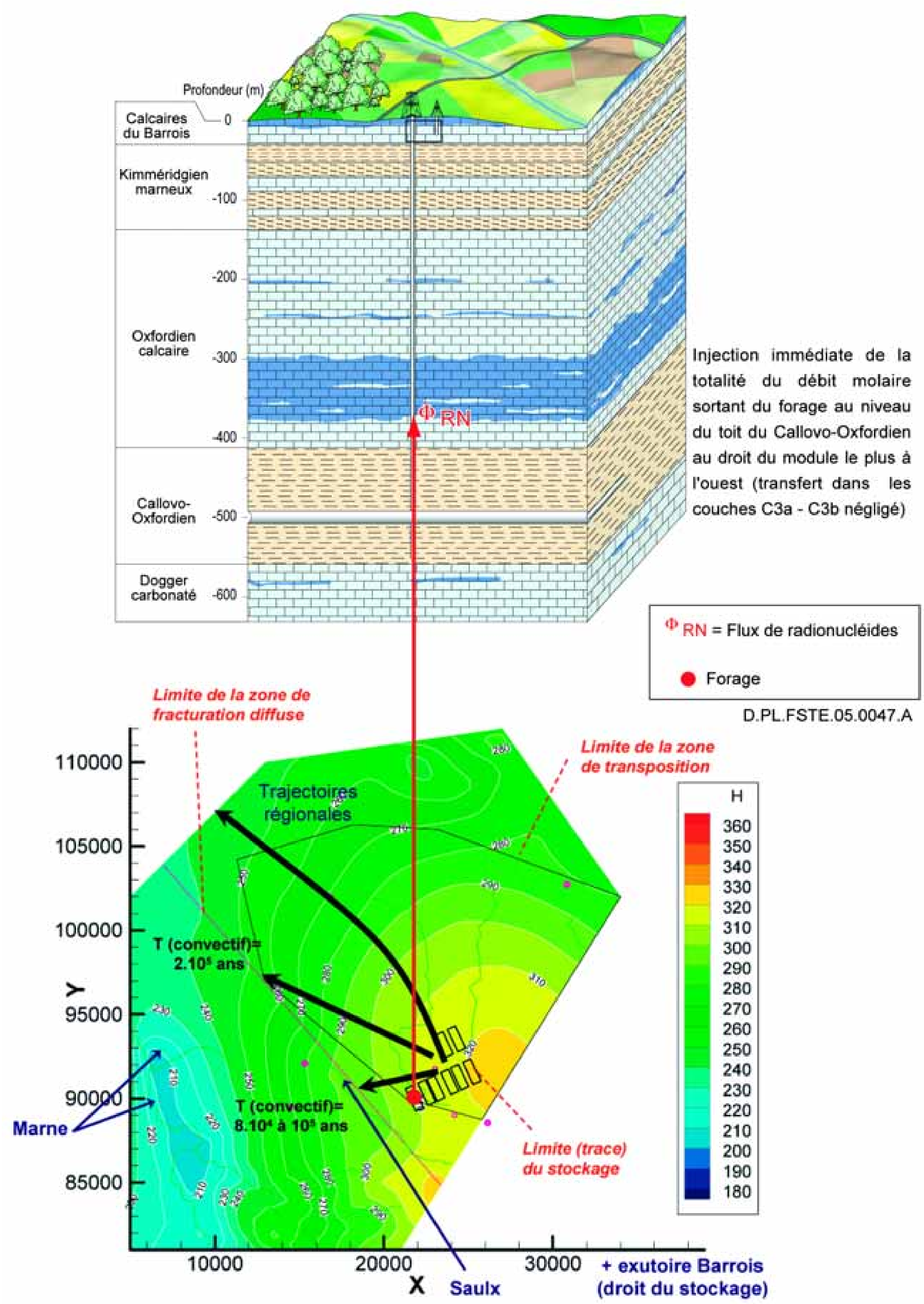


Figure 7.4-5 Abandoned borehole in the repository SEA – radiological impact calculation at the Saulx outlet of the Oxfordian limestone – present day hydrogeological model

Before the presence of a borehole, the repository follows a normal evolution. The vertical head gradients in the Callovo-Oxfordian formation are ascending ; they are 0.2 m/m. The head in the repository is in equilibrium with the one in the Callovo-Oxfordian formation at its depth.

The exploration borehole will locally disturb the repository and cause it to deviate from its normal evolution. In natural conditions for hydraulic head gradients and for transport through the thickness buffer in the geological barrier diffusion is the primary transport phenomenon. In the case of an exploration borehole intercepting a repository drift, this model will be modified : the exploration borehole can create two types of local disturbances :

- a disturbance in the hydraulic head charges. Within the repository, the borehole causes a head flow from the strongest heads to the weakest heads ;
- a disturbance in the chemical concentration gradients. The borehole leads to a virtually empty concentration of radionuclides along a vertical line that can cause a larger migration (horizontal, following a radial trajectory converging on the borehole) than the one proposed in SEN.

The borehole causes a rapid resaturation of the impacted module, if it is not already resaturated. Moreover, before resaturation, the flow is directed from the borehole toward the repository ; therefore, radionuclides can only move toward the surface with great difficulty via the borehole. We find ourselves, then, in a pessimistic situation in which the repository is completely resaturated before the borehole occurs.

From a chemical perspective, we can expect a slight modification in environmental conditions around the impact point if the water arriving through the borehole has clearly different characteristics from those in water from the Callovo-Oxfordian formation. For all cases in which the borehole is dug during the resaturation period, the water arriving will have a chemical composition characteristic of the surrounding formations (either Oxfordian or Dogger formations, depending on the case). This could influence the evolution of cement-based materials. Nevertheless, for B waste cells, the large quantity of concrete would allow for the preservation of alkaline chemistry within the cell. Moreover, C waste cells and spent fuel cells are largely resaturated when the borehole occurs. The convergence of water flowing from the cells toward the borehole should limit the advancement of potentially aggressive chemical species. In the case of spent fuels, the bentonite in the engineered barrier would act as an extra buffer. For the reference calculation then, we assimilate the chemical conditions within the cell to those used in the SEN.

From a mechanical point of view, the diameter of an exploration borehole is sufficiently small that we can ignore the borehole's influence on geomechanical processes. Further, the water rates related to the presence of a borehole, a few metres per year at most, should not cause any erosion problems. We can then consider that the borehole does not have a significant effect on the mechanical workings of the repository structures. Under these conditions, we presume that transport of structures and hydraulic performances are similar to those in the SEN.

Finally, the source term of vitrified waste addressed in SEN is valid only in diffusive conditions (see chapter 6). To take into account the uncertainties related to the borehole's local effects around the glass, which could disturb the balance with the silicon, the vitrified C waste release model used in the borehole scenario is a pessimistic model based on the initial dissolution rate of glass (V_o.S model).

The borehole intervenes in the thermal waste cells at a date at which the temperature's effect may still be non-negligible. Thus, the temperature's effect on permeability is taken into account. The process mode is identical to that in the « seal failure » SEA. The temperature's effect on the other transport parameters (diffusion, sorption) is similar to that in SEN.

7.4.3.2 Sensitivity analyses

A wide range of sensitivity studies has been planned to offer more comprehensive coverage of the effects of a hypothetical borehole. These studies can be classified into different categories.

- **Sensitivity studies on borehole position**

For vitrified waste and spent fuel, we focus on the case of a borehole which does not intercept the adjoining drift, but the cell it self (see Figure 7.4-6). This situation is studied on CU1 spent fuel for illustrative purposes. All other previously described parameters are the same as in the reference case.

It is supposed that the containers will be effective for 10,000 years. On the other hand, the borehole is located in close proximity to a package, offering maximum impact by placing it within the range of low-mobility radionuclides.

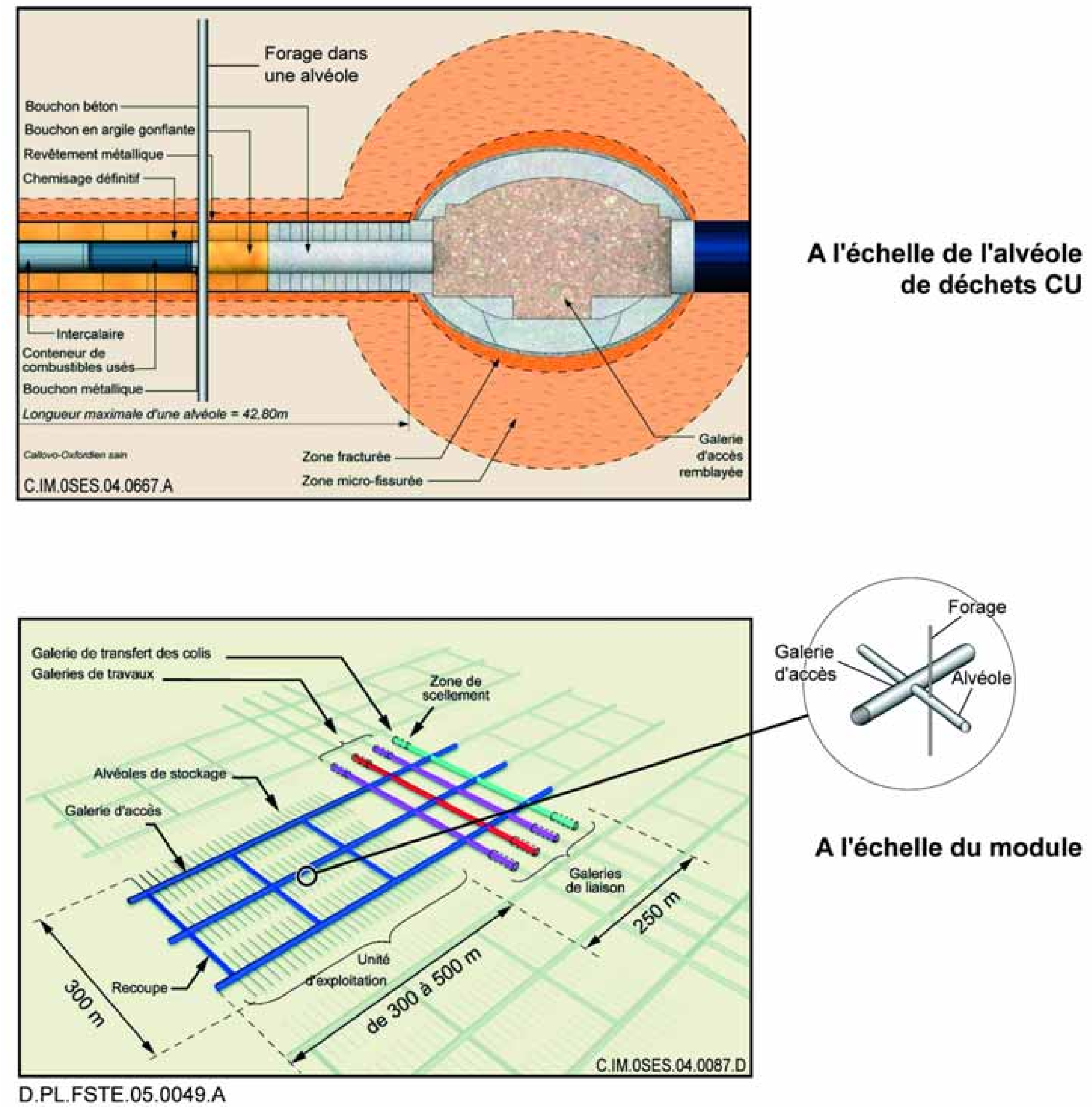


Figure 7.4-6 Abandoned borehole in the disposal SEA – Sensitivity to borehole position (borehole in a spent fuel cell)

The second sensitivity study corresponds to a borehole reaching the Dogger and crossing a secondary connecting drift near a B waste cell. The case of the B1x repository zone is addressed (cell containing the most pessimistic situation in terms of inventory). The situation is represented in Figure 7.4-7.

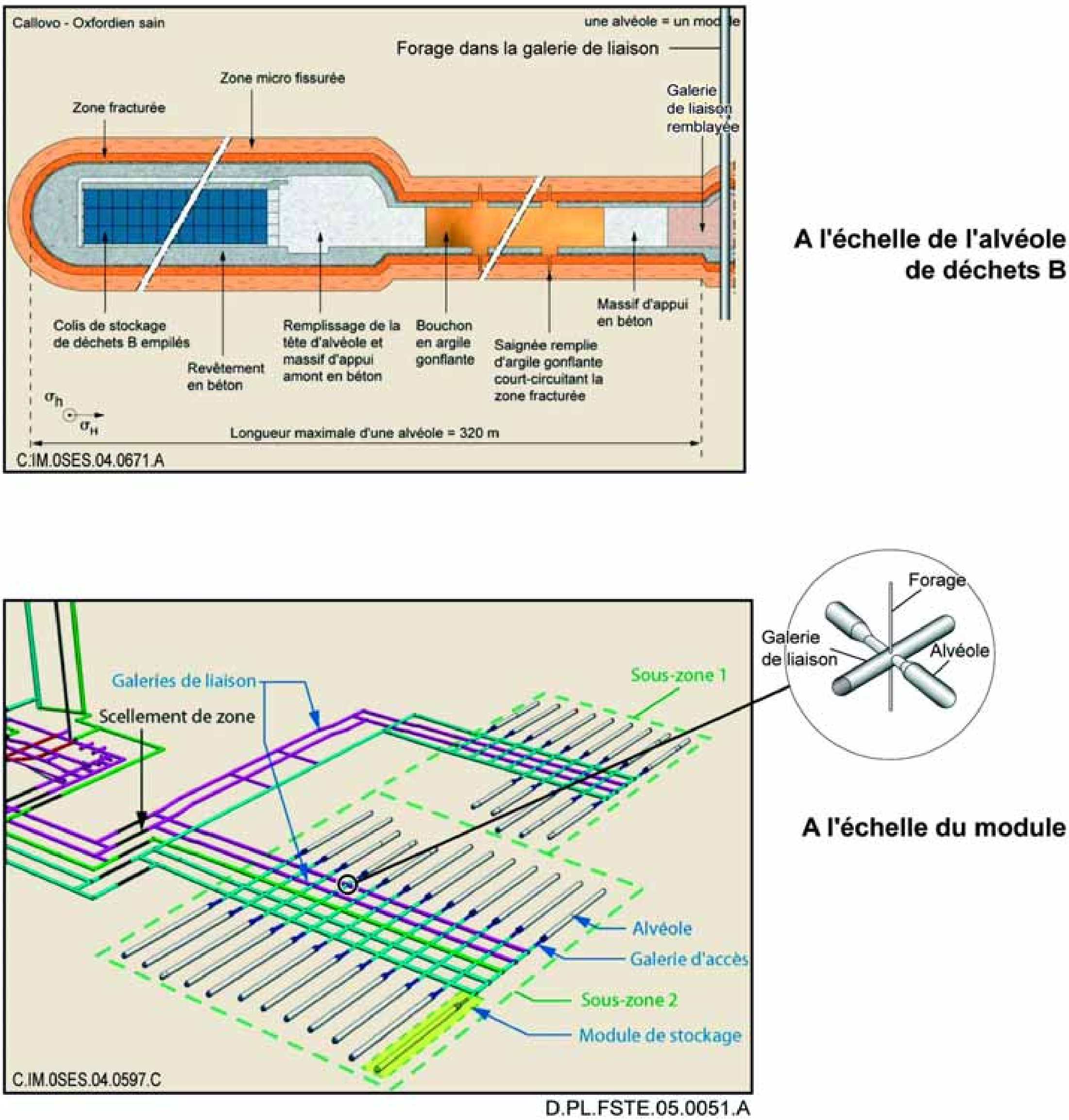


Figure 7.4-7 Abandoned borehole in the disposal SEA – sensitivity to borehole position (borehole in a secondary connection drift)

Finally, the case of a borehole doublet is studied, whereby two boreholes are placed near to each other, on either side in a drift. The aim is to study whether this layout can lead to « U-shaped » advective circulation of water between the boreholes. This situation is considered for the CU1 repository zone. Two boreholes intercept an access drift in a repository sub-zone. Their dimensions are identical to those of the borehole modelled in the reference calculation.

The hydraulic conditions between the two boreholes is governed by the natural hydraulic conditions of the overlying rocks (horizontal hydraulic head gradient) as simulated in the normal evolution scenario. The boundary conditions for the borehole doublet situation are given in Figure 7.4-8 (where H designates hydraulic head in NGF[74] metres, C radionuclide concentration and ∇ gradient).

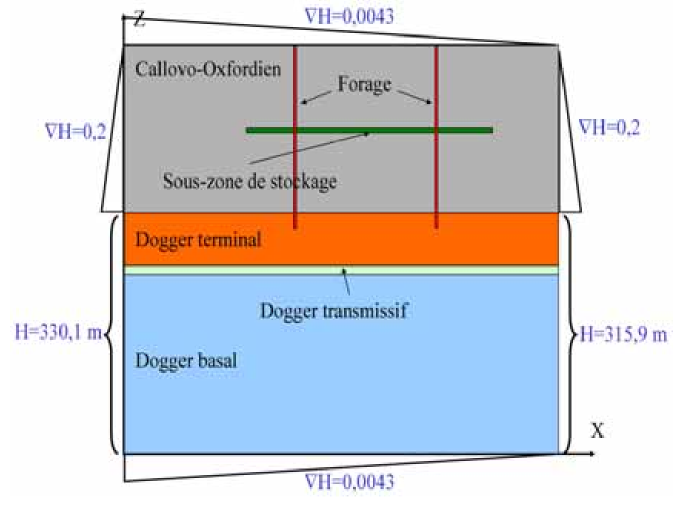
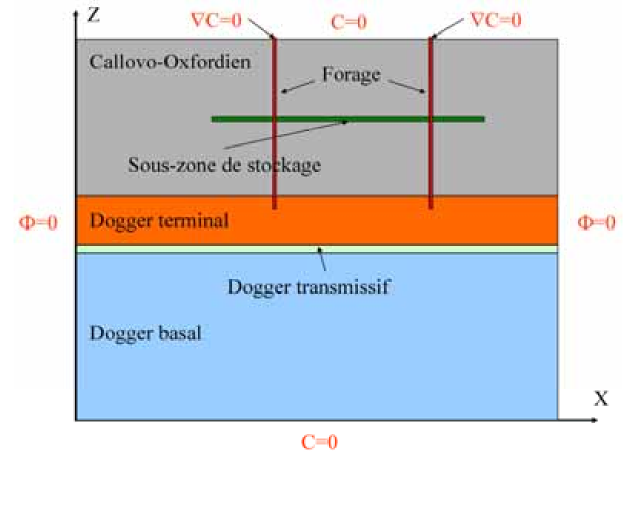
Hydraulic boundary conditions	Transport boundary conditions
	

Figure 7.4-8 Abandoned borehole in the disposal SEA – sensitivity in terms of the borehole doublet – repository model boundary conditions (borehole doublet)

Two situations are investigated (see Figure 7.4-9) :

- the first locates the two boreholes at the ends of a disposal drift to maximise the difference in hydraulic head between the two boreholes (situation 1) ;
- the second maximises the radiological influence of the two boreholes by placing them in the middle of two repository half-modules (situation 2).

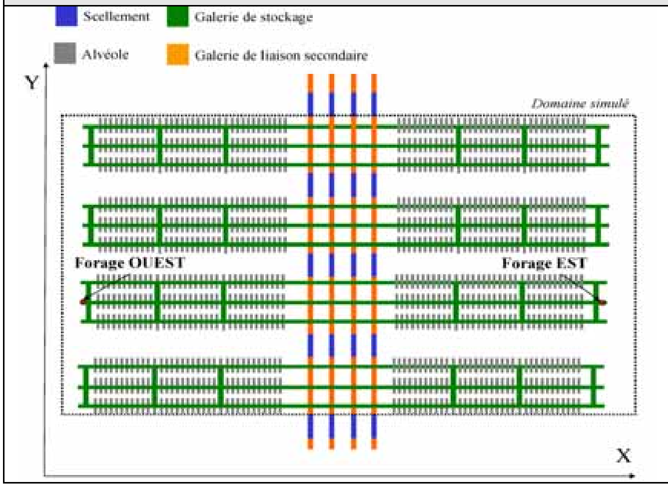
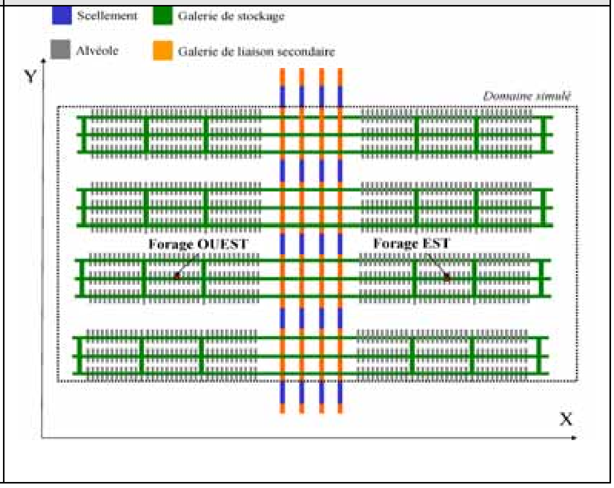
Situation 1 Borehole doublet at the ends of a disposal drift	Situation 2 Borehole doublet in the middle of two repository half-modules
	

Figure 7.4-9 Abandoned borehole in the disposal SEA – sensitivity in terms of the borehole doublet – positioning of borehole doublet

- **Sensitivity studies on the « barrier » role played by structures around the cell**

A sensitivity scenario assesses the potential benefits of introducing a swelling clay engineered barrier for C waste. Indeed, it is worthwhile investigating whether or not this component forms a redundant confinement barrier in the event of short-circuiting of the geological medium. This study is carried out for C2 waste.

- **Sensitivity studies on the model's hydraulic parameters**

Inasmuch as it can be anticipated that the damaged zone is a significant transfer pathway between the cells and the borehole, it also seemed worthwhile to degrade its properties to investigate whether this has an impact on radionuclide transfer.

A sensitivity study is performed in which EDZ parameters are degraded as indicated in Table 7.3-4 and Table 7.4-5. This study is carried out on C2 waste for illustration purposes.

	Phenomenological EDZ reference calculation	« Degraded » EDZ sensitivity calculation
Fractured zone	K = 5.10^{-9}m/s Anions : De_{Anions} = 1.10^{-11} m²/s ω_{Anions} = 0.15 Cations : $De_{Cations}$ = 5.10^{-10} m²/s $\omega_{Cations}$ = 0.20 Phenomenological geochemical retention	K = 10^{-6} m/s Pessimistic coefficient (Dp = De / ω =2.10^{-9} m²/s) No geochemical retention
Microfissured zone	K = 5.10^{-11} m/s Anions : De_{Anions} = 5.10^{-12} m²/s ω_{Anions} = 0.05 Cations : $De_{Cations}$ = $2.5.10^{-10}$ m²/s $\omega_{Cations}$ = 0.18 Phenomenological geochemical retention	K = $5.10^{-9\,m}$/s Anions : De_{Anions} = 1.10^{-11} m²/s ω_{Anions} = 0.04 Cations : $De_{Cations}$ = 5.10^{-10} m²/s $\omega_{Cations}$ = 0.21 Conservative geochemical retention

Table 7.4-4 Borehole SEA – Sensitivity to hydraulic, transfer and chemical retention parameter values in the EDZ – hydraulic and transport parameter values

	Half life [years]	Chemical retention parameters in the microfissured zone			
		Phenomenological geochemical retention		Conservative geochemical retention	
		R [-]	Csat [mol/m^3]	R [-]	Csat [mol/m^3]
^{10}Be	1 600 000	31 900	1,0.10^{-02}	1	10
^{10}Be (deltaT>20)	1 600 000	2003	1,0.10^{-02}	1	10
^{14}C	5 730	5,6	2,3.	1	9
^{36}Cl	302 000	1	soluble	1	Soluble
^{41}Ca	103 000	16	2,3	1	9
^{41}Ca (deltaT>20)	103 000	2,5	2,3	1	9
^{59}Ni	75 000	2 050	5,0.10^{-02}	1 100	1
^{79}Se	65 000	1	5,0.10^{-07}	1	5.10^{-4}
^{93}Zr	1 530 000	12 800	2,0.10^{-05}	1 100	3.10^{-3}
^{93}Mo	3 500	139	1,0.10^{-05}	1	1.10^{-3}
^{93m}Nb	16?4	53 400	2,0.10^{-04}	43 100	2.10^{-3}
^{94}Nb	20 300	53 400	2,0.10^{-04}	43 100	2.10^{-3}
^{99}Tc	213 000	128 000	4,0.10^{-06}	21 900	1.10^{-4}
^{107}Pd	6 500 000	8 950	4,0.10^{-04}	1 750	1.10^{-2}
^{126}Sn	100 000	179 000	1,010^{-05}	75 700	1.10^{-4}
^{129}I	15 700 000	1	soluble	1	Soluble
^{166m}Ho	1 200	639 000	1,0.10^{-04}	54 800	1.10^{-3}

Table 7.4-5 Abandoned borehole in the disposal SEA – Sensitivity to hydraulic, transfer and chemical retention parameter values in the EDZ – chemical retention parameter values in the microfissured zone

- **Sensitivity studies on the model's chemical parameters**

Two studies on sensitivity to model parameters have been carried from the reference situation to take into account uncertainty about the influence of the borehole on chemical and geochemical conditions.

- A first sensitivity study investigating conservative geochemical parameters for bentonite (swelling clay buffer and spent fuel cell plug, C waste cell plug) and module seals in the zone which may be affected by the borehole.
- A second sensitivity study in which conservative parameters are considered (higher fracturing rate) in addition to the already pessimistic release model chosen for vitrified C waste. This study is being carried out on C2 waste for illustration purposes.

Finally, to investigate whether the glass release model significantly influences borehole impact, a counter-situation is studied where the type C2 glasses are governed by a type (V_0.S → V_r) release model.

- **Summary**

Table 7.4-6 below summarises the various sensitivity studies carried out for the borehole scenario.

Sensitivity studies performed	**Reference package included in the sensitivity study**	**Page**
Sensitivity to borehole position		
Sensitivity to borehole position – in a spent fuel cell	CU1	569
Sensitivity to borehole position - in the B waste access drifts	B1x	572
Borehole doublet	CU1	575
Sensitivity on the barriers		
Sensitivity to the C waste concept – variant with clay engineered barrier	C2	579
Sensitivity on the hydraulic parameters		
Sensitivity to the EDZ parameter – « degraded EDZ »	C2	580
Sensitivity on the chemical parameters		
Sensitivity to the engineered barrier parameter (conservative geochemistry)	C2 C4 CU1 CU2	583
Sensitivity to the glasses release model : model $V_0.S \rightarrow V_r$ instead of reference model $V_0.S$	C2	586
Sensitivity to the vitrified waste release model parameter (conservative parameters)	C2 C4	588

Table 7.4-6 Abandoned borehole in the disposal SEA - List of sensitivity studies performed for the « borehole in the disposal » scenario

7.4.4 Effects on the safety functions

As for the normal evolution scenario, analysis of results has two objectives :

- firstly to understand the working of the system in the case of a borehole in the repository. The effectiveness of the functions implemented can be checked using interim indicators ;
- secondly to assess the radiological impact relating to the different reference and sensitivity situations envisaged.

The borehole will degrade or modify performance relating to the three safety functions defined in chapter 3 : « Resist to water circulation », « limiting the release of radionuclides and immobilising them in the disposal » and « delaying and reducing the migration of radionuclides ». Indeed, the scenario as defined in the reference situation totally or partially short-circuits the geological medium (Callovo-Oxfordian and possibly surrounding formations) located between the packages and humans. This leads to local deterioration in performance of the third safety function « delaying and reducing the migration of radionuclides », once the latter have been released by the packages. Moreover, it immediately results in a change in head and flow profiles in the repository zone affected ; it thus « disturbs » performance related to the « resist to water circulation » safety function.

The effects of the borehole on the « limiting the release of radionuclides and immobilising them in the repository » safety function stem only result from disturbances on the packages, and chiefly on the kinetics of radionuclide release by waste. Therefore, to integrate the potentially damaging effects of the borehole on the waste package, a conservative release model for the glasses has been considered in the altered evolution scenario, taking into account the potentially higher water flows circulating in the cell. The other aspects of the function (mainly radionuclide precipitation) are very little affected by the borehole given its minor influence on the chemical conditions in the cell (see section 7.4.3.1). It can however be noted that the sensitivity study planned on the geochemical properties of the fractured EDZ (endowing it with infinite solubility) chiefly covers a potential failure in the near-field immobilisation function.

The presentation thus focuses on the « resist to water circulation », then « delaying and reducing the migration of radionuclides » functions using intermediate indicators which are the same as those used in the normal evolution scenario (see section 5.5).

The results may differ depending on the waste zone, and radionuclide-specific analysis. Whenever required, the results are separated by waste type and by radionuclide. The special default case of the CU1 fuel zone and iodine-129, providing the strongest impact, is used when the aim is simply to illustrate a result.

7.4.4.1 Effect on the « resist to water circulation » function

- **Effect on hydraulic head**

The borehole will cause partial or total short-circuiting of the host formation by creating a high-permeability zone in its location from the top of the Callovo-Oxfordian to the terminal Dogger (see Figure 7.4-10).

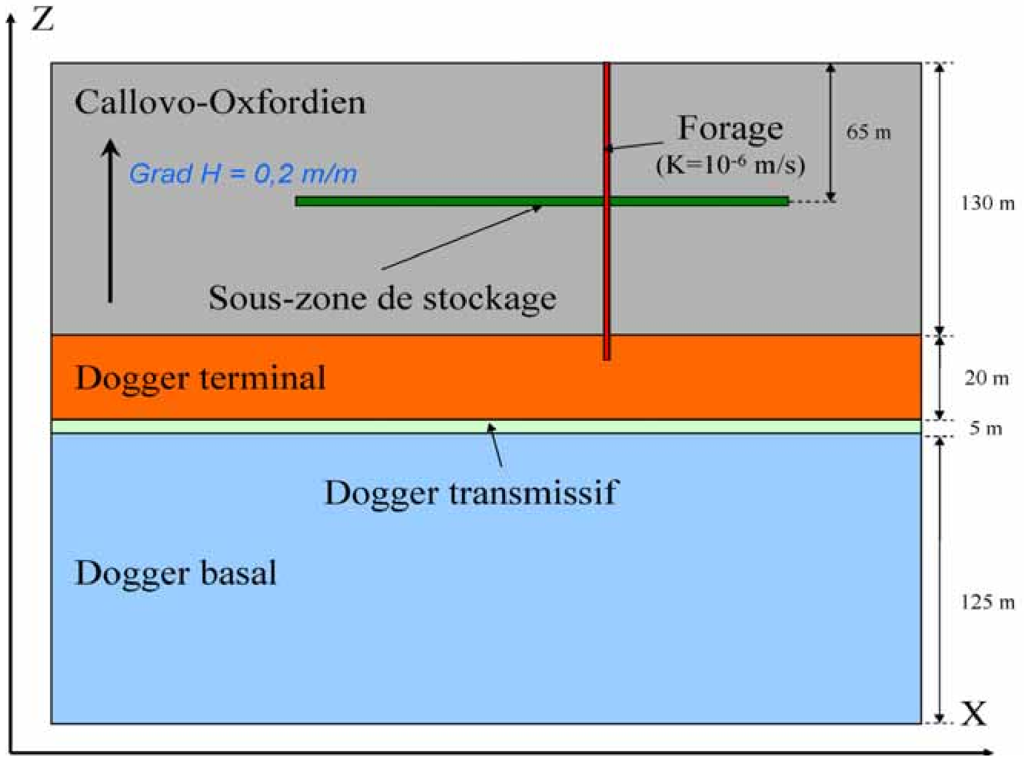


Figure 7.4-10 SEA for abandoned borehole in the repository – illustration of the borehole in the geological medium

Firstly, head in the borehole itself is considered, then the way in which it propagates inside the structures.

Hydraulic head in the borehole chiefly depends on its permeability, geometry (diameter in particular) and the aptitude of the surrounding system to supply water. In light of this, the high permeability of the borehole and the limited water supply from the surrounding medium cause locally reduced hydraulic head in the borehole. The calculation of hydraulic head highlights that, in equilibrium (See Figure 7.4-11) :

- in the lower part of the Callovo-Oxfordian where the water supply is limited, hydraulic head in the borehole is lower than in the Callovo-Oxfordian ;
- at the level of the repository, the head profile in the borehole is nearly constant. It ranges from hydraulic head below that of the Callovo-Oxfordian at the base of the repository to hydraulic head in equilibrium with that of the Callovo-Oxfordian in the upper part. This mainly stems from the repository's capacity to supply water to the borehole. Although it is nearly constant, hydraulic head in the borehole remains lower than or equal to (at the top of the repository) that of the Callovo-Oxfordian ;
- in the upper part of the Callovo-Oxfordian, the head profile in the borehole remains in equilibrium with that of the Callovo-Oxfordian.

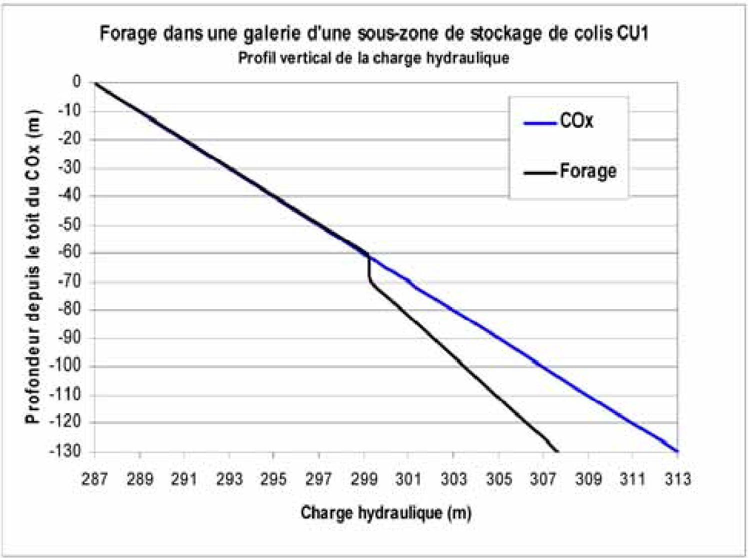


Figure 7.4-11 Abandoned borehole in the disposal SEA – hydraulic head profile in the borehole

The flows converge towards the borehole in the zones affected. At first, these flows tend to decrease hydraulic head in the repository until steady-state hydraulic conditions are established. The head in equilibrium in the repository module(s) affected by the borehole will then depend on :

- head in the borehole which will tend to propagate or balance out with that of the repository module affected ;
- the aptitude of the affected module(s) to supply the borehole hydraulically ; the weaker the water flows captured from the Callovo-Oxfordian by the modules, the more the head in the modules will be reduced ;
- seal efficiency ; their fairly low permeability with respect to the drifts limits the spread of head loss (see below) ;
- the hydraulic characteristics of each of the modules.

The water supply to low extent modules is more limited due to constraints in terms of the volume of water available in the module(s) and subsequently the quantity of water that the Callovo-Oxfordian can supply at structure walls. The water flow that the repository can supply is shown in Table 7.4-7 for each of the repository zones. They contribute to the total flow in a proportion of between 17 % and 31 % depending on the reference package and the location of the borehole. For the rest, the water drained by the borehole mainly comes from the Dogger.

Reference package or reference cell*	Borehole location	Water flow drained by the borehole at 100,000 years (m^3/yr)[102]{					Disposal / Total
		Dogger[103]	Callovo-Oxfordian under the disposal	Callovo-Oxfordian over the disposal	*Disposal*	Total	
Reference package C0	Drift	0.120	0.003	0.000	*0.025*	0.148	17 %
Reference package C2	Drift	0.111	0.002	0.000	*0.0500*	0.163	31 %
Reference package C4	Drift	0.111	0.002	0.000	*0.0506*	0.163	31 %
CU1 spent fuel	Drift	0.110	0.002	0.000	*0.0495*	0.162	31 %
CU2 spent fuel	Drift	0.111	0.002	0.000	*0.0490*	0.162	30 %
Cell B1x*	Cell	0.118	0.003	0.000	*0.024*	0.145	17 %
Reference package B2	Cell	0.115	0.003	0.000	*0.028*	0.146	19 %
* The B1x type cells receive non-organic reference package not giving off hydrogen : reference package B1, B5, B6 (excluding B5.1 and B6.4)							

Table 7.4-7 Abandoned borehole in the disposal SEA – contribution from model components to the water flow captured by the borehole

Seals limit the propagation of head loss from the module with the borehole to modules furthest away from the borehole due to their permeability. In the case of CU1 spent fuel, the calculated head is constant in the drifts of the affected module drifts (Figure 7.4-12) ; on the other hand, it rises at the seal (in the region of 0.15 metres over 50 metres of seals). Indeed, the hydraulic efficiency of the seal limits the propagation of head loss from the module with the borehole to adjacent modules, which in turn limits water circulation from adjacent modules to the affected module. Thus, the hydraulic influence of the borehole is even more limited if the modules are away from the borehole. Preliminary calculations have shown that hydraulic head variations over a seal fall by a factor of 2 on each seal passage. Thus, the repository modules furthest away from the borehole are less affected from a hydraulic point of view and the module intercepted by the borehole forms the main water supply, with approximately a third of the total flow in the case of C packages or spent fuel packages and 95 % in the case of B packages.

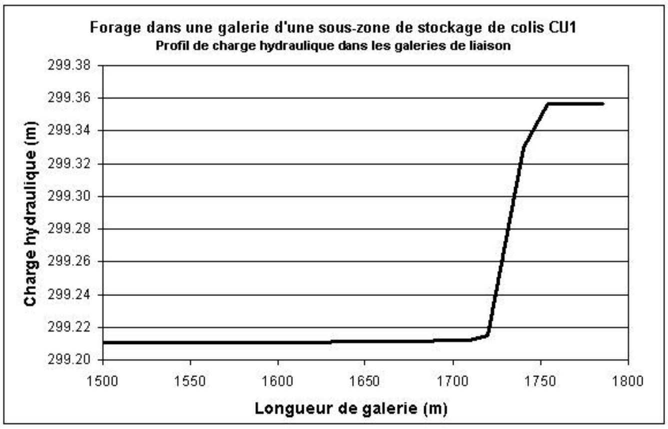


Figure 7.4-12 Abandoned borehole in the disposal SEA – hydraulic head profile in the drifts (right-hand end represents a seal passage)

[102] Example date
[103] A reminder that in the impact calculation, the water flows from the Dogger are not taken into account to dilute the radionuclide flows.

- **Effect on water circulation**

The water flows converge from the repository modules towards the borehole where the head is weaker. The Darcy velocity and Péclet number calculated at different points in the disposal show :

- that the conditions in the drift system remain diffusive (or in advective, diffusive co-dominance) with Péclet numbers between 0.2 and 1 in the drift. These were calculated by taking an average drift length of 250 metres, porosity of 40 % in the backfills and a molecular diffusion coefficient equal to 2.10^{-9} m^2/s. Figure 7.4-13 gives Darcy velocity values and Péclet numbers at different locations in the half-module affected or at the module seals. Also included (shaded rectangles) are the zones located around the half module affected, whose hydraulic influence has been modelled ;
- the conditions within the disposal cells remain diffusive for all the packages in the cell (see Figure 7.4-14). For example, the Péclet number in the structures was evaluated for CU1 spent fuel. It gives a value of 0.006 at the cell plug. The system becomes slightly advective in the fractured zone where permeability is highest. The Péclet number is 3 at the head of the cell in this zone. Moreover, transfer kinetics remain very limited with Darcy velocities of 10^{-4} m/yr maximum. For B waste, the Péclet number calculated at the cell head is in the region of 1.

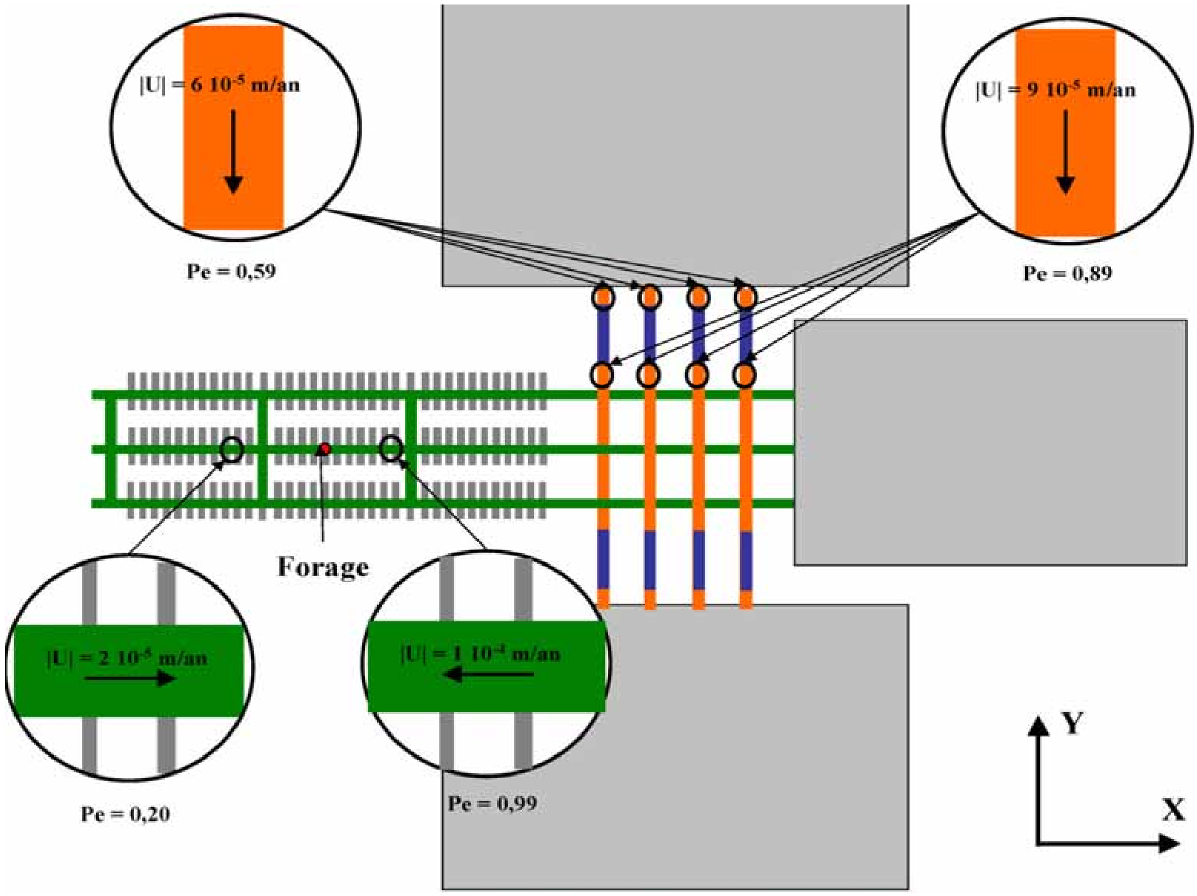


Figure 7.4-13 Abandoned borehole in the disposal SEA – Darcy velocity and Péclet number in the drifts in a CU1 package sub-zone

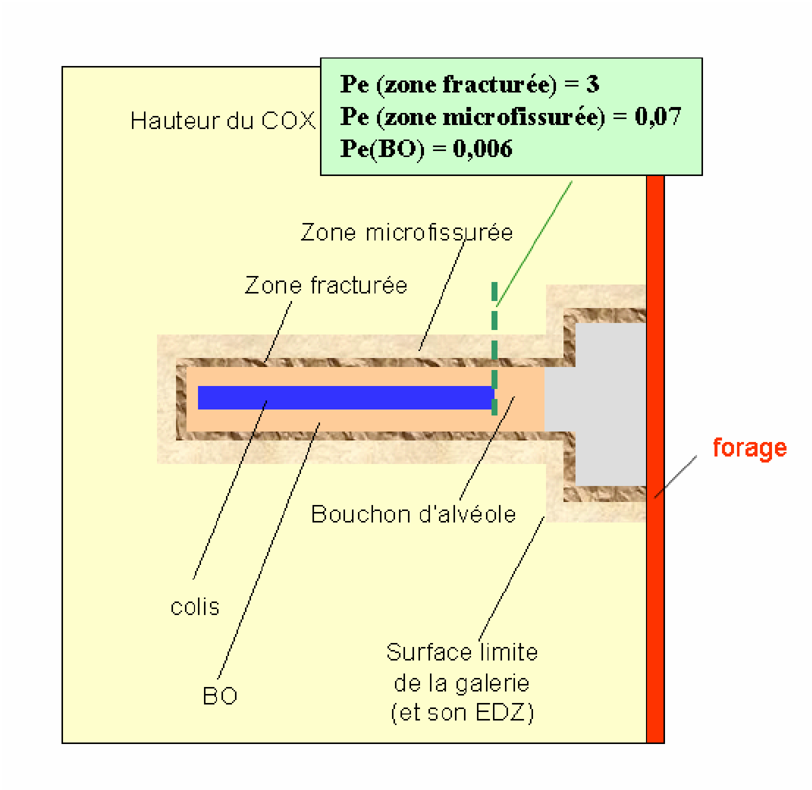


Figure 7.4-14 Abandoned borehole in the disposal SEA – Péclet number in the CU1 spent fuel cells

7.4.4.2 Effects on the « Delaying and attenuating radionuclide migration » function

The preceding analysis revealed that the system remains diffusive in the disposal modules in spite of the presence of the borehole. The effect of the borehole on radionuclide confinement should also be evaluated, particularly the aptitude of the repository and geological medium to delay and attenuate radionuclide flow.

Once released by the packages, radionuclides migrate into the cell body or plug. A fraction of the disposal cell's total activity reaches the borehole after passing through the structures or into the media between the packages and the borehole (see Figure 7.4-15). The activity reaching the borehole depends on :

- initial activity in the packages ;
- distribution between transfer pathways at the level of the cell itself. A fraction of this activity follows a horizontal trajectory (pathway 2) before reaching the access drifts, the other follows a vertical trajectory (pathway 1) before reaching and remaining in the Callovo-Oxfordian. Distribution between these two transfer pathways depends, among others, on the radionuclides' ability to be sorbed into the host formation and structures. Since the transport regime is diffusive (or codominantly diffusive-corrective), distribution between transfer pathways is comparable to that observed in SEN ;

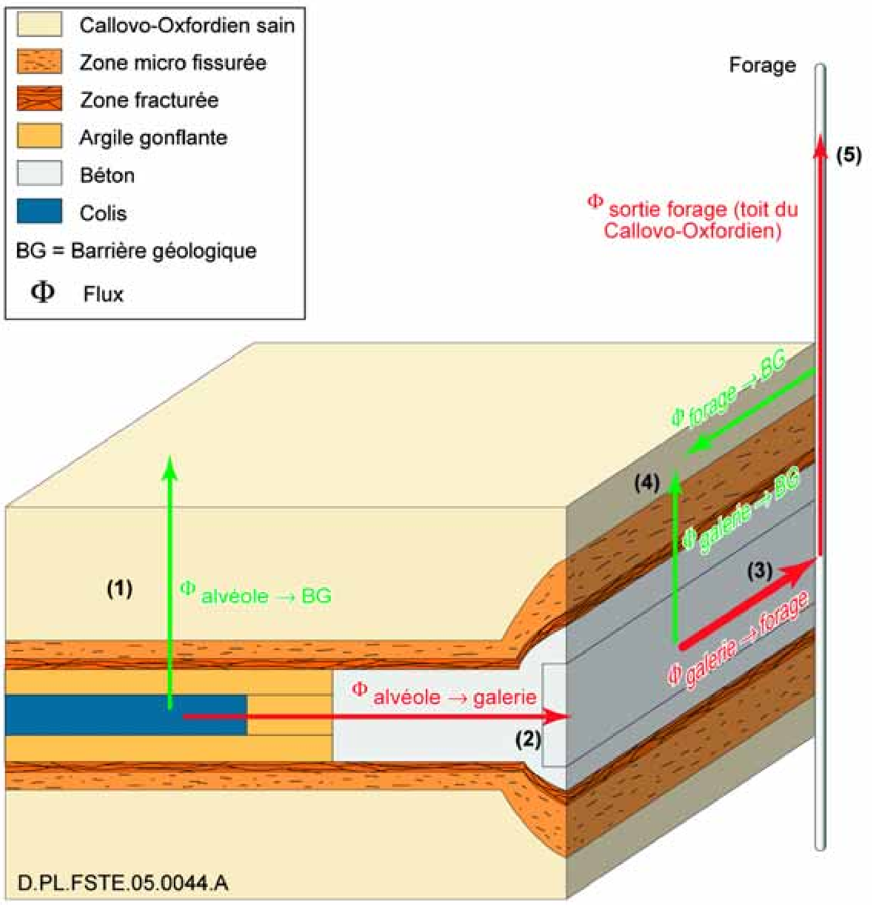


Figure 7.4-15 SEA borehole abandoned in the repository – transfer pathways for C waste or spent fuel cells

- the ability of the radionuclides which reach the access drifts to migrate and remain in them before reaching the borehole. Indeed, the flow of radionuclides which migrate into the drifts may be attenuated between the cell exit the borehole because the radionuclides may reach the Callovo-Oxfordian and migrate into it by diffusion or undergo radioactive decay before reaching the borehole. Because of this, the importance of attenuating flow between a cell and the borehole increases with the distance of the cell from the borehole. For reasons of simplification, the analysis presents the total amount of activity reaching the borehole, without splitting it into the respective contributions from each cell according to its position relative to the borehole. This is a mean value comparable to the total initial inventory disposed of in a zone, subzone, or module, or to activity in a cell or released from a cell into the access drifts. In fact, a limited number of cells contribute to supplying the borehole.

- **Spent fuel**

The reference situation consists of a borehole in an access drift contiguous to spent fuel cells (See Figure 7.4-15. Therefore, radionuclides reaching the borehole pass through the body of the clay engineered barrier, the plug or the Callovo-Oxfordian before reaching the access drifts, then the borehole. Only the horizontal fraction of this activity which follows on horizontal trajectory in the structures or associated damaged zones is likely to reach the borehole.

Radionuclide transfer follows different pathways depending on the chemical behaviour of the elements concerned. It has therefore been decided to present the results, distinguishing four different types of radionuclide :

- long-lived soluble and non- or very weakly sorbed radionuclides (^{129}I, ^{36}Cl, ^{41}Ca) ;
- moderately long-lived, weakly sorbed radionuclides and long-lived, moderately or strongly sorbed radionuclides (^{135}Cs, ^{14}C, ^{107}Pd ^{59}Ni, ^{93}Zr, ^{93}Mo) ;
- radionuclides which are not sorbed but precipitated in bentonite or argillite (in this case only ^{79}Se) ;
- moderate or long-lived, very strongly sorbed radionuclides (^{126}Sn, ^{99}Tc, ^{166m} Ho, ^{94}Nb).

long-lived soluble and non- or very weakly sorbed radionuclides

Distribution of the mass emitted by the packages in the cell near-field, over one million years, is detailed in Table 7.4-8 according to whether the radionuclides enter the adjacent drift or the geological medium.

Radionuclides	Mass leaving the field close to the cell (« entering and crossing the first few metres of Callovo-Oxfordian » + « entering the drift »)	Distribution of release from cells	
		Mass « entering and crossing the first few metres of sound Callovo-Oxfordian » (pathway 1) [104]	Mass « entering the drift » (pathway 2)[104]
^{129}I	100.00 %	42 %	58 %
^{36}Cl	86.07 %	40 %	46.07 %
^{41}Ca	97.45 %	57.99 %	39.46 %

Table 7.4-8 SEA borehole abandoned in the repository –distribution of transfer pathways in the cell near-field – mass emitted by different constituents over 1 million years - C waste package – long-lived radionuclides, soluble and non, or very weakly sorbed

It is noted that these radionuclides can leave the cell. The mass attenuation in the near field is very low, or zero. Iodine and chlorine are specified, since the mass of these emitted by the packages reaches the drifts preferentially via migration by diffusion through the structures between the packages and the access drift. It should be noted that conditions at the boundaries chosen relative to the horizontal transfer pathway (zero concentration in line with the plug) tend to favour horizontal migration for all radionuclides. Calcium 41, which is partly sorbed by the host formation, migrated along this pathway by preference.

The mass entering the drifts and reaching the borehole is highly dependent on the position of the cells relative to the borehole. The further the cells are from the borehole the less they contribute to borehole impact. Therefore, the ratio of the mass reaching the borehole compared with the total initial mass of a subzone is extremely small. These ratios are given in Table 7.4-9. Percentages are given relative to the initial subzone inventory for the repository concerned by the borehole.

Radionuclides	Mass % « entering the drift » (pathway 2)[104]	Mass % reaching the borehole via the drifts (pathway 3)[104]	Mass % reaching the top of the Callovo-Oxfordian via the borehole (pathway 5)[104]
^{129}I	58 %	1.455 %	1.42 %
^{36}Cl	46.07 %	0.82 %	0.79 %
^{41}Ca	39.46 %	0.025 %	0.005 %

Table 7.4-9 SEA borehole abandoned in the repository – transfer pathways from repository structures to the borehole – mass percent leaving the borehole over 1 million years. - long-lived soluble and non - or very weakly sorbed radionuclides

[104] See Figure 7.4-15

The fraction of iodine 129 activity reaching the borehole is greater, on a scale of one million years, than that of chlorine 36, because the latter undergoes radioactive decay. Indeed, it has been shown that the mass of iodine 129 captured by the borehole corresponds to the mass initially present in 15 cells, whereas for chlorine 36, it corresponds to the mass initially present in 8 cells.

The fraction of calcium 41 which reaches the borehole is more limited than that of iodine 129 and chlorine 36. This difference results from the effective coefficient of diffusion of cations in the host formation, which is stronger than that of anions ; it tends to promote radionuclide transfer from the drifts to the Callovo-Oxfordian rather than to the borehole.

Figure 7.4-16 shows molar flow rates for iodine 129 at various points between the package and the borehole, as well as distribution between the transfer pathways.

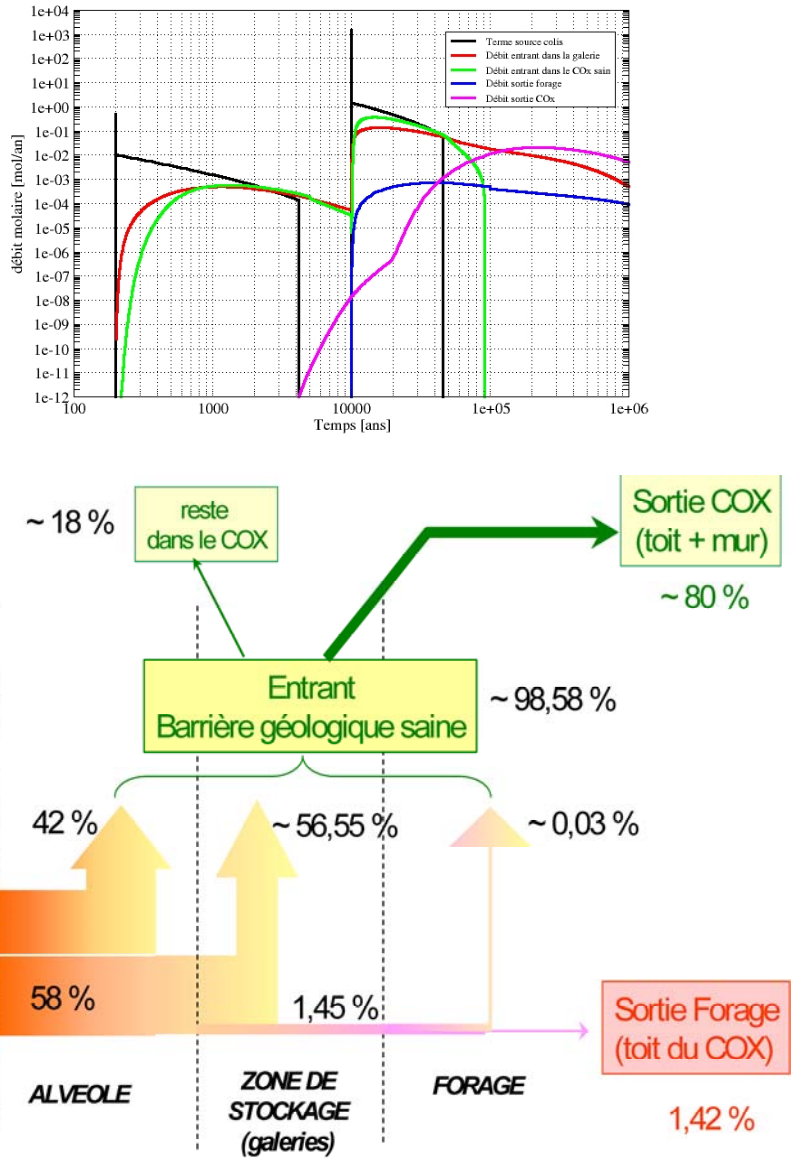


Figure 7.4-16 SEA borehole abandoned in the repository – History of molar flow rates in the near field and details of distribution of - ^{129}– CU1 reference packages - I transfer pathways

Moderately long-lived, weakly sorbed radionuclides and long-lived, moderately or strongly sorbed radionuclides

In the cells, moderately long-lived radionuclides which are very weakly or weakly sorbed (^{14}C, ^{93}Mo) and long-lived radionuclides which are moderately or strongly sorbed (^{135}Cs, ^{59}Ni, ^{107}Pd, ^{93}Zr) into the Callovo-Oxfordian and structures, profit from radioactive decay and/or remain essentially confined (at least 80 %) in the near field. Indeed, the mass emitted by the structures (See Table 7.4-10) varies from less than one percent (0.15 % for ^{93}Mo) to several tens of percents (14.46 % for ^{135}Cs) ; their sorption allows for radioactive decay and diffusive spreading through the cell near-field and the signal of the labile fraction emitted by the package is attenuated at the cell exit. Moreover, the radionuclides migrate nearly towards the host formation.

Radionuclides	Mass leaving the field close to the cell (« entering and crossing the first few metres of Callovo-Oxfordian » + « entering the drift »)	Distribution of release from cells	
		Mass « entering and crossing the first few metres of sound Callovo-Oxfordian » (pathway 1)[104]	Mass « entering the drift » (pathway 2)[104]
^{135}Cs	14.46 %	9.55 %	4.91 %
^{14}C	12.17 %	8.19 %	3.98 %
^{107}Pd	6.32 %	3.22 %	3.10 %
^{59}Ni	2.33 %	1.18 %	1.15 %
^{93}Zr	0.63 %	0.45 %	0.18 %
^{93}Mo	0.15 %	0.00012 %[105]	0.15328 %

Table 7.4-10 SEA borehole abandoned in the repository –distribution of transfer pathways in the field close to the cells – mass emitted by different constituents over 1 million years - CUI waste package – moderately long-lived, weakly sorbed radionuclides and long-lived, moderately or strongly sorbed radionuclides

Moderately long-lived, weakly sorbed anions (^{14}C, ^{93}Mo) only reach the borehole in limited quantities, because their capacity for sorption into the host formation promotes their migration from the drifts to the host formation and they also profit from radioactive decay when the cells are a certain distance from the borehole.

Radionuclides which are strongly sorbed into the host formation (^{135}Cs, ^{107}Pd, ^{59}Ni, ^{93}Zr, ^{126}Sn, ^{99}Tc, ^{166m}Ho, ^{94}Nb) migrate by diffusion from the access drifts to the geological medium, strongly limiting the mass of radionuclides which reach the borehole. For those radionuclides which do reach the borehole, their capacity for sorption by the host formation again promotes their return into the rock from the borehole and the molar flow rate leaving the borehole at the top of the shaft is extremely slow, or zero (See Table 7.4-11).

[105] although ^{93}Mo migrates preferentially towards the host formation, the mass crossing the first metres of Callovo-Oxfordian appears smaller than the one entering the drift. This is due to radioactive decay while crossing the first few metres of argilites.

Radionuclides	Mass % « entering the drift » (pathway 2)[104]	Mass % reaching the borehole via the drifts (pathway 3)[104]	Mass % reaching the top of the Callovo-Oxfordian via the borehole (pathway 5)[104]
^{14}C	3.98 %	0.0085 %	0.0075 %
^{93}Mo	0.15328 %	0.000165 %	0.000125 %
^{135}Cs	4.91 %	0.00046 %	0.00003 %
^{107}Pd	3.10 %	0.00026 %	0.00001 %
^{59}Ni	1.15 %	0.0001 %	0.000005 %
^{93}Zr	0.18 %	0.000015 %	Zero

Table 7.4-11 SEA borehole abandoned in the repository – transfer pathways from repository structures to the borehole – mass percent leaving the borehole over 1 million years. - moderately long-lived, weakly sorbed radionuclides and long-lived, moderately or strongly sorbed radionuclides

Figure 7.4-17 illustrates the migration of nickel 59.

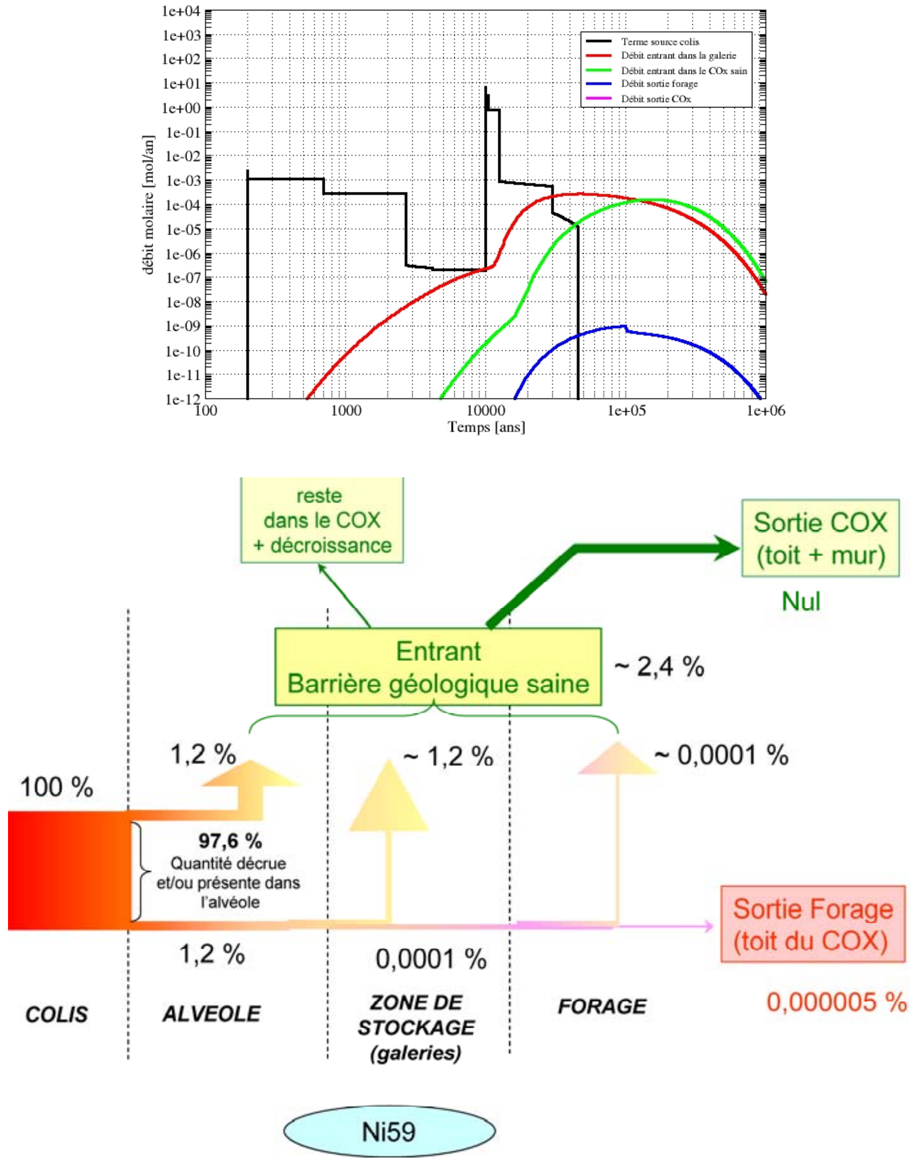


Figure 7.4-17 SEA borehole abandoned in the repository – History of molar flow rates in the near field and details of distribution of - Ni59 -- CU1 reference packages - transfer pathways

Radionuclides which are not sorbed but precipitated in bentonite or argillite

^{79}Se, is not sorbed but precipitates in the near field, displaying a highly attenuated rate of activity at the cell outlets. This attenuation is reflected in the mass released from the cell near-field over the period of analysis (1 million years). The transfer of selenium to the borehole is also limited (See Table 7.4-12 and Table 7.4-13).

Radionuclides	Mass leaving the field close to the cell (« entering and crossing the first few metres of Callovo-Oxfordian » + « entering the drift »)	Distribution of release from cells	
		Mass « entering and crossing the first few metres of sound Callovo-Oxfordian » (pathway 1)[104]	Mass « entering the drift » (pathway 2)[104]
^{79}Se	0.564 %	0.3347 %	0.2290 %

Table 7.4-12 SEA borehole abandoned in the repository –distribution of transfer pathways in the field close to the cells – mass emitted by different constituents over 1 million years - CUI waste package – radionuclides not sorbed but precipitated in bentonite or argillite

Radionuclides	Mass % « entering the drift » (pathway 2)[104]	Mass % reaching the borehole via the drifts (pathway 3)[104]	Mass % reaching the top of the Callovo-Oxfordian via the borehole (pathway 5)[104]
^{79}Se	0,229 %	0,00235 %	0,00225 %

Table 7.4-13 SEA borehole abandoned in the repository – transfer pathways from repository structures to the borehole – mass percent leaving the borehole over 1 million years. - radionuclides which are not sorbed but precipitated in bentonite or argillite

Moderately or long-lived, very strongly sorbed radionuclides

These radionuclides remain almost totally confined in the near field of the cell and/or profit from radioactive decay. For ^{126}Sn and ^{99}Tc, a very small quantity of the mass diffuses into the structures and reaches drifts adjacent to the cell plug. However, these elements are strongly sorbed into the host formation, and the latter continues to trap them on their way to the drifts, and the small amount of ^{99}Tc and ^{126}Sn which reaches the drifts cannot reach the borehole (See Table 7.4-14 and Table 7.4-15)

Radionuclides	Mass leaving the field close to the cell (« entering and crossing the first few metres of Callovo-Oxfordian » + « entering the drift »)	Distribution of release from cells	
		Mass « entering and crossing the first few metres of sound Callovo-Oxfordian » (pathway 1)[104]	Mass « entering the drift » (pathway 2)[104]
^{129}Sn	0.00075 %	Zero	0.00075 %
^{99}Tc	0.00009 %	Zero	0.00009 %
^{166m}Ho	Zero	Zero	Zero
^{94}Nb	Zero	Zero	Zero

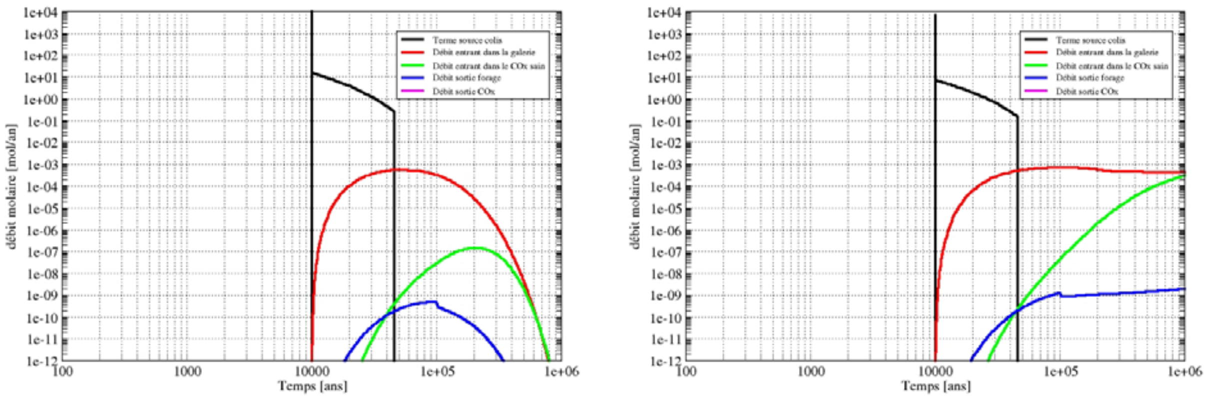
Table 7.4-14 SEA borehole abandoned in the repository –distribution of transfer pathways in the field close to the cells – mass emitted by different constituents over 1 million years - CUI waste package – moderately long-lived, very strongly sorbed radionuclides

Radionuclides	Mass % « entering the drift » (pathway 2)[104]	Mass % reaching the borehole via the drifts (pathway 3)[104]	Mass % reaching the top of the Callovo-Oxfordian via the borehole (pathway 5)[104]
^{126}Sn	0,00075 %	Zero	Not applicable
^{99}Tc	0,00009 %	Zero	Not applicable
^{166m}Ho	Zero	Not applicable	Not applicable
^{94}Nb	Zero	Not applicable	Not applicable

Table 7.4-15 SEA borehole abandoned in the repository – transfer pathways from repository structures to the borehole – mass percent leaving the borehole over 1 million years. - CUI spent fuel - moderate or long-lived, very strongly sorbed radionuclides

Actinides

Due to their high retention in the engineered structures and in the Callovo-Oxfordian, the actinides reaching the borehole are characterised by considerably reduced molar flow rates and late maxima. Only ^{240}Pu and ^{239}Pu, the quantity of which is relatively high, have a maximum molar flow rate out of the borehole towards the 100,000-year mark due to the fact that they are relatively less sorbed into the bentonite. The maximum molar flow rate of ^{239}Pu out of the borehole is around 5.10^{-9} mol/year after 90,000 years as illustrated in Figure 7.4-18. The maximum molar flow rates of the other actinides occur after a million years. Figure 7.4-18 also shows the case of ^{237}Np, the retardation coefficient of which in bentonite is identical to that of plutonium but does not benefit from a radioactive decay.



Molar flow rate history of ^{239}Pu (Kd = 1 m^3/kg in bentonite)

Molar flow rate history of ^{237}Np (Kd = 1 m^3/kg in bentonite)

Figure 7.4-18 Borehole scenario in an access drift – Molar flow rate histories – ^{239}Pu and ^{237}Np – Reference package CU1

Conclusion relative to the effect of the borehole for spent fuel

Analysis revealed that the hydraulic effect of the borehole is limited and that transport inside the repository, in both the cell and drifts, remains diffusive or is codominantly diffusive advective, with extremely slow transfer kinetics.

The radiological impact of the borehole is also strongly limited :

- partly because a fraction of the activity migrates by diffusion into the Callovo-Oxfordian and remains there. This concerns radionuclides which are sorbed into the host formation ;

- partly because the fraction of activity which migrates to the access drifts does not systematically reach the borehole. Indeed, particularly due to the efficacy of the sealing systems, cells located in adjacent modules contribute very little to the borehole's impact over the scale of time studied. In the same way, cells in the affected modules contribute less to impact, the further they are from the borehole. Furthermore, the fraction of activity captured by the borehole is specific to each radionuclide. It is higher for soluble, non-sorbed, long-lived radionuclides.

Thus, in the case of iodine (the radionuclide which has the greatest effect), the borehole drains a level of activity corresponding to a mean initial activity equal to 1.45 % of that of the repository subzone and less than 0.4 % of the total for the CU1 spent fuel zone. The effect of the borehole therefore remains extremely limited on the scale of the repository.

It should be noted that, for elements which are not sorbed into the structures, the maximum flow rate of activity in the borehole is recorded several thousand years after rupture of the spent fuel container. It should be noted that, for the most sorbed elements, the maximum flow rate is delayed and the influence of the labile fraction flattened.

- **C waste**

The results for C waste are similar to those for spent fuel, apart from the date of release and transfer of radionuclides, which takes place earlier (rupture of the overpack occurs at 4 000 years instead of 10 000 years for SF). The absence of a swelling clay buffer and a fractured zone in line with the plug would have little or no influence on the result in so far as :

- the argillite in the near-field for C cells plays a similar part to that of the clay engineered barrier for SF cells ;
- the influence of the microfissured zone, creating a direct short-circuit between packages and drift, is limited because transport remains diffusive in the body of the cell (or codominant diffusive-advective at the head of the cell).

Table 7.4-16 shows distribution between transfer pathways at cell level as far as the borehole for the main radionuclides studied.

Radionuclide	Distribution of release from cells		Mass % reaching the borehole via the drifts (pathway 3)[104]	Mass % reaching the top of the Callovo-Oxfordian via the borehole (pathway 5)[104]
	Mass « entering and crossing the first few metres of sound Callovo-Oxfordian » (pathway 1)[104]	Mass « entering the drift » (pathway 2)[104]		
^{129}I	39,82%	60,18%	1,28%	1,24%
^{36}Cl	51,23%	46,50%	0,78%	0,75%
^{41}Ca	49.62%	32.23%	Zero	Not applicable
^{135}Cs	7.06%	3.60%	0.00033%	0.000015%
^{14}C	4.15%	2.36%	0.0050%	0.0045%
^{107}Pd	1.95%	1.95%	0.00016%	0.0000043%
^{59}Ni	0.93%	0.85%	0.000077%	0.0000026%
^{93}Zr	0.62%	0.38%	0.000036%	0.0000013%
^{93}Mo	0.0000041%	0.00070%	0.00000092%	0.00000076%
^{79}Se	0.16%	0.11%	0.0011%	0.0010%
^{126}Sn	Zero	0,000059%	Zero	Not applicable
^{99}Tc	Zero	0,000031%	Zero	Not applicable
^{166m}Ho	Zero	Zero	Not applicable	Not applicable
^{94}Nb	Zero	Zero	Not applicable	Not applicable

Percentages are given relative to the initial total subzone inventory for the repository concerned by the borehole.

Table 7.4-16 SEA borehole abandoned in the repository –distribution of transfer pathways – mass emitted by different constituents of the disposal system and borehole over 1 million years – C2 waste package – reference calculations

- **B waste**

Calculations have been performed for B waste that are non organic and do note release hydrogen (B1x cells) and for bituminous waste. The following results deal with B1x cells, which have the higher impact (results are very similar for bituminous waste). All doses at the « Saulx » outlet are nevertheless presented in Table 7.4-19.

The reference situation studied differs from the one used for spent fuel or C waste in so far as the borehole is drilled directly at the head of the B waste cell (See Figure 7.4-19). This situation is particularly pessimistic because there is no plug or host formation between the packages and the borehole to delay and attenuate radionuclide flow. Furthermore, the primary containers of B waste are assumed not to be tight from the time the repository is closed, and the resaturation phase is neglected, so that a fraction of the activity is already in the cell outside the primary packages when the borehole occurs. The impact associated with the borehole may therefore appear not long after its occurence, in particular for radionuclides not sorbed into the concrete.

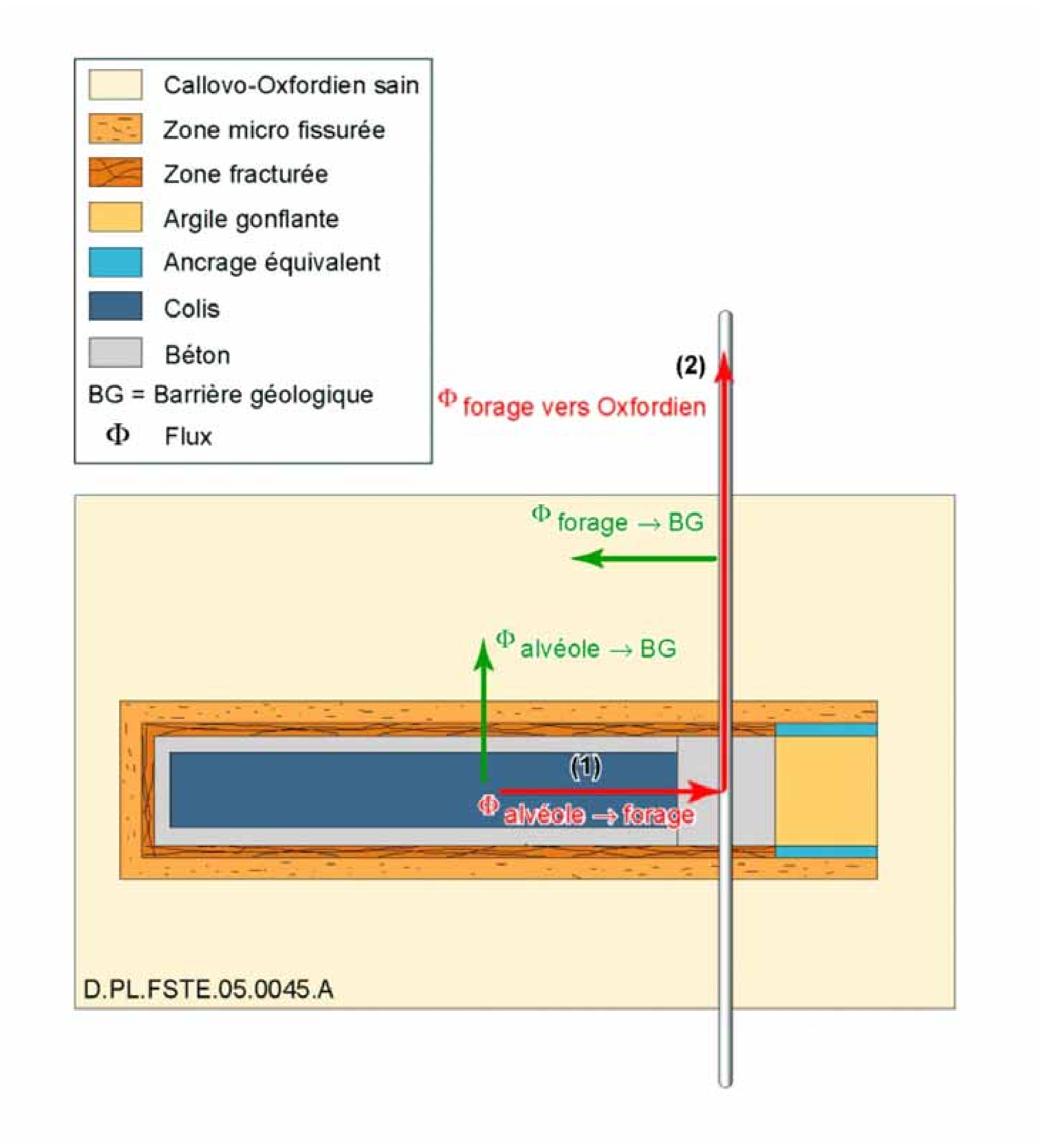


Figure 7.4-19 SEA borehole abandoned in the repository – Organisation of transfer pathways in the cell near-field area – B waste cell

The Peclet number at the head of B waste cells is around 1, so transport in the cell remains predominantly diffusive. The kinetics are still faster than for C waste or spent fuel cells.

In so far as the packages are not expected to be tight and can release activity as soon as the repository is closed, the impact of this situation has been evaluated for radionuclides with a half life greater than 30 years, the minimum half life being chosen arbitrarily with respect to the date of occurrence of the borehole and so as to include radionuclides such as cesium 137 and strontium 90 in relatively significant quantities in the packages ; this choice led to the selection of 54 radionuclides. It emerges that the fraction of activity released by the borehole, relative to the total initial activity of the module, is very limited or even zero for most radionuclides. This attenuation results from :

- on the one hand, the fact that most of the radionuclides are strongly sorbed into the concrete. This is increased for short-lived radionuclides which have already undergone radioactive decay during the 500 years before the borehole was made ;
- and on the other hand because most of the radionuclides migrate by diffusion into the Callovo-Oxfordian where they are sorbed.

Therefore, of all the radionuclides studied, only 27 of the 54 record mass released from the borehole which is not totally attenuated relative to the total initial mass in the cell. Furthermore, these radionuclides include some with extremely slow molar flow rates because they are present only in small quantities initially in the waste package.

Table 7.4-17 gives a list of radionuclides likely to display flow rates greater than zero on release from the borehole.

The fraction of activity released by the borehole, relative to the total initial activity of the cell, is given in Table 7.4-18 (as an example of waste package disposed of in cells holding non-organic packages not emitting hydrogen - B1x type cells).

Radionuclide	Period	Delay in the geological barrier [-]	Solubility [mol/m^3]	Concrete delay [-]
^{129}I	15 700 000	1	soluble	8
^{26}Al	720 000	1	3.10^{-04}	1
^{36}Cl	302 000	1	soluble	1
^{53}Mn	3 700 000	1	soluble	1
^{40}K	1 280 000 000	14	soluble	1
^{10}Be	1 600 000	31 900	soluble	8
^{87}Rb	48 000 000 000	30	soluble	71
^{194}Hg*	520	1	2.10^{-1}	1
^{93}Mo	3 500	1	7.10^{-04}	1
^{79}Se	65 000	1	$1,3.10^{-02}$	701
^{41}Ca	103 000	16	20	3 500
^{14}C	5 730	6	1.10^{-02}	3 500
^{135}Cs	2 300 000	Langmuir's Iso.	soluble	70
^{107}Pd	6 500 000	8 950	1.10^{-02}	4 200
^{59}Ni	75 000	2 050	$2,3.10^{-04}$	14 000
^{92}Nb	35 000 000	53 400	$2,4.10^{-04}$	70 000
^{93m}Nb	16,4	53 400	$2,4.10^{-04}$	70 000
^{94}Nb	20 3000	53 400	$2,4.10^{-04}$	70 000
^{99}Tc	213 000	128 000	soluble	210 000
^{93}Zr	1 530 000	12 8000	6.10^{-03}	280 000
* : radionuclide for which sorption and precipitation values are selected at arbitrarily to be pessimistic, owing to lack of data				

Table 7.4-17 SEA borehole abandoned in the repository – Main characteristics of radionuclide retention

Radionuclide	Period [years]	Release from the cell to the borehole	Release to the Oxfordian (borehole outlet)
^{129}I	15 700 000	22,400 %	21,70 %
^{26}Al	720 000	20,060 %	19,33 %
^{36}Cl	302 000	16,610 %	15,89 %
^{53}Mn	3 700 000	3,400 %	0,910 %
^{40}K	1 280 000 000	1,700 %	0,400 %
^{10}Be	1 600 000	1,4869 %	0,3515 %
^{87}Rb	48 000 000 000	0,5200 %	0,0350 %
^{194}Hg[106]	520	0,4000 %	0,3500 %
^{79}Se	65 000	0,2100 %	0,2000 %
^{93}Mo	3 500	0,2100 %	0,1600 %
^{41}Ca	103 000	0,0330 %	0,00500 %
^{107}Pd	6 500 000	0,0720 %	0,00100 %
^{135}Cs	2 300 000	0,0870 %	0,00074 %
^{14}C	5 730	0,000120 %	0,0001000 %
^{92}Nb	35 000 000	0,00210 %	0,0014000 %
^{93m}Nb	16,4	0,00130 %	0,0005000 %
^{93}Zr	1 530 000	0,0000810 %	0,0000130 %
^{59}Ni	75 000	0,00120 %	0,0000110 %
^{99}Tc	213 000	0,00400 %	0,0000020 %
^{94}Nb	20 300	0,0000030 %	0,0000012 %

Table 7.4-18 SEA borehole abandoned in the repository - Fraction of activity reaching the borehole, relative to the total initial activity of the cell

It was noted that :

- only three radionuclides (^{129}I, ^{26}Al, ^{36}Cl) have a mass released from the borehole representing 15 to 20 % of the total mass initially in the cell. These long-lived radionuclides are not precipitated in the concrete ; they are not sorbed by the host formation[107] and little or not at all by the concrete. Therefore, in so far as migration takes place essentially by diffusion, the distribution of mass between the horizontal and vertical transfer pathways leading to the Callovo-Oxfordian depends closely on the exchange areas developed at the cell / geological barrier interface. For those radionuclides which are little or not at all sorbed into the concrete, the maximum molar flow rate leaving the borehole is recorded at the time the borehole is drilled (500 years) or just afterwards for iodine which is weakly sorbed (See Figure 7.4-20 and Figure 7.4-21) ;

[106] Mercury 194 has a relatively short half-life (520 years). This radionuclide considered not to be sorbed into the concrete or argillite due to lack of data, displays a level of attenuation on release from the borehole of about 0.35 %, this being closely linked to the radioactive decay of ^{194}Hg in the waste package. This result (incomplete attenuation) is not representative with respect to pessimistic hypotheses applied for its geochemical characteristics.

[107] Note : for ^{26}Al, this hypothesis results from an arbitrary choise maide because of missing chemical data on it's behaviour.

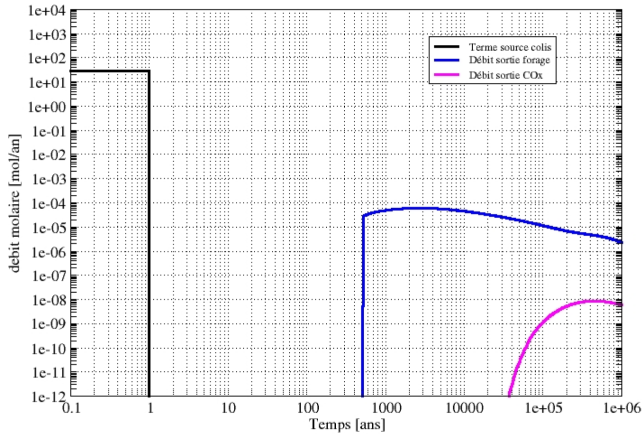


Figure 7.4-20 SEA borehole abandoned in a B waste cell – History of molar flow rates in the near field - B1x - ^{129}I packages

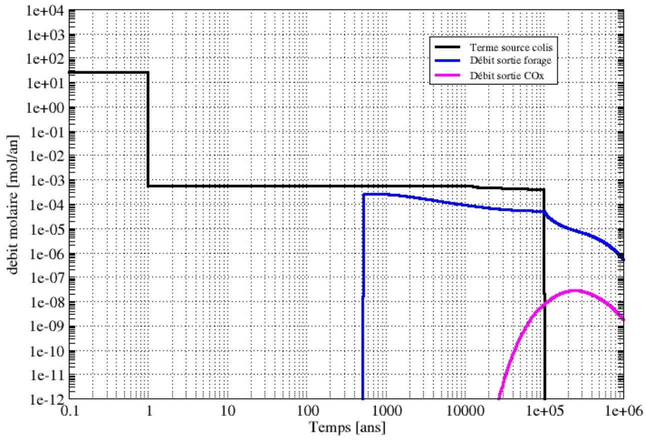


Figure 7.4-21 SEA borehole abandoned in a B waste cell – History of molar flow rates in the near field - B1x - ^{36}Cl packages

- for all the other radionuclides, the mass leaving the borehole is less than 1 % of the total initial mass in the module. This attenuation correlates with the intrinsic characteristics and behaviour of the radionuclides in the materials. These may include :
  (i) long-lived cations which are sorbed very little or not at all (and do not prepicitate) by the concrete and argillaceous host formation. In this case, the fraction of activity reaching and then leaving from the borehole is limited by the significant migration of radionuclides to the Callovo-Oxfordian. Indeed, the effective coefficient of diffusion of cations into the host formation, tends to promote radionuclide transfer to the Callovo-Oxfordian rather than to the borehole. This is in particular the case for ^{53}Mn, ^{40}K, ^{10}Be ;
  (ii) long-lived or moderately long-lived elements which are strongly sorbed into the cell concrete and therefore are alternated before reaching the borehole. This is in particular the case for ^{41}Ca, ^{107}Pd, ^{14}C, ^{92}Nb, ^{93}Zz (➔ ^{93}Nb), ^{59}Ni, ^{99}Tc and ^{94}Nb. For all these radionuclides (but ^{79}Se, ^{41}Ca and ^{14}C) this effect is amplified by a strong sorption in the geological barrier (sound and damaged). In such cases, concentration gradients between the cell and the host formation are strong and favour radial diffusion into the argillites.

Note the particular cases of :

- ^{135}Cs, weakly sorbed in concrete but which is alternated before reaching the borehole due to its sorbtion in the argillites ;
- ^{93}Mo, not sorbed into concrete or Callovo-Oxfordian argillites, for which the attenuation of the mass released at the borehole relative to the initial mass results from its precipitation in concrete and radioactive decay.

The amount of radionuclides released by the borehole to the top of the Callovo-Oxfordian is in any case inferior to the amount that is taken from the drift. Above the repository, elements that are sorbed in the host formation migrate preferentially towards it and therefore the mass reaching the top of the geological barrier is alternated : for example, for ^{107}Pd, ^{59}Ni, ^{135}Cs, ^{99}Tc or ^{93}Zr, the maximum molar flux decreases by an order of magnitude between the entrance and the exit of the borehole.

The impact of actinides released by a borehole into a B waste cell is zero. Indeed, Infact, the very high retention of actinides in the concrete of the disposal cells, combined with their high sorption in the Callovo-Oxfordian, attenuates their molar flow rate out of the borehole completely.

Conclusion relative to the effect of the borehole for B waste

Analysis revealed that the hydraulic effect of the borehole is limited and transport inside B waste cells remains diffusive or is codominantly diffusive / advective.

The radiological impact of the borehole is also strongly limited :

- partly because part of the activity migrates by diffusion into the Callovo-Oxfordian and remains there. This is all the more important in that the radionuclides are sorbed into the host formation ;
- partly because many of the radionuclides are strongly sorbed into the cell and package concrete. Therefore, for these radionuclides, activity flow rates are very slow and delayed in time (particularly for niobium, technetium and zirconium).

Radionuclides with the highest fractions of activity reaching the borehole are long-lived species which are soluble and either weakly or not at all sorbed by the concrete and Callovo-Oxfordian. These particularly include ^{129}I, ^{36}Cl and ^{26}Al. This last radionuclide is in small quantities in waste package. These geochemical properties have also been chosen to be pessimistic owing to lack of data. The result is therefore probably overestimated.

For all the other radionuclides, the fractions of activity reaching the borehole are less than 1 % of the total initial mass in the cell.

7.4.5 Impact assessment

Impact has been evaluated for the activity fraction resulting from altered evolution. The activity fraction not affected by the borehole evolves normally, it is covered by the results of the normal evolution scenario (SEN). We therefore proceed on the basis of the following data :

- The « Barrois limestone potable water pumping » outlet in the vicinity of the borehole is specific to the scenario. Impact at this outlet is therefore additional to the impact taken into account in a normal evolution scenario. Table 7.4-19 summarises the peak dose and the main contributors to the « drinking water supply (DWS) pumping » outlet ;

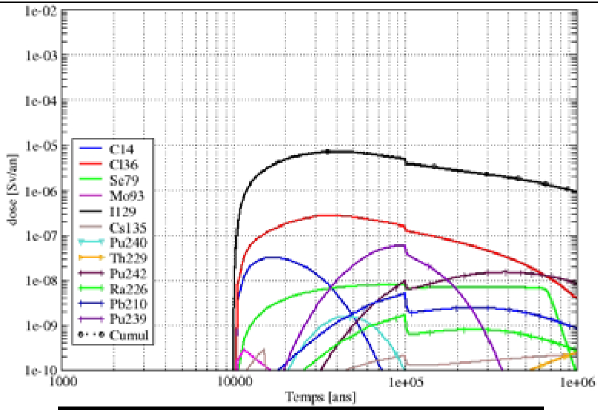
Reference package	Peak Dose [mSv/year]	Date of Peak Dose [Years]	Contributing Radionuclides
B1x - (Non-organic packages that release no gaseous hydrogen)	0.012	770	^{36}Cl ; ^{93}Mo
B2 - (Bituminised sludge)	0.00022	12,000	^{36}Cl ; ^{129}I
C2 Glass	0.00065	25,000	^{36}Cl ; ^{129}I
C4 Glass	0.00053	23,000	^{36}Cl ; ^{129}I
CU1 Spent fuel	0.0072	42,000	^{129}I
CU2 Spent fuel	0.0012	43,000	^{129}I

Table 7.4-19 SEA - Abandoned Borehole Penetration of the Repository - Total Dose - Peak Dose Dates and Main Contributors at the Barrois Limestone DWS Pumping Outlet (in the most pessimistic case) - All Waste Types

- The Saulx outlet is studied as a variant of the Barrois outlet defined above (See Section 7.4.3.1). The dose curves at the « Saulx » outlet (See Figure 7.4-22 to Figure 7.4-31), correspond to the impact associated with the activity fraction leaving the borehole, the evolution of which is therefore altered. In order to reconstitute the total impact at this outlet, it must be added to that of the normal evolution scenario. It will be observed that the additional impact of the borehole at Saulx is, in fact, negligible compared with that of the SEN.

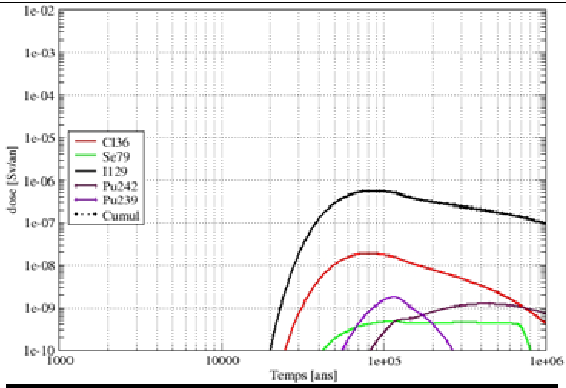
As transfer times in the surrounding formations are short and dispersion is limited, there is, in fact, only a small difference in terms of time between the impact at the two outlets, however ; that of Saulx is lower.

- **Spent Fuel CU1 (Uox3)**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	7.2E-06	4.2E+04
I129	6.9E-06	4.2E+04
Cl36	2.7E-07	3.6E+04
Pu239	5.9E-08	1.0E+05
C14	3.1E-08	1.7E+04
Pu242	1.5E-08	3.8E+05
Se79	8.0E-09	1.0E+05
Pb210	5.0E-09	1.0E+05
Ra226	1.7E-09	1.0E+05
Pu240	1.6E-09	4.5E+04
Mo93	2.8E-10	1.2E+04
Cs135	2.8E-10	1.5E+04
Th229	2.5E-10	1.0E+06

Figure 7.4-22 Dose Due to the Borehole at the Barrois DWS Pumping Outlet (All Radionuclides) - Borehole Penetrating an CU1-Uox3 reference Package Module



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	5.6E-07	8.7E+04
I129	5.4E-07	8.7E+04
Cl36	1.9E-08	8.0E+04
Pu239	1.7E-09	1.2E+05
Pu242	1.2E-09	4.3E+05
Se79	4.7E-10	1.1E+05

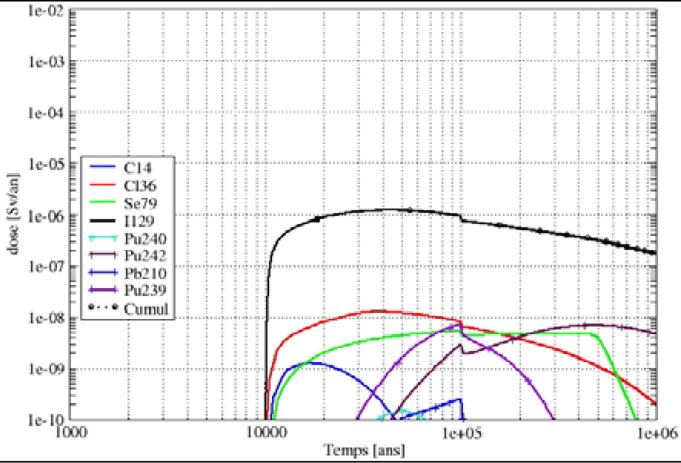
Figure 7.4-23 Dose Due to Borehole at the « Saulx » Outlet (Fission and Activation Products) - Borehole in a Standard CU1-Uox3 Package Module

The total peak dose is dominated by iodine 129 and, to a lesser extent, by chlorine 36.

At the DWS outlet the peak dose is of the order of 0.007 mSv/year and is reached in around 50,000 years' time. At the Saulx outlet studied as a variant, the peak dose resulting from the activity fraction leaving the borehole is of the order of 0.0006 mSv/year and is reached in around 90,000 years' time.

The dose associated with the activity fraction that evolves normally is not represented here. It is identical to the dose evaluated in SEN as the activity fraction whose evolution is altered is negligible compared with the total activity of the CU1 spent fuel zone. The peak dose of the activity fraction that evolves normally is reached at around 330,000 years and has a value of 0.02 mSv/year.

- **CU2 Spent Fuel**



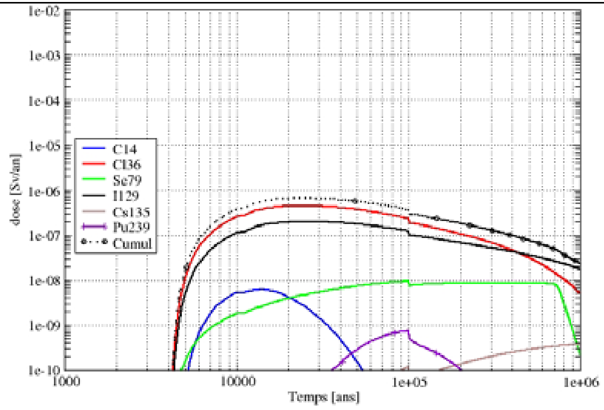
RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.20E-06	4.30E+04
I129	1.20E-06	4.30E+04
Cl36	1.30E-08	3.90E+04
Pu239	7.10E-09	1.00E+05
Pu242	6.80E-09	4.80E+05
Se79	5.30E-09	1.00E+05
C14	1.30E-09	1.70E+04
Pb210	2.50E-10	1.00E+05
Pu240	1.50E-10	5.00E+04

Figure 7.4-24 SEA - Abandoned Borehole Penetration of the Repository – Reference Calculation - Dose Due to the Borehole at the Barrois Limestone DWS Pumping Outlet (All Radionuclides) - Borehole Penetration of a Standard CU2 Package Module

As in the case of CU1 spent fuel, the main contributor to the impact of CU2 spent fuel is iodine 129. Impact at the DWS outlet only is represented. The peak dose of 0.0012 mSv/year is reached after around 50,000 years.

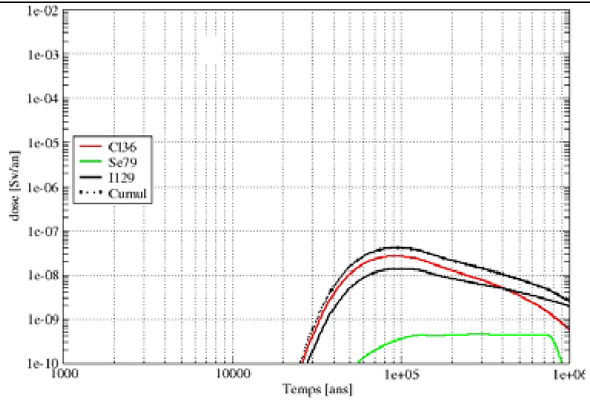
The dose associated with the activity fraction that evolves normally is not represented here. It is identical to the dose evaluated in the SEN ; the peak dose is reached at around 340,000 years and has a value of 0.0017 mSv/year.

- **C (C2 and C4) Waste**



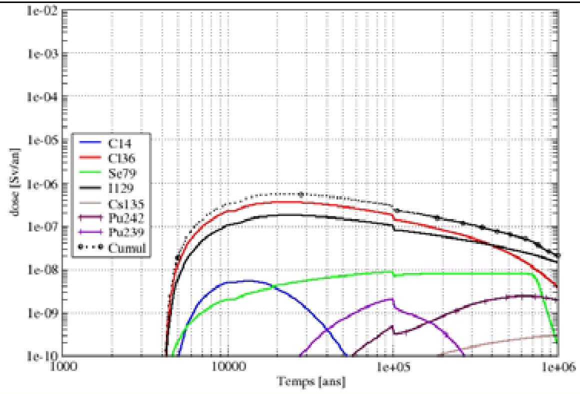
RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	6.5E-07	2.5E+04
Cl36	4.4E-07	2.3E+04
I129	2.0E-07	2.7E+04
Se79	9.4E-09	1.0E+05
C14	6.2E-09	1.4E+04
Pu239	7.4E-10	1.0E+05
Cs135	3.8E-10	1.0E+06

Figure 7.4-25 SEA - Abandoned Borehole Penetration of the Repository - Reference Calculation - Dose Due to the Borehole at the Barrois Limestone DWS Pumping Outlet (All Radionuclides) - Borehole Penetration of a C2 Reference package Module



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	4.0E-08	9.5E+04
Cl36	2.6E-08	9.2E+04
I129	1.4E-08	9.9E+04
Se79	4.4E-10	2.9E+05

Figure 7.4-26 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity Calculations - Dose Due to the Borehole at the « Saulx » Outlet - C2 reference packages



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	5.3E-07	2.3E+04
Cl36	3.5E-07	2.3E+04
I129	1.7E-07	2.4E+04
Se79	8.7E-09	1.0E+05
C14	5.2E-09	1.3E+04
Pu242	2.4E-09	6.3E+05
Pu239	2.0E-09	1.0E+05
Cs135	3.0E-10	1.0E+06

Figure 7.4-27 SEA - Abandoned Borehole Penetration of the Repository - Reference Calculation - Dose Due to the Borehole at the Barrois Limestone DWS Pumping Outlet (All Radionuclides) - Borehole Penetration of C4 Reference package

In the case of C waste, the main contributor is chlorine 36 which is found in greater quantities than in spent fuel.

The impact of C2 reference package is evaluated at the DWS outlet and the Saulx outlet studied as a variant. The impact of C4 reference package is only shown at the DWS outlet where the peak dose is of the order of 0.0005 mSv/year for C4 reference package and 0.0006 mSv/year for C2 reference package. This occurs after around 20,000 years which is 30,000 years earlier than in the case of spent fuel. This difference of 30,000 years is in part the result of the duration of water tightness of the spent fuel containers being longer than that of the C waste containers, and in part the result of the conservative release model taken into account in the case of vitrified waste which gives rise to a release of radionuclides over approximately 10,000 years as compared with approximately 35,000 years in the case of CU1 spent fuel.

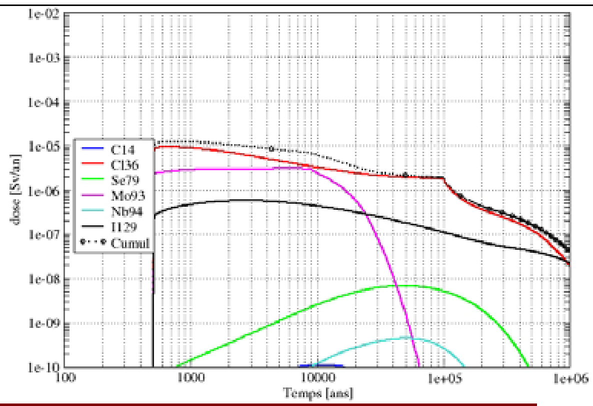
- **Impact Associated with Penetration of a B1x Cell**

The radiological impact is dominated in the short term by the following radionuclides : ^{36}Cl, ^{93}Mo, ^{129}I and to a lesser extent ^{79}Se et ^{94}Nb. In the longer term (over 50,000 years) chlorine 36 and iodine 129 dominate the total impact, as the activity of the others will have decayed. The spread out of labile iodine 129 by diffusion is very significant taking into account its delay in concrete. Conversely, the fuel cladding release model (10^{-5}/year) for the chlorine 36 activity concerned explains the discontinuity of the curves observed at 100,000 years. The labile activity of chlorine 36 spread by diffusion before the borehole is drilled explains the high concentration in the borehole from 500 years onwards.

At the « DWS pumping » outlet, the peak dose occurs when the borehole is drilled, i.e. at 500 years, with a total dose of the order of 0.01 mSv/year. Because the impact is associated with iodine released labilely in the case of packages disposed of in a B1x cell, this element is immediately available from the moment water arrives. Because the repository is considered to be resaturated from the outset, the borehole captures the elements that accumulated in the cell before drilling commenced.

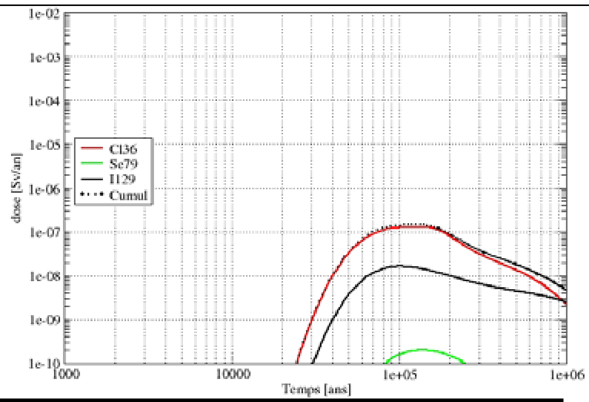
At the Saulx outlet studied as a variant, only iodine 129, chlorine 36 and selenium 79 contribute to impact because the other radionuclides will have benefited from radioactive decay. The peak dose of the order of 0.00015 mSv/year is reached at around 120,000 years.

The dose relating to the radionuclide inventory moving through the host formation is identical to that of the SEN (model at 1 million years) presented in chapter 5.5. It is of the order of 0.0003 mSv/year and occurs at around 300,000 years.



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.2E-05	7.7E+02
Cl36	9.3E-06	6.6E+02
Mo93	3.0E-06	5.8E+03
I129	5.7E-07	2.7E+03
Se79	6.7E-09	4.8E+04
Nb94	4.4E-10	5.1E+04
C14	1.1E-10	1.1E+04

Figure 7.4-28 SEA - Abandoned Borehole Penetration of the Repository - Dose Due to the Borehole at the Barrois DWS Pumping Outlet - B1x reference package

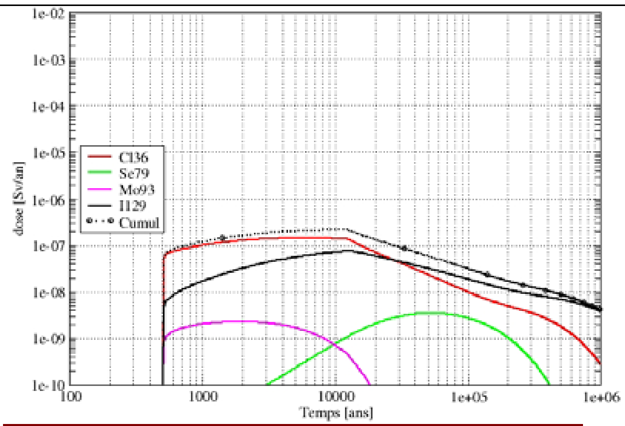


RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.5E-07	1.2E+05
Cl36	1.3E-07	1.3E+05
I129	1.6E-08	1.0E+05
Se79	2.0E-10	1.4E+05

Figure 7.4-29 SEA - Abandoned Borehole Penetration of the Repository - Dose Due to the Borehole - Pumping Outlet in the Vicinity of the Saulx - B1x reference package

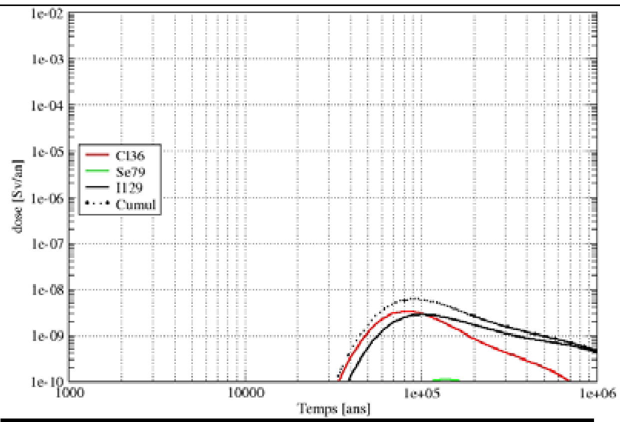
- **Impact Associated with the Borehole Penetration of a B2 Cell**

The radionuclides that give rise to the greatest impact are similar to those of the B1x waste cells (^{36}Cl and ^{129}I to a lesser extent). However, impact is lower because the inventory is smaller and the bituminised sludge release model gives rise to a slower release (COLONBO 3 model).



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	2.2E-07	1.2E+04
Cl36	1.5E-07	5.9E+03
I129	7.5E-08	1.3E+04
Se79	3.4E-09	5.2E+04
Mo93	2.3E-09	1.9E+03

Figure 7.4-30 SEA - Abandoned Borehole Penetration of the Repository - Dose Due to Borehole - Barrois DWS Pumping Outlet - B2 reference package



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	6.0E-09	9.2E+04
Cl36	3.3E-09	8.4E+04
I129	2.8E-09	1.0E+05
Se79	1.1E-10	1.4E+05

Figure 7.4-31 SEA - Abandoned Borehole Penetration of the Repository - Dose Due to Borehole -Pumping Outlet in the Vicinity of the Saulx - B2 Bitumen reference package

7.4.6 Sensitivity Analysis

7.4.6.1 Sensitivity to the Position of the Borehole

- **Sensitivity to the Position of the Borehole Penetration of an CU1 Spent Fuel Cell**

This study, carried out for illustrative purposes for the case of CU1 spent fuel, consists of assessing the long term consequences associated with the direct borehole penetration of a cell, unlike the reference calculation for C waste and spent fuel in which the borehole is assumed to have penetrated an access drift. Given such a situation is extremely unlikely, the borehole is assumed not to have penetrated a repository module containing a package that has failed, the influence of the latter is therefore not taken into account in respect of the borehole.

From a hydraulic point of view, it is notable that in Figure 7.4-32 the disturbance caused by the borehole extends to various drifts and cells of the repository zone.

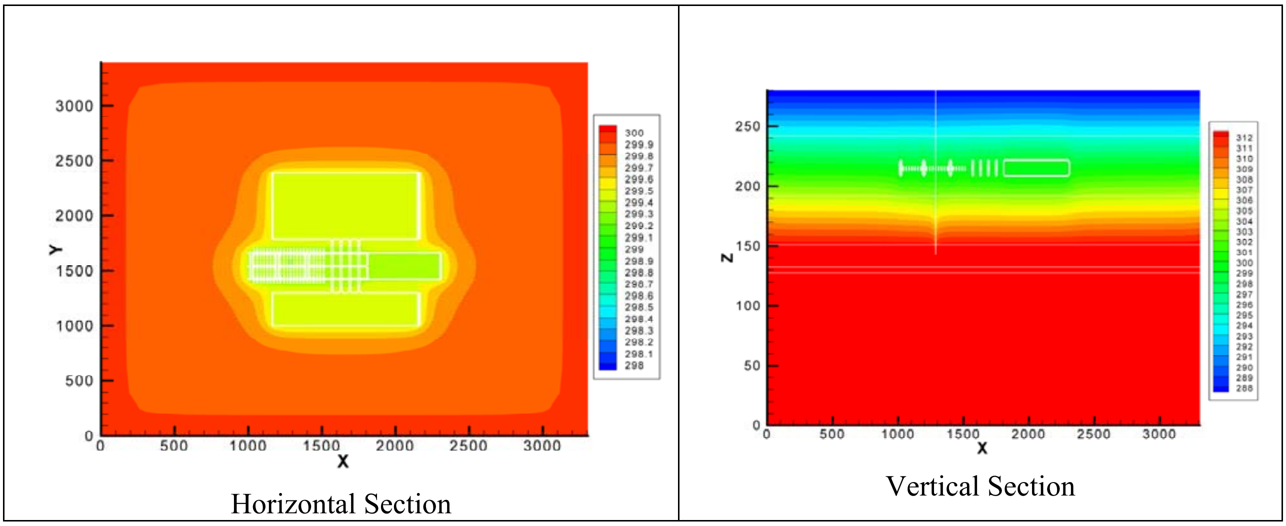


Figure 7.4-32 SEA - Borehole Penetration of the Repository - Sensitivity Study to the Position of the Borehole (Borehole Penetration of an CU1 Spent Fuel Cell) - Hydraulic Head Field in the Repository Zone

From the point of view of transport and impact when leaving the borehole, the results show that the peak molar flow of the majority of the radionuclides is reached when the containers fail. The borehole instantly « captures » part of the labile activity of the portion of package penetrated, then a large part of the inventory of the cell and eventually, in the long term in the case of the most mobile radionuclides, contributions from neighbouring cells via the drifts and the Excavation Damaged Zone (EDZ) of the cell penetrated.

In the case of the most mobile, soluble and non-sorbed radionuclides, iodine 129 and chlorine 36 in particular, it is notable that the quantity of activity captured by the borehole over a period of 1 million years is lower in the case of a borehole that penetrates a cell than in the case of a borehole that penetrates an access drift. In fact, it is verified that the activity of iodine 129 captured by the borehole corresponds to the equivalent of the initial activity contained in 9 cells in the case of a borehole that penetrates a cell compared with 15 in the case of a borehole penetrating an access drift (5 cells compared with 8 in the case of chlorine 36 which benefits from radioactive decay). However, the maximum molar flow captured by the borehole is significantly greater in terms of sensitivity. In fact, in the case of a borehole that penetrates a cell, the labile activity released by packages in the impacted cell is directly intercepted by the borehole. Similarly, the time at which the maximum molar flow appears is earlier than in the case of the reference calculations on account of the immediate availability of the inventory affected by the borehole (see Figure 7.4-33).

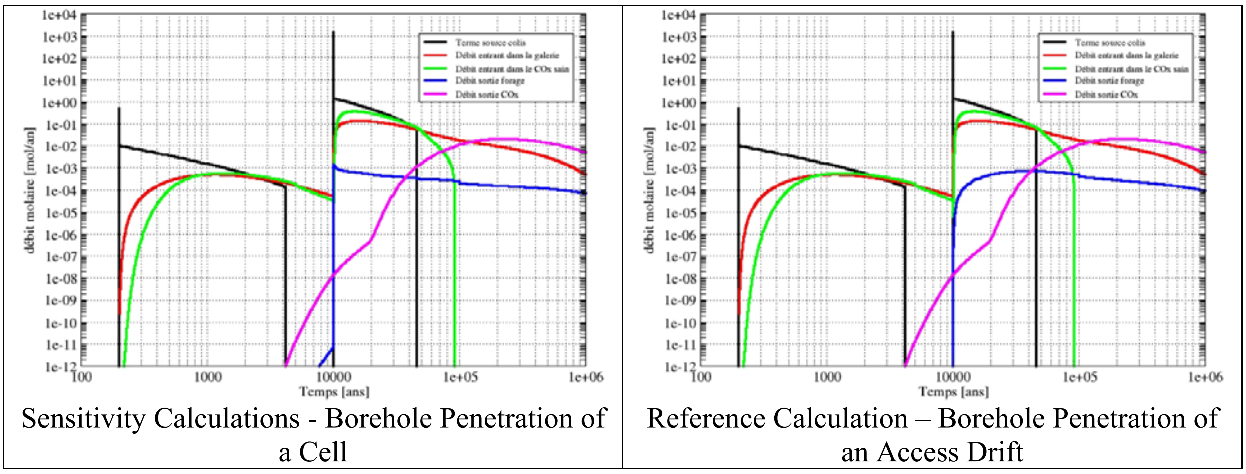
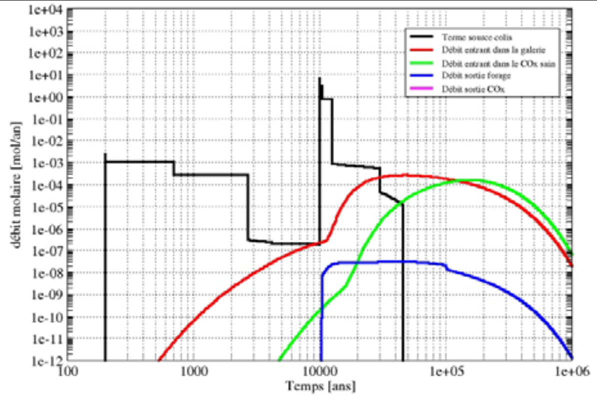
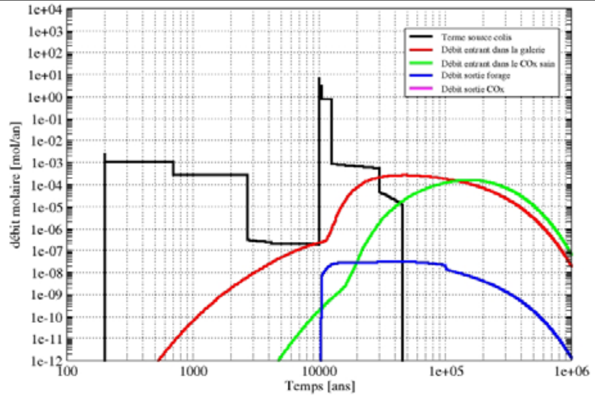


Figure 7.4-33 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the Position of the Borehole - History of the Molar Flow Rate CU1 - UOx3 - ^{129}I

In the case of radionuclides that are highly sorbed in the Callovo-Oxfordian argillites, the transfer pathway through the sound geological barrier is preponderantly over the borehole via the borehole. In addition, in the case of these radionuclides, the contribution of other cells is negligible partly because these radionuclides remain to a large extent confined in the near field of each of the cells and partly because the activity fraction that finally reaches the drifts is very significantly attenuated by the effect of diffusion towards the sound geological barrier before reaching the borehole. In the case of nickel 59, only 0.0025 % of the initial inventory of a repository sub-zone (which is 5 % of the inventory of a single cell) leaves via the borehole, this activity fraction essentially corresponds to that of the impacted cell (See Figure 7.4-34).



Sensitivity Calculations - Borehole Penetration of a Cell



Reference Calculation – Borehole Penetration of an Access Drift

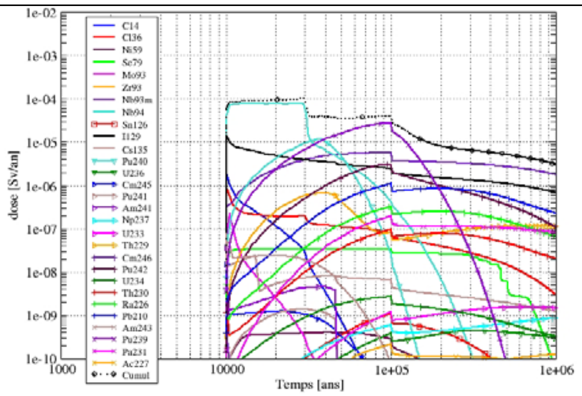
Figure 7.4-34 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the Position of the Borehole - History of the Molar Flow Rate SF - ^{59}Ni

As regards impact, the doses associated with the transfer pathway via the borehole are significantly higher than they are in the case of the reference calculation. The maximum dose at the DWS pumping outlet is of the order of 0.1 mSv/year at around 30,000 years compared with 0.007 mSv/year at around 40,000 years in the case of the reference calculation. Impact is mainly governed by labile release, the main contributors are niobium 94 initially and plutonium 239 in the longer term.

The maximum dose due to 239Pu and 240Pu is however moderate, of the order of 0.04 mSv/year after around 100,000 years (see Figure 7.4-35)

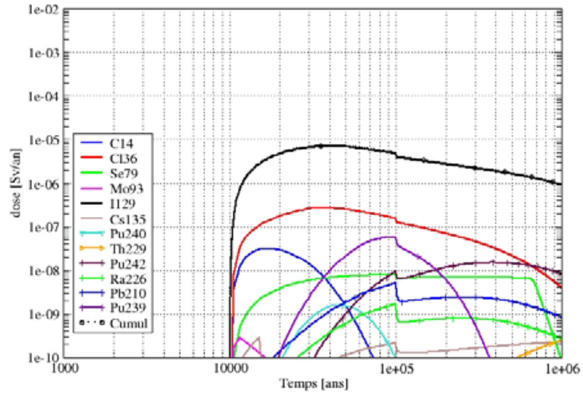
It should be noted, however, that the dose calculation at the « DWS pumping » outlet is very pessimistic. Indeed, taking into account a transfer time of several tens of thousands of years in the surrounding formations up to the Saulx outlet would be sufficient to delay and attenuate the impact of plutonium 239.

In conclusion, even in a very pessimistic situation in which the borehole penetrates a spent fuel cell in the immediate vicinity of a package, the impact observed is moderate (of the order of 0.1 mSv/year).



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Zr93	1.3E-10	1.0E+06
Np237	8.8E-10	1.0E+06
U233	1.6E-09	6.5E+05
U236	4.4E-10	3.1E+05
Nb93m	5.8E-06	1.0E+05
Sn126	1.1E-09	1.0E+05
U234	2.7E-09	1.0E+05
Th230	9.6E-08	1.0E+05
Ra226	3.2E-07	1.0E+05
Pb210	1.1E-06	1.0E+05
Pu239	2.7E-05	1.0E+05
Pa231	1.9E-07	1.0E+05
Ac227	2.2E-10	1.0E+05
Pu242	3.0E-06	9.2E+04
Se79	3.5E-08	4.6E+04
Ni59	4.0E-10	4.0E+04
Th229	6.7E-07	3.9E+04
Pu240	1.2E-05	3.7E+04
Cumul	1.0E-04	3.0E+04
Pu241	1.4E-09	2.9E+04
Nb94	8.0E-05	2.0E+04
Mo93	2.2E-07	1.1E+04
C14	1.9E-06	1.0E+04
Cl36	9.0E-07	1.0E+04
I129	1.4E-05	1.0E+04
Cs135	8.3E-08	1.0E+04
Cm245	1.5E-08	1.0E+04
Am241	3.7E-08	1.0E+04
Cm246	4.2E-09	1.0E+04
Am243	5.0E-08	1.0E+04

Sensitivity Calculations - Borehole Penetration of a Cell



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	7.2E-06	4.2E+04
I129	6.9E-06	4.2E+04
Cl36	2.7E-07	3.6E+04
Pu239	5.9E-08	1.0E+05
C14	3.1E-08	1.7E+04
Pu242	1.5E-08	3.8E+05
Se79	8.0E-09	1.0E+05
Pb210	5.0E-09	1.0E+05
Ra226	1.7E-09	1.0E+05
Pu240	1.6E-09	4.5E+04
Mo93	2.8E-10	1.2E+04
Cs135	2.8E-10	1.5E+04
Th229	2.5E-10	1.0E+06

Reference Calculation – Borehole Penetration of an Access Drift

Figure 7.4-35 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the Position of the Borehole - History of the Dose at the DWS Pumping Outlet - CU1 - UOx3

- **Sensitivity to the Position of the Borehole – in B (B1x) Waste Access Drifts**

This study, carried out for illustrative purposes for the case of non-organic waste that does not release hydrogen (B1x), consists of assessing the long term consequences associated with a borehole that penetrates a secondary connection drift, unlike the reference calculation in which the borehole is assumed to have penetrated a cell.

From a hydraulic point of view, it is notable that in Figure 7.4-36 the borehole gives rise to very low horizontal hydraulic head gradients in the access drifts that have the effect of limiting advective phenomena.

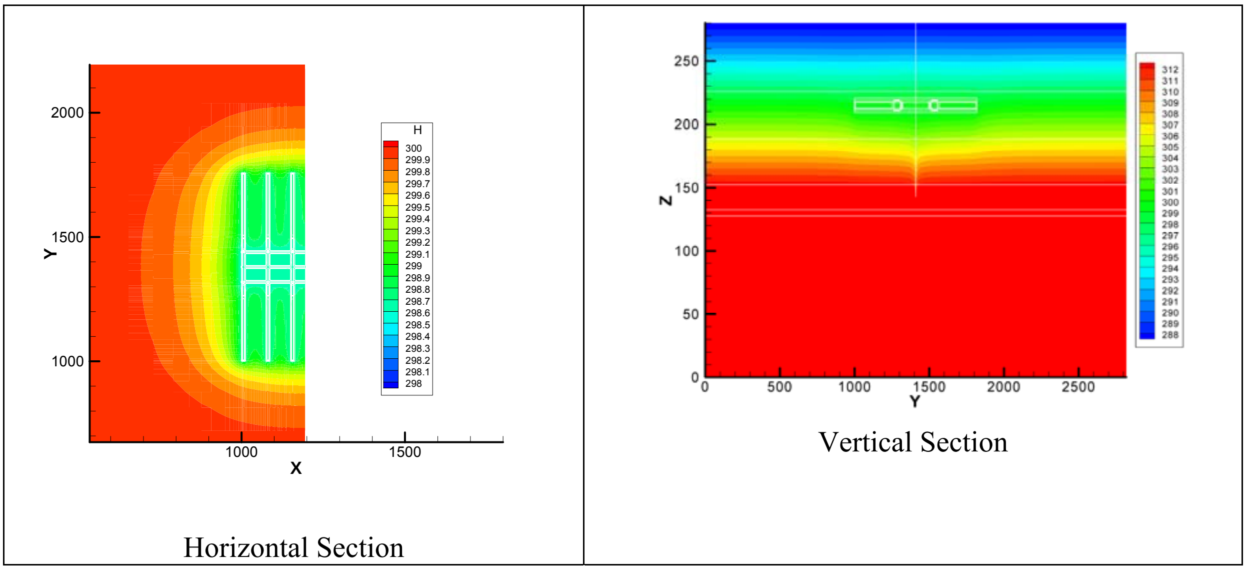


Figure 7.4-36 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity Study on the Position of the Borehole (Borehole Penetration of a B Waste Cell Access Drift - B1x reference package - Hydraulic Head Field in the Repository Zone

From the point of view of transport, the results show that the extremely slow advective kinetics combined with the significant distances to be covered from the packages to the borehole in the drift (20 to 250 meters in the cell depending on the location of the packages and 60 meters of drift to the borehole) give rise to major transfer times to reach the borehole.

The following points are therefore observed :

- Chlorine 36, which is soluble and not sorbed in the concrete, has a maximum molar flow rate on leaving the borehole very late at around 500,000 years, at which time it begins to benefit from radioactive decay (See Figure 7.4-37).

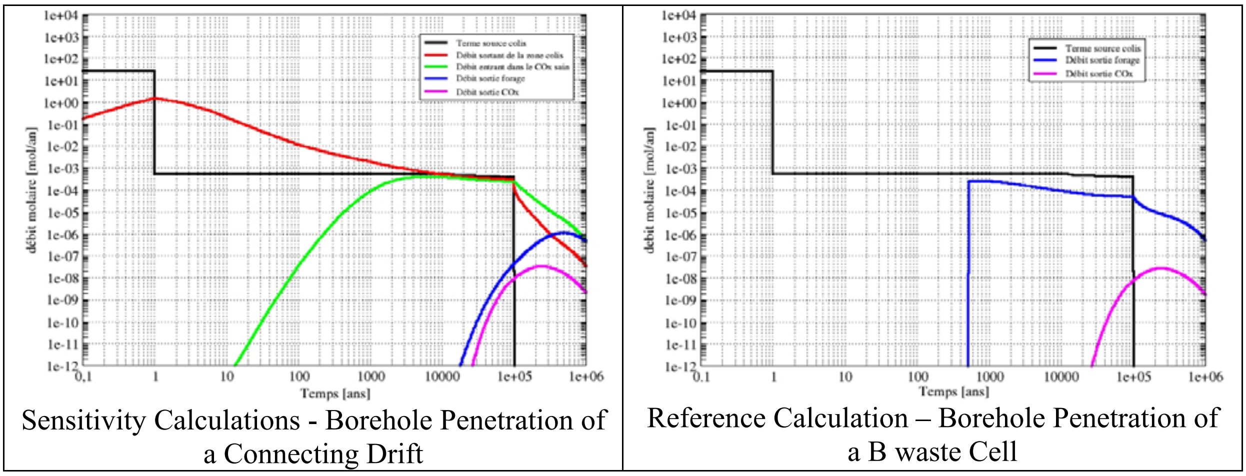
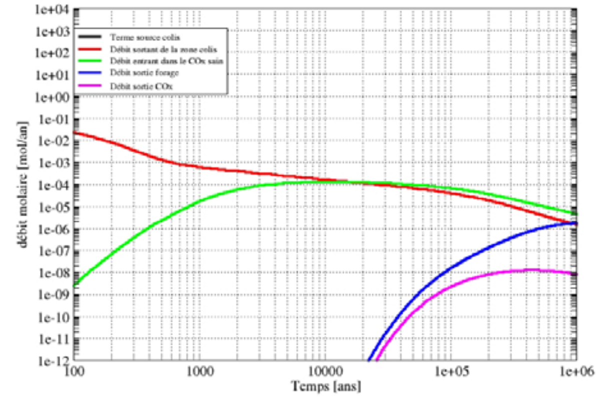
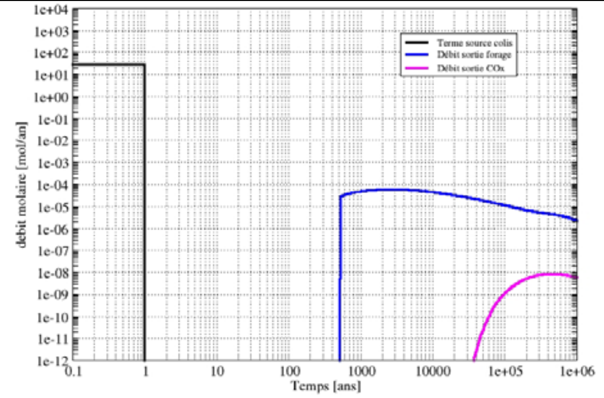


Figure 7.4-37 SEA - Abandoned Borehole Penetration of the Repository – Sensitivity to the Position of the Borehole - History of the Molar Flow Rate - Non-Organic reference package that Do Not Release Hydrogen - ^{36}Cl

- The low sorption of iodine 129 in concrete (retardation coefficient of 8) significantly increases its average transfer time from the packages to the borehole which favours heat exchange with the geological barrier. The maximum molar flow rate leaving the borehole therefore appears in excess of a million years and is significantly attenuated (See Figure 7.4-38).



