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National Cooperative for the
Disposal of Radioactive Waste

TECHNICAL REPORT 02-05

Project Opalinus Clay

Safety Report

Demonstration of disposal feasibility for spent fuel, vitrified high-level waste and long-lived intermediate-level waste (Entsorgungsnachweis)

December 2002

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Summary

This report presents a comprehensive description of the post-closure radiological safety assessment of a repository for spent fuel (SF), vitrified high-level waste (HLW) from the reprocessing of spent fuel and long-lived intermediate-level waste (ILW), sited in the Opalinus Clay of the Zürcher Weinland in northern Switzerland. This assessment has been carried out as part of the technical basis for Project *Entsorgungsnachweis*[1], which also includes a synthesis of information from geological investigations of the Opalinus Clay and a report on engineering feasibility. Project *Entsorgungsnachweis* is a milestone in the programme for the management of SF, HLW and ILW and represents an evaluation of the feasibility of the disposal of these wastes in Switzerland. It is also a major step on the way towards repository implementation.

The two main objectives of Project *Entsorgungsnachweis* are:

1. To demonstrate disposal feasibility of SF, HLW and ILW in the Opalinus Clay of the Zürcher Weinland in order to fulfil the requirements defined by the Federal Council in 1988 in its judgement of Project Gewähr 1985. This includes a demonstration that
   - a suitable geological environment for the repository exists (siting feasibility),
   - construction and operation of a repository is practicable in such an environment (engineering feasibility),
   - long-term safety from the hazards presented by the wastes is assured for such a repository (safety feasibility).
2. To provide a platform for discussion and a foundation for decision-making on how to proceed with the Swiss HLW programme. This includes a presentation of the key findings and results and a discussion of the underlying scientific basis. The excellent results obtained from the geological investigations led Nagra to propose to the Swiss Government to make a decision to focus future work for the waste management option "Disposal of SF / HLW / ILW in Switzerland" on the Opalinus Clay of the Zürcher Weinland[2]. Thus an additional objective is to provide the arguments to support such a decision. This proposal is supported by a long and systematic site selection procedure, which was developed in close cooperation with the regulatory authorities and with experts from the Federal Government. This stepwise procedure of narrowing down from (i) seven potential sedimentary host rock options to one (Opalinus Clay) and (ii) from two large regional investigation areas for Opalinus Clay to the potential siting area in the Zürcher Weinland is documented in several Nagra reports and summarised in a note by the regulator (see Chapter 1 of the present report for details). Discussion on the content and the timing of future work will follow the review of Project *Entsorgungsnachweis.*

Additional objectives of the project are:

3. To provide input for overall waste management planning (including cost estimates), to form a benchmark for assessing design alternatives and inventory variants (SF, HLW and various types of ILW) and to allow applied research and development priorities to be re-assessed to address any remaining safety-relevant issues and uncertainties.
4. To provide input for discussions of waste management issues with all stakeholders, most importantly with the public. Such discussions can contribute significantly to building an

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[1] The German term is also used in the English version of this report. The term translates into English as "demonstration of disposal feasibility".

[2] Disposal abroad is also an officially recognised option of the Swiss waste management strategy.

understanding of these issues, which in turn can lead to more constructive dialogue and perhaps increase public acceptance of waste management plans, including the implementation of repositories.

Specific aims of the safety assessment (this report) are:

1. To determine the suitability of the Opalinus Clay of the Zürcher Weinland as a host rock for the repository from the point of view of long-term safety.
2. To enhance the understanding of the multiple safety functions that the proposed disposal system provides.
3. To assess the robustness of the disposal system with respect to remaining uncertainties and the effects of phenomena that may adversely affect the safety functions.
4. To provide a platform for the discussion of a broad range of topics related to repository development. More specifically, the findings from the safety assessment, together with those from the regulatory authorities' review thereof, will provide guidance for future stages of repository planning and development.

Operational phase safety is not treated in this report, but is addressed in the engineering feasibility report on a qualitative level. The present report is restricted to post-closure radiological safety issues.

This report presents the arguments that together comprise the *safety case*, i.e. the case for the long-term safety of a repository for SF, HLW and ILW located in the Opalinus Clay of the Zürcher Weinland. The definition of this central term is as follows:

Safety Case

The safety case is the set of arguments and analyses used to justify the conclusion that a specific repository system will be safe. It includes, in particular, a presentation of evidence that all relevant regulatory safety criteria can be met. It includes also a series of documents that describe the system design and safety functions, illustrate the performance, present the evidence that supports the arguments and analyses, and that discuss the significance of any uncertainties or open questions in the context of decision making for further repository development.

Next, the key points are briefly introduced.

Chapter 1, as the introductory chapter, aims at **putting the present report into context[3].** With five nuclear power plants in operation (the first one since 1969), a substantial amount of radioactive wastes exists today in Switzerland that requires careful long-term management. A corresponding, detailed waste management concept is available: State-of-the-art interim storage facilities with enough capacity for all radioactive wastes of Swiss origin are in operation, and, for subsequent geological disposal of the wastes, there are detailed plans for two repositories (one for low-and intermediate level waste and one for SF, HLW and ILW). Key points of the legal framework include: (i) Radioactive waste arising in Switzerland shall, in principle, be disposed of in Switzerland (although the laws define the conditions under which, by way of an exception, an export licence for disposal of such wastes abroad may be granted); and (ii) in

[3] Status: August 31, 2002. The consequences of the negative outcome of the public referendum on the concession for an investigation gallery for the proposed L/ILW repository at Wellenberg are not discussed in this report.

Switzerland, the producers of radioactive waste are legally responsible for its safe management and disposal. To carry out their waste disposal responsibilities, the electricity supply utilities, which operate the nuclear power plants, and the Federal Government, which is responsible for the management of waste arising from medicine, industry and research, set up the National Cooperative for the Disposal of Radioactive Waste (Nagra) in 1972. Nagra is responsible for research and development work associated with final disposal. Other aspects of the waste management process, such as conditioning, interim storage and construction and operation of repositories, remain the responsibility of the individual waste producers or of organisations that may be set up by the producers specifically for these purposes. The Swiss Federal Nuclear Safety Inspectorate (HSK) is the supervisory authority. The Federal Commission for the Safety of Nuclear Installations (KSA) is responsible for evaluating projects for nuclear installations including radioactive waste repositories and for submitting to the Energy Department statements on the licensing applications and the reviews from HSK. In HSK's and KSA's guideline HSK-R-21, the protection objectives for disposal of radioactive waste are defined. One key argument of any Swiss safety case, including the current one, is the demonstration of compliance with this guideline. Finally, the step-wise approach to repository implementation is introduced; this includes a brief discussion of major past and future steps.

Chapter 2 is about **guidance and principles for choosing the disposal system and evaluating its long-term safety.** International guidance, the Swiss legal and regulatory framework and regulatory guidance are introduced and discussed. Key points include the draft of the revised Swiss Nuclear Energy Law (KEG[4]) and the Swiss regulatory guideline HSK-R-21. The KEG explicitly requires disposal in a geological repository, which must be monitored for some time before final closure. This requirement is based on the concept of "monitored long-term geological disposal" as proposed by the government advisory group, EKRA[5]. All these documents, all of which influence the approach to providing and analysing safety, are supplemented by additional guidance developed internally by Nagra based on its experience both in Switzerland and through its interaction with other organisations abroad. This includes a discussion of the concept of robustness, the role and treatment of the biosphere, the treatment of future human actions and the timescales of concern. All this information comes from many different sources and is rather heterogeneous. Therefore, the major findings are grouped and summarised in the following objectives and principles: (i) Objectives of geological disposal, (ii) objectives related to the system (including the *safety functions* of the disposal system, which are central to the development of the safety case), (iii) objectives related to stepwise implementation, (iv) assessment principles. Because of their important role, the safety functions are listed and defined explicitly below:

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4 Short for "Kernenergiegesetz" (Nuclear Energy Law).

5 Short for "Expertengruppe Entsorgungskonzepte für radioaktive Abfälle" (Expert Group on Disposal Concepts for Radioactive Waste) set up by the Swiss Government in June 1999.

Safety Functions

The disposal system performs a number of functions relevant to long-term security and safety. These are termed safety functions; they include:

- **Isolation from the human environment** – The safety and security of the waste, including fissile material, is ensured by placing it in a repository located deep underground, with all access routes backfilled and sealed, thus isolating it from the human environment and reducing the likelihood of any undesirable intrusion and misapplication of the materials. Furthermore, the absence of any currently recognised and economically viable natural resources and the lack of conflict with future infrastructure projects that can be conceived at present reduces the likelihood of inadvertent human intrusion. Finally, appropriate siting ensures that the site is not prone to disruptive events and to processes unfavourable to long-term stability.
- **Long-term confinement and radioactive decay within the disposal system** – Much of the activity initially present decays while the wastes are totally contained within the primary waste containers, particularly in the case of SF and HLW, for which the high integrity steel canisters are expected to remain unbreached for at least 10 000 years. Even after the canisters are breached, the stability of the SF and HLW waste forms in the expected environment, the slowness of groundwater flow and a range of geochemical immobilisation and retardation processes ensure that radionuclides continue to be largely confined within the engineered barrier system and the immediately surrounding rock, so that further radioactive decay takes place.
- **Attenuation of releases to the environment** – Although complete confinement cannot be provided over all relevant times for all radionuclides, release rates of radionuclides from the waste forms are low, particularly from the stable SF and HLW waste forms. Furthermore, a number of processes attenuate releases during transport towards the surface environment, and limit the concentrations of radionuclides in that environment. These include radioactive decay during slow transport through the barrier provided by the host rock and the spreading of released radionuclides in time and space by, for example, diffusion, hydrodynamic dispersion and dilution.

Chapter 3 defines the **methodology for developing the safety case**, i.e. the approach to the evaluation of long-term safety of the proposed repository, based on the assessment principles introduced in Chapter 2. Key points include: (i) The identification of the necessary steps in developing the safety case, and (ii) the definition of the lines of argument contributing to the safety case. These points are given in more detail below.

(i) The making of the safety case involves:

- the choice of a disposal system, via a flexible repository development strategy, that is guided by the results of earlier studies, including studies of long-term safety,
- the derivation of the system concept, based on current understanding of the features, events and processes (FEPs) that characterise, and may influence, the disposal system and its evolution,
- the derivation of the safety concept, based on well understood and effective pillars of safety,
- the illustration of the radiological consequences of the disposal system through the definition and analysis of a wide range of assessment cases, and

- the compilation of the arguments and analyses that constitute the safety case, as well as guidance for future stages of the repository programme.

(ii) Lines of argument that contribute to the safety case are related to:

- the strength of geological disposal as a waste management option,
- the safety and robustness of the chosen disposal system,
- the reduced likelihood and consequences of human intrusion,
- the strength of the stepwise repository implementation process,
- the good scientific understanding that is available and relevant to the chosen disposal system and its evolution,
- the adequacy of the methodology and the models, codes and databases that are available to assess radiological consequences,
- the multiple arguments for safety that include compliance with regulatory safety criteria, the use of complementary safety indicators, the existence of reserve FEPs and the lack of outstanding issues with the potential to compromise safety.

Chapter 4 documents the **current understanding of the characteristics of the proposed disposal system in the Opalinus Clay of the Zürcher Weinland at the time of repository closure.** The disposal system is sited and designed in accordance with the objectives and principles introduced in Chapter 2. The following elements are described in some detail: (i) key features of the site, (ii) key properties of the Opalinus Clay host rock, (iii) the repository layout, (iv) waste quantities and characteristics (SF, HLW and ILW), (v) the engineered barrier system and repository design. These elements are described in more detail below.

(i) Key features of the site include:

- The geological environment is simple, with predictable structural, hydrogeological and geochemical properties.
- The siting area is tectonically stable over the relevant timescales, with a low rate of uplift and associated erosion.
- The sediments overlying the basement in this region, and the basement rocks themselves, are not considered to have any significant natural resource potential.

(ii) Key properties of the Opalinus Clay host rock include:

- The Opalinus Clay has such a low hydraulic conductivity that solute movement through the formation is predominantly by diffusion rather than advection.
- The geochemical conditions in the Opalinus Clay are reducing, slightly alkaline and moderately saline and favour the preservation of the engineered barriers and radionuclide retention.
- The geochemical environment in the Opalinus Clay and surrounding formations is expected to remain effectively stable for several million years[6].
- The properties of the Opalinus Clay ensure that repository-induced and natural fractures will be of very low hydraulic conductivity due to the self sealing capacity of the Opalinus Clay; i.e. their effect on the hydraulic properties of the Opalinus Clay will be negligible.

[6] Only a minor salinity decrease is expected to occur over this timescale.

- The Opalinus Clay is an indurated claystone (clay shale) with reasonable engineering properties, allowing small, unlined tunnels and larger, lined tunnels to be constructed at depths of several hundred metres.

(iii) Key features of the repository layout include:

- an access ramp, construction and operations tunnels, central waste receiving facilities and a construction/ventilation shaft,
- an array of parallel, near-horizontal emplacement tunnels for SF / HLW, with a diameter of 2.5 m and a distance between tunnels of 40 m,
- three short horizontal emplacement tunnels for ILW located at a distance of several hundred metres away from the SF / HLW part of the repository,
- pilot and test facilities in accordance with the EKRA concept.

(iv) Key points regarding the waste include:

- Waste quantities are estimated based on the assumption of a 60 years operational lifetime of the five nuclear power plants currently in operation in Switzerland, corresponding to a total electricity production of 192 GWa(e).
- The quantities of SF on one hand and of HLW/ILW on the other hand are estimated based on the assumption that only that total amount of SF is reprocessed for which reprocessing contracts exist (1195 t_{IHM}).
- This leads to about 3200 t_{IHM} or 2065 canisters of SF, 730 canisters of HLW and about 7300 m^3 of (conditioned) ILW.

(v) Key features of the engineered barrier system include:

- The SF canister - the reference design concept is a cast steel body, with a machined central square channel fitted with crossplates to permit emplacement of either 4 PWR or 9 BWR fuel assemblies, with a minimal wall thickness of 15 cm, a length of about 5 m and a diameter of about 1 m.
- The HLW canister - the reference design is the same as in the Project *Gewähr* study; i.e. a steel canister containing one stainless steel flask of HLW glass, with a wall thickness of 25 cm, a length of about 2 m and a diameter of about 1 m.
- The ILW containers - the ILW drums are incorporated into concrete emplacement containers, with cementitious mortar used to fill the void spaces around the drums.
- Backfilling (SF / HLW emplacement tunnels) - the SF / HLW canisters are placed on highly compacted bentonite blocks, co-axially with the tunnel axis, with a distance of 3 m between the canisters, and the region around the canisters backfilled with granular bentonite.
- Backfilling (ILW emplacement tunnels) - after emplacement of the ILW containers in the ILW tunnels, the void regions within the emplacement tunnels are filled with a cementitious mortar.
- Backfilling (operations and construction tunnels, central area, ramp) - the operations and construction tunnels, the central area and the ramp are backfilled with a bentonite-sand mixture, with seals of highly compacted bentonite contained between bulkheads placed at several locations. The shaft will be sealed with highly compacted bentonite.

Chapter 5 is a **description of how the disposal system could evolve with time** after repository closure, taking into account the interactions of individual system components. This includes a discussion of the evolution of the site and of the engineered barriers. Besides describing the expected evolution of the disposal system, possible deviations from this course are also discussed. This is the basis for the development of the range of assessment cases that are quantitatively analysed in detail in this report.

Chapter 6 analyses **the relative importance of the different features and phenomena** and **identifies the assessment cases.** It includes deterministic sensitivity analyses of the consequences of possible deviations from the expected evolution, based on the *safety functions* defined in Chapter 2, as well as complementary probabilistic safety / sensitivity analyses. Based on a qualitative discussion and on insight from the quantitative analyses, the key features and phenomena contributing to the safety functions are identified; these are termed *pillars of safety.* Because of their important role in developing the safety case, they are listed and defined explicitly below:

Pillars of Safety

The pillars of safety are features of the disposal system that are key to providing the *safety functions:*

- **The deep underground location of the repository**, in a setting that is unlikely to attract human intrusion and is not prone to disruptive geological events and to processes unfavourable to long-term stability;
- **the host rock** which has a low hydraulic conductivity, a fine, homogeneous pore structure and a self-sealing capacity, thus providing a strong barrier to radionuclide transport and a suitable environment for the engineered barrier system;
- **a chemical environment** that provides a range of geochemical immobilisation and retardation processes, favours the long-term stability of the engineered barriers, and is itself stable due to a range of chemical buffering reactions;
- **the bentonite buffer (for SF and HLW)** as a well-defined interface between the canisters and the host rock, with similar properties as the host rock, that ensures that the effects of the presence of the emplacement tunnels and the heat-producing waste on the host rock are minimal, and that provides a strong barrier to radionuclide transport and a suitable environment for the canisters and the waste forms;
- **SF and HLW waste forms** that are stable in the expected environment;
- **SF and HLW canisters** that are mechanically strong and corrosion resistant in the expected environment and provide absolute containment for a considerable period of time.

Assuming first that the pillars of safety operate as expected, and then considering possible perturbations (based on the discussion in Chapters 4 and 5 and selected using the insight gained in the sensitivity analysis), the cases for quantitative assessment are identified.

Chapter 7 presents **the results of the analysis of the assessment cases** identified in Chapter 6. It starts with a description of the conceptualisation of the assessment cases, which are structured according to the various possible evolutions of the disposal system (scenarios) that determine the main pathway of radionuclide release. For a given scenario, different conceptualisations are considered, and for a given conceptualisation, data uncertainty is evaluated by parameter variations.

The starting point is the Reference Case, based on the Reference Conceptualisation of the Reference Scenario, which envisages a repository with a near field evolving according to the design functions of the engineered barriers, a geosphere based on the current understanding of the geological environment and a biosphere based on present-day geomorphological, hydrogeological and climatic conditions, with conservative assumptions regarding human behaviour and diet. Alternative conceptualisations of the Reference Scenario address phenomena in the near field and the geosphere where uncertainty exists about their importance for the reference radionuclide release pathway. Data uncertainty within alternative conceptualisations is investigated by parameter variations. The effects of uncertainty in the future evolution of the system is explored by means of alternative scenarios. As in the Reference Scenario, different conceptualisations and parameter variations are considered in the alternative scenarios. In order to test the robustness of the repository system, a category of "what if?" cases has been introduced addressing phenomena that are outside the range of possibilities supported by scientific evidence. To limit the number of "what if?" cases, they are restricted to those that explore perturbations to key properties of the pillars of safety. The list of "what if?" cases is not intended to be exhaustive, but is meant to illustrate system behaviour under extreme conditions.

Design and system options are evaluated separately, because they address conceptualisations where flexibility, rather than uncertainty, exists in the characteristics of the repository system.

The sensitivity of radionuclide transport in the biosphere is illustrated by a number of assessment cases related to alternative geomorphological and climatic conditions. In the framework of this group of cases, the focus is on illustrating the effect of uncertainties related to the biosphere using different (stylised) possibilities for the characteristics and evolution of the surface environment.

Chapter 8 is a **synthesis of the main arguments and results.** It represents the final step in compiling the safety case. It looks again at each of the lines of argument identified in Chapter 3 and summarises the findings that support them. Specifically:

- The strength of geological disposal as a waste management option is supported by (i) the internationally recognised fact that a well-chosen disposal system located at a well-chosen site fulfils the requirement of ensuring the safety and protection of humans and the environment, as well as security from malicious intervention, now and in the future, (ii) by the existence of suitable rock formations in Switzerland and elsewhere, (iii) by other safety assessments conducted world-wide, (iv) by observations of natural systems, and (v) by the relative advantage of geological disposal versus other options.
- The safety and robustness of the disposal system is ensured by (i) a set of passive barriers with multiple phenomena contributing to the safety functions, (ii) by the avoidance of uncertainties and detrimental phenomena through an appropriate choice of site and design, and (iii) by the long-term stability of the Opalinus Clay host rock and the repository due to a suitable geological situation.
- The reduced likelihood and consequences of human intrusion is supported by (i) the preservation of information about the repository, (ii) by the avoidance of resource conflicts (i.e. the absence of viable natural resources in the area proposed for the repository), and (iii) by the compartmentalisation of the repository and the solidification of the wastes.
- The strength of the stepwise repository implementation process is supported by (i) the fact that at a the current project stage, not all the details of the repository system need to be fixed and, therefore, the information basis only needs to be adequate for that particular stage, (ii) by the reliance on understood and reliably characterised components (site, EBS), (iii) by the involvement of stakeholders and the opportunities for feedback and improvements,

(iv) by the flexibility of the project, allowing to take into account new findings (e.g. regarding the detailed allocation of emplacement tunnels, choice of design options, placement of surface facilities, and even siting; i.e. other possible sites for the Opalinus Clay host rock option as well as for other host rock options exist), and (v) by the possibilities for monitoring and reversal (including retrieval of the wastes).

- The good scientific understanding that is available and relevant to the chosen disposal system and its evolution is supported by (i) the results from regional and local field investigation programmes, which included, as key elements, an extensive 3 D seismic campaign and an exploratory borehole in the potential siting area and the availability of information from other boreholes in the region, as well as complementary studies performed in the Mont Terri underground rock laboratory and in other laboratories, as well as observations of Opalinus Clay in a number of railway and road tunnels, (ii) by the findings from more than 20 years of experience in developing and characterising engineered barrier system components within the Swiss programme, as well as by the availability of a strong international information basis, and (iii) by the availability of a detailed model waste inventory for SF, HLW and ILW.

- The adequacy of the methodology and the models, codes and databases that are available to assess the radiological consequences for a broad spectrum of cases is supported by the adherence to the assessment principles outlined in Chapter 2 (especially by the adequacy of the consideration and treatment of uncertainties and the use of validated and verified models to evaluate the assessment cases).

- Compliance with legal and regulatory requirements is supported by (i) the fact that in all of the assessment cases considered, dose maxima are below the Swiss regulatory guideline; in many cases by several orders of magnitude, (ii) by the consistency of the proposed repository with the requirement that, at any time during a possible extended monitoring phase, it could be sealed within a few years, and that it does not rely for safety on any further measures after it has been sealed, and (iii) by the implementation of the concept of monitored long-term geological disposal.

- The use of alternative safety indicators includes, in addition to dose and risk, (i) radiotoxicity of the wastes as a function of time, which is compared with that of naturally occurring radionuclides, (ii) radiotoxicity fluxes due to radionuclides released from the repository in the course of time, which are compared with natural radiotoxicity fluxes in the surface environment, (iii) radiotoxicity concentrations originating from the repository at the top of the Opalinus Clay compared with natural radiotoxicity concentrations in Opalinus Clay, and (iv) the assessment of the distribution of radiotoxicity in the different system components as a function of time.

- Positive phenomena that are not included in the safety calculations due to limitations in the available tools (Reserve FEPs) are additional arguments for safety. Reserve FEPs that have been identified include (i) the co-precipitation of radionuclides with secondary minerals derived from spent fuel, glass and canister corrosion (except for co-precipitation of radium, which is included in all cases), (ii) sorption of radionuclides on canister corrosion products, (iii) natural concentrations of isotopes in solution in bentonite porewater, which could further reduce the effective solubilities of some radionuclides, (iv) irreversible sorption of radionuclides in the near field or in the geosphere (surface mineralisation), (v) long-term immobilisation processes in the geosphere (precipitation / co-precipitation), (vi) the delayed release of radionuclides, due to the slow corrosion rate of ILW metallic materials (e.g. hulls and ends), as well as a period of complete containment by ILW steel drums and emplacement containers, and (vii) the long resaturation time of the repository and its surroundings, which delays the commencement of corrosion and dissolution processes.

- Further reserves are available because of the simplified and conservative or pessimistic representation of the system in some of the assessment cases.
- Despite a detailed analysis of a wide range of assessment cases that were derived in a careful and methodical way, no outstanding issues with the potential to compromise safety have been identified.

Chapter 9 presents the **overall conclusions:**

1. Project *Entsorgungsnachweis* is a response to the request by the Swiss Government for a convincing demonstration of siting feasibility following the Government's review of the earlier Project Gewähr. The work described in the present report has shown that disposal is feasible from a safety point of view for the chosen system in the Opalinus Clay in the siting area in the Zürcher Weinland. Specifically the data and the analyses show that:

   - the reference site has properties that ensure sufficient safety. The safety case provides arguments that the repository is safe: there is sufficient safety for a broad spectrum of cases and the spectrum of analysed cases is broad enough to cover all reasonable possibilities;
   - the system is robust; i.e. remaining uncertainties do not put safety in question;
   - the information basis for the wastes and the engineered barrier system is adequate and draws on more than 20 years of work in Switzerland and wide experience abroad.

   In addition, as is extensively documented in the accompanying report on construction of facilities, the site properties and the design of the facility allow construction, operation and closure of the repository according to specifications, and thus to safety requirements.

   The information basis for the site is sufficient and the site is sufficiently well understood to support these statements on safety and engineering feasibility. The third report, the geosynthesis, illustrates that:

   - the geometry and structure of the host rock and confining units are well characterised with state-of-the-art 3 D seismics to identify a sufficiently large undisturbed area for allocation of the repository;
   - the host rock and confining units, which have been characterised with the deep borehole at Benken, have favourable properties that ensure long-term safety;
   - relevant processes have been investigated in detail in the underground rock laboratory at Mont Terri and in the laboratory and confirm and complement the findings from the borehole at Benken, thus the properties of the site and host rock and their future evolution can be bounded with confidence based on information from an extensive regional geological programme and on the fact that the overall situation of the site is reasonably simple.

2. Project *Entsorgungsnachweis* provides a platform for discussion and a foundation for decision-making on how to proceed with the Swiss HLW programme and to assess the role of the Opalinus Clay of the Zürcher Weinland in this programme. The excellent results obtained from the geological investigations and from the safety assessment in Project *Entsorgungsnachweis* have led Nagra to put forward a proposal, for consideration by the Swiss Government, to focus future work for the waste management option "Disposal of SF /

HLW / ILW in Switzerland" on the Opalinus Clay of the Zürcher Weinland[7]. This is justified by the facts that:

- a systematic screening of potential sedimentary host rocks[8] has indicated that Opalinus Clay has a number of particularly favourable properties, such as tightness, good retardation properties, a self-sealing capacity, good constructability, and good explorability;
- a systematic screening of situations[8] has indicated that the Zürcher Weinland has a number of favourable properties, such as a low tectonic activity, presence of Opalinus Clay at a suitable depth and with sufficient lateral extent, and the existence of confining units with similar properties as the host rock itself;
- Project *Entsorgungsnachweis* clearly indicates that, for a reference system in the Opalinus Clay of the Zürcher Weinland
  - a high level of safety can be expected,
  - construction, operation and closure of the repository is feasible,
  - the site has good qualities and offers sufficient flexibility.

The positive results obtained for this host rock and region does not imply that a safe system could not be implemented in other regions where Opalinus Clay is present, or in other host rocks. However, the technical arguments (based on safety, geological simplicity and predictability) that led to this region being preferred are considered to be plausible and well founded.

3. There are still many steps to be taken before a repository is definitively sited in Switzerland, underground explorations are carried through, final designs are agreed upon, and licensing activities are undertaken. A formal siting decision, which will be a milestone within the general licence process, is expected around the year 2020 at the earliest. This means that ample time is available to continue the investigations and to iterate on the repository design. Therefore, the level of detail documented in Project *Entsorgungsnachweis* is considered to be adequate for the current phase of the programme and sufficient to support the conclusions that disposal of SF / HLW / ILW in Switzerland is feasible (Statement 1 above) and that the choice of the Zürcher Weinland as the focus for future work, with the Opalinus Clay as host rock, is justified (Statement 2 above).

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7 Disposal abroad is an officially recognised option of the Swiss waste management strategy.

8 This step-wise appproach that lasted several years was done in close interaction, and in agreement, with the authorities and their advisory committees and is documented in several reports (see Chapter 1).

Zusammenfassung

Der vorliegende Bericht dokumentiert die Analyse und Beurteilung der Langzeitsicherheit eines geologischen Tiefenlagers[9] für abgebrannte Brennelemente (BE), hochaktive verglaste Abfälle (HAA) aus der Wiederaufarbeitung abgebrannter Brennelemente und langlebige mittelaktive Abfälle (LMA) im Opalinuston des potenziellen Standortgebiets im Zürcher Weinland in der Nordostschweiz. Diese Sicherheitsanalyse ist integraler Bestandteil der technischen Dokumentation für das Projekt *Entsorgungsnachweis*[10], die weiter eine Synthese der geowissenschaftlichen Untersuchungen und einen Bericht zur bautechnischen Machbarkeit umfasst. Das Projekt *Entsorgungsnachweis* ist ein wichtiger Meilenstein auf dem Weg zur Entsorgung der BE, HAA und LMA und beurteilt die grundsätzliche Machbarkeit der geologischen Tiefenlagerung dieser Abfälle in der Schweiz.

Die beiden Hauptziele des Projekts *Entsorgungsnachweis* sind:

1. Der Nachweis der grundsätzlichen Machbarkeit der geologischen Tiefenlagerung für BE, HAA und LMA im Opalinuston des potenziellen Standortgebiets im Zürcher Weinland. Damit sollen die 1988 durch den Bundesrat in seiner Beurteilung des *Projekts Gewähr 198*5 definierten Auflagen erfüllt werden. Dieser Nachweis soll aufzeigen, dass

   - ein genügend ausgedehntes Gebiet mit geeigneten geologischen Eigenschaften für die Aufnahme eines geologischen Tiefenlagers vorhanden ist (Ausdehnung eines geeigneten Wirtgesteinskörpers, *Standortnachweis*),
   - der Bau, Betrieb und Verschluss eines geologischen Tiefenlagers in diesem Gebiet technisch machbar ist (*Nachweis der bautechnischen Machbarkeit*) und
   - die Langzeitsicherheit eines solchen Lagers gewährleistet ist (*Nachweis der Langzeitsicherheit*).

2. Die Bereitstellung von Diskussions- und Entscheidungsgrundlagen zur Festlegung des weiteren Vorgehens im Programm zur Entsorgung der BE / HAA / LMA der Schweiz. Diese umfassen eine übersichtliche Darstellung der wichtigsten Aussagen und Resultate sowie eine Diskussion der zugrunde liegenden wissenschaftlichen Basis. Die bei den geowissenschaftlichen Untersuchungen erzielten ausgezeichneten Resultate und das durchgeführte systematische Auswahlverfahren veranlassten die Nagra, dem Bundesrat vorzuschlagen, künftige Untersuchungen im Hinblick auf die geologische Tiefenlagerung der BE / HAA / LMA in der Schweiz auf den Opalinuston und das potenzielle Standortgebiet im Zürcher Weinland[11] zu fokussieren. Daher ist ein weiteres Ziel, Grundlagen für einen solchen Entscheid bereitzustellen. Der Antrag zur zukünftigen Fokussierung stützt sich auf ein aufwändiges und systematisches Auswahlverfahren ab, das in Absprache und im Konsens mit den zuständigen Behörden und deren Fachexperten durchgeführt wurde. Das schrittweise Vorgehen bei der Eingrenzung (i) von sieben potenziellen sedimentären Wirtgesteinsoptionen auf eine (Opalinuston) und (ii) von zwei ausgedehnten Untersuchungsgebieten innerhalb des Opalinustons auf das potenzielle Standortgebiet im Zürcher Weinland ist in verschiedenen veröffentlichten Nagra-Berichten dokumentiert und in einem Bericht der Aufsichtsbehörde zusammengefasst (siehe Kap. 1 des vorliegenden Berichts). Nach erfolg-

[9] Der Begriff "geologisches Tiefenlager" wird im Entwurf des Kernenergiegesetzes definiert. Das geologische Tiefenlager wird nach dem Verschluss zum geologischen Endlager. Deshalb werden beide Begriffe verwendet.

[10] Der deutsche Ausdruck wird auch in der englischen Fassung des vorliegenden Berichts verwendet. Auf Englisch lautet dieser Begriff "demonstration of disposal feasibility."

[11] Die Entsorgung im Ausland ist ebenfalls eine offiziell anerkannte Option in der Schweizer Entsorgungsstrategie.

ter Beurteilung des Projekts *Entsorgungsnachweis* durch die Behörden sollen der detaillierte Inhalt und Zeitplan künftiger Untersuchungen festgelegt werden.

Zusätzliche Zielsetzungen des Projekts sind:

3. Die Bereitstellung von Grundlagen für die Planung der Entsorgung (inklusive Kostenschätzungen), für die Bewertung alternativer Auslegungsvarianten und für die Beurteilung verschiedener Abfallströme (BE, HAA und verschiedene Typen von LMA) sowie für die Planung der zukünftigen Forschungs- und Entwicklungsarbeiten.
4. Die Erarbeitung von Unterlagen für die Diskussion von Entsorgungsfragen mit allen Beteiligten und der Öffentlichkeit. Solche Diskussionen können zum besseren Verständnis der Entsorgung beitragen und so zu einer besseren Akzeptanz der geologischen Tiefenlagerung führen.

Die spezifischen Ziele der Sicherheitsanalyse (vorliegender Bericht) sind:

1. Beurteilung der Eignung des Opalinustons im Zürcher Weinland als mögliches Wirtgestein für ein Tiefenlager aus Sicht der Langzeitsicherheit.
2. Verbesserung des Kenntnisstands bzgl. der vielfältigen Sicherheitsfunktionen des betrachteten Lagersystems.
3. Bewertung der Robustheit des Lagersystems bzgl. der verbleibenden Ungewissheiten und der Auswirkung von Phänomenen, welche die Barrierenwirkung des Lagers beeinträchtigen könnten.
4. Bereitstellung einer Diskussionsgrundlage für eine Vielzahl von mit der Entsorgung zusammenhängenden Themen. Im Einzelnen bilden die in der Sicherheitsanalyse gemachten Aussagen zusammen mit den Stellungnahmen der zuständigen Behörden eine wichtige Grundlage im Hinblick auf die künftige Planung der Entsorgung.

Die Sicherheit während der Betriebsphase ist nicht Gegenstand dieses Berichts; sie wird qualitativ im Bericht über die bautechnische Machbarkeit abgehandelt.

Der vorliegende Bericht enthält die Argumente und Analysen, welche den Nachweis der Langzeitsicherheit eines Tiefenlagers für BE, HAA und LMA im Opalinuston des potenziellen Standortgebiets im Zürcher Weinland bilden. Der Begriff "Sicherheitsnachweis" wird nachfolgend umschrieben:

Sicherheitsnachweis

Der Sicherheitsnachweis besteht aus einer Reihe von Argumenten und Analysen, welche die Schlussfolgerung begründen, dass ein spezifisches Lagersystem sicher sein wird. Dazu gehört insbesondere der Nachweis, dass die behördlichen Schutzziele eingehalten werden. Der Sicherheitsnachweis und die dazu gehörenden Hintergrunddokumente beschreiben die Auslegung des Systems und seine Sicherheitsfunktionen und zeigen die Wirksamkeit der verschiedenen Barrieren und des Gesamtsystems auf. Die aufgeführten Argumente und Analysen werden begründet, und die Bedeutung von Ungewissheiten und offenen Fragen im Hinblick auf das weitere Vorgehen bei der Realisierung der Tiefenlagerung wird diskutiert.

Im Folgenden werden die wichtigsten Aspekte kurz vorgestellt.

Kapitel 1 hat als Einführungskapitel zum Ziel, den **vorliegenden Bericht in den nationalen Kontext**[12] zu stellen. Aus den fünf in der Schweiz in Betrieb stehenden Kernkraftwerken (das erste wurde 1969 in Betrieb genommen) und aus Medizin, Industrie und Forschung resultieren radioaktive Abfälle, deren zuverlässige Entsorgung sicherzustellen ist. Dazu wurde ein detailliertes Entsorgungskonzept entwickelt: Kürzlich wurde das zentrale Zwischenlager in Würenlingen in Betrieb genommen, womit heute eine genügende Kapazität für sämtliche aus der Schweiz stammenden radioaktiven Abfälle zu Verfügung steht, bis diese in ein geologisches Tiefenlager überführt werden können. Für die geologische Tiefenlagerung wurden zwei verschiedene Lager konzipiert (ein Lager für schwach- und mittelaktive Abfälle und ein weiteres für BE, HAA und LMA).

Die wichtigsten gesetzlichen Anforderungen an die Entsorgung lauten: (i) In der Schweiz anfallende radioaktive Abfälle sollen grundsätzlich in der Schweiz entsorgt werden (wobei die Gesetzgebung auch die Bedingungen definiert, unter denen für den Ausnahmefall einer Entsorgung im Ausland Exportbewilligungen erteilt werden) und (ii) in der Schweiz sind die Verursacher radioaktiver Abfälle gesetzlich zu deren sicheren Entsorgung verpflichtet. Um ihrer Verantwortung nachzukommen, gründeten die Elektrizitätsgesellschaften als Betreiber der Kernkraftwerke sowie die Schweizerische Eidgenossenschaft, die für die Entsorgung der aus der Medizin, Industrie und Forschung stammenden Abfälle verantwortlich ist, 1972 die Nationale Genossenschaft für die Lagerung radioaktiver Abfälle (Nagra). Die Nagra ist verantwortlich für die Forschungs- und Projektierungsarbeiten im Zusammenhang mit der geologischen Tiefenlagerung. Weitere Aufgaben bei der Entsorgung der Abfälle, wie die Konditionierung, Zwischenlagerung sowie der Bau und Betrieb der verschiedenen benötigten Anlagen, liegen in der Verantwortung der einzelnen Verursacher oder Organisationen, die von den Verursachern hierzu gebildet werden. Die Hauptabteilung für die Sicherheit der Kernanlagen (HSK) ist die Aufsichtsbehörde des Bundes. Die Eidgenössische Kommission für die Sicherheit von Kernanlagen (KSA) ist für die Beurteilung von Projekten zu Kernanlagen, einschliesslich der Lager für radioaktive Abfälle, verantwortlich. Sie erstellt auch Stellungnahmen zu Anträgen für die Erteilung von nuklearen Bewilligungen und zu Gutachten der HSK. In der Richtlinie R-21 der HSK und der KSA werden die Schutzziele für die Endlagerung radioaktiver Abfälle festgelegt. Ein wichtiges Ziel für jede Sicherheitsanalyse ist die Beurteilung, ob diese Schutzziele eingehalten werden können. In Kapitel 1 wird schliesslich das schrittweise Vorgehen zur Realisierung eines geologischen Tiefenlagers vorgestellt; dies schliesst eine kurze Diskussion der wichtigsten vergangenen und zukünftigen Schritte mit ein.

Kapitel 2 behandelt die **Vorgaben und Anforderungen für die Wahl und Auslegung des Lagersystems und für die Durchführung der Analysen der Langzeitsicherheit und die Bewertung deren Resultate.** Dazu werden die Anforderungen und Hinweise internationaler Organisationen und der Schweizer Gesetzgebung und behördlichen Richtlinien diskutiert. Dazu gehören insbesondere der Entwurf des neuen Kernenergiegesetzes (KEG)[13] und die Richtlinie HSK-R-21. Das KEG legt die Entsorgung in einem geologischen Tiefenlager fest, das vor seinem endgültigen Verschluss für eine gewisse Zeit überwacht und kontrolliert werden kann. Diese Vorgabe basiert auf dem Konzept der "Kontrollierten Geologischen Langzeitlagerung", wie es von der vom Bund eingesetzten Expertenkommission EKRA[14] vorgeschlagen wurde. Die

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[12] Status: 31. August 2002. Die Auswirkungen des negativen Ausgangs der Abstimmung über die Erteilung einer Konzession für einen Sondierstollen des vorgeschlagenen Lagers für die Entsorgung von SMA am Wellenberg werden im vorliegenden Bericht nicht diskutiert.

[13] Abkürzung für "Kernenergiegesetz" (in der englischen Version Nuclear Energy Law).

[14] Abkürzung für "Expertengruppe Entsorgungskonzepte für radioaktive Abfälle" , vom Bundesrat im Juni 1999 eingesetzt.

in den verschiedenen Dokumenten enthaltenen Vorgaben und Hinweise werden ergänzt durch zusätzliche Überlegungen, die auf der Erfahrung der Nagra sowohl durch ihre Arbeiten in der Schweiz als auch durch ihre Zusammenarbeit mit internationalen Organisationen und Partnerorganisationen beruhen. Dazu gehört auch die Diskussion des Konzepts der Robustheit des Lagersystems und das Vorgehen bei der Sicherheitsanalyse. Weiter werden im Hinblick auf die Sicherheitsanalyse die Rolle und Behandlung der Biosphäre, die Behandlung von künftigen menschlichen Aktivitäten und der zu betrachtende Zeithorizont diskutiert. Die Informationen in diesem Kapitel stammen aus verschiedenen Quellen und sind relativ heterogen. Deshalb werden die Schlüsselaussagen wie folgt gruppiert: (i) Ziele der geologischen Tiefenlagerung, (ii) Vorgaben für die Wahl eines geeigneten Lagersystems (einschliesslich Diskussion der *Sicherheitsfunktionen* des Lagersystems), (iii) Ziele und Vorgaben für eine schrittweise Realisierung, (iv) Anforderungen an die Sicherheitsanalyse. Aufgrund ihrer zentralen Bedeutung werden die Sicherheitsfunktionen nachfolgend aufgelistet und beschrieben.

Sicherheitsfunktionen

Das Lagersystem erfüllt verschiedene Funktionen, die für die Langzeitsicherheit relevant sind. Diese werden als Sicherheitsfunktionen bezeichnet und umfassen:

- Isolation vom menschlichen Lebensraum – Die Sicherheit für Mensch und Umwelt wird durch den Einschluss der radioaktiven Abfälle und des spaltbaren Materials in einem geologischen Tiefenlager gewährleistet. Nach Verfüllung und Versiegelung der Zugänge ist die Wahrscheinlichkeit eines unerwünschten menschlichen Eindringens und einer unerlaubten Verwendung des spaltbaren Materials sehr klein. Die Wahrscheinlichkeit des unabsichtlichen menschlichen Eindringens (z.B. Anbohren) reduziert sich durch die Wahl eines Lagergebietes, in dem aus heutiger Sicht keine wirtschaftlich nutzbaren Ressourcen vorhanden sind und keine Nutzungskonflikte bestehen. Schliesslich gewährleistet eine sorgfältige Standortwahl, dass keine für die Langzeitstabilität ungünstigen geologischen Ereignisse und ungünstig verlaufenden Prozesse zu erwarten sind.
- Langzeiteinschluss und radioaktiver Zerfall innerhalb des Lagersystems – Ein Grossteil der anfänglich vorhandenen Radioaktivität wird durch radioaktiven Zerfall abgebaut, während die Abfälle noch vollständig in den Abfallbehältern eingeschlossen sind. Dies gilt vor allem für BE und HAA, deren Stahlbehälter während mindestens 10'000 Jahren einen vollständigen Einschluss gewährleisten. Selbst nach späterem Versagen der Behälter sorgen die Langzeitstabilität der BE- und HAA-Abfallmatrizen, die äusserst geringe Grundwasserbewegung und eine Reihe geochemischer Immobilisierungs- und Rückhalteprozesse dafür, dass die Radionuklide weitgehend innerhalb des Systems der Sicherheitsbarrieren und des unmittelbar umgebenden Gesteins eingeschlossen bleiben und dort zerfallen.
- Verminderung der Freisetzung in die Umwelt – Obwohl ein vollständiger Einschluss nicht über die gesamte relevante Zeit und für sämtliche Radionuklide aufrechterhalten werden kann, sind die Freisetzungsraten von Radionukliden aus den Abfällen gering. Dies gilt insbesondere für die BE und HAA mit ihren stabilen Abfallmatrizen. Zudem verzögert bzw. vermindert eine Reihe von Prozessen die Freisetzung und den Transport der Radionuklide und begrenzt damit die Radionuklidkonzentration in der Umwelt. Dazu gehört auch der radioaktive Zerfall während des langsamen Transports durch die technischen und geologischen Barrieren.

Kapitel 3 beschreibt **die für die Sicherheitsanalyse verwendete Methode**, welche die in Kapitel 2 aufgelisteten Anforderungen berücksichtigt. Schlüsselaspekte sind: (i) die Beschreibung der einzelnen Schritte der Sicherheitsanalyse und (ii) die Festlegung der wichtigsten Themen, die für den Nachweis der Sicherheit zu behandeln sind. Die verschiedenen Aspekte werden nachstehend näher ausgeführt.

(i) Der Sicherheitsnachweis umfasst:

- Die Auswahl bzw. Festlegung des Lagersystems unter Berücksichtigung einer flexiblen Realisierungsstrategie. Dabei werden die Ergebnisse früherer Studien, inkl. Studien zur Langzeitsicherheit, berücksichtigt.
- Die Beschreibung des Systems (das sogenannte Systemkonzept) gemäss dem gegenwärtigen Kenntnisstand der Eigenschaften, Ereignisse und Prozesse ("features, events and processes", sog. FEPs), die das Lagersystem und dessen Entwicklung charakterisieren und möglicherweise beeinflussen können.
- Die Ableitung des Sicherheitskonzepts, das sich auf gut verstandene und wirksame Prozesse abstützt (die sogenannten Pfeiler der Sicherheit).
- Das Aufzeigen der möglichen radiologischen Konsequenzen des Lagersystems durch Auswahl und Analyse einer grossen Zahl und eines breiten, abdeckenden Spektrums von Rechenfällen.
- Die Zusammenstellung der wissenschaftlich belegten Argumente und Analysen, die den Sicherheitsnachweis bilden, sowie Hinweise für die künftigen Phasen des Entsorgungsprogramms.

(ii) Die Argumente, die zum Sicherheitsnachweis beitragen, betreffen :

- Die Eignung der geologischen Tiefenlagerung als Entsorgungsoption,
- die Sicherheit und Robustheit des gewählten Lagersystems,
- die geringe Wahrscheinlichkeit und die beschränkten Folgen eines menschlichen Eindringens,
- die Vorteile der schrittweisen Realisierung eines solchen Lagers,
- die für das gewählte Lagersystem und dessen Entwicklung vorhandenen guten wissenschaftlichen Kenntnisse,
- die Eignung der zur Verfügung stehenden Methode und der Modelle und Datensätze zur Beurteilung der Langzeitsicherheit und
- die breit abgestützte Begründung der Sicherheit, d.h. das Einhalten der behördlichen Schutzziele, die Resultate aus der Verwendung alternativer Sicherheitsindikatoren, das Bestehen von Sicherheitsreserven (Reserve-FEPs) und das Fehlen von offenen Fragen, welche die Sicherheit grundsätzlich in Frage stellen könnten.

Kapitel 4 dokumentiert **den gegenwärtigen Kenntnisstand und die Charakterisierung des potenziellen Lagersystems im Opalinuston des Zürcher Weinlands direkt nach Verschluss des Tiefenlagers.** Die Wahl des potenziellen Standorts und die Auslegung des Lagersystems erfolgt entsprechend den Zielsetzungen, Anforderungen und Hinweisen, wie sie in Kapitel 2 vorgestellt wurden. Dabei werden folgende Elemente diskutiert: (i) Schlüsselmerkmale des Standorts, (ii) Schlüsseleigenschaften des Wirtgesteins Opalinuston, (iii) die Auslegung des Lagers, (iv) die einzulagernden Abfallmengen und die Abfalleigenschaften (BE, HAA und

LMA), (v) das System der technischen Barrieren und deren Auslegung. Diese Elemente werden nachstehend detaillierter beschrieben:

(i) Schlüsselmerkmale des Standorts

- Die geologische Umgebung ist einfach; die strukturellen, hydrogeologischen und geochemischen Eigenschaften sind gut vorhersagbar.
- Das potenzielle Standortgebiet ist tektonisch stabil mit einer geringen Hebungs- und dadurch bedingten Erosionsrate.
- Sowohl das Kristallin als auch die überlagernden Sedimente in dieser Region haben kein bedeutendes Potential an natürlichen Rohstoffen.

(ii) Schlüsseleigenschaften des Wirtgesteins Opalinuston

- Der Opalinuston weist eine derart niedrige hydraulische Durchlässigkeit auf, dass der Transport gelöster Stoffe durch die Formation hauptsächlich durch Diffusion und nur untergeordnet durch Advektion erfolgt.
- Die geochemischen Bedingungen im Opalinuston sind reduzierend, leicht alkalisch und mässig salin und begünstigen sowohl die langfristige Wirksamkeit der technischen Barrieren als auch die Radionuklidrückhaltung.
- Es wird erwartet, dass die geochemischen Bedingungen im Opalinuston und in den Rahmengesteinen über den Zeitraum von einigen Millionen Jahre stabil bleiben[15].
- Die Eigenschaften des Opalinustons, insbesondere sein ausgeprägtes Selbstabdichtungsvermögen, gewährleisten, dass die durch den Lagerbau induzierten Klüfte (Auflockerungszone) sowie natürliche Störungszonen und Klüfte hydraulisch sehr gering durchlässig sind, d.h. deren Auswirkung auf die hydraulischen Eigenschaften des Wirtgesteins vernachlässigbar sein wird.
- Beim Opalinuston handelt es sich um ein verfestigtes Tonsediment mit günstigen bautechnischen Eigenschaften, die den Bau von Stollen mit kleinen Durchmessern ohne Ausbau und von grösseren Tunneln (mit Ausbau) in einer Tiefe von mehreren Hundert Metern ermöglichen.

(iii) Schlüsselmerkmale der Lagerauslegung sind:

- eine Zugangsrampe, Bau- und Betriebstunnel, zentrale Empfangsanlagen und ein Bau-/ Ventilationsschacht,
- die Anordnung paralleler subhorizontaler Lagerstollen für BE / HAA mit einem Durchmesser von 2.5 m und einem Stollenabstand von 40 m,
- drei kurze horizontale Lagertunnel für LMA einige hundert Meter entfernt vom BE / HAA-Bereich des Lagers und
- Pilot- und Testlager entsprechend dem EKRA-Konzept.

(iv) Schlüsselaspekte bezüglich der Abfälle

- Die angenommenen Abfallmengen basieren auf der Annahme einer 60-jährigen Betriebsdauer der fünf gegenwärtig in Betrieb stehenden Kernkraftwerke; dies entspricht einer Stromproduktion von insgesamt 192 GWa(e).

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15 In diesem Zeitraum wird lediglich eine geringfügige Salinitätsabnahme erwartet.

- Die Menge an BE und HAA/LMA basiert auf der Annahme, dass lediglich diejenige Menge an BE aufgearbeitet wird, für die heute Wiederaufarbeitungsverträge bestehen (ca. 1195 t_{IHM}).
- Dies führt zu ca. 3200 t_{IHM} BE bzw. 2065 BE-Behältern, 730 HAA-Behältern und ca. 7300 m^3 (konditionierten) LMA.

(v) Schlüsselmerkmale des Systems der technischen Barrieren sind:

- der BE-Behälter – beim Referenzkonzept handelt es sich um einen Stahlbehälter mit entweder 4 DWR- oder 9 SWR-Brennelementen, mit einer Mindestwandstärke von 15 cm, einer Länge von ca. 5 m und einem Durchmesser von ca. 1 m,
- der HAA-Behälter – das Referenzkonzept ist dasselbe, wie in der Projektstudie *Gewähr* dokumentiert, d.h. ein Stahlbehälter, der eine Edelstahlkokille mit verglastem HAA enthält, mit einer Wandstärke von 25 cm, einer Länge von ca. 2 m und einem Durchmesser von ca. 1 m,
- die LMA-Behälter – die LMA-Gebinde werden in Lagercontainer aus Beton verpackt, und die Zwischenräume zwischen den Gebinden werden mit Zementmörtel verfüllt,
- die Verfüllung der BE / HAA-Lagerstollen – die BE / HAA-Behälter werden auf kompaktierten Bentonitblöcken entlang der Stollenlängsachse platziert, mit einem Abstand von 3 m zwischen den Behältern; der Raum um die Behälter wird mit Bentonitgranulat verfüllt,
- die Verfüllung der LMA-Kavernen – nach Einlagerung der LMA-Lagercontainer in die LMA-Lagertunnel werden die Hohlräume zwischen den Lagercontainern mit Zementmörtel verfüllt und
- die Verfüllung und Versiegelung (Bau- und Betriebstunnel, Zentralbereich, Rampe) – die Bau- und Betriebstunnel, der zentrale Bereich und die Rampe (im Bereich des Wirtgesteins) werden mit einem Bentonit–Sand-Gemisch verfüllt, mit an mehreren Orten vorgesehenen Versiegelungsstrecken aus hochkompaktiertem Bentonit; auch der Schacht wird mit hochkompaktiertem Bentonit verfüllt.

Kapitel 5 enthält eine **Beschreibung der zeitlichen Entwicklung des Lagersystems** nach Verschluss des Lagers. Dabei werden die Wechselwirkungen zwischen den einzelnen Systemkomponenten berücksichtigt, und die Entwicklung des Standorts und der technischen Barrieren diskutiert. Neben der erwarteten Entwicklung des Lagersystems werden auch mögliche Abweichungen davon diskutiert. Dies bildet die Basis für die Ableitung einer Reihe von zu betrachtenden Fällen, die in dem vorliegenden Bericht im Detail quantitativ analysiert werden.

Kapitel 6 analysiert **die relative Bedeutung der verschiedenen Eigenschaften und Prozesse** des Lagersystems und **identifiziert die quantitativ zu analysierenden Fälle.** Dazu werden sowohl deterministische als auch probabilistische Sensitivitätsanalysen durchgeführt und die Effekte möglicher Abweichungen von der erwarteten Entwicklung untersucht, ausgehend von den *Sicherheitsfunktionen*, wie sie in Kapitel 2 definiert wurden. Anhand einer qualitativen Diskussion und dem aus den quantitativen Analysen gewonnenen Verständnis werden die Schlüsselmerkmale und -phänomene identifiziert, die zu den Sicherheitsfunktionen beitragen; diese werden als *Pfeiler der Sicherheit* bezeichnet. Aufgrund der wichtigen Rolle der Pfeiler der Sicherheit für den Sicherheitsnachweis werden sie nachstehend aufgelistet und definiert:

Pfeiler der Sicherheit

Die Pfeiler der Sicherheit sind Eigenschaften des Lagersystems, die eine Schlüsselrolle für die Gewährleitung der *Sicherheitsfunktionen* übernehmen:

- **Die Platzierung des Lagers im tiefen Untergrund**, in einem Umfeld, das menschliches Eindringen sehr unwahrscheinlich macht und das keinen die Langzeitstabilität gefährdenden geologischen Ereignissen und ungünstigen Prozessen ausgesetzt ist.
- **Das Wirtgestein,** das eine sehr geringe hydraulische Durchlässigkeit, eine homogene Porenstruktur und ein Selbstabdichtungsvermögen aufweist und somit eine wirkungsvolle Barriere gegenüber dem Radionuklid-Transport darstellt und eine geeignete Umgebung für das System der technischen Barrieren bildet.
- **Ein chemisches Umfeld,** das für eine Reihe von geochemischen Immobilisierungs- und Rückhaltungsprozessen günstig ist, das die Langzeitstabilität der technischen Barrieren begünstigt, und das seinerseits aufgrund einer Reihe von chemischen Pufferreaktionen langfristig stabil ist.
- **Die Bentonitverfüllung (für BE und HAA)** als gut definiertes Interface zwischen den Abfallbehältern und dem Wirtgestein, mit ähnlichen Eigenschaften wie das Wirtgestein. Die Bentonitverfüllung gewährleistet, dass die Auswirkungen der Lagerstollen und des wärmeproduzierenden Abfalls auf das Wirtgestein minimal sind. Sie bildet eine wirkungsvolle Transportbarriere für Radionuklide sowie eine geeignete Umgebung zur Gewährleistung eines günstigen Langzeitverhaltens der Behälter und der Abfallmatrizen.
- **Die BE- und HAA-Abfallmatrizen,** die unter den erwarteten Bedingungen sehr stabil sind.
- **Die BE- und HAA-Behälter,** die unter den erwarteten Bedingungen mechanisch stabil und korrosionsresistent sind und die für einen beträchtlichen Zeitraum einen absoluten Einschluss der Abfälle gewährleisten.

Um die Rechenfälle für die quantitative Sicherheitsanalyse zu identifizieren, wird angenommen, dass die Pfeiler der Sicherheit wie erwartet funktionieren, dass aber auch Abweichungen möglich sind (basierend auf den Ausführungen in Kapitel 4 und 5 und aufgrund der Erkenntnisse aus den Sensitivitätsanalysen).

Kapitel 7 erläutert **die Ergebnisse der quantitativen Analyse der in Kapitel 6 identifizierten Fälle**. Zuerst werden die betrachteten Fälle konzeptuell beschrieben. Dabei werden für ein bestimmtes Szenarium verschiedene Konzeptualisierungen betrachtet, und für eine bestimmte Konzeptualisierung werden Ungewissheiten in den Daten durch Parametervariationen berücksichtigt. Ausgangspunkt der Analysen ist das Referenzszenarium, welches ein Lagersystem annimmt, das sich entsprechend den Erwartungen verhält. Die Auswirkungen von Ungewissheiten im Verhalten und in der zukünftigen Entwicklung des Systems werden anhand von alternativen Szenarien untersucht.

Um die Robustheit des Lagersystems zu testen, wurde eine Reihe von *"what if?"*-Fällen untersucht, wobei Phänomene unterstellt bzw. Parameterwerte angenommen werden, die ausserhalb des Bereichs liegen, der aufgrund wissenschaftlicher Ergebnisse als möglich erachtet wird. Hier geht es also nicht darum, real denkbare Fälle zu analysieren, sondern einen vertieften Einblick in die Qualität des Lagersystems zu erhalten. Um die Zahl der *"what if?"*-Fälle einzuschränken, werden nur solche betrachtet, welche die Schlüsseleigenschaften der Pfeiler der

Sicherheit beeinträchtigen würden. Die Liste der *"what if?"*-Fälle ist nicht umfassend; trotzdem illustriert sie, dass das System selbst unter Extrembedingungen noch sicher ist.

Verschiedene Auslegungs- und Systemoptionen für den Bau des Lagers werden separat evaluiert. Die Sensitivität des Verhaltens von Radionukliden in der Biosphäre und die resultierenden Dosen werden durch eine Anzahl von Fällen untersucht, die sich aus alternativen geomorphologischen und klimatischen Bedingungen ableiten lassen. Das Spektrum möglicher Dosen wird anhand verschiedener (modellhafter) zukünftig denkbarer Eigenschaften des Lebensraums an der Erdoberfläche illustriert.

Kapitel 8 umfasst die **Synthese der wichtigsten Argumente und Resultate der Sicherheitsanalyse** und bildet den letzten Schritt des Sicherheitsnachweises. Alle Aspekte der in Kapitel 3 als wichtig identifizierten Argumente werden rekapituliert und die Feststellungen im Einzelnen zusammengefasst:

- Die Eignung der geologischen Tiefenlagerung als Entsorgungsoption wird begründet (i) durch die weltweit anerkannte Auffassung, dass ein sorgfältig gewähltes Lagersystem an einem günstigen Standort jetzt und in Zukunft Sicherheit und Schutz von Menschen und Umwelt gewährleistet und auch einen guten Schutz gegen unbefugte menschliche Eingriffe bietet, (ii) durch in der Schweiz und anderswo vorhandene geeignete Gesteinsformationen, (iii) durch die Resultate vieler weltweit durchgeführter Sicherheitsanalysen, (iv) durch Beobachtungen an natürlichen Systemen (natürliche Analoga) und (v) durch die Vorteile der geologischen Tiefenlagerung gegenüber anderen Optionen.

- Die Sicherheit und Robustheit des Lagersystems wird gewährleistet (i) durch eine Reihe wirksamer passiver Barrieren, welche mit verschiedenen Phänomenen zu Einschluss und Rückhaltung der Radionuklide beitragen, (ii) durch die Vermeidung von Ungewissheiten und ungünstigen Phänomenen mit der Wahl eines geeigneten Standorts und einer geeigneten Lagerauslegung und (iii) durch die Langzeitstabilität des Lagersystems in einer günstigen geologischen Situation.

- Die kleine Wahrscheinlichkeit und die mässigen Auswirkungen eines menschlichen Eindringens sind bedingt (i) durch die Archivierung von Informationen über das Tiefenlager, (ii) durch die Vermeidung von möglichen Rohstoffkonflikten durch die Standortwahl (d.h. das Fehlen nutzbarer natürlicher Ressourcen im vorgeschlagenen Lagergebiet) und (iii) durch die Bildung von Kompartimenten für die einzelnen BE-/HAA-Behälter (vollständige Umhüllung der Behälter mit Bentonit) und die Verfestigung der Abfälle.

- Die Realisierung eines geologischen Tiefenlagers erfolgt schrittweise, d.h. (i) dass in der gegenwärtigen Phase das System noch nicht in allen Details festgelegt werden muss (und deshalb noch nicht alle Fragen im Detail beantwortet sein müssen); (ii) dass sich aber der vorliegende Sicherheitsnachweis auf bereits heute gut verstandene und zuverlässig charakterisierte Komponenten (Standort, System der technischen Barrieren) abstützen kann, (iii) dass die Beteiligten einbezogen werden können und damit die Möglichkeit besteht, Anregungen und Verbesserungsvorschläge Dritter zu berücksichtigen, (iv) dass das Projekt die Flexibilität hat, neue Resultate und Wünsche zu berücksichtigen (z.B. bezüglich der detaillierten Anordnung der Lagerstollen, der Wahl zwischen verschiedenen Auslegungsoptionen, der Platzierung der Aussenanlagen und sogar bezüglich der Standortwahl; d.h. es existieren alternative Standortmöglichkeiten sowohl für die Wirtgesteinsoption Opalinuston als auch für andere Wirtgesteinsoptionen) und (v) dass Möglichkeiten zur Überwachung bestehen und gegebenenfalls Entscheidungen rückgängig gemacht werden können (im Extremfall die Rückholung der Abfälle).

- Der für das gewählte Lagersystem gute wissenschaftliche Kenntnisstand ist begründet (i) durch die guten und belastbaren Ergebnisse der regionalen und lokalen Felduntersuchungen, die als wichtigste Elemente die 3 D-Seismik und die Sondierbohrung Benken im potenziellen Standortgebiet sowie die Informationen aus weiteren Bohrungen in der näheren und weiteren Umgebung beinhalten; durch die Experimente im Felslabor Mont Terri und im Labor sowie durch Beobachtungen am Opalinuston in einer Reihe von Eisenbahn- und Strassentunneln, (ii) durch die Resultate und Erfahrungen aus mehr als 20 Jahren Entwicklung und Charakterisierung von Komponenten der technischen Barrieren in der Schweiz sowie durch die zur Verfügung stehende weltweite Informationsbasis und (iii) durch das vorhandene detaillierte modellhafte Inventar für BE, HAA und LMA.
- Methodik, Modelle, Rechencodes und Datensätze für die Analyse eines breiten Spektrums von Fällen haben sich bewährt und erfüllen die Anforderungen, die in Kapitel 2 vorgegeben wurden.
- Die behördlichen Vorgaben werden erfüllt: (i) in allen betrachteten Fällen liegen die Dosismaxima unterhalb des festgelegten Schutzzieles, in den meisten Fällen um einige Grössenordnungen; (ii) das vorgeschlagene Tiefenlager kann während der möglicherweise länger dauernden Beobachtungs- und Überwachungsphase jederzeit innerhalb weniger Jahre verschlossen werden, und die Sicherheit hängt nach Verschluss des Lagers nicht von weiteren Massnahmen ab und (iii) mit der vorgeschlagenen Lagerauslegung wurde das EKRA-Konzept der "Kontrollierten Geologischen Langzeitlagerung" umgesetzt.
- Die Verwendung alternativer Sicherheitsindikatoren umfasst zusätzlich zu Dosis- und Risikoabschätzung (i) einen Vergleich der Radiotoxizität der Abfälle als Funktion der Zeit mit der Radiotoxizität natürlicher Materialien, (ii) einen Vergleich potentieller Radiotoxizitätsflüsse aus dem Tiefenlager mit natürlichen Radiotoxizitätsflüssen in der Umwelt, (iii) einen Vergleich potentieller Radiotoxizitätskonzentrationen am oberen Rand des Opalinustons (von aus dem Tiefenlager stammenden Radionukliden) mit der natürlichen Radiotoxizitätskonzentration im Opalinuston und (iv) die Evaluation der Verteilung der Radiotoxizität in den verschiedenen Systemkomponenten als Funktion der Zeit.
- Einige positiv zur Sicherheit beitragende Phänomene (sog. Reserve-FEPs) wurden in den durchgeführten Analysen nicht berücksichtigt, da zur Zeit keine geeigneten Modelle zur Verfügung standen um sie quantitativ zu analysieren. Sie sind zusätzliche Argumente für die Sicherheit. Es sind dies: (i) die Mitfällung von Radionukliden mit sekundären Korrosionsprodukten der BE, HAA-Gläser und der Behälter (mit Ausnahme der Mitfällung von Radium, die in allen betrachteten Fällen enthalten ist), (ii) die Sorption von Radionukliden an Korrosionsprodukten der Behälter, (iii) die Konzentrationen natürlicher Isotope in der Porenwasserlösung von Bentonit, welche die effektive Löslichkeit einiger Radionuklide weiter vermindern könnten, (iv) die irreversible Sorption von Radionukliden im Nahfeld oder in der Geosphäre (Oberflächenmineralisation), (v) die Langzeit-Immobilisierungsprozesse in der Geosphäre (Ausfällung / Mitfällung), (vi) die verzögerte Freisetzung von Radionukliden aufgrund der niedrigen Korrosionsrate von metallischem LMA-Material (z.B. Hülsen und Endstücke) sowie eine Zeitspanne vollständigen Einschlusses durch LMA-Gebinde und -Lagerbehälter und (vii) die lange Wiederaufsättigungszeit des Lagers und seiner Umgebung, die den Beginn von Korrosions- und Auflösungsprozessen verzögert.
- Durch die vereinfachte konservative oder pessimistische Darstellung des Systems bei einigen der analysierten Fälle werden die berechneten Dosen überschätzt. Damit sind in diesen Fällen weitere Sicherheitsreserven vorhanden.
- Trotz der detaillierten Analyse eines grossen Spektrums von Fällen, welches sorgfältig und systematisch abgeleitet wurde, wurden keine ungeklärten Fragen identifiziert, welche die Sicherheit grundsätzlich in Frage stellen könnten.

Kapitel 9 beinhaltet die **Schlussfolgerungen:**

1. Das Projekt *Entsorgungsnachweis* erfüllt den Auftrag des schweizerischen Bundesrats für einen überzeugenden Standortnachweis, welchen er in seinem Entscheid zum damaligen Projekt Gewähr gegeben hat. Die im vorliegendem Bericht dokumentierten Arbeiten zeigen, dass das gewählte Lagersystem im Opalinuston im potenziellen Standortgebiet des Zürcher Weinlands langfristige Sicherheit bietet. Die Resultate und Analysen belegen nachvollziehbar, dass:

   - der Referenzstandort Eigenschaften aufweist, welche die Sicherheit gewährleisten: Für ein breites Spektrum von Rechenfällen ist die Sicherheit gegeben und das Spektrum der analysierten Fälle wurde genügend breit gewählt, um alle realistischerweise anzunehmenden Möglichkeiten abzudecken;
   - das System robust ist, d.h. dass die verbleibenden Ungewissheiten die Sicherheit nicht in Frage stellen können;
   - die Informationsbasis bezüglich der Abfälle und dem System der technischen Barrieren gut ist. Sie stützt sich auf mehr als 20 Jahre wissenschaftliche Untersuchungen in der Schweiz und grosse internationale Erfahrung.

   Wie im Bericht zur bautechnischen Machbarkeit ausführlich dokumentiert wurde, erlauben die Standorteigenschaften und die Auslegung der Anlage den zuverlässigen und sicheren Bau, Betrieb und Verschluss des Lagers entsprechend den Vorgaben.

   Die Synthese der geowissenschaftlichen Untersuchungen zeigt, dass:

   - die Geometrie und Struktur des Wirtgesteins und der Rahmengesteine durch die 3 D-Seismik gut und zuverlässig charakterisiert sind und dass ein ausreichend grosses ungestörtes Gebiet für die Erstellung eines Tiefenlagers vorhanden ist;
   - das Wirtgestein und die Rahmengesteine günstige Eigenschaften aufweisen, welche die Langzeitsicherheit gewährleisten;
   - die relevanten Prozesse im Detail im Felslabor Mont Terri und im Labor untersucht werden konnten und die Ergebnisse aus der Sondierbohrung Benken damit im Einklang stehen. Die Eigenschaften des Standorts und des Wirtgesteins sind gut bekannt. Die zukünftige Entwicklung kann aufgrund der Resultate des umfangreichen regionalen geowissenschaftlichen Untersuchungsprogramms genügend genau eingegrenzt werden, nicht zuletzt da der geologische Aufbau des potenziellen Standortgebiets vergleichsweise einfach ist.

2. Das Projekt *Entsorgungsnachweis* liefert wichtige Diskussions- und Entscheidungsgrundlagen für die Festlegung des zukünftigen Vorgehens im Programm zur Entsorgung der BE / HAA / LMA und für die Bewertung des potenziellen Wirtgesteins Opalinuston im Zürcher Weinland. Gestützt auf die ausgezeichneten Ergebnisse der geowissenschaftlichen Untersuchungen und die Resultate der Sicherheitsanalysen schlägt die Nagra dem Bundesrat vor, die zukünftigen Arbeiten für die Option "Entsorgung von BE / HAA / LMA in der Schweiz" auf den Opalinuston im Zürcher Weinland[16] zu fokussieren. Dies wird wie folgt begründet:

[16] Eine Entsorgung im Ausland ist eine offiziell anerkannte Option im Schweizer Entsorgungsprogramm.

- In der Schweiz wurden im Verlaufe des systematischen Einengungsverfahrens verschiedene potenzielle sedimentäre Wirtgesteine[17] betrachtet. Der Opalinuston weist eine Reihe besonders günstiger Eigenschaften auf: sehr geringe Durchlässigkeit, gute Rückhalteeigenschaften, gutes Selbstabdichtungsvermögen, vernünftige Baueigenschaften und gute Explorierbarkeit.
- Die systematische Einengung möglicher Standortgebiete[17] hat ergeben, dass das Zürcher Weinland eine Reihe günstiger Eigenschaften aufweist, wie geringe seismotektonische Aktivität, geeignete Tiefenlage des Opalinustons mit ausreichender ungestörter lateraler Ausdehnung sowie Rahmengesteine mit ähnlichen guten Eigenschaften wie das Wirtgestein selbst.
- Das Projekt *Entsorgungsnachweis* zeigt, dass für das Referenzsystem im Opalinuston im Zürcher Weinland:
  - ein hohes Mass an Sicherheit erwartet werden kann,
  - Bau, Betrieb und Verschluss des Lagers zuverlässig machbar sind und
  - das geeignete Standortgebiet eine genügende Ausdehnung aufweist.

Diese Folgerungen schliessen nicht aus, dass in anderen Gebieten auch ein sicheres Lager erstellt werden könnte, sei dies in anderen Gebieten mit Opalinuston oder in anderen Wirtgesteinen. Die Argumente, die zum Vorschlag der Fokussierung der zukünftigen Arbeiten auf den Opalinuston im potentiellen Standortgebiet des Zürcher Weinlands führten, sind aber aus Sicht der Nagra gut begründet.

3. Bevor ein geologisches Tiefenlager in der Schweiz realisiert werden kann, sind noch weitere Arbeiten nötig. Beispielsweise müssen die geologischen Untersuchungen vertieft, die endgültige Lagerauslegung festgelegt sowie verschiedene Bewilligungsverfahren durchlaufen werden. Ein formeller Entscheid über die Standortwahl wird als Teil des Rahmenbewilligungsverfahrens nicht vor dem Jahr 2020 erwartet. Damit steht genügend Zeit zur Verfügung, um die Untersuchungen fortzuführen und die endgültige Lagerauslegung umsichtig festzulegen. Der Detaillierungsgrad, welcher dem Projekt *Entsorgungsnachweis* zu Grunde liegt, erfüllt die Anforderungen der heutigen Phase des Entsorgungsprogramms und ist eine gute Basis für den Nachweis der sicheren Entsorgung von BE / HAA / LMA in der Schweiz (Aussage 1 oben) und für die Fokussierung künftiger Arbeiten auf das Wirtgestein Opalinuston und das potenzielle Standortgebiet im Zürcher Weinland (Aussage 2 oben).

[17] Dieses schrittweise Vorgehen dauerte mehrere Jahre, wurde in enger Zusammenarbeit und in Übereinstimmung mit den Behörden und deren Beratungskommissionen durchgeführt und ist in verschiedenen Berichten dokumentiert.

Exposé de synthèse

Ce rapport décrit l'analyse et l'évaluation de la sûreté radiologique à long terme d'un dépôt souterrain en profondeur[18], situé dans les Argiles à Opalinus du "Weinland zurichois" (Zürcher Weinland) dans le nord de la Suisse et destiné aux assemblages combustibles usés (AC), aux déchets de haute activité vitrifiés (DHA) issus du retraitement des assemblages combustibles, ainsi qu'aux déchets de moyenne activité à vie longue (DMAL). Cette analyse de la sûreté a été réalisée dans le cadre de l'étude de faisabilité intitulée *Entsorgungsnachweis*[19], qui comprend également une synthèse des résultats des études géologiques effectuées dans les Argiles à Opalinus, ainsi qu'un rapport concernant la faisabilité technique de la construction du dépôt. Le projet *Entsorgungsnachweis* constitue une phase essentielle du programme de gestion des AC, DHA et DMAL en Suisse, puisqu'il évalue la faisabilité du stockage géologique pour ce type de déchets. C'est également une étape importante vers la réalisation du dépôt.

Le projet *Entsorgungsnachweis* a deux objectifs principaux:

1. Démontrer que les AC / DHA / DMAL peuvent être stockés de manière sûre dans les Argiles à Opalinus du Weinland zurichois et remplir ainsi les exigences posées en 1988 par le Conseil fédéral dans l'arrêté concernant le projet *Garantie 1985*. Ces exigences sont les suivantes:

   - Démonstration de l'existence d'un site: prouver qu'il existe en Suisse un ou plusieurs sites adaptés au stockage des déchets du point de vue géologique (une formation d'accueil d'une étendue suffisante) et hydrogéologique.
   - Démonstration de la faisabilité technique: prouver qu'il est possible, dans l'état actuel des connaissances techniques, de construire, d'exploiter et de sceller un dépôt implanté sur un tel site.
   - Démonstration de la sûreté: prouver que la sûreté à long terme d'un tel dépôt est garantie.

2. Fournir une base de discussion et de décision pour la suite du programme de gestion des déchets de haute activité en Suisse, en présentant les principales conclusions et les bases scientifiques sur lesquels reposent les recherches. Les études géologiques sur les Argiles à Opalinus du secteur d'accueil potentiel dans le Weinland zurichois, réalisées à l'issue d'une procédure de sélection systématique, ont fourni d'excellents résultats et conduit la Nagra à proposer au Conseil fédéral de concentrer ses activités futures sur cette roche et ce secteur, dans le cadre de l'option de gestion des déchets "Stockage des AC / DHA / DMAL en Suisse"[20]. Le projet *Entsorgungsnachweis* doit par conséquent fournir des arguments étayant cette proposition, qui constitue l'aboutissement d'une procédure de sélection longue et systématique, élaborée en étroite collaboration avec les autorités de contrôle et les experts mandatés par la Confédération. Une approche progressive a permis de passer (i) pour la formation d'accueil potentielle, de sept roches sédimentaires à une seule (les Argiles à Opalinus) et (ii) pour les sites envisagés dans les Argiles à Opalinus, de deux grandes régions de prospection à un secteur potentiel pour l'établissement d'un dépôt dans le Weinland zurichois. Ce processus de sélection, qui a fait l'objet de plusieurs rapports rédigés par la Nagra, est également résumé dans un document émanant de l'autorité de

[18] La notion de "dépôt souterrain en profondeur" est définie dans le projet de loi sur l'énergie nucléaire. Après sa fermeture, le dépôt souterrain deviend un "dépôt final". Dans le texte qui suit, le terme de "dépôt" fait toujours référence à ce type d'infrastructure, également dénommé "dépôt en formation géologique profonde".

[19] Le terme allemand a été conservé dans les versions anglaise et française. Il peut être traduit par "Démonstration de la faisabilité du stockage à long terme".

[20] La possibilité d'un stockage à l'étranger figure officiellement dans la stratégie suisse de gestion des déchets.

contrôle (pour plus de précisions, se reporter au chapitre 1 du présent rapport). Le contenu des activités à venir et leur calendrier de réalisation seront déterminés à l'issue de l'examen du projet *Entsorgungsnachweis*.

Parmi les objectifs du projet figurent également:

3. La compilation des données nécessaires (i) à la planification générale du stockage des déchets (notamment les estimations des coûts), (ii) à la constitution d'une base scientifique et technique permettant d'évaluer des solutions alternatives pour l'architecture du dépôt et pour un inventaire différent (AC, DHA et plusieurs types de DMAL) et enfin (iii) à la planification des activités futures de recherche et de développement.
4. La mise à disposition de bases scientifiques et techniques sur le stockage des déchets en vue d'un dialogue avec toutes les parties en présence, en particulier avec le public. Un tel dialogue peut contribuer de façon non négligeable à une meilleure compréhension de certains aspects du problème (concepts de stockage, sûreté des dépôts, protection de l'environnement), et mener par conséquent à une discussion plus constructive, puis à une meilleure acceptabilité des programmes de stockage auprès de la population.