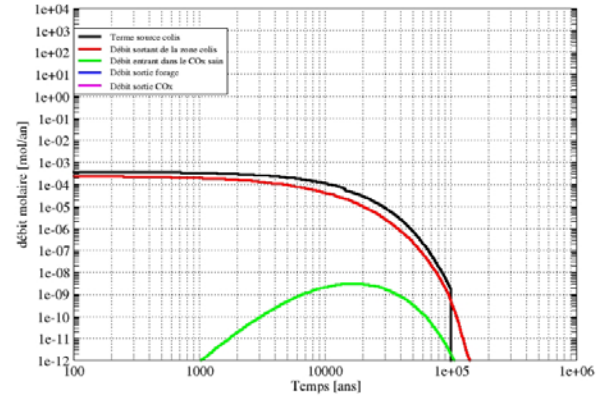
Sensitivity Calculations - Borehole Penetration of a Connecting Drift



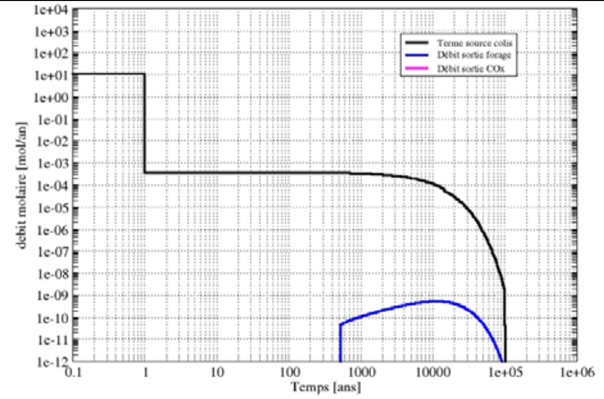
Reference Calculation – Borehole Penetration of a B waste Cell

Figure 7.4-38 SEA - Abandoned Borehole Penetration of the Repository – Sensitivity to the Position of the Borehole - History of the Molar Flow Rate - Non-Organic reference package that Do Not Release Hydrogen - ^{129}I

- The selenium 79 and carbon 14 inventory that reaches the borehole is significantly limited and delayed by sorption phenomena in the concrete



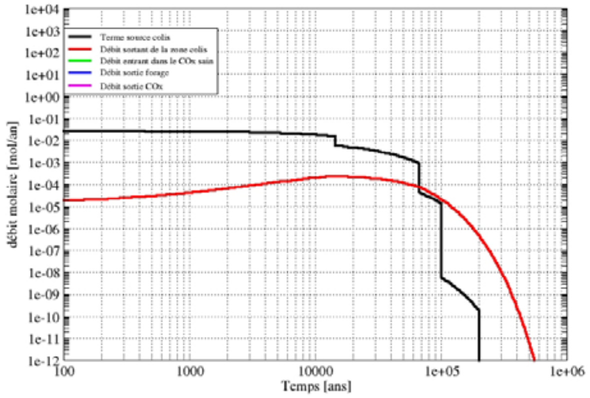
Sensitivity Calculations - Borehole Penetration of a Connecting Drift



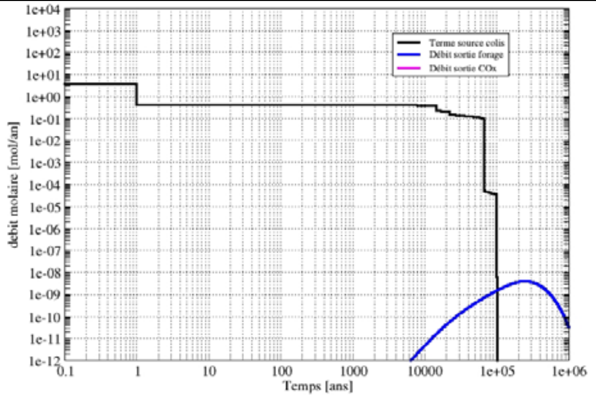
Reference Calculation – Borehole Penetration of a B waste Cell

Figure 7.4-39 SEA - Abandoned Borehole Penetration of the Repository – Sensitivity to the Position of the Borehole - History of the Molar Flow Rate - Non-Organic reference package that Do Not Release Hydrogen – ^{14}C

- In the case of the other radionuclides (^{59}Ni, ^{107}Pd, ^{126}Sn, ^{99}Tc, ^{94}Nb, actinides, etc.), their significant chemical affinity with the Callovo-Oxfordian argillites favours their migration in the geological barrier, in the drift then in the borehole. In the case of all these radionuclides, the molar flow rate is negligible (See Figure 7.4-40).



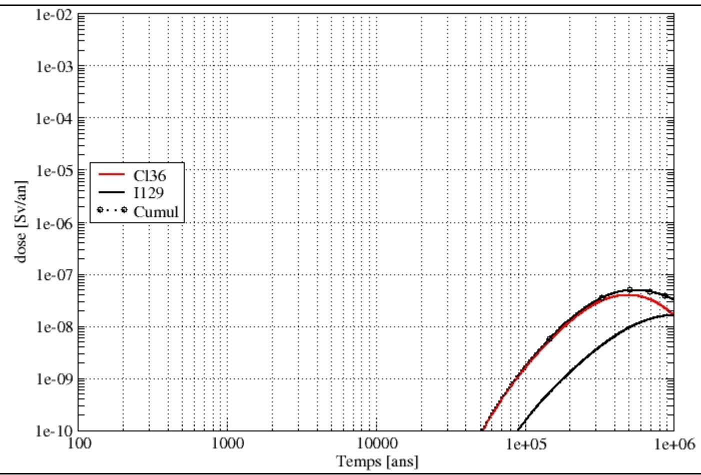
Sensitivity Calculations - Borehole Penetration of a Connecting Drift



Reference Calculation – Borehole Penetration of a B waste Cell

Figure 7.4-40 SEA - Abandoned Borehole Penetration of the Repository – Sensitivity to the Position of the Borehole - History of the Molar Flow Rate - Non-Organic reference package that Do Not Release Hydrogen ^{59}Ni

From the point of view of impact, a maximum dose of 0.00005 mSv/year is observed at around 550,000 years led by chlorine 36 and to a lesser extent iodine 129 (See Figure 7.4-41). The peak dose of iodine 129 is situated in excess of 1 million years. The other radionuclides all have doses lower than 10^{-7} mSv/year (10^{-10} Sv/year).



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	4.9E-08	5.5E+05
Cl36	4.0E-08	4.9E+05
I129	1.6E-08	1.0E+06

Figure 7.4-41 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the Position of the Borehole (Borehole Penetration of a B Waste Connection Drift) - History of the Dose at the DWS Pump Outlet - Non-Organic reference package that Do Not Release Hydrogen (B1x)

- **Two-Borehole Configuration - Penetrating an CU1 Spent Fuel Repository Module**

This case concerns a two borehole configuration, one borehole located to the east and the other to the west of an CU1 spent fuel module the horizontal head gradient from west to east is 0,0013 m/m (see Figure 7.4-8). As seen in Section 7.4.3.2, 2 two-borehole configuration situations are considered : one that maximises the hydraulic head differences between the two boreholes by positioning them at the ends of the access drifts (Situation 1) and another which maximises the radiological influence of the boreholes by positioning them in the middle of access drifts (Situation 2). Both two-borehole configurations have been carried out for illustrative purposes for the case of iodine 129.

Figure 7.4-42 and Figure 7.4-43 provide the hydraulic head fields in an CU1 spent fuel repository sub-zone and its neighbouring geological (Dogger Callovo-Oxfordian) formation at 100,000 years for both situations studied.

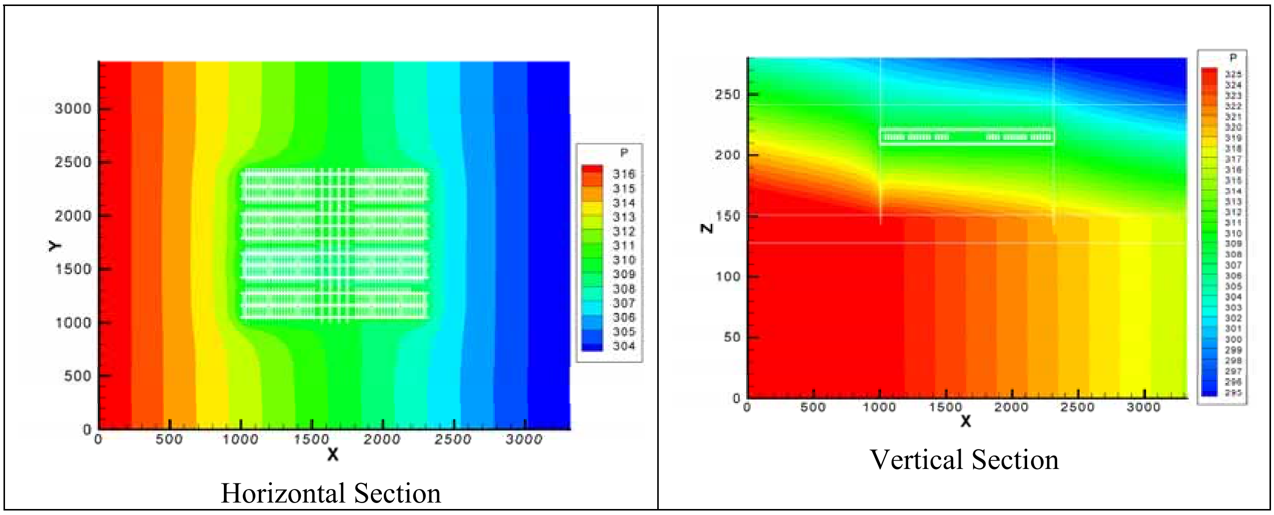


Figure 7.4-42 SEA - Borehole Penetration of the Repository - Sensitivity Study of the Two-Borehole Configuration Positioned at the Ends of the Spent Fuel Access Drifts - CU1 reference package - Hydraulic Head Field in the Repository Zone.

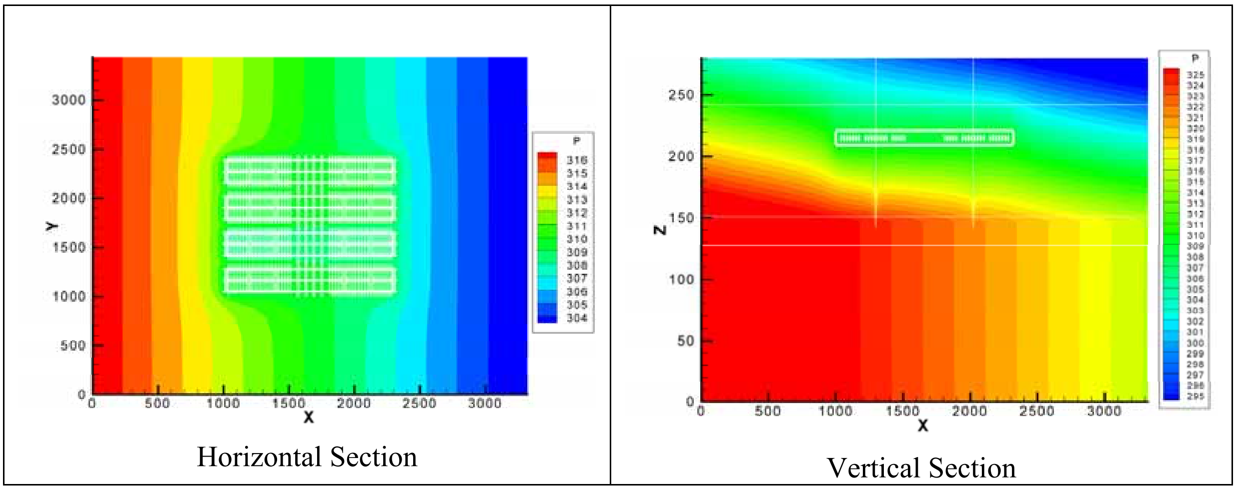


Figure 7.4-43 SEA - Borehole Penetration of the Repository - Sensitivity Study of the Two-Borehole Configuration Positioned at the Centre of the Spent Fuel Cell Access Drifts - CU1 reference package - Hydraulic Head Field in the Repository Zone.

As in the case of the reference calculations, the vertical sections of the hydraulic head fields show that the boreholes locally reduce the hydraulic heads under the repository. With respect to the repository, drawdown is greater in the module intercepted than in the other modules. The water flows drained by the boreholes in each of the main components of the model are given inTable 7.4-20. Comparison with the results of the reference calculation shows that the total water flow drained by the two-borehole configuration is approximately twice as high as it is in the case of the reference calculation.

Scenario		Water Flow Drained by Boreholes at 100,000 years (³/year)					Repository /Total
		Dogger	Callovo-Oxfordian Surmounted by Repository	Callovo-Oxfordian Overlying Repository	Repository	Total	
Reference	-	0.110	0.002	0.000	0.050	0.162	31 %
Two-Borehole Configuration (Situation 1)	West	0.131	0.006	0.000	-0.011[108]	0.126	-9 %
	East	0.087	0.003	0.000	0.110	0.200	55 %
Two-Borehole Configuration (Situation 2)	West	0.125	0.003	0.000	0.010	0.138	7 %
	East	0.100	0.002	0.000	0.076	0.178	43 %

Table 7.4-20 SEA - Abandoned Borehole Penetration of the Repository - Contribution of the Components of the Model to the Water Flow Drained by the Borehole into an CU1 Spent Fuel Repository Sub-Zone

As regards transport, the results of the two-borehole configuration reveal the following points (See Table 7.4-21 :

- The peak flow leaving the borehole is greater to the east of the repository module than to the west. This difference, at most of a factor of 2.5, is due to the horizontal hydraulic head gradient that favours the transfer of the radionuclides towards the east.
- For both two-borehole configuration situations, the most pessimistic contribution of the borehole (east borehole) in terms of molar flow rate and mass leaving the borehole over 1 million years remains lower than (but of the same order of magnitude as) that of the single borehole studied in the reference case. These results show that the limited horizontal hydraulic head gradients, the permeability of the geological barrier and the « cul-de-sac » arrangement of the cells significantly limit U-shaped flows. In the case of the first situation, the location of the borehole at the end of an access drift distances it altogether from the cells which significantly reduces the impact compared with the reference situation in which the borehole is situated at centre of the access drifts. In the case of the second situation, the results are comparable to those of the reference situation.
- For both two-borehole configuration situations, the mass leaving the two boreholes of the system over a million years is higher than but not double that of a single borehole. Indeed, the increase compared with reference calculation is of the order of 26 % in the case of Situation 1 (two-borehole configuration at the ends of a drift) and 62 % in the case of Situation 2 (two-borehole configuration at the centres of two repository half-modules).

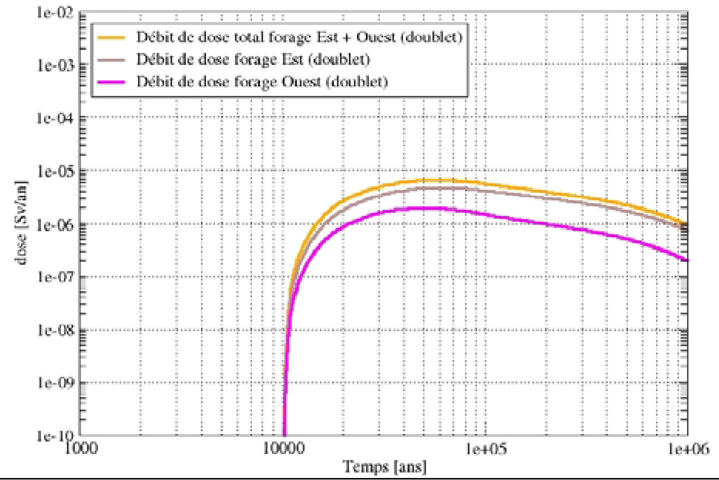
[108] Negative values means a loss of water from the borehole into the components that it passes through : the West Borehole of the Two-Borehole System (Situation 1) looses 0.011 m^3/year into the repository drifts whereas it drains 0.137 m^3/year into underlying formations (Dogger Callovo-Oxfordian surmounted by the repository). The reduction in hydraulic heads to the east of the model gives rise, in fact, to horizontal flows that result in losses of water from the borehole across permeable media such as the repository drifts.

	Scenario				
	Reference	Two-Borehole Configuration (Situation 1)		Two-Borehole Configuration (Situation 2)	
		West Borehole	East Borehole	West Borehole	East Borehole
Maximum Molar Flow Rate [mol/year]	$7.28{\cdot}10^{-4}$	$2.02{\cdot}10^{-4}$	$4.92{\cdot}10^{-4}$	$4.68{\cdot}10^{-4}$	$6.78{\cdot}10^{-4}$
Date of the Maximum Molar Flow Rate [year]	42,000	51,000	59,000	45,000	47,000
Accumulated molar flow over 1 M Yrs [mol]	213.64	65.97	202.43	135.69	209.62

Table 7.4-21 SEA - Abandoned Borehole Penetration of the Repository - Accumulated and Instantaneous Molar Flow Rates - CU1 Spent Fuel - ^{129}I

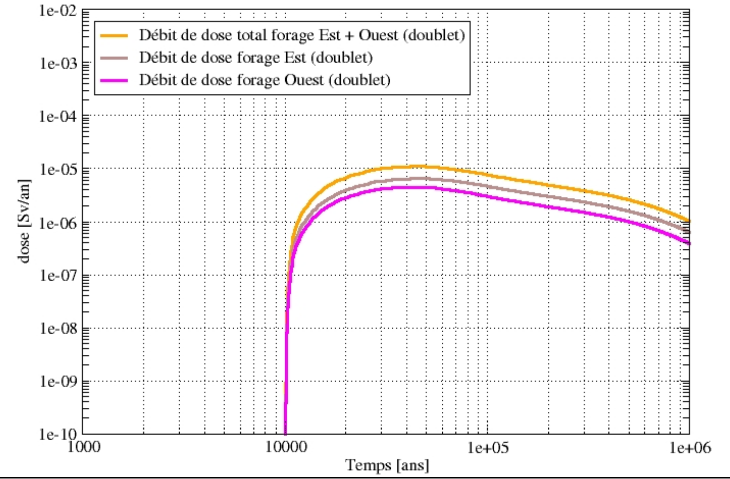
The results of radiological impact are consistent with the preceding observations (See Figure 7.4-45 and Figure 7.4-46) :

- In the case of the first situation, the peak dose due to the boreholes of the double configuration situated at the ends of the drifts is slightly lower than that obtained by the reference calculation (0.0066 mSv/year compared with 0,0069 mSv/a in the case of iodine 129 which is -4 %).
- In the case of the second situation, taking into account a two-borehole configuration at the centres of two repository half-modules, the maximum dose rate increases by approximately 60 % compared with the reference calculation.



Scénario	DoseMax[Sv/an]	t(DoseMax)[an]
Forage Est+Ouest (doublet)	6.6E-06	5.7E+04
Forage Est (doublet)	4.7E-06	5.9E+04
Forage Ouest (doublet)	1.9E-06	5.1E+04

Figure 7.4-44 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity - Two-Borehole Configuration at the Ends of a Drift (Situation 1) Dose History - CU1 Spent Fuel ^{129}I



Scénario	DoseMax[Sv/an]	t(DoseMax)[an]
Forage Est+Ouest (doublet)	1.1E-05	4.5E+04
Forage Est (doublet)	6.5E-06	4.7E+04
Forage Ouest (doublet)	4.5E-06	4.5E+04

Figure 7.4-45 SEA - Abandoned Borehole Penetration of the Repository – Sensitivity - Two-Borehole Configuration in the Middle of Two Repository Half-Modules (Situation 2) - Dose History - CU1 Spent Fuel^{129}I

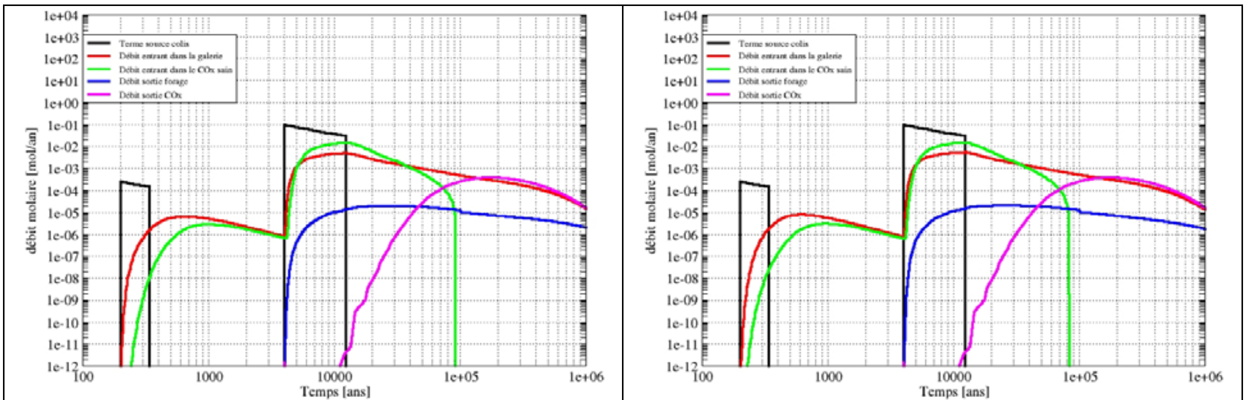
In conclusion, for both two-borehole configuration situations, the impact of each of the two-borehole configurations is less than or equal to the dose associated with the single borehole studied in the reference.

7.4.6.2 Sensitivity in Clay Engineered Barriers

- **Sensitivity to the C Waste Design - Variant with Clay Engineered Barrier (C2 Waste)**

This study consists of an analysis of the consequences associated with taking into account a variant in the case of C waste in which the cell has an 80-centimetre thick clay engineered barrier, the properties of which are identical to those of the plug (bentonite).

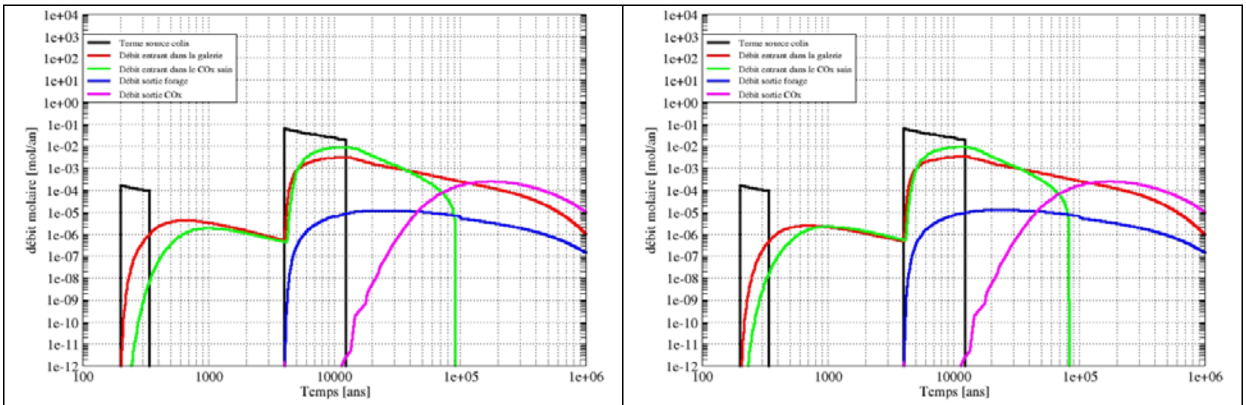
Molar flow rate histories (See Figure 7.4-46 to Figure 7.4-48) show that there is no significant difference in terms of release between the two designs. The small thickness of the engineered barrier does not change the nature of release in terms of the average diffusive travel time towards the drift.



Sensitivity Calculations : Short Tunnel with Clay Engineered Barrier

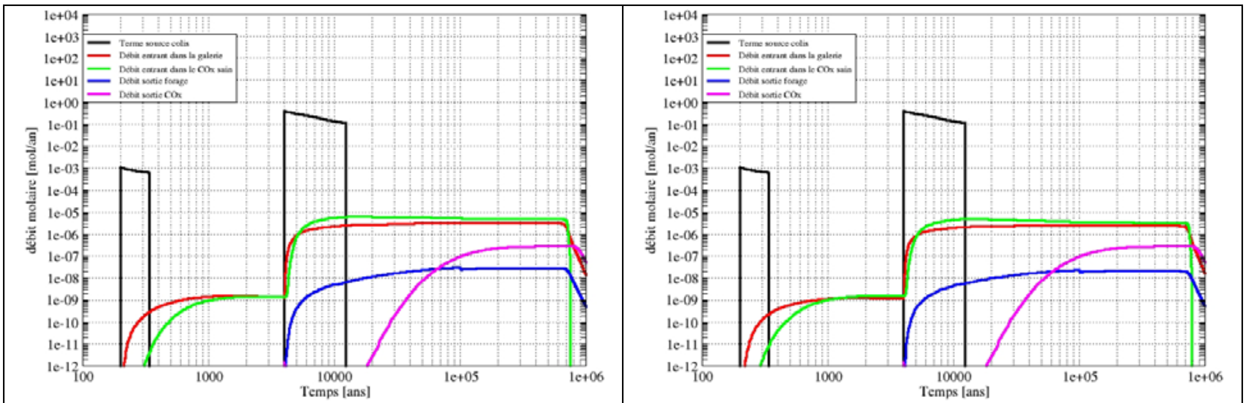
Reference Calculation : Short Tunnel without Clay Engineered Barrier

Figure 7.4-46 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the C Waste Design (Alterative Design with Clay Engineered Barrier) - Molar Flow Rate History Data - C2 reference package - ^{129}I



Sensitivity Calculations : Short Tunnel with Clay Engineered Barrier	Reference Calculation : Short Tunnel without Clay Engineered Barrier

Figure 7.4-47 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the C Waste Design (Alterative Design with Clay Engineered Barrier) - Molar Flow Rate History Data - C2 reference package - ^{36}Cl



Sensitivity Calculations : Short Tunnel with Clay Engineered Barrier	Reference Calculation : Short Tunnel without Clay Engineered Barrier

Figure 7.4-48 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the C Waste Design (Alternative Design with Clay Engineered Barrier) - Molar Flow Rate History - C2 reference package - ^{129}I

The doses are identical to those of the reference situation (see Section 7.4.5).

Contrary to what may have been thought, it appears, therefore, that the clay engineered barrier does not play a redundant barrier role of the geological medium in the case of a borehole. The fact not being considered does not diminish the robustness of the repository in the event of a borehole penetrating a drift.

7.4.6.3 Sensitivity to Hydraulic Parameters of the EDZ (« Degraded EDZ »)

The purpose of this study is to assess the consequences of a reduction in the hydraulic, transfer and retention characteristics of the EDZ in the event of a borehole being drilled. This study, which has been carried out for the case of the C2 reference package, consisted of taking a « degraded » EDZ into account whose parameter value details can be found in Table 7.4-4 with :

- A « pessimistic » fractured zone represented by a high degree of permeability (10^{-6} m/s), diffusion equal to that of a particle of water in water and an absence of geochemical retention properties ;
- A micro-fissured zone with « conservative » transfer and retention hydraulic parameters.

Table 7.4-22 gives the percentages of mass leaving the various components of the system over 1 million years' time.

Radionuclide	Distribution of Release from the Cells		% Mass to reach the borehole via the drifts (Pathway 3 [104])	% Mass to reach the roof of the Callovo-Oxfordian formation via the borehole (Pathway 5 [104])
	Mass « entering and passing through the first meters of the sound Callovo-Oxfordian formation » (pathway 1 [104])	Mass « Entering the Drift » (Pathway 2 [104])		
^{129}I	39.40%	60.60%	1.41%	1.37%
^{36}Cl	51.02%	46.88%	0.80%	0.78%
^{14}C	4.44%	2.65%	0.0057%	0.0052%
^{107}Pd	2.14%	2.10%	0.00018%	0.000005%
^{59}Ni	0.95%	0.91%	0.000084%	0.0000029%
^{93}Zr	1.30%	0.92%	0.000091%	0.0000035%
^{79}Se	0.20%	0.16%	0.0015%	0.0014%
^{126}Sn	Nil	0.00030%	Nil	Not applicable
^{99}Tc	Nil	0.00044%	Nil	Not applicable

The percentages are given in relation to the total initial inventory of the repository sub-zone affected by the borehole.

Table 7.4-22 SEA - Abandoned Borehole Penetration of the Repository - Distribution of Transfer Pathways - Mass Leaving the Various Components of the Repository System and the Borehole over a Period of One million Years - C2 reference package - Sensitivity Calculations at the EDZ (Degraded EDZ)

Due to the dead-end architecture, the low permeability of the host formation and the weak hydraulic influence of the borehole, degrading the EDZ and therefore diminishing the performances of the sealings do not have a strong influence on the hydraulic mode in the alls ; it remains mainly diffusive.

As regards the transport and distribution of the transfer pathways between the structures and the geological barrier, the following is observed :

- The behaviour of soluble and non-sorbed elements (^{129}I and ^{36}Cl) is similar to that of the reference situation (see Figure 7.4-49). The iodine 129 activity fraction leaving the borehole is 1.37 % of the total initial activity of the repository sub-zone whereas it is 1.24 % in the reference (See Table 7.4-16 and *Table 7.4-22*) ;

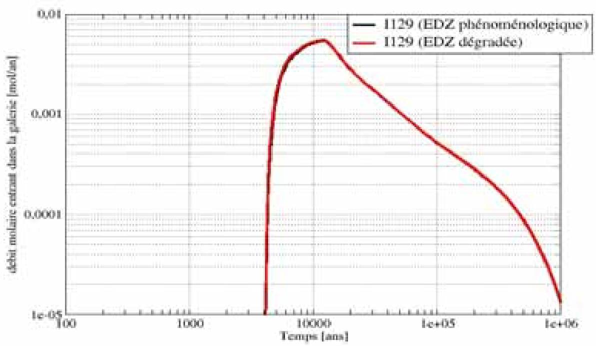


Figure 7.4-49 SEA - Abandoned Borehole Penetration of the Repository - Molar Flow Rate History as a Function of Time Entering the Drift - ^{129}I - C2 - Comparison of the (Best-Estimate) EDZ Model with the (Degraded) EDZ

- Differences can be detected in the case of sorbed elements (see Figure 7.4-50). Indeed, taking conservative retention values into account in the micro-fissured zone increases the mass entering the drifts via this pathway. For example, in the case of tin 126, the activity fraction entering the drifts corresponds to $3\cdot10^{-4}$ % of the initial activity of the sub-zone whereas it is $5.9\cdot10^{-5}$ % in the reference calculations. This result affects the activity fraction reaching the borehole but transport in the drifts does not give rise to a significant additional divergence from the reference calculations. Once in the drifts, the degraded EDZ has little influence on the transfer of radionuclides to the shafts, the latter mainly move through the backfill.

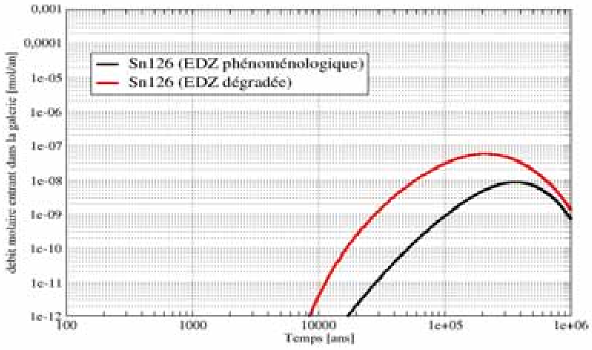
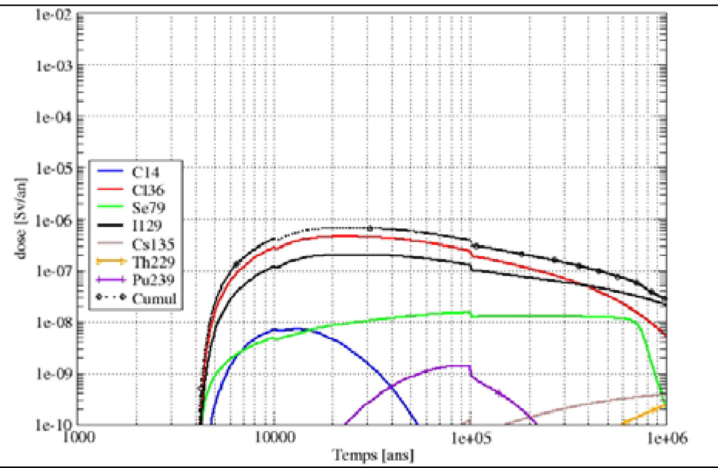


Figure 7.4-50 SEA - Abandoned Borehole Penetration of the Repository - Molar Flow Rate History as a Function of Time Entering the Drift - ^{126}Sn - C2 - Comparison of the (Best-Estimate) EDZ Model and the (Degraded) EDZ

A degraded EDZ has little influence on impact because the main contributors are radionuclides that are not sorbed in the argillites. Moreover, the results show that the dose due to the borehole is similar to that of the reference calculation with a peak dose of 0.00067 mSv/year compared with 0.00065 mSv/year in the reference at 25,000 years (see Figure 7.4-51).



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	6.7E-07	2.5E+04
Cl36	4.5E-07	2.3E+04
I129	2.0E-07	2.6E+04
Se79	1.5E-08	1.0E+05
C14	7.2E-09	1.3E+04
Pu239	1.4E-09	9.0E+04
Cs135	3.8E-10	1.0E+06
Th229	2.4E-10	1.0E+06

Figure 7.4-51 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the « Degraded » EDZ - Dose Due to the Borehole at the DWS Pumping Outlet - C2 reference package

7.4.6.4 Sensitivity to Chemical Parameters

- **Sensitivity to the Parameter of the Clay Engineered Barrier (Conservative Geochemical)**

As seen in Section 7.4.6.4, this study consists of taking into account the conservative geochemical parameters in the bentonite of the cell plug, the buffer, if applicable, and the seals. These parameters are given in Table 7.4-23 and Table 7.4-24.

The study has been carried out for reference package CU1, CU2, C2 and C4.

	Bentonite Engineered Barrier, Plugs and Seals						
	Reference Calculation					Sensitivity to the Geochemical Parameters	
	Period [years]	$\omega_{Diffusion}$ [-]	De [m²/s]	R [-]	Csat [Mol/m^3]	R [-]	Csat [Mol/m^3]
^{10}Be	1 600 000	0.36	5.10^{-10}	973	10^{-2}	1	10
^{10}Be (deltaT>20)	1 600 000	0.36	5.10^{-10}	98	10^{-2}	1	10
^{14}C	5 730	0.05	5.10^{-12}	1	2,3	1	9
^{36}Cl	302 000	0.05	5.10^{-12}	1	soluble	1	soluble
^{41}Ca	103 000	0.36	5.10^{-10}	6	2,3	1	9
^{41}Ca (deltaT>20)	103 000	0.36	5.10^{-10}	1,5	2,3	1	9
^{59}Ni	75 000	0.36	5.10^{-10}	2 430	5.10^{-2}	487	1
^{79}Se	65 000	0.05	5.10^{-12}	1	5.10^{-7}	1	5.10^{-4}
^{93}Zr	1 530 000	0.36	5.10^{-10}	486 000	2.10^{-5}	48 600	3.10^{-3}
^{93}Mo	3 500	0.05	$5.10-^{12}$	1	1.10^{-5}	1	1.10^{-3}
^{93m}Nb	16,4	0.05	5.10^{-12}	350 000	2.10^{-4}	35 000	2.10^{-3}
^{94}Nb	20 300	0.05	5.10^{-12}	350 000	2.10^{-4}	35 000	2.10^{-3}
^{99}Tc	213 000	0.36	5.10^{-10}	146 000	4.10^{-6}	48 600	1.10^{-4}
^{107}Pd	6 500 000	0.36	5.10^{-10}	4 380	4.10^{-4}	4 380	1.10^{-2}
^{126}Sn	100 000	0.36	5.10^{-10}	53 500	1.10^{-5}	14 600	1.10^{-4}
^{129}I	15 700 000	0.05	5.10^{-12}	1	soluble	1	soluble
^{166m}Ho	1 200	0.36	5.10^{-10}	58 300	1.10^{-4}	5 830	1.10^{-3}
^{135}Cs	2 300 000	0.36	5.10^{-10}	487	soluble	290	soluble
^{135}Cs (deltaT>20)	2 300 000	0.36	5.10^{-10}	50	soluble	30	soluble

Table 7.4-23 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity Reference Values of the Chemical Retention Parameters in the Swelling Clay of the Buffer, Plugs and Seals - Fission and Activation Products

	Bentonite Buffer, Plugs and Seals						
	Reference Calculation					Sensitivity to the Geochemical Parameters	
	Period [years]	$\omega_{diffusion}$ [-]	De [m²/s]	R [-]	Csat [Mol/m^3]	R [-]	Csat [Mol/m^3]
^{244}Cm	18,1	0.36	5.10^{-10}	58 000	4.10^{-04}	5 830	1.10^{-03}
^{240}Pu	6 560	0.36	5.10^{-10}	4 860	2.10^{-04}	3 400	2.10^{-04}
^{236}U	23 400 000	0.36	5.10^{-10}	486 000	7.10^{-04}	48 600	5.10^{-02}
^{232}Th	14 100 000 000	0.36	5.10^{-10}	14 600	6.10^{-04}	14 600	1.10^{-03}
^{245}Cm	8 500	0.36	5.10^{-10}	58 300	4.10^{-04}	5 830	1.10^{-03}
^{241}Pu	14,4	0.36	5.10^{-10}	4 860	2.10^{-04}	3 400	2.10^{-04}
^{241}Am	433	0.36	5.10^{-10}	58 300	4.10^{-04}	5 830	1.10^{-03}
^{237}Np	2 140 000	0.36	5.10^{-10}	4 860	4.10^{-06}	3 400	1.10^{-04}
^{233}U	159 000	0.36	5.10^{-10}	486 000	7.10^{-04}	48600	5.10^{-02}
^{229}Th	7 340	0.36	5.10^{-10}	14 600	6.10^{-04}	14 600	1.10^{-03}
^{246}Cm	4 730	0.36	5.10^{-10}	58 300	4.10^{-04}	5 830	1.10^{-03}
^{242}Pu	374 000	0.36	5.10^{-10}	4 860	2.10^{-04}	3 400	2.10^{-04}
^{238}U	4 470 000 000	0.36	5.10^{-10}	486 000	7.10^{-04}	48 600	5.10^{-02}
^{234}U	246 000	0.36	5.10^{-10}	486 000	7.10^{-04}	48 600	5.10^{-02}
^{230}Th	75 400	0.36	5.10^{-10}	14 600	6.10^{-04}	14 600	1.10^{-03}
^{226}Ra	1 600	0.36	5.10^{-10}	2 900	soluble	490	soluble
^{210}Pb	22,3	0.36	5.10^{-10}	15 100	4.10^{-03}	1 510	4.10^{-03}
^{243}Am	7 370	0.36	5.10^{-10}	58 300	4.10^{-04}	5 830	1.10^{-03}
^{239}Pu	24 100	0.36	5.10^{-10}	4 860	2.10^{-04}	3 400	2.10^{-04}
^{235}U	704 000 000	0.36	5.10^{-10}	486 000	7.10^{-04}	48 600	5.10^{-02}
^{231}Pa	32 800	0.36	5.10^{-10}	48 600	1.10^{-03}	4 860	1
^{227}Ac	21,8	0.36	5.10^{-10}	58 300	4.10^{-04}	5 830	4.10^{-04}

Table 7.4-24 SEA - Abandoned Borehole Penetration of the Repository - Bentonite Data Taken into Account in the Calculation - Actinides

CU1 Spent Fuel

Concerning the distribution of the transfer pathways is concerned, comparison of the results with the reference calculations reveals the following points :

- The radionuclides that are already soluble in the reference calculation and whose retardation coefficient does not change (^{129}I, ^{36}Cl, ^{14}C, ^{93}Mo, ^{107}Pd) have similar results to those of the reference calculation. The increase in the limit of solubility has no impact because these radionuclides are already soluble in the reference calculation ;
- The low variation of the retardation coefficient of caesium (barely factor 2) only very slightly modifies the molar flow rate entering the borehole. The high sorption in the sound geological barrier implies a predominant transfer pathway via the geological barrier and barely increases (less than 2 %) the amount of caesium 135 entering the drift ;
- Holmium 166m, Nobium 94 and Tin 26, whose contribution to the borehole is negligible in the reference taking into account their very high sorption in clayey materials, remain totally attenuated in the sensitivity study. Indeed, the reduction of their retardation coefficient by a factor of 10 is not sufficient to make their radiological contribution to the borehole significant : because of their half life, they are attenuated ;

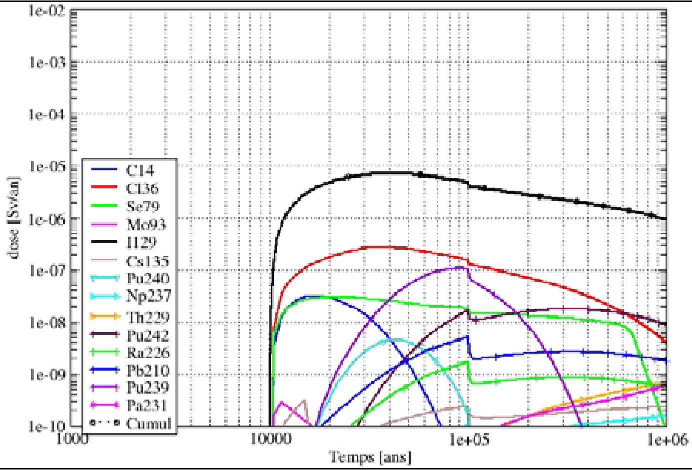
- The increase by three orders of magnitude of the solubility limit of selenium 79 in bentonite increases the molar flow rate leaving the borehole by a factor of three with a maximum reached around 20,000 years (instead of 80,000 years in the case of the reference calculation).

As regards the other radionuclides (^{59}Ni, ^{93}Zr, ^{99}Tc, actinides, etc.), taking a lower retardation coefficient into account bentonite significantly increases the transfer pathway via the drift compared with the reference calculation. However, because of the high sorption in the geological barrier, quantity of radionuclides leaving via the borehole is very low and the contribution of these radionuclides remains negligible (see Table 7.4-25).

Radionuclide	Distribution of Release from the Cells		% Mass to reach the borehole via the drifts (Pathway 3) [104]	% Mass to reach the roof of the Callovo-Oxfordian formation via the borehole (Pathway 5) [104]
	Mass « entering and passing through the first meters of the sound Callovo-Oxfordian Formation » (Pathway 1) [104]	Mass « Entering the drift » (Pathway 2) [104]		
^{129}I	42%	58%	1.45%	1.42%
^{36}Cl	40 %	46.07%	0.82%	0.79%
^{41}Ca	54.60%	36.93%	0.0215%	0.005%
^{135}Cs	9.73%	4.97%	0.0005%	0.000032%
^{14}C	8.19%	3.98%	0.0085%	0.0075%
^{107}Pd	3.22%	3.10%	0.00026%	0.000011%
^{59}Ni	1.99%	2.16%	0.00018%	0.000008%
^{93}Zr	2.87%	2.04%	0.00021%	0.0000115%
^{93}Mo	0.00012%	0.15%	0.00016%	0.00012%
^{79}Se	0.45%	0.44%	0.00415%	0.00395%
^{126}Sn	Nil	0.016%	0.0000010%	0.000000003%
^{99}Tc	Nil	0.0075%	0.00000047%	0.0000000025%
^{166m}Ho	Nil	0.0000023%	Nil	Not applicable
^{94}Nb	Nil	Nil	Not applicable	Not applicable

Table 7.4-25 SEA - Abandoned Borehole Penetration of the Repository - Distribution of Transfer Pathways - Mass Leaving the Various Components of the Repository System and the Borehole over a Period of One Million Years - CU1 reference package - Calculations for Sensitivity to Bentonite Geochemistry

As regards the results of impact (See Figure 7.4-52), the dose remains dominated by iodine 129 with a peak dose that is almost identical to that of the reference calculation (0.0073 mSv/year compared with 0.0072 mSv/year in the reference).

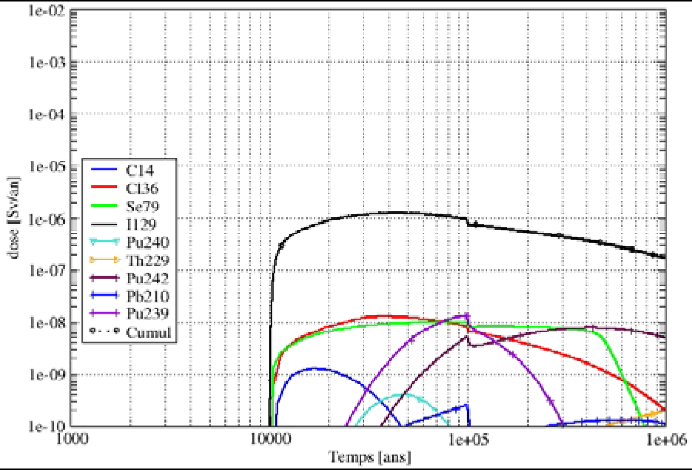


RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	7.3E-06	4.2E+04
I129	6.9E-06	4.2E+04
Cl36	2.7E-07	3.6E+04
Pu239	1.1E-07	9.1E+04
C14	3.1E-08	1.7E+04
Se79	3.0E-08	2.3E+04
Pu242	1.8E-08	3.2E+05
Pb210	5.1E-09	1.0E+05
Pu240	4.5E-09	4.2E+04
Ra226	1.7E-09	1.0E+05
Th229	6.6E-10	1.0E+06
Pa231	6.0E-10	1.0E+06
Cs135	3.1E-10	1.5E+04
Mo93	2.8E-10	1.2E+04
Np237	1.6E-10	1.0E+06

Figure 7.4-52 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the Conservative Geochemical Parameters of Bentonite - Dose Due to the Borehole at the DWS Pumping Outlet - CU1 reference package

CU2 spent fuel

Comments are qualitatively similar to the comments on CU1 spent fuel. Iodine-129 is the main contributor (See Figure 7.4-53) ; the peak dose is identical to the reference dose calculation (0.0012 mSv/year at 43,000 years).



RN	DoseMax[Sv/an]	t(DoseMax)[an]
I129	1.2E-06	4.3E+04
Cumul	1.2E-06	4.4E+04
Cl36	1.3E-08	3.9E+04
Pu239	1.3E-08	1.0E+05
Se79	1.0E-08	7.4E+04
Pu242	7.7E-09	4.2E+05
C14	1.3E-09	1.7E+04
Pu240	4.0E-10	4.8E+04
Pb210	2.5E-10	1.0E+05
Th229	2.0E-10	1.0E+06

Figure 7.4-53 SEA borehole abandoned in the repository – Sensitivity to the conservative geochemical parameters of the bentonite - dose due to the DWS pumping outlet at the borehole – CU2 reference package

Reference package C2 and C4

The dose remains governed by the soluble and non-sorbed radionuclides (^{36}Cl and ^{129}I) ; it is identical to the reference dose calculations for the two reference package.

- **Sensitivity to the glass release model : The « V_0.S ➔ V_r » model instead of the V_0.S model adopted in reference**

This study consists of taking into account a phenomenological vitrified waste release model that makes allowance for the residual dissolution rate (« V_0.S ➔ V_r ») rather than a conservative model V_0.S, based exclusively on the initial glass dissolution rate. This study has been carried out as an illustration for reference package C2. Because of the low Peclet numbers in the cell, that are characteristic of a co-dominant advective/diffusive regime at the cell head, the « V_0.S ➔ V_r » model is tested in this case. Release from the glass matrix increases from 8000 years to over 300,000 years as a result of taking the « V_0.S ➔ V_r » release model in account.

Incorporating a less conservative release model for transport from the waste package to the borehole does not have a significant influence on the distribution of mass between the structures and the geological barrier.

However there are wide contrasts between the two release models for the release kinetics of the transfer pathway via the drift. The release model of the sensitivity study results in molar flow rate peaks that are lower as they enter the drift. By way of example, the maximum molar flow rate of iodine-129 entering the drift is 10 times lower than that obtained with the V_0.S reference calculation model. It peaks after about 300,000 years instead of 12,000 years (See Figure 7.4-54).

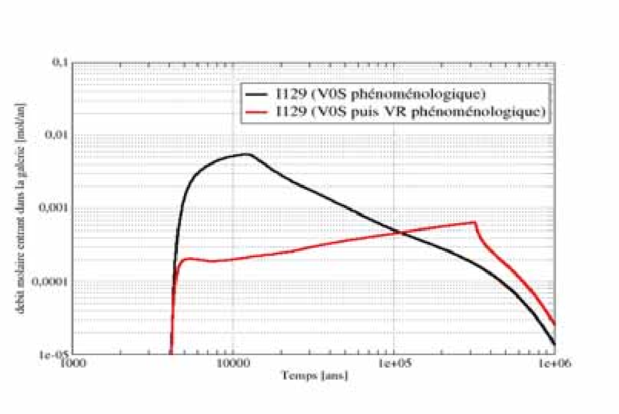


Figure 7.4-54 Borehole abandoned in the repository SEA – Molar flow rate history entering the drift plotted against time – ^{129}I – C2 – Comparison of the V_0.S and V_0.S → V_R models

Furthermore in the case of the V_0.S model, 26 % of the initial inventory of a cell has travelled through the repository after 50,000 years, as against 4 % with the « V_0.S ➔ V_r » model. The instantaneous contribution of the « V_0.S ➔ V_r » model becomes greater after 100,000 years, because in this case the source term is slower to deplete (See Figure 7.4-55)

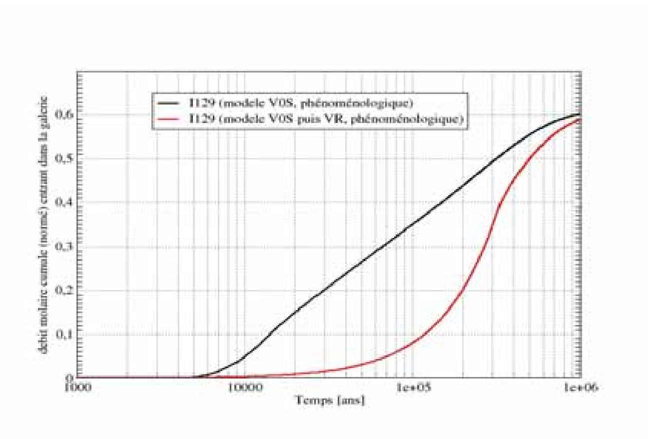


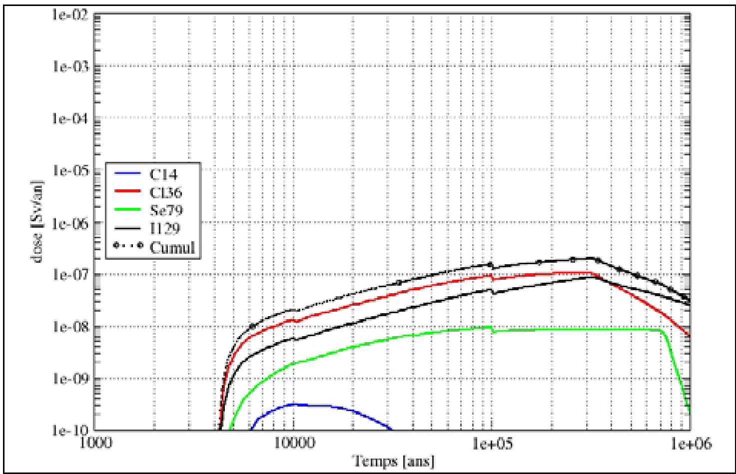
Figure 7.4-55 Borehole abandoned in the repository SEA – Accumulated and normalised molar flow rate entering the drift plotted against time – ^{129}I – C2 – Comparison of the V_0.S and V_0.S → V_r models

The drop in the peak molar flow rate of mobile elements has lower impact at the borehole. Thus, the peak molar flow rate presented by iodine-129 and chlorine-36 in the borehole is five times lower than the reference calculation.

The gain in impact is also visible on slightly sorbed elements in the plug, such as carbon 14. However diffusive transfer times in excess of one million years mask the influence of the slower release if the radionuclides are highly sorbed in the plug (^{126}Sn, ^{99}Tc, certain actinides…). The same is true of selenium 79, limited by precipitation, which masks the influence of the release model.

The impact results reveal that the main contributors to the impact are chlorine-36 and iodine-129 (See Figure 7.4-56). The dose maximum is around 0.0002 mSv/year after about 320,000 years, that is roughly three times lower than for the reference calculation.

These results demonstrate the usefulness of the waste matrix as an additional barrier in the event of borehole drilling.



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	2.0E-07	3.2E+05
Cl36	1.0E-07	2.6E+05
I129	8.8E-08	3.2E+05
Se79	9.3E-09	1.0E+05
C14	3.1E-10	1.0E+04

Figure 7.4-56 Borehole abandoned in the repository SEA – Sensitivity to the glass release model - dose due to drilling a borehole at the DWS pumping outlet – Reference package C2

- **Sensitivity to the vitrified waste release model parameters (conservative parameters)**

This study consisted of considering conservative parameters (glass fracturing and initial dissolution rates – See Table 7.4-26) of the V_0.S model, reducing the release period from 8000 years to 110 years. This calculation case simulates major damage to the glass for example.

	Reference calculations	Sensitivity calculations
Fracturing rate	5	40
Initial glass dissolution rate($g.m^{-2}.j^{-1}$)	0.89	4.45

Table 7.4-26 Glass fracturing and dissolution rates in reference and sensitivity calculations

The results reveal that :

- the molar flow rate peaks of mobile radionuclides entering the drift are increased roughly two-fold in the short term Figure 7.4-57). This is because the source term duration (110 years instead of 8000 years) influences the maximum release level, given the characteristically low diffusion times (for the quantity of radionuclides close to the plug and crossing the 3 metres of bentonite and the fractured zone). Peak occurs just after 4000 years instead of 12,000 years. After 40,000 years, the release levels with the adopted reference release parameters are identical and the influence of the release model is no longer visible ;

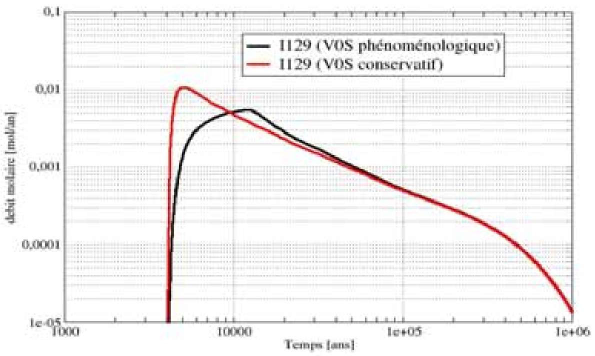
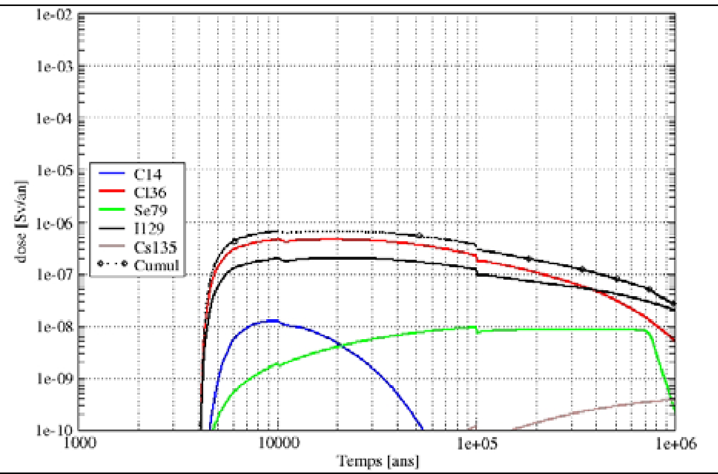


Figure 7.4-57 Borehole abandoned in the repository SEA – Molar flow rate history entering the drift plotted against time – ^{129}I – C2 – Sensitivity to the conservative parameters of the V_0.S glass release model

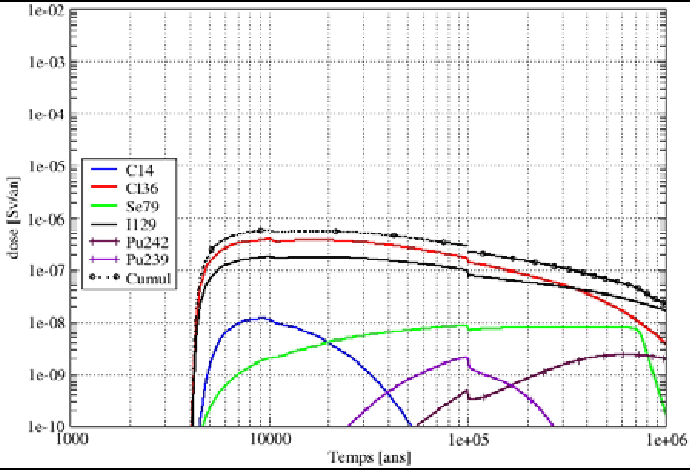
- in the case of the highly sorbed radionuclides (actinides, ^{99}Tc, ^{126}Sn, ^{135}Cs…), the release model does not influence the transfer pathway via the drift (and thus release by the borehole), as the transfer time is the overriding dominant factor. The same is true of not very soluble selenium-79, limited by precipitation, which masks the influence of the source term.

The impact results are provided below for reference package C2 and C4 (See Figure 7.4-58 and Figure 7.4-59). The dose maximum at the « DWS pumping » outlet relates to iodine-129 and chlorine-36. It is very close to the reference calculation, i.e. 0.00066 mSv/year for reference package C2 (vs. the reference value of 0.00065 mSv/year) and 0.00058 mSv/year for reference package C4 (vs. the reference value of 0.00053 mSv/year).



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	6.6E-07	1.9E+04
Cl36	4.5E-07	1.9E+04
I129	2.0E-07	2.1E+04
C14	1.2E-08	9.4E+03
Se79	9.4E-09	1.0E+05
Cs135	3.8E-10	1.0E+06

Figure 7.4-58 Borehole abandoned in the repository SEA – Sensitivity to the conservative parameters of the V0.S glass release model - dose due to drilling a borehole at the DWS pumping outlet – Reference package C2



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	5.8E-07	1.0E+04
Cl36	3.9E-07	1.0E+04
I129	1.8E-07	1.0E+04
C14	1.2E-08	9.4E+03
Se79	8.7E-09	1.0E+05
Pu242	2.4E-09	6.3E+05
Pu239	2.1E-09	1.0E+05

Figure 7.4-59 Borehole abandoned in the repository SEA – Sensitivity to the conservative parameters of the V0.S glass release model - dose due to drilling a borehole at the DWS pumping outlet – Reference package C4

Table 7.4-27 summarizes the results of the impact due to the borehole (at the AEP pumping outlet in the Barrois) combined with the reference calculations and sensitivity studies.

		Sensitivity studies			Reference calculation		
Reference package		Dose maximum (mSv/year)	Date of max. dose [years]	Contributing radionuclides	Dose maximum (mSv/year)	Date of max. dose [years]	Contributing radionuclides
B1x	Borehole in the access drift	$4.9.10^{-5}$	550,000	^{36}Cl ; ^{129}I	0.012	770	^{36}Cl ; ^{93}Mo
B2	No sensitivity study				0.0022	12,000	^{36}Cl ; ^{129}I
C2	Design with clay engineered barrier	Ditto reference			0.00065	25,000	^{36}Cl ; ^{129}I
	« Degraded EDZ »	0.00067	25,000	^{36}Cl ; ^{129}I			
	Phenomenological release model ($V_0.S \rightarrow V_r$)	0.0002	320,000	^{36}Cl ; ^{129}I			
	Conservative parameters of the V_0.S model	0.00066	19,000	^{36}Cl ; ^{129}I			
	Conservative geochemistry of bentonite	Ditto reference					
C4	Conservative geochemistry of bentonite	Ditto reference			0.00053	23,000	^{36}Cl ; ^{129}I
	Conservative parameters of the V_0.S model	0.00058	10,000	^{36}Cl ; ^{129}I			
CU1	borehole in cell	0.1	30,000	^{94}Nb ; ^{129}I ; $^{239.240}Pu$	0.0072	42,000	^{129}I
	Doublet at the ends of the drifts	0.0066	57,000	^{129}I			
	Doublet at the centre of the access drifts	0.011	45,000	^{129}I			
	Conservative geochemistry of bentonite	0.0073	42,000	^{129}I			
CU2	Conservative geochemistry of bentonite	0.0012	43,000	^{129}I	0.0012	43,000	^{129}I

Table 7.4-27 Bore hole abandoned in the repository SEA – Total dose – dates of dose maxima and main contributors to the DWS pumping outlet of the Barrois – reference and sensitivity calculations

7.4.7 Lessons drawn from the SEA borehole

The study of the « borehole » scenario has revealed that the repository system is robust when subject to the drilling followed by the abandonment of one or more boreholes in the repository.

This is because, regardless of the situations studied, even if the release models considered for vitrified waste and transport parameters are pessimistic and conservative for the EDZ or bentonite, for most situations (excluding direct boring into the cells) the impact combined with the activity fraction leaving the borehole is of the same magnitude as the impact combined with the rest of the normal evolution activity. However in the light of the borehole occurrence date which is deliberately pessimistic and consistent with the recommendations of Basic Safety Rule RFS.III.2.f, the dose maxima appear earlier. The most pessimistic situations are those in which the borehole is drilled into a waste cell (reference case for B waste and the sensitivity case study for spent fuel).

The results reveal that the borehole has limited hydraulic consequences, primarily because of the low permeability of the geological barrier and the effectiveness of the seal in « resist to the flow of water ». Even in the event of drilling a borehole doublet in the spent fuel sub-zone access drifts, the low permeability of the geological barrier, the « dead-end » architecture and horizontal head gradient would prevent a U-shaped flow developing, which would probably result in a significant rise in the contribution of one of the doublet boreholes ; as the latter individually present an impact lower or equal to that of a single borehole.

It is demonstrated that the main contributors to impacture soluble non-sorbed elements such as iodine and chlorine. Highly sorbed elements in the geological barrier often present negligible impact at the borehole, as they have either migrated preferentially into the geological barrier, or been subject to radioactive decay before reaching the borehole. The various sensitivity studies have revealed that the dose is not particularly sensitive to the conservative geochemistry of the bentonite nor to the degraded parameters of the EDZ.

Allowing for the alternative engineered core design for C waste does not offer any potential gain with regard to the impact either, as the latter is absolutely identical to the result obtained with the reference design. However the waste matrices are a useful barrier, as incorporating slower release models limits the dose.

To conclude, the results confirm that the system is robust when subjected to external events such as an intrusive borehole into the repository. As the geological barrier is partly bypassed, the « delay and attenuate the migration of radionuclides function » is certainly degraded, but offers some effectiveness for elements sorbed in the geological barrier by forcefully limiting the fraction of activity reaching the borehole.

7.5 The « severely degraded evolution » scenario

7.5.1 Definition and purpose of the scenario

The « severely degraded evolution » scenario is defined to radically lower the performance levels of the three main safety functions all together as they affect transfers by water. Models and parameters, which are generally less favourable than those used as the SEN reference are therefore simultaneously adopted for all repository components. This is a conventional scenario, which does not really present any physical validity, with the purpose of verifying the degree of complementarity of the functions. As it has already been stated that no altered scenarios led to impacts that would significantly diverge from that of the SEN, it appeared useful to carry out further tests on safety functions behaviour in a severely degraded multiple failure situation : by degrading all safety functions simultaneously, and by comparing the result with those of the normal evolution scenario ; we can observe whether minimum performance levels, apart from what is normally expected, complement each other sufficiently to enable the impact to be controlled.

This scenario is not subject to any particular dose constraint as it is solely conventional.

7.5.2 Scenario processing

The scenario is based on the general SEN pattern coupled with greatly degraded parameter values and pessimistic models. Various types of modifications have been made to the SEN, appropriate to the functions being degraded.

- **Modification to degrade the « resist to the flow of water » function**

It emerged in chapter 5 that up to permeability levels of around 10^{-12} m/s, the Peclet number in the argillites sustains a diffusive regime. Leaving aside the measurement values of experimental protocols which are not yet fully understood, no permeability measurement value in the Callovo-Oxfordian exceeds this 10^{-12} m/s threshold. Therefore the latter is adopted for vertical and horizontal permeability levels.

Furthermore, the « degraded » damaged zone model is adopted. Its hydraulic characteristics are given *in Table 7.5-1.*

On account of the highly pessimistic hydraulic characteristics adopted for the EDZ, the seals have low performance levels bordering on complete failure. Indeed, the permeability which is considered as equivalent as to the fractured EDZ, the hydraulic cut-off and their interfaces, over the length of the seal, is about 10^{-7} m/s and that of the microfissured zone surrounding it about 5.10^{-9} m/s. The possibility of an increase of gas pressure due to corrosion is not considered. This type of effect is not represented during the hydraulic transient.Additionally, degradation of back-fill permeability is allowed for, although its effects are expected to be minor (10^{-6} m/s).

Table 7.5-1 summarises the permeability values adopted in the various repository components and in the Callovo-Oxfordian.

	« Severely degraded evolution » scenario	SEN (Reminder)
Fractured zone	$K = 10^{-6}$ m/s	$K = 5.10^{-9}$m/s
Microfissured zone	$K = 5.10^{-9}$ m/s	$K = 5.10^{-11}$ m/s
Sound Callovo-Oxfordian	$Kv = Kh = 10^{-12}$ m/s	$Kv = 5.10^{-14}$ m/s ; $Kh = 5.10^{-13}$ m/s
Bentonite	$K = 10^{-11}$ m/s	$K = 10^{-11}$ m/s
Concrete	$K = 10^{-6}$ m/s	$K = 10^{-6}$ m/s
Backfill	$K = 10^{-6}$ m/s	$K = 10^{-8}$ m/s
* Owing to the high permeability of the EDZ (fractured and microfissured zone), the performance of the seals – even with a permeability of 10^{-11} m/s for bentonite – is very low : the overall effectiveness of their hydraulic cut-off is of the order of 10^{-7} m/s.		

Table 7-5-1 Severely degraded evolution SEA – Values of the hydraulic parameters in the sound Callovo-Oxfordian, the associated EDZ and the engineered structures in bentonite or concrete

In principle the transport pathway via the access structures cannot be left out of the computation. Consequently transport firstly through the drifts and shaft and secondly through the geological environment are represented explicitly in two different models. Bearing in mind the expected transfer times, which should be short, the transfer pathway via the structures is calculated using the hydrogeological present-day model. In this model, the gradient of the vertical heads ascending in the Callovo-Oxfordian is 0.2 m/m

At the same time the transfer pathway via the sound host formation is calculated. It has been decided to adopt the geoprospective model (1 million years) for this transfer pathway, which maximises the gradient of vertical heads ascending in the Callovo-Oxfordian (0.4 m/m). This is because transport in the host formation could become diffusive/advective codominant and thus sensitive to the gradient given the fact that for the vertical component, the permeability of the geological barrier is increased by a factor of 20. Thus incorporating the million-year period model maximises the molar flow rate leaving the Callovo-Oxfordian, and reduces the transfer time to the surrounding formations.

- **Modification to degrade the « limiting the release of radionuclides and immobilizing them in the repository » function**

Conservative release models are adopted for calculating all the reference packages of the inventory model :

- labile release for waste packages disposed of in B1x cells (reference packages B1, B5 excluding B5.1 and B6 excluding B6.1), This choice broadly covers all uncertainties relating to localised pitting corrosion hazards, the variable characteristics of the waste or the possible presence of oxidising elements from the radiolysis of water ;
- release after 1000 years for the bituminised sludge corresponding to the most pessimistic model in which the release is controlled by water take-up, by overcoming both the geometry of the embedded waste and the solubility of the radionuclides that it contains ;
- a V_0.S model with conservative parameters for vitrified waste. In particular, a rapid initial glass dissolution rate and a pessimistic fracturing rate of 40 (versus 5 in SEN) are adopted, leading to a release period of a few thousand years at most ;

- a conservative spent fuel release model. On the one hand, a radiolytic dissolution model with conservative parameters is adopted. It consists of adopting a conservative burnup fraction and multiplying the dissolution rates obtained by 10. In this case, the release times associated with radiolytic dissolution are a few thousand years. On the other hand, conservative labile activity percentages are adopted in the matrix.

The « severely degraded » scenario, just like the reference SEN, only has a few defective contains plugs. However, a sensitivity study involving failure of all the inventory waste package, identical to that considered for the « waste package failure » SEA is adopted to additionally degrade the « limiting the release of radionuclides and immobilizing them in the repository » function. The thermal load in the repository is the load evaluated with the conservative 2D model.

It is also pointed out that considering no geochemical properties of the fracture zone (infinite solubility) permits to simulate the effect of colloidal transport in that zone for fission and activation products subject to the calculation (see chapter 6). For the actinide clusters, it is considered highly unlikely that they can move over very long distances, under moderate advection conditions, without breaking it up.

- **Modification to degrade the « delay and attenuate the migration of radionuclides » function**

Consideration is made of :

- conservative transport parameters in the Callovo-Oxfordian, and in the microfissured part of the EDZ ;
- conservative retention parameters in the Callovo-Oxfordian, in the microfissured part of the EDZ and in the concrete and clay structures (engineered barrier).

The fractured zone is represented in a penalising manner : no chemical retention is considered in this zone and pessimistic transport parameters are adopted (diffusion coefficient of 2.10^{-9} m²/s).

These values are the same as those adopted for SEN sensitivity studies.

These values are the same as the ones adopted in the SEN sensitivity studies for the 15 radionuclides covered in the safety studies ; they are presented again in *Table 7.5-2*, Table 7.5-3 and Table 7.5-4 with regard to the sound Callovo-Oxfordian and the EDZ, the bentonite engineered structures and the concrete engineered structures respectively.

	Half-life [years]	ωDiffusion [-]	De [m²/s]	R [-]	Csat [mol/m³]
	Callovo-Oxfordian and microfissured zone (Fractured zone : $K = 10^{-6}$ m/s ; Dp =De/ωDiffusion = 2.10^{-9} m²/s and no chemical retention)				
^{10}Be	1,600,000	0.21	5.10^{-10}	1	10
^{10}Be (deltaT>20°C)	1,600,000	0.21	5.10^{-10}	1	10
^{14}C	5,730	0.04	10^{-11}	1	9
^{36}Cl	302,000	0.04	10^{-11}	1	soluble
^{41}Ca	103,000	0.21	5.10^{-10}	1	9
^{41}Ca (deltaT>20°C)	103,000	0.21	5.10^{-10}	1	9
^{59}Ni	75,000	0.21	5.10^{-10}	1,100	1
^{79}Se	65,000	0.04	10^{-11}	1	5.10^{-4}
^{93}Zr	1,530,000	0.21	5.10^{-10}	1,100	3.10^{-3}
^{93}Mo	3,500	0.04	10^{-11}	1	1.10^{-3}
^{93m}Nb	16.4	0.04	10^{-11}	43,100	2.10^{-3}
^{94}Nb	20,300	0.04	10^{-11}	43,100	2.10^{-3}
^{99}Tc	213,000	0.21	5.10^{-10}	21,900	1.10^{-4}
^{107}Pd	6,500,000	0.21	5.10^{-10}	1,750	1.10^{-2}
^{126}Sn	100,000	0.21	5.10^{-10}	65,700	1.10^{-4}
^{129}I	15,700,000	0.04	10^{-11}	1	soluble
^{166m}Ho	1,200	0.21	5.10^{-10}	54,800	1.10^{-3}
^{135}Cs	2,300,000	0.21	5.10^{-10}	R=f[Kd $_{25°C}$(Langmuir)*]	soluble
^{135}Cs (deltaT>20°C)	2,300,000	0.21	5.10^{-10}	$Kd_{\Delta T>25°C} = 0.1\ Kd_{25°C}$	soluble
Langmuir : $Kd_{25°C} = (1,85.10^{-7}/(4,76.10^{-7}+ Ceq)$ where Ceq = concentration in solution (mol/l)					

Table 7.5-1- Severely degraded evolution SEA – Transport and chemical retention parameter values in the sound Callovo-Oxfordian and in the microfissured zone of the EDZ

	Half-life [years]	ωDiffusion [-]	De [m²/s]	R [-]	Csat [mol/m^3]
		Bentonite engineered structures K = 10^{-11} m/s			
^{10}Be	1,600,000	0.36	5.10^{-10}	1	10
^{14}C	5,730	0.05	5.10^{-12}	1	9
^{36}Cl	302,000	0.05	5.10^{-12}	1	soluble
^{41}Ca	103,000	0.36	5.10^{-10}	1	9
^{59}Ni	75,000	0.36	5.10^{-10}	487	1
^{79}Se	65,000	0.05	5.10^{-12}	1	5.10^{-4}
^{93}Zr	1,530,000	0.36	5.10^{-10}	48,600	3.10^{-3}
^{93}Mo	3,500	0.05	5.10^{-12}	1	1.10^{-3}
^{93m}Nb	16.4	0.05	5.10^{-12}	35,000	2.10^{-3}
^{94}Nb	20,300	0.05	5.10^{-12}	35,000	2.10^{-3}
^{99}Tc	213,000	0.36	5.10^{-10}	48,600	1.10^{-4}
^{107}Pd	6,500,000	0.36	5.10^{-10}	4,380	1.10^{-2}
^{126}Sn	100,000	0.36	5.10^{-10}	14,600	1.10^{-4}
^{129}I	15,700,000	0.05	5.10^{-12}	1	soluble
^{166m}Ho	1,200	0.36	5.10^{-10}	5,830	1.10^{-3}
^{135}Cs	2,300,000	0.36	5.10^{-10}	290	soluble
^{135}Cs (deltaT>20°C)	2,300,000	0.36	5.10^{-10}	30	soluble

Table 7.5-2 Severely degraded evolution SEA – Transport and chemical retention parameter values in the bentonite engineered structures (cell plugs and seals)

	Half-life [years]	ωDiffusion [-]	De [m²/s]	R [-]	Csat [mol/m^3]
		Concrete engineered structures K= 10^{-6} m/s			
^{10}Be	1,600,000	0.3	6.10^{-10}	1.7	soluble
^{14}C	5,730	0.3	6.10^{-10}	1	$4{,}0.10^{-2}$
^{36}Cl	302,000	0.3	6.10^{-10}	1	soluble
^{41}Ca	103,000	0.3	6.10^{-10}	1	30
^{59}Ni	75,000	0.3	6.10^{-10}	14,000	$2{,}3.10^{-4}$
^{79}Se	65,000	0.3	6.10^{-10}	12	5.0
^{93}Zr	1,530,000	0.3	6.10^{-10}	280,000	$6{,}0.10^{-3}$
^{93}Mo	3,500	0.3	6.10^{-10}	1	$7{,}0.10^{-4}$
^{93m}Nb	16.4	0.3	6.10^{-10}	70,000	$2{,}4.10^{-4}$
^{94}Nb	20,300	0.3	6.10^{-10}	70,000	$2{,}4.10^{-4}$
^{99}Tc	213,000	0.3	6.10^{-10}	1	soluble
^{107}Pd	6,500,000	0.3	6.10^{-10}	4,200	$1{,}0.10^{-2}$
^{126}Sn	100,000	0.3	6.10^{-10}	70,000	$3{,}0.10^{-5}$
^{129}I	15,700,000	0.3	6.10^{-10}	1	soluble
^{166m}Ho	1,200	0.3	6.10^{-10}	700,000	$2{,}0.10^{-3}$
^{135}Cs	2,300,000	0.3	6.10^{-10}	4.5	soluble

Table 7.5-3 Severely degraded evolution SEA – Transport and chemical retention parameter values in the concrete engineered structures (B waste disposal packages)

Table 7.5-5 provides a summary of the hydraulic, transport and chemical retention values adopted for actinides.

	Callovo-Oxfordian Kh = Kv = 10^{-12} m/s De = 5.10^{-10} m²/s ωDiffusion = 0.21			Bentonite engineered structures K = 10^{-11} m/s De = 5.10^{-10} m²/s ωDiffusion = 0.3		
	Kd [m^3/kg]	R [-]	Csat [mol/m^3]	Kd [m^3/kg]	R [-]	Csat [mol/m^3]
Am, Cm, Ac	5	54,760	10^{-3}	1.2	5,830	10^{-3}
Np	0.6	6,570	10^{-4}	0.7	3,400	10^{-4}
Pb	0.16	1,750	4.10^{-3}	0.3	1,510	4.10^{-3}
Pu	0.6	6,570	2.10^{-4}	0.7	3,400	2.10^{-4}
Pa	0.5	5,480	1	1	4,860	1
Th	1	10,950	10^{-3}	3	14,600	10^{-3}
U	1.3	14,240	$5.\ 10^{-2}$	10	48,600	5.10^{-2}

Table 7.5-4 « Severely degraded evolution » SEA – Summary of transport and chemical retention parameters in the Callovo-Oxfordian and bentonite engineered structures for actinides

Figure 7.5-1 summarises the main hypotheses adopted with regard to hydraulics, transport and chemical retention on each repository component and the Callovo-Oxfordian.

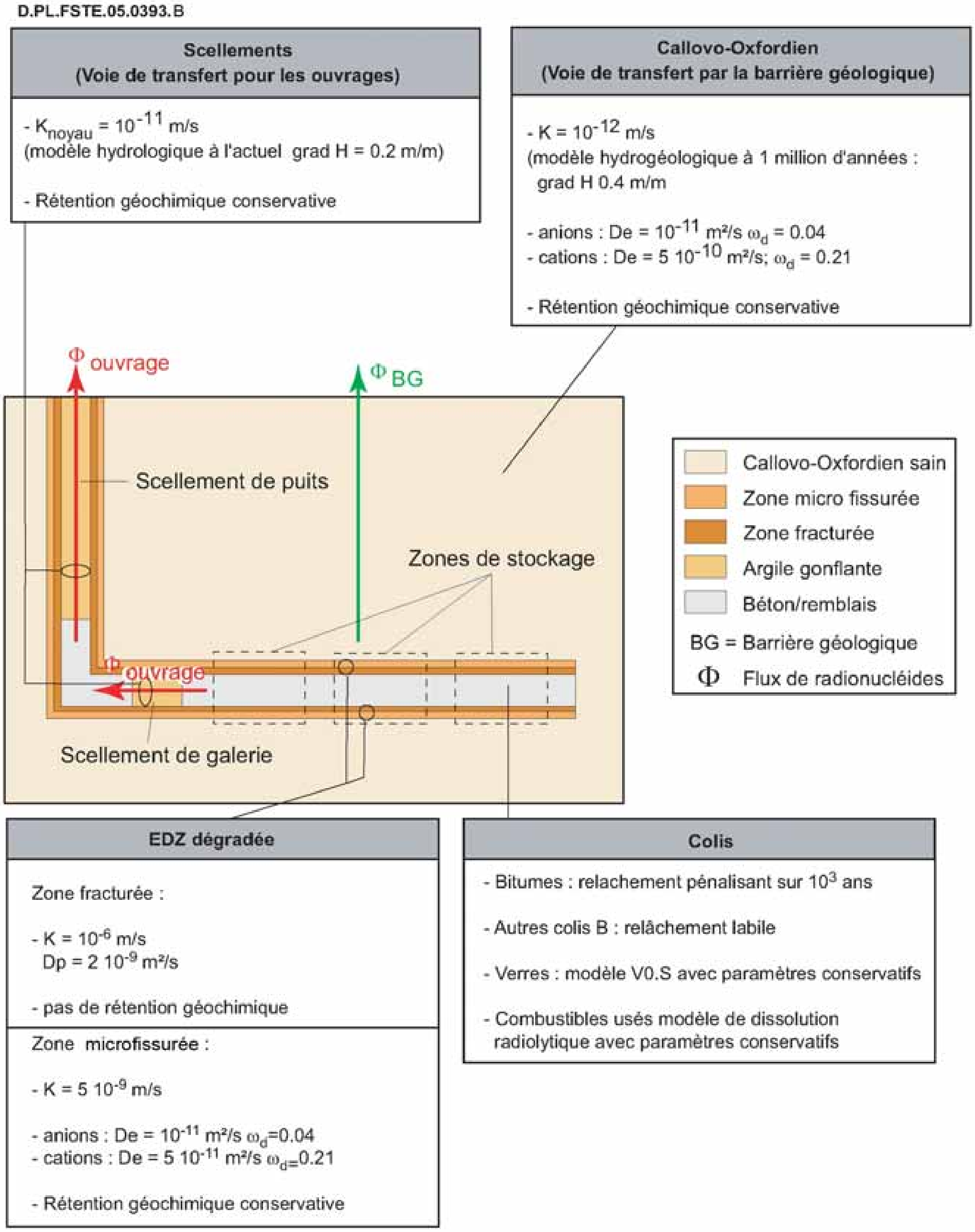


Figure 7.5-1 Main hypotheses adopted for each repository component

7.5.3 Effect on the safety functions

The overall results are presented, all functions taken together, adopting an approach of following the radionuclide transfer. The aim is to distinguish the transfer pathway via the structures from the transfer pathway via the geological environment.

7.5.3.1 Transfer pathway via the structures

Here the focus is to assess the consequences of incorporating a severely degraded situation on the magnitude of the transfer pathway via the access structures. The results are summarized hereafter.

The high permeability level of the EDZ and sound Callovo-Oxfordian (factor 20 on the vertical component) generates significant hydraulic disturbance. The hydraulic flow emerges from the repository via the shaft at a rate of 24 m^3/year (See Figure 7.5-2), that is 50 times higher than in an SEN, and 10 times higher than the « seal failure » altered scenario in the « all seals failed » case. This is because the high permeability of the Callovo-Oxfordian generates higher head gradients between the repository and the shaft, resulting in increasing the flow rates along the repository drifts and increasing transfers by advection

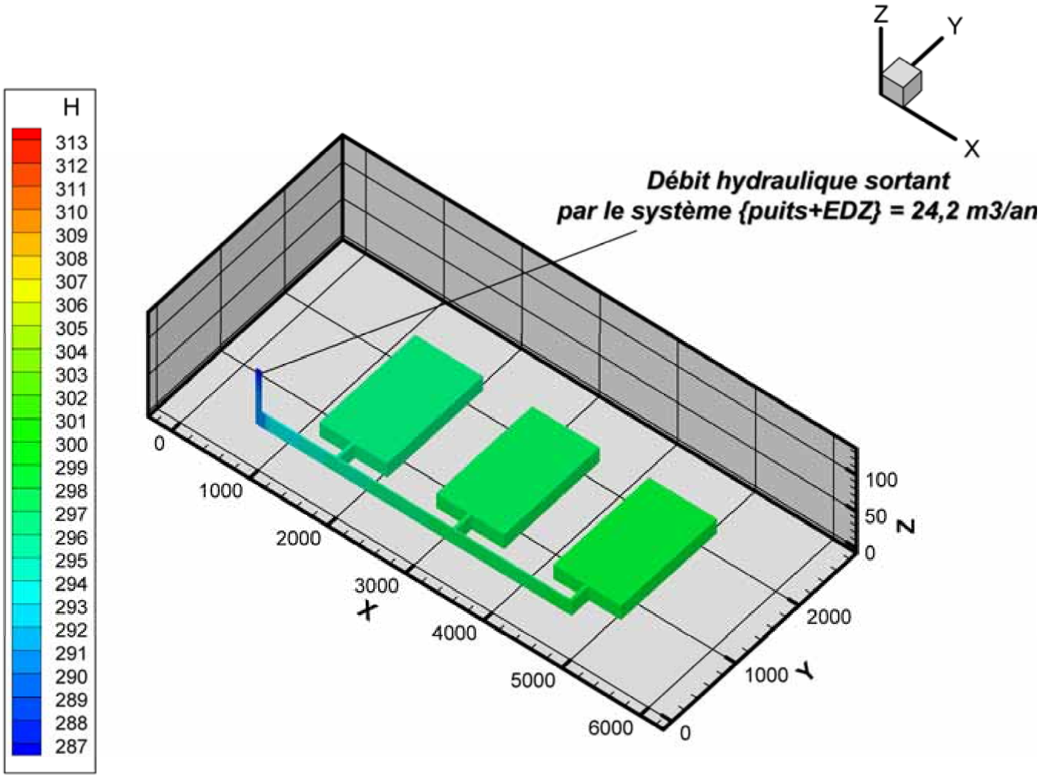


Figure 7.5-2 Severely degraded evolution SEA – Distribution of the hydraulic heads in the repository access structures – Severely degraded evolution – example of spent fuel

- **At the cells**

In contrast to the « seal failure » SEA, the « severely degraded evolution » scenario that combines a degraded EDZ with a higher geological barrier permeability level (See Figure 7.5-3), generates a advective system in the fractured zone at the head of the spent fuel cell with a 10-fold higher Peclet number (equal to 7 as opposed to 0.7 in the SEN). However the system remains diffusive in the cell where the Peclet number is 0.002 (as opposed to 7.10^{-4} in the SEN).

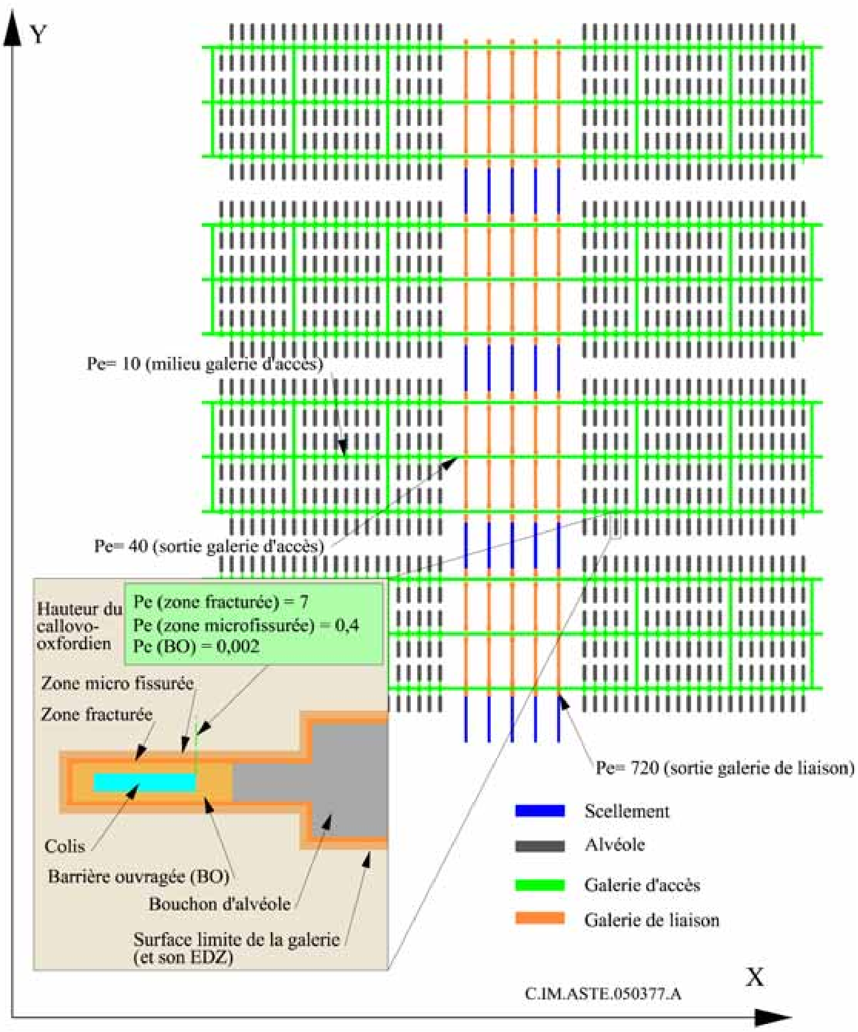


Figure 7.5-3 « Severely degraded evolution » SEA – Reference package CU1 – ^{129}I – Peclet number values in the structures and spent fuel cells

When combined with a conservative source term, the degraded parameters of the EDZ and the hydraulic parameters in the bentonite and host formation, contribute to increasing the maximum molar flow rate entering the drift by one order of magnitude, in relation to the SEN reference calculation, (in the example of iodine-129 of reference package CU1), and the mass of ^{129}I entering the drift over 1 million years by 7 % (66 % as against 59 %). Note that the mass entering the drift in is identical to the SEN sensitivity calculation (See section 5.5.6.1). The increase in permeability of the Callovo-Oxfordian boosts the advective kinematics in the structures and host formation. However, in comparison with the study of sensitivity to the degraded parameters of the EDZ, the advective flow in the cell head is too weak to increase the mass migrating from the cells to the drifts.

Even with high diffusion, advection also becomes the dominating factor over the 50 metres at the cell head and in high permeability zones (fractured and microfissured zones) for B waste cells (See Figure 7.5-4).

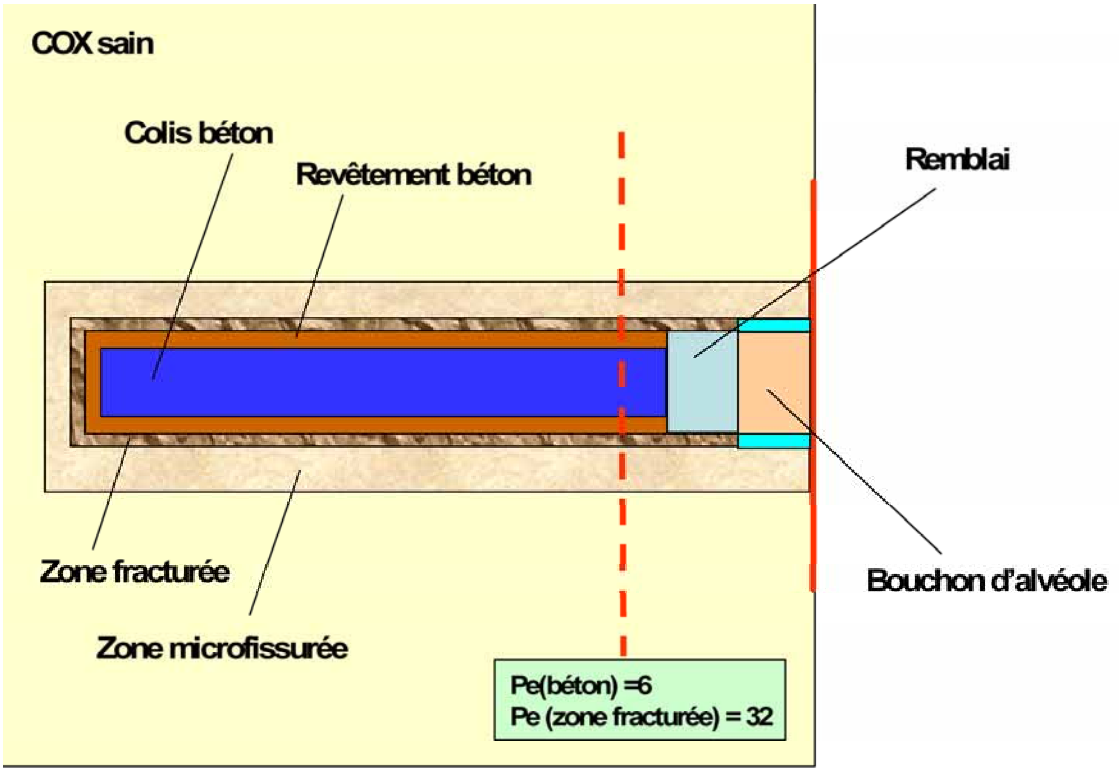


Figure 7.5-4 « Severely degraded evolution » SEA – B reference package disposed of in B1x cells – ^{129}I – Peclet number values in the cells

« Degradation » of the various transport and hydraulic parameters broadly boosts the diffusive and advective kinematics simultaneously.

- **In the repository zone**

Influenced by the upstream structures, the water flows become advective in the repository drifts of a zone (Peclet number of about 10 in the middle of the repository zone). The advective nature becomes stronger leaving the repository zone (Peclet number of about 720) with typically short transport times (See Figure 7.5-3). The molar flow rate leaving the repository zone peaks after 20,000 years as against 100,000 years in the SEN (See Figure 7.5-5).

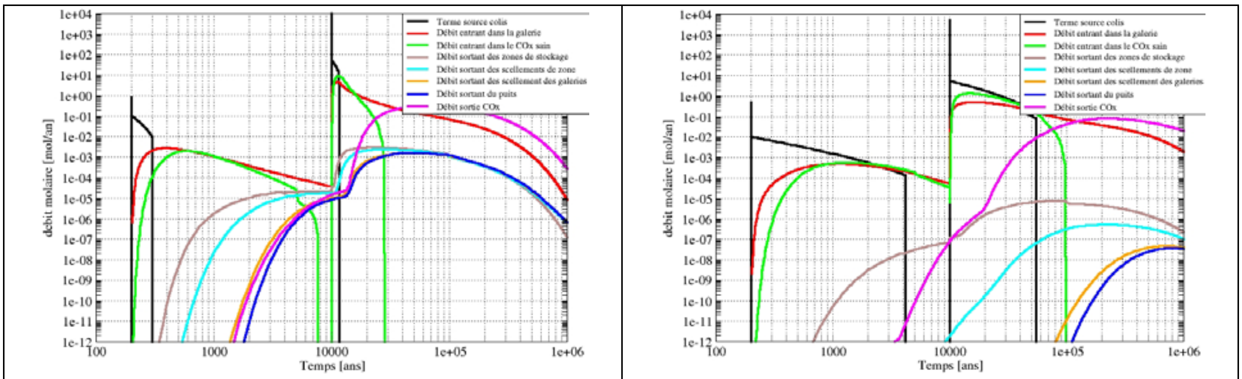
Consequently, in the example of the CU1 package (See Figure 7.5-5), the integrated amount of ^{129}I leaving the repository zone is much higher after 1 million years than in the SEN (factor of 60). The maximum molar flow rate leaving the repository zone is increased by a factor of about 70.

For the B waste âckages, the same tendency is observed (see Figure 7.5-6).

Nevertheless, the length travelled in the drifts are long (several hundreds of meters) and exchanges with the host formation mitigate the importance of this transfer pathway. This contributes to reducing the quantity of radionuclides leaving the repository zone : only 0.5 % of the initial quantity of iodine-129 leaves this zone.

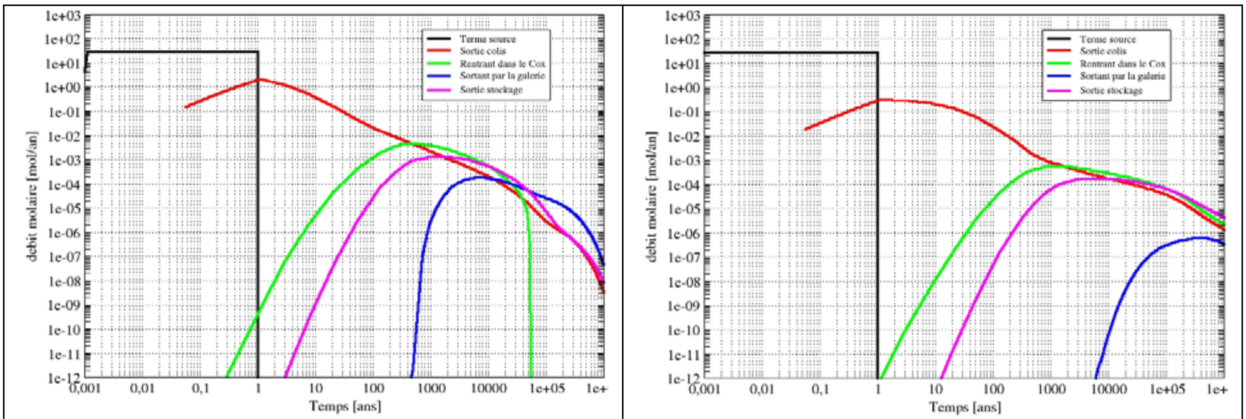
- **In the access structures**

The Iodine 129 mass leaving via the shaft increases by a factor of 10,000 (0.38 % of the initial mass vs. 3.10^{-5} % in the SEN as advection is very dominant in the main connecting drifts, with very short transfer times – See Figure 7.5-7) than in the SEN, peaking after about 50,000 years instead of after 800,000 years (See Figure 7.5-5).



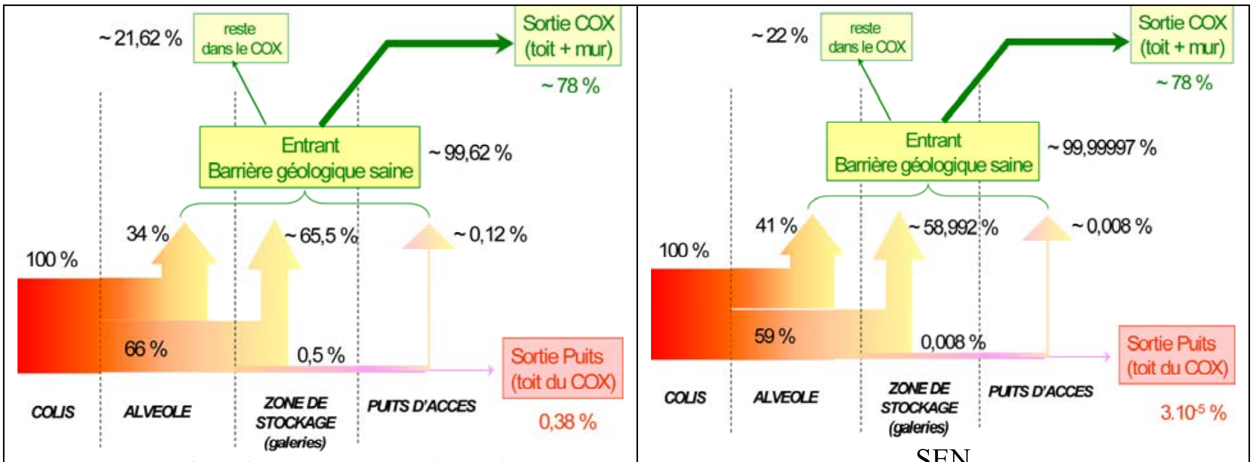
« Severely degraded evolution » SEA — SEN

Figure 7.5-5 « Severely degraded evolution » scenario – Reference package CU1 – ^{129}I – Molar flow rate history



SEA « fonctionnement très dégradé » — SEN

Figure 7.5-6 « Severely degraded evolution » scenario – Reference package B1x – ^{129}I – Molar flow rate history



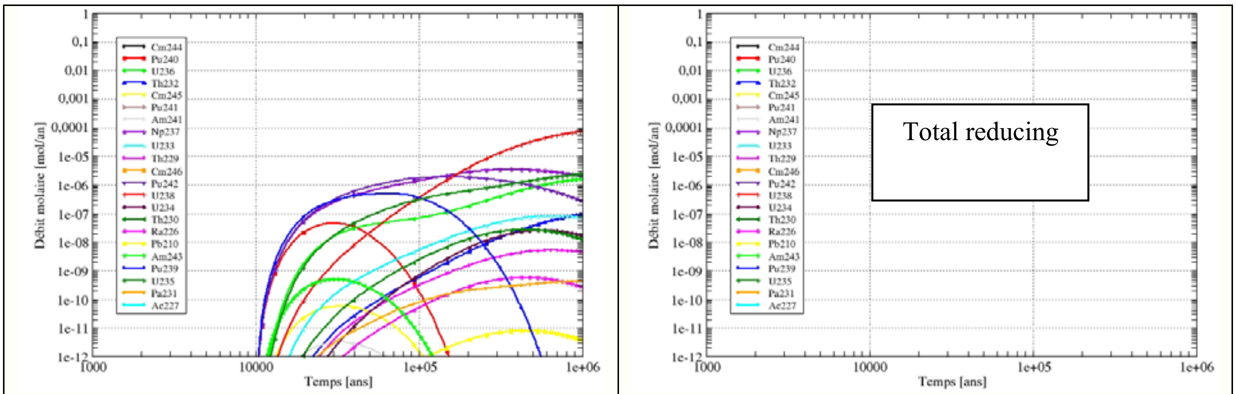
SEA « fonctionnement très dégradé » SEN

Figure 7.5-7 « Severely degraded evolution » scenario – Reference package CU1 – ^{129}I – Distribution of transfer pathways

Thus the allowance made for conservative transport, hydraulics and geochemical performance levels of the host formation, clay engineered barrier and EDZ increase the fraction of activity that migrates towards the access drifts. This is because the pessimistic setting of the host formation permeability level at 10^{-12} m/s (for the vertical and horizontal component) combined with a degraded EDZ (which equates to ineffective seal anchorages), generates significant advection inside the structures.

Note that the molar flow rate leaving the shaft increases significantly in relation to the SEN reference calculation. Moreover it appears earlier than the peak molar flow rate leaving the host formation. By way of example for CU1 spent fuel, the peaks molar flow rate leaving the Callovo-Oxfordian is about 10^{-1} mol/year after 200 000 years as against about 10^{-3} mol/year after 50,000 years when it leaves the shaft. However the mass leaving the shaft over the whole analysis period is still low by comparison with the mass that travels via the host formation. This is because it only accounts for 0.38 % of the total inventory, the other fraction of the mass travelling via the Callovo-Oxfordian transfer pathway.

The results highlight that actinide sorption in the Callovo-Oxfordian is sufficient for them to take the transfer pathway through the geological barrier as a matter of preference. Consequently the attenuation of the molar flow rate of the actinides out of the zone seal is complete as illustrated in Figure 7.5-8.

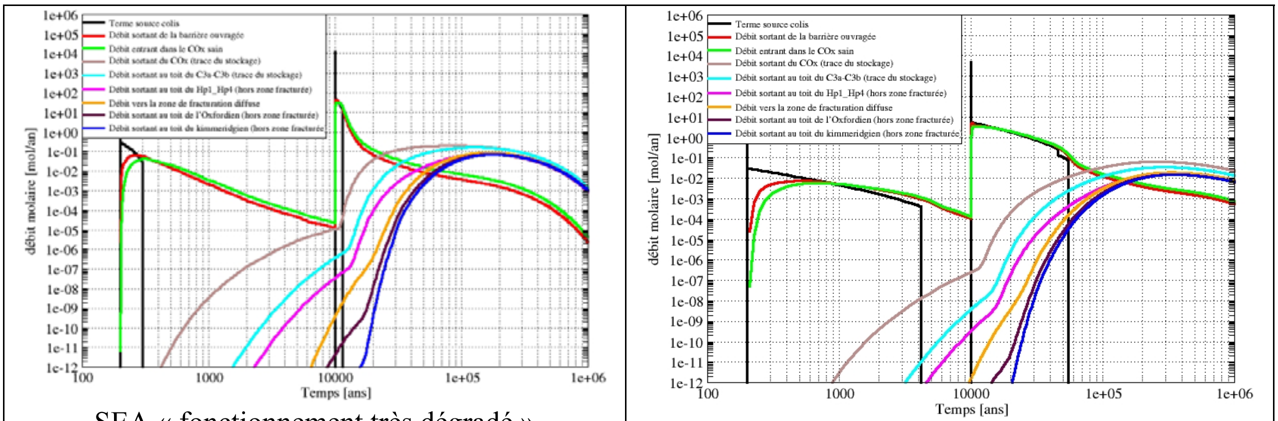


Molar flow rate into the drift (CU2 cell) Molar flow rate out of the CU2 zone seal (all cells)

Figure 7.5-8 : Severely degraded evolution scenario – history of molar flow rates into the drift (one CU2 cell) and out of a CU2 repository zone seal (all cells).

7.5.3.2 Transfer pathway by the host formation

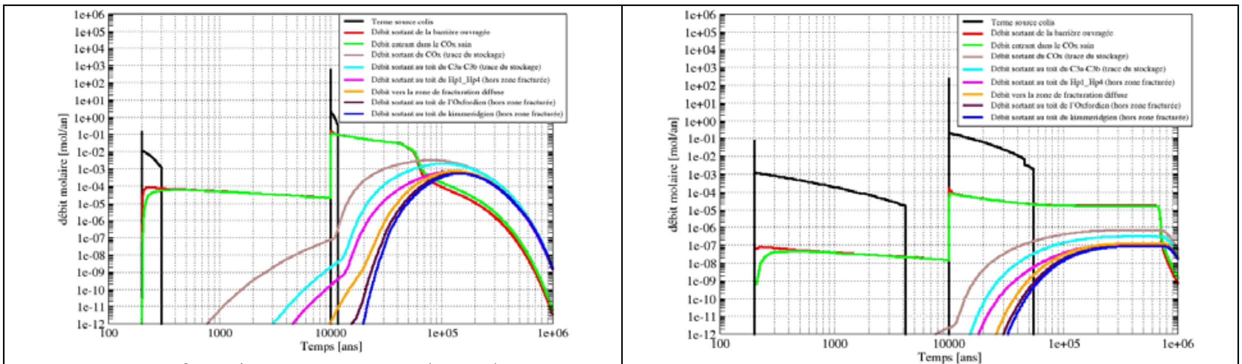
In terms of the transfer pathway by the host formation, the results of the calculations are similar to those already found during the sensitivity study of the SEN conducted on the transfer and retention parameters (See section 5.5.6.1) ; the molar rates are given as an example for the ^{129}I of CU1 spent fuels (See Figure 7.5-9). The maximum molar rate coming out of the Callovo-Oxfordian appears significantly earlier than in the SEN (breakthrough time of the order of 90 000 years). Remember, that because a higher solubility limit was used, ^{79}Se is not limited in the calculation by its solubility (See Figure 7.5-10). It also benefits less from radioactive decay, because of the shorter transfer time. For wastes where it occurs in higher quantities, it becomes the main contributor to the impact.



SEA « fonctionnement très dégradé »

SEN

Figure 7.5-9 « Severely degraded evolution » scenario - CU1 reference package– ^{129}I – History of molar rates –« geological barrier » transfer pathway



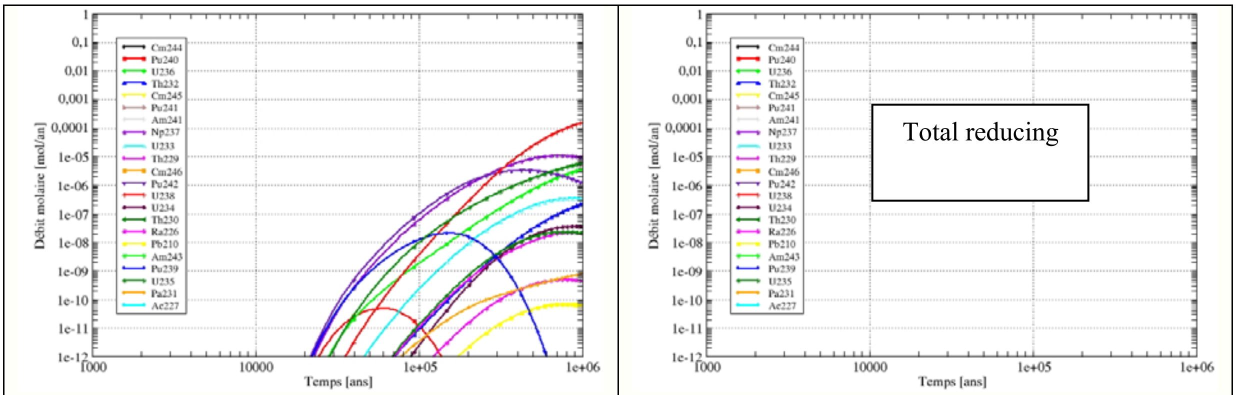
SEA « fonctionnement très dégradé »

SEN

Figure 7.5-10 « Severely degraded evolution » scenario - CU1 reference package– ^{79}Se – History of molar rates –« geological barrier » transfer pathway

Furthermore, the slightest retention of actinides in the Callovo-Oxfordian argillites is insufficient to induce a contribution to the impact from the actinides. Even if the attenuation in the molar flow rates in the Callovo-Oxfordian at a distance of 7 metres from the cells is less than the attenuation observed in the SEN, it is total at the limit of the Callovo-Oxfordian (see Figure 7.5-11). The theoretical characteristic times of migration by diffusion calculated for a depth of 60 metres of transport in the Callovo-Oxfordian remain very high, and much higher at the time scale considered for the analysis. These times have no physical significance, but do indicate that the actinides cannot contribute to the impact before a million years.

T_D (neptunium, plutonium) = $3.2.10^8$ years
T_D (protactinium) = $2.6.10^8$ years
T_D (thorium) = $5.3.10^8$ years
T_D (uranium) = $6.8.10^8$ years
T_D (actinium, americium, curium) = $2.6.10^9$ years



Molar flow rate out of the repository (7 metres from the cells)

Molar flow rate out of the Callovo-Oxfordian

Figure 7.5-11 Severely degraded evolution scenario – history of molar flow rates out of the repository at a distance of 7 metres from the cells and out of the Callovo-Oxfordian for a CU2 cell

7.5.4 Impact assessment

The results of the impact assessment confirm that the dose caused by the transfer pathway through the unaltered geological barrier is much higher than the dose caused by the transfer pathway through the built structures (See Figure 7.5-12).

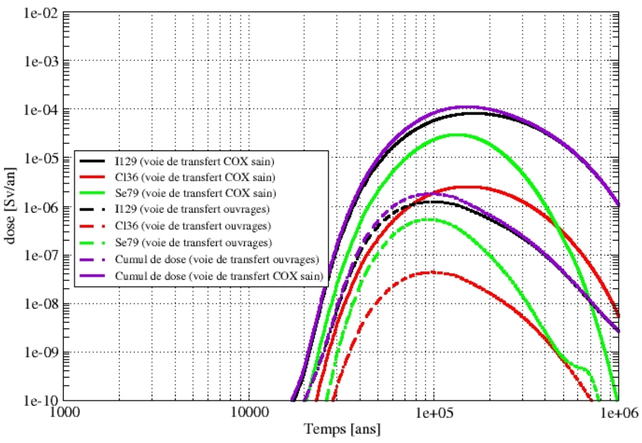


Figure 7.5-12 « Severely degraded evolution » SEA - Comparison of the doses at the Saulx reference outlet of the CU1 reference package for the transfer pathways through the structures (present-day model) and unaltered geological barrier (model at 1 million years)

For the CU1 spent fuel, the maximum dose is 0.11 mSv/year at 150 000 years. Iodine 129 and selenium 79 co-dominate the dose. For the vitrified C wastes, the contribution of selenium 79 is even more pronounced as it occurs in larger quantities (50 %), whereas the quantity of iodine 129 is lower.

The maximum dose, governed by selenium, is in this case 0.0055 mSv/year at about 170 000 years. After 500 000 years, iodine, chlorine and selenium co-dominate the impact (selenium starting to benefit from radioactive decay) but with very low dose levels of not more than 0.001 mSv/year.

For B wastes, the contribution of the proportion of activity that comes out through the structures is greater because the cell is directly connected to the main connecting drifts, which reduces the average transfer time of the radionuclides through the structures to the shaft and therefore limits the exchanges with the Callovo-Oxfordian. The highest dose is however still associated with the fraction of activity migrating through the geological barrier with a maximum dose (for the waste disposed of in the type B1x and B2 cells) of the order of 0.003 mSv/year at about 160 000 years.

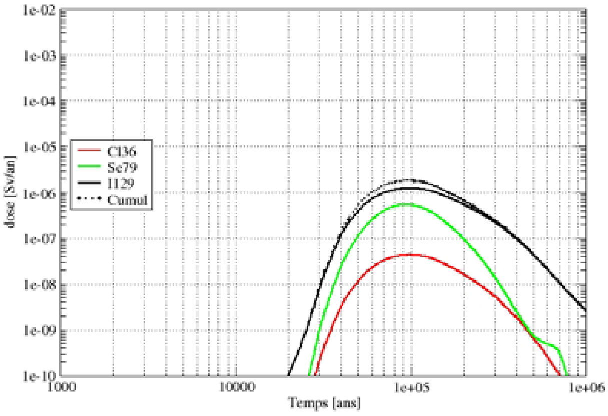
The doses at the Saulx outlet of the one-million years model are given in Table 7.5-5 for the fraction of activity that migrates through the geological barrier. As previously seen, these doses dominate over those associated with the fraction of activity that migrates through the structures.

Package-types	Maximum dose [mSv/year]	Date of the maximum [years]	Contributing radionuclides
The « Saulx » outlet (the worst case)			
B1x - (inorganic package not giving off gaseous hydrogen)	0.0023	160 000	^{36}Cl (^{129}I to a lesser extent)
B2 - (bituminized sludges)	0.00022	170 000	^{129}I, ^{36}Cl, ^{79}Se
Total B1x/B2 wastes (scenario S1b)	0.0025	ca. 160 000	^{36}Cl (^{129}I to a lesser extent)
C1 and C2 glasses (Scenario (S1b)	0.03	170 000	^{79}Se (^{129}I, ^{36}Cl at very long term)
C3 and C4 glasses (Scenario S1a)	0.024	170 000	^{79}Se (^{129}I, ^{36}Cl at very long term)
CU1 Spent fuel	0.11	150 000	^{129}I, ^{79}Se
CU2 Spent fuel	0.0089	160 000	^{129}I, ^{79}Se
CU3 Spent fuel			^{129}I, ^{79}Se
Total CU1/CU2 spent fuels	0.12	ca. 150 000	^{129}I

Table 7.5-5 SEA Severely degraded evolution – Total dose – dates of maximum dose and main contributors at the Saulx outlet of the Oxfordian (the worst case) – model at 1 million years – all wastes

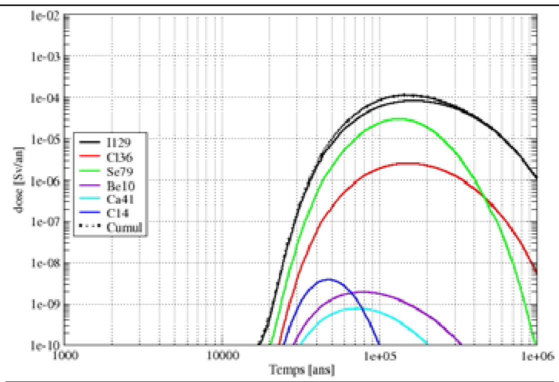
From Figure 7.5-13 to Figure 7.5-19, the doses for the radionuclides transiting by the structures and the doses resulting from transfer of radionuclides through the geological barrier are given.

- **CU1 Spent fuel**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cl36	4.3E-08	9.9E+04
Se79	5.3E-07	9.4E+04
I129	1.2E-06	9.9E+04
Cumul	1.8E-06	9.6E+04

Voie de transfert par les ouvrages

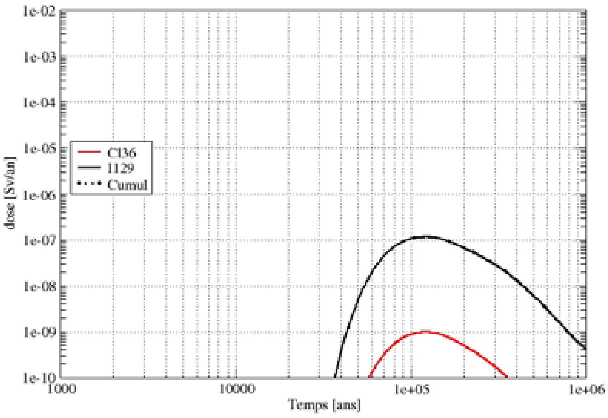


RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.1E-04	1.5E+05
I129	8.2E-05	1.7E+05
Se79	3.0E-05	1.3E+05
Cl36	2.5E-06	1.5E+05
C14	3.8E-09	4.8E+04
Be10	1.9E-09	7.9E+04
Ca41	7.7E-10	7.3E+04

Voie de transfert par la formation hôte

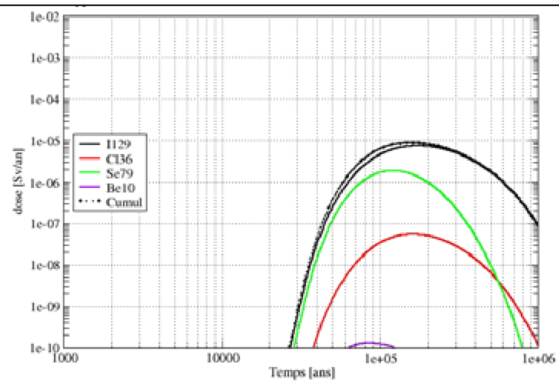
Figure 7.5-13 « Severely degrade evolution » scenario - Dose at the Saulx outlet of the Oxfordian - CU1 spent fuel reference packages

- **CU2 Spent fuel**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cl36	9.7E-10	1.2E+05
I129	1.1E-07	1.2E+05
Cumul	1.2E-07	1.2E+05

Voie de transfert par les ouvrages

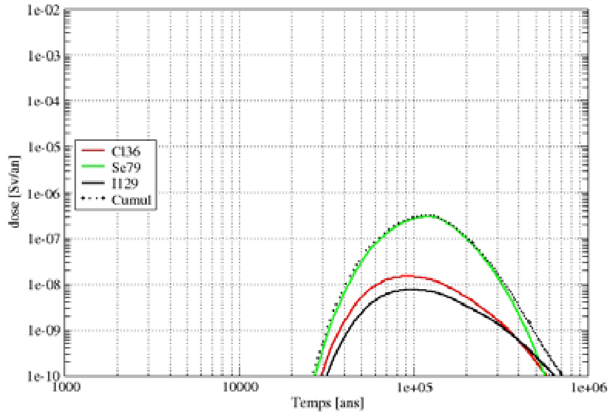


RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	8.9E-06	1.6E+05
I129	7.4E-06	1.7E+05
Se79	1.8E-06	1.2E+05
Cl36	5.5E-08	1.6E+05
Be10	1.3E-10	8.8E+04
Ca41	dose < 1e-10	sans objet
C14	dose < 1e-10	sans objet

Voie de transfert par la formation hôte

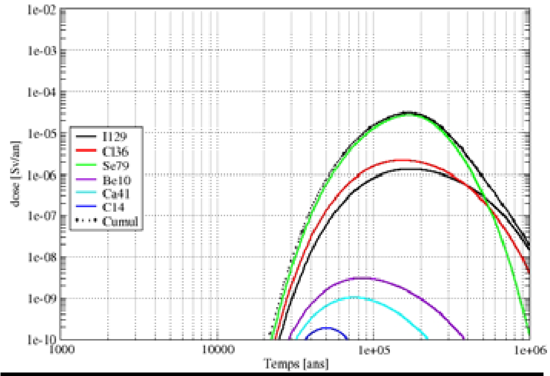
Figure 7.5-14 « Severely degraded evolution » scenario - Dose at the Saulx outlet of the Oxfordian - CU2 spent fuel referencee packages

- **Vitrified wastes**



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cl36	1.5E-08	9.2E+04
Se79	3.0E-07	1.2E+05
I129	7.5E-09	9.8E+04
Cumul	3.2E-07	1.2E+05

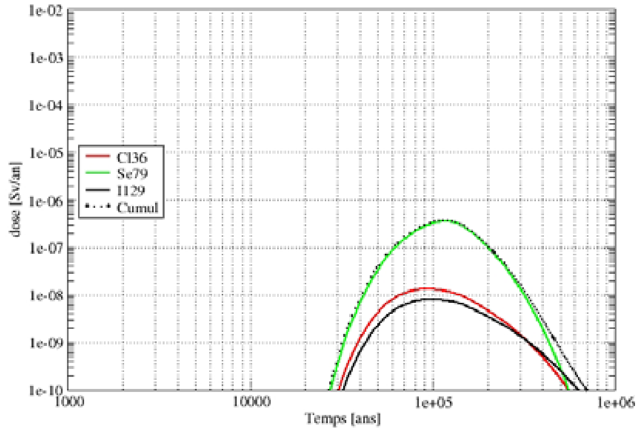
Voie de transfert par les ouvrages



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	3.0E-05	1.7E+05
Se79	2.7E-05	1.7E+05
Cl36	2.2E-06	1.5E+05
I129	1.3E-06	1.7E+05
Be10	3.0E-09	8.5E+04
Ca41	1.0E-09	7.6E+04
C14	1.9E-10	5.1E+04

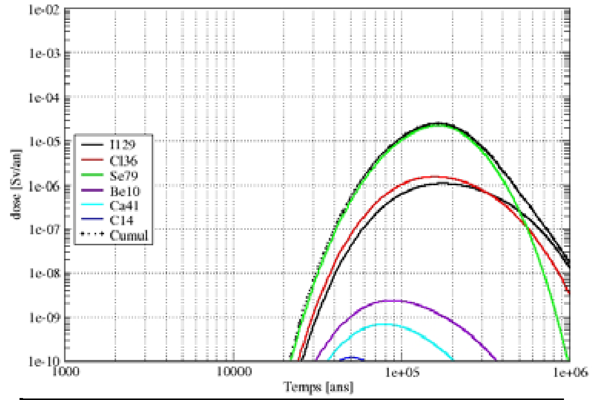
Voie de transfert par la formation hôte

Figure 7.5-15 « Severely degraded evolution » scenario - Dose at the Saulx outlet of the Oxfordian - C1/C2 reference packages



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cl36	1.3E-08	9.4E+04
Se79	3.5E-07	1.2E+05
I129	7.9E-09	1.0E+05
Cumul	3.7E-07	1.2E+05

Voie de transfert par les ouvrages

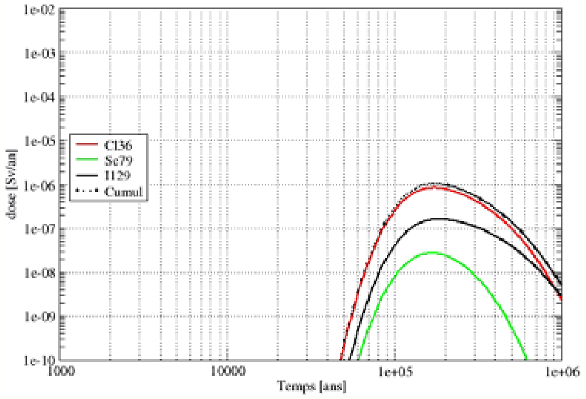


RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	2.4E-05	1.7E+05
Se79	2.2E-05	1.7E+05
Cl36	1.5E-06	1.6E+05
I129	1.1E-06	1.8E+05
Be10	2.3E-09	8.8E+04
Ca41	6.7E-10	8.0E+04
C14	1.2E-10	5.1E+04

Voie de transfert par la formation hôte

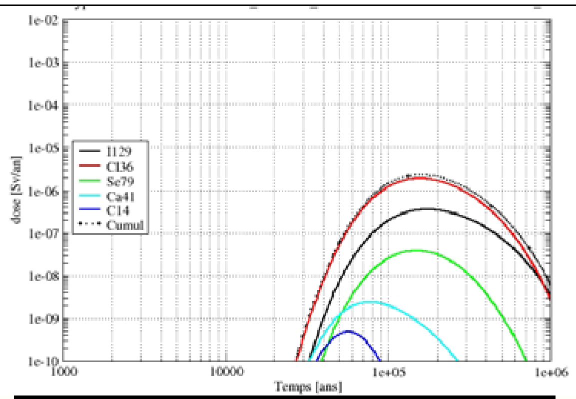
Figure 7.5-16 « Severely degraded evolution » scenario - Dose at the Saulx outlet of the Oxfordian - C3/C4 reference packages

- **B wastes**



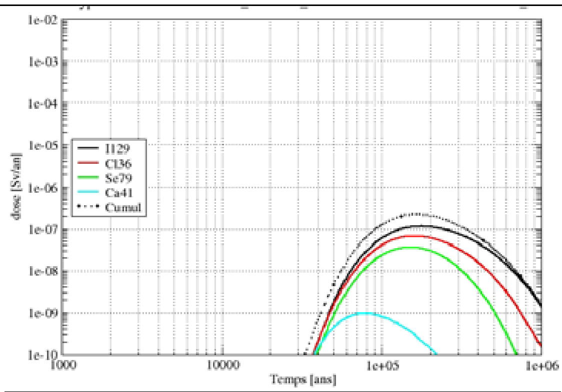
RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.00E-06	1.70E+05
Cl36	8.40E-07	1.70E+05
I129	1.60E-07	1.90E+05
Se79	2.70E-08	1.70E+05

Figure 7.5-17 « Severely degraded evolution » scenario - Dose at the Saulx outlet of the Oxfordian - B1x reference package



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	2.3E-06	1.6E+05
Cl36	1.9E-06	1.6E+05
I129	3.5E-07	1.7E+05
Se79	3.8E-08	1.5E+05
Ca41	2.4E-09	7.7E+04
C14	4.8E-10	5.7E+04
Be10	dose < 1e-10	sans objet

Colis reference package B1x



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	2.2E-07	1.7E+05
I129	1.1E-07	1.8E+05
Cl36	6.7E-08	1.6E+05
Se79	3.6E-08	1.5E+05
Ca41	9.6E-10	7.8E+04
C14	dose < 1e-10	sans objet

Colis reference package B2

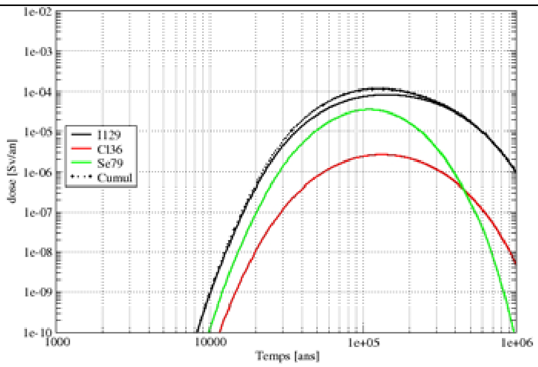
Figure 7.5-18 « Severely degraded evolution »scenario - Dose at the Saulx outlet of the Oxfordian - B2 reference package

7.5.5 Sensitivity analyses

To the « severely degraded evolution » SEA the premature failure of all the spent fuel containers is added here.

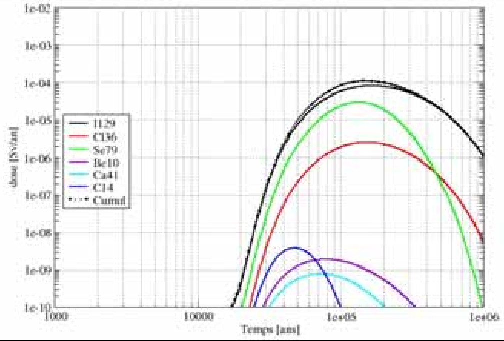
After calculation, it is found that by taking failed containers into account there is no change to the main results, either in terms of distribution of the transfer pathways through the structures and through the unaltered Callovo-Oxfordian, on the molar rates escaping from the host formation or even more on the maximum dose and the time of appearance of this dose. Indeed, the duration of the thermal transient remains limited compared to the transfer times in the Callovo-Oxfordian. During this first period, the mean thickness traversed in the Callovo-Oxfordian is small compared to the total thickness of the geological barrier. Therefore, for the main radionuclides contributing to the impact (^{129}I, ^{36}Cl

and [79]Se), the decreased transfer time caused by the parameters of the « severely degraded evolution » altered scenario, combined with a premature release of the package (i.e. an accelerated transfer under the influence of high temperatures), is insufficient to significantly increase the impact. The maximum dose is very similar to that of the reference case of the « severely degraded evolution » scenario (0.12 mSv/year compared to 0.11 mSv/year). The dates of the maxima occur slightly earlier.



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cl36	2.60E-06	1.30E+05
Se79	3.50E-05	1.10E+05
I129	8.20E-05	1.40E+05
Cumul	1.20E-04	1.30E+05

Sensitivy calculation – all defective packages – transfer pathhway to the host formation



RN	DoseMax[Sv/an]	t(DoseMax)[an]
Cumul	1.1E-04	1.5E+05
I129	8.2E-05	1.7E+05
Se79	3.0E-05	1.3E+05
Cl36	2.5E-06	1.5E+05
C14	3.8E-09	4.8E+04
Be10	1.9E-09	7.9E+04
Ca41	7.7E-10	7.3E+04

reference calculation – transfer pathway thaough the host formation

Figure 7.5-19 « Severely degraded evolution » scenario – sensitivity to the premature failure of the CU1 spent fuel containers- Dose at the Saulx outlet of the Oxfordian

7.5.6 Findings of the « severely degraded evolution » SEA

This scenario does not correspond to any of the situations demonstrated by the risk analysis. It is however possible to draw some useful conclusions on the robustness of the repository in the face of uncertainties. Because of the accumulation of pessimistic hypotheses taken into account, this scenario and its sensitivity studies is equivalent to consider effective only the following components :

- the host formation, which retains a degraded geochemistry but is nevertheless capable of stopping the radionuclides in their migration ;
- the dead-end repository architecture, which despite the relatively high permeability of the Callovo-Oxfordian, is capable of limiting the flow. The system is nevertheless advective within the repository ;
- the matrices of B2, C and spent fuel wastes, whose performances are lessened but not zero (in contrast to other B wastes).
- Thus reduced to the minimum expression of its safety functions, the repository appears to be robust. The impact increases, but in proportions that remain low (not more than 0.12 mSv/year).

7.6 Initial probabilistic approach

7.6.1 Objectives

By testing the influence of a set of determined parameters on the performances of the repository system, the results of the SEN and SEA calculations enabled to identify the most influential elements and to deduce the lessons learnt on the role of the components with regard to the main safety functions.

In addition to these results, a probabilistic study was carried out taking into account the simultaneous variation spectrum of the various parameters [109]. That consists inn a sensitivity analysis exercise, conducted by way of an illustration on the iodine and selenium of C1/C2 glass, which is designed to back up the lessons learnt of the deterministic studies and assess the effects of joint variations of several parameters. From this type of calculation, it is possible to deduce information on the uncertainty of the result by situating the position of the various deterministic calculations on an overall distribution curve. It is however difficult to draw direct lessons from this type of assessment as it depends on the probably distribution laws that were adopted. Consequently, the objective adopted by Andra at the stage of this initial methodological exercise is first and foremost to identify the parameters which, due to their uncertainty, have the greatest influence on the uncertainty of the result. This does not particularly mean proceeding with a probabilistic treatment of the impact of the repository. In accordance with RFS.III.2.f [2], the safety approach remains deterministic. The calculation is limited to the indicators as such the molar flow rates out of the Callovo-Oxfordian and access structures and the distribution of radiological impact is not assessed.

7.6.2 Selected adopted parameters, distribution laws and correlations

The choice was made to treat the two transfer pathways of radionuclides – through the engineered structures and through the geological barrier – together for the same set of data. To cover the results of both transfer pathways simultaneously, the components concerned by the probabilistic study are engineered components in bentonite (seals and plugs) and the Callovo-Oxfordian (sound and damaged). The properties of the backfill and the release models are considered to be constant. A conservative release model based solely on the initial dissolution rate of the glass (model V_0.S) is adopted, thereby limiting the influence of the source term on the overall transport kinetics in the repository.

Consequently, for each of these components, probability distribution laws are established for the following parameters (see Figure 7.6-1) :

- water permeability K [m/s],
- kinematic porosity ωc [-],
- effective diffusion coefficient De [m^2/s],
- porosity accessible to the diffusion of anions ωd_Anions and cations ωδ_Cations [-],
- specific solubility limit of the studied element [mol/m^3],
- specific partition coefficient of the studied element [m^3/kg].

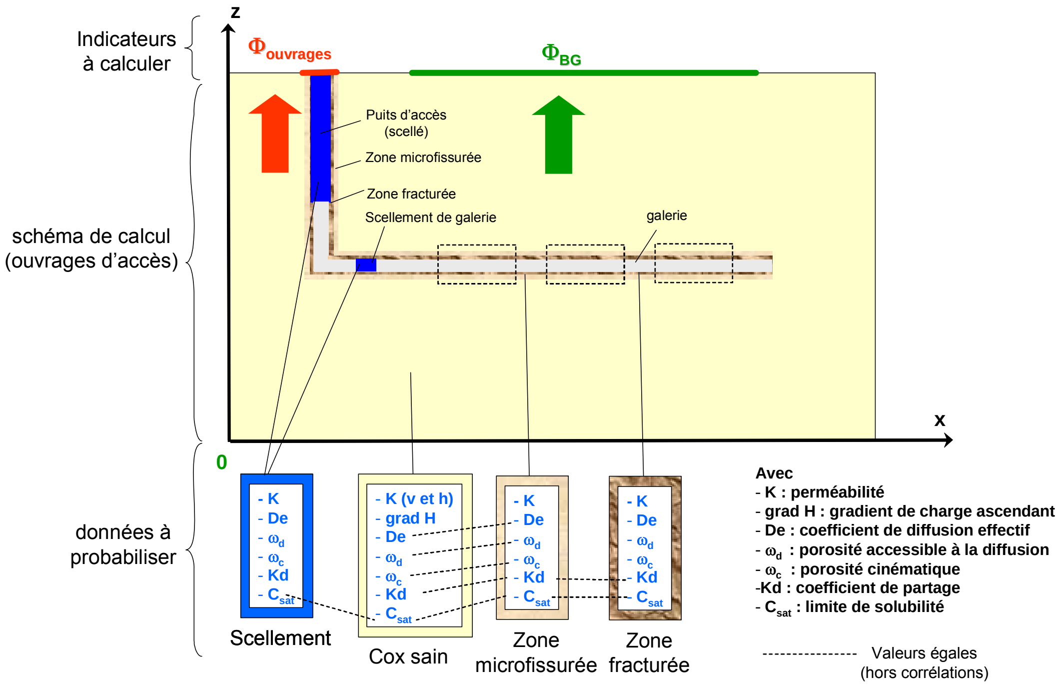


Figure 7.6-1 Probabilistic calculation – Summary of the parameters involved in a probability distribution

The probability distribution laws of each of the parameters and the correlations between parameters are established on the basis of the variability of experimental data and their correlations as well as possible constraints imposed by the Alliances tools (available distribution laws, types of correlations that could be considered, etc.).

Through the spectrum of parameter values taken into account, the purpose of the probabilistic calculation is to cover all results of the SEN and mose of the « severely degraded evolution » scenario. As the latter corresponds in principle to the most pessimistic case, the distribution laws are truncated : the most pessimistic value of a parameter corresponds generally to the value taken in a « severely degraded evolution » scenario ; for the other bound, the order of magnitude immediately below (respectively above) the lowest (respectively the highest) measured item of data is adopted. Note that taking into account of the range of variation in the EDZ parameters and seal permeability levels, this study also covers seal failure situations.

The process of selecting distribution laws for each of the characteristic parameters of the Callovo-Oxfordian is illustrated below. The process is similar for the other components. Finally, the distribution laws adopted for all of the components are presented in summary tables (see Table 7.6-1 to Table 7.6-3).

7.6.2.1 Distribution laws adopted in the Callovo-Oxfordian

- **Vertical permeability**

The vertical permeability distribution law is a log-normal law established on the basis of 49 values resulting from measurements on samples.

The distribution law is truncated to :

- 10^{-12} m/s for its upper value,
- 10^{-15} m/s for its lower value which corresponds to the order of magnitude immediately below the lowest-value data.

- **Horizontal permeability**

The horizontal permeability was deduced from the vertical permeability on the basis of a statistical distribution law.

Data deduced from borehole tests and permeability measurements on samples (72 data items) highlight the fact that the horizontal permeability also follows a log-normal law, the characteristics of which are significantly different to those of vertical permeability due to anisotropy. Such anisotropy cannot however be assessed deterministically due to the fact that data are not available on the same samples in the two directions.

The horizontal permeability distribution law is therefore the result of the combination of the law applied to the vertical permeability with an anisotropy coefficient, itself probabilised according to a log-normal law, varying from 1 to 10. The probabilistic calculation can therefore, in certain cases, represent situations that are slightly less favourable than the « severely degraded evolution » SEA, where the horizontal permeability amounted to 10^{-12} m/s, but this parameter is not expected to be very sensitive. It has been checked that the horizontal permeability distribution law thus established reproduces infact the experimental measurements (see Figure 7.6-2).

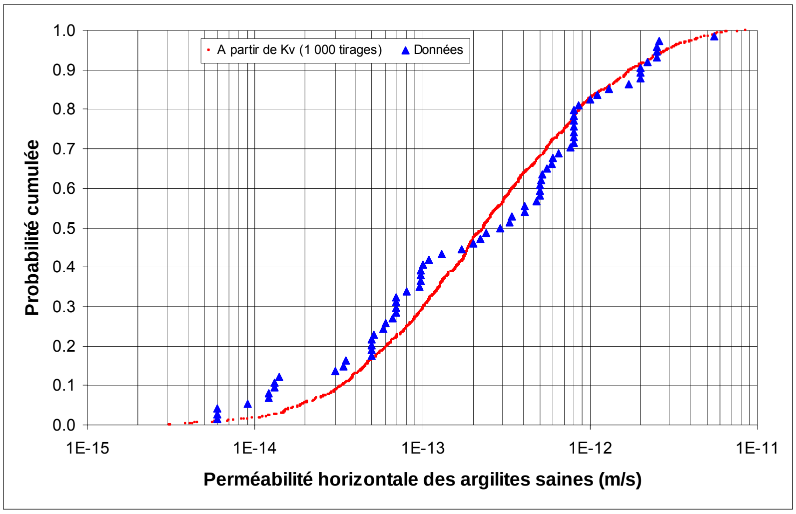


Figure 7.6-2 Probabilistic calculation – Representation of the distribution function of horizontal permeability in sound argillites, deduced firstly from data and secondly from the vertical permeability distribution law (1000 samples) by using the transition law.

- **Ascending vertical head gradient**

The vertical head gradient in the Callovo-Oxfordian depends on the position of the repository (which cannot be probabilised) and the geodynamic evolution. In order to take account of the latter parameter, it was decided to adopt a uniform distribution law over the 0.2 m/m to 0.4 m/m interval encompassing the values adopted for the SEN. Note that these values are also used in the « severely degraded evolution » scenario.

- **Effective diffusion coefficient**

The distribution law for the anion diffusion coefficient was established on the basis of 15 values resulting from diffusion tests on samples from various facies penetrated by boreholes EST 104 and EST 205. This is a normal law.

The distribution law is truncated to :

- 10^{-11} m²/s for its upper value, which corresponds to the maximum value adopted for the « severely degraded evolution » scenario (see chapter 7.5),
- 10^{-13} m²/s for its lower value.

As the number of values beyond these bounds is low, truncation does not affect the characteristics of the distribution law.

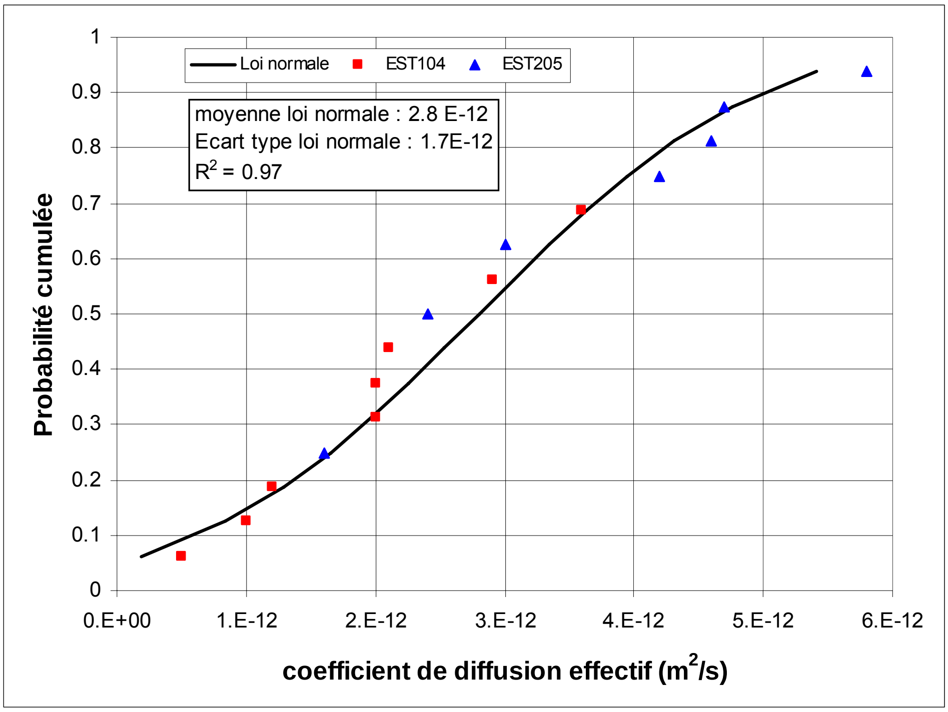


Figure 7.6-3 Distribution of cumulative probability for the effective diffusion coefficient of the iodide ion – comparison with experimental data acquired in the boreholes.

- **Porosity accessible to anion diffusion**

The distribution law for porosity accessible to anions is a uniform law based on available data items (3 for iodine and 3 for chlorine). The calculation is only conducted on anions, but it was also ascertained that a uniform law enabled the porosity accessible to cation diffusion to be represented, for which there exists a large number of data items (39).

The bounds of the uniform law are 3 % and 6 % respectively.

- **Partition coefficient and solubility limit**

Several measurements have been made on iodine retention in the Callovo-Oxfordian, either on the basis of Kd batch or column tests or on the basis of diffusion tests. The analysis of all data (167 measurements) reveals two groups of values :

- Values of Kd egual to zero representing around 20 % of measurements,
- non-zero Kd measurements representing around 80 % of measurements and which may be adjusted by a log-normal law.

It is difficult today to interpret these experimental results [71]. Iodine sorption may be partly linked to a contribution of organic matter from the argillites (this could be more or less available or altered according to the techniques and protocols), but also to long reaction kinetics causing the measurements to depend on contact times and therefore the techniques used. With regard to these two hypotheses, it is reasonable to think that the most representative experiments are the diffusion measurements performed on intact rock. These experiments indicate a non-zero chemical retention of iodine, but the processes governing this have not been determined to date[109].

Two cases have been treated :

- a case studied as a reference that can be termed phenomenological. This case contains the 167 available data on the basis of which a distribution law is defined. This law allocates a probability of 20 % to the zero Kd value, and adopts a log-normal law for the remaining 80 % ;
- a more cautious case used in the sensitivity study, based only on the zero Kd values.

109 In the absence of a clear understanding of the sorption mechanisms, zero sorption has been adopted for the deterministic reference calculation of the SEN, even if a predictive sensitivity study has enabled the consequences relating to possible iodine sorption in the Callovo-Oxfordian to be assessed.

For selenium, the Kd distribution law obeys the same logic.

No solubility limit for iodine has been adopted in the calculation. The selenium solubility limit follows a discrete law corresponding to different modes of inspecting the selenium in solution. A relatively higher weight is given to the solubility limit corresponding to 5.10^{-7} mol/$m^{3,}$ the value considered most likely in the repository context [21].

7.6.2.2 Summary of the distribution laws

A summary is presented below of the values and distribution laws adopted for the sound Callovo-Oxfordian (Table 7.6-1), for the EDZ (Table 7.6-2) and for the bentonite engineered structures (Table 7.6-3).

Parameter		Deterministic calculations		Probabilistic calculations
		SEN Reference value	Maximum value*	Distribution law proposals
Permeability (m/s)	K_v	$5\ 10^{-14}$	10^{-12}	Log-normal law, Mode[110] and standard deviation in Napierian logarithm : m = -30.2 ; σ = 1.6 Bounds in Napierian logarithm : [-34.5 ; -27.6]
	K_h	$5\ 10^{-13}$	10^{-12}	Deduced from Kv by a log-normal transition law, additive on the Napierian logarithms of Kv values Mode and standard deviation in Napierian logarithm : m=1.5, σ = 1.2 Bounds in Napierian logarithm : [0 ; 2.3]
Kinematic porosity (%)	ωc	9	9	Unique value 9
Diffusion coefficient (m^2/s)	De anion	$5\ 10^{-12}$	10^{-11}	Normal law defined by m = 2.8 10^{-12}, σ = 1.7 10^{-12}, Bounds [10^{-13} ; 10^{-11}]
Porosity accessible to diffusion (%)	ωDiffusion	5	4	Uniform law [3 ; 8]
Iodine solubility limits (mol/m^3)	C_{sat} (I)	« infinite »	« infinite »	« infinite »
Iodine partition coefficients (m^3/kg)	Kd (I)	0	0	Case 1 : 20% of Kd zero 80% according to log-normal law ; mode and standard deviation in Napierian logarithm : « m » = -8.48 and σ = 2.388 ; bounds in Napierian logarithm [-12.42 ; -4.73] Case 2 : unique Kd zero value
Selenium solubility limits (mol/m^3)	C_{sat} (Se)	5.10^{-7}	5.10^{-4}	Discrete law with three values : 70 % of cases at 5.10^{-7} mol/m^3 15 % of cases at 1.10^{-10} mol/m^3 15 % of cases at 5.10^{-4} mol/m^3
Selenium partition coefficients (m^3/kg)	Kd (Se)	0	0	20 % of Kd zero 80% according to log-normal law ; mode and standard deviation in Napierian logarithm : « m » = -8.48 and σ = 2.388 ; bounds in Napierian logarithm [-12.42 ; -4.73] (law identical to iodine case 1 but with independent samples)
Vertical head gradient (m/m)	Grad H	0.2 and 0.4	0.4	Uniform law over [0.2 ; 0.4]
* Value adopted for SEN sensitivity study or in severely degraded evolution scenario (see chapter 7.5)				

Table 7.6-1 Probabilistic calculation – Summary table of distribution laws of various parameters for the sound Callovo-Oxfordian

110 The mode corresponds to the highest probability value. This is the best represented data item which (with the standard deviation) characterises the « normal » and « log-normal » distribution laws.

Parameter		Deterministic calculations		Probabilistic calculations
		SEN Reference value	**Maximum value**	**Distribution law proposals**
Fractured zone				
Permeability (m/s)	Kv =K_h	5 10^{-9}	10^{-6}	Log-normal law Mode and standard deviation in Napierian logarithm : m = -22.5 ; σ = 2.6 Bounds in Napierian logarithm : [-34.5 ; -13.8]
Kinematic porosity (%)	ωc	10	-	Unique value : 9
Diffusion coefficient (m^2/s)	De anion	10^{-1}1	2 10^{-9}	Normal law where m = 5.6 10^{-1}2, σ = 3.4 10-12, [2 10^{-13} ; 2 10^{-9}]
Porosity accessible to diffusion (%)	ωDiffusion	15	-	Uniform law with bounds [10 ; 20]
Solubility limits (mol/m^3)	C_{sat} (I)	« infinite »	« infinite »	Same as sound Callovo-Oxfordian without resampling
Partition coefficients (m^3/kg)	Kd (I)	0	0	Same as sound Callovo-Oxfordian without resampling
Solubility limits (mol/m^3)	C_{sat} (Se)	5.10^{-7}	5.10^{-4}	Same as sound Callovo-Oxfordian without resampling
Partition coefficients (m^3/kg)	Kd (Se)	0	0	Same as sound Callovo-Oxfordian without resampling
Microfissured zone				
Permeability (m/s)	K_v=K_h	5 10^{-11}	5 10^{-9}	Log-normal law Mode and standard deviation in Napierian logarithm : m = -26.4 ; σ = 2.4 ; Bounds in Napierian logarithm : [-34.5 ; -19.1]
Kinematic porosity (%)	ωc	9	-	Unique value : 9
Diffusion coefficient (m^2/s)	De anion	5 10-12	10-11	Same distribution law as sound Callovo-Oxfordian, specific sampling
Porosity accessible to diffusion (%)	ωDiffusion	5	4	Same distribution law as sound Callovo-Oxfordian, specific sampling
Solubility limits (mol/m^3)	C_{sat} (I)	« infinite »	« infinite »	Same as sound Callovo-Oxfordian without resampling
Partition coefficients (m^3/kg)	Kd (I)	0	0	Same as sound Callovo-Oxfordian without resampling
Solubility limits (mol/m^3)	C_{sat} (Se)	5.10^{-7}	5.10^{-4}	Same as sound Callovo-Oxfordian without resampling
Partition coefficients (m^3/kg)	Kd (Se)	0	0	Same as sound Callovo-Oxfordian without resampling

Table 7.6-2 Probabilistic calculation – Summary table of distribution laws of various parameters for the EDZ (microfissured zone and fractured zone)

Parameter		Deterministic calculations		Probabilistic calculations
		SEN Reference value	Maximum value	Distribution law proposals
Bentonite of cell plugs and seals				
Permeability (m/s)	K_v=Kh	10^{-11}	10^{-9}	Log-normal law Mode and standard deviation in Napierian logarithm : m = -27.3 ; σ = 2 Bounds in Napierian logarithm : [-30 ; -20,7]
Kinematic porosity (%)	ωc	18	-	Uniform law over [13 ; 23] range
Diffusion coefficient (m^2/s)	De anion	5 1^{0-12}	-	Log-normal law Mode and standard deviation in Napierian logarithm : m = -28 ; σ = 1.1 Bounds in Napierian logarithm : [-30 ; -25.3]
Porosity accessible to diffusion (%)	ωDiffusion	18	-	Uniform law over [3 ; 7] range
Solubility limits (mol/m^3)	C_{sat} (I)	« infinite »	« infinite »	Same as sound Callovo-Oxfordian without resampling
Partition coefficients (m^3/kg)	Kd (I)	0	0	Normal law centred on 0 (σ = 0.25.10-3) with 10^{-3} as maximum value
Solubility limits (mol/m^3)	C_{sat} (Se)	5.10^{-7}	5.10^{-4}	Same as sound Callovo-Oxfordian without resampling
Partition coefficients (m^3/kg)	Kd (Se)	0	0	20 % of Kd zero, 20 % of Kd at 6.10-3 m^3/kg, 60 % according to uniform law over [0 ; 6.10^{-3}] range.

Table 7.6-3 Probabilistic calculation – Summary table of distribution laws of various parameters for the bentonite of the plugs and seals.

7.6.2.3 Correlations between parameters and constraints

- **Correlation between parameters**

The models implemented comprise several parameters which are not all mutually independent. Consequently, for each set of data to be consistent from a physical point of view, correlations have been defined between the parameters.

By identifying the calculation parameters, it has been possible to assess the correlations between them. A correlation between the diffusion coefficient, porosity accessible to diffusion and vertical permeability in the Callovo-Oxfordian has been taken into account ; the correlation between these three parameters has also been extended to the other repository components (EDZ and bentonite). Table 7.6-4 provides the values of the correlation coefficients between these three parameters in the Callovo-Oxfordian.

Correlated parameters	Correlation coefficient
Vertical permeability logarithm – Effective anion diffusion coefficient	0.85
Effective anion diffusion coefficient – Porosity accessible to anion diffusion	0.6
Vertical permeability logarithm – Porosity accessible to anion diffusion (Correlation coefficient deduced from first two)	0.55

Table 7.6-4 Probabilistic coefficient - Correlation coefficients of parameters in the Callovo-Oxfordian (permeability, diffusion coefficient and porosity accessible to diffusion)

The permeability distribution laws in the EDZ and in the bentonite were established in the same way as horizontal permeability in the Callovo-Oxfordian, on the basis of a log-normal transition law which combines with the vertical permeability values in the Callovo-Oxfordian. Under such conditions, a static correlation does in fact exist between the permeability levels of the various repository components.

- **Constraints**

Several constraints have been defined between the variables in order to respect the physical validity of the data sets ; this involves, for example, the diffusion coefficient in the microfissured zone being systematically higher than or equal to the one in the Callovo-Oxfordian. This type of constraint is represented by means of inequalities between already correlated variables. These inequalities are subsequently verified after each sampling and the values of the parameters are modified if necessary in order to comply with them.

Table 7.6-5 shows all of the constraints taken into account as an inequality and the order in which they were applied.

Order of constraints	Constraints to be observed	Consequences if the constraints are not observed*
1	$K_{\text{microfissured EDZ}} \geq K_{\text{horizontal of sound argillite}}$	$K_{\text{microfissured EDZ}} \rightarrow K_{\text{horizontal of sound argillite}}$
2	$K_{\text{fractured EDZ}} \geq K_{\text{microfissured EDZ}}$	$K_{\text{fractured EDZ}} \rightarrow K_{\text{microfissured EDZ}}$
3	$K_{\text{microfissured EDZ}} \geq$ 10 Kh sound argillite	$De_{\text{microfissured EDZ}} \rightarrow De_{\text{sound argillite}}$
4	$De_{\text{microfissured EDZ}} \geq De_{\text{sound argillite}}$	$De_{\text{microfissured EDZ}} \rightarrow De_{\text{sound argillite}}$
5	$K_{\text{fractured EDZ}} \geq 10\ Kh_{\text{sound argillite}}$	$De_{\text{fractured EDZ}} \rightarrow De_{\text{sound argillite}}$
6	$De_{\text{fractured EDZ}} \geq$ De microfissured EDZ	$De_{\text{fractured EDZ}} \rightarrow De_{\text{microfissured EDZ}}$
7	$K_{\text{microfissured EDZ}} \geq 10\ K_{h\ \text{sound argillite}}$	$\omega_{\text{Diffusion of fissured EDZ}} \rightarrow \omega_{\text{Diffusion of sound argillite}}$
8	$\omega_{\text{Diffusion of microfissured EDZ}} \geq \omega_{\text{Diffusion of sound argillites}}$	$\omega_{\text{Diffusion of microfissured EDZ}} \rightarrow \omega_{\text{Diffusion of sound argillites}}$
9	$K_{\text{fractured EDZ}} \geq 10\ K_{h\ \text{sound argillite}}$	$\omega_{\text{Diffusion of fractured EDZ}} \rightarrow \omega_{\text{Diffusion of sound argillite}}$
10	$\omega_{\text{Diffusion of fractured EDZ}} \geq \omega_{\text{Diffusion of microfissured EDZ}}$	$\omega_{\text{Diffusion of fractured EDZ}} \rightarrow \omega_{\text{Diffusion of microfissured EDZ}}$
*The « $\rightarrow$ » symbol in the « consequences » column means that the first parameter takes on the value of the second		

Table 7.6-5 Probabilistic calculation – Physical constraints taken into account in the Latin hypercube statistical sampling

The application of constraints to define the sampled data sets could in principle lead to significant changes to the distribution laws selected initially. We have however checked, by comparing the theoretical distributions and those obtained after sampling, that the influence of the constraints on the sampling was low apart from in the case of porosity accessible to anions (see Figure 7.6-4)) subjected to constraints 7 and 8 in the above table.

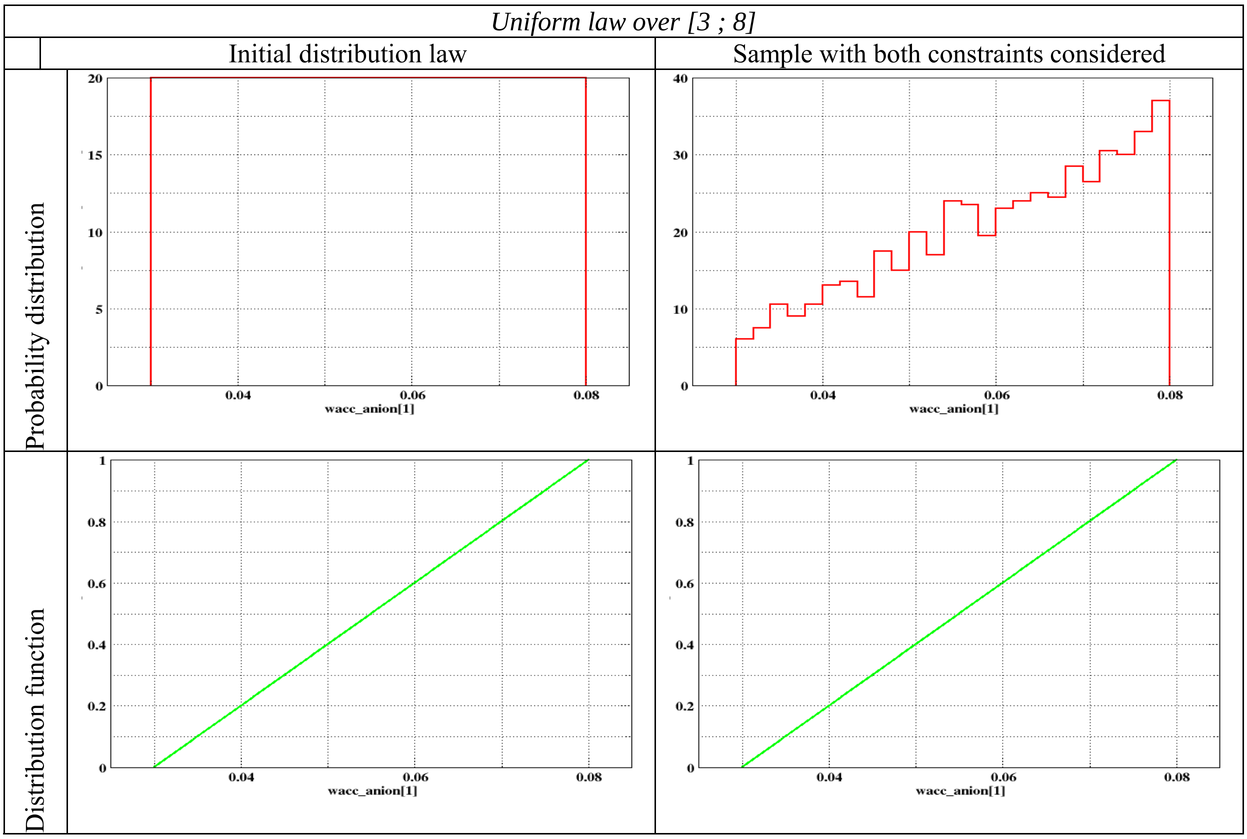


Figure 7.6-4 Probabilistic calculation – Porosity accessible to anion diffusion in the microfissured zone – Distribution law and distribution function (initial law and sample with constraints)

7.6.2.4 Calculation methods

The sampling method adopted is Latin Hypercube Sampling (LHS) which has the advantage of guaranteeing good coverage of the space of the variables in a minimum of samples (1000 samples have been retrained in order to have a satisfying confidence interval). On account of the number of samples that have been realized and the needed calculation times, the various compartments arranged in sequence into the deterministic safety calculations (see section 5.4) have been simplified. In particular, the model the calculation compartments relating to the cell, the repository zone studied (C1/C2 waste in this case) and the contribution of the other repository zones and access structures. The various parallel connecting drifts and access drifts to the cells are represented in a simplified manner by a single macro-drift, while retaining the sections of the EDZs. A validation exercise performed for the various calculation cases has shown the good matching of simplified model with the safety model for the following indicators :

- water flow out of the repository subzones and shafts,
- molar flow rate of iodine : in particular its maximum, its date of occurrence and the number of moles released upstream and downstream of the various seals (especially the shafts).

7.6.3 Results

Remember that the results of a probabilistic study can give rise to two types of information :

- uncertainty studies used to quantify the overall uncertainty of the various indicators and situate a specific deterministic calculation case with in a probability distribution ;
- sensitivity studies showing the relative influence of the uncertainty of each parameter on the uncertainty of the result.

The latter are less dependent on the probability laws adopted and are therefore the most robust. The former are more tick list to interpret.

For each case studied (iodine with variable Kd and zero Kd, and selenium), the results, which can be obtained from an uncertainty analysis are presented at first. Secondly, the results of the sensitivity study are presented.

7.6.3.1 Calculation indicators

The two main calculation indicators are the molar flow rate and the masses (integrals of the flow rates) ; the latter are calculated out of the Callovo-Oxfordian and out of the shafts in order to assess the impact on the two transfer pathways. In order to gain a better knowledge of the behaviour of the system, intermediate indicators were considered in the analysis : the molar flow rate out of the repository zone (before and after the seals) and the molar flow rate out of the connecting drift seals (before the shaft seals).

7.6.3.2 Case of iodine with chemical retention considered (variable Kd)

- **Uncertainty analysis**

Comparison of transfer pathways

As illustrated in Figure 7.6-5, the results show that in all cases, the molar flow rate out of the top of the Callovo-Oxfordian formation, after transfer into the sound argillites, is clearly higher than the molar flow rate out of the access structures and their EDZ (by a factor of 50 at least). The two transfer pathways are however studied to assess whether the variation in the input parameters has an influence on each of them.

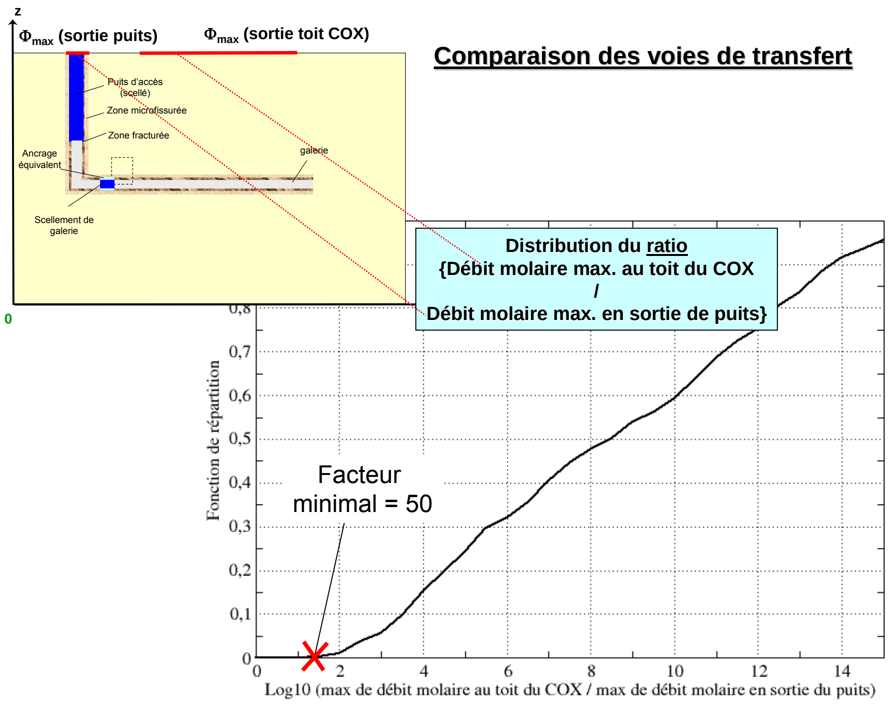


Figure 7.6-5 Probabilistic calculation – Comparison of transfer pathways – Ratio of the maximum molar flow rate out of the top of the Callovo-Oxfordian formation and the maximum molar flow rate out of the shafts : reference packages C1/C2 – ^{129}I

Transfer pathway through the sound Callovo-Oxfordian

Peclet number values and transfer times

Figure 7.6-6, which represents the distribution function of the Peclet number values in the Callovo-Oxfordian (calculated on the basis of data from the 1000 samples), shows that diffusion is dominant in 70 % of cases and codominant with advection in 26 % of cases. Advection is dominant in only 4 % of cases, with a maximum Peclet number of 2.7. This maximum value is obtained with a high vertical permeability of 10^{-12} m/s and a steep head gradient of 0.4 m/m.

The Peclet number in a SEN appears to be at value 0.3 of the distribution function. Note that, as the Peclet number is a ratio between diffusive and advective times, alone it is not a safety indicator and must be utilized with an indicator of transfer times. It might be indeed envisaged to have an advection slightly dominant, but as the advective transfer times are so slow, safety is not compromised as a result. Consequently, this result must be analysed in conjunction with the results of the distribution function relative to the date on which the maximum molar flow rate out of the Callovo-Oxfordian occurs (see Figure 7.6-7).

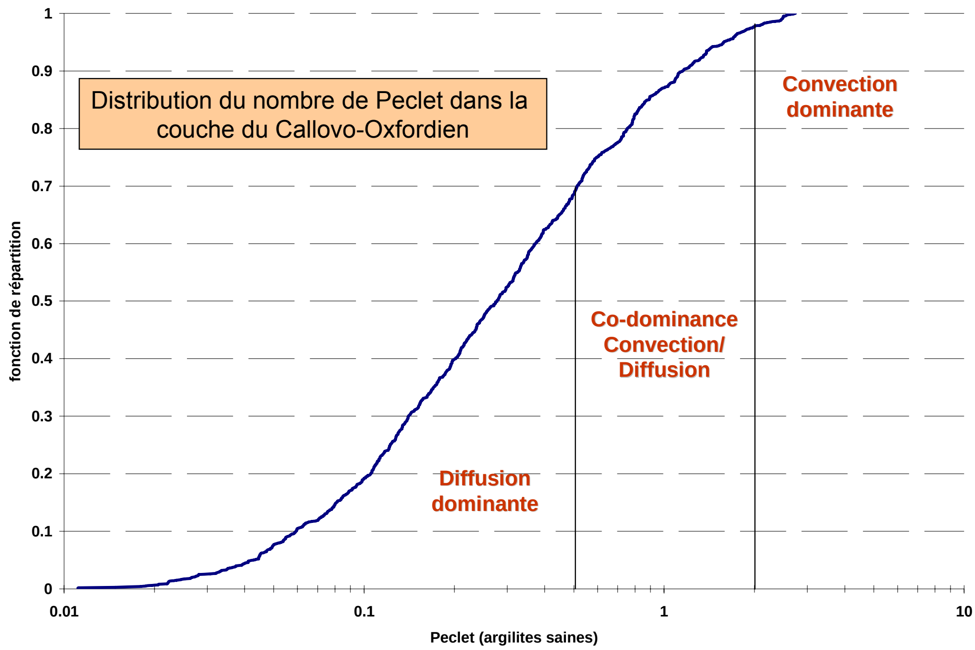


Figure 7.6-6 Probabilistic calculation – Distribution of the Peclet number values in the Callovo-Oxfordian argillite layer : Reference packages C1/C2 – ^{129}I

Figure 7.6-7 shows that the distribution of the dates on which the maximum molar flow rate out of the top of the Callovo-Oxfordian formation occurs is staggered over wide variation ranges. The shortest time is of the order of 50,000 years and the longest of the order of several million years, when the effect of the radioactive decay of iodine-129 has a significant influence. Occurrence times of maximum molar flow rates out of the Callovo-Oxfordian formation exceeding 5 million years all correspond to data sets in which the iodine is sorbed into the Callovo-Oxfordian.

As an indication, this result can be compared to the one coming from the SEN sensitivity study on a model based only on the initial dissolution of the glass (which is the model considered in the probabilistic study) ; it can be seen that around 95 % of the maximum molar flow rate occurrence dates are later than that obtained from the sensitivity study (250,000 years - see Table 5.5-24). This confirmes, cautious accompanying an uncertainty study, the conservative nature of the SEN.

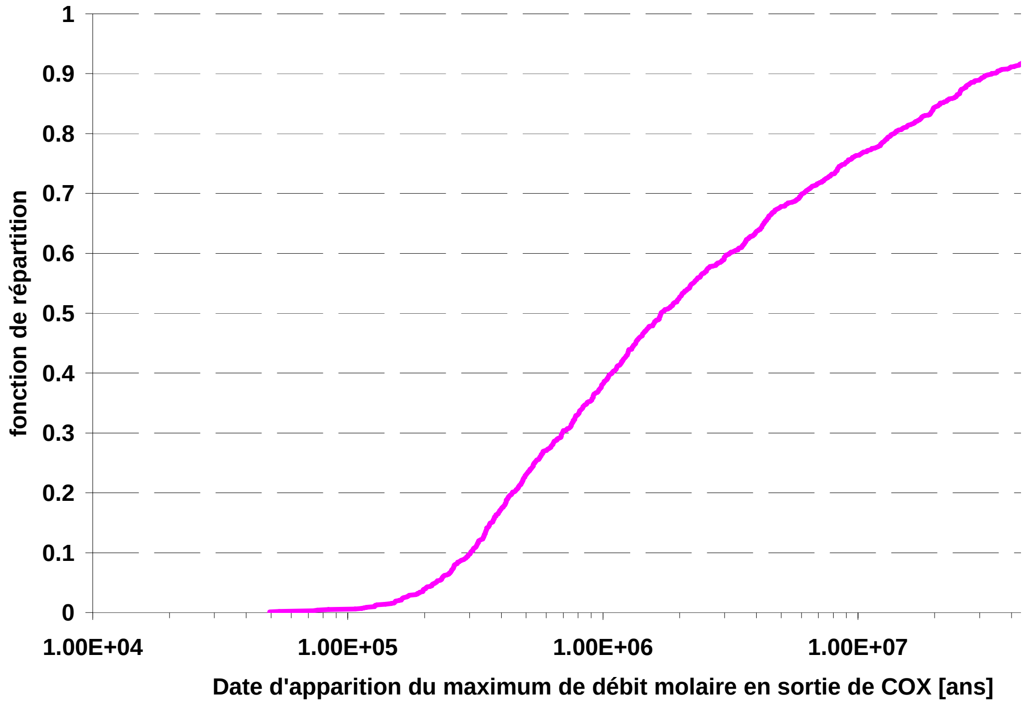


Figure 7.6-7 Probabilistic calculation - Distribution of the dates of occurrence of the maximum molar flow rate out of the top of the Callovo-Oxfordian argillite layer : reference packages C1/C2 – ^{129}I (with retention considered)

Molar flow rate out of the top of the Callovo-Oxfordian formation

Figure 7.6-8 provides various quantiles (1^{st}, 25^{th}, 50^{th} (median), 75^{th} and 99^{th} percentiles) as well as the average and the maximum curve (envelope recreated for each instant) of the molar flow rates out of the top of the Callovo-Oxfordian formation. This figure also recalls the deterministic calculation curves relating to the normal evolution scenario and to the « several degraded evolution » scenario. The comments are as follows :

- a quarter (25th percentile) of the numerical simulations performed lead to an iodine release of less than 10^{-10} mol/year (i.e. negligible) up to one million years ;
- more than a half of the releases at the Callovo-Oxfordian output present on occurrence time of the maximum over one million years ;
- the deterministic reference curve of the SEN coincides with the 99th percentile. This point confirms the conservative nature of the deterministic calculation for which the values of the parameters are generally slightly more pessimistic than the average of the distributions considered. This result is also due to the fact that a careful zero Kd value is taken into account in the deterministic reference calculation. We shall see later that considering an exclusively zero Kd has a significant influence on the result but does not raise to question the conservative nature of the deterministic calculations (see section 7.6.3.3) ;
- the scattering of maximum molar flow rates out of the Callovo-Oxfordian is low : 49 % of the results (from the median to the 99th percentile) are situated within a range of slightly more than an order of magnitude ;
- the deterministic curve relative to « severely degraded evolution » (see chapter 7.5) has a higher maximum molar flow rate than that of the envelope maximum curve of the various simulations. As an indication, Table 7.6-6 compares the values of the hydraulic, transfer and retention parameters in the Callovo-Oxfordian for the « severely degraded evolution » scenario and for the worst case resulting from the sampling.

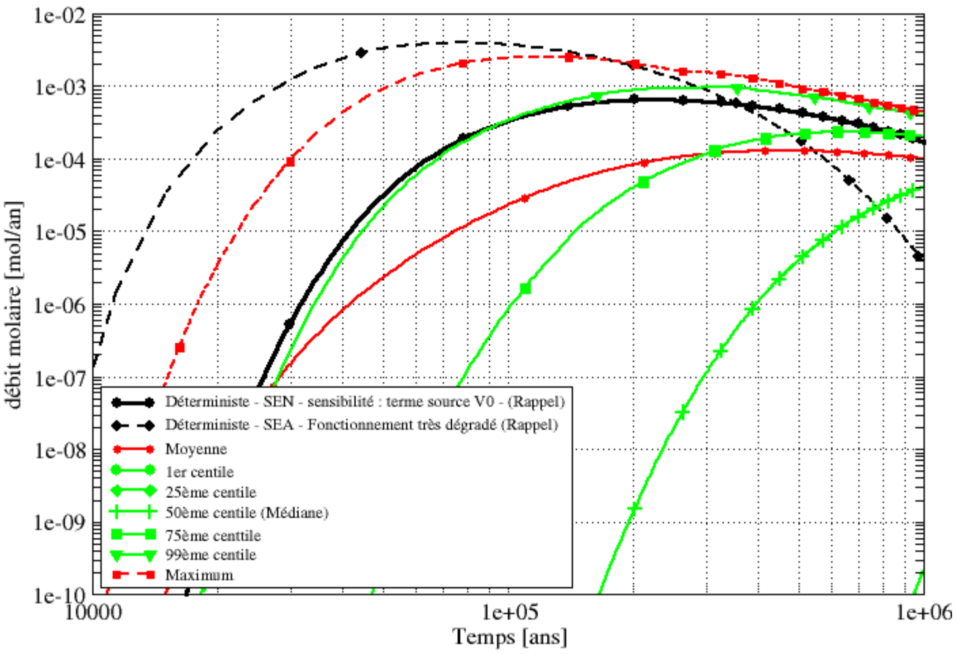


Figure 7.6-8 Probabilistic calculation – Quantiles of molar flow rate out of the top of the Callovo-Oxfordian : reference packages C1/C2 – ^{129}I (with retention considered)

Parameter values in the Callovo-Oxfordian in the « severely degraded evolution » scenario (see Chapter 7.5)	Sampled parameter values in the Callovo-Oxfordian leading to the absolute maximum molar flow rate
K^v = 10^{-12} m/s K^h = 10^{-12} m/s De = 10^{-11} m²/s ωDiffusion = 0.04 GradH = 0.4 m/m R = 1 Other values of little influence	K^v = $7.5.10^{-13}$ m/s K^h = $1.7.10^{-12}$ m/s De = $5.6.10^{-12}$ m²/s ωDiffusion = 0.03 GradH = 0.3 m/m R = 1 Other values of little influence

Table 7.6-6 Probabilistic calculation – Comparison between of the values of the hydraulic and transport parameters of the « severely degraded evolution » scenario and those leading to the maximum molar flow rate out of the Callovo-Oxfordian : reference packages C1/C2 – ^{129}I (with retention considered)

Transfer pathway through the repository structures

<u>Peclet number values</u>

Figure 7.6-9 illustrates the distribution function for Peclet number values in the secondary connecting drift backfill, at the zone seal outputs. The Peclet number value incorporates the hydraulics of the entire repository zone.

It shows that diffusion is dominant or codominant with advection in approximately 60 % of the cases. An examination of the data sets shows that the lowest Peclet number values are reached with the lowest permeability values of the microfissured zone, which confirms the strong correlation of results with the permeability of the microfissured zone.

Remember that the Peclet number values of the deterministic reference calculations of the SEN (all seals effective), the « seal failure » scenario (all seals defective) and the « severely degraded evolution » scenario are of the order of 10, 25 and 720 respectively. Respectively 20 %, 8 % and no data set of the probabilistic study lead to higher Peclet values. However, the Peclet number does not directly constitute a safety indicator ; no conclusion on the conservative nature of the scenarios can therefore be drawn at this stage.

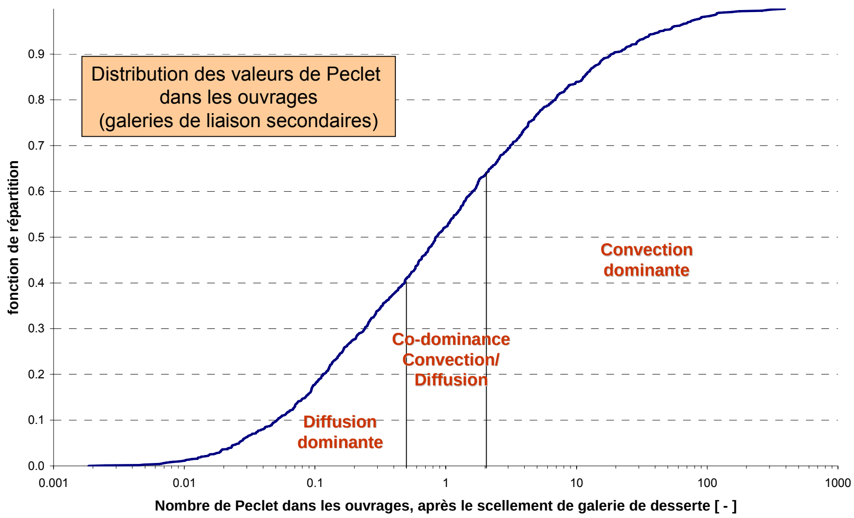


Figure 7.6-9 Probabilistic calculation – Distribution of the Peclet number values in the secondary connecting drifts : reference packages C1/C2 – ^{129}I

<u>Molar flow rate out of the shaft</u>

Figure 7.6-10 gives various quantiles (1st, 25th, 50th (median), 75th and 99th percentiles) as well as the average and the maximum curve (envelope recreated for each instant) of the molar flow rates out of the shafts and their EDZ. This figure also recalls the deterministic calculations relating to the normal evolution scenario and to the « several degraded evolution » scenario. The comments are as follows :

- approximately three-quarters (75th percentile) of the numerical simulations performed lead to an iodine release at the shaft output of less than 10^{-12} mol/year before one million years ;
- the scatter of maximum molar flow rates out of the shafts up to one million years is high : 24 % of the results (between the 75th and 99th percentile) are situated within a range of more than four orders of magnitude ;
- an examination of the curves presenting a significant release at the shaft output shows that the corresponding data sets are thoses comprising high EDZ permeability values, and low iodine retardation coefficients. This point will be examined further in the analysis of sensitivity to parameters presented later ;
- the deterministic curve of the « severely degraded evolution » scenario has a higher maximum molar flow rate which occurs earlier than the maximum curve of the various simulations. As an indication, Table 7.6-7 compares the values of the hydraulic, transfer and retention parameters in the various repository components for the « severely degraded evolution » scenario and for the least favourable case resulting from the sampling. Note that the data set leading to a maximum release from the transfer pathway through the engineered structures differs from the one relating to the transfer pathway through the sound argillites (See table 7.6-6)

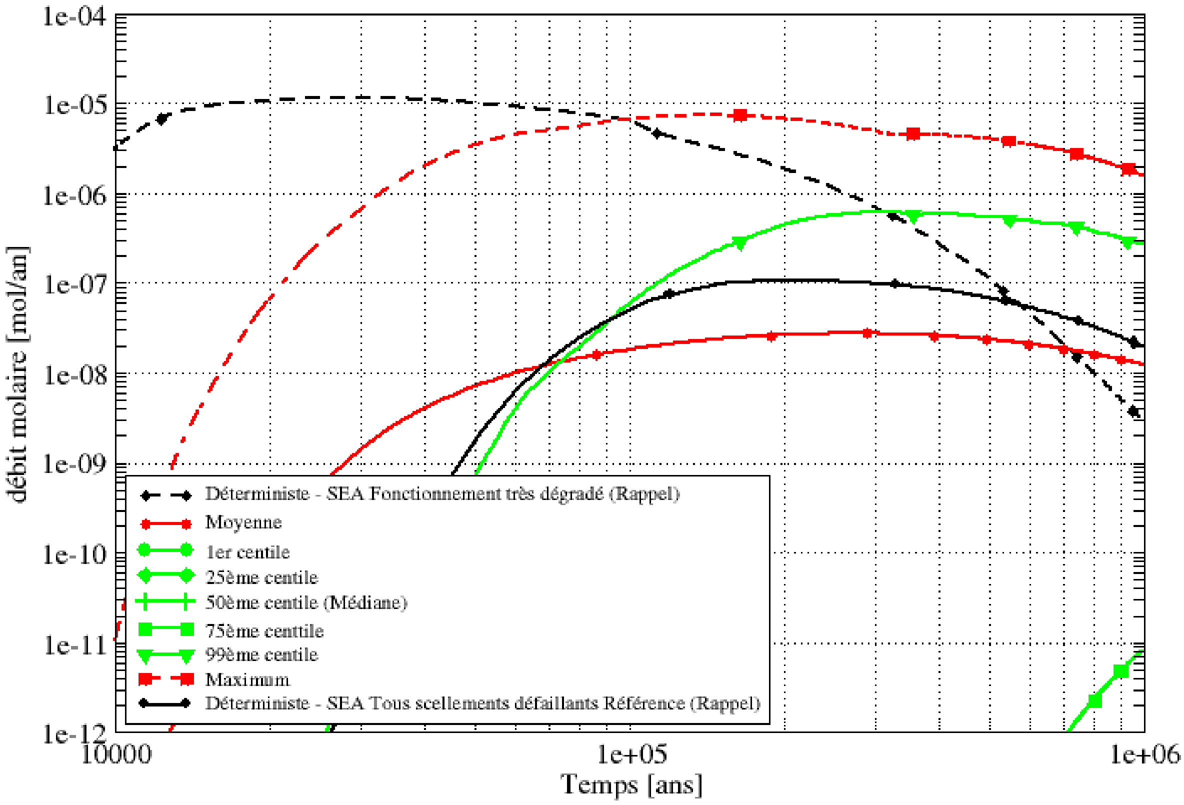


Figure 7.6-10 Probabilistic calculation – Quantiles of molar flow rate out of the shafts : reference packages C1/C2 – ^{129}I (with retention considered)

Parameter values in the « severely degraded evolution » scenario (see Chapter 7.5)	Sampled parameter values leading to the absolute maximum molar flow rate
Callovo-Oxfordian	
K^v = 10^{-12} m/s K^h = 10^{-12} m/s De = 10^{-11} m²/s ωDiffusion = 0.04 GradH = 0.4 m/m R = 1 Other values of little influence	K^v = $2.7.10^{-13}$ m/s K^h = 10^{-12} m/s De = 5.10^{-12} m²/s ωDiffusion = 0.067 GradH = 0.35 m/m R = 1.7 Other values of little influence
Microfissured EDZ	
K = 5.10^{-9} m/s De = 10^{-11} m²/s ωDiffusion = 0.04 Other values of little influence	K = $4.5.10^{-9}$ m/s De = 8.10^{-12} m²/s ωDiffusion = 0.08 Other values of little influence
Fractured EDZ	
K = 10^{-6} m/s De = 4.10^{-10} m²/s ωDiffusion = 0.15 Other values of little influence	K = 8.10^{-8} m/s De = 10^{-11} m²/s ωDiffusion = 0.2 Other values of little influence

Table 7.6-7 Probabilistic calculation – Comparison of the values of the hydraulic and transport parameters of the « severely degraded evolution » scenario and those involving the maximum molar flow rate out of the shaft and its EDZ : reference packages C1/C2 – ^{129}I (with retention considered)

- **Sensitivity analysis**

General

The purpose of the probabilistic sensitivity analysis conducted here is to determine the parameters of the model whose uncertainty has the greatest influence on the uncertainty of the result. The indicators that were utilised are the same as the ones adopted previously : the mass and the molar flow rate out of the shaft and out of the sound Callovo-Oxfordian formation.

For each physical indicator, three statistical coefficients have been calculated according to time and for the maxima of the indicators[111] :

- Spearman's correlation coefficient ;
- the partial rank correlation coefficient (PRCC). This coefficient is used to assess the influence of parameter uncertainty on the result over time ;
- the standardised regression coefficient (SRRC).

The nearer the value of these indicators is to 1 (or –1), the stronger the positive (or negative) correlation between the input parameter and the result. A value close to 0 indicates a weak correlation.

The absolute value has been calculated for each of the statistical coefficients. Consequently, in order to meet the objective of ranking the parameters whose uncertainty has a greatest influence on the uncertainty of the result (see Figure 7.6-12), the statistical indicator is defined as the average of the absolute values of the statistical coefficients[112] applied to the maximum molar flow rate over the time scale extending up to one million years :

$$I = \frac{|SPEAR| + |PRCC| + |SRRC|}{3}$$

Transfer pathway through the sound Callovo-Oxfordian

Figure 7.6-11 shows, for example, the scattered plots representing the maximum molar flow rate according to the value of the parameter of three types of variable :

- the effective diffusion coefficient (De) of the sound Callovo-Oxfordian. Correlation is positive (the « leading » slope of the scatter diagram is positive) which means that, overall, the value of the result increases as the diffusion coefficient increases. A certain scattering of the results can however be seen, revealing the influence of other parameters (in this case, Kd of iodine) ;
- the gradient of the vertical heads (gradH) ascending in the sound Callovo-Oxfordian. Correlation is nil which means that the result is highly scattered (over several orders of magnitude), whatever the variation of this parameter ;
- the partition coefficient (Kd) of iodine in the Callovo-Oxfordian. Correlation is negative (the « leading » slope of the scatter diagram is negative) which means that the value of the result decreases as the value of the parameter increases. The limited scatter of the results indicates strong correlation, especially when Kd values are low. The 20 % of zero Kd provide the strongest results.

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[111] The following analysis is conducted on the molar flow rate indicator insofar as a preliminary study has revealed similar results between the molar flow rate and the mass (integral of the molar flow rate).

[112] At this stage, no preference is given to one coefficient over another provided that they give the same tendencies.

Figure 7.6-11 Probabilistic calculation – Scattered plots linking the indicator and an example of variables : reference packages C1/C2 – ^{129}I (with retention considered)

Using the method based on the correlation coefficients explained previously (average of the absolute values of the Spearman coefficient, the PRCC and the SRRC), the parameters whose uncertainty has the greatest influence on the uncertainty of the result are identified (maximum molar flow rate out of the top of the Callovo-Oxfordian formation – see Figure 7.6-12). The latter are, in order of importance :

- the distribution coefficient of iodine in the Callovo-Oxfordian (which clearly appears as the dominant parameter) ;
- the effective diffusion coefficient in the Callovo-Oxfordian (the uncertainty of this coefficient is restricted, but it has a major influence on the molar flow rate),
- the vertical permeability of the Callovo-Oxfordian (for opposite reasons to the diffusion coefficient : the parameter has a little influence, but has a broad variation range) ;
- the porosity accessible to diffusion in the Callovo-Oxfordian.

We can also conclude that the contribution of the other magnitudes (ascending head gradient, horizontal permeability, EDZ parameters, etc.) is negligible with regard to the indicator corresponding to of the molar flow rate out of the top of the Callovo-Oxfordian formation, either as intrinsically they have a little influence or as their variation interval is too restricted for them to have an influence.

These results confirm those of the deterministic sensitivity studies of the SEN, which show a significant effect of the iodine Kd (sensitivity 1.4 of Table 5.5-17) and the diffusion coefficient (sensitivity 1.3 of Table 5.5-17). They also help in ranking the parameters for which a reduction in uncertainty would lead to a reduction in the uncertainty of the result.

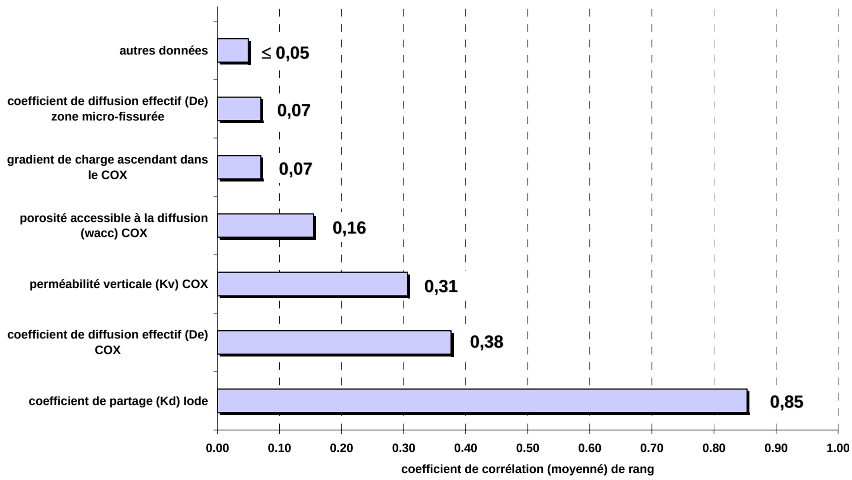


Figure 7.6-12 Probabilistic calculation – Histogram of correlation coefficient values between the maximum molar flow rate out of the top of the Callovo-Oxfordian formation and the various input variables : reference packages C1/C2 – ^{129}I (with retention considered)

Transfer pathway through the repository structures

The same type of analysis as above is conducted on the transfer pathway through the engineered structures. Based on the calculation of the correlation coefficients provided in Figure 7.6-13, the parameters whose uncertainty has the greatest influence on the uncertainty of the result (molar flow rate out of the shaft) are as follows, in descending order of importance :

- the permeability of the microfissured zone,
- the iodine partition coefficient,
- the permeability of the fractured zone,
- the effective diffusion coefficient in the Callovo-Oxfordian,
- the permeability of the bentonite,
- the ascending head gradient in the Callovo-Oxfordian.

We can also conclude that the contribution of the other magnitudes (permeability of the Callovo-Oxfordian, EDZ and bentonite diffusion, etc.) is negligible with regard to the indicator of the molar flow rate out of the shafts.

The importance of EDZ permeability levels on the sensitivity of the results is noted since they control the advective transfer pathway to the access shafts. In particular, the permeability of the microfissured zone (of which 90 % of the values lie within the 10^{-13} to 10^{-10} m/s range) influences the nature of the hydraulic regime in the engineered structures significantly. The variation in the permeability of the fractured zone (of which 90 % of the values lie within the 10^{-11} to 10^{-8} m/s range) has a more ustricted influence owing to a smaller cross-section and the existence of a cut-off at the seals (remember that these seals on the other hand are not represented as anchored in the microfissured zone).

The importance of the iodine partition coefficient on the sensitivity of the results is also noted. As retention is taken into account in the EDZ and in the sound Callovo-Oxfordian, but not in the drift backfill, it restricts the transfer of radionuclides through the EDZ. It also indirectly influences the migration of iodine through the drift due to of the rock's « absorbing barrier » effect.

An examination of the results also shows the significant sensitivity of the diffusion in the Callovo-Oxfordian on the molar flow rate out of the shafts with a negative correlation : as the diffusion coefficient increases, the molar flow rate out of the shafts decreases. This point is explained by the fact that a significant part of the mass passing through the EDZ along the drifts migrates radially into the sound Callovo-Oxfordian. The permeability of the bentonite has also a great influence on the sensitivity of the results : this permeability (of which 90 % of the values range from 2.10^{-13} to 4.10^{-11} m/s) helps to limit the hydraulic disturbances and the transfer into the drifts, facing the variation in the other hydraulic parameters. Slight sensitivity of the results to the ascending head gradient in the Callovo-Oxfordian is noted, the variation of which is masked by those of the other parameters.

The influence of the other parameters (Callovo-Oxfordian permeability levels, diffusion coefficient of the EDZ and bentonite, etc.) can be considered as insignificant.

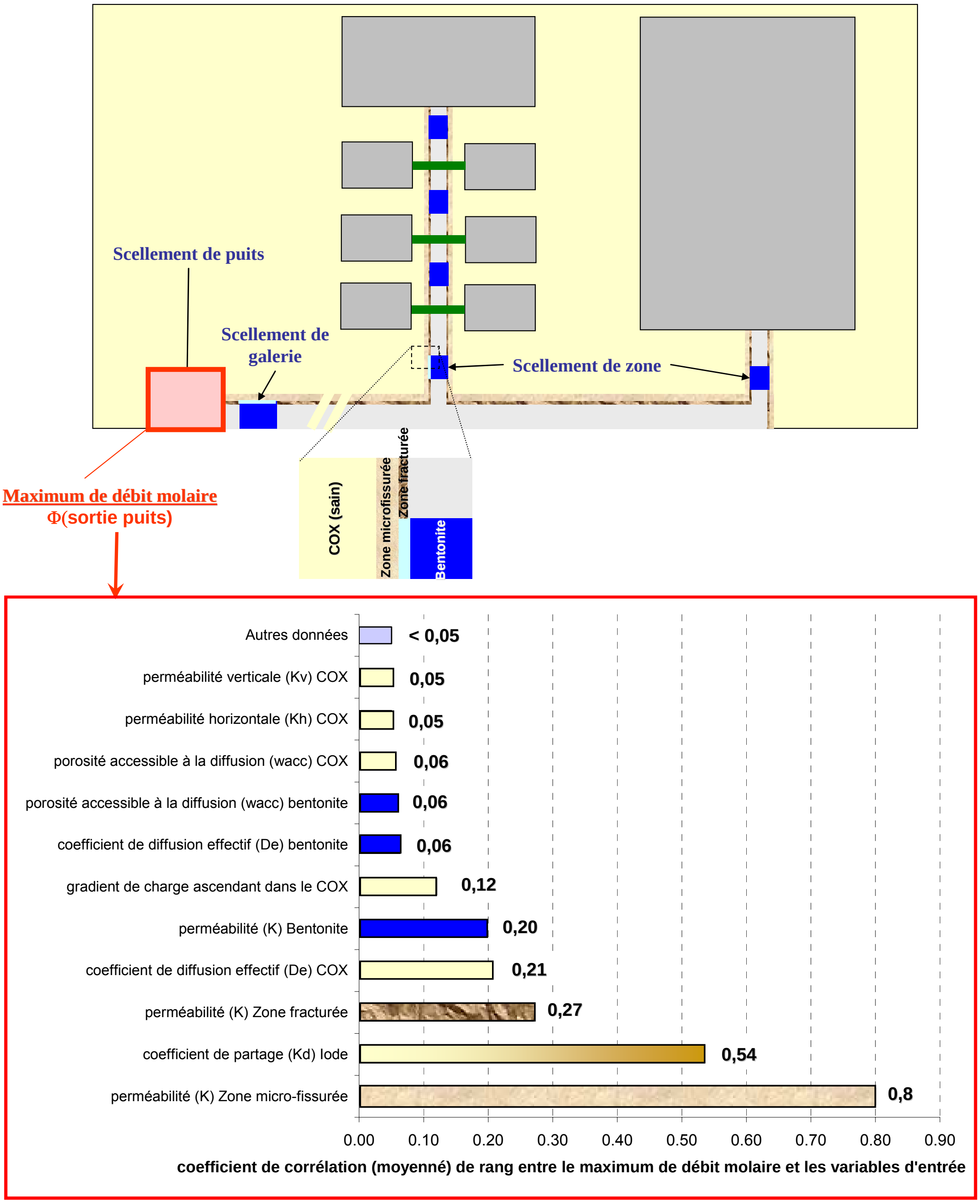


Figure 7.6-13 Probabilistic calculation – Histogram of correlation coefficient values between the molar flow rate out of the shafts and their EDZ and each input variable : reference packages C1/C2 – ^{129}I (with retention considered)

To illustrate the above points, Figure 7.6-14 provides the example of the scattered plot (values of the parameter represented with the corresponding maximum molar flow rate) of the permeability of the microfissured zone, showing a strong correlation (positive).

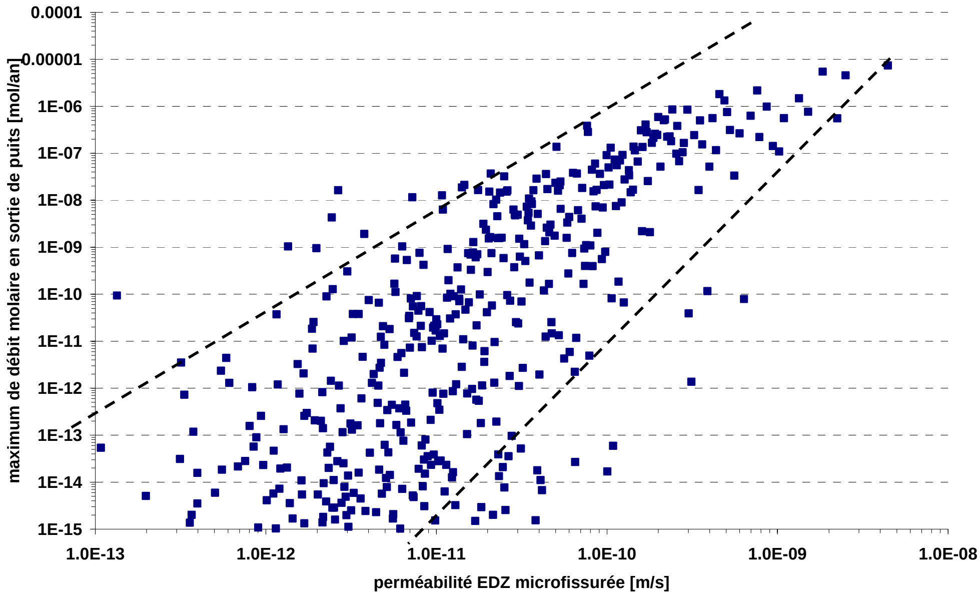


Figure 7.6-14 Probabilistic calculation – Scatter diagram linking the maximum molar flow rate out of the shafts and the permeability of the microfissured zone : reference packages C1/C2 – ^{129}I (with retention considered)

7.6.3.3 Case of iodine with no chemical retention considered (zero Kd value)

As a precaution and to assess the conservative nature of the SEN calculation even in the absence of chemical retention of the iodine, a second probabilistic calculation was performed in which the 1000 samples taken previously were all allocated a zero partition coefficient value.

- **Uncertainty analysis**

Figure 7.6-15 compares the distribution of the occurrence dates of the maximum molar flow rates out of the Callovo-Oxfordian with taking a zero Kd or a variable Kd. The results are clearly less scattered for the zero Kd since 90 % of the maxima are reached before one million year versus 35 % in the case of a variable Kd. As shown by the results of the deterministic study, this point illustrates the strong dependency of the results on the iodine Kd.

Note that even when the Kd is egal to zero, the deterministic value of the occurrence time of the maximum release out of the Callovo-Oxfordian (approximately 250,000 years) is situated around the first quartile of the distribution (versus the 5th percentile at a non-zero Kd). This point confirms the conservative nature of the deterministic calculation of the SEN.

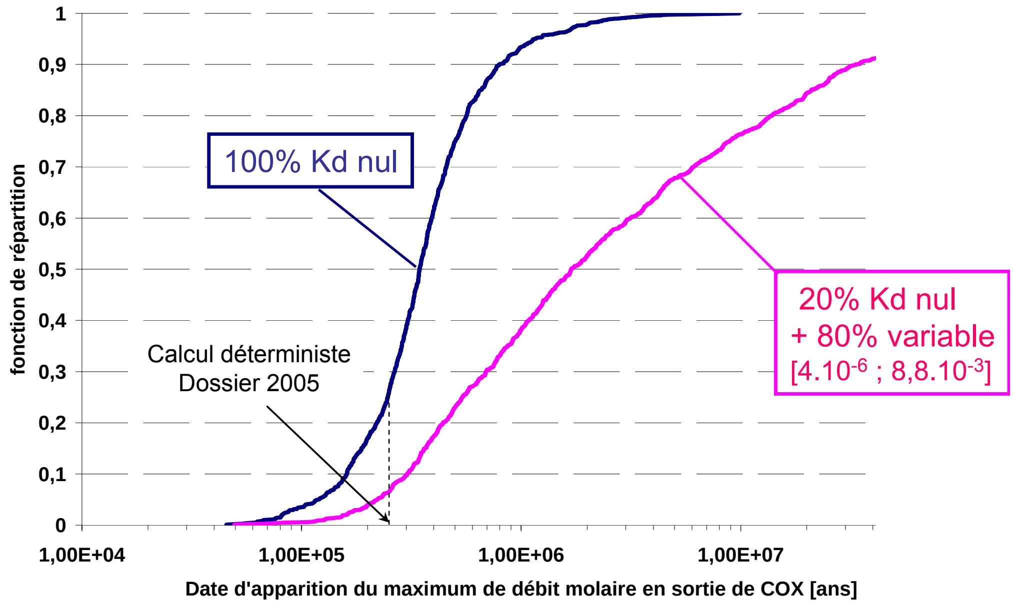
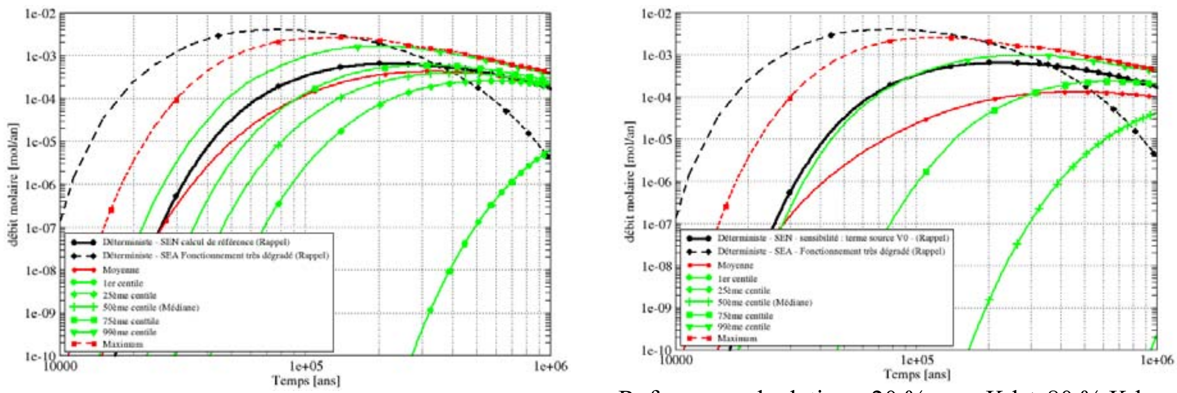


Figure 7.6-15 Probabilistic calculation – Comparison of times for the maximum molar flow rate out of the Callovo-Oxfordian to occur, with and without consideration of iodine sorption : reference packages C1/C2

From Figure 7.6-16, it can be concluded that :

- where Kd is zero, the scattering of maximum molar flow rates out of the Callovo-Oxfordian is low : 80 % of the results are situated within a range of slightly less than an order of magnitude. This limited scattering can be explained by the strong correlation of the results with diffusion (Figure 7.6-17), and by its low variability ;
- if the deterministic reference curve of the SEN fits with the 99th percentile in cases with a variable Kd, it coincides with the 75th percentile of the curve at zero Kd. This point confirms the conservative nature of the SEN reference calculation ;
- the curve of the « severely degraded evolution » scenario has a maximum impact in excess of the envelope maximum curve.



Sensitivity calculation : zero Kd

Reference calculation : 20 % zero Kd + 80 % Kd between [4.10-6 ; 8.8.10-3]

Figure 7.6-16 Probabilistic calculation – Percentiles of molar flow rate out of the top of the Callovo-Oxfordian with variable Kd and zero Kd : reference packages C1/C2 – ^{129}I

- **Sensitivity analysis**

The results show that in the absence of iodine retention (zero Kd), the ranking of the other parameters remains unchanged ; on the other hand, the correlations are necessarily stronger (Figure 7.6-17).

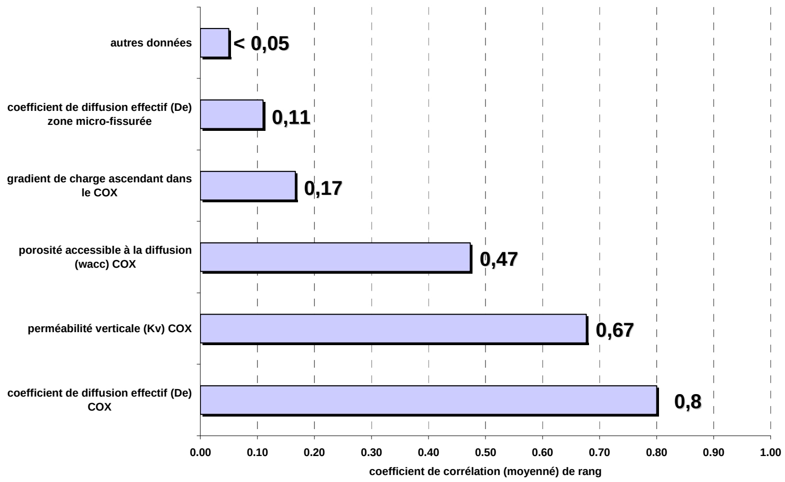


Figure 7.6-17 Probabilistic calculation – Histogram of correlation coefficient values between the maximum molar flow rate out of the top of the Callovo-Oxfordian formation and the various input variables : reference packages C1/C2 – ^{129}I (with no iodine retention considered – zero Kd)

7.6.3.4 Case of selenium-79

- **Uncertainty analysis**

Transfer pathway through the sound Callovo-Oxfordian

Figure 7.6-18 shows the distribution of molar flow rates of selenium-79 at the top of the Callovo-Oxfordian formation, according to the same principle as for iodine.

The results are similar to those observed for iodine (with variable Kd) since around half of the numerical simulations that were conducted lead to a negligible ^{79}Se release and the deterministic reference curve of the SEN fits with the 90th percentile (versus the 99th percentile for iodine). This result is due to the fact that, in the deterministic reference calculation, the parameter values considered are generally slightly more pessimistic than the average of the distributions considered and that a zero Kd value is considered whereas, in the probabilistic calculation, the sorption of selenium is taken into account in 80 % of cases (as for iodine).

On the other hand, the maximum molar flow rates out of the Callovo-Oxfordian are significantly more scattered that in the case of iodine since 49 % of the results (between the median and the 99th percentile) lie within a range of approximately seven orders of magnitude (compared with slightly over one order of magnitude for iodine). This extensive scattering is due to the relatively short half-life of selenium-79[113], which makes it more sensitive to variations in transfer time and to variability in the solubility limit (from 5.10^{-10} to 5.10^{-4} mol/m^3). This becomes a sensitive parameter as soon as the sampled Kd values are low.

The severely degraded evolution scenario appears to be very pessimistic for selenium also, in view of all of the simulations.

[113] The calculation was conducted with the half life of Sélénium 79 coming from the data base JEFF 2.2 ; which is 65 000 years.

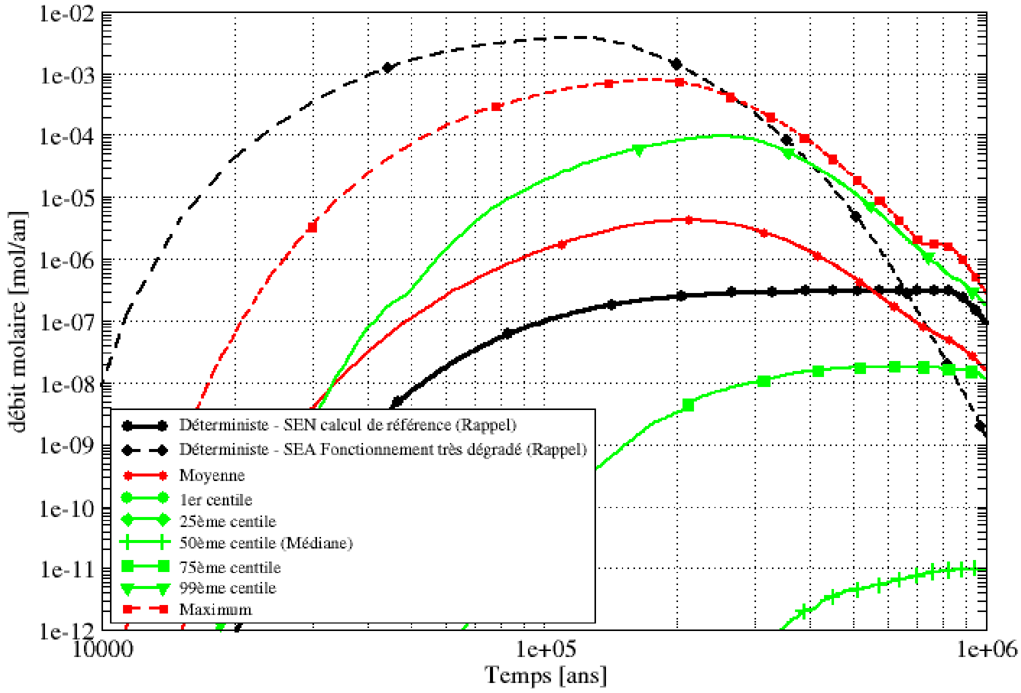


Figure 7.6-18 Probabilistic calculation – Quantiles of molar flow rate out of the top of the Callovo-Oxfordian : reference packages C1/C2 - ^{79}Se

Transfer pathway through the repository structures

Figure 7.6-19 shows that, considering the significant transfer times in engineered structures, selenium with its shorter half-life than iodine-129 benefits more from the radioactive decay before reaching the shafts. Thus, approximately 90 % of the numerical simulations performed lead to a release of ^{79}Se at the shaft output of less than 10^{-12} mol/year (versus approximately 75 % for iodine).

Out of the 25 % of curves with a significant molar flow rate out of the shafts, the most pessimistic are the ones associated with high EDZ permeability values (microfissured and fractured zone) and low ^{79}Se distribution coefficient values.

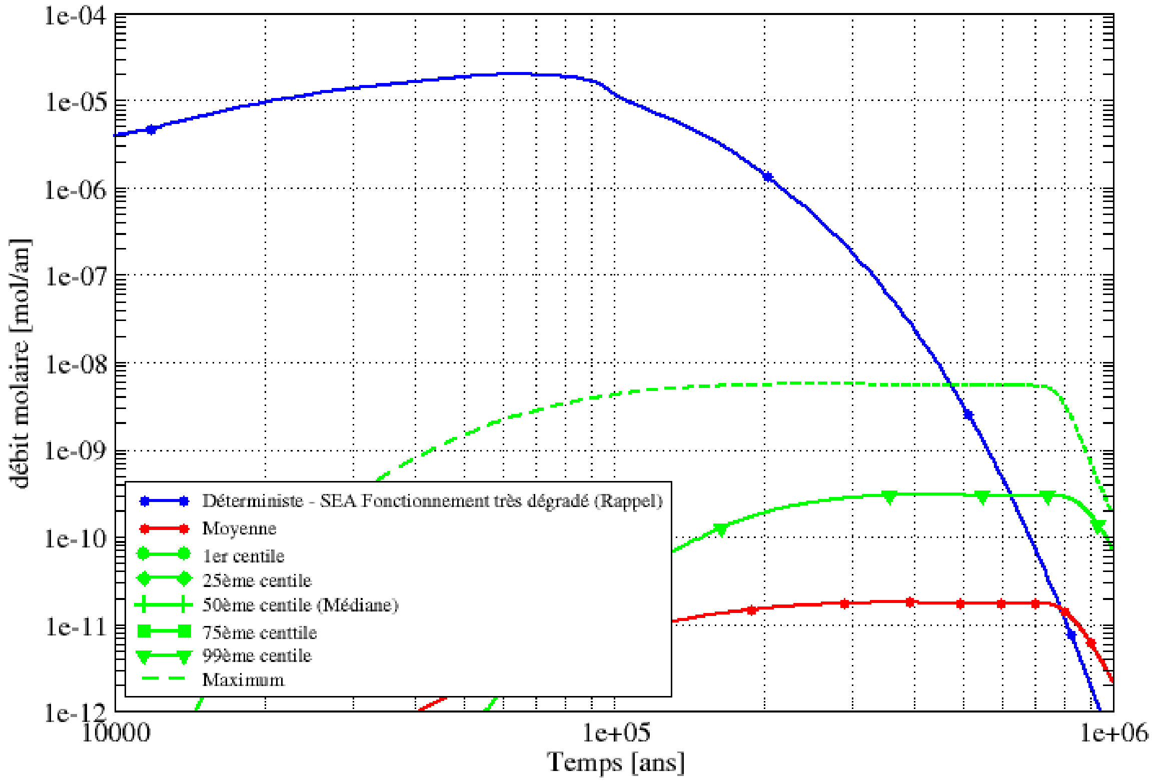


Figure 7.6-19 Probabilistic calculation – Quantiles of molar flow rate out of the shaft : reference packages C1/C2 - ^{79}Se

- **Sensitivity analysis**

Transfer pathway through the sound Callovo-Oxfordian

Figure 7.6-20 provides histograms of correlation coefficient values which reveal a parameter ranking similar to the one observed in the case of iodine. In the case of selenium, the solubility limit is a further parameter, which appears to be of equal importance to the diffusion coefficient with regard to the uncertainty of the calculation result. The weaker the sorption, the more solubility influences the transfer. Had we conducted a probabilistic calculation forcing the selenium Kd to a zero value, we would have observed a proportionally more significant contribution of the solubility limit.

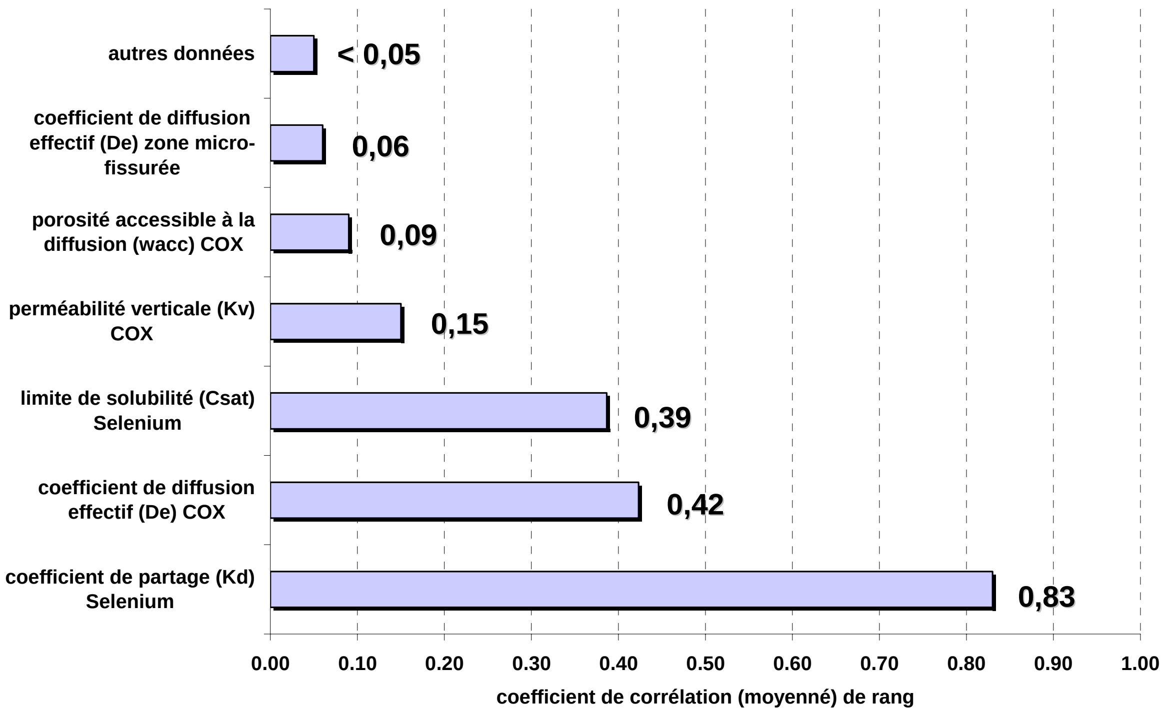


Figure 7.6-20 Probabilistic calculation – Histogram of correlation coefficient values between the maximum molar flow rate out of the top of the Callovo-Oxfordian formation and the various input variables : reference packages C1/C2 - 79Se

- **Transfer pathway through the repository structures**

The same type of analysis as above is conducted on the transfer pathway through the engineered structures as illustrated in Figure 7.6-21. The ranking and remarks are globally identical to those applicable to iodine. The main three parameters (permeability of the microfissured and fractured zones, and selenium Kd) help control the advective transfer pathway and mask the direct influence of the other parameters (notably the solubility limit). Note that the influence of Callovo-Oxfordian diffusion is more limited than for iodine (weak correlation between the result and effective diffusion) due to the significant influence of selenium solubility in the materials.

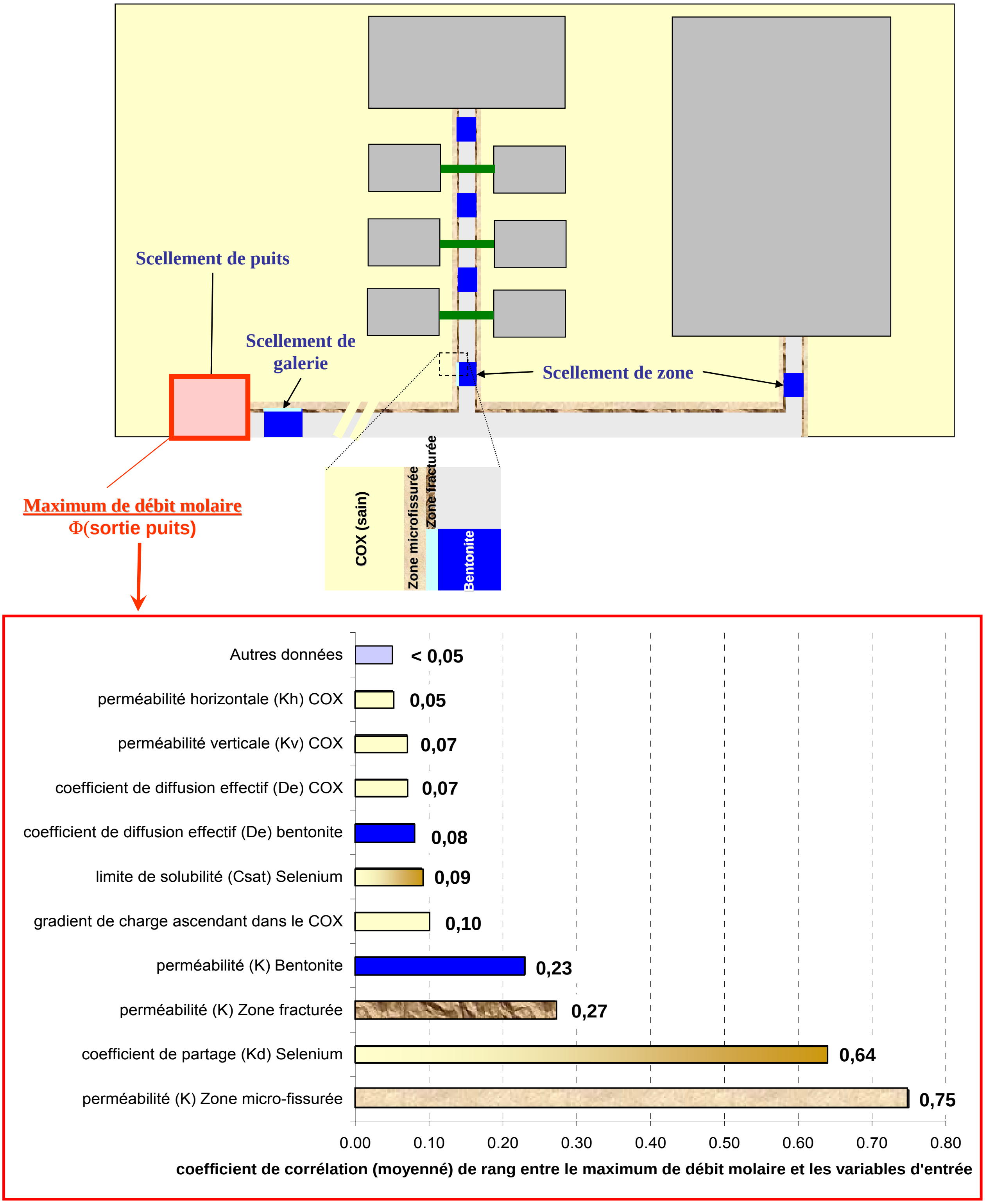


Figure 7.6-21 Probabilistic calculation – Histogram of rank correlation coefficient values between the molar flow rate out of the shafts and their EDZ and each input variable : reference packages C1/C2 - ^{79}Se

7.6.4 Conclusion of the probabilistic study

The purpose of the probabilistic calculations was to detail the results of the deterministic studies by ranking the most influential parameters. Additionally, they highlight the conservative nature of the SEN with respect with the various data taken into consideration. They also show that there is a margin on the severely degraded evolution scenario in relation to all of these calculation cases.

Due to the distribution functions adopted, the calculations cover a domain, which includes the SEN but also the results of the « highly degraded evolution » scenario and situations close to the « seal failure » scenario.

Concerning the distribution of the transfer pathways are concerned, the results confirm that the geological barrier is the preferential radionuclide transfer pathway in all cases.

The calculations also emphasise the gain achieved by iodine sorption into the Callovo-Oxfordian. The study involving a variable iodine Kd does indeed show this parameter to be the most influential : its correlation coefficient with the maximum molar flow rate out of the Callovo-Oxfordian is 0.85 (see *Figure 7.6-12*).

A part from the influence of the Kd, the parameters whose uncertainty has the greatest influence on the uncertainty of the result are the transport parameters (diffusion coefficient and solubility limits) and vertical permeability in the Callovo-Oxfordian. The latter point needs a special comment : as the probabilistic case study adopted a permeability variation law based on measurements on samples, it presents a variability of several orders of magnitude for this parameter. Consequently, the sensitivity analysis stresses the uncertainty on permeability to be high. This confirms the advantage of reducing the range of variation of this parameter, especially by measurements in situ and on a larger scale, in the underground laboratory (see chapter 6). However, this does not call into question the conclusions of the deterministic study : at a given variation range, the permeability remains a parameter of little influence compared with the diffusion coefficient.

Finally, concerning the transfer pathway through the engineered structures is concerned, the results confirm the significant role of the hydraulic performance of the microfissured EDZ, as already observed in the studies on the sensitivity to the degraded EDZ performance adopted in the various scenarios. However, note that the scenarios represent seals anchored only in the fractured zone. The influence of the permeability of the seal body, already visible on the results presented, would certainly be increased if the seal were represented with a longer hydraulic cut-off.

7.7 Conclusion for all scenarios

As the findings from each altered evolution scenario were given at the end of the sections of the present chapter, the conclusion will only return to a few points that seem to be the most significant.

In all the scenarios for which calculations were made, including the borehole scenario, the Callovo-Oxfordian comes out as being an important component in terms of controlling the impact. Depending on the case it is complementary or redundant compared to the other components. The low permeability of the rock at a large scale allows the water inputs to be controlled, even in situations where the seals are lost, the damaged zone is degraded or there is an intrusion of a borehole. The geochemical properties of the rock limit the progression of sorbed radionuclides, which in most situations limits the impact to only those elements that are insensitive to these properties (iodine and chlorine). It should be noted that the architecture contributes to make the most of the favourable properties of the rock : in particular the choice of « dead-end » structures effectively limits water circulation. Due to the relatively high uncertainty on the permeability values, the probabilistic calculation stresses the influence of this parameter on the results. However, analysis of the « seal failure » and « severely degraded evolution » scenarios shows that the rates of water flowing in the repository are not significant in any configuration.

Nevertheless, the seals have been shown to be components with a sensitive safety role. If a borehole were to penetrate, they help in fractionating the structure limiting the influence of hydraulic factors and the head drops. They enable possibly very soluble elements to be immobilised, which could if the seals were not present migrate through the drifts. Moreover, probabilistic processing of the SEN, covering calculation cases close to the « seal failure » SEA, emphasise the sensitivity of the radionuclide transfers to the permeability of the microfissured zone and the seal itself.

The bentonite barrier built around the spent fuel packages does not appear to be a « barrier » in the normal sense of the term. It does not play the role of limiting transfers of radionuclides, since even in the borehole scenario the geological environment substitutes for it. Neither does it appear more crucial for maintaining a diffusive system in the cells. The hydraulic system is controlled by the sealings of the shaft and drift. In the end, the role of the bentonite barrier is mainly to take into account uncertainties about the thermal effects, mostly thermo-mechanical effects in the cells.

The waste packages have a qualitatively important role, but in the impact calculation itself, as the maximum doses are not very sensitive to their performance. It should however be noted that they delay the releases from vitrified wastes, that are temperature-sensitive. The packaging matrices are an effective barrier in situations such as the borehole, provided that their performance can be mobilized indeed.

In terms of the impact, it is seen that this remains controlled, even in « severely degraded evolution » scenario, in which the repository's safety functions either have minimal performance (for the geological environment and bentonite structures) or no performance at all (for the packages). This result emphasizes the repository system's robustness. The doses resulting from water transfers are at most 0.12 mSv/year in the case of the « severely degraded evolution » scenario assuming a weakened rock permeability, a short-circuit of the seals, a failure of all the packaging components and poor or zero performance of all the waste matrices.

Concerning potential actinide impact, the results show that their contribution is zero in the « seal failure », « package failure » and « severely degraded evolution » scenarios. The conclusions relating to the transport of actinides in the geological barrier remain identical to those of the SEN : their retention in the argillites leads to a total attenuation of the molar flow rates out of the Callovo-Oxfordian over the duration of the analysis, even in a situation where the geochemistry is taken as conservative (as in the case of the « severely degraded evolution » scenario). The only contribution of actinides to the radiological impact is observed in the « borehole » scenario where this occurs in or near to a C waste or spent fuel cell. In the latter case, the contribution of actinides remains generally low compared with the contribution of fission and activation products. Note that in the case of a borehole in a B waste cell, the very high actinide sorption into the concrete makes a negligible impact.

Inset 9 Consequences linked to the change in the half-life of selenium

Following the May 2005 update of the JEFF nuclear database, used as a reference by Andra for radioactive half-lives, the Agency assessed the consequences of such an evolution on radiological impact calculations.

From the15 radionuclides studied as a priority in the SEN, three have a longer half-life in the new version of JEFF : ^{93}Mo, ^{126}Sn and ^{79}Se. Table 7.5-7 summarises the changes to the half-life of these radionuclides between JEFF versions 2.2 and 3.1.

Radionuclides	Half-life [years] in JEFF 2.2	Half-life [years] in JEFF 3.1	Ratio TJEFF3.1 / TJEFF2.2
^{93}Mo	3 500	4 000	1,14
^{126}Sn	100 000	230 000	2,3
^{79}Se	65 000	1 100 000	16,9

Change in the half-life of 93Mo, 126Sn and 79Se between JEFF versions 2.2 and 3.1

As ^{93}Mo is retained in the first few metres of the Callovo-Oxfordian, it does not contribute to the radiological impact in the SEN, even in the conservative configurations studied in terms of sensitivity. The slight extension of its half-life does not therefore appear likely to change its radiological impact significantly.

Tin-126 is highly sorbed into the geological barrier, even in conservative configurations (« severely degraded evolution » scenario, sensitivity to conservative geochemistry). Therefore in the absence of radioactive decay, the maximum dose would in any case occur well beyond the million-year mark. Taking into account the half-life of 126Sn, the molar flow rates out of the Callovo-Oxfordian (and therefore the contribution to the impact of tin) are considerably attenuated beyond the million-year mark and an increase in the half-life by a factor of 2.6 should not lead to any significant changes in its contribution either.

As for ^{79}Se, one of the three main contributors to the radiological impact, its half-life is 16.9 times longer in the new version of JEFF. Additional calculations have therefore been carried out on this radionuclide in particular.

It is noted that the change to the half-life of a radionuclide can influence the impact calculation in different ways :

- in certain cases, the half-life is one of the parameters used to evaluate the initial inventory of the radionuclide. In particular, in the main waste (CSD-V and CSD-C) and spent fuels, the inventory is assessed on the basis of calculations of fuel evolution in the reactor. These calculations involve the fission yields and specific activities of the fission products. This final parameter is correlated to the half-life ;
- the half-life is considered of course in the transport calculations, through the phenomenon of radioactive decay ;
- the half-life can also be involved in the biosphere model, in certain specific cases linked for example to radionuclide build-up situations in soils. But for very long-lived radionuclides (over 10,000 years), the factors of transfer into the various compartments of the biosphere do not depend on radioactive decay.
- In practice, the increase by a factor of approximately 17 in the half-life of ^{79}Se results :
- on the one hand, in a reduction in the specific activity by a ratio proportional to the increase in the half-life (i.e. a reduction in the initial inventory in Becquerels by a factor of approximately 17) ;
- and, on the other hand, in delaying the radioactive decay leading to a lesser attenuation of the molar flow rate out of the geological barrier.

As these two phenomena have opposite effects on the radiological impact, it is not easy to anticipate the results. Various calculations have therefore been carried out in order to assess the consequences of the increase in the half-life of ^{79}Se on the impact.

- **Calculation cases**

^{79}Se precipitates in the SEN but becomes soluble in more pessimistic configurations where conservative geochemistry is considered (sensitivity to geochemistry of the SEN, severely degraded evolution scenario). The analysis consisted of assessing the impact of the ^{79}Se (with the new half-life from the JEFF 3.1 database) in the SEN and for three additional cases :

- the sensitivity to geochemistry of the SEN where the selenium is soluble,
- the « severely degraded evolution » scenario (see chapter 7.5) where the selenium is also soluble,
- the « borehole » scenario (see chapter 7.4).

The impact has been assessed for C1/C2 reference packages which present the highest total initial inventory in ^{79}Se of all reference packages studied (around 7.10^{14} Bq for 32,100 C1/C2 reference packages) with a half-life of 1.1 million years.

All of the hypotheses are identical to those of the calculations already conducted apart from the half-life of ^{79}Se which increases by a factor of 17 and the initial activity in Becquerels which is reduced by the same ratio, due to the change in specific activity.

- **Calculation results**

Regarding the four calculation cases studied, the indicators provided (with the half-lives of JEFF 2.2 and JEFF 3.1) are :

- the molar flow rate through the diffuse fracturing zone (or through the borehole in the specific case of the « borehole » SEA). This indicator is interesting insofar as it allows for reasoning in terms of a constant number of initial moles and therefore enables the effect of only radioactive decay in the medium to be assessed ;
- the rate of activity through this same zone (or through the borehole in the specific case of the « borehole » SEA). This involves converting the molar flow rate (mol/year) into a rate of activity (Bq/year) insofar as this is the indicator used to calculate the radiological impact. In addition to the effect linked to radioactive decay in the medium, this indicator incorporates the consequences due to the decrease of the specific activity ;
- the dose at the outlet leading to the highest impact : the Saulx for the SEN and for the « severely degraded evolution » SEA, pumping at drinking water outlets in the Calcaires du Barrois (Tithonian) for the « borehole » SEA.

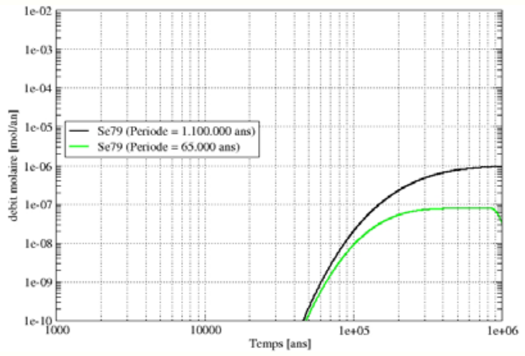
In all of the calculation cases, the increased half-life of ^{79}Se led to the following results for each of the indicators listed above (see Figure A) :

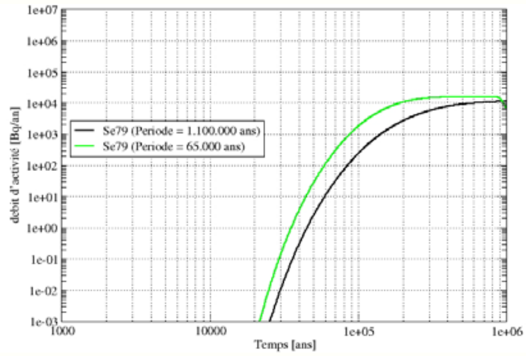
- the molar flow rate reaching the outlet is increased and prolonged due to lesser decay in the medium. In the specific case of the borehole, where the transfer times are short in the face of the old and new half-life of 79Se, there is little effect on the attenuation of the molar flow rate at the borehole ;
- the conversion of the number of moles into Becquerels leads in all cases to a lower rate of activity (linked to the half-life of 1.1 million years) than the one associated with the shorter half-life of 65,000 years. Indeed, the drop in specific activity (by a factor of 17) largely compensates in all cases for the lower decay in the medium. On the other hand, the flow is prolonged over time.

The dose, proportional to the rate of activity, is lower in the case of the longer half-life and the maxima occur later (see Figure B).

Normal evolution scenario – Reference calculation

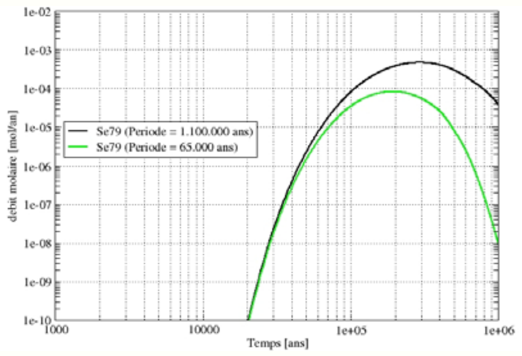
Molar flow rates reaching the outlet	Rates of activity reaching the outlet

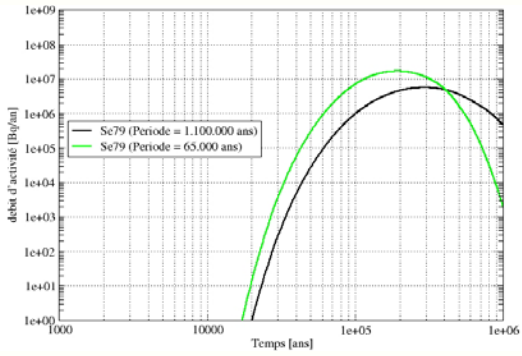




Normal evolution scenario – Study of sensitivity to geochemistry

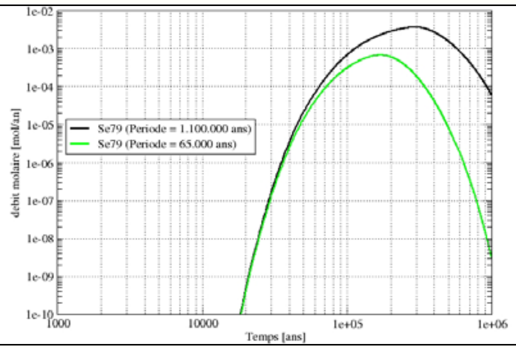
Molar flow rates reaching the outlet	Rates of activity reaching the outlet

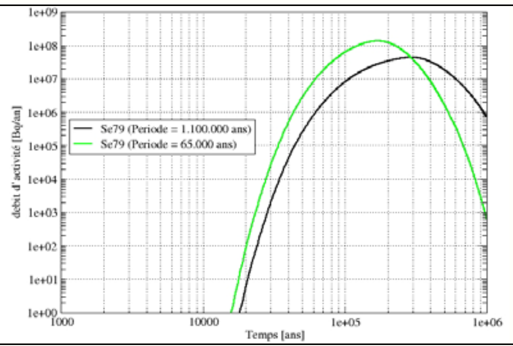




« Severely degraded evolution » scenario

Molar flow rates reaching the outlet	Rates of activity reaching the outlet





« Borehole » scenario

Molar flow rates reaching the outlet	Rates of activity reaching the outlet

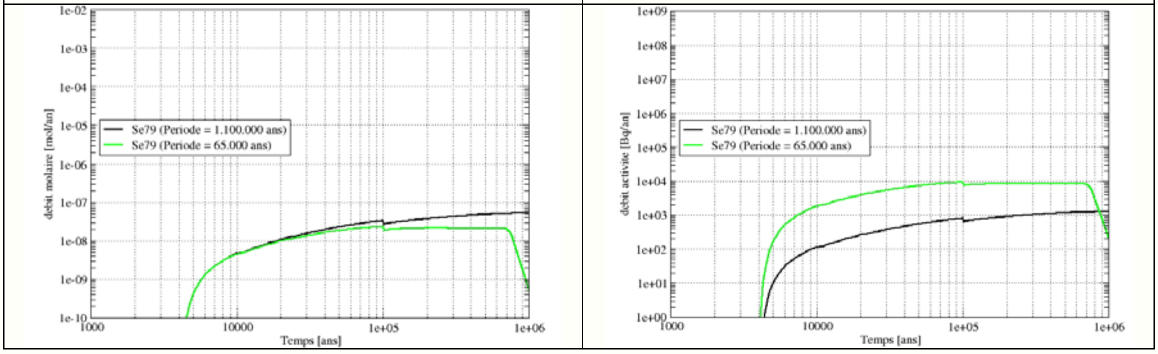
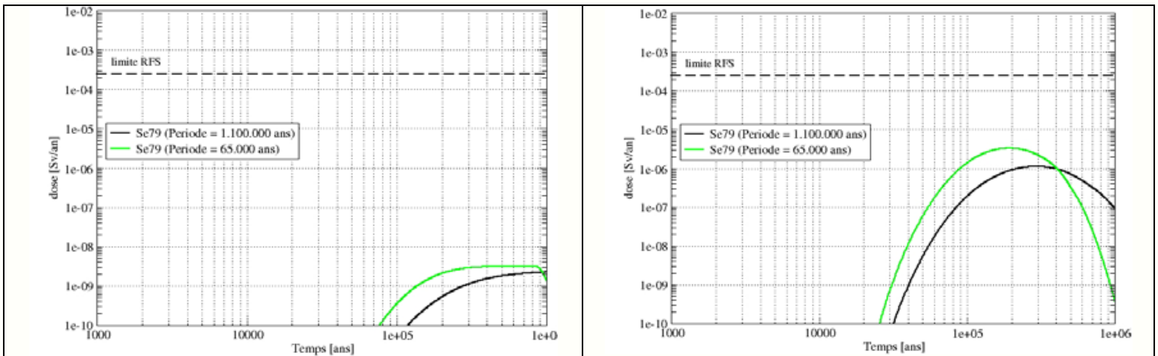
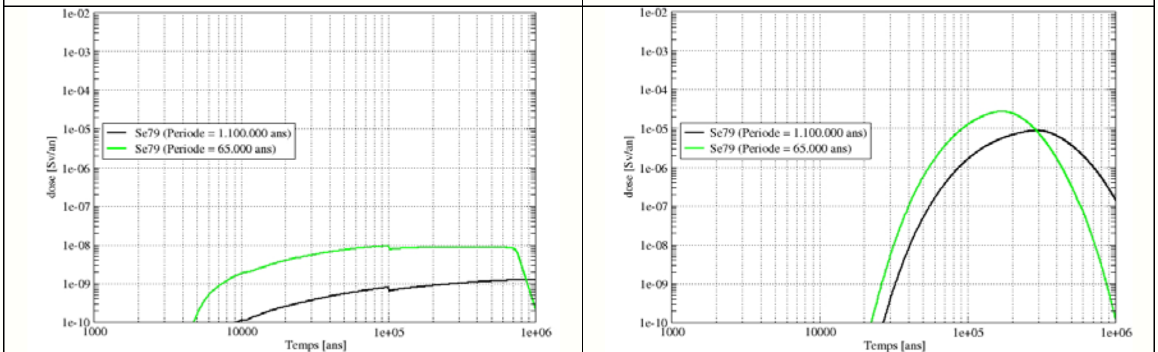


Figure A - Effects of the increased half-life of ^{79}Se – Dose history at the outlet – Reference packages C1/C2



SEN – reference calculation	SEN – sensitivity to geochemistry



« Borehole » scenario	« Severely degraded evolution » scenario

Figure B - Effects of the increased half-life of ^{79}Se – Dose history at the outlet – Reference packages C1/C2

- **Conclusion**

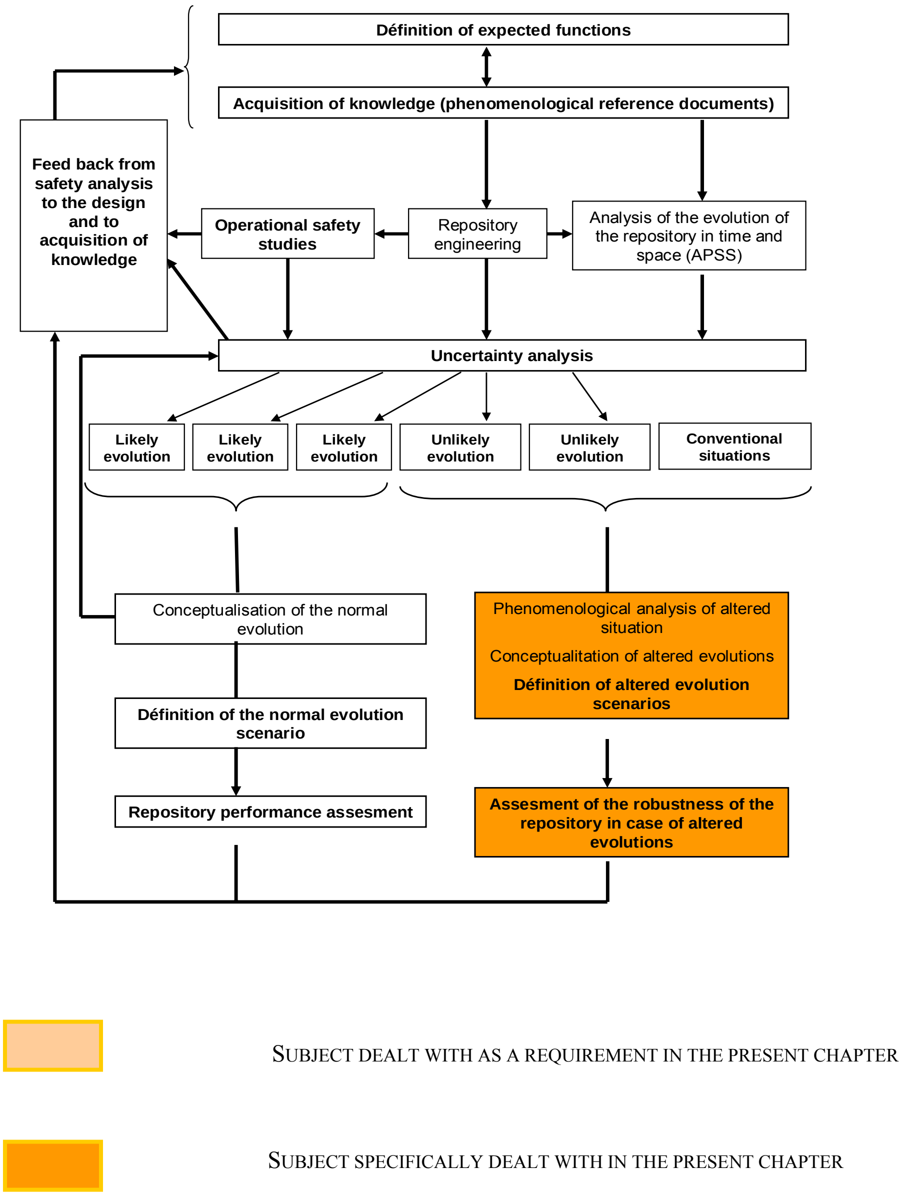
The seventeen-fold increase in the half-life of ^{79}Se leads to a reduction in initial activity by the same ratio, which compensates in all cases the lesser decay in the medium. As a consequence, even if the dose histories are prolonged with a half-life of 1.1 million years, the radiological impact is lower and the maximum dose later.

The dose values due to ^{79}Se alone in the various calculation cases are recalled in the table below.

	Dose associated with a half-life of 65,000 years		Dose associated with a half-life of 1.1 million years	
Calculation case	Maximum dose [mSv/year]	Date of maximum [years]	Maximum dose [mSv/year]	Date of maximum [years]
SEN	$3{,}2.10^{-6}$	800 000	$2{,}2.10^{-6}$	1 000 000
SEN – study of sensitivity to geochemistry	$3{,}3.10^{-3}$	190 000	$1{,}1.10^{-3}$	300 000
Borehole scenario – reference calculation	$9{,}4.10^{-6}$	100 000	$1{,}1.10^{-6}$	1 000 000
Severely degraded evolution scenario	$2{,}7.10^{-2}$	170 000	$8{,}8.10^{-3}$	300 000

Comparison of doses and dates of maximum doses due to selenium with a half-life of 65,000 years (JEFF 2.2) and 1.1 million years (JEFF 3.1) at the Saulx outlet of the Oxfordian (most pessimistic case) – 1 million year model – C1/C2 waste

8 Conclusions



Block Diagram 8-1 Representation of the connections between the different stages of the analyis (see Block Diagram 1-1) Subject : Conclusions

The conclusions of the « safety assessment » document have a double aim. Firstly this involves summarising and reviewing the results of the various stages of the analysis, so as to provide an overall assessment of the safety of the proposed repository design. This assessment is supported by the safety analyses in the operation – observation stage (chapter 4) and after closure (chapter 5 and following). It is based not only on the calculations of performance (chapters 5 and 7) but also on the more qualitative analyses, dealing with the manner in which the concept is defined to fulfil the safety objectives (chapter 3) and also on the analysis of its robustness in the face of uncertainties (chapter 6).

The second objective is to draw conclusions for the later stages of the program. On this latter point, it is important to emphasize that the conclusions do not intend to prejudge the decisions that could be taken at the outcome of the parliamentary debate which must take place in 2006. They sketch out the perspectives for a future research program, by identifying some important aspects for safety.

8.1 The lessons of the safety analysis

The feasibility of the repository seems to have been determined with a reasonable degree of confidence. In particular, the safety assessment shows that the radiological protection objectives assigned to the repository are complied with. Moreover, the repository's safety relies on a multi-functional concept, involving the complementary properties of the site, the design provisions (not only the addition of engineered components but also the general organisation of the repository, divided into sections and in a dead-end layout) and the primary package. This gives the repository a good robustness. There are few altered evolution situations that emerge from the analysis and they refer to a limited number of well-identified scenarios. These do not lead to a significantly higher radiological impact than that of the normal evolution scenario.

8.1.1 The host formation

The host formation has been shown to be an essential component of the system, to the extent that it contributes to all of the functions that have been defined :

- To the function « prevent water circulation «, because of its low permeability. This determines not only the predominance of the diffusive regime in a normal situation but proves to be an essential parameter in case of an altered situation, such as a defective seal or a bore-hole. It limits the inputs of water in a circulation short-circuiting the rock, whether these are the access routes to the repository or a route created by human intrusion or an undetected heterogeneity. In addition, the proposed repository's design, in dead-end topology, helps to mobilise these favourable properties.
- To the function « limit the release of radionuclides and immobilise them in the repository », by imposing favourable physico-chemical conditions. The rock helps to maintain a pH close to neutrality and reducing conditions that are suitable for the control of releases, especially for vitrified wastes. The solubility of many elements is low under such conditions. Its low porosity also helps limit the risks of colloidal transport.
- To the function « delay and attenuate the migration of radionuclides »: the low diffusion rates, and the high sorption capacities of the argillites, are the main determinants for the performance of this function. Provided the access structures to the repository do not constitute a favoured circulation route for the radionuclides, the host rock controls the great majority of these flows. In normal evolution situations it totally delays and attenuates all the inventory of radionuclides, with the sole exception of four radionuclides.

The rock's properties are now well-known because of a major characterisation program in bore-holes and in the laboratory ; the main properties do not have an uncertainty margin of more than a factor of 10. The experimental data acquired within the drifts of the Meuse/Haute-Marne laboratory, after completing the performance calculations, reinforce and supplement the data acquired on samples. The experiments should continue over longer periods to obtain further data ; at this stage, they show good consistency with the values considered for Dossier 2005.

The conclusions of the safety analysis are relatively insensitive to the residual uncertainties. Including, in the « sealing failure » altered evolution scenario stronger gradients than those measured, a significantly higher permeability in the « very degraded operation » altered evolution scenario and the creation of a short-circuit in the « bore-hole » altered evolution scenario, in no case led to an unacceptable impact.

It is also important to limit the perturbations suffered by the host formation due to the exogenous materials placed within it. They have formed the subject of a survey covering all of the thermal, hydraulic, mechanical and chemical processes. On the basis of current knowledge, their potential extent and their effects on the properties of the argillites have been identified. The repository performance assessment can, in certain cases, take explicitly account of the effect of these perturbations, or ignore them provided that a preliminary analysis has shown that it was possible to proceed in this way. Design arrangements have been proposed to manage such perturbations. They are not likely to undermine the performance of the repository system

All provisions have been taken in the dossier to minimise the role of the surrounding formations. These do not have a safety function, and only constitute transfer paths for the radionuclides. Furthermore, the choice of models systematically tends to minimise the importance of the transfer times and conditions in these formations : choice of outlet situated near to the site and upstream of natural outlets, adoption of a model leading to « inflate » the mass of radionuclides coming out of the repository toward the outlet. Despite these conservative choices, the impact calculation does not lead to significant doses.

8.1.2 The seals and engineered barriers

The Dossier 2001 had identified the problem of the damaged zone and the operation of the seals as a subject whose importance with respect to safety should be the topic of a significant effort for the Dossier 2005. All of the works accomplished, and especially the more precise definition of the design of these structures (development of hydraulic cutoffs, more detailed definition of the installation conditions) now give good confidence in the ability to obtain the required performances from these components. The safety assessment was cautiously conducted, and only adopted reduced performances for these components. The outlooks derived from engineering studies make it likely that the properties will be much better.

The assessment of the normal evolution scenario showed that the seals effectively reconstitute the hydraulic continuity of the environment, the access structures not being favoured transfer paths. In the hypothetical situations of human intrusion, the division of the repository into separate sections by seals helps prevent the hydraulic influence of a bore-hole from propagating.

The failure of the seals does not however appear to be a very damaging event for the repository's safety. The lengths of the drifts, the low gradients, the low quantity of water provided by the geological environment, the dead-end design and the role of the « absorbent barrier » fulfilled by the rock along the path of the radionuclides in the structures, all help to limit the impact of such a situation. However important the sealing devices are, they do not on their own determine the repository's safety.

The clay engineered barrier has been implemented in the spent fuel cells so as to mitigate over the thermal effect during several thousands years. It does not play a significant role as regards radionuclides transfers, in any of the scenarios considered.

8.1.3 Packaging elements

The over-packing of B waste is defined to be as durable as possible, not only with the aim of repository reversibility but also with the aim of ensuring long-term safety.

Analysis show them to be interesting devices in the qualitative plan : the presence of hydraulic transients in the first thousand years after the repository is closed makes safety assessment more complex, and it could eventually prove useful to have a barrier guaranteeing that the radionuclides are confined during this period. Over-packing also implies protection close to the primary waste, as meant by recommendations of the basic safety rule RFS III.2.f. On the other hand, quantitatively the package durability has no visible benefit due to long transfer time in geological medium.

Metallic containers are defined vis-à-vis the thermal phase. Their failure does not seem crucial vis-à-vis impact, since the host formation guards the good properties even in degraded conditions. This appreciation however rests on the assessment of transfer properties in temperature related Argillite, a research that needs to be followed up. However, container is a precautionary measure against uncertainties.

In case of vitrified waste, the container also enable significant reduction in release, which are slower in moderate temperature conditions. This effect is adequately large to be « visible » on the characteristic duration scale of safety calculations.

8.1.4 Primary waste

The waste matrices mostly present the confinement capabilities, of which some are operated through performance calculations and other are neglected as a precaution. Either as a reference or in sensitivity studies, Andra has retained conservative models, often providing large safety margins with respect to those that originate from the most recent research.

For B wastes, only metallic waste offers an adequate life span so that its performance can be included in the calculations. For other wastes, as actually known, the margin is more qualitative. In the most pessimistic situation, the integrally labile waste will however not induce an impact much higher than in normal situation.

For vitrified waste, life span from models (some hundreds of thousands of years) are of a magnitude equivalent to the transfer time in geological barrier. These matrices therefore fulfil a complementary function of the geological medium. This observation validates the interest in taking measures to favour the slow release through glasses (to guarantee a diffusive regime in disposal cells, distance the disturbance sources of pH).

For spent fuel, models used in performance calculations (retaining radiolytic dissolution and increasing the effects of D3AI) lead to higher releases than for other types of waste and due to this have a greater impact by one order of magnitude, while still within the radiological protection constraint limits.

8.2 Possible lines of progress

Without pre-empting the decisions which will be taken in 2006 regarding research work on the deep geological repository, it is possible to draw a few lines of progress from the conclusions of the safety analysis with a view to possible future work. These focus on several areas : consolidation of the data acquired within the Meuse/Haute-Marne laboratory, full-scale technological tests to support more detailed engineering studies, work to explore the transposition zone on a larger scale and a more precise quantification of the safety margins through more thorough knowledge of the phenomenology. On this final point, it is important to stress that the representation of the processes and their inclusion in the safety assessment of Dossier 2005 involves simplified, conservative models in certain cases. It would be important in a later phase to represent them in a more precise manner in order to increase the confidence that can be placed in the assessments. It is in this particular area that the lines of work mentioned in this section are being concentrated. They can only be temporary and partial. On the one hand, the construction of a work programme for the years post-2005 depends on decisions from the public authority ; on the other, it depends in part on the result of the assessment of the dossier (see section 1.5.5) and the recommendations arising from it.

Pursuit of the geological medium characterisation may allow one to define more precisely the conceptual model at the transposition zone scale, based on the directions already stated at the end of chapter 6. Especially, additional investigations will allow establishing the flow directions in the Dogger in greater detail and specifying the role of regional faults and their environment in the Hydrogeological model. This will enable us to review the impact calculation conditions of the repository and some conservative choices, especially the outlets retained as of date in the upstream of the « diffuse fracturing zone ».

It could be possible to continue providing further details on the performance of the Callovo-Oxfordian (permeability, diffusion, retention, etc.) from the acquisition of data on the formation in situ in the Meuse / Haute-Marne laboratory

Following a more detailed study of the properties that the horizons at the bottom of the Oxfordian present and especially C3a and possibly the Kimmeridgian, could call for assigning retention properties to them that offer additional margin, and allow if deemed pertinent, to attribute a safety function to them that complete the performances of the only Callovo-Oxfordian.

Dossier 2005 also marks progress compared with the previous dossiers produced by Andra in that, for the first time, it explicitly envisages the influence of climate changes on the hydrogeological model and on the biosphere. A finer appreciation of climate sequencing could result in greater detail being provided for these assessments. It must however be emphasised that any effort in this area must be set against the uncertainties weighing on the evolution of the surface environment, encouraging the adoption of very robust and partly stylised approaches

Characterisation of the transport properties of the excavation damaged zone, immediately after sinking, then their evolution under effect of mechanical or even thermo-mechanical constraints in the concerned disposal cells, is an important subject for which the underground laboratory has already started and will continue to contribute important information. Today the EDZ assessment is conducted by modelling ; the data obtained during experiments will enable specifying the mechanical behaviour of the rock with the aim of optimising the concepts.

Modelling of transient phases also requires pursuing the works for representation of the coupled phenomena. The Dossier 2005 has already built on the transport-chemistry coupled calculations that allowed specifying the phenomena extension. Detailed understanding of the earliest phases of life takes place through the pursuit of modelling work on couplings, including those induced by heat (thermo-mechanical behaviour of EDZ, pursuit of studies on the heat-transport coupling). Representation of coupling due to hydraulic transients –particularly models in an unsaturated medium - will also enable refining the control and understanding of the initial centuries of the repository's evolution. with particular attention paid to controlling the conditions in which the materials change over time in the repository.

The continuation of studies into the conditions under which corrosion develops within the repository should enable the conceivable speed ranges to be reduced by approaches which are both theoretical (for example, coupling with modelling in an unsaturated medium) and experimental (with possible experiments in situ on metallic materials). It is possible that we could therefore revise the corrosion gas pressure build-up assessments downwards and, through this, the influence of the gases on the hydraulic transient. Furthermore, by studying the various hydrogen migration pathways, it will be possible to provide further detail for the overall evolution diagram, based here too on modelling and a more experimental approach.

Finally, it will be noted that, as a result of the qualitative safety analysis, it has been possible to draw up an initial list of processes, the implementation of which during the operating phase could restrict the duration of this phase from the point of view of long-term safety. Reversibility appears possible over a few centuries (typically two or three hundred years) or potentially longer periods. The design approach adopted by Andra, privileging joint, homogeneous treatment of the questions of safety and reversibility, leads to architectures in which these two notions do not appear to compete with each other. The same approach could be continued in the future.

From the point of view of waste behaviour and release models, we have already had the opportunity to see the behaviour of spent fuel

If it is decided to continue investigating the possibility of managing in repository all or part of the spent fuel of the water pressure reactor park, pursuit of efforts is required to better characterise the long-term behaviour and profit from important benefits that a release dominated by standard dissolution and not by radiolytic dissolution will bring.. The expected gains are highlighted by the analysis in chapter 5 : the spent fuel could have a release level comparable to that of other waste types.

For vitrified waste, the change from a dissolution model only driven by the initial rate to a « $V_0.S \rightarrow Vr$ » type model brings appreciable and quantifiable gain in terms of reduction and spread in impact time. Finer understanding of the underlying mechanisms, and possible extension of the validity domain of this model with larger number of waste type or in greater physico-chemical conditions, will bring a greater sturdiness to the safety analysis.

8.3 General conclusion

The safety assessments are not an autonomous domain of the repository feasibility study. They form an inseparable set with engineering and research studies on the phenomenological evolution of the repository. Each of these three fields have a separate document in the Dossier 2005. The documentary structure, in which division by discipline is mandatory, should not for all this mask the close links that unite these three approach modes. We have especially seen that some passages of this document were common with the « architecture and management of a reversible repository » document or with the « phenomenological evolution of a repository » document. Other use the same information by presenting it in a different perspective. The « safety case » is constituted by the collection of three works.

For all this, we have tried to make the « safety assessment » document a coherent set. It can be read on its own and presents the safety approach process that guided the engineering studies, research programme, evaluation of performance and uncertainty management. This approach allows arriving at the conclusion of the first stage, that of the repository principle feasibility. The repository impact actually appears controlled and conforms to the objectives in spite of the cautious assessment conditions. It is sturdy against uncertainties.

Qualitative analysis lessons and safety calculations also allow to sketch the broad outlines of a future research programme for specifying the architecture and minimising the uncertainties that subsist, if it should be decided to go ahead with the deep repository works. The conclusions of this document, as well as those of all other works presented in the Dossier 2005, constitute a solid knowledge base on which an engineering study and research program post 2005 can be based, as part of the same iterative approach that was responsible for the production of this document.

Appendix of Curves

In order to present results more clearly, some curves included in the body of this volume are shown in thumbnail format to avoid breaking up the text too often.

This appendix presents each of these thumbnail images in a larger format to allow the reader to take a closer look at a given curve if he/she wishes.