L'évaluation de la sûreté (le présent rapport) a les objectifs suivants:

1. Déterminer si, du point de vue de la sûreté à long terme, les Argiles à Opalinus du secteur d'accueil potentiel dans le Weinland zurichois ont les qualités nécessaires pour servir de roche d'accueil.
2. Mettre en lumière les multiples fonctions de sûreté fournies par le système de stockage proposé.
3. Evaluer la "robustesse" du système de stockage au regard des incertitudes qui subsistent, ainsi que l'impact potentiel des phénomènes susceptibles d'affecter les fonctions de sûreté.
4. Fournir une base scientifique pour aborder une série de questions relatives à la conception d'un dépôt. Les résultats de l'évaluation de la sûreté, ainsi que les commentaires des autorités de contrôle, serviront en particulier à la préparation des étapes ultérieures de planification et de développement.

La sûreté de la phase d'exploitation n'est pas traitée dans ce rapport. En revanche, elle est abordée du point de vue qualitatif dans le rapport sur la faisabilité technique.

Ce rapport présente une série d'arguments qui, dans leur ensemble, constituent la *démonstration de la sûreté*, c'est-à-dire la démonstration qu'un dépôt pour AC / DHA / DMAL situé dans les Argiles à Opalinus du Weinland zurichois peut être considéré comme sûr. Cette notion-clé se définit de la façon suivante:

La démonstration de la sûreté

La démonstration de la sûreté est composée d'un ensemble d'arguments et d'analyses qui justifient la conclusion selon laquelle un système de dépôt spécifique peut être considéré comme sûr. Elle contient en particulier la preuve que les critères de sûreté figurant dans le cadre légal peuvent être respectés. Elle comprend également une série de documents qui décrivent l'architecture du système et les fonctions de sûreté, détaillent les performances du système, présentent les données qui étayent les arguments et les analyses, et estiment l'importance relative des éventuelles incertitudes et questions demeurées sans réponse dans le contexte des décisions à prendre pour la poursuite des travaux.

Les différents chapitres sont brièvement présentés ci-après.

Le chapitre 1 est une introduction visant **à replacer le rapport dans le contexte national**[21]**.** Les cinq réacteurs actuellement en activité en Suisse (le plus ancien fonctionne depuis 1969), de même que la médecine, l'industrie et la recherche, produisent des déchets radioactifs dont la gestion à long terme doit être assurée. Un concept détaillé de gestion des déchets a par conséquent été élaboré. Grâce au site d'entreposage de Würenlingen, récemment entré en service, on dispose d'une capacité suffisante pour accueillir tous les déchets produits en Suisse, jusqu'à ce que ceux-ci soient transférés dans un dépôt géologique. Pour le stockage géologique, la construction de deux dépôts (l'un pour les déchets de faible et moyenne activité et l'autre pour les AC / DHA / DMAL) a été planifiée en détail.

Selon les dispositions légales en vigueur, (i) les déchets radioactifs produits en Suisse doivent être, en principe, éliminés sur le territoire national (toutefois, la loi définit certaines conditions selon lesquelles, à titre exceptionnel, une licence d'exportation pour le stockage des déchets à l'étranger pourrait être attribuée), et (ii) en Suisse, les producteurs de déchets radioactifs doivent faire en sorte que ceux-ci soient gérés et évacués de manière sûre. Afin de répondre à ces exigences, les compagnies électriques, qui exploitent les centrales nucléaires, et le gouvernement fédéral, responsable de la gestion des déchets provenant de la médecine, de l'industrie et de la recherche, ont créé en 1972 la Société coopérative nationale pour l'entreposage de déchets radioactifs (Nagra). La Nagra est chargée des travaux de recherche et de développement relatifs au stockage final des déchets. D'autres aspects inhérents à la gestion des déchets, tels que le conditionnement, l'entreposage, ainsi que la construction et l'exploitation des infrastructures nécessaires, sont du ressort des producteurs de déchets eux-mêmes ou d'organismes qu'ils auront créés dans cet objectif. L'autorité de contrôle de la Confédération est la Division principale de la sûreté des installations nucléaires (DSN). La Commission fédérale pour la sûreté des installations nucléaires (CSA) est chargée quant à elle d'examiner les projets d'installations nucléaires, y compris ceux qui concernent les dépôts de déchets radioactifs, et de prendre position vis-à-vis du Département fédéral de l'énergie sur les dossiers et les expertises effectuées par la DSN. La directive HSK-R-21, rédigée par la DSN et de la CSA, définit les objectifs de protection pour l'évacuation des déchets radioactifs. En Suisse, toute démonstration de la sûreté, y compris celle qui est présentée ici, se doit de prouver que ces objectifs de protection sont respectés. Le chapitre 1 présente enfin la stratégie élaborée pour la réalisation du dépôt et résume les principales étapes passées et futures.

Le chapitre 2 concerne les **recommandations et les principes qui président au choix du système de stockage et permettent d'évaluer sa sûreté à long terme**. Il présente tout d'abord les recommandations formulées par les organisations internationales et les dispositions du cadre légal et réglementaire suisse, en particulier le nouveau projet de loi sur l'énergie nucléaire (LENu) et la directive suisse HSK-R-21. Selon la LENu, les déchets doivent être mis en dépôt dans une formation géologique appropriée et la fermeture du site doit être précédée d'une phase de surveillance. Ces exigences reposent sur le concept de "stockage géologique durable contrôlé" proposé par le groupe d'experts mandaté par le Conseil fédéral, EKRA[22]. L'ensemble de ces documents est complété par des principes élaborés par la Nagra au plan interne, sur la base de l'expérience acquise à la fois en Suisse et dans le cadre de ses collaborations avec des organismes étrangers. Ces aspects spécifiques concernent la notion de "robustesse" et la procédure à suivre pour réaliser une analyse de la sûreté, mais aussi le rôle et le traitement de la biosphère, le

[21] Etat au 31 août 2002. Les conséquences du vote négatif sur la concession pour une galerie de sondage au Wellenberg, site envisagé pour le dépôt destiné aux déchets de faible et moyenne activité, ne sont pas évoquées dans ce rapport.

[22] "Expertengruppe Entsorgungskonzepte für radioaktive Abfälle" (Groupe d'experts pour les modèles de gestion des déchets radioactifs) nommé par le Conseil fédéral en juin 1999.

traitement des activités humaines futures et les échelles de temps à envisager. Du fait que ces informations émanent de sources diverses et sont par conséquent assez hétérogènes, les principaux résultats ont été groupés et résumés autour des objectifs et principes suivants: (i) les objectifs du stockage géologique, (ii) les objectifs en rapport avec le système (comprenant les *fonctions de sûreté* du système de dépôt, essentielles pour la démonstration de la sûreté), (iii) les objectifs en rapport avec une réalisation par étapes et (iv) les principes d'évaluation. Les fonctions de sûreté jouent un rôle primordial dans la démonstration de la sûreté et sont définies ci-dessous:

Les fonctions de sûreté

Le système de dépôt doit assurer un certain nombre de fonctions relatives à la sécurité et à la sûreté à long terme, que l'on appelle fonctions de sûreté:

- **Isolement des êtres humains et de l'environnement** – La sûreté et la sécurité des déchets, y compris des matières fissiles, sont garanties par leur mise en dépôt dans des formations géologiques profondes, suivie du remblayage, puis du scellement de toutes les voies d'accès, de manière à les isoler des êtres humains et de l'environnement, et à réduire la probabilité d'une intrusion humaine non souhaitée (par exemple un forage) et d'un usage illicite des matériaux. Cette probabilité peut être également limitée si l'on s'assure que, selon l'état des connaissances actuelles, le site ne renferme pas de ressources naturelles économiquement viables et n'est pas susceptible d'être englobé dans un projet de construction futur. Enfin, le processus de sélection doit vérifier l'absence de phénomènes géologiques pouvant remettre en question la stabilité à long terme.
- **Confinement à long terme et décroissance de la radioactivité au sein du système de dépôt** – Une part importante de l'activité présente à l'origine décroît alors que les déchets sont encore confinés dans leur emballage primaire, en particulier dans le cas des assemblages combustibles et des déchets de haute activité pour lesquels les conteneurs en acier ont été conçus pour résister pendant au moins 10 000 ans. Même après la perte d'étanchéité des conteneurs, la stabilité des AC et des DHA dans l'environnement du dépôt, la lenteur des mouvements d'eau, ainsi qu'un ensemble de processus géochimiques provoquant l'immobilisation et le retard du relâchement des radionucléides, font en sorte que la radioactivité demeure pour une large part confinée à l'intérieur du système de barrières ouvragées et de la roche avoisinante et peut, par conséquent, continuer de décroître.
- **Limitation du relâchement dans l'environnement** – Bien qu'il soit impossible d'assurer le confinement total, de tout temps, de l'ensemble des radionucléides, les taux de relâchement restent infimes, notamment pour les AC et les DHA, immobilisés dans des matrices stables. Par ailleurs, plusieurs processus contribuent à ralentir le relâchement des radionucléides au cours de leur migration vers la surface, puis à limiter leur concentration dans l'environnement. Citons à ce propos la décroissance de la radioactivité lors la très lente migration des radionucléides au travers de la roche d'accueil, et l'étalement du relâchement dans le temps et dans l'espace par le biais de processus tels que la diffusion, la dispersion hydrodynamique et la dilution.

Le chapitre 3 définit la **méthodologie utilisée pour élaborer la démonstration de la sûreté**, c'est-à-dire l'approche choisie pour évaluer la sûreté à long terme du dépôt envisagé, sur la base des principes d'évaluation présentés au chapitre 2. Elle comprend (i) l'identification des étapes nécessaires pour réaliser la démonstration de la sûreté et (ii) la définition des arguments qui étayent cette démonstration. Ces deux points sont décrits en détail ci-dessous.

(i) Les étapes de la démonstration de la sûreté comprennent:

- le choix du système de dépôt, par le biais d'une stratégie de développement modulable, basée sur les résultats d'études antérieures, en particulier d'études sur la sûreté à long terme,
- la description du concept de dépôt, élaboré sur la base des connaissances actuelles concernant les événements et processus (Features, Events and Processes ou FEPs) qui caractérisent le système de dépôt et peuvent avoir une influence sur son évolution,
- la définition du concept de sûreté, à partir de "garants de la sûreté[23]" aux mécanismes bien maîtrisés et à l'efficacité reconnue,
- la démonstration de l'impact radiologique du système de dépôt par le biais de la définition et de l'analyse de nombreuses "situations"[24], et enfin
- la compilation des arguments et analyses qui constituent la démonstration de la sûreté, suivie de recommandations pour les étapes ultérieures du programme de stockage.

(ii) La démonstration de la sûreté se fonde sur les arguments suivants:

- le bien-fondé du choix du stockage géologique pour la gestion à long terme des déchets radioactifs
- la sûreté et la robustesse du système de stockage choisi,
- la minimisation de la probabilité et des conséquences d'une intrusion humaine dans le dépôt,
- le bien-fondé d'un processus de réalisation par étapes,
- les bonnes connaissances scientifiques disponibles relatives au système de dépôt choisi et à son évolution,
- la pertinence de la méthodologie et des modèles, codes et bases de données utilisés pour l'évaluation de l'impact radiologique,
- divers arguments, notamment le respect des critères de sûreté fixés dans le cadre légal, l'utilisation d'indicateurs de sûreté complémentaires, l'existence d'une réserve d'événements et de processus (FEPs) et l'absence de questions non résolues susceptibles d'affecter la sûreté à long terme.

Le chapitre 4 présente l'état des connaissances actuelles concernant **le système de dépôt envisagé dans les Argiles à Opalinus du Weinland zurichois et ses caractéristiques à l'époque de sa fermeture.** Le choix de l'emplacement et la conception du système de dépôt respectent les objectifs et les principes énoncés au chapitre 2. Les éléments suivants font l'objet d'un développement particulier: (i) les caractéristiques principales du site, (ii) les propriétés principales des Argiles à Opalinus en tant que roche d'accueil, (iii) la disposition générale des ouvrages, (iv) les quantités et caractéristiques respectives des déchets (AC, DHA et DMAL),

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[23] En anglais "pillars of safety", littéralement "piliers de la sûreté". Voir définition ci-après, chapitre 6.

[24] Le terme de "situation" fait dans ce texte référence à l'anglais "assessment case", littéralement "cas sur lequel porte l'évaluation".

(v) le système de barrières ouvragées et l'architecture du dépôt. Ces éléments sont décrits ci-dessous.

(i) Les caractéristiques principales du site sont les suivantes:

- L'environnement géologique est simple et possède des propriétés structurelles, hydrogéologiques et géochimiques prévisibles.
- La région envisagée est stable du point de vue tectonique pour les périodes envisagées et présente un faible taux de soulèvement et d'érosion.
- Selon les connaissances actuelles, les sédiments qui recouvrent le socle rocheux de cette région, de même que le socle cristallin lui-même, ne constituent pas une ressource naturelle de quelque importance.

(ii) Les Argiles à Opalinus ont les propriétés suivantes:

- Leur conductivité hydraulique est extrêmement basse, si bien que les mouvements de solutés au travers de la formation sont plutôt dus à des processus de diffusion que d'advection.
- Du point de vue géochimique, les Argiles à Opalinus constituent un milieu réducteur, légèrement alcalin et modérément salin, favorable à la conservation des barrières ouvragées et à la rétention des radionucléides.
- L'environnement géochimique dans les Argiles à Opalinus et les formations avoisinantes devrait rester stable pendant plusieurs millions d'années[25].
- Grâce aux propriétés des Argiles à Opalinus, en particulier leur capacité d'auto-cicatrisation, les fissures naturelles ou causées par le dépôt auront une conductivité hydraulique très basse, ce qui signifie qu'elles n'affecteront que très peu les propriétés hydrauliques du site lui-même.
- Les Argiles à Opalinus sont des argiles indurées (argilites) qui autorisent le creusement d'ouvrages souterrains à plusieurs centaines de mètres de profondeur, notamment l'aménagement de petites galeries sans revêtement et de plus grandes galeries dotées d'un revêtement.

(iii) Le dépôt comprend les éléments suivants:

- une rampe d'accès, des galeries servant à la construction et à l'exploitation du dépôt, une aire centrale pour la réception des colis de déchets et un puits pour la phase de construction et la ventilation,
- une série de galeries de stockage parallèles, pratiquement horizontales, pour les AC et les DHA, d'un diamètre de 2,5 m et espacées entre elles de 40 m,
- trois tunnels de stockage courts et horizontaux pour les DMAL, situés à plusieurs centaines de mètres des galeries réservées aux AC et DHA,
- un dépôt pilote et un dépôt test, selon les spécifications du concept EKRA.

[25] Sur cette durée, la seule modification prévisible serait une faible baisse de la salinité.

iv) Du point de vue quantitatif, les déchets se caractérisent de la façon suivante:

- Les estimations reposent sur une durée d'exploitation de 60 ans pour les cinq réacteurs actuellement en activité en Suisse, ce qui correspond à une production totale d'électricité de 192 GWa(e).
- Pour estimer les quantités respectives d'AC et de DHA / DMAL, on pose comme hypothèse que seuls seront retraités les éléments combustibles irradiés pour lesquels il existe actuellement des contrats de retraitement (environ 1195 t_{MLI}).
- On obtient ainsi le chiffre d'environ 3200 t_{MLI} d'assemblages combustibles usés – soient 2065 conteneurs –, 730 conteneurs de déchets de haute activité et environ 7300 m^3 de déchets de moyenne activité à vie longue (conditionnés).

(v) Le système de barrières ouvragées comprend les composants suivants:

- Le conteneur pour AC: selon le concept de référence, il s'agit d'une coque en acier permettant d'accueillir soit 4 assemblages combustibles pour REP, soit 9 assemblages combustibles pour REB, avec des parois d'une épaisseur minimale de 15 cm, une longueur d'environ 5 m et un diamètre d'environ 1 m.
- Le conteneur pour DHA: de conception identique à celui qui figure dans le rapport sur le projet *Garantie*, il est composé d'un conteneur en acier renfermant un emballage en acier inoxydable où est placée la matrice de déchets vitrifiés. Les parois du conteneur ont une épaisseur de 25 cm, sa longueur est d'environ 2 m pour un diamètre d'environ 1 m.
- Les conteneurs pour DMAL: les fûts contenant les DMAL sont placés dans des conteneurs de stockage en béton. Un mortier de ciment est utilisé pour combler les espaces entre les fûts.
- Le remblayage des galeries pour AC/DHA: les conteneurs de AC/DHA sont placés sur des blocs de bentonite hautement compactée, parallèlement à l'axe de la galerie et espacés de 3 m entre eux. L'espace restant est comblé à l'aide de bentonite sous forme de granulés.
- Le remblayage des tunnels pour DMAL: les conteneurs de DMAL sont déposés dans les tunnels correspondants et l'espace restant est remblayé à l'aide de mortier de ciment.
- Le remblayage des galeries servant à l'exploitation et à la construction, de l'aire de réception centrale et de la rampe d'accès: il est effectué à l'aide d'un mélange de sable et de bentonite. Pour obturer les galeries, on utilise de la bentonite hautement compactée placée entre des cloisons étanches à plusieurs endroits du dépôt. Le puits est également comblé à l'aide de bentonite compactée.

Le chapitre 5 décrit **l'évolution du dépôt dans le temps** après sa fermeture, en tenant compte des interactions entre les différents éléments du système. L'évolution probable du site et des barrières ouvragées est évoquée, de même que divers scénarios d'évolution altérée. Sur cette base sont élaborées les "situations" qui seront ensuite analysées en détail du point de vue quantitatif.

Le chapitre 6 analyse **l'importance relative des différents éléments et phénomènes** et **identifie les situations qui vont faire l'objet d'une évaluation.** A l'aide d'analyses de sensibilité déterministes, on étudie l'impact possible des scénarios d'évolution altérée, au regard des *fonctions de sûreté* décrites au chapitre 2. Ces études sont complétées par des analyses probabilistes de sûreté et de sensibilité. La prise en compte des aspects qualitatifs et des analyses quantitatives permet d'identifier les principaux éléments et phénomènes qui sont déterminants pour la sûreté et que l'on dénomme *garants de la sûreté*. Du fait du rôle important qu'ils jouent dans l'élaboration de la démonstration de la sûreté, ces éléments et phénomènes sont définis ci-dessous:

Les garants de la sûreté

Il s'agit des éléments essentiels qui permettent au système de remplir les *fonctions de sûreté:*

- **La localisation du dépôt à une grande profondeur**, sur un site où la probabilité d'une intrusion humaine est faible et où l'on ne prévoit aucun phénomène géologique ou processus susceptible d'affecter la stabilité à long terme;
- **une roche d'accueil** caractérisée par une faible conductivité hydraulique, une texture fine et homogène et une bonne capacité d'auto-cicatrisation, agissant par conséquent comme une barrière efficace face à la migration des radionucléides et constituant un environnement adéquat pour le système de barrières ouvragées;
- **un environnement chimique** qui comprend une série de processus géochimiques d'immobilisation et de retard, favorise la stabilité à long terme des barrières ouvragées et possède lui-même une stabilité suffisante en raison de ses bonnes capacités de tampon chimique;
- **le remplissage et remblayage de bentonite (pour les AC/DHA)**. Interface entre les conteneurs et la roche d'accueil, la bentonite possède des caractéristiques connues et des propriétés similaires à celles de la roche d'accueil, elle assure que l'impact des galeries de stockage et des déchets exothermiques soit aussi faible que possible sur la roche; elle constitue à la fois une barrière efficace à la migration des radionucléides et un environnement adéquat pour les conteneurs et les déchets;
- **des AC et DHA** conditionnés de telle manière qu'ils demeurent stables dans l'environnement du dépôt;
- **des conteneurs pour AC/DHA** solides et résistants à la corrosion dans l'environnement du dépôt, et aptes à confiner efficacement les déchets sur de très longues périodes.

En posant tout d'abord l'hypothèse que les garants de la sûreté fonctionnent de la manière prévue, puis en évoquant les perturbations éventuelles (sur la base des observations faites aux chapitres 4 et 5 et en utilisant les résultats des analyses de sensibilité pour sélectionner les phénomènes pertinents), on peut identifier les situations nécessitant une évaluation quantitative.

Le chapitre 7 présente **les résultats de l'analyse des situations** identifiées au chapitre précédent. Il débute par une description de la conceptualisation des situations, structurées selon les différentes évolutions possibles du système de dépôt (scénarios) qui déterminent la voie principale de relâchement des radionucléides. Pour chaque scénario, différentes conceptualisations sont prises en compte et pour chaque conceptualisation, on utilise les variations de paramètres pour évaluer les incertitudes. Le point de départ est constitué par le "scénario de référence", qui comprend: un dépôt où les barrières ouvragées fonctionnent comme prévu, une géosphère basée sur la connaissance actuelle de l'environnement géologique et une biosphère dérivée des conditions géomorphologiques, hydrogéologiques et climatiques actuelles, en posant des hypothèses conservatrices pour ce qui concerne les activités humaines et la nourriture. Ensuite, une série de scénarios d'évolution altérée permet d'évaluer les conséquences des incertitudes qui subsistent quant au comportement et à l'évolution du système.

Afin de tester la robustesse du système de dépôt, une catégorie de situations de type "Qu'arriverait-il si..." a été créée pour prendre en compte des phénomènes situés en dehors de la gamme des possibilités scientifiquement prouvées. Il ne s'agit pas ici d'analyser des situations plausibles, mais plutôt de tester le comportement du système dans des conditions extrêmes. Afin de limiter le nombre de ces scénarios, seuls les phénomènes susceptibles d'affecter les propriétés principales des garants de la sûreté ont été retenus. La liste de ces situations n'a donc pas l'ambition d'être exhaustive.

Les options relatives à l'architecture et au système de dépôt font l'objet d'une évaluation séparée, car elles concernent des conceptualisations où le système est caractérisé plutôt par la flexibilité que par l'incertitude.

La sensibilité de la migration des radionucléides dans la biosphère est illustrée par plusieurs situations relatives à des conditions géomorphologiques et climatiques différentes. Dans le cadre de ces situations, on a cherché à montrer l'impact des incertitudes relatives à la biosphère en utilisant différentes possibilités (stylisées) pour les caractéristiques et l'évolution de l'environnement.

Le chapitre 8 constitue **une synthèse des principaux arguments et résultats.** Il représente l'étape finale pour la compilation de la démonstration de la sûreté. En reprenant chacun des arguments généraux énoncés au chapitre 3, il fournit un résumé des résultats pour chacun d'entre eux, notamment:

- Le bien-fondé du stockage géologique pour la gestion des déchets radioactifs est confirmé par (i) le fait qu'il est reconnu au niveau international qu'un système de stockage correctement conçu, implanté sur un site bien choisi, garantit la sûreté et la protection des êtres humains et de l'environnement ainsi que la sécurité face à des interventions non souhaitées, aujourd'hui et à l'avenir, (ii) l'existence, en Suisse et dans le monde, de formations géologiques d'accueil appropriées, (iii) les évaluations de la sûreté effectuées dans d'autres pays, (iv) l'observation de systèmes naturels (les "analogues naturels"), et (v) les avantages présentés par le stockage géologique par rapport aux autres options.
- La sûreté et la robustesse du système de stockage sont assurées (i) par un système de barrières passives où un ensemble de phénomènes garantissent les fonctions de sûreté, (ii) en évitant, par le choix d'un site et d'une architecture de stockage appropriés, les incertitudes et les phénomènes affectant la sûreté et (iii) par la stabilité à long terme du dépôt et des Argiles à Opalinus en tant que roche d'accueil.
- La probabilité et les conséquences d'une intrusion humaine peuvent être limitées (i) en faisant en sorte que l'information nécessaire concernant le dépôt soit correctement archivée et transmise aux générations futures, (ii) en vérifiant l'absence de ressources naturelles exploitables dans la région envisagée pour l'implantation du site, (iii) en plaçant les conteneurs d'AC et de DHA dans des compartiments séparés les uns des autres, et enfin en ne stockant que des déchets solidifiés.
- La réalisation du dépôt s'effectue par étapes, c'est-à-dire (i) que dans la phase actuelle, le système n'a pas besoin d'être défini dans les moindres détails, (ii) que l'on ne met en œuvre que des composants parfaitement maîtrisés et caractérisés (site, barrières ouvragées), (iii) que l'on peut engager le dialogue avec les parties concernées, ce qui permet d'obtenir des commentaires et des suggestions d'améliorations, (iv) que le projet est facilement adaptable et peut incorporer au fur et à mesure les résultats de nouvelles études (concernant par exemple la répartition des galeries et tunnels de stockage, le choix de l'architecture de dépôt, l'implantation des installations de surface et même le choix du site, c'est-à-dire qu'il existe d'autres sites possibles, ou d'autres formations rocheuses appropriées) et (v) qu'il est possible d'exercer une surveillance à long terme sur le dépôt et de réviser, si nécessaire, certaines décisions (cette possibilité allant par exemple jusqu'à la récupération des déchets).

- Les connaissances scientifiques disponibles, relatives au système de dépôt choisi et à son évolution sont basées sur (i) les résultats des programmes de recherches sur le terrain au niveau local et régional, comprenant une vaste campagne d'études sismiques en trois dimensions et un forage profond dans le secteur d'accueil potentiel, ainsi que des études complémentaires réalisées dans des laboratoires souterrains, notamment au Mont Terri, et des observations effectuées sur les Argiles à Opalinus dans plusieurs tunnels routiers et ferroviaires, (ii) les résultats de plus de 20 ans d'expérience dans l'élaboration et l'étude d'éléments du système de barrières ouvragées au sein du programme suisse, couplés à l'ensemble des données disponibles au niveau international et (iii) l'existence d'un modèle d'inventaire détaillé des déchets (AC, DHA et DMAL).
- La pertinence de la méthodologie et des modèles, codes et bases de données disponibles pour évaluer les conséquences radiologiques pour un large éventail de situations est confirmée par le fait qu'ils respectent les principes d'évaluation énumérés au chapitre 2.
- Les dispositions légales et réglementaires sont respectées: (i) dans toutes les situations envisagées, les doses maximales sont inférieures (souvent de plusieurs ordres de grandeur) au seuil fixé par la directive de l'autorité suisse de sûreté nucléaire, (ii) conformément à cette même directive, le dépôt proposé peut, à n'importe quel moment d'une éventuelle phase de surveillance prolongée, être fermé en quelques années, et la sûreté du dépôt ne nécessite pas de mesures supplémentaires après la fermeture et (iii) l'architecture de dépôt retenue répond aux exigences du concept de stockage géologique durable contrôlé.
- Les indicateurs de sûreté complémentaires, utilisés conjointement aux indicateurs «classiques» (dose efficace et risque radiologique), comprennent (i) la radiotoxicité des déchets en fonction du temps, comparée à celle des radionucléides existant naturellement (ii) les flux de radiotoxicité dus au relâchement progressif des radionucléides, comparés aux flux de radiotoxicité naturellement présents dans l'environnement de surface, (iii) les concentrations de radiotoxicité émanant du dépôt à l'interface géosphère-biosphère, comparées aux concentrations de radiotoxicité dans les Argiles à Opalinus et (iv) l'évaluation de la répartition de la radiotoxicité dans les différents éléments du système en fonction du temps.
- Certains événements et processus (FEPs) ayant une influence positive sur la sûreté du dépôt n'ont pas été traités, car il n'existait pas, lors de l'étude, de modèles permettant leur analyse quantitative. Parmi ces "FEPs de réserve", qui constituent des arguments supplémentaires en faveur de la sûreté, figurent (i) la co-précipitation des radionucléides avec des minéraux secondaires issus de la corrosion des assemblages combustibles, du verre et des conteneurs (à l'exception de la co-précipitation du radium, qui a été prise en compte dans toutes les situations), (ii) la sorption des radionucléides sur les produits de corrosion des conteneurs, (iii) les concentrations naturelles d'isotopes en solution dans l'eau interstitielle de la bentonite, susceptibles de réduire encore plus la solubilité effective de certains radionucléides, (iv) la sorption irréversible des radionucléides dans le champ proche ou la géosphère (minéralisation de surface), (v) les processus d'immobilisation à long terme dans la géosphère (précipitation/co-précipitation), (vi) le retard du relâchement des radionucléides dû à la lenteur de la corrosion des matériaux métalliques contenus dans les DMAL (par exemples les gaines et les embouts), ainsi qu'une période de confinement complet des DAML assurée par les fûts en acier et les conteneurs de stockage et (vii) la lenteur de la resaturation du dépôt et de ses environs, qui retarde le début des processus de corrosion et de dissolution.
- Du fait que dans plusieurs des situations envisagées, on a opté pour une représentation simplifiée et conservatrice ou pessimiste du fonctionnement du système, les doses calculées sont plus élevées qu'elles ne le seraient dans une situation réelle. Ceci fournit une marge de sûreté supplémentaire.

- A l'issue de l'analyse détaillée d'une large gamme de situations dérivées de façon précise et méthodique, aucune question susceptible d'affecter la sûreté du dépôt n'est restée sans réponse.

Le chapitre 9 présente les **principales conclusions:**

1. Le projet *Entsorgungsnachweis* répond à une demande émanant du Conseil fédéral qui souhaitait, à l'issue de l'examen du projet *Garantie*, obtenir une démonstration convaincante de la faisabilité du stockage géologique. Les travaux présentés dans ce rapport montrent que la sûreté peut être garantie pour le système choisi, dans les Argiles à Opalinus et le secteur d'accueil potentiel du Weinland zurichois. Les données et analyses démontrent notamment que:

   - le site de référence possède des propriétés suffisantes pour assurer la sûreté du dépôt. La démonstration de la sûreté fournit des arguments à l'appui de cette affirmation: la sûreté est suffisante pour une large palette de situations et la gamme des situations analysées est suffisamment étendue pour couvrir toutes les éventualités;

   - le système est robuste, c'est-à-dire que les incertitudes qui subsistent ne remettent pas la sûreté en question;

   - les informations disponibles pour les déchets et le système de barrières ouvragées sont suffisantes pour une prise de décision. Elles reposent sur plus de 20 ans de recherches en Suisse, ainsi que sur l'expérience acquise par d'autres pays.

   De plus, comme le montre en détail le rapport de la même série traitant de la construction du dépôt, les caractéristiques du site et l'architecture envisagée permettent la construction, l'exploitation et la fermeture du dépôt conformément aux spécifications, et par conséquent aux contraintes de sûreté.

   Les informations sur le site sont suffisantes et le site est suffisamment bien compris pour étayer les affirmations concernant la sûreté et la faisabilité technique. Le troisième rapport de la série concerne les aspects plus spécifiquement géologiques et démontre que:

   - les méthodes de recherche les plus récentes (sismique 3 D) ont permis de bien caractériser la géométrie et la structure de la roche d'accueil et des roches avoisinantes (dite "encaissantes"), si bien que l'on a pu identifier un espace non perturbé suffisamment étendu pour l'implantation d'un dépôt;

   - la roche d'accueil et les roches avoisinantes, caractérisées à l'aide du forage de Benken, ont les propriétés nécessaires pour assurer la sûreté à long terme;

   - les processus susceptibles d'affecter la sûreté ont fait l'objet de recherches détaillées au laboratoire souterrain du Mont Terri, ainsi qu'en laboratoire, et les résultats confirment et complètent les résultats obtenus par le forage de Benken. Les propriété du site et de la roche d'accueil sont bien connues. Leur évolution peut donc être prédite de manière fiable, sur la base d'études géologiques couvrant une région étendue, et grâce à une situation générale relativement simple.

2. Le projet *Entsorgungsnachweis* fournit une base de discussion et de décision pour la poursuite des travaux dans le cadre du programme suisse de gestion des déchets de haute activité et pour l'évaluation du rôle des Argiles à Opalinus du Weinland zurichois au sein de ce programme. Les excellents résultats obtenus lors des recherches géologiques sur le terrain et par l'évaluation de la sûreté pour le projet *Entsorgungsnachweis* ont conduit la

Nagra à proposer au Conseil fédéral de concentrer ses travaux sur les Argiles à Opalinus du Weinland zurichois pour l'option de gestion des déchets intitulée "Stockage des AC / DHA / DMAL en Suisse"[26]. Ceci est justifié par[27]:

- une étude systématique des formations sédimentaires susceptibles de servir de roche d'accueil a montré que les Argiles à Opalinus possédaient un ensemble de caractéristiques particulièrement favorables, telles qu'une structure compacte, de bonnes capacités de retardation, des propriétés d'auto-cicatrisation, et une bonne aptitude à la construction et à l'exploration souterraines;
- une étude systématique des régions d'accueil potentielles a montré que le secteur envisagé du Weinland zurichois possédait une série de qualités favorables, telles qu'une faible activité tectonique, la présence d'une formation non perturbée d'Argiles à Opalinus à une profondeur et sur une étendue suffisantes, et l'existence de roches avoisinantes possédant des qualités similaires à celles de la roche d'accueil;
- les résultats du projet *Entsorgungsnachweis* montrent clairement que, pour un système de référence dans les Argiles à Opalinus du Weinland zurichois,
  - un niveau de sûreté élevé est garanti,
  - la construction, l'exploitation et la fermeture d'un site sont réalisables,
  - le site est doté de bonnes caractéristiques et offre une étendue suffisante.

Les résultats positifs obtenus pour cette roche d'accueil et ce secteur ne signifient pas qu'un système tout aussi sûr ne pourrait pas être implanté dans d'autres régions où les Argiles à Opalinus sont également présentes, ou même dans d'autres formations rocheuses. Cependant, les arguments techniques (basés sur la sûreté, la simplicité de la géologie et le caractère prévisible de l'évolution) qui ont présidé au choix de cette région sont plausibles et reposent sur des bases solides.

3. Un grand nombre d'étapes sont encore nécessaires avant qu'un dépôt ne soit réalisé en Suisse: campagne d'exploration du sous-sol, choix de l'architecture définitive du site et octroi des autorisations nécessaires. Une décision officielle concernant la sélection du site – un jalon décisif au sein du processus d'autorisation – ne sera pas rendue avant au moins 2020. Cela signifie qu'il reste suffisamment de temps pour continuer les recherches et perfectionner le concept de stockage. De ce fait, le degré de précision atteint par les rapports du projet *Entsorgungsnachweis* est adapté à la phase actuelle du programme et il est suffisant pour étayer les conclusions selon lesquelles (i) le stockage géologique des AC / DHA / DMAL en Suisse est réalisable (affirmation n° 1 ci-dessus) et (ii) le choix du Weinland zurichois en tant que secteur d'étude, et celui des Argiles à Opalinus en tant que roche d'accueil, est justifié (affirmation n° 2 ci-dessus).

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[26] Le stockage à l'extérieur des frontières nationales est une possibilité qui figure officiellement dans la stratégie suisse de gestion des déchets.

[27] Pour les deux premiers points, cette approche systématique s'est déroulée par étapes sur plusieurs années, en étroite collaboration et avec l'aval des autorités de contrôle et de leurs experts. Elle a fait l'objet de plusieurs rapports (voir chapitre 1).

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1 Introduction

1.1 Background and scope of this report

This report presents a comprehensive description of the post-closure radiological safety assessment of a deep geological repository. This project considers a repository sited in the Opalinus Clay[28] of the Zürcher Weinland region in northern Switzerland that is designed for the disposal of:

- spent fuel (SF[29]), in the form of fuel assemblies containing UO_2 or mixed-oxide (MOX) fuel,
- vitrified high-level waste (HLW[30]) from the reprocessing of spent fuel, and
- long-lived intermediate-level waste (ILW[31]).

For over twenty years, Nagra has been developing its concepts for implementing such repositories and for performing the necessary analyses of the long-term safety that they can offer. The disposal strategy that Nagra has refined over the years is fully compatible with the concept of monitored long-term geological disposal[32], as recently described in guidance and requirements documented by Swiss governmental bodies and their experts (KEG 2001, EKRA 2000). This safety assessment[33] has been carried out as part of the technical basis for Project *Entsorgungsnachweis*[34], which also includes a synthesis of information from geological investigations of the Opalinus Clay (Nagra 2002a), and a report on engineering feasibility (Nagra 2002b). Project *Entsorgungsnachweis* is a milestone in the programme for the long-term management of SF, HLW and ILW and represents an evaluation of the feasibility of the disposal of these wastes in Switzerland. It is also a major step on the way towards repository implementation. It is recognised internationally that a stepwise or staged approach to the development of a deep geological repository is the most appropriate way of addressing the technical and societal challenges involved (see, e.g., NEA 1999a, NRC 2001). The stages chosen and the corresponding breadth and depth of the scientific programmes and their accompanying documentation will vary from country to country, as will the timing of the overall disposal programme. The following sections in this chapter discuss the Swiss radioactive waste management planning, the step-wise repository implementation process as foreseen in Switzerland, the milestones of the SF / HLW / ILW programme, the aims of Project *Entsorgungsnachweis* in general, and of the post-closure safety assessment in particular. In the succeeding chapters, arguments are derived that together comprise the case for the long-term safety of the repository, i.e. the *safety case*[35]. The exact definition of this term, as used throughout Project *Entsorgungsnachweis* and as given below, is based on the results of interactions with other waste management organisations and on

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[28] Opalinus Clay is a shale (claystone) formation present in large areas of northern Switzerland.

[29] The German term, used in Switzerland, is BE (abgebrannte **B**renn**e**lemente).

[30] Termed HAA (**h**och**a**ktive **A**bfälle) in German.

[31] Termed LMA (**l**anglebige **m**ittelaktive **A**bfälle) in German. This waste form is broadly similar to the waste category sometimes referred to as TRU – transuranic-containing waste – even though the transuranics may not be the most safety-relevant radionuclides in such waste.

[32] The concept of monitored long-term geological disposal is discussed further in Chapter 2.

[33] The present report, which aims at providing a transparent description of the safety case, is accompanied by a report that describes in more detail the models, codes and data used in this assessment (Nagra 2002c). The aim of this accompanying report is to provide traceability.

[34] The German term is also used in the English version of this report. The term translates into English as "demonstration of disposal feasibility".

[35] Some definitions of key terms are given in the text immediately after the position where they appear for the first time, and are placed in boxes. A summary of definitions of key terms is also given in Appendix 5.

the work of international groups, such as the NEA ad hoc group on validation and confidence building (NEA 1999b) and the group on integrated performance assessments of deep repositories (IPAG) (NEA 1997, 2000a and 2002a), but has been adapted for our own purposes.

Safety Case

The safety case is the set of arguments and analyses used to justify the conclusion that a specific repository system will be safe. It includes, in particular, a presentation of evidence that all relevant regulatory safety criteria can be met. It includes also a series of documents that describe the system design and safety functions, illustrate the performance, present the evidence that supports the arguments and analyses, and that discuss the significance of any uncertainties or open questions in the context of decision making for further repository development.

1.2 Swiss radioactive waste management planning

1.2.1 Overview

A schematic overview of the sources of radioactive waste in Switzerland and the waste management strategy considered[36] is presented in Fig. 1.2-1.

Radioactive waste in Switzerland arises from the operation and decommissioning of nuclear power plants, from the fuel cycle of the power plants and from medicine, industry and research. To estimate waste volumes and inventories, in previous studies, a 40-year operational lifetime of the power plants was assumed. In *Project Entsorgungsnachweis*, however, a reference power plant lifetime of 60 years is assumed; this corresponds to a total nuclear power production of 192 GWa(e)[37]. This should be seen as a cautious approach towards estimating waste volumes, taking into account recent international trends to extend the lifetime of existing power plants if applicable safety standards can be met at acceptable costs. This is also in line with the government draft of Switzerland's revised nuclear energy law (KEG 2001). Enough flexibility is, however, maintained to cope even with the possibility of an extension of the existing nuclear power programme.

A part of the spent fuel from nuclear power production is being reprocessed. The fissile material recovered in the process is used to fabricate new fuel elements. The wastes arising from reprocessing (HLW and ILW) are being returned to Switzerland. The Swiss electricity utilities have placed contracts with BNFL (UK) and COGEMA (France) for the reprocessing of spent fuel from about 40 GWa(e) of nuclear power generation. For the purposes of the present safety assessment, it is assumed that the spent fuel not covered by the existing reprocessing contracts will be disposed of without reprocessing ("direct disposal")[38].

36 Status: August 31, 2002. The consequences of the negative outcome of the public referendum on the concession for an investigation gallery for the proposed L/ILW repository at Wellenberg are not discussed in this report.

37 Calculated by multiplying the net annual power capacity of the Swiss nuclear power plants given in Tab. 1.2-1 by 60 years.

38 This is in line with the government draft of Switzerland's revised nuclear energy law (KEG 2001), which prohibits any future reprocessing.

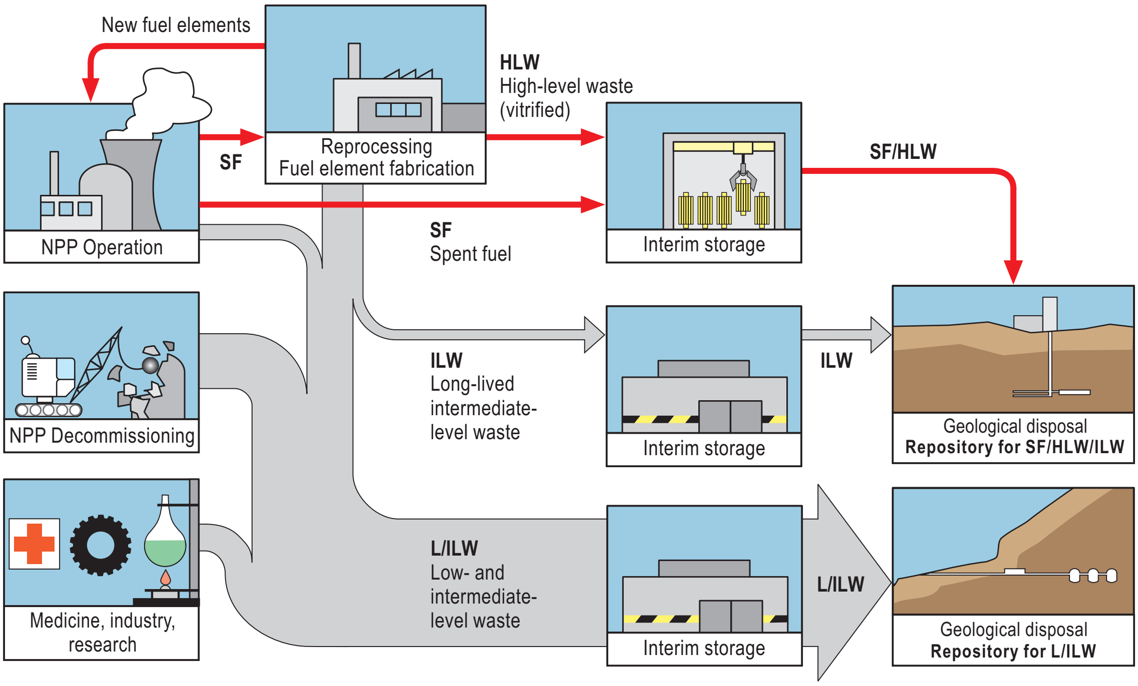

Fig. 1.2-1: Overview of the sources of radioactive waste in Switzerland and the waste-management strategy considered

Tab. 1.2-1: The nuclear power plants currently operating in Switzerland

Plant	Reactor type [1]	In operation since	Net power capacity [2] [MW(e)]
Beznau I	PWR	1969	365
Beznau II	PWR	1971	365
Mühleberg	BWR	1971	355
Gösgen	PWR	1979	970
Leibstadt	BWR	1984	1145

[1] BWR = boiling-water reactor, PWR = pressurised-water reactor.
[2] Status: January 2002 (SVA 2002).

1.2.2 Legal framework

In Switzerland, management of radioactive waste is regulated by a legal framework consistent with the requirements of the IAEA Joint Convention on the Safety of Spent Fuel and on the Safety of Radioactive Waste Management (IAEA 1997). At the highest level, the Federal Constitution (Art. 90) states that atomic energy legislation is a federal matter.

The Atomic Law of 1959 (AtG 1959) forms the legal basis on which management of radioactive waste arising from the peaceful use of nuclear energy (i.e. from the operation and decommissioning of nuclear power plants) is founded. According to the Law, facilities for disposing radioactive waste must be licensed and supervised by the Federal Government. The Law is supplemented by the Federal Government Act on the Atomic Law of 1978 (BB/AtG 1978), which embodies the principle that the producers of radioactive waste are responsible for its safe disposal and all associated costs. The AtG and the BB/AtG also define the different licences required in the step-wise approach to repository implementation (see Section 1.2.5). The Atomic Law is currently being revised; the revision is termed "Kernenergiegesetz" (KEG 2001). The law as proposed by the Federal Government[39] includes, in addition to the above points, the following key elements: (i) the option of nuclear power is kept open (no limits on nuclear power plant (NPP) lifetimes, new NPPs are in principle possible); (ii) decisions on new NPPs are subject to an optional national referendum ("fakultatives Referendum"); (iii) any future reprocessing of spent fuel not covered by existing contracts is prohibited; (iv) radioactive waste has to be disposed of in a deep geological facility ("geologisches Tiefenlager").

The Radiological Protection Law of 1991 (StSG 1991) and the Radiological Protection Ordinance of 1994 (StSV 1994) govern the management of all other radioactive materials; i.e. all except those arising from the operation and decommissioning of nuclear power plants.

Both Laws state that radioactive waste arising in Switzerland shall, in principle, be disposed of in Switzerland. The draft KEG and the StSG define the conditions under which, by way of an exception, an export licence for disposal of such wastes abroad may be granted.

The Swiss Federal Nuclear Safety Inspectorate (HSK) is the supervisory authority. The Federal Commission for the Safety of Nuclear Installations (KSA) is responsible for evaluating projects for nuclear installations including radioactive waste repositories and for submitting to the Energy Department statements on the licensing applications and on the reviews from HSK. In HSK's and KSA's guideline HSK-R-21, the protection objectives for disposal of radioactive waste are defined (HSK & KSA 1993). HSK-R-21 is further discussed in Chapter 2.

In Switzerland, the producers of radioactive waste are legally responsible for its safe management and disposal (BB/AtG 1978, KEG 2001). To carry out their waste disposal responsibilities, the electricity supply utilities, which operate the nuclear power plants, and the Federal Government, which is responsible for the management of waste arising from medicine, industry and research, set up the National Cooperative for the Disposal of Radioactive Waste (Nagra) in 1972. Nagra is responsible for research and development work associated with final disposal. Other aspects of the waste management process, such as conditioning, interim storage and construction and operation of repositories, remain the responsibility of the individual waste producers or of organisations that may be set up by the producers specifically for these purposes.

1.2.3 Types of repositories

Two types of repository are foreseen in Switzerland:

- A repository for the disposal of low- and intermediate-level waste (L/ILW[40]) arising from the operation and decommissioning of Swiss nuclear power plants, from medicine, industry and research and from those operations in reprocessing that produce only low-level technological waste. The repository will consist of mined caverns under a mountain, with horizontal access, located in a suitable host rock.

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[39] By the time of the deadline for editorial changes to this report, the debate in Parliament regarding the "government draft" of the KEG (KEG 2001) was still ongoing.

[40] Termed SMA (**s**chwach- und **m**ittelaktive **A**bfälle) in German.

- A repository for the disposal of SF, HLW and long-lived ILW (primarily resulting from fuel reprocessing). The repository will be located in a deep geological formation and will consist of a tunnel system for SF and HLW and separate tunnels for ILW, with access via a ramp and/or vertical shafts, depending on the repository location and the host rock that is selected.

Prior to disposal, SF and HLW will be held in interim storage for a period of at least 40 years, in order to allow radiogenic heat production to decrease. The interim storage facilities ZWILAG[41], ZWIBEZ[42], BZL[43] and others[44] provide enough capacity for storage of all radioactive wastes of Swiss origin.

1.2.4 The programme for the management of spent fuel, high-level waste and long-lived intermediate-level waste in Switzerland

The planning and development of a repository for SF, HLW and ILW is a novel and complex task that requires many years to complete. As in other countries where deep geological disposal of radioactive waste is being considered, repository development in Switzerland is proceeding in stages. The major past and future steps in the Swiss case are illustrated in Fig. 1.2-2 and summarised below.

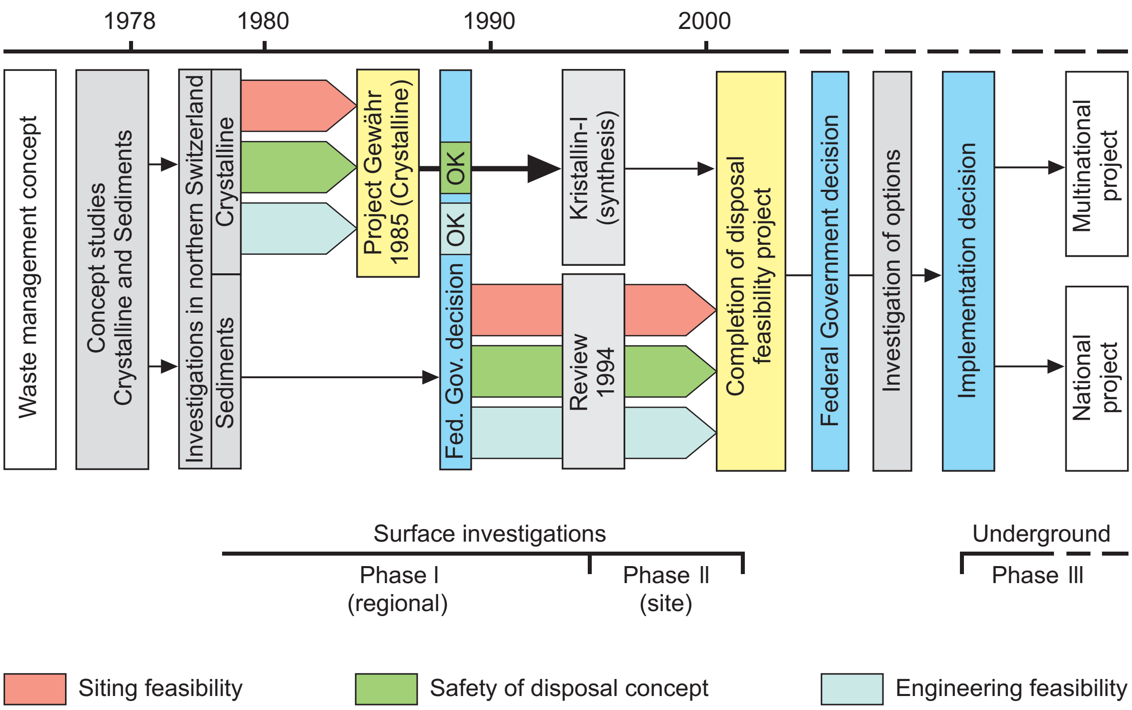

Fig. 1.2-2: History and future of the Swiss programme for spent fuel, vitrified HLW and long-lived ILW

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41 A recently completed centralised interim storage facility for all waste types.

42 A facility for SF, HLW and L/ILW operated by the Beznau nuclear power plant.

43 "Bundeszwischenlager", a facility for radioactive waste from medicine, industry and research, located at the same site as ZWILAG and operated by the Paul Scherrer Institute (PSI) on behalf of the Swiss Federal Office of Health (BAG).

44 Facilities for operational waste located on the grounds of the nuclear power plants.

- As early as 1978, the Swiss electricity supply utilities and Nagra formulated key aspects of the Swiss nuclear **waste management concept** in a landmark report (VSE et al. 1978). Both crystalline formations and sediments (e.g. clay, marl, salt, anhydrite) were identified as potential host rocks and first design concepts covering both crystalline rocks and sediments were developed (Nagra 1980).
- An extensive **regional field investigation programme in northern Switzerland**, including seven deep boreholes and two seismic surveys, was launched in the early 1980s. The focus was mainly on crystalline rocks, but sediments were also investigated.
- In parallel, an extensive **technical programme** was launched, with activities in, e.g., the areas of development, testing and application of safety assessment models, waste inventory characterisation, HLW glass matrix stability, HLW canister design, characterisation of bentonite, the development of geochemical databases, etc. In addition, the underground rock laboratories Grimsel Test Site (in operation since 1984) and Mont Terri (in operation since 1996) were established as centrepieces of Nagra's research and development programme.
- **Synthesis "Project Gewähr" 1985:** This project (Nagra 1985) was requested by the Federal Government to demonstrate the feasibility of safe disposal of all categories of radioactive waste, including low- and intermediate-level waste. Such a demonstration was made a condition for the further operation of existing nuclear power plants and the licensing of new plants (BB/AtG 1978). The prime questions posed by the Federal Government for the project concerned the feasibility of constructing and operating a deep geological repository, and the long-term safety of the facility, with the stipulation that the analyses be based on real geological data. A specific site was not sought, however, and it was recognised that all field data for the project would be gathered by exploration from the surface. The host rock selected for examination in Project Gewähr 1985 for the disposal of HLW was the crystalline basement of northern Switzerland[45]. The government-specified deadline for submitting Project Gewähr 1985 fell before the completion of all field work (the Siblingen borehole was completed only in 1989 because of extensive opposition) and hence it was not possible to completely integrate all relevant geological data within this project. The regional field studies provided a large amount of valuable results and led to new insights about the geology of the basement in northern Switzerland. Due to the large Permocarboniferous Troughs identified in these studies the area potentially available for siting a repository is much smaller than originally anticipated and the remaining area is also more heterogeneous than originally expected.
- **Synthesis "Kristallin-I":** This project completed the evaluation of data collected on the crystalline basement in the regional field investigation programme and presented the results in the form of a geological synthesis (Thury et al. 1994). The project also included an updated assessment of the post-closure safety of a high-level waste repository sited in the crystalline basement (Nagra 1994a). The overall conclusion of the safety assessment was that, although considerable work would be needed for the detailed characterisation of potential sites and on the optimisation of repository layout and engineered barrier design at a selected site, a high-level waste repository sited in the crystalline basement of northern Switzerland is feasible from the point of view of long-term safety. Kristallin-I is currently[46] still under review by HSK.

  Based on the results of the Kristallin-I study, Nagra proposed a programme for future work for the crystalline option. This was discussed in depth with the safety authorities and their experts, resulting in a modified programme reflecting the consensus that was reached. This

[45] For L/ILW, the host rock selected for examination was marl (Oberbauenstock, Canton of Uri).

[46] i.e. at the time of the deadline for editorial changes to the present report.

led to additional field work in the Mettauer Tal (Vorwaldscholle). The discussions also confirmed the reservations of the Swiss safety authorities and their experts about the explorability of the crystalline bedrock (see next bullet point).

- In its **decision on Project Gewähr**, which was handed down in 1988, the Federal Government judged that two out of three conditions for demonstration of HLW disposal feasibility were satisfied:

  (i) construction feasibility was given; safe construction and operation are possible with current technologies;

  (ii) long-term safety was shown to be achievable, provided that the database used in the analyses could be confirmed to be applicable to a sufficiently extensive potential disposal area;

  (iii) the highly localised nature of the geological field data did not allow one to say with confidence that sufficiently large areas of crystalline rock with the required properties could be found in Switzerland; a specific suitable site had certainly not been identified and thus siting feasibility was not fully demonstrated.

  The government required that the siting feasibility be more convincingly demonstrated and also that sediments be investigated more intensively as alternative potential host rocks for the disposal of HLW. Nagra subsequently expanded the geological investigations, construction feasibility studies and safety studies of sedimentary host-rock options in parallel with its work on crystalline rock. Key stages for the investigations of the sediment option are summarised under the following bullet point.

- On the basis of existing data (from both its own and other programmes), Nagra's first step was to screen adequate geological situations and potential host-rock options and, as a result, the **Opalinus Clay** and the **Lower Freshwater Molasse** (Untere Süsswassermolasse) were identified as **potential sedimentary host formations**. The work carried out is documented in two interim reports (Nagra 1988 and 1991) and a status review (Nagra 1994b). Key issues for safety were identified in the earlier interim report, and worked on subsequently. The favoured option was Opalinus Clay, a shale (claystone) formation, and a field-investigation programme, including seismics, was carried out for this potential host rock (seismic campaign 1991/1992). The Lower Freshwater Molasse was also investigated in a desk study and with supplementary field data obtained from third parties (e.g. Burgdorf geothermal field, Bassersdorf geothermal borehole). As part of the status review of 1994 (Nagra 1994b), the selection of the Opalinus Clay as the favoured sediment option was confirmed by both Nagra and the authorities, with the Lower Freshwater Molasse being kept as a reserve option. Agreement was also reached that the field results obtained indicated that the Zürcher Weinland should be the area to be investigated with first priority.

- The next milestone in Nagra's programme for SF, HLW and ILW is **Project *Entsorgungsnachweis***, the completion of the demonstration of disposal feasibility in Switzerland. The main thrust of the project concerns the aspect of siting feasibility which was judged to have been left open by Project Gewähr. Since Project *Entsorgungsnachweis* is based on a repository in Opalinus Clay, however, the engineering feasibility question must also be revisited and, of course, the siting feasibility is intimately related to long-term safety, which thus needs special consideration. Accordingly, Project *Entsorgungsnachweis* includes, in addition to the extensive geological synthesis, an engineering report and also a post-closure safety assessment that forms the subject of the present report.

  An additional objective of Project *Entsorgungsnachweis* is to provide a platform for discussion and a foundation for decision-making on how to proceed with the Swiss HLW programme and to assess the role of the Opalinus Clay in the Zürcher Weinland in this programme. The aims of Project *Entsorgungsnachweis* are discussed further in Section 1.3.

- Besides the Opalinus Clay of the Zürcher Weinland, siting alternatives have been identified for the Opalinus Clay host rock with the siting regions "Nördlich Lägern" and "Jura-Südfuss" (Nagra 1994b) and for the host rock option "crystalline basement" the potential siting region "Mettauer Tal" (Vorwaldscholle) has been selected and characterised by 2 D seismics. Finally, for the reserve host rock option "Lower Freshwater Molasse" preferred regions have been identified on a preliminary basis (Nagra 1994b).
- In the period between Project *Entsorgungsnachweis* and the **implementation decision**, state-of-the-art expertise will be maintained and used to further develop the Opalinus Clay project and also to allow an assessment of the relative advantages and disadvantages of the options "national project" and "multinational project". Further work in the Zürcher Weinland at a later stage would also include planning studies and environmental assessments aimed at fixing a specific site, which need not be at the exact location of the initial test drilling at Benken. In this process local and regional interests will be considered and the corresponding bodies will be involved. Since a repository for SF, HLW and ILW is not required for several decades, there is enough time to address relevant issues in greater detail than in Project *Entsorgungsnachweis* where necessary.

In summary, for Project *Entsorgungsnachweis,* Nagra can make use of extensive experience gained in the course of the programmes for SF, HLW and ILW on one hand and L/ILW on the other hand, both of which have been ongoing for more than twenty years. Relevant fields of expertise include:

- regional field investigations;
- selection of potential siting areas;
- specific investigations in potential siting areas, including many deep boreholes and several seismic campaigns, one of which involved a very advanced three dimensional survey;
- compilation of site investigation results in synthesis reports;
- facility design in hard rocks and sediments;
- development of concepts for robust engineered barrier systems (EBS);
- establishment and operation of underground rock laboratories;
- development of methodologies for safety assessments and data bases and their practical application.

During all of the work, intensive co-operation with other national programmes and with international organisations has been a key feature of the Swiss waste management strategy. Nagra staff have headed, or participated in, a very wide range of joint working groups and collaborative projects involving studies, lab work and field investigations in Switzerland and in many other countries around the world.

1.2.5 Important elements in the step-wise repository implementation process foreseen in Switzerland

The step-wise implementation process foreseen in Switzerland requires periodic decision-making. In this decision-making, one of the most important aspects that is continually revised and reviewed is the assessment of the level of safety provided by the repository system and of the level of confidence in safety. This process is formalised by developing and evolving a safety case for the system, as described in detail in Chapter 3. However, the safety case will not be the only information that is used for decision-making but other elements will also be important. The safety case is strongly interrelated with these elements that include:

- the **implementation strategy** (RD+D programme[47], site selection & characterisation, repository design);
- societal confidence in, and support for, the **concept of geological disposal** for the long-term management of radioactive wastes;
- the development of a **decision-making process** that finds sufficiently broad acceptance.

The important elements in the step-wise and iterative[48] implementation process and their interrelation, which are illustrated in Fig. 1.2-3, are briefly described below.

The implementation strategy determines the system to be considered and includes site selection and system design. Timescales for repository implementation are also an important part of the strategy. The implementation strategy determines what RD+D projects are needed at what times for the chosen system. The current understanding with respect to the site and repository design as well as key results from the corresponding RD+D programme are described in more detail in Chapters 4 and 5.

Societal confidence in the concept chosen for the long-term management of radioactive waste is crucial for a successful repository implementation. Although geological disposal has been the chosen option in Switzerland for many years, and still is today (AtG 1959, KEG 2001), there have been demands, especially from non-governmental organisations (NGOs) for a broad review of all options. In recognition of this fact, the Head of the Federal Department for the Environment, Transport, Energy and Communication initiated two activities:

- In order to enhance public participation in the discussion of disposal concepts, the Working Group "Energie-Dialog Entsorgung"[49] was set up in February 1998. This Working Group included NGOs as well as representatives of the relevant Swiss Government Departments, the nuclear industry, regulatory bodies and Nagra. It provided a forum for intense discussions (seven meetings were held, including a full-day closed meeting); however, no agreement could be reached on disposal issues. These discussions are documented in a final report (EDE 1998).
- An Expert Group on Disposal Concepts for Radioactive Waste (EKRA) was set up in June 1999. In their work, EKRA explicitly discussed societal needs when evaluating different options. The concept of monitored long-term geological disposal proposed by EKRA, which has also been included in the draft nuclear energy law (KEG 2001), was designed to consider both long-term safety and societal needs. This is discussed further in Chapter 2.

Finally, a decision-making process that ensures participation of all stakeholders is also of key importance. Existing legislation, as well as the draft nuclear energy law (KEG 2001), foresees such a participation, with extensive rights of objection, and will provide the framework for repository implementation.

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[47] RD+D: Research, development and demonstration.

[48] Iterative in this context means that each of the elements shown in Fig. 1.2-3 may be adapted as one moves from step *n* to step *n+1*, based on the results from step *n*.

[49] Energy Dialogue on Disposal (of radioactive waste).

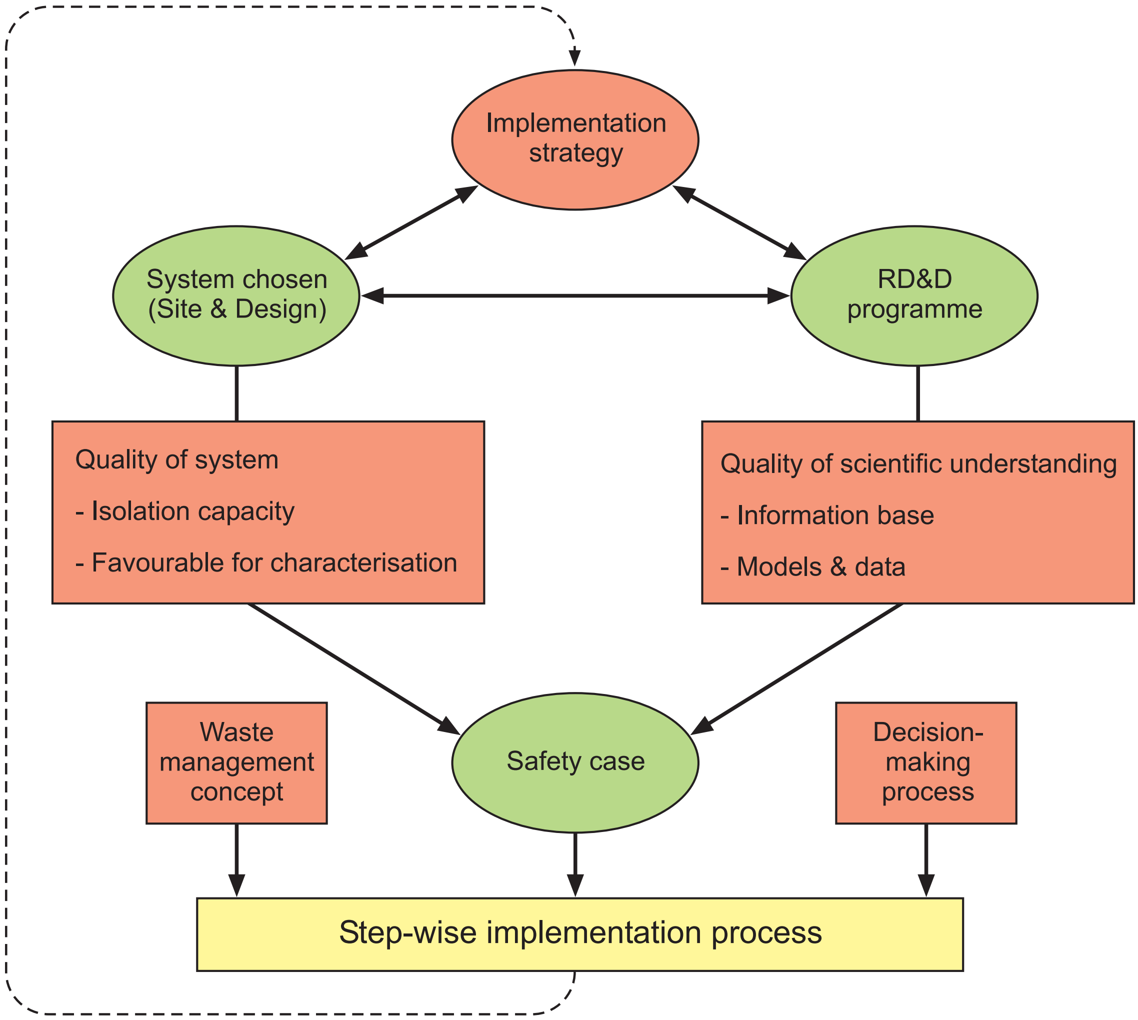

Fig. 1.2-3: Important elements of decision-making for the step-wise and iterative repository implementation process

The **implementation strategy** determines the **system** to be considered and includes the definition of a **site** and **system design**. The implementation strategy also determines what **RD+D** projects are needed for the chosen system. These represent essential input to the **safety case**. Societal confidence in the **concept** chosen for the long-term management of radioactive waste (in Switzerland: geological disposal) is crucial for a successful repository implementation. Finally, a **decision-making process** that ensures participation of all stakeholders is also of key importance and is foreseen in the existing legislation as well as in the draft nuclear energy law (KEG 2001).

1.3 Aims of Project Entsorgungsnachweis and of the safety assessment

The two main objectives of Project *Entsorgungsnachweis* are:

1. To demonstrate disposal feasibility of SF, HLW and ILW in the Opalinus Clay of the Zürcher Weinland in order to fulfil the requirements defined by the Federal Council in 1988 in its judgement of Project Gewähr 1985. This includes a demonstration that
   - a suitable geological environment for the repository exists (siting feasibility),
   - construction and operation of a repository is practicable in such an environment (engineering feasibility),
   - long-term safety from the hazards presented by the wastes is assured for such a repository (safety feasibility).
2. To provide a platform for discussion and a foundation for decision-making on how to proceed with the Swiss HLW programme. This includes a presentation of the key findings and results and a discussion of the underlying scientific basis. The excellent results obtained from the geological investigations led Nagra to propose to the Swiss Government to make a decision to focus future work for the waste management option "Disposal of SF / HLW / ILW in Switzerland" on the Opalinus Clay of the Zürcher Weinland[50]. Thus an additional objective is to provide the arguments to support such a decision. Project *Entsorgungsnachweis* represents a key milestone in a long and systematic site selection procedure, which was developed in close co-operation with the regulatory authorities and with experts from the Federal Government. The stepwise procedure of narrowing down from (i) seven potential sedimentary host rock options to one (Opalinus Clay) and (ii) from two large regional investigation areas for Opalinus Clay to the potential siting area in the Zürcher Weinland is documented in several Nagra reports (Nagra 1988, 1991 and 1994b) and summarised in HSK (2001). The choice of the Opalinus Clay as a host rock is based on its excellent barrier properties, its homogeneity and good explorability. The Zürcher Weinland was chosen because in that area the Opalinus Clay is situated at a suitable depth, with significant lateral extent and in a tectonically favourable and stable region (slightly compressive regime). Furthermore, in the Zürcher Weinland the overlying low-permeable layers provide a significant additional barrier to radionuclide migration, in contrast to the situation in western Switzerland (i.e. west of Baden), where these low-permeable layers are absent and the layer immediately above the Opalinus Clay is an aquifer (the Hauptrogenstein).

   Because of this second objective of Project *Entsorgungsnachweis*, in the safety report much emphasis is put on the evaluation of the level of confidence available for the option "Opalinus Clay in the Zürcher Weinland". This includes the identification of key phenomena with respect to long-term safety, an evaluation of the current understanding of the performance of these phenomena, and calculations of overall system performance for a broad spectrum of cases covering all realistically conceivable possibilities for the characteristics and evolution of the barrier system. In order to perform this analysis in a systematic manner, early in the report corresponding objectives and principles are developed for guiding the qualitative and quantitative evaluation of the chosen disposal system.

   Project *Entsorgungsnachweis* has thus also to provide the information to guide future work. Discussion on the content and the timing of future work, however, is expected to follow the reviews of the present project.

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[50] Disposal abroad is also an officially recognised option of the Swiss waste management strategy.

It is important to note that implementation of a deep repository for SF / HLW / ILW in Switzerland, the corresponding first licensing step and the political decision to site and construct such a repository (general licence and corresponding parliamentary decision) lie still far in the future (at the earliest around the year 2020). The current project, therefore, need not be, and cannot be expected to be, of the depth that would be needed for the licensing process that must precede repository implementation.

Additional objectives of the project are:

3. To provide input for overall waste management planning (including cost estimates), to form a benchmark for assessing design alternatives and inventory variants (spent fuel, HLW and various types of ILW) and to allow applied research and development priorities to be re-assessed to address any remaining safety-relevant issues and uncertainties.
4. To provide input for discussions of waste management issues with all stakeholders, and most importantly with the public. Such discussions can contribute significantly to building an understanding of these issues, which in turn can lead to a more constructive dialogue and perhaps increase public acceptance of waste management plans, including the implementation of repositories.

Based on these broad project-specific aims the following detailed aims of the safety assessment (this report) are derived:

1. To determine the suitability of the Opalinus Clay of the Zürcher Weinland as a host rock for a repository for SF / HLW / ILW from the point of view of long-term safety.
2. To enhance the understanding of the multiple safety functions that the proposed disposal system provides.
3. To assess the robustness of the disposal system with respect to remaining uncertainties and the effects of phenomena that may adversely affect the safety functions.
4. To provide a platform for the discussion of a broad range of topics related to repository development. More specifically, the findings from the safety assessment, together with those from the regulatory authorities' review thereof, will provide guidance for future stages of repository planning and development.

Operational phase safety is not treated in this report, but is addressed in the engineering feasibility report (Nagra 2002b) on a qualitative level. The present report is restricted to post-closure radiological safety issues.

1.4 Hierarchy of Project Entsorgungsnachweis reports and target audiences

Documentation of Project *Entsorgungsnachweis* is presented in a series of reports. At the highest level, there are three key technical project reports, which are primarily aimed at a technical audience (Swiss safety authorities, the Swiss scientific and technical community, technical bodies such as implementers and regulators in other countries, but also the technically interested non-specialist reader). These three reports are:

- a project report providing a synthesis of geological information on Opalinus Clay and on the geology of northern Switzerland and, specifically, of the region of the Zürcher Weinland (Nagra 2002a),
- a project report describing the design, construction, operation and closure of the proposed facilities (Nagra 2002b),

- a safety assessment report, which is divided into two parts: (i) the present report ("Main report" in Fig. 1.4-1), addressing long-term safety, and (ii) a report on models, codes and data (Nagra 2002c) that serves as a technical back-up for the main report.

The three project reports, in turn, are backed up by more detailed technical "Reference reports". This is indicated for the safety assessment in Fig. 1.4-1.

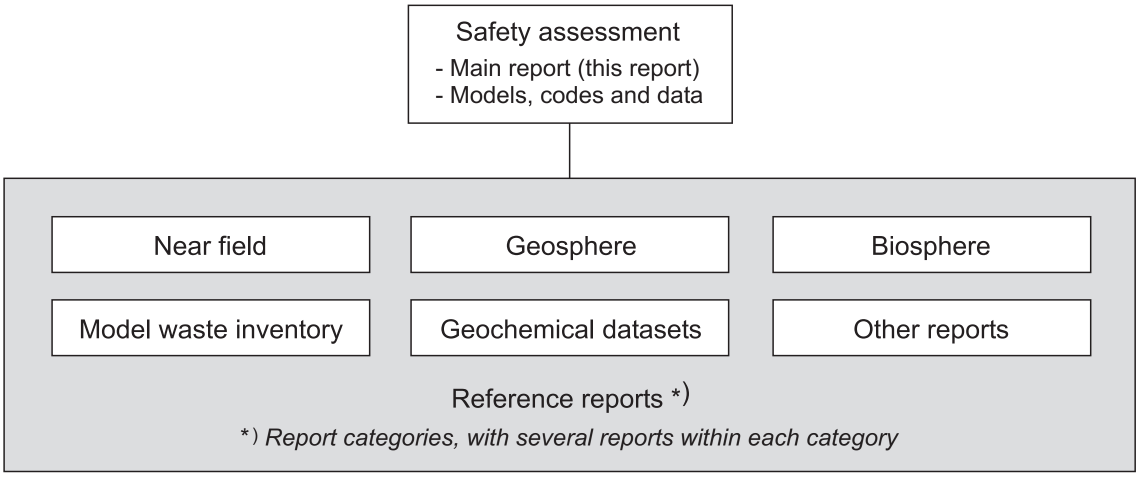

Fig. 1.4-1: Reporting structure for the safety assessment of Project *Entsorgungsnachweis*

The safety assessment report is divided into two parts to satisfy the two important requirements of transparency and traceability: The present report aims at pulling together as transparently as possible the arguments and analyses that make up the safety case, without, however, giving all the detailed formulae and data used. The Models, Codes & Data report (Nagra 2002c) should provide a traceable route that allows the interested reader to independently re-calculate the results shown in the present report. Furthermore, although the present report contains a summary of geological information relevant to the safety case, the reader who is looking for more in-depth geological information and justification for key geological data is referred to the geosynthesis report (Nagra 2002a).

The present report is broader than just a safety assessment and also summarises key information from other reports of Project *Entsorgungsnachweis* (see next section). This is for two reasons: On one hand the present report is the only one available in English and thus one of its roles is to provide the international community with an overall picture of the project; on the other hand it is felt that a comprehensive safety case should also provide key information from other disciplines that are of relevance to the overall project conclusions. However, in order to fully understand all the details the reader is also referred to the other project reports.

1.5 Organisation of this report

This report presents the arguments that together comprise the safety case, i.e. the case for the long-term safety of a repository for SF, HLW and ILW located in the Opalinus Clay of the Zürcher Weinland. Fig. 1.5-1 shows a simplified flowchart for the development of the safety case. It also serves as a guide through the report, with the chapters in which each topic is discussed indicated. In order to ensure that individual chapters are sufficiently self-contained to allow a rounded treatment of the topic in question, a certain degree of repetition has been deliberately introduced into the report.

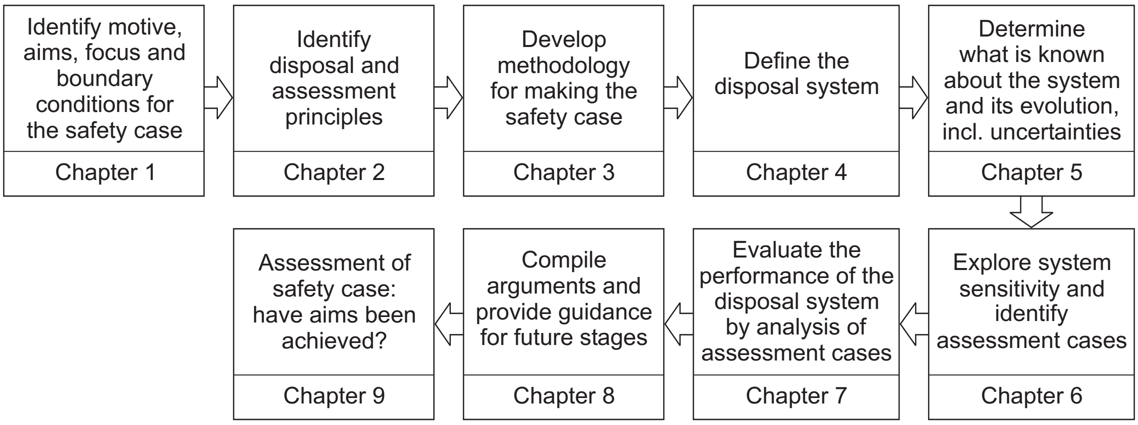

Fig. 1.5-1: Flowchart for the development of the safety case and guide through the safety assessment report, with the chapters in which each topic is discussed indicated

This flowchart is developed further in Chapter 3.

Chapter 2 is about guidance and principles for choosing the repository system and evaluating its long-term safety. The international view, the Swiss law and regulatory framework and regulatory guidance (including documentation by the regulator on their concept for evaluating the acceptability of Project *Entsorgungsnachweis*) are considered. These documents, all of which influence the approach to providing and analysing safety, are supplemented by additional guidance developed internally by Nagra based on its experience both in Switzerland and through its interaction with other organisations abroad. Finally, the safety functions of the repository system, which are central to the development of the safety case, are introduced and defined.