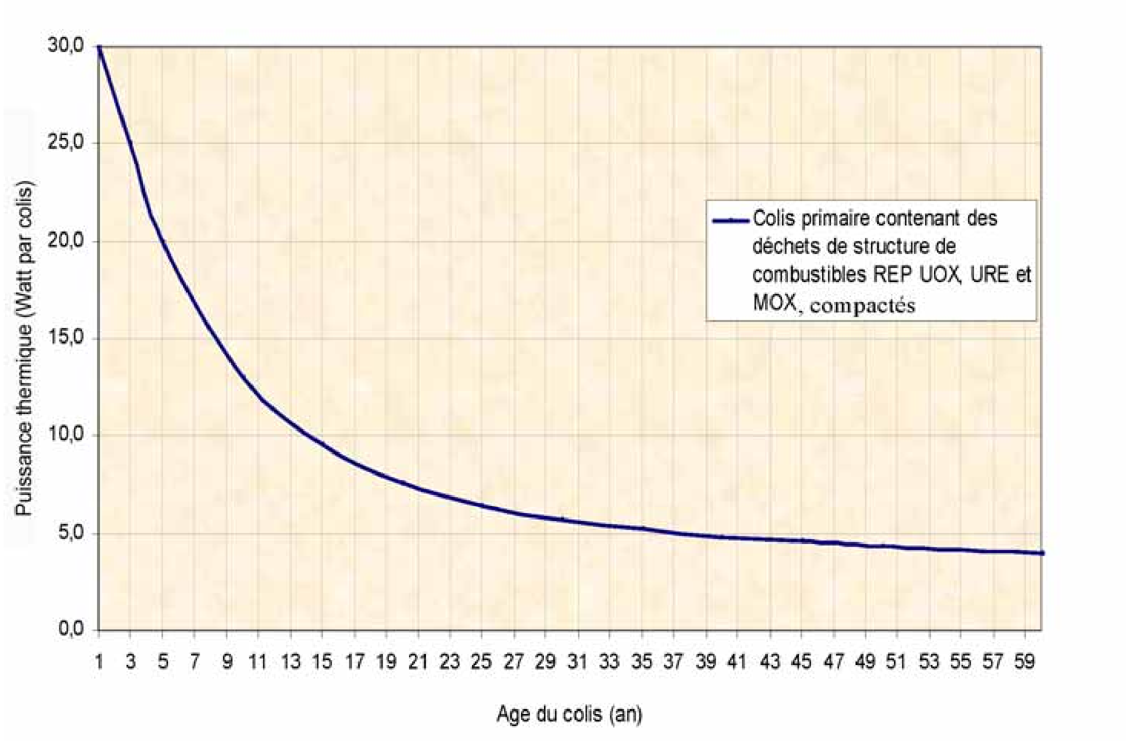


Figure 3.2-1 Thermal power of REP UOx, URE and MOX fuel structural waste packages

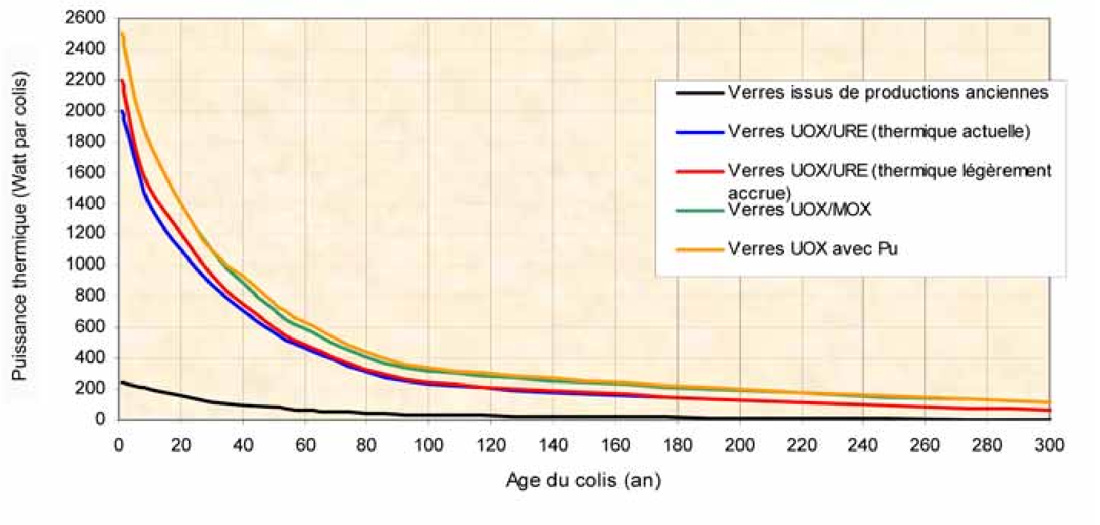


Figure 3.2-1 Thermal power of waste packages

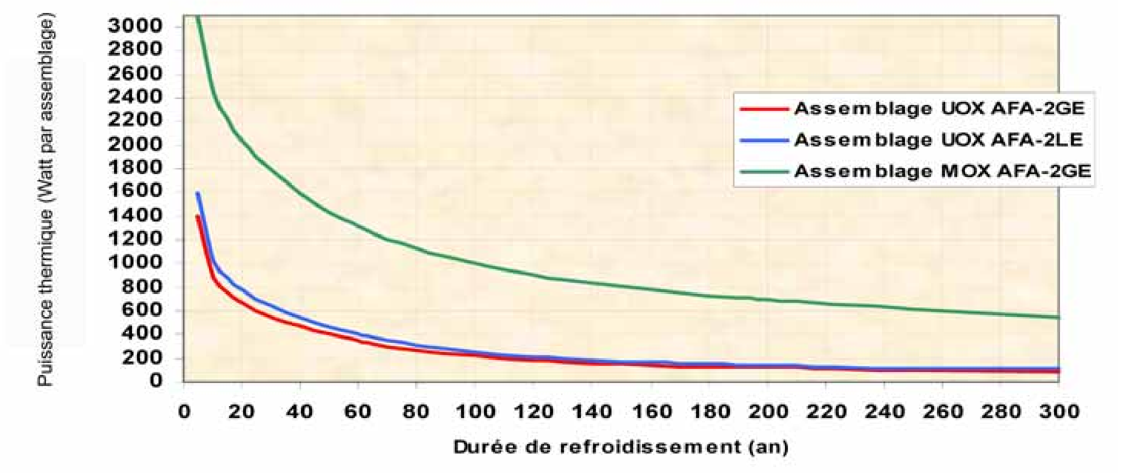


Figure 3.2-1 Thermal power of waste packages

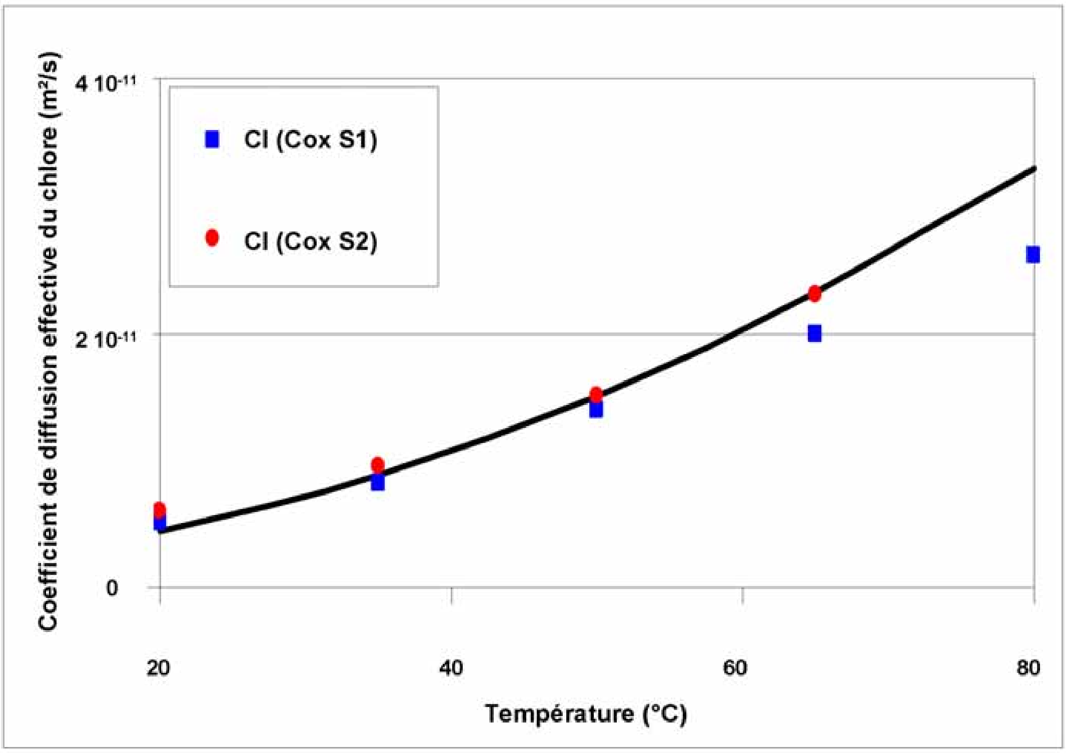


Figure 3.7-12 Experimental results for the evolution of the effective coefficient of diffusion for chlorides in function of temperature for two samples of argillite (Callovo-Oxfordian S1, Callovo-Oxfordian S2)

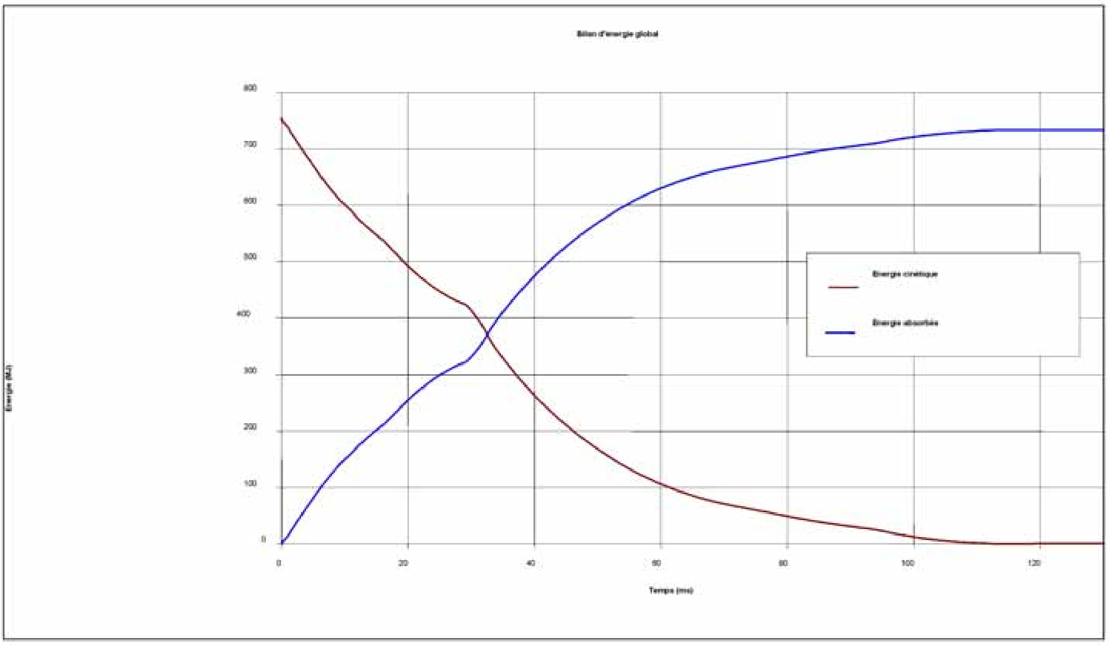


Figure 4.7-2 Energy balance of a fall in a shaft of a cage transporting a transfer cask with C wastes

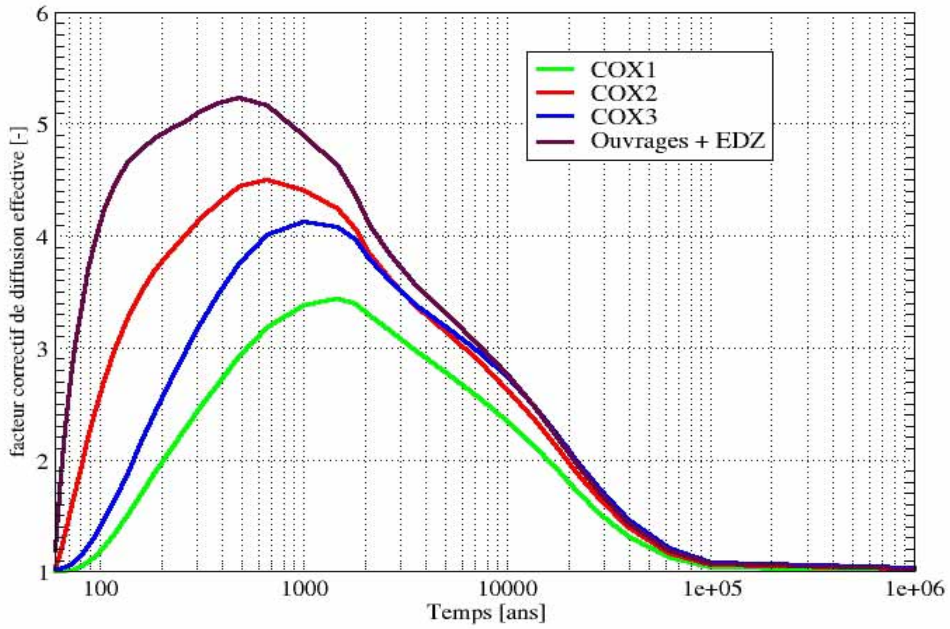


Figure 5.3-12 *Corrective factor applied to the effective diffusion coefficient at 20°C for taking into account the temperature – spent fuel CU1 (COX = Callovo-Oxfordian)*

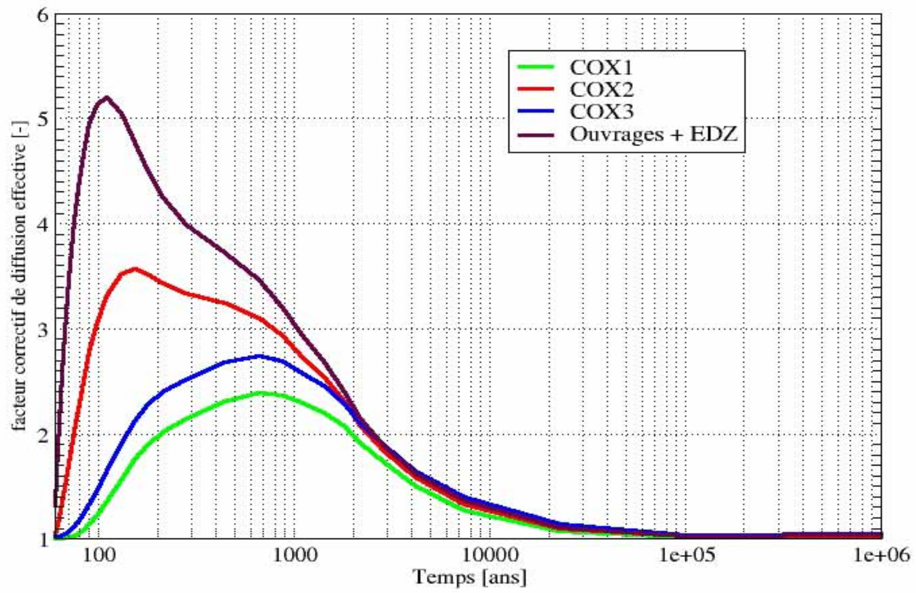


Figure 5.3-13 *Corrective factor applied to the effective diffusion coefficient at 20°C for taking into account the temperature - spent fuel CU2 and vitrified waste C - Reference packages C1 + C2*

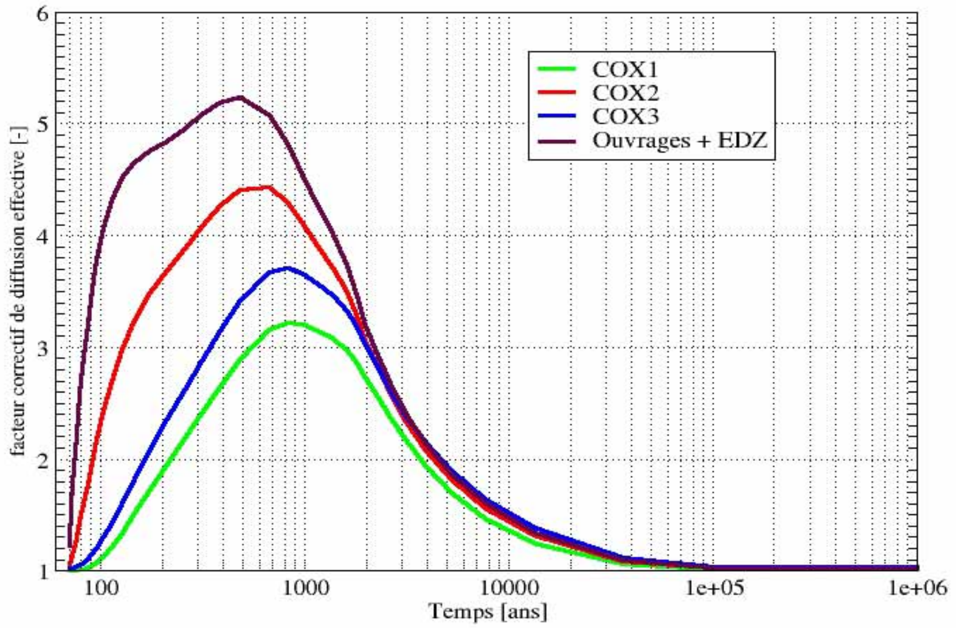


Figure 5.3-13 Corrective factor applied to the effective diffusion coefficient at 20°C for taking into account the temperature - spent fuel CU2 and vitrified waste C - Reference packages C3 + C4

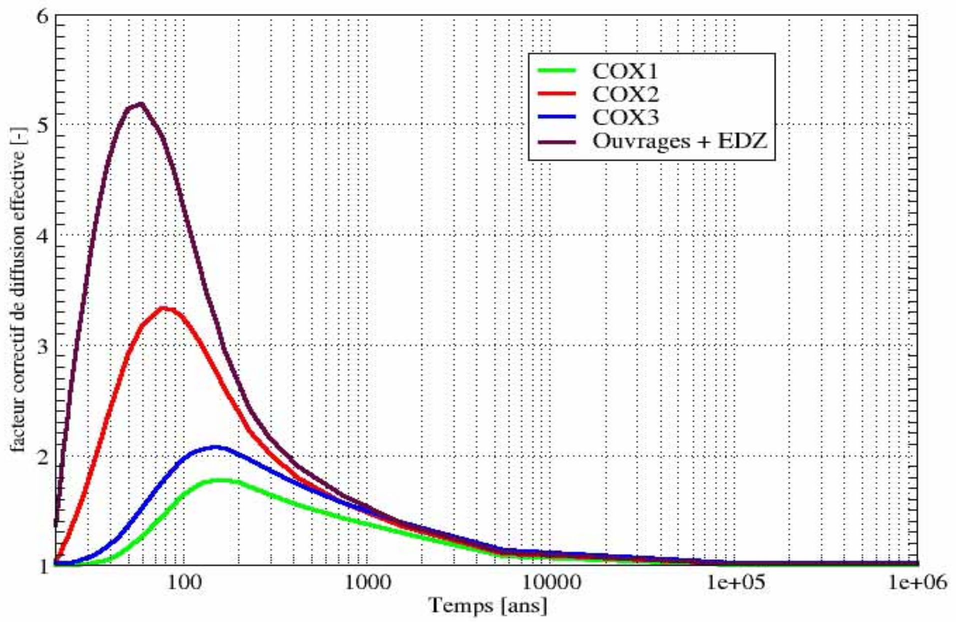


Figure 5.3-13 Corrective factor applied to the effective diffusion coefficient at 20°C for taking into account the temperature - spent fuel CU2 and vitrified waste C- Reference package C0

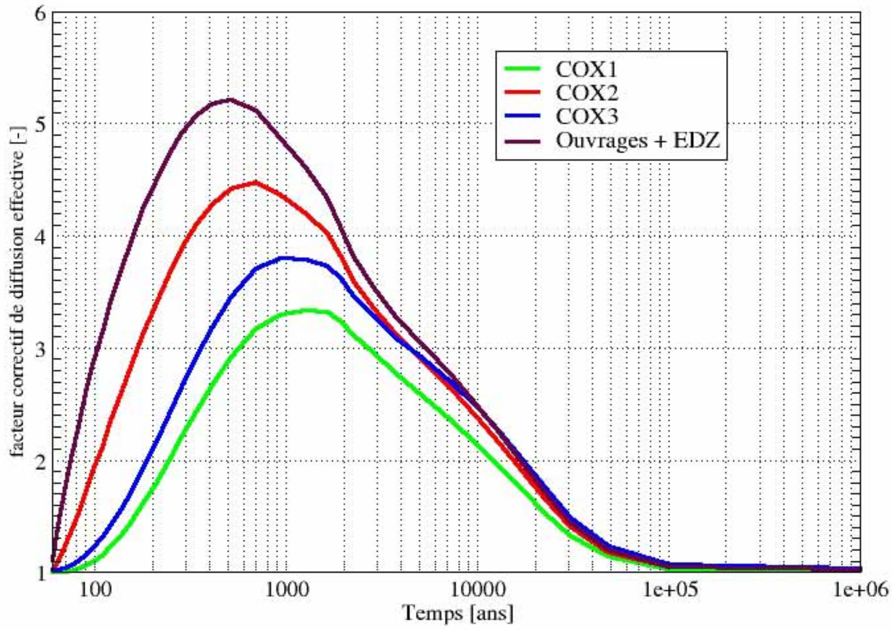


Figure 5.3-13 *Corrective factor applied to the effective diffusion coefficient at 20°C for taking into account the temperature - spent fuel CU2 and vitrified waste C - Reference package CU2*

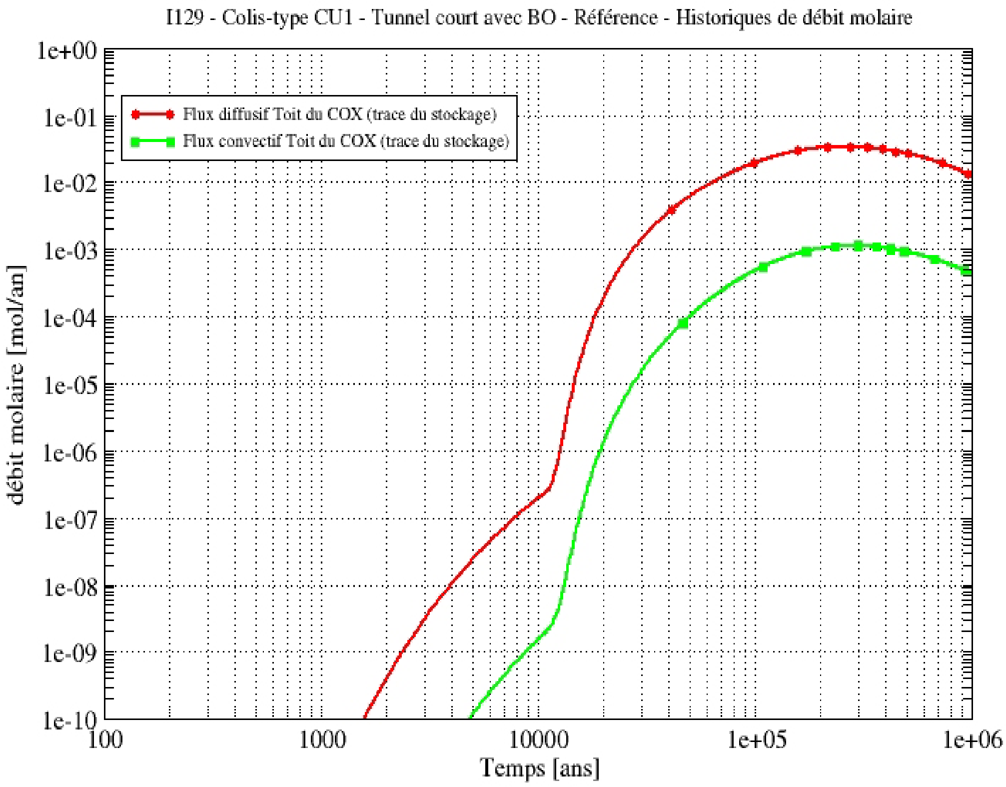


Figure 5.5-2 *SEN – History of the molar flow rate at the interface COX/overlying formation – example of iodine 129 of reference package CU1 (BO = swelling clay buffer, COX = Callovo-Oxfordian)*

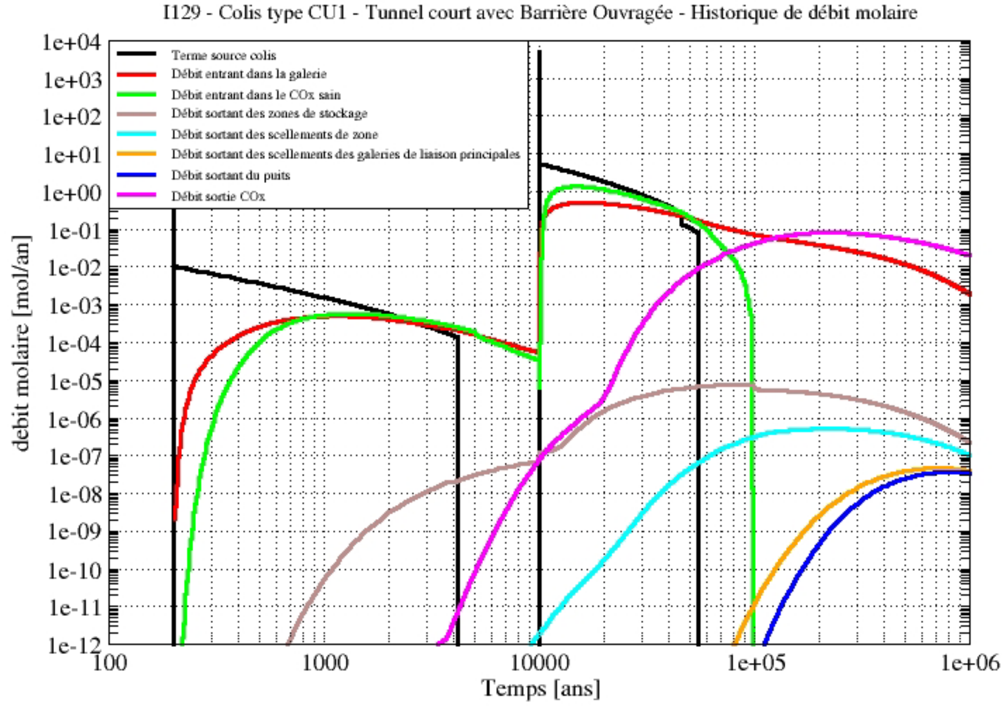


Figure 5.5-5 *SEN – Evolution of molar flows through the repository – ^{129}I – CU1 (Cox = Callovo-Oxfordian, COX exit : exit through the top and botton of the formation)*

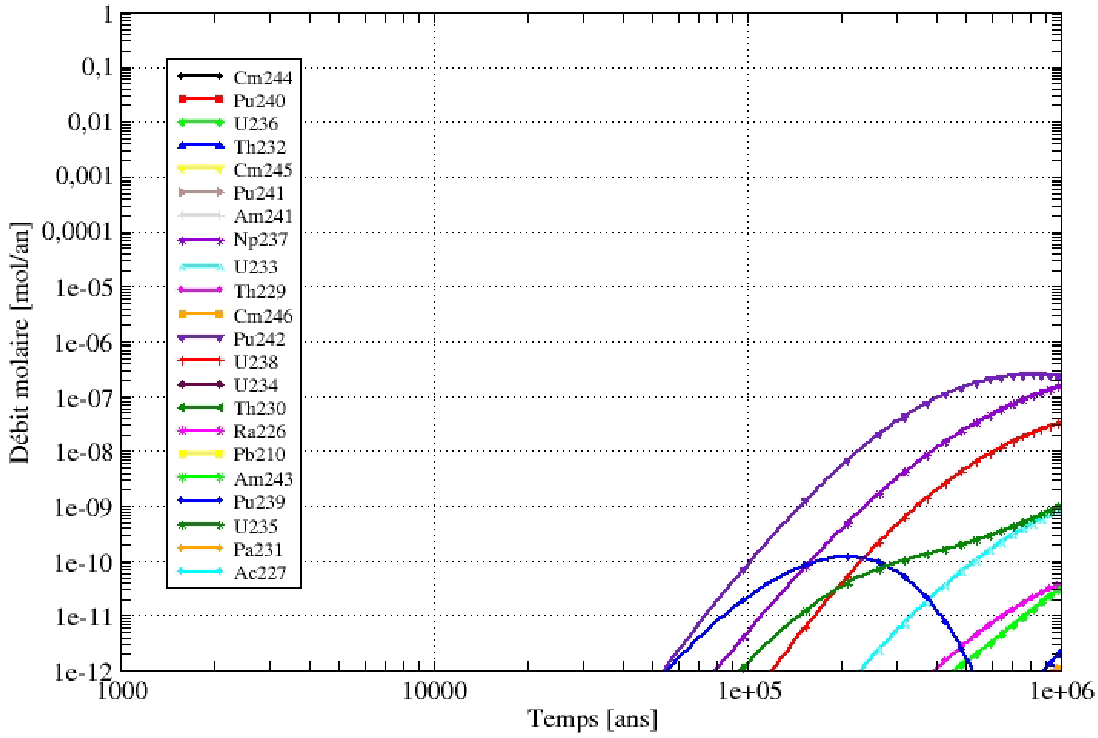


Figure 5.5-9 *Historical molar flow rates out of the repository at 7 metres from the cells – Reference package CU1 Uox3*

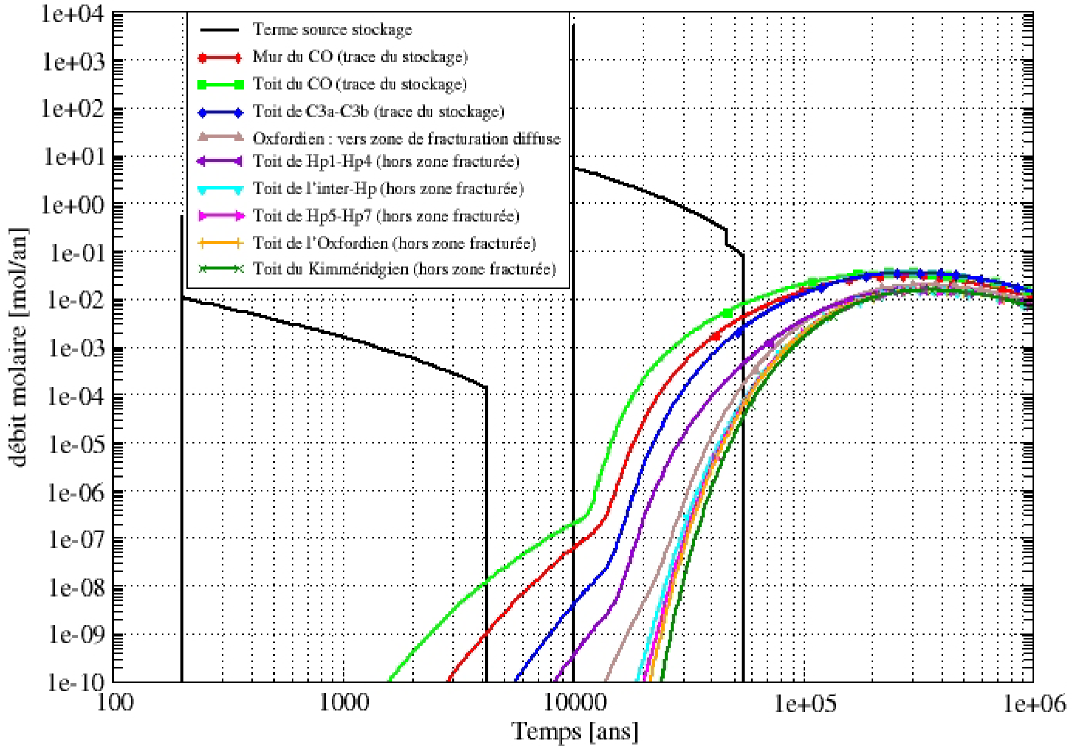


Figure 5.5-14 SEN - CU1 reference package – History of molar flows through different surfaces of the geological medium and quantification of the attenuation of different formations for the million-year model - ^{129}I

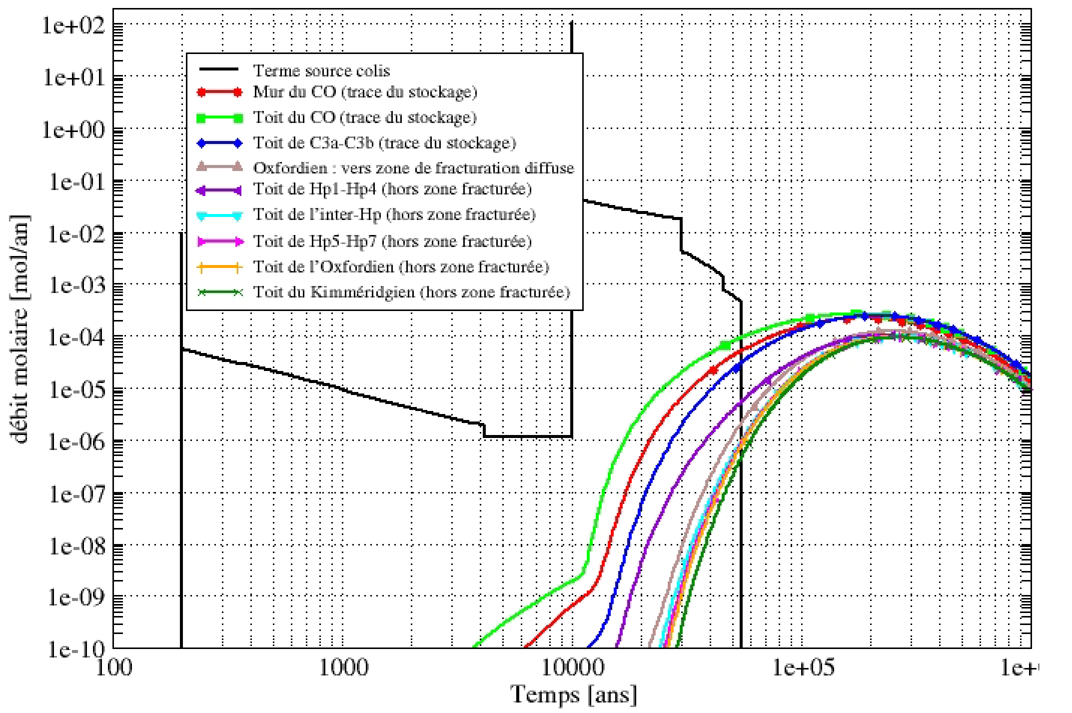


Figure 5.5-14 SEN - CU1 reference package – History of molar flows through different surfaces of the geological medium and quantification of the attenuation of different formations for the million-year model - ^{36}Cl

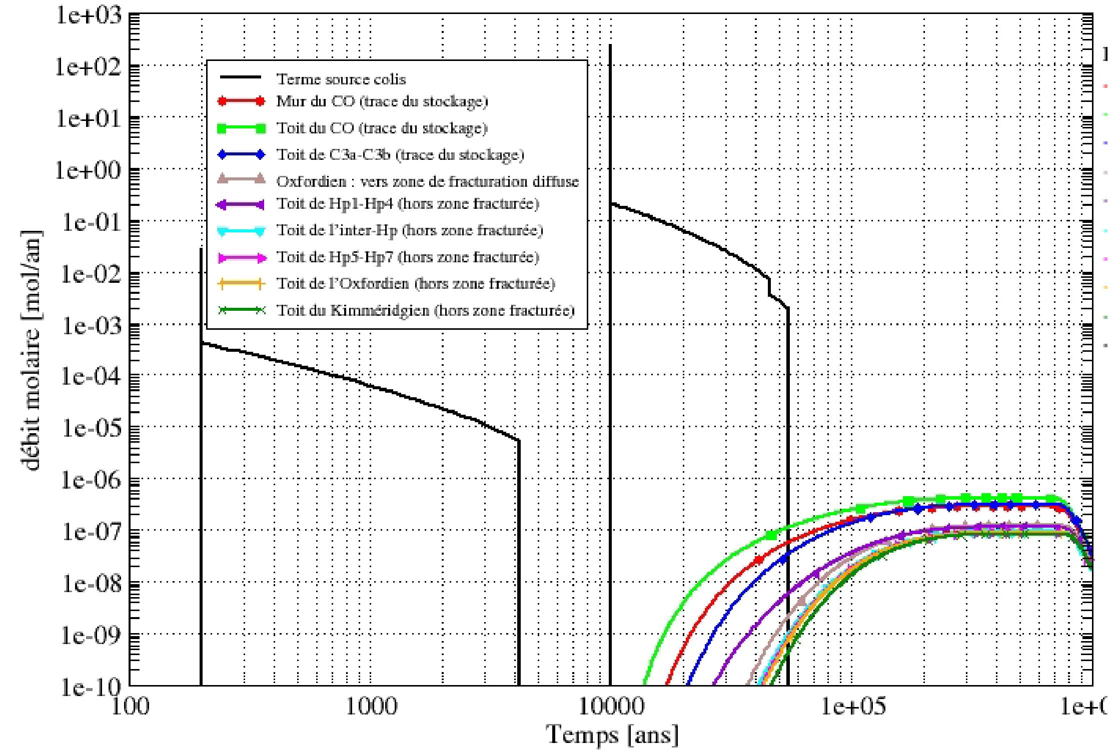


Figure 5.5-14 *SEN - CU1 reference package – History of molar flows through different surfaces of the geological medium and quantification of the attenuation of different formations for the million-year model -* ^{79}Se

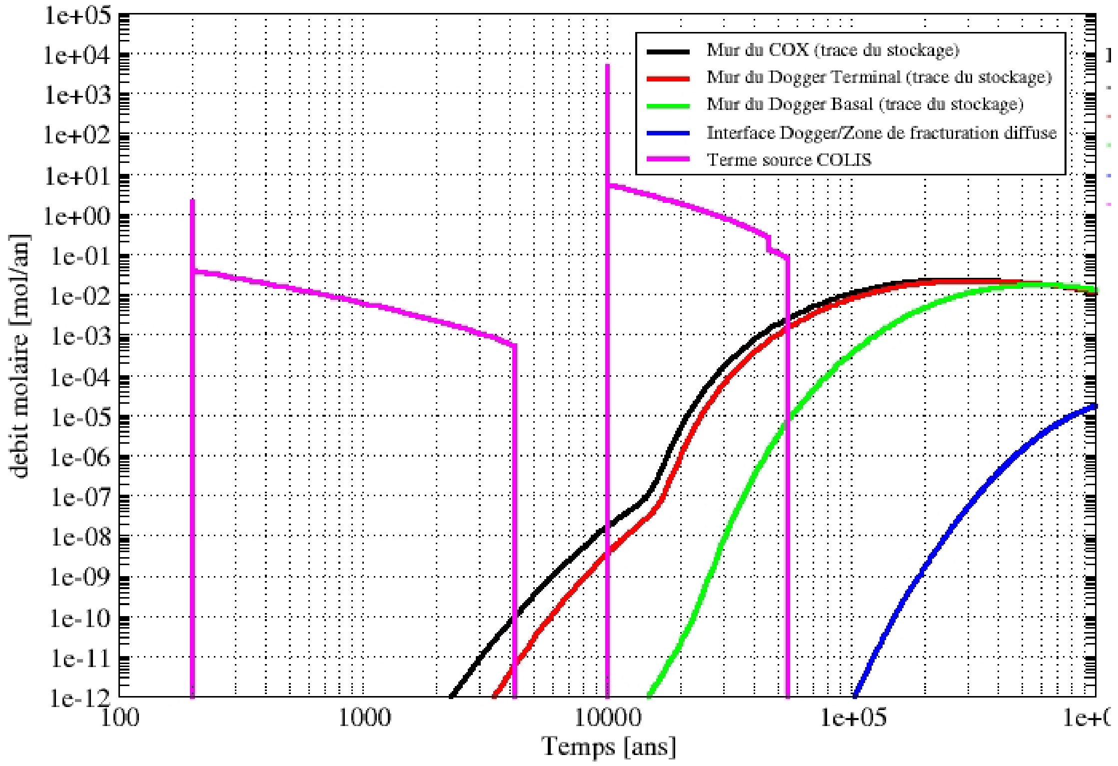


Figure 5.5-15 *SEN - Concentration plumes in different Dogger horizons (vertical section) at different dates and history of the molar flow through the different surfaces - Dogger model– CU1 reference package –* ^{129}I

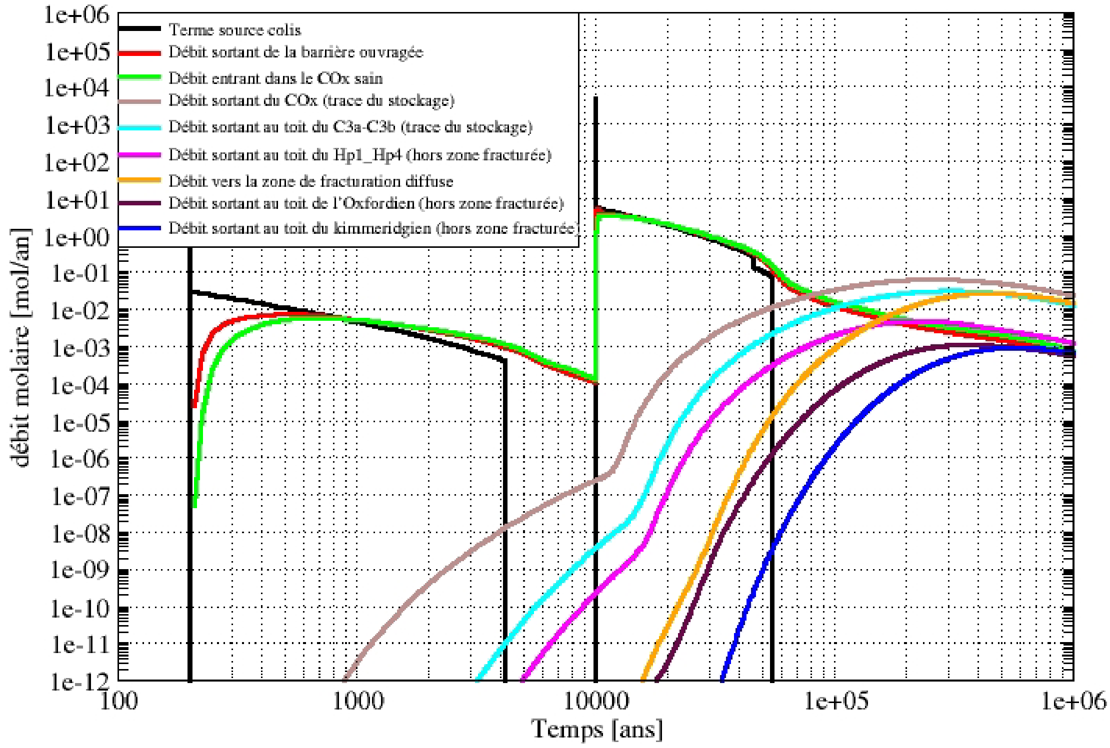


Figure 5.5-17 SEN – CU1 reference package - 129I – history of the molar flow in both configurations of the Hydrogeological model - Current model

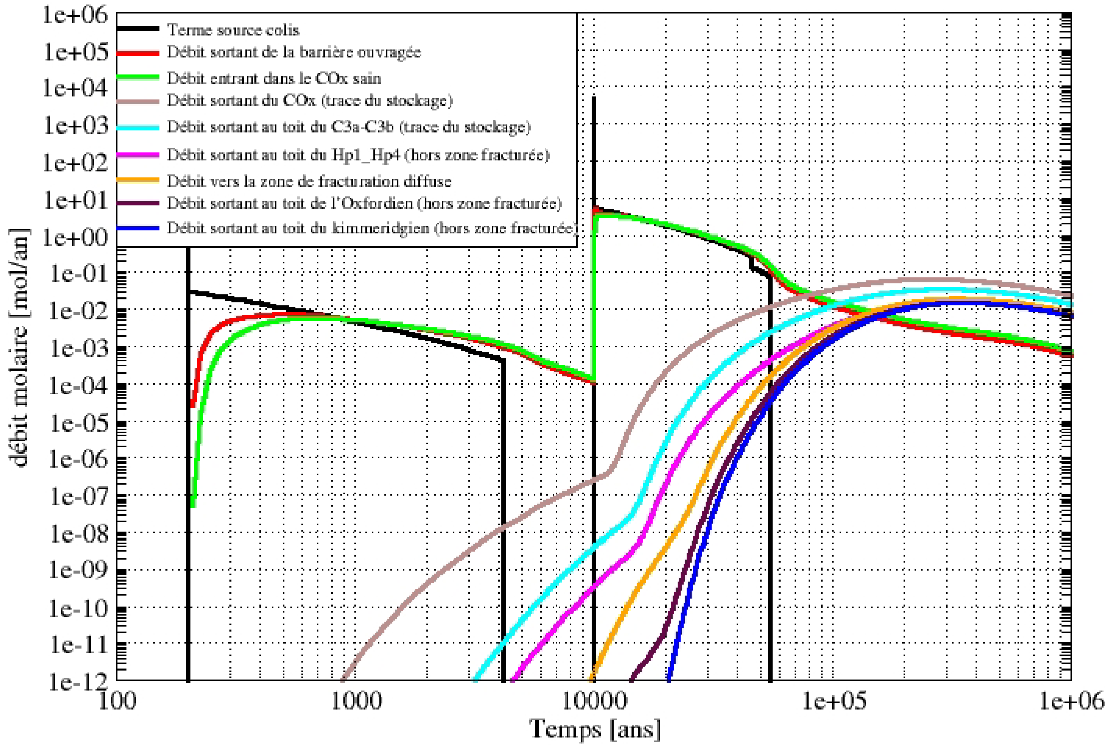


Figure 5.5-17 SEN – CU1 reference package - 129I – history of the molar flow in both configurations of the Hydrogeological model - 1 million-year model

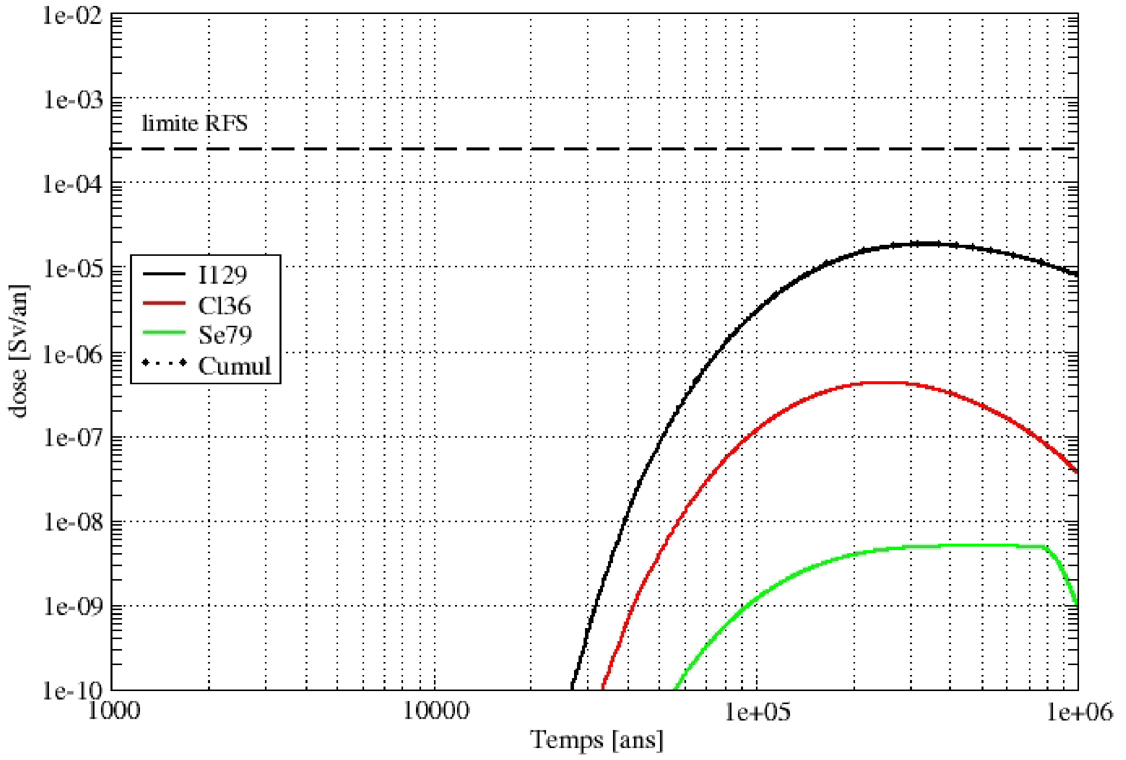


Figure 5.5-18 SEN – Reference calculation – million year model - Doses at the Saulx outlet of the Oxfordian – CU1 reference package

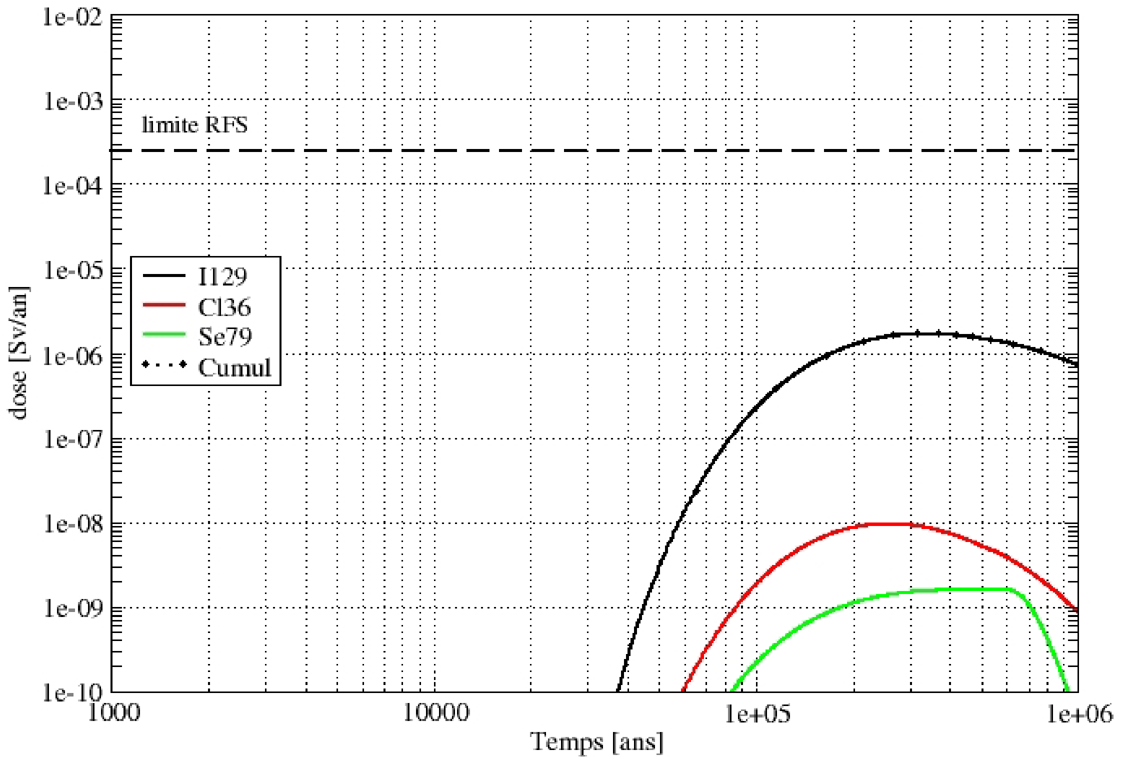


Figure 5.5-19 SEN – Reference calculation – million year model - Doses at the Saulx outlet of the Oxfordian – CU2 reference package

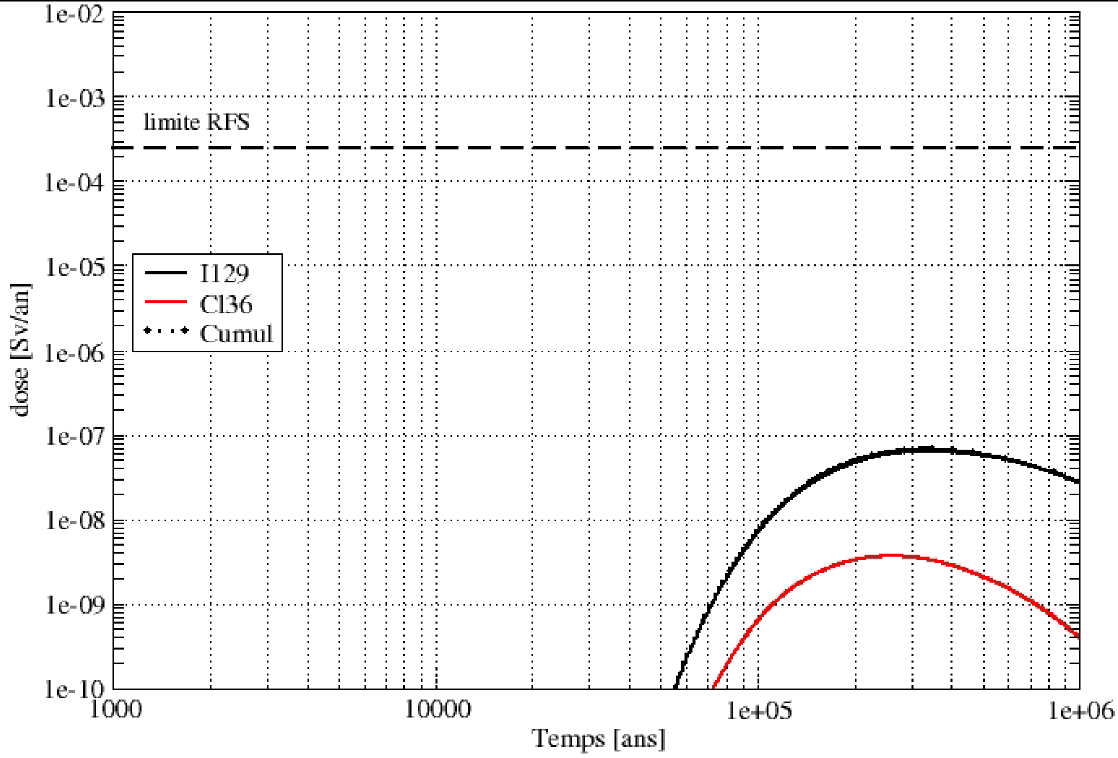


Figure 5.5-20 *SEN – Reference calculation – million year model - Doses at the Saulx outlet of the Oxfordian – CU3 reference package*

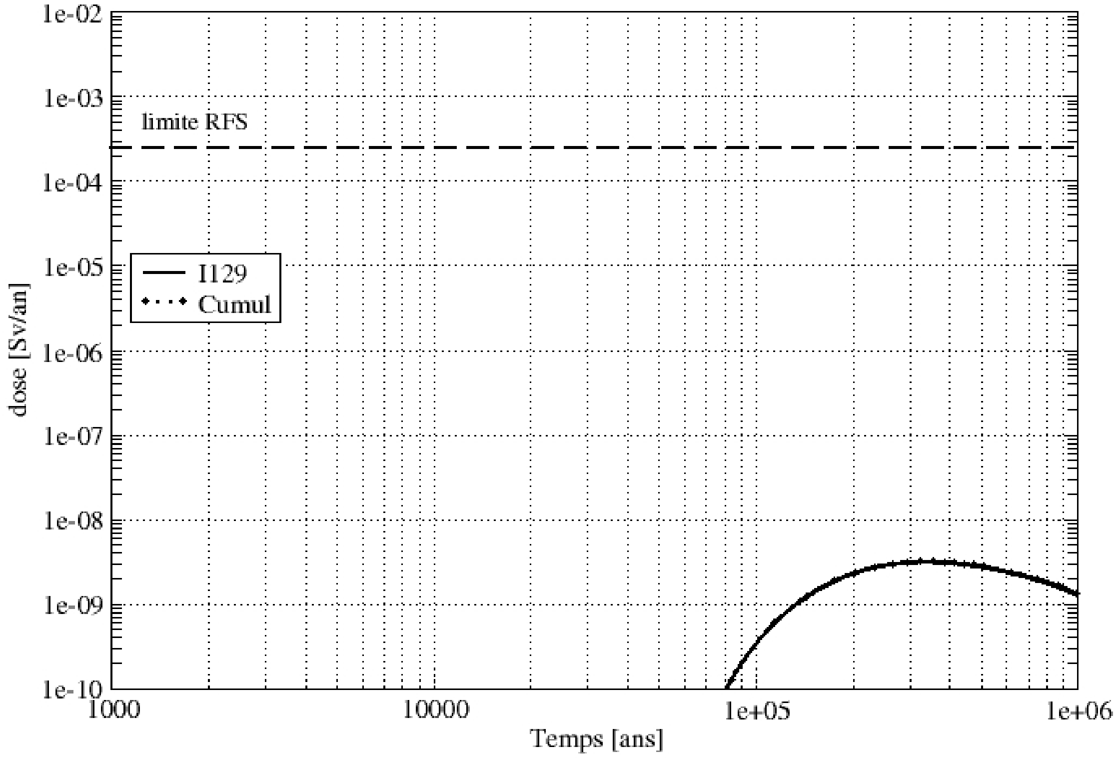


Figure 5.5-21 *SEN – million year model - Doses at the Saulx outlet of the Oxfordian – C0 reference packages*

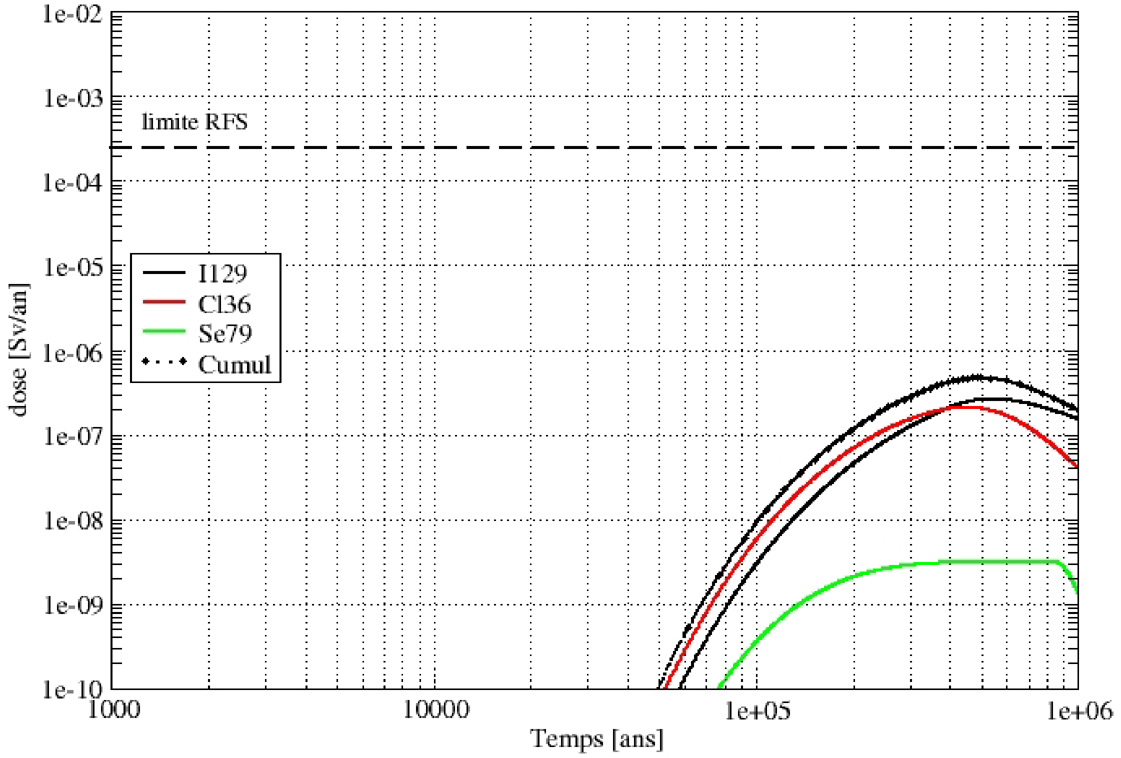


Figure 5.5-22 SEN – million year model - Doses at the Saulx outlet of the Oxfordian – Reference packages (C1+C2)

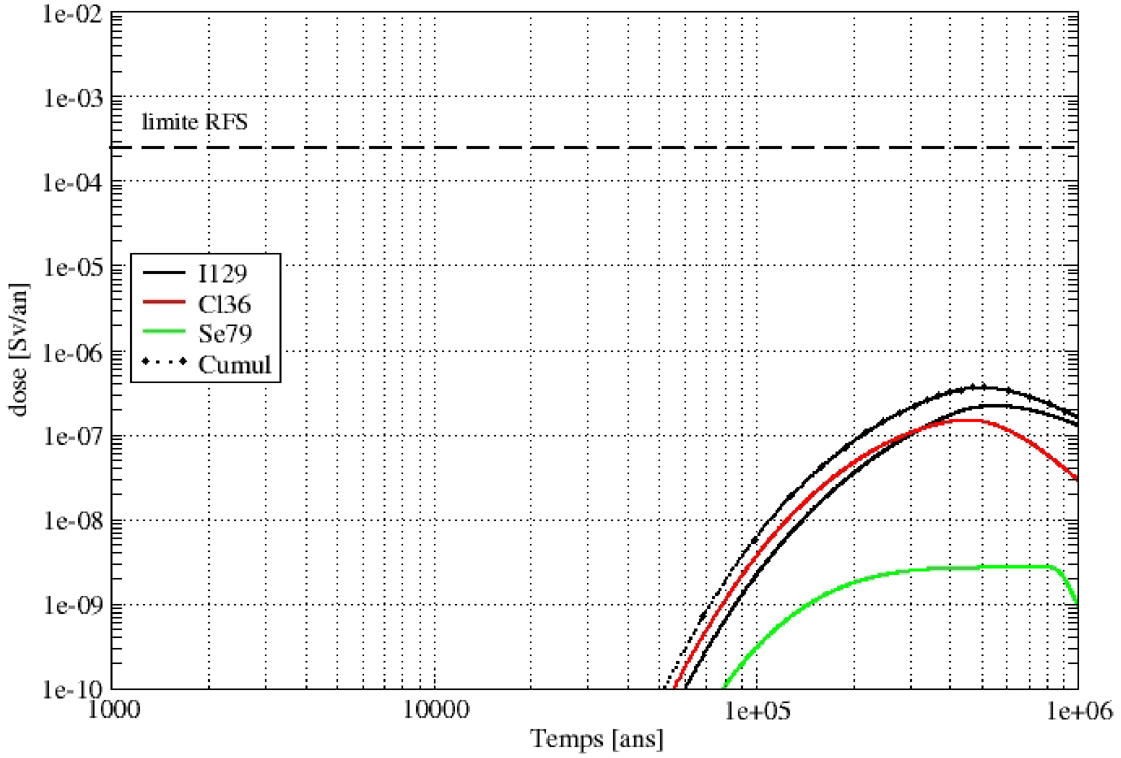


Figure 5.5-23 SEN – Reference calculation – million year model - Doses at the Saulx outlet of the Oxfordian – Reference packages (C3+C4)

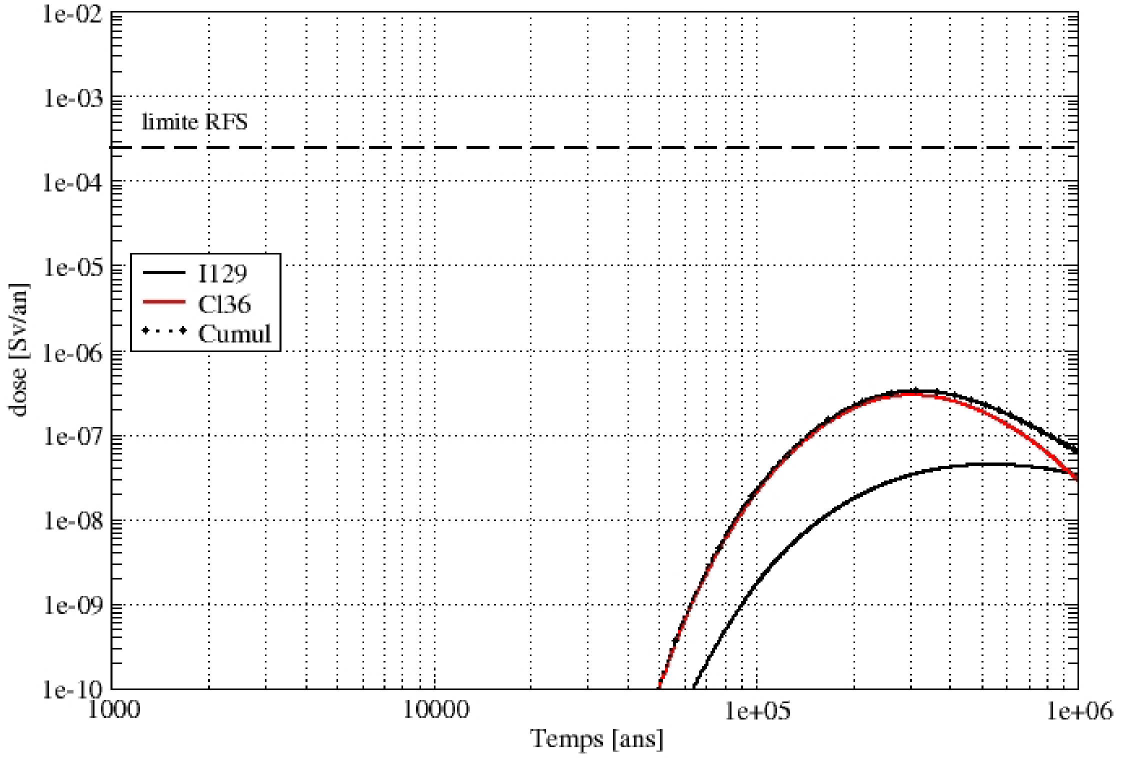


Figure 5.5-24 SEN – Reference calculation – million year model - Doses at the Saulx outlet of the Oxfordian – Reference packages (B1x)

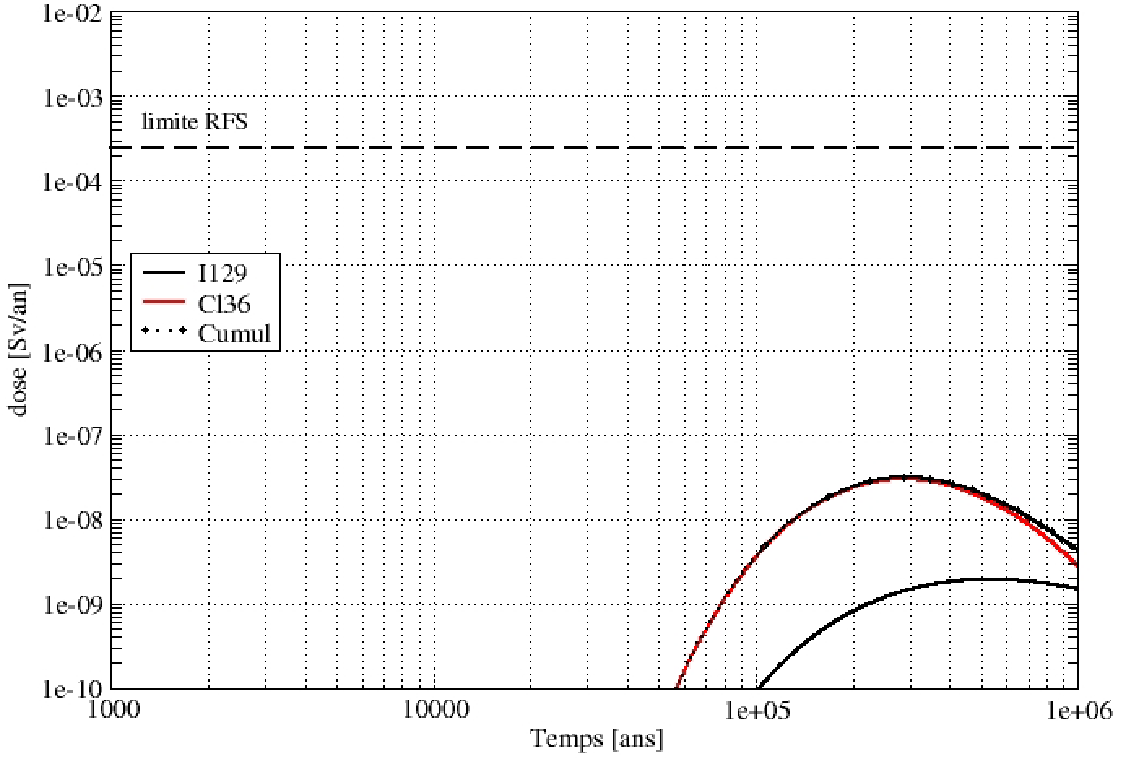


Figure 5.5-25 SEN – Reference calculation – million year model - Doses at the Saulx outlet of the Oxfordian – Reference packages (B1h)

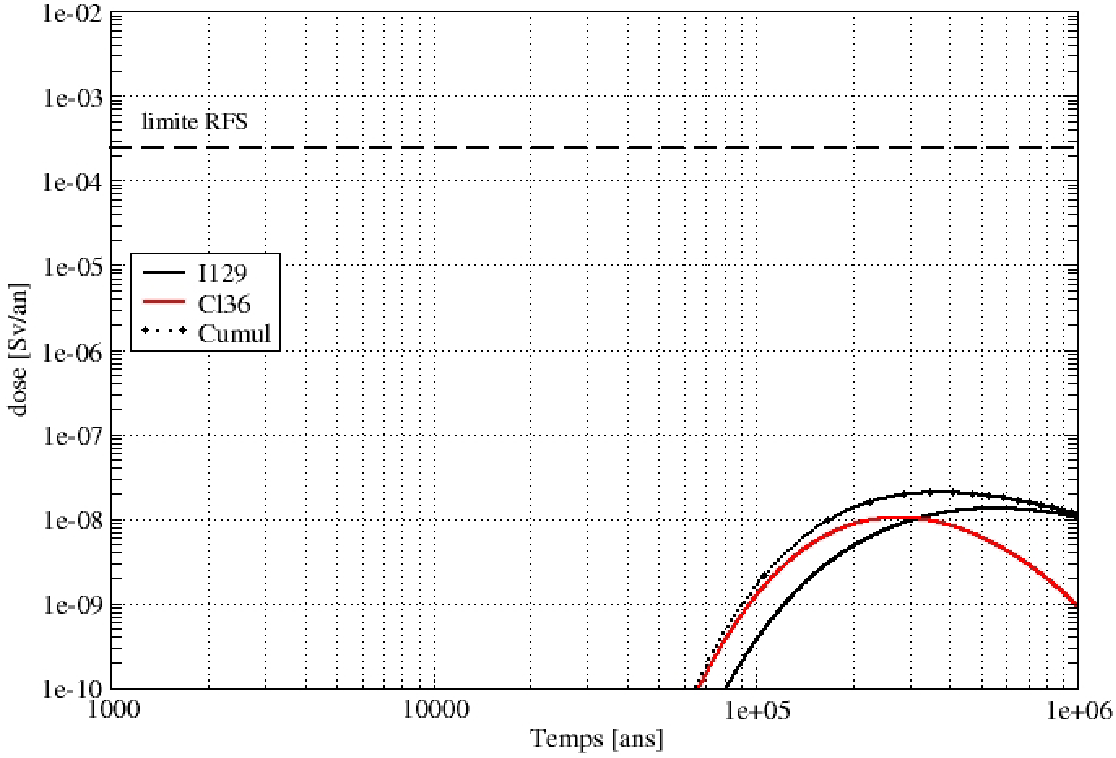


Figure 5.5-26 SEN – Reference calculation – million year model - Doses at the Saulx outlet of the Oxfordian – B2 Reference package

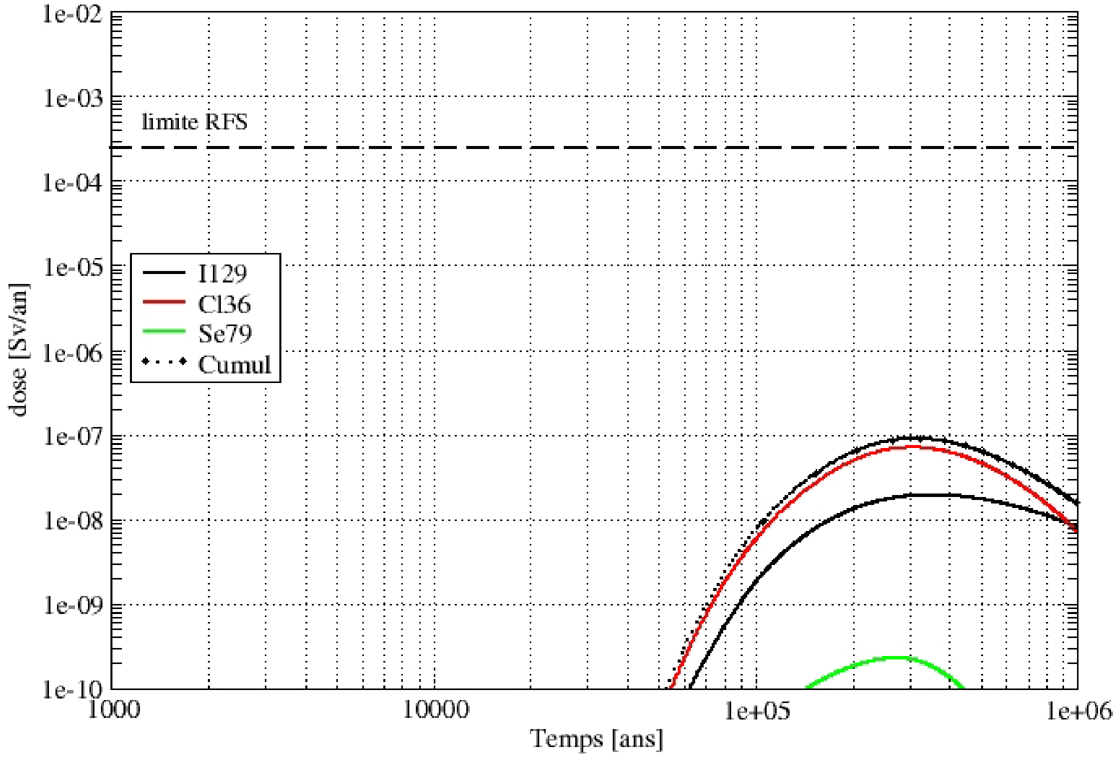


Figure 5.5-27 SEN – Reference calculation – million year model - Doses at 1 million years at the Saulx outlet of the Oxfordian – other B reference packages

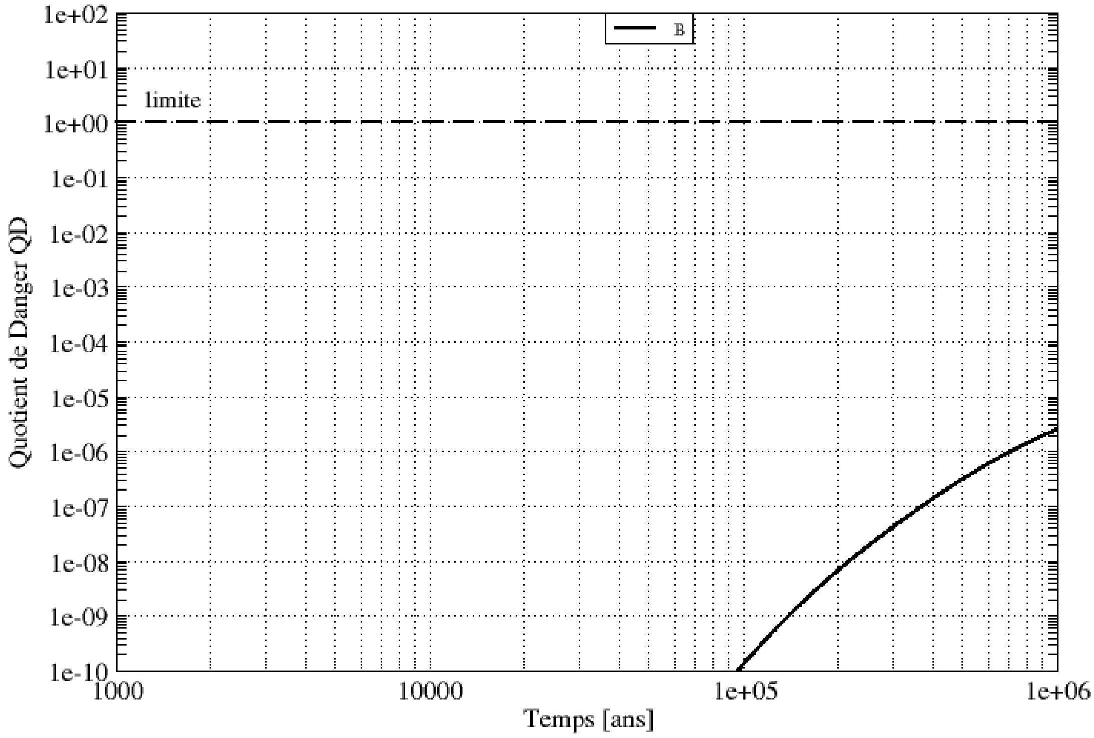


Figure 5.5-29 SEN – Reference calculation – model to 1 million years – Reference package C1/C2 – Boron – Chemical impact at Saulx outlet of Oxfordian linked to non-cancerigenous effects, for ingestion (QD)

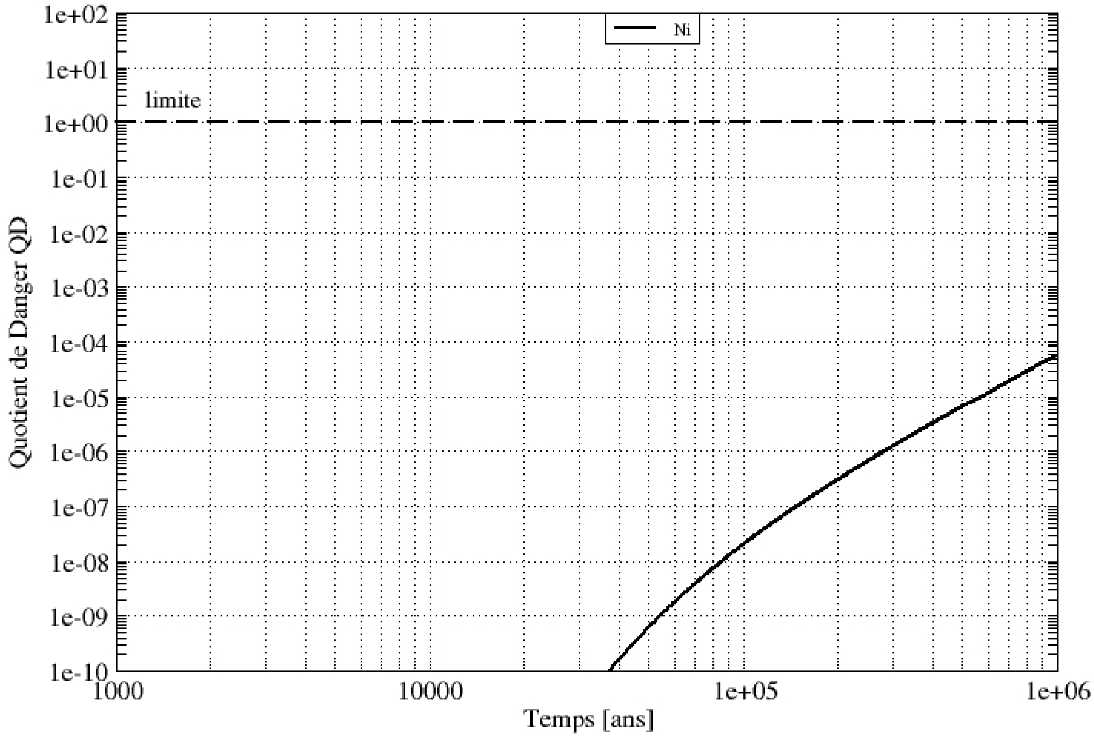


Figure 5.5-30 SEN – Reference calculation – model to 1 million years – Reference package C1/C2 Nickel – Chemical impact at Saulx outlet of Oxfordian linked to non-cancerigenous effects, for ingestion (QD)

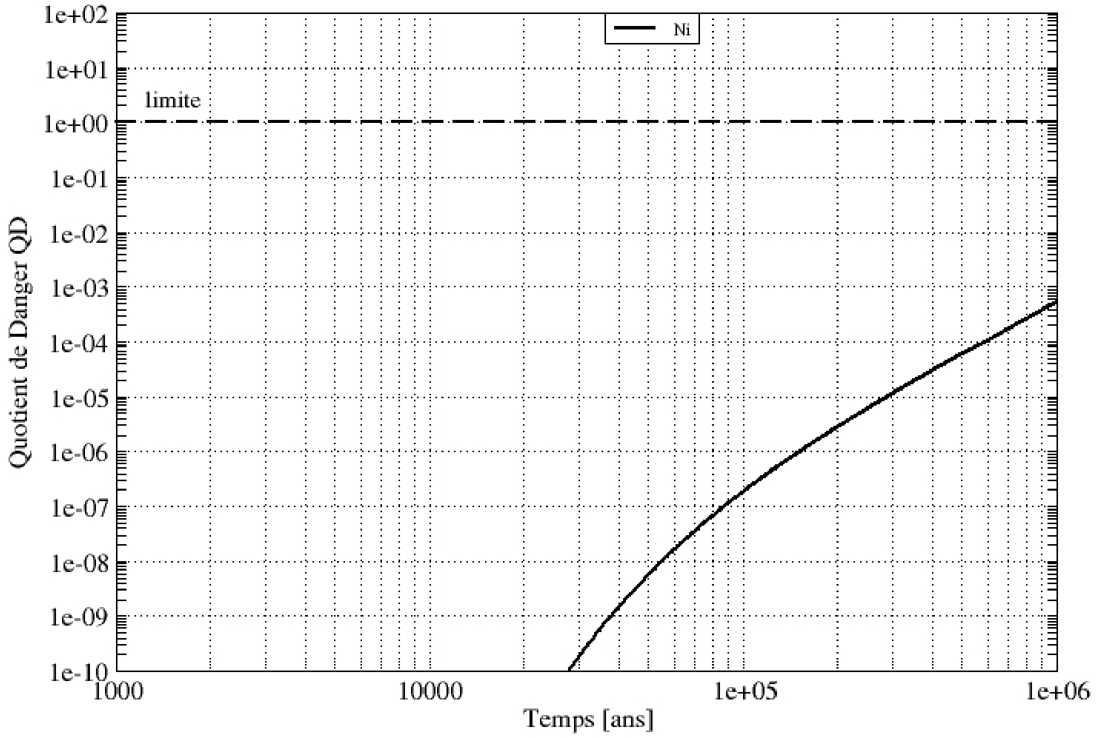


Figure 5.5-31 SEN – Reference calculation – model to 1 million years – Reference package C1/C2 Nickel – Chemical impact at Saulx outlet of Oxfordian linked to non-cancerigenous effects, for inhalation (QD)

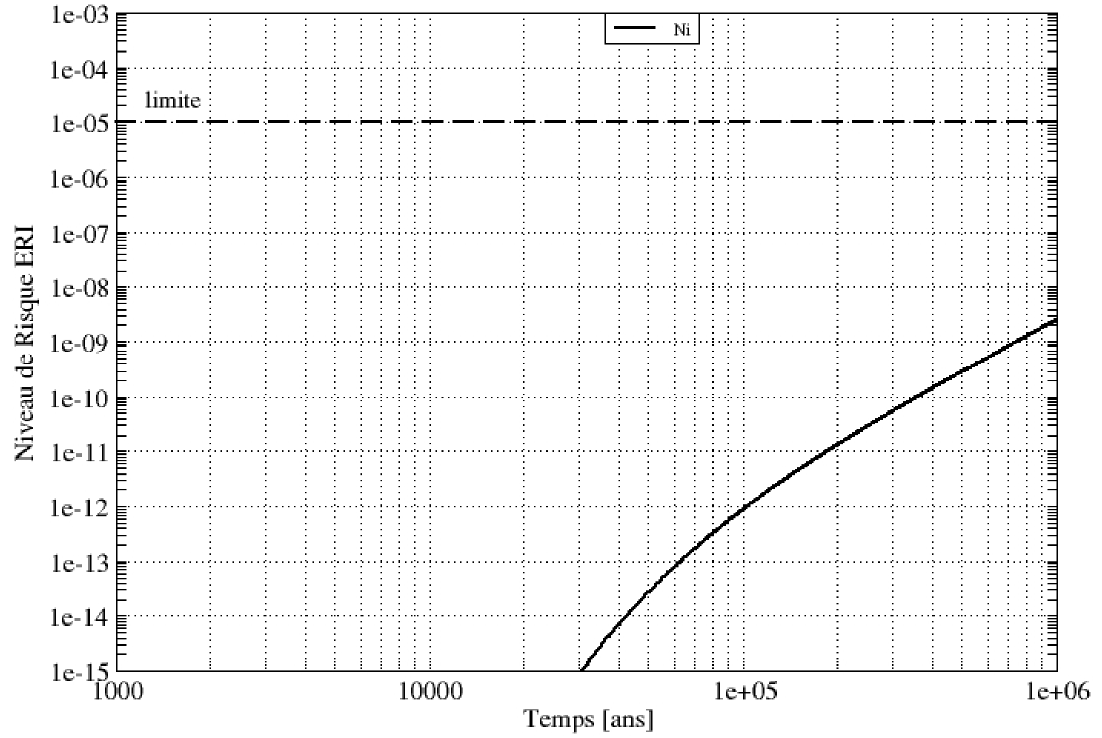


Figure 5.5-32 SEN – Reference calculation – model to 1 million years – Reference package C1/C2 Nickel – Chemical impact at Saulx outlet of Oxfordian linked to non-cancerigenous effects, for inhalation (ERI)

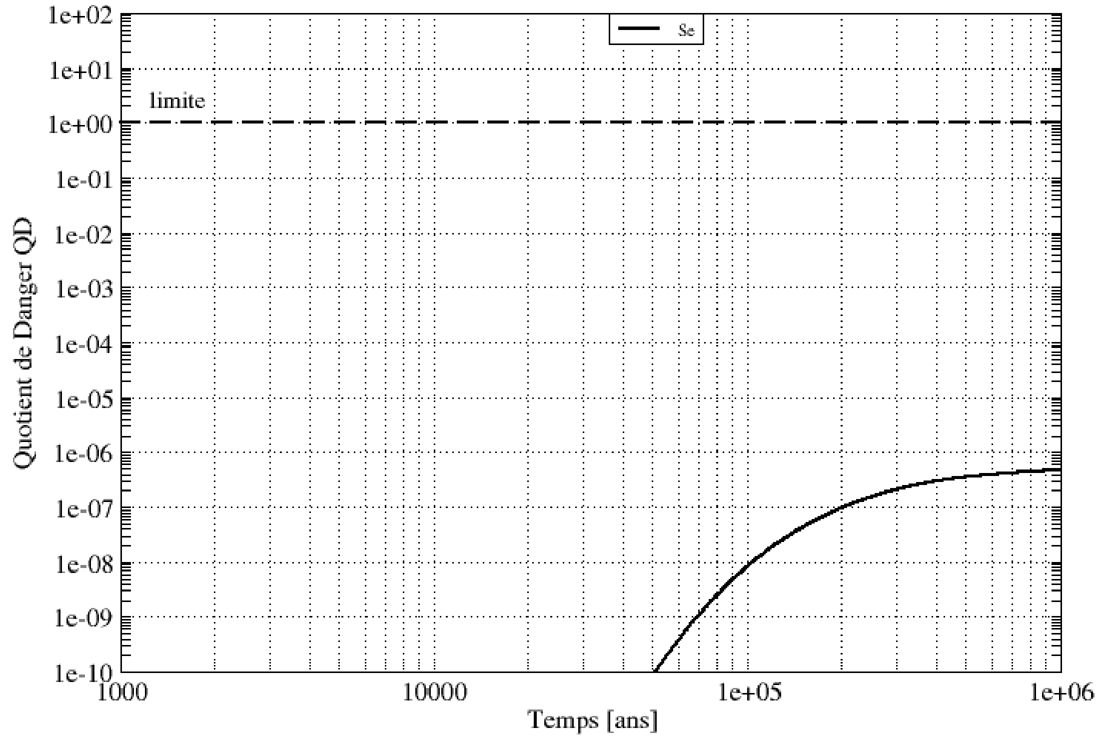


Figure 5.5-33 SEN – Reference calculation – model to 1 million years – Reference package C1/C2 Selenium – Chemical impact at Saulx outlet of Oxfordian linked to non-cancerigenous effects, for ingestion (QD)

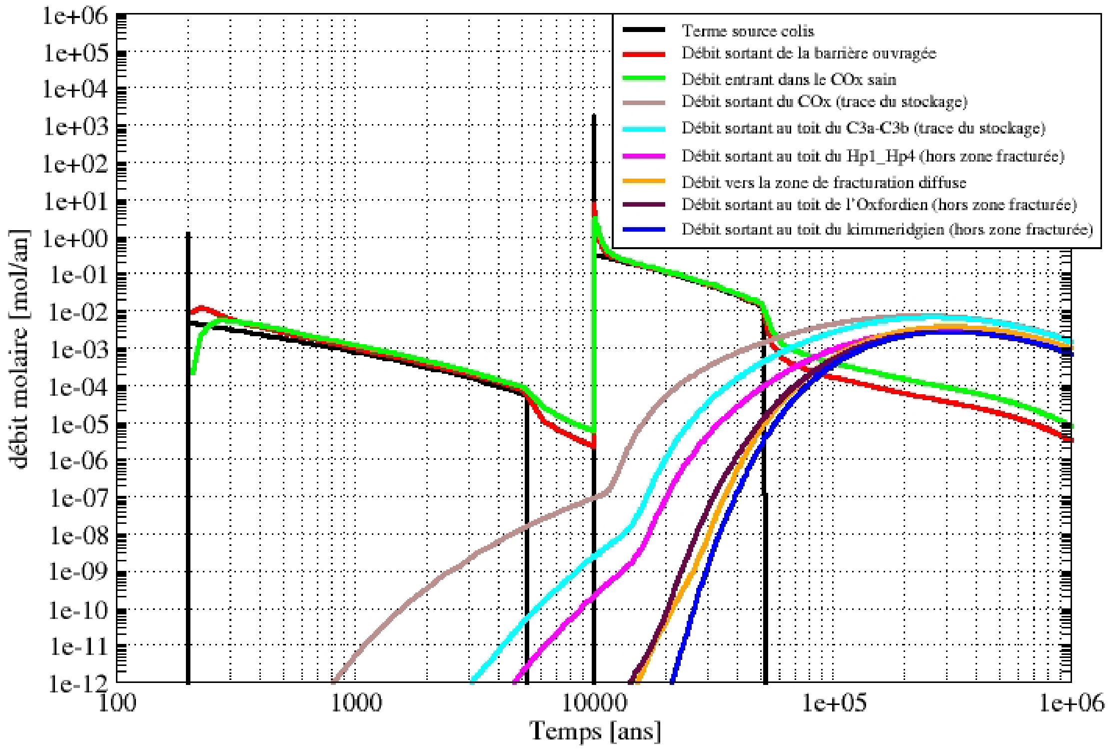


Figure 5.5-34 SEN – Sensibility to hydraulic parameters : CU2 reference package – ^{129}I– molar flow history [Sensitivity to hydraulic parameters - CU2 reference package – ^{129}I]

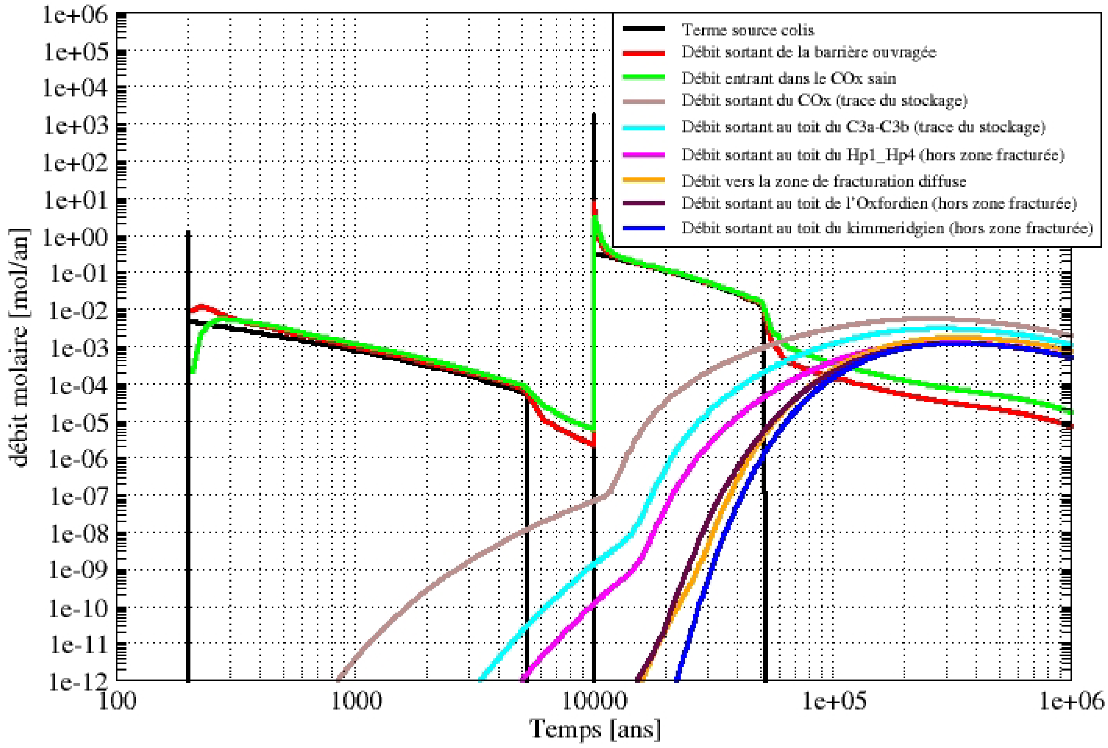


Figure 5.5-34 SEN – Sensibility to hydraulic parameters : CU2 reference package – ^{129}I– molar flow history [Reference calculation (REMINDER) - CU2 reference package – ^{129}I]

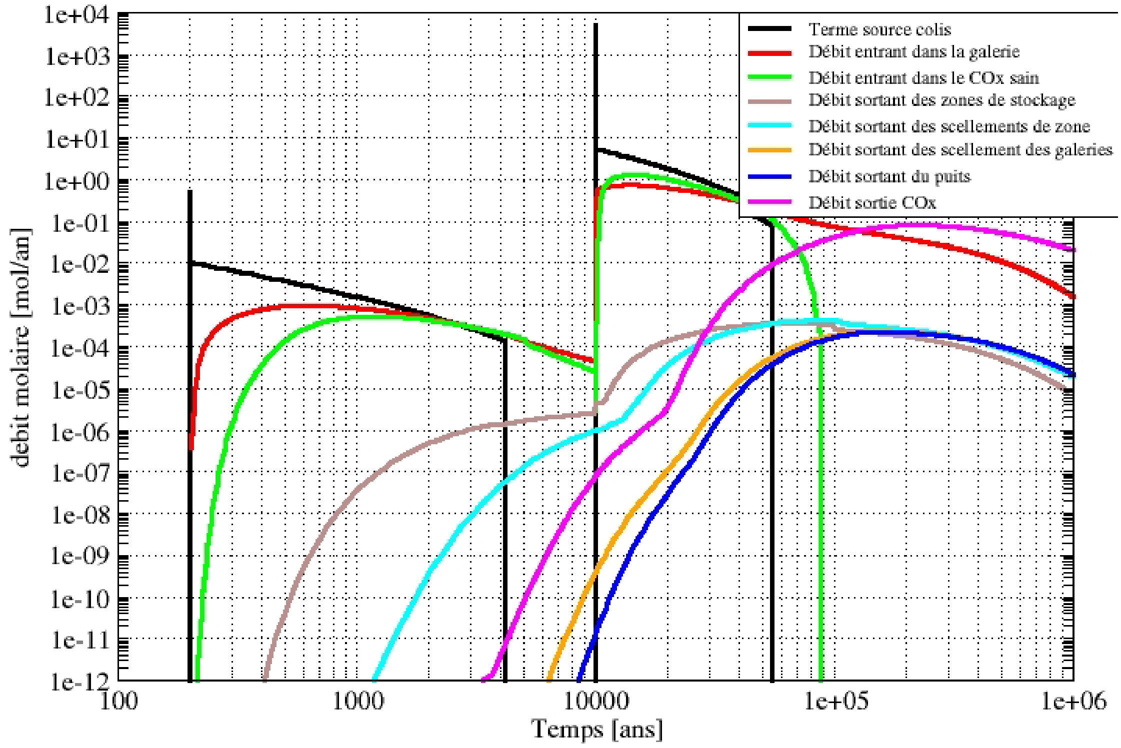


Figure 5.5-37 SEN - Sensitivity : « Degraded » EDZ (Pessimistic fractured zone, conservative micro-fissured zone) - CU1 reference package – ^{129}I - Molar flow history [Sensitivity study]

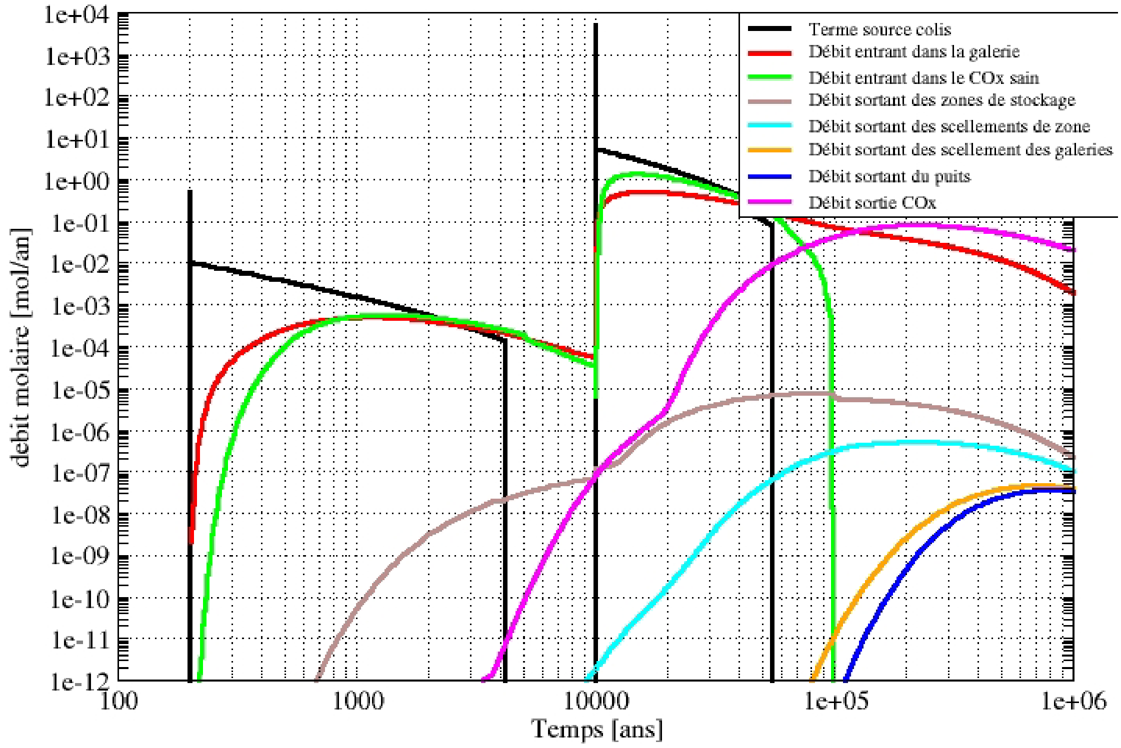


Figure 5.5-37 *SEN - Sensitivity : « Degraded » EDZ (Pessimistic fractured zone, conservative micro-fissured zone) - CU1 reference package – ^{129}I - Molar flow history [Reference calculation]*

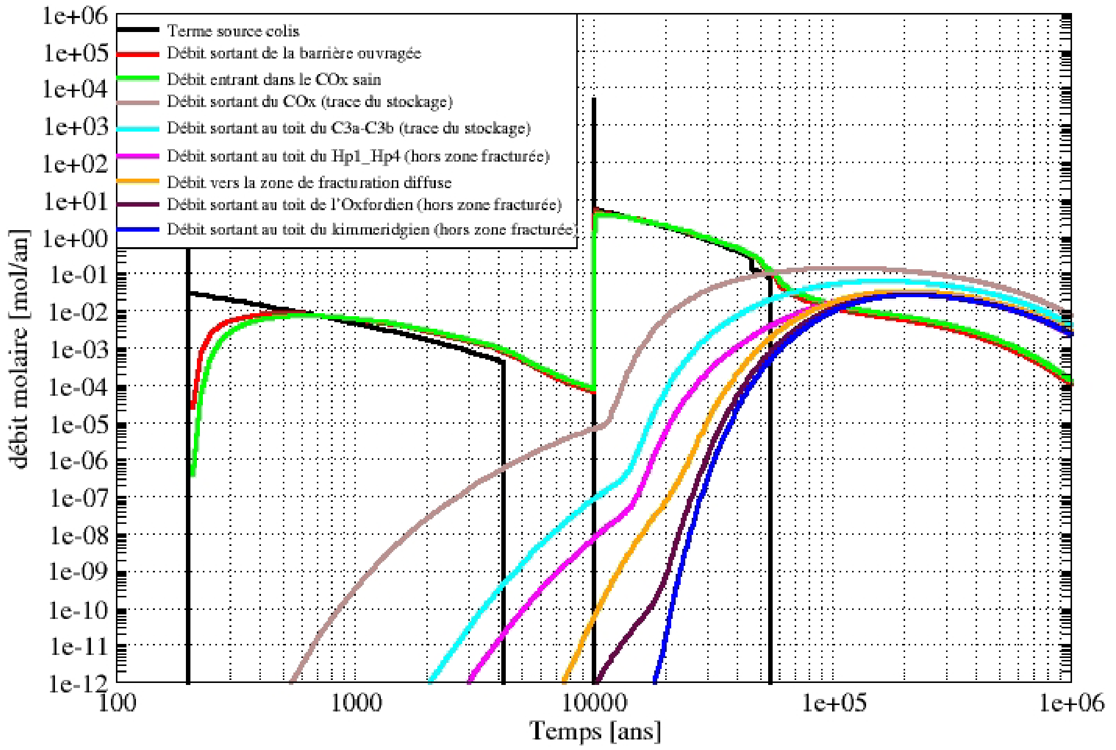


Figure 5.5-39 *SEN – Sensitivity to transport and retention parameters : CU1 reference package – ^{129}I – History of molar flow [Sensitivity study]*

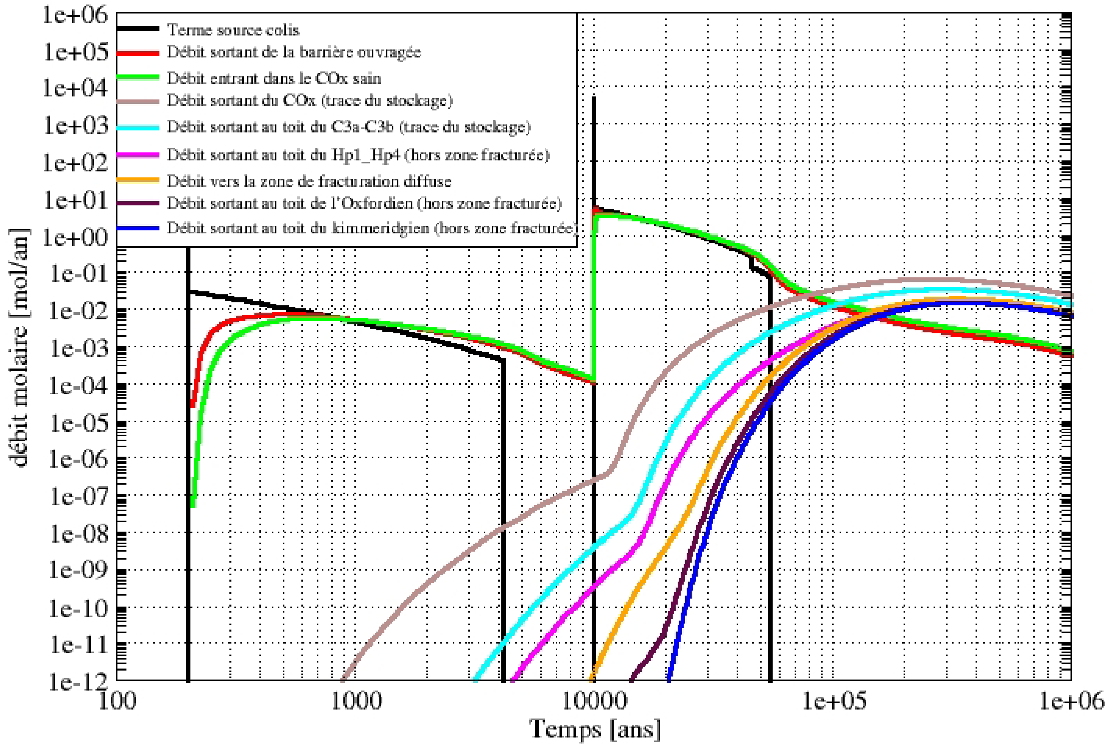


Figure 5.5-39 SEN – Sensitivity to transport and retention parameters : CU1 reference package – ^{129}I – History of molar flow [Reference calculation]

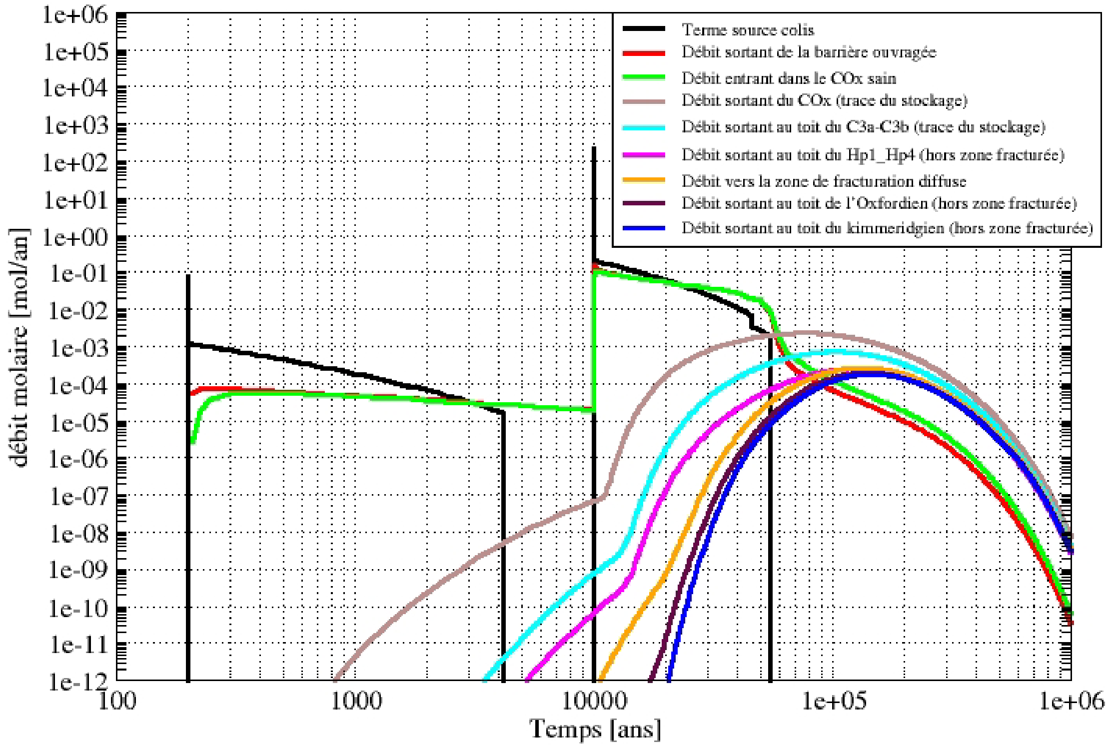


Figure 5.5-40 SEN – Sensitivity to transport and retention parameters : CU1 reference package – ^{79}Se– History of molar flow [Sensitivity study]

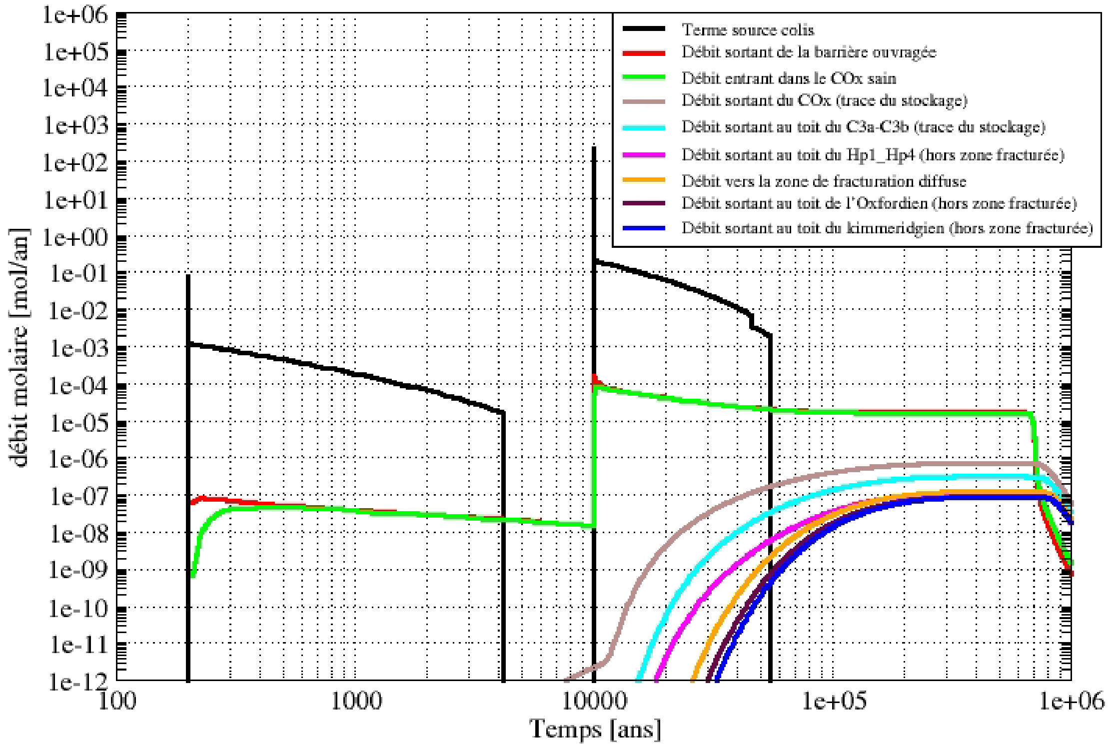


Figure 5.5-40 SEN – Sensitivity to transport and retention parameters : CU1 reference package – $^{79}Se^{-}$ History of molar flow [Reference calculation]

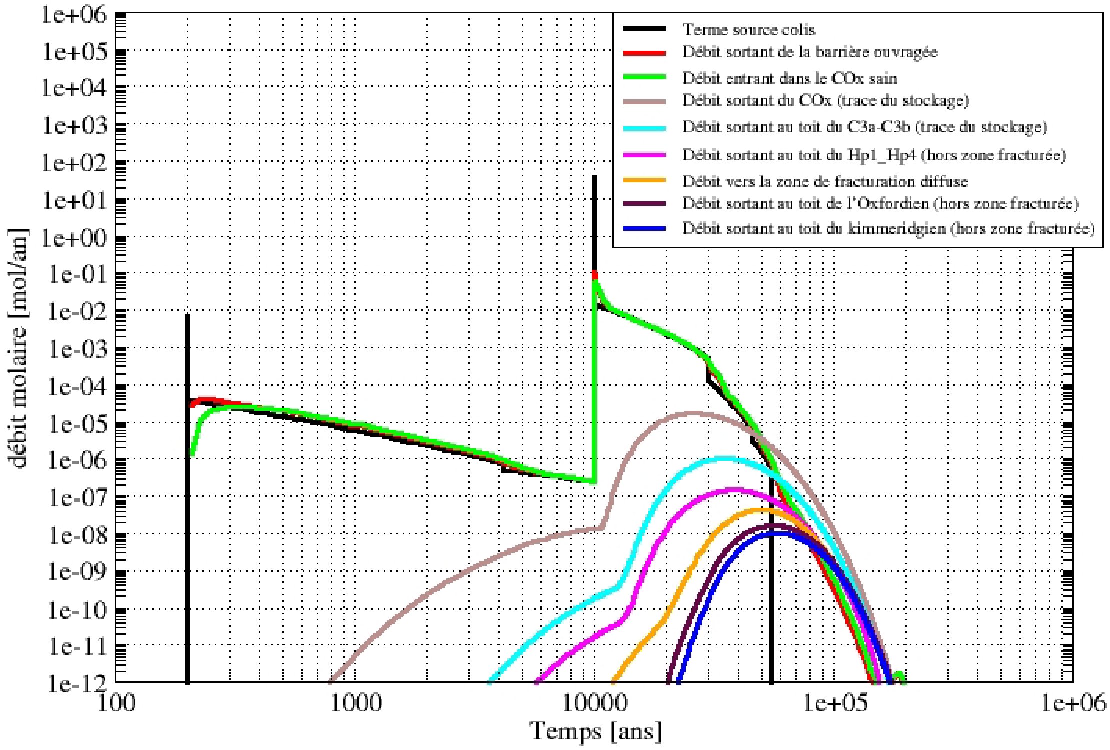


Figure 5.5-41 SEN – Sensitivity to transport and retention parameters : CU1 reference package – $^{14}C^{-}$ History of molar flow [Sensitivity study]

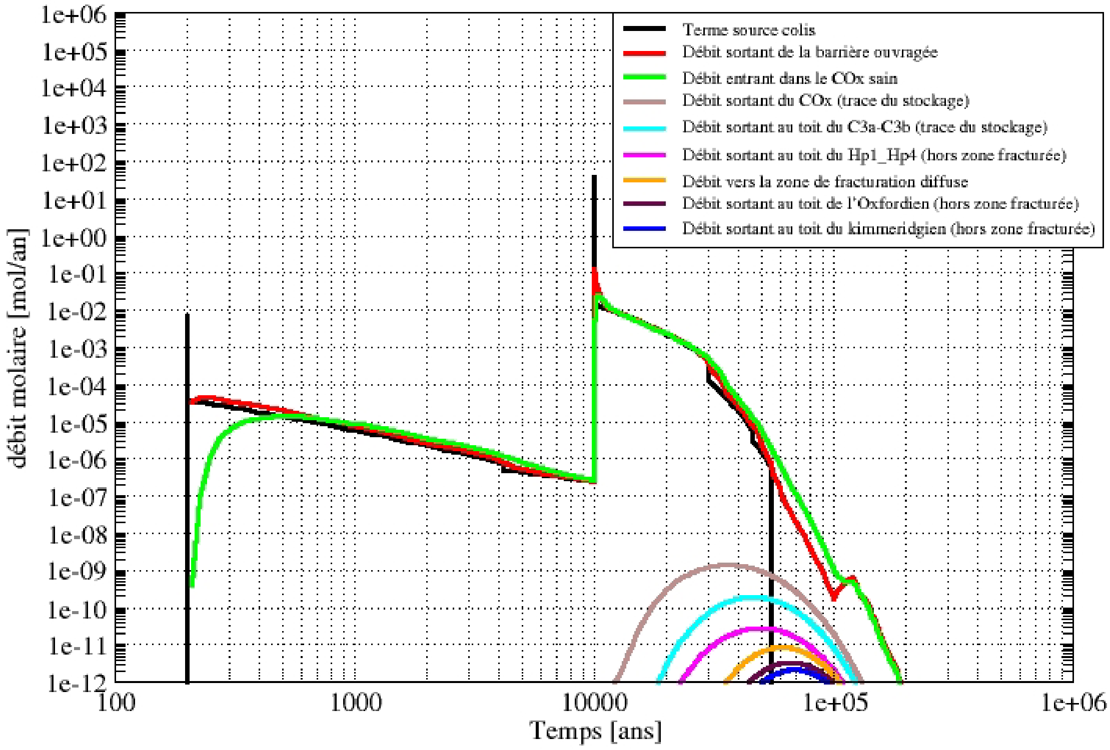


Figure 5.5-41 SEN – Sensitivity to transport and retention parameters : CU1 reference package – 14C– History of molar flow [Reference calculation]

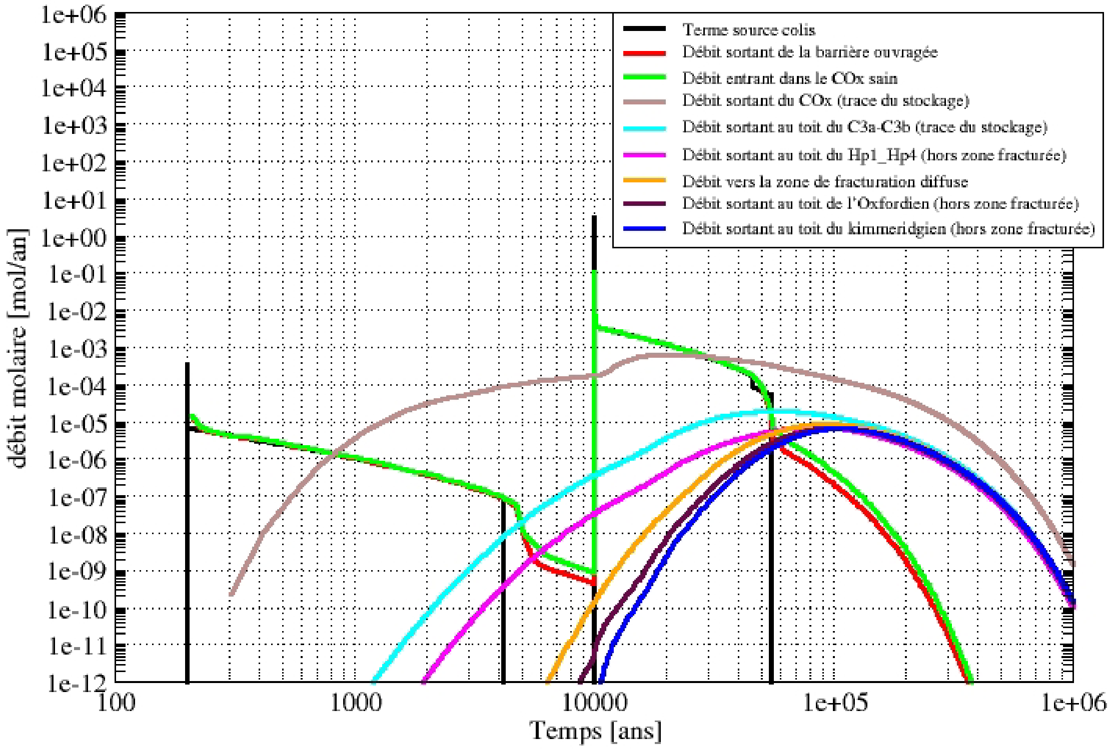


Figure 5.5-42 SEN – Sensitivity to transport and retention parameters : CU1 reference package – 10Be– History of molar flow

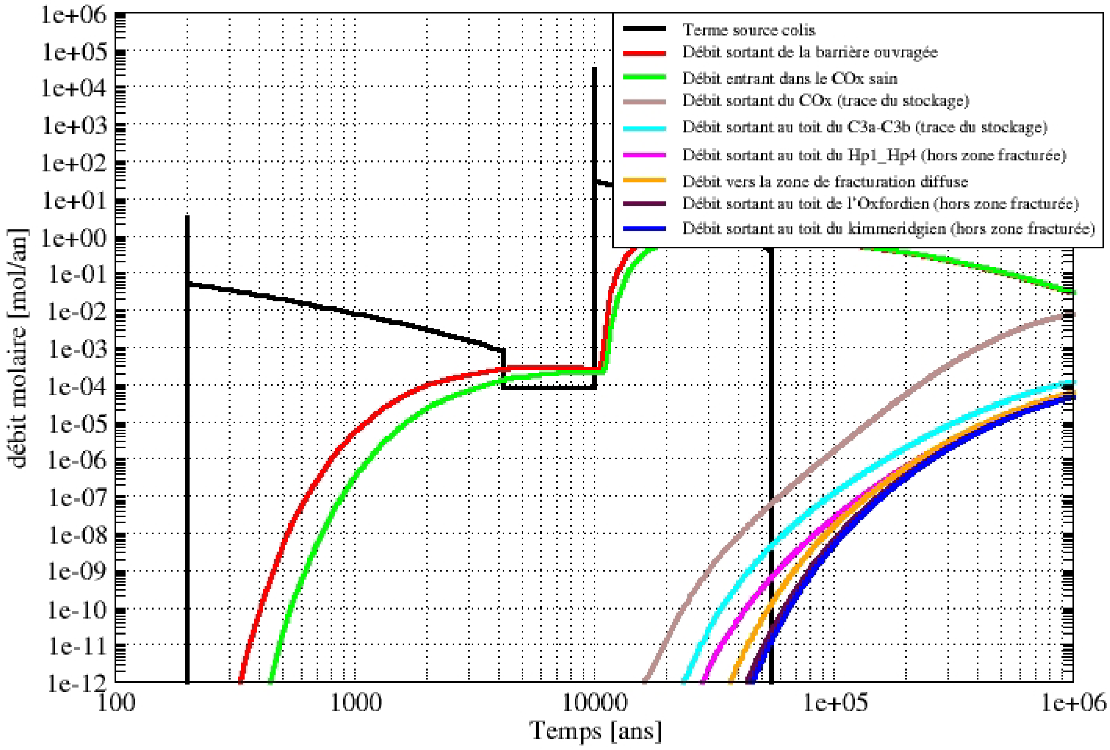


Figure 5.5-43 SEN – Sensitivity to transport and retention parameters : CU1 reference package – ^{93}Zr– History of molar flow

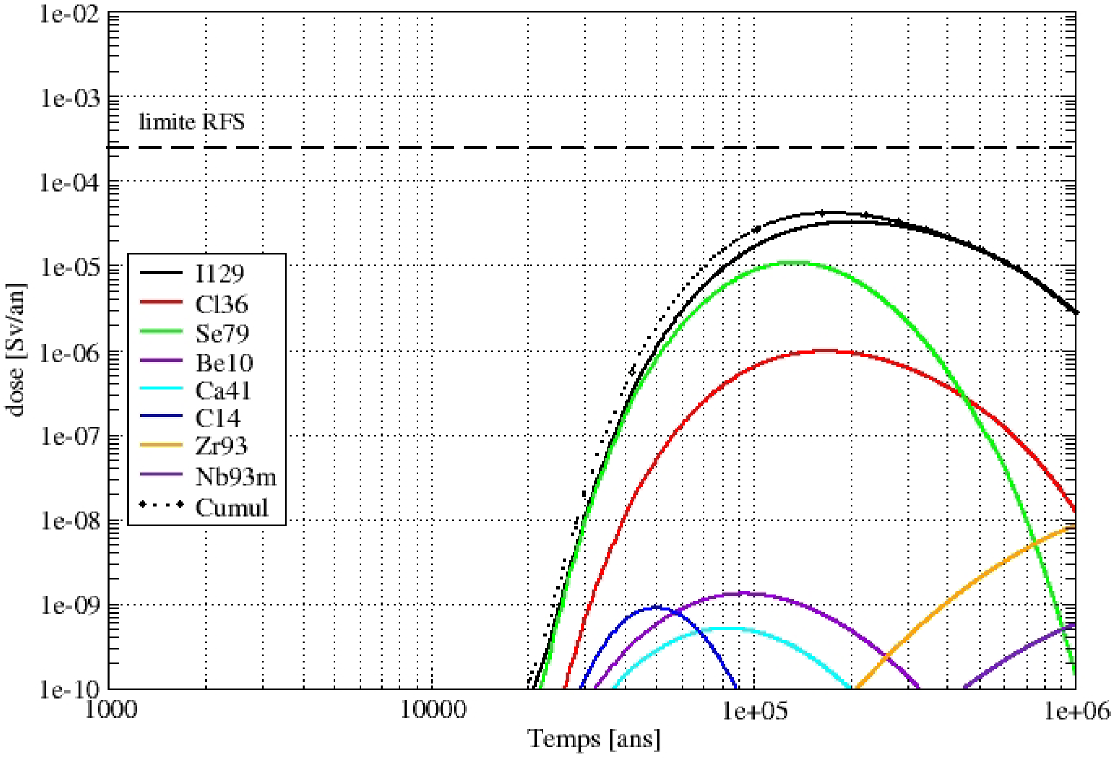


Figure 5.5-44 SEN – Sensitivity to transport and retention parameters (conservative values) - Dose at the Saulx outlet of the Oxfordian – CU1 reference package – 1 million year model [Sensitivity study]

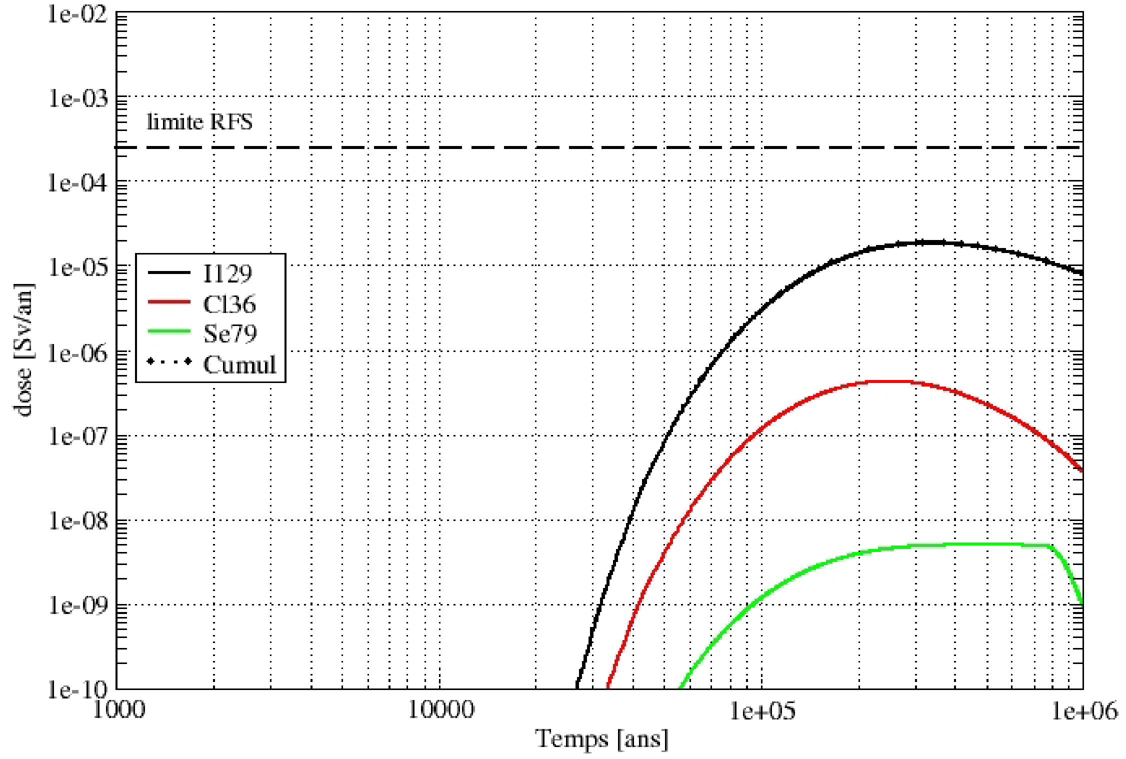


Figure 5.5-44 *SEN – Sensitivity to transport and retention parameters (conservative values) - Dose at the Saulx outlet of the Oxfordian – CU1 reference package – 1 million year model [Reference calculation – REMINDER]*

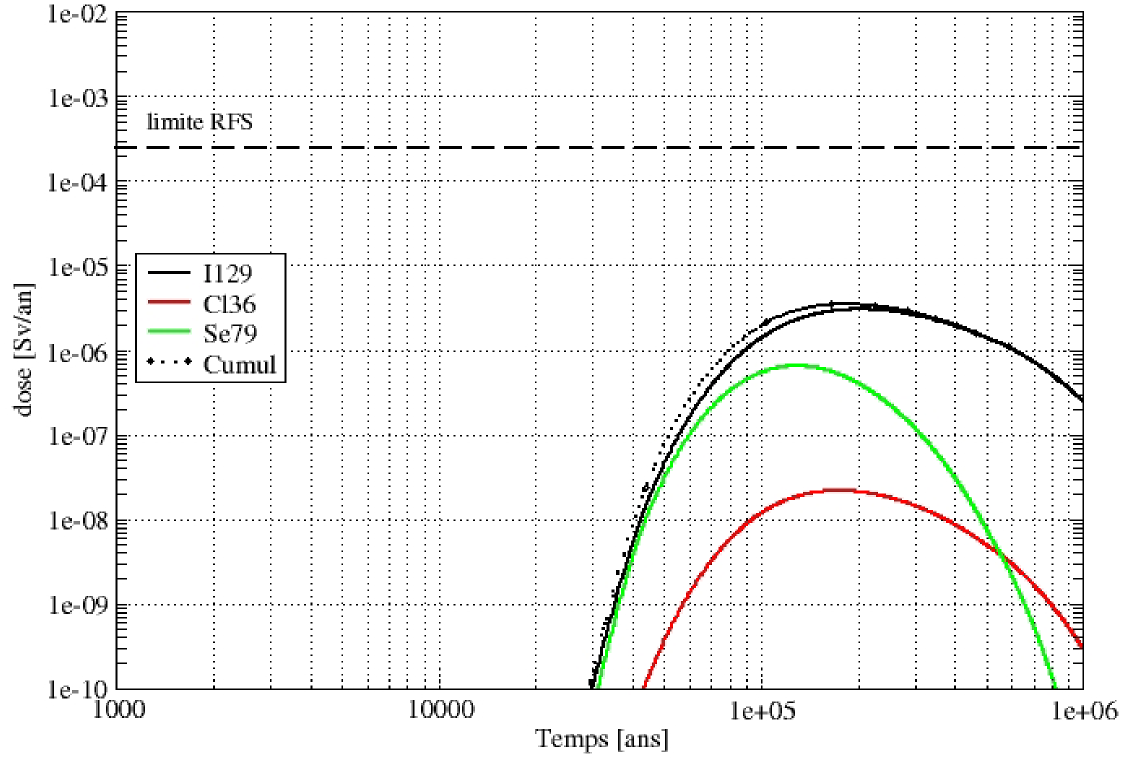


Figure 5.5-45 *SEN – Sensitivity to transport and retention parameters (conservative values) - Dose at the Saulx outlet of the Oxfordian – CU2 reference package – 1 million year model [Sensitivity study]*

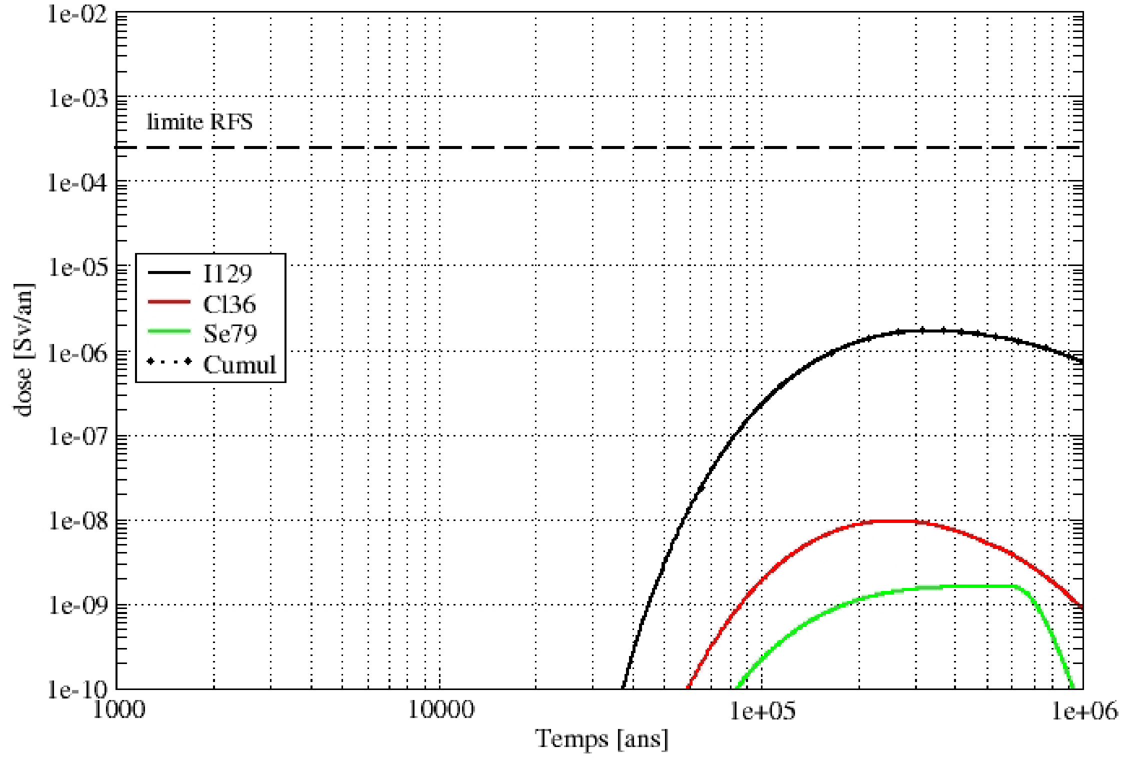


Figure 5.5-45 SEN – Sensitivity to transport and retention parameters (conservative values) - Dose at the Saulx outlet of the Oxfordian – CU2 reference package – 1 million year model [Reference calculation – REMINDER]

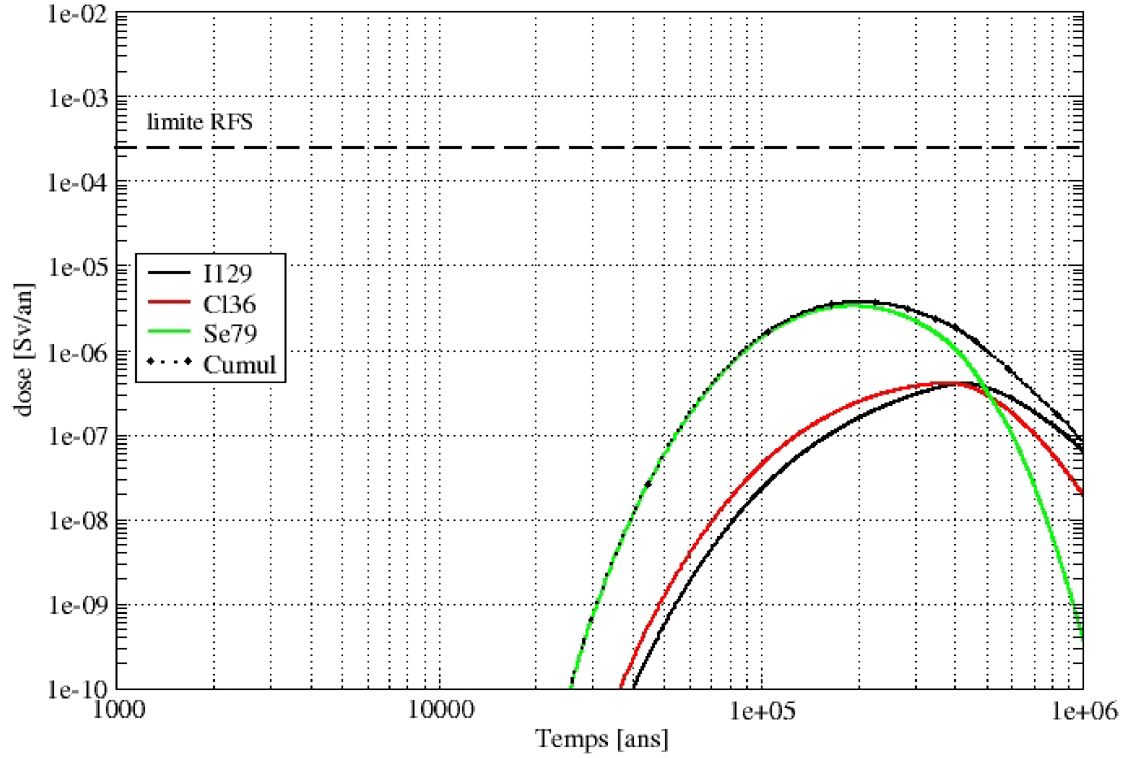


Figure 5.5-46 SEN – Sensitivity to transport and retention parameters (conservative values) - Dose at the Saulx outlet of the Oxfordian – Reference packages (C1+C2) [Sensitivity study]

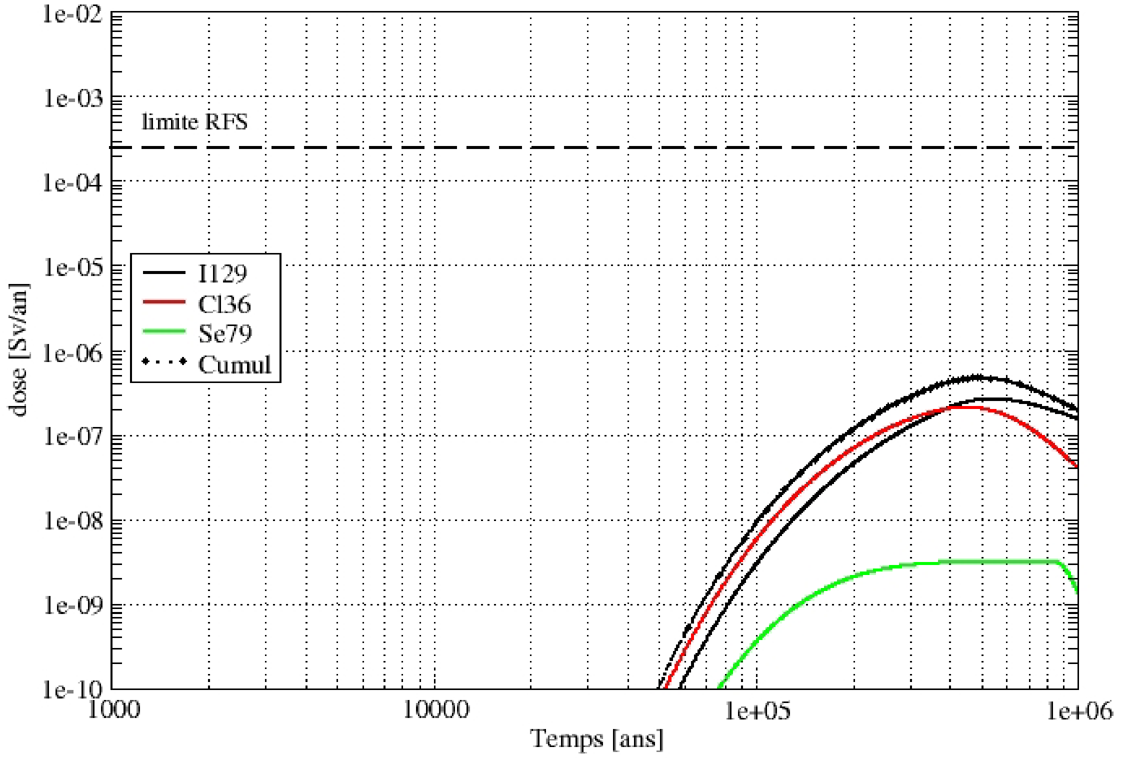


Figure 5.5-46 *SEN – Sensitivity to transport and retention parameters (conservative values) - Dose at the Saulx outlet of the Oxfordian – Reference packages (C1+C2) [Reference calculation - REMINDER]*

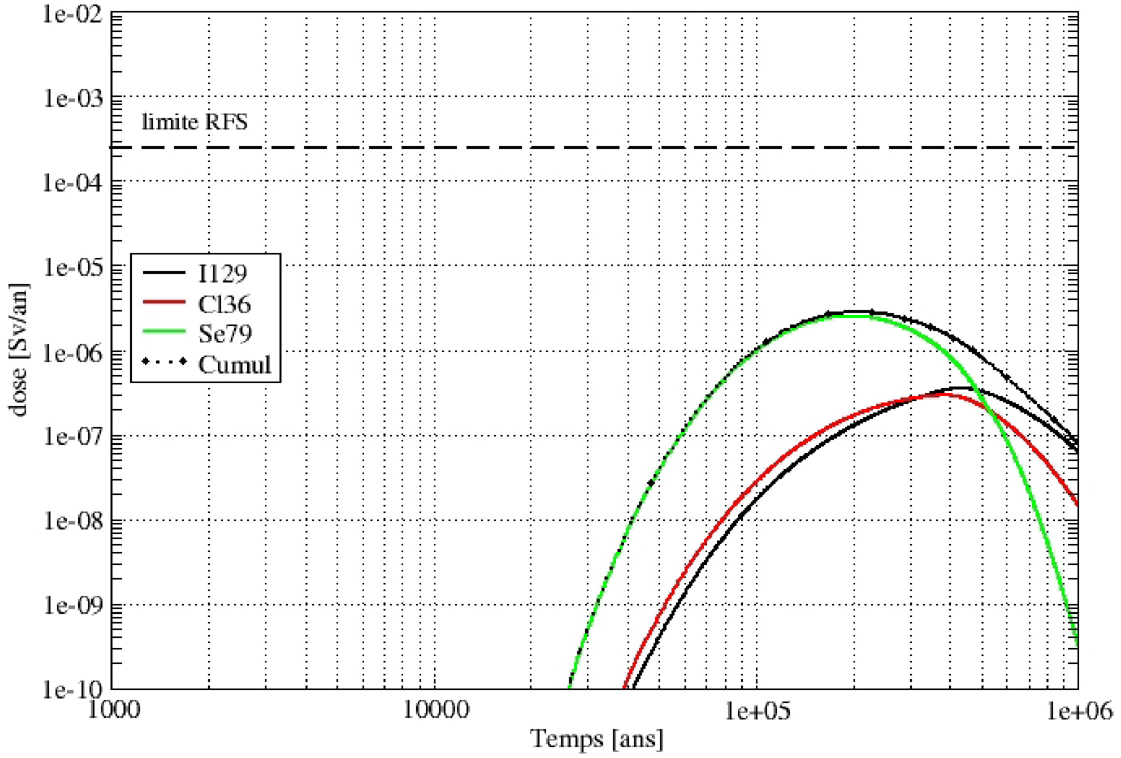


Figure 5.5-47 *SEN – Sensitivity to transport and retention parameters (conservative values) - Dose at the Saulx outlet of the Oxfordian – Reference packages (C3+C4) [Sensitivity study]*

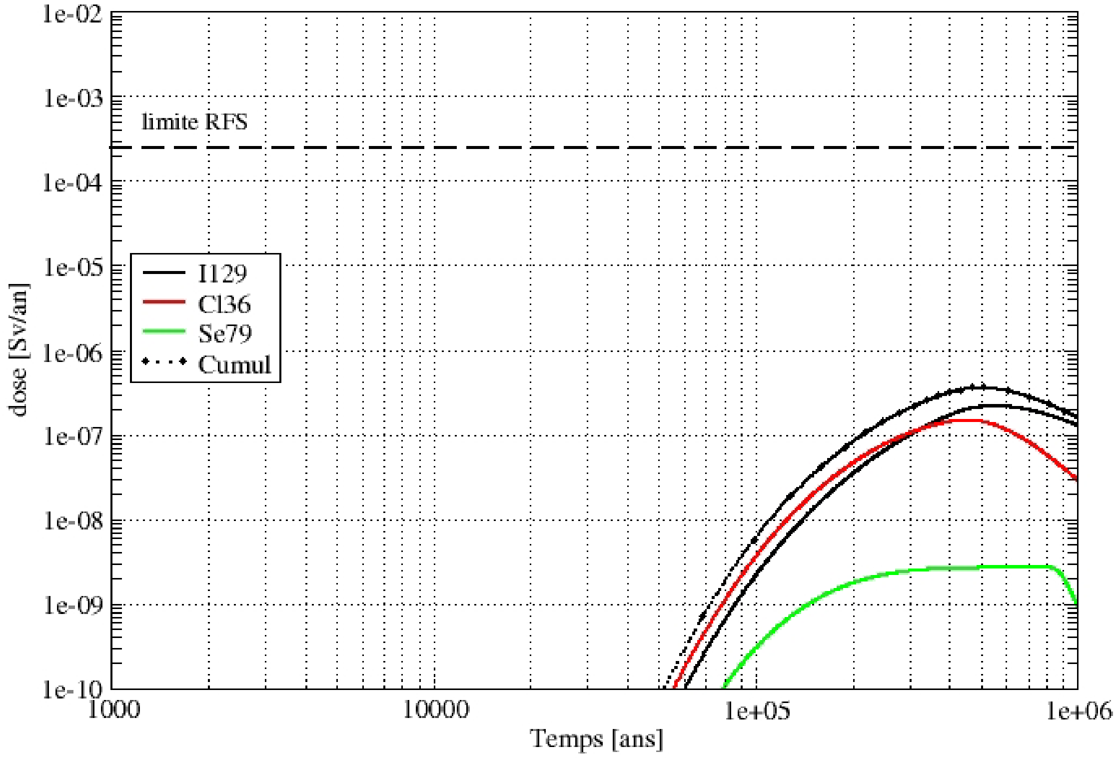


Figure 5.5-47 *SEN – Sensitivity to transport and retention parameters (conservative values) - Dose at the Saulx outlet of the Oxfordian – Reference packages (C3+C4) [Reference calculation - REMINDER]*

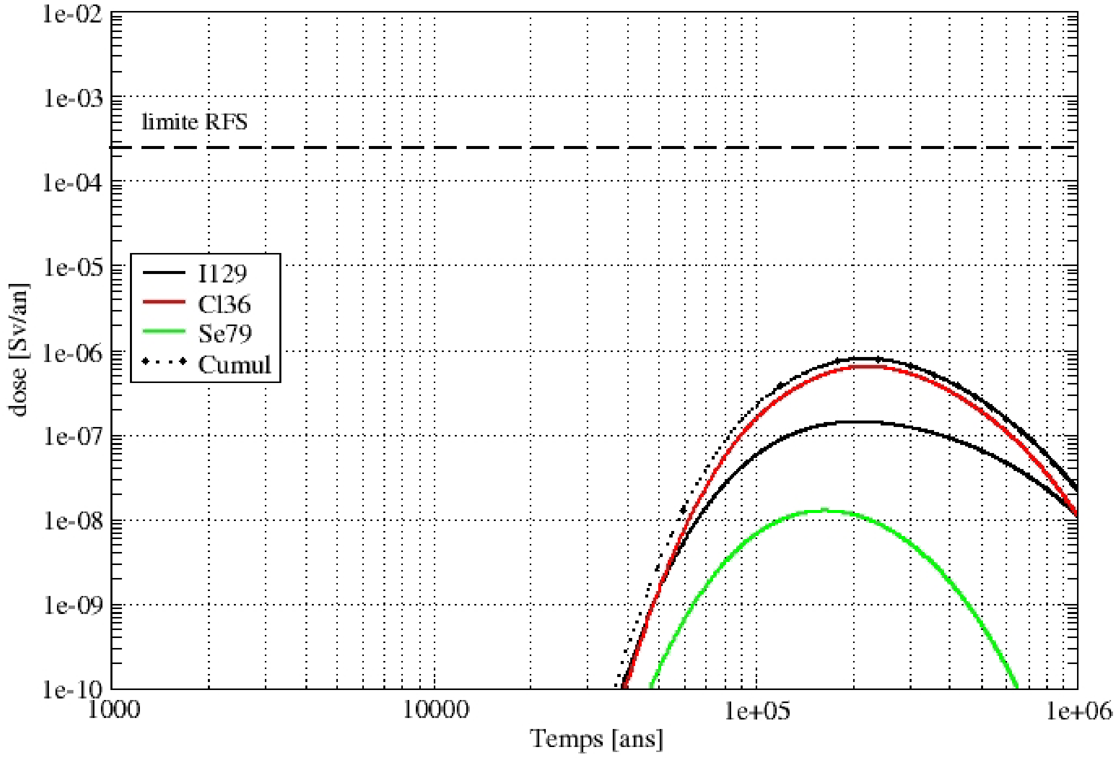


Figure 5.5-48 *SEN – Sensitivity to transport and retention parameters (conservative values) - Dose at the Saulx outlet of the Oxfordian – Reference waste packages B (B1x) [Sensitivity study]*

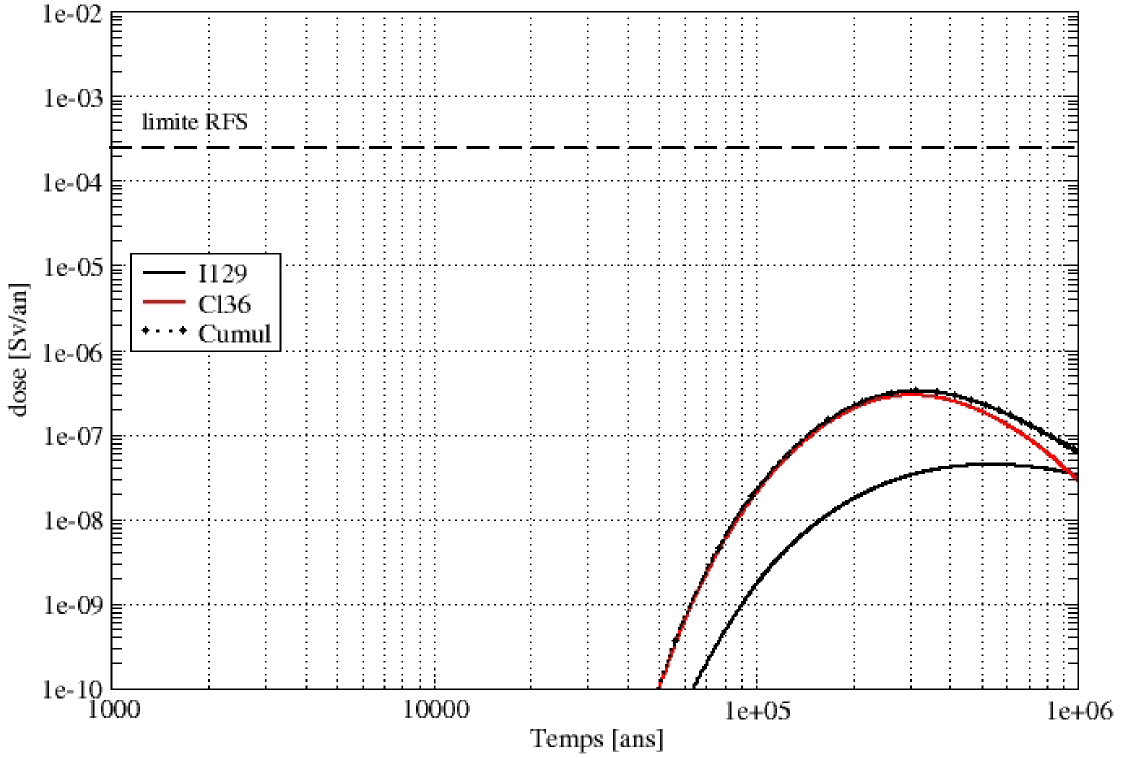


Figure 5.5-48 *SEN – Sensitivity to transport and retention parameters (conservative values) - Dose at the Saulx outlet of the Oxfordian – Reference waste packages B (B1x) [Reference calculation - REMINDER]*

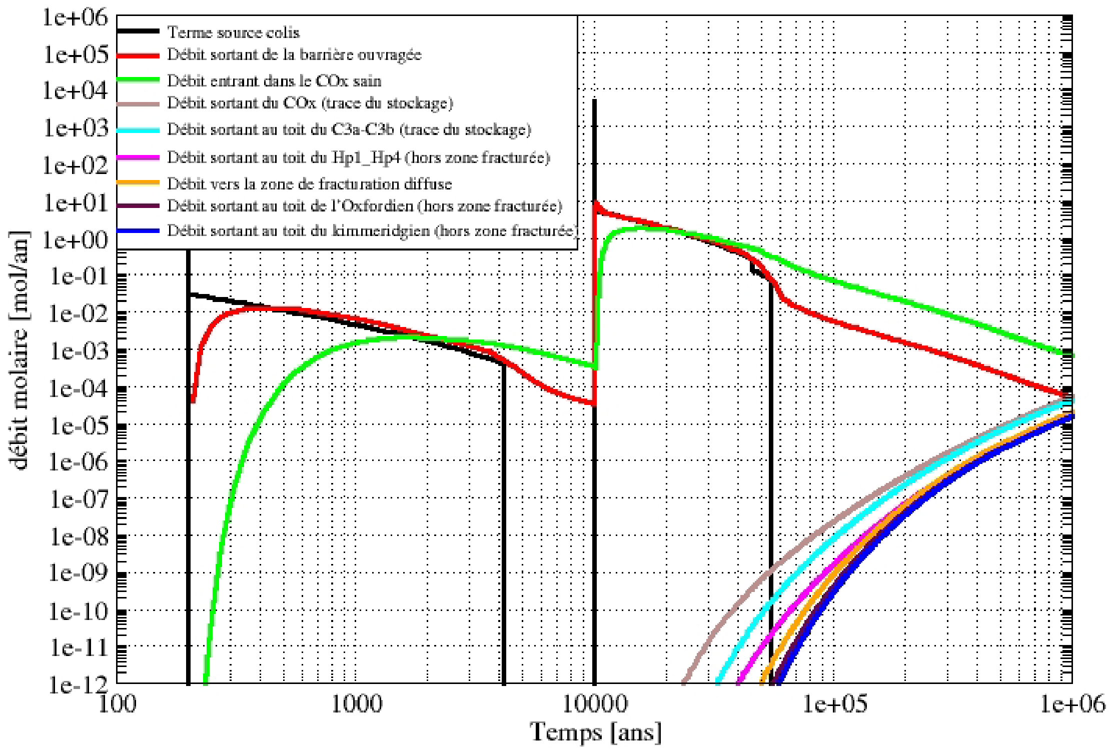


Figure 5.5-49 *SEN – Sensitivity to the iodine retardation coefficient in the Callovo-Oxfordian : CU1 reference package – ^{129}I– History of molar flow [Sensitivity study (Kd = 10^{-3} m^3/kg)]*

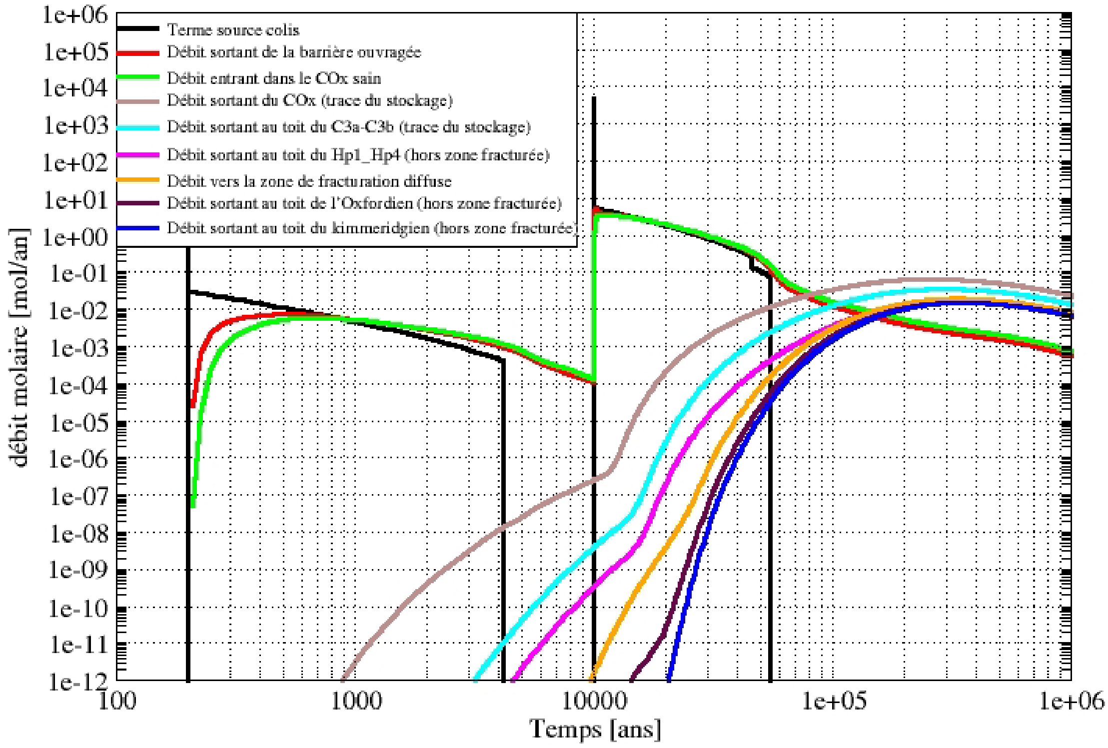


Figure 5.5-49 *SEN – Sensitivity to the iodine retardation coefficient in the Callovo-Oxfordian : CU1 reference package – ^{129}I– History of molar flow [Reference calculation (Kd = 0-3 m^3/kg)]*

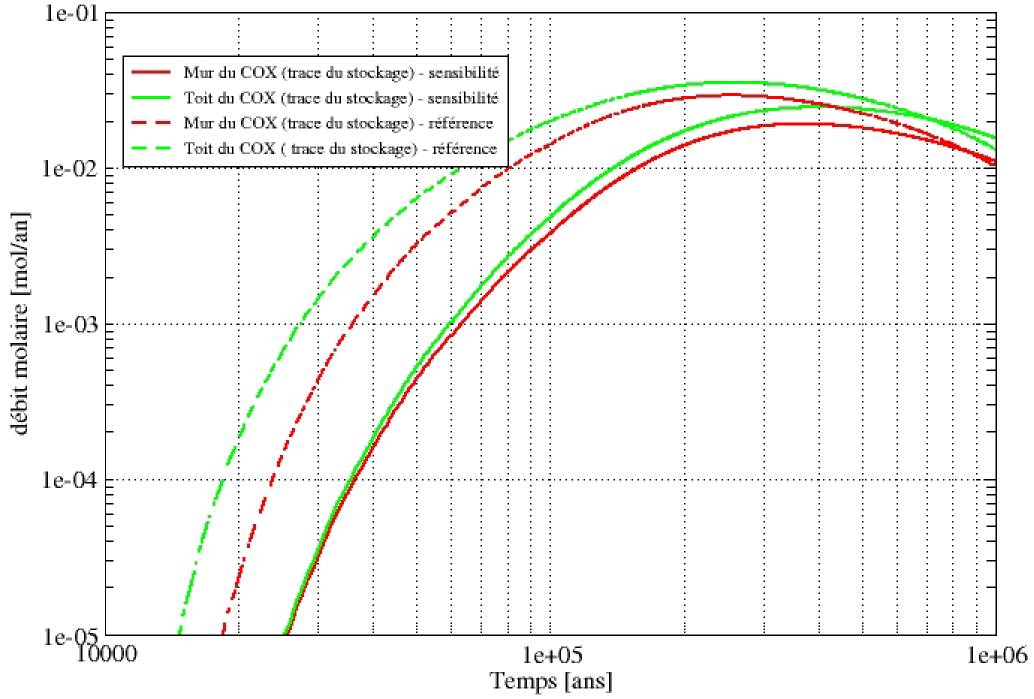


Figure 5.5-50 *SEN – Sensitivity to the thickness of the Callovo-Oxfordian (150 meters instead of the 130 meters in the reference calculation) - CU1 reference packages – ^{129}I – History of molar flow. (COX = Callovo-Oxfordian)*

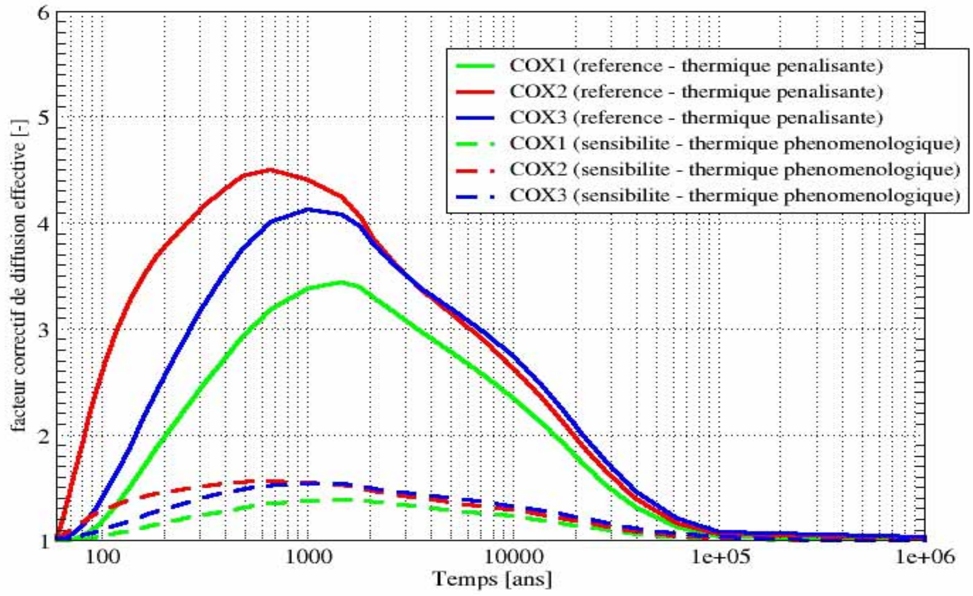


Figure 5.5-51 *SEN - Comparison of diffusion correction factors (phenomenological and pessimistic thermal effects) – CU1 reference package*

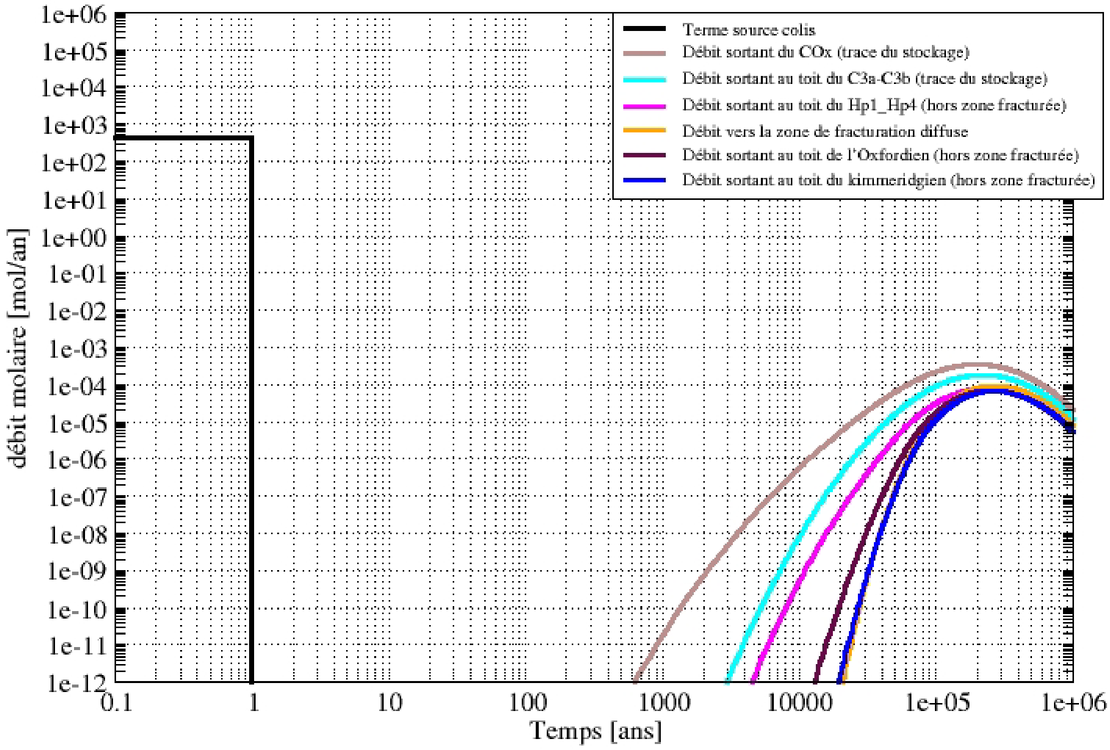


Figure 5.5-52 *SEN – Sensitivity to reference model used for inorganic packages that do not release hydrogen (pessimistic model) - B1x reference package – ^{36}Cl – history of molar flow [Sensitivity study]*

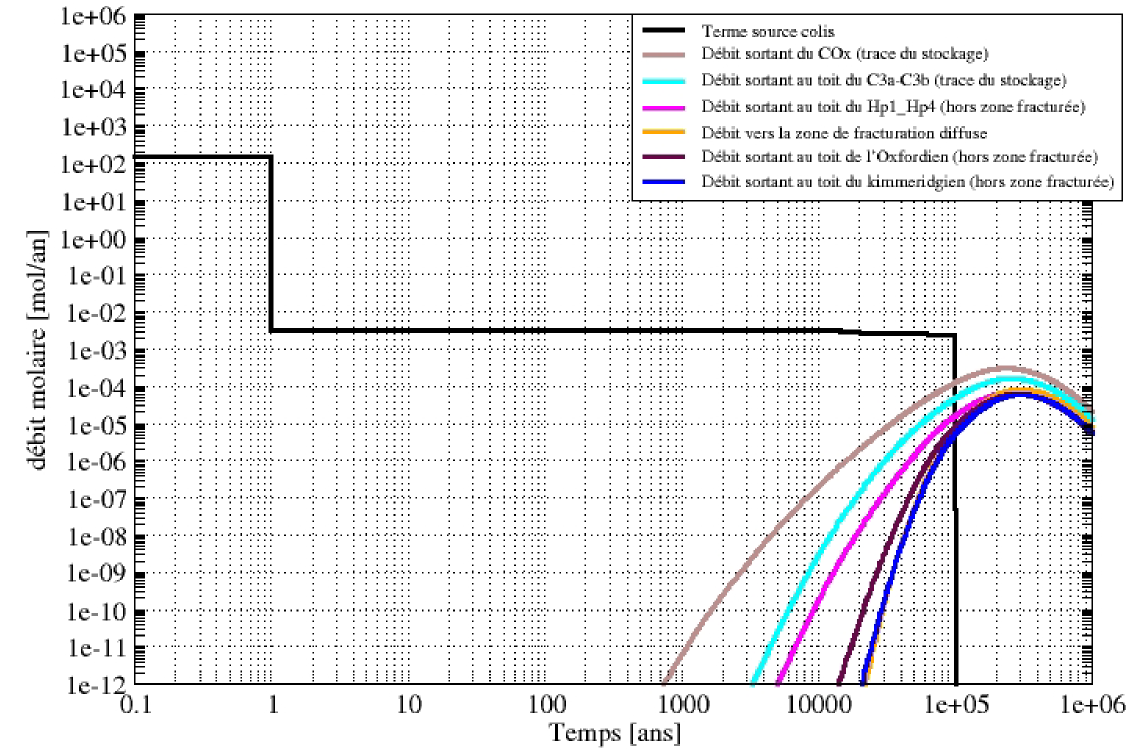


Figure 5.5-52 SEN – Sensitivity to reference model used for inorganic packages that do not release hydrogen (pessimistic model) - B1x reference package – ^{36}Cl – history of molar flow [Reference calculation]

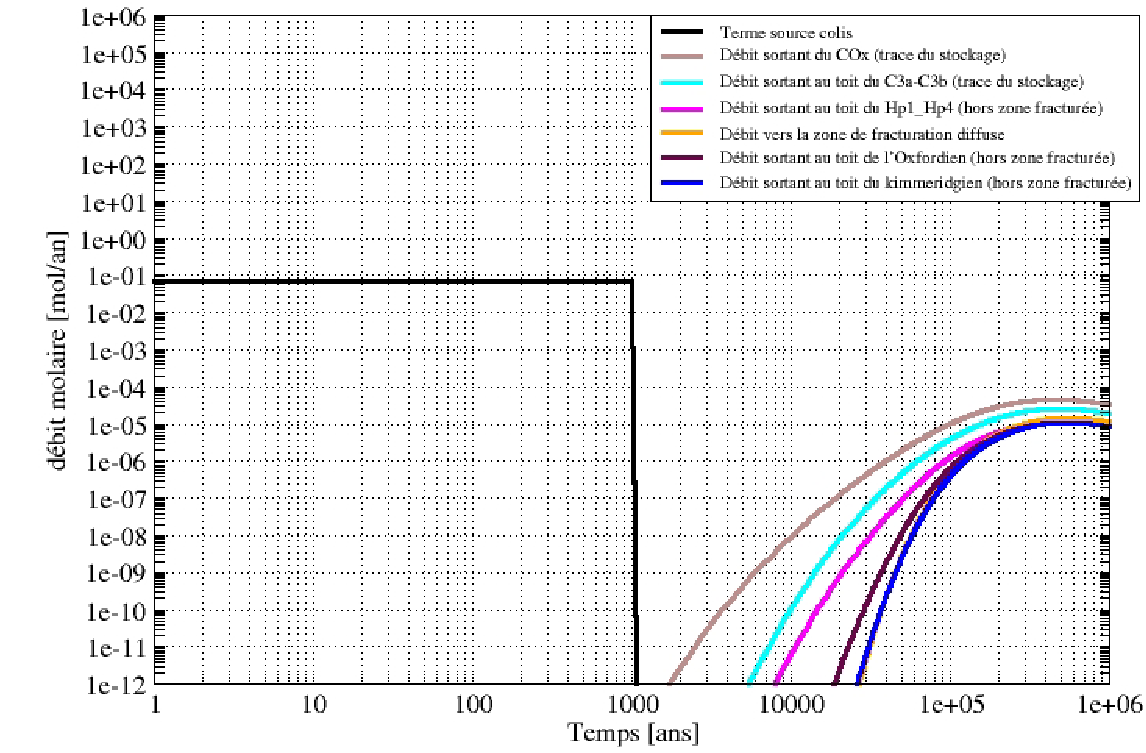


Figure 5.5-53 SEN – Sensitivity to reference model used for the B2 reference package (pessimistic model) - B2 reference package – ^{129}I – history of molar flow [Sensitivity study]

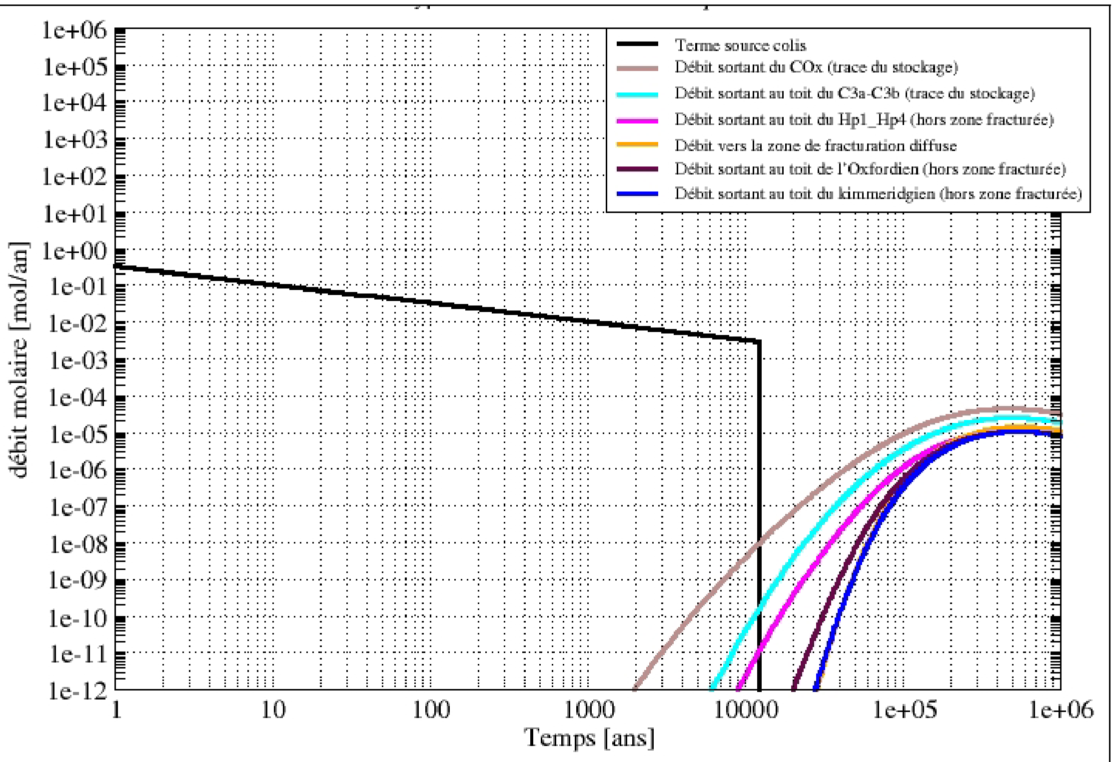


Figure 5.5-53 SEN – Sensitivity to reference model used for the B2 reference package (pessimistic model) - B2 reference package – ^{129}I – history of molar flow [Reference calculation]

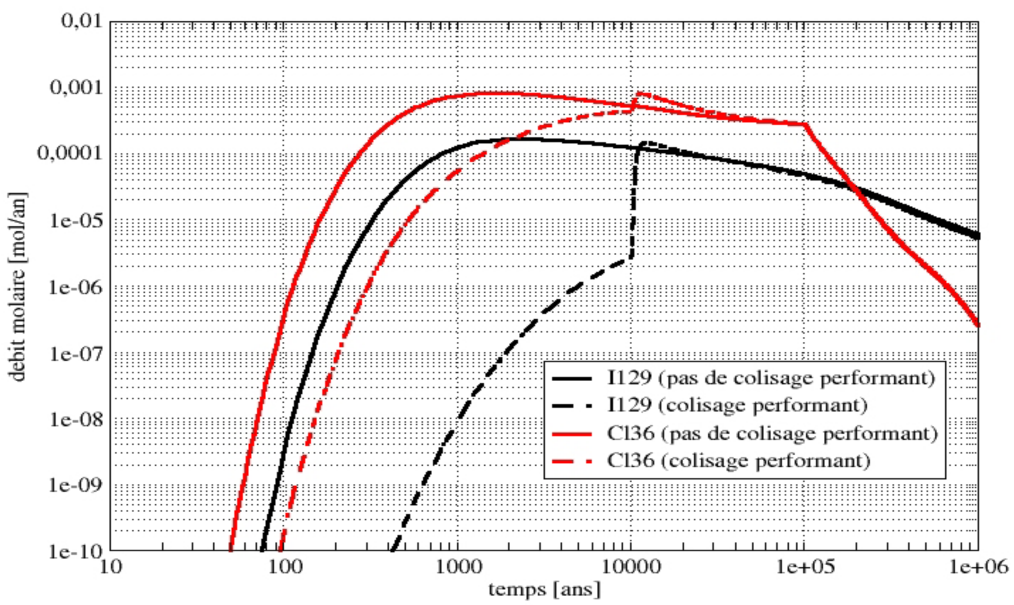


Figure 5.5-54 SEN – Changes in molar flow exiting B1x waste cells for ^{129}I and ^{36}Cl – comparison with and without effective containers

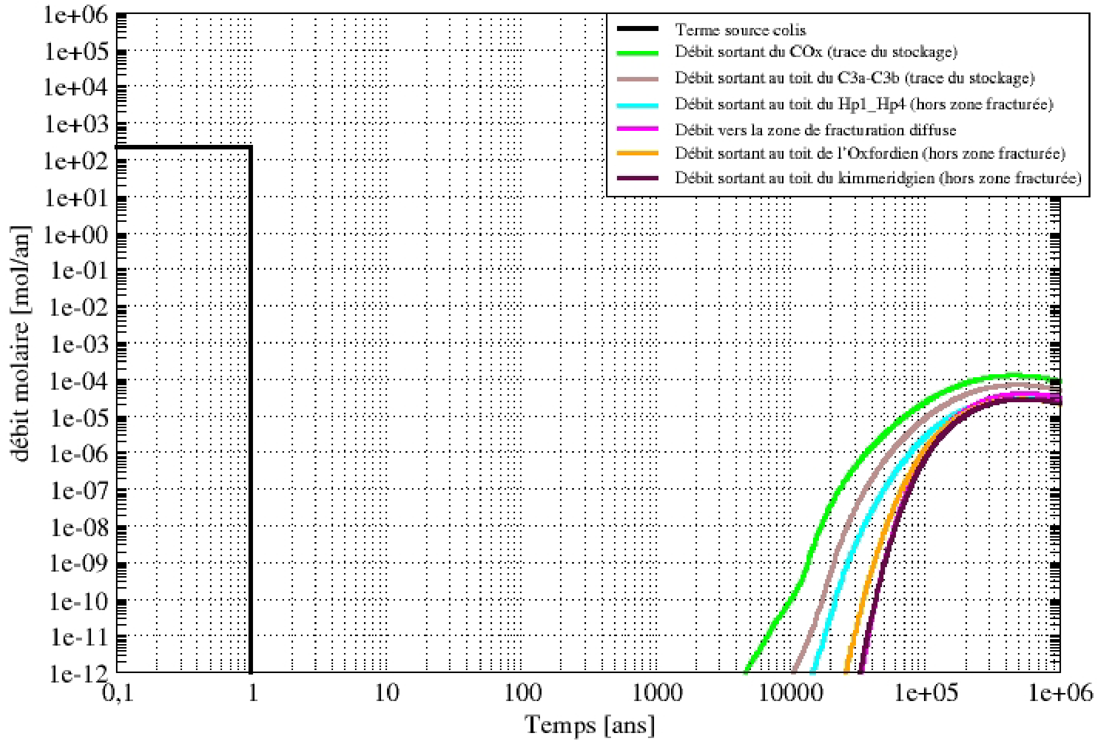


Figure 5.5-45 SEN – Sensitivity to transport and retention parameters (conservative values) - Dose at the Saulx outlet of the Oxfordian – CU2 reference package – 1 million year model[Sensitivity study]

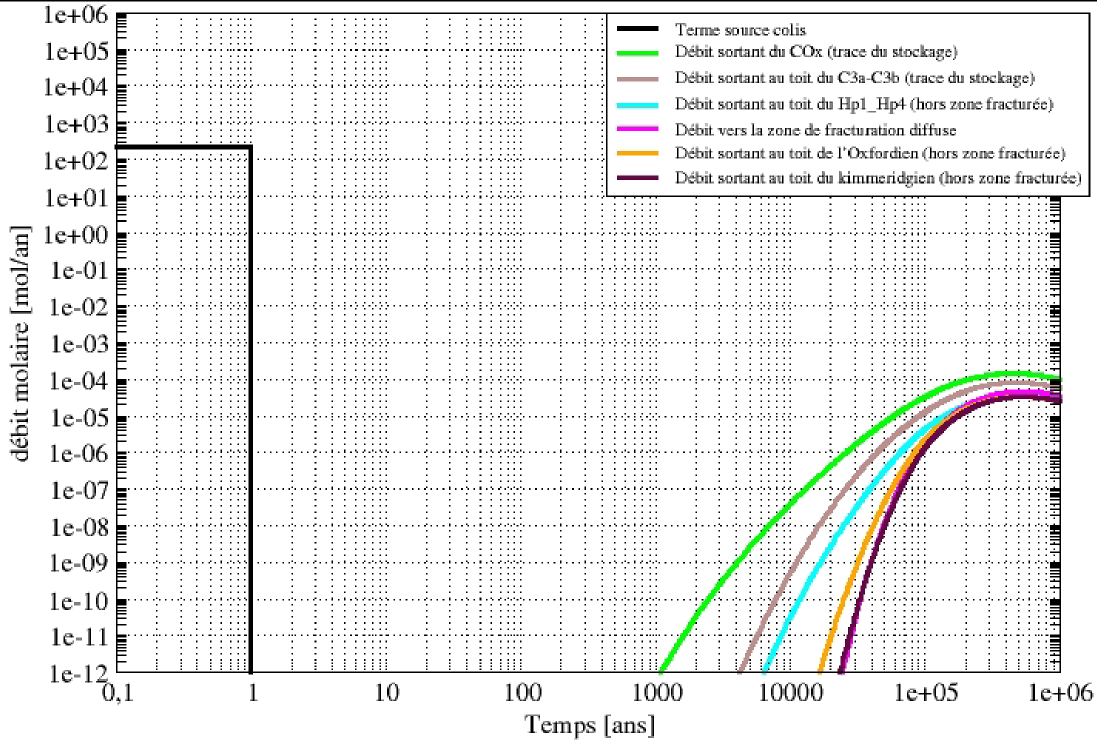


Figure 5.5-55 B1x reference package – ^{129}I – molar flow history [Reference calculation]

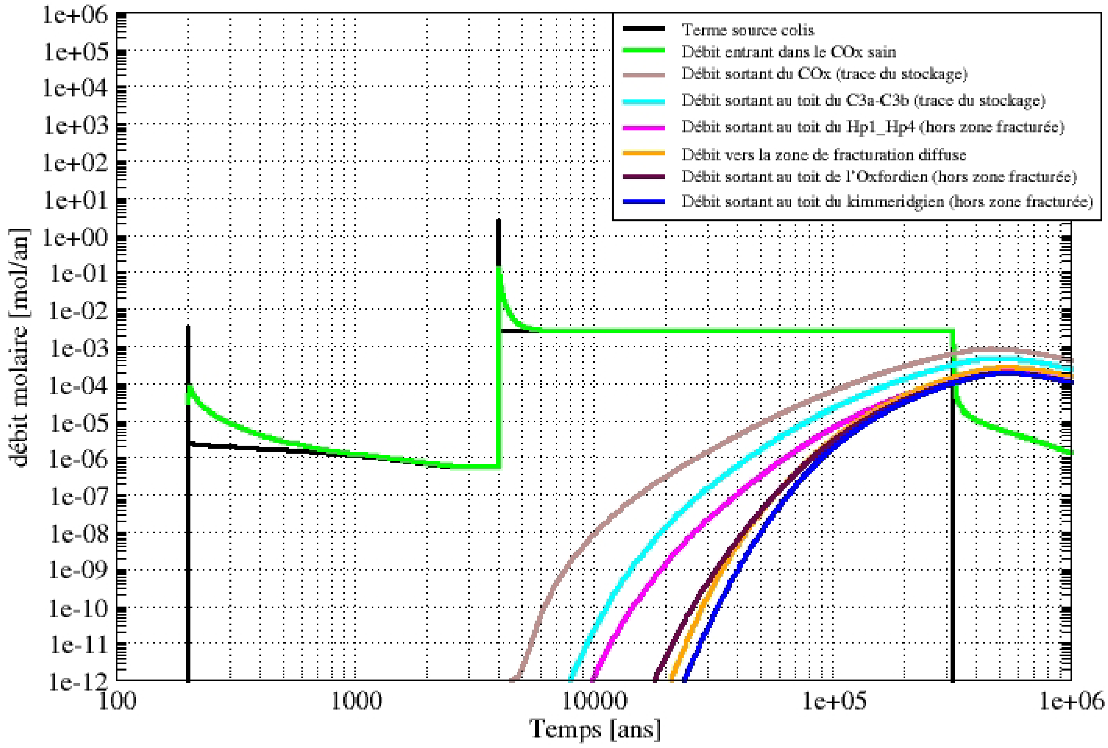


Figure 5.5-56 SEN – Sensitivity to parameters of reference source term for C waste packages (model $V_0.S \rightarrow V_r$) - C1 + C2 reference packages – ^{129}I – history of molar flow [Sensitivity study]

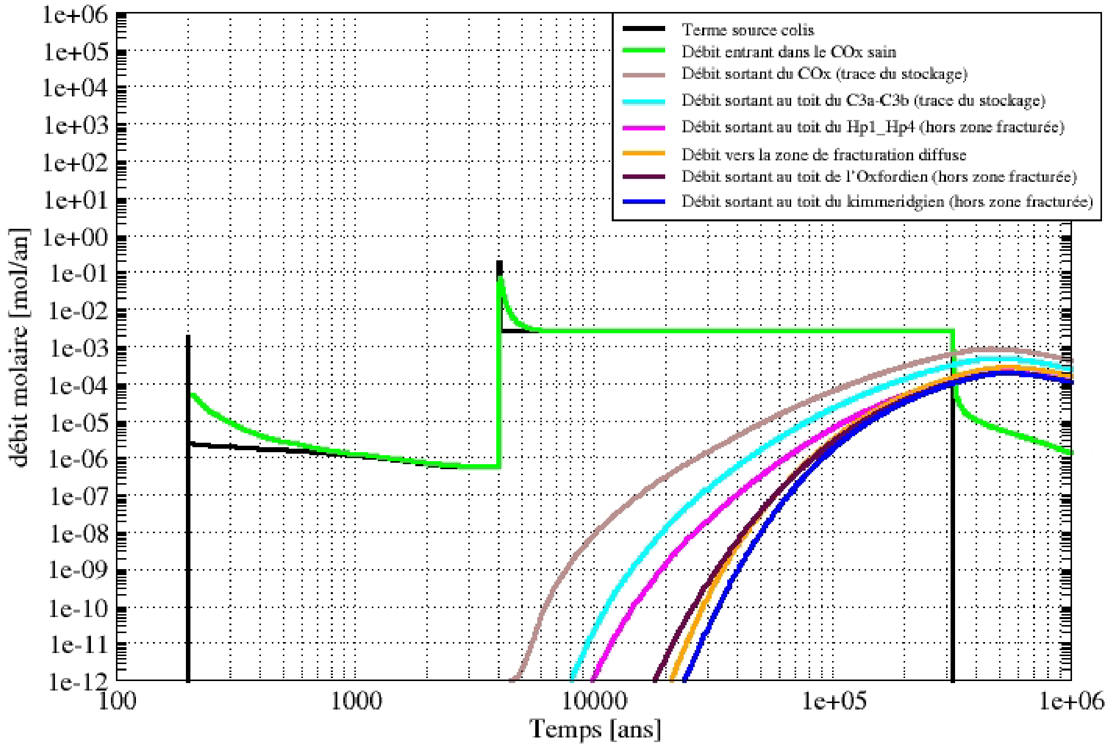


Figure 5.5-56 SEN – Sensitivity to parameters of reference source term for C waste packages (model $V_0.S \rightarrow V_r$) - C1 + C2 reference packages – ^{129}I – history of molar flow [Reference calculation]

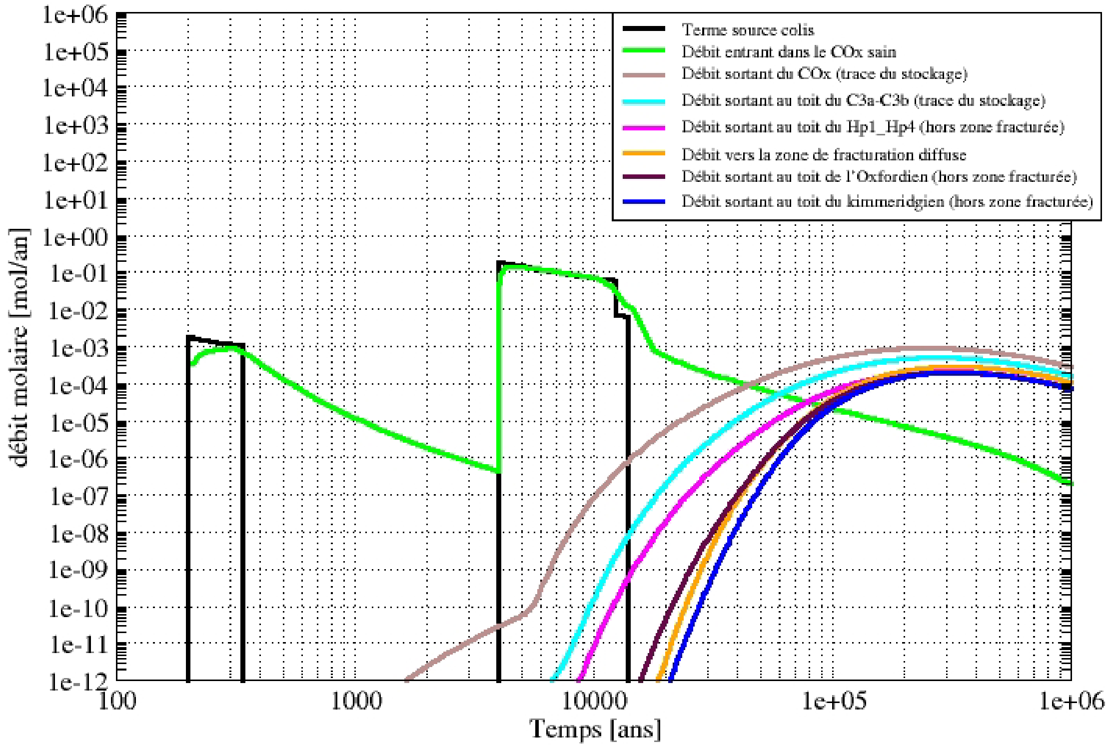


Figure 5.5-57 SEN – Sensitivity to release model of C waste (model $V_0.S$) - C1 + C2 reference package – ^{129}I – History of molar flow [Sensitivity study]

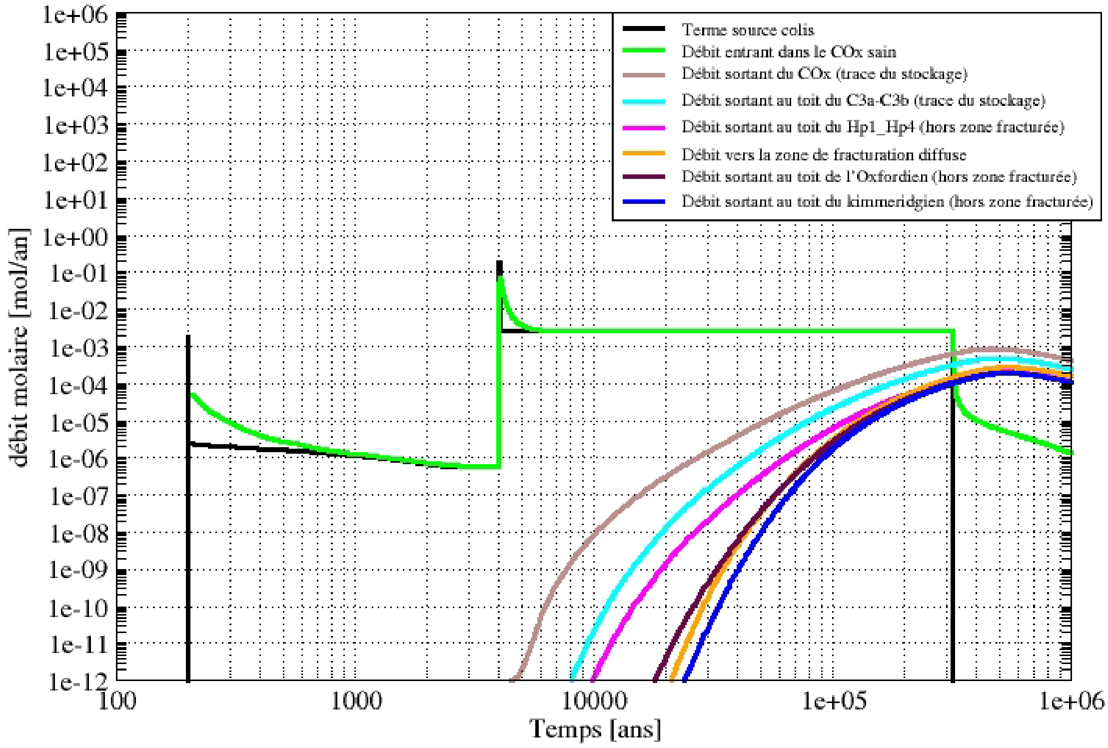


Figure 5.5-57 SEN – Sensitivity to release model of C waste (model $V_0.S$) - C1 + C2 reference package – ^{129}I – History of molar flow[Reference calculation]

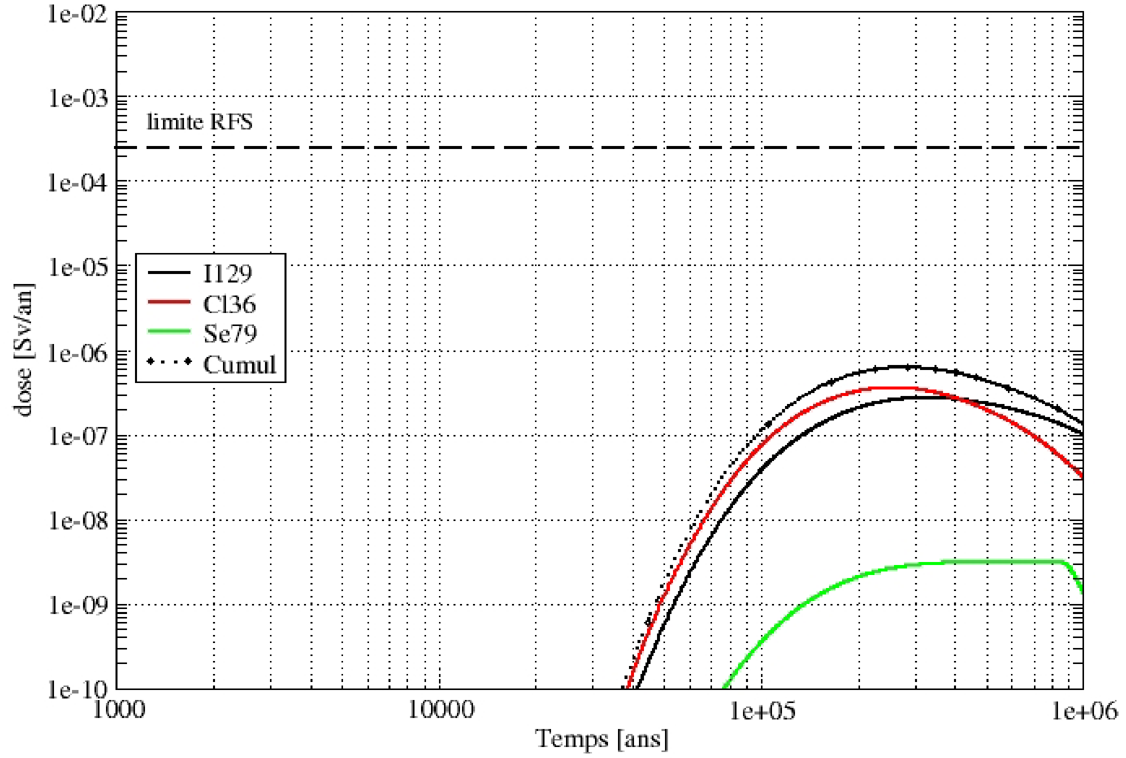


Figure 5.5-58 SEN – Sensitivity to release model of C waste (model V_0.S)– Dose at Saulx outlet of Oxfordian - C1 + C2 reference package [Sensitivity study]

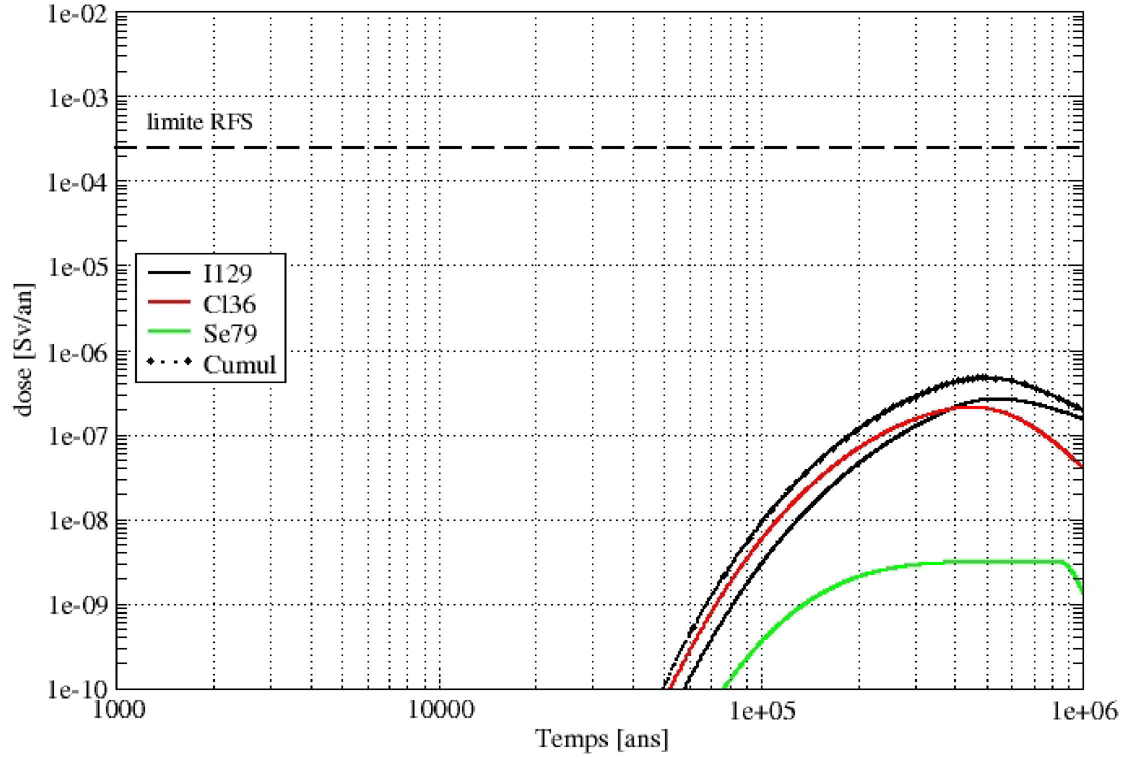


Figure 5.5-58 SEN – Sensitivity to release model of C waste (model V_0.S)– Dose at Saulx outlet of Oxfordian - C1 + C2 reference package [Reference calculation]

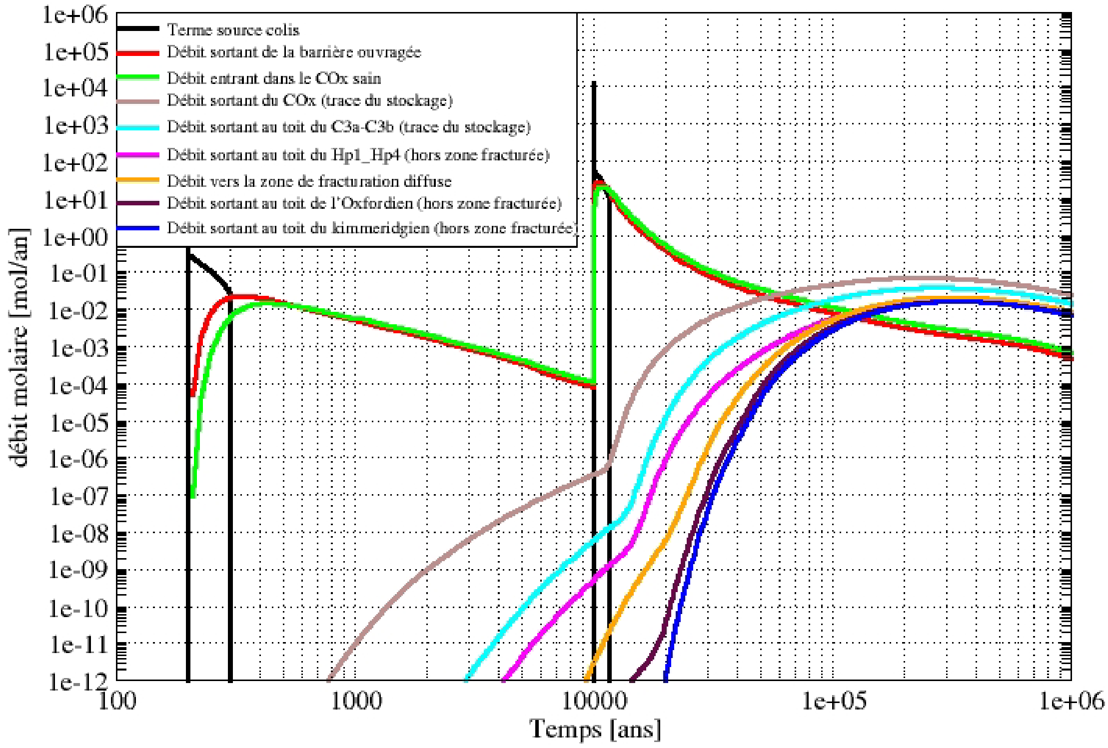


Figure 5.5-59 SEN – Sensitivity of spent fuel packages to parameters of the reference release model (conservative values) - CU1 reference package – ^{129}I - history of molar flow – [Sensitivity study]

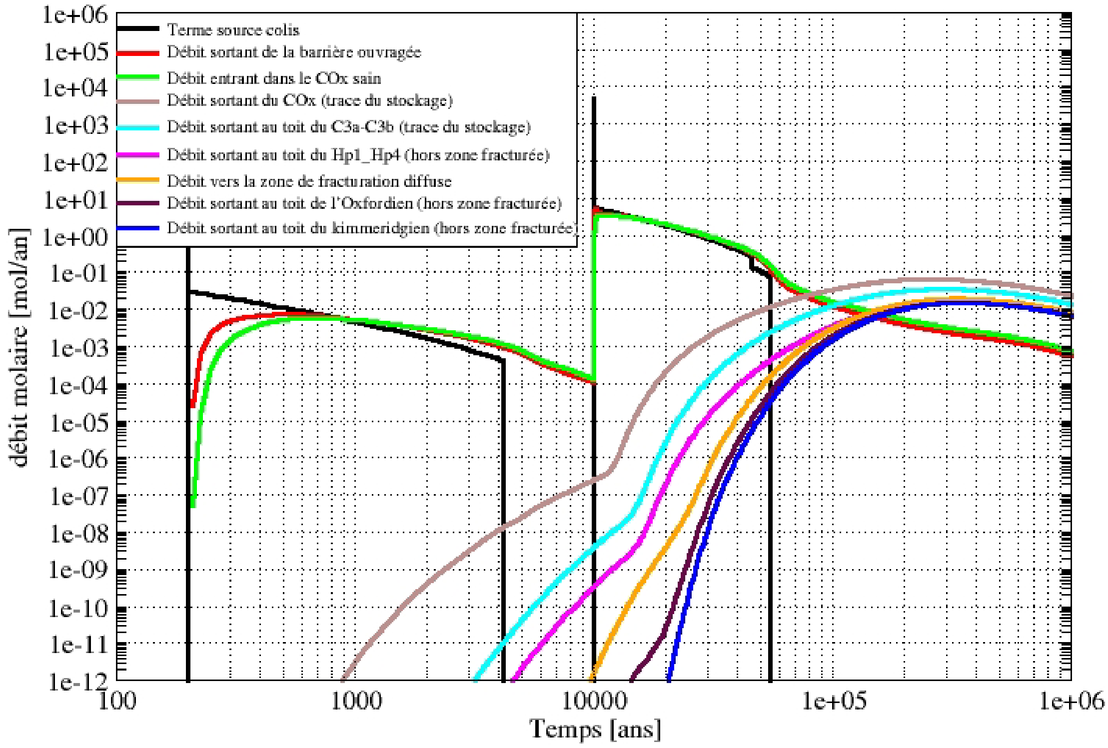


Figure 5.5-59 SEN – Sensitivity of spent fuel packages to parameters of the reference release model (conservative values) - CU1 reference package – ^{129}I - history of molar flow – [Reference Calculation]

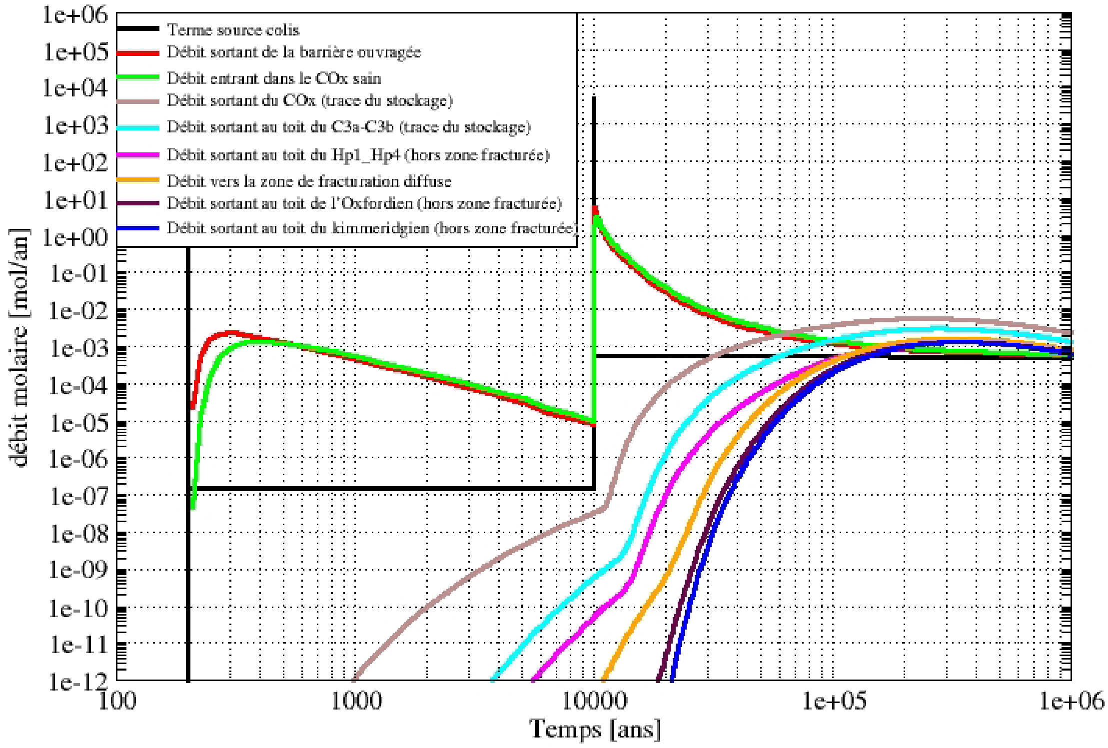


Figure 5.5-60 SEN – Sensitivity of spent fuel packages to release model (monitoring by uranium solubility) - CU1 reference package – ^{129}I – history of molar flow [Sensitivity study]

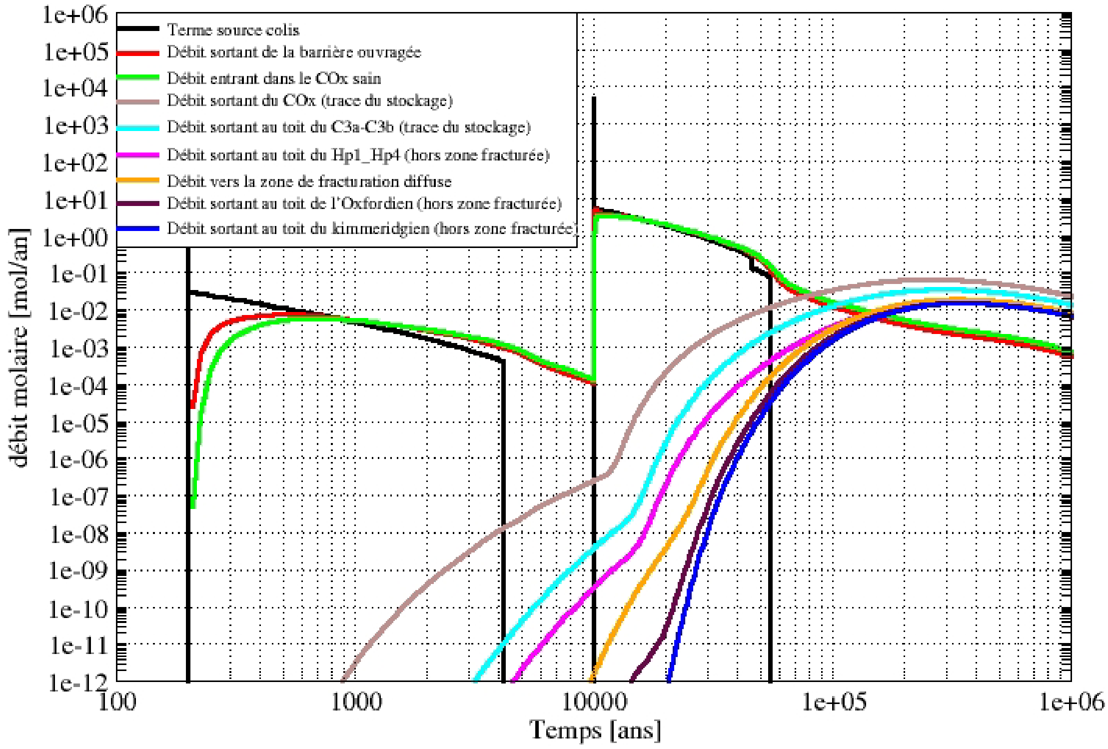


Figure 5.5-60 SEN – Sensitivity of spent fuel packages to release model (monitoring by uranium solubility) - CU1 reference package – ^{129}I – history of molar flow [Reference calculation]

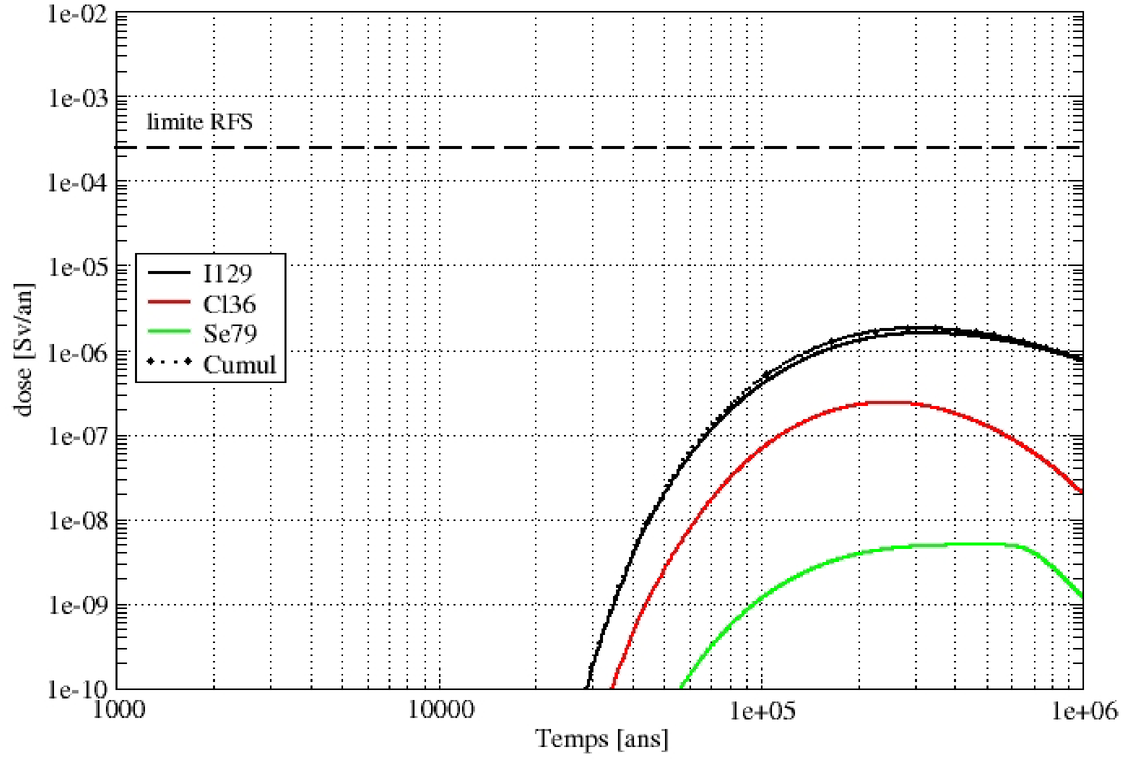


Figure 5.5-61 SEN – Sensitivity of spent fuel packages to release model (controlled by uranium solubility) - CU1 reference package – ^{129}I – Dose at Saulx outlet of Oxfordian [Sensitivity study]

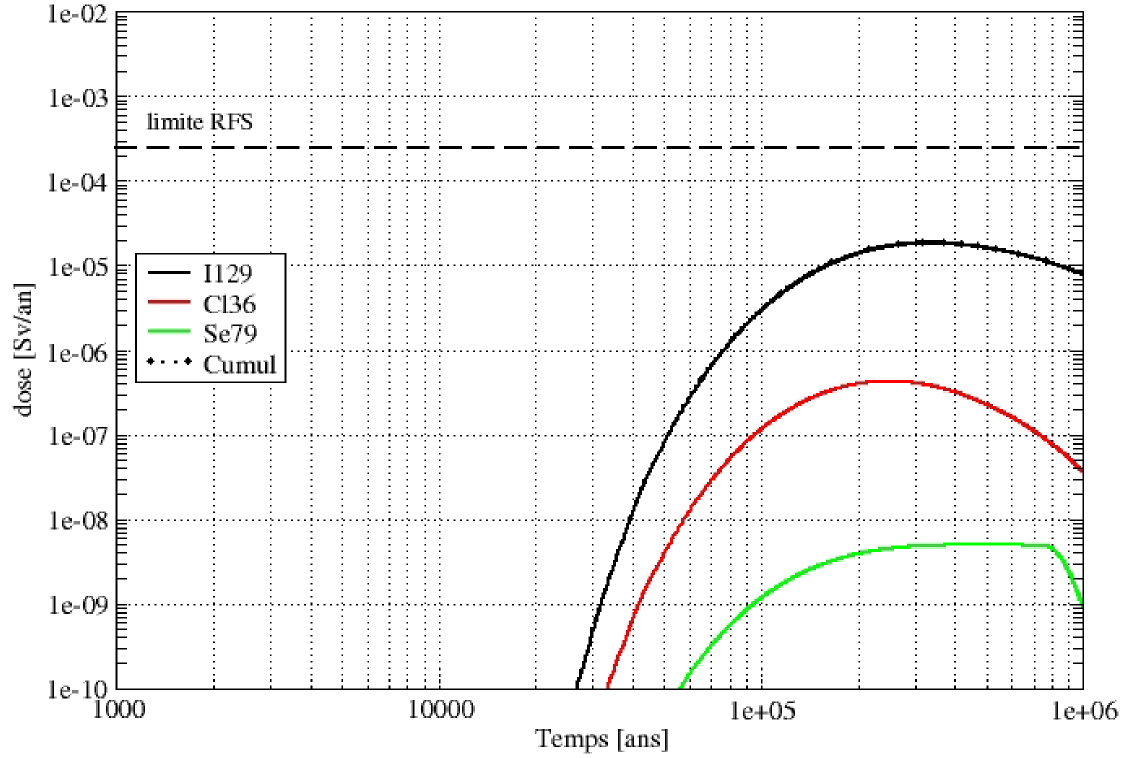


Figure 5.5-61 SEN – Sensitivity of spent fuel packages to release model (controlled by uranium solubility) - CU1 reference package – ^{129}I – Dose at Saulx outlet of Oxfordian [Reference calculation]

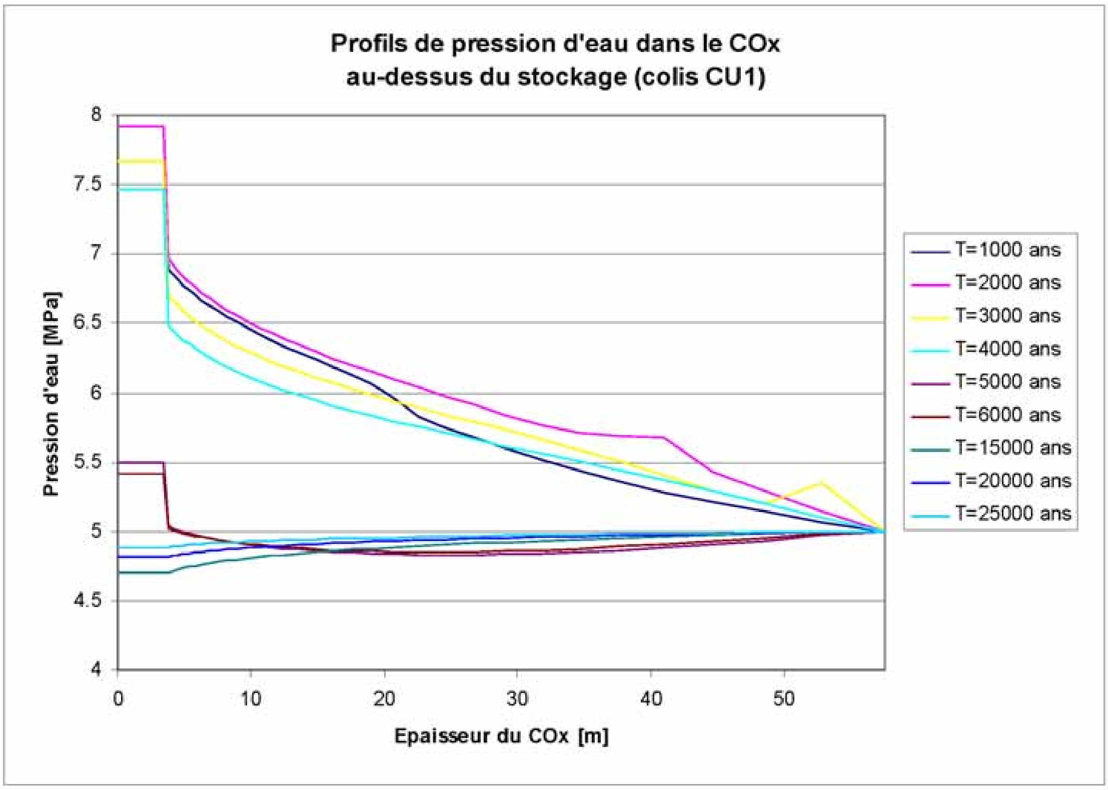


Figure 5.5-62 Water Pressure in the Callovo-Oxfordian (COX) above the Repository – Reference package CU1

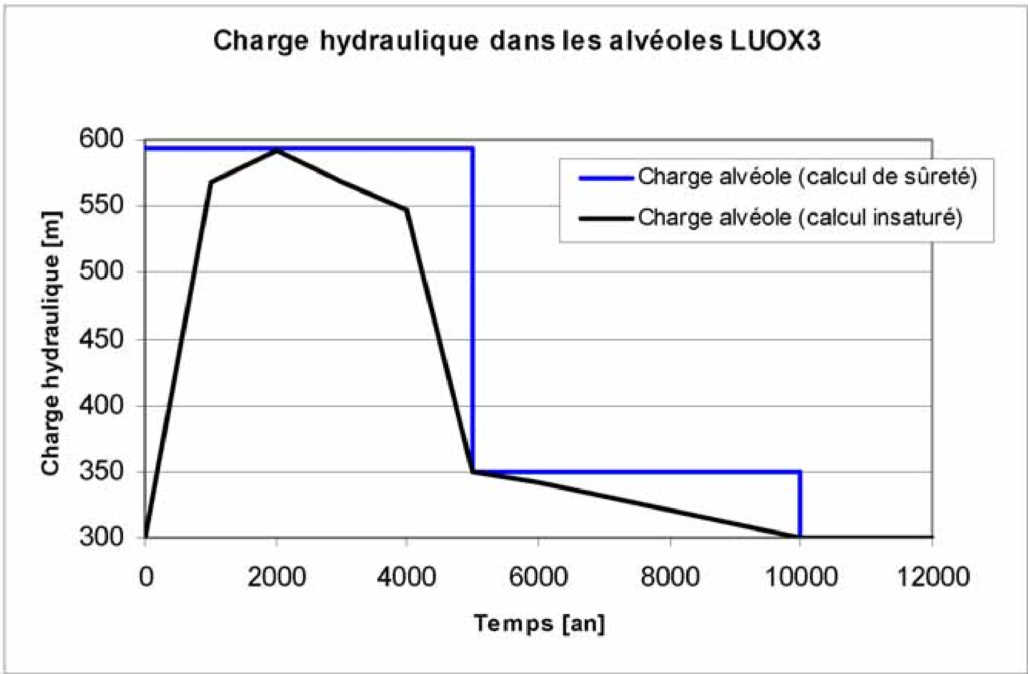


Figure 5.5-63 SEN – Sensitivity calculation on the influence of the hydraulic transient – Hydraulic head imposed during the overpressure phase – Reference package CU1

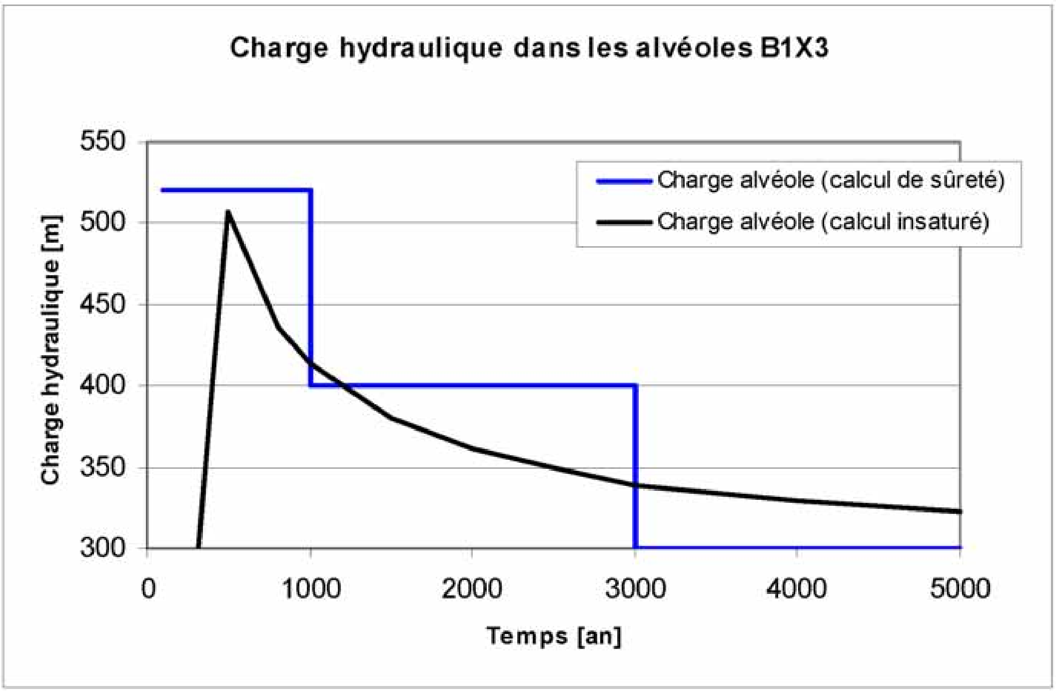


Figure 5.5-64 *SEN – Sensitivity calculation on the influence of the hydraulic transient – Hydraulic head imposed during the overpressure phase – Reference package B1x*

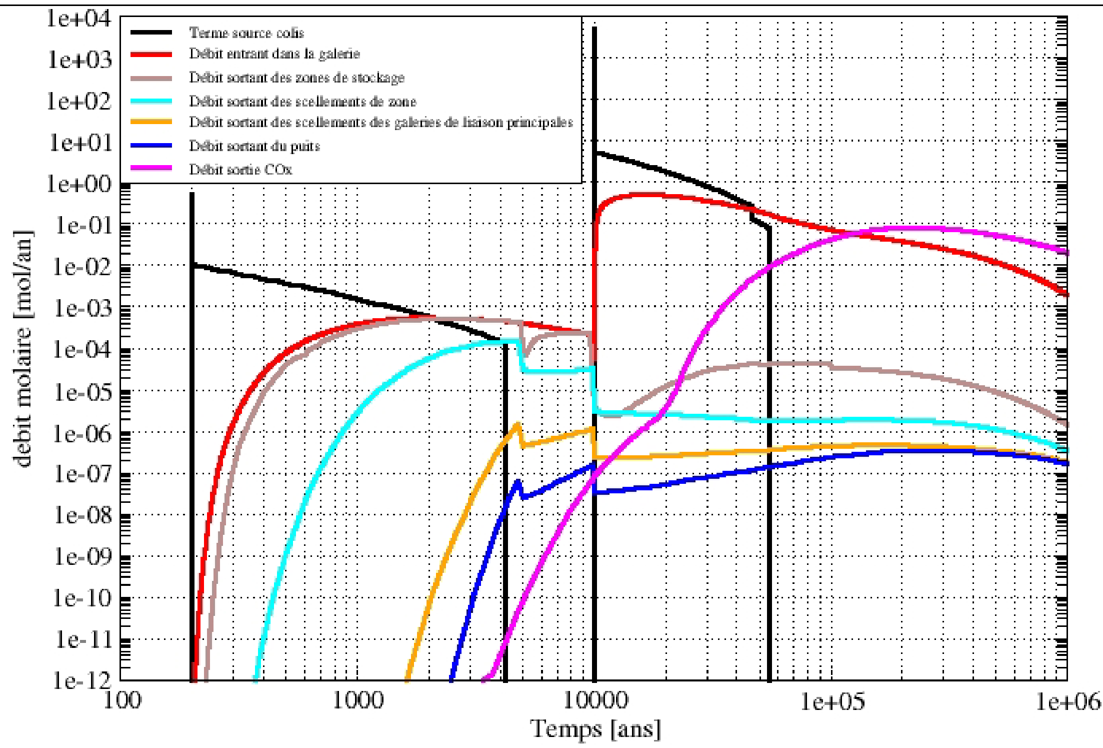


Figure 5.5-66 *Normal Evolution Scenario – Sensitivity calculation on the influence of the hydraulic transient – molar flow rate history – ^{129}I – spent fuel CU1– Sensitivity calculation*

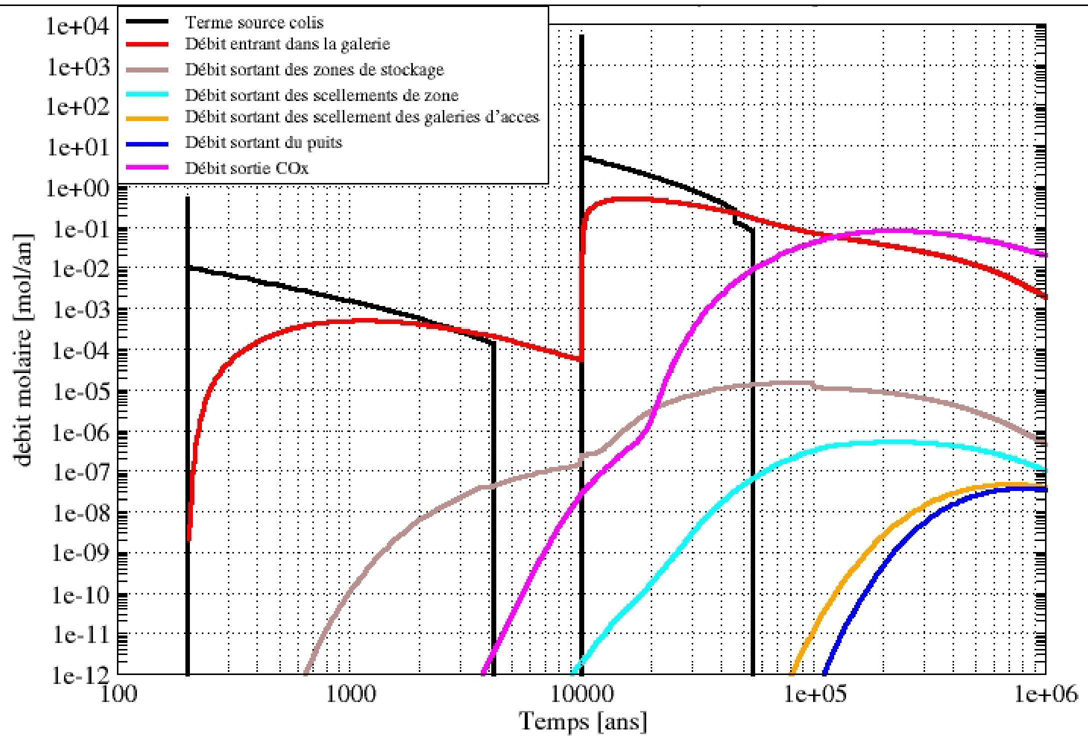


Figure 5.5-66 *Normal Evolution Scenario – Sensitivity calculation on the influence of the hydraulic transient – molar flow rate history – ^{129}I – spent fuel CU1 – reference calculation*

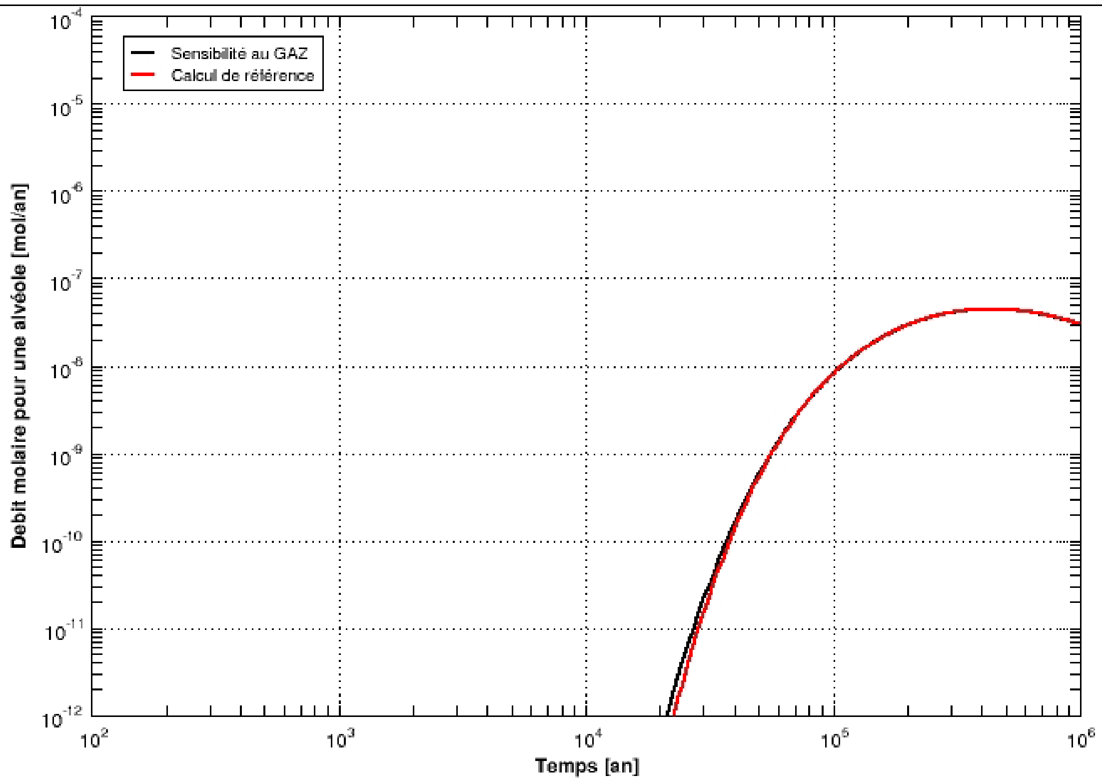


Figure 5.5-67 *Normal Evolution Scenario – sensitivity calculation on the influence of the hydraulic transient – historical molar flow rates out of Callovo-Oxfordian for a B1x waste cell - comparison of cases with and without hydraulic overpressures – ^{129}I*

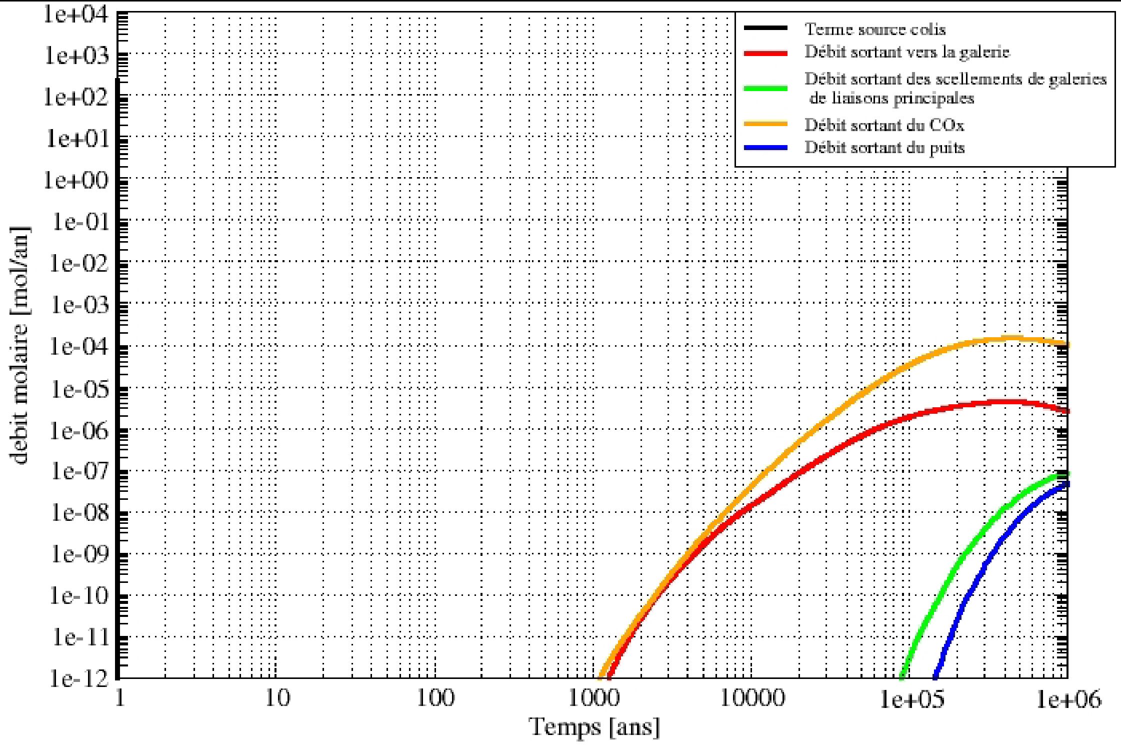


Figure 5.5-69 Normal Evolution Scenario – Sensitivity calculation on the influence of the hydraulic transient – molar flow rate history – ^{129}I – Non-organic reference packages not releasing gaseous hydrogen. – Sensitivity calculation

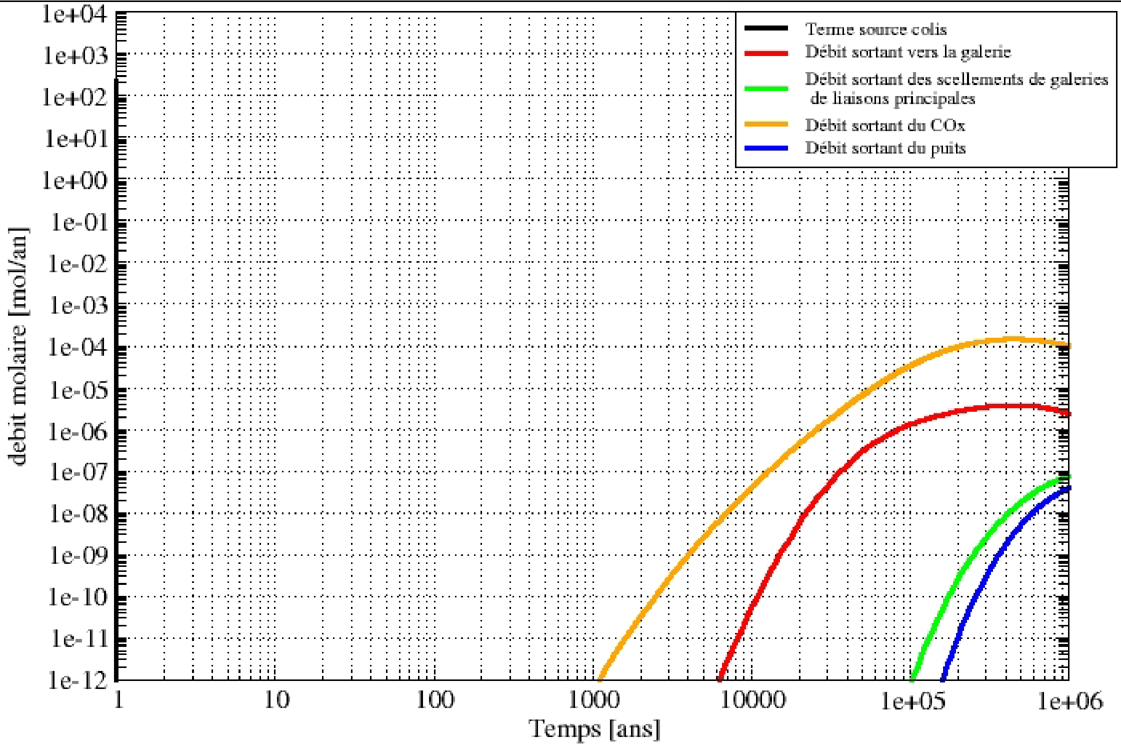


Figure 5.5-69 Normal Evolution Scenario – Sensitivity calculation on the influence of the hydraulic transient – molar flow rate history – ^{129}I – Non-organic reference packages not releasing gaseous hydrogen. – Sensitivity calculation

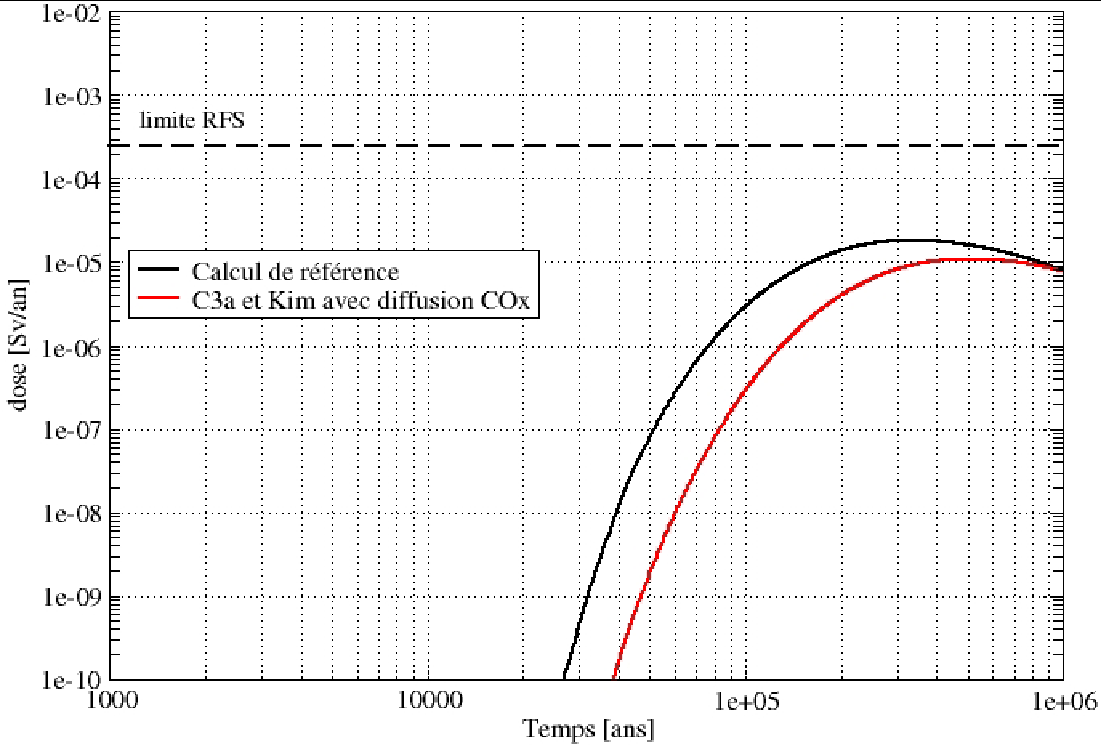


Figure 5.5-70 Normal Evolution Scenario - Sensitivity to the diffusive properties of semi-permeable layers in the overlying surrounding rocks - Reference package CU1 – ^{129}I -Dose at the Saulx and Ornain outlet of the Oxfordian – Reference calculation comparison / Sensitivity study [Saulx Outlet]

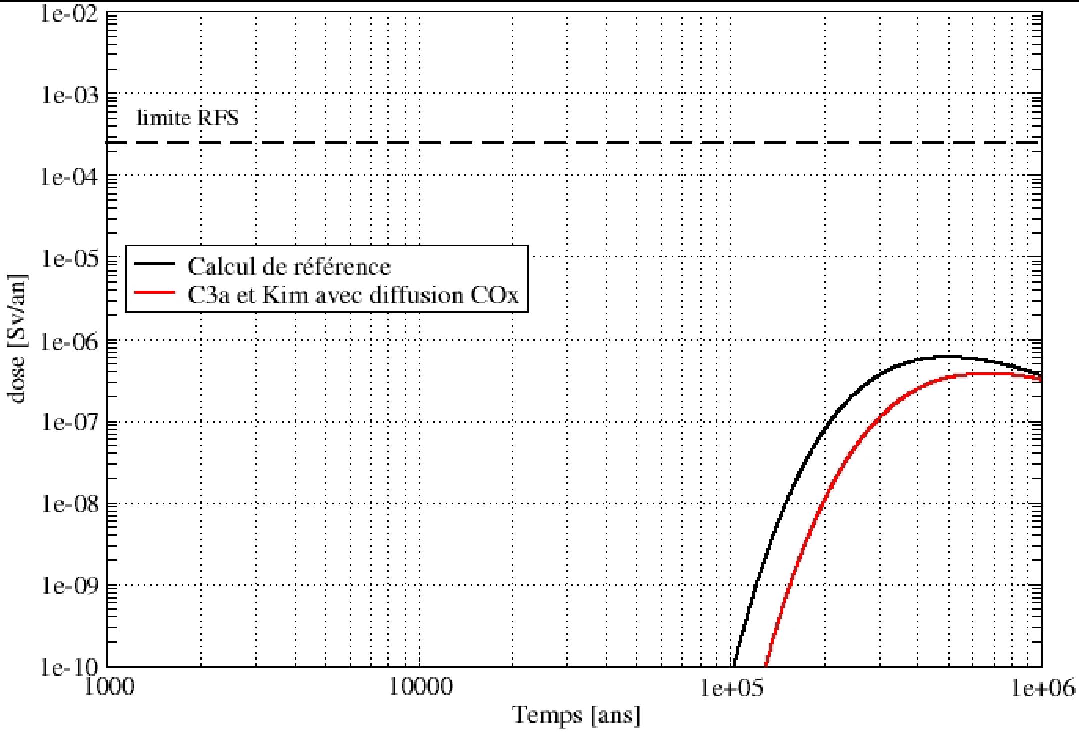


Figure 5.5-70 Normal Evolution Scenario - Sensitivity to the diffusive properties of semi-permeable layers in the overlying surrounding rocks - Reference package CU1 – ^{129}I -Dose at the Saulx and Ornain outlet of the Oxfordian – Reference calculation comparison / Sensitivity study [Ornain Outlet]

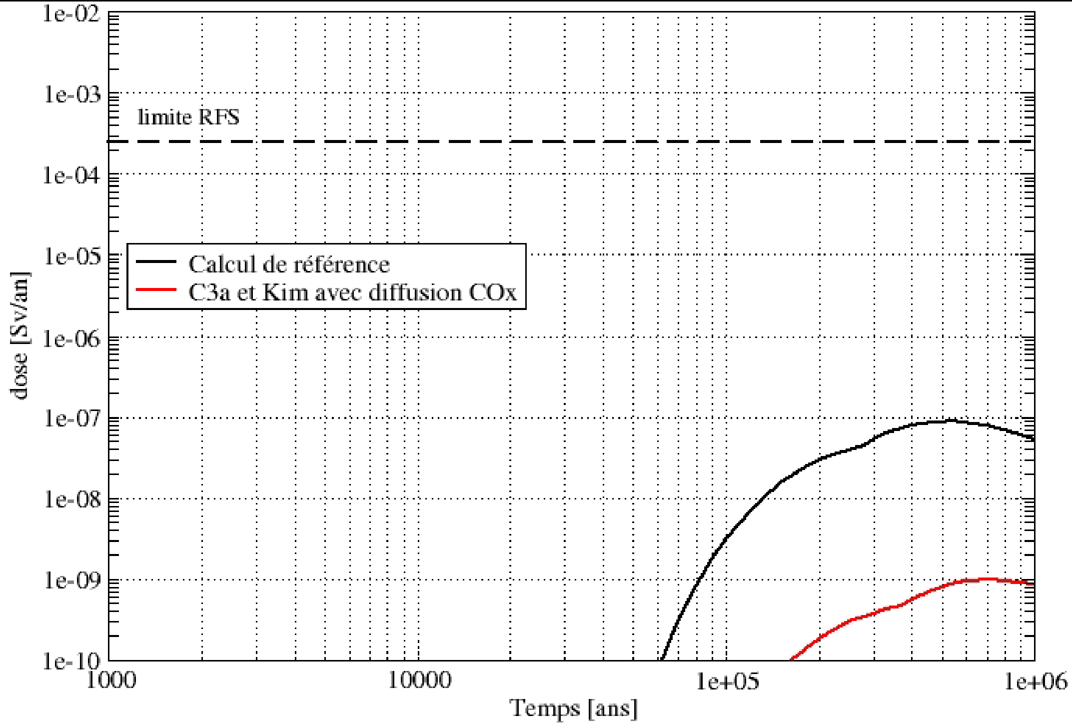


Figure 5.5-71 *Normal Evolution Scenario - Sensitivity to the diffusive properties of semi-permeable layers in the overlying surrounding rocks - Reference package CU1 – ^{129}I -Dose at the Barrois – Reference calculation comparison / Sensitivity study (H= hydraulic loads in NGF meters)*

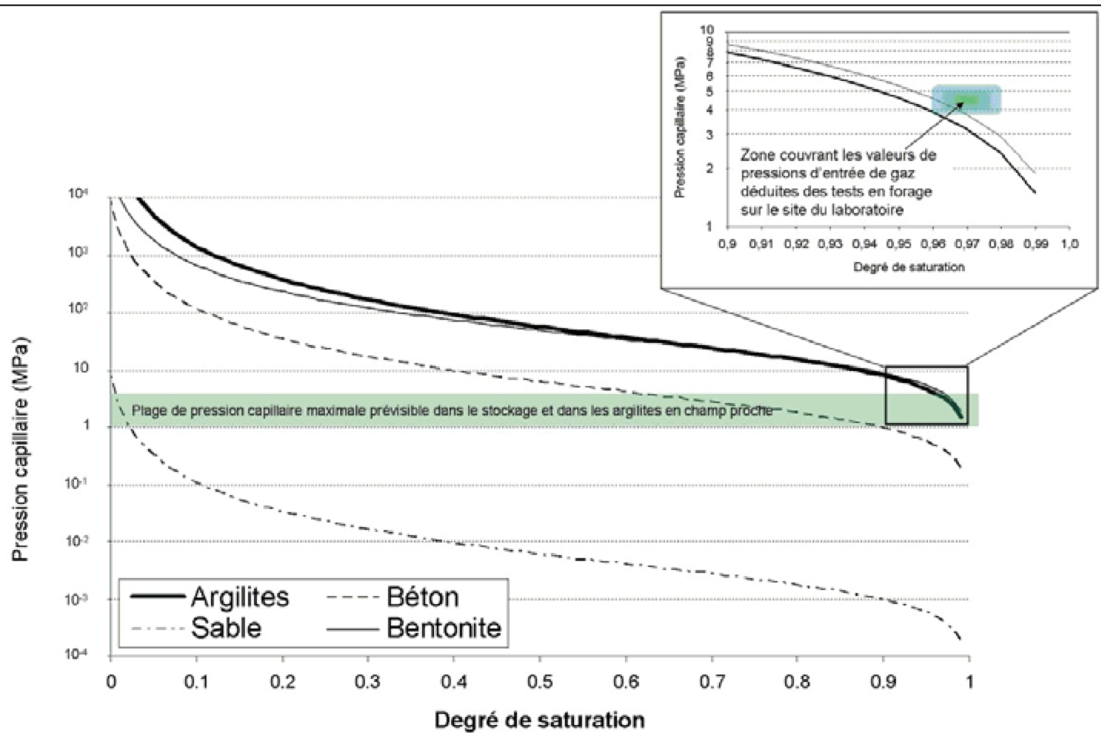


Figure 6.2-7 *Suction in various porous media versus water saturation level*

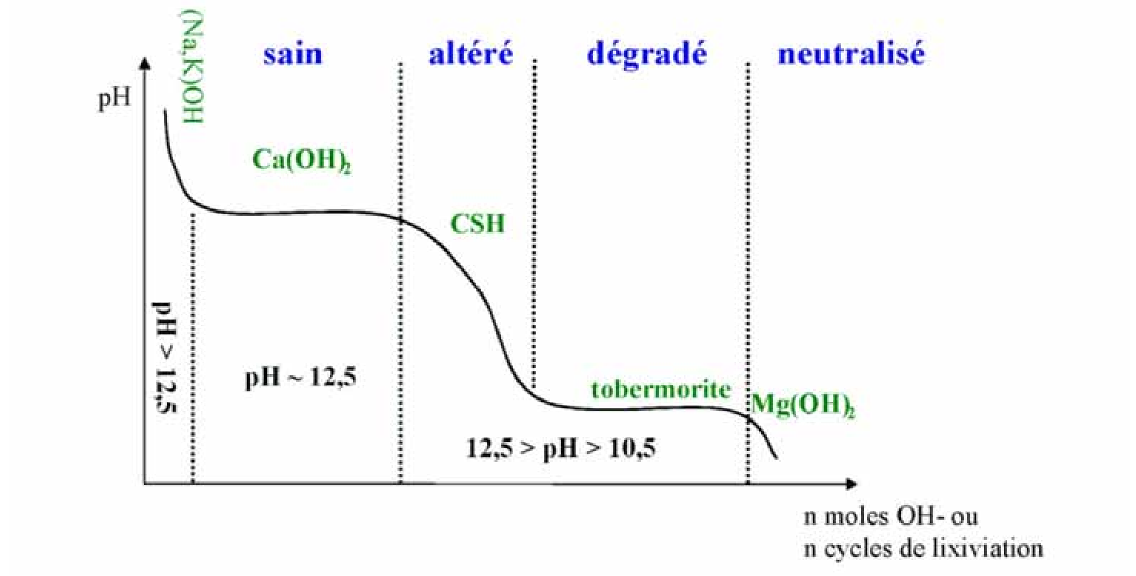


Figure 6.2-13 Evolution of the chemical state of cement-based materials

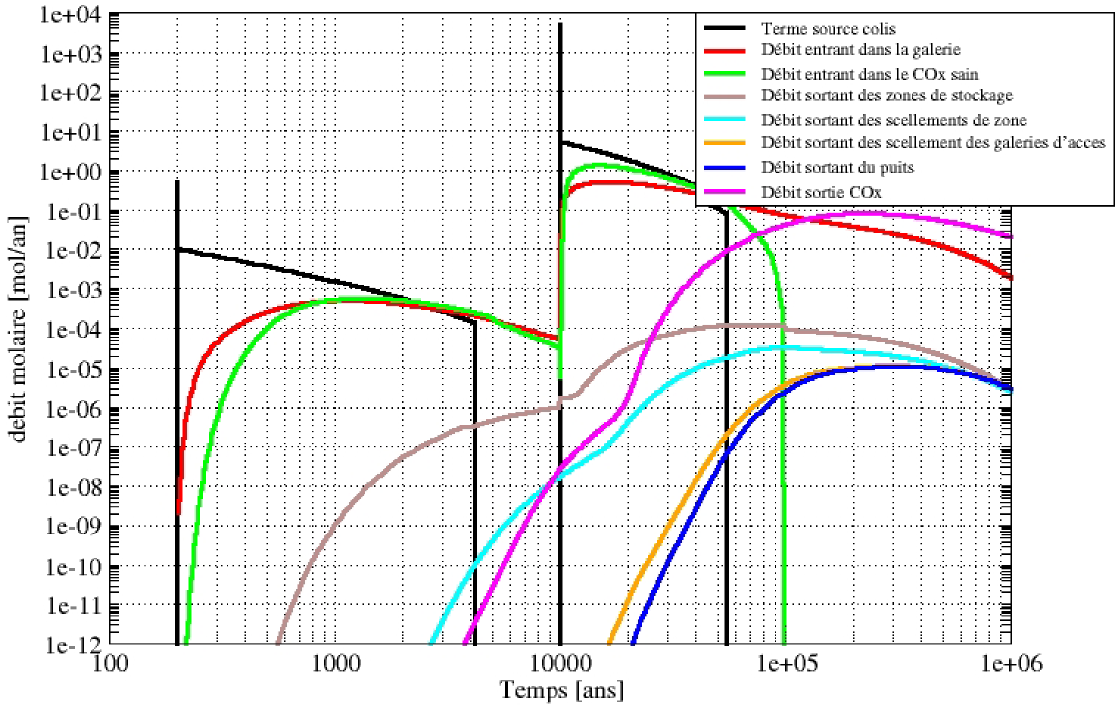


Figure 7.2-16 'Seal failure' altered evolution scenario – All failure situations – Molar flow rate history – Reference calculation - CU1 reference package - ^{129}I - [All seals defective]

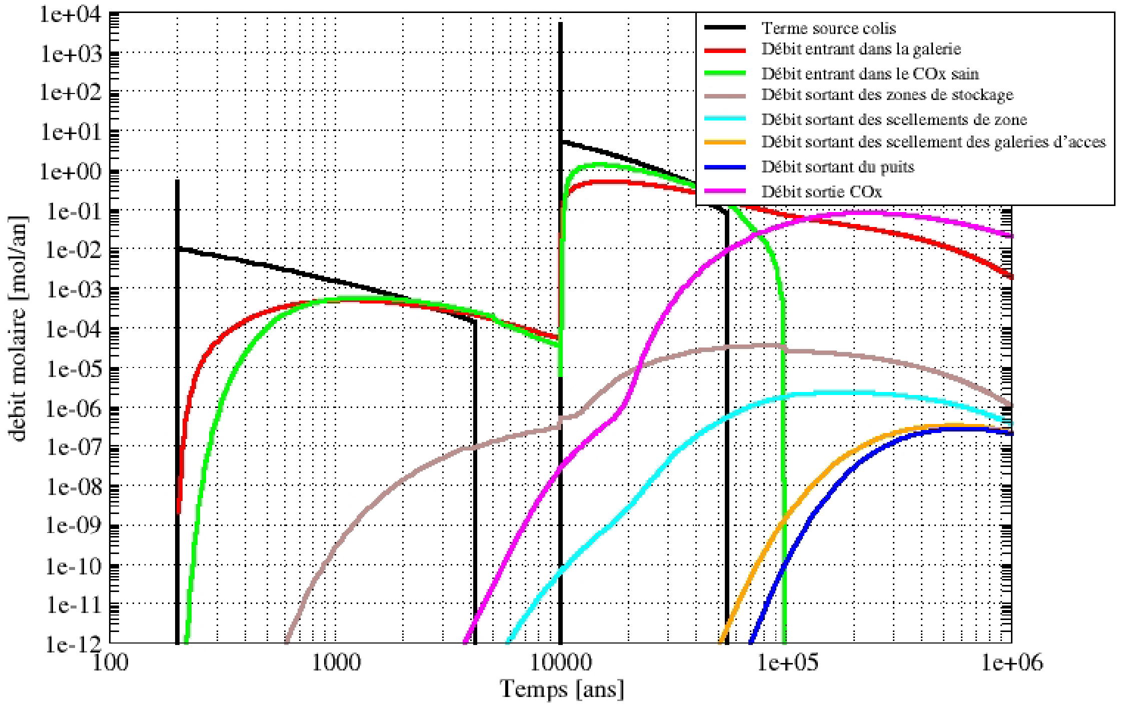


Figure 7.2-16 'Seal failure' altered evolution scenario – All failure situations – Molar flow rate history – Reference calculation - CU1 reference package - ^{129}I - [Drift seals defective]

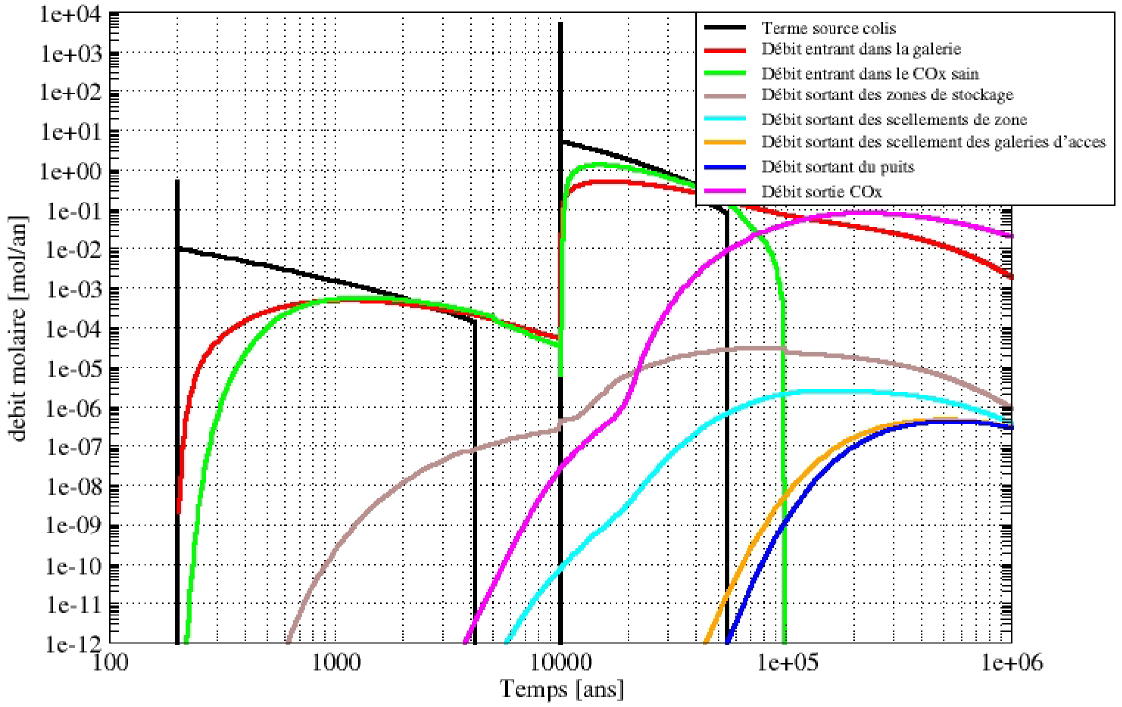


Figure 7.2-16 'Seal failure' altered evolution scenario – All failure situations – Molar flow rate history – Reference calculation - CU1 reference package - ^{129}I - [Shaft seals defective]

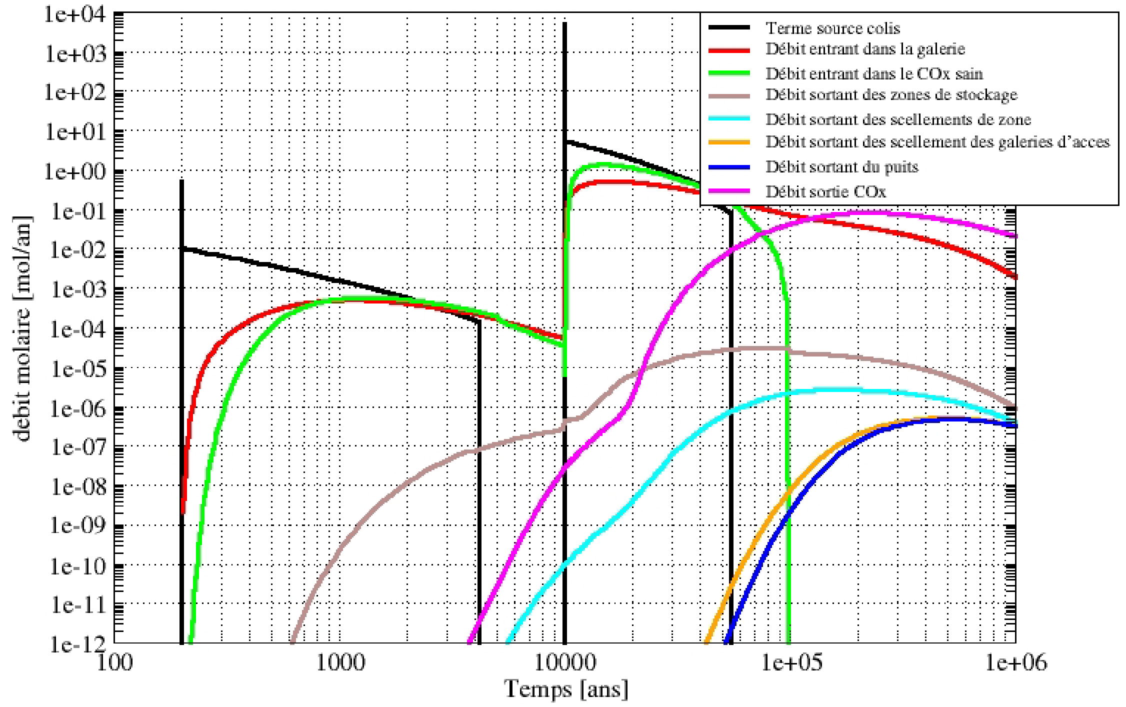


Figure 7.2-16 'Seal failure' altered evolution scenario – All failure situations – Molar flow rate history – Reference calculation - CU1 reference package - ^{129}I – [Shaft seals desertion]

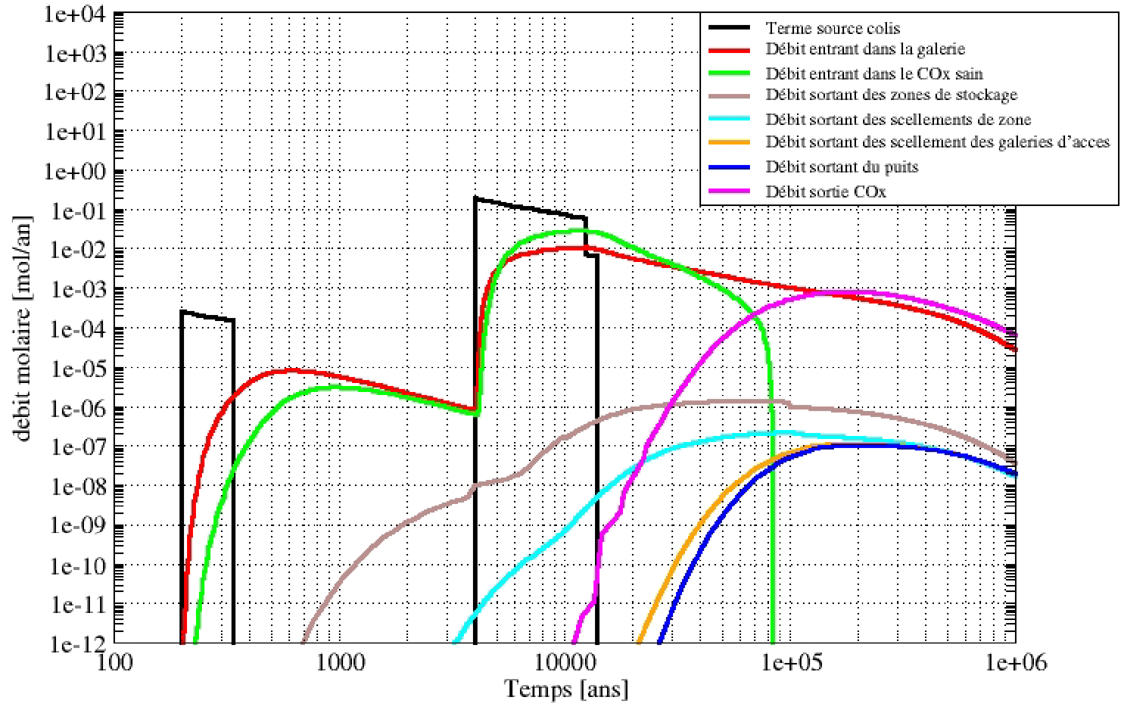


Figure 7.2-18 Seal failure altered evolution scenario – All seals defective' situation – Molar flow rate history – Reference calculation – C1/C2 reference package - ^{129}I - [All seals defective]

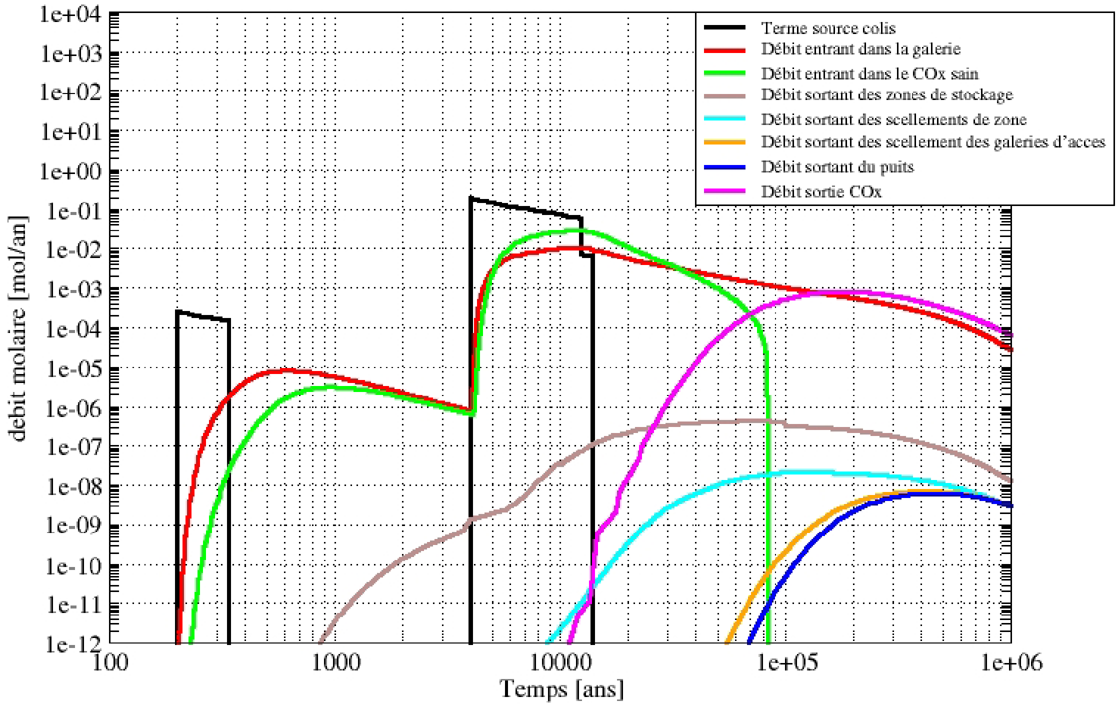


Figure 7.2-18 Seal failure altered evolution scenario – All seals defective' situation – Molar flow rate history – Reference calculation – C1/C2 reference package - ^{129}I - [Shaft seals defective]

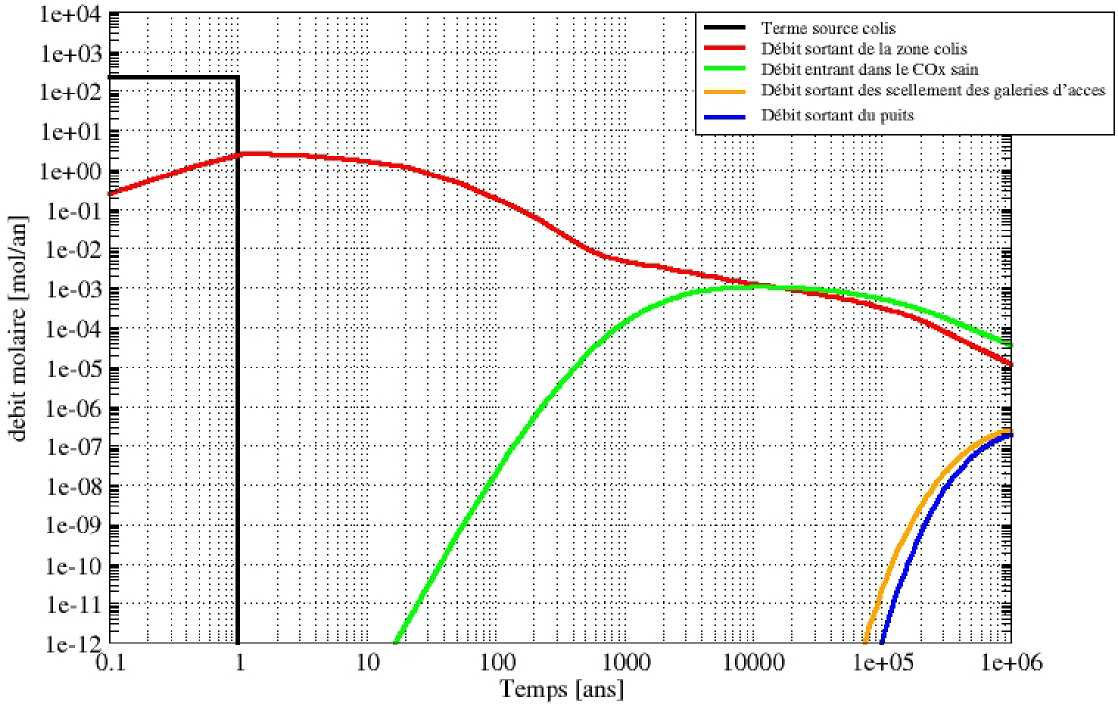


Figure 7.2-20 Seal failure altered evolution scenario – All seals defective - Reference calculation – Molar flow rate history – Non-organic packages not releasing hydrogen - ^{129}I and ^{36}Cl - [^{129}I]