Chapter 3 introduces the methodology for making the safety case, i.e. the approach to the evaluation of long-term safety of the proposed repository. This includes a definition of the lines of argument, the approach for processing information, the iterative nature of the safety assessment process with its strong interaction with other disciplines, and the consideration of uncertainties.

Chapter 4 documents the characteristics of the proposed disposal system in the Opalinus Clay of the Zürcher Weinland at the time of repository closure. This includes a description of the wastes to be emplaced, the EBS and key features of the site (i.e. the host rock, the confining units and the aquifers). In addition, a brief description of the near-surface environment is also given.

Chapter 5 is a description of how the disposal system evolves with time after repository closure, taking into account the interactions of individual system components. This includes a discussion of waste-related processes for SF, HLW and ILW, the evolution of the engineered barriers and the evolution of the site. Besides describing the expected evolution of the disposal system, possible deviations from this course are also discussed. This is the basis for the development of the range of assessment cases that are quantitatively analysed in detail in this safety report.

Chapter 6 analyses the relative importance of the different features and phenomena, as well as the consequences of possible deviations from the expected evolution (sensitivity analysis). Based on a qualitative discussion, the key features and phenomena contributing to the safety functions are identified; these are termed "pillars of safety". Assuming first that the pillars of safety operate as expected, and then considering possible perturbations (based on Chapters 4 and 5 and selected using the insight gained in the sensitivity analysis), the assessment cases are identified.

Chapter 7 starts with a description of the conceptualisation of the assessment cases identified in Chapter 6. It then presents the results of the quantitative analyses of the assessment cases and evaluates the performance of the disposal system.

Chapter 8 is a synthesis of the main arguments and results. It presents the case for the long-term safety of the proposed repository. It includes a comparison of the assessment results with the requirements given in the Swiss regulatory guidelines and an evaluation of system understanding. It also discusses how the safety assessment provides a platform for the discussion of a broad range of topics related to repository development and how the findings from the safety assessment, together with those from the regulatory authorities' review thereof, will provide guidance for future stages of repository planning and development.

Chapter 9 presents the overall conclusions. It is argued that the evaluation of the Opalinus Clay of the Zürcher Weinland as a host rock for a repository for SF, HLW and ILW leads to a positive conclusion from the point of view of long-term safety, i.e. that the main objectives set out in Section 1.3 are fulfilled. Furthermore, it is concluded that a good system understanding has been developed, that the system is robust (i.e. uncertainties do not call safety into question) and finally that although a thorough analysis has been performed, no critical open issues have been identified.

Appendix 1 outlines the approach for choosing parameter values for deterministic and probabilistic assessment calculations, in **Appendix 2** key data characterising the disposal system are given, **Appendix 3** presents the background to the alternative safety and performance indicators discussed in this report, **Appendix 4** discusses the role and interaction of different groups in making the safety case, and **Appendix 5** gives the definitions of key terms.

2 Guidance and Principles for Choosing the Disposal System and Evaluating Safety

2.1 Aims and structure of this chapter

A number of sources of guidance and principles are taken into account in carrying out Project *Entsorgungsnachweis*. International consensus exists on many of the overall principles governing such disposal projects. More specific guidance and principles are set out by the Swiss regulatory authorities and in the draft of the revised Swiss Nuclear Energy Law ("Kernenergiegesetz" or short: KEG). Principles relevant to the choice of disposal system and the carrying out of safety assessments have also been derived internally by Nagra, based on the experience and information acquired in the course of studies and investigations carried out both in Switzerland and internationally. As illustrated in Fig. 2.1-1, the aim of this chapter is to present the various sources of guidance and principles and to indicate how these are considered in the approaches taken in Project *Entsorgungsnachweis.* The issues discussed in this report relate to post-closure safety of a repository, thus in this chapter the main emphasis is on developing guidance and principles with respect to post-closure safety. However, because the evaluation of post-closure safety is strongly interlinked with the process of repository implementation, the objectives related to stepwise implementation are also discussed.

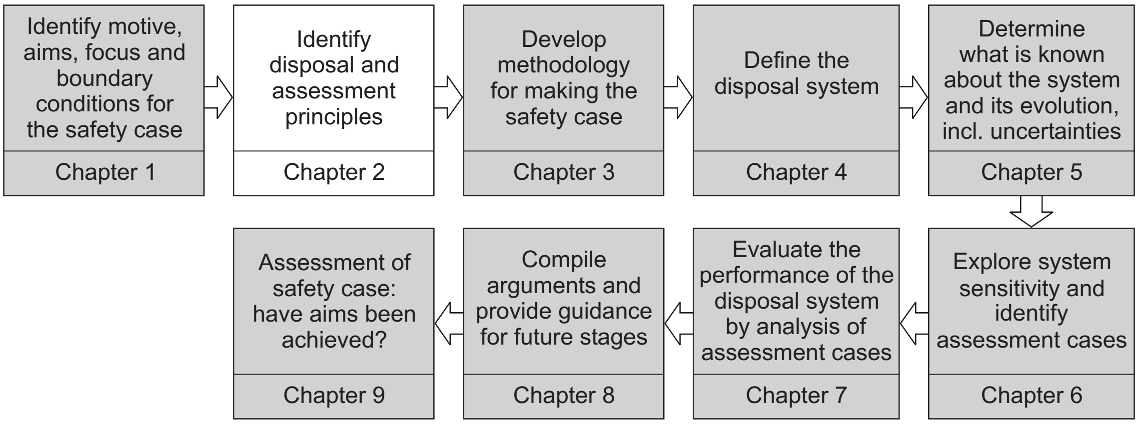

Fig. 2.1-1: The role of the present chapter in the sequence of tasks involved in developing the safety case

The sources of guidance and principles are described in Section 2.2. Guidance and principles related to the overall objectives of waste disposal in Switzerland, to the siting, design and stepwise implementation of the repository and to safety assessment are discussed in Sections 2.3, 2.4 and 2.5, respectively. In each of these sections, the international view is summarised, followed by guidance and principles from Swiss regulations and elsewhere. Because information comes from different sources, Sections 2.3 to 2.5 are rather heterogeneous. Therefore, in Section 2.6, the major findings are grouped and summarised in the categories "Principal objectives of deep geological disposal: Security and long-term safety" (2.6.1), "Objectives related to the system" (2.6.2), "Objectives related to stepwise implementation" (2.6.3) and "Assessment principles" (2.6.4). These objectives and principles provide the framework for producing the safety case documented in this report.

2.2 Sources of guidance and principles

2.2.1 International guidance

The strategy of disposal of long-lived radioactive wastes in geological repositories and the corresponding technical and scientific aspects have been discussed for many years within international fora such as the IAEA and NEA and in all countries that have nuclear energy programmes or use radioactive materials in the fields of medicine, industry and research[51]. A large amount of information is available that provides guidance on the disposal of long-lived radioactive wastes and the assessment of its safety. It is internationally accepted that geological disposal is an ethically and environmentally sound waste management solution (NEA 1995a), for which considerable practical experience has been developed (NEA 2000b), which can be flexibly developed (NEA 2001a), and the safety of which can be evaluated (NEA, IAEA & CEC 1991). Geological disposal is in accord with the internationally established principles for waste management (IAEA 1995) and the terms of the Joint Convention on the Safety of Spent Fuel and on the Safety of Radioactive Waste Management (IAEA 1997). International standards have been developed that are applicable to geological disposal (IAEA 1989, IAEA 2002a) which are in turn consistent with the basic safety standards for radiological protection (IAEA 1996). The ICRP has also given guidance on the application of its general recommendations (ICRP 1991) to disposal of solid radioactive waste including geological disposal (ICRP 1985, ICRP 1997, ICRP 1998).

2.2.2 The Swiss HSK-R-21 regulatory guideline

As described in Section 1.2.2, the Swiss Federal Nuclear Safety Inspectorate (HSK) is responsible for supervising safety and radiation protection in nuclear installations. HSK is also charged with providing the licensing authorities with reviews of projects for nuclear installations including radioactive waste repositories. The Federal Commission for the Safety of Nuclear Installations (KSA) is responsible for evaluating projects for such installations and for submitting statements on the licensing applications and the reviews from HSK.

When evaluating and reviewing projects, HSK and KSA base their analyses on the relevant provisions in the Federal Laws on atomic energy and radiation protection and associated decrees, as identified in Section 1.2. HSK and KSA have provided guidance which is specific to facilities for the disposal of radioactive waste in the guideline document HSK-R-21 (HSK & KSA 1993). The guideline sets out the overall objective of disposal, the principles to be applied and safety requirements, as well as giving explanatory comments on the scope of, and expectations from, safety analyses. Key parts of HSK-R-21 are summarised in later sections of this chapter.

Compliance with guidance and principles set out in HSK-R-21 is regarded as an essential part of any future licence application for a repository in Switzerland. Although Project *Entsorgungsnachweis* represents only one stage in the long stepwise process towards possible licence applications that could lie decades in the future, it will be subject to a formal, rigorous review by the Swiss regulatory authorities. Thus, the siting and design of the repository and the safety assessment are, even at this stage, evaluated for compliance with the regulatory guidelines.

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[51] References given in Section 2.2 are merely representative, and are not intended to be comprehensive.

2.2.3 HSK's assessment concept for the siting feasibility project

Early in 1999, HSK published an outline of their assessment concept for the siting feasibility project (HSK 1999). The document begins with an extended definition of the term "demonstration of siting feasibility[52]", the issue that was judged after Project *Gewähr* to be still open (see Section 1.2.4). HSK points out that this demonstration represents a major intermediate step on the path towards possible repository implementation, and acknowledges that this step is to be based on investigations from the surface; i.e. no underground facilities are required at this stage. The document also lists a number of basic requirements on the siting area; these are all addressed in the geosynthesis report (Nagra 2002a).

2.2.4 The revised Swiss Nuclear Energy Law

The specific features of the draft of the revised Swiss Nuclear Energy Law that most directly affect the disposal strategy are that the law explicitly requires disposal in a geological repository, which must be monitored for some time before final closure (KEG 2001). This requirement is based on the concept of "monitored long-term geological disposal" as proposed by EKRA (EKRA 2000) and as described in Section 2.4.4.

2.3 Discussion of general objectives and principles of geological disposal

2.3.1 The choice of geological disposal as a waste management option

In common with other countries that have to deal with long-lived radioactive wastes, Switzerland attaches great importance to their safe management. The objectives of waste management are the protection of humans and the environment from the hazards associated with radioactive substances and the securing of the fissile material contained in spent nuclear fuel from any undesirable misapplication (safeguards). For safeguards purposes, in the different phases of the repository (operational phase, observation phase, post-closure phase) different measures will be necessary. Such measures are being discussed and are under development within the framework of international organisations (see, e.g., IAEA 1998).

Waste can be stored safely in interim storage facilities for several decades. The safe operation of interim storage facilities, however, requires continuous monitoring and maintenance activities. Effective monitoring and maintenance presume the continued economic and political stability of our society, as well as the availability of the necessary technical know-how; however, these cannot be indefinitely guaranteed. For these reasons, very long-term (e.g. over centuries) storage in such facilities cannot be considered as a substitute for the final disposal of waste in a deep geological repository (see, e.g., EKRA 2000).

As discussed in Section 2.2.1, disposal in geological repositories conforms to international recommendations. It is also the currently adopted waste management strategy in several countries, including those that, in addition to geological disposal, are pursuing the options of partitioning and transmutation and long-term interim storage. The view that geological disposal is the only method available today for ensuring long-term safety without placing undue burdens on future generations is also expressed in a recent publication of the US National Research Council (NRC 2001). Geological disposal may be viewed as the final step in the management of radioactive waste by a "concentrate and confine" strategy, which is the preferred strategy over the alternative of "dilute and disperse" in the environment (IAEA 1995). As pointed out,

[52] "Standortnachweis" in German.

however, in recent ICRP recommendations (ICRP 1998), the "concentrate and confine" strategy leads to a concentration of the hazard and eventually some release of radionuclides to the environment is inevitable. The aim of geological disposal within a "concentrate and confine" strategy is to delay any releases for long enough to ensure that radioactive decay will greatly reduce the level of activity and also to ensure that residual releases, should they occur, can never be at concentrations harmful to humans or the environment.

The discussions about the concept of deep geological disposal also include ethical considerations and the need to develop stakeholder confidence in geological disposal. These are not considered in detail here; the reader is referred to NEA (1995a), NEA (2000c) and EKRA (2000) for an extended discussion. The key ethical principles are incorporated directly into the Swiss guidelines described in the following section.

2.3.2 Regulatory principles and protection objectives in Switzerland

Principles

The draft KEG (KEG 2001) states that radioactive waste should be placed in a deep geological facility ("geologisches Tiefenlager") that, following a phase of monitoring, should be closed and sealed, at which point it becomes a final disposal facility ("Endlager"). The overall objective of such a facility is, according to the Swiss regulatory guideline HSK-R-21 (HSK & KSA 1993),

"... to eliminate radioactive waste in such a way that:

- human health and the environment are protected in the long term against the ionising radiation from the waste, and
- no undue burdens are imposed on future generations."

This objective is achieved by the implementation of the following principles (quoted from HSK-R-21), which are compatible with, for example, the IAEA Safety Principles (IAEA 1989):

Principle 1:	The additional radiation dose to the population resulting from radioactive waste disposal shall be low.
Principle 2:	When disposing of radioactive waste, environmental protection shall be ensured in a way that living species will not be endangered and the use of mineral resources will not be unnecessarily restricted.
Principle 3:	The risks to humans and the environment arising from radioactive waste disposal shall not, at any time in the future and anywhere abroad, exceed the levels which are permissible today in Switzerland.
Principle 4:	The long-term safety of a repository shall be ensured by a system of multiple passive safety barriers.
Principle 5:	Any measures which could facilitate surveillance and repair of a repository or retrieval of the waste shall not impair the functioning of the passive safety barriers.
Principle 6:	The provisions for radioactive waste disposal are the responsibility of the present society which benefits from the waste-producing activities and shall not be passed on to future generations.

Protection objectives

HSK-R-21 gives three specific Protection Objectives that a repository should be shown to satisfy. Protection Objectives 1 and 2 are derived from Principles 1, 2 and 3 (above) and relate to the long-term safety of the repository for its expected evolution and for less likely situations, respectively. Protection Objective 3 is derived from Principles 4, 5 and 6 and relates to avoiding undue burdens on future generations.

Protection Objective 1:	The release of radionuclides from a sealed repository subsequent upon processes and events reasonably expected to happen shall at no time give rise to individual doses which exceed 0.1 mSv per year.
Protection Objective 2:	The individual radiological risk of fatality from a sealed repository subsequent upon unlikely processes and events not taken into consideration in Protection Objective 1 shall, at no time, exceed one in a million per year.
Protection Objective 3:	After a repository has been sealed, no further measures shall be necessary to ensure safety. A repository must be designed in such a way that it can be sealed within a few years.

2.4 Discussion of guidance and principles for siting, design and staging

2.4.1 International guidance

International guidance on siting has been issued by the IAEA. The IAEA broad siting guidelines for a deep waste repository issued in 1994 (IAEA 1994a) give some points of guidance with respect to site selection. With respect to repository design, there is international consensus that the safety of geological repositories should be ensured by a system of passive barriers, which is capable of functioning reliably without any supervision or maintenance measures. This does not necessarily mean that the possibility of monitoring is excluded; the disposal system must, however, be capable of providing a sufficient degree of safety without relying on institutional control and monitoring (IAEA 1989 and 2002a).

Based on Swiss and international experience, the following three broad categories of safety functions are essential for the performance and safety of a repository: (i) isolation of the wastes from the human environment, (ii) long-term confinement and radioactive decay, within the disposal system, of a large fraction of the activity initially present in the wastes, and (iii) attenuation of releases to the environment for the small fraction of radionuclides that do not decay to insignificance within the disposal system. These safety functions are discussed in more detail in Section 2.6.2.1.

There is also broad consensus that implementation of a repository should preferably be done in a stepwise manner, see, e.g., NEA (1999a and 2002) and NRC (2001). This is also the case in Switzerland, see discussions in Chapter 1. Such a stepwise approach provides several opportunities for review and should allow for modifications and changes to future plans and – if this were required – should also allow the reversal of decisions in the course of the implementation process. This would include – in the most extreme case – the retrieval of emplaced wastes.

2.4.2 Designing the system for robustness

In Switzerland, the construction, operation and, in particular, closure of a repository for SF, HLW and ILW are likely to take place only in the relatively distant future and, to date, not all issues and uncertainties are fully resolved. Indeed, as for all technical systems, some uncertainties concerning their future behaviour will remain unresolved and decisions must be taken in the face of such inherent uncertainties. The approach taken in waste disposal to counter this problem is to choose the disposal system such that no open issues are likely to undermine safety and our ability to demonstrate safety, and none of the uncertainties that need to be addressed in order to demonstrate safety are likely to place undue demands on the site characterisation and research and development programmes. Such a disposal system is termed robust. The principles involved in choosing a robust disposal system are summarised in Fig. 2.4-1.

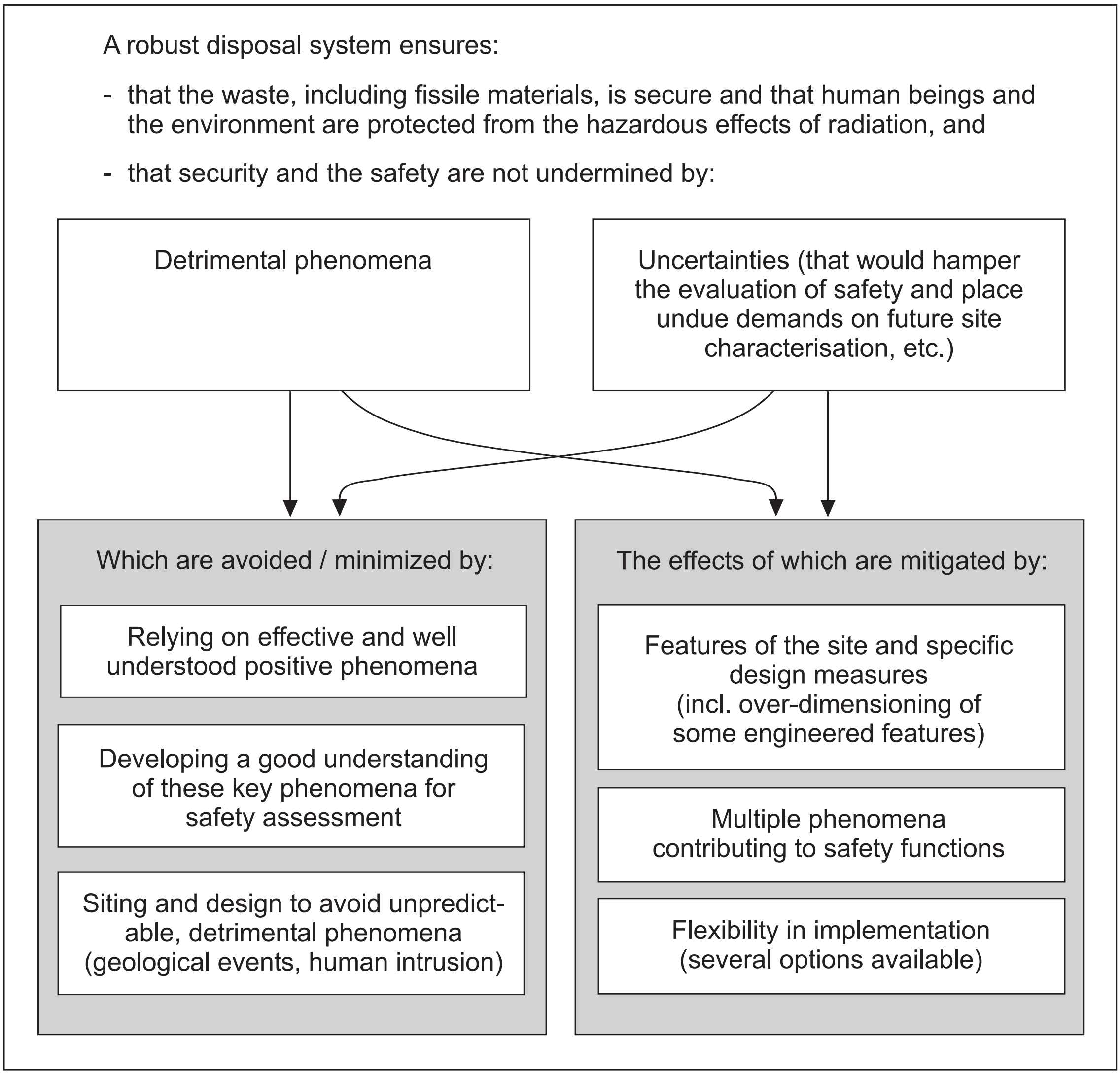

Fig. 2.4-1: The principles involved in choosing a robust disposal system

Where possible, phenomena that would be detrimental to safety, as well as sources of uncertainty that would hamper the evaluation of safety are avoided or reduced in magnitude, likelihood or impact. This can be achieved by choosing a system that is not prone to unpredictable detrimental phenomena and that provides safety via processes that are (i) well understood, (ii) for which the necessary detailed information for safety assessment can readily be acquired, and (iii) that are likely to operate adequately, even in the event of unfavourable developments and a pessimistic interpretation of uncertainties. This includes the choice of a design in which the materials used are – in addition to providing sufficient performance – also compatible in their interactions and in which wastes that are potentially unfavourable in their interactions are segregated. Because of the limited thickness of the Opalinus Clay the design of the facility should be chosen in such a way that the disturbances of the host rock in the vertical direction are small in order not to unnecessarily reduce the vertical migration distance. Furthermore, the site should, if possible, be amenable to characterisation, to the extent required for safety assessment, at an early stage of the project, i.e. it should be readily explorable.

Not all detrimental phenomena and uncertainties can be avoided, but their impacts can be reduced by a suitable choice of natural (geological) and engineered barriers, with multiple and to some extent independent phenomena contributing to safety. The availability of design options to address specific safety concerns also contributes to robustness.

2.4.3 Swiss regulatory guidance on siting and design in HSK-R-21

Overall system design

The requirement for a system of multiple passive barriers is laid down in the Swiss HSK-R-21 guideline (HSK & KSA 1993), which is discussed in Section 2.3.2 (Principle 4 and Protection Objective 3).

Criteria for site properties: Explorability and predictability and absence of resources

HSK-R-21 gives no specific criteria for site properties that relate to long-term safety, although a general preference is expressed for "... sites at which conditions are easy to predict, both in time and space, for the purpose of the safety assessment".

The Regulatory Principle 2 discussed in Section 2.6.2 implies that sites are preferred that lack exploitable mineral resources since this helps to keep resources available to future generations and to minimise the likelihood of inadvertent intrusion.

Requirements on the engineered barrier system: The need for initial complete containment

HSK-R-21 is intended for application to disposal facilities for all types of radioactive waste and it specifies protection objectives for the repository system as a whole. This allows the system components to be adapted to specific characteristics of the waste and the geological environment under consideration. HSK-R-21 states, however, that

"... in the case of HLW disposal, there is a particularly high hazard potential during the initial phase (around 1000 years). During this phase complete containment of the radionuclides within the repository should be aimed at."

This implies that it is necessary to design waste canisters with sufficient corrosion resistance to prevent groundwater reaching the wastes for at least this time.

Safety enhancing measures

HSK-R-21 states:

"Even if compliance with Protection Objectives 1 and 2 is demonstrated, the radiological consequences from the repository have to be reduced by appropriate measures as far as feasible and justifiable with current status of science and technology. However, owing to the uncertainties involved in determining potential radiation exposure, no quantitative optimisation procedure is required."

This implies that subjective judgement can be used in designing the individual barriers that ensure safety, provided that the overall requirements are met and that there are indeed multiple contributors to safety.

In HSK-R-21 it is further stated:

"Protection Objective 3 requires that no post-closure measures shall be necessary in order to ensure long-term safety of a repository. It is thus to be assumed in the safety analysis that future generations will not take measures to protect themselves from the exposure to radionuclides released from the repository. The applicant should, nevertheless, take measures to preserve information on the repository, including its location, design and wastes which have been emplaced. This is intended to reduce the likelihood of an unintentional intrusion into the repository."

The requirement for passive safety is critical for the choice of the system concept. Information preservation is not a critical issue at the present development stage, but will be important at licensing and especially at closure.

2.4.4 The EKRA concept of monitored long-term geological disposal

As discussed above, the draft KEG requires disposal in a geological repository, which must be monitored for a certain time before closure (KEG 2001). This requirement is consistent with the concept of "monitored long-term geological disposal" that was formulated in some detail in the report of the government advisory group, EKRA (EKRA 2000). This concept combines the need for passive safety, as ensured by geological disposal, with a cautious, stepwise approach to implementation, intended to address societal demands and also technical uncertainties. The approach involves an extended period of monitoring, during which retrieval of the waste is relatively easy, and the emplacement of a representative fraction of the waste in a pilot facility to test predictive models and to facilitate the early detection of any unexpected undesirable behaviour of the system should this occur. Thus, opportunities are provided for review and possible reversal of decisions, including the retrieval of emplaced wastes.

EKRA's concept of monitored long-term geological disposal foresees the following system elements (see Fig. 2.4-2):

- the main facility;
- the test facility;
- the pilot facility;
- a tunnel system connecting the different system components, including tunnels for the near field and environmental monitoring programmes.

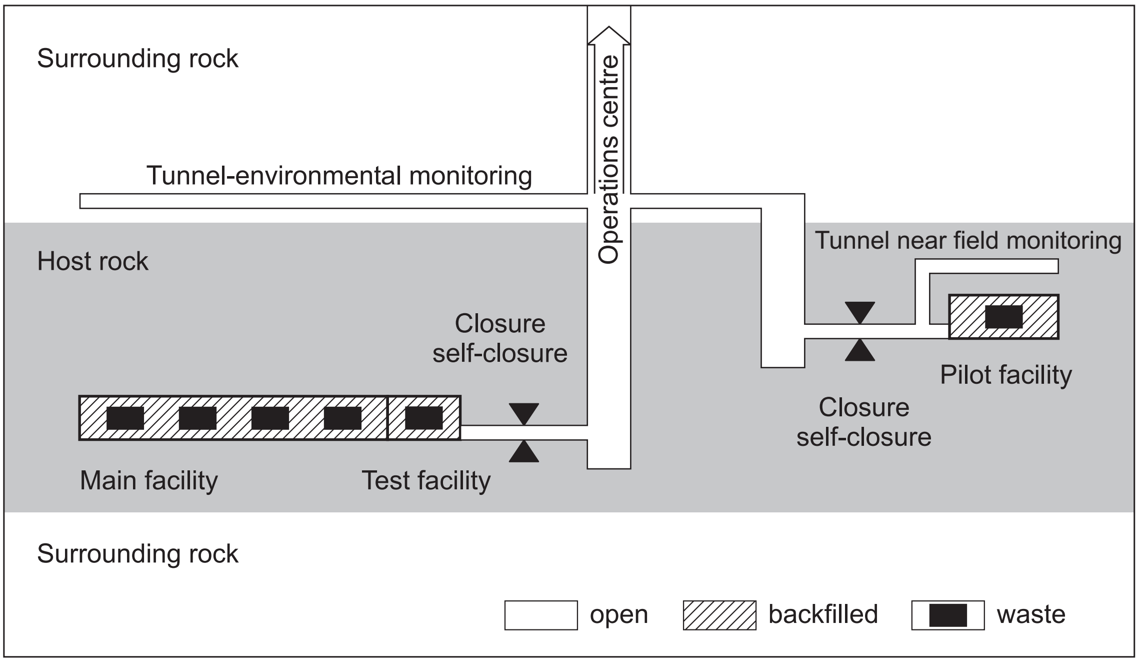

Fig. 2.4-2: Overview of the key components of the concept of monitored long-term geological disposal (taken from EKRA 2000)

Note that this is a schematic representation of a generic facility. A repository for SF, HLW and ILW in Opalinus Clay that is consistent with the concept of monitored long-term geological disposal is described in Chapter 4.

The function of the *test facility* is to provide the information required before the main facility can start operation, in so far as this information is not already available from site investigations. This includes information relevant to post-closure safety as well as information for construction and operation of the main facility. The test facility is not a single facility, but is rather a series of experiments at different locations, and may be regarded as a site-specific underground rock laboratory (URL). These *in-situ* experiments are complemented by laboratory programmes and by generic (non-site-specific) investigations, for example in other URLs.

The aim of the *pilot facility,* which contains a small but representative fraction of the waste, is to provide information on the behaviour of the barrier system and to check predictive models. It also serves as a demonstration facility that provides input for decisions regarding closure of the entire facility. In addition, it should allow early detection of any unexpected and undesirable system evolution. It is foreseen that the pilot facility will be backfilled and sealed without delay, in order that it is in a state that represents as closely as possible the foreseen final state of the repository. It will, however, be monitored from a series of observation boreholes. The possibility also exists for carrying out (destructive) sampling. The pilot facility and its access routes are arranged in such a way that the facility can continue to be monitored for a long period after closure of the main facility. As is the case for the main facility, the pilot facility should not represent a significant risk if, during a time of crisis, it should be abandoned without the access route being closed according to plan.

The majority of the waste is emplaced in the *main facility.* All of the components of the main facility are designed in such a manner that, together with the geological barrier provided by the host rock, they will ensure passive safety in the post-closure phase, i.e. after closure of the whole facility. The layout of the main facility has also to ensure a high degree of robustness in the extreme event that the facility is abandoned during the extended monitoring phase (e.g.

during a time of crisis). In particular, immediately following emplacement of the waste, backfilling of voids and sealing of the main facility has to ensure that passive long-term safety is achieved even if closure of the access routes is not carried out according to plan. In addition, provisions may be made for easy and rapid closure of the access tunnels (EKRA discusses a design that would close automatically if the repository were abandoned). Furthermore, according to EKRA the design of the repository has to allow for reasonably easy retrieval of the wastes if this were deemed necessary.

Both monitoring and retrievability are also being studied in many countries and have been (and still are) discussed in several international fora (see, e.g., IAEA 2000, EC 2000, NEA 2001a, IAEA 2001a, EC 2002). The requirement of the draft KEG for a period of monitoring before repository closure and the corresponding detailed recommendations of EKRA are considered in the proposed layout of the underground facilities for Project *Entsorgungsnachweis*, in particular by the inclusion in the design of a test facility and a pilot facility. This is described in Chapter 4 and in more detail in the Facilities Report (Nagra 2002b).

2.4.5 Swiss guidance on a stepwise approach to repository implementation

The existing Swiss law and also the draft KEG foresees a stepwise approach to repository implementation. Major steps are defined by the several licences that are required for implementing a repository and for its eventual closure (see Section 1.2.2). Even before licensing, there are several milestones with corresponding safety reports (see Section 1.2.4). The stepwise approach is also addressed in HSK-R-21:

"The applicant has to submit a safety analysis at each stage of the licensing procedure (general, construction, operating and closure licence). Safety relevant information on the repository system obtained from preliminary investigations should be supplemented by ongoing investigations during the construction and operation of the repository. The safety analysis for the post-closure phase should be refined in accordance with the improved knowledge of the repository system."

Thus, the iterative nature of the safety assessments is very important: For each major decision-point a new safety assessment has to be prepared which will increase in detail as the project progresses and more details become available. A key issue in such a stepwise approach is how the level of confidence of all stakeholders changes through the steps. How soon does the information base become sufficiently reliable that no doubts remain about the fundamental feasibility? For a stepwise procedure in which early confirmation of basic feasibility is a goal, this indicates the importance of the explorability of a system: A geological environment for which a reliable dataset can be obtained already in the early phases is advantageous.

After construction of the repository periodic re-evaluation continues. The concept of monitored geological disposal proposed by EKRA provides a good framework for such re-evaluations which will be partially based on experimental evidence from the pilot facility.

However, in such a stepwise approach with multiple decision points it is important that enough flexibility exists so that modifications and changes can be made if considered to be necessary at one of these decision-points. Thus, a project has to provide flexibility with possibilities for changes.

2.5 Discussion of guidance and principles for safety assessment

2.5.1 International guidance on safety assessment

Before the selected disposal system can be implemented, its long-term safety has to be clearly demonstrated. This principle is also laid down in the IAEA Joint Convention on the Safety of Spent Fuel and on the Safety of Radioactive Waste Management (IAEA 1997), to which Switzerland is a signatory. There has been a long history of international co-operation in developing the approaches and methods for analysing the long-term safety of geological disposal. This work is documented in numerous publications of the IAEA and the NEA (e.g. IAEA 1981, 1983 and 1985, NEA, IAEA & CEC 1991, NEA 1997, 2000d and 2002). A key conclusion, stated in many of the documents, is that although absolute proof of safety is not possible for the long timescales considered, a reasonable expectation of long-term safety is a prerequisite for the implementation of a disposal system, and that safety assessment (or, more generally, the development of a convincing safety case) is the procedure by which this is tested. Evaluations of the radiological consequences of a disposal system made in the course of a safety assessment are always subject to uncertainty. In practice, conclusions with respect to safety are based not on an exact prediction of the actual future evolution of the system, but rather on the consideration of a broad range of representative cases addressing different assumptions regarding the possible characteristics and evolution of the system.

In addition to the guidance gained by the study of safety documents from international organisations, much can be learned from the numerous national safety assessments made, submitted and reviewed over the past years.

2.5.2 The need to produce a robust safety case

One of the key roles of a safety assessment is to show how a disposal system could evolve over the course of time and to demonstrate that sufficient safety is achievable for the system envisaged and that the regulatory requirements can be met. The resulting safety case needs to be robust, i.e. it needs to address all reasonably conceivable issues, and be based on arguments for safety that are reliable. Features of the safety assessment that promote robustness of the safety case are discussed further in Chapter 3.

2.5.3 Swiss regulatory guidance on safety assessment in HSK-R-21

HSK-R-21 provides broad guidance on how the safety assessment should be conducted. The following issues are covered.

2.5.3.1 Predictive modelling and predictions into the distant future

The timescale of interest for dose and risk calculations (as required to satisfy Protection Objectives 1 and 2) is not specified by HSK-R-21. Rather,

"... dose and risk calculations should be carried out for the distant future, at least for the maximum potential consequences from the repository, despite the uncertainties related to the condition of the biosphere and the existence of a population".

HSK-R-21 acknowledges that there is inevitable uncertainty in model calculations and the further into the future predictions are made, the greater the uncertainty. The implementer has to show what processes and events could affect the repository over the course of time and then to derive and evaluate potential evolution scenarios from these.

2.5.3.2 Excluded processes and events

HSK-R-21 identifies some classes of processes and events that need not be analysed in a safety assessment. In particular, processes and events need not be considered if they have either (i), an extremely low probability of occurrence, or (ii), considerably more serious non-radiological consequences (e.g. a large meteor uncovering the repository). Assessment of intentional human intrusion into the repository (for example, to recover the waste) is also explicitly excluded.

2.5.3.3 Treatment of uncertainty

When calculating dose or risk, the ranges of uncertainty for relevant data due to incomplete knowledge of the properties of the system and due to incomplete understanding, or simplified modelling should be estimated and variability in space and time should be considered. Conservative assumptions can be used to treat uncertainty and to simplify variability for modelling purposes.

2.5.3.4 Verification of codes and validation of models[53]

HSK-R-21 defines verification as

"... demonstrating that a given computer code is error-free"

and validation as

"... providing confidence that a computer code used in safety analysis is applicable for the specific repository system".

Each computer code used in the safety analysis has to be verified. In addition, it should be shown that all the models used are applicable for the specific repository system (validation), taken both individually and as an overall model chain.

2.5.3.5 Treatment of the biosphere

There is particularly large uncertainty in the evolution of the surface environment (the biosphere) and the habits of humans living in the future. HSK-R-21 comments that:

"... Dose calculations for the distant future are not to be interpreted as effective predictions of radiation exposures of a defined population group. They are, in fact, much more in the nature of indicators for evaluating the impact of a potential release of radionuclides into the biosphere."

"… For such calculations, reference biospheres and a potentially affected population group with realistic, from the current point of view, living habits should be assumed."

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53 In recent years there has been considerable international debate on the meaning of verification and validation in the context of models for post-closure assessment, e.g. see NEA (1999b). The above definitions are, however, adopted in this report.

2.5.4 Specific issues in safety assessment

Some of the issues raised in Section 2.5.3 on regulatory guidance for safety assessment involve topics that have been discussed for many years and have a strong judgmental component with respect to their implementation in the analysis of long-term repository performance. In the present section, these and other particularly subtle issues are discussed at more length.

2.5.4.1 Timescales of concern

HSK-R-21 states that the doses and risks related to a sealed repository "shall at no time" exceed the values set in Protection Objectives 1 and 2 (Section 2.3.2). This leads to a potential difficulty: The better the disposal system is, the further into the future will any significant release occur, and the more difficult will it be to estimate doses and risks with confidence when they eventually arise. It is important, therefore, to understand the evolution of the hazard posed by radioactive waste so as to determine over what time period the geological disposal system should perform and, also, to focus safety assessments on the time period during which the waste poses an unusual hazard.

A useful measure of the potential hazard of radioactive material is the radiotoxicity index (RTI), which is defined in Appendix 3. Fig. 2.5-1 shows the RTI of the total radionuclide inventory of the three waste categories to be emplaced in the proposed repository as a function of time[54]. This is compared with the RTI of the natural radionuclides contained in 1 km^3 of Opalinus Clay ("OPA") and with that of a volume of natural uranium ore corresponding to the volume of the SF / HLW / ILW emplacement tunnels. In the latter case, three uranium concentrations (uranium ore grades) are considered. These are 3 %, which is the average uranium concentration of the small uranium ore body of La Creusa, Switzerland, 8 %, which is a representative value for the Cigar Lake uranium deposit in Canada, and 55 %, which is near the upper end of observed concentrations in uranium ore bodies (see also Appendix 3). Fig. 2.5-1 shows that after one million years, the radiotoxicity of even the most toxic waste, the spent fuel, has dropped to well below that of a volume of natural uranium ore sufficient to fill the SF / HLW / ILW emplacement tunnels.

Another way of putting the potential hazard of spent fuel in perspective is to compare it with that of the quantity of natural uranium that was used to produce the fuel. This was done in a study by Hedin (1997), where it is noted that it takes about 8 tonnes of natural uranium to produce 1 tonne of nuclear fuel suitable for PWR/BWR reactors. Fig. 2.5-2 shows that, after about 300 000 years, the radiotoxicity of spent fuel has dropped to that of the natural uranium from which it was produced (assumed to be in equilibrium with its daughters)[55].

Both Figs. 2.5-1 and 2.5-2 indicate that the timescale over which the spent fuel presents a hazard that needs special attention is of the order of about one million years. Thus, it is considered that the disposal system should provide effective isolation of the spent fuel from the human environment also for about this period of time. Note that this does not imply a necessity for complete containment within the waste packages; the surrounding geological media are part of the isolation system. The remainder of this report will compile arguments to support the statement that a disposal system providing the required isolation capability for at least one million years is, in fact, feasible. The analyses are complemented with arguments that the good performance of the system will continue beyond one million years for at least another few

[54] Time is measured from the end of waste emplacement (see Chapter 4 for details).

[55] The production of one tonne of enriched uranium, suitable for PWR/BWR fuel, from eight tonnes of natural uranium also results in radioactive products which are removed during uranium milling and extraction, and seven tonnes of depleted uranium produced during uranium enrichment. These materials are the responsibility of their respective producers.

million years. Additionally, more qualitative illustrations are provided for the different evolutionary stages of the repository for the period beyond these few million years. However, these illustrations are considered to be of far less importance than the detailed analyses for the period when the waste still presents a higher hazard and they are, therefore, kept rather short.

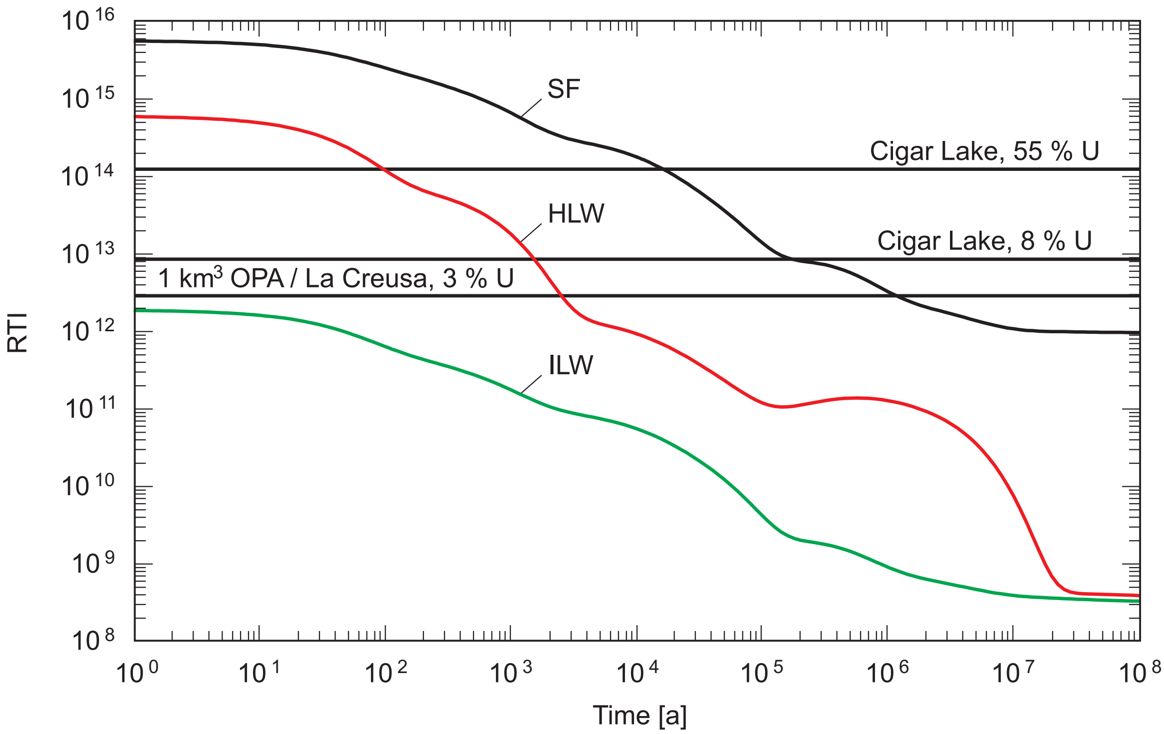

Fig. 2.5-1: Radiotoxicity index (RTI) of spent fuel (SF), vitrified high-level waste (HLW) and long-lived intermediate-level waste (ILW) as a function of time

As a comparison, the RTI of the natural radionuclides contained in 1 km^3 of Opalinus Clay ("OPA") and in a volume of natural uranium ore corresponding to the volume of the SF / HLW / ILW emplacement tunnels are also given. In the latter case, three uranium concentrations are considered. These are 3 %, which is the average uranium concentration of the small uranium ore body of La Creusa, Switzerland, 8 %, which is a representative value for the Cigar Lake uranium deposit in Canada, and 55 %, which is near the upper end of observed concentrations in uranium ore bodies. Note that the levels for 1 km^3 of Opalinus Clay and 3 % U coincide.

2.5.4.2 The role and treatment of the biosphere

The key role of the barrier system (i.e. the EBS together with the host rock) is to limit radionuclide release to the biosphere, and the performance of the barrier system is an "adjustable parameter" within certain bounds via siting and design. In contrast, the role of the biosphere in a typical safety assessment is to provide a "measuring stick" to convert radionuclide releases from the barrier system into a dose, and the corresponding scale is only marginally adjustable. In addition, the possible evolutions of the barrier system for a well-sited repository and a well-chosen host rock can be bounded with reasonable confidence over about one million years into the future, and its performance can be evaluated with a reasonable level of reliability over this period (Fig. 2.5-3). The same is not true for the factors that must be taken into account for evaluating the meaning of any radionuclide releases from the host rock and for modelling radionuclide transfer in the biosphere and uptake by man; i.e. processes in the surface environment and, especially, human behaviour that affect the radiological exposure pathways. For these factors, the timescale over which reliable statements can be made are of the order of only a few hundred years in the case of the natural environment and human populations and, perhaps, a few tens of years in the case of individual human behaviour. Moreover, much of the uncertainty concerning these factors is irreducible. In spite of this, there are approaches available that allow an evaluation of a range of future scenarios regarding the potential harm that might result from a repository and to measure this against safety standards.

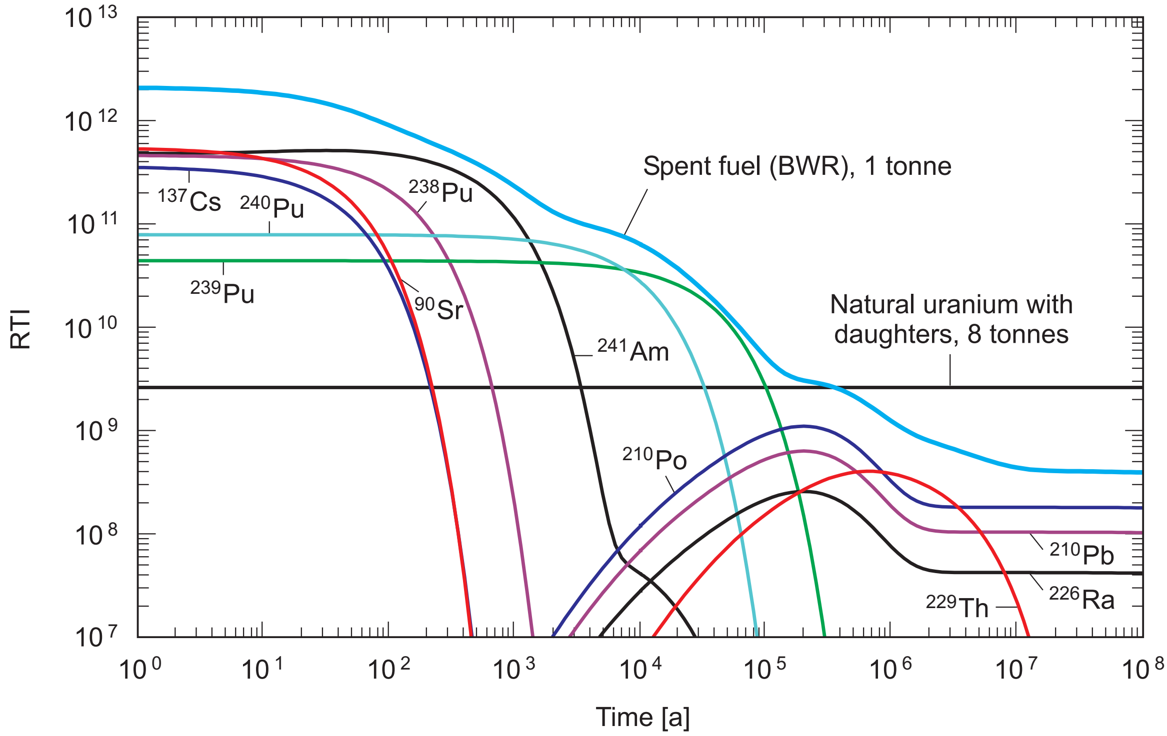

Fig. 2.5-2: Radiotoxicity index of 1 tonne of representative Swiss spent fuel (BWR) with a burnup of 48 GWd/t and of 8 tonnes of natural uranium

To fabricate 1 tonne of fuel, about 8 tonnes of natural uranium are required. The different fractions in the nuclear fuel cycle include, in addition to the natural uranium and the fuel, the depleted uranium and the uranium daughters that are separated in the uranium mill (not shown in this figure). Adapted from Hedin (1997).

The role and treatment of the biosphere in long-term safety assessment in the light of this uncertainty and taking into account its limited importance relative to the barrier system has been discussed extensively in international fora. The international consensus that has developed suggests that a reasonable approach is to separate the assessment of the biosphere from that of the barrier system, as proposed, for example, by a NEA *ad hoc* working group (NEA 1999c) and to develop a range of credible illustrations for the biosphere, thereby exploring the uncertainty related to the biosphere (Sumerling et al. 2001). This is the approach taken in the current safety assessment. For the vast majority of the release calculations (i.e. those that focus on the barrier system) the same stylised biosphere situation is chosen to convert releases into dose. The sensitivity of calculated doses to uncertainty related to the biosphere is then investigated in stand-alone calculations for a broad range of biosphere situations.

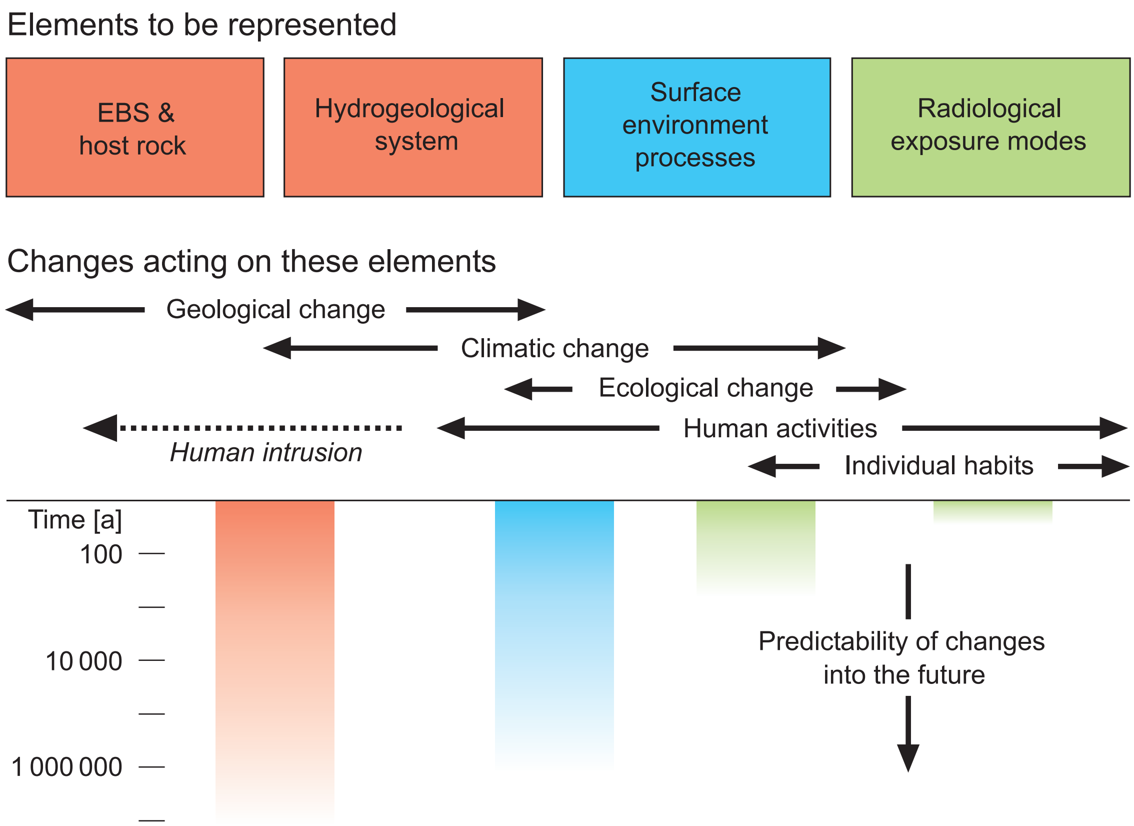

Fig. 2.5-3: Schematic illustration of the limits of predictability of the different elements of a geological disposal system (NEA 1999c)

2.5.4.3 Treatment of future human behaviour

In the present assessment, doses are calculated to average members of hypothetical critical groups chosen to be representative of the individuals or population groups that might receive the highest doses as a result of the presence of the repository and assuming current living habits and diets. The calculated doses are regarded as indicators of the level of protection, rather than estimates of actual future doses (see ICRP Publication 81 (ICRP 1998) for specific recommendations for the definition of critical groups in relation to solid radioactive waste disposal).

2.5.4.4 Treatment of future human actions

The issues surrounding the assessment of future human actions affecting a deep geological repository are, in some respects, similar to those surrounding the treatment of the biosphere. Any statement about the actions that humans might take in the far future is largely speculative. The relationship between the assessment of such actions and the assessment of the quality of a site and a design is problematic because human actions have the potential to create exposure paths that by-pass the normal safety functions of the repository. These issues have been discussed internationally by a NEA Working Group (NEA 1995b), within the NEA IPAG exercise (NEA 2000d) and, most recently, within IAEA Specialists' Meetings (IAEA 2001b). The ICRP has also given guidance for the assessment of future human actions (ICRP 1998). The international consensus on these issues guides the treatment of future human actions in this safety assessment as follows:

- The decisions to dispose of waste in a deep geological repository, and to site the repository in a host rock that does not have any other obvious resource potential, are made to reduce the likelihood of inadvertent human interference.
- It is acknowledged that human technology and society will change over the timescales of relevance for repository safety assessment. These changes are unpredictable. To limit speculation, it is assumed that human technological capabilities and societal patterns observed today, and in the past, provide a reasonable model to assess the safety of the repository in the future. Thus, only future human actions that could be undertaken with present-day technology are considered.
- Exploratory drilling through the repository horizon can be considered as an illustrative case for the purpose of testing the resilience of the disposal system to such an event. The main concerns are whether there is significant degradation of the long-term performance of the disposal system following such an event, and the possible dose to human individuals dwelling near to the site. The ICRP has suggested that doses to such individuals could be viewed against the levels at which intervention would be considered, i.e. in the range 10 to 100 mSv (ICRP 1998). Individuals involved in a drilling that actually intercepts a spent fuel or HLW canister could be subject to very high doses, but this is an unavoidable hazard arising from the decision to concentrate and contain the waste, and should be judged against the generally very high level of protection that the disposal system offers.
- In accordance with HSK-R-21, which states that intentional human intrusion into the repository need not be discussed in a safety assessment, there is no consideration of deliberate acts of intrusion into the repository undertaken with knowledge of the repository location and content. Such actions, which could include retrieval of the waste or other materials and malicious acts, are considered to be the responsibility of future generations[56].

2.6 Summary and conclusions: Objectives and principles

The key points from Chapters 1 and 2 that determine how a repository system should be developed and then analysed to determine whether it fulfils its safety goals are recapitulated in the following sections and then summarised in two concluding overview tables (Tab. 2.6-1 and 2.6-2). The points are collected, grouped and summarised into the following broad categories:

- Objectives of geological disposal
- Objectives related to the system
- Objectives related to stepwise implementation
- Assessment principles.

2.6.1 Principal objectives of deep geological disposal: Security and long-term safety

The disposal system that is selected must ensure that the waste, including fissile materials, is secure and that human beings and the environment are protected from the hazardous effects of radiation.

56 However, all necessary actions will be taken to prevent malicious acts within the framework of safeguards.

Disposal in a deep geological repository, with any access routes backfilled and sealed, intrinsically favours both security and safety. Wastes emplaced in such a repository are kept remote from human beings and the surface environment, and the repository affords protection against both deliberate and inadvertent human intrusion, as well as surface phenomena.

Deep geological disposal is also considered to be in accordance with ethical principles.

2.6.2 Objectives related to the system

The disposal system considered in the present study is selected according to the following objectives, which are based on legal requirements, on regulations, on internal considerations by Nagra and on international consensus.

The system objectives are divided into the following broad objectives:

- **Safety and robustness of the system** – The disposal system has to ensure that the waste, including fissile materials, is secure and that human beings and the environment are protected from the effects of radiation for the time period of about one million years during which the wastes (and especially spent fuel) pose an unusual hazard. The system has to be robust: its performance may not be unduly affected by residual uncertainties that may result from the spectrum of realistically conceivable possibilities regarding its future evolution. For such a system, a convincing safety case can be made, despite existing uncertainties.
- **Reduced likelihood and consequences of human intrusion** – Measures should be taken to minimise the risk of human intrusion. Should intrusion nevertheless occur, the repository should be designed in such a manner that degradation of performance after intrusion is limited.

A disposal system that considers these broad system objectives will provide a number of safety functions that are of key relevance to long-term safety and security. These safety functions can be considered as functional objectives of a disposal system; they are summarised in Section 2.6.2.1, below.

The broad objectives related to the system discussed above are achieved by adherence to the following principles, which are divided into three categories, namely:

- Principles related to the overall system (Section 2.6.2.2),
- Principles related to siting (Section 2.6.2.3), and
- Principles related to repository design (Section 2.6.2.4).

2.6.2.1 Functional objectives: The safety functions of the disposal system

A disposal system sited and designed according to the system objectives set out above performs a number of functions relevant to long-term security and safety. These functions are:

- **Isolation from the human environment** – The safety and security of the waste, including fissile material, is ensured by placing it deep underground, with all access routes backfilled and sealed, thus isolating it from the human environment and reducing the likelihood of any undesirable intrusion and misapplication of the materials. Furthermore, the absence of any currently recognised and economically viable natural resources and the lack of conflict with future infrastructure projects that can be conceived at present reduces the likelihood of inadvertent human intrusion. Finally, appropriate siting ensures that the site is not prone to disruptive events and to processes unfavourable to long-term stability.

- **Long-term confinement and radioactive decay within the disposal system** – Much of the activity initially present decays while the wastes are totally contained within the primary waste containers, particularly in the case of SF and HLW, for which the high integrity steel canisters are expected to remain unbreached for at least 10 000 years. Even after the canisters are breached, the stability of the SF and HLW waste forms in the expected environment, the slowness of groundwater flow and a range of geochemical immobilisation and retardation processes ensure that radionuclides continue to be largely confined within the engineered barrier system and the immediately surrounding rock, so that further radioactive decay takes place.
- **Attenuation of releases to the environment** – Although complete confinement cannot be provided over all relevant times for all radionuclides, release rates of radionuclides from the waste forms are low, particularly from the stable SF and HLW waste forms. Furthermore, a number of processes attenuate releases during transport towards the surface environment, and limit the concentrations of radionuclides in that environment. These include radioactive decay during slow transport through the barrier provided by the host rock and the spreading of released radionuclides in time and space by, for example, diffusion, hydrodynamic dispersion and dilution.

2.6.2.2 Principles related to the overall system

Achievement of the two objectives formulated in Section 2.6.2 (Safety and robustness and reduced likelihood and consequences of human intrusion) depends upon satisfying a series of principles, as described below.

Safety and robustness is achieved by:

- **Multiple passive barriers** – The repository and its environment must ensure long-term safety through a system of multiple passive barriers providing multiple safety functions. The barriers should be provided both by the repository itself (the EBS) and the geological setting and a range of different mechanisms should contribute to safety. The use of passive barriers means that, once sealed, no further measures are required in order to ensure safety. Measures to facilitate surveillance and repair of the repository or retrieval of the waste are not excluded, but the functioning of the barriers must not be impaired by such measures.
- **Stability and longevity** – The location of the wastes in emplacement rooms deep underground in a tectonically stable situation and the use of long-lived materials for key components of the engineered barriers that are compatible in their interaction is essential to safety and robustness.
- **Avoidance of and insensitivity to detrimental phenomena** – The repository may, in principle, be adversely affected by a range of phenomena. A system is thus preferred that avoids such phenomena. In practice, however, not all detrimental phenomena can be avoided, and systems are thus also preferred with properties that mitigate the effects of detrimental phenomena. These issues are further discussed with respect to siting and design.
- **Reduced likelihood and consequences of human intrusion,** i.e. the reduction of likeli-hood of human intrusion and the mitigation of degradation of system performance in the case of intrusion is achieved by:
  - **Preservation of information** – Measures must be taken to ensure that information regarding the purpose, location, design and contents of the repository are preserved so

that future generations are made aware of the consequences of actions they may choose to take that might affect the performance of the disposal system.

- **Avoidance of resource conflicts** – A site should be chosen so that any foreseeable resource conflict is avoided and so that there is no conflict with future infrastructure projects that can be conceived at present, in order to minimise the likelihood of future inadvertent human intrusion.
- **Compartmentalisation and solidification of wastes** – A repository design, where in the case of intrusion only a small part of the repository is affected, is beneficial. This can be achieved by compartmentalisation; i.e. each SF / HLW waste package forms an isolated compartment with no shortcut to the next one and waste emplacement tunnels are widely spaced. A limited size of the ILW emplacement tunnels has a similar effect. Solidification of the wastes ensures that only the small fraction of radionuclides released from the waste form that are in solution can be transported to the surface instantaneously.

2.6.2.3 Principles related to repository siting

The primary safety-relevant requirements on the geological setting for a repository are that it provides a stable and protected environment for the engineered barriers, promoting their longevity, and also an effective barrier to the migration of any radionuclides that might escape from the engineered barriers. It should furthermore be possible, in practice, to demonstrate convincingly that the geological setting meets these requirements. Thus, a site should conform to the following principles:

- **Stability** – A stable and protected environment is provided by a site that is tectonically stable (e.g. with a low incidence of faulting) and that has a low rate of uplift and associated erosion.
- **Favourable host rock properties** – Suitable hydrogeological and geochemical properties are essential both for the satisfactory performance and evolution of the engineered barriers and for the performance of the host rock as a barrier to migrating radionuclides. Equally important are the host rock properties that ensure engineering feasibility of constructing, operating and closing the repository.
- **Avoidance of detrimental phenomena** – The geological setting of any repository may, in principle, be adversely affected by geological events and repository-induced effects. Sites are thus preferred that are geologically quiet and whose safety-relevant properties are largely unaffected by the presence of the repository.
- **Insensitivity to detrimental phenomena** – In practice not all detrimental phenomena can be avoided, and sites are thus also preferred with properties that mitigate the effects of detrimental phenomena and perturbations.
- **Explorability** – It should be possible to characterise the selected site reliably to the extent required for a convincing safety assessment. Simplicity of a site and homogeneity of the host rock can make characterisation easier and more reliable.
- **Predictability** – The geological history of the site should be well understood, at least over the timescale of interest to safety assessment. This constrains the possible paths of evolution of the site over a similar period in the future. Predictability is also favoured by the quietness of the geological setting.

2.6.2.4 Principles related to repository design and implementation

The repository is required to provide a set of engineered barriers that act in a complementary manner with the natural geological barrier to contribute to the safety functions of "confinement" and "attenuation" introduced in Section 2.6.2.1. The principles of predictability, avoidance of detrimental phenomena and insensitivity to detrimental phenomena apply to the engineered barriers as well as to the geological setting.

The repository design and implementation should conform to the following principles:

- **Confinement and attenuation** – The engineered barriers have to contribute through their physical and geochemical properties to the key safety functions of the repository system; i.e. the confinement of radionuclides and, for those nuclides that are released, attenuation of releases.
- **Initial complete containment for SF and HLW** – The design should ensure substantially complete containment of the radionuclides associated with SF and HLW for a period of a thousand years or more, consistent with the HSK-R-21 regulatory guideline. The purpose of this principle is to provide a straightforward primary argument for safety in the period during which the radiotoxicity of the wastes is highest, and transient phenomena such as the elevated temperatures around the repository due to radiogenic heating and the resaturation of the repository mean that radionuclide release would be more difficult to predict. Detailed radionuclide migration calculations during this transient phase can thus be circumvented by sufficiently long complete containment. Furthermore, complete containment is effective in mitigating the effects of inadvertent intrusion in the period when SF and HLW are most toxic.
- **Redundancy to ensure insensitivity to uncertainties** – A cautious approach should be adopted in the choice of barriers and the dimensioning of particular components of the EBS. There may be barriers or processes that only make a significant contribution to safety if some parts of the system do not perform according to expectations. The existence of such barriers or phenomena, however, contributes to the overall safety case.
- **Avoidance of and insensitivity to detrimental phenomena** – Through an adequate choice of materials and a careful design detrimental phenomena and perturbations are avoided as far as possible; e.g. the materials used for the disposal system have – besides providing adequate performance – to be compatible in their interaction, incompatible wastes are segregated and specific design and administrative measures are taken to ensure sub-criticality. In practice, however, not all detrimental phenomena can be avoided, and designs are thus preferred with properties that mitigate the effects of detrimental phenomena and perturbations.
- **Reliability of implementation** – The site and design should be selected such that the properties that favour safety can be relied upon to exist, in a sufficiently unperturbed form, when the repository is implemented, without placing excessive demands on novel engineering technology and allowing for reliable quality assurance. On the other hand, the project should provide enough flexibility that new developments in science and technology can be taken into account. This also applies to possible future changes of waste properties (e.g. due to new conditioning / packaging technologies). In order to facilitate the structured interaction between waste producers and repository implementer the implementation of preliminary waste acceptance criteria and a corresponding waste acceptance process ("acceptance in principle") may be advantageous; it also ensures that no unsuitable waste products will arise and that the necessary information on the wastes will be available.

- **Reliability of closure of the repository** – The repository must be designed in such a way that it can be sealed within a few years (Protection Objective 3 in HSK-R-21, see Section 2.3.2).
- **Predictability** – In order to favour the predictability of their evolution, the engineered structures of the repository should employ a small number of simple, well-understood materials, with materials that interact in a complex manner with each other, or with the geological setting, avoided as far as possible.

Principles related to the safe construction and operation of the repository must also be considered, but this is not the subject of the present report.

2.6.3 Objectives related to stepwise implementation

The stepwise repository implementation process, as discussed in Chapter 1, contains several elements that should be fulfilled to ensure successful progress. The first of these is the quality of the information base available at a given stage that has to be sufficient to support a decision to proceed to the next stage. This requirement is most easily met with a system that is characterised by proven good explorability and quiet geology providing ample space for the suggested repository, with good predictability of the future development of the potential siting area. Another key element for a successful programme is the participation of all stakeholders in the decision-making process, which is in accordance with existing legislation as well as with the draft nuclear energy law (KEG 2001). This also includes the provision of a monitoring phase after waste emplacement, with the possibility to retrieve the waste, as proposed by EKRA (EKRA 2000) and as reflected in the KEG (KEG 2001).

Key objectives in stepwise implementation are:

- **Commitment to systematic learning** – During the sequential steps, there is an increase in the body of available information, including scientific, technical, societal, institutional, and operational knowledge. Needs and questions to be addressed are made explicit at the outset. Information gained will be accepted and incorporated into the available knowledge base. One needs enough information to proceed with confidence to the next step, even if the ultimate goal is some way off. Project *Entsorgungsnachweis* is, in this respect, an important, but not yet final or irreversible step on the way to definitive siting and realisation. This commitment to systematic learning includes a QA system that ensures that the relevant information is made available to, and used in, future steps.
- **Involvement of stakeholders** – The stepwise approach requires the development of a safety report at each milestone. Such a safety report and other related documents (e.g. reviews) provide an excellent platform for interaction with all stakeholders, notably the regulator and policy makers, but also the scientific community and the public; and they provide an opportunity for feedback.
- **Possibilities for modification** – The approach has to provide flexibility with respect to new findings in the process of implementation. A site with spatial reserves that allows for optimal allocation of the emplacement rooms and the availability of several design options provide flexibility. As discussed in Chapter 1, the Swiss waste management strategy also provides alternative options with respect to siting areas and host rocks. The design of the repository, conforming to the concept of monitored long-term geological disposal as proposed by EKRA, also allows the reversal of decisions in the course of the implementation process, including the retrieval of emplaced wastes.

- **Possibilities for monitoring** – The design of the repository according to the concept of monitored long-term geological disposal, as proposed by EKRA, provides opportunities for monitoring and review and possible reversal of decisions in the course of the implementation process.
- **Implementation of specific measures to ensure security during the different phases of a repository** – Specific security and safeguards measures will be implemented to ensure security in the operational phase, during the observation phase and also in the post-closure phase. Such measures are currently being discussed in international fora and are under development by international organisations. Because implementation of the Swiss HLW repository is still far away no detailed discussion of these issues is considered to be necessary in Project *Entsorgungsnachweis* and security issues are therefore not further discussed.
- **Reliance on simple, well-understood and reliably characterised components** – Sites and engineered barrier systems are preferred that are simple and that can be characterised reliably to the extent required for a robust safety assessment even at an early stage of the repository. Site characterisation should preferably be possible by surface-based techniques (good explorability) and the system of engineered barriers should rely on well understood materials. This should ensure adequate predictability of the overall system. Construction of the repository and the emplacement of the EBS must be feasible with proven technology. For development of the repository adequate quality assurance measures should be used to ensure reliable implementation.

2.6.4 Assessment principles

A key role of safety assessment is to gain quantitative and qualitative information that allows one to judge the safety of the proposed system and to provide guidance on the future repository development programme. The assessment has to show how a disposal system could evolve over the course of time and discuss what level of safety can confidently be expected. The resulting safety case needs to be robust, i.e. it needs to address all reasonably conceivable issues, and be based on arguments for safety that are reliable.

These aims can be achieved if the following broad principles are obeyed:

- **Focus of the safety case** – The focus depends upon the stage that has been reached in the overall programme. For the Swiss HLW programme the focus is currently on assessing the feasibility of disposal and on providing guidance of future work by providing a platform for discussion, e.g. with respect to Nagra's proposal to focus future work on the Opalinus Clay in the potential siting area in the region of the Zürcher Weinland. Besides assessing the expected level of safety, the main emphasis is on the evaluation of the reliability and robustness of the system: Is there enough confidence that no outstanding issues exist with the potential to undermine the safety case and thus to justify focusing future work on the Opalinus Clay of the Zürcher Weinland?
- **Sufficient scientific understanding** – Scientific understanding is the core of a safety case and thus, it has to be convincingly shown that the understanding is adequate for the stage of the programme, i.e. that it is appropriate for the focus of the safety case.
- **Systematic and defined method** – The safety case has to be developed in a systematic manner based on a clearly defined method. This also contributes to transparency and traceability.

- **Multiple arguments for safety** – The discussion of safety should not only look at compliance with regulatory criteria but also other arguments for safety should be provided.
- **Documentation** – The development of the safety case and its results should be documented in manner that provides transparency and traceability.

For each of these broad principles more detailed principles are defined, which are again based in part on regulations and in part on internal considerations by Nagra, guided by international consensus. These are described in the next sections.

2.6.4.1 The focus of the safety case

The focus of the safety case defines where to put the main emphasis of the study and also indicates the level of rigour required in the different areas. It thus defines the scope and boundary conditions of the assessment. The focus depends on the stage of the repository programme, and moves from assessing broad feasibility and providing guidance towards a more detailed and precise consideration of dose and risk impacts as the programme progresses. The focus of the safety case must be defined by the assessor. It also needs to be explained by putting the safety report in context with the overall repository development programme.

- **Assessment of feasibility** – In the current phase of the Swiss programme it is important to assess the feasibility of the project and to discuss its robustness. No excessive demands are placed on the level of detail for quantifying uncertainties and for those uncertainties that are difficult to quantify with respect to their likelihood of occurrence, bounding assessments are considered to be acceptable. Thus, the main aim is to be comprehensive with respect to the identification of sources of uncertainty.
- **Guidance for future stages** – At the current early stage of the repository programme, an important role of safety assessment is to provide a platform for the discussion of a broad range of topics related to repository development. More specifically, the findings from the safety assessment, together with those from the regulatory authorities' review thereof, will provide guidance for future stages of repository planning and development. In order to provide an adequate platform for discussion, in the safety report much emphasis is put on the evaluation of the level of confidence available for the option "Opalinus Clay of the Zürcher Weinland". This includes the identification of key phenomena with respect to long-term safety and an evaluation of current understanding about the performance of these phenomena, as well as calculations of overall system performance for a broad spectrum of cases covering all realistically conceivable possibilities for the characteristics and evolution of the barrier system. These quantitative analyses are complemented with qualitative evaluations.

2.6.4.2 Scientific understanding

Scientific understanding is the basis for each safety analysis and determines which phenomena can be relied upon in developing the safety case.

- **Documentation of the scientific basis** – The scientific basis for the key assumptions of the safety analysis needs to be clearly described and discussed. This also includes the referencing to key reports.
- **Evidence for key phenomena** – The safety assessment will allow identification of the phenomena most important to safety. For these phenomena the scientific evidence for their reliable operation needs to be discussed.

- **Resolution of open issues** – The assessment of the information base for describing and quantifying key phenomena and the level of confidence in their reliable operation will indicate if there are any outstanding issues that have the potential to undermine the safety case. If such issues exist, it is important to assess the possibilities for their satisfactory resolution in the course of the project.

2.6.4.3 Systematic and defined method for conducting the analyses

Principles and objectives relevant to the carrying out of the analyses[57] are:

- **Systematic approach to information collection, treatment and abstraction** – The assessment has to rely on a systematic approach that ensures that no important issues are overlooked and that their treatment (including abstraction) is done in an adequate manner.
- **Ensurance of completeness** – All reasonably foreseeable possibilities for the characteristics and evolution of the disposal system must be considered in developing safety case arguments, and the various sources of bias must be recognised when conducting and interpreting the analyses. Intentional human intrusion, and processes and events that are either extremely unlikely or have considerably more serious non-radiological consequences are, however, excluded, in accordance with the HSK-R-21 regulatory guideline.
- **Rigour in consideration and treatment of uncertainty** – Although, as far as possible, all potential sources of uncertainty must be considered, the degree of detail required in quantifying uncertainty will depend on the stage of the programme, and will increase as the proposed site and design become more firmly fixed. In the current early phase of the project bounding analyses are in many cases considered to be appropriate.
- **Irreducible uncertainties** – A stylised approach is adopted for the modelling of the evolution of the biosphere and the nature of future human behaviour and actions, on account of the largely irreducible and unquantifiable uncertainties associated with predictions, even over relatively short timescales.
- **Development and validation of models and databases** – A range of measures must be adopted to ensure that the models and databases developed and applied in the analyses are suitable for their intended purpose, in accordance with the requirement for validation in the HSK-R-21 regulatory guideline.
- **Verification of codes** – Each computer code used to perform analyses must be verified, in accordance with the HSK-R-21 regulatory guideline.
- **Internal and external reviews** – All important issues are subject to internal and/or external review.
- **Timescales to be considered in the assessment** – The safety of the disposal system must be assessed for as long as the waste poses a significant hazard. Quantitative analyses must be carried out into the distant future, as required by the HSK-R-21 guideline. The limits to the applicability of models, particularly at distant times, must, however, be recognised when interpreting results. In this analysis, as in Nagra's previous safety assessment for HLW (Nagra 1994a), the main emphasis is on the period up to one million years. However, the analyses are complemented with arguments that the good performance of the system will continue beyond one million years for at least another few million years. Therefore, the quantitative calculations are carried out for the period up to 10 million years although it is

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[57] In addition to the principles and objectives mentioned below, for completeness' sake it should be mentioned that all activities in carrying out a safety assessment are performed under strict quality assurance measures.

recognised that at times so far in the future some of the assumptions may no longer hold. Additionally, in Section 5.2.2.3 a more qualitative discussion is provided for the different evolutionary stages of the repository that may arise in the period beyond a few million years.

2.6.4.4 Multiple arguments for safety

Safety will be assessed using several indicators and arguments:

- **Compliance with regulatory criteria** – Compliance with regulatory criteria is essential for judging safety. For Project *Entsorgungsnachweis*, the criteria are defined in HSK-R-21 (Protection Objectives 1 and 2, see Section 2.3.2).
- **Complementary safety indicators** – In addition to the criteria defined in HSK-R-21, other indicators will be used for judging the safety of the system. These include the radiotoxicity of the wastes compared with naturally occurring radiotoxicity, the spatial distribution of radiotoxicity as a function of time, a comparison of radiotoxicity fluxes from the repository system with naturally occurring radiotoxicity fluxes and radiotoxicity concentrations in the host rock due to the repository compared with natural radiotoxicity concentrations. The characteristics of these alternative indicators are such that they give an appropriate indication of the performance of the system and are less sensitive to the uncertainties in surface conditions than dose calculations.
- **Existence of reserve FEPs** – Reserve FEPs are a qualitative argument for safety because their existence indicates that in reality performance will be positively affected by phenomena that have been excluded from the current safety assessment.
- **No outstanding issues with the potential to compromise safety** – The safety assessment will allow the identification of those phenomena that are key to the safety of the repository. The evidence available that provides confidence in their reliable operation will show if the argument can be made that no outstanding issues exist that have the potential to undermine the safety case.

2.6.4.5 Principles relevant to the documentation of the safety case

The documentation of the safety case should be guided by the (in some respects, conflicting) principles of transparency and traceability. In consideration of this requirement, and as discussed in Chapter 1, the safety assessment report is divided into two parts; i.e. this report, which aims at a transparent presentation of the safety case, and the Models, Codes and Data report (Nagra 2002c) which aims to provide traceability of the calculations that support the safety case:

- **Transparency** – The arguments and analyses that make up the safety case, and the process by which these are derived, must be readily understood. Too many details can result in a loss of transparency. This is why the present report aims at pulling together the arguments and analyses that make up the safety case without giving all the detailed formulae and data used in the evaluation of the assessment cases.
- **Traceability** – The documentation of the arguments and analyses that make up the safety case should provide the information required for a reviewer, for example, to reproduce key calculations, to understand the reasoning leading to particular model assumptions and to trace the source of parameter values employed in the analyses. The Models, Codes and Data report (Nagra 2002c) is designed to provide traceability in this sense.

2.6.5 Summary tables

Tab. 2.6-1 and 2.6-2 summarise the most important of the hierarchy of objectives and principles described throughout Section 2.6.

Tab. 2.6-1: Objectives and principles related to system and staging

Objectives	**Overall disposal principles**	
Objectives of geological disposal	• passive safety and security through deep geological disposal	
Objectives related to system	**Principles related to site**	**Principles related to design**
Safety and robustness	• multiple passive barriers to provide for safety • multiple safety functions provided by barrier system • stability and longevity of barrier system • avoidance of and insensitivity to detrimental phenomena • predictability	
	• stability • favourable host rock properties • explorability	• confinement and attenuation • initial complete containment • redundancy • reliability of implementation • reliability of closure
Reduced likelihood and consequences of human intrusion	• preservation of information	
	• avoidance of resource conflicts	• compartmentalisation and waste solidification
Objectives related to stepwise implementation		
Commitment to systematic learning	• information gained is incorporated into available knowledge base	
Involvement of stakeholders	• staging with multiple reviews and feedback by different stakeholders	
Possibilities for modifications	• flexibility for siting	• flexibility: design alternatives • possibility for easy retrievability during operation and observation phase
Possibilities for monitoring	• concept of monitored geological disposal	
Security in all phases of the repository	• application of recognised security and safeguards measures	
Reliance on understood and reliably quantified components	• simplicity • predictability	
	• explorability	• reliance only on proven technology and materials • application of adequate QA measures

Tab. 2.6-2: Objectives and principles related to safety assessment

Objectives	Principles
Focus of safety case	• assessment of feasibility and discussion of robustness • guidance for future stages
Sufficient scientific understanding	• documentation of scientific basis • existence of evidence for key phenomena • possibilities for resolution of open issues that are of importance to system performance
Systematic and defined method	• systematic approach to information collection & treatment/abstraction • ensurance of completeness • rigorous consideration and treatment of uncertainties • treatment of irreducible uncertainties (biosphere, future human actions) • use of internal/external reviews • development and application of codes (incl. validation, verification) • timescale considered in assessment
Multiple arguments for safety	• compliance with safety criteria • use of complementary safety indicators • no outstanding issues with the potential to compromise safety • existence of reserve FEPs
Documentation	• ensuring transparency and traceability

3 Methodology for Developing the Safety Case

3.1 Aims and structure of this chapter

The safety case consists of a set of arguments and analyses that support the view that a repository will be safe and in particular that all relevant safety criteria can be met. The process of developing such arguments and carrying out the analyses is termed safety assessment. This chapter describes the safety assessment methodology used to develop the safety case for the proposed SF / HLW / ILW repository in the Opalinus Clay of the Zürcher Weinland.

Section 3.2 presents the background to the methodology and Section 3.3 outlines its aims and discusses how the principles described in Chapter 2 are taken into account. Section 3.4 discusses the uncertainties to be addressed in developing the safety case. Section 3.5 summarises the broad tasks to be carried out in order to develop the safety case and Section 3.6 defines the foundations on which the safety case is built. The methodology for putting the safety case together is described in Section 3.7, with particular emphasis on defining and analysing assessment cases, and the management of information on features, events and processes (FEPs). Key messages from this chapter are given in Section 3.8.

The methodology described in this chapter is developed from experience acquired by Nagra in previous safety studies and from interaction with the Swiss regulatory authorities. It also draws on the results of interactions with other waste management organisations and on insights from participating in committees of international organisations (e.g. NEA's Radioactive Waste Management Committee, its Performance Assessment Advisory Group and its Integration Group for the Safety Case) and on the work of international groups, such as the NEA ad hoc group on validation and confidence building (NEA 1999b) and the group on integrated performance assessments of deep repositories (IPAG) (NEA 1997, 2000a and 2002).

3.2 Background to the methodology

Many countries have derived detailed concepts for deep geological repositories for SF, HLW and ILW, developed and in some cases demonstrated the technologies necessary for implementation, and investigated their safety through the application of rigorous safety assessment methods. The deep disposal facility at the Waste Isolation Pilot Plant (WIPP) for ILW in the USA is now in operation, and facilities for SF and HLW in the USA and Europe are advanced in the planning and licensing process, supported by safety assessments. Furthermore, several repositories for L/ILW are now, and have been for many years, in operation around the world.

In Switzerland, safety assessments have been carried out within both the HLW programme (Nagra 1985 and 1994a) and the L/ILW programme (Nagra 1985, 1993 and 1994c). The quality of individual safety assessments has, in many cases, been evaluated through independent review by regulatory bodies.

International groups have concluded that the technology to implement deep geological disposal is available (NEA 1999a), and this conclusion is supported by experience from test facilities.

As stated in the NEA/IAEA/CEC Collective Opinion document of 1991 (NEA, IAEA & CEC 1991) and re-affirmed in the findings of the first IPAG report (NEA 1997), methodologies exist for adequately assessing the long-term safety of deep geological repositories. Furthermore, the extensive experience acquired from safety assessments carried out in different countries has been reviewed in further phases of IPAG (NEA 2000a and 2002b) and in the course of the development of a new IAEA Safety Standard for geological disposal (IAEA 2001b and 2002a). A greater understanding has also been reached of the application of radiological protection principles in the context of long-term safety of waste disposal (ICRP 1997 and 1998). As a result, there is an increasing degree of consensus within the waste management community regarding the main attributes that a safety assessment and a safety case should possess, as well as consensus that geological repositories can be developed and their safety evaluated consistent with the accepted principles for radioactive waste management (IAEA 1995).

3.3 Aims of the methodology

Safety assessment is used to show how the proposed disposal system could evolve over the course of time, to test whether adequate levels of safety are to be expected based on what is known about the system, and to assess whether all the circumstances in which safety might be compromised can be reasonably ruled out. Particularly in the early stages of a repository programme, the main emphasis is on assessing the general feasibility of the project and on the robustness of the proposed disposal system. The safety assessment also provides a platform for discussion of a broad range of topics related to repository development.

The methodology adopted in Project *Entsorgungsnachweis* is based on the following *assessment principles,* as discussed in Chapter 2:

- **Focus of safety case** – The methodology focuses on issues appropriate to the stage of the repository programme. For the current phase of the Swiss HLW programme, these are: (i) the assessment of feasibility of safe disposal of SF / HLW / ILW in Switzerland and (ii) the guidance of future work by providing a platform for discussion, e.g. with respect to Nagra's proposal to focus future work on the Opalinus Clay in the potential siting area in the region of the Zürcher Weinland. Such a decision should be based on the results of an evaluation of the feasibility of safe disposal in this region and host rock, including a comprehensive consideration of all remaining uncertainties[58], as well as a consideration of a range of design options. Phenomena that are critical to long-term safety should be identified and the level of current understanding of these phenomena should be assessed. The performance of the repository system and its components should be evaluated for a broad spectrum of cases covering all realistically conceivable possibilities for the characteristics and evolution of the system. The quantitative analyses are complemented by more qualitative discussions on why the system is considered to favour safe disposal.
- **Sufficient scientific understanding** – The methodology is designed to make full use of the available scientific understanding of the repository system and its setting, and, where necessary, to provide guidance on how it could be improved to the required level. The methodology thus reflects the iterative nature of repository planning and development.
- **Systematic and defined method** – The methodology also aims to provide a systematic approach to building the safety case. This helps achieve *completeness* in that all reasonably conceivable possibilities for the characteristics and evolution of the system are considered in developing safety case arguments. The *degree of rigour* in quantifying uncertainty is chosen to be appropriate to the stage of the repository programme. *Irreducible uncertainties* related to the biosphere and future human actions are treated by examining stylised and illustrative cases. The extensive use of internal and external *reviews* provides valuable guidance in developing the safety case and ensures that no undue bias is introduced into the safety case (see also Appendix 4). A range of measures are adopted in developing models

[58] In the current phase the main aim is not to quantify uncertainties and their effect on system performance as precisely as possible, but rather to ensure that all relevant uncertainties have been considered and a reasonable upper bound of their consequences is established.

and databases to ensure that they are suitable for their intended purpose *(validation),* and each computer code used to perform analyses is *verified.* Finally, the *timescales* considered in the assessment are discussed and defined (see Chapter 2).

- **Multiple arguments for safety** – A further important assessment principle is the use of *multiple arguments for safety.* This includes *compliance with safety criteria*, the *use of complementary safety and performance indicators*, the screening of the safety analysis for *outstanding issues with the potential to compromise safety*, and the *existence of reserve FEPs*[59].
- **Documentation** – Finally, the methodology aims to *document* the safety case in a manner that is *traceable* and *transparent*.

3.4 Uncertainties to be addressed in developing the safety case

For a complex problem such as analysing the future evolution and performance of a geological repository and its environment, it must be acknowledged that exact predictions can never be achieved and that some level of uncertainty can be tolerated provided the uncertainties do not compromise safety.

Uncertainties differ in the degree to which they can be quantified or bounded, the degree to which they can be avoided or reduced in the course of a repository programme, and in the extent to which they affect the evaluated performance of the system. Uncertainty in safety assessments is discussed, for example, in NEA (1997).

In the present study, uncertainties are divided into the following broad categories:

- *Scenario uncertainty* is uncertainty in the broad evolution of the repository and its environment. This can also be considered as the uncertainty related to inclusion, exclusion or alternative realisations of FEPs that may affect this broad evolution[60].
- *Conceptual uncertainty*[61] is uncertainty in the assumptions or conceptual model[62] used to represent a given scenario or set of FEPs, including uncertainty related to the existence of plausible alternative conceptual models.
- *Parameter uncertainty* is the uncertainty in parameter values used in a model. Parameter uncertainty can be due to spatial variability and evolution over time of relevant properties and to uncertainty in the extrapolation of observations from laboratory or natural system conditions and scales of space and time to the conditions and scales relevant to the repository and its environment. Parameter uncertainty can also arise from uncertainty in the models used to interpret the raw data used to derive the parameters required for safety assessment.

There is, in addition, a category of uncertainty termed *completeness uncertainty*, arising from the possibility that some important FEPs may have been overlooked or are unknown.

It is recognised that there is an overlap between the different categories of uncertainty and allocation to a particular category is an operational matter that may involve a subjective decision.

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[59] The term "reserve FEP" is defined in Section 3.6.5 and in Appendix 5.

[60] In practice, some possibilities can be excluded if, for example, regulations do not require them to be considered.

[61] The term "model uncertainty" is also commonly used for this category.

[62] Conceptual model: ".... a set of qualitative assumptions used to describe a system or subsystem for a given purpose." (NEA 1995c).

A number of *design options* are considered in the present study, because the exact amounts and types of waste to be disposed, and the detailed location and design of the repository, including the choice of materials, are matters for future consideration. The existence of design options is addressed separately from other sources of uncertainty, since the options considered are largely under the control of the waste management programme.

The treatment of the different categories of uncertainty and design options is discussed in Sections 3.7.3 and 3.7.4.

3.5 Broad tasks to develop the safety case

The following broad tasks are carried out in order to develop the safety case:

- choice of the *disposal system*, i.e. definition of the site and design for the repository, constrained by the disposal objectives and principles defined in Chapter 2,
- derivation of the *system concept*, i.e. a description of what is known about the disposal system and its evolution, including a discussion of relevant uncertainties,
- derivation of the *safety concept*, i.e. how the disposal system is expected to provide demonstrable safety,
- illustration of the radiological consequences of the disposal system across the range of uncertainty, by defining and analysing a broad range of *assessment cases*,
- compilation of the *arguments and analyses* that constitute the safety case.

The items indicated in italics are the foundations of the safety case. Each item is described in more detail in Section 3.6. The methodology for performing the different tasks and thus constructing the safety case is described in Section 3.7.

3.6 Foundations of the safety case

3.6.1 The disposal system

The disposal system is defined as the chosen repository design and its geological setting. The disposal system in the case of the proposed SF / HLW / ILW repository in the Opalinus Clay of the Zürcher Weinland, including a number of design options, is described in Chapter 4. The repository design is developed iteratively in the course of repository planning and development. In particular, earlier studies indicate ways in which the robustness of the disposal system can be enhanced by modifications that avoid, reduce, or mitigate the effects of particular detrimental FEPs and uncertainties.

3.6.2 The system concept

The system concept is a description of what is known about the disposal system and its evolution, developed for the purpose of safety assessment. It includes a description of the key features of the system, as well as events and processes that may affect its evolution (features, events and processes are collectively referred to as FEPs), and broad conceptualisations of the possible paths that its evolution might take. It also includes a description of uncertainties. The system concept for the proposed SF / HLW / ILW repository is presented in Chapters 4 and 5.

3.6.3 The safety concept

The safety concept is the conceptual understanding of why the disposal system is safe. The disposal system performs the broad safety functions described in Section 2.6 via a range of features and associated processes that vary in their effectiveness and in the level of understanding that is available. The safety concept is built on a limited number of effective and well-understood features that ensure that the disposal system is safe and that safety can be demonstrated, even allowing for the various uncertainties and detrimental events and processes that might affect its evolution. A safety concept is developed initially based on the relevant scientific understanding and the intended functions of the design, i.e. to fulfil the safety functions described in Section 2.6. During the safety assessment, estimates of performance are made and an understanding is developed of which elements of the disposal system actually provide safety under various conditions, thus refining the safety concept. The refined safety concept for the proposed SF / HLW / ILW repository is described in Chapter 6, which also identifies the most effective and well-understood features which are termed "pillars of safety".

3.6.4 Identification of assessment cases

An assessment case is a specific set of assumptions regarding the broad evolution of the repository and its environment, the conceptualisation of individual FEPs relevant to the fate of radionuclides within the disposal system and the parameters used to describe these FEPs. In the present safety assessment, a broad range of assessment cases is defined and analysed in order to illustrate the impact of various detrimental FEPs and uncertainties on the level of safety provided by the disposal system. This range of cases has to be representative of all realistically conceivable possibilities for the characteristics and the evolution of the system. The identification of assessment cases and the underlying reasoning is documented in Chapter 6. The cases and the results of their analysis are described in Chapter 7.

3.6.5 Arguments and analyses

The safety case includes more information than numerical values of calculated dose or risk (see also the discussion of assessment principles in Section 2.6). The arguments and analyses that constitute the safety case for the proposed SF / HLW / ILW repository are brought together in Chapter 8, where the following broad lines of argument are followed.

- **The strength of geological disposal as a waste management option** – Arguments are made that safe geological disposal is possible provided a suitable site and design are chosen, and the positive attributes of geological disposal as a long-term waste management option are discussed.
- **The safety and robustness of the chosen disposal system** – The level of safety provided by the chosen system is illustrated and arguments are made that support the robustness of the chosen system. These mainly qualitative arguments, which complement the quantitative analyses, draw on evidence from nature, engineering experience, and laboratory and field experiments.
- **The reduced likelihood and consequences of human intrusion** – Arguments are made that the likelihood of human intrusion is small for the chosen site, and that, if human intrusion were to occur, the consequences would be minimised by the chosen design of the repository.
- **The strength of the stepwise repository implementation process** – Arguments are made in support of the stepwise repository implementation process as far as these are related to the safety case. A key characteristic of the stepwise process is the periodic assessment of the

project. This explicitly includes the possibility for modifications. This means that in the earlier phases of the project assessments need to be made based on preliminary information and the quality of the assessment will depend upon the reliability of this information base. If it is possible to rely for safety on well understood and well characterised repository components already in the early phases, this means that a robust safety case can be made even at such an early stage of the programme. Thus, although it is acknowledged that for a licence application the current information base needs to be extended, it is argued that the current information base is adequate to provide enough confidence that there are no critical deficiencies that could undermine the safety case.

Another argument in support of the stepwise repository implementation process is that it favours the identification and subsequent consideration of previously unrecognised safety-relevant issues. This is (i) because of the involvement of all stakeholders at each step (who may point out such issues) and (ii) because, via monitoring, possible perturbing factors are likely to be detected. That such factors can be adequately addressed is ensured by flexibility in the design and the possibility for modifications.

The operational phase, the observation phase and the post-closure phase together cover a rather long period of time and require careful consideration of security and safeguards measures as these are also essential for safety. The arguments related to this issue are not, however, developed in detail given the current early stage of the programme.

- **The good scientific understanding that is available and relevant to the chosen disposal system and its evolution** – Arguments are made that the available system understanding and characterisation provide a strong scientific foundation for carrying out the safety assessment.
- **The adequacy of the methodology and the models, codes and databases that are available to assess radiological consequences** – Arguments are made that the methodology is clearly defined, has been properly applied and can provide a safety case that is transparent, traceable, complete and robust, in accordance with the assessment principles. Arguments are also made that the models, codes and databases used to analyse the assessment cases are well supported, and have been properly applied.
- **Multiple arguments for safety** – Several lines of argument are used to discuss the level of safety provided by the disposal system. These are compliance with regulatory safety criteria, comparison with complementary safety indicators, the existence of reserve FEPs and the lack of outstanding issues with the potential to compromise safety. These are briefly discussed below.

  ***Compliance with safety criteria*:** Analyses are carried out that test whether the safety criteria defined in HSK-R-21 (HSK & KSA 1993) are fulfilled for the range of representative assessment cases. The cases are defined in such a way as to illustrate a broad range of possibilities for the characteristics and evolution of the disposal system, taking into account the various sources of scenario, conceptual and parameter uncertainty.

  Complementary safety and performance indicators: In order to place the results of the analyses in a broader perspective, comparison is made not only with dose criteria, but also with other safety indicators (IAEA 1994b, 2002b).

  In the current safety case the following safety and performance indicators are used (see also Appendix 3):

  - annual dose received by an average individual in the population group likely to be most affected by the potential releases from the repository, as a function of time, which is compared with Protection Objective 1 specified in HSK-R-21,

- individual radiological risk as defined in Protection Objective 2 of HSK-R-21 is used in the graphical representations of results from the probabilistic safety analysis (Chapter 7),
- the radiotoxicity of the wastes, which is evaluated as a function of time and compared with that of naturally occurring mineral deposits and rocks,
- radiotoxicity fluxes due to radionuclides released from the repository in the course of time, which are compared with natural radiotoxicity fluxes in the surface environment,
- radiotoxicity concentrations originating from the repository at the top of the Opalinus Clay, as a function of time, compared with natural radiotoxicity concentrations in Opalinus Clay, and
- the inventories of radionuclides in different components of the repository system, which are evaluated as functions of time, illustrating the fate of radionuclides and, in particular, the degree to which they decay before reaching the surface environment.

Reserve FEPs: Some FEPs that are considered likely to occur and to be beneficial to safety are deliberately excluded from the assessment cases, or at least from their analysis, when the level of scientific understanding is insufficient to support quantitative modelling, or when suitable models, codes or databases are unavailable. Such FEPs are termed reserve FEPs, since they may be mobilised at a later stage of repository planning if the level of scientific understanding is sufficiently enhanced, and the necessary models, codes and databases are developed. The existence of reserve FEPs constitutes an additional, qualitative argument for reserves of safety beyond those indicated by the quantitative analysis.

In addition to the reserve FEPs, in some assessment cases there are further reserves due to a number of simplifying pessimistic or conservative assumptions that have to be made for the quantitative analysis because of the limitations of available codes and / or data.

Lack of outstanding issues with the potential to compromise safety: The safety analysis allows the identification of those phenomena that are key to the safety of the chosen system. An evaluation of current understanding of these phenomena has to show that they are sufficiently well understood and adequately reflected in the safety assessment and that there are no outstanding issues that might undermine the safety case.

3.7 Constructing the safety case

3.7.1 Overview

The broad lines of argument that form the safety case are shown on the left-hand side of Fig. 3.7-1, and the procedure for constructing the safety case is summarised on the right-hand side. The figure shows which of the broad tasks within this procedure contributes to particular lines of argument, and also shows the chapters of the present report where each individual task is described. Although shown as a linear procedure of sequential tasks, there is also considerable iteration between the tasks. The right-hand side of the figure is repeated in a slightly modified format at the start of Chapters 1, 2 and 4 to 8, in order to serve as a "road map" to indicate to the reader which of the tasks is described in each chapter.

Fig. 3.7-2 shows the procedure for constructing the safety case in greater detail. Arrows linking the various boxes depict the flow of information and dependencies between activities and their respective input and output. Numbered arrows indicate important paths for possible iterations in which activities may be modified and repeated, if necessary, as explained in the following sections.

The carrying out of the broad tasks involved in constructing the safety case is described in Section 3.7.2. The tasks are performed by various groups, the roles and contributions of which are discussed in Appendix 4. The assessment and treatment of uncertainties are discussed in Section 3.7.3. The evaluation of system behaviour and consequence analysis, including the use of probabilistic and deterministic approaches and the derivation and analysis of assessment cases, are described in more detail in Section 3.7.4. FEP management procedures indicated by shaded boxes in Fig. 3.7-2 are described in Section 3.7.5. It is important to note that the activities shown in Fig. 3.7-2 are not performed in a strictly sequential manner. The main aim of Fig. 3.7-2 is to show the interdependencies of activities. The italicised text in the following sections corresponds to items in the boxes in Fig. 3.7-2.

3.7.2 Carrying out the broad tasks to develop the safety case

Choosing the disposal system

The disposal system is chosen according to the *disposal principles* defined in Chapter 2 in order to provide passive safety and robustness, and to ensure that information critical for the demonstration of safety can be obtained. The choice of the *disposal system* is the result of an iterative *repository development strategy* (see Fig. 1.2-3). Refinements to the disposal system at any given stage are guided by earlier studies, including studies of long-term safety. The disposal principles also have elements that are derived from previous experience (arrow (1) in Fig. 3.7-2). Refinements to the disposal system, and also to relevant scientific understanding, are expected to continue until a safety case can be made that is adequate for the corresponding licence application. The stepwise approach adopted in the programme for the management of SF, HLW and ILW in Switzerland is described in Chapter 1.

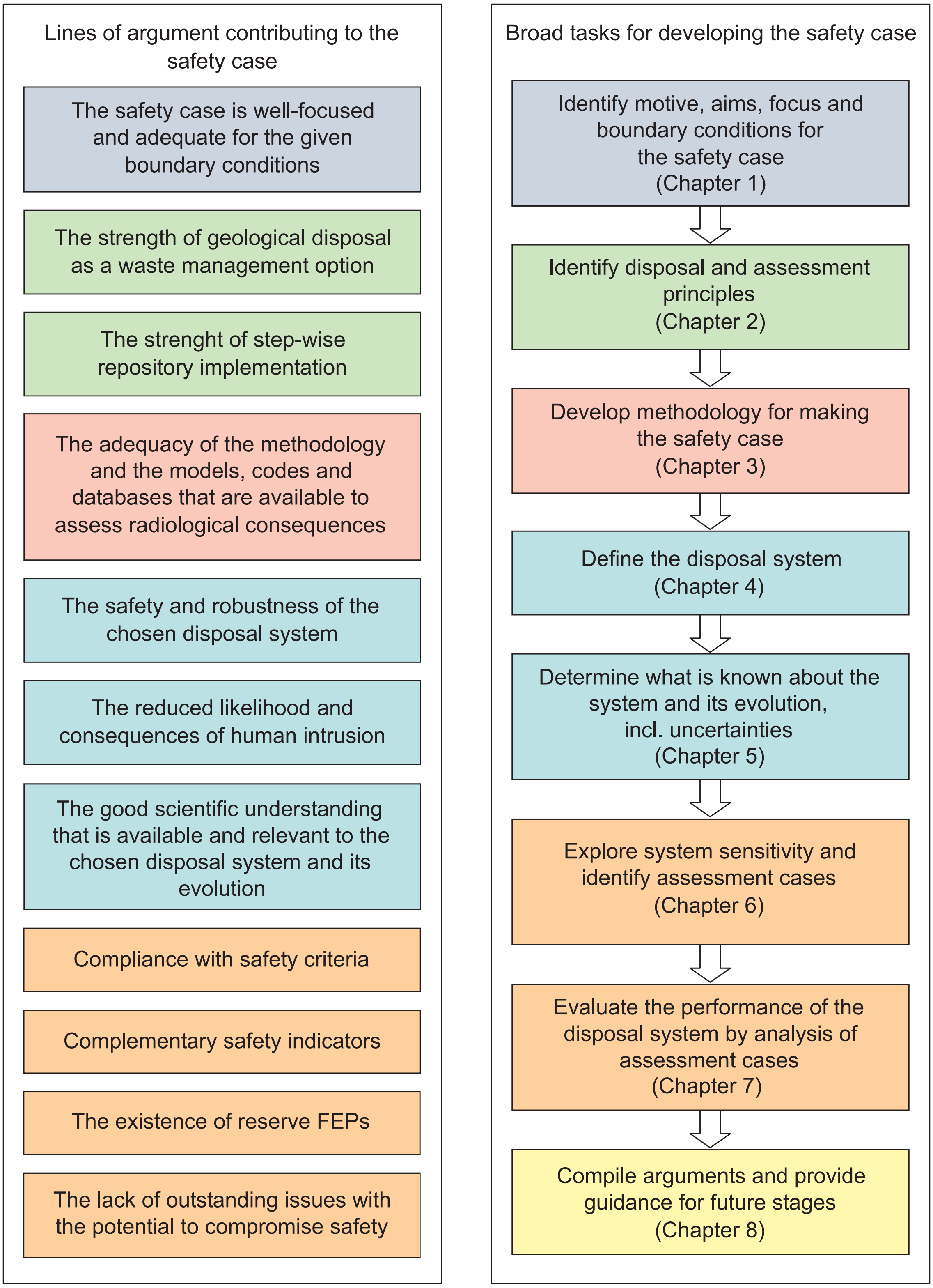

Fig. 3.7-1: The lines of argument contributing to the safety case and their relationship to the broad tasks for developing the safety case

Colours are used to indicate which of the broad tasks within this procedure contribute to particular lines of argument.

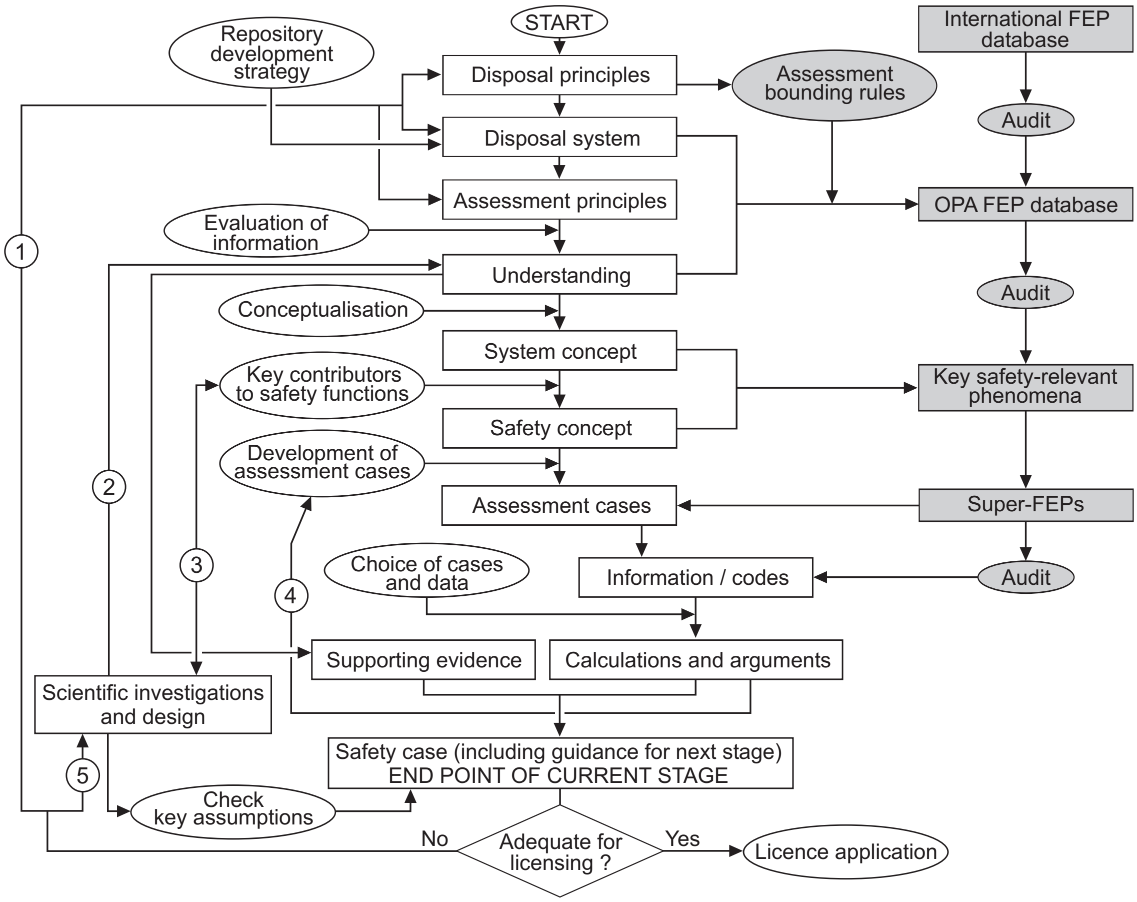

Fig. 3.7-2: The procedure for constructing the safety case

FEP management procedures indicated by shaded boxes are described in Section 3.7.5.

Deriving the system concept

Having chosen the disposal system, a safety assessment is carried out in accordance with the *assessment principles*. The starting point for the assessment is an *evaluation of information* that is relevant to the system, drawing on the expertise that is available to the project, and the results of *scientific investigations and design* studies (arrow (2) in Fig. 3.7-2). The resulting *under-standing* of the FEPs that characterise, and may influence, the disposal system and its evolution, together with a broad *conceptualisation* of the possible paths that its evolution might take and a comprehensive evaluation of uncertainties, is termed the *system concept*.

Deriving the safety concept

The derivation of the *safety concept* involves identifying the elements (i.e. physical or chemical features and their ensemble of associated processes) of the disposal system that are *key contributors to the safety functions* (as defined in Section 2.6.2). *Scientific investigations and design* studies are used to provide understanding and support for the reliability of these features (arrows (2) and (3) in Fig. 3.7-2), as well as evidence for their reliability and long-term effectiveness. Their importance and actual contribution to safety is investigated by means of mathematical modelling studies, including deterministic and probabilistic sensitivity analyses. Those elements that are eventually confirmed as providing robust safety are termed "pillars of safety".

Illustrating radiological consequences

A range of representative *assessment cases* is defined in order to illustrate the radiological consequences and their ranges of uncertainty. The results of preliminary calculations, including sensitivity analyses, are used to investigate the behaviour of the system with respect to perturbations and to identify key uncertainties, in order to restrict the assessment cases to a manageable number (arrow (4) in Fig. 3.7-2). A decision is made as to which cases are to be qualitatively discussed, and which need to be analysed quantitatively, and the data on which to base the discussions and analyses are compiled (*choice of cases and data* in Fig. 3.7-2). The *calculations* are then performed, and qualitative *arguments* made, using the assessment capability (i.e. the *information and codes*) that has been acquired. The majority of analyses are performed using a deterministic approach, but this is complemented by a number of probabilistic calculations.

Consequence analysis, including the use of deterministic and probabilistic calculations and the definition and analysis of assessment cases, is discussed in more detail in Section 3.7.4.

Compiling arguments and providing guidance for future stages

The results of the analyses of the assessment cases are combined with a range of *supporting evidence* and complemented with qualitative arguments in order to construct the safety case. The arguments and analyses aim to provide a clear indication as to whether the disposal system has been well chosen, in that it has good prospects of providing the level of safety required by Swiss regulations, and thus deserves further consideration. Through the definition and analysis of a broad range of representative assessment cases, the expected levels of safety of the proposed site and design options are assessed, and the consequences of current uncertainties are evaluated. As discussed below, many of the sources of uncertainty are either avoidable or amenable to reduction. The analysis of the assessment cases allows the detrimental FEPs and uncertainties with the greatest impact to be identified, giving guidance for future project stages (arrow (5) in Fig. 3.7-2), which may include specific modifications to the repository design (arrow (1) in Fig. 3.7-2), until ultimately the safety case is adequate to support the corresponding licence applications.

3.7.3 Evaluation and treatment of uncertainties

Evaluation of uncertainties in terms of relevance to safety

Uncertainties are identified in the course of deriving the system concept and are evaluated in terms of their potential relevance to safety. Specifically, their potential to perturb overall system performance or the evolution of the key safety-relevant phenomena is evaluated, based on scientific understanding, taking into account mitigating factors provided by the choice of site and design, and using the findings of sensitivity analyses. On the basis of this evaluation, the consequences of some uncertainties are judged to be very small, or irrelevant to safety, or the likelihood of occurrence or degree of belief of some potentially perturbing events and processes is judged to be negligible. These uncertainties and phenomena are either not considered in further analyses, or, in a few instances, are examined in a class of assessment cases termed "what if?" cases, the motivation for which is described in Section 3.7.4 below.

The remaining safety-relevant uncertainties may affect the characteristics and evolution of the disposal system itself, or may affect evaluated doses or risks due to their impact on the surface environment. These different types of uncertainty are treated separately, as detailed below.

Treatment of safety-relevant uncertainties affecting the barrier system

The treatment of uncertainties that affect the characteristics and evolution of the disposal system depends on:

- whether their effects on radiological consequences can be explored using scientifically sound models and data, and verified computer codes, and
- whether they are, in principle, avoidable or amenable to reduction.

The first of these points relates to the possibility of reliable modelling. The second relates to the usefulness of the results in terms of guidance for future project stages.

If suitable models, codes and data are available to explore the effects of a specific uncertainty, and the uncertainty is either avoidable or amenable to reduction, then either a probabilistic approach and/or one or more assessment cases are used to explore its radiological consequences. If, however, suitable models, codes and data are unavailable, or the uncertainty is neither avoidable nor amenable to reduction, then pessimistic or conservative assumptions and parameters are used to ensure that the uncertainty does not lead to the under-estimation of radiological consequences. Pessimism is here defined as the use of assumptions and parameter choices that give rise to calculated radiological consequences that are towards the high end of the range of possibilities supported by current understanding. Conservatism is defined as the use of conceptual assumptions and parameter choices that over-predict radiological consequences, and are known to lie outside the range of possibilities.

A conservative approach in some cases involves deliberately omitting FEPs that are favourable to safety, the so-called reserve FEPs, see Section 3.6.5.

Treatment of uncertainties affecting the surface environment

Many uncertainties affecting the surface environment and especially future human actions are largely unavoidable and not amenable to reduction. The full range of possibilities that may arise is a matter of speculation over the long time periods of interest in safety assessment, although some bounds may be set based on human resource and dietary needs. The probabilistic approach is not applied to uncertainties affecting the surface environment. Rather, uncertainties are treated by defining a range of different stylised conceptualisations in agreement with international consensus (see Chapter 2), and these are applied in a range of deterministic assessment cases.

3.7.4 Exploring system behaviour and consequence analysis

Deterministic and probabilistic approaches

Different and complementary approaches are used to explore system behaviour and evaluate the consequences of the different types of uncertainty discussed above. On the one hand deterministic analyses are conducted for a broad range of cases, while, on the other hand, probabilistic methods are used to explore systematically the consequences of different combinations of parameters that fall within the ranges of uncertainty. In the present study, the main emphasis is on deterministic analyses for a broad range of cases that are representative for all realistically conceivable possibilities for the characteristics and the evolution of the system. The objectives are to develop system understanding, to illustrate the possible radiological consequences of the repository, to evaluate uncertainties and design / system options in terms of their impact on the radiological consequences of the disposal system, and to determine whether existing uncertainties are acceptable, or need to be addressed in the course of future stages of the programme. These deterministic calculations are, however, complemented by probabilistic calculations that aim to enhance system understanding, ensure that no unfavourable combinations of parameters are overlooked, and to test whether there are sudden or unexpected changes in performance as parameters are varied, which might not be detected using a deterministic approach. The probabilistic calculations also provide input to the evaluation of the compliance of the repository performance with protection objectives.

The Reference Case and sensitivity analyses

The starting point for exploring system behaviour and for consequence analyses is to define and analyse a Reference Case. The Reference Case is based on the reference design / system and on the assumption that the likely / expected broad evolutionary path of the disposal system is followed (this is termed the Reference Scenario). It is also based on a number of assumptions regarding the conceptualisation for modelling purposes of key FEPs associated with the various system components (the Reference Conceptualisation), together with a reference set of parameters. The Reference Conceptualisation and reference parameter set are generally based on the expected characteristics and evolution of the system, but some pessimistic or conservative conceptual assumptions and parameter values are also used (see above). Quantitative models are used to examine the fate of radionuclides in the Reference Case, and to perform deterministic as well as probabilistic sensitivity analyses, both within and beyond the constraints of the Reference Scenario and the Reference Conceptualisation. The sensitivity analyses provide understanding of the behaviour of the system with respect to perturbations and the extent to which deviations from the likely / expected characteristics and evolution of the disposal system affect overall performance and the performance of individual system components. They provide insight into the robustness of the system chosen, guide the definition of alternative assessment cases and assist in the interpretation of results. Probabilistic analyses around the Reference Case provide also an indication of compliance with regulatory criteria taking into account the combined effects of uncertainties.

Groupings of alternative assessment cases

Alternative cases address other possibilities and stylised conceptualisations and are divided into a number of groups, according to the issues or uncertainties that they address. In particular, there are cases that address:

- the range of possibilities arising from uncertainties affecting the disposal system, where this range can be bounded with reasonable confidence on the basis of available scientific understanding,
- "what if?" cases,
- design / system options, and
- different (stylised) possibilities for future human actions and for the characteristics and evolution of the surface environment (the biosphere).

Like the Reference Case, each alternative case is defined in terms of a scenario (the broad evolutionary path that the disposal system follows), a number of conceptual assumptions for modelling key FEPs, and a set of parameters. Issues and uncertainties are assessed as to whether they (i), significantly affect the broad path of evolution of the disposal system described by the Reference Scenario, in which case they generate alternative scenarios, or whether they only affect (ii), the conceptualisation of FEPs within a given scenario, or, (iii), the assignment of parameter values within a given conceptualisation of a scenario. The result is a number of scenarios, within each of which there may be alternative conceptualisations of particular FEPs. Furthermore, for each conceptualisation, there may be a range of alternative parameter sets, including a Base Case parameter set. This hierarchy of scenarios, conceptualisations and parameter sets is illustrated in Fig. 3.7-3.

Combinations of multiple, highly unlikely possibilities are, for the most part, excluded from the assessment cases, although their consequences are screened by the use of probabilistic analyses.

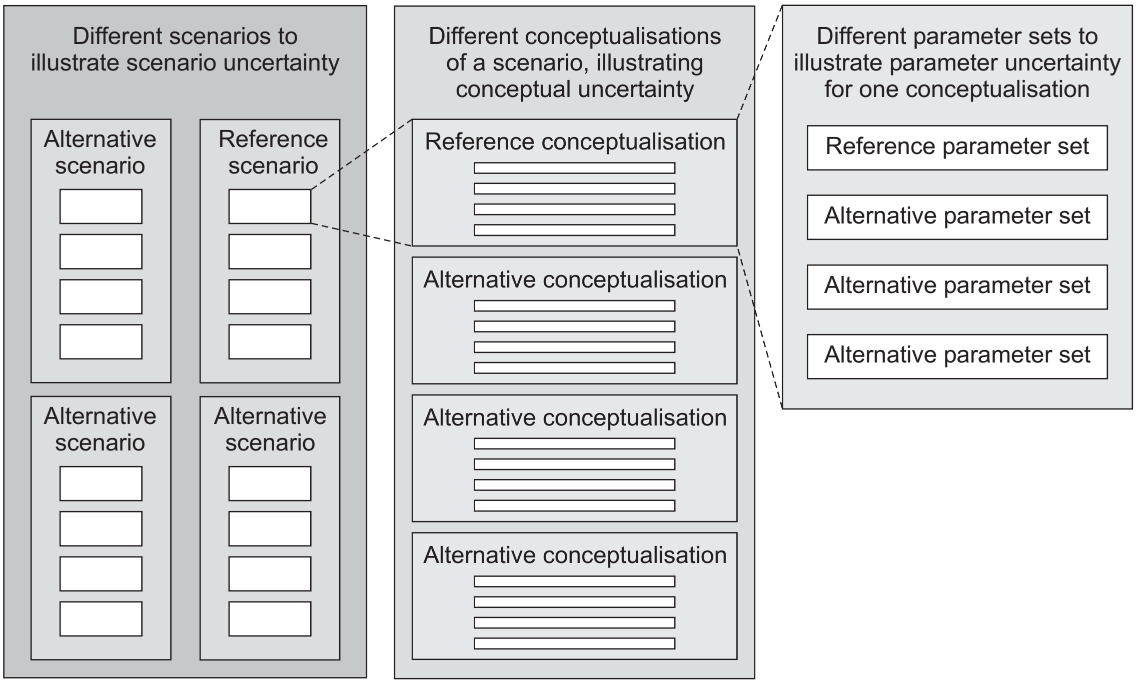

Fig. 3.7-3: The hierarchy of scenarios, conceptualisations and parameter sets

"What if?" cases

A "what if?" case is an assessment case set up to test the robustness of the disposal system. In a "what if?" case, a particular set of assumptions and / or parameter values is adopted which lies outside the range of possibilities supported by scientific evidence. In order to limit the number of such cases, they are restricted to those that test the effect of perturbations to key properties of the pillars of safety. It is not the aim to derive a comprehensive list of all conceivable "what if?" cases, but rather to select and analyse a few typical cases in order to illustrate system behaviour under extreme conditions. A limited number of probabilistic sensitivity analyses are carried out in which "what if?" assumptions and / or parameter values are assumed, with other parameters being varied within the range of possibilities supported by scientific evidence.

Analysing the assessment cases

A chain of computer codes (the reference model chain) is used to analyse the Reference Case and most of the alternative cases, as well as to perform probabilistic analyses. Parameter values required by the codes are assigned reference values in the Reference Case, and alternative (generally more pessimistic) values in the alternative cases. Parameters that are considered in probabilistic analyses are also assigned probability density functions (PDFs). In these analyses, large numbers of calculations are performed using parameter values sampled at random from the PDFs. The reference and alternative parameter values and the PDFs are presented in Appendix 2. All of the models, codes and data used in the present study in order to analyse the assessment cases and perform probabilistic analyses are described in detail in Nagra (2002c).

For a few cases, alternative conceptualisations are considered that do not fall within the scope of the reference model chain, and alternative codes or analytical solutions are employed. Furthermore, in order to examine particular aspects of Reference Case system performance in more detail, and to investigate sensitivity to key system properties, results using the reference model chain are complemented by the results of simplified models that focus on a limited number of FEPs. These are termed "insight models", and are also described in Nagra (2002c).

In order to keep the amount of information to be processed to a manageable size, all release and dose calculations are limited to safety-relevant radionuclides, i.e. radionuclides which have the potential to make a significant contribution to overall radiological releases to the environment. These are identified by carrying out simplified release and consequence calculations which are described and documented in Nagra (2002c).

3.7.5 Assurance of completeness and FEP management

A prime concern in the development of all assessments of the long-term safety of radioactive waste is the issue of phenomenological "completeness" of the assessment, see, e.g., NEA (2001b). That is:

- Has a thorough identification been made of all the features, events and processes (FEPs) that could affect the long-term safety?
- Have these features, events and processes been adequately treated in the safety assessment?

Although it can never be proved that an assessment is complete, actions can be taken to encourage the identification of all potentially relevant phenomena and, equally important, once identified, to assure that the safety-relevant phenomena have been represented appropriately in the assessment.

In the present safety assessment, an assurance of phenomenological completeness and appropriate treatment is provided by:

- the development of a comprehensive database of FEPs that may be relevant to the safety assessment – the Opalinus Clay FEP Database or *OPA FEP Database*, and
- a careful accounting of the phenomenological scope of assessment cases and the models by which they are represented, in a process termed *FEP management*.

These activities are described in detail in Nagra (2002d). The general strategy is illustrated in Fig. 3.7-4.

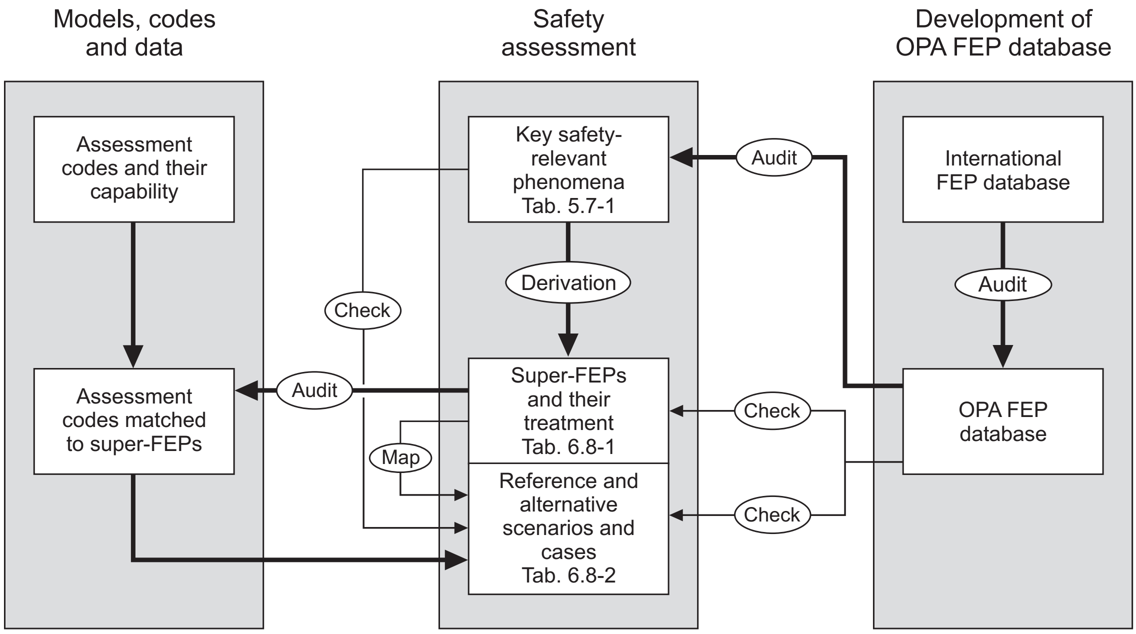

Fig. 3.7-4: Strategy for assurance of phenomenological completeness and appropriate treatment by FEP management

First, it is recognised that adequate characterisation of the disposal system and thorough scientific understanding of the safety-relevant processes that act therein is the essential foundation. From this basis, three separate lines of work can be identified.

- Assessment codes have been developed, often to represent quite generic systems but in some cases to be capable of representing processes that are specific to the disposal system for SF / HLW / ILW in Opalinus Clay; the assessment codes and their capabilities are described in Nagra (2002c).
- The thorough synthesis of the scientific knowledge concerning the Opalinus Clay disposal system and its performance has been assembled in the main stream of the safety assessment as described in Chapters 4 and 5 of this report. This forms the basis for the selection of scenarios and cases that are treated quantitatively as described in Chapters 6 and 7 of this report.
- In addition, at an early stage of the safety assessment, a comprehensive identification was made of individual FEPs that might be relevant to the safety of the Opalinus Clay disposal system – the OPA FEP Database, as described in Nagra (2002d). The comprehensiveness of this database is encouraged by an audit of the database against the NEA International FEP Database (NEA 2000e) and the database is also updated as the assessment proceeds.

Within the main stream of the safety assessment, key safety-relevant phenomena are identified bearing in mind the understanding of the disposal system and its functions, and judgements are made regarding relevance or importance in the *system context*. Similarly, later, the so-called super-FEPs and assessment cases are identified with a view to defining a set of calculations that encompass the possible overall system behaviour or to providing relevant illustrations. By contrast the development of the OPA FEP Database is done with the aim of identifying all possibly relevant *individual FEPs*. As a result, the key safety-relevant phenomena and super-FEPs identified in the safety assessment tend to be agglomerations of the individual FEPs contained in the OPA FEP Database. The aim of FEP management procedures are then:

- to assure that no individual FEP from the OPA FEP Database has been excluded without reason from consideration in the main stream of the assessment – this is achieved by audit of the key safety-relevant phenomena, and checks of the super-FEPs and assessment cases against the OPA FEP Database; and
- to assure that the assessment codes that are used are appropriate to represent the various assessment cases and are used appropriately – this is achieved by audits of the assessment codes against the super-FEPs and checks that each of the assessment cases is correctly represented by a set of codes.

In addition, the derivation of super-FEPs from key safety-relevant phenomena and the mapping of the assessment cases to their component super-FEPs and key safety-relevant phenomena has been carefully checked. Tables which detail all of the above audits and checks are presented in Nagra (2002d).

3.8 Key messages from this chapter

- The methodology described in this chapter is designed to test the adequacy of the proposed site and design options in terms of long-term safety and to produce a safety case that is transparent and complete, recognises sources of bias and minimises inadvertent bias, and is robust.
- The methodology is also designed to identify issues that might usefully be addressed in the course of future stages of the project, in order to further enhance the robustness of the safety case.
- The documentation of the safety case is designed to provide transparency and traceability of the arguments and analyses that constitute the safety case.
- Lines of argument that contribute to the safety case are related to:
  - the strength of geological disposal as a waste management option,
  - the safety and robustness of the disposal system,
  - the reduced likelihood and consequences of human intrusion,
  - the strength of the stepwise repository implementation process,
  - the good scientific understanding that is available and relevant to the chosen disposal system and its evolution,
  - the adequacy of the methodology and the models, codes and databases that are available to assess radiological consequences,
  - the multiple arguments for safety that include compliance with regulatory safety criteria, complementary safety indicators, the existence of reserve FEPs and the lack of outstanding issues with the potential to compromise safety.
- The making of the safety case involves:
  - the choice of a disposal system, via a flexible repository development strategy, that is guided by the results of earlier studies, including studies of long-term safety,
  - the derivation of the system concept, based on current understanding of the FEPs that characterise, and may influence, the disposal system and its evolution,
  - the derivation of the safety concept, based on well understood and effective pillars of safety,

  - the illustration of the radiological consequences of the disposal system through the definition and analysis of a wide range of representative assessment cases, and
  - the compilation of the arguments and analyses that constitute the safety case and provide a platform for discussing future stages of the repository programme.

- Measures, including the use of independent experts who participated in a series of review meetings, as well as adherence to the relevant Nagra Quality Rules, have been implemented to ensure that the models, codes and data used to analyse the assessment cases are scientifically sound and have been properly applied (Nagra 2002c).
- FEP management procedures are used to promote completeness of the safety assessment, to assure that all FEPs identified are considered in the safety assessment and that those carried forward in quantitative analyses are represented appropriately.
- In addition to the FEP management procedures, completeness and the avoidance of inadvertent bias are promoted by information exchange within the team involved in Project *Entsorgungsnachweis* and with other organisations that have conducted safety assessments of disposal systems with at least some features in common with the system under consideration. Interaction with the scientific community also occurs via presentations at technical conferences, publications in scientific journals and external peer review.

4 Description of the Disposal System for SF / HLW / ILW in Opalinus Clay

4.1 Aims of the chapter

In this chapter, a description of the features of the disposal system is given to set the stage for discussion of system evolution and development of assessment cases, which are presented in subsequent chapters. The information base on the status of the disposal system as implemented (the present chapter) as well as its expected evolution and the way it has been conceptualised to allow evaluation of long-term safety (Chapter 5) constitutes the system concept as defined in Chapter 3 (Fig. 4.1-1).

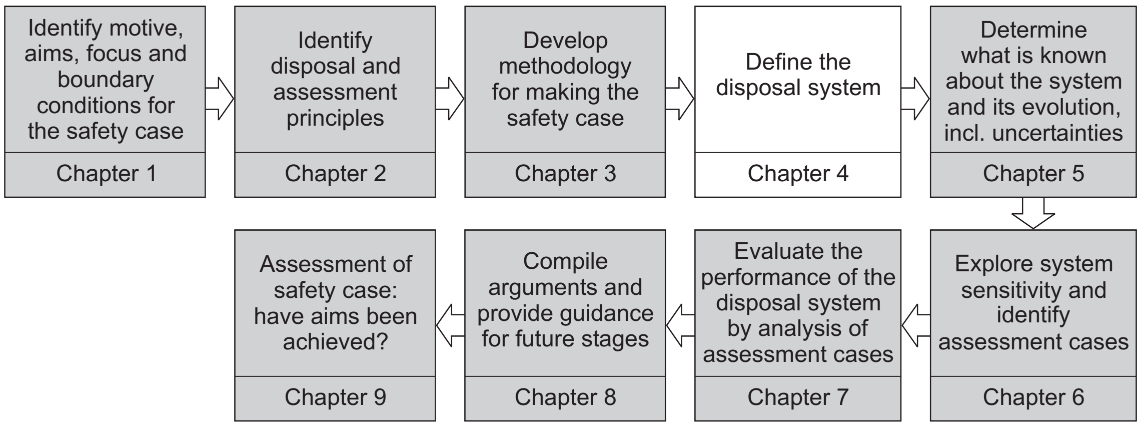

Fig. 4.1-1: The role of the present chapter in the sequence of tasks involved in developing the safety case

The geological setting and the surface environment in their present state are summarised in Sections 4.2 and 4.3 based on the information in Nagra (2002a). The repository layout, quantities of waste and their characteristics, and features and properties of the engineered barriers are discussed in Sections 4.4 and 4.5.

4.2 The geological environment of the repository

The present project assesses the Opalinus Clay of the Zürcher Weinland[63] as a host rock for a repository for long-lived wastes. The principal reasons why this formation and region have been chosen are:

- **Simplicity** – The geological environment is simple, with predictable structural, hydrogeological and geochemical properties over a scale of several kilometres. The Opalinus Clay is sufficiently homogeneous to allow confident prediction of its behaviour on the time and space scales of interest for repository safety.
- **Stability** – The potential siting area is tectonically stable on a timescale of the next few million years, being characterised by a low rate of uplift and associated erosion and average *in situ* stresses and heat flows.

[63] including the Murchisonae beds in Opalinus Clay facies (see Fig. 4.2-2 and Tab. 4.2-1)

- **Plasticity/Self-sealing Capacity** – The mechanical properties of Opalinus Clay ensure that repository-induced or natural discontinuities are self-sealed, thus they will not significantly influence hydraulic properties.
- **Negligible Advective Water Flow** – The Opalinus Clay has such a low hydraulic conductivity that solute movement through the formation is predominantly by diffusion rather than advection.
- **No Resource Potential** – The sediments overlying the basement in this region, and the basement rocks themselves, are not considered to have any significant natural resource potential.
- **Geochemical Stability and Retention Capacity** – The geochemical environment in the Opalinus Clay and surrounding formations has been stable for many millions of years, with no identifiable perturbations resulting from climate change (including Quaternary glaciations). In addition, the geochemical conditions in the Opalinus Clay are reducing, which favours preservation of the engineered barriers and low solubilities/strong sorption of many important radionuclides.
- **Engineering Feasibility** – The Opalinus Clay is an indurated claystone (clay-shale) with good engineering properties, allowing small, unlined tunnels and larger, lined tunnels to be constructed at depths of several hundred metres at reasonable cost with existing technology.

Tab. 4.2-1: Definition of principal stratigraphic components overlying and underlying the Opalinus Clay host rock

System component	Stratigraphy (see Fig. 4.2-7)	Barrier function
Malm aquifer (Regional aquifer)	Malm limestones, Plattenkalke to Hornbuck beds	No barrier function, but dilution
Upper confining unit	Effingen beds to Wedelsandstein Formation *Sequence of low permeability beds with intercalations of potentially water-conducting units of limited inter-connectedness*	Upper supplementary geological barrier
Host rock	Opalinus Clay and Murchisonae beds in Opalinus Clay facies *Sequence of low permeability beds*	Primary geological barrier
Lower confining unit	Lias to Keuper *Sequence of low permeability beds partly with minor aquifers of limited interconnectedness*	Lower supplementary geological barrier
Muschelkalk aquifer (Regional aquifer)	Upper Muschelkalk (including dolomite of anhydrite group)	No barrier function, but dilution

Combined, these features indicate excellent isolation potential for a repository within the Opalinus Clay. Additionally, the so-called confining units (see Table 4.2.1) will also contribute to radionuclide retention. There is also the likelihood of significant dilution of any such releases in groundwaters in the permeable formations above and below the confining units, before they reach surface aquifers in the biosphere, where further dilution takes place.

This section outlines briefly the main evidence for each of these functions (see Nagra 2002a for a complete description). It begins by placing the potential siting area in its regional geological context.

4.2.1 Regional geological setting

The Opalinus Clay was deposited some 180 million years ago by the sedimentation of fine clay, quartz and carbonate particles in a shallow marine environment. It is part of a thick sequence of Mesozoic and Tertiary sediments in the Molasse Basin (see Figs. 4.2-1 and 4.2-2), which overly Palaeozoic sediments and crystalline basement rocks. The overlying Tertiary sediments thicken considerably into the Molasse Basin to the south. In the Zürcher Weinland the Mesozoic sediments containing the Opalinus Clay are of uniform thickness over several kilometres, almost flat-lying (dipping gently to the south east) and little affected by faulting. To the north-east, towards the Hegau-Bodensee Graben structure and in the Jura mountains to the west, the sedimentary rocks become faulted and folded (Nagra 2002a). The Zürcher Weinland consequently represents a structurally simple region on the northern edge of a deep basin, bounded by deformed sedimentary rocks to the north-east and west.

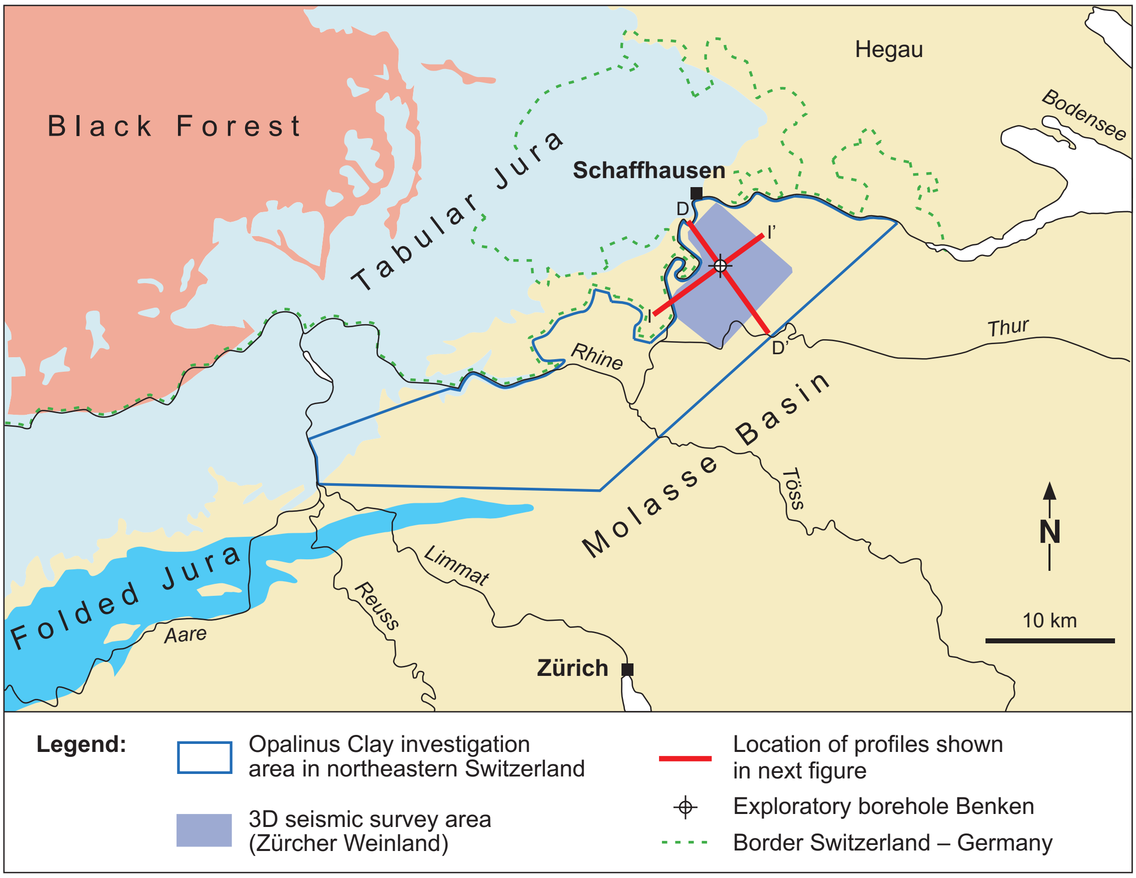

Fig. 4.2-1: Principal tectonic units of northern Switzerland and adjacent areas

The Opalinus Clay investigation area lies in the tectonically stable Tabular Jura and is predominately covered by the sediments of the Molasse Basin. Also shown is the 3 D seismic investigation area in the Zürcher Weinland and the location of the borehole at Benken.

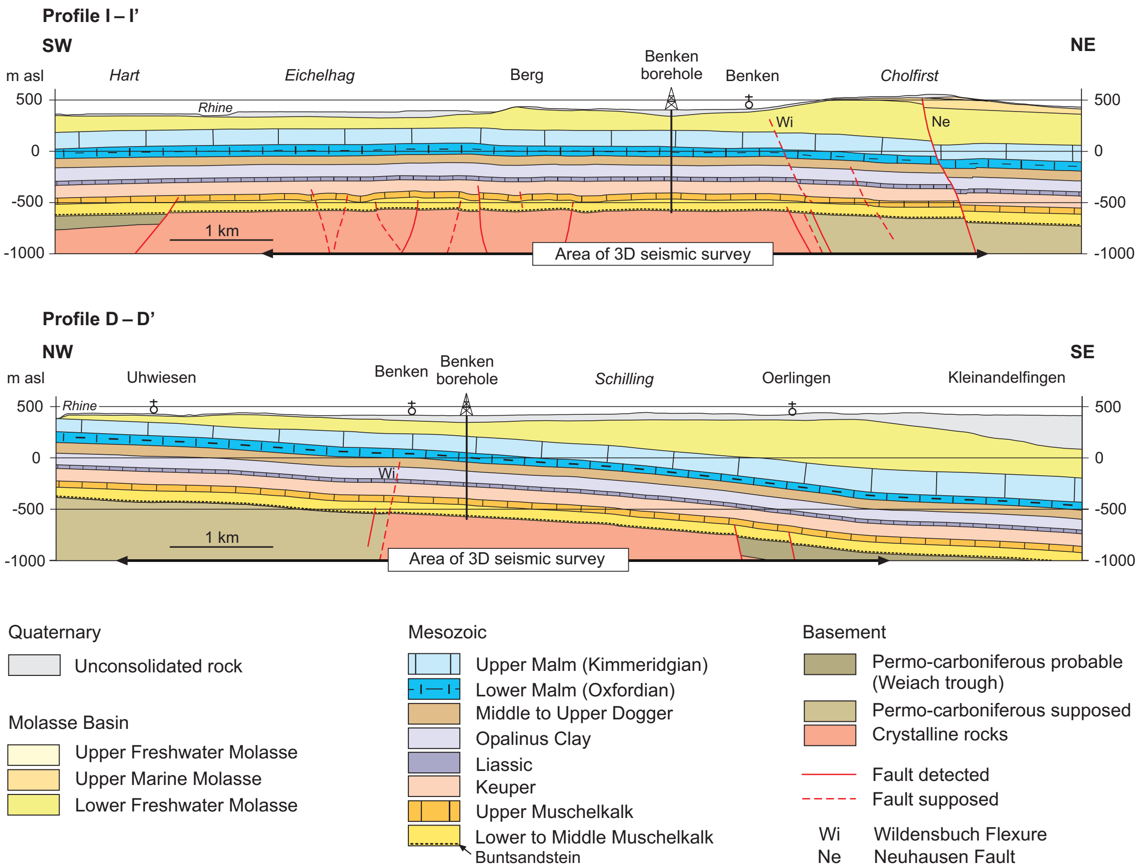

Fig. 4.2-2: Schematic geological profiles from SW to NE (top) and NW to SE (bottom) through the sedimentary rocks in the Zürcher Weinland

The potential host rock consists of the Opalinus Clay formation and the Murchisonae beds in Opalinus Clay facies.

The burial and compaction history of the sedimentary formations in the central part of the Zürcher Weinland is shown in Fig. 4.2-3. It can be seen that the Opalinus Clay reached a burial depth of about 1000 m during the Cretaceous, between about 65 and 120 million years ago. Northern Switzerland was affected by the rifting of the Rhine Graben (starting about 40 million years ago) and by differential vertical movements that resulted in considerable uplift of the area in the mid-Tertiary, followed by down-warping and burial under late Tertiary sediments (Nagra 2002a). This resulted in the Opalinus Clay reaching its greatest burial depth, of about 1700 m below the surface, in the Miocene stage of the Alpine orogeny, the last major tectonic event to affect the region. From about 10 million years ago, Alpine uplift and erosion brought the Opalinus Clay progressively up to its present burial depth of about 600 to 700 m in the region of interest. The most recent stage of the orogeny involved the thrusting and folding of the Jura Mountains, which, along with the uplift of the Alps and the updoming of the Black Forest massif, continues today. The thermal gradient in the investigation area is ~ 4.2 °C per 100 m. The present temperature of the Opalinus Clay at 650 m at Benken is 38 °C; during its burial and compaction history it has experienced maximum temperatures of about 85 °C.

Currently, a compressive stress field is observed in the Zürcher Weinland, with the maximal component oriented horizontally in a N-S direction which is consistent with the regional trend. The region is characterised by low seismic activity and a low uplift rate compared with other regions in Switzerland (e.g. Alps, Folded Jura).

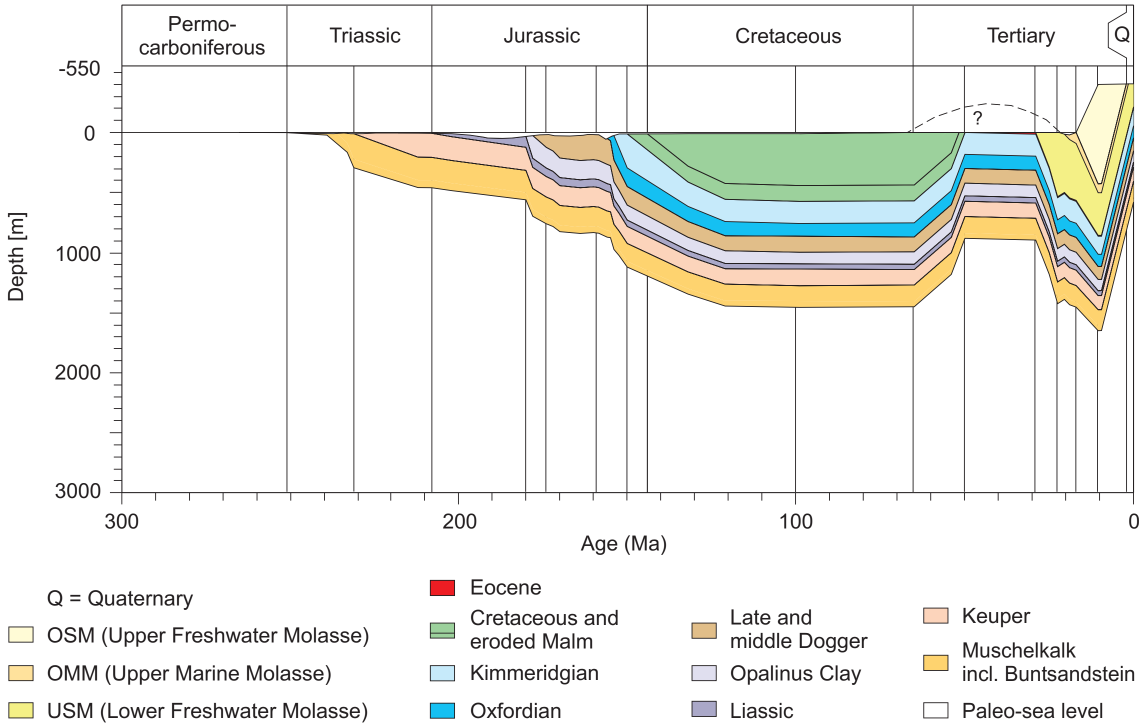

Fig. 4.2-3: Burial history of the sedimentary formations in the Zürcher Weinland

4.2.2 Nagra studies in the Zürcher Weinland and Opalinus Clay

Regional geological investigations in northern Switzerland over the past 20 years, including more focused studies in the area over the last ten years, have provided a clear picture of the geological and hydrogeological structure and properties of the Zürcher Weinland. This information has come principally from intensive work in a 1000 m deep borehole at Benken and a 3 D seismic survey of the surrounding area. The seismic campaign confirmed the remarkable homogeneity and lateral extent of the Opalinus Clay (see Fig. 4.2-4). In addition, geological, hydrogeological, hydrochemical and isotopic data have been synthesised on a regional basis and a new, local hydrodynamic model has been developed. Comprehensive and detailed geological information is presented in the "geosynthesis" report (Nagra 2002a).

In addition to specific information obtained in the area of interest, much has been learned from Nagra's other deep boreholes in north-central Switzerland (drilled as part of the earlier study of the crystalline basement as a potential repository host formation) and from studies in the Opalinus Clay where it intersects underground excavations (e.g. road and railway tunnels) elsewhere in Switzerland. In particular, the underground rock laboratory at Mont Terri, in the Jura mountains of north-west Switzerland, has provided detailed data on rock engineering, hydrogeology and geochemistry, and has helped to develop fundamental understanding of processes relevant to repository safety.

Fig. 4.2-5 shows the investigation area of the Zürcher Weinland with those larger scale features that need to be considered in locating the repository waste emplacement tunnels. The Neuhausen fault in the north-east is considered as a "no-go feature" (respect distance approx. 200 m). This leaves an area of ~ 35 km^2. Considering long-term evolution (uplift and erosion) and ease of construction, an overburden of between approximately 600 m and 750 m is preferred. This still leaves an area of about 22 km^2. Within this area a first priority repository zone was delimited for the "Reference Case" in Project *Entsorgungsnachweis* by application of the criterion that the crystalline basement should directly underlie the Mesozoic cover. This results in a preferred area for the location of waste emplacement tunnels of about 8 km^2, which is several times greater than the required area of about 2 km^2.

4.2.3 The geological isolation concept of the Opalinus Clay environment

The potential repository host rock identified consists of the Opalinus Clay and the Murchisonae beds within the Opalinus Clay facies, and is more than 100 m thick. Fig. 4.2-6 shows the lithological sequence in the north-central part of the region of interest, as obtained from the Benken borehole. The repository is foreseen as consisting of a series of waste emplacement tunnels constructed roughly in the mid-plane of the Opalinus Clay.

The host rock, a well-compacted, moderately over-consolidated claystone, is expected to have exceptional isolation properties resulting from its very low hydraulic conductivity. Evidence for the high isolation potential is discussed in Section 4.2.5, below. In Figs. 4.2-6 and 4.2-7 it can be seen that the host formation is overlain and underlain by further thicknesses (100 to 150 m) of clay-rich formations (confining units) which, although not as impermeable and homogeneous as the Opalinus Clay (they contain thin, probably discontinuous sandstones and carbonate rocks), also possess good isolation properties, with limited groundwater movement and good sorption potential.

Above and below these clay-rich formations, the sequence contains large, regional aquifers in the carbonate rocks of the Malm and the Muschelkalk. There is significant groundwater flow in some regions of these formations.

The geological component of the isolation concept is thus as follows (see Fig. 4.2-7):

- The absence of significant advective groundwater flow in the host formation, which is thick enough to extend for more than 40 m above and below a repository (Nagra 2002a, Birkhäuser et al. 2001), will ensure that the rate of movement of radionuclides out of the engineered barriers and through the undisturbed host rock will be extremely small. Any such movement will be controlled by diffusion and will be so slow that only the most mobile and longest-lived radionuclides can reach the edge of the formation.
- The surrounding clay-rich sediments are rocks of the confining units and have the additional potential to retard the movement of any radionuclides that escape from the host rock. Although there are thin, more permeable horizons in these clays (see Section 4.2.4), based on isotopic and hydrochemical evidence, flows are rather small due to limited hydraulic interconnectedness, and potential pathways to the biosphere are long (15 – 25 km, if they exist). Furthermore, the surrounding formations (Upper Dogger/Lower Malm above, and Lias/Keuper below) have good sorption properties.
- Any radionuclides that migrate through the clay-rich formations of the Upper Dogger/ Lower Malm and Lias/Keuper (i.e. are not transported laterally along the thin, water-conducting horizons), will enter the regional aquifers of the Malm (above) and the Muschelkalk (below). Neither of these aquifers or permeable horizons are exploited in this region, and, with the exception of the Muschelkalk, the waters have relatively high salinities and are non-potable[64]. The current discharge area for the Muschelkalk aquifer is some 30 km to the west, with the Malm discharging a few km to the north.

[64] Here non-potable is defined as groundwaters with salinities exceeding 5 g l^{-1}

- If radionuclides enter the regional aquifers, they will be significantly dispersed and diluted. An additional stage of dilution will occur when the deep aquifers discharge to the more dynamic freshwater flow systems of near surface gravel aquifers, or to river waters. Groundwaters directly discharging to springs are also considered.

The safety assessment described later in this report looks at various possible behaviours and pathways for radionuclides within this conceptual system. The following sections briefly outline the key evidence supporting the safety functions of the geological environment that have been discussed above.

4.2.4 Regional and local groundwater flow in the sediment sequence

In the Zürcher Weinland sedimentary rock sequence, deep groundwater movement is dominated by the two **regional aquifers** of the Malm and the Muschelkalk, which are hydraulically separated by the low conductivity of the Opalinus Clay and upper/lower confining units, through which no significant flow occurs. Fig. 4.2-8 shows the hydraulic conductivity and hydraulic head profiles for the formations encountered in the Benken borehole. The generally low conductivity of the Dogger and Lias formations above and below Opalinus Clay is evident, as are the two higher conductivity regional aquifers above and below. A minor aquifer (Stubensandstein Formation, see K1 in Fig. 4.2-8) exists in the Keuper below the host rock and a water-conducting layer of rather low permeability was encountered in the Dogger (Wedelsandstein Formation) above the host rock.

The closest recharge regions of both regional aquifers are in the Black Forest area of Germany; however, for the Malm aquifer in the region of interest, recharge from the SE (Molasse basin) is probably more significant (see Fig 4.2-9b). In the Benken area, the underlying Muschelkalk aquifer (Fig. 4.2-9a) contains slightly mineralised but potable waters, with flow being generally to the south, swinging round towards the west for the shortest path to discharge[65], which is in the area near the confluence of the Rhine-Aare rivers, 20 – 30 km to the west.

In the overlying Malm aquifer (Fig. 4.2-9b), flow through the immediate area of interest is generally towards the north-west, with discharge in the Rhine valley a few km to the north, downstream from the Rhine Falls. The Malm aquifer is heterogeneous, locally fractured and partly karstic (only the upper part). In the area of interest, the groundwater in the Malm is practically stagnant and contains mineralised and non-potable saline waters. The Malm up-dips, as the formation approaches the surface near its discharge area and the Malm waters become potable (and are exploited) as a result of mixing with younger meteoric waters, infiltrating through the shallow overlying formations.

Discharge from both Muschelkalk and Malm takes place into the base of thin (tens of metres) Quaternary gravel aquifers in the river valleys.

[65] The groundwater divide evident on Fig. 4.2-9a indicates a possible, but longer drainage path to the north-east.

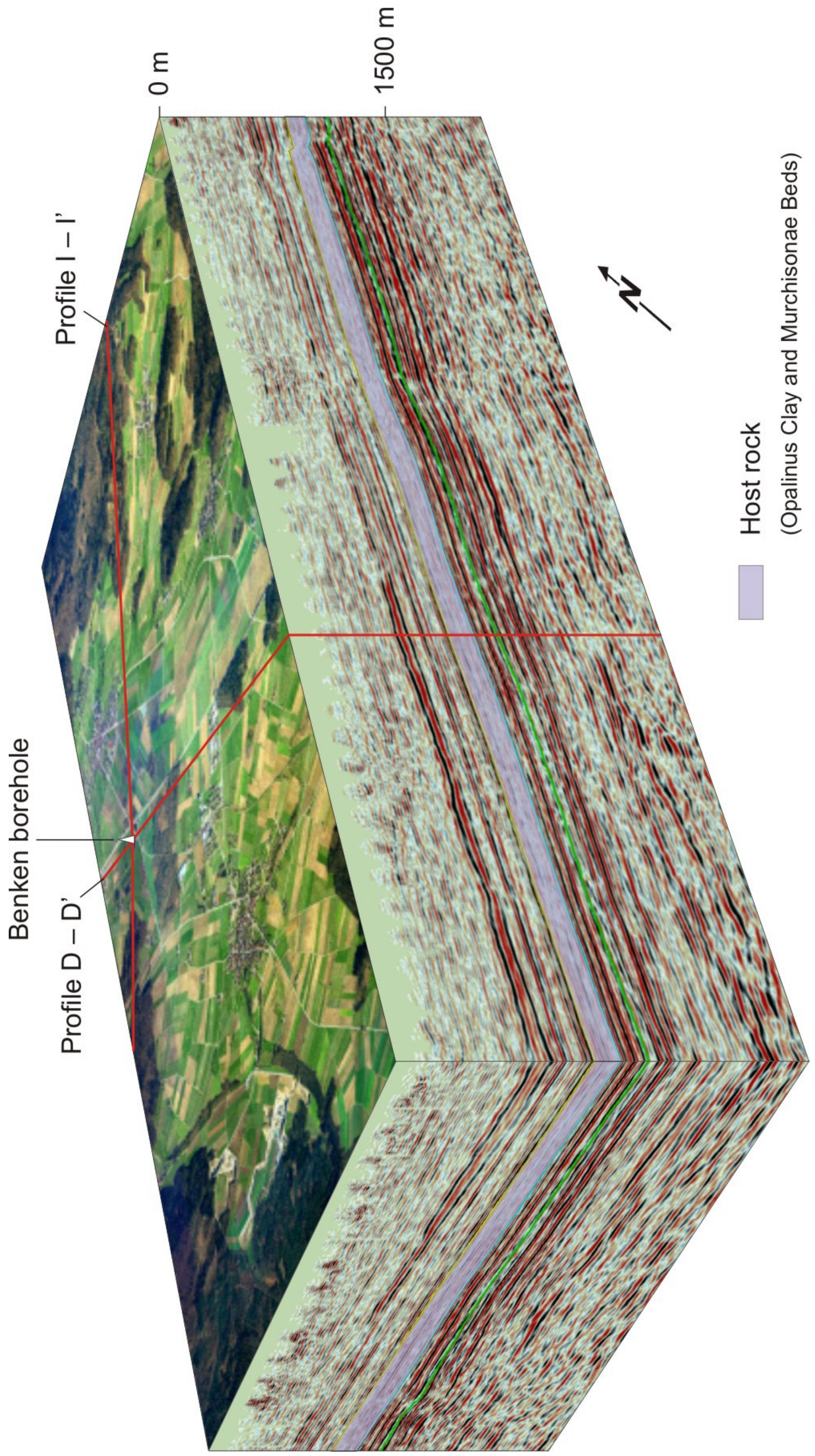

Fig. 4.2-4: Geological structure of the sedimentary sequences in the vicinity of the Benken borehole, based on 3 D seismic data

For the location of the investigation area and profiles see Fig. 4.2-1.