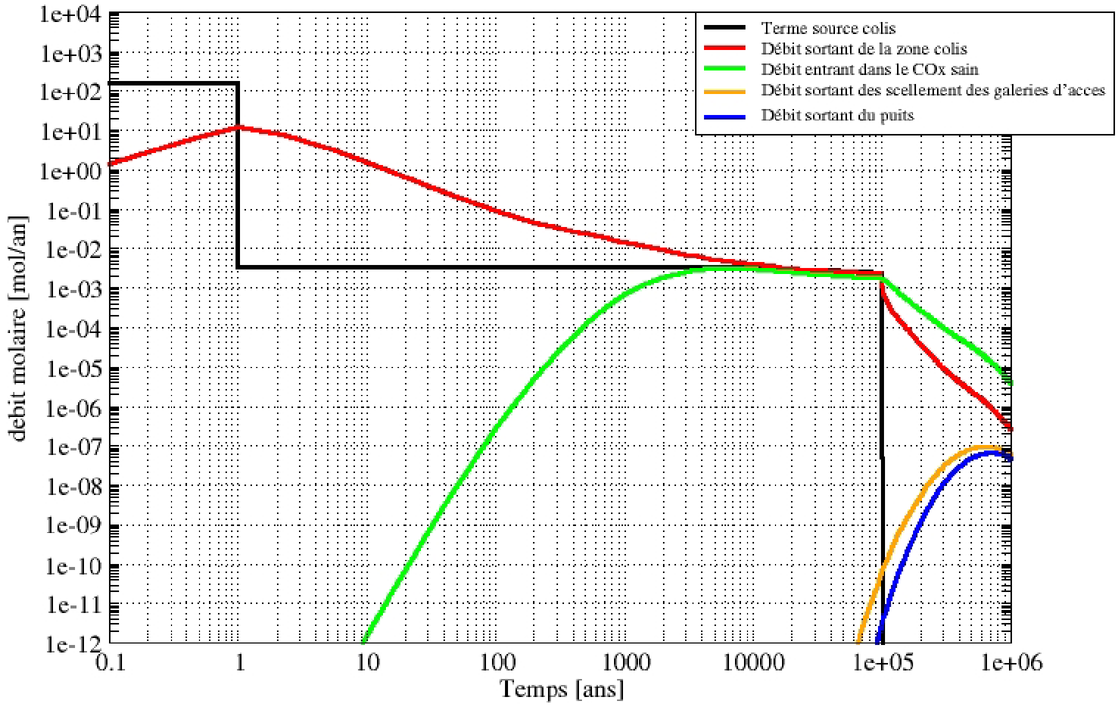


Figure 7.2-20 *Seal failure altered evolution scenario – All seals defective - Reference calculation – Molar flow rate history – Non-organic packages not releasing hydrogen - ^{129}I and ^{36}Cl - [^{36}Cl]*

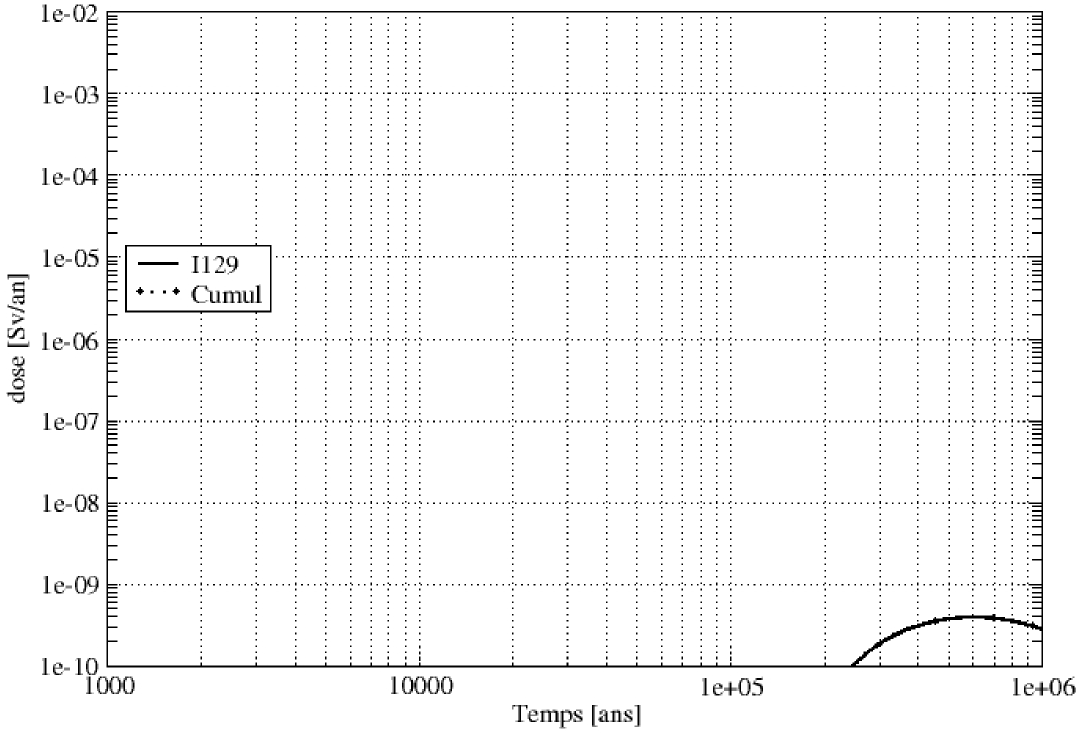


Figure 7.2-21 *Seal failure altered evolution scenario – Shaft seals defective – Reference calculation – Dose rate history for the transfer pathway through the engineered structures – CU1 reference package*

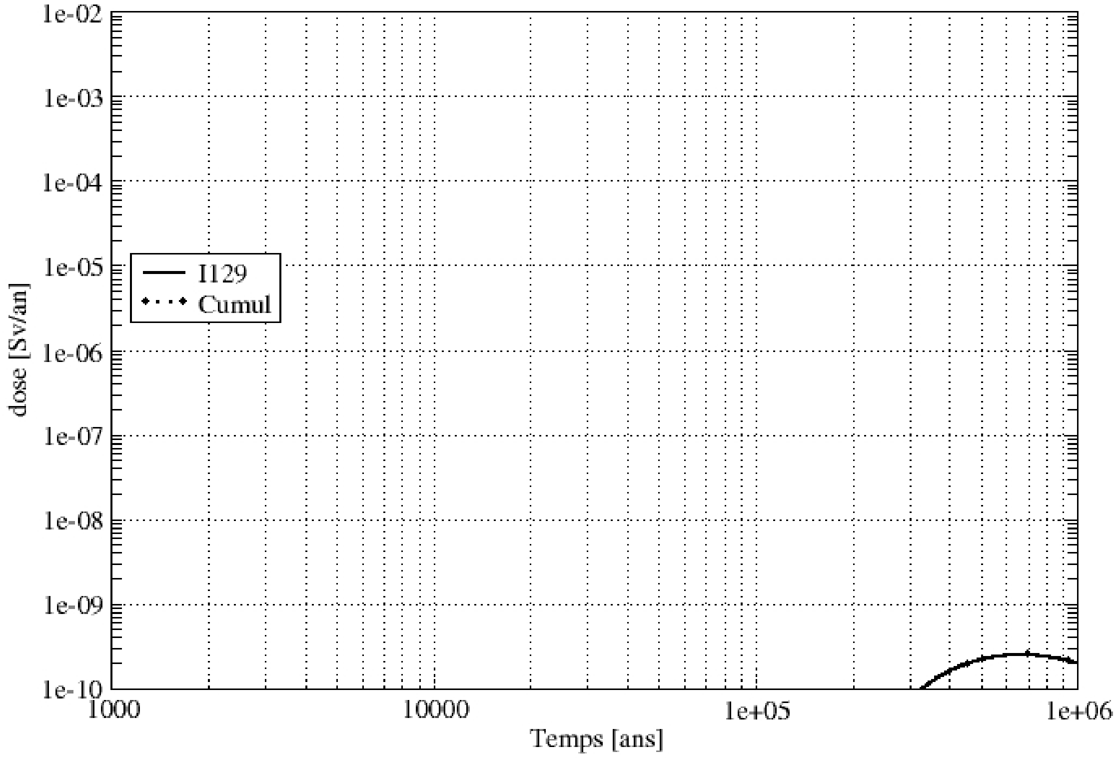


Figure 7.2-22 *Seal failure altered evolution scenario – Drift seals defective – Reference calculation – Dose rate history for the transfer pathway through the engineered structures – CU1 reference packages*

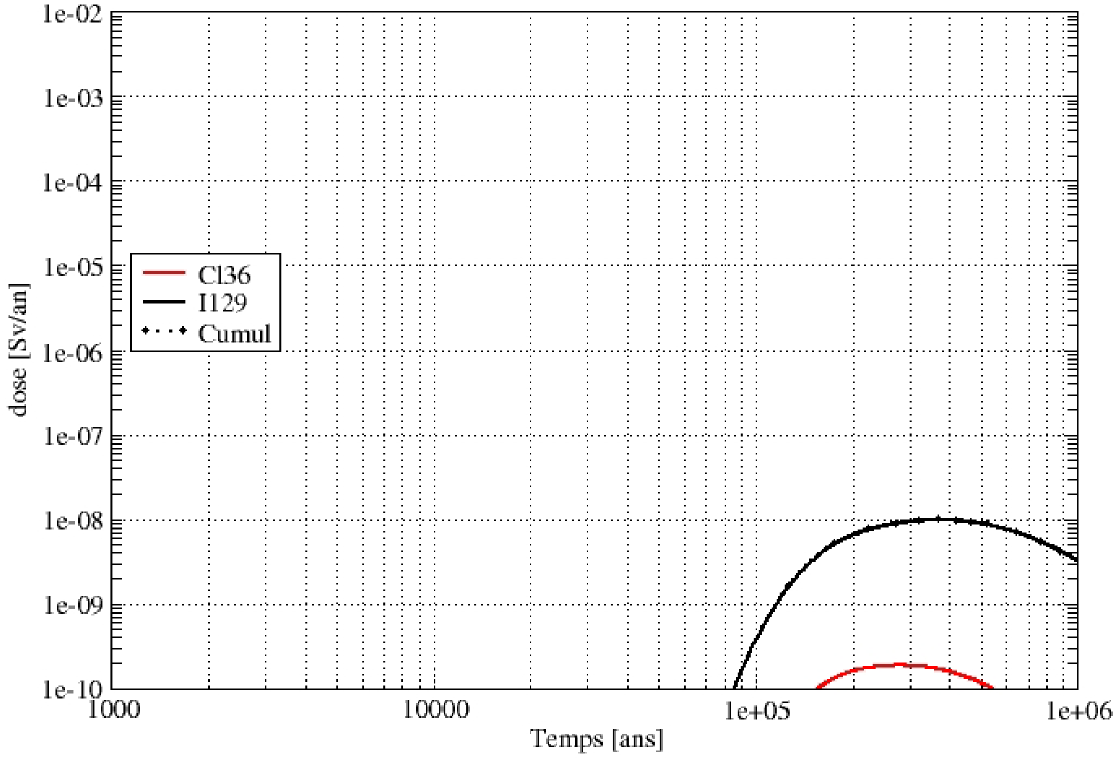


Figure 7.2-23 *Seal failure altered evolution scenario – All seals defective – Reference calculation – Dose rate history for the transfer pathway through the engineered structures – CU1 reference package*

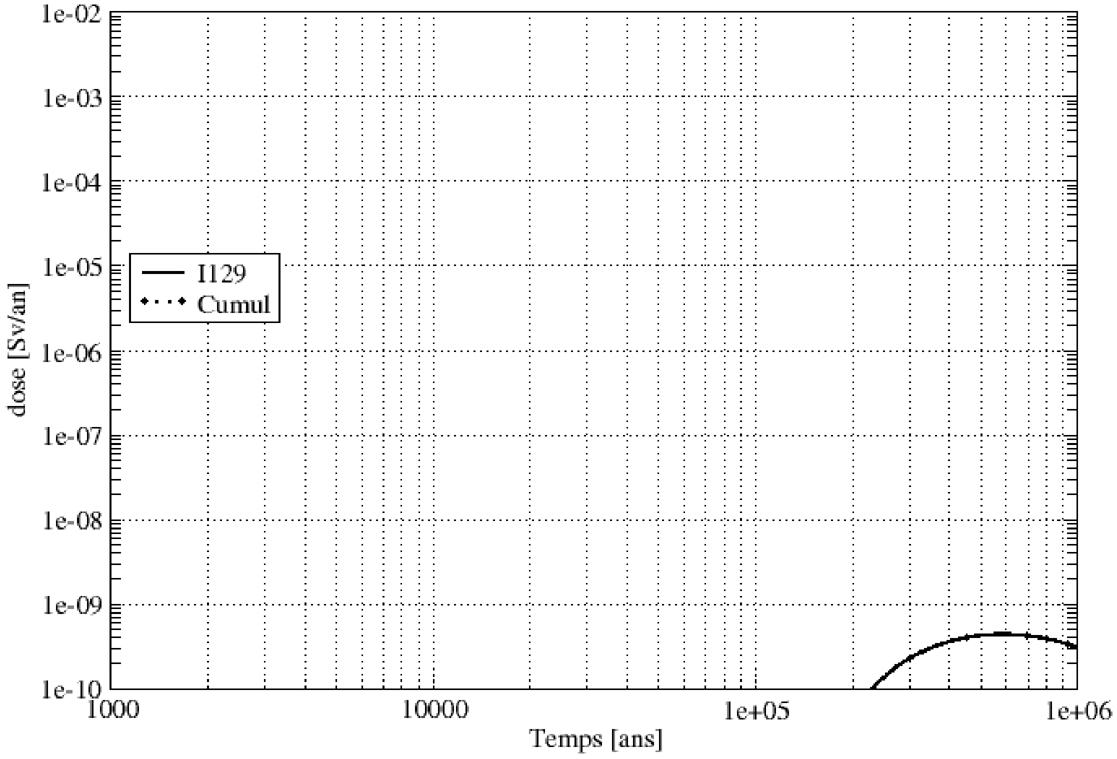


Figure 7.2-24 *'Seal failure altered evolution scenario – Abandonment of shaft seal – Reference calculation – Dose rate history for the transfer pathway through the engineered structures – CU1 reference package*

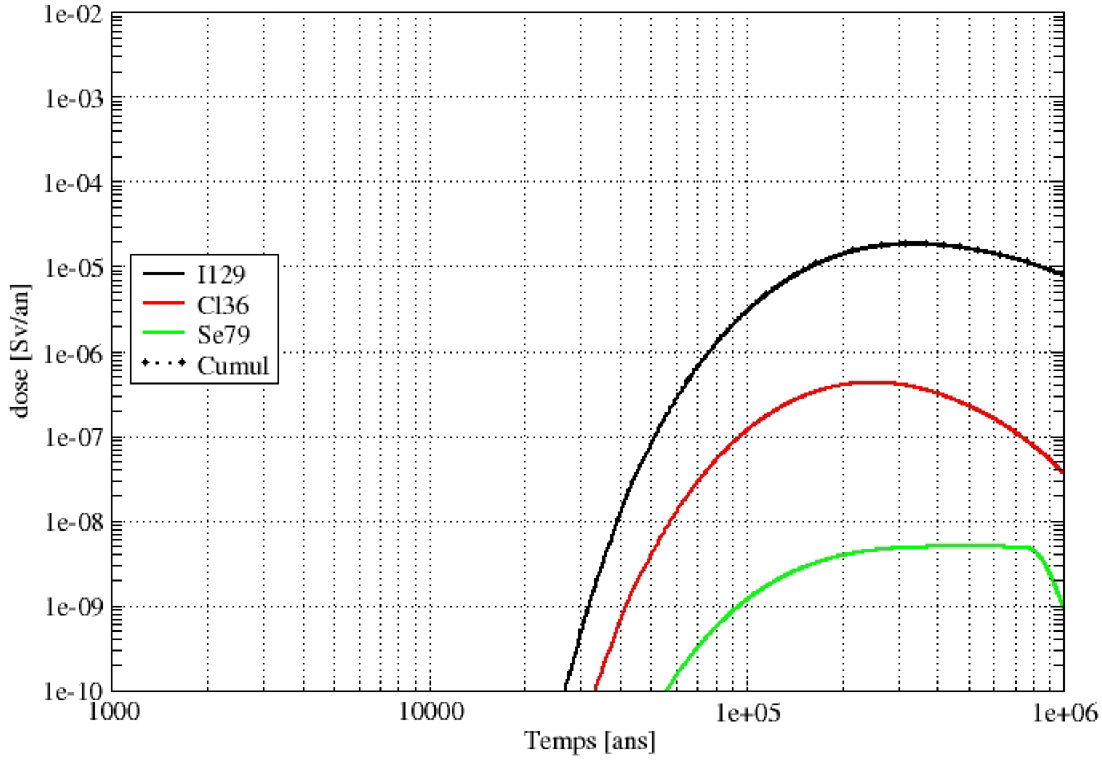


Figure 7.2-25 *Seal failure altered evolution scenario – All situations – Reference calculation – Dose rate history for the transfer pathway through the host formation – CU1 reference package*

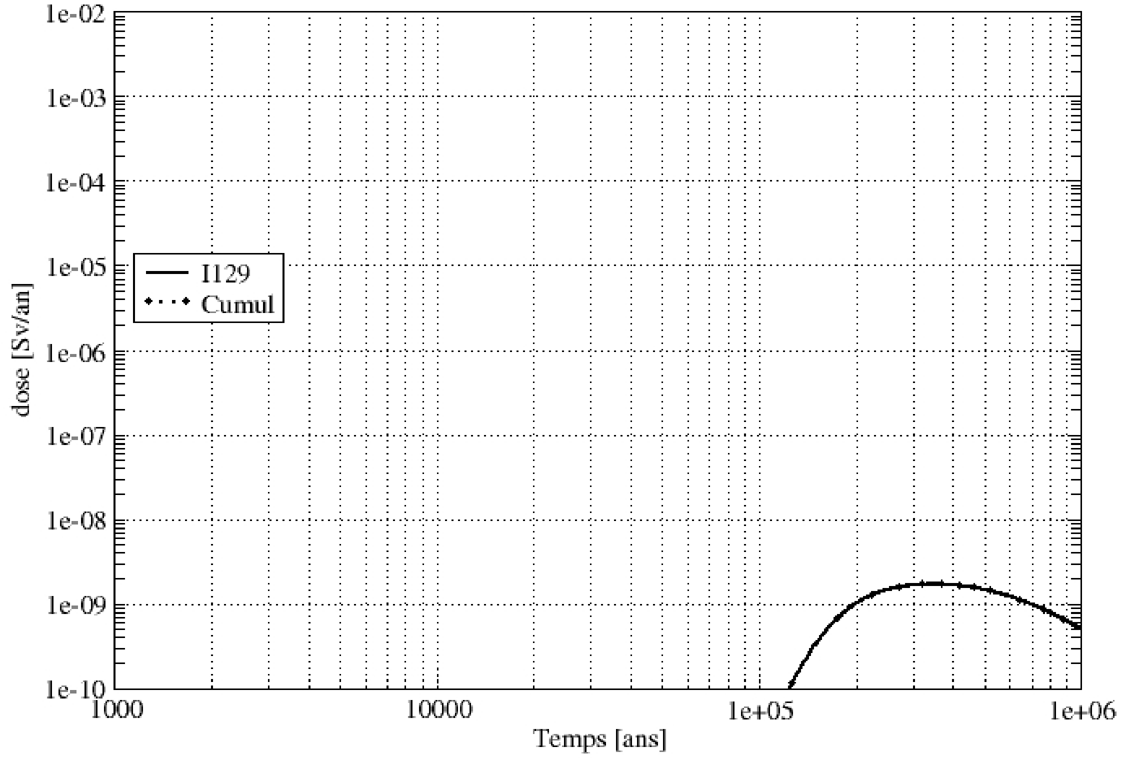


Figure 7.2-26 *Seal failure altered evolution scenario – All seals defective – Reference calculation – Dose rate history for the transfer pathway through the engineered structures – CU2 reference package*

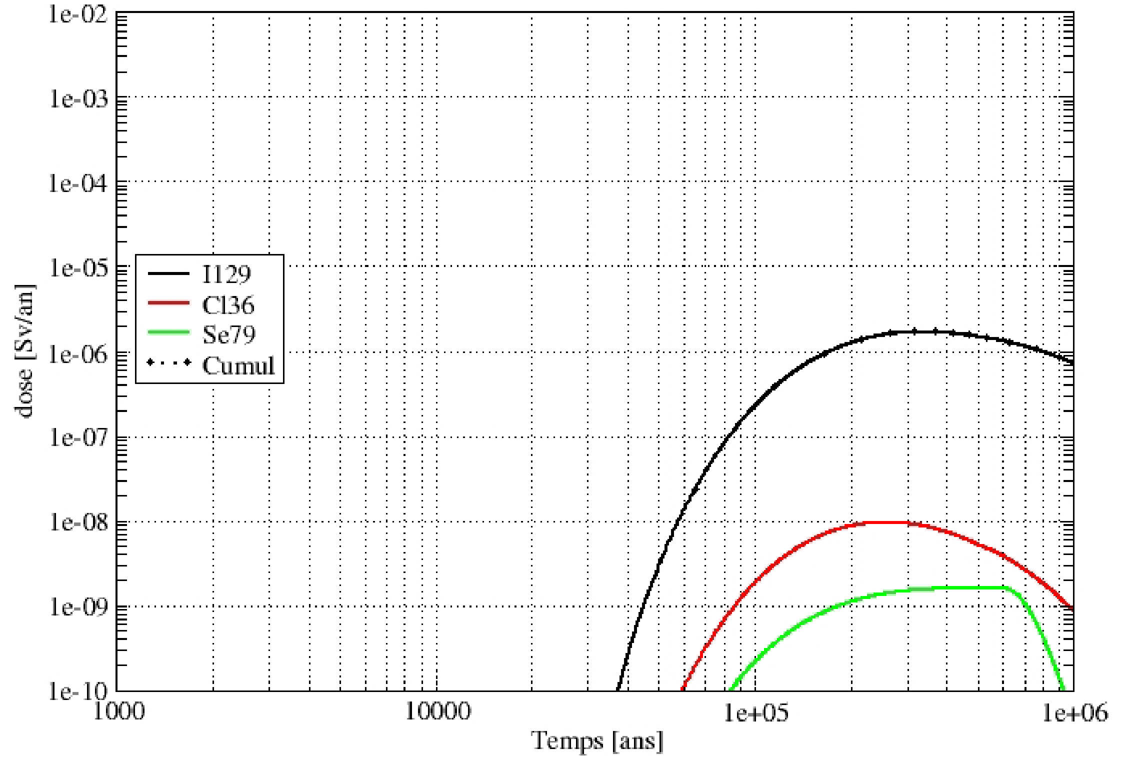


Figure 7.2-27 *Seal failure altered evolution scenario – All situations – Reference calculation – Dose rate history for the transfer pathway through the host formation – CU2 reference package*

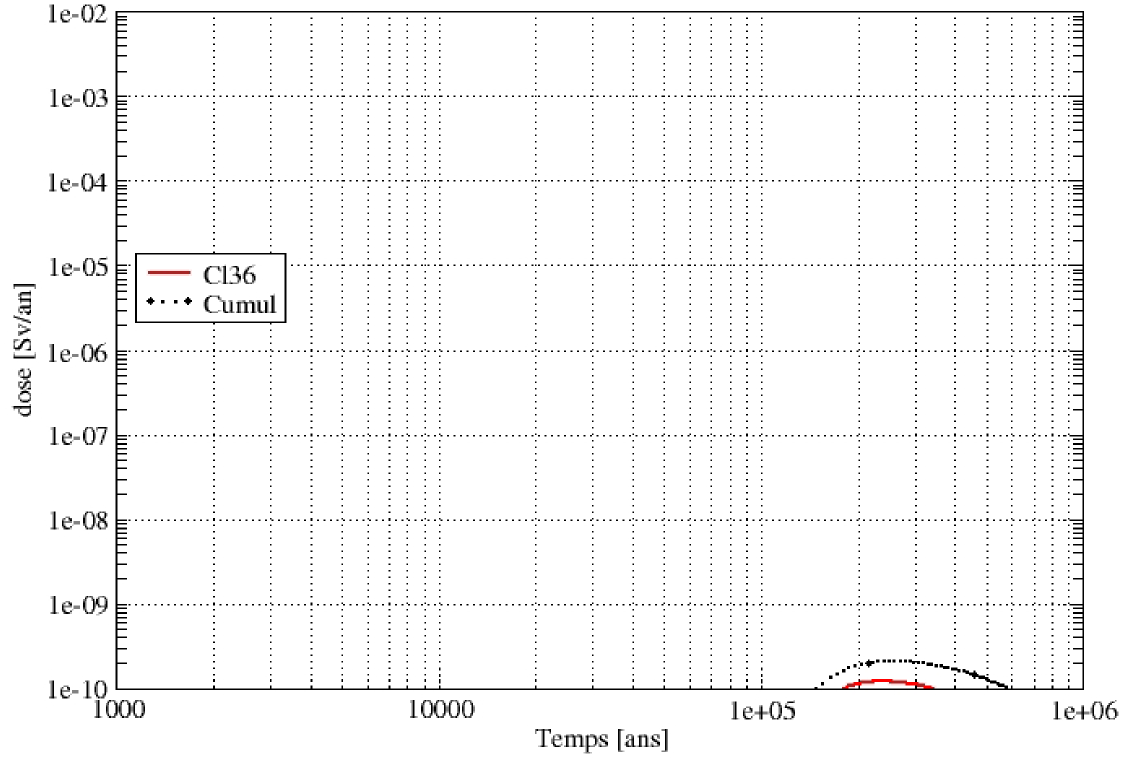


Figure 7.2-28 *Seal failure altered evolution scenario – All seals defective – Reference calculation – Dose rate history for the transfer pathway through the engineered structures – C1/C2 reference package*

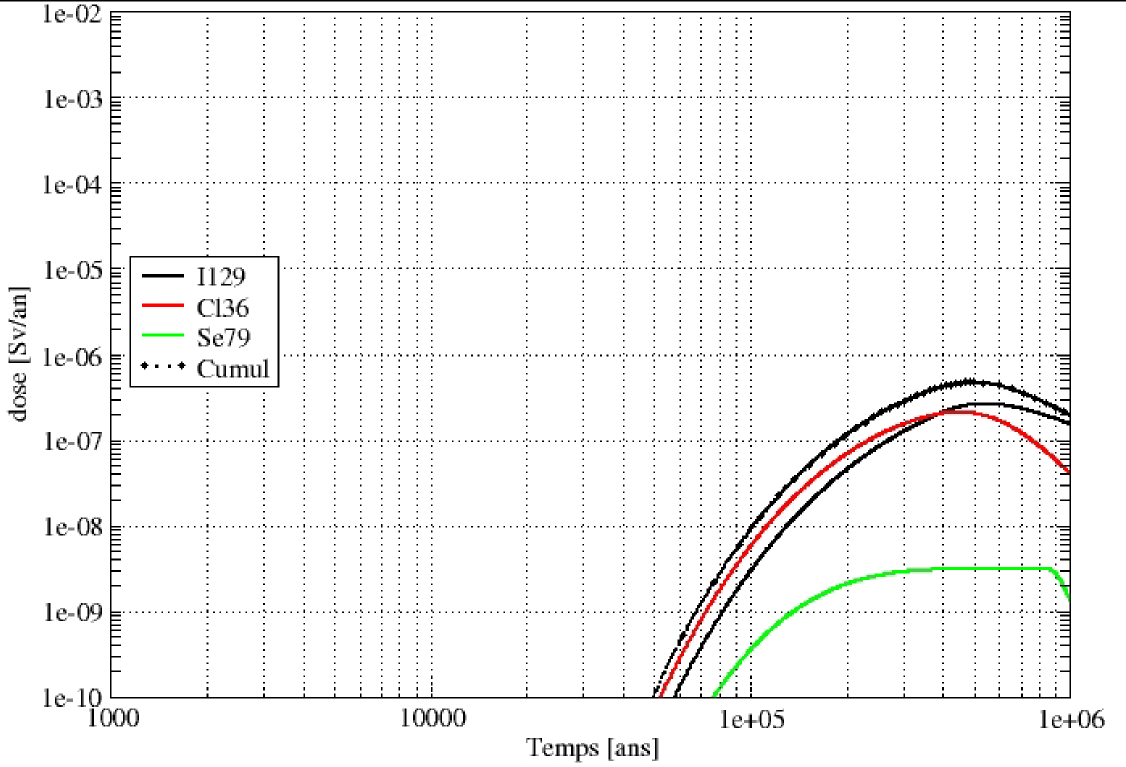


Figure 7.2-29 *Seal failure altered evolution scenario – All situations – Reference calculation – Dose rate history for the transfer pathway through the host formation – C1/C2 reference package*

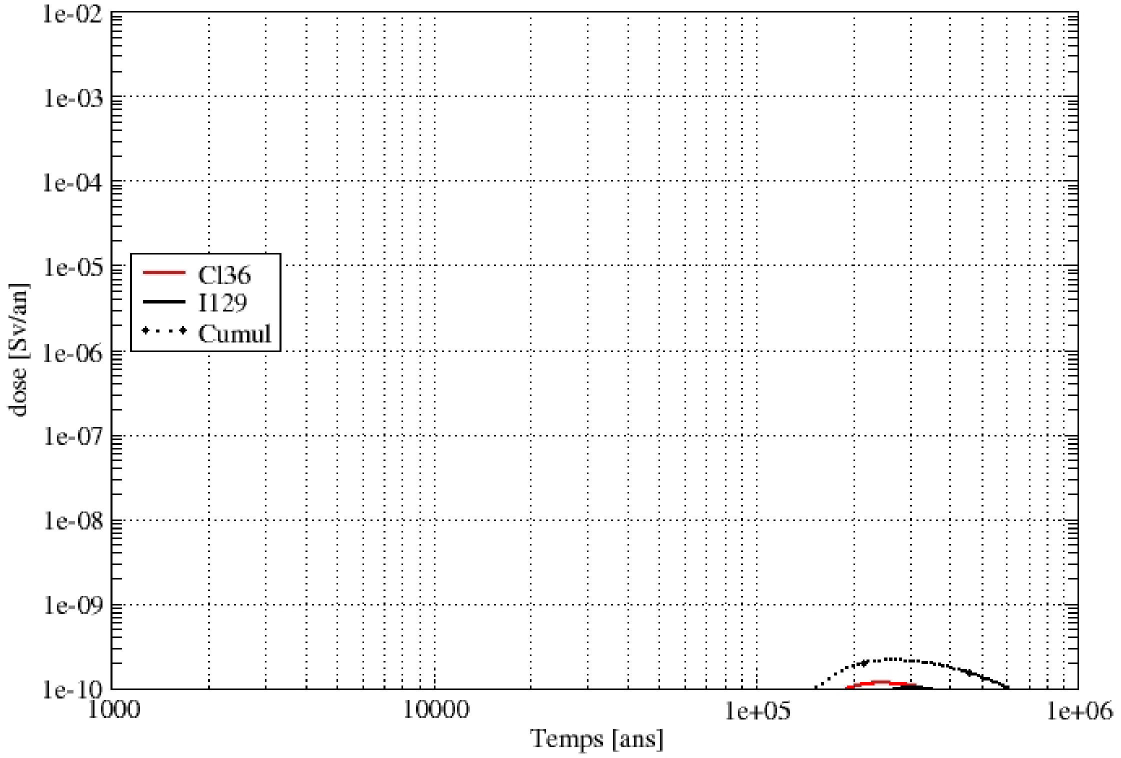


Figure 7.2-30 *Seal failure altered evolution scenario - all situations – reference calculation – history of transfer pathway doses through the structures – C3/C4 reference package*

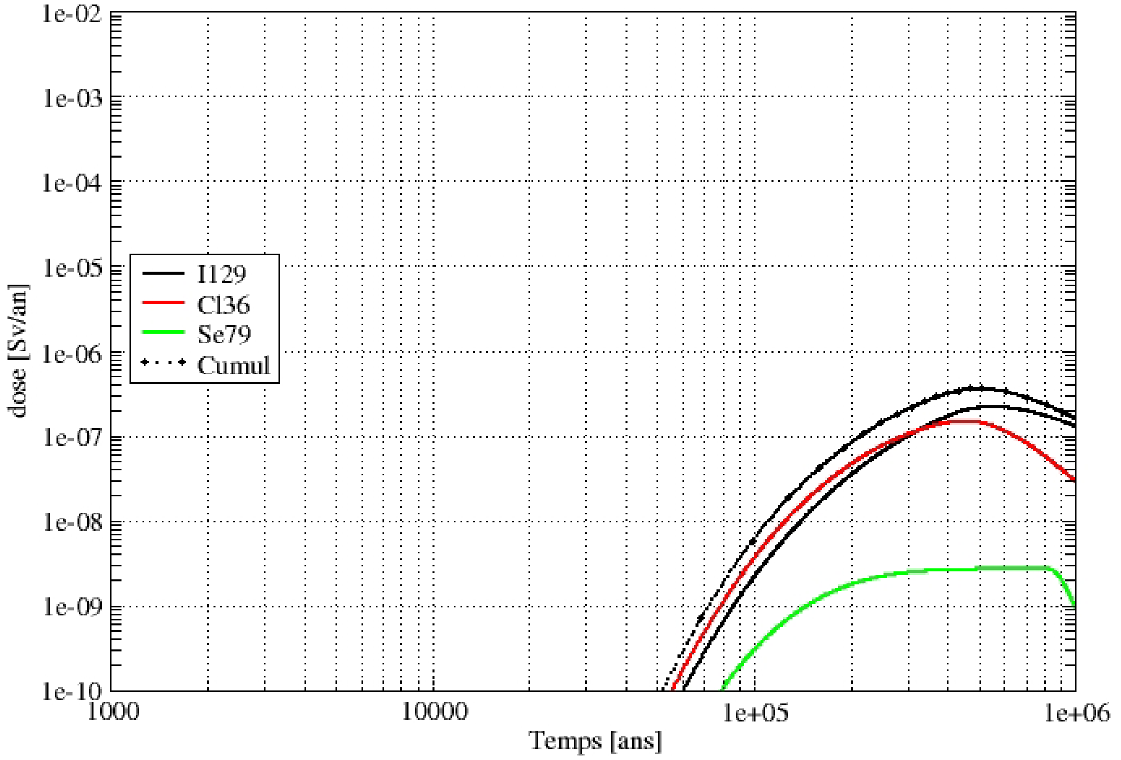


Figure 7.2-31 *Seal failure altered evolution scenario - all situations – reference calculation – history of transfer pathway doses through the formation host – C3/C4 reference package*

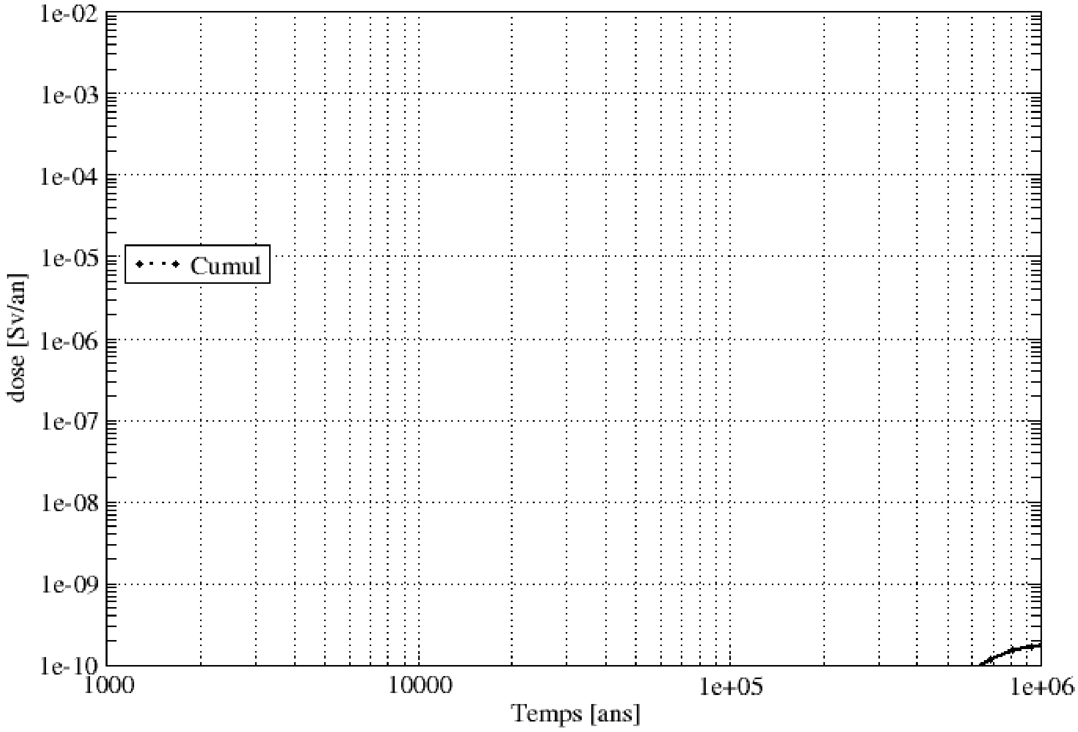


Figure 7.2-32 Seal failure altered evolution scenario – Shaft seals defective – Reference calculation – Dose rate history for the transfer pathway through the engineered structures – B1x reference package

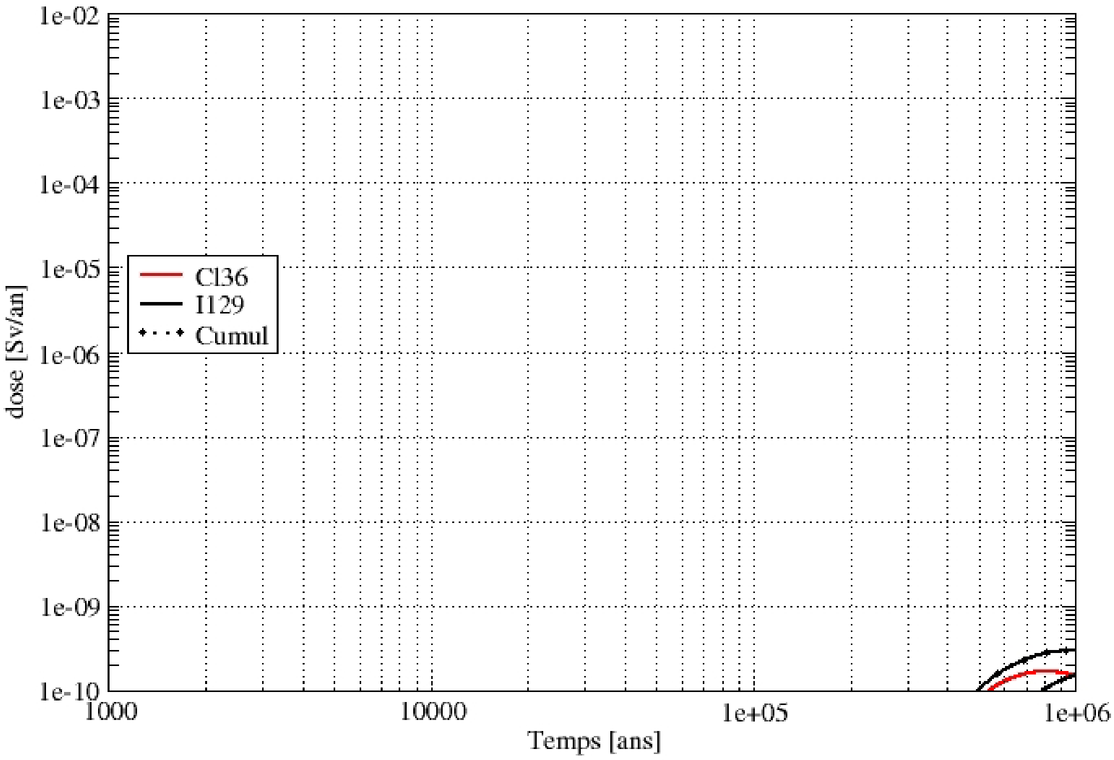


Figure 7.2-33 Seal failure' altered evolution scenario – All seals defective – Reference calculation – Dose rate history for the transfer pathway through the engineered structures – B1x reference package

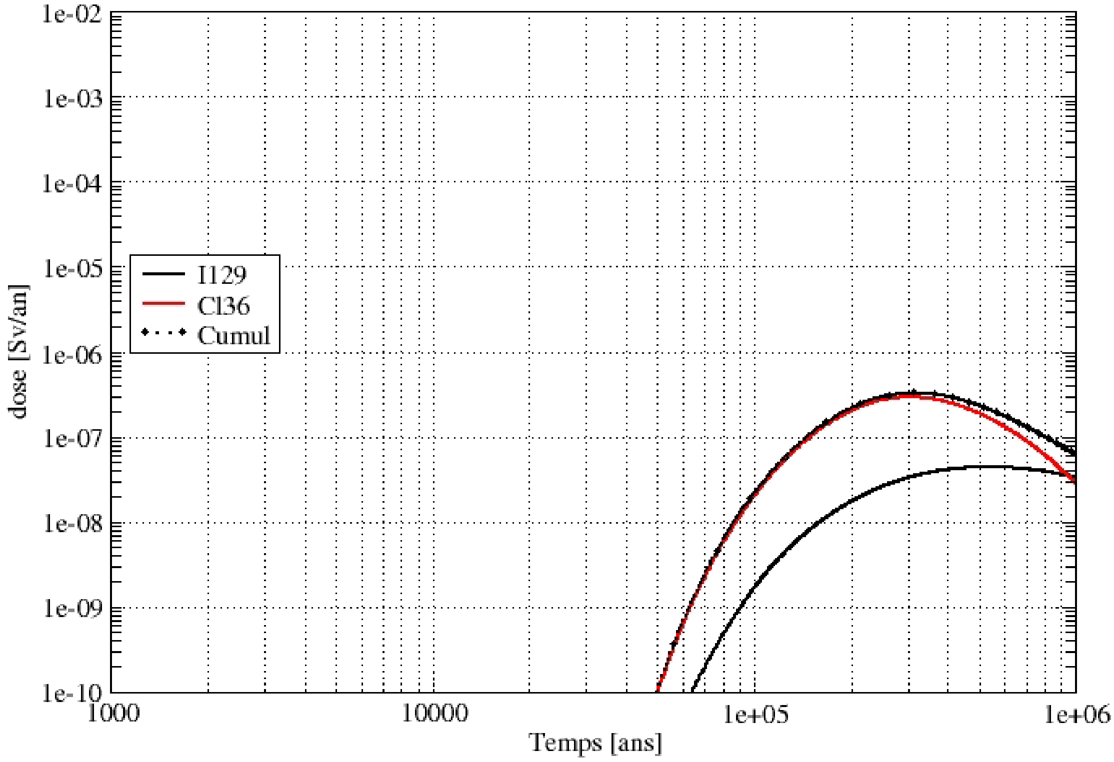


Figure 7.2-34 Seal failure' altered evolution scenario – All situations – Reference calculation – Dose rate history for the transfer pathway through the host formation – B1x reference package

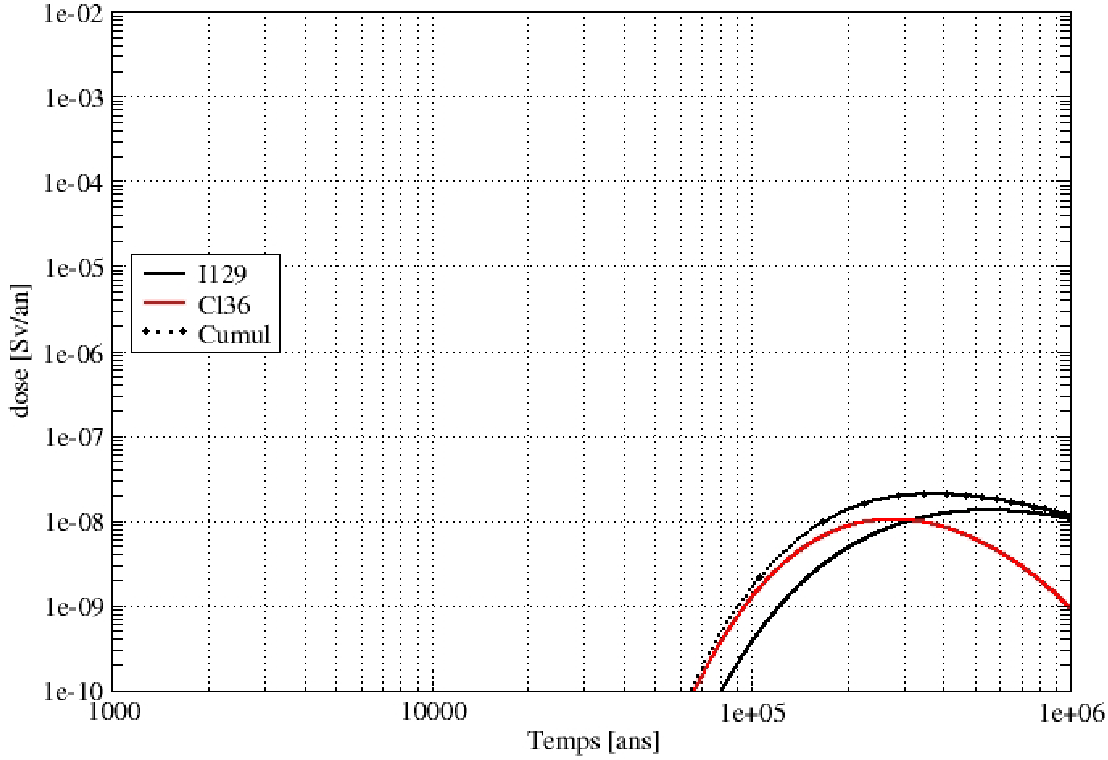


Figure 7.2-35 Seal failure altered evolution scenario - All situations - Reference calculation - dose histories of the transfer pathway via the host formation - B2 reference package

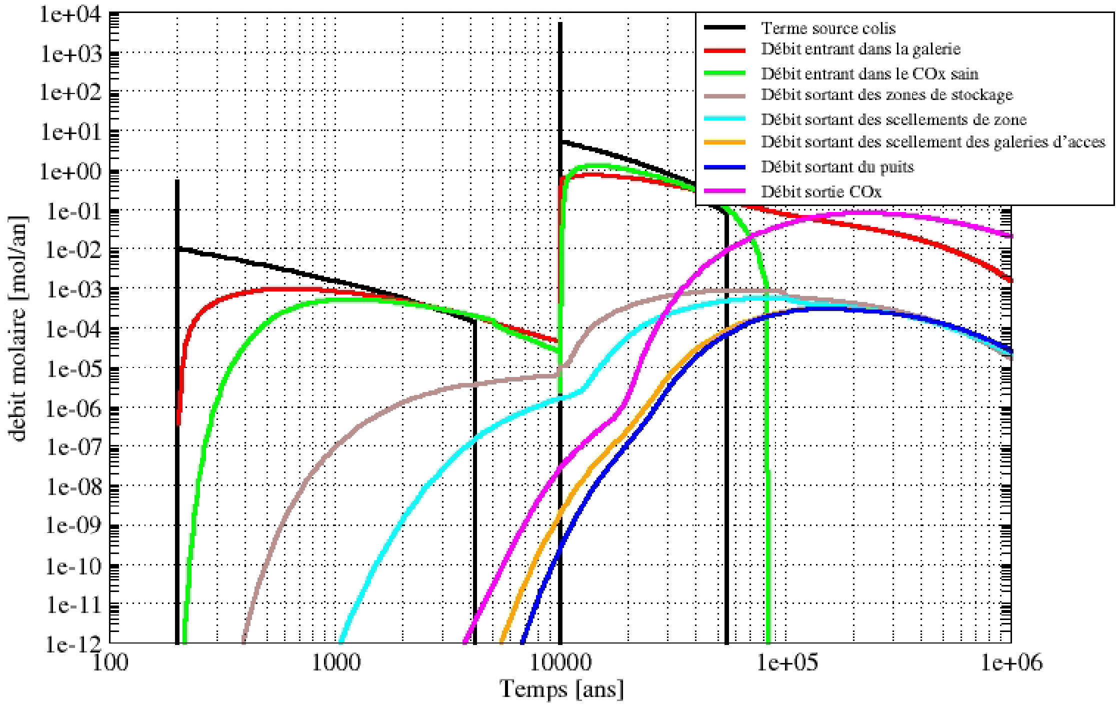


Figure 7.2-37 Seal failure altered evolution scenario - molar flow history - All seals failed : EDZ and seal core performance sensitivity : CU1 waste reference package - ^{129}I, ^{36}Cl, ^{79}Se, ^{14}C – [^{129}I]

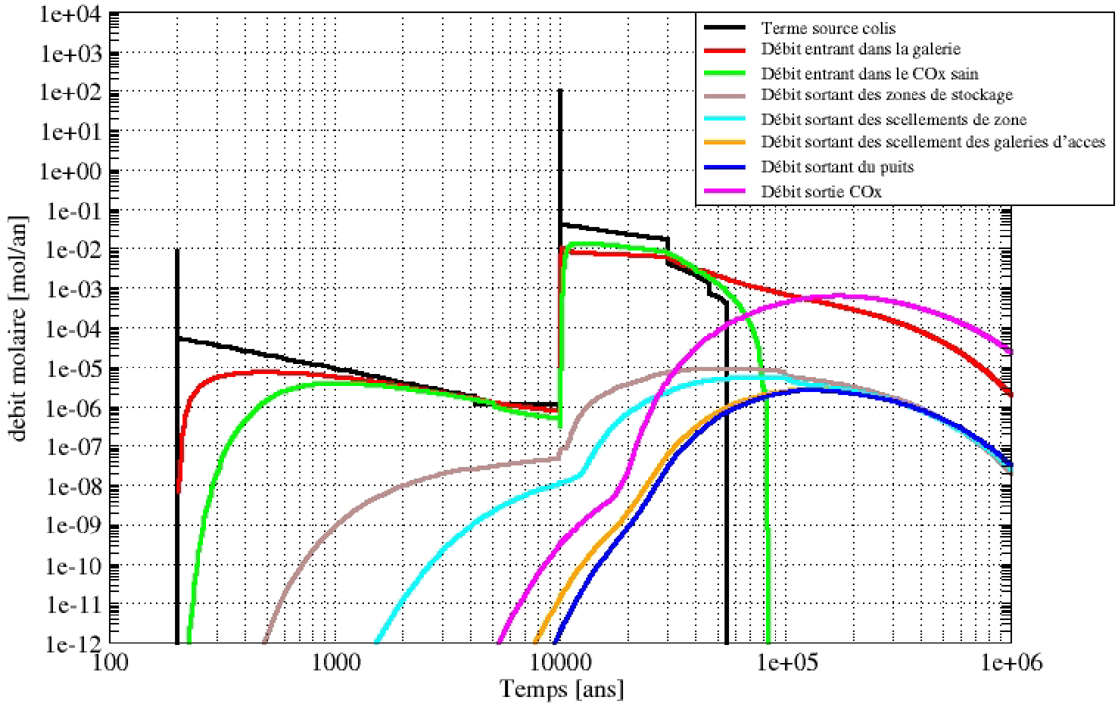


Figure 7.2-37 Seal failure altered evolution scenario - molar flow history - All seals failed : EDZ and seal core performance sensitivity : CU1 waste reference package - ^{129}I, ^{36}Cl, ^{79}Se, ^{14}C – [^{36}Cl]

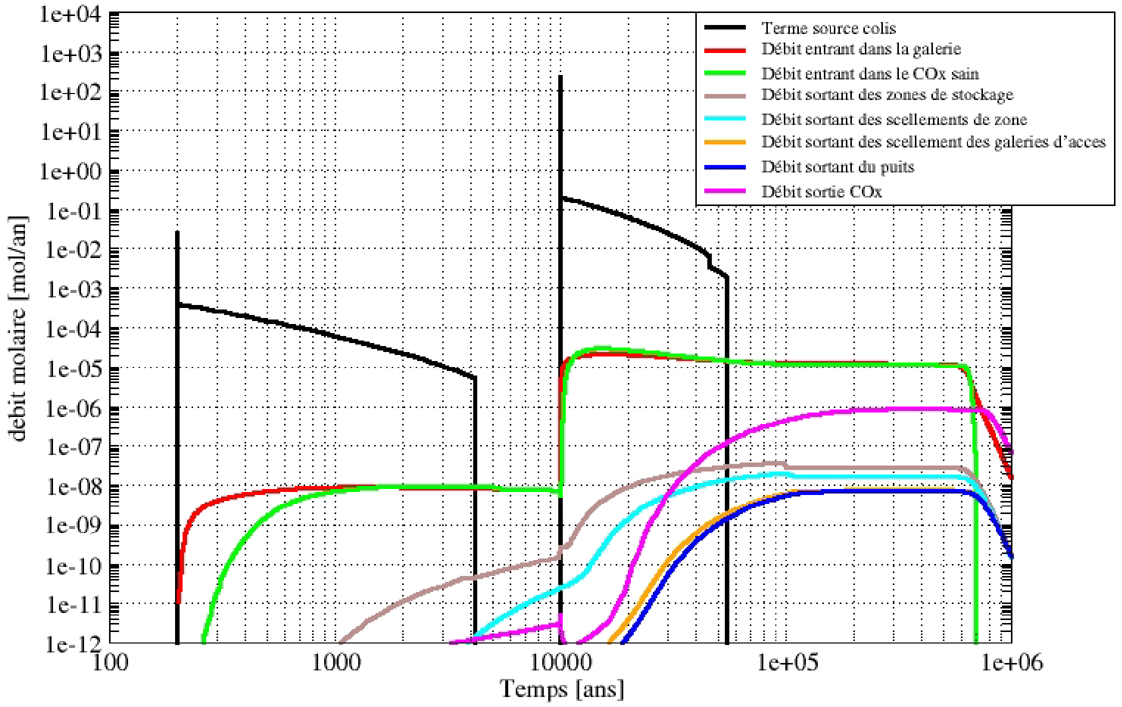


Figure 7.2-37 Seal failure altered evolution scenario - molar flow history - All seals failed : EDZ and seal core performance sensitivity : CU1 waste reference package - ^{129}I, ^{36}Cl, ^{79}Se, ^{14}Ce - [^{79}Se]

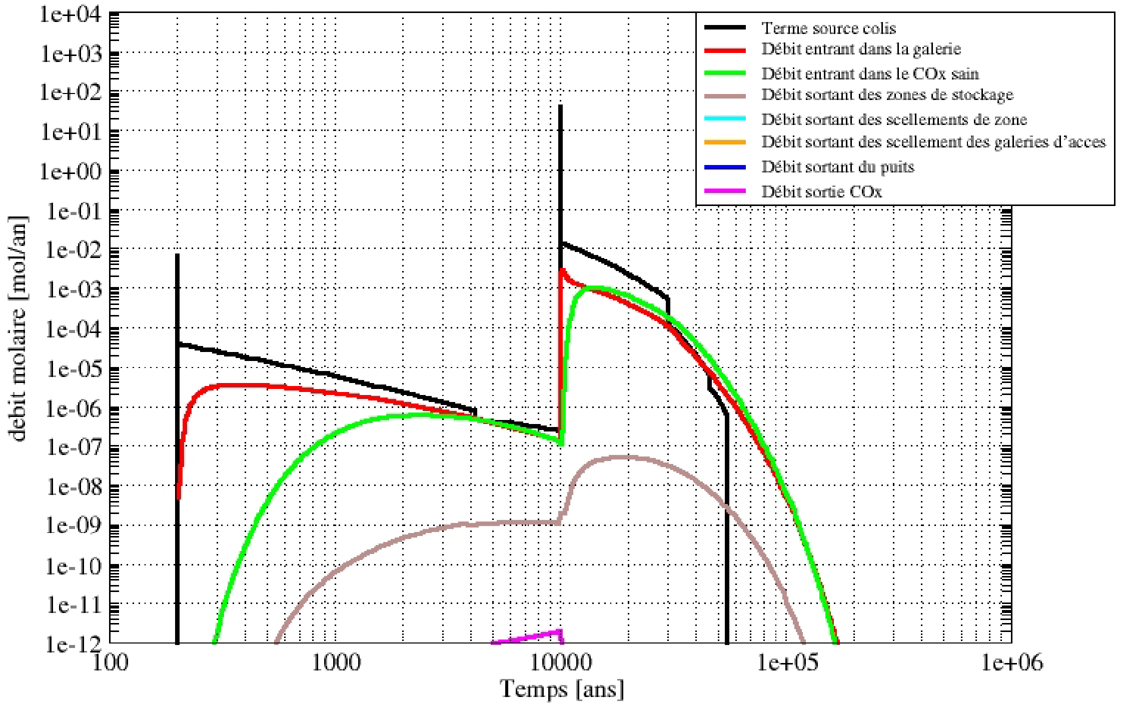


Figure 7.2-37 Seal failure altered evolution scenario - molar flow history - All seals failed : EDZ and seal core performance sensitivity : CU1 waste reference package - ^{129}I, ^{36}Cl, ^{79}Se, ^{14}Ce - [^{14}C]

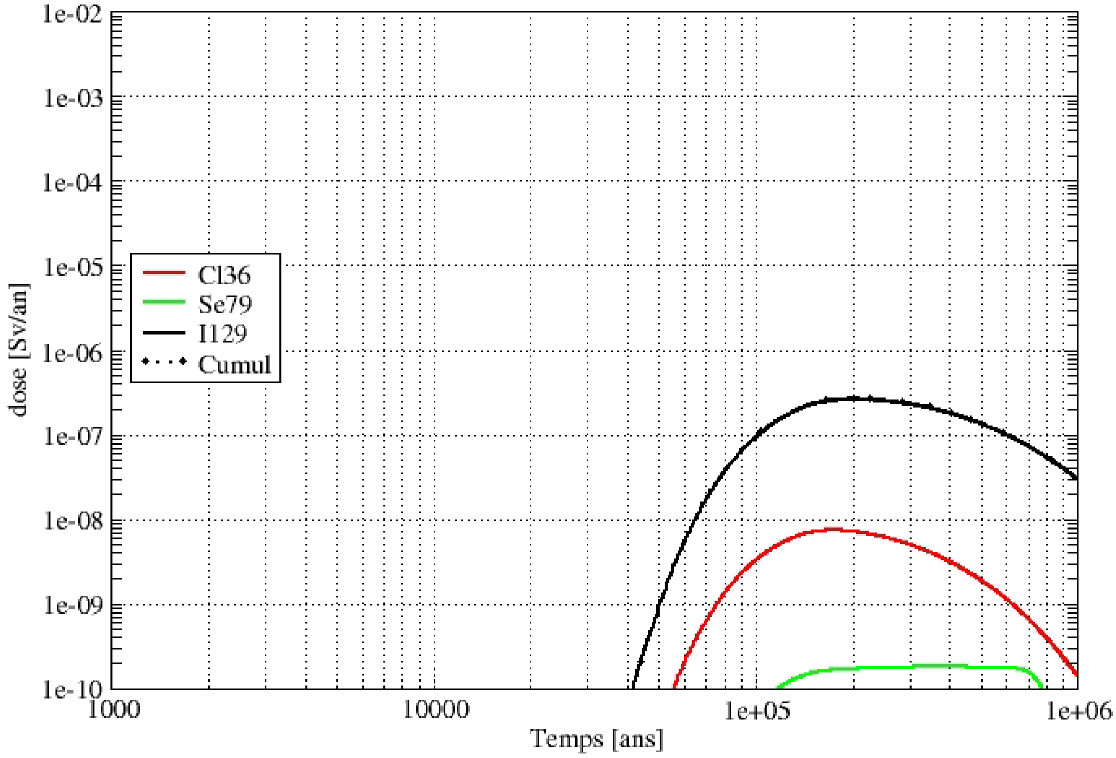


Figure 7.2-38 *Seal failure altered evolution scenario - All seals failed - EDZ and seal core performance sensitivity - dose histories of the transfer pathway through the structures - CU1 reference package*

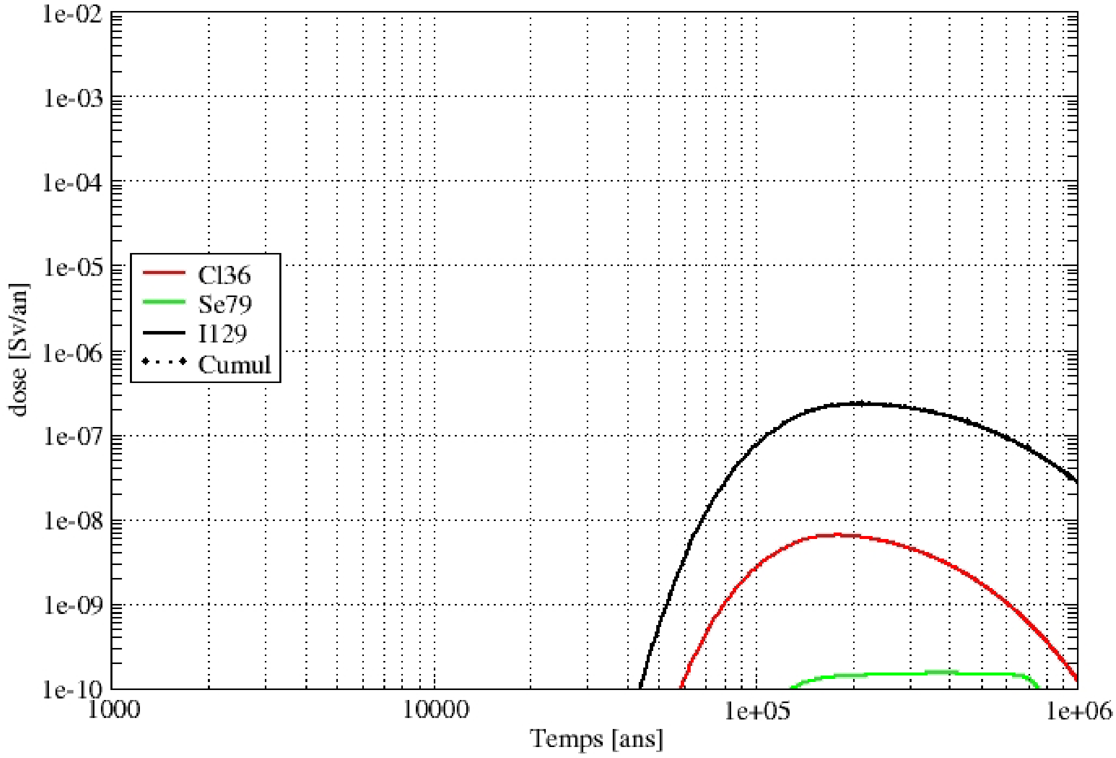


Figure 7.2-39 *Seal failure altered evolution scenario - Shaft seals failed - EDZ and seal core performance sensitivity - dose histories of the transfer pathway through the structures - CU1 reference package*

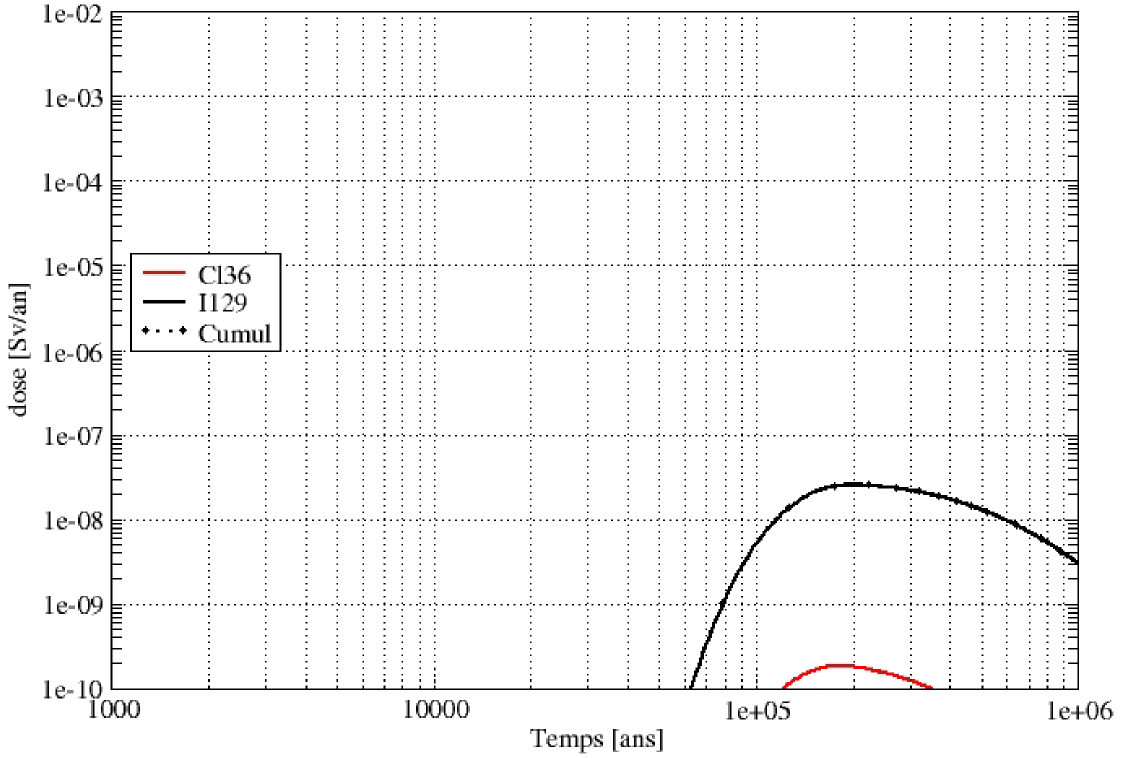


Figure 7.2-40 *Seal failure altered evolution scenario - All seals failed - EDZ and seal core performance sensitivity - dose histories of the transfer pathway through the structures - CU2 reference package*

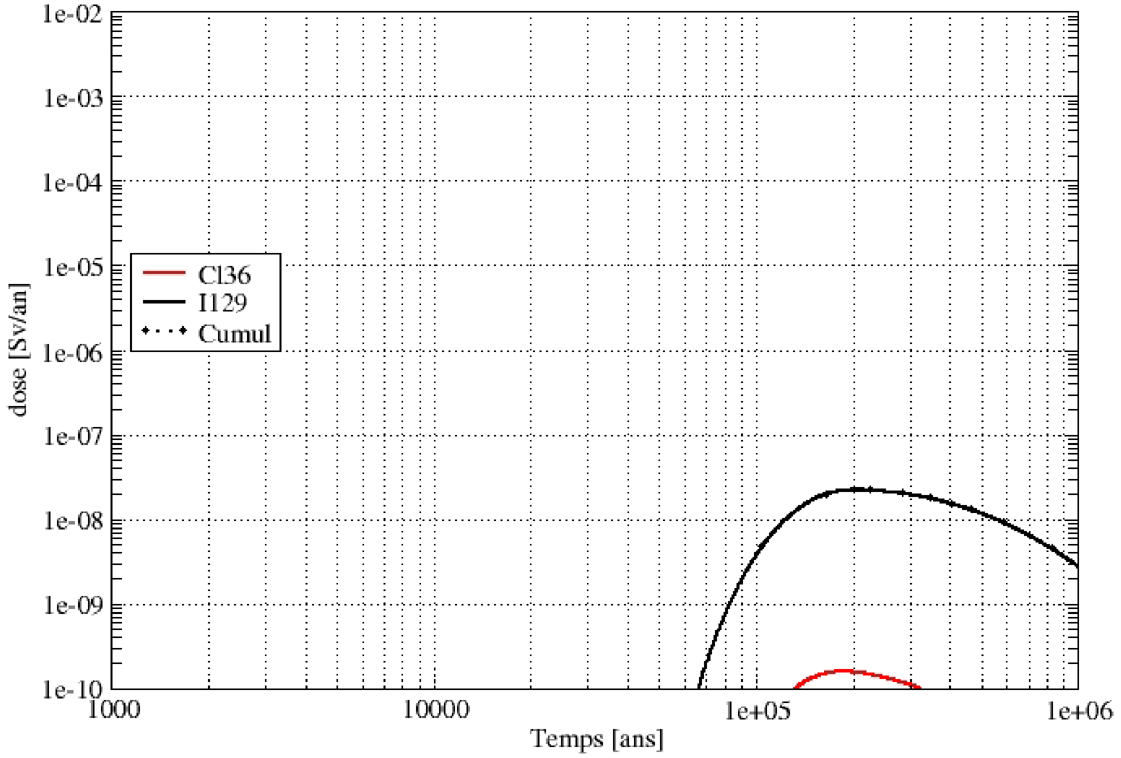


Figure 7.2-41 *Seal failure altered evolution scenario - Shaft seals failed - EDZ and seal core performance sensitivity - dose histories of the transfer pathway through the structures - CU2 reference package*

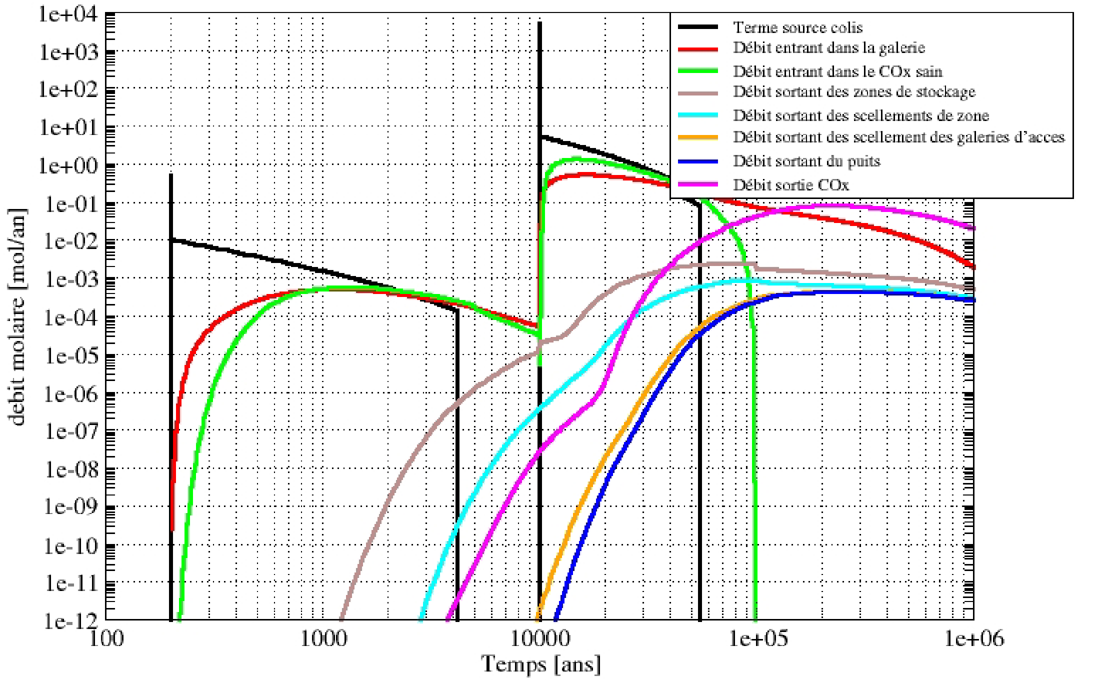


Figure 7.2-42 *Seal failure altered evolution scenario - hydraulic head gradient sensitivity study - molar flow histories - all seals failed situation - CU1 spent fuel - ^{129}I - [sensitivity 1 : 0.4 m/m hydraulic head gradient]*

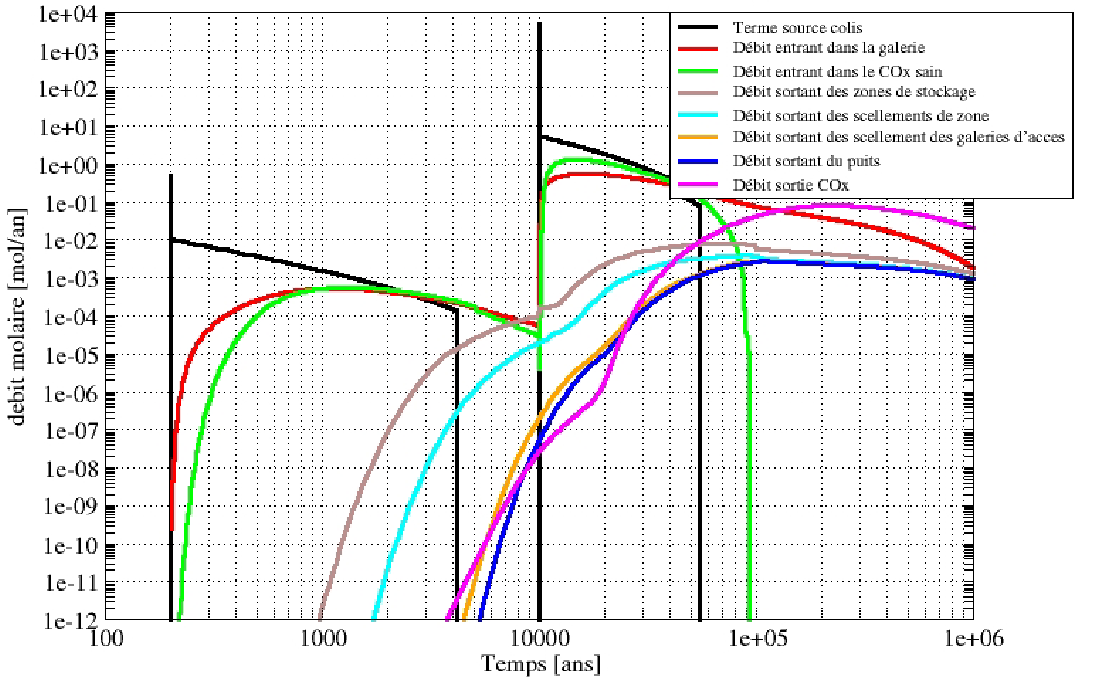


Figure 7.2-42 *Seal failure altered evolution scenario - hydraulic head gradient sensitivity study - molar flow histories - all seals failed situation - CU1 spent fuel - ^{129}I - [sensitivity 2 : 1 m/m hydraulic head gradient]*

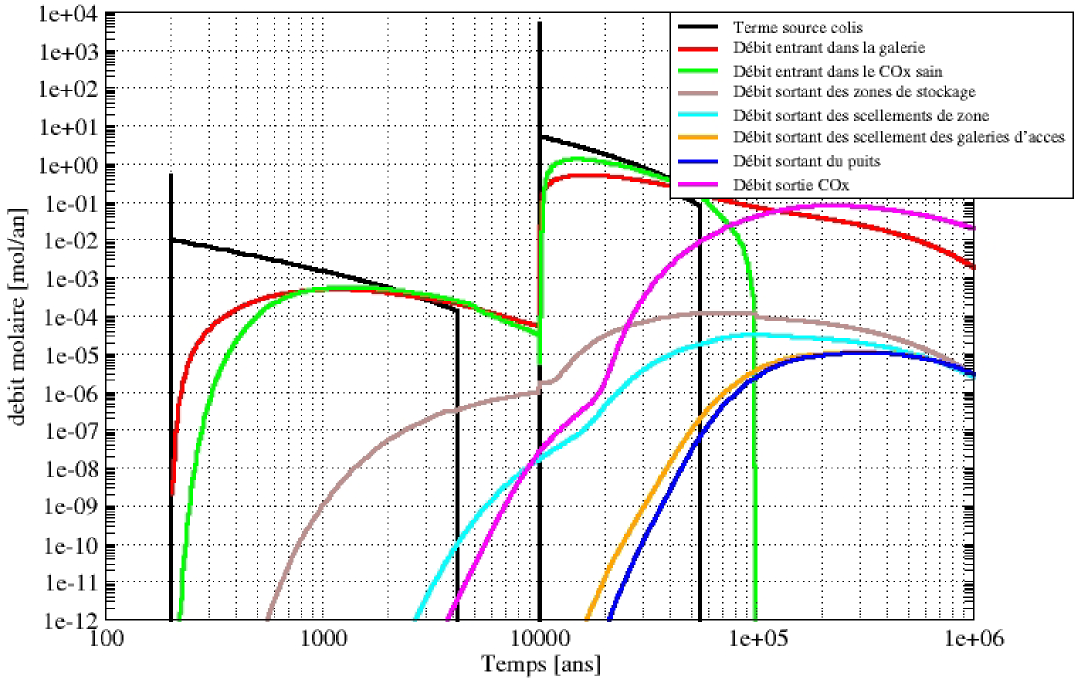


Figure 7.2-42 *Seal failure altered evolution scenario - hydraulic head gradient sensitivity study - molar flow histories - all seals failed situation - CU1 spent fuel - ^{129}I - [Reference calculation : 0.2 m/m hydraulic head gradient]*

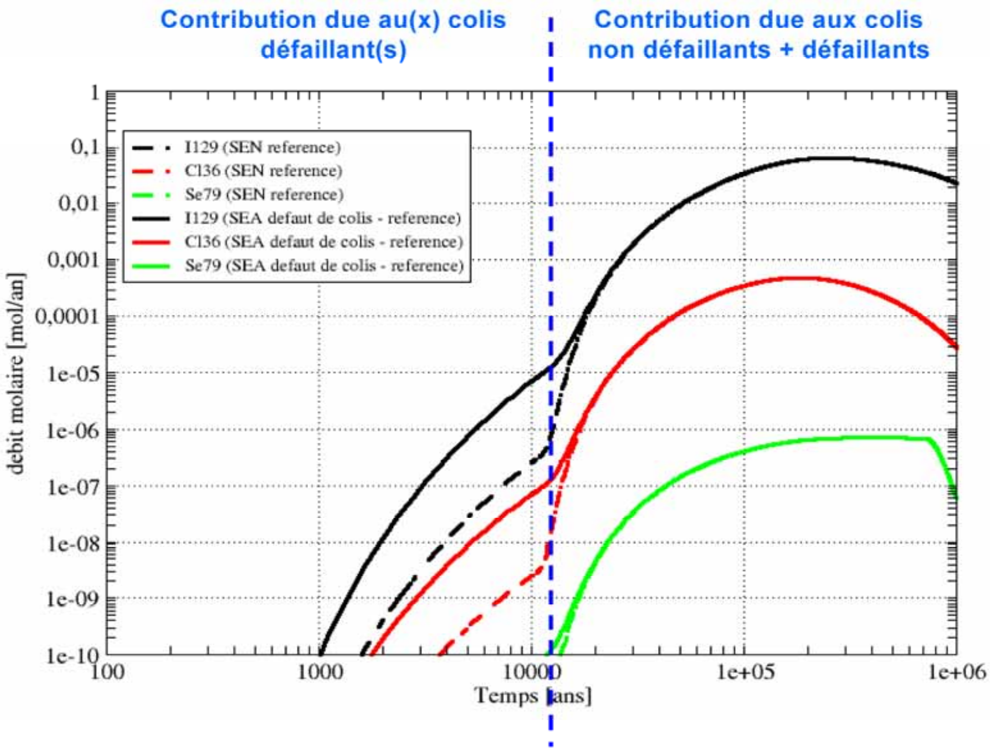


Figure 7.3-1 *Package failure SEA – Molar flow curves leaving the Callovo-Oxfordian formation – comparison ^{129}I, ^{36}Cl et ^{79}Se between the SEN situation (1 defective package) and package failure SEA (reference – 30 defective packages) - CU1 spent fuels*

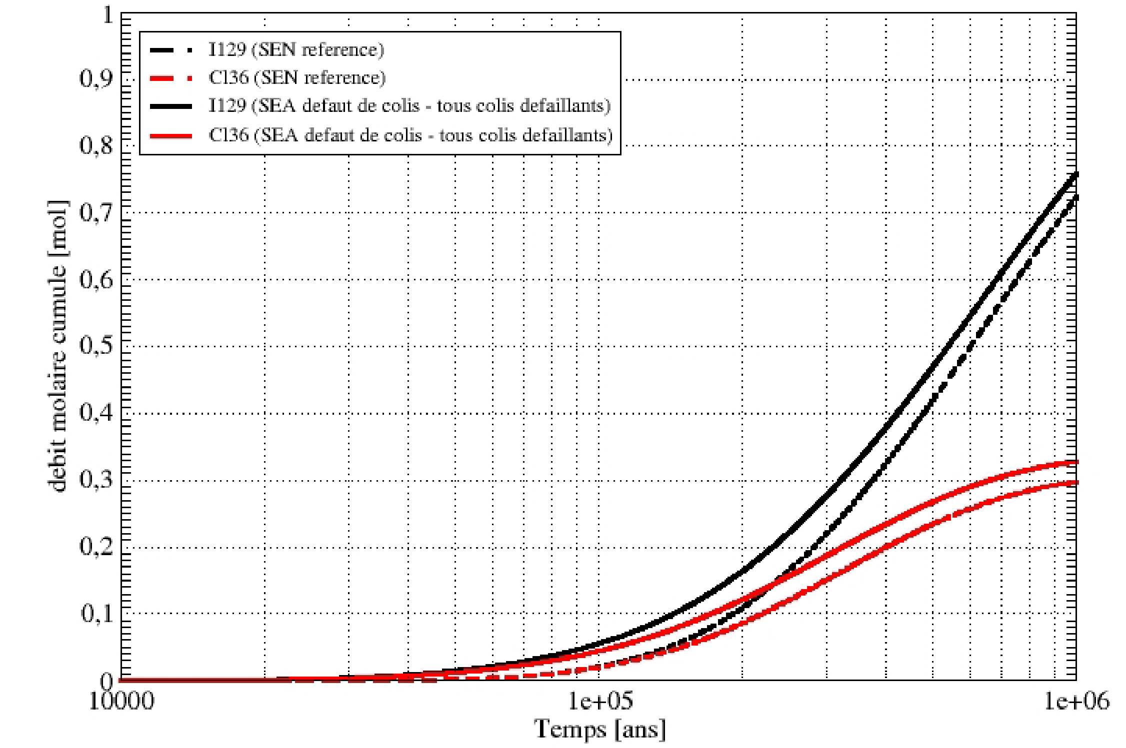


Figure 7.3-3 Package failure SEA - Evolution of cumulative molar flow rate leaving the Callovo-Oxfordian formation - comparison of « all defective packages » and SEN – CU1 spent fuels

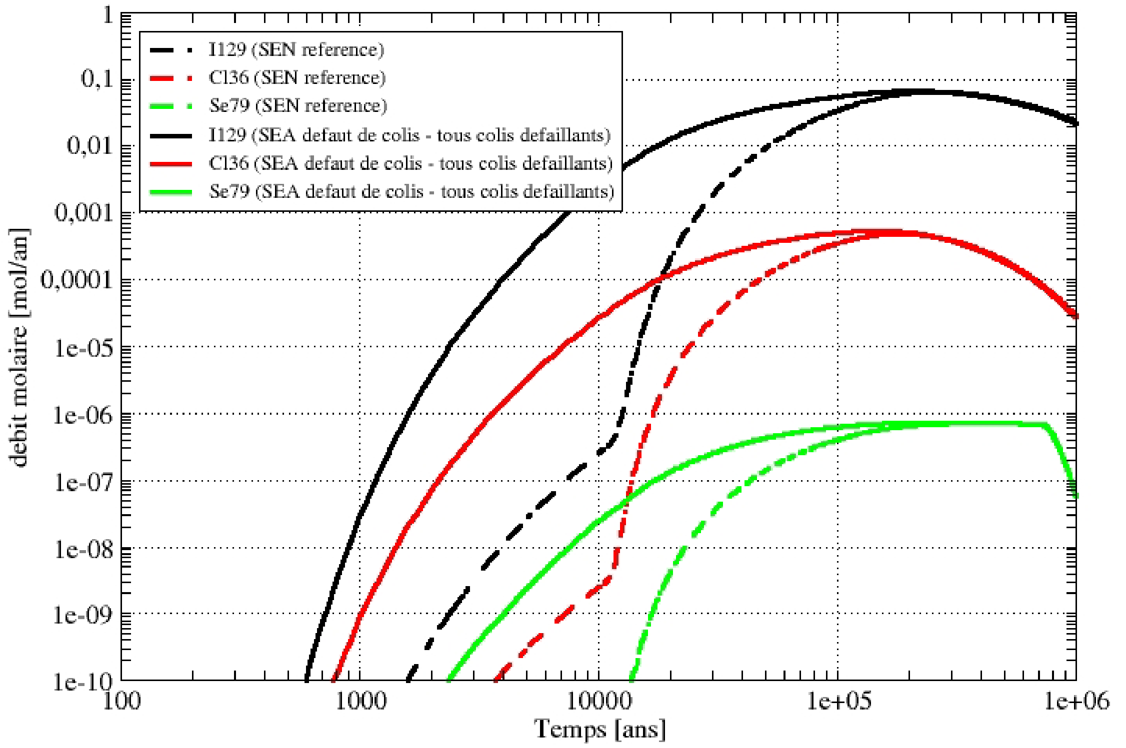


Figure 7.3-4 Package failure SEA - Evolution of molar flow rate leaving the Callovo-Oxfordian formation - comparison of « all defective packages » and SEN – CU1 spent fuels

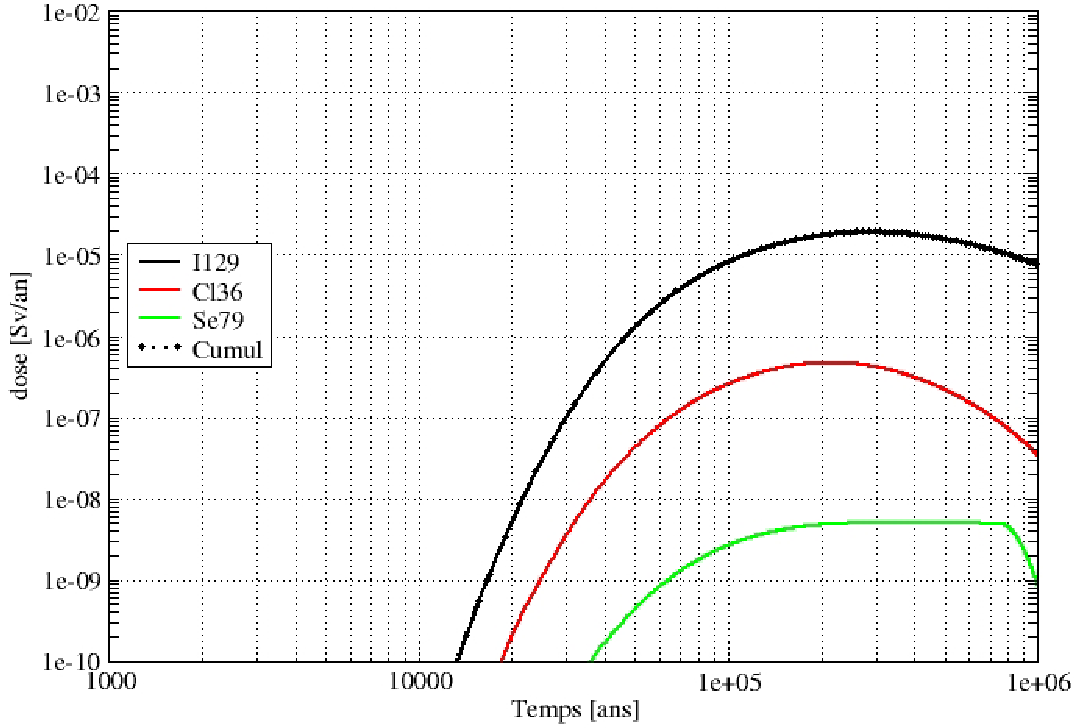


Figure 7.3-5 *Package failure SEA – Sensitivity calculation for all defective packages – Dose at the Oxfordian Saulx outlet – CU1 reference package*

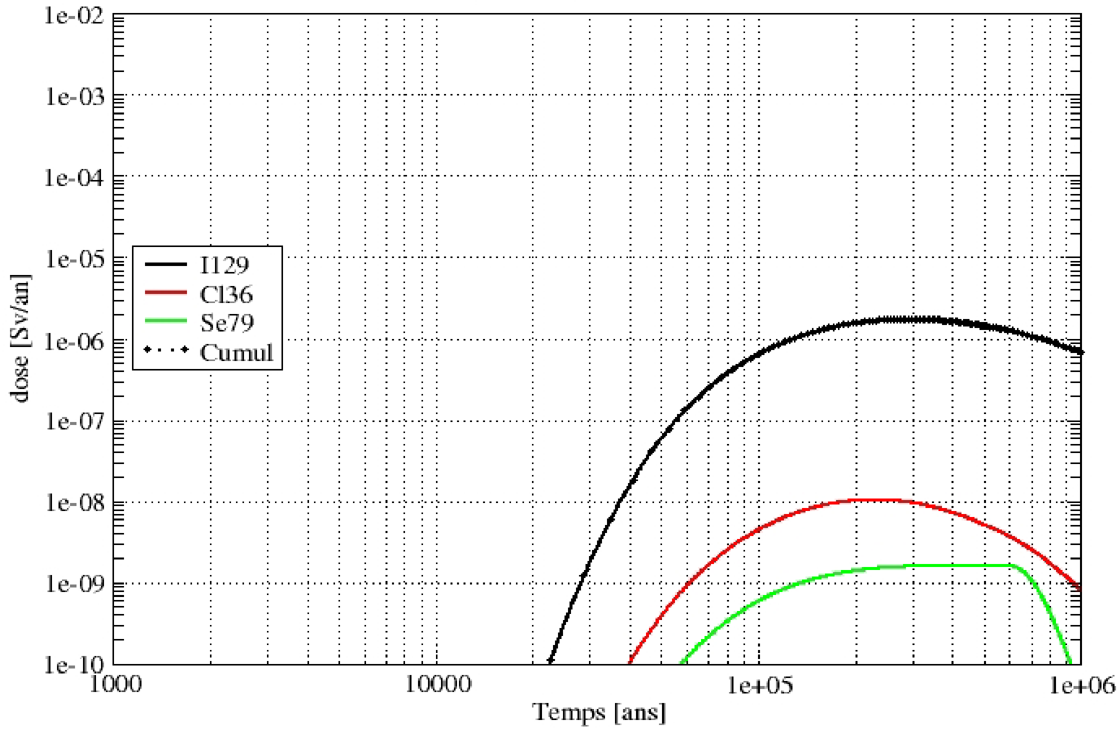


Figure 7.3-6 *Package failure SEA – Sensitivity calculation for all defective packages – Dose at the Oxfordian Saulx outlet – CU2 reference package*

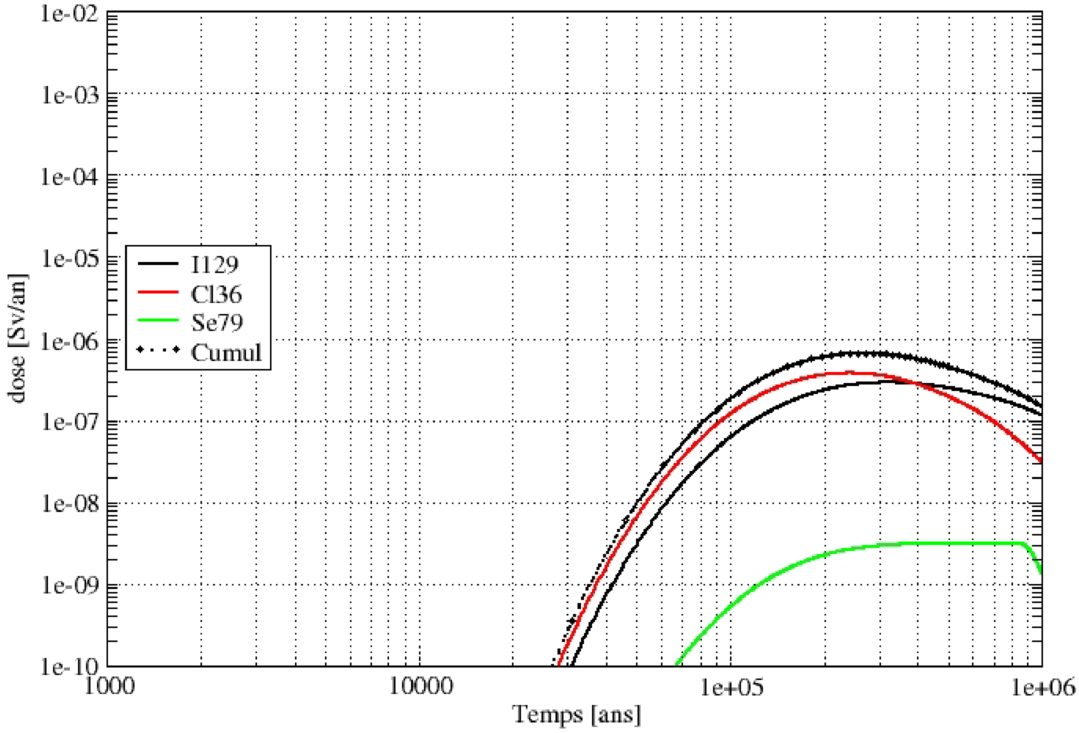


Figure 7.3-7 *Package failure SEA – Sensitivity calculation for all defective packages – Dose at the Oxfordian Saulx outlet – C1/C2 reference package*

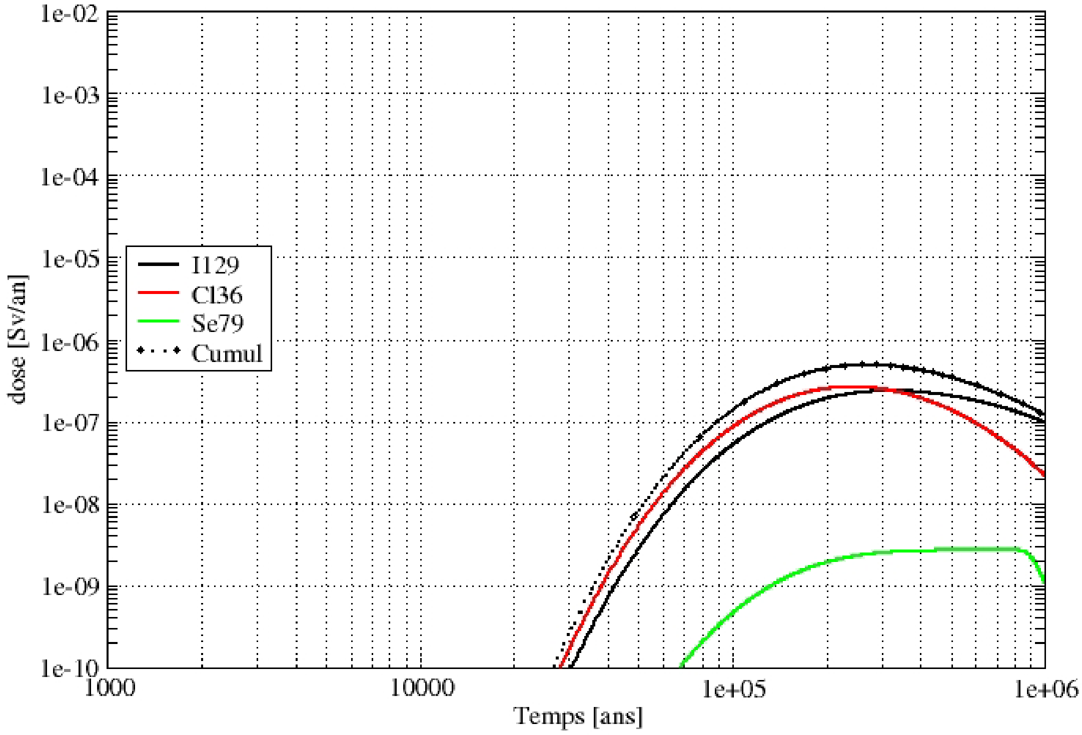


Figure 7.3-8 *Package failure SEA – Sensitivity calculation for all defective packages – Dose at the Oxfordian Saulx outlet – C3/C4 reference package*

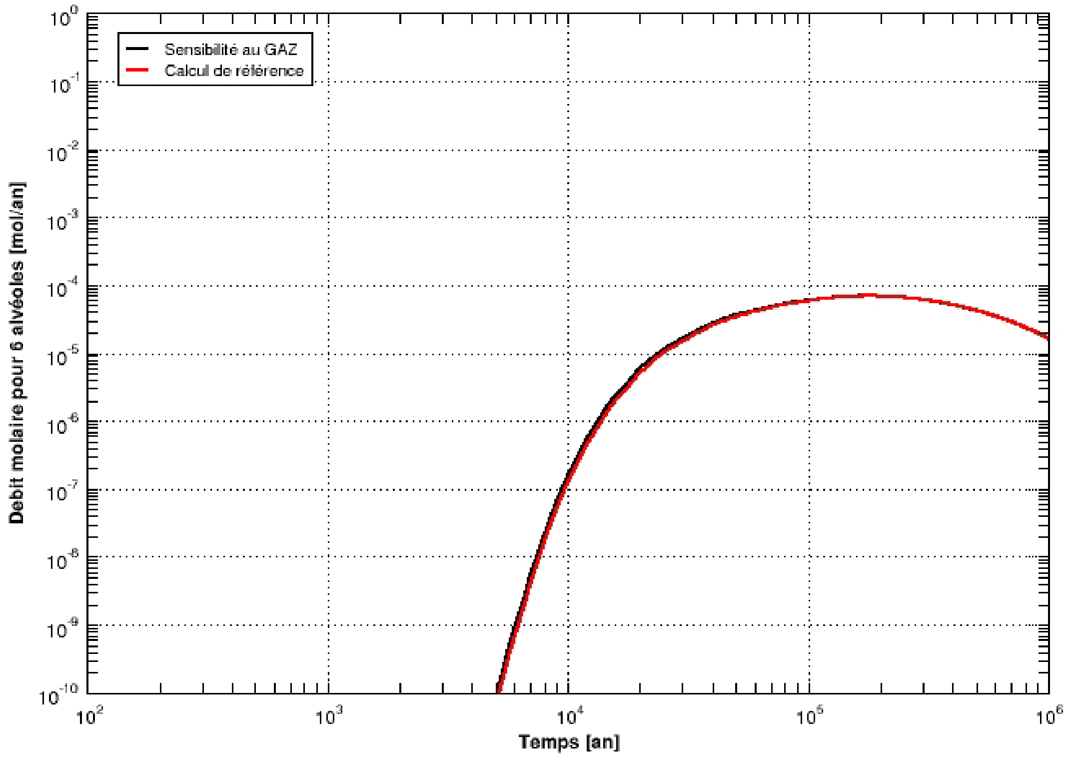


Figure 7.3-9 *Package failure SEA – sensitivity calculation of hydraulic transient influence – records of the molar flow rate leaving the Callovo-Oxfordian formation due to 6 CU1 cells – comparison of cases with and without hydraulic overpressure (^{129}I)*

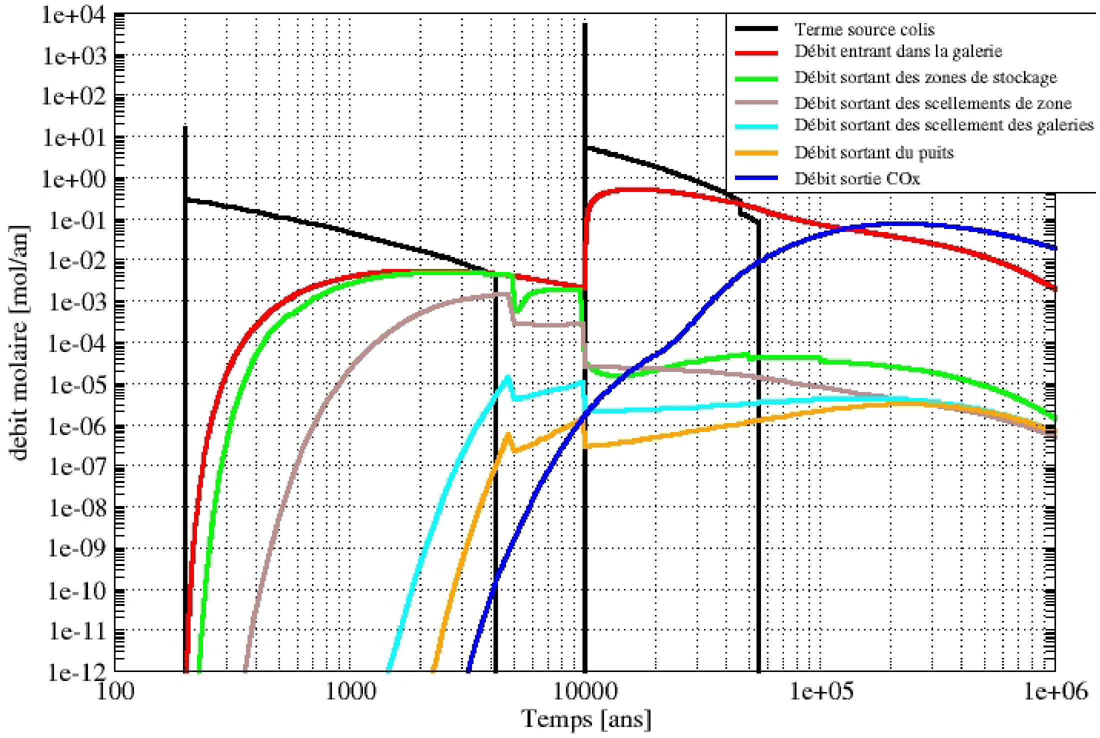


Figure 7.3-10 *« Package failure » SEA – sensitivity calculation of hydraulic transient influence – record of molar flow rate – ^{129I} – CU1 spent fuels. - [sensitivity calculation]*

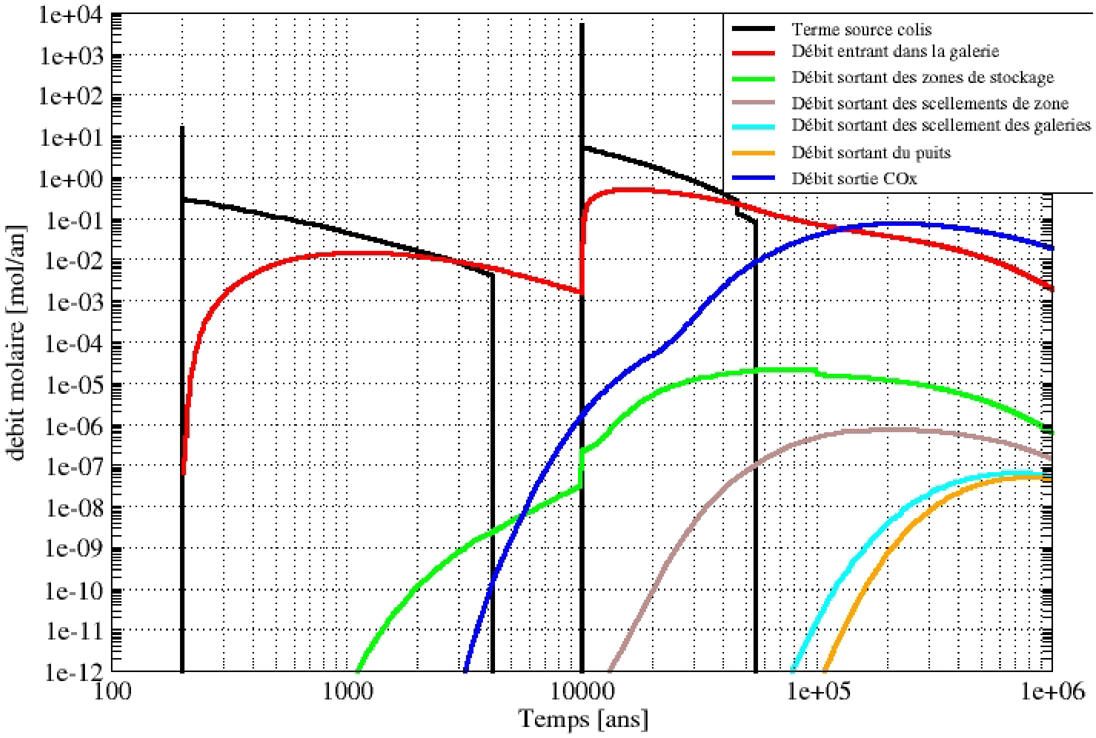


Figure 7.3-10 *« Package failure » SEA – sensitivity calculation of hydraulic transient influence – record of molar flow rate – 129I – CU1 spent fuels. - [reference calculation]*

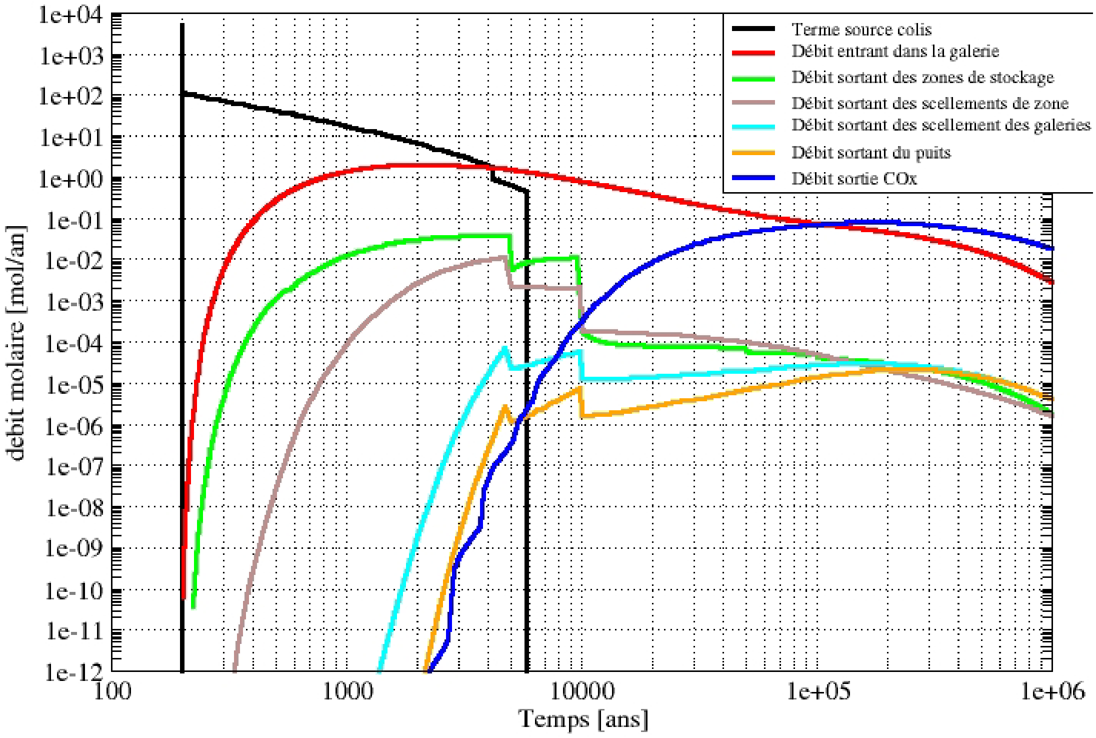


Figure 7.3-11 *« Package failure » SEA – sensitivity calculation of hydraulic transient influence – record of molar flow rate – case of all defective packages- ^{129}I – CU1 spent fuels. - [Sensitivity calculation – all defective packages]*

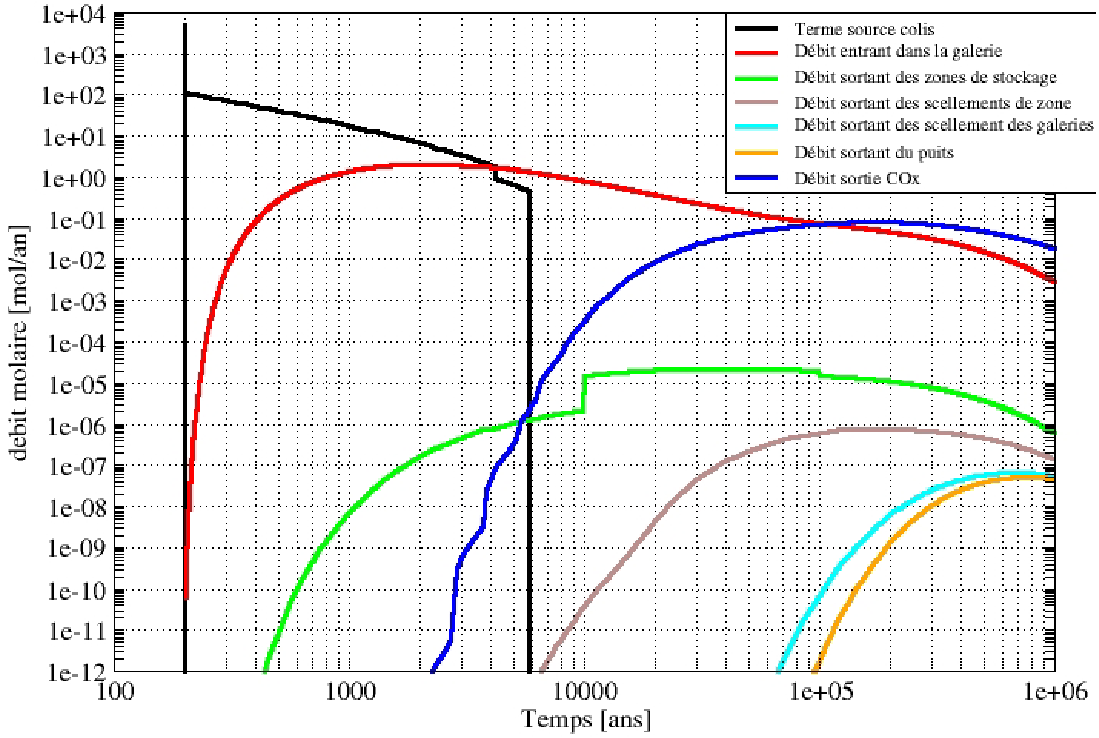


Figure 7.3-11 *« Package failure » SEA – sensitivity calculation of hydraulic transient influence – record of molar flow rate – case of all defective packages- ^{129}I – CU1 spent fuels. - [reference calculation – all defective packages]*

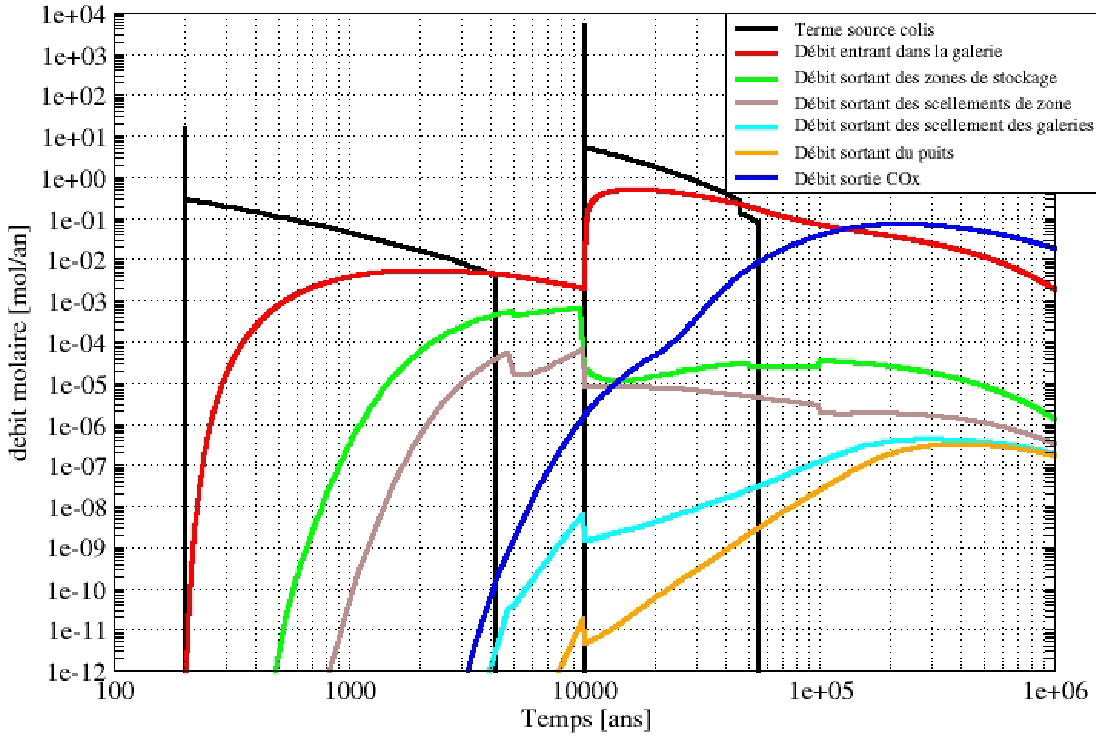


Figure 7.3-12 *« Package failure » SEA – sensitivity calculation of hydraulic transient influence – record of molar flow rate – taking the degree of saturation in the structures on the permeability value into account - ^{129}I – CU1 spent fuels. [Sensitivity calculation — fraction of defective packages]*

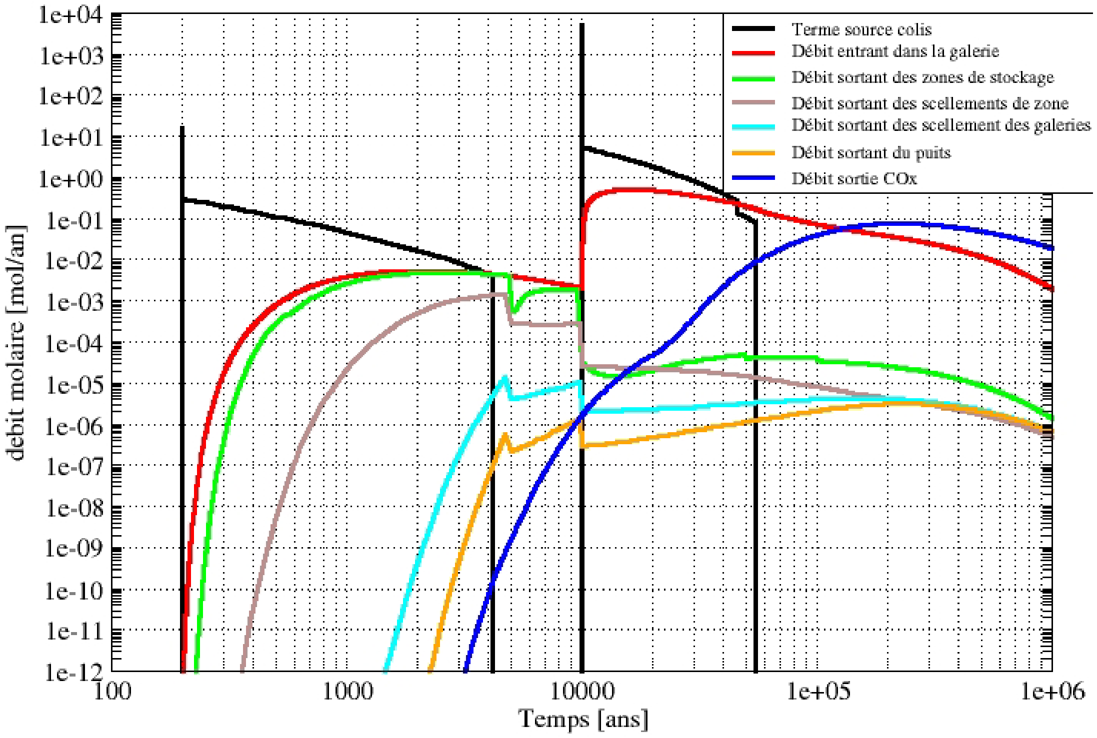


Figure 7.3-12 *« Package failure » SEA – sensitivity calculation of hydraulic transient influence – record of molar flow rate – taking the degree of saturation in the structures on the permeability value into account - ^{129}I – CU1 spent fuels. - [Sensitivity calculation — fraction of defective packages]*

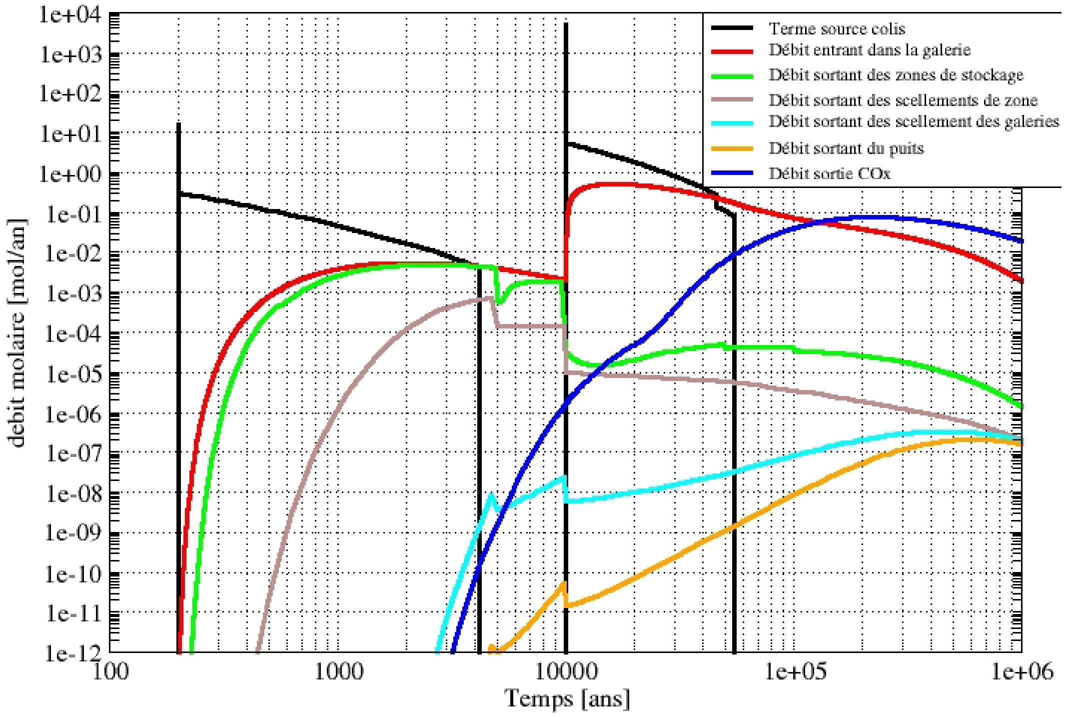


Figure 7.3-13 *« Package failure » SEA – sensitivity calculation on the influence of the hydraulic transient – record of molar flow rate – taking into account the permeability of the equivalent anchoring of seals : 10^{-10} m/s - ^{129}I – CU1 spent fuels. - [Sensitivity calculation – Taking into account the equivalent permeability of anchorings : 10^{-10} m/s – Fraction of defective packages]*

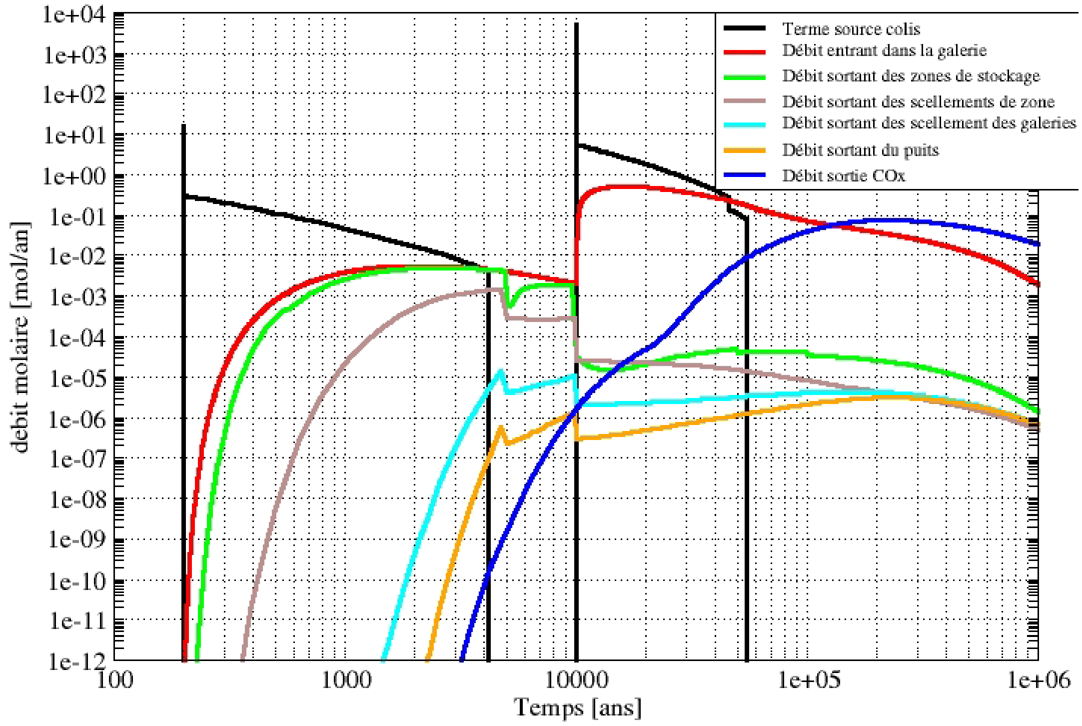


Figure 7.3-13 *« Package failure » SEA – sensitivity calculation on the influence of the hydraulic transient – record of molar flow rate – taking into account the permeability of the equivalent anchoring of seals : 10^{-10} m/s - ^{129}I – CU1 spent fuels. - [Sensitivity calculation (basic case) – Permeability of equivalent anchorings : 10^{-9} m/s – Fraction of defective packages]*

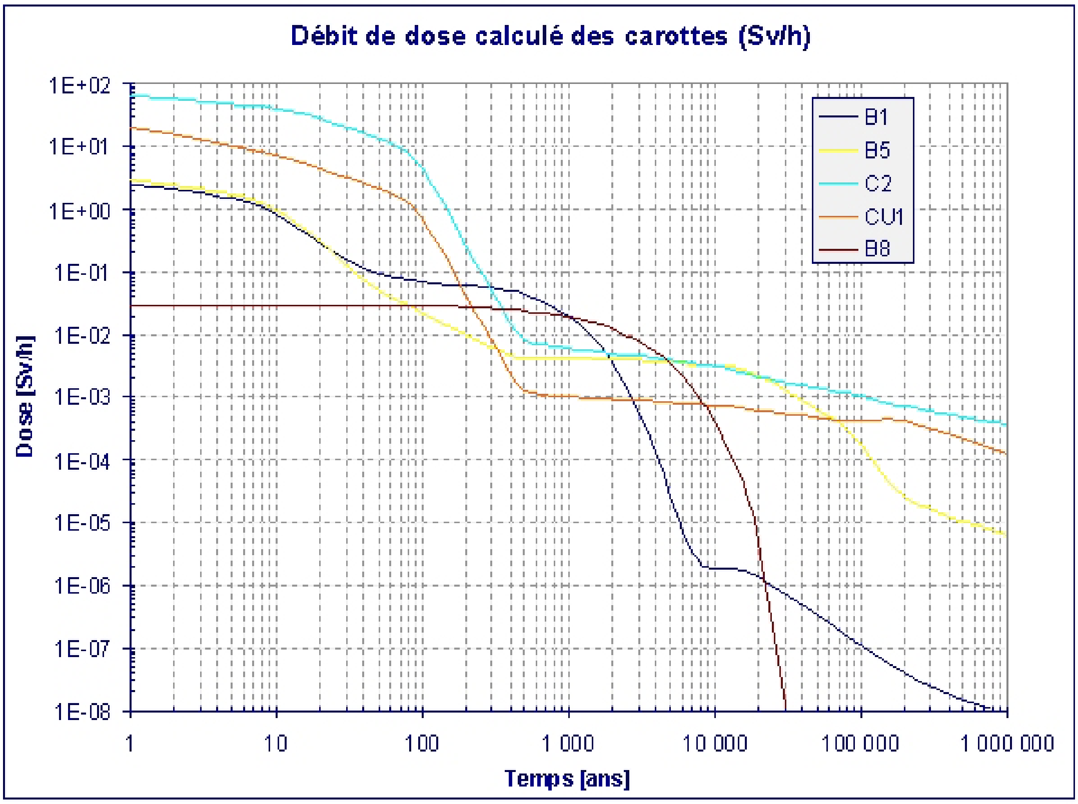


Figure 7.4-1 *Borehole in the repository with resurfacing of core waste SEA – Dose rate of core waste – B1, B5, C2, CU1, B8 reference package*

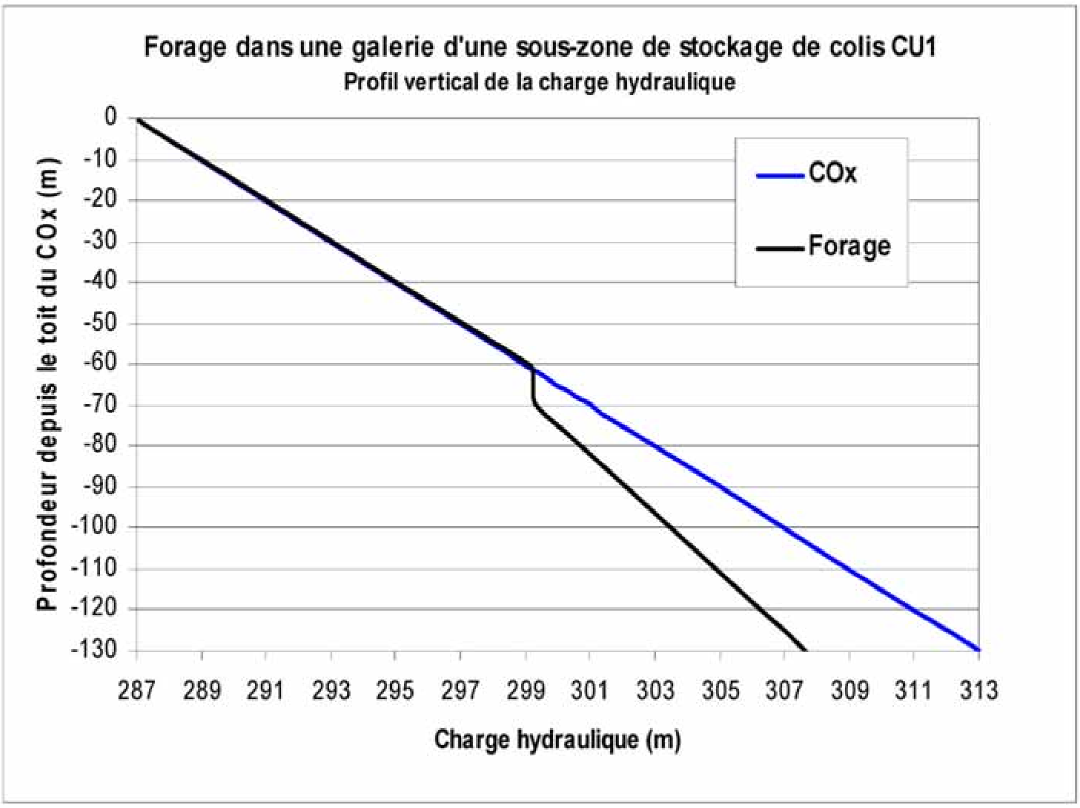


Figure 7.4-11 *Abandoned borehole in the disposal SEA – hydraulic head profile in the borehole*

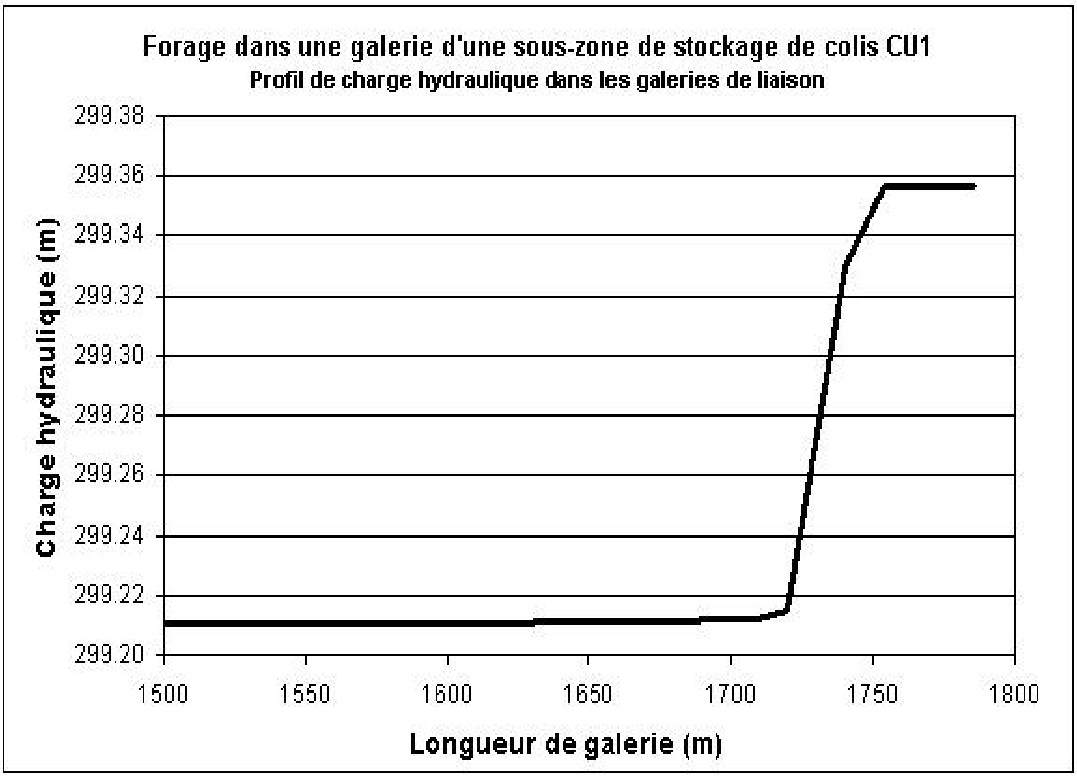


Figure 7.4-12 *Abandoned borehole in the disposal SEA – hydraulic head profile in the drifts (right-hand end represents a seal passage)*

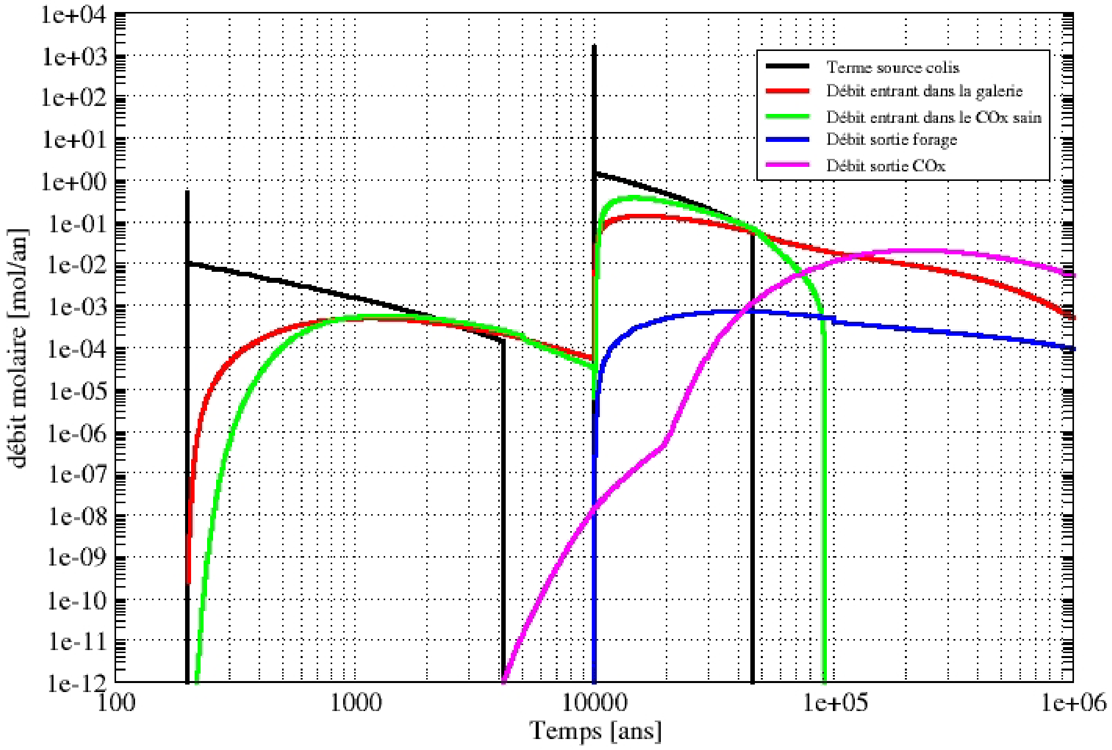


Figure 7.4-16 *SEA borehole abandoned in the repository – History of molar flow rates in the near field and details of distribution of - 129– CU1 reference packages - I transfer pathways*