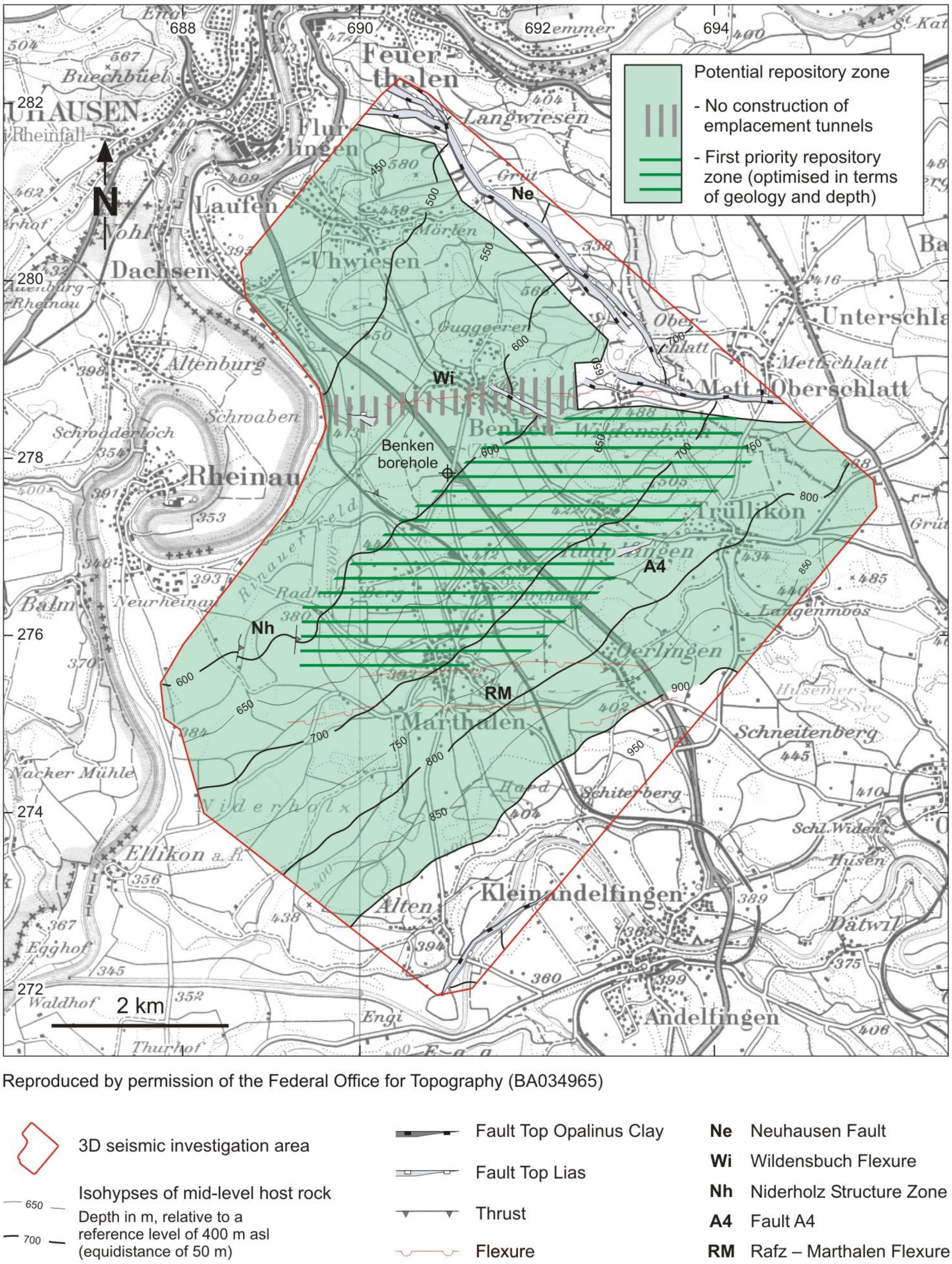

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Fig. 4.2-5: Potential area for a repository in Opalinus Clay in the Zürcher Weinland

The area chosen for the reference repository is ~ 8 km^2 and is marked with green horizontal lines.

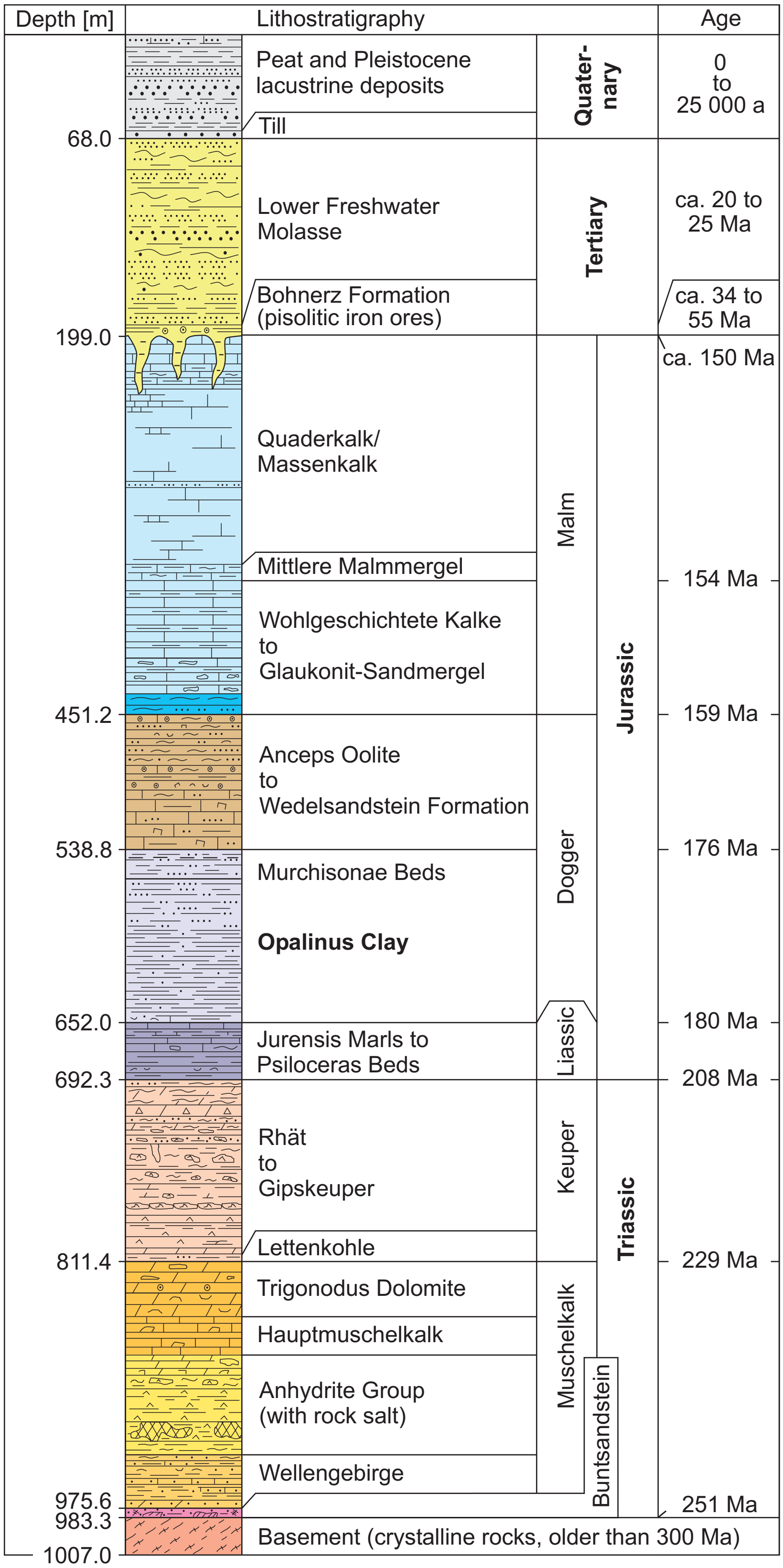


Fig. 4.2-6: Lithological sequence based on information from the Benken borehole

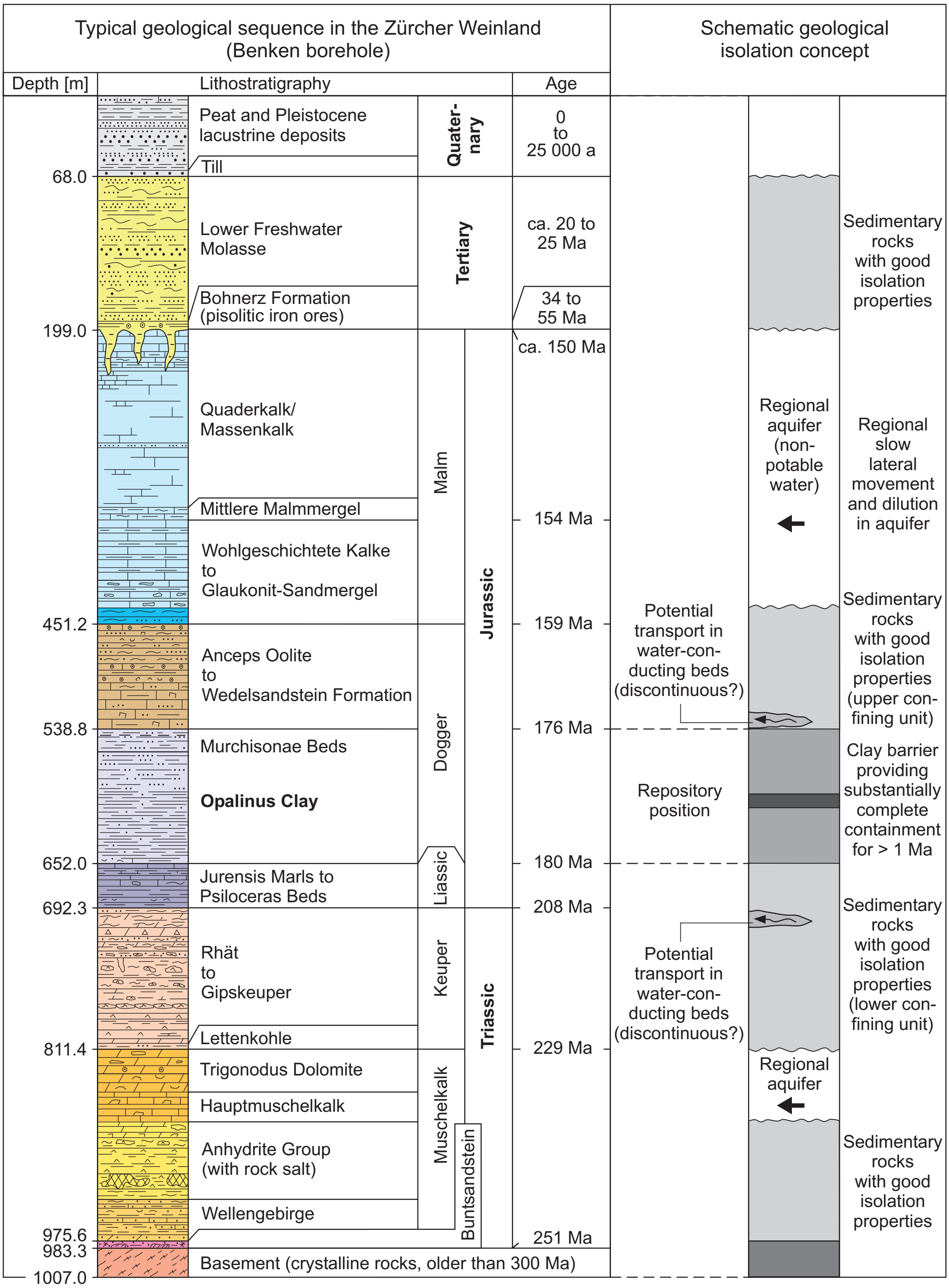


Fig. 4.2-7: The geological sequence in the Benken borehole (left) and the simplified features illustrating the isolation concept (right)

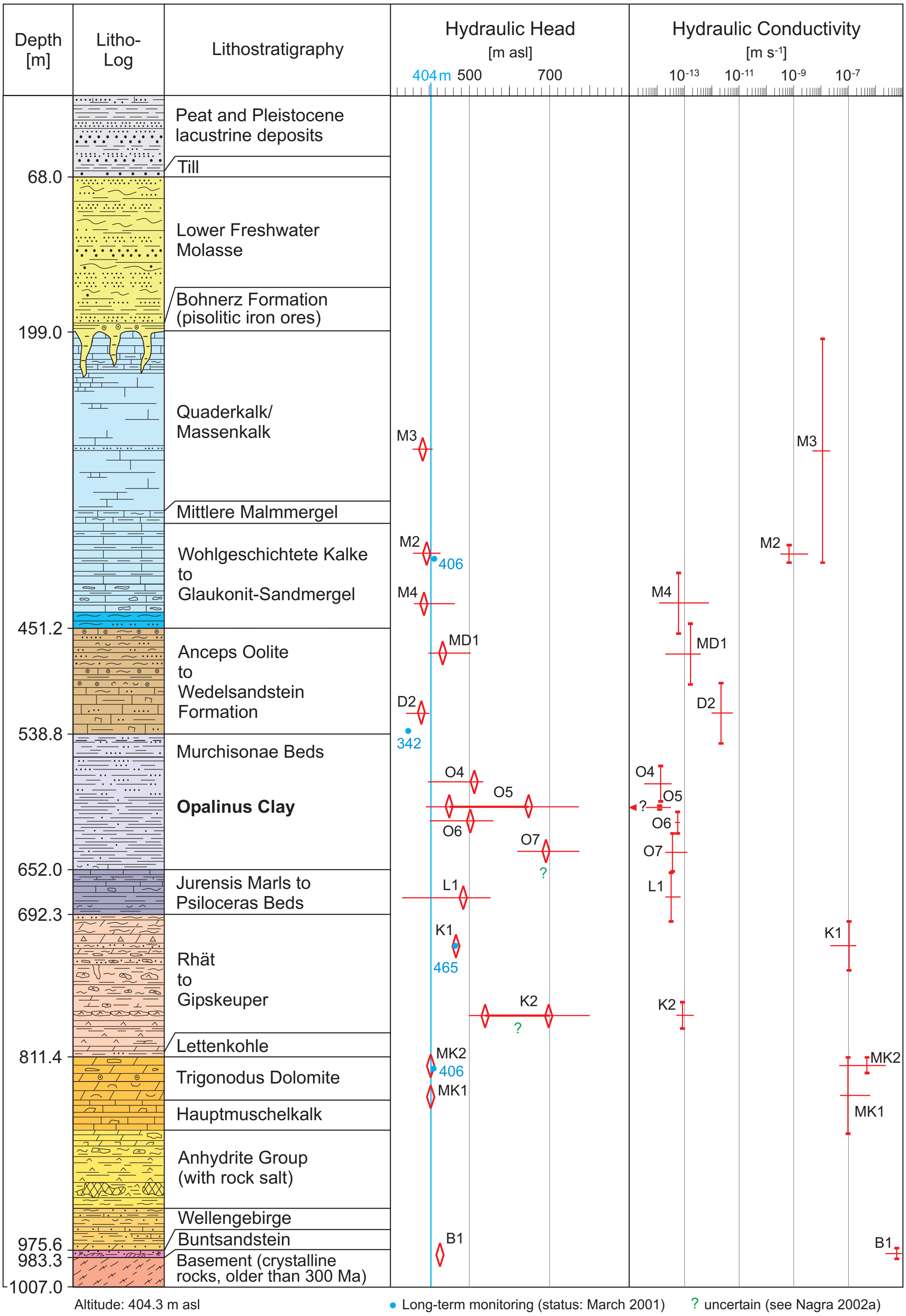


Fig. 4.2-8: Lithological sequence in the Benken borehole and the measured hydraulic heads and hydraulic conductivities

As noted earlier, there are thin, discontinuous water-conducting horizons within the low conductivity units of the middle and upper Dogger (the Wedelsandstein Formation) and the Keuper (the Stubensandstein Formation), comprising sandstones and carbonate sediments. The most permeable part of the Stubensandstein Formation at Benken is a dolomite breccia with a thickness of 5 to 6 metres. The lateral extent of this facies is unknown (e.g. it is not found in the Weiach borehole to the West). In other regions of northern Switzerland, water-flow in the Keuper occurs in the Schilfsandstein formation or in the Gansingen Dolomite. This complex system of partly interconnected, partly disconnected permeable sandstone and carbonate layers forms a hydrogeological unit, the so-called "Sandsteinkeuper" (Nagra 2002a). A similar situation is supposed for the Wedelsandstein, which also forms a rather heterogeneous hydrogeological unit. These sandwich the Opalinus Clay, with the Wedelsandstein Formation directly above and the Stubensandstein Formation some sixty metres below. At Benken, the Wedelsandstein Formation has a rather low hydraulic conductivity (~ 10^{-10} m s^{-1}), with that of the Stubensandstein Formation significantly higher (~ 10^{-7} m s^{-1}; K1 in Fig. 4.2-8). The higher conductivity Stubensandstein Formation contains mineralised and non-potable water and the low conductivity Wedelsandstein Formation is also saline. Their large-scale lateral hydraulic connectivity is not known, although that of the "Sandsteinkeuper" unit is considered to be one or two orders of magnitude lower than local values measured at Benken. This formation discharges locally into a Quaternary gravel aquifer some 15 km to the west in the Rhine valley (Klettgau). It is believed that the discontinuous nature of both formations precludes regional-scale flow or discharge to surface groundwater systems, although there are no data to verify this. If they were sufficiently connected that they could act as continuous pathways for radionuclide transport, path lengths to discharge would be about 25 km (Wedelsandstein unit) to 15 km ("Sandsteinkeuper").

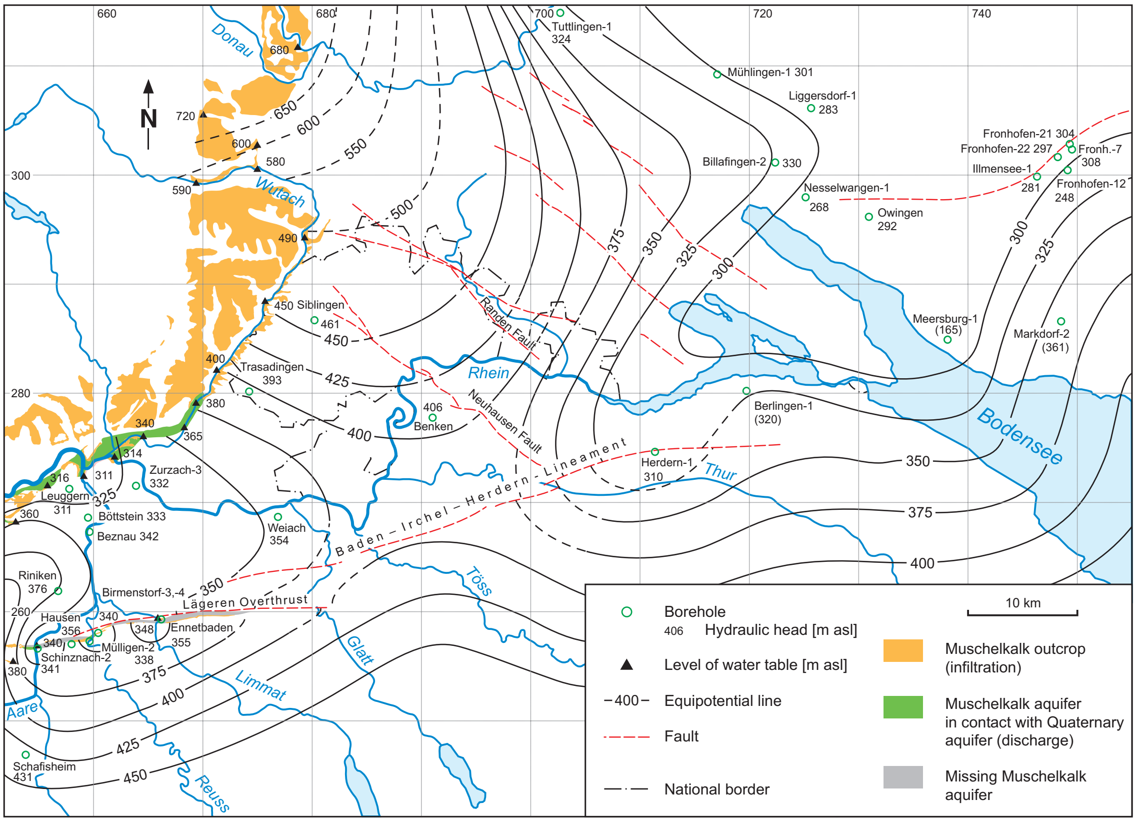


Fig. 4.2-9a: Infiltration areas, discharge areas and hydraulic heads in the Muschelkalk aquifer in the region surrounding the Zürcher Weinland

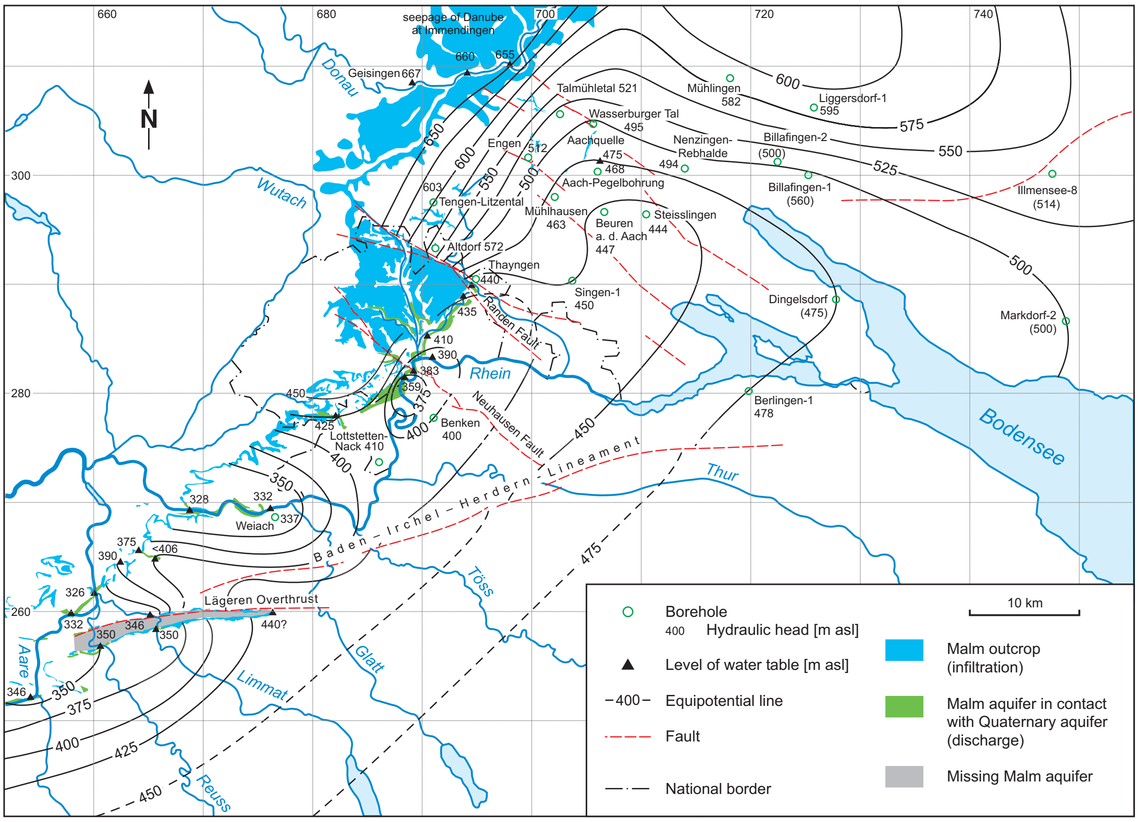


Fig. 4.2-9b: Infiltration areas, discharge areas and hydraulic heads in the Malm aquifer in the region surrounding the Zürcher Weinland

The predominantly argillaceous Lias directly below the host rock formation contains a thin, discontinuous limestone (Arietenkalk) that is not thought to have any significant impact on the flow regime outlined above. If it did, its significance for isolation is expected to be much the same as that of the Stubensandstein Formation and its potential impact can be assessed using the same assumptions.

The overall properties of the aquifers and significant water conducting formations in the Keuper and Dogger are outlined (for the area of interest for repository siting) in Tab. 4.2-2.

Based on the information above, on regional considerations (see Chapter 3 in Nagra 2002a) and on the results of the hydrodynamic model (see Chapter 6 in Nagra 2002a), a broad spectrum of possible groundwater transport paths have been identified on the regional scale (Fig. 4.2-10). In Fig. 4.2-10a, which represents the most plausible case, it is assumed that solutes which diffuse through the host rock are advected laterally in the minor aquifers in the confining units ("Sandsteinkeuper" and Wedelsandstein unit). If neither of these minor aquifers is continuous, then vertical transport is likely to occur through the confining units into the Muschelkalk and Malm aquifers and from there laterally (Fig. 4.2-10b). On the regional scale it cannot be excluded that larger fracture zones occur that off-set the minor aquifers in the confining units. Thus, even if these minor aquifers are continuous, transport may go vertically through these off-sets into the Malm and Muschelkalk aquifers (Fig. 4.2-10c).

Tab. 4.2-2: Properties of regional aquifers and water-conducting formations (minor aquifers) above and below the host rock formation in the area of interest. TDS = total dissolved solids

Water conducting formation	Properties	Water chemistry
Malm (regional aquifer)	Limestone Average hydraulic conductivity ~10^{-8} m s^{-1}	Highly mineralised Na-Cl-(SO_4) (TDS ~ 10 g l^{-1}) and non-potable
Wedelsandstein Formation (Wedelsandstein unit at regional scale)	Sandstones Hydraulic conductivity ~10^{-10} m s^{-1}	High salinity, similar to formation waters in the Opalinus Clay
	Host rock	
Stubensandstein Formation ("Sandsteinkeuper" at regional scale)	Dolomite breccia Hydraulic conductivity ~10^{-7} m s^{-1} locally; probably ~10^{-8} – 10^{-9} m s^{-1} on a regional scale	Highly mineralised Na-SO_4-(Cl) (TDS ~ 10 g l^{-1}) and non-potable
Muschelkalk (regional aquifer)	Dolomitic limestone Average hydraulic conductivity ~10^{-6} m s^{-1}	Slightly mineralised Ca-Mg-SO_4-(HCO_3) (TDS ~ 2.4 g l^{-1}) and potable

4.2.5 Water and solute movement in the host formation

Hydraulic regime in Opalinus Clay

The potential for vertical movement of water across the sedimentary formations in the Zürcher Weinland is controlled by the hydraulic properties of the formations and the hydraulic driving forces. Fig. 4.2-8 indicates overpressures of 50 – 300 m (best estimate reference value 100 m based on hydraulic test interpretation) above hydrostatic head in the Opalinus Clay. Similar overpressures have been measured in the argillaceous horizons in the Lias and possibly also in the low permeability gypsum horizon in the Keuper below (see Figs. 4.2-6 and 4.2-8). The Wedelsandstein Formation exhibits significant underpressure (60 m below hydrostatic), showing it to be hydraulically separated from the overpressured formations above and below, whereas the Stubensandstein Formation is overpressured (60 m above hydrostatic). The hydraulic head values are close to hydrostatic within or near to the regional aquifers. There are, however, a limited number of hydraulic head measurements in the region, and this needs to be considered when discussing the flow regime.

Today, and until the overpressures are dissipated, the hydraulic gradient at the mid-plane of the Opalinus Clay is oriented vertically upwards and downwards in the lower parts. It is assumed that these overpressures are inherited from an earlier period of rapid burial or, alternatively, ongoing lateral stress, for which drainage and compaction of the Opalinus Clay has not yet resulted in a state of pressure equilibrium. The long-term effects of clay compaction on the performance of the geological barrier are discussed in Section 5.2.2.

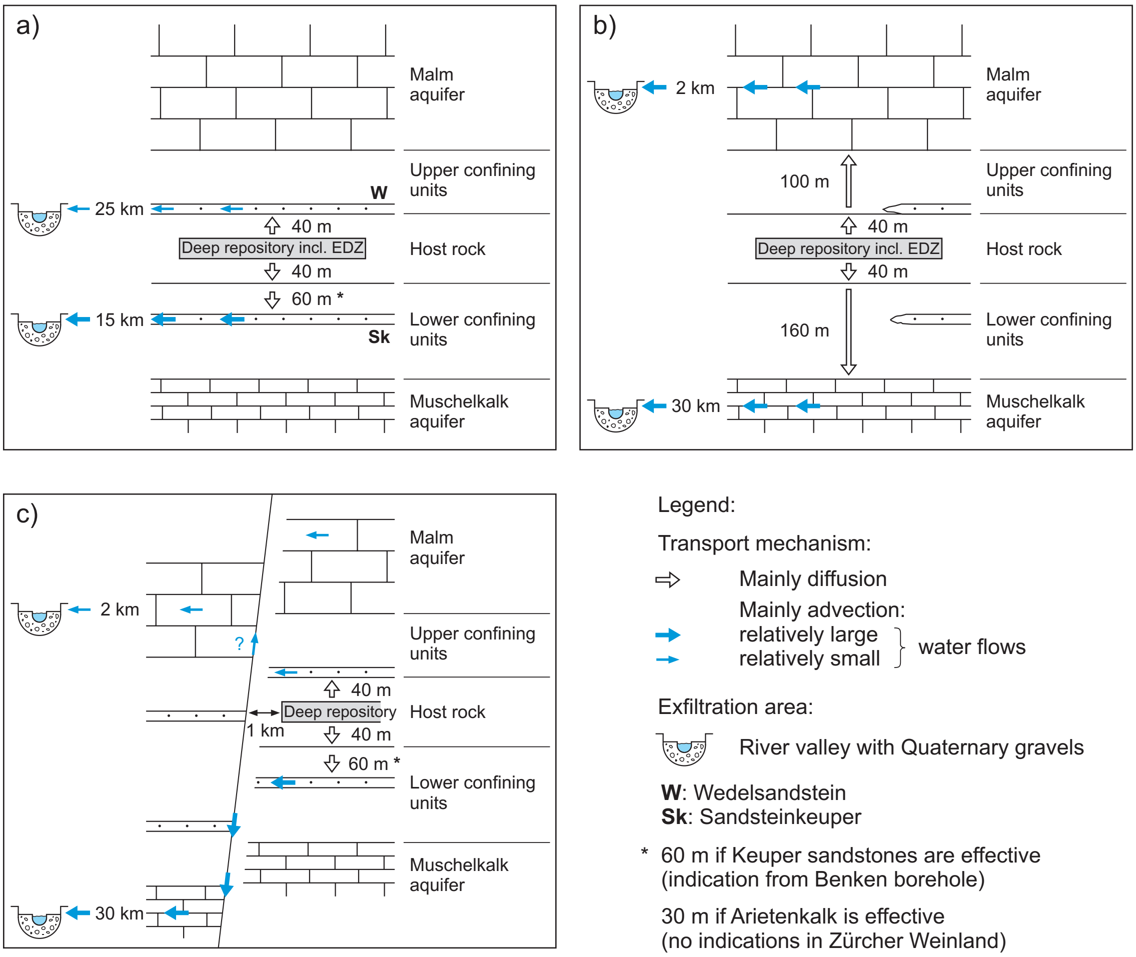


Fig. 4.2-10: Potential groundwater transport paths based on siting information and hydrodynamic modelling

a) lateral transport occurs in minor aquifers in confining units, b) discharge occurs through the regional aquifers after vertical transport through upper and lower confining units, c) after lateral transport through minor aquifers in confining units for at least 1 km (200 m in the pessimistic case), discharge occurs through regional aquifers due to a large fault connecting the confining unit and regional aquifers.

There are various ways of characterising the vertical hydraulic pressure gradient across the host rock formation, which can be treated as variants in subsequent sensitivity analysis calculations. At the largest scale, the hydraulic gradient within the Opalinus Clay will be controlled by the head difference between the two regional aquifers. This will also be the case in future, once the overpressures are naturally dissipated. Currently, the observed hydraulic head difference between the regional aquifers is relatively small, producing a hydraulic gradient of 0.05.

At a smaller scale, the gradient across the host formation is controlled by the head difference between the Stubensandstein Formation (60 m above hydrostatic) and the Wedelsandstein Formation (60 m below hydrostatic), resulting in a gradient of about unity. Yet on a more local scale, a gradient of 5 might be considered between the even more highly overpressured host rock formation and the Wedelsandstein Formation. Active hydraulic connections between the regional aquifers or the discontinuous water-conducting formations may exist only through sub-vertical water-conducting faults, and no such features have been observed in the potential repository area (Fig. 4.2.5).

Scenarios for hydraulic gradients that could drive vertical advection are indicated in Tab. 4.2-3.

Tab. 4.2-3: Hydraulic gradients (in m m^{-1}) between different formations in the Zürcher Weinland sedimentary rock sequence

Formation	**Hydraulic gradient [m m^{-1}]**
Muschelkalk to Malm aquifers (ignoring high overpressure in Opalinus Clay)	0.05
Stubensandstein Formation to Wedelsandstein Formation (ignoring high overpressure in Opalinus Clay)	1
Centre of Opalinus Clay (repository horizon) to Wedelsandstein Formation (accounting for high overpressure in Opalinus Clay)	5

The Opalinus Clay has an extremely low hydraulic conductivity: From the field tests in Benken and from core samples typical hydraulic conductivities in the range 10^{-13} to 10^{-14} m s^{-1} have been determined, suggesting a best estimate of ~ 10^{-13} m s^{-1} parallel to the bedding and ~ 2×10^{-14} m s^{-1} normal to the bedding. These values are also consistent with tests and long-term observations from Mont Terri and with isotopic profile data (see below). It is worth mentioning that hydraulic tests are carried out at elevated hydraulic gradients (typically > 50), which are not observed in the natural system. Furthermore, these values represent rock properties on a relatively small scale and, therefore, other observations are used to test the applicability of these conductivities at larger scales ("upscaling"). For this purpose, the observed overpressures were modelled with specific basin models investigating different possible mechanisms for generating overpressures (effect of consolidation from rapid burial, effect of lateral thrust). The results of these models clearly indicate that if the overpressures represent the large-scale porewater pressure in the host rock formation, then the hydraulic conductivity of the Opalinus Clay has either to be very low (10^{-15} m s^{-1} or lower) or a non-Darcian flow regime exists where a threshold gradient has to be exceeded before flow starts or – most likely a combination of both a very low hydraulic conductivity and a threshold gradient. If one or both of these were not the case, then the observed overpressures could not be sustained over geological time scales. The existence of such threshold gradients is consistent with all existing observations but cannot be quantified with confidence. Conductivities of 10^{-15} m s^{-1} or lower cannot easily be justified because measurements indicate higher values. A value of 10^{-13} m s^{-1} is considered an upper limit. In any case, whether or not the overpressures are real, the water is quasi-stagnant.

A hydrodynamic model, discussed in detail in Nagra (2002a), was developed to help in the interpretation of the local hydrodynamic situation. The apparent inconsistency between measured conductivities and observed overpressures and our current inability to quantify threshold gradients with confidence necessitated the use of pessimistic parameters in the model. In all calculational cases, the measured hydraulic conductivities of 10^{-13} and 2×10^{-14} m s^{-1} were used and threshold gradients were neglected, except for case RF2 where conductivities of 10^{-15} and 2×10^{-16} m s^{-1} were used (Nagra 2002a). To investigate the effects of existing uncertainties regarding boundary conditions and properties of some formations, different cases were modelled applying Darcy's law. The results show different vertical gradients through the Opalinus Clay, but in all cases, only a small vertical flux upward is observed through the repository horizon. The results for the spectrum of cases analysed with the hydrodynamic model are summarised in Tab. 4.2-4 and are briefly discussed below.

Tab. 4.2-4: Key results from the hydrodynamic model for the different cases analysed (see also Nagra 2002a)

Case	Hydraulic conductivity Opalinus Clay [m s^{-1}]		Specific flux [m s^{-1}]	Key characteristics
	K_H	K_V		
RF0	10^{-13}	2×10^{-14}	10^{-14}	Regime determined by Malm and Sandsteinkeuper
RF1	10^{-13}	2×10^{-14}	10^{-14}	Regime determined by Sandsteinkeuper, and enhanced permeability in Wedelsandstein
RF2	10^{-15}	2×10^{-16}	10^{-16}	Regime determined by Wedelsandstein and Sandsteinkeuper, low K in Opalinus Clay
RF3	10^{-13}	2×10^{-14}	10^{-14}	Regime determined by Wedelsandstein and Sandsteinkeuper (assumption: Sandsteinkeuper exfiltrates towards Neckar)
RF4	10^{-13}	2×10^{-14}	10^{-14}	Overpressures in Opalinus Clay
RF5	10^{-13}	2×10^{-14}	9×10^{-15}	Regime determined by Wedelsandstein and Sandsteinkeuper
RF6	10^{-13}	2×10^{-14}	4×10^{-15}	Regime determined by Wedelsandstein and Sandsteinkeuper; Neuhausen fault: K = 10^{-8} m s^{-1}; Wildensbuch flexure: K = 10^{-8} m s^{-1}
RF7	10^{-13}	2×10^{-14}	4×10^{-15}	Regime determined by Wedelsandstein and Sandsteinkeuper; Neuhausen fault: K = 10^{-6} m s^{-1}; Wildensbuch flexure: K = 10^{-6} m s^{-1}
RF8	10^{-13}	2×10^{-14}	2×10^{-14}	Heads calibrated to measured values in Benken borehole (inverse modelling)

The smallest flow through the Opalinus Clay is observed for the case where a very low conductivity is allocated to the Opalinus Clay (case RF2). For the other cases (RF0, RF1, RF3, RF4, RF5, RF8), the flow rates are similar, except for those in which both the Neuhausen fault and the Wildensbuch flexure are given higher transmissivities (case RF6 - transmissivity of 10^{-7} m^2 s^{-1}, case RF7 - transmissivity of 10^{-5} m^2 s^{-1}), which leads to a slight decrease in the gradient across the Opalinus Clay. In Benken, the observed heads cannot be reconciled with these latter two cases, thus the transmissivity of these features must be smaller than 10^{-7} m^2 s^{-1}.

Some further perspective on the validity of the measured hydraulic conductivity values for Opalinus Clay can be obtained by comparing the data with values obtained for other claystones as well as for shales and unconsolidated clays (Fig. 4.2-11). The data for Opalinus Clay are fully consistent with the trend of decreasing permeability with decreasing porosity.

Solute transport through the Opalinus Clay

With conductivities of 2×10^{-14} m s^{-1} (vertical) and 10^{-13} m s^{-1} (horizontal) or lower and the possible existence of a threshold gradient, the porewater will be effectively stagnant. For such stagnant porewaters, transport is dominated by diffusion.

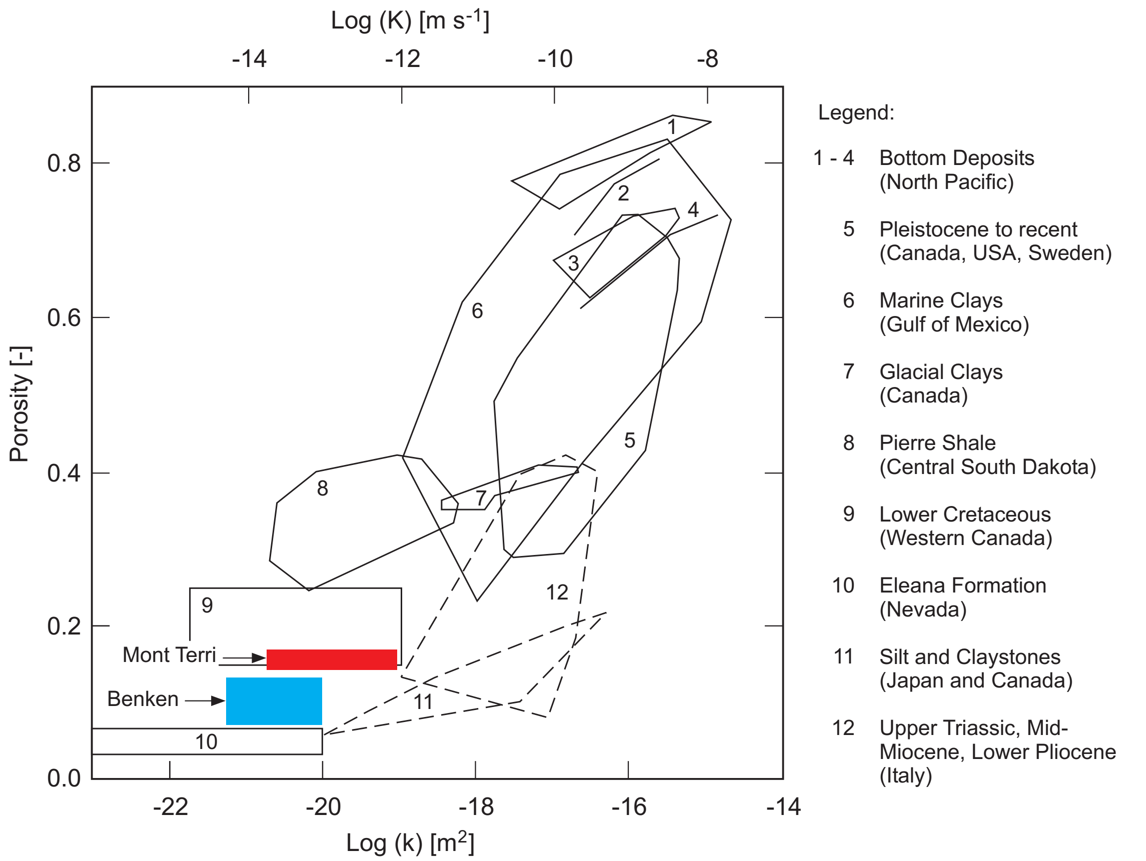


Fig. 4.2-11: Compilation of porosity and permeability data from argillaceous rocks of different maturity after Neuzil (1994) and comparison with results from the Benken borehole and the Mont Terri investigation tunnel

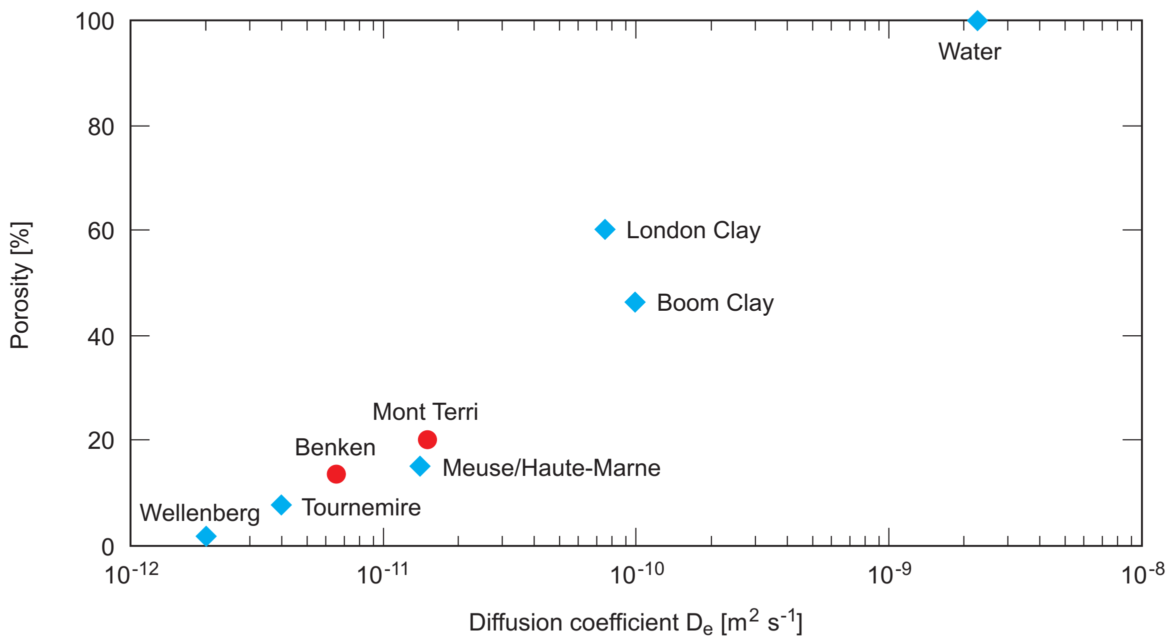


Fig. 4.2-12: Diffusion coefficient and porosity for tritium (perpendicular to bedding) in different argillaceous rocks (Nagra 2002a)

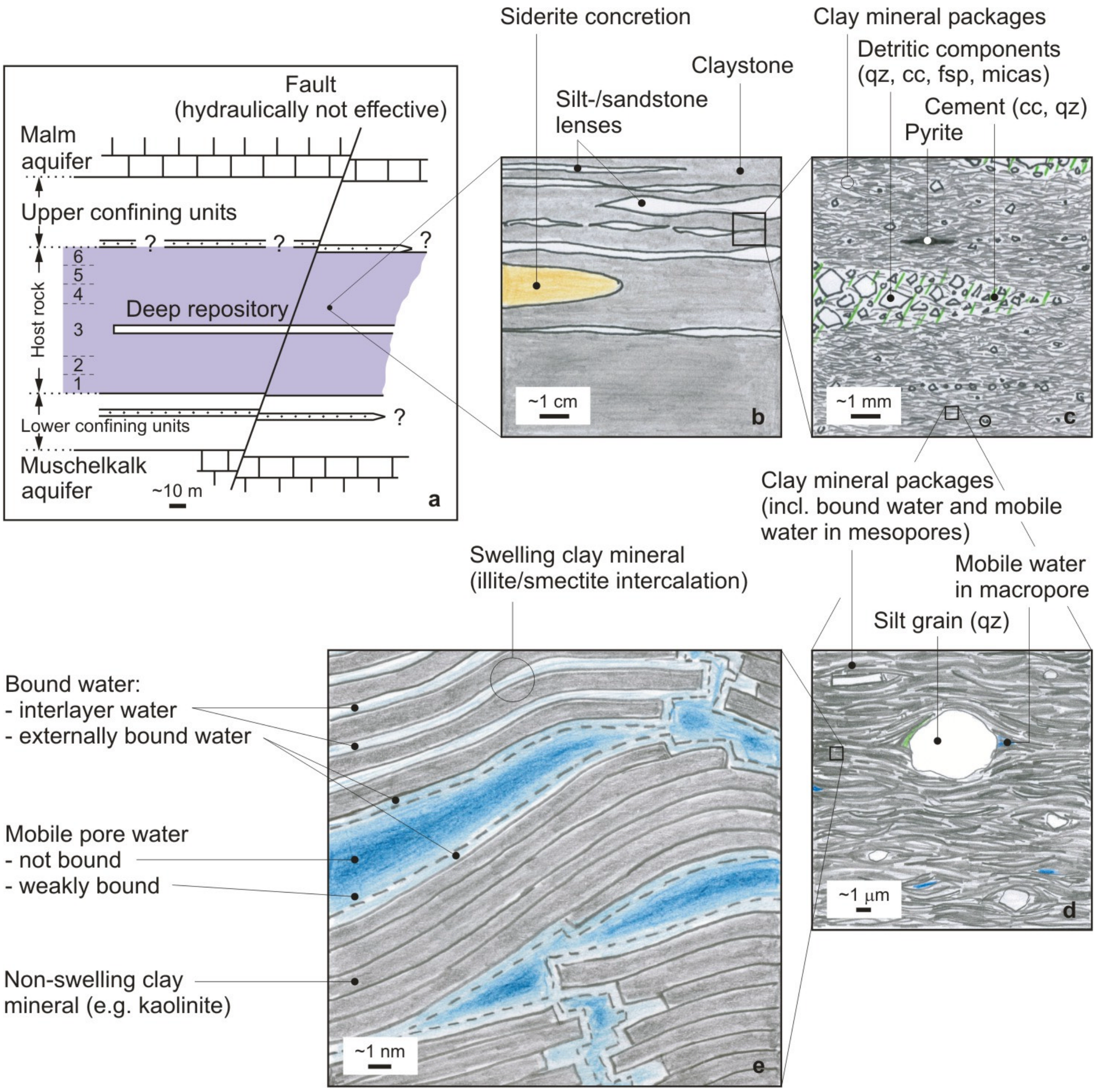


Fig. 4.2-13: Texture and structure of the Opalinus Clay, illustrating characteristics on various scales that give rise to anisotropy in transport properties (see text and Nagra 2002a)

The diffusion properties of Opalinus Clay have been investigated in both laboratory and field studies. The latter includes migration experiments and analysis of the large scale distribution of solutes and isotopes carried out in the Mont Terri Rock Laboratory. The obtained diffusion data show general agreement at different temporal and spatial scales (Nagra 2002a). The laboratory diffusion constants for Benken samples are somewhat lower than those obtained for Mont Terri samples (Van Loon et al. 2002 and 2003). The effective diffusion coefficients of tritium for Benken samples obtained from laboratory measurements are about 6×10^{-12} $m^2 s^{-1}$ perpendicular to the bedding plane. For anions, effective diffusion constants are 10 times lower. The diffusion constants parallel to bedding are about five times higher, consistent with the strong anisotropy of the medium. The diffusion accessible porosity for tritium was found to be in the range of 0.12 - 0.15, whereas anions show significantly lower values of about 0.06. Very few diffusion data exist for cations. Preliminary data (Van Loon et al. 2003) suggest that effective diffusivities for Na^+ are somewhat higher (a factor of 2) compared to tritium. Comparison of diffusion data for Opalinus Clay with those from other clay formations reveals a correlation between diffusion constants and porosity. Thus claystones with a similar degree of consolidation, such as the Callovo-Oxfordian of Meuse/Haute-Marne and the Toarcian of Tournemire, show similar diffusion coefficients, as shown in Fig. 4.2-12.

The observed low conductivities and the anisotropies in conductivity and diffusivity are due to the structure of Opalinus Clay, which is illustrated in Fig. 4.2-13. On a medium scale (~ 1 mm) the clay particles with their plate-like geometry (length 10 – 1000 times their thickness), which on the average make up about 50 % of the minerals, are horizontally bedded. This bedding is responsible for the observed anisotropies in conductivity and diffusivity. On a nm-scale, the specific properties of the clay minerals and their interaction with the porewater are most important. Approximately 75 % of the pores have apertures in the range of 1 – 25 nm. This very fine pore structure is the reason for the very low hydraulic conductivities despite the significant water content of the Opalinus Clay. Because of the interaction of the porewater with the surfaces of the clay minerals, only a fraction (approx. 50 %) is free water, the remainder being bound.

On a larger scale (10 m), fracture zones (faults) must be considered. Large faults are not present within the potential repository area (see Section 4.2.2), but smaller fracture zones cannot be ruled out. However, none of the discontinuities observed in Nagra's deep boreholes at Benken and Schafisheim revealed enhanced permeabilities in comparison to the rock matrix. Faults, shear-zones and joints have been observed in Opalinus Clay at different locations and depths in tunnels intersecting the formation. However, significant transmissivities (~ 10^{-10} m^2 s^{-1}) are not observed at locations with overburdens larger than 200 m, a fact that can be explained by the efficient self-sealing capacity of Opalinus Clay (Gautschi 2001). A more detailed description of the properties of Opalinus Clay for the different scales can be found in Nagra (2002a).

Evidence for the high isolation capacity of the Opalinus Clay and its strong barrier function within the hydrogeological system of the sedimentary sequence comes from detailed analyses of porewater chemistry, and models of how this has evolved with time. Fig. 4.2-14 shows measured concentration profiles across the Opalinus Clay and adjacent rock strata of the two naturally occurring stable isotopes, ^{18}O and ^{2}H, from core samples from the Benken borehole. The isotopes were assumed to be originally distributed approximately uniformly across the Opalinus Clay ~ 1 million years ago, before the current regional groundwater flow system was established. ^{18}O and ^{2}H have subsequently migrated outwards into the over- and underlying aquifers, modifying the uniform concentration profiles. The reason for the movement of these isotopes is changing water composition in the more permeable formations above and below, caused by flushing of the aquifers with younger waters. This caused a concentration gradient away from the centre of the claystone sequence, leading to diffusive transport of the isotopes outwards into the aquifers. The measured profiles are compared in Fig. 4.2-14 to model profiles for which it is assumed that diffusion has occurred for 0.25, 0.5 or 1 Ma. Calculations illustrate that significant deviations from the measured isotope profiles occur for flows in excess of 10^{-12} m s^{-1} (Gimmi & Waber 2003).

Fig. 4.2-14 provides compelling evidence for the dominant role of diffusion in controlling porewater compositions in the host clay formations and, by analogy, in controlling the movement of any radionuclides released into those porewaters from a repository.

Appendix 2 (Tabs. A2.8 and A2.9) summarises the key hydraulic parameters relevant to water and solute movement in Opalinus Clay in its present state. The potential evolution of these hydraulic properties over time, including the effects of repository-induced disturbances, is discussed in Section 5.5.3.

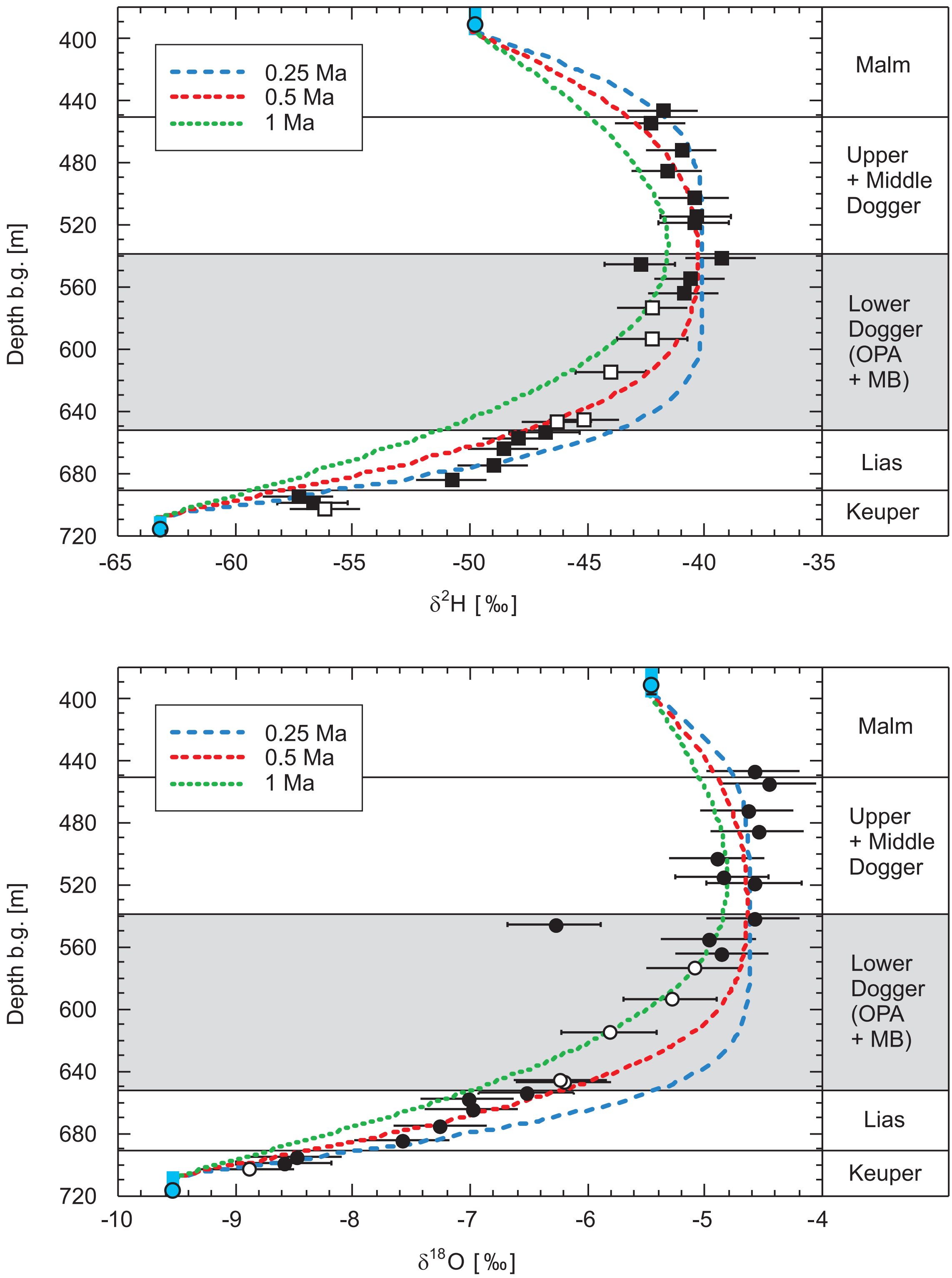


Fig. 4.2-14: Isotope concentration profiles in porewater across the Opalinus Clay (OPA) and adjacent rock strata due to diffusion that occurred for 0.25, 0.5 and 1 Ma.

Results are based on measured data (Nagra 2001, 2002a) obtained under various conditions (data points) and modelling (Nagra 2002a, Gimmi & Waber 2003) assuming diffusion only, presented as δ values relative to an international standard water (V-SMOW).

4.2.6 Geochemistry and mineralogy of the Opalinus Clay

Porewater chemistry in the host clay formation is of central interest when evaluating the evolution of the engineered barriers and the retention of any escaping radionuclides. Porewater chemistry and mineralogy affect, among other things, the sorption and retardation potential of the host formation.

The average mineralogical composition (reference mineralogy) of the host rock is presented in Tab. 4.2-5. The presence of pyrite and siderite, which show no signs of oxidation, indicate the reducing nature and the high redox-buffering capacity of the Opalinus Clay. The mineralogy in shear zones does not differ from that of the rock matrix (Nagra 2002a).

Tab. 4.2-5: Average mineralogy of the Opalinus Clay (Nagra 2002a)

Mineral	**wt % (average)**	**Standard deviation (1σ) wt %**
Illite	18	± 6
Kaolinite	17	± 6
Illite/smectite mixed layer	14	± 4
Chlorite	5	± 2
Quartz	20	± 5
Calcite	16	± 10
Dolomite/ankerite	1	± 0.4
Siderite	4	± 2.4
Feldspar	3	± 1.3
Pyrite	1.1	± 1
Organic carbon	0.6	± 0.3

Due to its marine origin, the porewater of the Opalinus Clay is relatively saline and sodium-chloride dominated. Based on investigations on cores from Benken, extensive studies at Mont Terri and geochemical modelling, Pearson (2002) derived the so-called reference porewater chemistry for Opalinus Clay (see also Nagra 2002a). This composition is given in Tab. 4.2-6 and it corresponds to the most probable water composition based on current understanding. The pH conditions are expected to be near-neutral but the uncertainties are rather large mainly because the partial pressure of CO_2 cannot presently be precisely constrained. Bounding pH values of 6.9 to 8.2 have been proposed (Pearson 2002). Redox conditions are reducing as evidenced by the large amounts of unoxidised pyrite and siderite. From mineralogical observations, Eh measurements performed at Mont Terri and geochemical modelling, redox potentials of about -170 mV (SHE) for the reference water and bounding values of about -140 to -240 mV were derived. The derivation is based on the assumption of thermodynamic equilibrium between pyrite, sulphate and siderite. The assumption of sulphate/pyrite equilibrium is considered reasonable, given the very long residence time of the porewater (Grenthe et al. 1992), even though the role of sulphate-reducing bacteria in Opalinus Clay is uncertain (Nagra 2002a, Stroes-Gascoyne 2002).

Tab. 4.2-6: Reference water chemistry of the Opalinus Clay at the Benken site (Pearson 2002)

pH	7.24
Eh [V]	-0.167
temperature [°C]*	25.0
log pCO_2	-2.2
concentrations [mol l^{-1}]	
CO_3 (tot)	2.70×10^{-3}
Na	1.69×10^{-1}
K	5.65×10^{-3}
Mg	7.48×10^{-3}
Ca	1.05×10^{-2}
Sr	3.04×10^{-4}
S(VI)	2.40×10^{-2}
S(-II)	1.41×10^{-11}
F	1.67×10^{-4}
Cl	1.60×10^{-1}
Br	2.40×10^{-4}
Fe (II)	4.33×10^{-5}
Mn	2.42×10^{-5}
Si	1.78×10^{-4}

* The temperature refers to the model water composition of Pearson (2002); the actual temperature of the formation at 650 m at Benken is 38 °C

The amount of dissolved organic carbon (DOC) is from 3 to 250 mg l^{-1}, with a low humic and fulvic fraction. The highest DOC values are probably caused by contamination by drilling. The largest part is composed presumably of low-molecular weight organic molecules or organic macromolecules (Glaus et al. 2001). The role of colloids in Opalinus Clay porewater has been assessed by Voegelin & Kretzschmar (2002). The small pore size in Opalinus Clay (the majority are in the 1 to 25 nm range) is expected to be a determining factor in limiting colloid mobility. Microbial activity in the Opalinus Clay is considered to be low due to the small size of the pores, their poor interconnectivity and the restricted availability of water (Stroes-Gascoyne 2002). Nevertheless, a catalytic influence of microbes on redox processes is conceivable.

4.3 The surface environment

4.3.1 Regional description

Any radionuclides released from the repository that do not decay during transport through the engineered and natural barriers would be likely to reach the surface environment by advective transport in deep groundwaters within the aquifers overlying and underlying the Opalinus Clay. For present-day conditions, discharge of deep groundwaters passing nearby the planned repository site is most likely to occur:

- in the Rhine valley below the Rhine Falls for the case of lateral transport in the Malm aquifer,
- in the Rhine valley further downstream for the case of lateral transport in the Wedelsandstein ("W" in Fig. 4.2-10a),
- into the Klettgau aquifer for the case of lateral transport in the Sandsteinkeuper ("Sk" in Fig. 4.2-10a),
- in the area near the confluence of Aare and Rhine[66] for the case of lateral transport in the Muschelkalk aquifer.

Discharge of deep groundwaters occurs predominantly either into rivers or into Quaternary gravels laid down in the valley bottoms. Discharge into valleys filled with impermeable lake sediments or into higher lying areas is thus very unlikely. In some places, however, deep groundwater discharge into springs located at valley sides has been observed. In the area of interest, these springs are all located north of the Rhine valley, with elevations higher than the deepest discharge points in the bottom of the Rhine valley (the basin of the Rhine Falls, see Nagra 2002a). For this reason, radionuclide discharge into springs located at the valley sides, conveyed by deep groundwaters from the Malm aquifer, is unlikely.

The present-day conditions in these discharge areas serve as a basis for the definition of different stylised possibilities for the characteristics and evolution of the surface environment in the distant future, including the possibility of discharge into a tributary river valley.

The surface elevation in the region of interest is 300 to 700 m above sea level and the topography of the Rhine valley is characterised today by relatively gently sloping valley bottoms, with gravel terraces laid down in the Quaternary period (Fig. 4.3-1). The river Rhine forms the major drainage feature in the region and would thus also constitute the main transport path for radionuclides in the surface environment away from the potential discharge areas. The other major river in the region (the Aare) has similar characteristics. Various tributaries of the Rhine river exist in the region of interest (e.g. Thur, Töss, Glatt).

The mean volumetric water flow rates in the Rhine at the Rhine Falls is 1.2×10^{10} m^3 a^{-1}, in the Aare at the village of Stilli is 1.8×10^{10} m^3 a^{-1}, and in the Rhine at the town of Rheinfelden is 3.3×10^{10} m^3 a^{-1}. The mean volumetric water flow rates in tributary rivers to the Rhine are: Thur (Andelfingen) 1.5×10^9 m^3 a^{-1}, Töss (Neftenbach) 2.5×10^8 m^3 a^{-1}, and Glatt 2.7×10^8 m^3 a^{-1} (LHG 1999).

The sub-surface water flow rates in the Quaternary gravels along the Rhine river downstream of the Rhine Falls and along the Aare near the confluence with the Rhine are in the order of 10^5 to

66 Another possibility is discharge into the Neckar river in the Stuttgart area. This case is not considered further, as it represents an extremely long flow path with corresponding additional retention and decay.

10^7 m^3 a^{-1} (Nagra 2002c and 2003b). The water flow rates in the gravels along the tributary rivers of the Rhine between Koblenz and the Rhine Falls vary between 5×10^5 and 10^7 m^3 a^{-1}. These water flows are due to inflow from gravels in upstream areas, from the river, from overlying soils (rainfall), from valley sides and from regional aquifers (deep groundwaters). The distance from the groundwater table to the surface of the gravel terraces in the Rhine and Aare valley is typically 20 m, but in some places the water table is near to the surface. In tributary valleys this depth is in general much smaller.

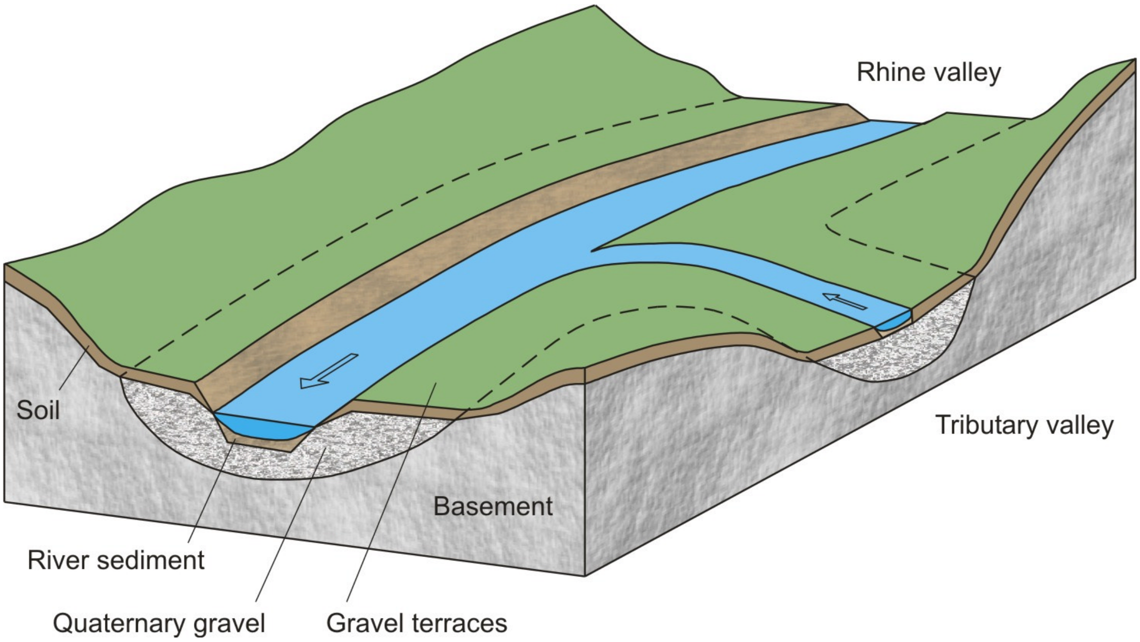


Fig. 4.3-1: Schematic illustration of the surface environment in the Rhine valley, indicating the main topographic and near-surface geological features

4.3.2 Climate

Over the long timescales considered in the safety assessment, climatic changes are expected. These are discussed in Section 5.2.1. The present-day climate is taken as a starting point for the definition of reference biosphere conditions in the future.

The present climate in the region is temperate and shows little spatial variation along the section of the Rhine valley of interest in this study. The mean annual temperature is approximately 9 ^{o}C (Atlas der Schweiz 1965 – 1990, Sheet 11), the annual rainfall varies between 800 and 1200 mm a^{-1} (Spreafico et al. 2001, Sheet 2.2) and the annual evapotranspiration rate is 550 to 600 mm a^{-1} (Spreafico et al. 2001, Sheet 4.4), all showing considerable seasonal variation.

4.3.3 Soils and the natural environment

In the region of interest, the predominant soils are neutral "Parabraunerde" (parabrownearths) in the valley bottoms and weakly acidic "Braunerde" (brownearths) in the valley sides (Atlas der Schweiz 1965 – 1990). The Parabraunerde has coarse particle sizes, being derived from the terraces and moraines, shows good drainage properties and exhibits translocation of clay. In addition, both soil types can contain pockets of reduced iron minerals (pseudogley). Original vegetation is almost entirely absent from the region. The present rural landscape is the result of human cultivation over the past centuries. On the basis of soil types and topography, the original vegetation is considered to have been deciduous forests (e.g. beech and oak).

4.3.4 Human habitation and economy

The human population has had a large influence on the regional environment. The present population density varies between 50 and 500 persons km^{-2} (Atlas der Schweiz 1965 – 1990, Sheet 24b). Subsistence agriculture[67] has been practised in the region until comparatively recent times. Based on today's agricultural productivity, about 1 hectare of agricultural land is required to feed 1 person, whereas in subsistence agriculture about 2 hectares per person are needed. (Nagra 2003b). At present, the regional non-agricultural economy consists mainly of small businesses and light industry with a strong service sector (Atlas der Schweiz 1965 – 1990, Sheets 60, 61 and 62).

4.3.5 Land use and agriculture

Agriculture is important to the regional economy (Atlas der Schweiz 1965 – 1990, Sheets 60, 61 and 62). This includes animal husbandry, arable farming and forestry. The agricultural land is devoted to arable and fruit production and livestock grazing (pasture land) or for growing animal foodstuffs (predominantly maize).

4.3.6 Water resources and usage

The near-surface hydrology is determined by climate and water flow in the Quaternary deposits. In addition, the near-surface hydrology is locally influenced by the presence of dams used for electrical power generation. In the region of interest, the Quaternary gravels in the valley bottoms comprise important aquifers. These water resources are used extensively (Stäuble 1993); they would be likely water sources for future communities, such as those considered in the biosphere model employed in this assessment.

There are several extraction sites from Quaternary aquifers in the region, with flow rates of up to 5000 l min^{-1} (2.6 x 10^6 m^3 a^{-1}) (Grundwasserkarten 1995). In addition, there are a number of mineral and thermal water wells (e.g. Zurzach, with a flow rate of 600 l min^{-1}).

4.4 Overview of the repository layout and the multi-barrier system

Fig. 4.4-1 illustrates a possible layout for the repository, which would be placed in the centre of the Opalinus Clay formation, and Figs. 4.4-2, 4.4-3 and 4.4-4 provide overviews of the system of safety barriers for each of the three waste types SF, HLW and ILW. In each case, the principal components of the multi-barrier system are listed, together with the key attributes that contribute to the safety functions, defined in Chapter 2 and discussed further in Chapter 6.

SF assemblies consist of ceramic uranium oxide (UO_2 or MOX) pellets contained in a zirconium alloy (Zircaloy) cladding, along with other structural materials such as steel alloys. The radionuclides of HLW are incorporated into borosilicate glass. The ILW, which contains much lower activity than SF or HLW, is embedded within a cement or, in some cases, a bitumen solidification matrix in steel drums.

[67] Subsistence agriculture means that humans produce all their dietary needs locally.