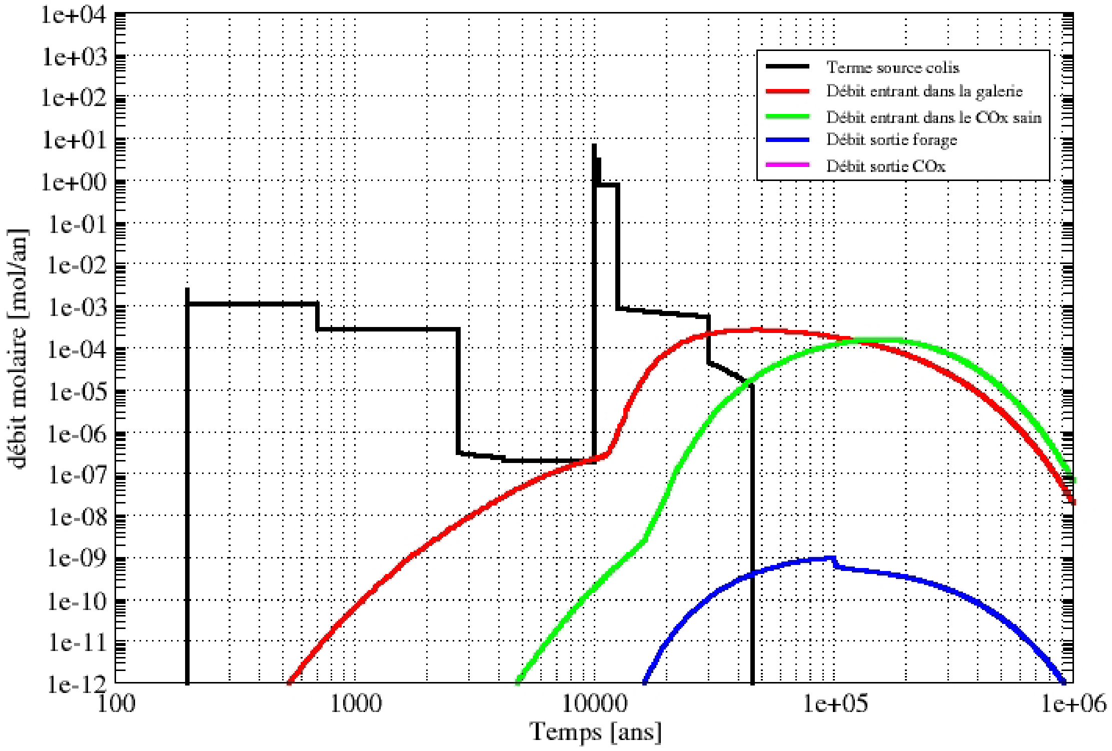


Figure 7.4-17 *SEA borehole abandoned in the repository – History of molar flow rates in the near field and details of distribution of - Ni59 -- CU1 reference packages - transfer pathways*

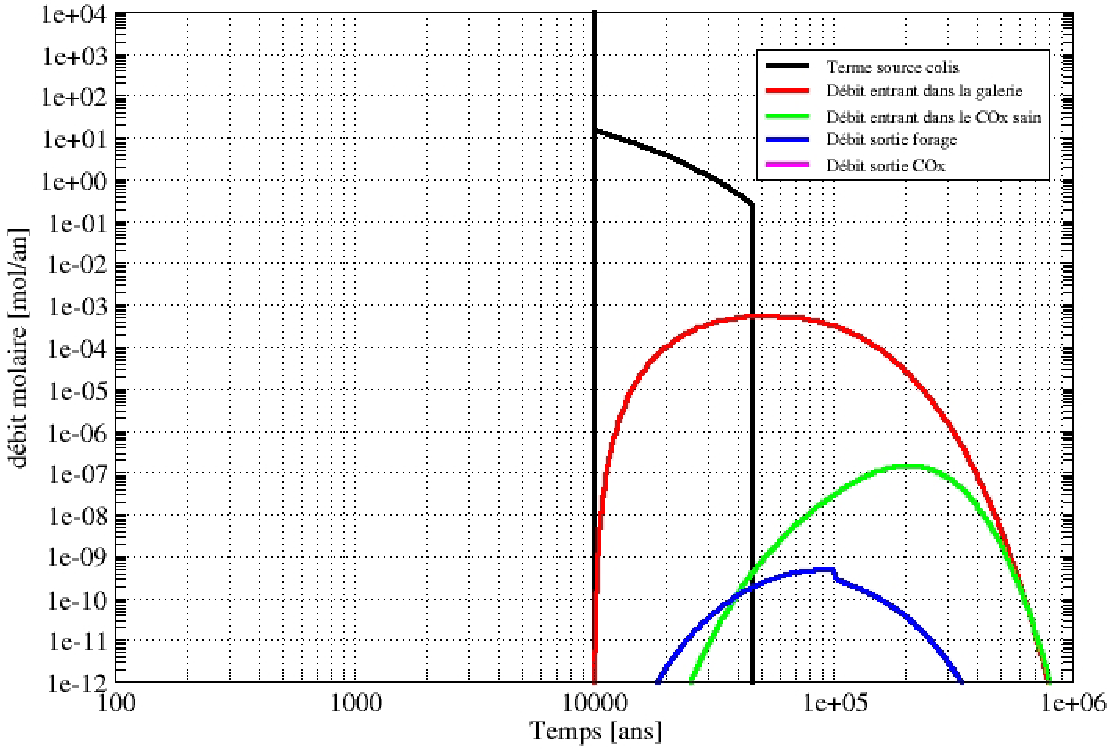


Figure 7.4-18 Borehole scenario in an access drift – Molar flow rate histories – ^{239Pu} and ^{237Np} – Reference package CU1 [^{239}Pu]

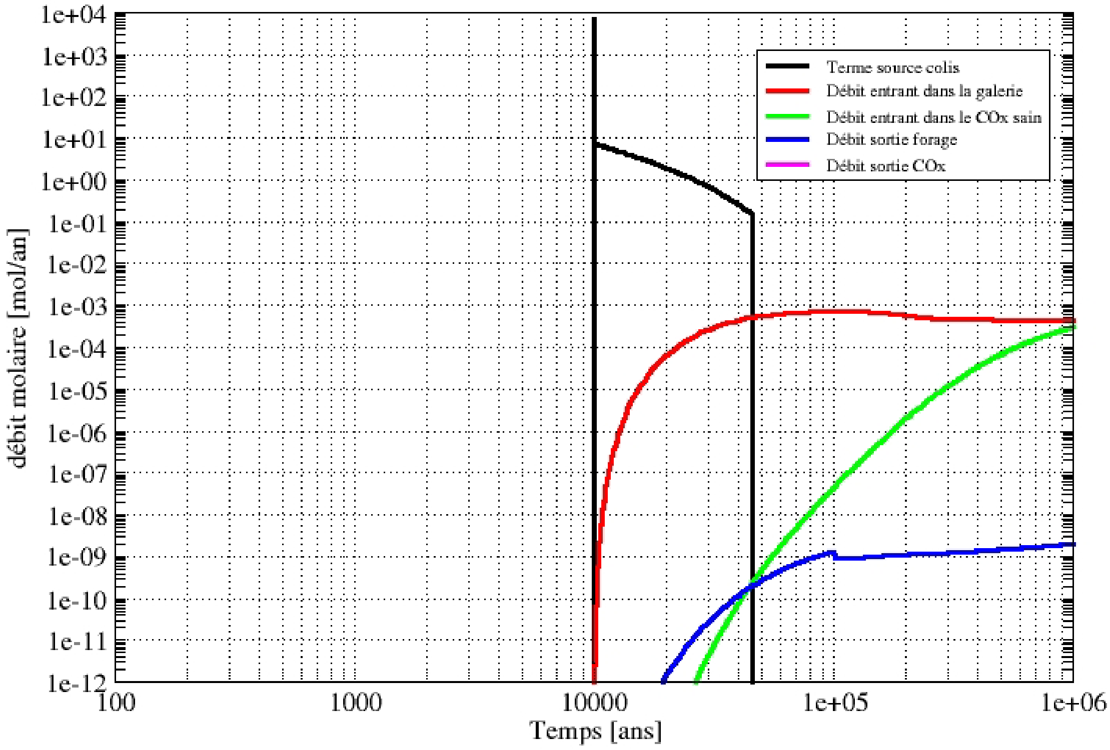


Figure 7.4-18 Borehole scenario in an access drift – Molar flow rate histories – ^{239Pu} and ^{237Np} – Reference package CU1 [^{237}Np]

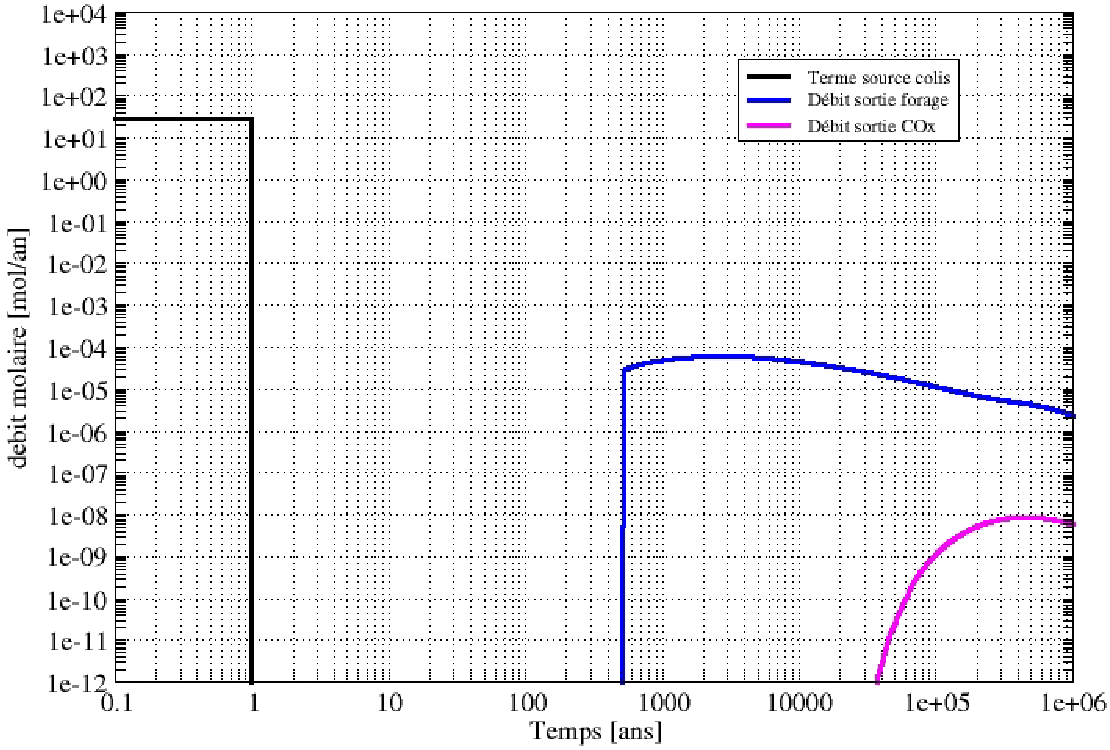


Figure 7.4-20 SEA borehole abandoned in a B waste cell – History of molar flow rates in the near field - B1x - ^{129}I packages

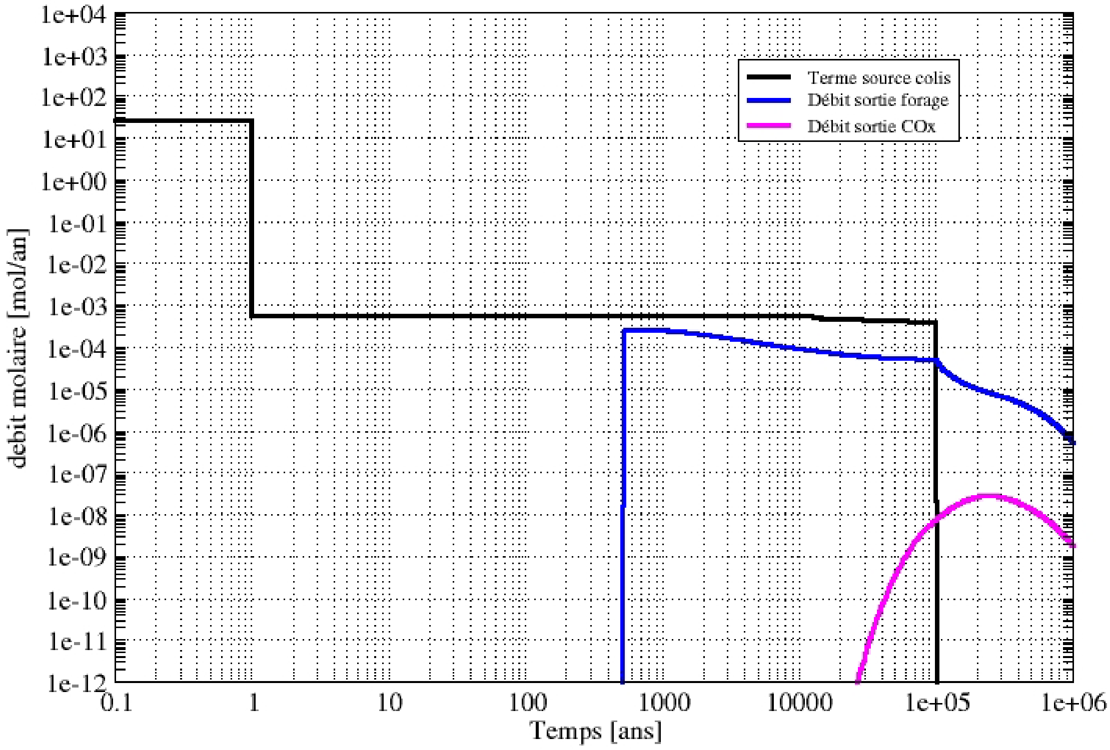


Figure 7.4-21 SEA borehole abandoned in a B waste cell – History of molar flow rates in the near field - B1x - ^{36}Cl packages

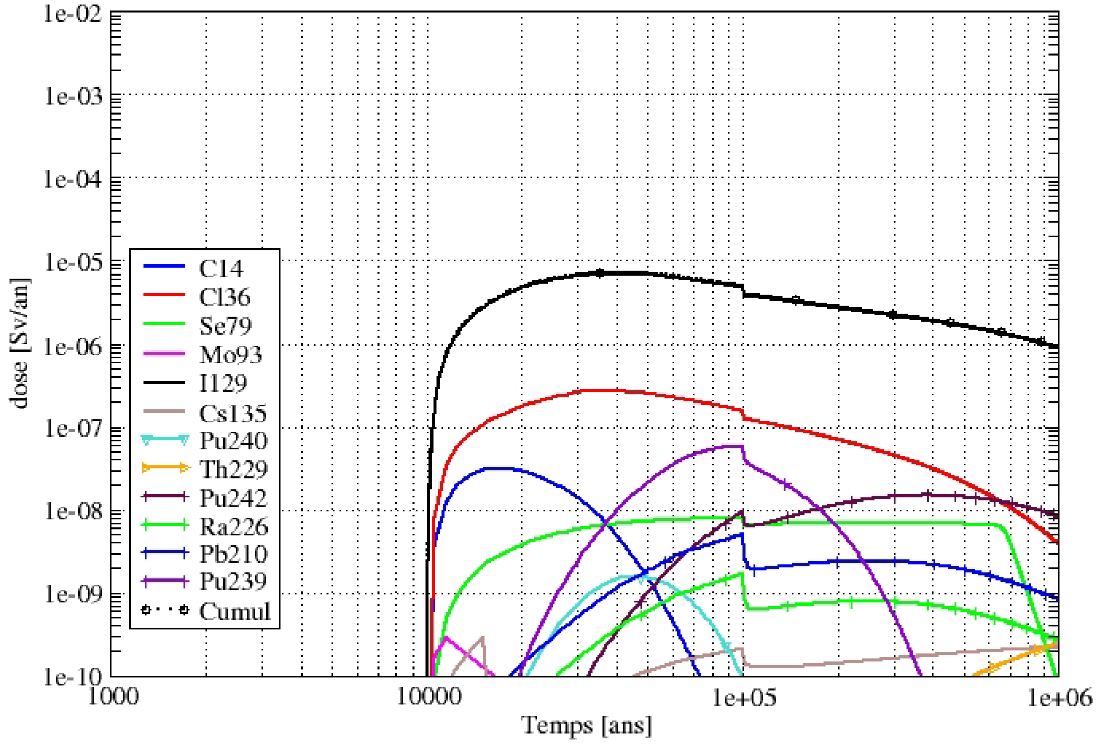


Figure 7.4-22 Dose Due to the Borehole at the Barrois DWS Pumping Outlet (All Radionuclides) - Borehole Penetrating an CU1-Uox3 reference Package Module

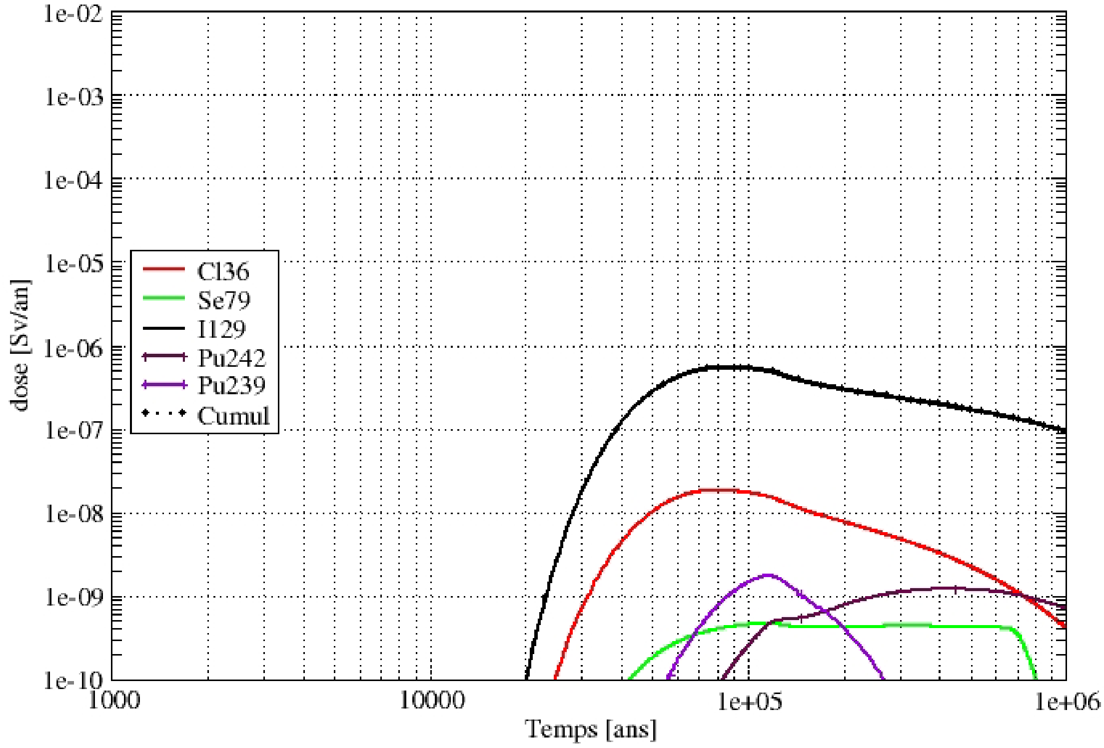


Figure 7.4-23 Dose Due to Borehole at the « Saulx » Outlet (Fission and Activation Products) - Borehole in a Standard CU1-Uox3 Package Module

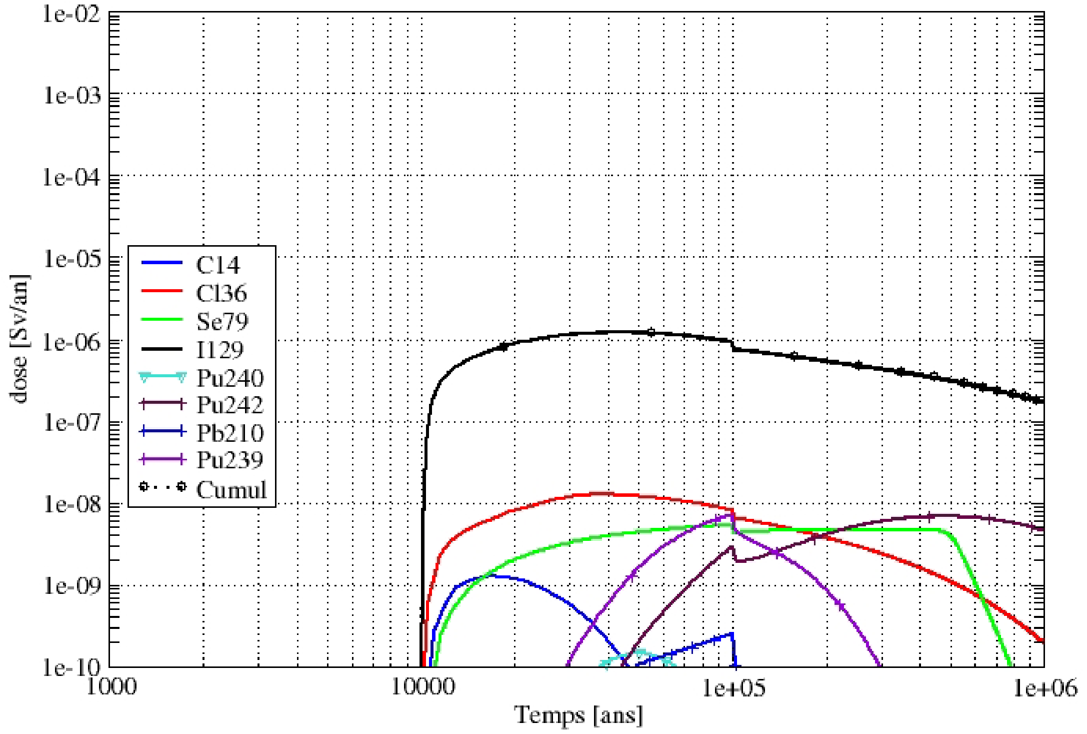


Figure 7.4-24 *SEA - Abandoned Borehole Penetration of the Repository – Reference Calculation - Dose Due to the Borehole at the Barrois Limestone DWS Pumping Outlet (All Radionuclides) - Borehole Penetration of a Standard CU2 Package Module*

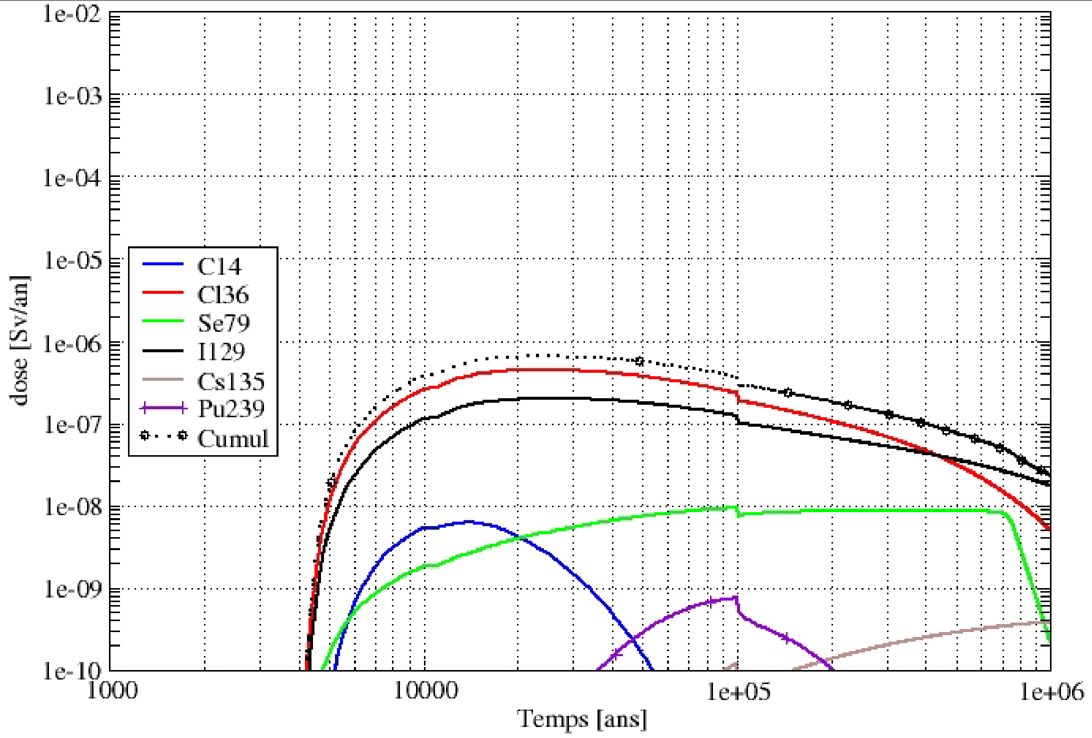


Figure 7.4-25 *SEA - Abandoned Borehole Penetration of the Repository - Reference Calculation - Dose Due to the Borehole at the Barrois Limestone DWS Pumping Outlet (All Radionuclides) - Borehole Penetration of a C2 Reference package Module*

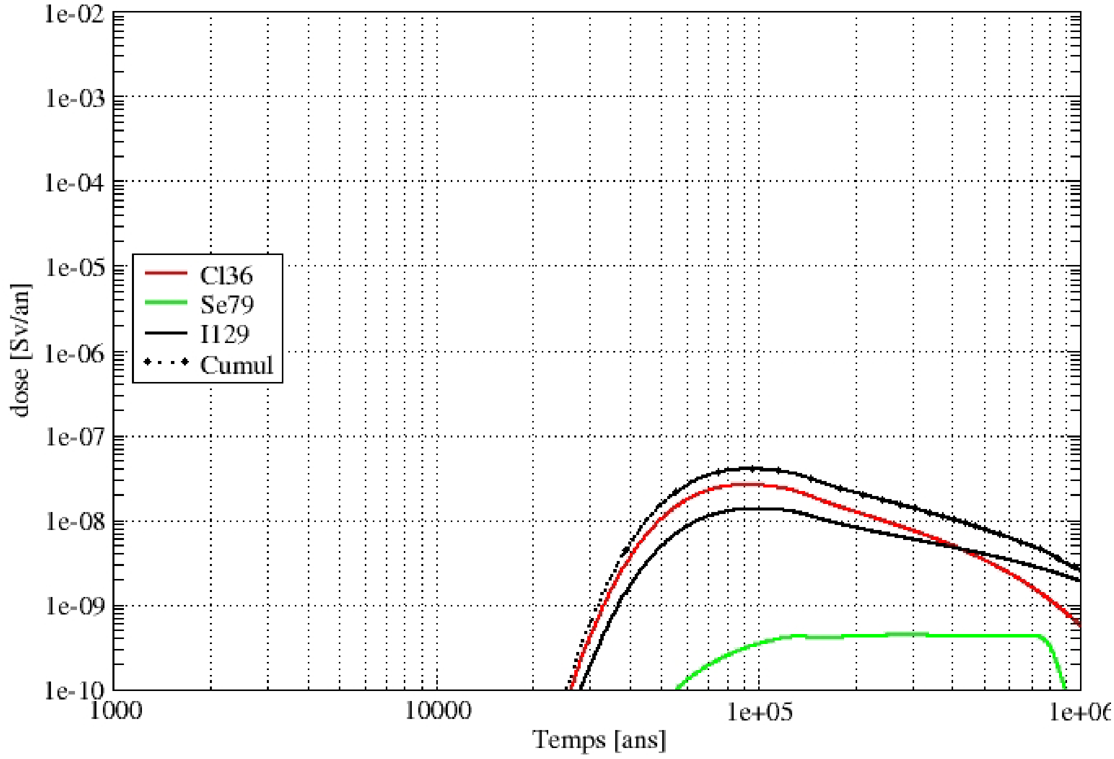


Figure 7.4-26 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity Calculations - Dose Due to the Borehole at the « Saulx » Outlet

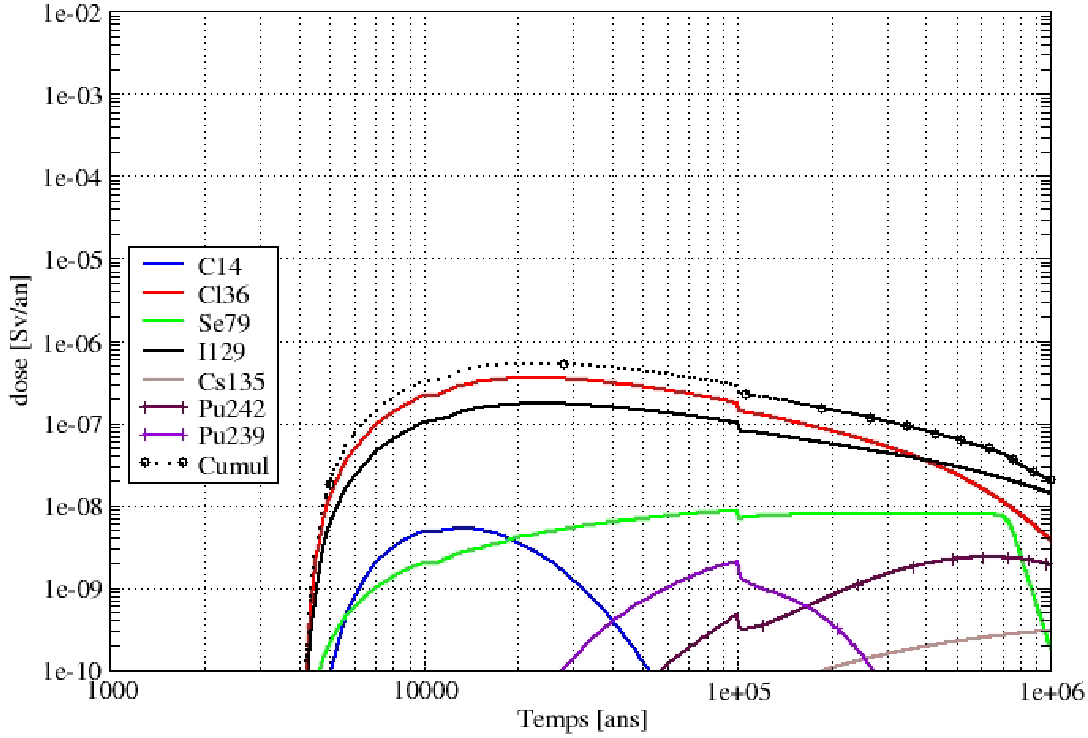


Figure 7.4-27 SEA - Abandoned Borehole Penetration of the Repository - Reference Calculation - Dose Due to the Borehole at the Barrois Limestone DWS Pumping Outlet (All Radionuclides) - Borehole Penetration of C4 Reference package

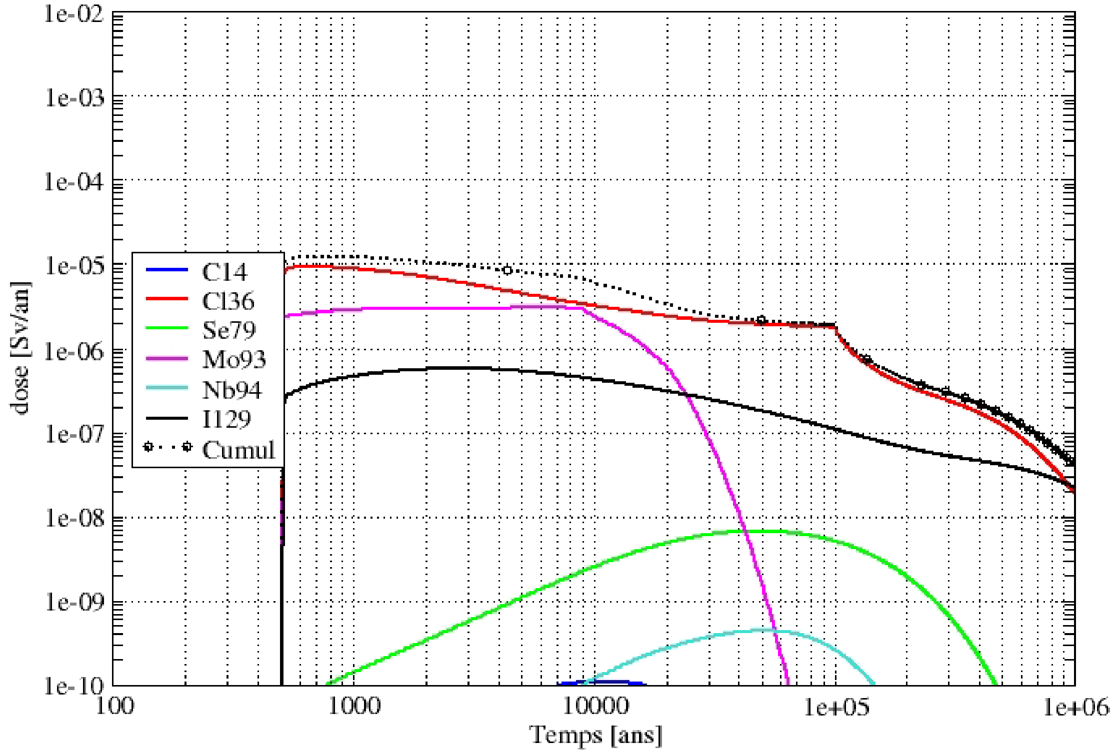


Figure 7.4-28 *SEA - Abandoned Borehole Penetration of the Repository - Dose Due to the Borehole at the Barrois DWS Pumping Outlet - B1x reference package*

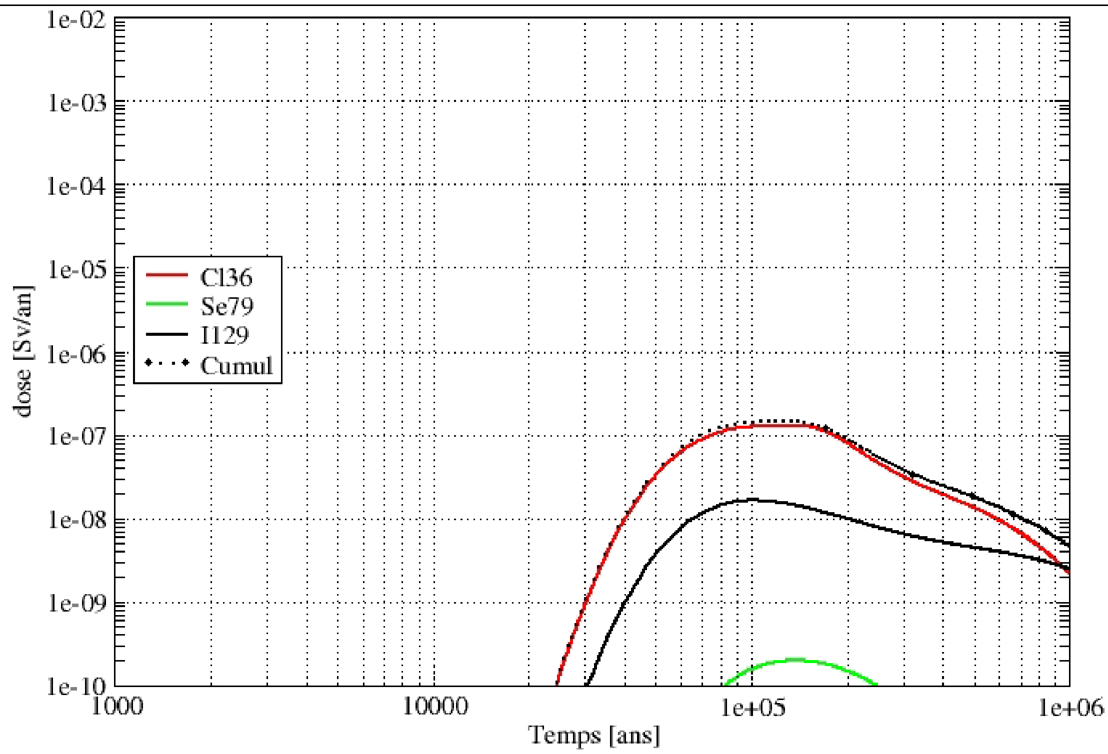


Figure 7.4-29 *SEA - Abandoned Borehole Penetration of the Repository - Dose Due to the Borehole -Pumping Outlet in the Vicinity of the Saulx - B1x reference package*

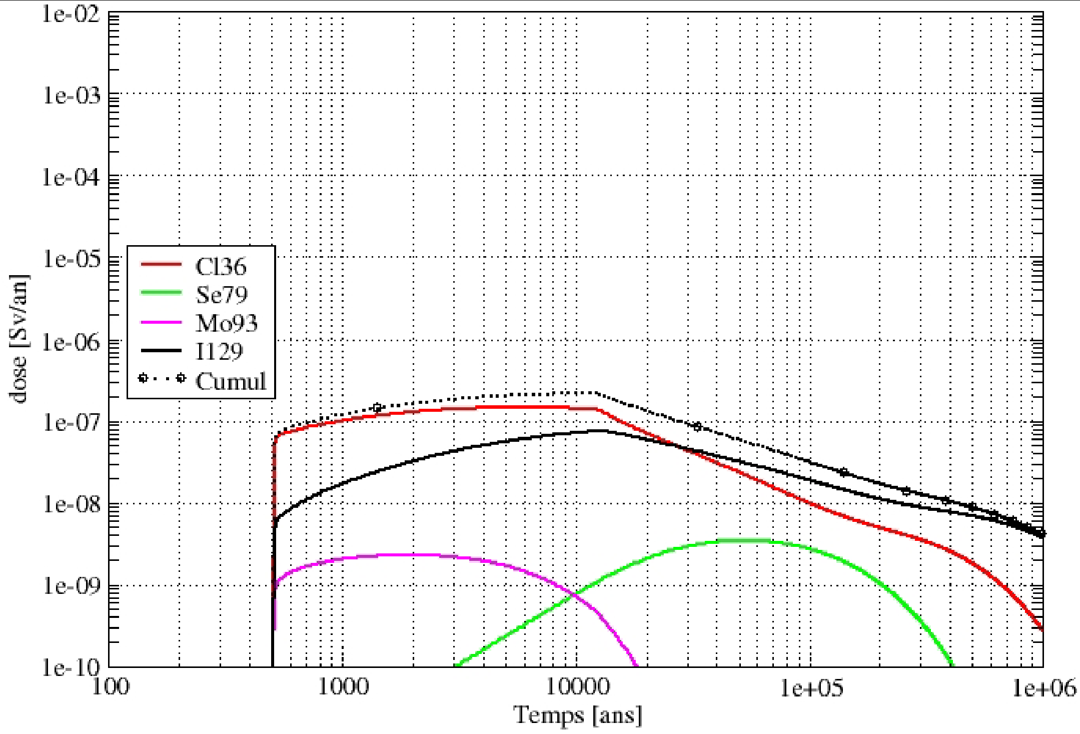


Figure 7.4-30 *SEA - Abandoned Borehole Penetration of the Repository - Dose Due to Borehole - Barrois DWS Pumping Outlet - B2 reference package*

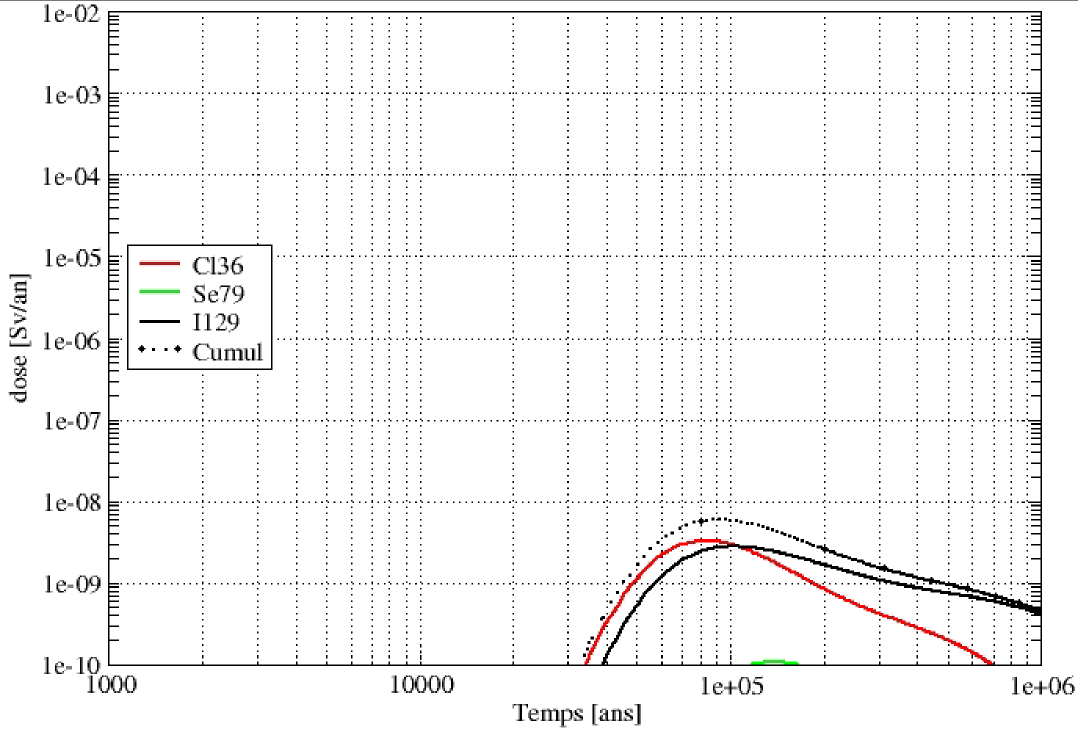


Figure 7.4-31 *SEA - Abandoned Borehole Penetration of the Repository - Dose Due to Borehole -Pumping Outlet in the Vicinity of the Saulx - B2 Bitumen reference package*

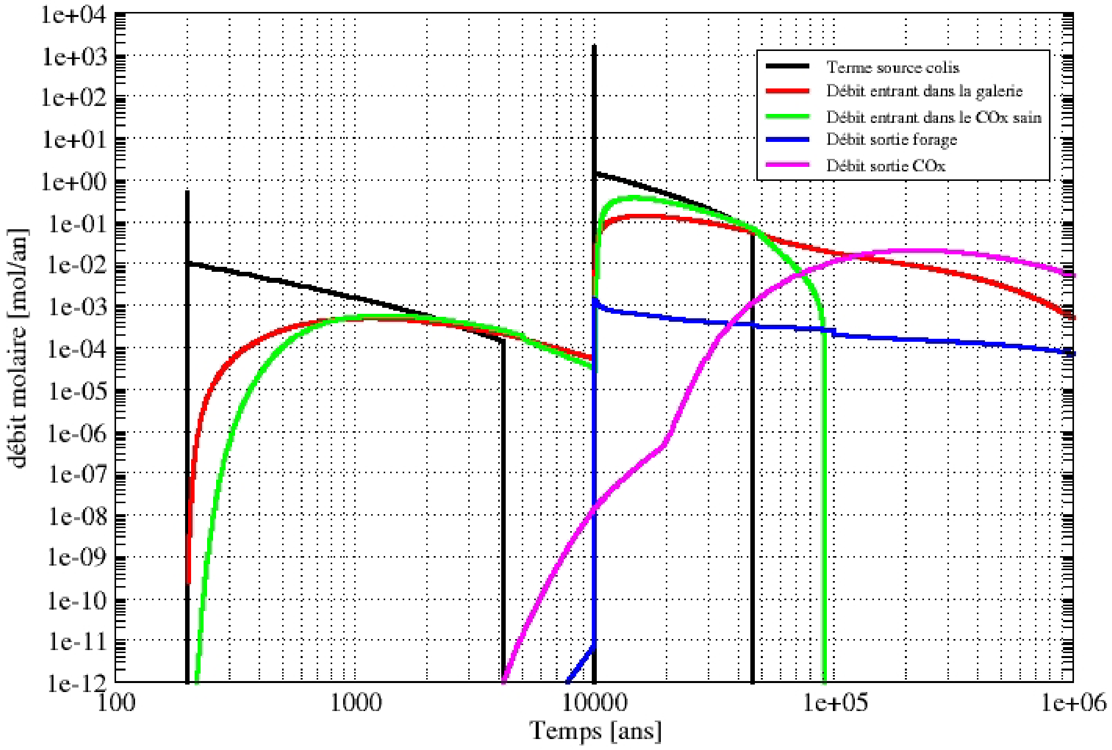


Figure 7.4-33 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the Position of the Borehole - History of the Molar Flow Rate CU1 - UOx3 - ^{129}I - [Sensitivity Calculations - Borehole Penetration of a Cell]

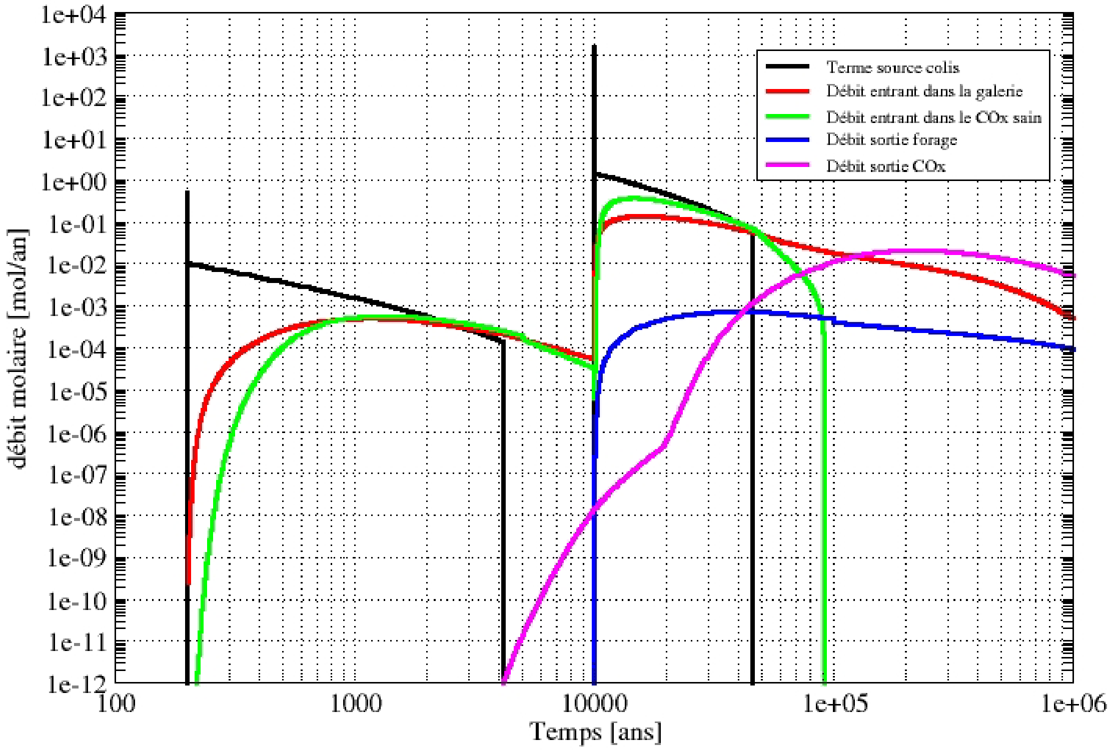


Figure 7.4-33 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the Position of the Borehole - History of the Molar Flow Rate CU1 - UOx3 - ^{129}I - [Reference Calculation – Borehole Penetration of an Access Drift]

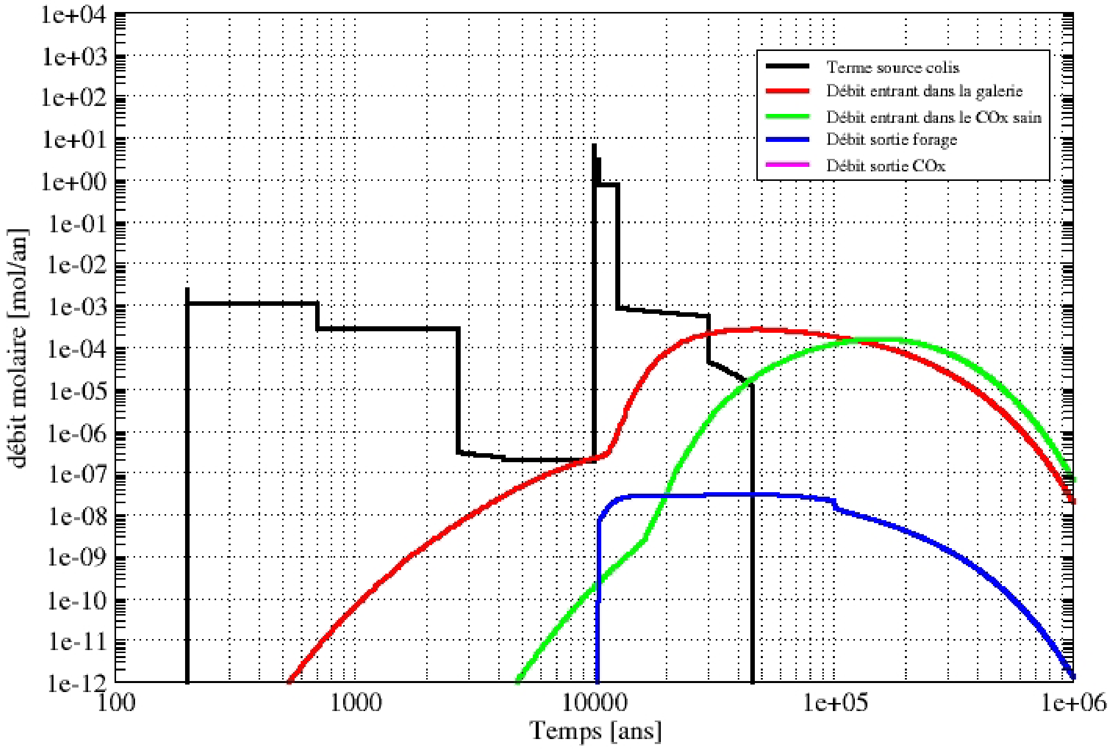


Figure 7.4-34 *SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the Position of the Borehole - History of the Molar Flow Rate SF - ^{59}Ni - [Sensitivity Calculations - Borehole Penetration of a Cell]*

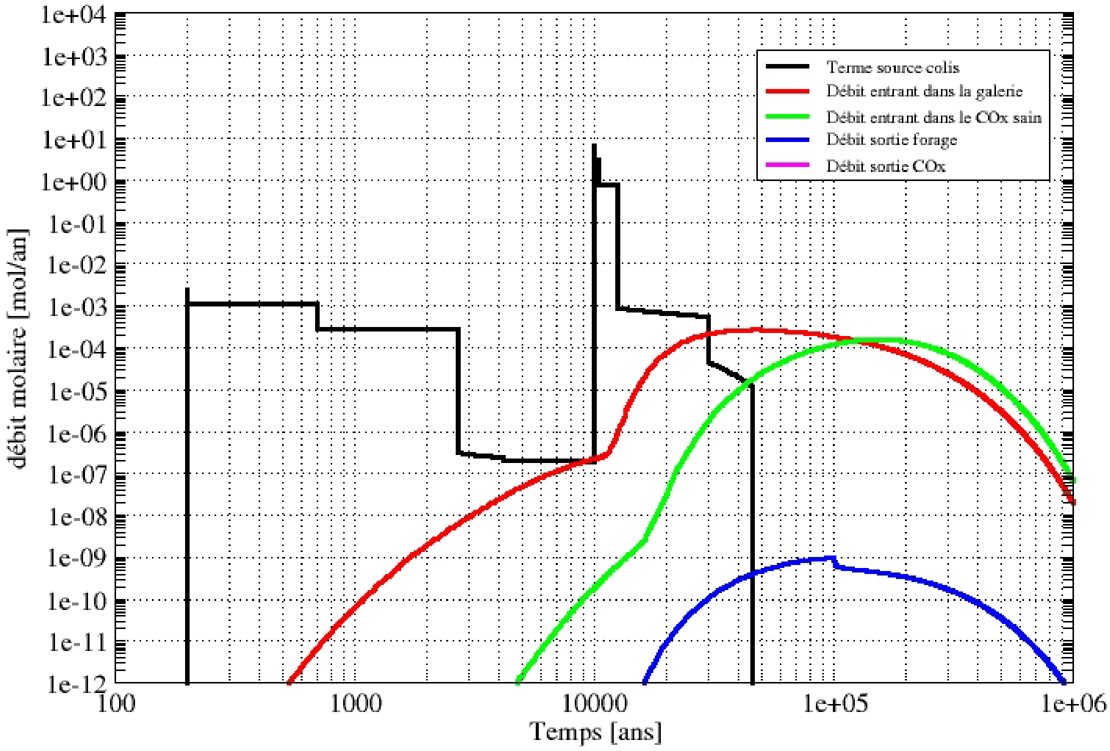


Figure 7.4-34 *SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the Position of the Borehole - History of the Molar Flow Rate SF - ^{59}Ni - [Reference Calculation – Borehole Penetration of an Access Drift]*

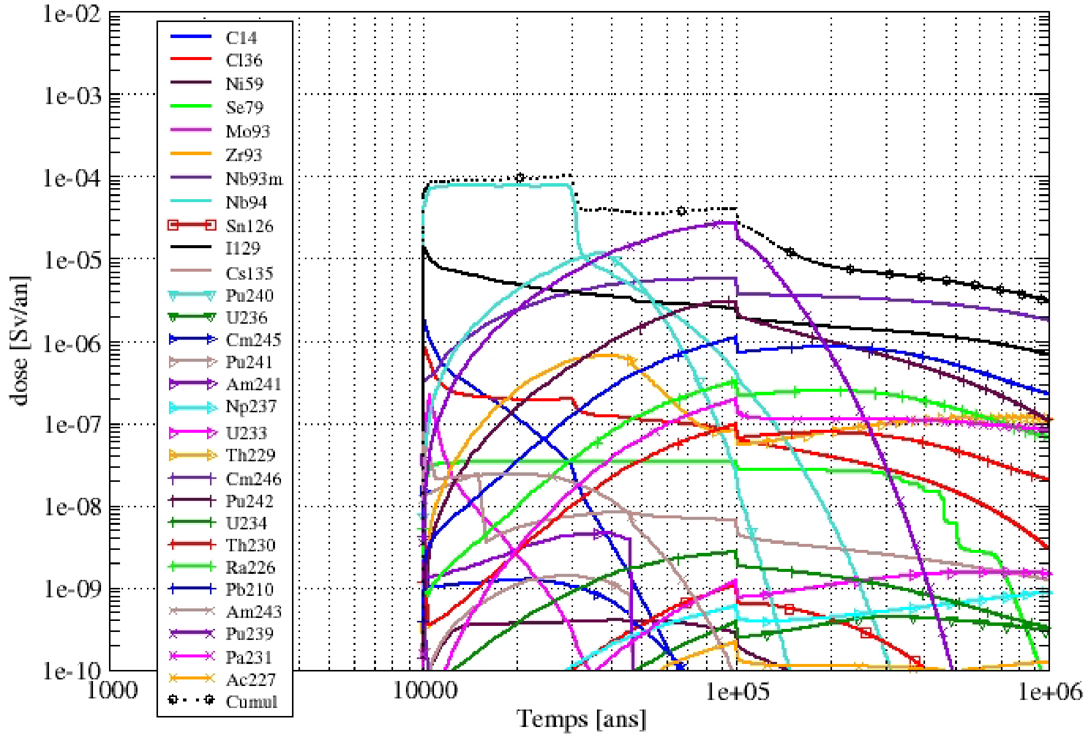


Figure 7.4-35 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the Position of the Borehole - History of the Dose at the DWS Pumping Outlet - CU1 - UOx3 - [Sensitivity Calculations - Borehole Penetration of a Cell]

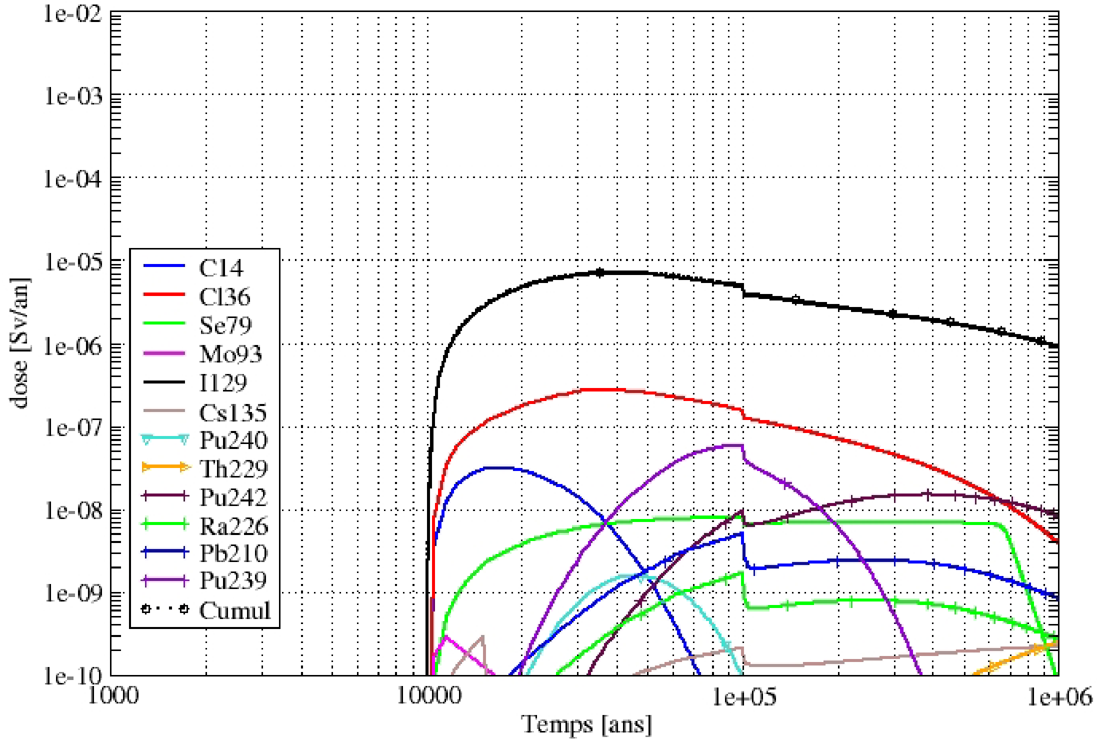


Figure 7.4-35 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the Position of the Borehole - History of the Dose at the DWS Pumping Outlet - CU1 - UOx3 - [Reference Calculation – Borehole Penetration of an Access Drift]

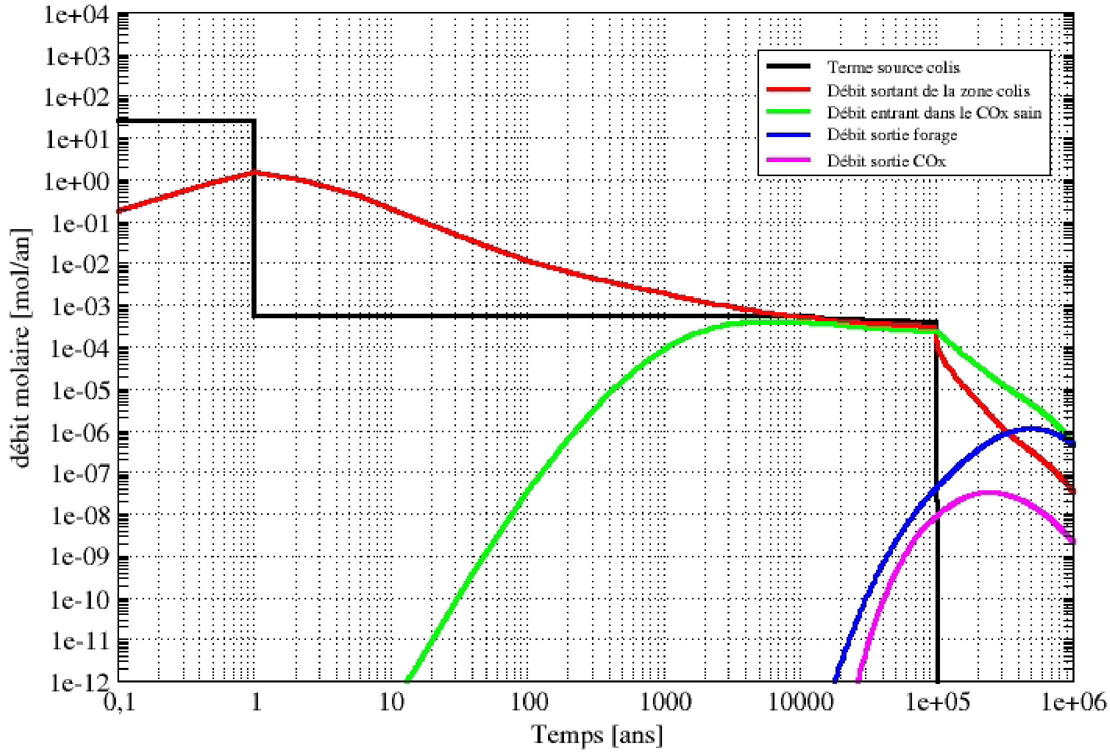


Figure 7.4-37 SEA - Abandoned Borehole Penetration of the Repository – Sensitivity to the Position of the Borehole - History of the Molar Flow Rate - Non-Organic reference package that Do Not Release Hydrogen - ^{36}Cl - [Sensitivity Calculations - Borehole Penetration of a Connecting Drift]

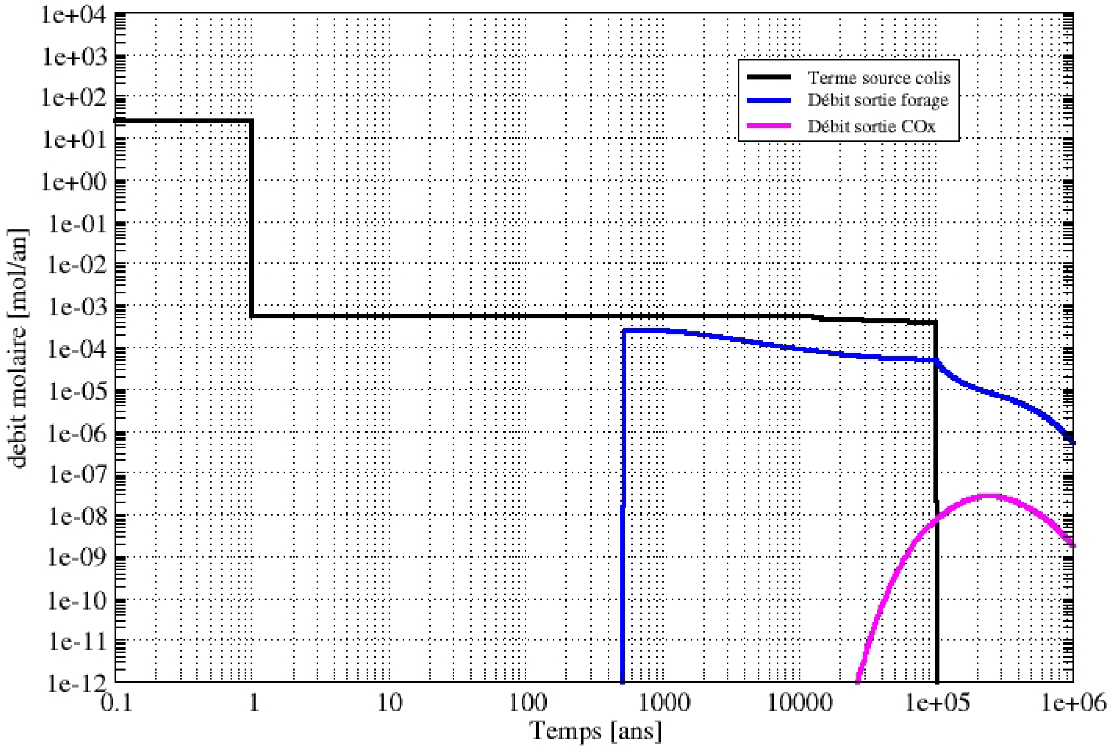


Figure 7.4-37 SEA - Abandoned Borehole Penetration of the Repository – Sensitivity to the Position of the Borehole - History of the Molar Flow Rate - Non-Organic reference package that Do Not Release Hydrogen - ^{36}Cl - [Reference Calculation – Borehole Penetration of a B waste Cell]

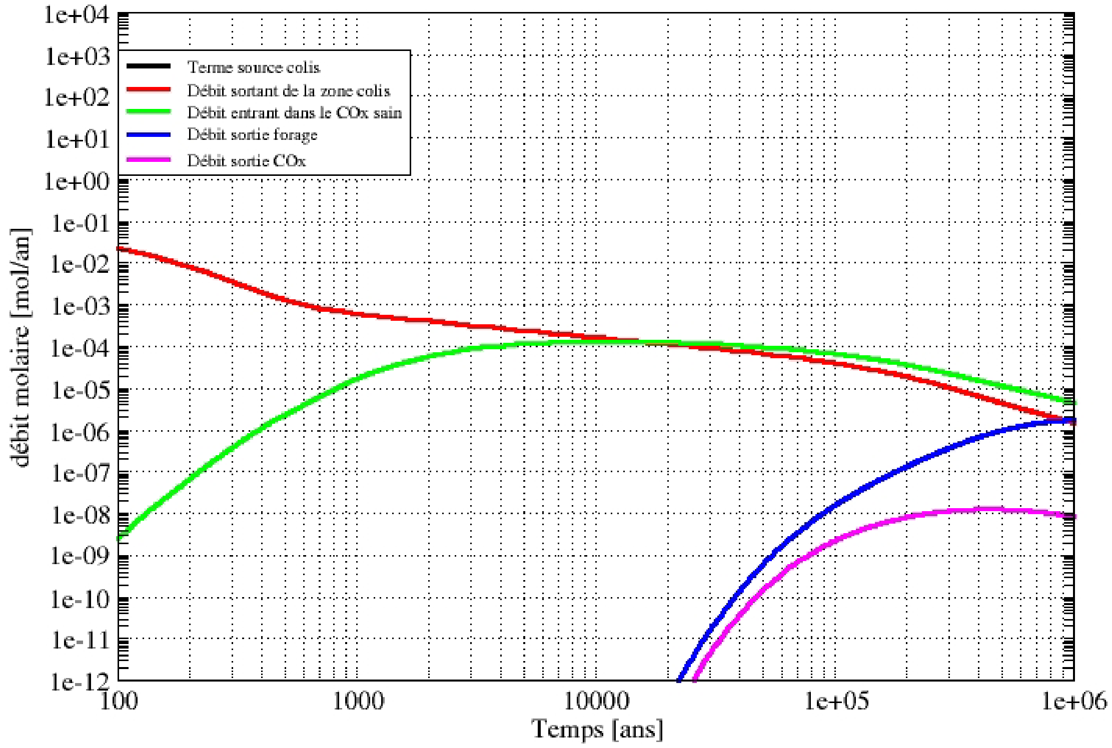


Figure 7.4-38 SEA - Abandoned Borehole Penetration of the Repository – Sensitivity to the Position of the Borehole - History of the Molar Flow Rate - Non-Organic reference package that Do Not Release Hydrogen - 129I - [Sensitivity Calculations - Borehole Penetration of a Connecting Drift]

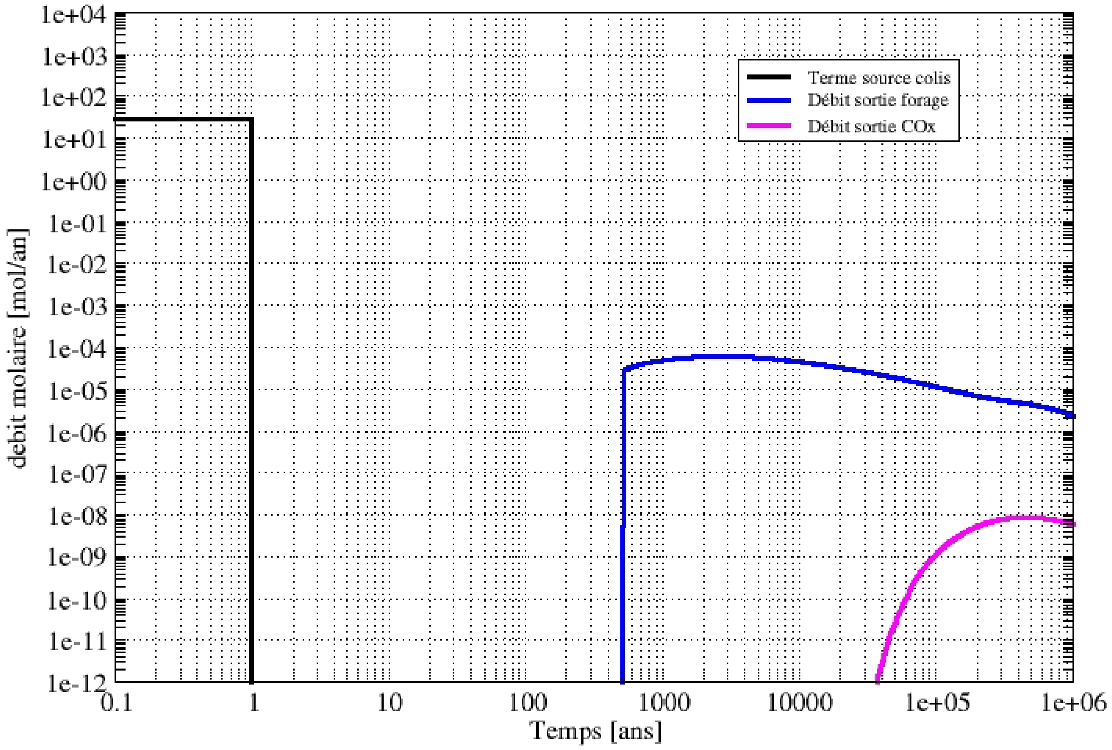


Figure 7.4-38 SEA - Abandoned Borehole Penetration of the Repository – Sensitivity to the Position of the Borehole - History of the Molar Flow Rate - Non-Organic reference package that Do Not Release Hydrogen - 129I - [Reference Calculation – Borehole Penetration of a B waste Cell]

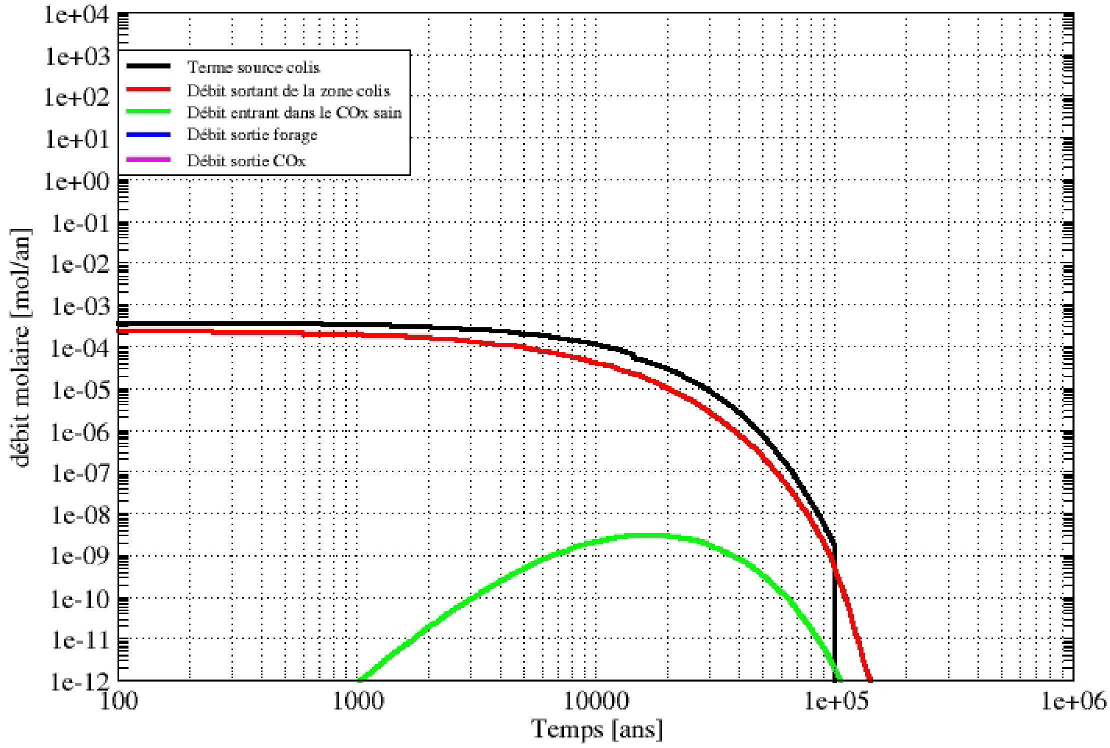


Figure 7.4-39 *SEA - Abandoned Borehole Penetration of the Repository – Sensitivity to the Position of the Borehole - History of the Molar Flow Rate - Non-Organic reference package that Do Not Release Hydrogen –* ^{14}C *- [Sensitivity Calculations - Borehole Penetration of a Connecting Drift]*

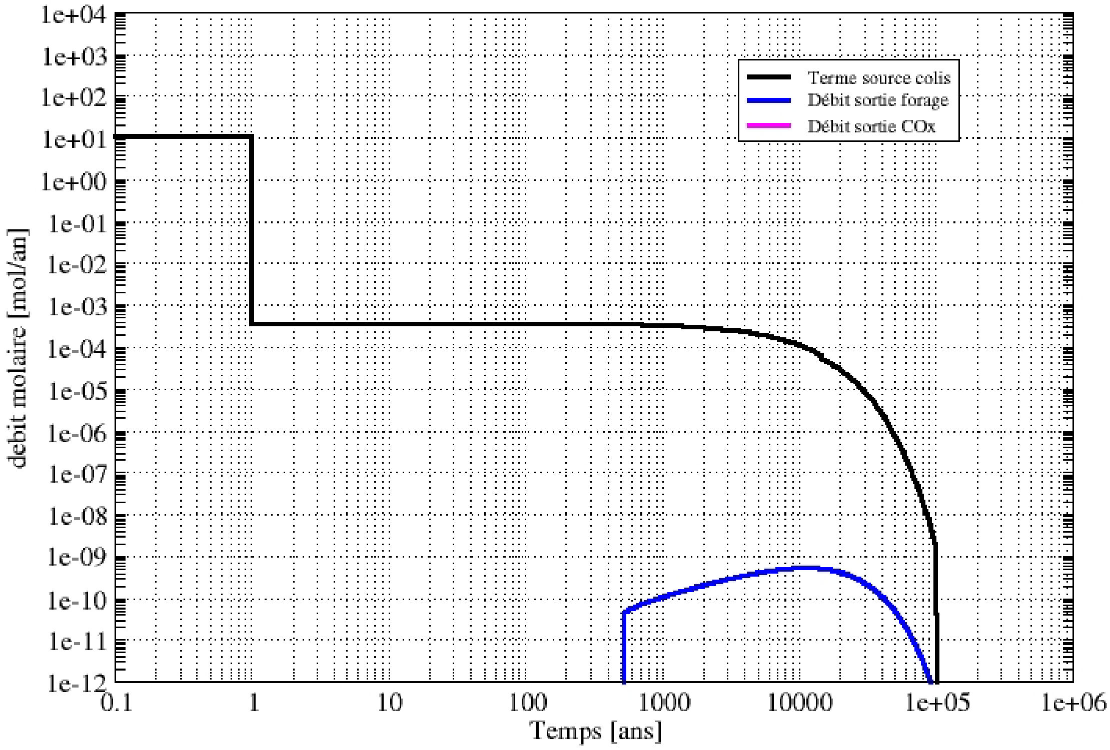


Figure 7.4-39 *SEA - Abandoned Borehole Penetration of the Repository – Sensitivity to the Position of the Borehole - History of the Molar Flow Rate - Non-Organic reference package that Do Not Release Hydrogen –* ^{14}C *- [Reference Calculation – Borehole Penetration of a B waste Cell]*

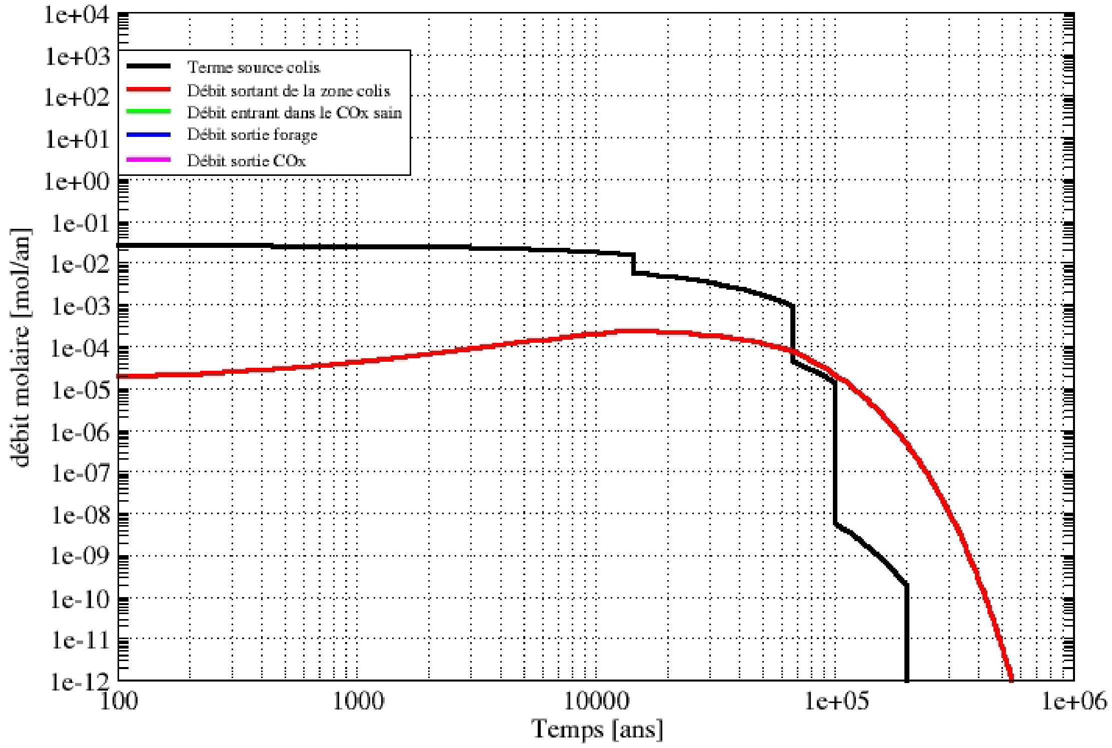


Figure 7.4-40 SEA - Abandoned Borehole Penetration of the Repository – Sensitivity to the Position of the Borehole - History of the Molar Flow Rate - Non-Organic reference package that Do Not Release Hydrogen ^{59Ni} *- [Sensitivity Calculations - Borehole Penetration of a Connecting Drift]*

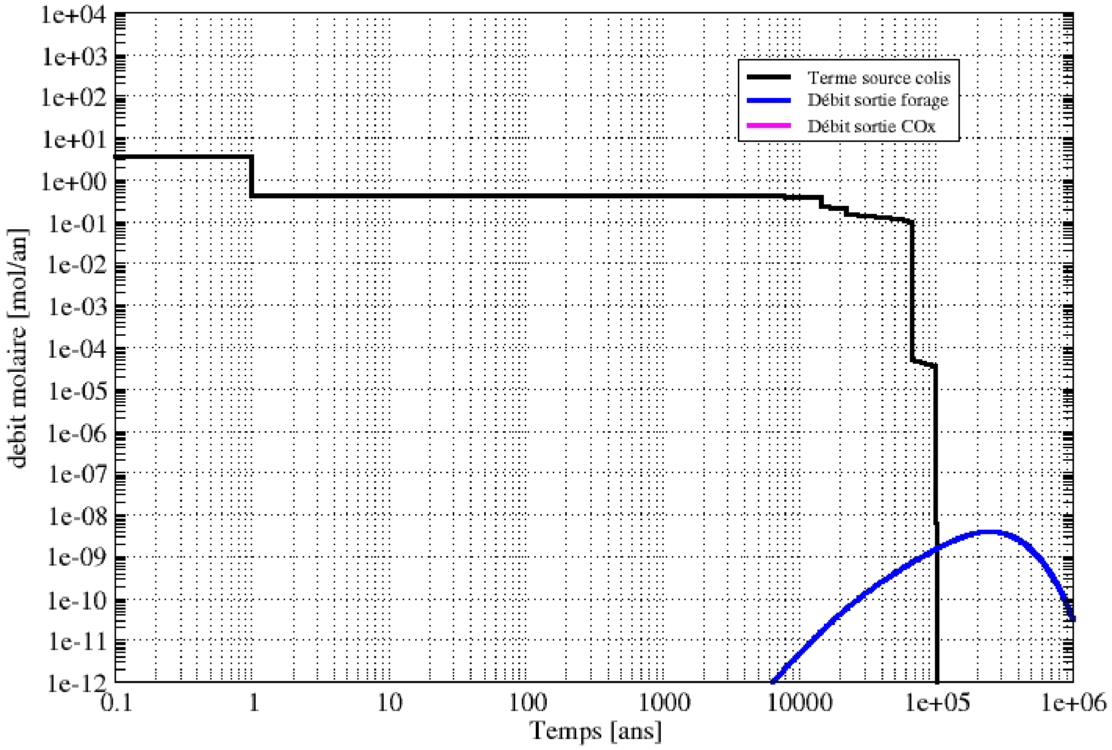


Figure 7.4-40 SEA - Abandoned Borehole Penetration of the Repository – Sensitivity to the Position of the Borehole - History of the Molar Flow Rate - Non-Organic reference package that Do Not Release Hydrogen ^{59Ni} *- [Reference Calculation – Borehole Penetration of a B waste Cell]*

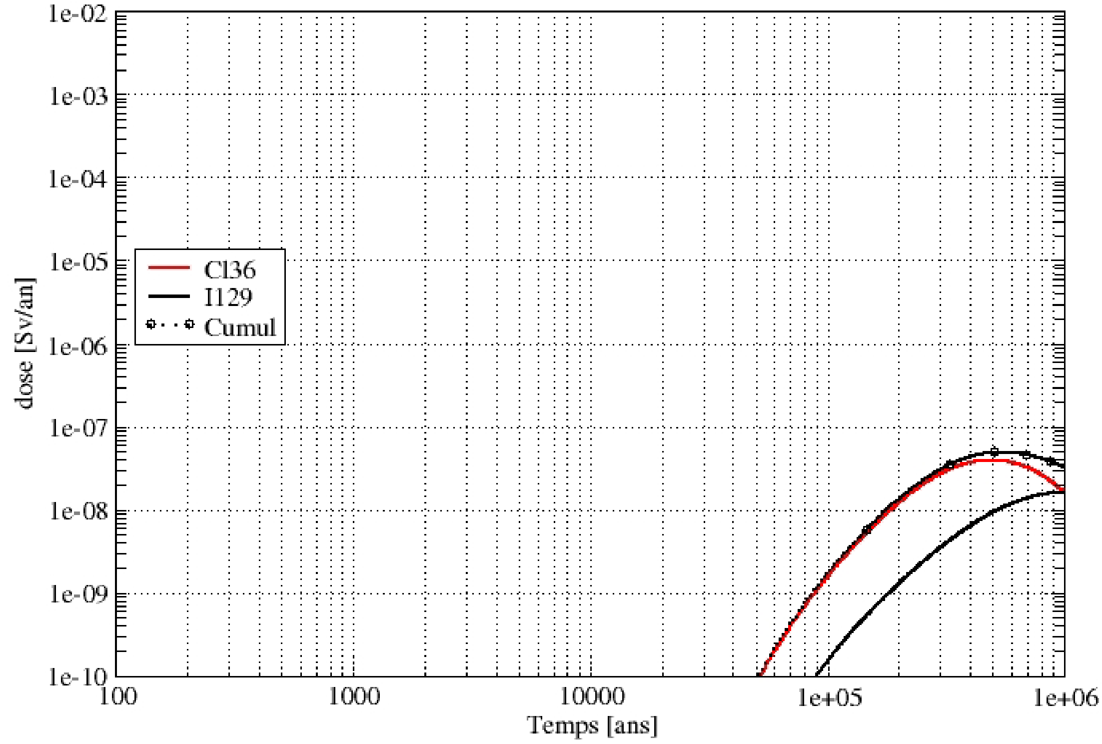


Figure 7.4-41 *SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the Position of the Borehole (Borehole Penetration of a B Waste Connection Drift) - History of the Dose at the DWS Pump Outlet - Non-Organic reference package that Do Not Release Hydrogen (B1x)*

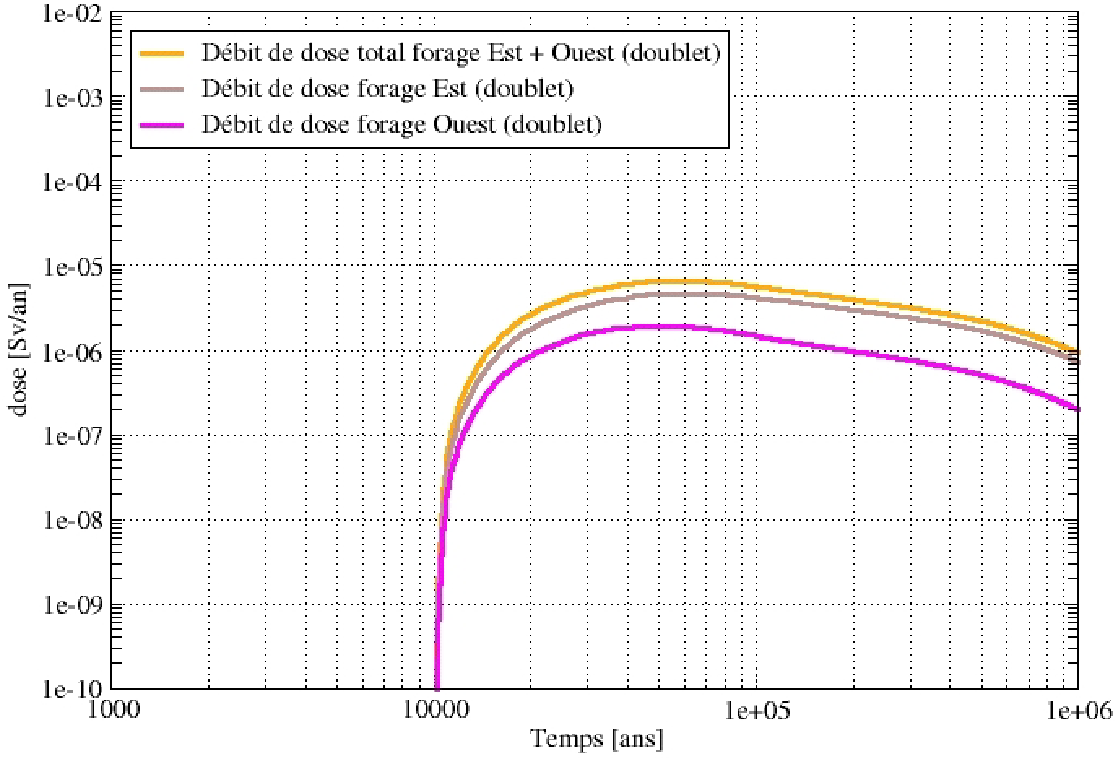


Figure 7.4-44 *SEA - Abandoned Borehole Penetration of the Repository - Sensitivity - Two-Borehole Configuration at the Ends of a Drift (Situation 1) Dose History - CU1 Spent Fuel* ^{129}I

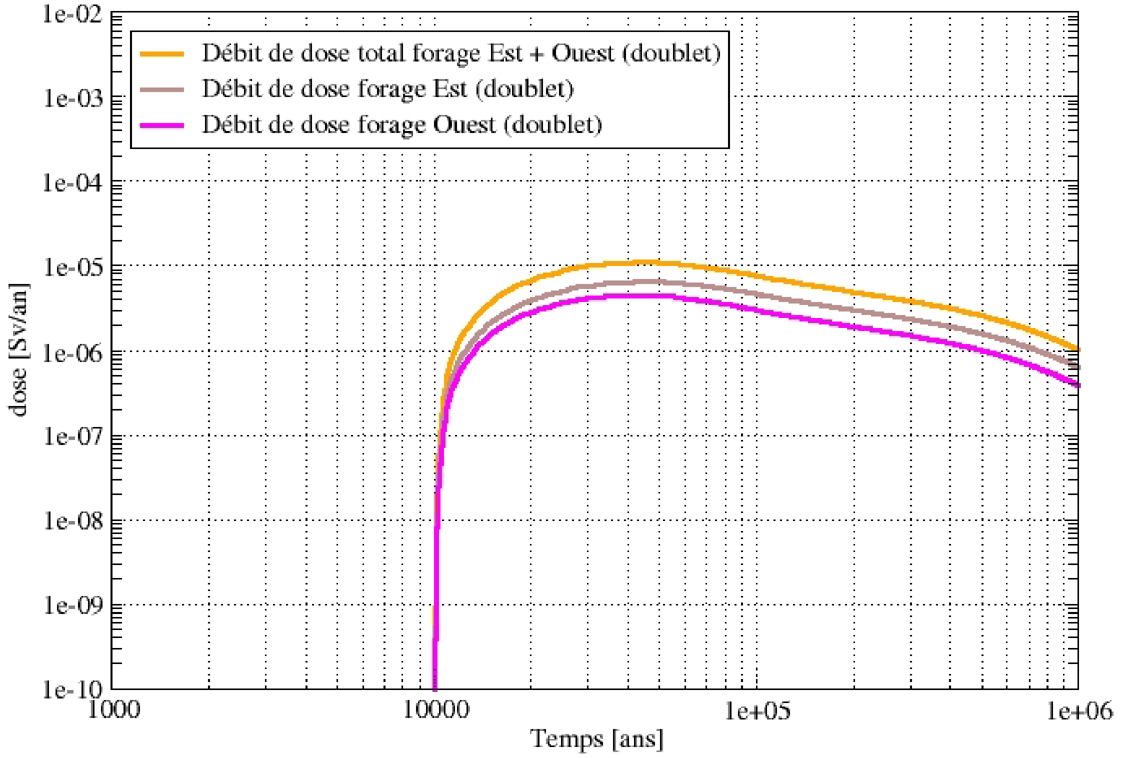


Figure 7.4-45 *SEA - Abandoned Borehole Penetration of the Repository – Sensitivity - Two-Borehole Configuration in the Middle of Two Repository Half-Modules (Situation 2) - Dose History - CU1 Spent Fuel ^{129}I*

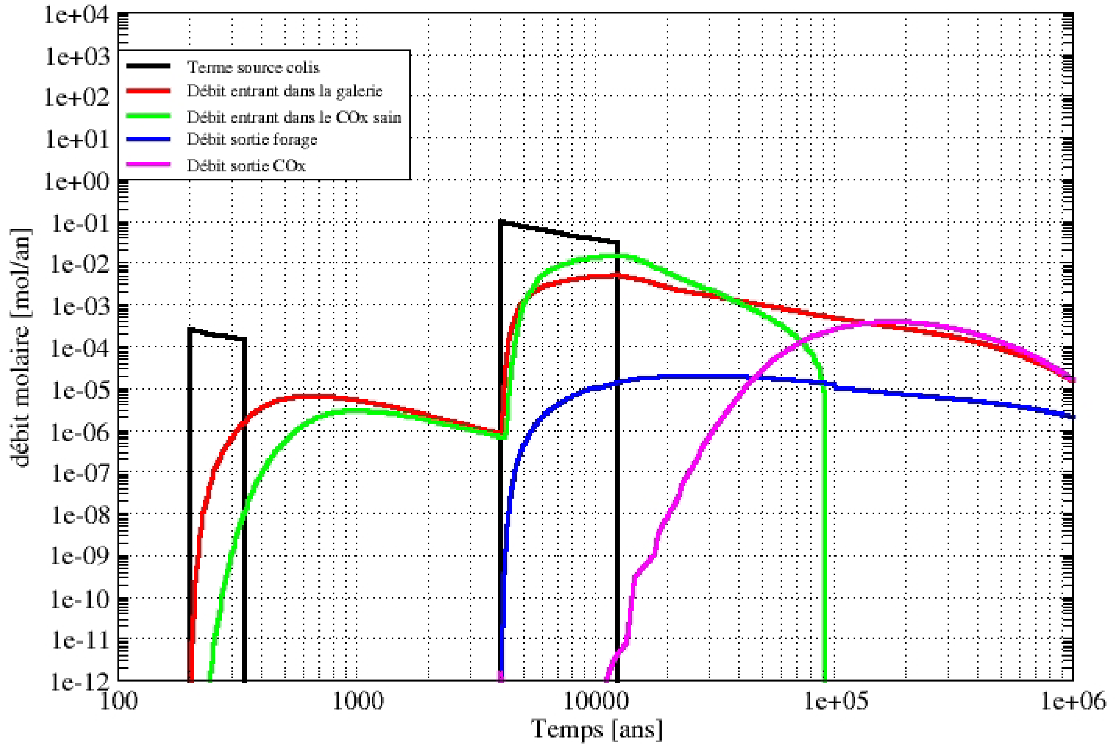


Figure 7.4-46 *SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the C Waste Design (Alterative Design with Clay Engineered Barrier) - Molar Flow Rate History Data - C2 reference package - ^{129}I - [Sensitivity Calculations : Short Tunnel with Clay engineered barrier]*

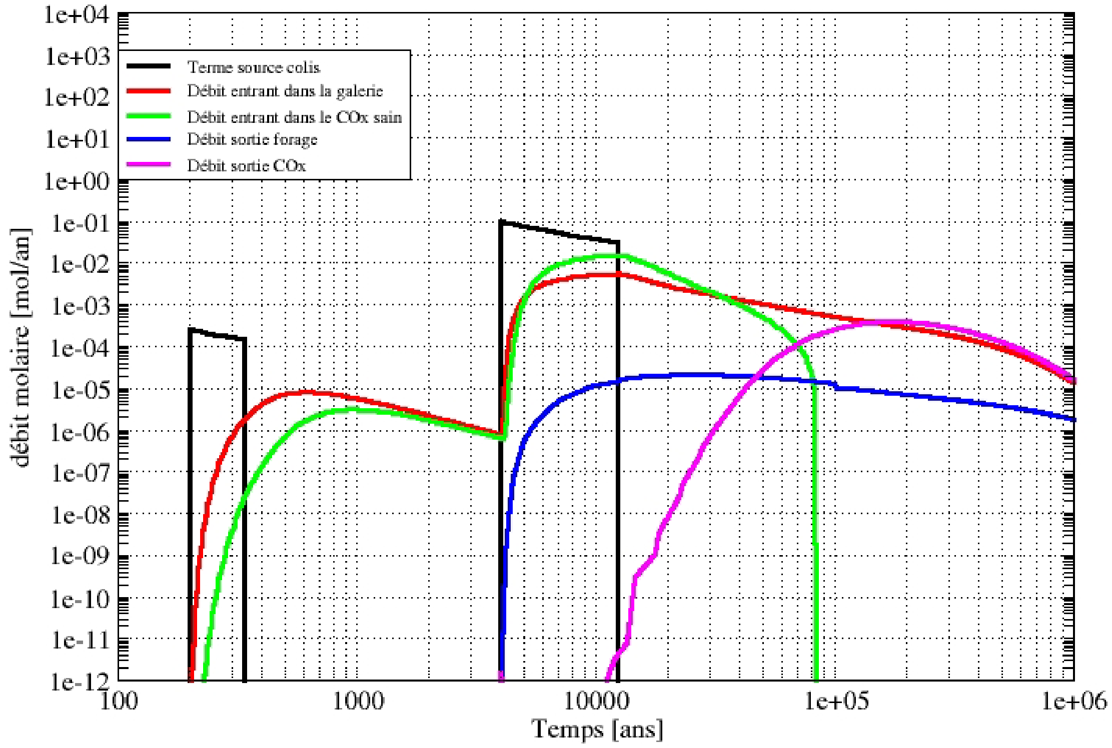


Figure 7.4-46 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the C Waste Design (Alterative Design with Clay Engineered Barrier) - Molar Flow Rate History Data - C2 reference package - ^{129}I - [Reference Calculation : Short Tunnel without Clay engineered barrier]

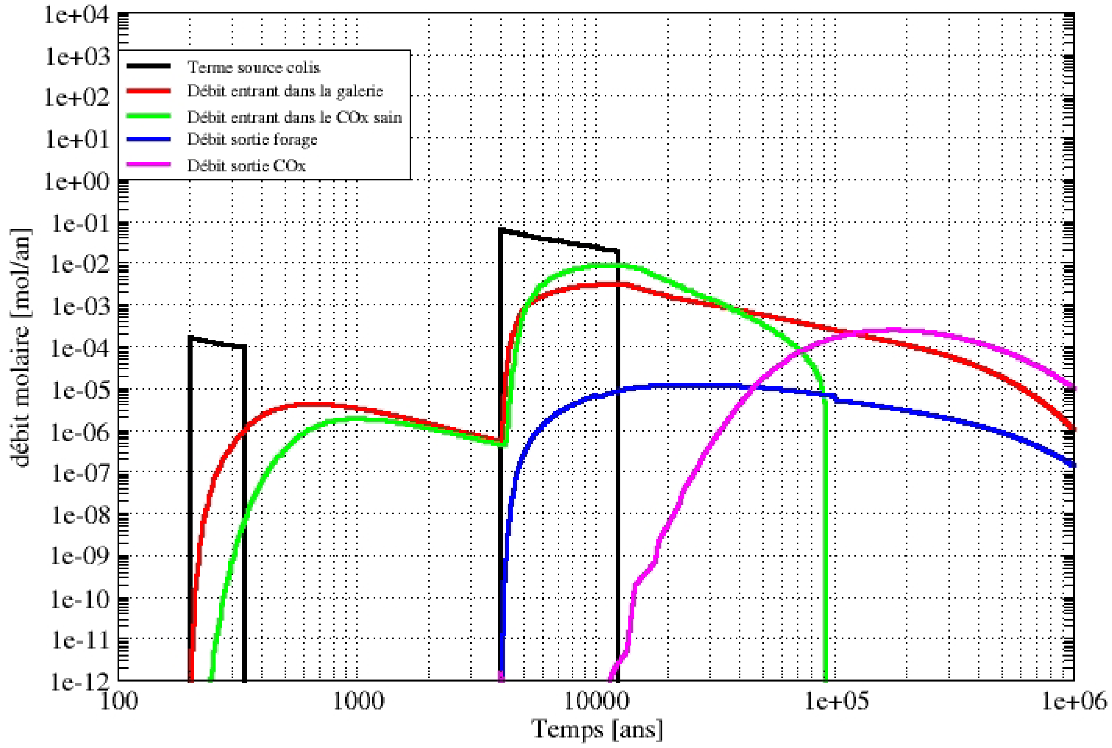


Figure 7.4-47 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the C Waste Design (Alterative Design with Clay Engineered Barrier) - Molar Flow Rate History Data - C2 reference package - ^{36}Cl - [Sensitivity Calculations : Short Tunnel with Clay engineered barrier]

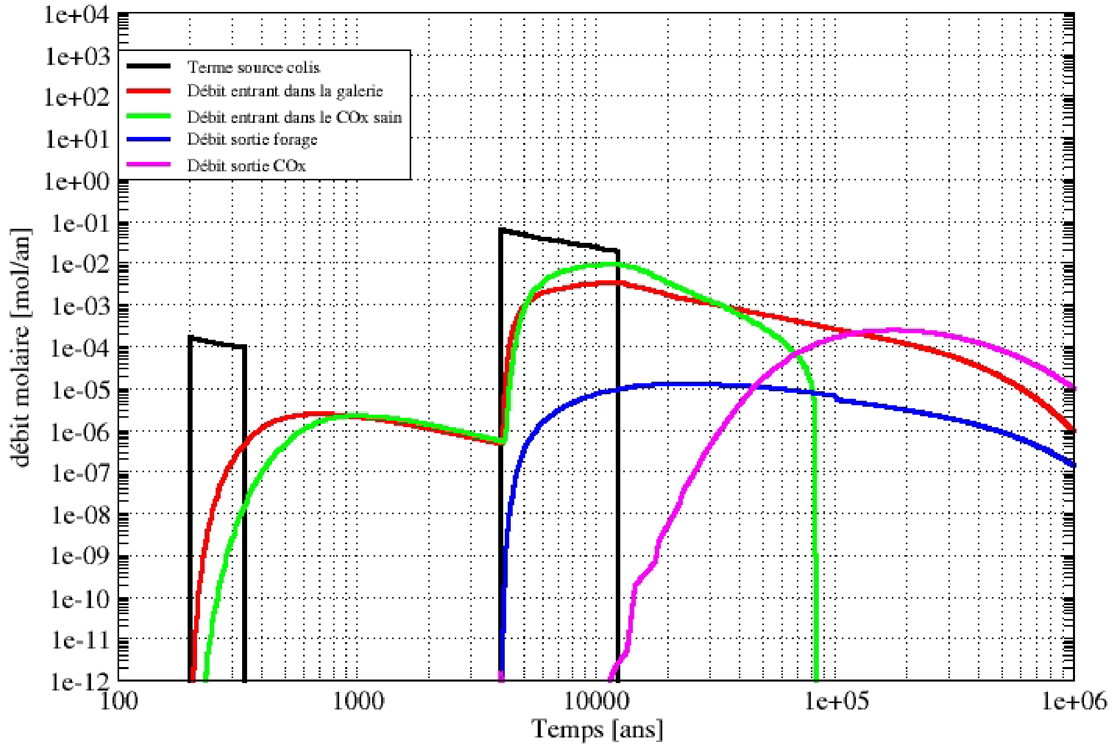


Figure 7.4-47 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the C Waste Design (Alterative Design with Clay Engineered Barrier) - Molar Flow Rate History Data - C2 reference package - ^{36}Cl - [Reference Calculation : Short Tunnel without Clay engineered barrier]

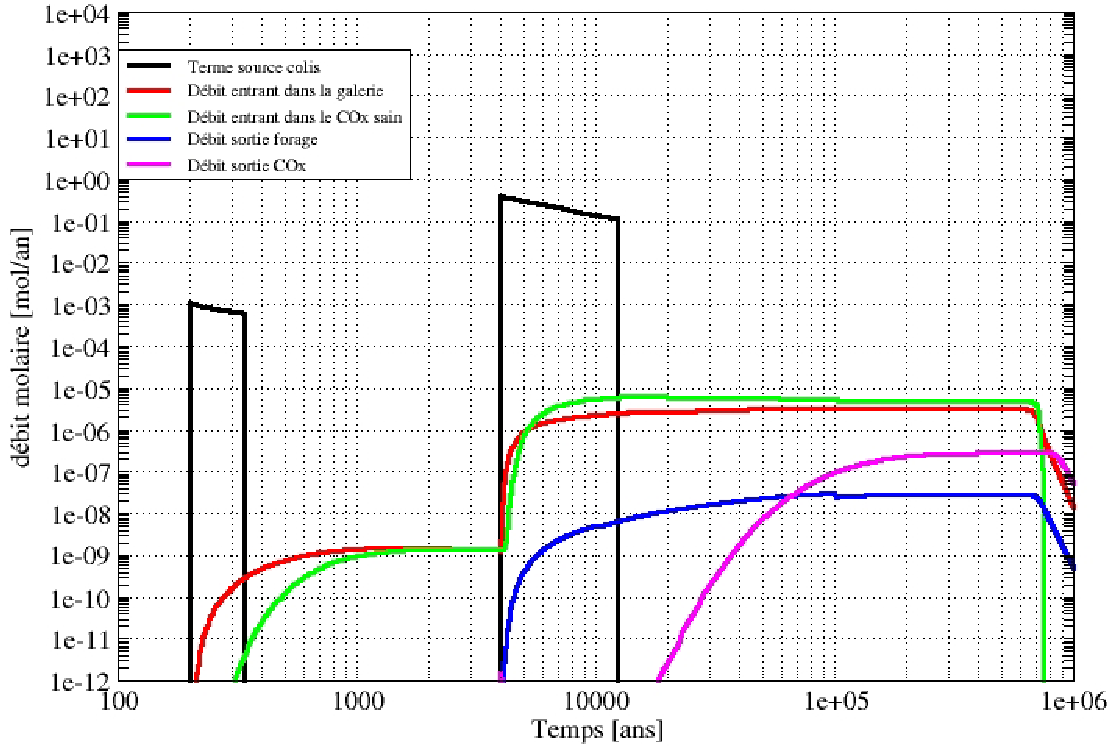


Figure 7.4-48 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the C Waste Design (Alternative Design with Clay Engineered Barrier) - Molar Flow Rate History - C2 reference package - ^{129}I - [Sensitivity Calculations : Short Tunnel with Clay engineered barrier]

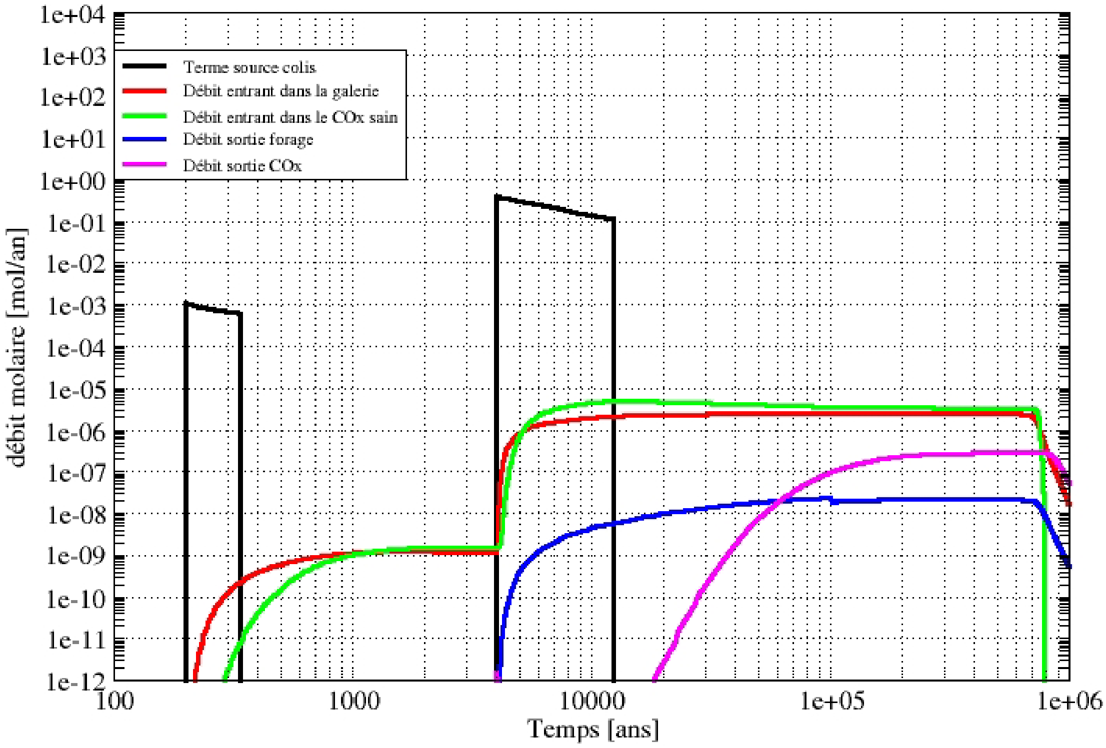


Figure 7.4-48 *SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the C Waste Design (Alternative Design with Clay Engineered Barrier) - Molar Flow Rate History - C2 reference package - ^{129}I - [Reference Calculation : Short Tunnel without Clay engineered barrier]*

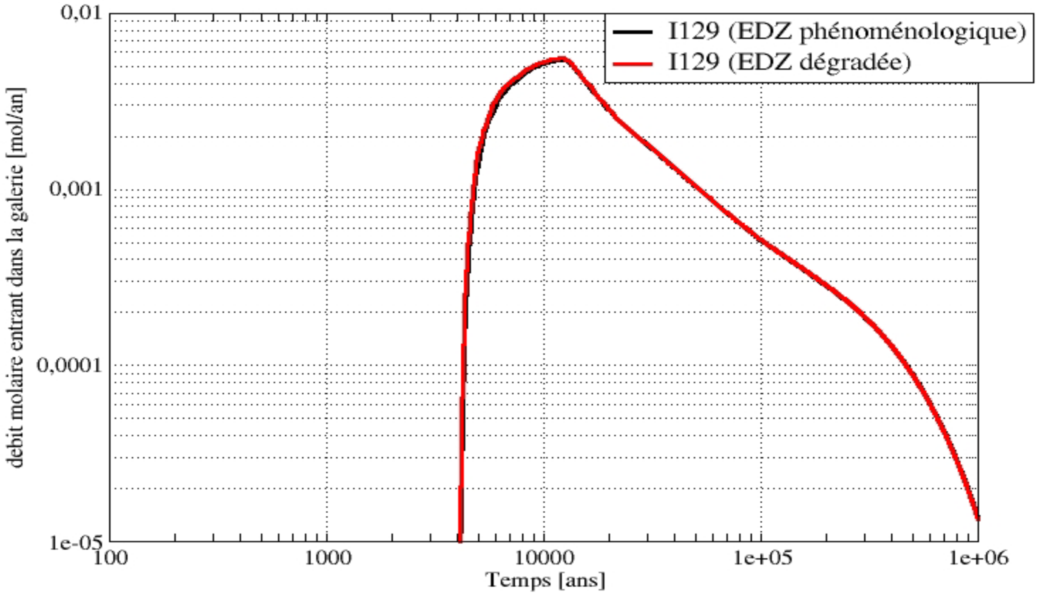


Figure 7.4-49 *SEA - Abandoned Borehole Penetration of the Repository - Molar Flow Rate History as a Function of Time Entering the Drift - ^{129}I - C2 - Comparison of the (Best-Estimate) EDZ Model with the (Degraded) EDZ*

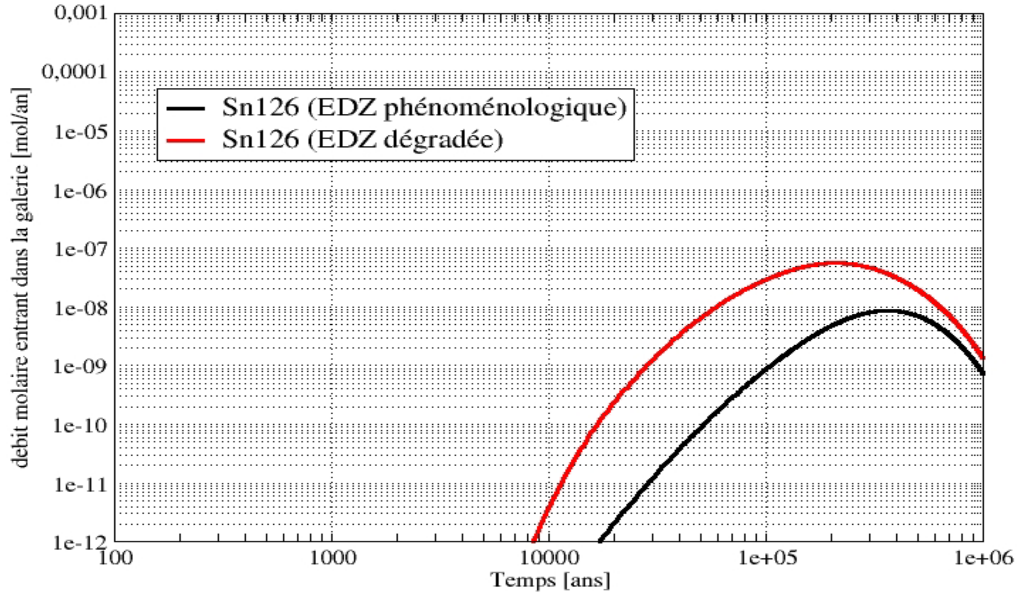


Figure 7.4-50 SEA - Abandoned Borehole Penetration of the Repository - Molar Flow Rate History as a Function of Time Entering the Drift - ^{126}Sn - C2 - Comparison of the (Best-Estimate) EDZ Model and the (Degraded) EDZ

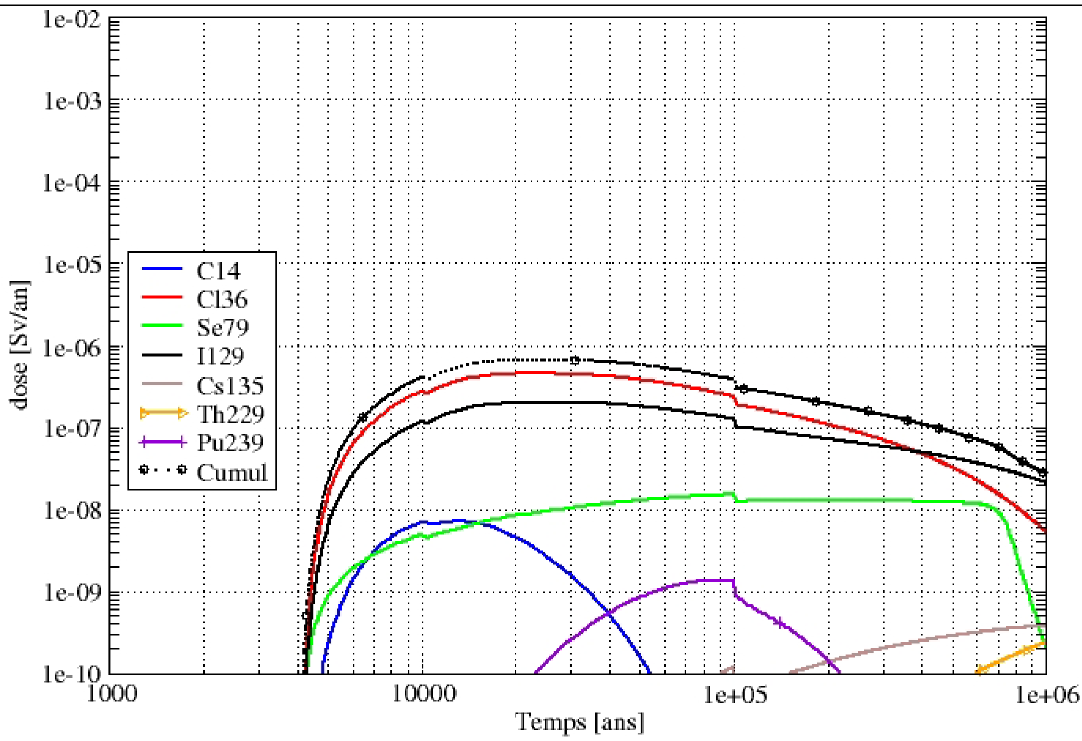


Figure 7.4-51 SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the « Degraded » EDZ - Dose Due to the Borehole at the DWS Pumping Outlet - C2 reference package

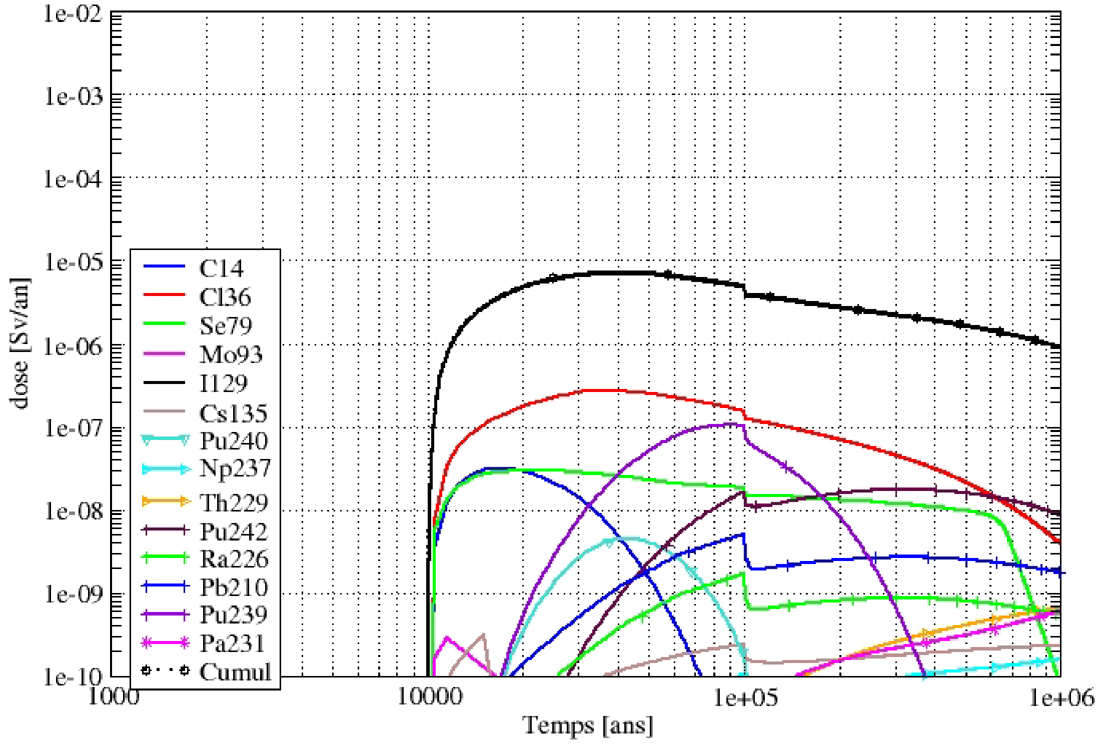


Figure 7.4-52 *SEA - Abandoned Borehole Penetration of the Repository - Sensitivity to the Conservative Geochemical Parameters of Bentonite - Dose Due to the Borehole at the DWS Pumping Outlet - CU1 reference package*

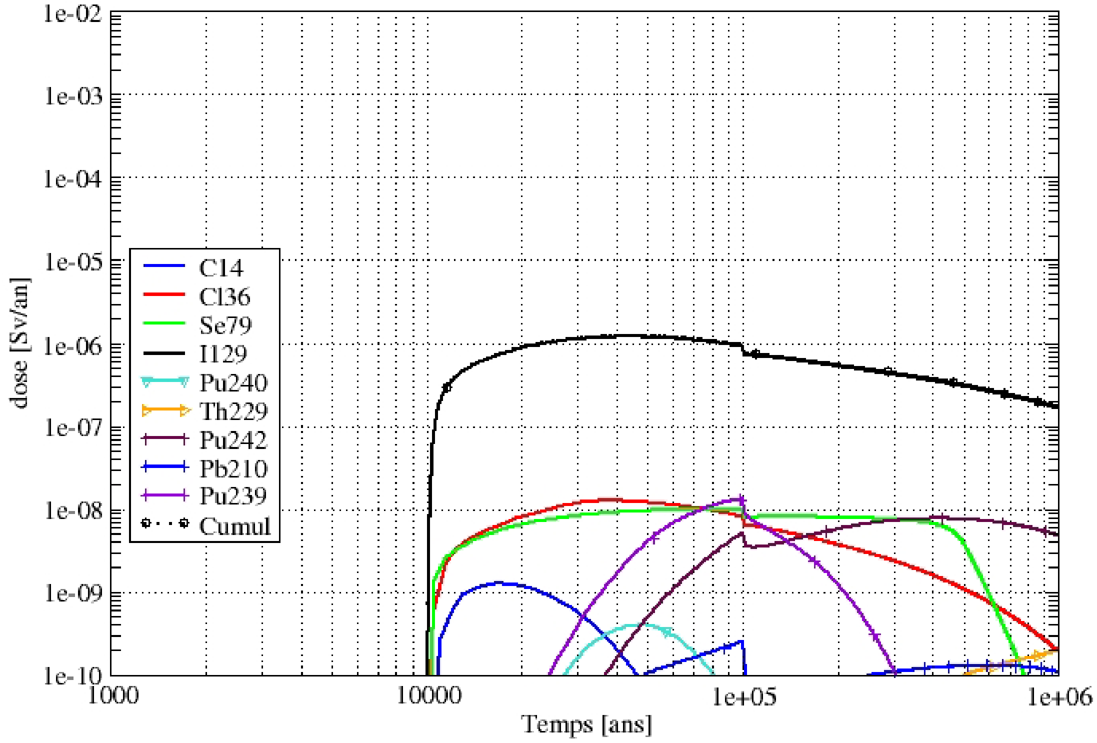


Figure 7.4-53 *SEA borehole abandoned in the repository – Sensitivity to the conservative geochemical parameters of the bentonite - dose due to the DWS pumping outlet at the borehole – CU2 reference package*

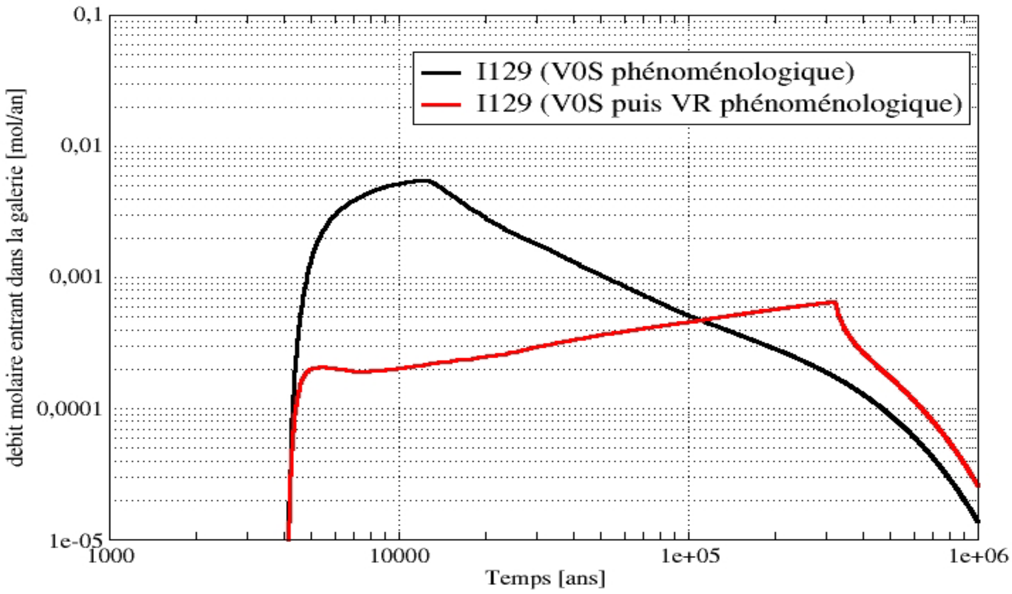


Figure 7.4-54 *Borehole abandoned in the repository SEA – Molar flow rate history entering the drift plotted against time – ^{129}I – C2 – Comparison of the $V_0.S$ and $V_0.S \rightarrow V_R$ models*

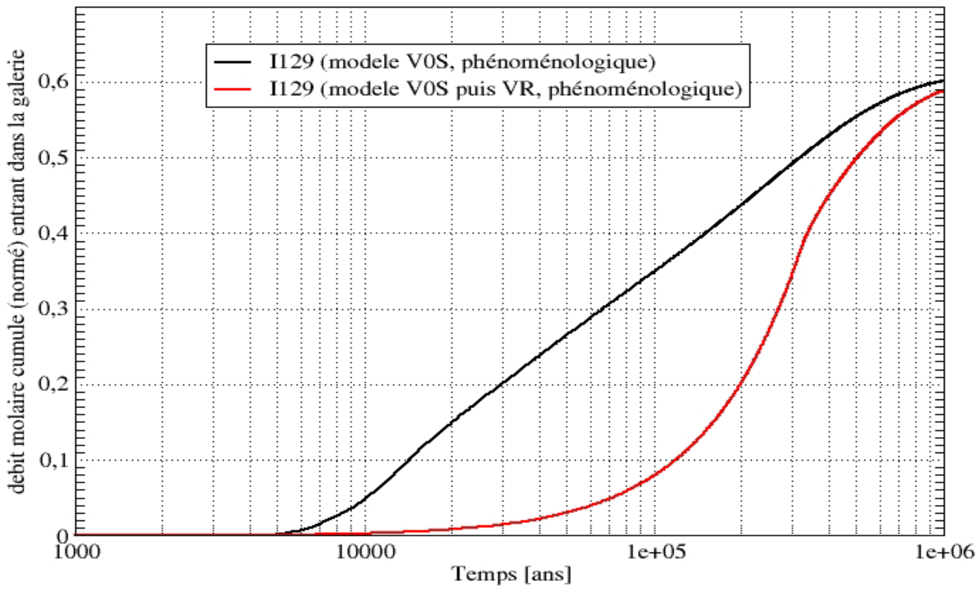


Figure 7.4-55 *Borehole abandoned in the repository SEA – Accumulated and normalised molar flow rate entering the drift plotted against time – ^{129}I – C2 – Comparison of the $V_0.S$ and $V_0.S \rightarrow V_r$ models*

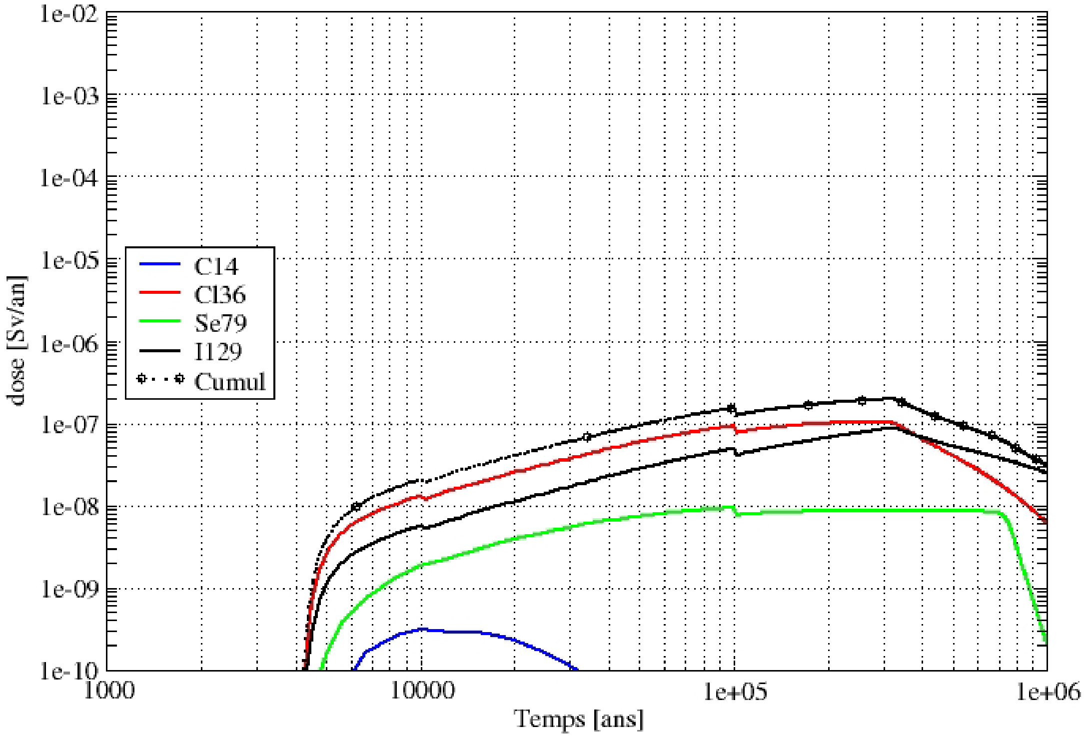


Figure 7.4-56 *Borehole abandoned in the repository SEA – Sensitivity to the glass release model - dose due to drilling a borehole at the DWS pumping outlet – Reference package C2*

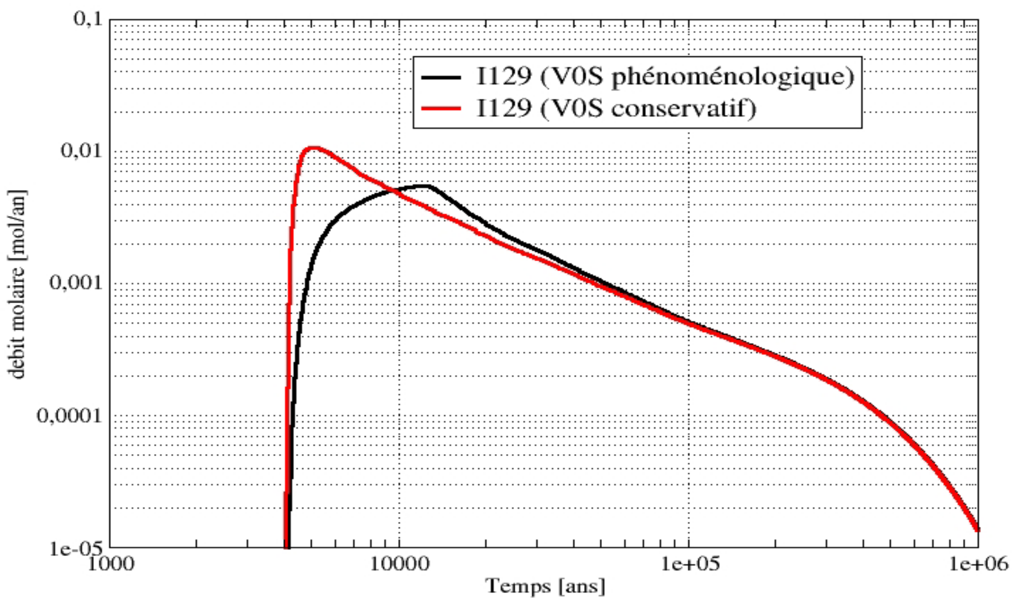


Figure 7.4-57 *Borehole abandoned in the repository SEA – Molar flow rate history entering the drift plotted against time – ^{129}I – C2 – Sensitivity to the conservative parameters of the $V_0.S$ glass release model*

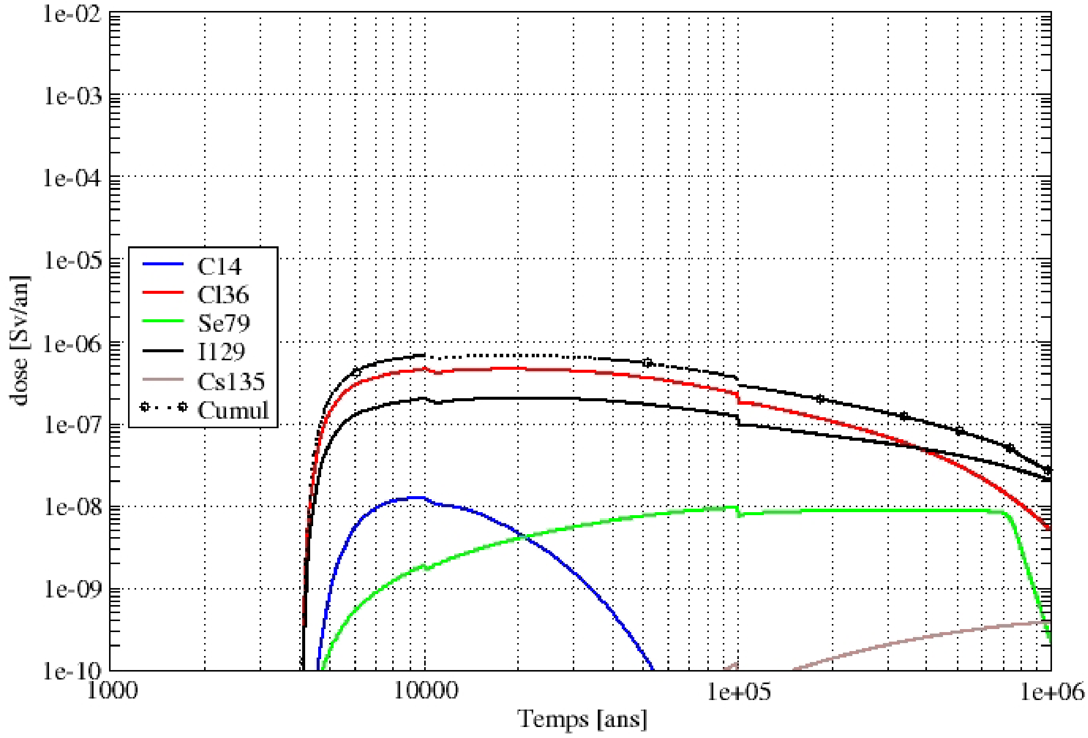


Figure 7.4-58 *Borehole abandoned in the repository SEA – Sensitivity to the conservative parameters of the V0.S glass release model - dose due to drilling a borehole at the DWS pumping outlet – Reference package C2*