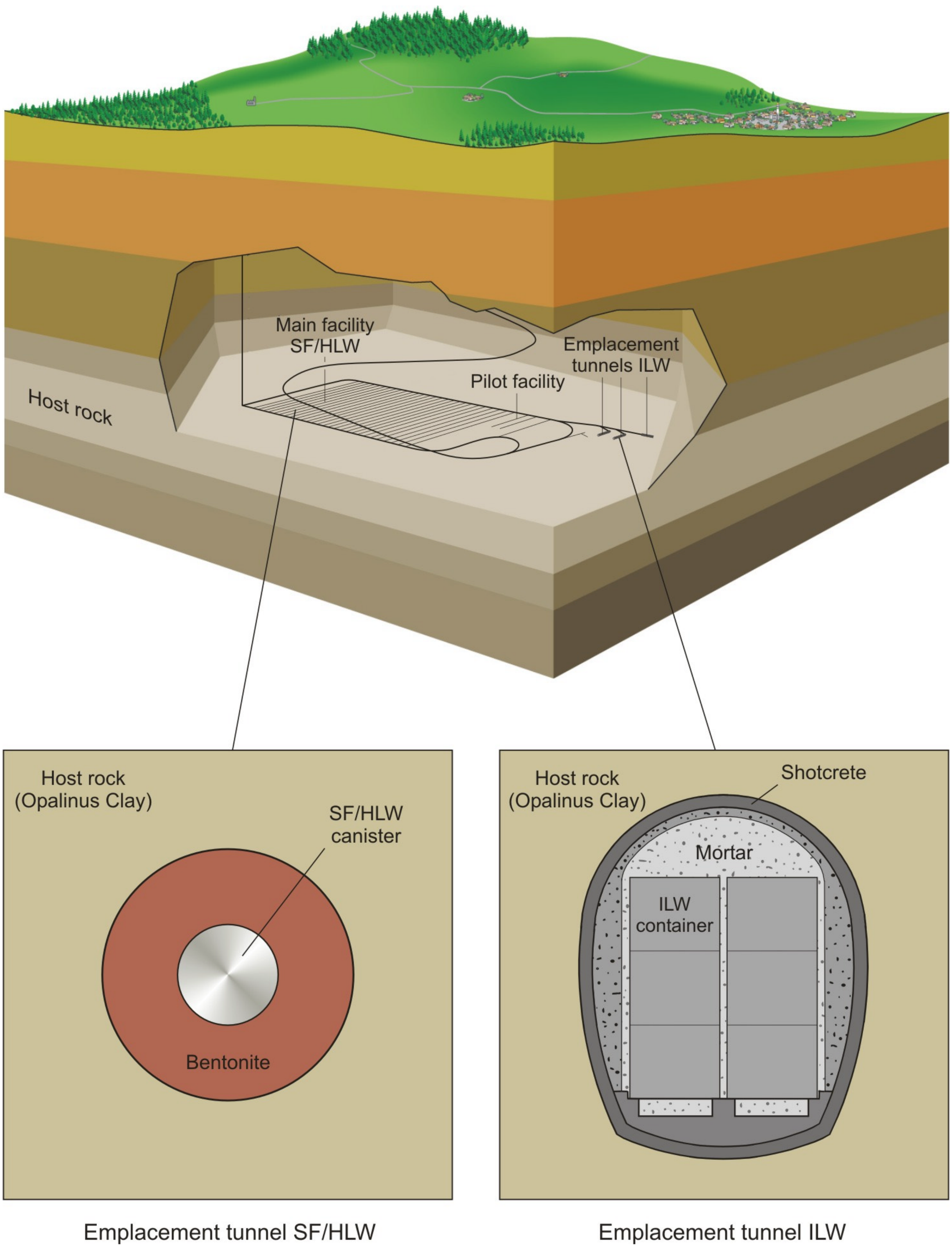
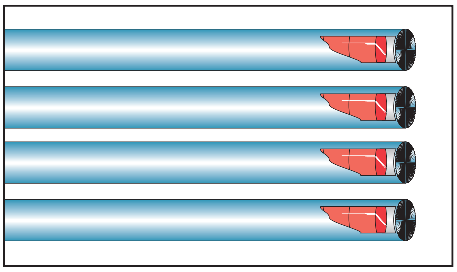


Fig. 4.4-1: Possible layout for a deep geological repository for SF / HLW / ILW in Opalinus Clay

Safety barrier system for spent fuel

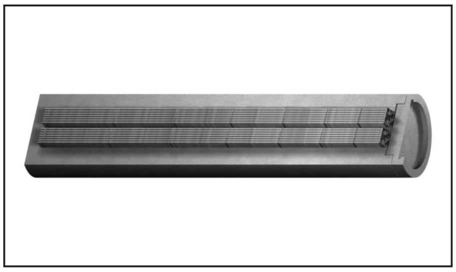
Spent fuel assemblies

- Confinement
  - Containment of radionuclides in spent fuel pellets and Zircaloy cladding
- Attenuation of releases
  - Low corrosion rates of spent fuel pellets and Zircaloy



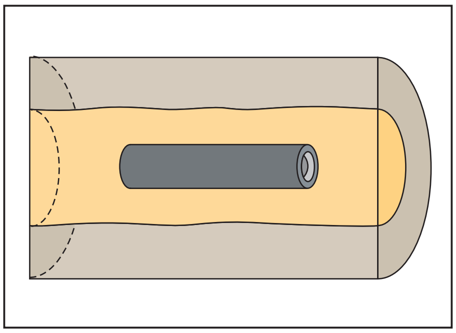
Steel canister

- Confinement
  - Prevents inflow of water and release of radionuclides from spent fuel for several thousand years
- Attenuation of releases
  - Corrosion products act as reducing agent (giving low radionuclide solubilities)
  - Corrosion products take up radionuclides



Bentonite backfill

- Confinement
  - Long resaturation time
  - Plasticity (self-sealing following physical disturbance)
- Attenuation of releases
  - Low solute transport rates (diffusion)
  - Retardation of radionuclide transport (sorption)
  - Low radionuclide solubility in pore water



Geological barriers

Host rock

- Confinement
  - Absence of water-conducting features
  - Mechanical stability
- Attenuation of releases
  - Low groundwater flux
  - Retardation of radionuclide transport (sorption and colloid filtration)

Geosphere

- Confinement
  - Physical protection of the engineered barriers (e.g. from glacial erosion)
- Attenuation of releases
  - Retardation of radionuclide transport (sorption)
  - Dispersion

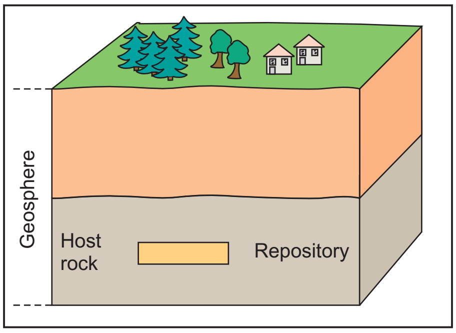
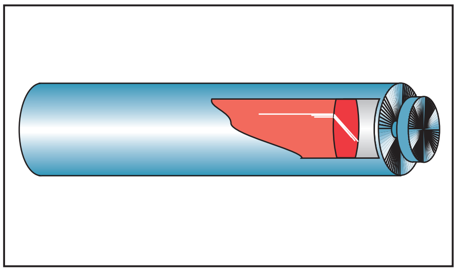


Fig. 4.4-2: The system of safety barriers in the case of SF

Safety barrier system for vitrified HLW

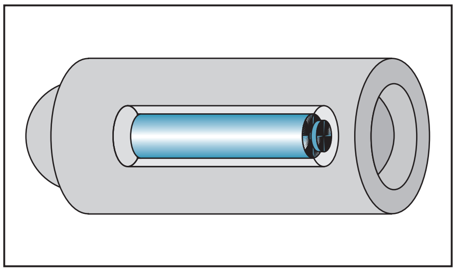
Glass matrix (in steel flask)

- Confinement
  - Containment of radionuclides in glass
- Attenuation of releases
  - Low corrosion rate of glass



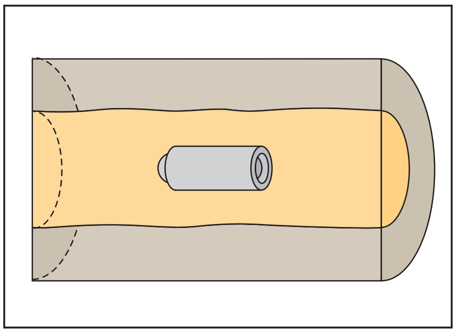
Steel canister

- Confinement
  - Prevents inflow of water and release of radionuclides from waste for several thousand years
- Attenuation of releases
  - Corrosion products act as reducing agent (giving low radionuclide solubilities)
  - Corrosion products take up radionuclides



Bentonite backfill

- Confinement
  - Long resaturation time
  - Plasticity (self-sealing following physical disturbance)
- Attenuation of releases
  - Low solute transport rates (diffusion)
  - Retardation of radionuclide transport (sorption)
  - Low radionuclide solubility in pore water



Geological barriers

Host rock

- Confinement
  - Absence of water-conducting features
  - Mechanical stability
- Attenuation of releases
  - Low groundwater flux
  - Retardation of radionuclide transport (sorption and colloid filtration)

Geosphere

- Confinement
  - Physical protection of the engineered barriers (e.g. from glacial erosion)
- Attenuation of releases
  - Retardation of radionuclide transport (sorption)
  - Dispersion

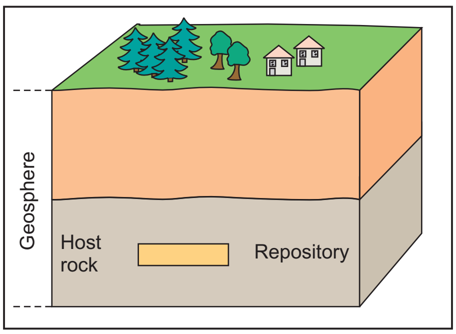
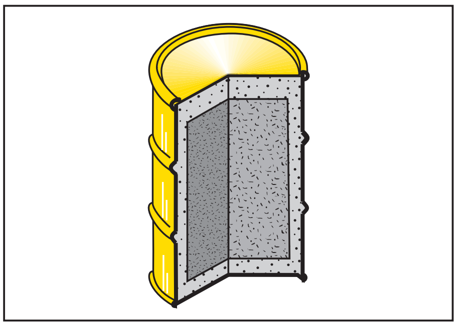


Fig. 4.4-3: The system of safety barriers in the case of HLW

Safety barrier system for ILW

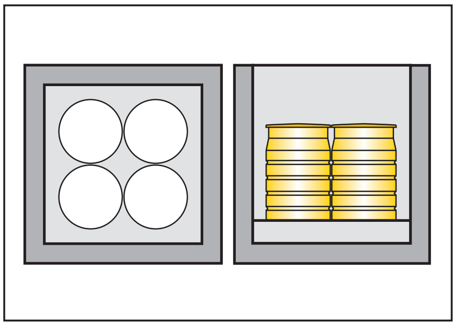
Solidification matrix
(cement)

- Confinement
  - Fixation of radionuclides in the cement / bitumen matrix
- Attenuation of releases
  - Low leaching rate of cement matrix
  - Low corrosion rate of metallic waste components



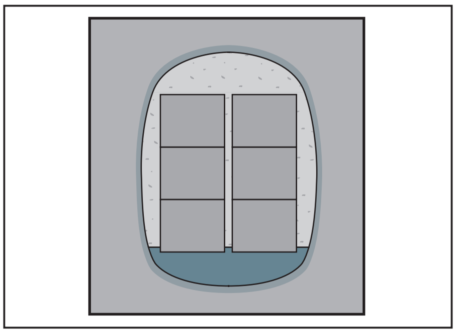
Emplacement containers, backfill and waste drums
(concrete / cement grout / steel)

- Confinement
  - Containment of radionuclides in steel waste drums and isolation with grout
- Attenuation of releases
  - Sorption of radionuclides on the grout
  - Low radionuclide solubilities in pore water



Emplacement tunnel and surrounding host rock
(repository zone)

- Confinement
  - Isolation of wastes with grout and liner
  - Mechanical stability
- Attenuation of releases
  - Radionuclide sorption
  - Low radionuclide solubilities
  - Low water flux



Geological barriers

Host rock

- Confinement
  - Absence of water-conducting features
  - Mechanical stability
- Attenuation of releases
  - Low groundwater flux
  - Retardation of radionuclide transport (sorption and colloid filtration)

Geosphere

- Confinement
  - Physical protection of the engineered barriers (e.g. from glacial erosion)
- Attenuation of releases
  - Retardation of radionuclide transport (sorption)
  - Dispersion

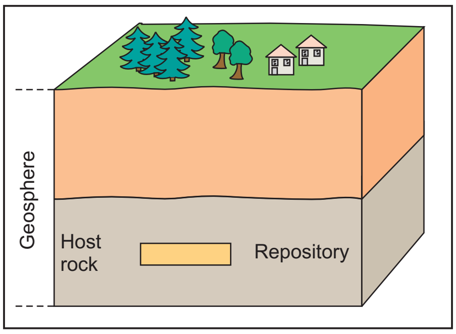


Fig. 4.4-4: The system of safety barriers in the case of ILW

In the proposed repository, carbon steel canisters containing either SF or HLW from reprocessing are emplaced co-axially within a system of parallel tunnels that are excavated in the mid-plane of the Opalinus Clay layer. For SF, a design variant of a copper canister of the design proposed by SKB (Werme 1998) is also considered. The tunnels are backfilled with highly compacted bentonite (blocks and granules) forming a buffer around the canisters.

The ILW drums are placed within concrete containers. These containers are placed in tunnels with a larger cross-section than those for SF / HLW. Void spaces within and around the containers are filled with cementitious grout.

Access to the system of waste emplacement tunnels is provided, during construction and operation, by a spiral ramp and operations and construction tunnels. A vertical shaft, also used for construction during enlargement of the facility, provides ventilation and the required emergency escape facility. During each year of the operational phase, it is foreseen that two SF / HLW tunnels will be excavated, waste emplaced and the tunnels sealed. At the end of the operational phase, the operations and construction tunnels and the shaft are backfilled with a bentonite-sand mixture and sealed with bentonite seals. More details are given in the following sections.

4.5 Detailed description of the disposal system at the time of end of waste emplacement

4.5.1 Repository layout

A plan view of the repository is shown in Fig. 4.5-1. The main elements include:

- an access ramp, construction and operations tunnels, central waste receiving facilities and a shaft,
- pilot and test facilities (see Chapter 2),
- an array of SF / HLW emplacement tunnels of 800 m length, spaced 40 m apart, and
- three short emplacement tunnels for ILW.

The waste emplacement tunnels and the operations and construction tunnels will be excavated at a depth of ~ 650 m in the centre of the Opalinus Clay formation, which dips gently to the south-east. Horizontal emplacement in tunnels was selected rather than borehole emplacement because of ease of emplacement and because the Opalinus Clay formation is 105 – 125 m thick, thus tunnel emplacement maximises the length of the radionuclide transport path to adjacent formations. Local variations in the dip of the formation mean that the tunnels are not always absolutely centered in the formation, thus the transport path may be slightly less than 50 m in some locations. The maximum principal stress is 22-23 MPa and is nearly horizontal with a N-S orientation. The minimum stress and lithostatic stress are 15 and 16 MPa, respectively. The SF / HLW tunnels (diameter = 2.5 m) are oriented in the direction of the maximum stress so as to maximise tunnel stability. These tunnels are expected to be self-supporting, based on experience with excavations in Opalinus Clay, although rock bolts and a light mesh are required for operational safety (Nagra 2002a and b). The possibility that some lining may be required is discussed in Section 4.5.3.4. All other tunnels, including the ILW emplacement tunnels and the access ramp, require concrete liners to ensure tunnel stability. The design and operations of the repository are described in detail in Nagra (2002b).

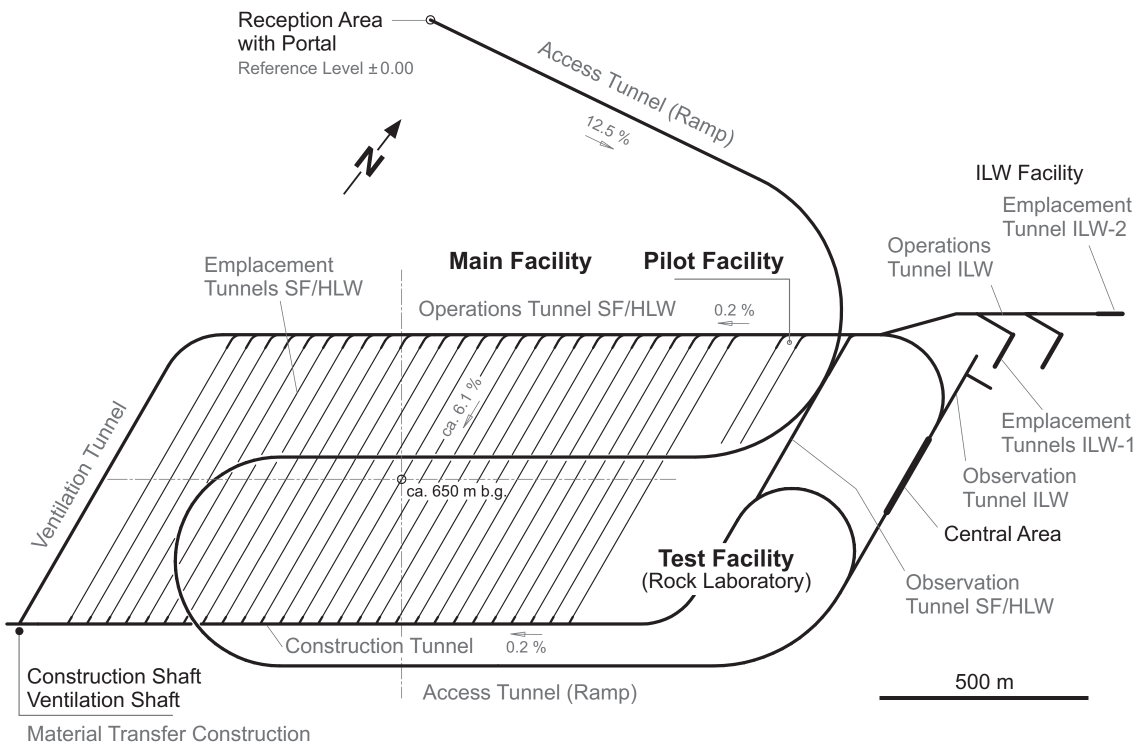


Fig. 4.5-1: Plan view of the repository for SF / HLW / ILW in the Opalinus Clay (see also Nagra 2002b)

4.5.2 Waste quantities and characteristics

The waste arisings are evaluated (McGinnes 2002) by assuming that the five power reactors in operation in Switzerland operate for 60 years (equivalent to 192 GWa (e)). The total SF arisings over this period are expected to be 4412 t_{IHM}[68], of which about 1195 t_{IHM} will have been reprocessed, resulting in the production of about 292 t of HLW glass. The remaining 1443 t_{IHM} of PWR UO_2 fuel, 1629 t_{IHM} of BWR UO_2 fuel and 145 t_{IHM} of MOX fuel, all in the form of complete fuel assemblies, are assumed to be conditioned for disposal in unreprocessed form. A variant inventory that corresponds to a 300 GWa (e) power system is also considered in both the design of the repository and in safety assessment calculations, in order to illustrate the impacts of a much larger inventory of waste. The model inventories of SF and HLW requiring disposal for these two variants are summarised in Tab. 4.5-1.

There are a variety of PWR and BWR fuel assemblies, varying in length, mass, numbers of fuel rods and fuel composition (MOX and UO_2). For the purposes of safety assessment calculations and to avoid discussion of a large number of different disposal canister loadings, some simplifications are made. As shown in Tab. 4.5-2, the Reference Case SF inventory in the repository is represented as 2065 canisters, 935 of which contain BWR UO_2 fuel (9 assemblies per canister), 680 of which contain 4 PWR UO_2 assemblies and 450 of which contain 1 MOX and 3 UO_2 assemblies. The radionuclide inventories of each of the canister types is given in Appendix 2, Tabs. A2.1.1, A2.1.2 and A2.1.3.

[68] The unit t_{IHM} refers to tonnes of initial heavy metal and relates to the original mass of U or U+Pu in the fuel assemblies.

Tab. 4.5-1: Model inventories of SF and HLW requiring disposal

Waste Type	Reference case inventory – 60 years power plant operation (192 GWa (e))	Variant inventory – 300 GWa (e)
SF	t_{IHM}	t_{IHM}
PWR UO_2	1443	2700
BWR UO_2	1629	2731
PWR MOX	128	128
BWR MOX	17	17
Total SF	3217	5576
HLW	t_{glass}	t_{glass}
	292 *	292 *

* representing reprocessing of about 1195 t_{IHM} of spent fuel

Tab. 4.5-2 Rounded number of disposal canisters of SF and HLW

Waste Type	Reference case inventory – Projected number of canisters for 60 years power plant operation (192 GWa (e))	Variant inventory – Projected number of canisters for a 300 GWa (e) nuclear power plant system
SF		
BWR UO_2	935	1630
PWR UO_2	680	1500
PWR (3 UO_2+1 MOX)	450	450
Total	2065	3580
HLW		
COGEMA	460	460
BNFL	270	270
Total	730 *	730 *

* representing reprocessing of about 1195 t_{IHM} of spent fuel

Additional wastes from reprocessing of SF are received from the COGEMA and BNFL facilities in the form of solidified ILW residues. The quantities of ILW requiring disposal are discussed in Section 4.5.2.3. In addition to these, small quantities of additional ILW are produced in research and, eventually, will arise from operation of the fuel encapsulation facility required for packaging the SF for disposal. In the design study (Nagra 2002b), some additional reserves (2900 m^3) are foreseen for wastes that may not be suitable for the low and intermediate-level waste repository. However, because the activity of this waste will be very small compared to the ILW discussed above, these wastes are neglected in the present study.

4.5.2.1 Spent UO_2 and MOX fuel

Quantities and burnup

The total masses of the various types of SF for disposal are given in Tab. 4.5-1. By far the largest quantities are UO_2 fuel, with the remainder being MOX. The present average burnup of fuel assemblies is 48 GWd/t_{IHM}, although burnups range from ~ 30 to 65 GWd/t_{IHM}[69]. The potential for higher average burnups exists in the future (burnups up to 75 GWd/t_{IHM} are presently envisaged, McGinnes 2002).

Structure of fuel assemblies

The nuclear fuel considered consists of cylindrical ceramic pellets of UO_2. Some fuel assemblies contain MOX fuel pellets, fabricated from a blend of PuO_2 (5 to 6 %) and UO_2 powders. Stacks of pellets are placed in tubes of zirconium alloy cladding (Zircaloy), which are then sealed with welds. A large number of fuel rods (~ 100 for BWR fuel and ~ 200 for PWR fuel, the exact numbers depending on the design) are then integrated to form a fuel assembly, which is held together with spacers and plates made of corrosion-resistant stainless steel and nickel alloys. The various non-fuel materials in the assemblies (Zircaloy cladding and other alloys) are collectively referred to as structural materials.

Radionuclide content and heat output

Irradiation of the fuel assemblies produces a large number of radionuclides. These include fission products, arising from fission of uranium and plutonium in the fuel pellets, and activation products, arising from neutron absorption. Some of the activation products, such as ^{14}C and ^{36}Cl, are present in both the fuel pellets and structural materials, whereas others, such as various actinide nuclides, are contained in significant amounts only within fuel pellets. Certain radionuclides are enriched at grain boundaries in the fuel, at pellet cracks and in the fuel / sheath gap as a result of thermally driven segregation during irradiation of the fuel in the reactor, as illustrated in Fig. 4.5-2. The quantities of the most important radionuclides in a typical canister containing either 9 BWR or 4 PWR UO_2 fuel assemblies, or 1 MOX PWR plus 3 UO_2 PWR fuel assemblies after decay for 40 years are given in Appendix 2 (Tabs. A2.1.1, A2.1.2 and A2.1.3) for fuel irradiated to an average burnup of 48 GWd/t_{IHM}. The heat output of a typical canister of SF is approximately 1500 W after 40 years of cooling (the minimum assumed pre-disposal storage period). The total α activity of MOX fuel after decay for 40 years is approximately 5 times larger than that of UO_2 fuel, while the fission product activity is approximately the same. An important consequence of this is the higher heat output of MOX fuel at longer times, as discussed in Section 5.3.1. The implications of this in relation to heat output constraints for SF canisters are discussed in Section 4.5.3.5.

4.5.2.2 Vitrified HLW

Production

HLW is the highly active residue from the reprocessing of nuclear fuel to recover U and Pu, and contains most of the radionuclides from the irradiated fuel. The HLW from reprocessing, in the form of a highly active liquid, is evaporated, calcined and mixed at high temperature with

[69] Burnup also varies along the length of a fuel rod as well as from one fuel rod to the next.

borosilicate-glass forming additives. Reprocessing is carried out by COGEMA in France and BNFL in the United Kingdom, producing HLW glasses with two different sets of characteristics. Quality control of the glass manufacturing ensures a homogeneous glass-waste matrix. The molten glass is poured into stainless steel fabrication flasks which are then sealed. To allow for differential expansion and to avoid spillage, a void space is left at the crown of each flask. Details of the composition of HLW glass are given in McGinnes (2002).

Radionuclide content and heat output

The radionuclide inventories of the COGEMA and BNFL vitrified HLW are known in detail and are described by McGinnes (2002). The compositions and radionuclide contents of the two glasses are similar. Appendix 2 (Tabs. A2.1.4 and A2.1.5) gives the radionuclide content of average flasks of BNFL and COGEMA HLW glasses after 40 years of decay. The decay heat of an average canister of BNFL HLW glass at the time of production is ~ 3500 W, decreasing to ~ 690 W after approximately 40 years of cooling. The COGEMA reference glass has a similar composition, with a decay heat and total activity at the time of production that is, for an average flask, ~ 20 % lower than of the BNFL glass. The evolution of decay heat and radionuclide inventory of the HLW after emplacement in the repository is discussed in Section 5.3.1.

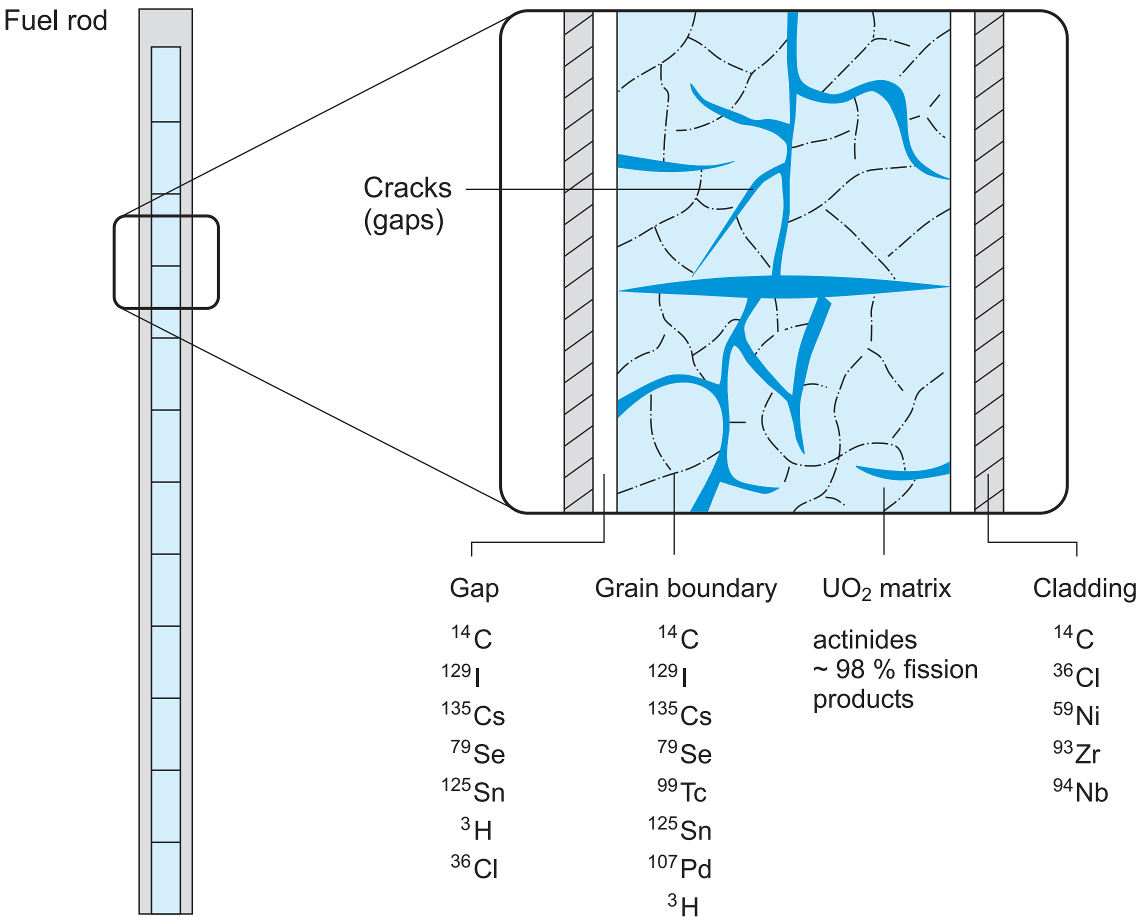


Fig. 4.5-2: Schematic illustration of the distribution of radionuclides within a fuel rod (based on Johnson & Tait 1997)

4.5.2.3 ILW

Quantities and types of ILW

A variety of ILW types are produced as a result of reprocessing of SF. These wastes include metals, organics and inorganic materials. They are mixed with either cementitious materials or bitumen and placed in steel and fibre-cement waste drums (McGinnes 2002) as shown in Fig. 4.5-3. The formally specified quantities and types of wastes to be returned from reprocessing are referred to as the "cemented waste" option. A second scenario, referred to as the "high force compacted waste" option, differs principally in that full compaction of fuel structural materials (hulls and ends) and technological wastes takes place (McGinnes 2002).

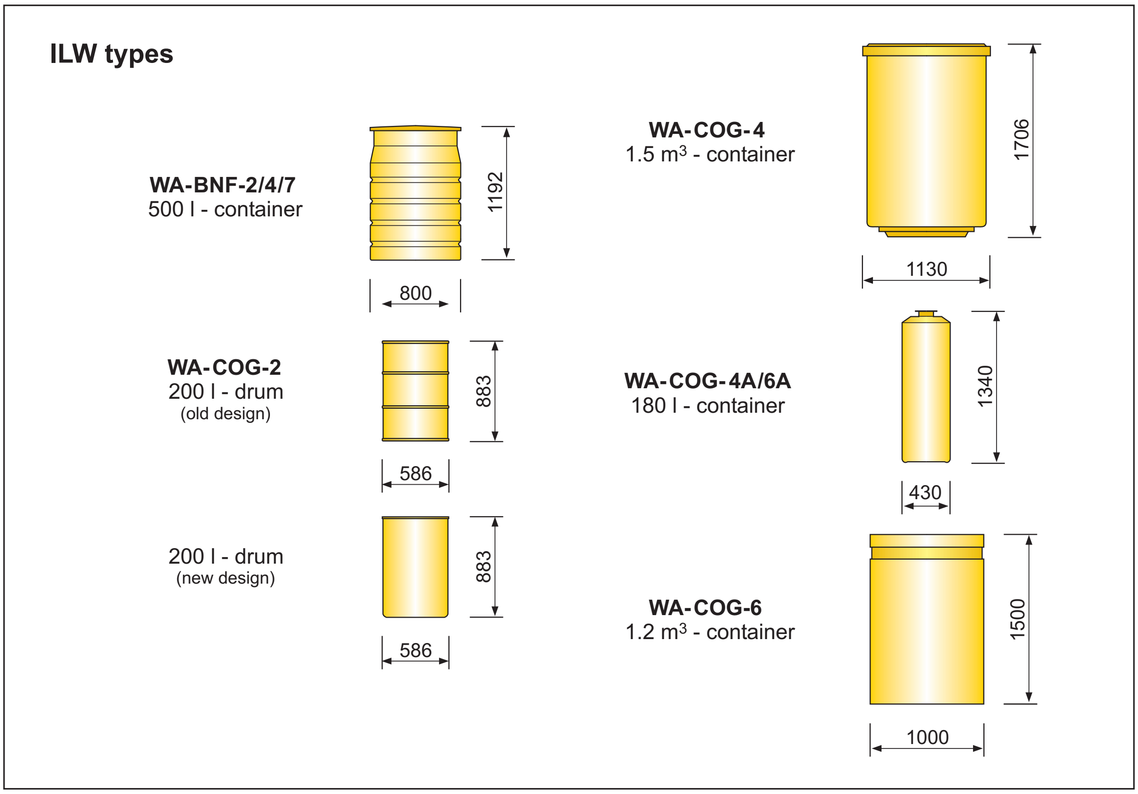


Fig. 4.5-3: The various waste drums containing ILW, see also McGinnes (2002)

Dimensions are in mm.

The inventory of materials present in the drums, including the concrete and bitumen solidification matrices, are summarised in Tab. 4.5-3 for the cemented waste option.

The mortar used to fill the void spaces both between the waste drums within the concrete emplacement containers and between the emplacement containers within the ILW tunnels has a porosity of about 25 %, thus providing some storage volume for gas generated as a result of corrosion and waste degradation processes after closure of the repository. The total numbers of waste drums are summarised in Tab. 4.5-4 for both the cemented waste and high force compacted waste options.

Tab. 4.5-3: The total inventories of materials in the ILW, excluding emplacement containers (cemented waste option, see McGinnes 2002)

Material	Mass [kg]	Material	Mass [kg]
Metals	8.9×10^5	High mol. wt. organics	7.4×10^4
Steels	5.6×10^5	Bitumen	4.7×10^4
Al/Zn	3.4×10^3	Cellulose	5.1×10^2
Inconel	1.6×10^4	Plastics	2.7×10^4
Zircaloy	3.0×10^5	Low mol. wt. organics	8.4×10^2
Other	3.2×10^3	Detergents	7.5×10^2
Inorganics	2.2×10^6	Flocculants	6.5
Salts	3.0×10^4	Complexing agents	9.1×10^1
Glass	3.1×10^2		
Cementitious material	2.1×10^6		
Other	2.3×10^3	*Total*	3.1×10^6

Tab. 4.5-4: The rounded number of the various waste drums for disposal of ILW

		Number of containers	
Waste Type	**Waste Sort**	**Cemented waste option**	**High force compacted waste option**
$BaCO_3$/Crud	WA-BNF-2	30	30
H&E* Cement	WA-BNF-4	270	270
Cent. Cake and Non Specified	WA-BNF-7	170	170
Bitumen	WA-COG-2	380	380
H&E Cement	WA-COG-4	320	-
H&E +Tech. ILW Compacted	WA-COG-4/6	-	1030
Tech. ILW	WA-COG-6	510	-

* hulls and ends

The ILW emplacement containers are divided between two different types of disposal tunnels in the repository, ILW-1 and ILW-2, which can be seen in Fig. 4.5-1. A small portion of the wastes, the COG-2 containers, a total of ~ 84 m^3 (300 m^3 packaged in disposal containers), contains materials such as nitrates and various chelating agents (Schwyn et al. 2003). In order to minimise the impact that these materials may have on other wastes in the repository, e.g. increasing radionuclide solubilities or decreasing sorption, they are emplaced in the smaller ILW-2 tunnel, which is well separated from the ILW-1 tunnels.

Radionuclide contents and heat output

The radionuclide inventories and heat outputs of the various waste container types are given in detail by McGinnes (2002). The combined decay heat of all containers at the time of waste emplacement is only ~ 1500 W, distributed over ~ 190 m of tunnel length. The total radionuclide inventory in the repository is given in Appendix 2 (Tabs. A2.1.6 and A2.1.7), apportioned between the ILW-1 and ILW-2 tunnels.

4.5.2.4 Waste acceptance criteria

Even though it is many years until a SF / HLW / ILW repository might be licensed, it is important to ensure that wastes being produced today could be disposed of in the future. This is ensured by the development of preliminary waste acceptance criteria (see IAEA (1990) for a general discussion of such criteria) and the collection of data from waste producers to ensure compliance with these criteria. Nagra has developed and implemented a set of rules regarding waste documentation and acceptance and formal disposability assessments have been performed since 1989 (Zuidema et al. 1997). The main emphasis is on proper quality assurance and documentation of waste properties relevant to disposal and on the feasibility of determining waste package properties. This provides to the waste producers at an early stage the necessary boundary conditions to prevent costly corrective actions. Preliminary waste acceptance criteria also provide some boundary conditions for developing the repository design and operational procedures.

The preliminary waste acceptance system takes into account the provisions of the regulatory guidelines (HSK 1988). Nagra has evaluated the waste specifications for the ILW and HLW considered in the present study based on this strategy. In the case of SF, formal acceptance criteria related to disposal have yet to be developed. Nonetheless, in order to evaluate the safety of a specific repository design, some quantitative preliminary waste acceptance criteria must be assumed at a very early stage and are related to the specifics of the repository design and safety assessment calculations. The assumed criteria are in some cases specific to the repository design, in particular the thermal power of waste packages. The most important criteria related to waste acceptance in the context of the present study and their relationship to assessment of the reference disposal system are:

Defined contents of radionuclides in the waste packages – Each of the waste types has a well defined radionuclide inventory that is known with reasonable accuracy. The uncertainties in radionuclide inventories are discussed by McGinnes (2002).

Defined maximum thermal power of waste packages – Thermal output at the time of disposal is limited so that the properties of the waste form and other repository components are not adversely affected. For ILW, the thermal output is very low, such that close packing of the waste emplacement containers discussed in Section 4.5.3.4 leads to low temperatures (maximum of ~ 50^{o} C). For SF and HLW, the thermal criteria (maximum heat output per canister) are discussed in detail by Johnson et al. (2002) and in Section 4.5.3.5. The criteria are derived based on an analysis of the temperatures reached by various repository components (waste form, canister, backfill, and rock) and an assessment of the impacts on performance. Techniques for deriving the radiation levels and associated thermal output of SF and HLW canisters are available. In the case of SF assemblies, it is recognised that scanning using neutron and γ measurement techniques may have to be performed prior to loading into disposal canisters. Such scanning techniques, which are presently available (Lebrun & Bignan 2001), permit burnup to be measured with a 5 % error range. This would provide sufficient information to ensure that both thermal and sub-criticality criteria are met (see below).

Design of waste canisters and arrays of waste canisters to ensure nuclear sub-criticality – This is a particularly important issue for SF, as discussed in Section 5.3.1.1, where it is noted that SF assemblies should reach a certain minimum level of burnup before they are acceptable for disposal in the reference canister from the perspective of avoiding criticality. It is also noted that other design measures can be taken in the rare cases where this minimum burnup might not be reached. The burnup of SF assemblies would be determined by non-destructive methods, as noted above.

Radiation effects, including dose rates (radiation safety) and radiation damage to materials – Dose rates on the outside of conditioned waste packages are calculated and considered in design of the repository waste handling and emplacement procedures, such that operational safety requirements would be met (Nagra 2002b). Radiation damage is considered in the selection and assessment of the various barrier materials; dose rates to materials are low enough that degradation of their properties is not expected, thus no quantitative criteria have been considered in the present study.

Gas generation – Gases are produced by a variety of processes occurring in waste packages. In the case of SF canisters, the production of such gases is low enough that the integrity of the canisters is not affected (Johnson & King 2003), thus no criteria are established for individual waste canisters in the present study. For ILW package types, gas generation may occur in storage and the primary canisters are designed with this gas production in mind. The post-disposal effects of gas generated within the disposal system are evaluated in Chapter 5.

Chemical durability – For HLW and ILW, detailed specifications are given by the waste producers COGEMA and BNFL, as discussed in McGinnes (2002). Detailed acceptance criteria and associated quality assurance procedures for SF disposal remain to be developed.

4.5.3 Engineered barrier system and repository design

4.5.3.1 SF canisters

Steel canister

The reference design concept for SF canisters involves a cast steel body, with a machined central square channel fitted with crossplates to permit emplacement of either 4 PWR or 9 BWR fuel assemblies, as shown in Fig. 4.5-4 (Johnson & King 2003). The canisters have a wall thickness of ~ 150 mm, a void volume of ~ 0.7 m^3 (when loaded) and weigh ~ 26 t. The design studies of the canister are at this stage conceptual in nature, with the feasibility of manufacturing the canister body and the basic structural performance (isotropic loading of the body) and corrosion performance evaluated to establish proof-of-concept. The details of design of the closure have not yet been established, but may involve a threaded seal followed by a closure weld seam, as in the case of the HLW canister (Steag & Motor-Columbus 1985). Detailed investigations of welding, stresses in the weld region and anisotropic loading scenarios have not yet been investigated, because it is judged that good industrial experience with carbon steel will lead to adequate understanding of these issues during a future optimisation stage. Although welding procedures have not been specified, there is good experience with TIG (tungsten inert gas) welding and electron beam procedures and this is an area where continuous improvements in technology can be expected (see the discussion in JNC 2000). The minimum design lifetime of the canister is 1000 years, although a 10 000 year lifetime might be more realistically expected (see Section 5.3.4.4).

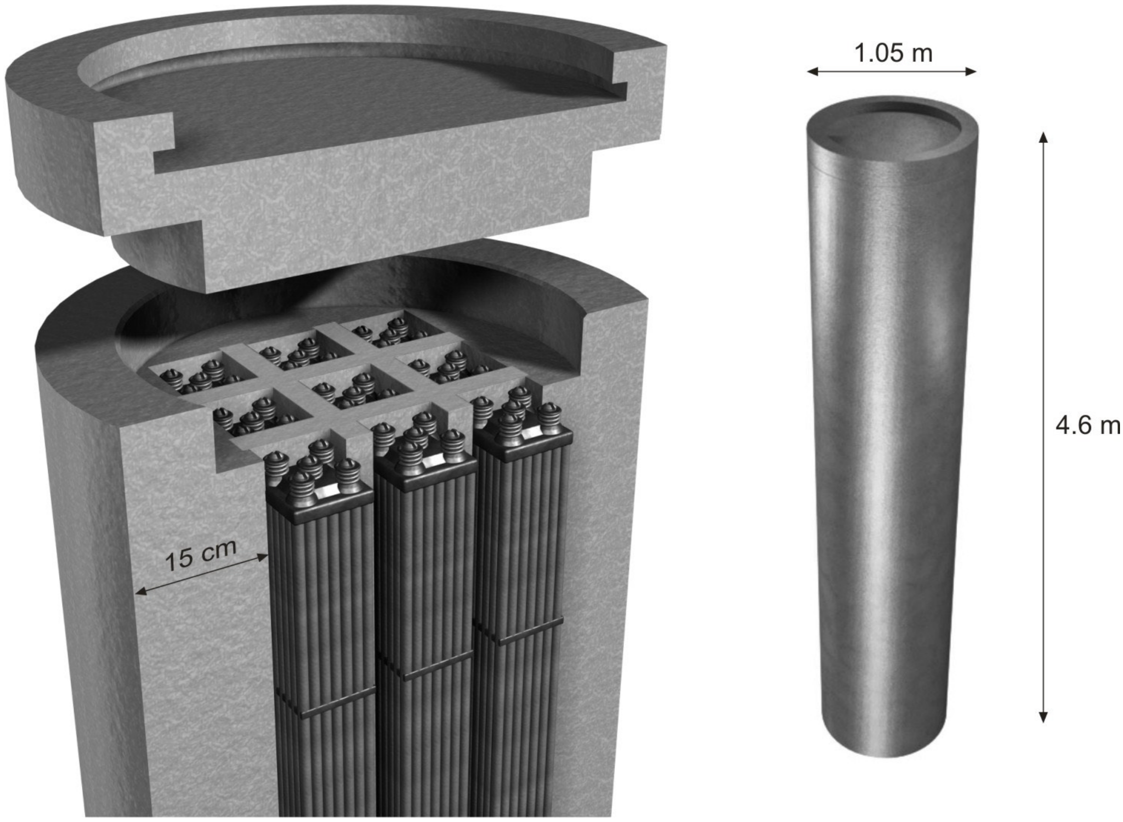


Fig. 4.5-4: Canister for the disposal of spent BWR fuel

The canister for PWR is similar in design, with channels to hold four of the larger PWR fuel assemblies (Johnson & King 2003).

Copper canister

An alternative design that incorporates a Cu shell with a steel insert is also considered, based on the SKB design (Werme 1998). A Cu canister of this type has a design lifetime of 100 000 years, although a corrosion evaluation suggests a lifetime in excess of 1 million years (King et al. 2001, Johnson & King 2003). In addition to the benefit of an increased canister lifetime, the choice of this canister design also essentially eliminates corrosion-induced H_2 production, except in the rare case of a prematurely breached canister.

Alternative materials

Options exist to use other materials for canister construction, in the event that it is considered necessary to completely eliminate gas production as a result of canister corrosion (see Section 5.5.2.3 for a discussion of the effects of gas production and buildup). It may be possible to design a chemically inert insert (e.g. ceramic or bronze) for a copper canister for SF so that no significant gas production would occur in the SF emplacement tunnels although this has not been considered in the present study.

4.5.3.2 HLW canister

The steel HLW canister is designed to contain a single stainless steel flask of HLW glass[70] and is shown in Fig. 4.5-5. The design is the same as that described in Steag & Motor-Columbus (1985) for the Projekt Gewähr study (Nagra 1985). The weight of the canister is 8.4 t and the wall thickness is 25 cm. The minimum design lifetime of the canister is 1000 years, although a 10 000 year lifetime might realistically be expected. Details of the performance of the canister are discussed in Section 5.3.4.4. For both the HLW and the SF canister, the steel also provides a chemical barrier after canister failure, ensuring reducing conditions and providing iron corrosion products that may strongly sorb many radionuclides.

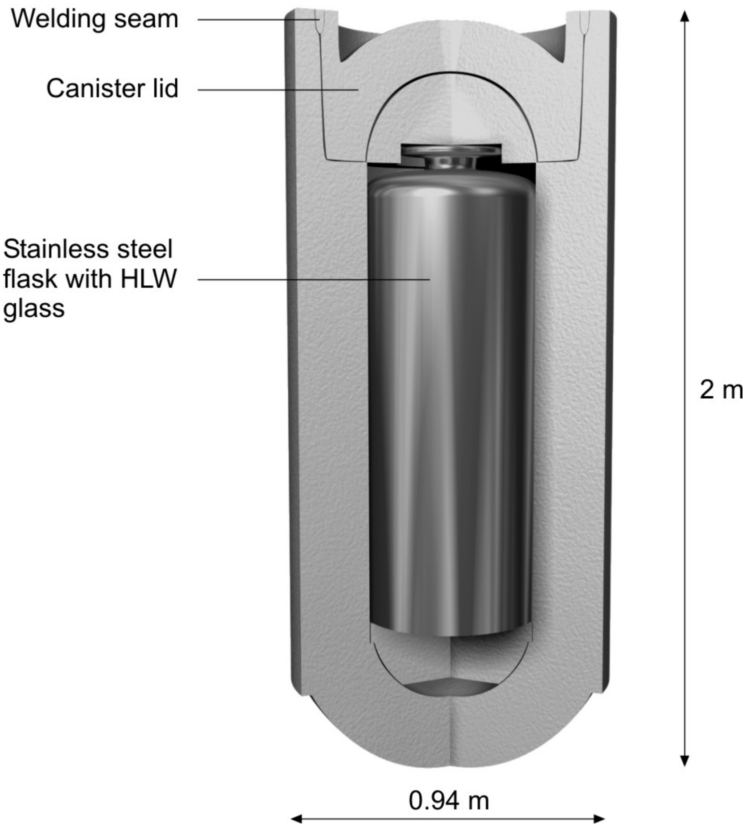


Fig. 4.5-5: Steel canister for the disposal of vitrified HLW (Steag & Motor-Columbus 1985)

4.5.3.3 ILW containers

The ILW drums are incorporated into concrete emplacement containers, as shown in Fig. 4.5-6, with cementitious mortar used to fill the void spaces around the drums.

4.5.3.4 Emplacement of wastes, backfilling, monitoring and repository sealing

The overall layout of the underground facilities is shown in Fig. 4.5-1. The major part of the facilities is the main facility with the emplacement tunnels for SF and HLW canisters. Prior to

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70 A canister of 3.25 m length containing two HLW flasks is also an option.

emplacement of SF / HLW canisters in the main repository, some canisters are emplaced in the pilot facility, which consists of a few short emplacement tunnels and an observation tunnel, but is otherwise constructed along the same principles and with the same materials as the main facility. The pilot facility is designed to permit monitoring of near field rock and the emplaced engineered barriers (see Chapter 2).

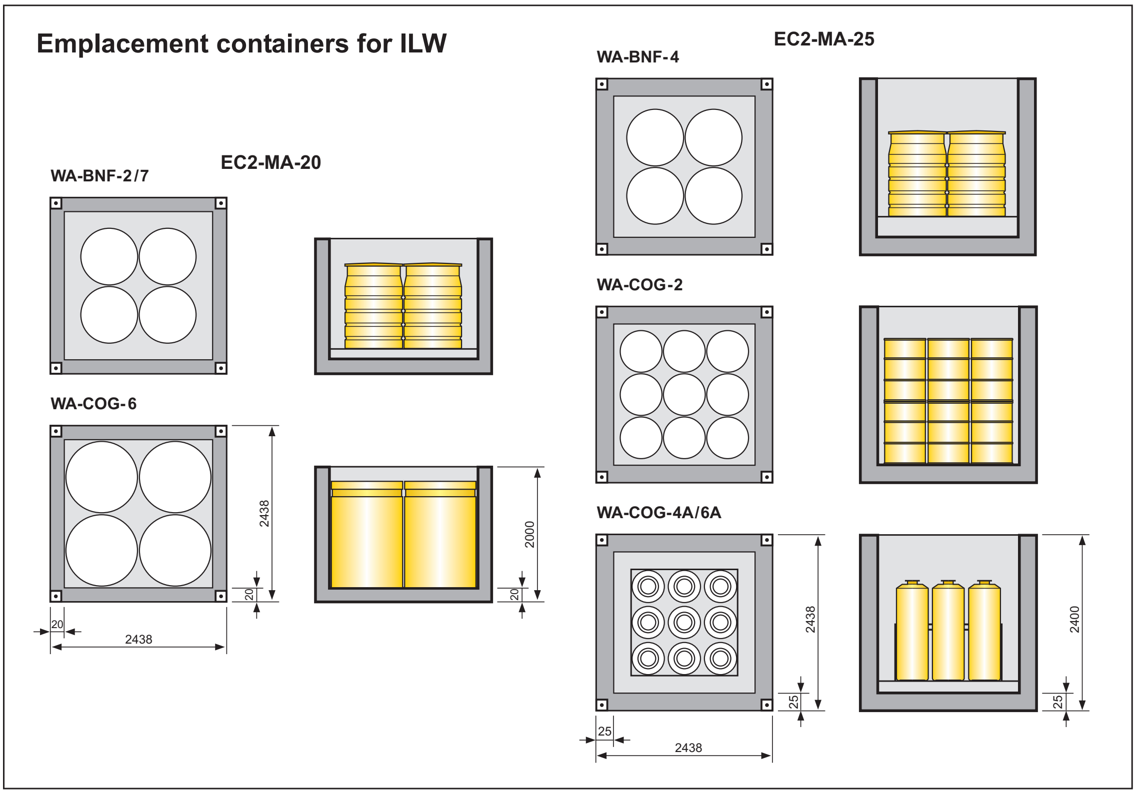

Fig. 4.5-6: Emplacement containers for ILW

Dimensions are in mm.

Emplacement of SF / HLW canisters

The emplacement tunnels for SF / HLW have a diameter of 2.5 m and are optimally aligned with respect to the regional stress field in order to reduce excavation-induced damage arising from stress relief. Design studies based on field measurements of geotechnical properties of Opalinus Clay suggest that under these conditions, the rock will be self-supporting, although a light reinforcing mesh and rock bolts will be required to ensure operational safety. Nonetheless, the possibility that the SF / HLW tunnels will require a liner cannot be completely excluded. In such a circumstance the interaction of liner material (e.g. concrete or synthetic polymer) with bentonite and Opalinus Clay would have to be evaluated.

Canisters of SF / HLW, contained in a transport shield, are moved by rail into position at the entrance of emplacement tunnels. Here the canisters are transferred onto bentonite blocks on an emplacement wagon, which is moved by rail to the final emplacement position (Nagra 2002b). The distance between canisters is 3 m.

Backfilling

The highly compacted bentonite blocks which support the canisters have a dry density of ~ 1.75 Mg m^{-3} (emplaced at an assumed initial moisture content of ~ 10 %). The region around the canisters is backfilled with granular bentonite using a pneumatic or conveyor system. The granular backfill has a moisture content of ~ 2 % (Naundorf & Wollenberg 1992), and comprises ~ 80 % by volume very dense granules (~ 2.1 – 2.2 Mg m^{-3}) and 20 % bentonite powder. Upon emplacement, the material is expected to have an average dry density of 1.5 Mg m^{-3} (Röski 1997). An important consequence of using high density, low moisture content granules is low thermal conductivity[71], as discussed in Section 5.3.2. The sequence of operations involving excavation, waste emplacement, backfilling and tunnel sealing ensures that a given emplacement tunnel is open for only one to two years, thus avoiding significant alteration of the Opalinus Clay at the tunnel periphery. The emplacement tunnels at the completion of backfilling are illustrated in Fig. 4.5-7. Note that both the terms buffer and backfill are used to describe the bentonite backfill surrounding SF / HLW canisters.

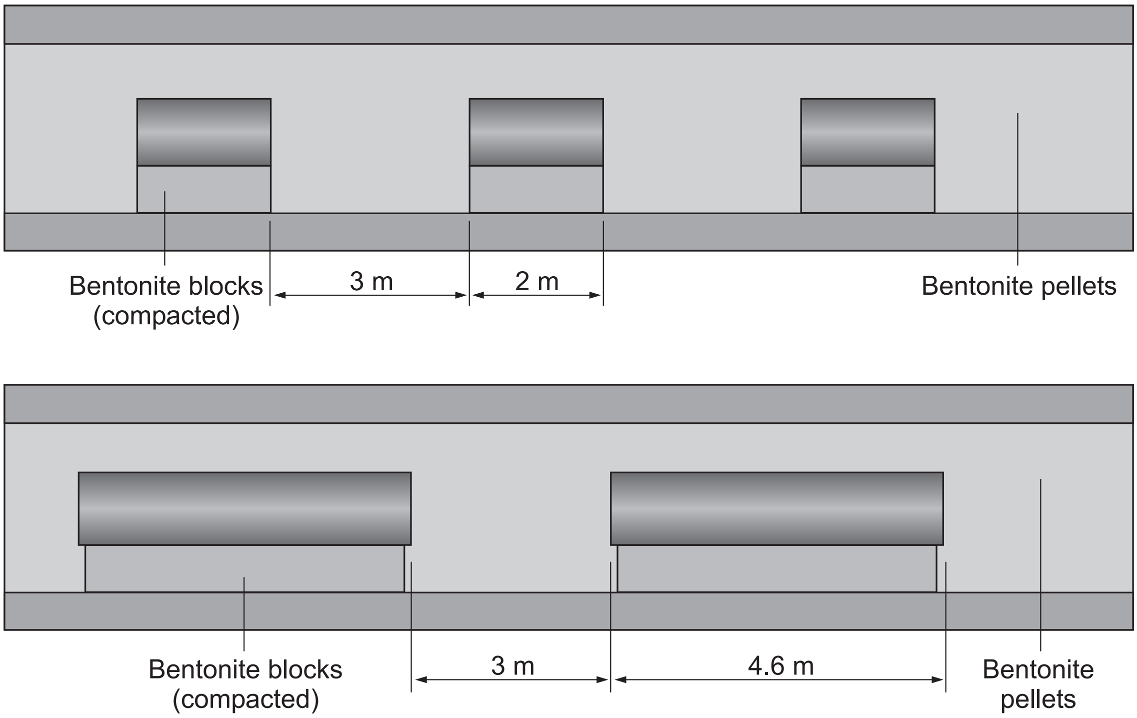

Fig. 4.5-7: Longitudinal section through disposal tunnels for canisters of HLW (top) and SF (bottom) at completion of canister and backfill emplacement

The tunnel diameter is 2.5 m.

Emplacement of ILW

The ILW emplacement tunnels are approximately 9 m × 7 m in cross-section (open space after lining) and are supported by a concrete liner and rock bolts. They are located at the end of the operations tunnel, ~500 m away from the SF / HLW tunnels (see Fig. 4.5-1). Cross-sections

[71] If required, the thermal conductivity of the bentonite can be significantly increased by the addition of 5 % graphite, which would likely decrease the maximum temperature at the canister/bentonite interface by ~ 20 – 30 °C.

through the ILW emplacement tunnels are shown in Fig. 4.5-8. A smaller diameter ILW emplacement tunnel (ILW-2) for disposal of ILW that have significant concentrations of low molecular weight organics and other complexing agents is located nearby, but well separated from the ILW-1 tunnels (Fig. 4.5-1). After emplacement of the ILW, the void regions within the emplacement tunnels are filled with a cementitious mortar. Most of the ILW would be emplaced during the first phase of repository operation, at the same time as wastes are emplaced in the pilot facility.

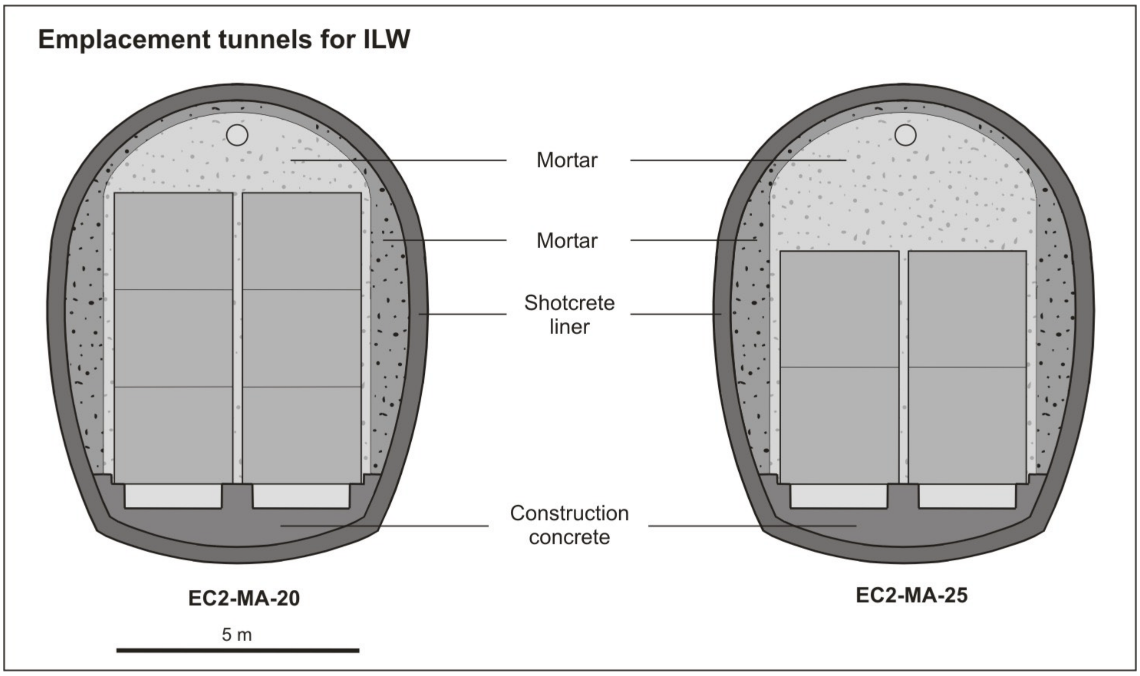

Fig. 4.5-8: Cross-sections of ILW emplacement tunnels at the end of waste emplacement, with different types of waste emplacement containers

Monitoring phase

The status of the disposal facility after the completion of emplacement of all wastes is shown in Fig. 4.5-9. At this time, all wastes are in backfilled and sealed emplacement tunnels and the main operations tunnels are sealed. The wastes in the pilot facility are in backfilled emplacement rooms that are accessible to monitoring using the adjacent observation tunnel and monitoring boreholes. The shaft is sealed with ~ 40 m of highly compacted bentonite, above which is backfill and an additional bentonite seal is placed where the shaft exits the Wedelsandstein Formation, to prevent nuclides that may have migrated through the Opalinus Clay from taking a shortcut path through the overlying confining units to the overlying aquifer. The SF / HLW and ILW operations tunnels and the construction tunnel are backfilled with a 30 % bentonite/70 % sand mixture and sealed with ~ 40 m of highly compacted bentonite (Sitz 2002).

Closure and final sealing

Final closure of the facility would involve emplacement of two ~ 40 m long seals of highly compacted bentonite and backfilling the ramp. The main seal at the repository horizon is to be placed at the construction branchoff of the access tunnel (Fig. 4.5-10), the second where the ramp intersects the overlying Wedelsandstein Formation. These long-term seals are designed with the objective of ensuring that the main tunnels and access ramp have hydraulic properties similar to those of the undisturbed host rock. This will be achieved by the following steps:

1. The concrete tunnel liner and steel rails are removed within the seal section.
2. A ring of Opalinus Clay is excavated to a depth of ~ 1 m to remove the excavation-damage zone, which may have been altered by contact with the concrete and with air.
3. A bulkhead is installed across the tunnel.
4. An approximately 40 m section of highly-compacted blocks of bentonite is installed, keyed into the Opalinus Clay. The blocks would have a density sufficient to give a swelling pressure of 9 MPa, to counteract tunnel convergence and the formation of a new EDZ.
5. The second bulkhead is emplaced; the ramp is then backfilled with a mixture of bentonite and sand.

Details of the design of seals are given by Sitz (2002).

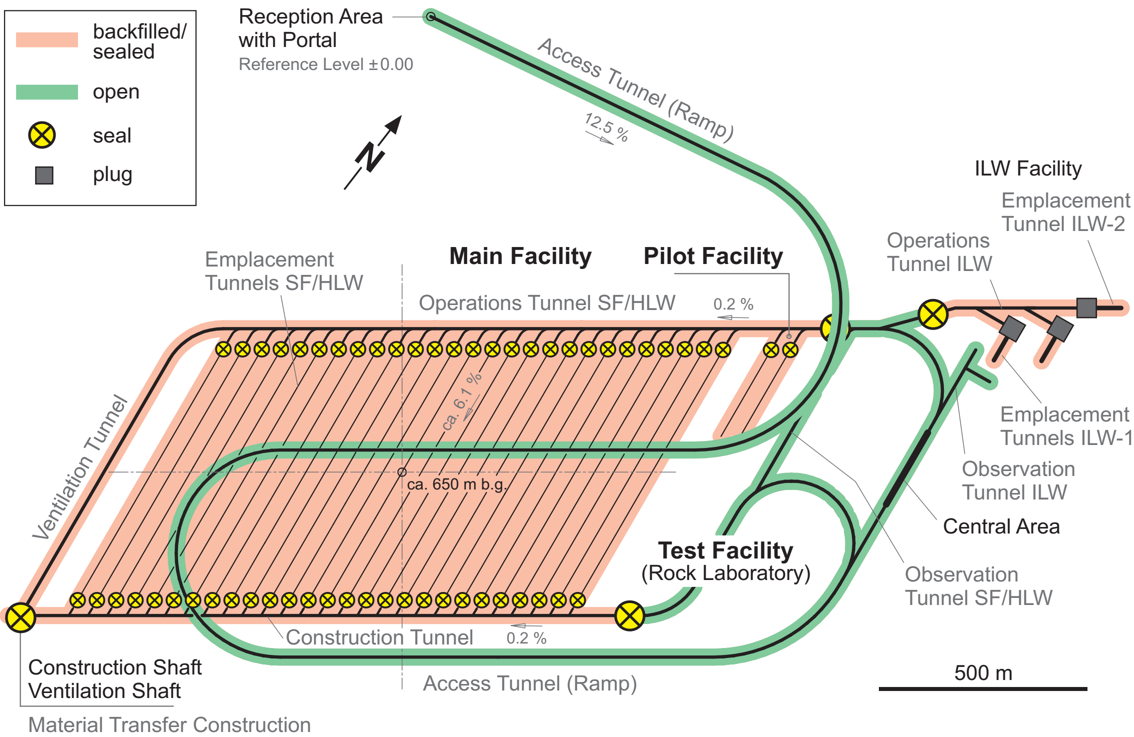

Fig. 4.5-9: Status of the repository during the monitoring phase, when waste emplacement is complete, but before final sealing and closure of the facility

Note that seals are considered to comprise highly compacted bentonite, along with a concrete bulkhead. Plugs at the entrances to ILW emplacement tunnels are composed of concrete.

4.5.3.5 Thermal constraints on repository design

SF and HLW canisters generate sufficient heat that thermal effects must be considered in repository design as they can affect a variety of processes. Maximum temperatures in the repository are determined by the ambient temperature (38 °C at 650 m in Opalinus Clay at the potential site), the thermal properties of Opalinus Clay and bentonite backfill, as well as a number of parameters that can be controlled, such as canister loading, the type and age of the waste at emplacement, the separation distance between canisters and the separation distance between waste emplacement tunnels. Several repository safety studies have argued that an important constraint related to thermal loading of the repository is the need to keep bentonite temperatures low enough that detrimental changes to its desirable swelling properties and plasticity do not occur (SKB 1999, Nagra 1994a, Johnson et al. 1994). At a temperature of 150 °C, for the reference granular bentonite with an initial dry density of 1.5 Mg m^{-3}, a ~ 50 % drop in swelling pressure occurs due to cementation, whereas at 125 °C, only minor reduction in swelling is observed (Pusch et al. 2002). As discussed in Section 5.3.2 and in Johnson et al. (2002), because of the low thermal conductivity of bentonite prior to resaturation and the relatively high ambient temperature, it is difficult to keep temperatures below 125 °C throughout the entire bentonite barrier. As a result, the thermal constraint specified for the design of the SF / HLW emplacement tunnels is that the outer half of the bentonite should remain below ~ 125 °C, so as to retain maximum swelling capacity in at least the outer half of the bentonite to ensure a good quality hydraulic seal around each canister. For canisters of HLW, this is achievable for both the reference waste glasses (BNFL and COGEMA) provided they are cooled for 30 to 40 years prior to disposal. For SF canisters, studies have shown that a maximum heat output of 1500 W per canister at the time of waste emplacement will satisfy the temperature constraint (Johnson et al. 2002). This will require that UO_2 SF with a burnup of 48 GWd/t_{IHM} has decayed for at least 40 years prior to canister emplacement in the repository. For the case of PWR MOX fuel, in order to meet the thermal constraint it is necessary to combine it with UO_2 assemblies (e.g., in the case of PWR fuel, placing 1 MOX assembly and 3 UO_2 assemblies in a canister). Calculations indicate that for ILW, heat generation is significantly smaller, and thus thermal constraints need not be considered. The temperature evolution of the repository is discussed in detail in Section 5.3.2.

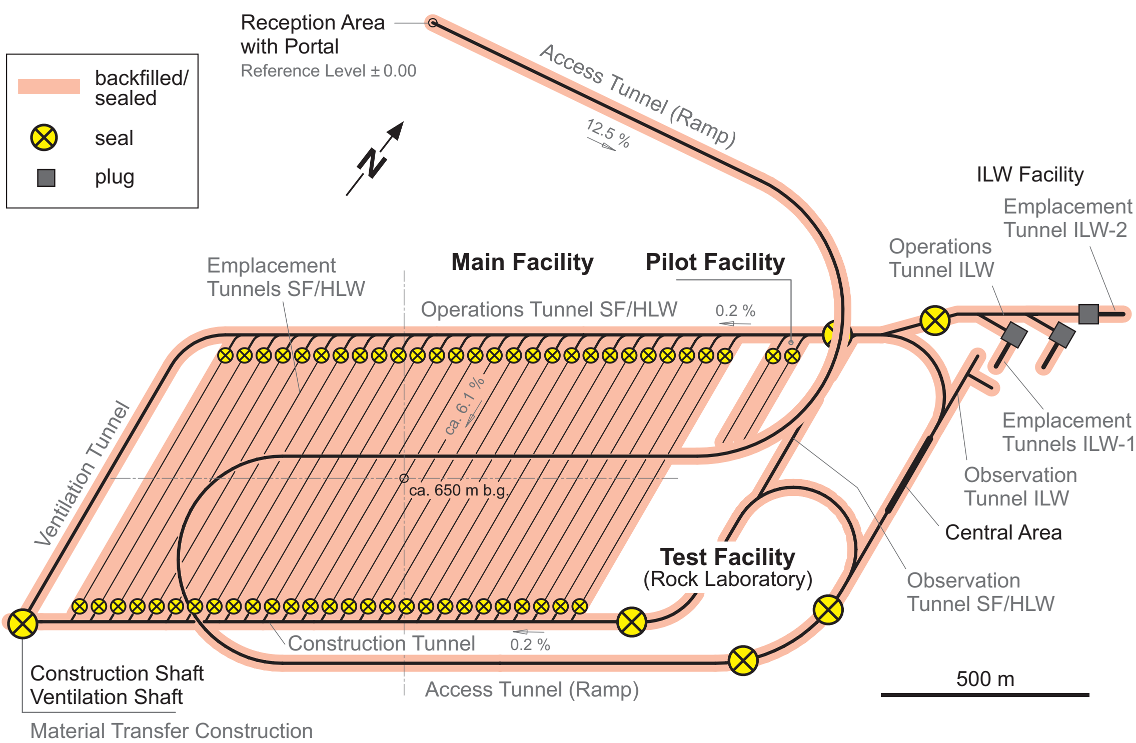

Fig. 4.5-10: Status of the repository after final sealing and closure of the facility

4.5.3.6 Summary of design variants and associated uncertainties

Uncertainties exist in various areas associated with quantities of waste, choice of canister materials and aspects of repository design. The principal uncertainties and their implications for assessment of repository performance are summarised in Tab. 4.5-5.

Tab. 4.5-5: Summary of design variants associated with wastes, engineered barrier system and repository design and their implications for safety assessment

System component	Reference assumption	Variant assumption	Implications of variant assumption
SF (quantity for direct disposal)	3217 t_{IHM} in 2065 canisters (60 years power plant operation)	5576 t_{IHM} in 3580 canisters (300 GWa (e) nuclear power system)	increased radionuclide inventory and size of repository
SF (average burnup)	average burnup of 48 GWd/t_{IHM}	higher burnups (up to 75 GWd/t_{IHM})	higher heat output (longer decay times required, or reduced mass loading per canister); more heterogeneous radionuclide distribution
SF canisters	steel canisters	copper canisters (with steel insert); possibility of bronze insert to lower gas production rate	long lifetime (> 10^5 a), no H_2 gas production until canister failure occurs (see Section 5.5.2.3 for discussion of gas production and migration in SF/HLW tunnels)
ILW inventory	cemented waste option	high force compacted waste option	not significant (inventories of radionuclides are basically the same), except for possibility for localised radiolysis
Reserves and miscel-laneous waste *	repository design considers volume of additional waste, but no assessment calculations are performed		not significant (radionuclide inventory is small relative to that of other ILW)
Bentonite backfill	granules with low thermal conductivity	improved thermal conductivity by addition of graphite	significantly lower temperatures (20 – 30°C) during the resaturation phase
ILW emplace-ment tunnel seals	concrete plug at emplacement tunnel entrance	high gas permeability plug to enhance gas leakage into access tunnel	low gas pressure ensured in ILW emplacement tunnels (see Section 5.5.2.2 for discussion of gas production and migration in ILW tunnels)
SF/HLW emplace-ment tunnel support	rock bolts and mesh	(unlikely) requirement for tunnel support with polymer or thin concrete liner because of poor rock conditions	no assessment done, but a thin layer is considered unlikely to significantly degrade the bentonite or Opalinus Clay

* some wastes from research, fuel encapsulation, etc.

5 System Evolution

5.1 Objectives and scope of this chapter

The objective of this chapter is to outline the evolution of the disposal system beginning at the time of emplacement of the wastes in the repository (i.e. the system concept, as shown in Fig. 5.1-1). The intention is to illustrate the importance of various safety-relevant phenomena in relation to disposal system behaviour.

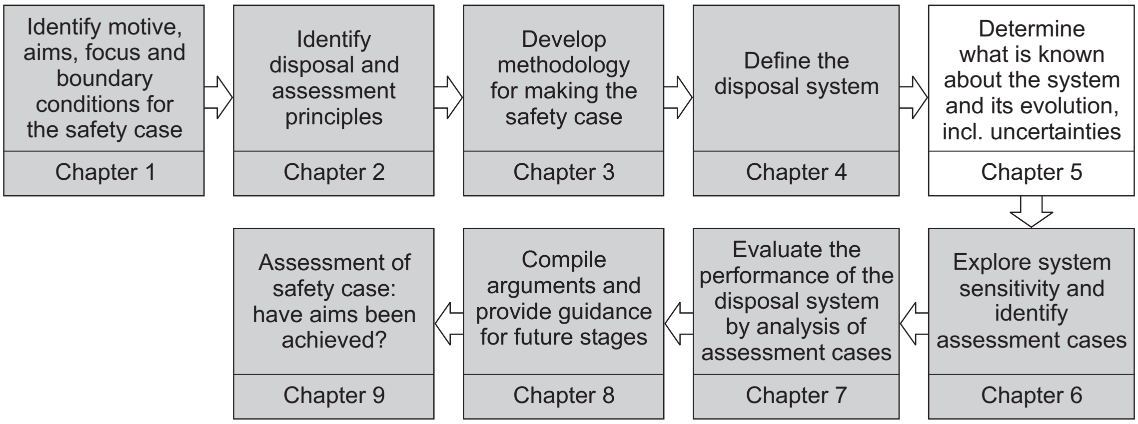

Fig. 5.1-1: The role of the present chapter in the sequence of tasks involved in developing the safety case

As discussed in Section 2.5.4, the time frame for safety assessment calculations is of the order of about one million years. The present chapter focuses on describing how all the components of the disposal system evolve over this time frame. For each major system component, the key phenomena (temperature-related, radiation-related, hydraulic, mechanical, chemical and biological) and their expected evolution over time are described and the expected effects on the various components are discussed. This results in an expected or normal evolutionary path for the repository. Also discussed are uncertainties in the evolution that are termed "possible deviations from the expected evolution" which are based on the degree of uncertainty ascribed to the various phenomena, their time of occurrence and their interaction with other phenomena. They form the basis for a discussion of the key phenomena (pillars of safety) that contribute to the safety functions, leading to the development (Chapter 6) and evaluation (Chapter 7) of assessment cases. The climatic, surface environmental and geological boundary conditions of the system evolution are presented first. This also includes long-term processes such as uplift/erosion, climatic changes, neotectonic events, etc. The evolution of the near field and far field, driven by the presence of the waste and influenced by the boundary conditions, are then discussed, including all aspects relevant to radionuclide transport. The near field is defined as the EBS as well as the region of rock immediately around waste emplacement tunnels (extending a few metres from the tunnels) that is significantly affected by thermal, hydraulic, chemical and mechanical changes induced by the presence of the waste and excavations. The far field is considered the region of host rock and geosphere beyond this region, in which such effects are substantially smaller or negligible.

The description of the expected evolution of the disposal system has been based on analysis and expert judgement of the phenomena most relevant to safety. These phenomena were identified through the FEP (Features, Events and Processes) procedure outlined in Sections 3.7.4 and 3.7.5 and through discussions with various experts. The FEP procedure is discussed in more detail in

Nagra (2002d). Following a description of the evolution of the disposal system in Sections 5.2 through 5.7, in Section 5.8 the reference disposal system is examined from the perspective of the objectives and principles for the selection of such a system that are outlined in Section 2.6. This illustrates the manner in which the system meets the proposed objectives.

5.2 Climatic, surface environmental and geological setting

The natural environment sets the boundary conditions for the SF / HLW / ILW repository constructed deep underground in the Opalinus Clay of the Zürcher Weinland. This section discusses those aspects of the evolution of the natural environment that may exert an influence on the long-term performance of the repository system.

The fundamental controls on the natural (non repository-induced) evolution of the repository site are long-term changes in climatic and geological conditions. These changes are linked – e.g. overburden and hydrogeology are dependent on uplift/erosion and the periodic formation of ice-sheets in northern Switzerland to be expected in the future. The climate controls the long-term surface environmental conditions, particularly the erosion regime and the dilution of radionuclides, and may affect the hydrogeological regime (recharge).

Geological evolution affects the hydrogeological properties of the host rock through its effect on the stress regime, neotectonics, compaction, uplift/erosion, and seismic activity.

Understanding future changes in climatic and geological conditions is, therefore, important in gaining understanding of the evolution of the site. The information collected in the following sections is based on the geological synthesis report for the Opalinus Clay in northern Switzerland (Nagra 2002a).

5.2.1 Evolution of climate, surface environment and assumptions about future human behaviour

5.2.1.1 Introduction

Future climatic developments are relevant to the evolution of the surface environment. They may also affect to some extent the evolution of the geological setting by driving erosion and groundwater movement. The climate also affects radionuclide transfer in the surface environment and uptake in the food chains.

5.2.1.2 Evolution of climate

The complexity of the earth's climatic system is such that the effects of changes in any part of the system cannot be predicted accurately. However, historic monitoring data, complemented by long-term information recorded in rocks, groundwater, ice-caps and sea-sediments, can all be used to derive climate models which provide both an understanding of past climate and broad-scale predictions of future climate development (IPCC 2001).

The present-day climate of northern Switzerland is described as temperate and is representative of interglacial conditions. Since the late Pliocene and throughout the Quaternary (the last 2.6 million years), the Earth has experienced numerous cycles of cooling and warming that have led to considerable variations in northern hemisphere ice-cover ("icehouse" climate). There is strong evidence that these cycles are related to temporal and spatial changes of the radiation of the sun directly related to orbital parameters (Milankovitch cycles). Until 0.9 million years before present, climate changes appeared to be controlled by the 41 000-years-cycle of the orientation of Earth's axis relative to its orbit. During a relatively short period (0.9 – 0.6 million years before present), the evolution of climate seemed to be in a transient regime. Since 0.6 million years, periodic climate changes have been triggered most likely by the 100 000-years-cycling of the eccentricity of Earth's orbit.

At times of greatest cooling, the northern hemisphere has been extensively glaciated and global sea levels have fallen considerably below the present level (e.g. 100 – 120 m during the maximum of the Würm glaciation, according to Burga & Perret (1998)). Arctic sea ice was connected to continental ice caps that covered Scandinavia and much of northern Europe. The Alps have acted as the core of major ice-sheets that extended far into the foreland. The build up to full glacial conditions involved partial freezing of the uppermost part of the sediment sequence (permafrost). During interglacial times, such as the present period, ice sheets retreat until only the upper levels of mountain glaciers remain, permafrost melts and surface drainage patterns become re-established. After each glaciation, there are some changes in the locations of lakes and courses of rivers as a result of glacial and periglacial erosion and sedimentation, but the overall arrangement of the main drainage valleys is believed to have remained more or less constant since the beginning of the Pleistocene. Throughout these cycles, precipitation rates vary in response to major changes in the European and North Atlantic weather system.

Expected evolution of climate

The last glacial period ended about 10 000 years ago. The Earth is presently in the midst of an interglacial period (the Holocene), with sea-levels near their maximum. Permanent snow and ice cover and permafrost in Switzerland are limited to mountainous regions above about 2500 m. The present climate is temperate, with significant influence from the Atlantic Ocean.

Tab. 5.2-1: Expected climatic evolution in northern Switzerland for the next one million years

Climate regime	Climate states	Likelihood of occurrence of climate regime
Glacial/interglacial cycling ("Icehouse") Orbitally forced climate changes (Milankovitch cycles with a period of 100 000 years)	Interglacial (present-day, possibly with higher/lower temperatures, precipitation and evapotranspiration)	Very likely (expected evolution)
	Periglacial climate (e.g. tundra climate)	
	Glacial (ice cover)	

There is strong evidence that within the next one million years, the climate in northern Switzerland will continue to oscillate between glacial and interglacial periods ("icehouse", see Tab. 5.2-1), at a rate of one glaciation every 100 000 years, as in the past 600 000 years. This expectation is based on the assumption that past and future human activities may perturb, but not completely alter, the long-term major climatic cycles. Recent modelling indicates that the present interglacial period may continue for about 50 000 years (Loutre & Berger 2000). This may involve periods with higher/lower temperatures, precipitation and evapotranspiration compared to present-day conditions. 10 major glaciations are expected to occur in the next one million years, with a possible ice thickness in the Zürcher Weinland of up to 400 m and an average duration of ice cover of 20 000 years. During the transition between interglacial and glacial periods, a periglacial climate (e.g. tundra climate) would develop in northern Switzerland (Burga & Perret 1998).

It is obvious that glacial cycling will affect the geological setting, the surface environment and, especially, human activities. The effects of repeated glacial loads on the hydromechanical conditions in the Opalinus Clay are discussed in Section 5.2.2.2. The different effects of glacial cycling on the surface environment and human activities are addressed in Section 5.2.1.3 and 5.2.1.4.

Possible deviations from expected evolution of climate

Long-term global changes from the "icehouse" regime to alternative climatic regimes cannot be ruled out. Conceivable alternative climates are:

A Cessation of glaciations and transition to permanent humid temperate climate

B Change to permanent humid warm climate induced by anthropogenic activity ("greenhouse")

C Cessation of glaciations and transition to permanent dry climate.

These alternative climatic regimes are unlikely to occur in the next one million years. Moreover, the possible effects of these alternative climates on the surface environment are considered to be covered by the range of climate states involved in the "icehouse" climatic regime: Type A is similar to the present-day climate and type B is similar to interglacial periods involving warmer and more humid conditions. Dry climatic conditions may occur during the transition from glacial to interglacial periods, as can be shown for the late Würm glaciation.

For this reason, the effects of alternative climatic regimes on the surface environment can be evaluated by analysis of the "icehouse" climatic regime, and the alternative climates A – C need not be investigated any further.

5.2.1.3 Evolution of the surface environment

The long-term evolution of the surface environment is far less predictable on a local scale than the evolution of the geological setting. This is because the rates of change at the surface are much larger than those deep underground and are affected by a wider range of phenomena. Rates of change of the surface environmental characteristics depend on climate (rainfall/ evapotranspiration), geological processes (tectonic movement, erosion), catastrophic events (e.g. volcanic eruptions, meteorite impact), and human activities.

In northern Switzerland, deep groundwaters from regional aquifers generally discharge at the lowest levels of terrain (valley bottoms), where deep aquifers are in direct contact with gravel aquifers or surface waters (rivers, lakes). Discharge of deep groundwaters in valleys filled with impermeable lake sediments or in higher lying areas is very unlikely. In most river valleys of present-day northern Switzerland, valley sections with gravel aquifers alternate with sections where the river either flows directly on the bedrock or is embedded in impermeable lake sediments or glacial till (Fig. 5.2-1). As a consequence, river sections where significant groundwater flow occurs alternate with sections where groundwater flow is insignificant. The distance between water table and terrain surface varies significantly along the river valley. It is greatest in zones with gravel terraces and may decrease to zero in transition zones where the river bed passes from gravel aquifers to impermeable ground. In these transition zones, wetlands may exist. Furthermore, valley sections where net erosion of solid materials takes place alternate with sections with net deposition.