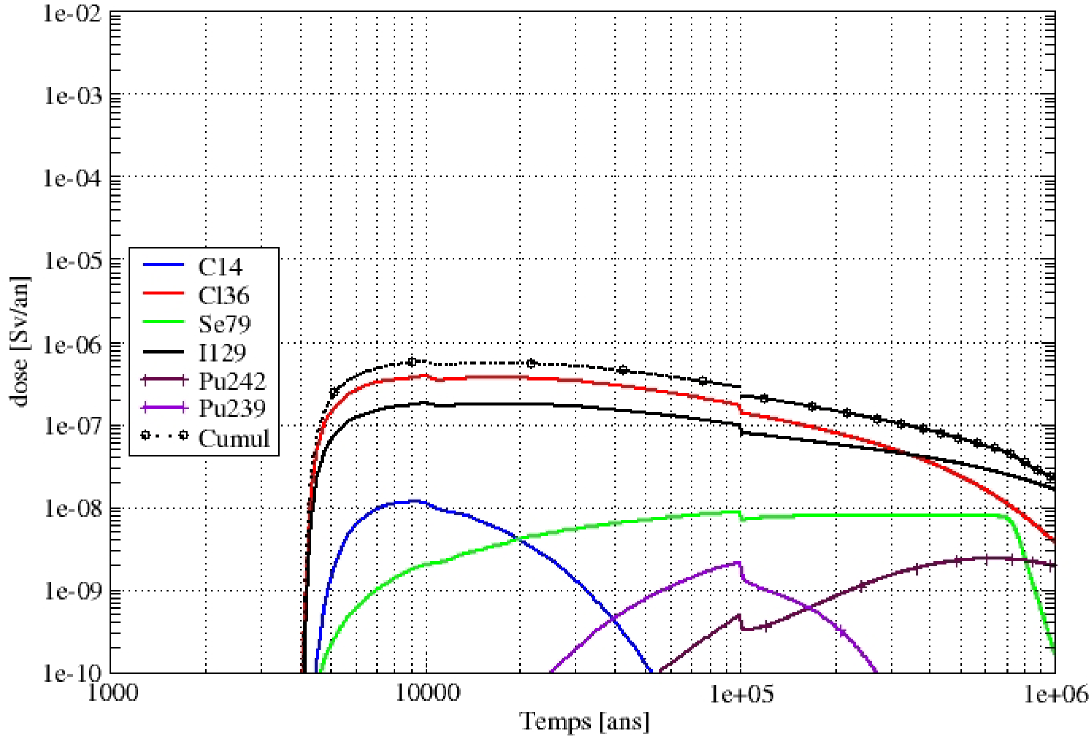


Figure 7.4-59 *Borehole abandoned in the repository SEA – Sensitivity to the conservative parameters of the V0.S glass release model - dose due to drilling a borehole at the DWS pumping outlet – Reference package C4*

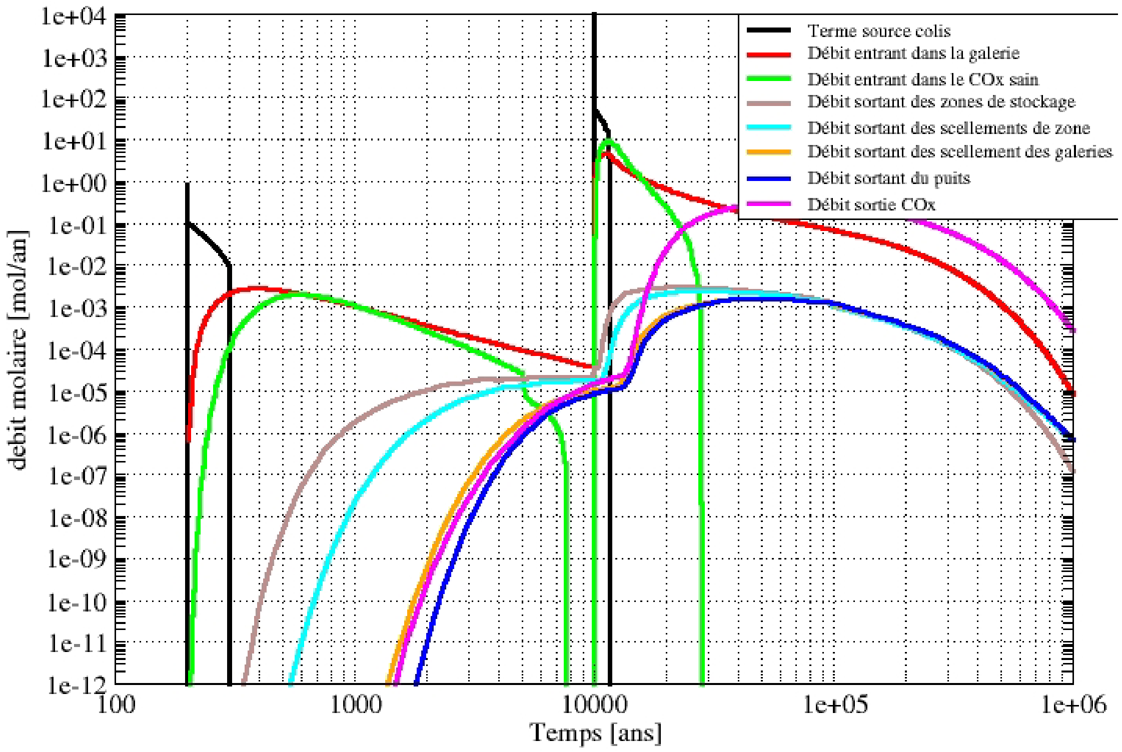


Figure 7.5-5 *« Severely degraded evolution » scenario – Reference package CU1 – ^{129}I – Molar flow rate history - [« Highly degraded operation » SEA]*

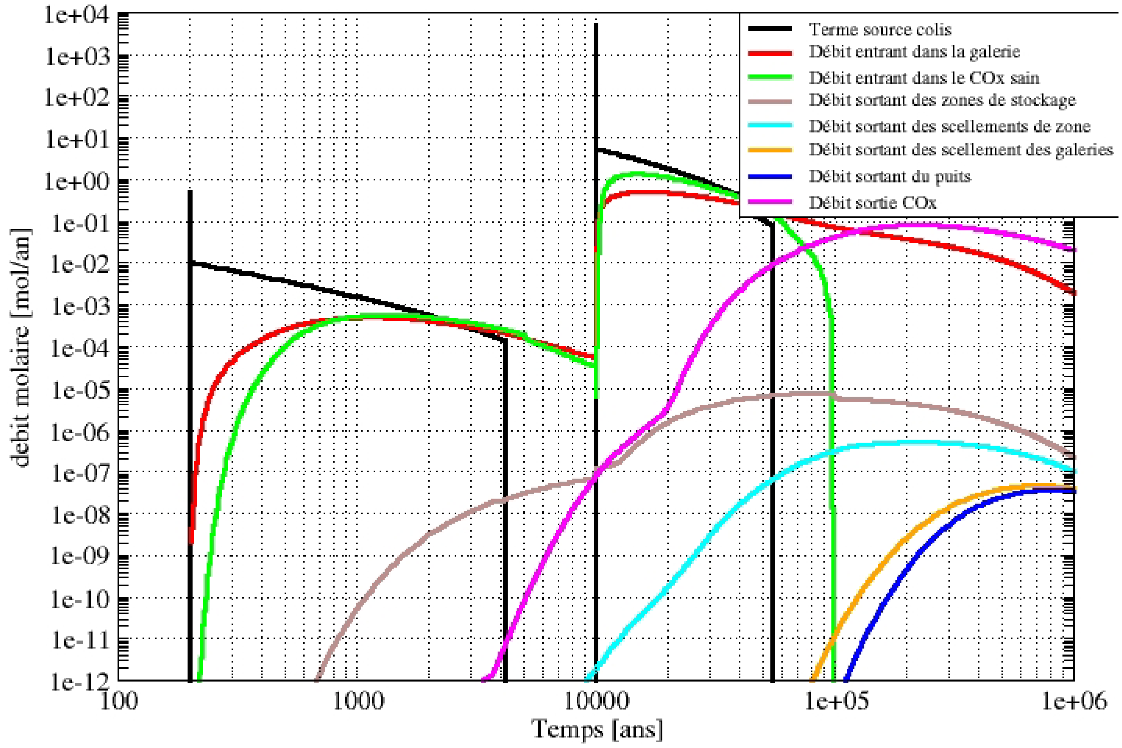


Figure 7.5-5 *« Severely degraded evolution » scenario – Reference package CU1 – ^{129}I – Molar flow rate history - [SEN]*

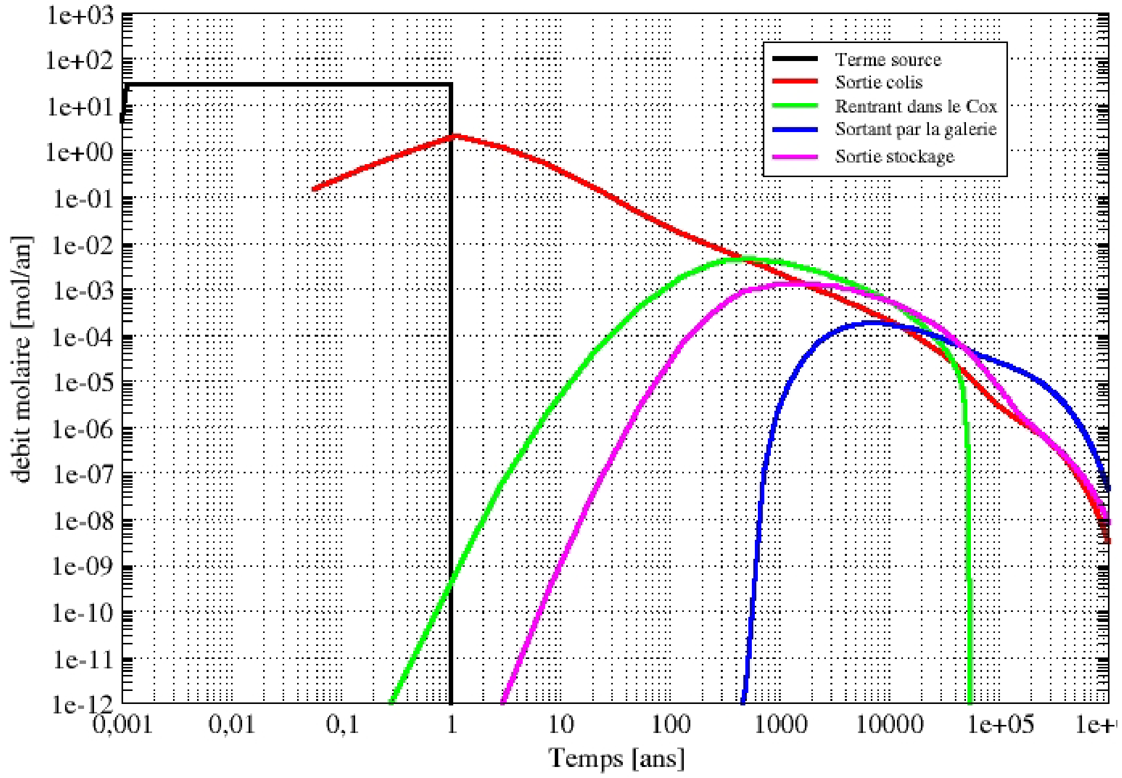


Figure 7.5-6 *« Severely degraded evolution » scenario – Reference package B1x – ^{129}I – Molar flow rate history - [Highly degraded operation » SEA]*

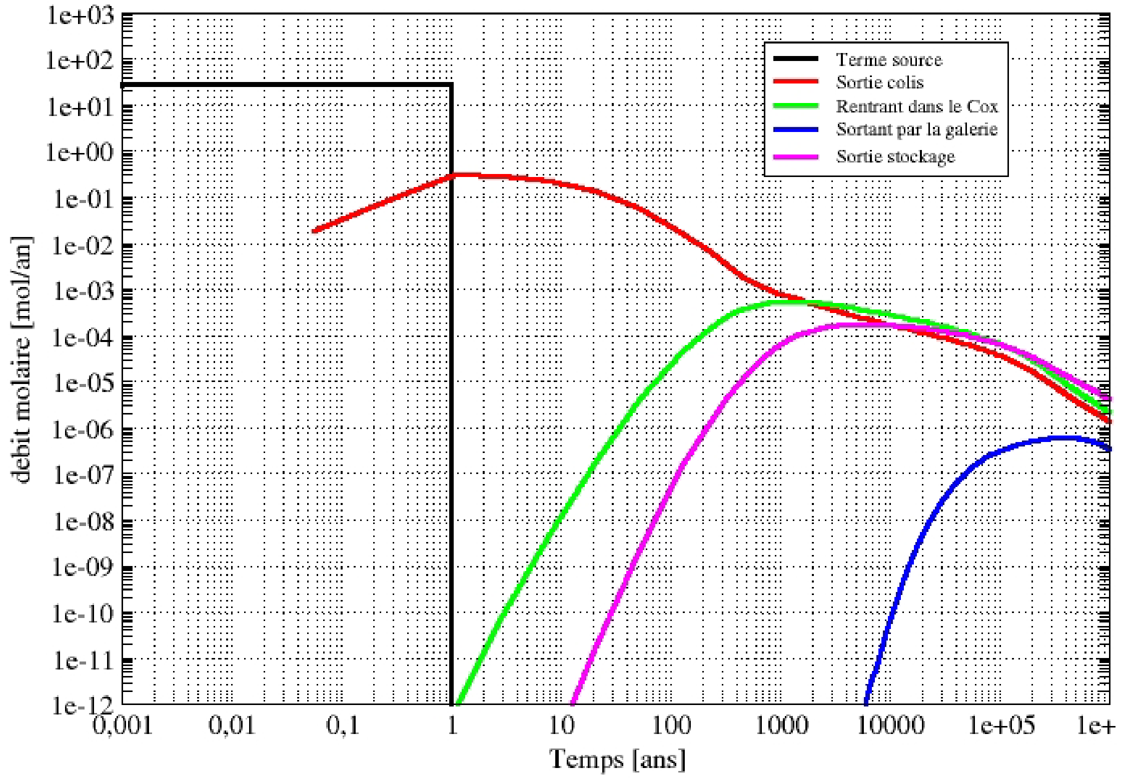


Figure 7.5-6 *« Severely degraded evolution » scenario – Reference package B1x – ^{129}I – Molar flow rate history - [SEN]*

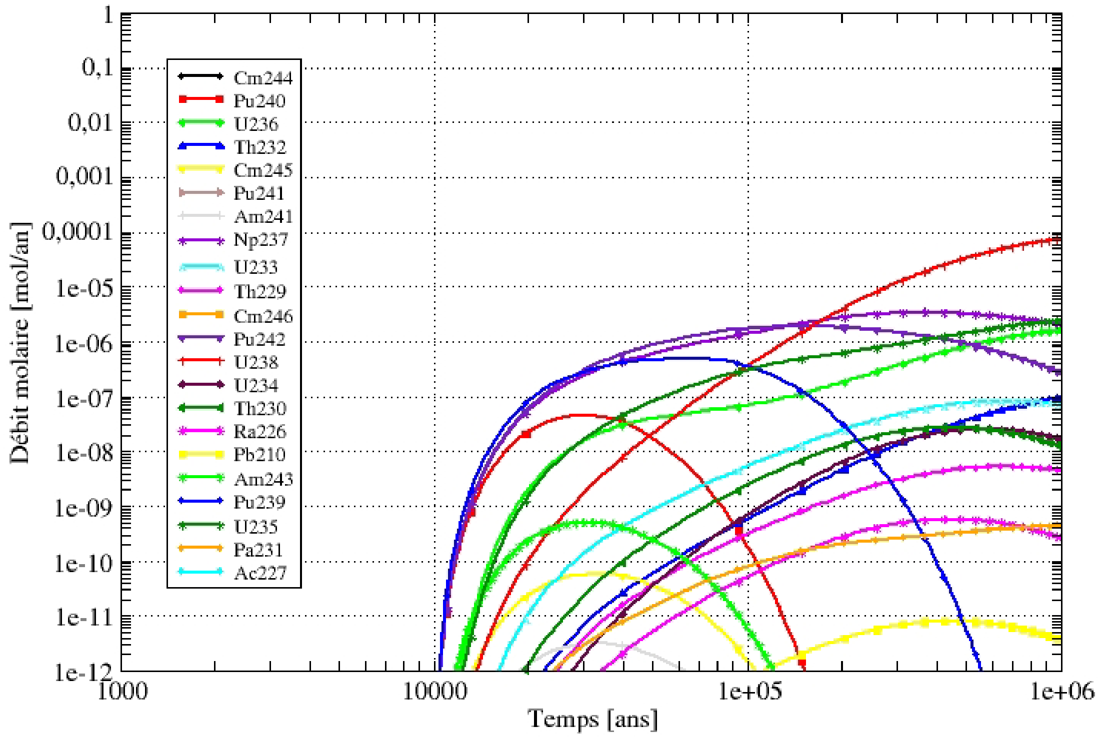


Figure 7.5-8 : *Severely degraded evolution scenario – history of molar flow rates into the drift (one CU2 cell) and out of a CU2 repository zone seal (all cells).*

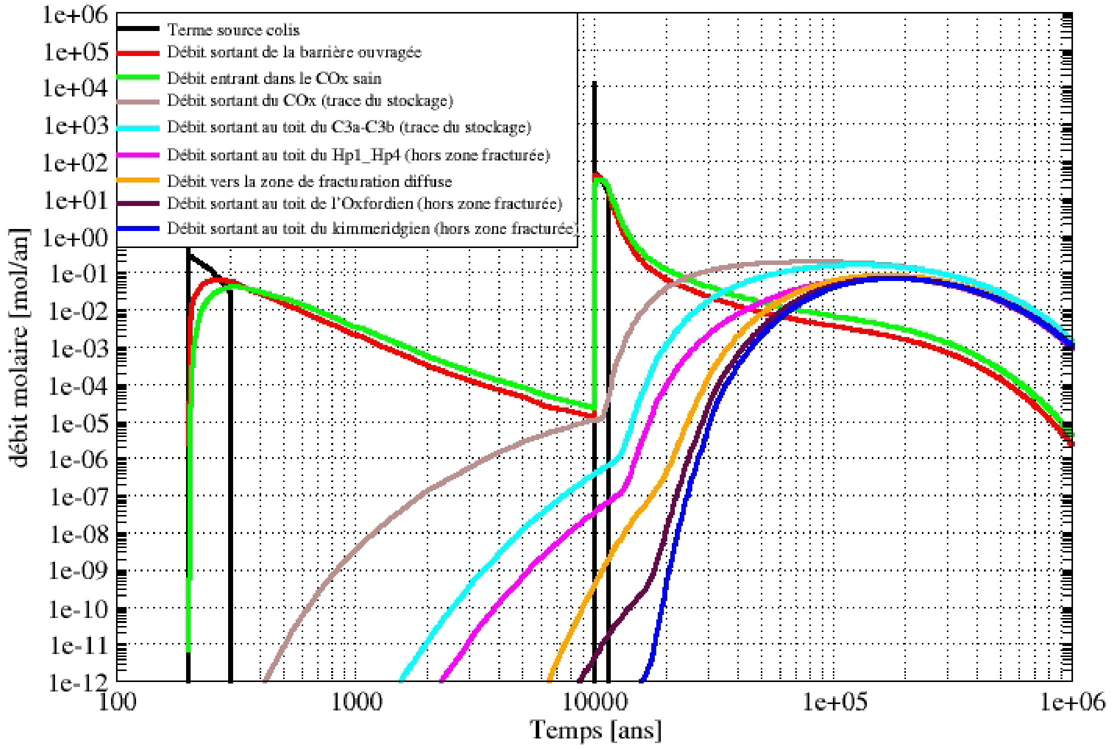


Figure 7.5-9 *« Severely degraded evolution » scenario - CU1 reference package– ^{129}I – History of molar rates –« geological barrier » transfer pathway - [Highly degraded operation » SEA]*

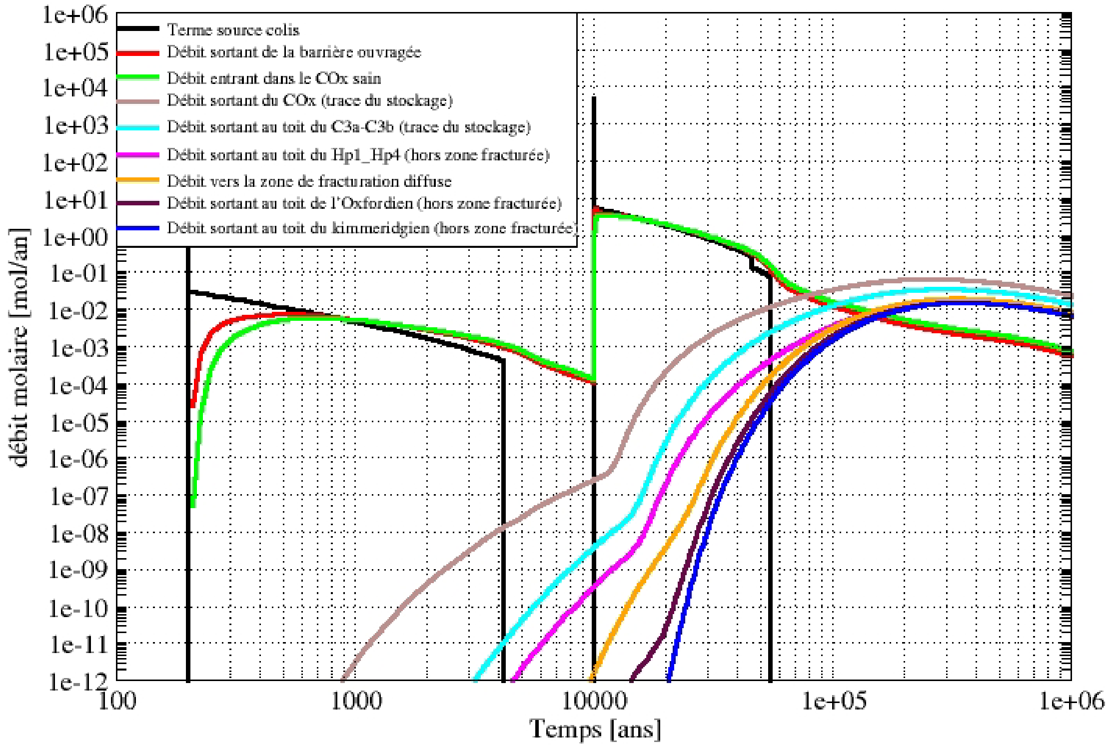


Figure 7.5-9 *« Severely degraded evolution » scenario - CU1 reference package– ^{129}I – History of molar rates –« geological barrier » transfer pathway - [SEN]*

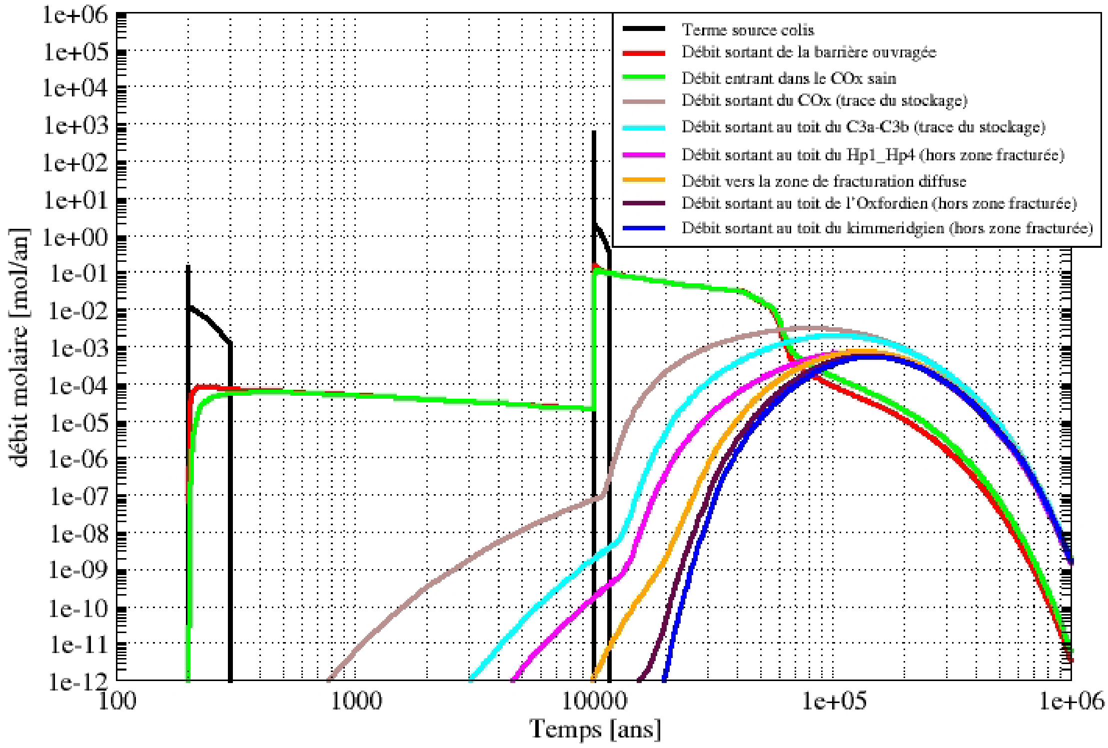


Figure 7.5-10 *« Severely degraded evolution » scenario - CU1 reference package– ^{79}Se – History of molar rates –« geological barrier » transfer pathway - [Highly degraded operation » SEA]*

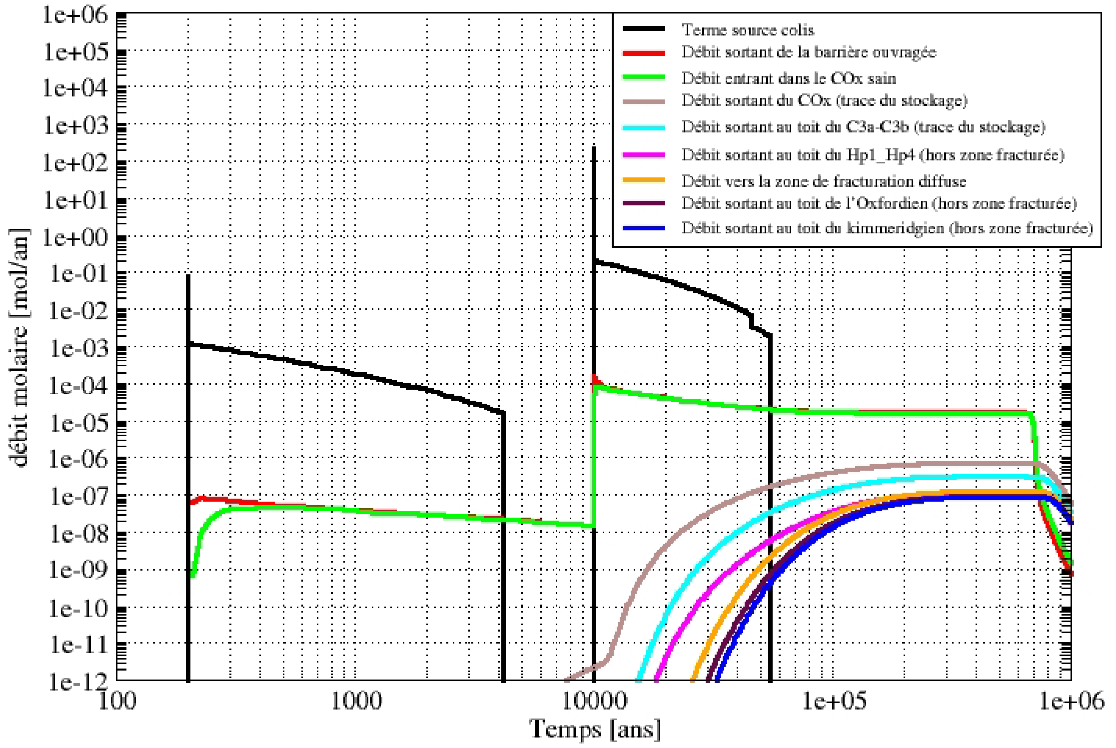


Figure 7.5-10 *« Severely degraded evolution » scenario - CU1 reference package– ^{79}Se – History of molar rates –« geological barrier » transfer pathway - [SEN]*

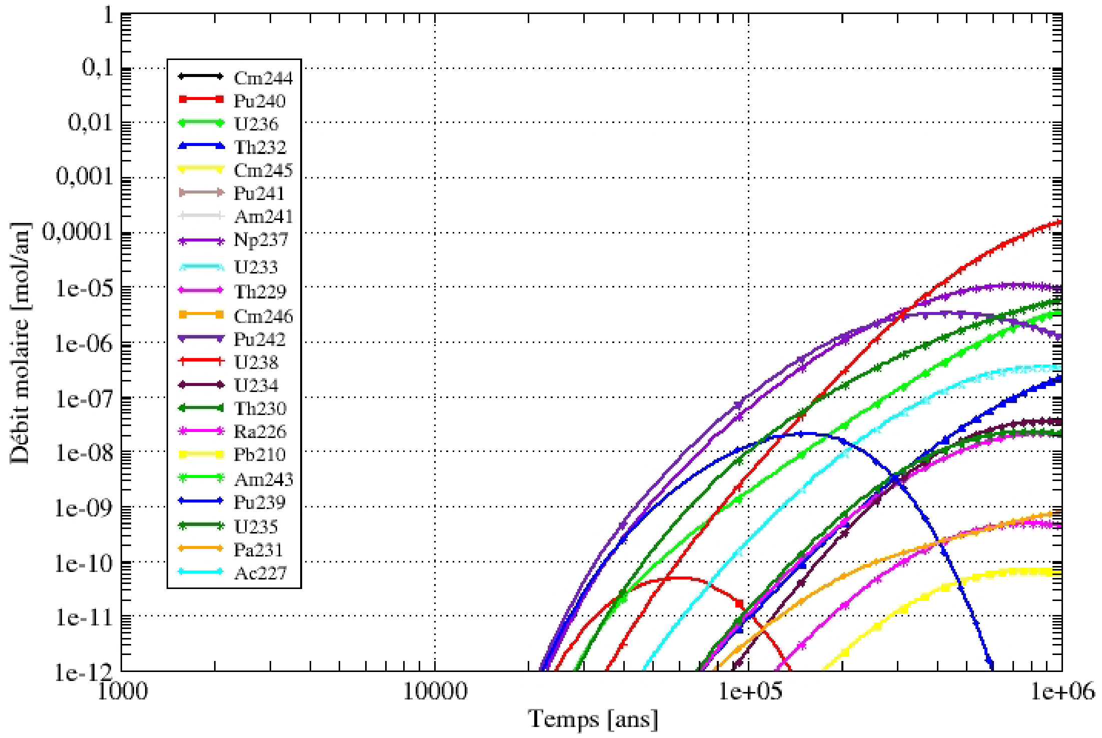


Figure 7.5-11 *Severely degraded evolution scenario – history of molar flow rates out of the repository at a distance of 7 metres from the cells and out of the Callovo-Oxfordian for a CU2 cell*

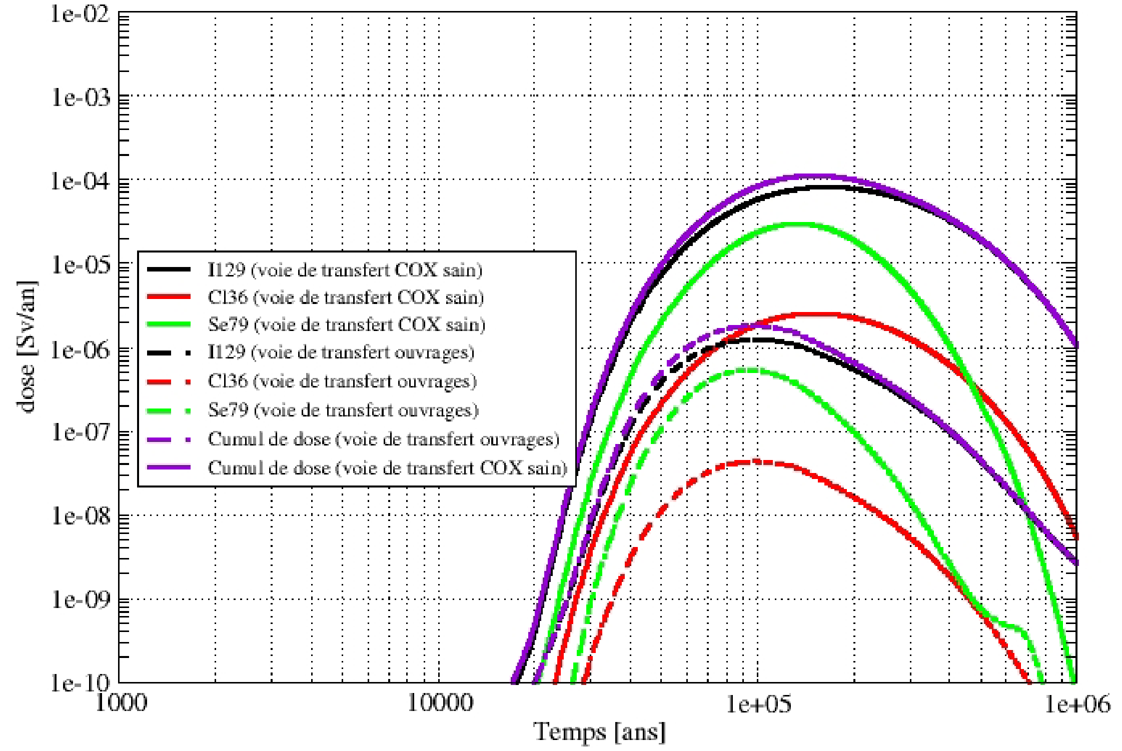


Figure 7.5-12 *« Severely degraded evolution » SEA - Comparison of the doses at the Saulx reference outlet of the CU1 reference package for the transfer pathways through the structures (present-day model) and unaltered geological barrier (model at 1 million years)*

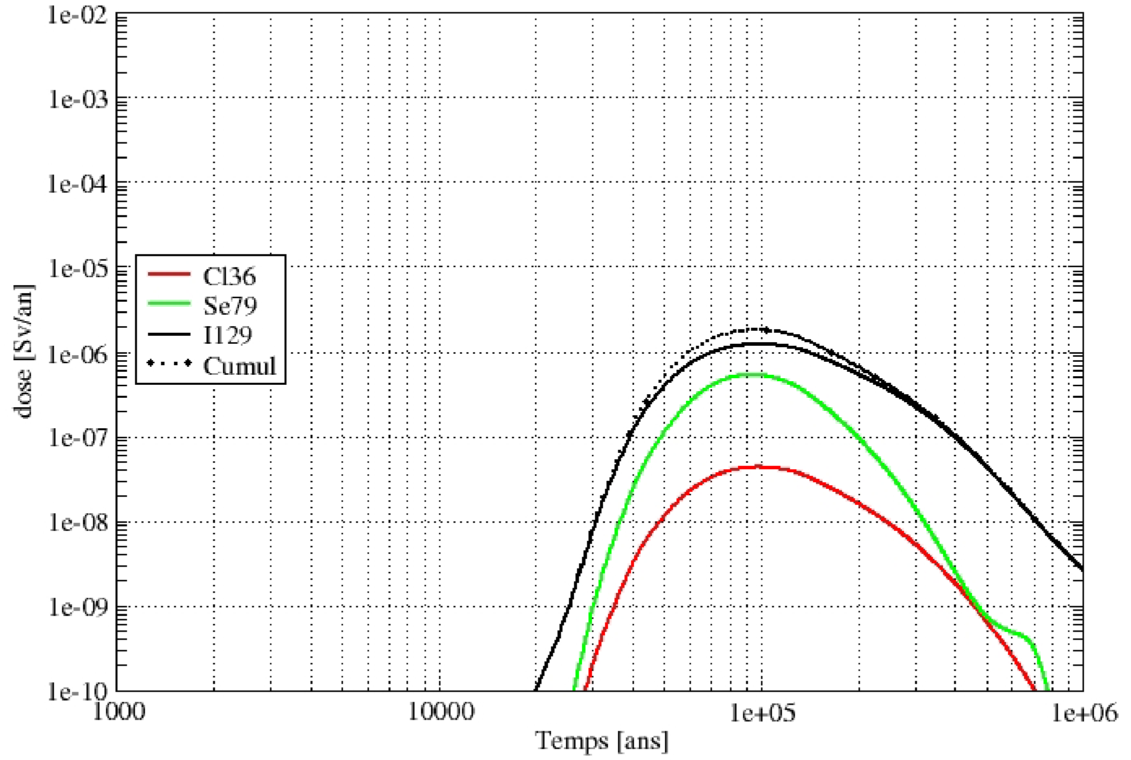


Figure 7.5-13 *« Severely degrade evolution » scenario - Dose at the Saulx outlet of the Oxfordian - CU1 spent fuel reference packages - [Transfer path through the structures]*

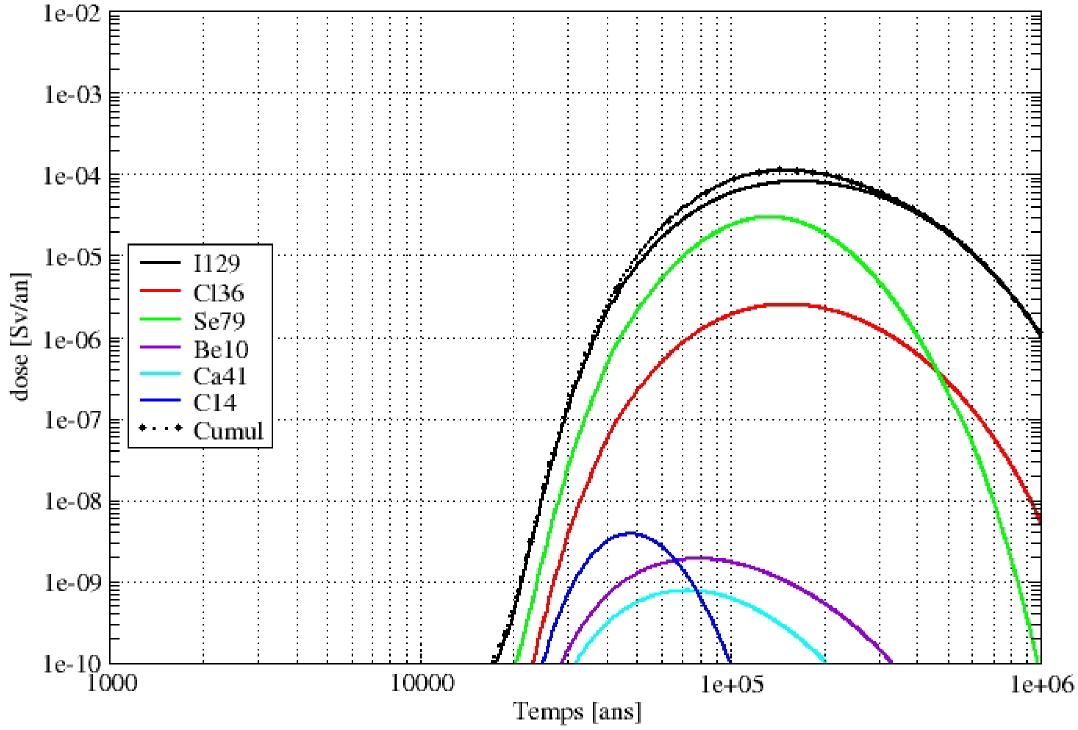


Figure 7.5-13 « Severely degrade evolution » scenario - Dose at the Saulx outlet of the Oxfordian - CU1 spent fuel reference packages - [Transfer path through the host formation]

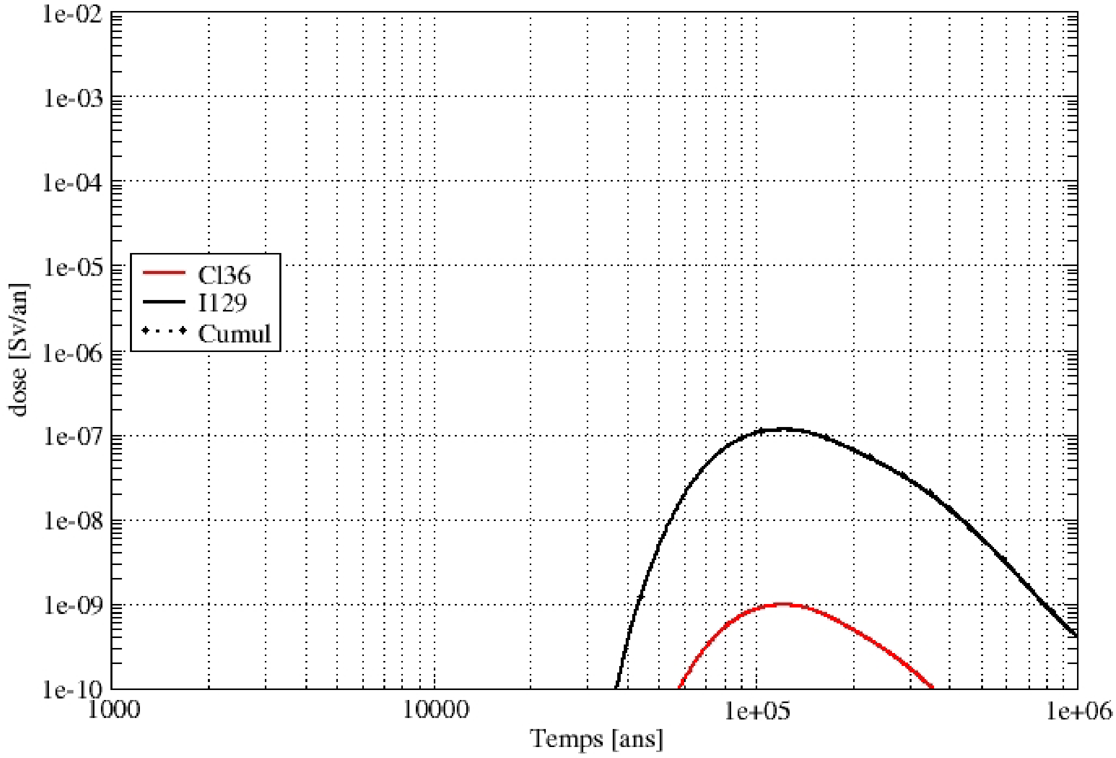


Figure 7.5-14 « Severely degraded evolution » scenario - Dose at the Saulx outlet of the Oxfordian - CU2 spent fuel referencee packages - [Transfer path through the structures]

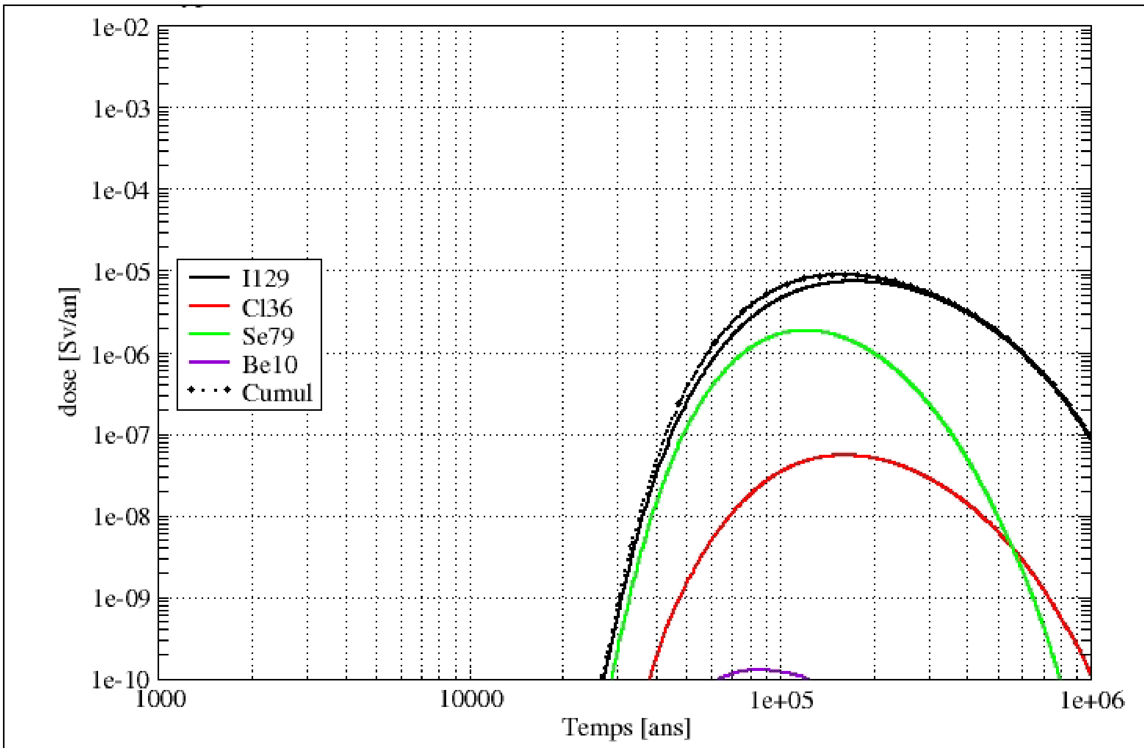


Figure 7.5-14 *« Severely degraded evolution » scenario - Dose at the Saulx outlet of the Oxfordian - CU2 spent fuel referencee packages - [Transfer path through the host formation]*

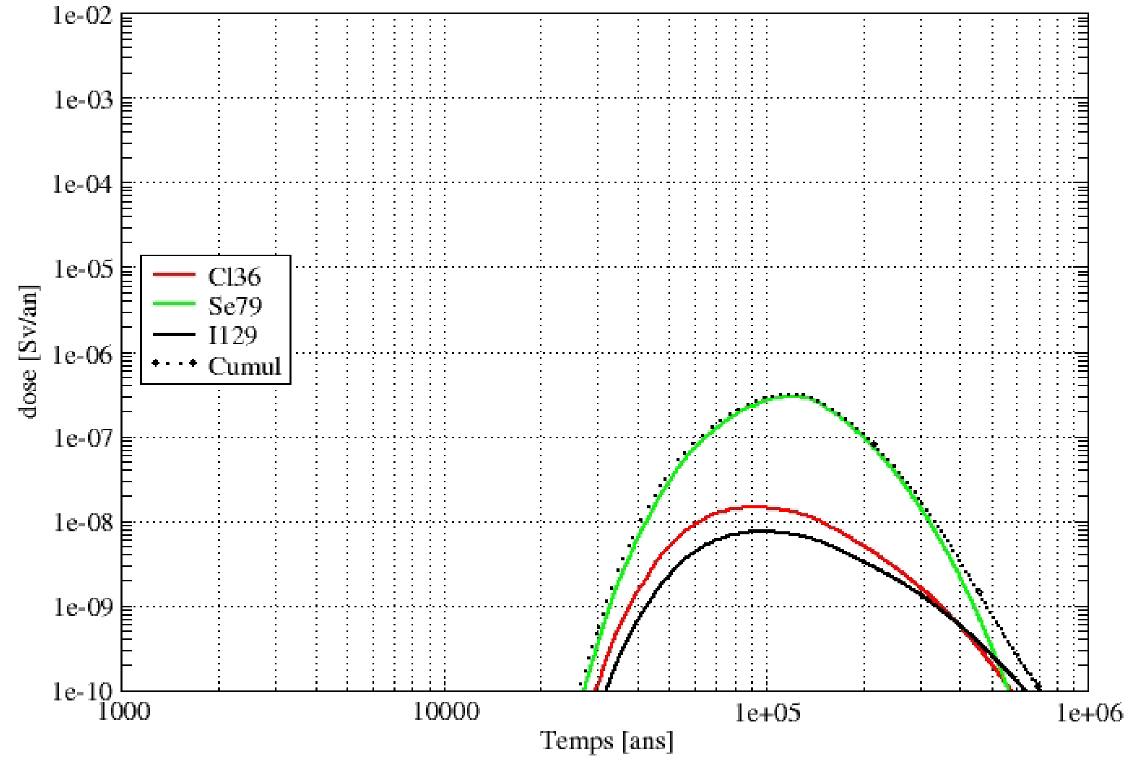


Figure 7.5-15 *« Severely degraded evolution » scenario - Dose at the Saulx outlet of the Oxfordian - C1/C2 reference packages - [Transfer path through the structures]*

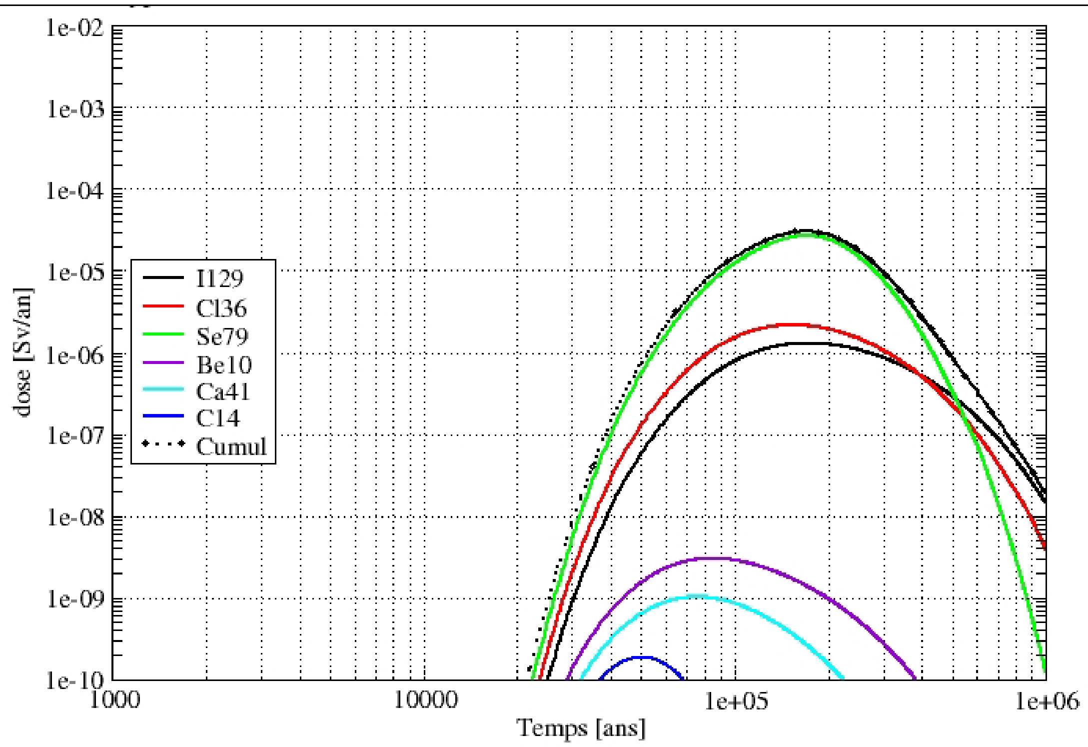


Figure 7.5-15 *« Severely degraded evolution » scenario - Dose at the Saulx outlet of the Oxfordian - C1/C2 reference packages - [Transfer path through the host formation]*

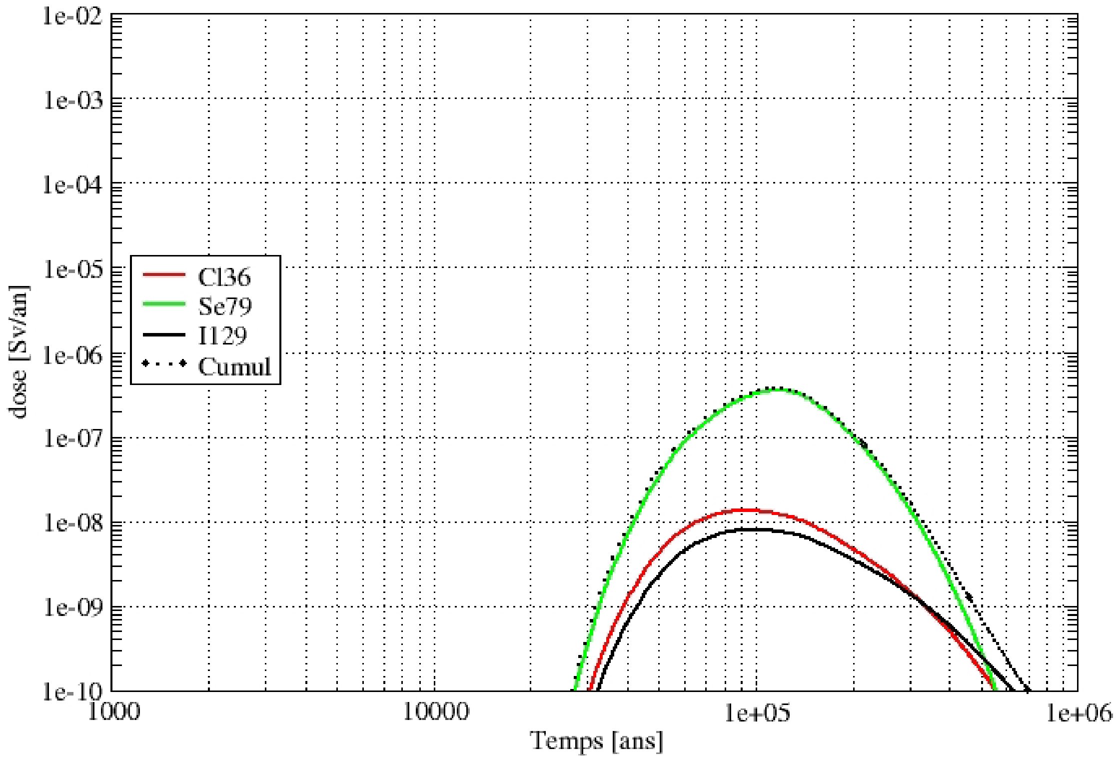


Figure 7.5-16 *« Severely degraded evolution » scenario - Dose at the Saulx outlet of the Oxfordian - C3/C4 reference packages - [Transfer path through the structures]*

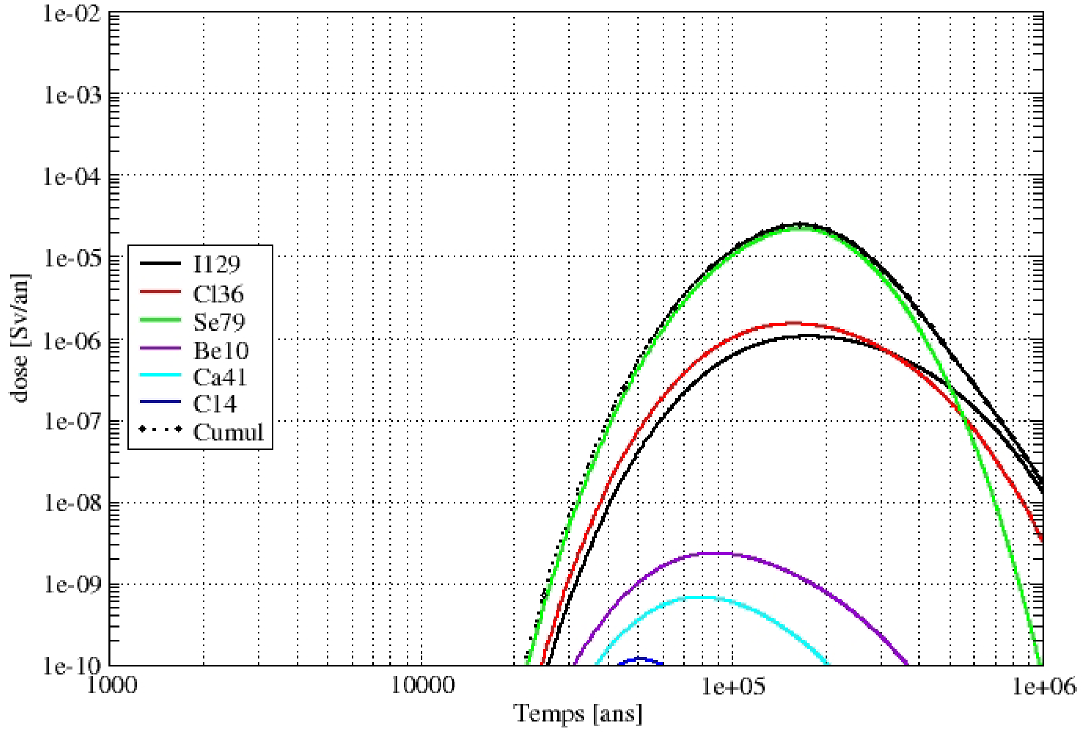


Figure 7.5-16 « Severely degraded evolution » scenario - Dose at the Saulx outlet of the Oxfordian - C3/C4 reference packages - [Transfer path through the host formation]

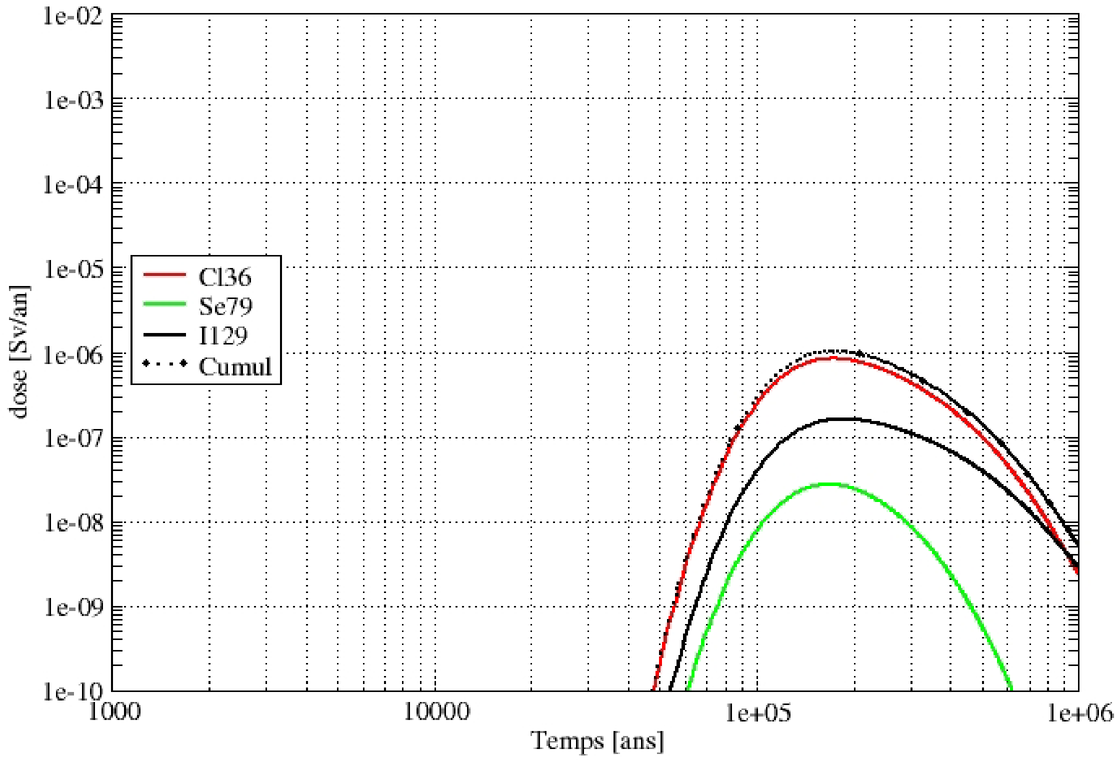


Figure 7.5-17 « Severely degraded evolution » scenario - Dose at the Saulx outlet of the Oxfordian - B1x reference package - [Transfer path through the structures]

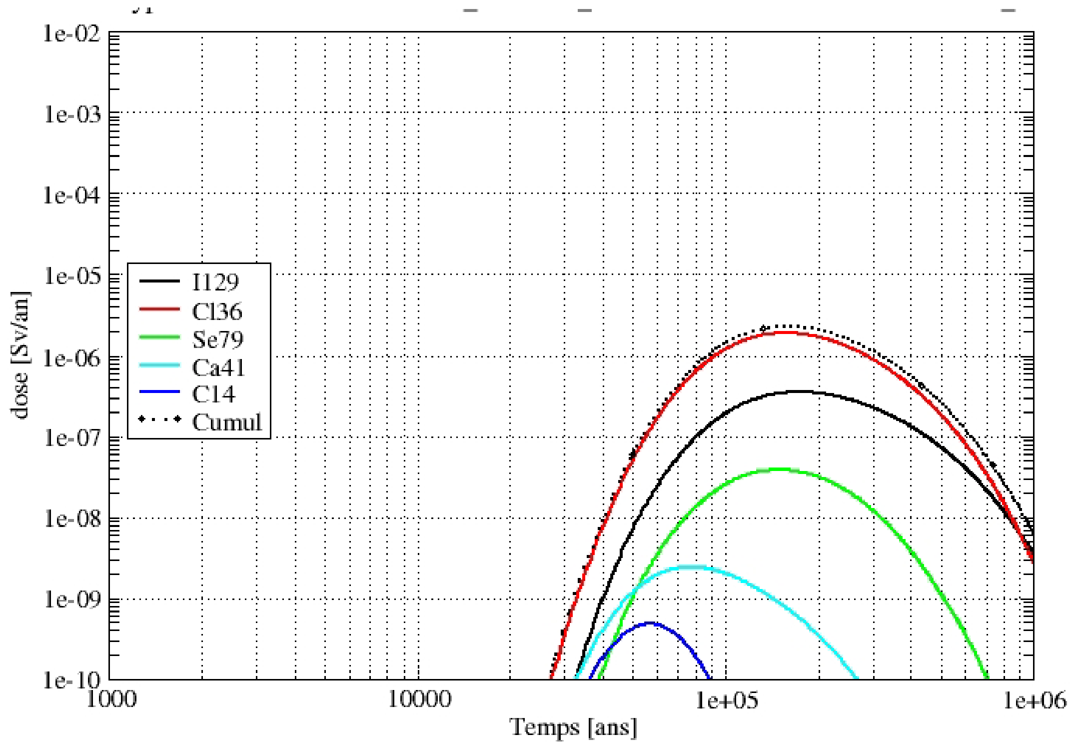


Figure 7.5-18 « Severely degraded evolution »scenario - Dose at the Saulx outlet of the Oxfordian - B2 reference package

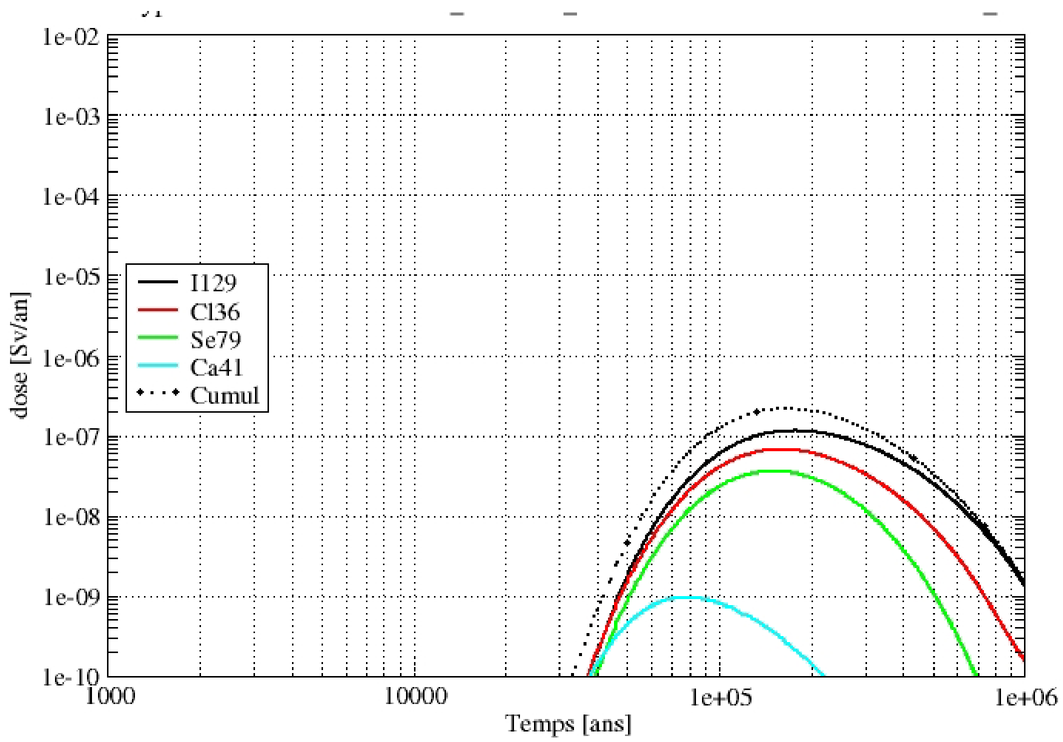


Figure 7.5-18 « Severely degraded evolution »scenario - Dose at the Saulx outlet of the Oxfordian - B2 reference package

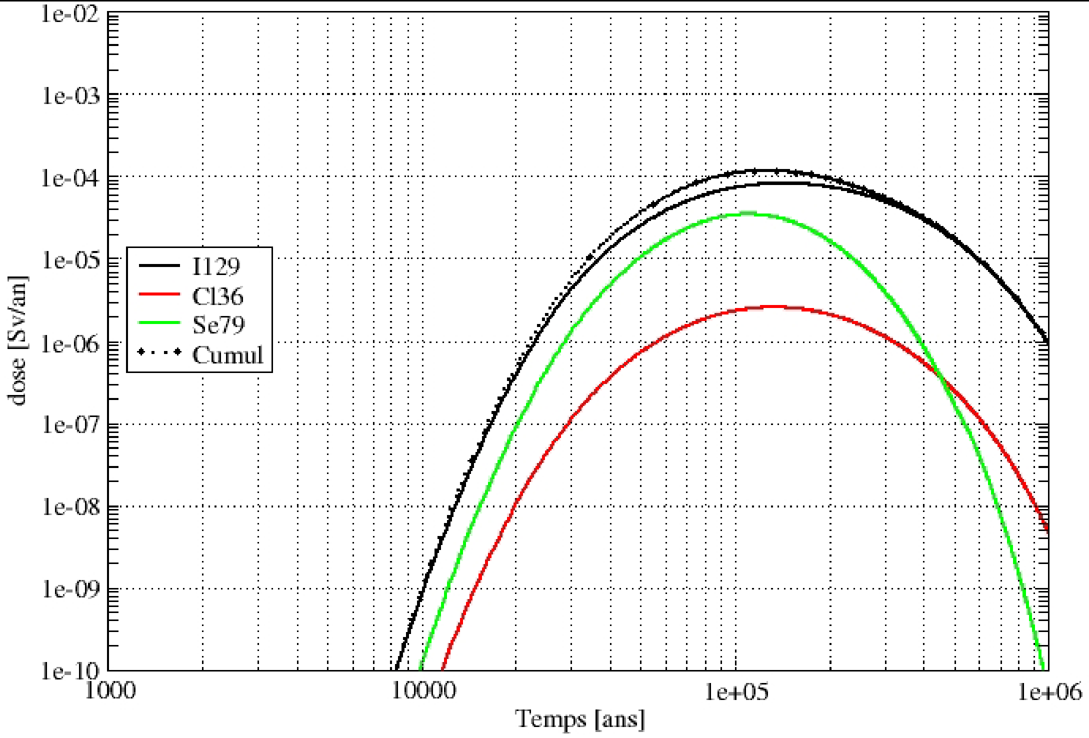


Figure 7.5-19 « Severely degraded evolution » scenario – sensitivity to the premature failure of the CU1 spent fuel containers- Dose at the Saulx outlet of the Oxfordian - [Sensitivity to failure of the containers – transfer path through the geological barrier]

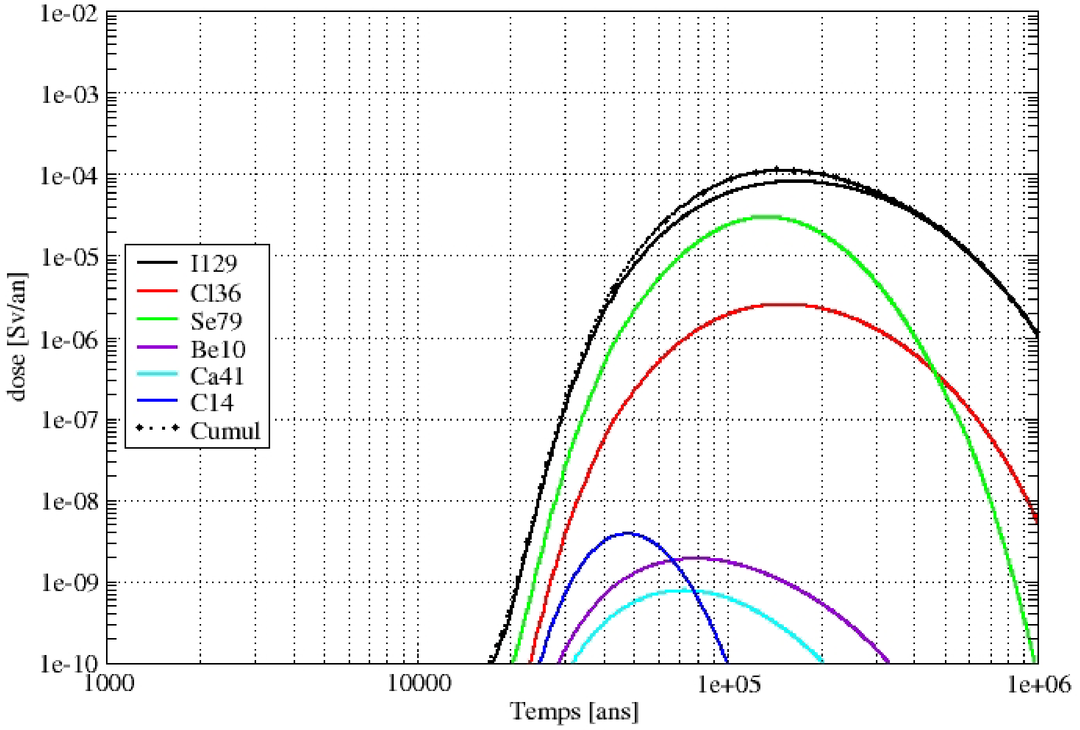


Figure 7.5-19 « Severely degraded evolution » scenario – sensitivity to the premature failure of the CU1 spent fuel containers- Dose at the Saulx outlet of the Oxfordian - [Reference calculation - transfer path through the geological barrier]

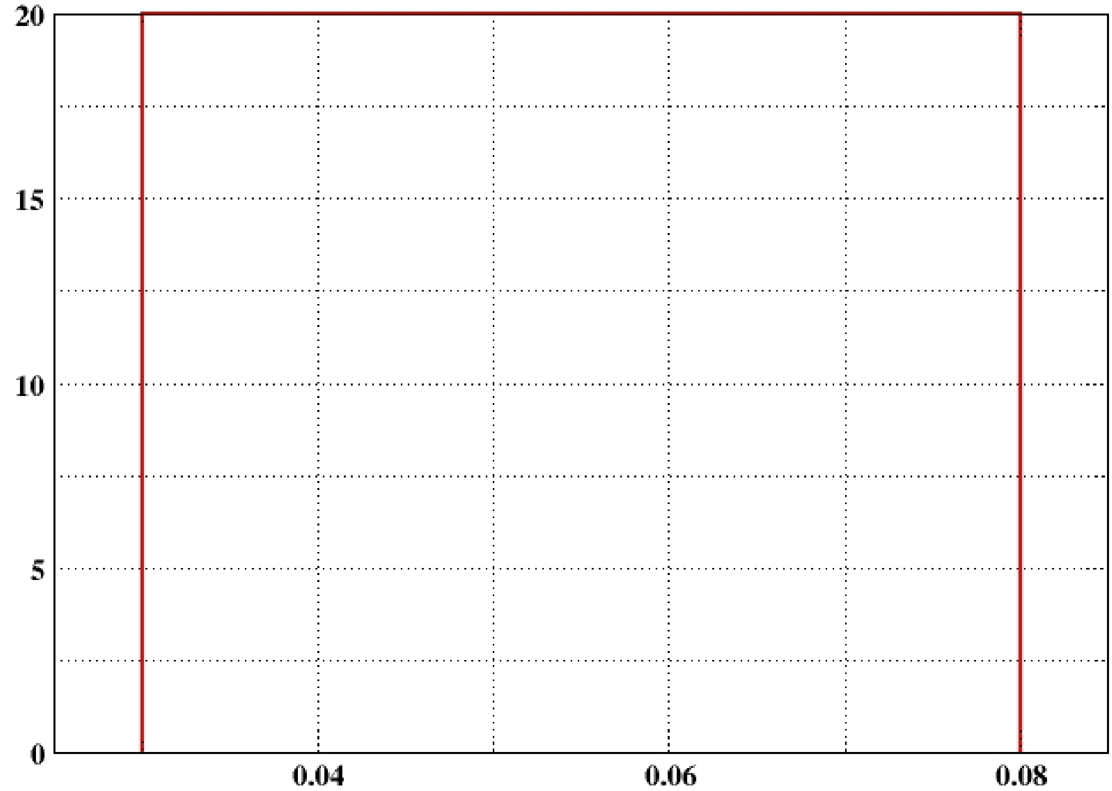


Figure 7.6-4 *Probabilistic calculation – Porosity accessible to anion diffusion in the microfissured zone – Distribution law and distribution function (initial law and sample with constraints) [Normal distribution]*

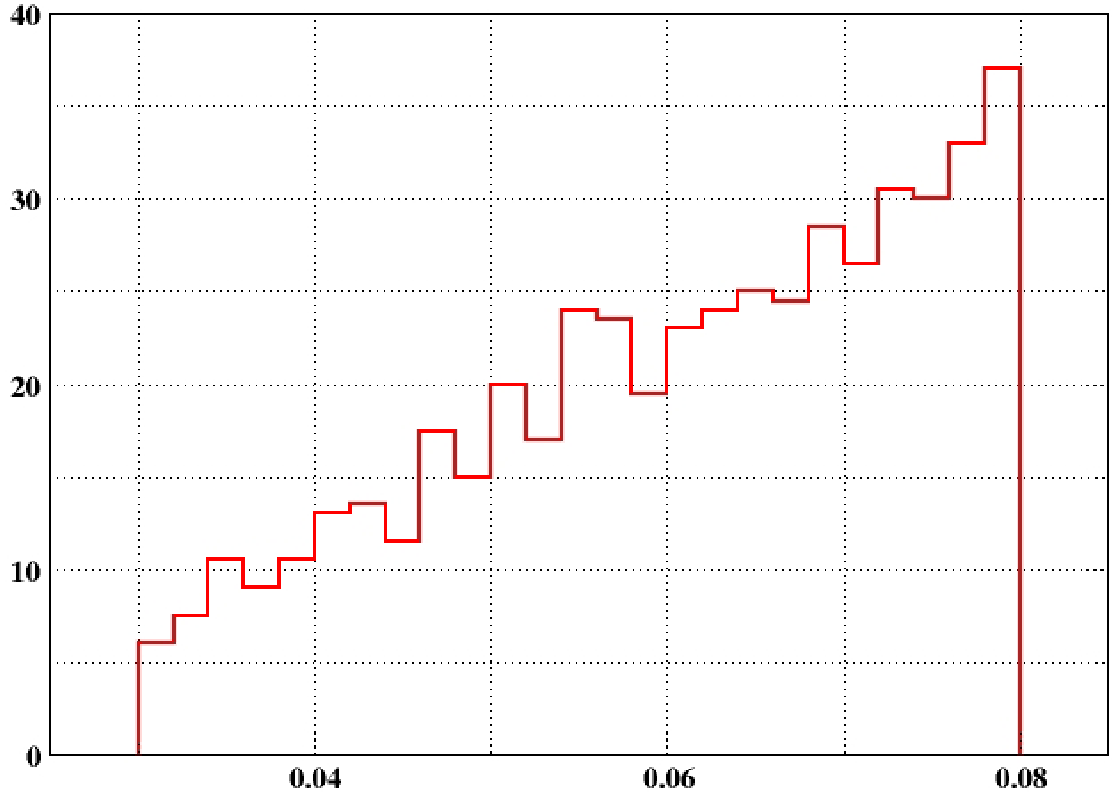


Figure 7.6-4 *Probabilistic calculation – Porosity accessible to anion diffusion in the microfissured zone – Distribution law and distribution function (initial law and sample with constraints) [Probabilist distribution]*

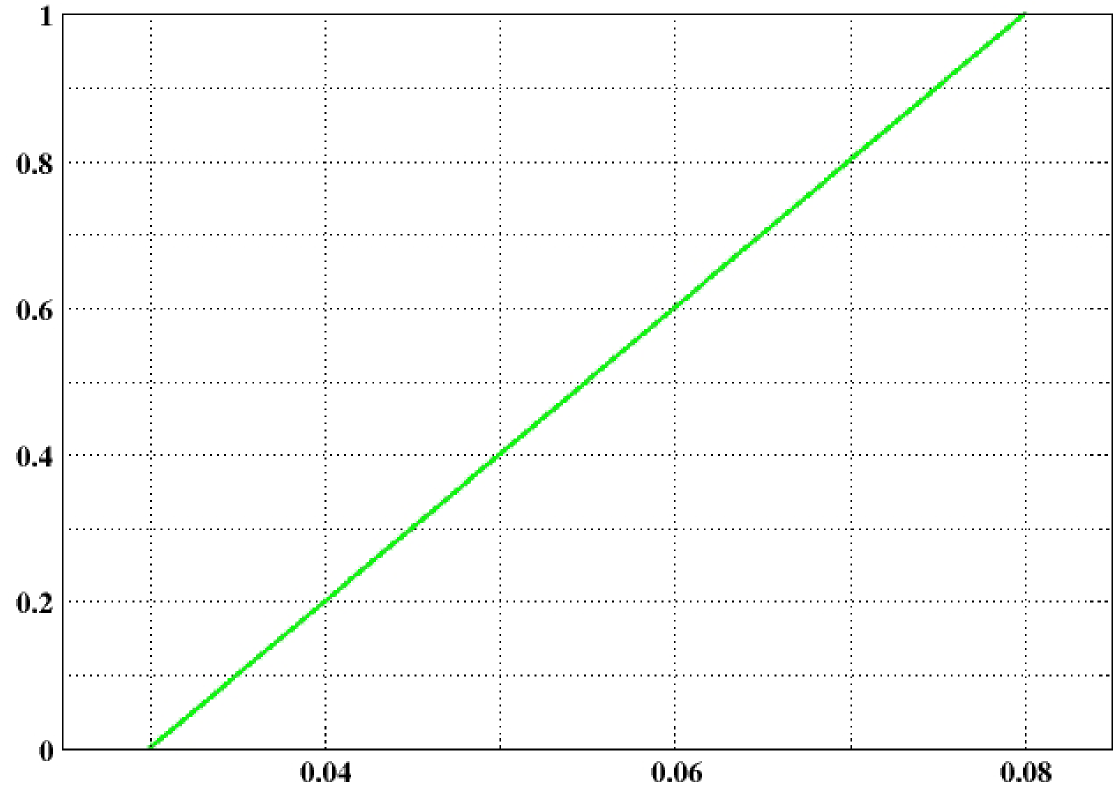


Figure 7.6-4 *Probabilistic calculation – Porosity accessible to anion diffusion in the microfissured zone – Distribution law and distribution function (initial law and sample with constraints) [Normal distribution and repartition]*

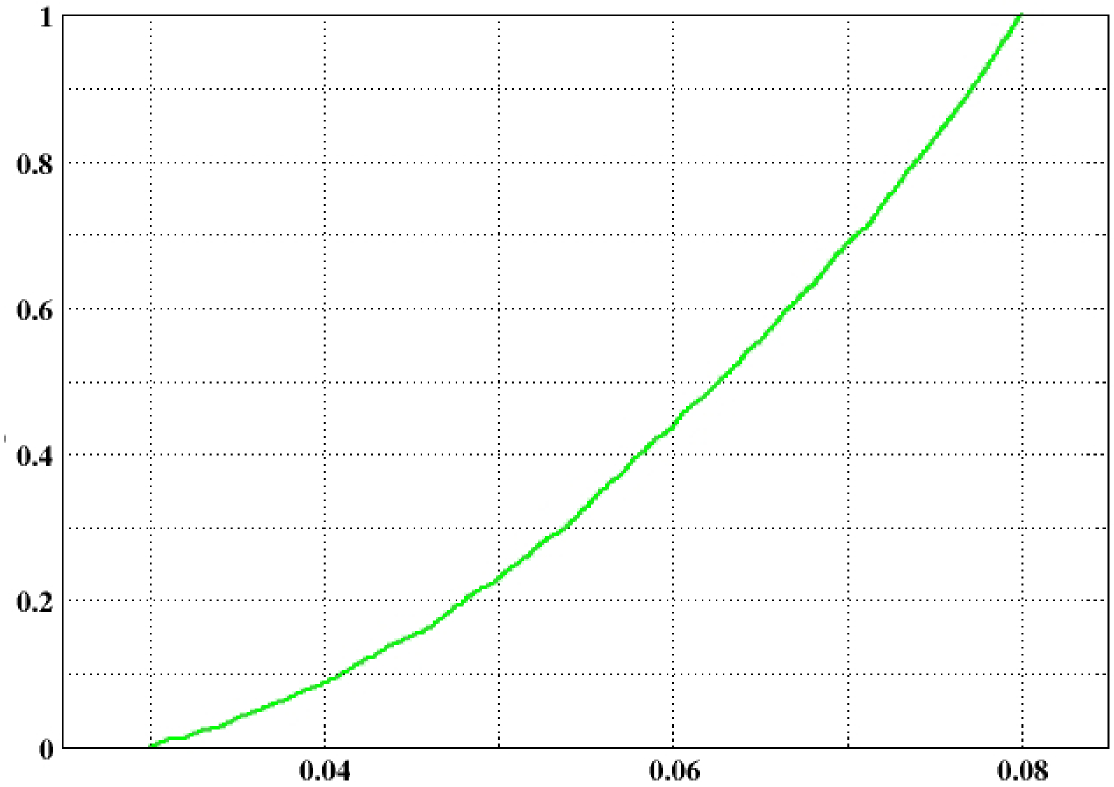


Figure 7.6-4 *Probabilistic calculation – Porosity accessible to anion diffusion in the microfissured zone – Distribution law and distribution function (initial law and sample with constraints) [Repartition function]*

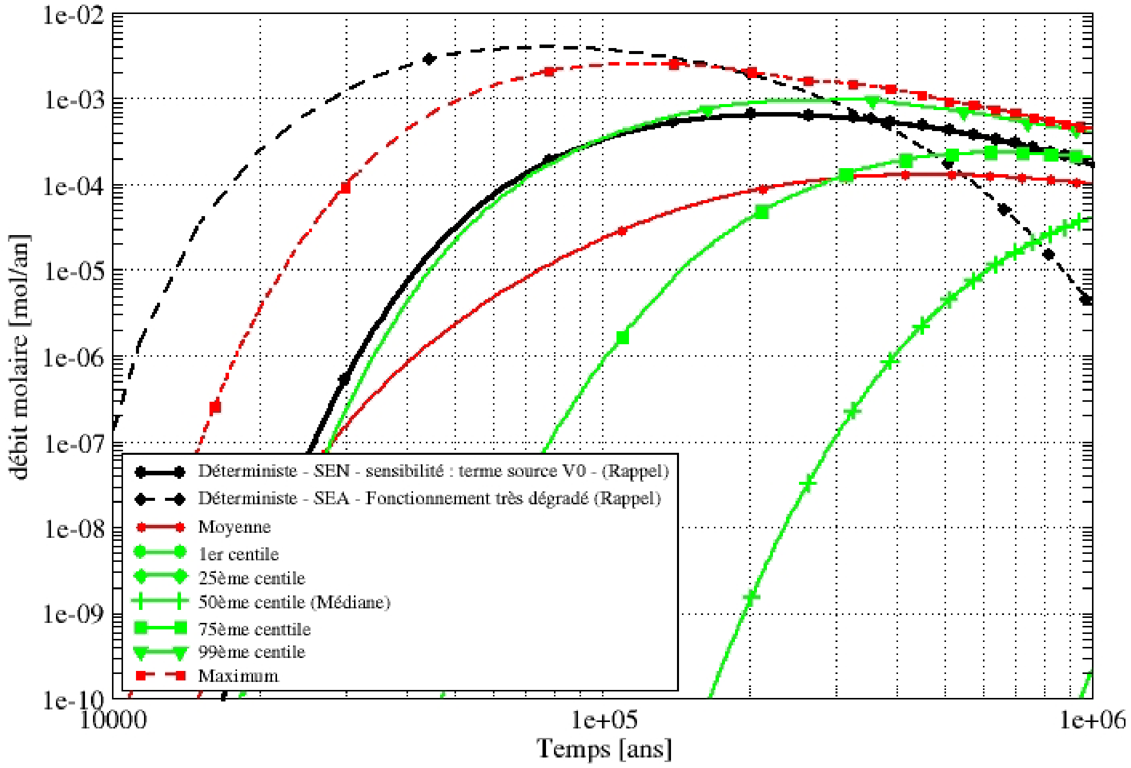


Figure 7.6-8 *Probabilistic calculation – Quantiles of molar flow rate out of the top of the Callovo-Oxfordian : reference packages C1/C2 –* ^{129}*I (with retention considered)*

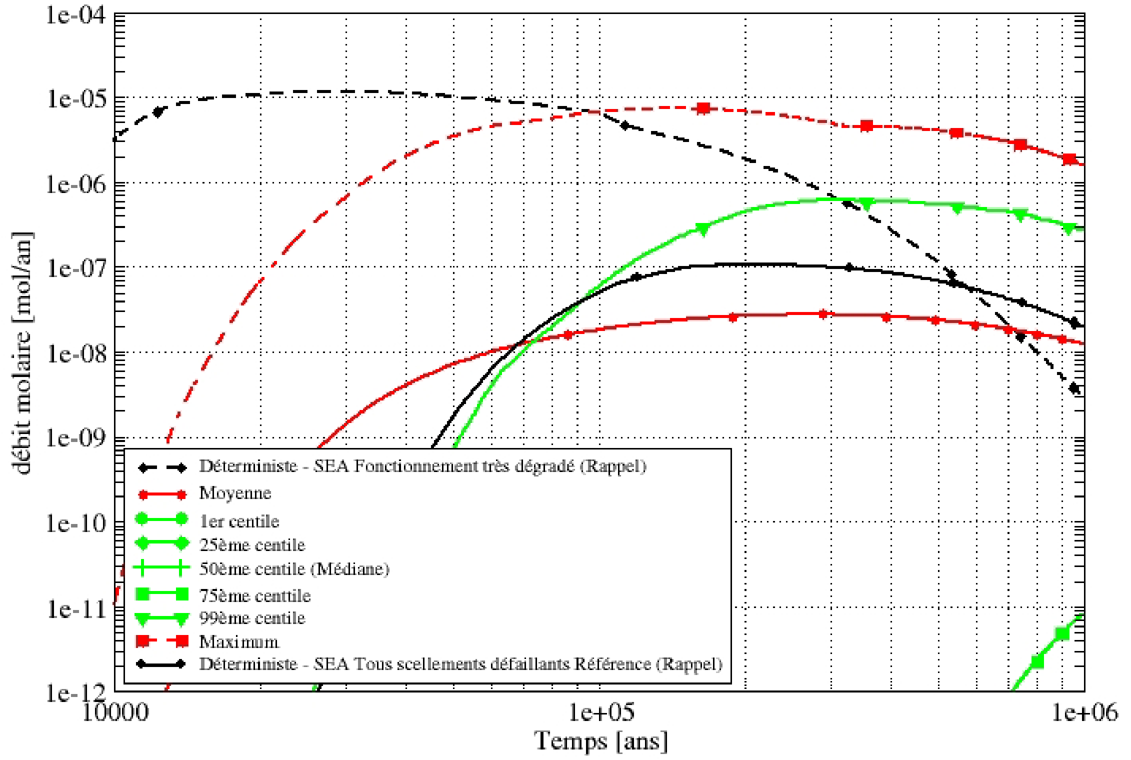


Figure 7.6-10 *Probabilistic calculation – Quantiles of molar flow rate out of the shafts : reference packages C1/C2 –* ^{129}*I (with retention considered)*

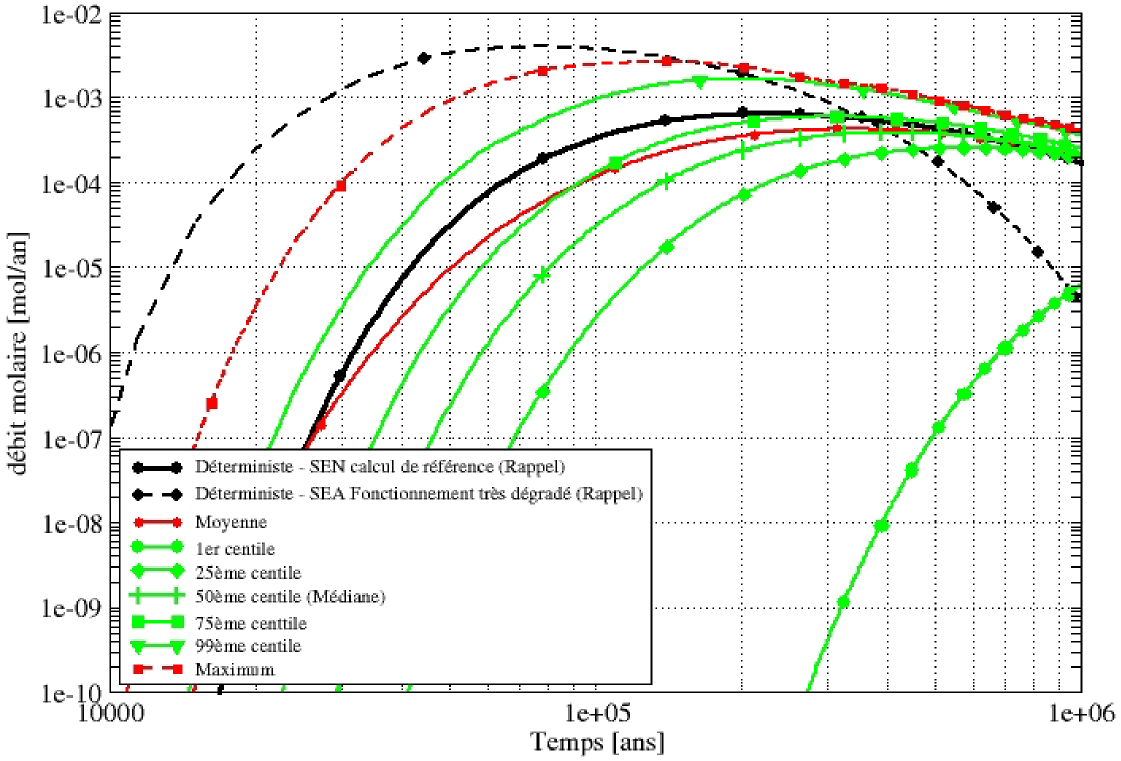


Figure 7.6-16 Probabilistic calculation – Percentiles of molar flow rate out of the top of the Callovo-Oxfordian with variable Kd and zero Kd : reference packages C1/C2 – ^{129}I [Sensibility]

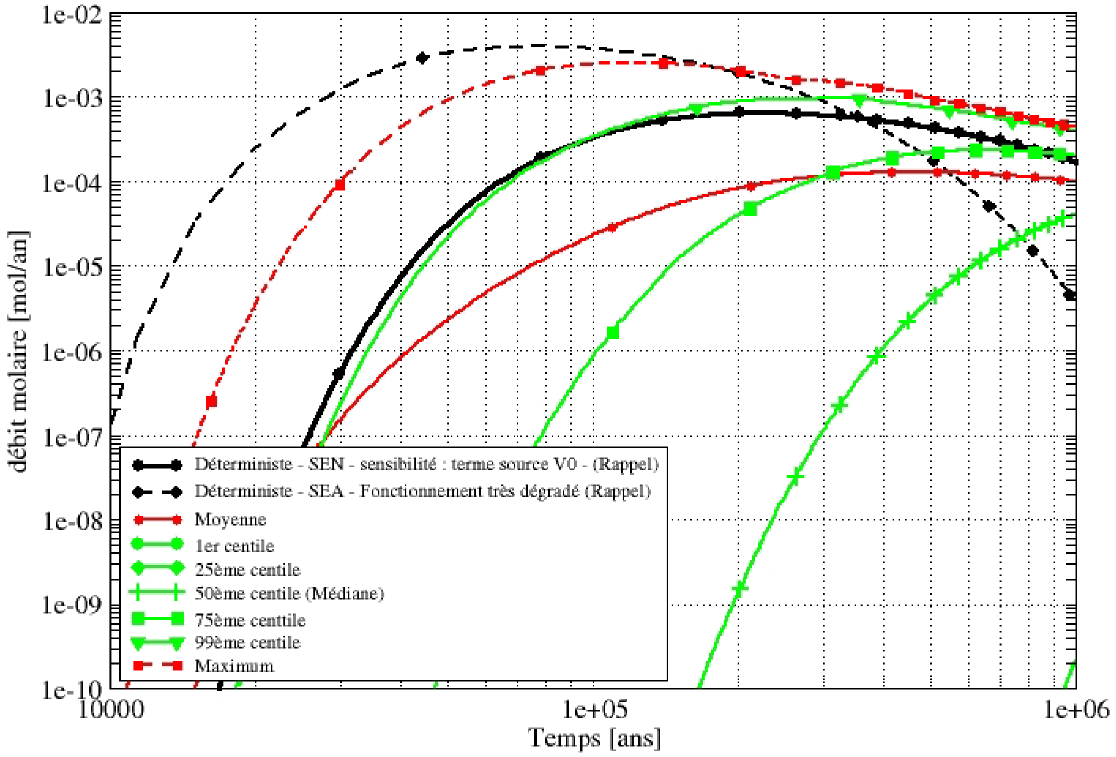


Figure 7.6-16 Probabilistic calculation – Percentiles of molar flow rate out of the top of the Callovo-Oxfordian with variable Kd and zero Kd : reference packages C1/C2 – ^{129}I [Reference]

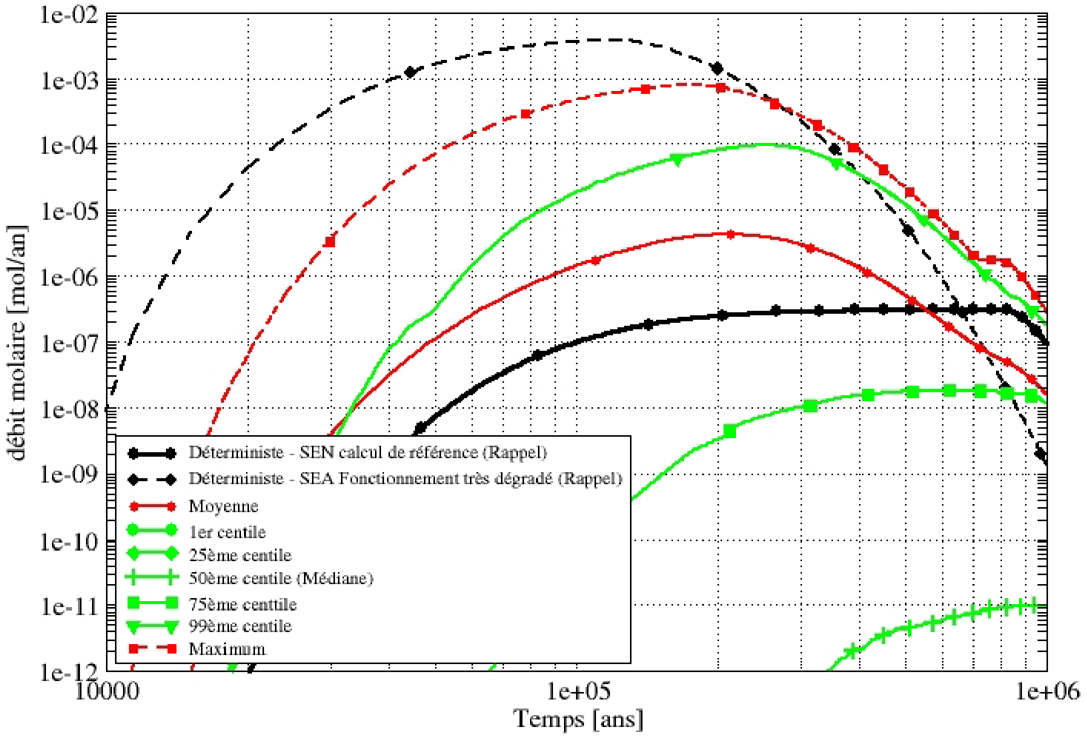


Figure 7.6-18 Probabilistic calculation – Quantiles of molar flow rate out of the top of the Callovo-Oxfordian : reference packages C1/C2 - ^{79}Se

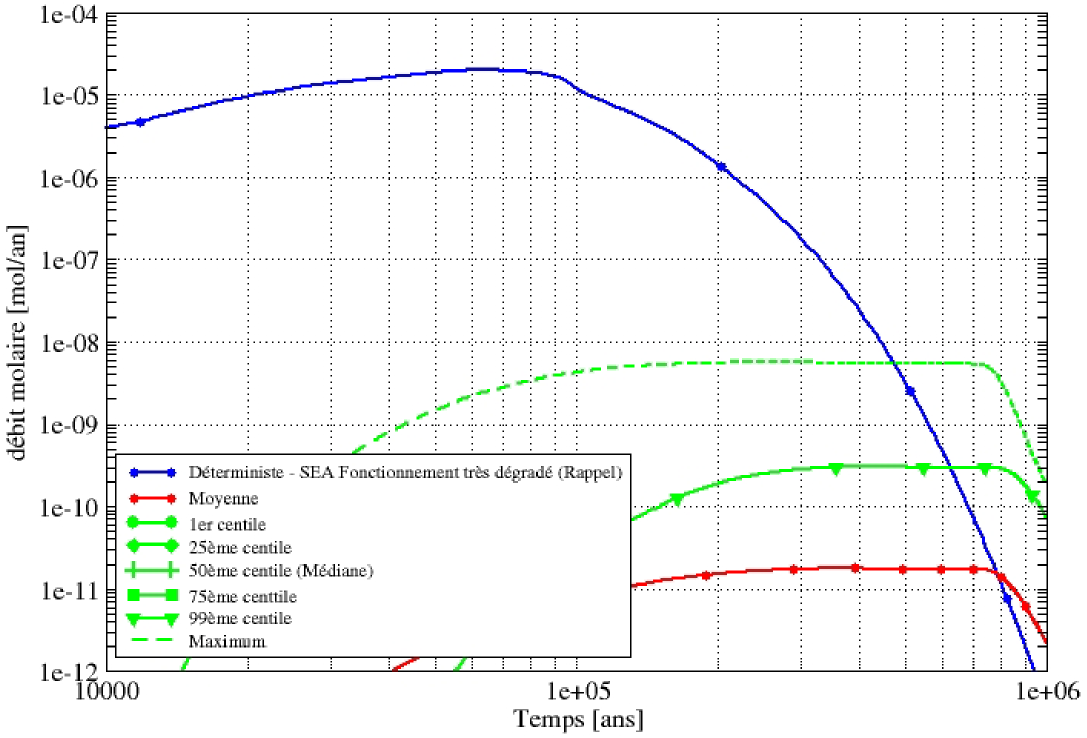


Figure 7.6-19 Probabilistic calculation – Quantiles of molar flow rate out of the shaft : reference packages C1/C2 - ^{79}Se

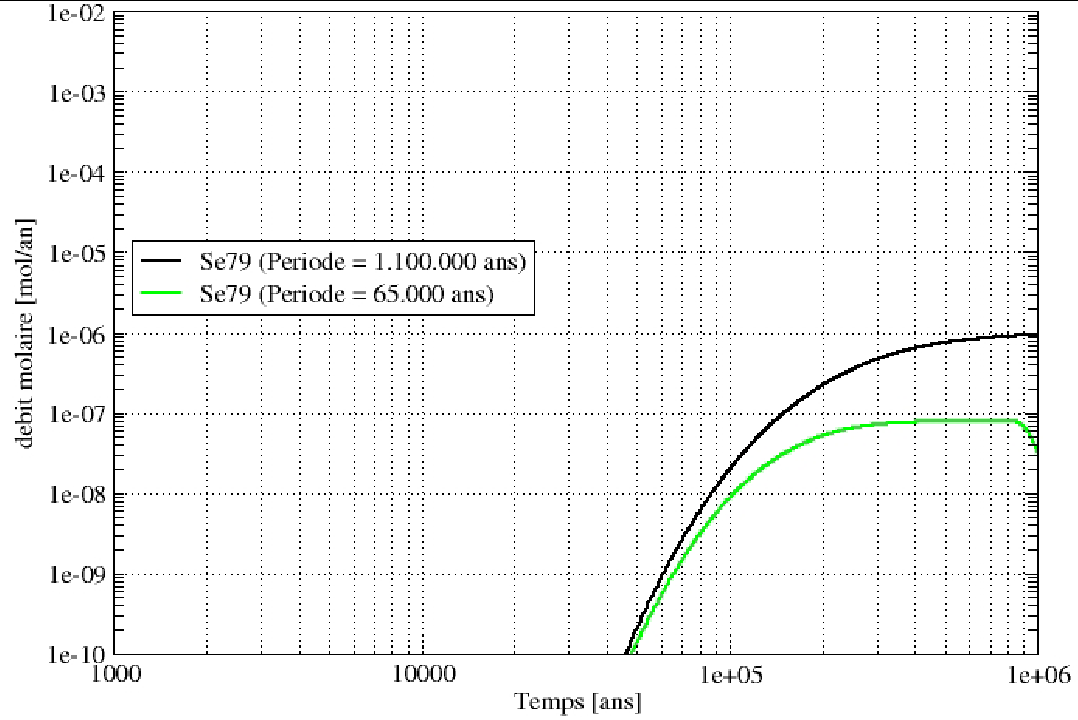


Figure A - Effects of the increased half-life of ^{79}Se – Dose history at the outlet – Reference packages C1/C2 [Normal evolution – Reference – Molar]

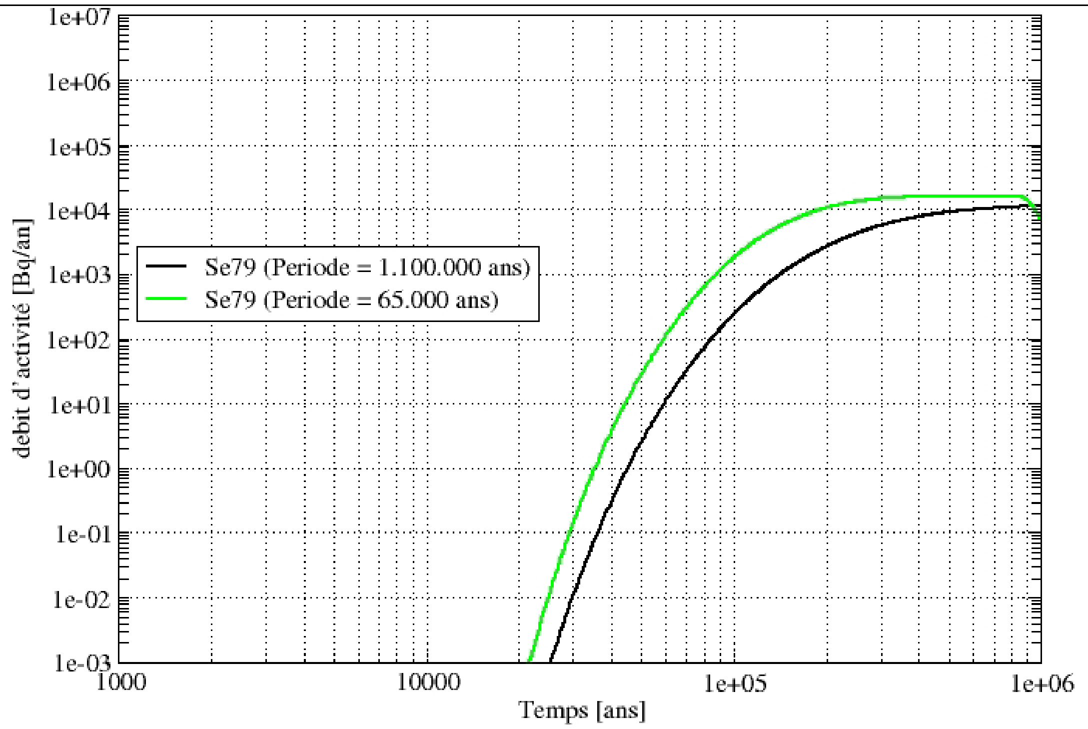


Figure A - Effects of the increased half-life of ^{79}Se – Dose history at the outlet – Reference packages C1/C2 [Normal evolution – Reference –Activity]

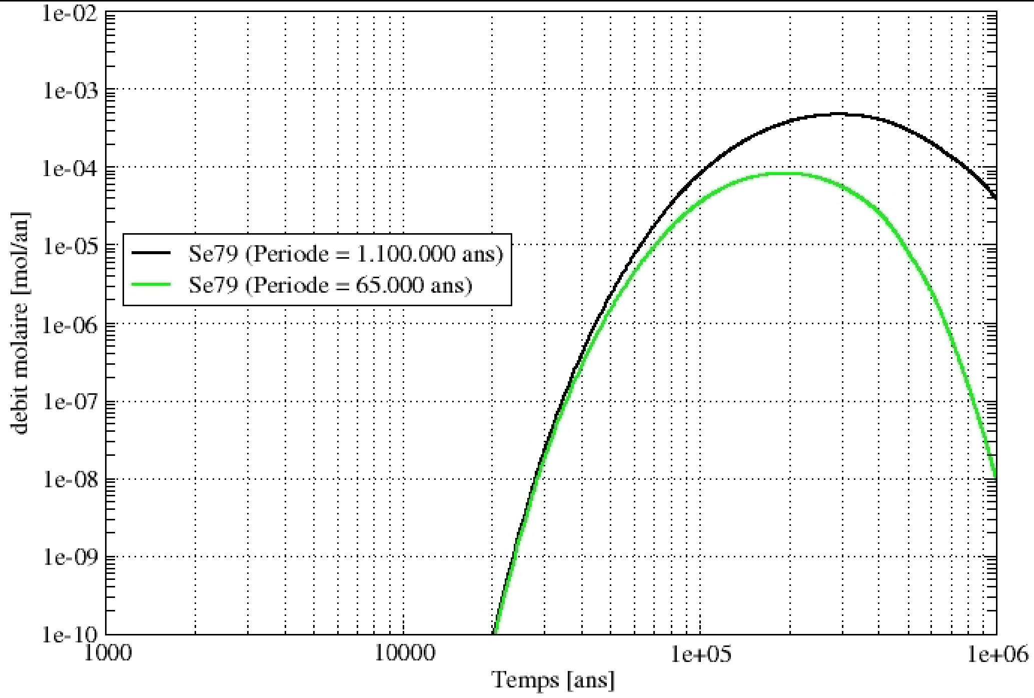


Figure A - Effects of the increased half-life of ^{79}Se – Dose history at the outlet – Reference packages C1/C2 [Normal evolution – Sensibility –Molar]

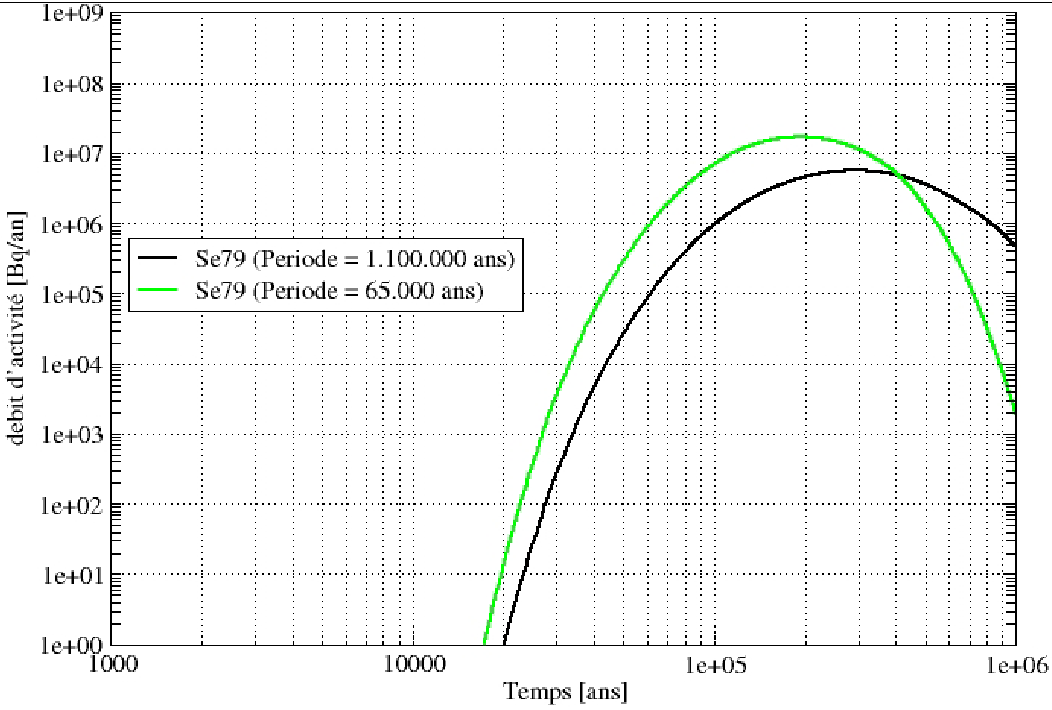


Figure A - Effects of the increased half-life of ^{79}Se – Dose history at the outlet – Reference packages C1/C2 [Normal evolution – Sensibility –Activity]

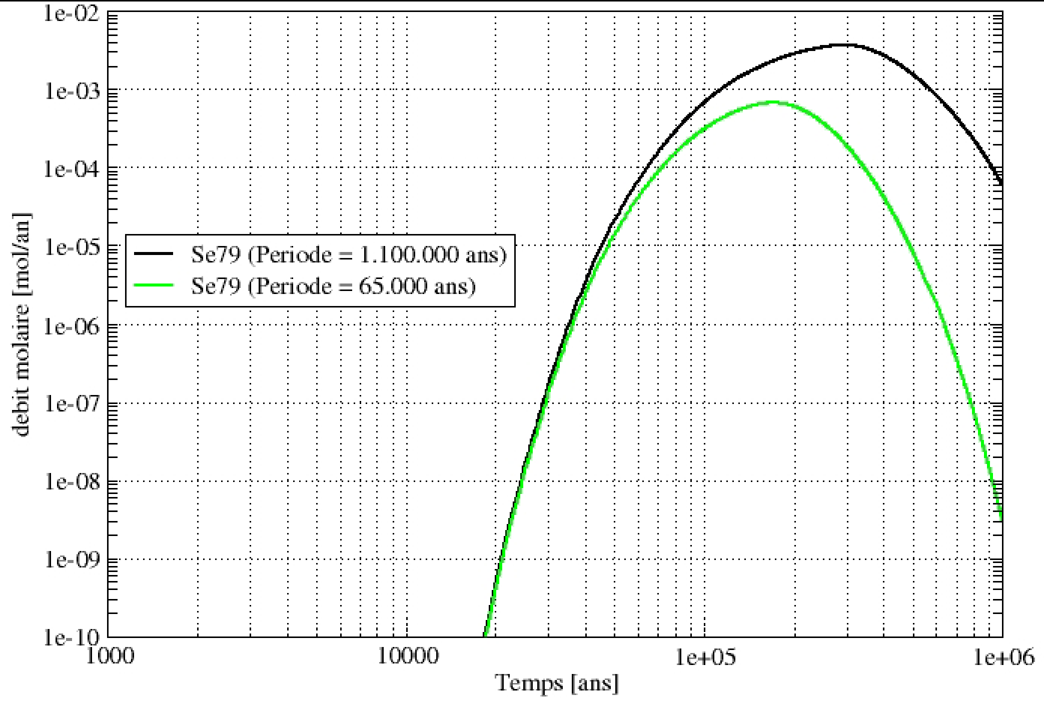


Figure A - Effects of the increased half-life of ^{79}Se – Dose history at the outlet – Reference packages C1/C2 [Hight degrated evolution – Molar]

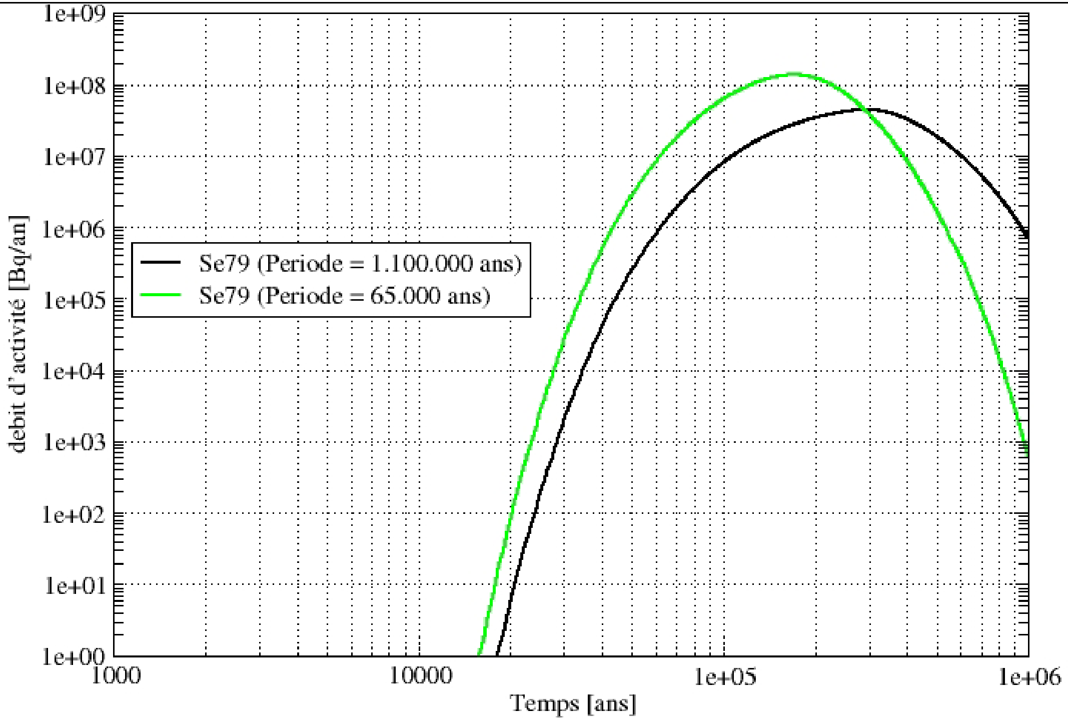


Figure A - Effects of the increased half-life of ^{79}Se – Dose history at the outlet – Reference packages C1/C2 [Hight degrated evolution – Activity]

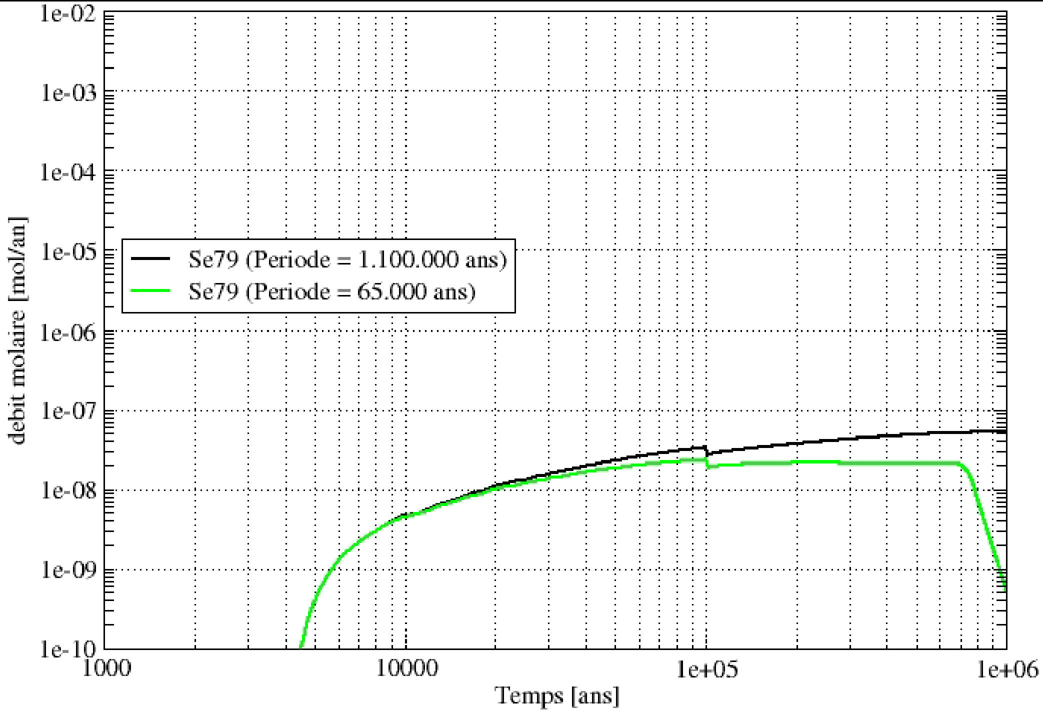


Figure A - *Effects of the increased half-life of* ^{79}*Se – Dose history at the outlet – Reference packages C1/C2 [Drill evolution – Molar]*

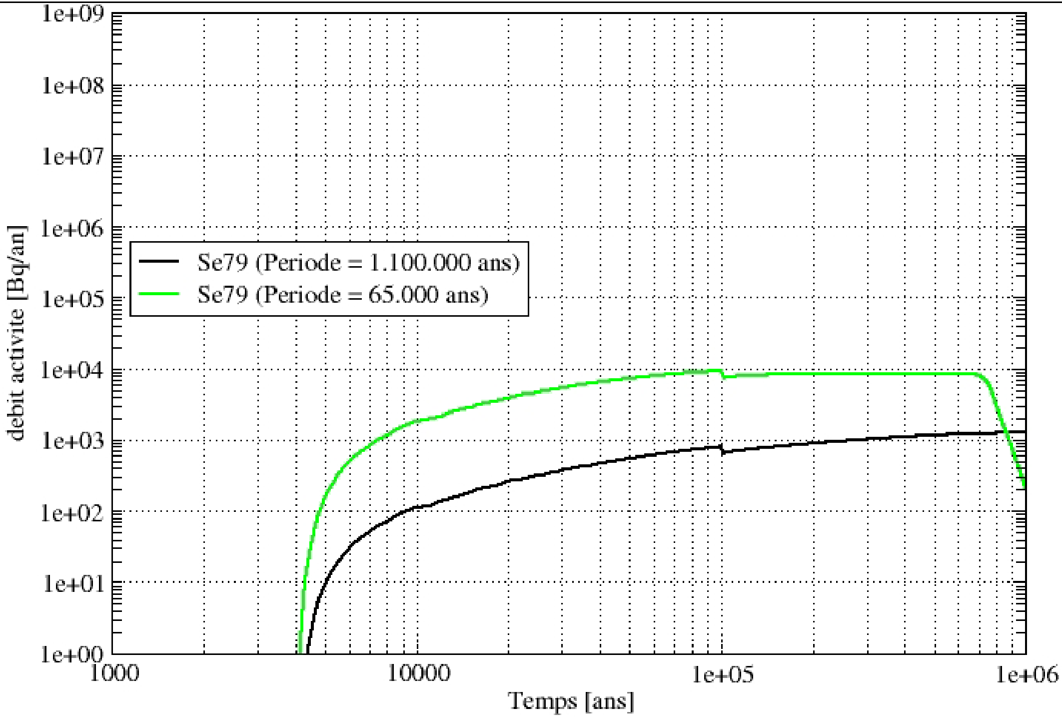


Figure A - *Effects of the increased half-life of* ^{79}*Se – Dose history at the outlet – Reference packages C1/C2 [Drill evolution – Activity]*

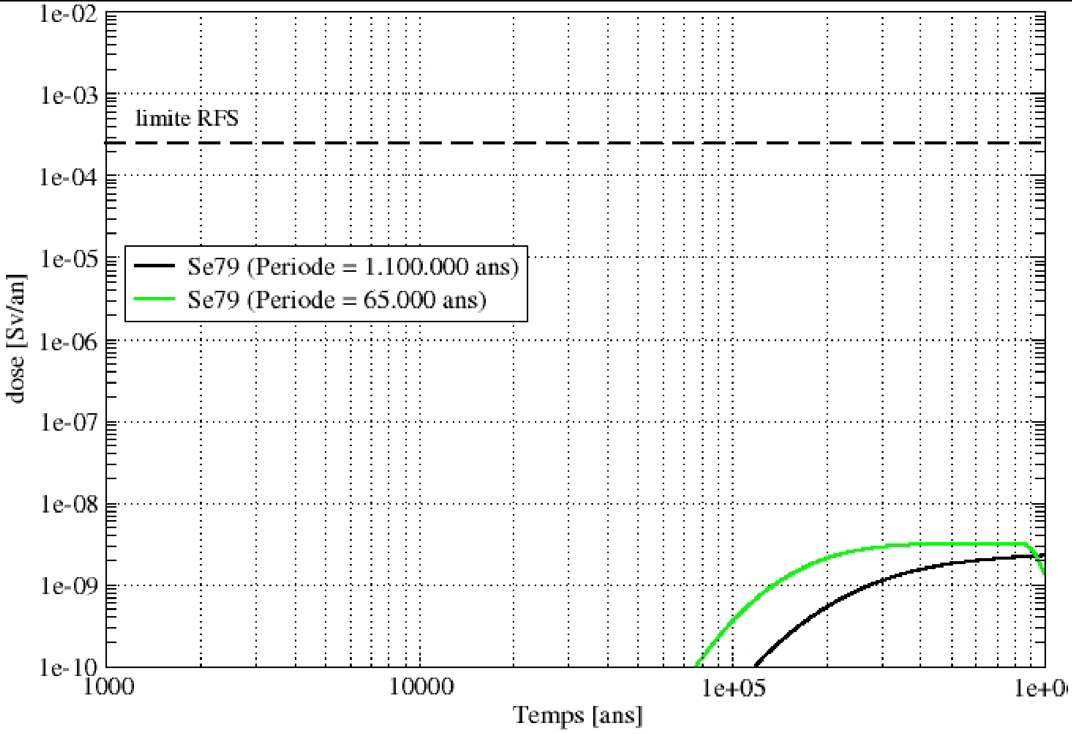


Figure B - *Effects of the increased half-life of ^{79}Se – Dose history at the outlet – Reference packages C1/C2 [SEN – Reference]*

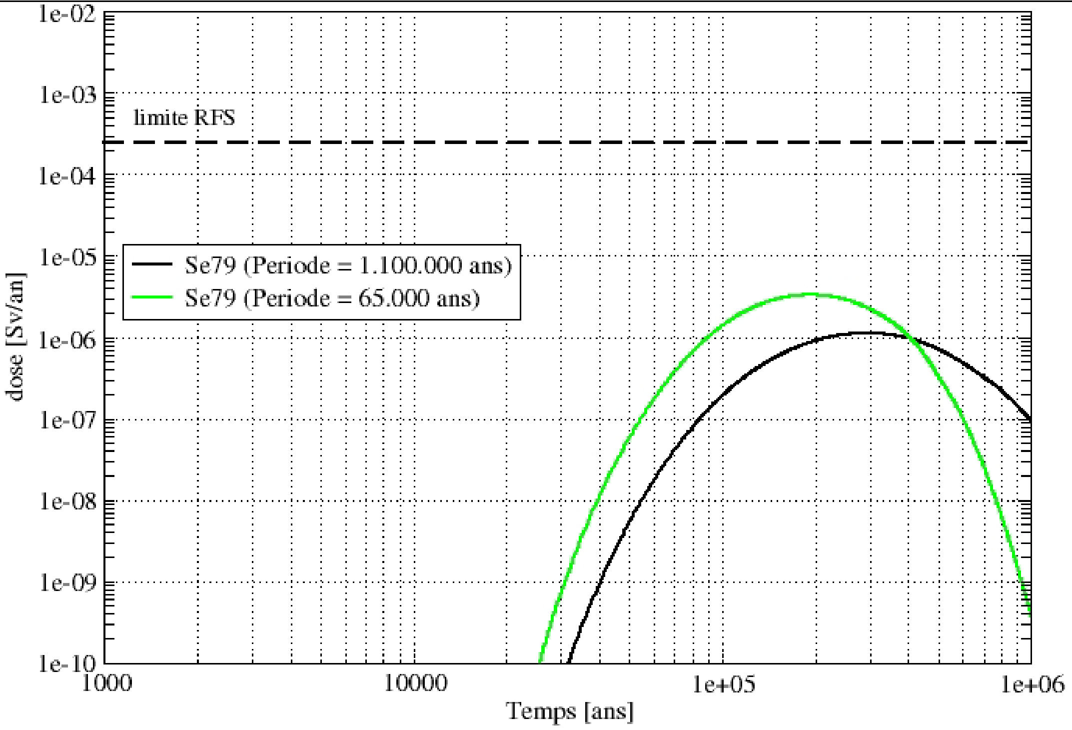


Figure B - *Effects of the increased half-life of ^{79}Se – Dose history at the outlet – Reference packages C1/C2 [SEN – Sensibility]*

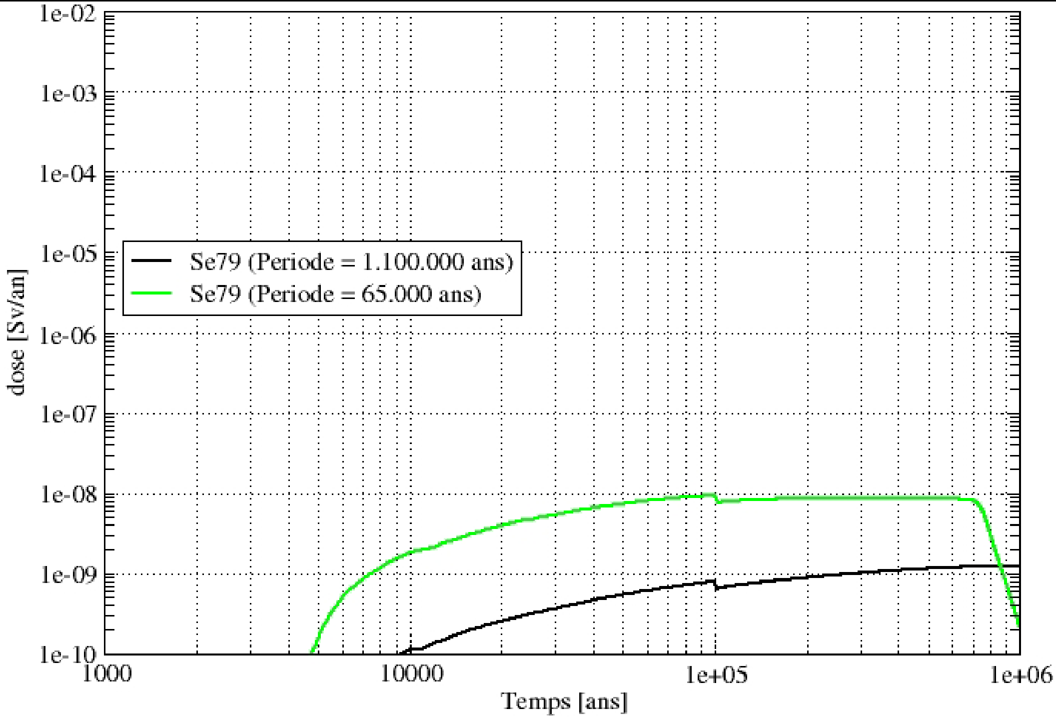


Figure B - *Effects of the increased half-life of ^{79}Se – Dose history at the outlet – Reference packages C1/C2 [SEA - Drill]*

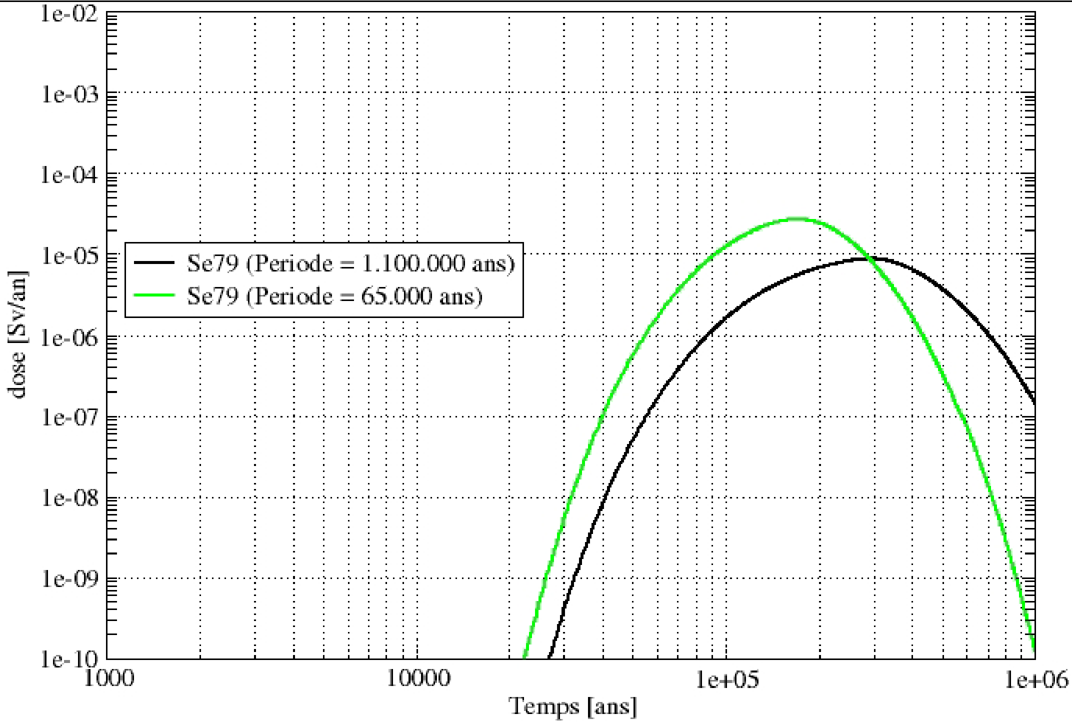


Figure B - *Effects of the increased half-life of ^{79}Se – Dose history at the outlet – Reference packages C1/C2 [SEA – Hight degrated]*

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