Although deep groundwaters generally discharge into valley bottoms, it cannot be ruled out that, in special circumstances, discharge occurs into springs located at valley sides. This situation is depicted in the leftmost transverse profile in Fig. 5.2-1, where discharge from the regional aquifer does not occur in the impermeable valley infill but higher up at the side of the valley.

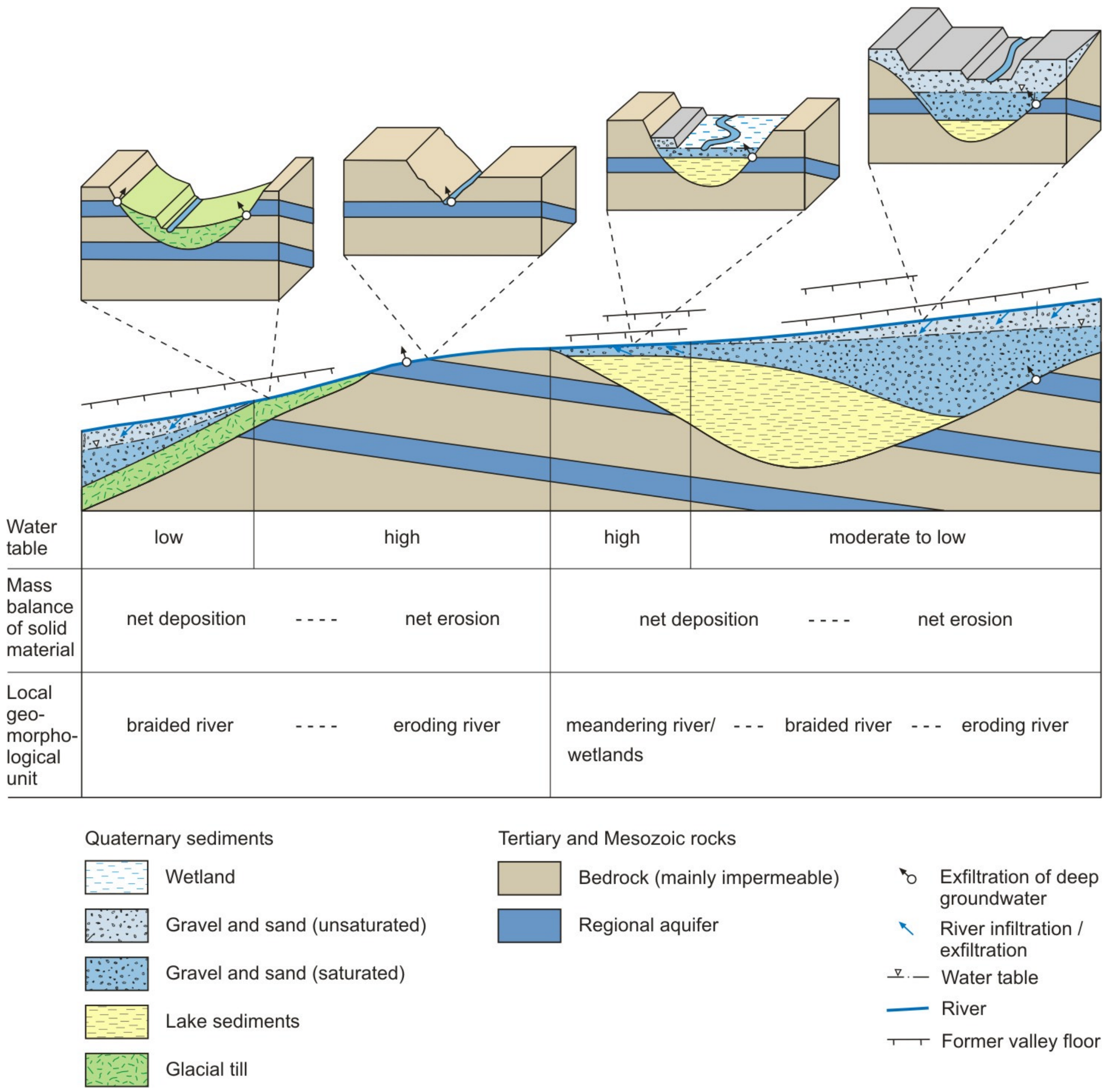

Fig. 5.2-1: Schematic cross-sections (rotated 90°) through the valley bottoms along a typical river valley in northern Switzerland that correspond to local geomorphological units (see Tab. 5.2-2)

It is thus important to classify the various valley types in northern Switzerland where discharge of deep groundwaters occurs today and may continue to occur in the future. These discharge areas, termed *local geomorphological units* for the purpose of biosphere description in the present report, are listed and described in Tab. 5.2-2.

The mass balance of eroding river sections, such as the Rhine river in the region of interest, is characterised by a small turn-over rate and a net loss of solid materials to downstream locations. In contrast, braided rivers, meandering rivers, river deltas, lakes and wetlands are local geomorphological units with medium to large turn-over rates, where net deposition of solid materials takes place[72].

Climate evolution controls geomorphological and hydrological conditions in the surface environment. Relevant processes affected by climate are: precipitation, evapotranspiration, glacial and fluvial erosion, river discharge, groundwater flow, capillary rise, and contaminant dilution. The relationship between climatic and geomorphological/hydrological conditions is illustrated in Tab. 5.2-3.

Tab. 5.2-2: Local geomorphological units representing possible discharge areas of deep groundwater in northern Switzerland

Local geomorphological unit	**Description**	**Mass balance of solid material**	**Example**
Eroding river	Relatively narrow, cut-in river section where solid material balance is dominated by erosion (e.g. V-shaped valleys, gravel terraces). Eroding rivers cause linear erosion and act as regional base level for denudation.	Small turn-over rate, net erosion	Majority of Rhine river sections between Bodensee and Basel
Braided river	Relatively flat network of river arms that continuously changes with time due to flooding and deposition.	Large turn-over rate, net sedimentation	Historical Thur between Weinfelden and Thalheim
Meandering river	Relatively flat river section with a sequence of bends; local erosion of river banks and episodic short-cuts between bends.	Medium turn-over rate, net sedimentation	Historical Rhine below Basel
River delta in lake	Sedimentation area near inflow of river to lake.	Medium/large turn-over rate, net sedimentation	Rhine delta in Bodensee
Lake (open water)	Large surface water body with relatively low turn-over rate. Lakes act as solid material sinks and significantly reduce the suspended solid load between inlet and outlet rivers. Under oligotrophic (low-nutrient) conditions, lakes are eventually filled with sediments carried by inlet river. Under eutrophic (nutrient-rich) conditions, intensive growth of plants can take place and can turn a lake into land.	Small turn-over rate, net sedimentation	Bodensee
Wetland (including swamps)	Areas with water table at or near the surface, subject to frequent flooding. Deposition of inorganic and organic material.	Small turn-over rate, net sedimentation	Rhine valley east of Rüdlingen

72 Net sedimentation of valley-infill in northern Switzerland may be limited to a certain time window in the icehouse climate regime, e.g. during the transition zone from glacial to interglacial conditions.

Tab. 5.2-3: Relationship between climate and local geomorphological/hydrological conditions in the surface environment

Climate regime	**Climate state**	**Local geomorpho-logical units**	**Likelihood of occurrence of local geomorpho-logical units**	**Typical distance from water table to terrain surface**	**Dilution in aqueous phase** **P-ETP** [1]	**Dilution by mixing with solid material**
Glacial/inter-glacial cycling ("Icehouse") Orbitally forced climate changes (Milankovitch cycles with a period of 100 000 years)	Interglacial (present-day, possibly with higher/lower temperatures, precipitation and evapotrans-piration)	Eroding river	likely	> 10 m	> 500 mm a^{-1}	significant [2]
		Braided river	less likely	< 2 m	> 500 mm a^{-1}	significant
		Meandering river	likely	< 2 m	> 500 mm a^{-1}	significant
		River delta in lake	likely	< 2 m	> 500 mm a^{-1}	significant
		Lake	likely	NA [3]	> 500 mm a^{-1}	NA
		Wetland	likely	< 2 m	> 500 mm a^{-1}	insignificant
	Periglacial climate (e.g. tundra climate)	Eroding river	unlikely	> 10 m	< 500 mm a^{-1}	significant [2]
		Braided river	likely	2 to 10 m	< 500 mm a^{-1}	significant
		Meandering river	unlikely	2 to 10 m	< 500 mm a^{-1}	significant
		River delta in lake	likely	2 to 10 m	< 500 mm a^{-1}	significant
		Lake	likely	NA	< 500 mm a^{-1}	NA
		Wetland	less likely	< 2 m	< 500 mm a^{-1}	insignificant
	Glacial (ice cover)	Under glacial conditions, population density and agricultural productivity are either zero (areas covered by ice) or low (periglacial areas). During deglaciation large redistribution of sediments occurs, resulting in significant dilution.				

1 P is the precipitation rate and ETP is the evapotranspiration rate

2 If a lake is present upstream from the investigated biosphere area, then the majority of the suspended solid load is sedimented in the lake, and dilution by mixing with solid material is low.

3 NA = Not applicable

All geomorphological units listed in Tab. 5.2-2 may exist during future interglacial periods within the region of interest. Significant dilution of radionuclides in the aqueous phase and by mixing with solid material is expected to take place in the surface environment, even for drier/warmer than present-day conditions during interglacial periods. Eroding rivers deserve special attention because of the ongoing uplift/erosion in northern Switzerland (Section 5.2.2.3). It can be anticipated that the major drainage features, including those of present-day Rhine, Aare, Reuss, Limmat and Thur, will be partly covered by young terraced gravel bodies during interglacial periods. In such cases, a relatively large distance between the groundwater table and the terrain surface is expected, inhibiting radionuclide transport from the gravel aquifer to the top soil by capillary rise. Dilution by mixing with solid material is significantly reduced if a lake is located at some distance upstream, because lakes act as efficient sinks for suspended solid material carried by rivers. Direct discharge of deep groundwaters in lakes may be hindered by the presence of impermeable lake sediments.

In all climate states, the local geomorphological units braided river, meandering river and river delta in lake show similar characteristics with respect to hydrological conditions (e.g. dilution, distance between water table and terrain surface etc.) and can thus be represented by one single model area, termed *sedimentation area*.

The "icehouse" climate regime involves changes to the biosphere in the region of interest from present-day temperate conditions, through progressively cooler conditions (tundra climate) and possibly covering the land finally with ice. Braided rivers, lakes and deltas in lakes will be the most likely local geomorphological units for periglacial conditions. Dilution of dissolved radionuclides is expected to be lower compared to interglacial periods due to lower precipitation rates, but dilution by mixing with solid material from upstream locations may be significant.

During glacial periods, ice cover and partial permafrost affect both local and regional groundwater flow, as hydraulic gradients change in magnitude and direction in response to ice cover, recharge is effectively cut off in some areas and increased in others, and discharge rates and locations vary with time. Erosion beneath glaciers may strip out Quaternary deposits and cut into the bedrock, deepening existing river valleys.

Variations in the dilution potential of Quaternary aquifers and rivers are important for the safety assessment. Periodic increases in flows (above present-day conditions) would be a positive feature with respect to radionuclide concentrations. When the ground is frozen or inaccessible under ice, water flows are expected to be at their lowest. If summer temperatures remain above freezing, then a zone of seasonally unfrozen ground may exist in areas where there is a localised permafrost layer, which may support migratory animals and herdsmen. Thus, the biosphere may be partially isolated from the deep groundwater system. Significant groundwater discharge zones, the courses of major rivers and lakes are, however, liable to remain free from permafrost due to the heat from discharging groundwater and insulating effect of large water bodies. Thus, while permafrost may tend to reduce recharge, it may focus groundwater discharge to the remaining unfrozen zones.

Expected evolution of the surface environment

As discussed in the previous section, it is expected that the "icehouse" climate regime will continue over the next one million years. Interglacial periods, with moderate temperatures as today, will be interrupted by cold periods (e.g. average temperatures during maximum of Würm glaciation 10 – 15 °C lower than today, according to Burga & Perret 1998) during which permafrost may develop locally[73] and parts of northern Switzerland may be covered by ice.

Under "icehouse" conditions, the surface environment is subject to repeated far-reaching alterations that affect the geomorphological and hydrological conditions. Four main phases may be distinguished:

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[73] There is no evidence for extended deep (> 100m) permafrost in northern Switzerland (Nagra 2002a).

- **Interglacial period** – Duration 20 000 – 50 000 years. Little changes to geomorphology, surface hydrology and base level of erosion. Predominant redistribution and displacement of solid material and suspended solids with a general tendency to linear erosion due to the on-going uplift. Few lakes, significant groundwater flow, good retention of meteoric water, extended aquifers with relatively large separation of water table and surface, locally some areas with shallow water table.
- **Transition from interglacial to glacial period** – Duration up to 50 000 years. Significant changes to geomorphology with important redistribution and accumulation of solid material in main valleys. Lakes nearly absent, significant flow of water both at the surface and underground, few areas with shallow water table, poor retention of meteoric water.
- **Glacial period** – Duration 10 000 – 50 000 years. Foreland partly covered with ice. Changes to geomorphology and hydrology governed by sub-glacial processes. No lakes, poor water retention. Conditions in periglacial areas similar to conditions prevailing during transition from interglacial to glacial period. Permafrost mainly in the vicinity of border of glaciers, but not completely extending over large areas. Infiltration and discharge patterns not significantly changed with respect to interglacial periods. Relatively dry climate, partly cold desert with important wind erosion.
- **Transition from glacial to interglacial period** – Duration of a few thousand years. Melting of glaciers in foreland. Important changes to geomorphology, with rapidly changing hydrological and topographical conditions. Numerous lakes and areas with shallow water table, huge redistribution and displacement of solid material, equilibration of river network caused by erosion and accumulation. Large amounts of water, large surface discharge.

5.2.1.4 Assumptions about future human behaviour

Attempting to define the full range of possibilities related to future human behaviour over the long time periods of interest in safety assessment is a matter of speculation, although some bounds may be set based on human resource and dietary needs.

The types of agriculture that are possible and the sources of water will change in each stage of the "icehouse" climate regime. In the Reference Case biosphere, a human life-style based on the present-day diet is assumed. The diet consists of vegetables, grain, fruit, milk and dairy products, meat, eggs, fish and water. Drinking water for humans is extracted from wells in the Quaternary aquifer, or, in alternative cases, from a spring at the valley side where deep groundwater from the Malm exfiltrates or from a deep well drilled into the Malm aquifer[74]. In contrast to today's agricultural practices, local food production is assumed for the future. The members of the critical group[75] are assumed to spend their entire life-time within the biosphere area under consideration, yielding higher than realistically expected doses due to ingestion, inhalation and external radiation. Key biosphere data is discussed in Nagra (2003b).

In periglacial areas during glacial periods, natural and semi-natural ecosystems prevail. The production rates of food would be significantly lower than during interglacial periods. Thus a significantly larger area would be required for food production for the same number of humans. As a consequence, the average activity concentrations in food are reduced.

[74] This would lead to less dilution of radionuclides compared to the situation of extracting drinking water from a shallow Quaternary aquifer.

[75] The doses are calculated for average members of the critical group chosen to be representative of the individuals or population groups that might receive the highest doses as a result of the presence of the repository (Chapter 2).

5.2.2 Evolution of geological setting

5.2.2.1 Introduction

As discussed in Section 4.2.5, overpressures are observed in the Opalinus Clay. Their evolution is discussed in Section 5.2.2.2 below. In the very long run, uplift and erosion will – if they continue – become more important. The effects of continuing uplift on the geological situation are discussed in Section 5.2.2.3. Besides these slow and continuous geologic processes, the likelihood and effects of unlikely (rare) geological events need to be discussed. This is done in Section 5.2.2.4.

5.2.2.2 Compaction of Opalinus Clay and evolution of hydraulic overpressures

The Opalinus Clay is overpressured (see Section 4.2.5), even though the overlying sediments are currently being eroded. It is assumed that these overpressures are the result of an earlier period of rapid burial or, alternatively, ongoing lateral thrust, during which drainage and compaction of the Opalinus Clay has not yet resulted in a state of pressure equilibrium. Thus, the process of compaction and dissipation of overpressures due to porewater drainage is expected to continue (if the hypothesis of a threshold gradient holds then overpressures will be maintained). This will also be affected by reduction of the total stress imposed (erosion of overburden) or increases in stress (loading of ice during glaciation).

Expected evolution

From the safety point of view, porewater drainage is important because of possible advective radionuclide transport through the Opalinus Clay (and/or through the access tunnel system of the repository – for the effects due to the presence of the repository see Section 5.5.3). Calculations show that very low hydraulic conductivities (10^{-15} m s^{-1} or less) and/or the existence of a threshold gradient are required to reproduce the observed current overpressures of the Opalinus Clay (see Section 4.2.5 and Nagra 2002a). If no threshold gradient exists, the time for complete overpressure dissipation is expected to be in the order of one million years and is characterised by extremely low flow rates – for the case with a threshold gradient the overpressures will not completely dissipate.

As stated in Section 5.2.1, it is expected that the glacial/interglacial climate will continue to be dominant for the next one million years. As a result, several glaciations are expected to take place during this time period. Within the Opalinus Clay, the additional (to lithostatic) load temporarily imposed by the presence of 200 – 400 m of ice (equivalent to a load increase of about 20 – 30 %) will initially be carried predominantly by the clay porewaters. It will be transferred only gradually to the grain-to-grain contacts of the solid matrix (the rock frame) as a result of drainage and compaction. Due to the extremely low hydraulic conductivity of the Opalinus Clay, the relaxation time of glaciation-induced overpressures is expected to be much longer than the actual duration of the glaciation (tens of thousands of years). The palaeohydrogeological evidence for the stability of the Opalinus Clay environment (Section 4.2.5) indicates that the six previous Quaternary cycles of glaciation have had no perceptible impact on the transport processes of solutes in the host formation.

Thus, while it could be expected that the rates of water movement in the two regional aquifers will be affected, as will the composition of shallow recharge waters, the response time of the Opalinus Clay hydrogeological system is so slow that the relatively short-term (tens of thousands of years) cyclic changes in external hydraulic boundary conditions will be substantially smoothed out within the proposed repository rock volume.

Possible deviations from expected evolution

Due to the possibility that the measured overpressures are only artefacts and have dissipated much faster than indicated above, it cannot be ruled out that for future loading the transient involving compaction of the Opalinus Clay takes place much faster, leading to higher specific water flow rates.

5.2.2.3 The effect of uplift and erosion on the properties of the host rock and on the hydrogeological situation

As discussed in Section 4.2.1, northern Switzerland and the Zürcher Weinland are considered to be influenced by both the alpine orogeny and the updoming of the Black Forest and thus, for the time being, continuous uplift can be expected to occur as it has for the past few millions of years.

Long-term evolution of the topography (rivers and relief) in the case of uplift is characterised by a dynamic equilibrium between uplift and erosion. According to geological and geomorphological data, the evolution of the landscape of northern Switzerland including the Zürcher Weinland can thus be described as a steady state process for the past several millions of years. The river Rhine (or its equivalent in the long term) acts as the (local) level of erosion and cuts down into the bedrock at a rate similar to that of uplift.

In addition to this linear erosion, the local relief will also develop with time. The evolution of the local relief (e.g. drainage pattern, steepness of hillslopes) depends upon the climate, the bedrock and – since historic times – upon anthropogenic effects (farming, deforestation etc.). Furthermore, the effects of glacial erosion need to be considered as a major factor of landscape and relief development. Future glacial excavation along valleys is expected to be of a magnitude similar to that of excavations in past ice ages and is thus of limited extent in the Zürcher Weinland.

Expected evolution

The geological record of northern Switzerland provides a good data base for estimating the long-term regional uplift. Information from geomorphological studies, basin modelling and also from high-precision leveling gives a consistent picture and indicates that the uplift is in the order of 0.1 mm a^{-1}.

Taking into account the evolution of both the base level of erosion and the local relief, a maximum reduction of repository overburden of 200 m after one million years is possible (Nagra 2002a), with the expected regional uplift in the order of 0.1 mm a^{-1}, resulting in 100 m erosion. Additionally, a one-time down cutting of the base level of erosion of about 100 m has been assumed, which may be caused both by back-erosion of the Rhine Falls and more importantly by the relief adjustment because of the Bodensee (see Fig. 8.3-1 of Nagra 2002a).

With a remaining overburden of 450 m or more for the period of primary interest of about one million years, no changes are expected in the hydraulic properties of the Opalinus Clay.

Erosion will change the outcrop areas of the regional aquifers (Malm, Muschelkalk) and of the smaller aquifers in the confining units (Wedelsandstein Formation, Stubensandstein Formation) and will thus give rise to some changes in flow patterns and location of discharge areas. Although changes in the location of discharge are possible, they are still expected to be in the Rhine valley or its future equivalent.

Possible deviations from expected evolution

Although there is some uncertainty in the future evolution of the uplift rates, significant changes to the geological environment of the repository can be ruled out for at least a few million years. The estimates referred to above are based on a pessimistic interpretation of the available information.

The long-term changes in the regional hydrogeological circulation system cannot be predicted in all details; in particular, it cannot be excluded that discharge takes place in a tributary to the Rhine or its future equivalent.

Possible evolution for a time period exceeding one million years

If uplift continues, erosion will continue to cut down into the bedrock and eventually, after more than a few million years, the repository may be exhumed. During the very gradual uplift and erosion, the repository overburden would be reduced to about 200 m after about 4 million years, if present uplift rates continue. As the overburden is reduced further, the permeability of the Opalinus Clay would slowly increase with time, although it would remain relatively low and redox conditions would remain reducing. Calculations assuming an increased permeability and a reduced overburden indicate that doses would continue to be well below the regulatory guideline. After in excess of 5 million years, erosion can be expected to expose some tunnels, but the large area of the repository and the development of topographic relief due to erosion would lead to only localised exposure of repository tunnels. Chemical conditions would remain reducing in the vicinity of the decayed wastes until erosion reduces the overburden to approximately 10 – 20 m. The radiotoxicity of the spent fuel, the most hazardous of the wastes, will have declined by this time to a value similar to that of the uranium ore from which the fuel was produced (see Section 2.5.4). No rigorous assessment of doses for times beyond several million years has been carried out, but some illustrative calculations of releases have been performed that indicate that sufficient safety is maintained even after millions of years.

5.2.2.4 Potential effects of infrequent geological events

Expected evolution

The potential role in repository evolution of various infrequent geological events (e.g. earthquakes, neotectonic movements) needs to be assessed. Such an assessment includes, on the one hand, an evaluation of whether or not such events can occur at all in the immediate surroundings of the repository and on the other hand, an assessment of the potential impact of such events.

The events considered are (Nagra 2002a, Chapter 8.4):

- re-activation of existing or formation of new discontinuities;
- effects of earthquakes;
- magma intrusion.

The formation of significant new fracture zones is considered to be very unlikely because neotectonic movements will occur along pre-existing structures. Therefore, a respect distance will be kept between the repository and already existing larger-scale fracture zones such as the Neuhausen Fault. In the current design, it is also envisaged to stay away from the Wildensbuch Flexure, the inactive fault zone delimiting the crystalline and Permo-carboniferous basement and some minor faults in the Opalinus Clay identified by 3 D seismics, although these are not expected to be no-go areas. The precautionary decision to avoid these features is due to the limited information available on them. There is no indication of new fault systems currently being developed away from the above-mentioned tectonic features. Based on current knowledge, as in the past few million years, negligible neotectonic activity is expected in the Zürcher Weinland within the next one million years.

Small fracture zones are expected in the vicinity of the repository, these are currently not water-conducting and it is expected that their hydraulic properties will not be affected by any neotectonic events. In particular due to the self-sealing capacity of the Opalinus Clay, such fractures will not have enhanced transmissivity.

Seismic analysis shows that there is only minor seismic activity in the Zürcher Weinland (Nagra 2002a). Furthermore, once the repository is closed, no mechanical damage to the barrier system is to be expected, even in the unlikely case of a large earthquake. In some areas with permeable geological formations and earthquakes of large magnitudes, there are indications of episodic, earthquake-induced fluid migration, especially in the vicinity of vertical/sub-vertical faults (e.g. Muir-Wood & King 1993). Similar effects can be excluded in the case of the Opalinus Clay, because there are no indications either of active water-conducting features or of any significant fluid movements in the past.

The possibility of magmatic intrusion can be excluded by geologic reasoning and by the geothermal maps, the latter indicating that the magma is deep enough to be of no concern.

Possible deviations from expected evolution

Based on current scientific understanding, the existence of preferential pathways due to small fracture zones affected by neotectonic events can be excluded. However, small fracture zones with an enhanced transmissivity (10^{-10} m^2 s^{-1}) are included in the safety analysis as a "what if?" case (see Section 3.7.4), to test the effect of such enhanced transmissivities on system performance.

5.3 Evolution of the SF / HLW near field

5.3.1 Radiation-related processes

5.3.1.1 Spent fuel

Radioactive decay and decay heat

As noted in Section 4.5.2, individual canisters of SF and HLW will have various heat outputs at the time of emplacement in the repository. A thermal constraint of a maximum of 1500 W per canister at the time of waste emplacement has been selected, based on calculations of repository temperature evolution (see Section 4.5.2). This can be met for PWR and BWR UO_2 fuel canisters by allowing 40 years storage prior to emplacement for fuel with an average burnup of 48 GWd/t_{IHM}. In the case of canisters with 3 UO_2 assemblies and 1 MOX assembly of the same burnup, 55 years cooling are required for the canister decay heat to decrease to ~ 1500 W. The time-dependent decay heat of canisters of SF and HLW is illustrated in Fig. 5.3-1.

Fig. 5.3-2 shows the total α and total β/γ activities for a canister containing 1 MOX assembly and 3 UO_2 assemblies and for a HLW canister with BNFL glass. For canisters with 4 PWR or 9 BWR UO_2 assemblies, the β/γ activity is similar, but the α activity and decay heat are approximately 30 % lower than for the 1 MOX / 3 UO_2 canister in the time period from 40 to 10^5 years (McGinnes 2002).

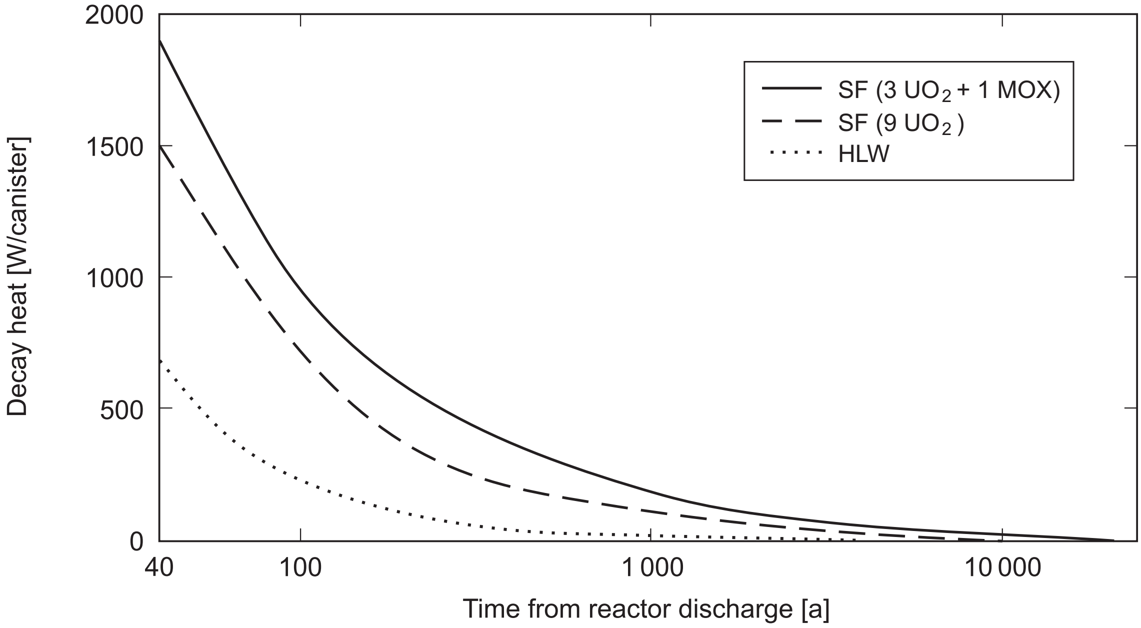

Fig. 5.3-1: Calculated time-dependent heat production of disposal canisters as a function of time from reactor discharge for: a SF canister with 3 UO_2 assemblies and 1 MOX assembly (1.50 t_{IHM}), a SF canister with 9 BWR UO_2 assemblies (1.60 t_{IHM}), all with an average burnup of 48 GWd/t_{IHM} and a canister with an average flask of BNFL HLW glass (based on McGinnes 2002)

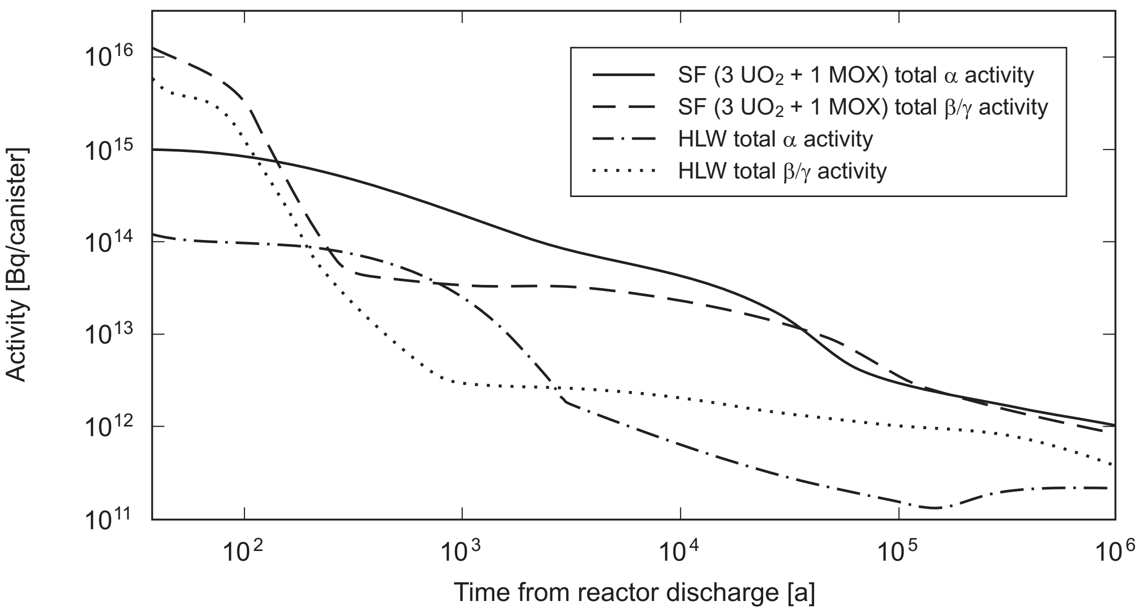

Fig. 5.3-2: Calculated time-dependent total α and total β/γ activity of i) a SF canister with 3 UO_2 assemblies and 1 MOX assembly (1.50 t_{IHM}), all with an average burnup of 48 GWd/t_{IHM}, and ii) an average flask of BNFL HLW glass (based on McGinnes 2002)

Nuclear criticality

Nuclear criticality is a sustained nuclear chain reaction, in which fissioning atoms release neutrons which induce further fission reactions. The reactions produce radiation, heat and new fission products. In order to occur, criticality requires sufficient quantities of fissile isotopes in a suitable geometry and a neutron moderator, such as water, to slow the neutrons. The principal fissile isotopes present in spent fuel are ^{235}U, ^{239}Pu and ^{241}Pu, which are present at levels that are sufficient to obtain criticality under certain conditions. Criticality, were it to occur, could produce elevated temperatures (several hundred degrees) in the EBS, which could potentially damage canisters and backfill material and induce groundwater movement. It is necessary, therefore, to select a layout and adopt other measures that ensure sub-criticality and to perform calculations to demonstrate that, for the spent fuel compositions and geometries in question, including any evolution of the EBS, criticality will not occur.

Calculations for MOX and UO_2 fuel of average burnup indicate that the reactivity[76] decreases for the first few hundred years after discharge of the fuel from the reactor due to ingrowth of ^{241}Am, a strong neutron absorber, from decay of ^{241}Pu. It gradually increases back to its initial value over hundreds of thousands of years as ^{235}U grows back in. This means that the potential for criticality requires evaluation for the entire period for which assessment calculations are performed. An analysis summarised in Kühl et al. (2003) shows that, when loaded with spent UO_2 fuel, canisters would be sub-critical, both when intact, and after failure when the void space in the canister would eventually become filled with water, provided a burnup of at least 15 GWd/t_{IHM} is reached (22 GWd/t_{IHM} for canisters with both MOX and UO_2 fuel)[77]. In the rare cases in which the minimum burnup is not attained by a fuel assembly, the co-placement of high burnup assemblies in the canister or the use of inert filler (e.g. sand) in the void space in the canister would reduce reactivity to a sub-critical level. For the long term, calculations show that changes in geometry, such as diffusion of large quantities of uranium into the bentonite, followed by precipitation within pore spaces, would not result in criticality. Similar conclusions have been reached by Oversby (1996), whose results are based on observations from the Oklo natural reactor.

The identification of a minimum burnup requirement to preclude criticality represents a waste acceptance criterion related to the operation of a spent fuel encapsulation facility, an issue which is discussed in more detail in Section 4.5.2.4.

Radiation damage

Uraninite is regarded to be highly resistant to radiation damage, as evidenced by studies of natural uraninites. Although spent fuel will sustain an even higher radiation dose over hundreds of thousands of years, this will not lead to loss of crystallinity, as discussed by Weber (1981) and Johnson & Shoesmith (1988).

5.3.1.2 HLW

The time-dependent heat production of a canister of BNFL HLW glass after 40 years of cooling is shown in Fig. 5.3-1. The α- and β/γ- activity of BNFL glass is shown in Fig. 5.3-2. As noted

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76 Reactivity is a measure of the number of excess neutrons produced by fission

77 The use of burnup credit in criticality calculations is reviewed by Dyck (2001). It is recognised that considerable developments are required before it is applied in the context of spent fuel disposal. The principles as well as the technologies to implement it are nonetheless well established.

in Section 4.5.2.2, the COGEMA glass has a slightly lower activity and decay heat. More detailed data are available in McGinnes (2002).

Radiation damage

Accumulated radiation damage effects appear to cause only small increases (a factor of 2 to 3) in the dissolution rate of HLW glass (Lutze 1988). These small effects are considered in the selection of the reference glass dissolution rates discussed in Section 5.3.4.6.

5.3.1.3 Possible deviations from expected radiation-related behaviour

Since radiation-related properties can generally be modelled with high confidence[78], calculated radionuclide inventories can be considered to be quite accurate, as can calculated radiation fields and decay heats. In the case of the activation product ^{14}C, the inventory in spent fuel is likely to be overestimated because the ^{14}N content was assumed to be 25 ppm, whereas analysis of spent CANDU UO_2 fuel indicates that pre-irradiation N impurity levels are approximately 10 ppm. As result, the doses due to ^{14}C reported in Chapters 6 and 7 are overestimated by a factor of 2 to 3.

There appears to be no possibility of criticality occurring within a water-filled canister, provided the fuel is irradiated to a moderate burnup, or as a result of concentration of U or Pu in the surrounding bentonite over long periods of time.

5.3.2 Temperature evolution in the SF / HLW near field

5.3.2.1 Expected evolution

Decay heat from SF and HLW canisters will increase temperatures within and around the repository for long periods of time. The maximum temperatures achieved in the various disposal system components, and the time dependency of the temperatures, are determined principally by the heat output of the wastes, the selected repository layout and the thermal properties of the bentonite backfill and the surrounding rock. Maximum temperatures can be limited by limiting canister heat output (reducing canister loading and increasing the storage time of the SF / HLW prior to emplacement in the repository) and/or increasing the spacing between canisters and between emplacement tunnels[79]. In developing the repository design and layout, the principal constraint is the desire to limit the maximum temperature experienced by the outer half of the bentonite barrier to less than approximately 125 °C, to preclude significant thermal alteration that might degrade its desirable swelling and hydraulic properties (see Section 5.3.4.3).

Calculations of temperature distribution within and around the repository have been performed using heat output data for reference SF and HLW canisters as well as site-specific thermal properties of the Opalinus Clay at a depth of 650 m (Johnson et al. 2002). For the calculations, waste emplacement tunnels are assumed to be 40 m apart, with either low thermal conductivity (0.4 W m^{-1} K^{-1}) "dry" bentonite or high thermal conductivity (1.35 W m^{-1} K^{-1}) saturated bentonite surrounding the canisters to represent the limiting cases (note that rapid saturation of bentonite, especially near the canister, is not expected (Section 5.3.3.1), thus only the results for

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78 There have been some significant corrections made to the half lives of some long-lived radionuclides, however (e.g. ^{79}Se), which illustrates the importance of evaluating the uncertainties in basic nuclear data. These uncertainties have been recently assessed by McGinnes (2000).

79 Although changing spacing between tunnels has very little effect on temperatures in the immediate vicinity of the canister.

the low thermal conductivity case are discussed here in detail). Tunnels are assumed to contain either SF canisters or HLW canisters, with a spacing between canisters of 3 m. Time-dependent temperatures in the near field for the case of dry bentonite are shown in Fig. 5.3-3 for SF canisters (3 UO_2 assemblies plus 1 MOX assembly) and Fig. 5.3-4 for HLW canisters. The maximum temperature at the bentonite-canister interface is reached in about 10 years in both cases. Initial temperatures within the bentonite are similar to Fig. 5.3-3 for the case of canisters with 4 UO_2 fuel assemblies, although the duration of the period over which the temperatures in the inner part of the bentonite exceed 125 °C is considerably reduced.

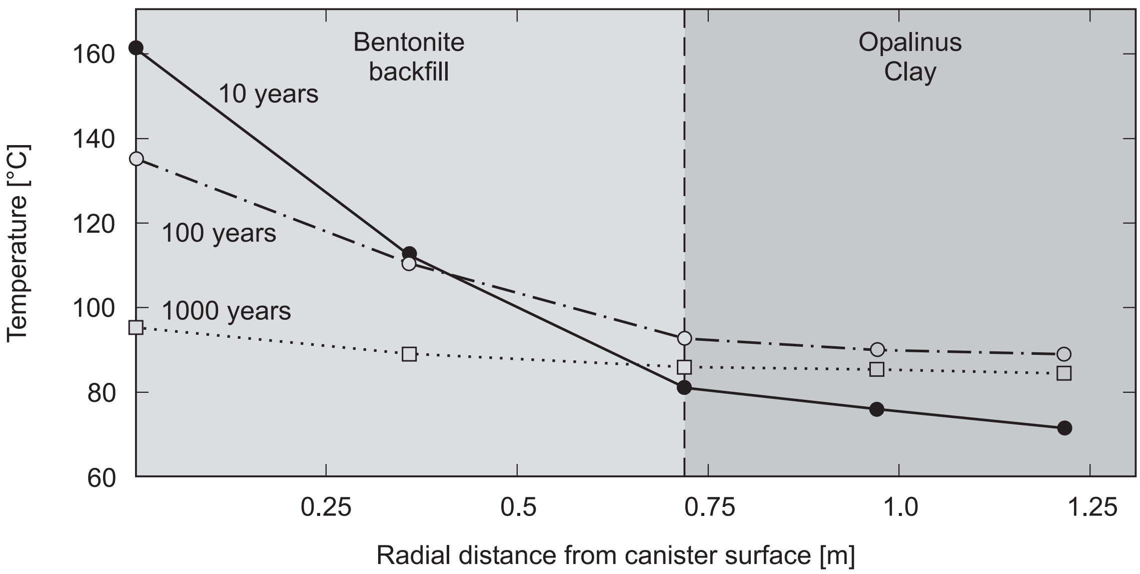

Fig. 5.3-3: Time-dependent temperature evolution at various positions within the engineered barrier system and surrounding rock for canisters containing 4 PWR SF assemblies (3 UO_2 plus 1 MOX)

The bentonite is assumed to have a thermal conductivity of 0.4 W m^{-1} K^{-1} and a heat capacity of 1.2 MJ m^{-3} K^{-1}. The initial ambient temperature is 38 °C. Canisters have a heat output of 1490 W at the time of waste emplacement in the repository (Johnson et al. 2002).

The temperature of the bentonite at the canister surface and throughout the bentonite is greatly influenced by the thermal conductivity of the bentonite, which is a function of the density and water content. For the expected initial water content of ~ 2 %, the corresponding low bentonite thermal conductivity leads to a maximum temperature of the bentonite at the canister surfaces of 140 to 160 °C for both SF and HLW. In the case of HLW, the temperature at the canister surface will decrease to ~ 80 °C within 100 years, whereas the temperature at the surfaces of SF canisters will remain above 100 °C for almost 1000 years. The outer half of the bentonite will always remain below ~ 115 °C in both cases. The 3 m spacing between canisters along the tunnel axis is sufficient to ensure that the maximum temperature at the mid-point between canisters is always < 95 °C for UO_2/MOX canisters (Fig. 5.3-5), < 85 °C for UO_2 canisters and < 80 °C for HLW canisters. Should the bentonite saturate rapidly (i.e. within decades), peak temperatures at the canister surface would be only slightly reduced, as the maximum canister surface temperature is reached within ten years of canister emplacement. Rapid saturation of the bentonite close to the canister is extremely unlikely, because of the very low hydraulic conductivity of the Opalinus Clay, which limits the water inflow rate, and because the thermal gradient in the vicinity of canisters drives moisture away from the canister surface (see Section 5.3.3). Possible changes in the properties of bentonite arising from elevated temperatures are discussed in Section 5.3.4.3.

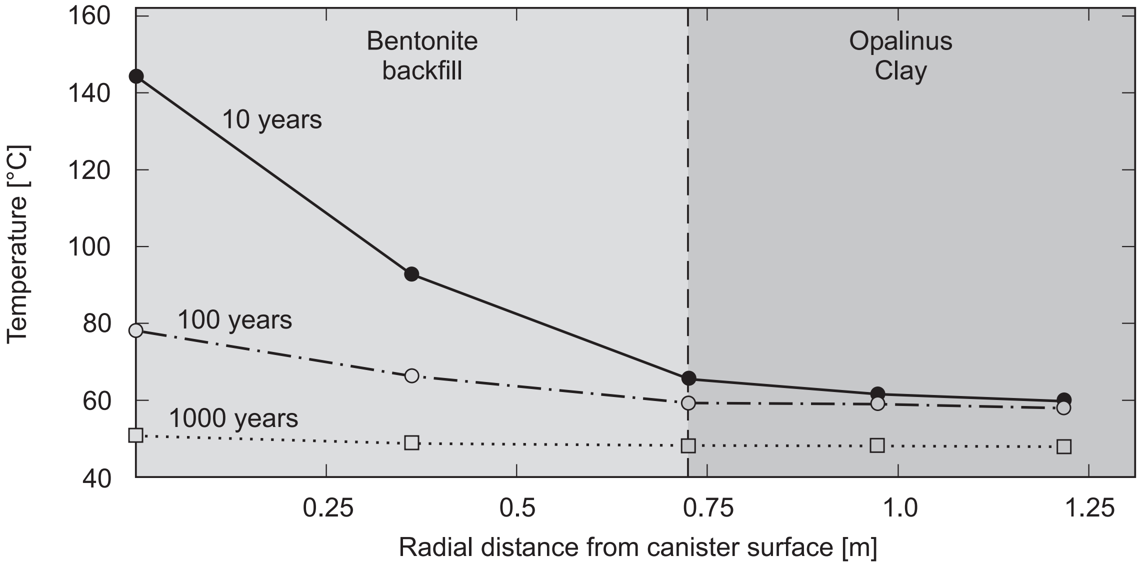

Fig. 5.3-4: Time-dependent temperature evolution at various positions within the engineered barrier system and surrounding rock for canisters of BNFL HLW glass

The canisters are assumed to have a heat output of 688 W upon emplacement in the repository, corresponding to 40 years cooling of the HLW glass prior to emplacement in the repository (Johnson et al. 2002). Ambient temperature, bentonite thermal conductivity and heat capacity as in Fig. 5.3-3.

5.3.2.2 Possible deviations from expected thermal behaviour

The strong dependence of near field temperatures on bentonite thermal conductivity leads to considerable uncertainty in temperatures within the bentonite because groundwater inflow rates are uncertain. Saturation of the near field within decades would lead to significantly lower temperatures (10 to 20 °C lower at the mid-bentonite position). Because the maximum temperature at the bentonite-canister interface is reached in about 10 years, i.e. when this region is dry, there is relatively little uncertainty regarding the maximum projected temperatures at this location. Smaller uncertainties in predicted temperatures (< 10 °C) arise from uncertainty in the values of thermal conductivity assumed for the host rock.

5.3.3 Hydromechanical evolution of SF / HLW near field

5.3.3.1 Evolution of the near field rock and bentonite backfill system

Effects of excavation

The region of rock immediately surrounding the emplacement tunnels (the excavation-disturbed zone or EDZ) will become partially desaturated as a result of evaporation due to ventilation during the construction and operation phase. Stress re-distribution due to excavation will lead to the formation of micro- and macro-scale fractures in the EDZ and desaturation will result in stiffening of the clay. For unsupported SF / HLW emplacement tunnels, the rock is sufficiently strong that the deformation immediately upon excavation and during the operational phase (1 to 2 years for a given tunnel) is limited (approximately 1 to 2 % convergence), although fracturing may extend outwards approximately 1.6 tunnel radii, equivalent to 2 m, from the roof and floor of the tunnels (Nagra 2002a).

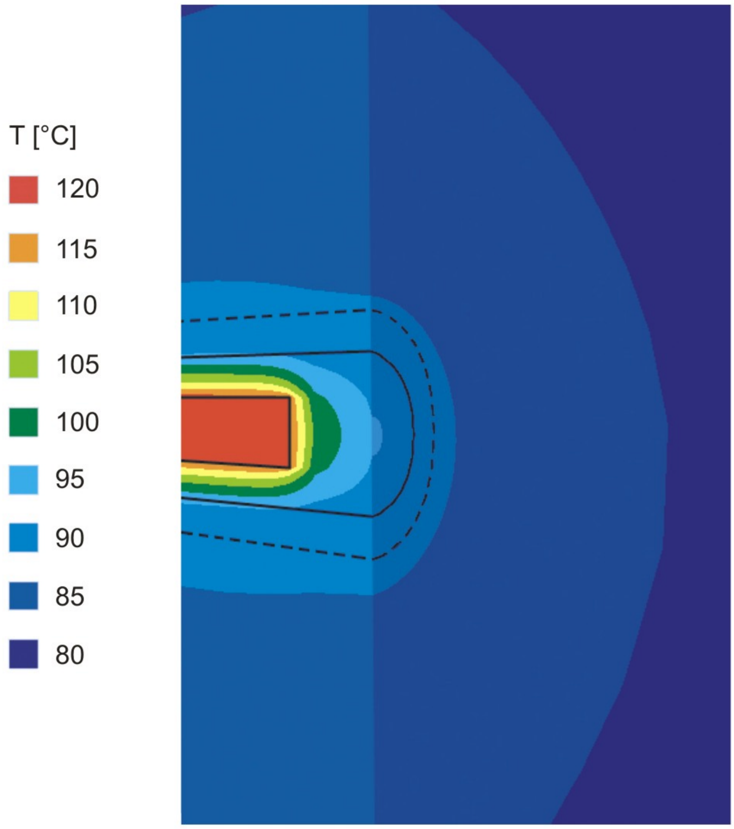

Fig. 5.3-5: Temperature distribution for a disposal tunnel with canisters containing 4 PWR SF assemblies (3 UO_2 plus 1 MOX) at 268 years after emplacement, for a bentonite thermal conductivity of 0.4 W m^{-1} K^{-1}

The canister is represented by the rectangle at the centre of the tunnel and the solid line and dashed line represent the tunnel boundary (R = 1.25 m) and the assumed outer boundary of the EDZ (R = 1.75 m), respectively. The time of 268 years represents the time of maximum temperature of the bentonite at the mid-point between canisters and in the surrounding rock. The vertical cross-sections are at mid-tunnel and midway between canisters (Johnson et al. 2002).

Some oxidation of pyrite may occur on fracture surfaces, leading to the formation of small amounts of gypsum and iron hydroxide, which has been studied in detail. In the case of SF / HLW emplacement tunnels, which will be open for only one to two years, estimates of the extent of oxidation have been derived from various field studies. Results indicate that gypsum formation in the EDZ is limited to open fracture surfaces (Mäder & Mazurek 1998). Calculations based on field studies of tunnels open from a few years (Mont Terri) to over 100 years (Hauenstein railway tunnel) permit bounds to be placed on the extent of oxidation. For SF / HLW emplacement tunnels, only about 1 % of the pyrite originally present will be altered, thus long-term impacts will be insignificant (Mäder 2002, Nagra 2002a).

Near field saturation

After emplacement of the bentonite, resaturation of the partially desaturated EDZ will gradually occur. The fracturing of the rock upon excavation and its low strength when resaturated, combined with the ~ 1 to 7 % swelling capacity of Opalinus Clay (Nagra 2001 and 2002a, Meier et al. 2000) is expected to result in effective homogenisation and self-sealing of the EDZ and gradual convergence of the tunnels. This deformation process is illustrated on a small scale in Fig. 5.3-6, which shows the behaviour of a small diameter (8 mm) borehole in Opalinus Clay under a confining stress. When the hole is dry, the material is sufficiently strong to be self-supporting with a circumferential stress of 30 MPa. When water is present, weakening of the material leads to creep and borehole convergence.

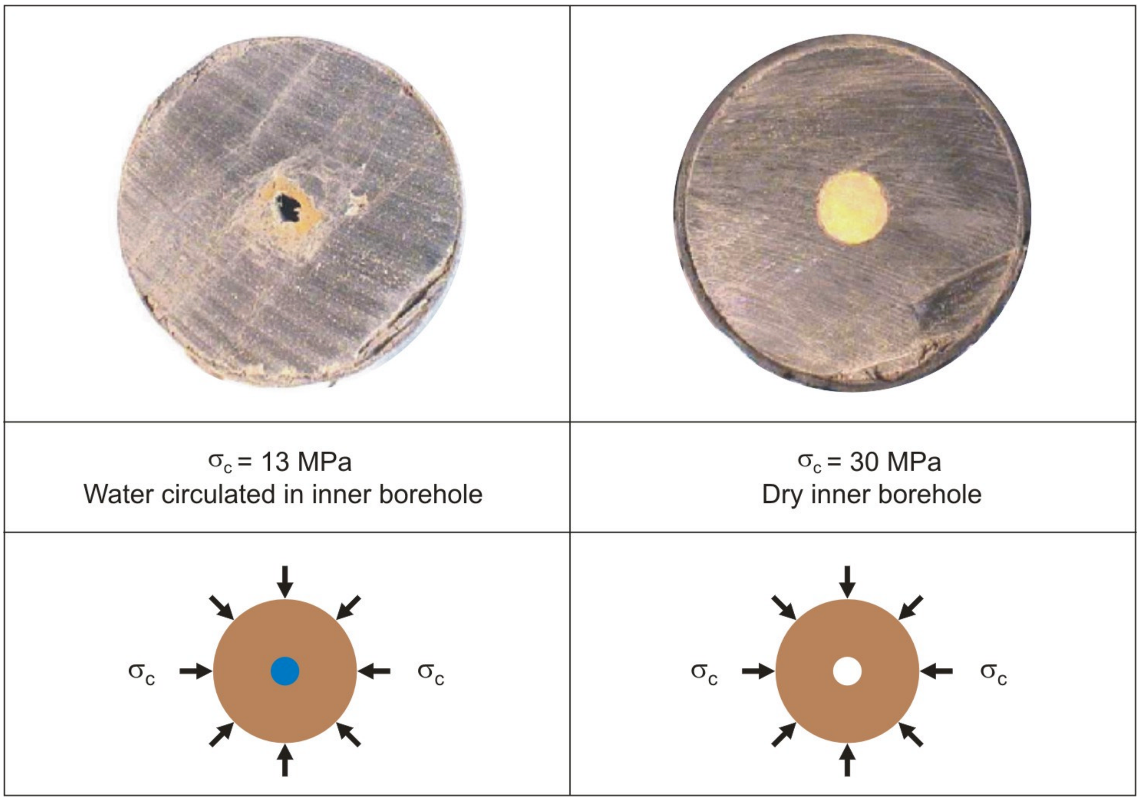

Fig. 5.3-6: Small-scale demonstration of strength reduction of Opalinus Clay due to water weakening of stressed material at an excavation boundary

Sample diameter = 3 cm (Nagra 2002a). In both cases, a confining stress is applied to the sample.

The reduction in strength of the Opalinus Clay when saturated is greater at elevated temperatures. The impact of the tunnel convergence and self-sealing process on the hydraulic properties of the EDZ have not yet been examined at full scale[80], but, based on the process understanding and hydraulic modelling results, the long-term effective hydraulic conductivity of the 2 m thick EDZ is expected to be increased by approximately one order of magnitude relative to the undisturbed rock (Nagra 2002a).

High temperatures near the canisters would initially cause water vapour to migrate away from the canister surfaces (Börgesson & Hernelind 1999). As the bentonite approaches approximately 50 % saturation, the highly compacted granules will begin to swell and a homogeneous microstructure will quickly develop once saturation is achieved, as has been observed in studies by Pusch et al. (2002) and Dereeper & Volckaert (1999). At full saturation, a swelling pressure of ~ 2 to 4 MPa will develop and the hydraulic conductivity will be 10^{-12} m s^{-1} (Pusch et al. 2002) .

80 The SELFRAC experiment, which is examining these processes, was still underway at Mont Terri when this chapter was developed.

The tunnel convergence process will compact the bentonite to a higher density, likely in concert with the resaturation process. The final saturated density of the bentonite will not exceed ~ 2.15 Mg m^{-3}, because at this density the swelling pressure of ~ 15 MPa will approximately balance the external stress field (~ 16 MPa vertical and 15 MPa minimum horizontal). Assuming that this stress balance defines the limiting state of convergence, the tunnel radius will be reduced from 1.25 to 1.15 m. Concurrent with the slow compaction of the bentonite, its porosity would decrease from ~ 0.45 to ~ 0.36, and its permeability from 10^{-12} m s^{-1} to less than 10^{-13} m s^{-1} (Dixon 2000).

Because of the very low hydraulic conductivity of the Opalinus Clay ($\leq 10^{-13}$ m s^{-1}), water uptake by the bentonite will be slow. The range of possible timescales for saturation of the bentonite has been estimated using various models, including a transient analytical model that does not include temperature effects and TOUGH-2, a hydraulic model partially considering thermal effects (Nagra 2002a). The models suggest resaturation times ranging from ~ 100 years to many hundreds of years, reflecting, in particular, the uncertainties in some of the values of the hydraulic parameters of the Opalinus Clay and bentonite. The impact of uncertainty in water inflow rates on canister corrosion and bentonite behaviour is discussed in Section 5.3.4.

The hydraulic conductivity of bentonite increases with increasing salinity, although the effects are not significant for the salinity of Opalinus Clay porewater at saturated densities of bentonite greater than 1.9 Mg m^{-3} (Dixon 2000). The porewater salinity may initially be slightly increased (~ 2 %) as a result of evaporation during the operating phase (Gribi & Gautschi 2001), but this does not affect the conclusion.

It is possible that some compaction of the bentonite backfill by tunnel convergence will occur while its moisture content remains low, because of the slow water inflow rate. Under such conditions, the partially saturated backfill may be compacted to a higher dry density (to a maximum of ~ 1.7 Mg m^{-3}, cf. 1.5 Mg m^{-3} at emplacement). This represents a saturated density similar to the values noted above, thus the same stress balance is expected to result in the long term.

The temperature rise in the Opalinus Clay surrounding the disposal tunnels is expected to produce a thermal expansion of porewater, which would increase fluid pressures and compressive stresses. Calculations based on the approach described by Horseman (1994) suggest that excess pore pressures may reach about 5 MPa after several hundred years (Nagra 2002a). The pressures are likely to dissipate gradually with cooling and are not sufficient to reactivate existing discontinuities or fractures in the Opalinus Clay (Nagra 2002a).

Gas production and transport through bentonite

Various gases will be produced in the SF / HLW near field as a result of metal corrosion, radiolysis of water and radioactive decay. The gas produced in largest quantity is H_2, arising predominantly from anaerobic steel corrosion, with lesser amounts from α-radiolysis of water after canister breaching. Other gases produced in much smaller quantities after breaching of the canister include Rn, He, Xe, Kr and, possibly ^{14}C (as CH_4 or CO_2), all present in, or arising from, radioactive decay of the waste. Once the near field becomes partially saturated with groundwater (after some hundreds of years), anaerobic corrosion of steel canisters at a rate of 1 μm a^{-1} (Johnson & King 2003) will produce H_2 at a rate of ~ 4 mol a^{-1} per canister. The comprehensive measurements of Smart et al. (2001) suggest that lower rates are more likely, but at a pH of ~ 7.5 a rate of 1 μm a^{-1} cannot be excluded, because the passive film of magnetite may be more porous than at higher pH values. Hydrogen will initially dissolve in the porewater, but calculations show (Nagra 2003a) that for a corrosion rate of > 0.1 μm a^{-1}, the production rate exceeds the diffusive transport rate of dissolved gas, because of the low diffusion rates of dissolved H_2 in both bentonite and the surrounding Opalinus Clay. Thus the concentration of hydrogen in bentonite porewater at a hydrostatic pressure of 7.5 MPa will reach the solubility after several hundred years, after which a gas phase will form (see Section 5.5.2).

Laboratory studies show that gas breakthrough within bentonite occurs at a pressure approximately equal to the sum of the bentonite swelling pressure and the hydrostatic pressure (Pusch et al. 1985, Horseman et al. 1999 and Tanai et al. 1997). The swelling pressure of bentonite will increase from several MPa at a time shortly after saturation to ~ 15 MPa in the very long term after compaction caused by deformation and creep of the surrounding Opalinus Clay under the lithostatic load. There is some uncertainty about whether the gas breakthrough mechanism in bentonite involves capillary flow (displacement of water in capillaries) or microfracturing and pathway dilation (Rodwell et al. 1999), but recent studies discussed in Swift et al. (2001) provide some evidence for the latter mechanism. Irrespective of uncertainties about the details of the mechanism, there is agreement that there is a threshold pressure required for gas breakthrough, that gas entry and gas flow result in very little desaturation of the clay, and that gas pathways will reseal if the gas pressure drops and water is available (Swift et al. 2001). The discrete gas pathways (or capillaries) formed by gas breakthrough are believed to have a diameter of < 1 µm. Ortiz et al. (1997) also present evidence that the extent of water expulsion is very small (< 1 %). The transport of gas through host rock is discussed in Section 5.5.2.

Possible deviations from expected hydromechanical behaviour

The uncertainties related to the hydromechanical behaviour of the near field are largely related to the time over which the various processes take place. Although the duration of tunnel convergence and the associated compaction of bentonite to a higher density is uncertain, the process is expected to occur concurrently with resaturation and be largely complete within several thousand years, i.e., prior to breaching of the waste canisters.

A possible, although unlikely phenomenon, involves potential gas-driven accelerated release of porewater containing radionuclides from SF canisters, in the event that breaching of the canister wall occurs only on the underside. In this case, flow of water into the canister could occur, followed by H_2 gas production from corrosion of internal canister surfaces, which could expel water from the canister.

5.3.3.2 Hydromechanical evolution of the SF and HLW canisters

As the bentonite saturates, it will develop a swelling pressure of ~ 2 to 4 MPa, increasing to a maximum of ~ 15 MPa as a result of convergence of the tunnels. This pressure, combined with the hydrostatic pressure of 7.5 MPa, will impose a load of ~ 22 MPa on the canisters. The load may be somewhat anisotropic initially, because of differences between the swelling pressure of highly compacted bentonite blocks (~ 10 MPa) and granular bentonite backfill (2 to 4 MPa), but will gradually become isotropic because of the plasticity of saturated bentonite. Canister sinking as a result of consolidation and creep is expected to be very limited (~ 1 cm), based on the information presented by Pusch & Adey (1999).

The HLW canister is designed to withstand an isotropic load of 30 MPa with safety margins higher than required by the corresponding ASME pressure vessel code, even after reduction of the wall thickness by 5 cm due to corrosion (Steag & Motor Columbus 1985).

The cast steel SF canister design is described in Section 4.5.3.1 and has a minimum wall thickness of ~ 15 cm. Calculations for an external pressure of 40 MPa indicate that when the wall thickness is eventually reduced to ~ 5 cm by corrosion, stresses may reach the failure criterion (Johnson & King 2003). No analysis has been performed for cases where loads on the canister might be anisotropic, such as during resaturation or as a result of tunnel convergence. Nonetheless, the results summarised below for Cu/cast iron canisters suggest this will not lead to canister damage.

Stresses in excess of the sum of lithostatic and hydrostatic load could arise eventually as a result of the volume of canister corrosion products, because magnetite has a lower density than steel. The effects of volume increase on stresses have been examined in the H-12 study (JNC 2000) for the case of a steel canister surrounded by bentonite. The results show that the additional forces generated are relatively small for bentonite thicknesses of > 50 cm; this process is therefore considered unimportant for the Nagra steel canister designs and tunnel diameters.

The Cu/cast iron canister of SKB, considered as a design variant, has a Cu shell with a thickness of 5 cm and a cast iron insert. Calculations discussed in SKB (1999) indicate that it has a collapse pressure of 80 MPa in the case of isotropic loading. For various anisotropic loading scenarios, the stresses in the insert were found to lie far below the yield strength in all cases.

Gases released from spent fuel while the canisters remain unbreached include fission gases, He and Rn, all of which may accumulate in the void space in the canister if the Zircaloy cladding is breached by, e.g. creep rupture. Internal pressures in the canister would be expected to reach only ~ 1 MPa in 100 000 years.

It is thus clear that both the SF and HLW canisters have a large initial margin of safety in relation to structural strength. Nonetheless, structural failure is expected to occur after > 10 000 years, because the wall thickness will be reduced by corrosion. The various corrosion processes considered and their projected rates are discussed in Section 5.3.4.4.

Possible deviations from expected hydromechanical behaviour of canisters

Large safety margins exist in relation to structural integrity of SF and HLW canisters; there therefore appears to be little likelihood of structural failure until extensive corrosion has occurred. The remote possibility of stress-corrosion cracking in the weld region causing a reduced lifetime is discussed in Section 5.3.4.4.

5.3.4 Chemical evolution of the SF / HLW near field

5.3.4.1 Evolution of the porewater chemistry of the SF / HLW near field

Overview

The mineralogy of Opalinus Clay and the composition of its porewater have been discussed in Section 4.2.6. In the far field, removed from the influence of elevated temperatures and repository materials such as bentonite and steel, the porewater chemistry is expected to change only very slowly (a small decrease in salinity may occur in the long term (Nagra 2002a)). In the near field, by contrast, chemical changes will occur as a result of temperature-dependent interactions. The air entrapped in bentonite at the time of repository closure will lead to a short period of oxidising conditions, and some mineral impurities present in bentonite will dissolve in the groundwater that gradually saturates the near field. The effects of these changes on repository performance are discussed in the following sections.

Evolution of redox conditions in the near field

Redox conditions in the near field will initially be oxidising, as a result of the presence of air trapped in pores in the bentonite. MX-80 bentonite contains a small amount of pyrite (0.3 wt %) and siderite (0.7 wt %) (Müller-Vonmoos & Kahr 1983)[81], oxidation of which would be expected to speed up the consumption of oxygen. Other factors affecting the redox state include microbial activity, which can catalyse attainment of reducing conditions, corrosion of the steel canisters and the presence of pyrite (~ 1 wt %) and other reduced minerals in the Opalinus Clay.

Microbial studies suggest that oxygen consumption by microbes is far more rapid than by inorganic reactions. For example, Pedersen (2002) suggests that it will occur in less than a year in sealed tunnels in crystalline rock after repository closure. Nonetheless, evidence for the lack of significant microbial activity in highly compacted bentonite (West et al. 2002, Pedersen 2002) indicates some uncertainty regarding whether this will occur as rapidly in the case of slowly resaturating SF / HLW emplacement tunnels in Opalinus Clay.

The low moisture content of bentonite will initially hinder consumption of O_2 by the steel canister; if saturation is slow, this situation may last for a considerable time. If moisture inflow is relatively fast, much of the O_2 is likely to be rapidly consumed by the canister and redox-active minerals in the bentonite. In the expected case of slow water inflow rates, the relatively high reactivity of pyrite towards O_2 (Wersin et al. 1994a), the high moisture content of Opalinus Clay (compared to the bentonite), and the high rate of O_2 diffusion in unsaturated bentonite will favour consumption of O_2 by pyrite present in the EDZ of the Opalinus Clay and by steel in the emplacement tunnels (rails, mesh). Depending on the rate of saturation of the bentonite, the consumption of O_2 may take a few years (if bentonite saturation is rapid) to a few decades (for a longer unsaturated period). Mass balance calculations indicate that corrosion of the steel canisters to a depth of tens of μm or oxidation of pyrite contained in a ~ 2 cm thick region of Opalinus Clay at the excavation boundary would be sufficient to consume all the O_2 in the near field (Wersin et al. 2003).

Once reducing conditions are established, a number of redox reactions may influence the redox potential, but the most important couples are expected to be Fe(metal)/Fe_3O_4 at the canister surface, Fe(II)/green rust[82] (Cui & Spahiu 2002), Fe(II)/ Fe_3O_4 and H_2/H_2O. The H_2/H_2O couple is generally considered to be rather unreactive, but the high partial pressure of H_2 (> 10 MPa), resulting from anaerobic corrosion of steel, may considerably increase its reactivity (Spahiu et al. 2000). The surface of the canister will become covered with Fe_3O_4, but galvanic coupling between steel corrosion and reduction of Fe(III) oxide phases that might form on Fe_3O_4 will prevent passivation and the surface of the magnetite is expected to remain reactive.

After canister breaching, additional influences on the redox chemistry are radiolysis at the spent fuel / water interface and redox reactions involving uranium. As noted in Section 5.3.4.5, high partial pressures of H_2 are seen to suppress fuel oxidation very effectively in the presence of radiation (King et al. 1999, Spahiu et al. 2000 and Röllin et al. 2001), implying that reduced U(IV) species may be dominant in solution. The presence of some U(VI) in solution is, however, considered possible, because of radiolysis at the fuel surface. The reduction of such oxidised species is expected to occur on magnetite covering the steel surfaces during diffusion out of the canister and on bentonite clay surfaces. This is confirmed by the studies of Morrison et al. (2001), which show that U(VI) is reduced to U(IV) during transport through a permeable Fe(0) barrier and Cui & Spahiu (2002), who found similar results for green rust formed on

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81 The actual bentonite used would be analysed to determine which of these minerals are present, and a similar reducing mineral could be added if deemed necessary.

82 Green rust is an Fe(II)/Fe(III) hydroxy-carbonate compound of variable composition.

corroding steel. The possibility of an oxidised zone (redox front) migrating from the canister thus appears very remote, even if relatively high dissolution rates are assumed for spent fuel (Johnson & Smith 2000).

There are significant difficulties in defining a unique redox potential in the bentonite porewater. Close to the canister surface, Fe(0) forces a very low redox potential leading to the decomposition of water and the formation of corrosion products (anaerobic iron corrosion). However, it is unlikely that the solutes in the porewater are in equilibrium with respect to the redox potential at the canister surface. Close to the host rock, redox conditions in the bentonite porewater will be influenced by diffusion of redox-active species from the surrounding Opalinus Clay. Based on thermodynamic modelling, Wersin et al. (2003) derived a bentonite porewater Eh range of -127 mV to -282 mV. This range includes uncertainties in pH as well as those related to the mineral phases controlling the Fe(III)/Fe(II) equilibria. Table 5.3-1 shows the derived Eh values for the magnetite/Fe(II) equilibrium where the dissolved Fe(II) concentration is assumed to be the same as in the surrounding Opalinus Clay. The results indicate that the uncertainty in pH affects redox potentials to a larger extent than do uncertainties related to Fe-bearing minerals.

Tab. 5.3-1: Calculated redox potentials within bentonite under the assumption of magnetite/ Fe(II) equilibrium for Fe (II) concentrations equal to those estimated for Opalinus Clay (Wersin et al. 2003)

Parameter	Nominal case	Lower pH limit	Higher pH limit
pH	7.25	6.9	7.8
Eh [mV]	-193	-127	-282

Possible deviations from expected redox chemistry in the near field

The possibility that oxidising species produced by radiolysis diffuse out of the canister and into the surrounding bentonite cannot be completely excluded, although this could only occur in the case of passivation of magnetite such that it becomes unreactive (Johnson & Smith 2000).

5.3.4.2 Porewater composition in the bentonite

Evolution of the bentonite porewater in the near field

Water taken up from the surrounding Opalinus Clay will induce dissolution of mineral impurities in the bentonite, such as gypsum, NaCl, carbonates and quartz, and in the EDZ, e.g. gypsum from pyrite oxidation. Interaction with the clay fraction and other silicate impurities will cause only slight changes in porewater composition. On the other hand, ion exchange and protonation/deprotonation reactions occurring at the montmorillonite surfaces will strongly affect porewater composition. Furthermore, dissolution of calcite and dissociation of carbonic acid are important reactions which will effectively buffer the solution against pH variations. Because of the similar geochemical properties of bentonite and Opalinus Clay, the compositions of their porewaters are not expected to be very different. Tab. 5.3-2 shows a modelled bentonite porewater composition (Curti & Wersin 2002) and, for comparison, the porewater composition of Opalinus Clay as presented in Section 4.2.6. The geochemical modelling assumed an open system concerning CO_2 partial pressure, i.e., it was assumed to be in equilibrium with Opalinus Clay. Tab. 5.3-2 also gives the expected extreme ranges of composition of bentonite porewater. The main uncertainty is related to the pCO_2 conditions in the surrounding host rock which have not been precisely determined (cf. Section 4.2.6). The slightly increased sulphate content in bentonite compared to Opalinus Clay is due to dissolution of gypsum impurities.

Tab. 5.3-2: Compositions of Opalinus Clay reference water (Pearson 2002) and bentonite porewater (Curti & Wersin 2002)

The bentonite reference water was derived by a thermodynamic model which includes ion exchange and surface complexation reactions and corresponds to an early stage after saturation. The expected maximum variations of bentonite porewater composition are also given. Total concentrations of dissolved components are in mol l^{-1}. Redox potentials are given in Tab. 5.3-1.

			Maximum expected variation	
	Opalinus Clay reference water	**Bentonite * reference water**	**Bentonite * low pH**	**Bentonite * high pH**
pH log pCO_2 [bar]	7.24 -2.2	7.25 -2.2	6.90 -1.5	7.89 -3.5
ionic strength [eq l^{-1}]	2.28×10^{-1}	3.23×10^{-1}	3.65×10^{-1}	2.63×10^{-1}
CO_3	2.70×10^{-3}	2.83×10^{-3}	6.99×10^{-3}	5.86×10^{-4}
Na	1.69×10^{-1}	2.74×10^{-1}	2.91×10^{-1}	2.49×10^{-1}
Ca	1.05×10^{-2}	1.32×10^{-2}	1.33×10^{-2}	1.34×10^{-2}
Mg	7.48×10^{-3}	7.64×10^{-3}	8.91×10^{-3}	6.15×10^{-3}
K	5.65×10^{-3}	1.55×10^{-3}	1.67×10^{-3}	1.38×10^{-3}
SO_4	2.40×10^{-2}	6.16×10^{-2}	6.39×10^{-2}	5.59×10^{-2}
Cl	1.60×10^{-1}	1.66×10^{-1}	2.06×10^{-1}	8.61×10^{-2}
Fe	4.33×10^{-5}	4.33×10^{-5}	7.74×10^{-5}	8.00×10^{-6}
Al	2.17×10^{-8}	1.92×10^{-8}	1.53×10^{-8}	7.55×10^{-8}
Si	1.78×10^{-4}	1.80×10^{-4}	1.80×10^{-4}	1.84×10^{-4}

* Since the applied model does not distinguish between the neutral external water and the diffuse double layer, the porewaters are slightly positively charged which is compensated by the negatively charged clay surface.

The temporal evolution of the porewater has been assessed with two simple models, a water exchange cycle model and a diffusion-reaction model (Curti & Wersin 2002). The results of both calculations indicate that the change in composition is small. This is because of the similar chemistry and the chemical stability of the argillaceous environment. Redox reactions, such as those involving Fe(II) and Fe(III), will not have a significant effect on the chemistry of the major ions.

Microbial activity in highly compacted bentonite is expected to decrease gradually with time, because pore sizes are smaller on average than typical cell diameters of microbes and because pores are poorly connected (Stroes-Gascoyne 2002). This is confirmed by experiments performed with a number of species, including sulphate-reducing bacteria (SRB), that might contribute to corrosion of metal canisters. For example, Pedersen et al. (2000a) have shown in experiments lasting 15 months at temperatures of up to 70 °C that only spore-forming bacteria survive in highly compacted bentonite and that their numbers are reduced markedly over time. In other studies, the activity of SRB was observed to cease at saturated densities higher than 1.5 Mg m^{-3} (Pedersen et al. 2000b). Furthermore, Pusch (1999) has shown that SRB are immobile in bentonite with a saturated density exceeding 1.9 Mg m^{-3}, thus they will not be able to migrate towards the canister from the surrounding rock. As a result, it is expected that bacteria will have a negligible impact on canister corrosion and on radionuclide transport in highly compacted bentonite.

Possible deviations from expected porewater composition in the near field

The similar mineralogical and porewater compositions of bentonite and Opalinus Clay suggest that the porewater composition in the near field will remain close to the Reference Case in Tab. 5.3-2. Both geochemical systems display a high buffering capacity towards acid-base and redox reactions. The main uncertainty is related to pH and pCO_2 conditions of the Opalinus Clay formation, which at present can be only roughly estimated. However, strong changes of these parameters with time are not expected. No realistic mechanism exists for intrusion of a groundwater of a significantly different composition for a repository closed and sealed as planned.

5.3.4.3 Mineralogical changes in bentonite

Maintaining the swelling properties and plasticity of at least the outer half of the bentonite barrier is considered important in relation to its functions of providing a low permeability diffusion barrier around the canister and providing a degree of swelling to limit the deformation of Opalinus Clay surrounding the excavations. There are several types of processes that might degrade swelling and reduce the plasticity of bentonite backfill over time. These include:

- dissolution and precipitation of silica and soluble trace minerals (e.g. $CaCO_3$, $CaSO_4 \cdot 2H_2O$, and $FeCO_3$);
- ion exchange of Ca in porewater with Na that is initially present in montmorillonite;
- alteration of montmorillonite to other clay minerals such as illite;
- reaction between Fe(II), from magnetite dissolution, and silica or montmorillonite;
- alteration of swelling properties by heating in the unsaturated state.

These processes and their consequences are summarised here.

Minerals such as $CaCO_3$, $CaSO_4 \cdot 2H_2O$, and $FeCO_3$ are present in rather small quantities in bentonite (~ 1 wt %). Some dissolution of these minerals can be expected, along with ion exchange of some Ca with Na present on the exchange complex of montmorillonite, although Na remains the dominant cation. In the unlikely case that early saturation of the buffer occurs, then the temperature gradient in the bentonite may also lead to precipitation of calcium sulphates at the canister surface. This has been observed in four-year long heater tests at maximum temperatures of 180 °C (Pusch et al. 1992), above the temperatures expected at the canister surface in the proposed repository. The resulting cementation of the clay in their study extended only a few centimetres from the canister surface. It is also possible that silica dissolving from quartz and smectite near the hot canister surface may migrate to cooler regions of the backfill and precipitate as chalcedony or quartz. The possible alteration of smectite to illite, a clay with very limited swelling capacity, also needs to be considered. Because of low potassium contents in bentonite and in Opalinus Clay porewaters (~ 5×10^{-3} mol l^{-1}), the supply of which is required for the reaction to proceed, the extent of illitisation will be negligible. Even if saturation occurs immediately after waste emplacement, the degree of illitisation of bentonite at the canister surface calculated with the method of Pusch & Madsen (1995) is only 5 % after ~ 10^5 years. As a result, illitisation is expected to be very limited. The most important effects of illitisation are reduced swelling capacity and the release and subsequent precipitation of silica, which can lead to an increase in strength (Pusch et al. 1998), presumably due to cementation between crystals. Nonetheless, the effects are considered to be unimportant for bentonite in the repository, because the amount of silica released and precipitated will be small. Observations from natural analogues such as the Kinekulle bentonite confirm this (Pusch et al. 1998). This bentonite, although experiencing even higher temperature than that expected in the proposed repository, leading to 20 to 40 % conversion to illite, still retains a plasticity and swelling capacity comparable to high density bentonite.

Magnetite will be formed on the steel canister surface as a product of the anoxic corrosion reaction. Under reducing conditions, magnetite may dissolve as Fe(II), which could favour formation of nontronite, a smectite with reduced swelling capacity, and other Fe-silicate phases (Grauer 1986). However, Müller-Vonmoos et al. (1991), in experiments performed at 80 °C over 6 months, found no evidence for Fe uptake by montmorillonite contacted with magnetite.

Studies by Couture (1985) and Oscarson & Dixon (1989) show clearly that uncompacted bentonite loses some of its swelling capacity due to silica cementation after even a few days of heating in a partially saturated state at temperatures above 110 °C. Compacted bentonite, in contrast, does not lose its swelling capacity at temperatures of 90 to 125 °C (Oscarson & Dixon 1990, Pusch 2000). Studies of the reference bentonite backfill, made up of ~ 80 % dense granules and ~ 20 % powder, also show that the swelling pressure is not reduced by exposure to steam at 125 °C, although it decreases by 50 % at 150 °C (Pusch et al. 2002).

The maximum temperature reached in the bentonite midway between the canister and the tunnel wall is 115 °C; midway between canisters along the tunnel axis, the maximum value reached is ~ 95 °C (Section 5.3.2). Thus, despite the likelihood of some local bentonite degradation, each canister is effectively surrounded by tens of cm of bentonite that will experience no significant degradation over time. Bentonite will also be used in the construction of seals at the ends of emplacement rooms and in other locations in the repository. All these seals will experience maximum temperatures of no higher than ~ 70 °C, thus the bentonite would not be thermally altered.

With respect to the significance of reduced swelling capacity of some of the bentonite closest to the canisters, the effect on near field hydraulic conductivity and radionuclide transport is likely to be small, because the outer portion of the bentonite will maintain its high swelling capacity, and because reduced average bentonite swelling capacity is likely to be compensated by increased tunnel convergence.

Possible deviations from expected physical and chemical behaviour of bentonite

The extent of alteration of bentonite is expected to be rather small, and is unlikely to effect its plasticity or swelling pressure, except close to the canister surface. It is difficult to quantify the effects or extent, but embrittlement seems possible near the canister surface only if early saturation of bentonite occurs. In addition, in the region of bentonite that may reach temperatures above ~ 125 °C (approximately the inner third), some reduction of swelling pressure could occur from cementation due to unsaturated heating effects. At distances greater than 25 cm from the canister, where temperatures are approximately 110 °C or less, bentonite clay is expected to remain essentially unaltered.

5.3.4.4 Corrosion of SF and HLW canisters

It is known that, in air, corrosion of mild steel is extremely slow provided the relative humidity is less than a critical value of ~ 60 % (Brown & Masters 1982). The 2 % initial moisture content of the bentonite backfill corresponds to a relative humidity of ~ 5 % (Marshall & Holmes 1979). In contact with bentonite, the critical relative humidity for initiation of aqueous corrosion may be reduced to about 30 to 40 % due to absorption of water by hygroscopic salts, in particular, trace quantities of $CaCl_2$ or NaCl present in bentonite (Mansfeld & Kenkel 1976). Such a high moisture level at the canister surface is unlikely to be reached for many years (see Section 5.3.3.1), because the high temperature gradient in the bentonite maintains low moisture levels in the hottest part of the bentonite, even when saturation of the outer bentonite is approached (Börgesson & Hernelind 1999). The corrosion of the steel canisters is therefore expected to be limited (< 100 μm) for the first decades, because the surfaces of canisters will remain dry until the humidity increases sufficiently that a thin film of water can condense and initiate both local and general corrosion (Johnson & King 2003). As discussed in Section 5.3.4.1, during this time period, much of the oxygen initially present in the bentonite will be consumed by reaction with pyrite and siderite in the bentonite and in the Opalinus Clay immediately surrounding the tunnel (Wersin et al. 2003) and by other steel materials present in emplacement tunnels (e.g. mesh and rails).

Corrosion of the canisters due to sulphide appears to be extremely improbable in the long term, because of the inability of sulphate-reducing bacteria (SRB) to thrive and be mobile in bentonite backfill (see Section 5.3.4.2). If a steady-state flux of sulphide to the canister surface is maintained as a result of SRB activity in the adjacent Opalinus Clay, this would lead to less than 1 mm corrosion in 10 000 years (Johnson & King 2003, Wersin et al. 1994b).

The effects of γ-radiation on corrosion of the SF canisters (wall thickness ~ 15 cm, compared to 25 cm for the HLW canister) are expected to be insignificant, because the radiation field at the canister surface is only ~ 35 mGy hr^{-1} at the time of canister emplacement (Kühl et al. 2003). This is well below the critical dose rate of 3 Gy hr^{-1}, which the studies of Marsh & Taylor (1988) suggest is the threshold for enhanced corrosion due to radiolysis.

The possibility that stress-corrosion cracking (SCC) could occur in the weld region has been examined. A summary of studies of SCC by JNC (2000) suggests that its occurrence is highly unlikely. This is supported by the discussion in Johnson & King (2003), which notes that, of the various forms of SCC, only high and low pH SCC in HCO_3^-/CO_3^{-2} need be considered in a repository environment. This type of SCC occurs only at slightly acid (pH ~ 6) and moderately alkaline (pH ~ 10 – 11) conditions, significantly below and above the expected bentonite pore-water pH of ~ 7.3. Furthermore, cyclic loading, which would not occur in a repository, is believed to be necessary for crack propagation, thus SCC of the weld region is considered to be only a remote possibility.

After several decades or longer, water reaching the canister surface will initiate corrosion. In the unlikely event that oxygen still remains at this time, it will cause rapid aerobic corrosion at a rate of tens of μm a^{-1} until all the oxygen is consumed, a time period of only a few years (the amount of oxygen is limited to that trapped in the pore spaces of bentonite around each canister). Pitting corrosion is possible during this period of oxic corrosion, thus penetration depths may be somewhat greater. Studies based on long-term field burial tests discussed in JNC (2000) and Johnson & King (2003) illustrate that the degree of localisation decreases as corrosion progresses, thus initially large pitting factors (~ 100) decrease to ~ 10 when general corrosion has progressed to a depth of 0.3 mm. Following this, anaerobic corrosion will proceed at a rate of ~ 1 $\mu m\ a^{-1}$ (Smart et al. 2001). Such low long-term corrosion rates are consistent with results of natural analogue studies of iron and steel archaeological artefacts (Miller et al. 1994), which yield rates of 0.1 to 10 $\mu m\ a^{-1}$, with the higher end of the range likely to be representative of steels exposed to aerated sediments. Based on these rates, and considering the short-term occurrence of pitting, corrosion to depths greater than a few cm in less than 10^4 years appears very unlikely. Mechanical breaching may eventually occur by collapse after a time in excess of 10^4 years. A detailed evaluation of corrosion of the steel canister is given in Johnson & King (2003).

The performance of the alternative of long-lived copper canisters (lifetime > 10^5 a) for the disposal of SF is discussed by Johnson & King (2003).

Possible deviations from expected corrosion behaviour of canisters

There is a possibility that some steel canisters will be manufactured with defects that escape detection during inspection and which could affect canister integrity (e.g., through-wall or near through-wall welding defects). Such failures are extremely improbable with good quality control of canister welding and inspection. For example, in the case of long-lived copper canisters (lifetime > 10^5 years), the possibility of a pinhole defect at a rate of 1 in 4500 canisters is assumed in Andersson (1999), arising from an undetected manufacturing flaw. Uncertainties regarding the corrosion performance of canisters are largely related to uncertainties in the rate of water inflow to the near field. If water inflow is more rapid, corrosion will begin earlier, but rapid saturation (within decades) appears to be extremely improbable. Nonetheless, should it occur, the overall impact would be small. There is a remote possibility of canister breaching by SCC of the weld region after a period of perhaps thousand years.

5.3.4.5 Chemical processes within a SF canister after breaching

Steel canisters are expected to be breached after a period of > 10^4 years. By this time the temperature of the fuel and canister would be ~ 50 to 60 °C. Following breaching of canisters, water inflow through penetrations would be slow because of the low hydraulic conductivities of both the bentonite and the Opalinus Clay host rock (see Section 5.3.3.1) and gas pressure generated by corrosion occurring when water enters the canister. Zircaloy cladding would be largely intact, although some failures during operation or during the period prior to canister breaching may occur in the form of narrow cracks as a result of hydrogen-induced cracking or creep rupture (Poinssot et al. 2001). Because of uncertainties in assessing such processes over the long term, containment credit by Zircaloy cladding is not generally assumed in safety analysis calculations (e.g. Andersson 1999, Johnson et al. 1994). The main processes controlling radionuclide release from the SF assemblies (see Fig. 4.5-2) include:

- Corrosion of Zircaloy cladding, releasing entrained activation products such as ^{14}C and ^{36}Cl. Zircaloy has a high corrosion resistance in spite of the possibility of local failure, so release of radionuclides will occur over tens of thousands of years.
- Corrosion of other metal parts of the fuel assembly composed of Ni alloys, releasing activation products such as ^{59}Ni and ^{63}Ni. This is also a slow process because of the corrosion resistance of Ni alloys.
- Preferential leaching of a fraction of the inventory of some radionuclides, e.g. ^{129}I, ^{36}Cl, ^{14}C and ^{135}Cs, located in the gap between the fuel pellets and the cladding and at grain

boundaries within the fuel (see Fig. 4.5-2). Release of leached radionuclides is a rapid process, occurring over a period of a few days (gap) to years (grain boundaries).

- Dissolution of the fuel pellets, releasing uranium, other actinides and entrained fission and activation products. The fuel matrix has a low solubility under anoxic repository conditions, thus dissolution occurs very slowly, although it may be influenced by α-radiolysis of water which produces oxidants such as H_2O_2. Corrosion of the steel canister under anoxic conditions produces magnetite (Fe_3O_4) and H_2, which are effective reducing agents that contribute to slow dissolution of the fuel.

Corrosion of Zircaloy and other metal parts

In the presence of water or water vapour, a thin highly protective oxide film forms on Zircaloy. As a result, release of radionuclides entrained in the alloy is very slow. The corrosion rate of Zircaloy in groundwater is expected to be < 0.01 $\mu m\ a^{-1}$ (Johnson & McGinnes 2002). The outer porous oxide film on Zircaloy, formed while the fuel is in the reactor, may contain up to 20 % of the ^{14}C, possibly present in organic form (Yamaguchi et al. 1999). This can be released more rapidly by diffusion out of the oxide film. Other metal parts of the fuel assemblies are made of Ni-based alloys, which are expected to have very low-corrosion rates (< 0.01 $\mu m\ a^{-1}$) in groundwater (Wada et al. 1999).

Preferential leaching of fission and activation products

The concentrations of some radionuclides are enriched on fuel pellet and crack surfaces and in the fuel/cladding gap as a result of enhanced release from the fuel matrix during in-reactor irradiation. Several percent of the inventory of a number of radionuclides is present in such locations and this fraction is leached rapidly in the presence of water (Johnson & McGinnes 2002). These radionuclides also have enhanced concentrations at fuel grain boundaries. Release from grain boundaries occurs over a longer time frame because water penetrates them more slowly. The release rates from grain boundaries are, however, generally considered to be rapid with respect to other release and transport processes and are combined with the gap release to give the so-called instant release fraction (IRF) (Johnson & Tait 1997). It is emphasised that the IRF concept implicitly assumes that the fuel has a network of open pathways connecting all grain boundaries. As a result, should microcracking of fragments occur, (for example, by He pressure buildup in bubbles at grain boundaries in the UO_2 (Poinssot et al. 2001)), IRF values will not increase. Estimated IRF values for UO_2 and MOX fuel are given in Appendix 2, Tab. A2.2.1. No data are available for MOX fuel, thus bounding IRF values are based on fission gas release measurements for UO_2, as the fractional gas release typically exceeds the release of other radionuclides (Johnson & McGinnes 2002).

Dissolution of the fuel matrix

The largest proportion of the radionuclide inventory is present within the fuel grains and its release is limited by the rate of fuel dissolution. The principal factor controlling the dissolution rate of the UO_2 matrix under disposal conditions is the oxidant supply. After an initial period of containment the main source of oxidants within a water-saturated steel canister is α-radiolysis, which produces principally molecular oxidants (H_2O_2) and reductants (H_2). For MOX fuel, which has an α-activity of several times that of UO_2 fuel, the production rate of radiolysis species is proportionately higher. The net yield of oxidants from α-radiolysis is expected to be quite low in the presence of dissolved Fe(II) and magnetite, and the high hydrogen partial pressures resulting from corrosion of the steel canister, due to consumption of oxidizing radicals and H_2O_2 by hydrogen (Johnson & Smith 2000, Christensen 1998). This is illustrated by experimental measurements under strongly reducing conditions with hydrogen overpressures, which indicate that radiolytic oxidative dissolution of SF does not occur. Fig. 5.3-7 (Röllin et al. 2001) shows SF dissolution rates measured in flow-through experiments under aerated conditions and with a hydrogen overpressure of 0.1 MPa. Dissolution rates under reducing conditions are several orders of magnitude lower than rates for aerated conditions. In static experiments of SF dissolution with a hydrogen partial pressure of 5 MPa (Spahiu et al. 2000), dissolved U concentrations are < 10^{-8} mol l^{-1}, indicative of the predominance of less soluble U(IV), rather than the more soluble U(VI), in solution. Overall, the results suggest solubility-controlled dissolution, rather than interface reaction-rate controlled oxidative dissolution. Other direct evidence for strongly reducing conditions in the presence of H_2 is given by King et al. (1999), who show that potentials on a UO_2 surface are in the -500 to -800 mV range in the presence of a 5 MPa H_2 pressure and γ-radiation. Such potentials can only be consistent with a reduced surface. Thus, provided H_2 pressures in the near field remain high, which should be the case until well beyond the time at which the canisters are breached (i.e. a period of several hundred thousand years), fuel dissolution rates are expected to remain extremely low. By this time the Pu and Am isotopes responsible for the high α-dose rate at the fuel surface will have decayed away, and the subsequent effects of radiolysis on fuel oxidation can be considered negligible, based on evidence of the extremely long-term (10^8 – 10^9 years) geochemical stability of natural uraninite deposits in reducing environments (Cramer & Smellie 1994).

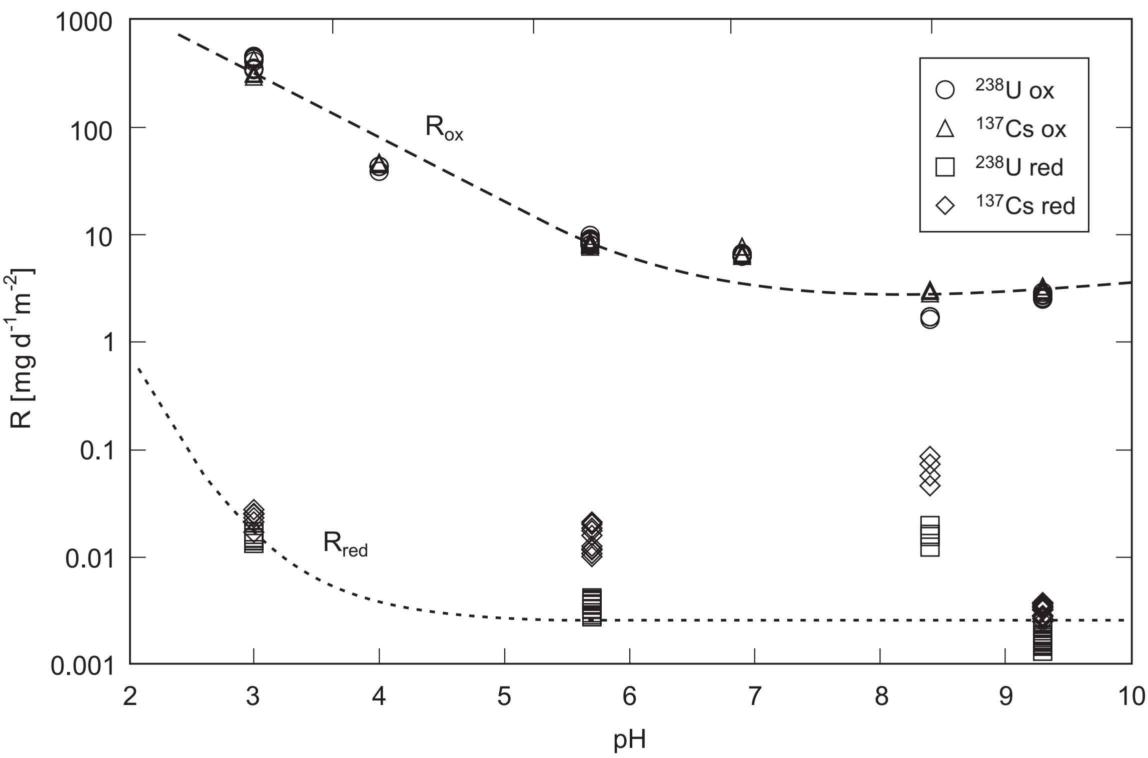

Fig. 5.3-7: Normalised dissolution rates (R) of ^{238}U and ^{137}Cs from spent fuel in 0.01 mol l^{-1} $NaHCO_3$ solution under oxidising (R_{ox}) (0.02 MPa O_2) and reducing (R_{red}) (0.1 MPa H_2) conditions (Röllin et al. 2001)

Note that the gap inventory was leached from the fuel sample prior to the experiment.

Formation of volatile radionuclides

Of the radionuclides present in the fuel and cladding, only ^{129}I, ^{14}C, and ^{3}H need be considered as having potential for formation of volatile forms. Studies of spent fuel solid-state chemistry summarised by Johnson & Shoesmith (1988) as well as leaching studies (Stroes-Gascoyne et al. 1995 and Gray 1999) confirm that ^{129}I is released to solution as iodide, a non-volatile form. Conversion to a volatile form (e.g. CH_3I or I_2) appears unlikely, as such compounds are reactive or hydrolyse readily to reform iodide. The case of ^{14}C is more complex. For Zircaloy cladding there is evidence of release of $^{14}CO_2$ from cladding in air under oxidising conditions (Ahn 1994), but leaching of an unidentified organic compound in anoxic alkaline solution (Yamaguchi et al. 1999) has also been observed. In the case of SF itself, the chemical form of released ^{14}C is unknown, because studies of release of ^{14}C from SF normally involve oxidative treatment of the solution to capture $^{14}CO_2$ for chemical analysis (Stroes-Gascoyne et al. 1994). About 1 % of the inventory of ^{3}H may be rapidly released from the fuel as a volatile species (Johnson & McGinnes 2002); however, the release is unimportant because of the short half-life.

Possible deviations from expected radionuclide release behaviour of SF

The processes controlling radionuclide releases from SF are reasonably well understood, but uncertainties remain regarding the magnitude of certain effects. The principal uncertainties that give rise to alternative model assumptions are judged to be:

- The chemical form of ^{14}C that is released relatively rapidly from the oxide layer of the Zircaloy and slowly by uniform corrosion of the underlying metal. In particular, ^{14}C may be released in organic form (Yamaguchi et al. 1999), which may allow it to be converted to a volatile form ($^{14}CH_4$) that could be more easily transported from the repository near field. Similarly, the ^{14}C released from spent fuel may also be organic in form.
- The importance of α-radiolysis in relation to oxidative dissolution of SF. There is a broad range of views expressed on the importance of the contribution of radiolysis to dissolution. Oxidants produced by radiolysis may be more reactive (i.e. less recombination) than noted above, leading to a greater rate of dissolution. A model in which the fuel dissolution rate is proportional to the α activity of the fuel is discussed by Johnson & Smith (2000), and estimated dissolution rates are given in Appendix 2, Tab. A2.2.2. Note that in Section 6.3.3, this more pessimistic model is adopted for the Reference Case, rather than the more realistic solubility-limited model discussed above. The hypothesis that radiolysis is substantially more aggressive than implied by the reference radiolytic dissolution model can be examined by increasing the rates in Appendix 2 (Tab. A2.2.2) by up to 100 times.

5.3.4.6 Dissolution of HLW

Chemical processes within an HLW canister after failure

Steel canisters are expected to be breached after a period of ~ 10^4 years (see Section 5.3.4.4). After canister breaching, slow groundwater inflow will lead to dissolution of the HLW glass. The dissolution rate of the glass is principally determined by temperature and solution composition. Initially, elements such as alkali-metals are selectively leached, but after a short period, all species in the glass are released at the same rate (congruent dissolution). Some radionuclides may precipitate readily due to low solubilities or be adsorbed on or incorporated into alteration phases forming on the glass surface. For a given temperature, pH and glass composition, the dissolution rate is controlled by the dissolved silica concentration, with the rate decreasing as the dissolved silica concentration increases (Curti et al. 1993). Sorption of silica onto Fe corrosion products decreases dissolved silica concentrations, increasing the glass dissolution rate (JSS 1988). A similar effect is observed with silica sorption on bentonite (Curti et al. 1993, Advocat et al. 1999). These effects are expected to be small, based on the analysis of Curti (2003), thus they will not significantly affect the long-term dissolution rate. The present study considers disposal of two different HLW glasses, with different compositions and dissolution rates. The estimated dissolution rates are 5.5×10^{-4} and 7.3×10^{-5} kg m^{-2} a^{-1} for the BNFL and COGEMA glasses, respectively (Curti 2003).

In estimating the dissolution rate of glass blocks from these rates per unit wetted surface area, the degree of cracking of the glass must be considered. Cracking occurs during cooling after vitrification and increases the surface area available for leaching. A factor of 15 relative to the original geometric surface area is assumed, slightly greater than the value of 12.5 used in Nagra (1994a).

For the estimated dissolution rates, including the effects of cracking, the BNFL and COGEMA glasses would dissolve at a rate of ~ 1 part in 10^5 and 1 part in 10^6 per year, respectively.

Possible deviations from expected corrosion behaviour of HLW glass

Difficulties with long-term extrapolation of laboratory data lead to some uncertainties in the long-term glass corrosion rate. As a result, a dissolution rate that is approximately two orders of magnitude higher or lower is not completely discounted. The very high rate is extremely pessimistic and is based on the assumption that the high initial rates measured in laboratory experiments prior to buildup of silica in solution are maintained, or alternatively, that rates remain high due to continuing sorption of silica on bentonite with an unrealistically high sorption coefficient for silica (Curti 2003).

5.3.5 Chemical processes and radionuclide migration in the SF / HLW near field

When canister breaching occurs, it most probably will involve small penetrations through which water inflow will be slow. The rate of water inflow is difficult to estimate because of uncertainty about various phenomena, including the number of penetrations and the build-up of gas pressures inside the canister as a result of steel corrosion that may reduce water inflow (SKB 1999). For steel canisters, the evolution of both the number of penetrations and their size over time is subject to considerable uncertainty. When water contacts the SF / HLW, radionuclide release will be controlled by the rate of waste form dissolution. Released radionuclides are subject to several chemical retention processes in the near field barrier systems depicted in Fig. 4.4-2 for SF and 4.4-3 for HLW, including:

- corrosion products either from the glass or the fuel pellets may take up radionuclides;
- large amounts of corrosion products from the steel canisters will trap many radionuclides;
- the chemical conditions in the near field porewater will favour low solubilities of many radionuclides;
- sorption will retard diffusion through the compacted bentonite;
- anion exclusion may retard diffusion rates of anionic radionuclides;
- solid solutions of radionuclides with minerals in the bentonite may be formed;
- colloids and high molecular weight species will be filtered due to the small pore size of bentonite.

Precipitation of radionuclides in the canister/bentonite environment

A factor limiting the release of many radionuclides from the canister after breaching is their low solubilities. The solubilities will be determined by the nature of the precipitating phase and by the porewater composition, which will be conditioned by the bentonite and corroding canister (see Section 5.3.4). Many safety-relevant radionuclides display low solubilities under the geochemical conditions expected in the near field. In particular, these include actinides (Th, Pa, U, Np, Pu, Am, Cm), most of their decay products (e.g. Ra and Ac) and some long-lived fission and activation products (e.g. Tc, Se and Ni). On the other hand, some long-lived radionuclides, such as ^{36}Cl, ^{129}I, ^{135}Cs and ^{14}C, are not expected to form insoluble phases and thus will dissolve according to their inventories and the rate of waste form corrosion. Solubility values used in assessment calculations for the former group do not incorporate co-precipitation and solid solution effects (except in the case of radium); such effects may further reduce solubilities. In the case of the HLW, the glass alteration products will also act as efficient scavengers for some trace elements (Pirlet 2001). The extent of co-precipitation and solid solution formation is difficult to quantify for most elements because of limited thermodynamic data on the phases formed and slow kinetics. An exception is radium which is known to co-precipitate with Ba and Sr in the presence of carbonate and other mineral phases (Bruno et al. 1997, Berner 2002).

The chemical speciation and solubilities of the radionuclides are influenced by the composition of the water that migrates from the bentonite into the breached canister. Ligands such as hydroxide and carbonate significantly affect the speciation of many nuclides. The speciation and solubilities have been evaluated (Berner 2002, Hummel and Berner 2002) using the recently updated Nagra/PSI Thermochemical Database (Hummel et al. 2002), which incorporates the thermodynamic data for U, Np, Pu, Am and Tc from the internationally recognised NEA TDB project. In addition, the Nagra/PSI Thermochemical Database update contains reviewed data for Th, Sn, Eu, Zr, Ni and Se. A list of the expected solubilities of all safety-relevant radionuclides including estimated uncertainties is given in Appendix 2, Tab A2.4. In the case of the actinides, amorphous oxide rather than the crystalline phases are assumed to be stable and solubility limiting in the near field. This assumption is based on both experimental results and field data from natural analogue studies (Bruno et al. 1997, Berner 2002). The canister corrosion process will result in very reducing conditions, which is an important factor in decreasing the solubilities of a number of elements, including U, Np and Tc. The presence of magnetite and possibly also green rust at the canister surface provides reactive sites for the reduction and precipitation of redox-sensitive actinides and fission products. Various experimental studies indicate that reactive Fe(II)-bearing minerals can reduce U(VI) and Tc(VII) on laboratory timescales (Liger et al. 1999; Cui & Spahiu 2002, Cui & Eriksen 1996, Lee & Bondietti 1983), thus confirming that both kinetics and thermodynamics favour the formation of low solubility reduced phases of these important elements.

Diffusion and sorption in the bentonite backfill

After saturation of the bentonite by groundwater, the hydraulic conductivity will be $< 10^{-12}$ m s^{-1}, thus transport of radionuclides will take place predominantly by diffusion through pores in the bentonite. Colloids bearing radionuclides may be formed during SF / HLW dissolution or subsequently by precipitation of insoluble radioelements or by coprecipitation and sorption processes. The small pore size in dense saturated bentonite, however, prevents transport of colloids (Kurosawa et al. 1997). Bentonite is therefore an effective transport barrier for radionuclides.

The diffusion of radionuclides through bentonite backfill is commonly described by Fick's law. During diffusive transport, radionuclides are retarded to varying degrees by sorption processes.

The degree of sorption is often quantified through the use of K_d values, in which the K_d is the ratio of the quantity of the element sorbed per unit mass of solid to the concentration in solution at equilibrium. K_d values are often obtained from batch sorption experiments at low bentonite/-water ratios (Bradbury & Baeyens 2003a). Elements with a low solubility, e.g. actinides, usually show strong sorption on bentonite, while anions such as I^- have only a very small affinity for mineral surfaces. Alternatively, diffusion experiments with compacted bentonite can provide apparent diffusion coefficients (D_a), in which the degree of sorption is incorporated in the diffusion coefficient term (Yu & Neretnieks 1997 and references therein, Sato & Yui 1997 and Sato 1998).

Consideration of the microstructure of smectite, the main constituent in bentonite, gives a qualitative picture of the process of solute transport (see e.g. Horseman et al. 1996): the main volume of porewater is located in the interlamellar spaces. The negative surface charges of these clay units and the narrow interlamellar spacing result in anion exclusion effects; as a result anions diffuse predominantly through the larger interparticle pore spaces. Ion exchange of cations occurs on the negatively charged surfaces in the interlamellar spaces and at the surfaces of the clay particles. Covalent bonding of heavy metals is thought to take place at the edges of the clay particles (edge sites). Both sorption processes result in retardation during diffusive transport.

As noted above, D_a values can either be obtained from diffusion measurements on compacted clay systems or they can be derived from K_d values obtained from batch sorption measurements on dilute systems, combined with appropriate effective diffusion coefficients (D_e values) and accessible porosities (ε) according to the relation:

$$D_a = \frac{D_e}{\varepsilon + \rho_d K_d}$$

where ρ_d is the clay dry density. In the present study the latter approach is used for the following reasons:

- A representative set of batch sorption measurements is available to derive a comprehensive K_d database. The method, including the adjustment to in-situ mineralogy and porewater chemistry, is described in Bradbury & Baeyens (2003a, 2003b) for bentonite and Opalinus Clay, respectively.
- Although the direct measurement of D_a values may be straightforward, obtaining good data for strongly sorbing elements on a laboratory timescale is difficult.
- Diffusion experiments in Opalinus Clay have been performed only for weakly sorbing elements; nonetheless, a consistent treatment of diffusive transport in the bentonite barrier and the Opalinus Clay had to be achieved.

It cannot be taken for granted that complete consistency will be observed between apparent diffusivities derived from batch K_ds and those directly determined in diffusion experiments, because of differences in porewater chemistry between highly compacted bentonite and more dilute systems. Bradbury & Baeyens (2002) derived a set of K_d values for five elements (Cs, Ni, Am, Zr and Np) from diffusion experiments with compacted Kunigel V1 bentonite (Sato & Yui 1997 and Sato 1998), using values for effective diffusivity (D_e) and accessible porosity of water derived from diffusion measurements with HTO. The data set had the advantage of covering the valence states from +1 to +5. Batch K_d values were adjusted for the bentonite porewater chemistry and mineralogy in the diffusion experiments using the approach discussed by Bradbury & Baeyens (2003a). The K_d values derived using the two different approaches are compared in Tab. 5.3-3.

Tab. 5.3-3: Comparison of K_d values of various elements derived from diffusion experiments with Kunigel V1 bentonite at a dry density of 1800 kg m^{-3} (Sato & Yui 1997 and Sato 1998) with those predicted from batch sorption data (Bradbury & Baeyens 2002).

Element	Apparent diffusion coefficient D_a from diffusion experiments [m^2 s^{-1}]	K_d derived from D_a from diffusion experiments [m^3 kg^{-1}]	K_d from batch experiments [m^3 kg^{-1}]
Cs(I)	$\sim 6.5 \times 10^{-12}$	$\sim 9.7 \times 10^{-3}$	$\sim 2.2 \times 10^{-2}$
Ni(II)	$\sim 1.6 \times 10^{-13}$	$\sim 4.0 \times 10^{-1}$	$\sim 4.3 \times 10^{-1}$
Am(III)	$\sim 1.3 \times 10^{-15}$	~ 49	44
Zr(IV)	$\sim 3.0 \times 10^{-15}$	~ 25	21
Np(V)	$\sim 4.0 \times 10^{-13}$	$\sim 1.6 \times 10^{-1}$	4.6 to 9.0×10^{-2}

The good agreement observed suggests that the use of K_d values from batch sorption experiments in safety assessment calculations is a valid approach, although it is acknowledged that some important discrepancies remain unresolved. A comparison with a broader spectrum of elements and diffusion experiments reveals reasonable agreement between the K_d values derived using the two different approaches, considering the rather scattered data from diffusion experiments. For some elements, however, K_d values listed by Bradbury & Baeyens (2003a) are either lower or higher than the ones derived from diffusion experiments in compacted bentonite. In particular, Bradbury & Baeyens (2002) calculate higher K_d values for Tc and U under reducing conditions than values derived from diffusion experiments. This may be because these two elements are extremely sensitive to oxidation, thus diffusion experiments may yield K_d values representing an unwanted mixture of oxidation states. This argument is supported by the observation that the apparent diffusivities of quadrivalent ions measured in compacted bentonite is chemically inconsistent: Tc(IV) and U(IV) diffuse unexpectedly faster than Th(IV), Np(IV) and Zr(IV).

Cations, typically sorbing by an ion exchange mechanism, appear to diffuse faster through compacted bentonite than their K_d values, measured in dilute systems, would predict. Interpretation of this phenomenon by "surface diffusion" (see Yu & Neretnieks 1997, Ochs et al. 1998) is, however, not uniformly supported by experimental evidence, some of which suggests the effect is negligible when the porewater composition is carefully adapted to the in-situ conditions of compacted bentonite (Bradbury & Baeyens 2002).

In the case of Sn, for which data from diffusion experiments are not available, batch sorption experiments give K_d values that are much higher than those used in other performance assessment studies. The reasons for this are not known.

The above discussion raises the question of the comparative reliability of sorption data derived by the two different approaches. Further data are required to soundly link batch sorption and diffusion experiments. In the present study, the uncertainties are explicitly considered by using correspondingly low K_ds as pessimistic values, which are given along with the reference data in Appendix 2, Tab. A2.6.

For performance assessment calculations the reference sorption values derived from measurements were not truncated at 5 $m^3 kg^{-1}$, in contrast to, e.g. Kristallin-1 (Nagra 1994a), where this was done because of limitations of sensitivity of the measurements, thus actual values well above 10 $m^3 kg^{-1}$ are retained in the present data set (e.g. for actinides). An effective diffusivity of 2×10^{-10} $m^2 s^{-1}$ is assumed for all radioelements except anions, along with a porosity of 0.36. For anions, a value of 3×10^{-12} $m^2 s^{-1}$ is assumed, along with a porosity of 0.05 (Yu & Neretnieks 1997). The diffusion coefficients and porosities are also summarised in Appendix 2, Tab. A2.6. Temperatures in the near field after some thousands of years will decline to about 50 °C; in free water this implies a factor of two increase in diffusivity relative to room temperature. This is effectively incorporated in the pessimistic selection of a D_e value.

Additional transport processes (so-called off-diagonal Onsager processes) are considered to be unlikely to affect radionuclide transport in bentonite (Soler 2001). Thermal diffusion will not be significant because the thermal gradient in bentonite will be negligible by the time canister breaching occurs.

Possible deviations from expected radionuclide transport behaviour in the near field

The possibility that oxidising species produced by radiolysis diffuse out of the canister and into the surrounding bentonite is not completely excluded, but penetration through a significant fraction of the bentonite thickness can be ruled out (see Section 5.3.4.1).

5.4 Evolution of the ILW near field

5.4.1 Radiation-related processes

The total activity in the ILW disposal tunnels as a function of time is shown in Fig. 5.4-1. Detailed inventories of the ILW-1 and ILW-2 waste emplacement tunnels are given in Appendix 2, Tabs. A2.1.6 and A2.1.7. Note that the initial activity in a single SF canister (Fig. 5.3-2) is approximately the same as the activity of the total ILW inventory. The impacts of radiolysis of water on redox processes and H_2 production are expected to be small for most of the ILW, as the activities are relatively low and the largest proportion of the decay energy is expected to be absorbed in solids. One exception may be the compacted fuel hulls and ends containers (a waste type from COGEMA), which contain most of the α-activity of the ILW. In this case, H_2 will be produced in quantities large enough to cause gas accumulation, although the amount produced by metal corrosion is expected to be much greater. In terms of redox effects, H_2O_2 will be produced by α-radiolysis, which could result in locally oxidising conditions, although reducing conditions are expected eventually as a result of corrosion of steel, because the total reduced iron available greatly exceeds the integrated production of oxidants from radiolysis (Wersin et al. 2003).

5.4.2 Temperature evolution of the ILW near field

Heat in the ILW tunnels is produced by cement hydration processes and radioactive decay. By the time of repository closure, thermogenic cement hydration processes will be complete. The release of hydration heat from the mortar, used to close the repository, would occur within a few months. Longer-term heat production will occur as a result of radioactive decay. Together these processes will lead to a temperature increase of a maximum of ~ 12 °C over a period of several decades (Johnson et al. 2002). On average, the decay heat per unit length of an ILW emplacement tunnel is a small fraction of that of the SF emplacement tunnels. The ILW container with the largest decay heat is that containing compacted fuel hulls and ends, with an initial decay heat of about ~ 30 W per metre length of tunnel, compared to an initial value of 190 W per metre length of the smaller diameter SF emplacement tunnel. As a result, temperatures in the long term in the ILW tunnels will remain relatively low (~ 5 to 10 °C above the ambient value of 38 °C), thus temperature effects on chemical and physical processes can be assumed to be negligible. Calculations of the maximum temperature in the ILW near field are summarised by Johnson et al. (2002).

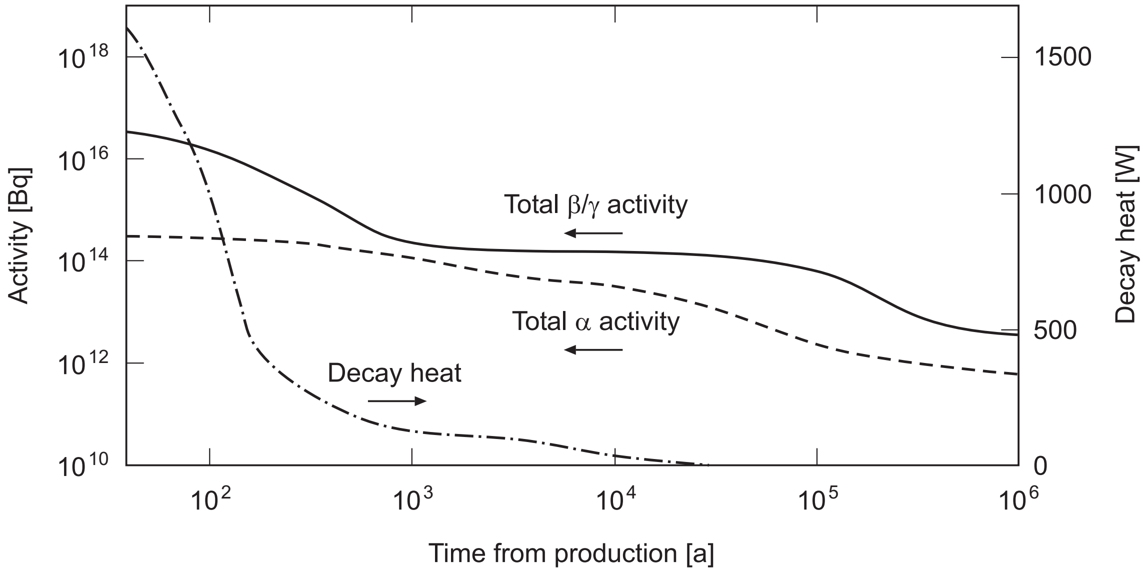

Fig. 5.4-1: Time-dependent total α and total β/γ activity and decay heat of the combined ILW (ILW-1 and ILW-2) inventory for the cemented waste option (McGinnes 2002)

5.4.3 Hydromechanical evolution of the ILW near field

Timescale of near field saturation

Both analytical calculations and coupled thermo-hydraulic calculations incorporating two-phase flow using TOUGH-2 (Nagra 2002a) give estimated saturation times for ILW emplacement tunnels of ~ 500 years for the maximum hydraulic conductivity of Opalinus Clay of 10^{-13} m s^{-1}. There remains some uncertainty regarding the hydraulic conductivity, however, and values as low as 10^{-15} m s^{-1} may also be possible (see Tab. 4.2-4), which would significantly increase the resaturation time.

Gas production

Gases such as H_2, CH_4 and CO_2 will be produced by anaerobic corrosion of metals and by microbial degradation of organic matter in the ILW containers (see Section 4.5.2.3 for materials inventories of ILW). Unlike the case of the HLW/SF tunnels, in which dry bentonite and the slow water inflow rates are likely to delay the initiation of corrosion, in the case of ILW some moisture (porewater in cementitious materials) will be present at the time of tunnel closure, permitting the immediate initiation of corrosion of at least some of the steel waste containers. Aerobic corrosion, with no gas generation, may occur for a short time, but the heterogeneous nature of the environment will probably lead to localised regions where anaerobic corrosion occurs even at early times. After all oxygen is consumed, anaerobic corrosion will proceed until metals and easily degraded organic matter have reacted (Nagra 2003a). The rate of H_2 production is expected to be low, as steels are passivated under alkaline conditions and corrode at a rate of < 10 nm a^{-1} (stainless steels) to 100 nm a^{-1} (mild steels) in cement porewater (Grauer et al. 1991, Kreis 1991, Wada et al. 1999). Breaching of the waste containers (and the existence of vents in some waste container types) will expose other metals and organic materials to water, increasing the gas production rates as more H_2 as well as CH_4 and CO_2 are produced. Small quantities of Al and Zn are expected to corrode rapidly (1 mm a^{-1}) if they have not already done so prior to disposal, whereas fuel cladding and other fuel assembly materials will corrode extremely slowly (< 10 nm a^{-1} (Wada et al. 1999)). Microbially-mediated degradation of organic waste materials will produce approximately equimolar amounts of CH_4 and CO_2, for which a simplified reaction can be written as (Stumm & Morgan 1996):

$$2\{CH_2O\} \rightarrow CH_4 + CO_2$$

where $\{CH_2O\}$ represents an average composition of organic materials. Carbon dioxide would dissolve in the alkaline porewater and precipitate as calcite. The rate of microbial degradation of organic matter depends on the nature of the material, with cellulose more susceptible to rapid degradation, and is uncertain because the environment, in particular the high pH, is considered poor for optimum microbial activity, although it is unlikely to prevent it from occurring. The maximum rates of degradation of organic matter to CH_4 and CO_2 in the ILW tunnels have been estimated based on values given by Wiborgh et al. (1986) and are 0.7 mol per kg of cellulose and 0.05 mol per kg of other organic material.

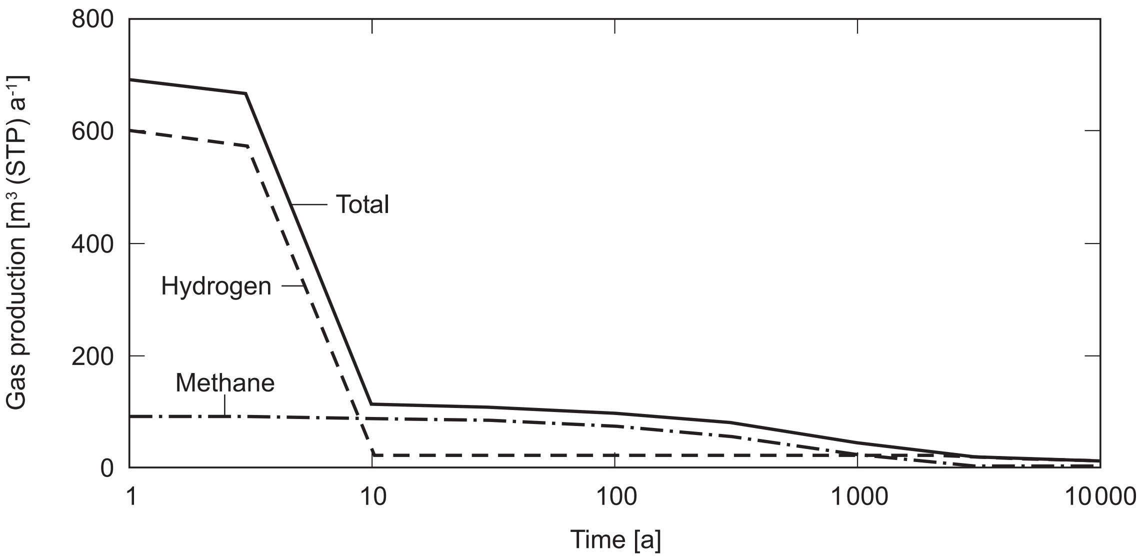

Fig. 5.4-2: The time-dependent total gas production rate (STP) in the ILW emplacement tunnels, based on metal corrosion rates and organic decomposition rates discussed in the text

Total ILW gas production rates, illustrated in Fig. 5.4-2, are in the range of ~ 100 $m^3 a^{-1}$ (STP) after the small quantities of Zn and Al have corroded, gradually decreasing to ~10 $m^3 a^{-1}$ after 10 000 years (Nagra 2003a). Calculations of water inflow rates suggest that, in spite of the low permeability of Opalinus Clay, water supply is unlikely to be the rate-limiting factor for gas production. In the calculations, the assumption is made that all waste materials are exposed to water immediately upon sealing of the emplacement tunnels. The solubilities of H_2 and CH_4 in the porewater would be reached rapidly, thus pressurisation of the ILW emplacement tunnels is likely to occur by gas generation within a time frame of several thousands of years after closure of the tunnels. The mechanisms of release of gas from both ILW and SF / HLW emplacement tunnels are discussed in Section 5.5.2 and further details on gas production rates are given in Nagra (2003a).

Porewater transport in ILW tunnels

The ILW tunnels contain a porous grout backfill with a much higher permeability than the surrounding rock. Nonetheless, because groundwater flow rates are so low, diffusion is still the dominant transport mechanism. The special case of the buildup of gas pressure and its effect on advective transport of dissolved radionuclides from the ILW emplacement tunnels needs to be considered in radionuclide transport calculations (see Section 5.5.2.2 for a discussion of displacement of porewater due to gas buildup).

Convergence of the ILW tunnels

Resaturation of the EDZ and the consequent creep and restoration of the original stress conditions in the Opalinus Clay surrounding the excavations will result in a lithostatic stress of ~ 15 MPa. The tunnel liner is not designed to withstand this stress, which would be transferred to the porous grout backfill. This has a relatively low strength and may fracture, although a significant reduction in porosity (initial porosity ~ 30 %) does not appear possible, because of bridging of the aggregate grains. The grout within and around containers is a more conventional lower porosity (20 %) pumpable grout. These materials would, even if chemically altered or cracked, have the strength to prevent tunnel convergence thus preventing the associated phenomenon of significant porewater expulsion from the ILW tunnels. There is, however, some voidage in waste drums. Assuming that waste drums ultimately collapse after extensive corrosion and that all voidage is eliminated by tunnel convergence, a reduction in void volume could occur equivalent to ~ 0.6 m^3 per m of tunnel length.

Possible deviations from expected hydromechanical behaviour of the near field

The most significant uncertainties in the hydromechanical behaviour of the ILW near field relate to the time frame for saturation with porewater relative to gas production rates and the potential for convergence of the ILW tunnels with concurrent reduction of void volume of degraded waste containers.

Gas build-up may lead to gradual displacement of porewater from the grout. Depending on the extent to which groundwater has leached radionuclides from the waste canisters, this may lead to slow gas-driven flow of porewater containing radionuclides from the near field into the EDZ or surrounding rock. Because the mortar backfill is expected to be mechanically stable (resist compaction), the only porosity reduction that might arise would come from collapse of degraded waste containers after extensive corrosion.

5.4.4 Chemical evolution of the ILW near field

The ILW waste tunnels contain waste solidified mainly with cement grout, surrounded by a relatively high permeability cement-based mortar with a porosity of ~ 25 %. About 75 % of the mass in the ILW tunnels is concrete (~ 80 % aggregate and 20 % cement) with about 17 % steel, present both in the waste and as structural material (Schwyn et al. 2003). Other materials present in smaller quantities include other metals (e.g. Al, Zn and Zircaloy) and organic waste components such as bitumen. These materials, together with the porewater from the host rock, determine the chemical conditions in the near field. Materials such as nitrate, which are present in some of the waste containers, may also influence these conditions. As noted in Section 4.5.2.3, wastes containing significant quantities of organic material and NO_3^- would be located in tunnel ILW-2, to limit their possible detrimental effects on the chemical conditions in the ILW-1 tunnels.

Porewater and mineralogy

Concrete consists of aggregates and cement paste. The latter is a hard gel with a large specific surface area and consists of portlandite ($Ca(OH)_2$) and calcium aluminum silicate hydrates (CSH-, CASH-phases; Taylor 1990; Lea 1970). Freshly set cement contains alkaline hydroxides (NaOH, KOH) that give rise to an initial porewater pH value higher than 13. The alkaline porewater, together with the cement minerals, provide an environment beneficial for the retention of many radionuclides because of low solubilities and strong sorption.

Cement paste is thermodynamically metastable and could therefore be altered by the porewater of the host rock. Initially, the alkali metal hydroxides (NaOH, KOH) are leached (degradation phase I). Experiments and modelling have shown that after about 10 exchanges of the porewater, the pH is reduced from > 13 to 12.5, where it stabilises because of buffering by $Ca(OH)_2$ (degradation phase II). During this second degradation phase, which may prevail for more than 1000 water exchange cycles, portlandite, $Ca(OH)_2$, is dissolved and secondary minerals, mainly calcium carbonate, are formed (Neall 1994, Schwyn et al. 2003). Due to the low hydraulic conductivity of the Opalinus Clay surrounding the repository, near field transport is dominated by diffusion and water exchange is very slow, as the characteristic diffusion time[83] (t_c) of OH^- in Opalinus Clay is in the order of one thousand years at a scale of about one metre.

Dissolved sodium, potassium and calcium hydroxide will diffuse out of the ILW near field while bicarbonate in the Opalinus Clay porewater will diffuse towards the near field. Calcium carbonate is expected to precipitate at the interface of the cement and Opalinus Clay. This process of self-sealing will further slow down transport and cement degradation (Berner 1998, Pfingsten 2001). Phase I of cement degradation (pH > 13) may continue for thousands of years and phase II (pH ca. 12.5) is expected to prevail for hundreds of thousands of years. Even in the case of advective flow through the porous grout, the amount of water flowing through the porous mortar will not be sufficient to fully degrade the cement within the relevant lifetime of the repository.

Effect of high pH plume on Opalinus Clay

Alkaline cement porewater, initially dominated by sodium and potassium hydroxides and later by calcium hydroxide, will diffuse from the cementitious near field and react with the minerals in the host rock (Nagra 2002a). Experiments involving interaction of fresh cement porewater with Opalinus Clay cores suggest that the early high pH plume development involves dolomite and kaolinite dissolution as well as precipitation of calcite, Mg-silicate/aluminate-hydroxide and illite. The reactions involve a net positive molar volume change leading to a porosity reduction at the contact zone with the cement. There are still some unresolved discrepancies with column experiments concerning the secondary minerals. Nevertheless, they indicate that porosity reduction and self-sealing will tend to reduce porosity in the EDZ of ILW emplacement tunnels. The occurrence of such processes in experiments is clear in both experimental and geochemical

[83] $t_c = x^2/D$, where x is the diffusion distance and D is the apparent diffusion coefficient ($\sim 10^{-11}$ m^2 s^{-1}) for OH^-.

modelling studies (Mäder 2003, Pfingsten 2001). Modelling of interaction of alkaline fluids with bentonite also indicate porosity reduction at the contact zone, although some porosity increase is observed further into the clay (Savage et al. 2002).

Although this first phase of pH plume development will last for thousands of years, at longer timescales, the alkaline hydroxides will be exhausted and the porewater diffusing out of the cement will be dominated by calcium hydroxide buffered at a pH of about 12.5. The dissolution and precipitation reactions are expected to be similar to the ones described above for a young cement porewater. The major difference, however, is the probable formation of CSH phases and zeolites as a result of alteration of minerals in the Opalinus Clay. The tendency of the EDZ around the ILW tunnels to seal will remain or even increase (Mäder 2003).

Evidence for the above postulated mineral changes and porosity reduction also comes from the natural analogue study at Maqarin, Jordan (Miller et al. 2000, Linklater 1998). There the high pH plume of a naturally evolved cement zone is in contact with a marl type rock.

The kinetics and details of the alteration reactions occurring in the reaction zone are poorly known, although it is clear that mineral dissolution and precipitation is rapid relative to diffusive transport (Savage et al. 2002). Nonetheless, a limiting case for the spreading of a pH plume in a diffusion-dominated system involves the assumption that all of the cement in the near field will degrade and that the hydroxide released will diffuse into the Opalinus Clay and react with its components. Using such a mass balance basis, a high pH plume would reach about 4 metres into the host rock, consuming about 10 % of the total buffering capacity of the Opalinus Clay components (Mäder 2003). Such conditions can probably only be reached after an extremely long time span (millions of years).

There is evidence from experimental and modelling studies that the reaction front, described above, may be preceded by a disturbed porewater chemistry zone, i.e. a zone with an increased pH in the porewater (between 10 and 11), but without major mineral changes. A rough extrapolation from these studies (Mäder 2003) indicates that such a front cannot proceed further than 10 metres into the rock within one million years.

An interaction of high pH porewater from the ILW waste tunnels with the relatively remote SF / HLW emplacement tunnels does not appear plausible under any conceivable groundwater transport conditions.

It is worth noting that in a diffusion-dominated system, a high pH plume may be to some degree beneficial, in that transport is slowed down due to sealing effects and the radionuclide retention properties of the secondary minerals are at least as good as those of the original minerals (Bradbury & Baeyens 1997 and 2003c). Moreover, an increased pH value leads to stronger sorption for many radionuclides.

Cement-based backfills are unlikely to experience significant carbonation by the carbon dioxide in the ventilation air during the operational phase, because ILW tunnels will be sealed shortly after waste and mortar backfill emplacement. Carbonation, if it occurs, could lead to a somewhat reduced pH and to slightly less favourable radionuclide sorption (Nagra 1998).

Corrosion of waste container materials, wastes and evolution of redox conditions

The different waste types are packed into stainless steel and fibre-cement containers. Not all of these containers have a tight seal, some being fitted with sintered metal vents to prevent pressurisation by gas generated during interim storage. As a result, corrosion of container materials and further corrosion of the wastes themselves may occur relatively soon after emplacement of the waste containers because water vapour will be able to contact the wastes. After closure of the repository, the residual oxygen will gradually be consumed by corrosion processes. Strongly reducing conditions will develop and slow anaerobic corrosion of steel will occur. This corrosion is expected to buffer the redox potential in the near field (Wersin et al. 2003). Together with the corrosion of construction steel and the metals in the waste, the corrosion of the waste packages will be the main contributor to gas (H_2) production (see Section 5.4.3). After all iron is consumed, the redox potential will slowly adjust to the conditions prevailing in the host rock porewater, which are also expected to be reducing and buffered by the pyrite and siderite in the Opalinus Clay.

Because the waste is conditioned, usually with cement and in some cases with bitumen, resistance of the waste form to corrosion is expected, but it is difficult to quantify the effectiveness of this physical barrier for most wastes. For fuel cladding and fuel assembly materials, ^{14}C present in the oxide film may be released relatively rapidly, but the extremely low corrosion rate of the alloys (< 10 nm a^{-1}) would lead to slow release of radionuclides. Nitrate is present in some waste packages, which can react with organic material, e.g. bitumen. Depending on the rate of this oxidation, which may be catalysed by microbes, more oxidising conditions may prevail in the vicinity of such waste packages (ILW-2 tunnel), temporarily overriding the redox buffering by corroding steel (Wersin et al. 2003).

Possible deviations from expected evolution of chemistry, corrosion and redox conditions

One particular waste type, referred to as compacted hulls and ends, contains relatively large quantities of α-emitting radionuclides, which may lead to significant production of radiolytic oxidants. It is possible that the corrosion of the high quality stainless steels may not be sufficiently rapid for the resultant Fe^{2+} to buffer this and that locally oxidising conditions may initially prevail within the vicinity of these waste packages, leading to some oxidation of redox-sensitive radionuclides. In the longer term, the steel corrosion products are expected to consume these radiolytic oxidants.

5.4.5 Radionuclide transport processes in the ILW near field

Although the cementitious near field, in particular the backfill, is porous and permeable, transport will be dominated by diffusion, because of the low permeability of the surrounding Opalinus Clay, which is further supported by the effective backfilling and sealing of access tunnels.

The very slow transport rates of radionuclides will be further decreased by the high retention capacity of the cementitious near field for many radionuclides. The hardened cement takes up these radionuclides by sorption (Wieland & Van Loon 2002) or precipitation (generally forming solid solutions). The high pH also leads to low solubilities for some radionuclides (Berner 2003). Retention capabilities may be reduced by organic complexants such as cellulose degradation products and additives. The cellulose and organic ligand content of the waste, however, is too low to have a significant detrimental effect (Wieland & van Loon 2002). A small part of the waste types contains inorganic ligands like cyanide which may diminish sorption and increase solubility of some radioelements (Schwyn et al. 2003). Potentially oxidising conditions, caused by reactions of nitrate, reduce sorption and increase solubility of some redox-sensitive radioelements. Potentially problematic waste will be disposed of separately (ILW-2) to avoid an influence on other waste types (see Section 4.5.2.3). Radionuclide solubilities and sorption coefficients in cementitious backfill are given in Appendix 2, Tabs. A2.5 and A2.7.

Possible deviations from expected evolution

The convergence process (see Section 5.4.3) may also occur for ILW emplacement tunnels, although it is less likely because of the poor compactibility of the cement-based liner, mortar and emplacement container system, compared to the bentonite backfill. Any drainage of porewater to the access tunnel could accelerate release of radionuclides from the emplacement tunnels.

5.5 Evolution of the far field

5.5.1 Evolution of shaft and tunnel seals

Operations tunnels and the access ramp will be backfilled with 30 % / 70 % bentonite / sand backfill with a hydraulic conductivity of $< 5 \times 10^{-11}$ m s^{-1} (Sitz 2002). As discussed in Section 4.5.3.4, the main seals in the access ramp will incorporate a ~ 40 m long section of blocks of highly compacted bentonite, located between bulkheads. Within the ramp section in which the seal is to be emplaced, the concrete liner and the EDZ will be removed. It is expected that the seal will have a long-term hydraulic conductivity of $< 10^{-13}$ m s^{-1}, approximately the same value as that of undisturbed Opalinus Clay. A similar behaviour is expected for the bentonite seal to be emplaced in the vertical ventilation shaft. The EDZ may retain an enhanced hydraulic conductivity (up to ~ 10 times that of undisturbed Opalinus Clay) if the swelling pressure of the highly compacted bentonite in the seal and the deformation of the EDZ by convergence does not fully reduce its porosity and permeability.

Possible deviations from expected evolution

The limited data on the large-scale permeability of the EDZ in Opalinus Clay and the time-dependent changes in EDZ properties suggests that a highly pessimistic assumption of a hydraulic conductivity of about 1×10^{-10} m s^{-1} be considered as a parameter variation.

5.5.2 Gas migration in the Opalinus Clay

5.5.2.1 Gas transport mechanisms in Opalinus Clay

Several mechanisms can contribute to gas transport in saturated claystone. The relative importance of the mechanisms for any given situation depends on several factors, including the porewater pressure, which determines the solubility of the gas, the confining stresses, the microstructure and texture of the rock, the intrinsic permeability of the rock, and the rate of increase of gas pressure with time. The transport mechanisms are illustrated in Fig. 5.5-1 and include:

- **Advection and diffusion of dissolved gas** (top left in Fig. 5.5-1) – Scoping calculations discussed in Nagra (2003a) illustrate that, at the gas production rates likely to occur in a repository in Opalinus Clay, neither advection nor diffusion are capable of transporting dissolved gas away from emplacement tunnels sufficiently rapidly that a gas phase is prevented from forming.
- **Two-phase flow** (top right in Fig. 5.5-1) – When the gas pressure exceeds the threshold for two-phase flow (pore pressure of 6.5 MPa plus gas entry pressure), leakage of gas into the Opalinus Clay takes place. This process is non-dilatant, i.e. the intrinsic properties of the rock are not affected by the gas transport process. Gas entry pressures range from

~ 2-5 MPa for the EDZ to 5-10 MPa for core samples taken from a depth of 650 m in the Benken borehole (Nagra 2002a). The broad range of entry pressures may relate to the micro-scale heterogeneity of Opalinus Clay (see Fig. 4.2-13).

- **Pathway dilation** (bottom left in Fig. 5.5-1) – If gas pressures increase further because two-phase flow cannot transport gas quickly enough, then microscopic gas pathways may form. This is a dilatant and slow process, in which the gas generation rate is compensated by the creation of new pore space. As discussed in Nagra 2002a, this may occur simultaneously with two-phase flow, as both these transport mechanisms are sensitive to microstructural and textural characteristics of the rock.
- **Formation of macroscopic gas fracs (tensile mode)** (bottom right in Fig. 5.5-1) – Tensile gas fracs are to be expected when the pressure build-up in the rock is rapid, in which case microscopic pathway dilation cannot keep up with the gas production rate. Macroscopic gas fracs require gas pressures in the order of the minimum tensile strength. At Benken, the frac pressure is 17.5 MPa. As will be shown, macroscopic gas fracs cannot occur in the case of gas buildup in the repository, due to the slow increase in gas pressure, the effectiveness of other gas transport mechanisms and the low creep rate involved in pathway dilation.

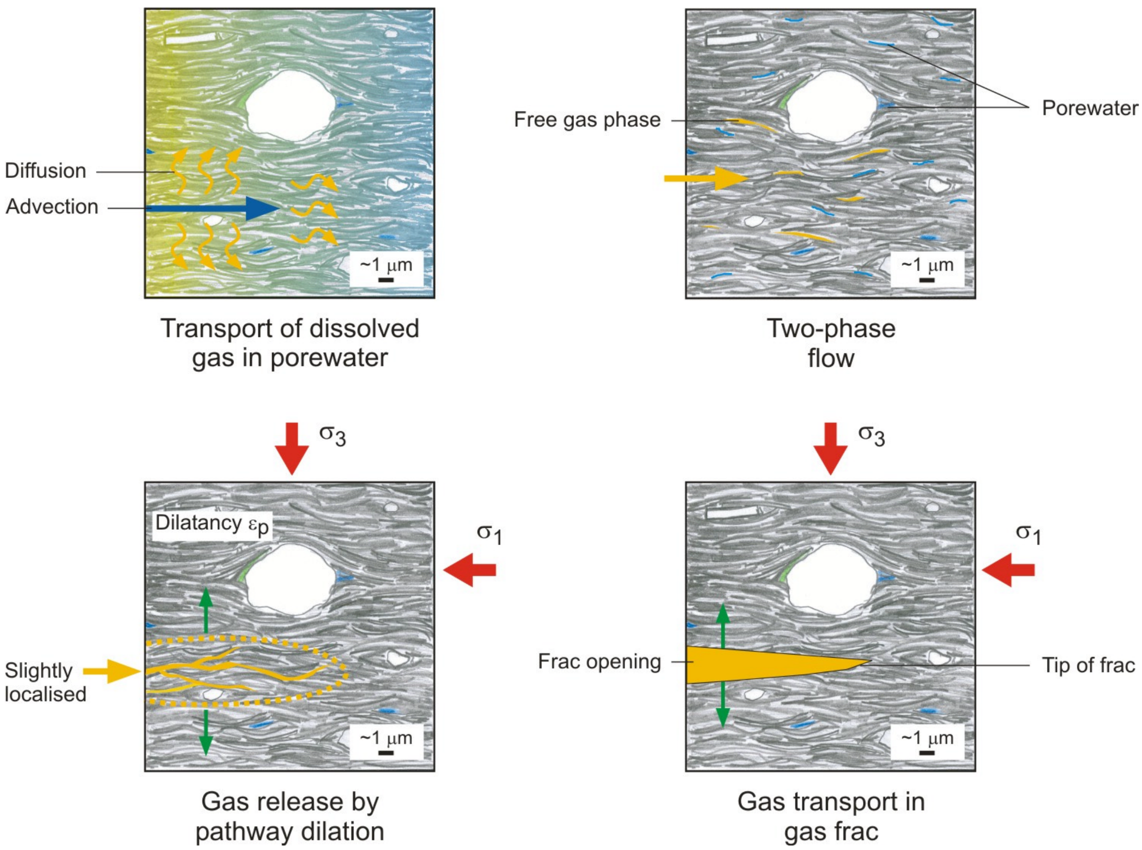

Fig. 5.5-1: Gas transport mechanisms in claystone (Nagra 2002a)

See Fig. 4.2-13 for a detailed discussion of the structure and texture of Opalinus Clay.

5.5.2.2 Gas buildup in and migration from ILW emplacement tunnels

As discussed in Section 5.4.3, significant quantities of gas are produced in the ILW emplacement tunnels as a result of anaerobic corrosion of metals and degradation of organic matter. The following discussion and calculations are based on Nagra (2003a) and Nagra (2002a,c). The total quantity of gases that will be produced by corrosion of metals and degradation of organic matter is ~ 5 x 10^5 m^3 (STP). This will be produced over a period of ~ 100 000 years. The quantity exceeds the storage capacity in the pore space of the cementitious near-field by a factor of approximately two at the ambient pore pressure of 6.5 MPa (assuming the Opalinus Clay is not overpressured, as discussed in Section 4.2.5), thus a pressure rise could occur if there were no effective mechanisms for transport of gas from the emplacement tunnels. Several processes exist that may contribute to limiting the buildup of gas pressures, including:

1. Storage of gas in pore spaces in the mortar and waste packages and dissolution of gas in porewater during the resaturation phase. With a total porosity of ~ 30 % in an ILW tunnel and slow water inflow, this considerably delays gas pressure buildup.
2. Diffusion of dissolved gas into the surrounding rock. This is an ineffective process at the expected gas production rates because of the low diffusivity of dissolved gases in undisturbed Opalinus Clay, as noted above.
3. Displacement of water from partially or fully saturated pores in the mortar as gas pressure increases. This is a very effective mechanism for reducing gas pressure buildup, which initiates when pore pressures in the mortar exceed the formation porewater pressure of 6.5 MPa.
4. Displacement of water from pores in the EDZ or undisturbed Opalinus Clay (two-phase flow). When the gas pressure exceeds the threshold for two-phase flow (pore pressure of 6.5 MPa plus gas entry pressure), leakage of gas into the Opalinus Clay takes place. Leakage is proportional to the product of the effective gas permeability and the hydraulic gradient (Darcy's law). This process may make a significant contribution to gas transport from emplacement tunnels.
5. Formation of microscopic gas pathways (pathway dilation) in the host rock around the plug at the exit of the emplacement tunnel. When gas pathways form through the sealing zone (including the EDZ), gas also escapes to the access tunnel system where significant pore volumes are available for additional gas storage.
6. Formation of microscopic gas pathways (pathway dilation) in the EDZ and the undisturbed rock. Such microscopic pathways are likely to propagate horizontally, because of the low tensile strength of bedding planes, and can form at a pressure of approximately 13 MPa, slightly below the minimum principal stress (~15 MPa at 650 m depth). These pathways would be expected to initiate at the top of the tunnel and grow at a rate governed by the rate of gas production. There is a smooth transition from two-phase flow to pathway dilation. This is an effective mechanism for increasing the volume of rock into which diffusion and two-phase flow can occur.

The buildup of gas pressure with time in the ILW emplacement tunnels, taking into account diffusion and porewater displacement and without consideration of gas leakage past the plug, is shown in Fig. 5.5-2. The threshold pressures for two-phase flow and gas pathway formation in undisturbed Opalinus Clay are shown on the y-axis. Initially, it is assumed that the pore volume in the emplacement tunnels (principally in pore space in the mortar) is 50 % saturated. Porewater displacement and diffusion keep pressures below that of two-phase flow (assumed here to occur at 11.5 MPa) for about 6 000 years. The other curves illustrate the hypothetical pressure buildup in the case of no gas transport and no porewater displacement and gas transport by diffusion only. The calculations further show that significant displacement of water from the porous grout into the Opalinus Clay starts at roughly 10 000 years.

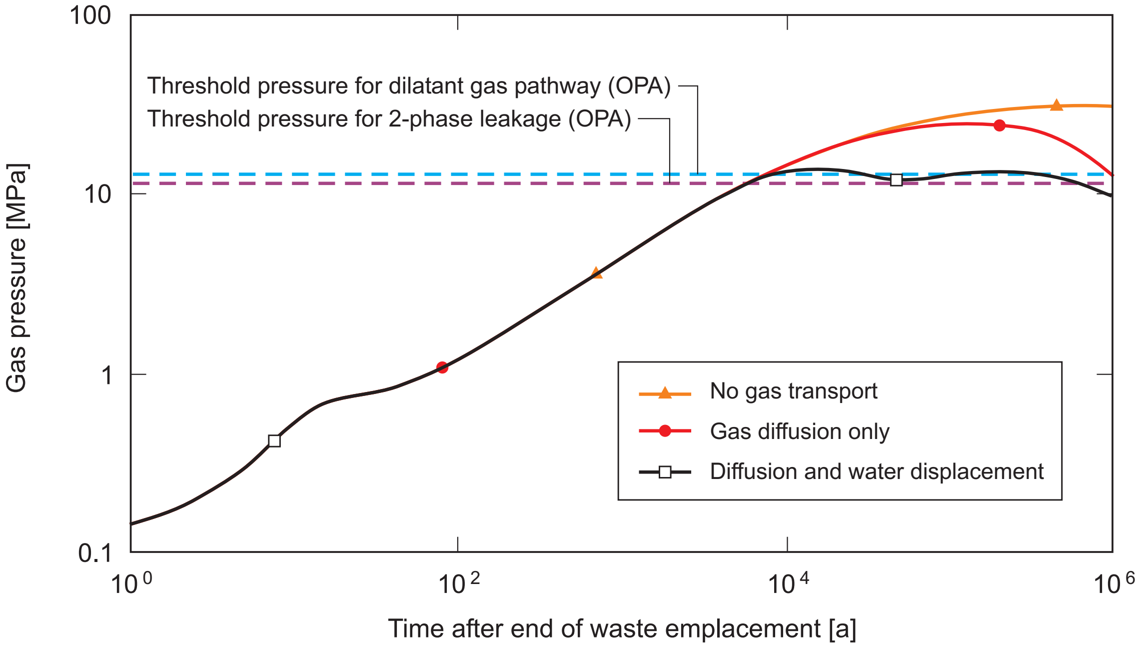

Fig. 5.5-2: Evolution of gas pressure in the ILW emplacement tunnels assuming an initial saturation of 50 %, taking into account gas diffusion / porewater displacement, but without consideration of two-phase flow / pathway dilation (Nagra 2003a)

Calculations that consider the occurrence of diffusion / porewater displacement / two-phase flow leakage and pathway dilation (starting at 13 MPa) show that a combination of these processes can prevent excessive pressure buildup. For an expected gas permeability of Opalinus Clay in the range of 10^{-22} to 10^{-23} m^2, a network of horizontal gas pathways would be expected to form over a few tens of thousands of years, extending a few metres from the waste emplacement tunnels (Nagra 2003a). Such pathways are microscopic and thus do not represent significant storage of gas, but rather provide a mechanism that enhances the surface areas for both vertical and horizontal gas transport due to two-phase flow and diffusion, without significantly affecting the hydraulic transport characteristics of the rock in the vertical direction. The effectiveness of the mechanisms of water displacement from the mortar and of two-phase flow in maintaining low gas pressures suggests that the maximum propagation rate of discrete pathways is about 0.2 cm a^{-1}. Furthermore, this propagation rate cannot be sustained because of the falling gas production rate (see Fig. 5.4-2).

The results support the view that a variety of mechanisms will contribute to gradual dissipation of gas generated in ILW emplacement tunnels. As a result, the formation of macro-scale gas fracs is not considered possible. Similar conclusions have been reached based on gas transport calculations performed using the TOUGH-2 code (Nagra 2002a and Nagra 2003a).

5.5.2.3 Gas buildup in and gas migration from SF / HLW emplacement tunnels

Anaerobic corrosion of SF and HLW canisters is expected to occur at a maximum rate of 1 µm a^{-1}. As discussed in Section 5.3.3.1, the solubility of hydrogen in bentonite porewater will be reached relatively quickly (several hundred years) and gas will be released through the bentonite. No significant gas storage in pore space is possible in highly compacted bentonite, in contrast to the case of mortar in ILW tunnels. The gas production rate is sufficiently high that the low diffusivity of dissolved hydrogen in Opalinus Clay contributes little to preventing gas pressure buildup. Gas buildup is thus expected to lead to pressures that cause two-phase flow and pathway dilation into the Opalinus Clay. For a corrosion rate of 1 $\mu m\ a^{-1}$, the thresholds for two-phase flow and pathway dilation are reached within about 1 000 years. The results suggest that pathway dilation is likely to occur unless the corrosion rate is ~ 0.1 $\mu m\ a^{-1}$ or less.

Gas production rates are thus expected to be sufficiently high that horizontal gas pathways will form in the Opalinus Clay, extending horizontally from the emplacement tunnels. Vertical pathway formation is not expected (except within the EDZ) because (horizontal) bedding planes represent preferred pathways. Calculations of the time-dependent propagation of such pathways suggest that they will extend for tens of metres after several thousands to tens of thousands of years; thus in the regions between tunnels (tunnel separation is 40 m) such microscopic pathways may coalesce. Calculations show that the gas pathway propagation rates are likely to be slow (~ 0.2 $cm\ a^{-1}$), because the rate of pressure increase (due to slow corrosion of steel) is slow. Such pathways will not result in a continuous gas phase, but nonetheless enhance vertical two-phase flow and diffusion into the overlying rock. Eventually some pathways are likely to propagate to the ramp with upwards gas flow along the ramp to the more permeable Wedelsandstein Formation.

It is important to note that the gas production rates for both ILW and SF / HLW emplacement tunnels are very low relative to the gas injection rates used to produce macro-scale gas fracs in the oil and gas industry (Nagra 2002a). Thus the discrete pathways that may arise represent a "creep" rather than a classical fracture process. Once the pathway forms, the pressure is relieved by gas flow, and resealing is expected to occur, until the pressure rises again. There is evidence that pathways in Opalinus Clay reseal after gas breakthrough (Nagra 2002a), and this is also observed for Boom Clay (Volckaert et al. 1995) and bentonite (Horseman et al. 1999, Harrington & Horseman 1999). Both creep and the 1 – 7 % swelling strain of Opalinus Clay (Nagra 2002a) are expected to play a role in closing such gas pathways. The amount of water displaced on pathway formation appears to be so small that it is difficult to measure in laboratory experiments (Ortiz et al. 1997; Rodwell 2000).

Because there are effective mechanisms for dissipating gas from SF, HLW and ILW emplacement tunnels and because gas pathways will not create preferential water flow paths due to resealing, the principal impact of gas production in waste emplacement tunnels is the creation of sustained gas pressure in the near field which may affect water flow in the ramp and shaft and thus the transport of dissolved radionuclides.

5.5.2.4 Possible deviations from expected evolution

The case of the potential formation of a volatile ^{14}C species such as methane (Section 5.3.4.5) and its transport with the hydrogen also needs to be considered. Slow release of gas from the Opalinus Clay into the overlying Wedelsandstein Formation and finally to the Malm aquifer, both through the ramp/shaft system and vertically by two-phase flow and diffusion, would gradually lead to release of this species, although the half-life of 5730 years coupled with the delay and dispersion is likely to substantially reduce the dose impact. The details of such a scenario are discussed in Chapter 7.

Because the number of gas transport experiments performed on Opalinus Clay is rather limited and because there is some uncertainty in gas generation rates, there remain some uncertainties regarding the details of mechanisms and time-dependency of long-term gas release. Due to these uncertainties, studies of gas transport mechanisms in similar rock were reviewed (Nagra 2002a). In the fields of hydrocarbon exploration and natural gas storage, no evidence was found for sudden reservoir depletion due to tensile fracturing. This suggests that there is no possibility of highly conductive macroscale gas pathways being generated and connecting to the overlying formations. Nonetheless, the limited experimental data obtained at a relevant scale suggests that a "what if?" case of rapid gas breakthrough to the overlying rock should be evaluated, even if it is not expected to occur. Such a case is discussed in Chapter 7, in the context of release of ^{14}C in the form of methane.

Evidence for the utilisation of H_2 by methanogenic and sulfate-reducing bacteria has been discussed by Rodwell (2000). This could substantially reduce the volume of gas produced (by a factor of up to four). This may also occur in Opalinus Clay but has not been considered in the present assessment.

Should new information come to light suggesting problems with gas transport from the repository, prospects for modifying the choice of materials used in the repository to reduce gas generation exist in the case of the SF / HLW emplacement tunnels. The use of long-lived canisters with an external shell of Cu (or a highly corrosion-resistant alloy such as titanium) eliminates gas generation for extended periods of time (except in the rare case of prematurely failed canisters). Furthermore, it is feasible to produce an insert from a more corrosion-resistant material (e.g. stainless steel) or an inert material (e.g. ceramic). In the case of ILW emplacement tunnels, the relatively short tunnels contain gas permeable grout, suggesting the possibility of enhancing gas leakage towards the access tunnel, the volume of which is so large that continuous leakage into it would maintain gas pressures in the ILW emplacement tunnels at a low (sub-pathway dilation or frac) level.

5.5.3 Evolution of radionuclide transport conditions in the far field

5.5.3.1 Hydrogeological situation after resaturation of repository

After construction of the repository and emplacement of the wastes the repository will eventually be closed and resaturation will occur. After a few tens of thousand years, a new pseudo-steady state hydrogeological situation will be established (Nagra 2002a). The presence of the repository, including such factors as the temperature transient, gas generation and transport, and convergence of excavations, may influence the hydrogeology within the Opalinus Clay.

This new pseudo-steady state will also have transient features because compaction of Opalinus Clay and dissipation of overpressures will continue if no threshold gradient exists (see Section 4.2.5). The evolution of the overpressures will also be affected by climate change, which may lead to loading and unloading by future glaciations and unloading due to erosion.

As discussed in Section 4.2.5, the observed overpressures indicate that the porewater is nearly stagnant because of the low conductivity (10^{-15} m s^{-1} or lower) and/or a non-Darcian flow regime, which requires that a threshold gradient has to be exceeded before flow starts. The existence of such threshold gradients is consistent with all existing observations but cannot be quantified with any degree of confidence. Conductivities of 10^{-15} m s^{-1} or lower cannot easily be justified because measurements under high gradient conditions indicate higher values.

The inherent uncertainty in hydraulic conductivity estimates and our current inability to introduce threshold gradients is treated in a conservative manner in the hydrodynamic model in Nagra (2002a): in all cases, defensible hydraulic conductivities of 10^{-13} m s^{-1} (horizontal) and 2×10^{-14} m s^{-1} (vertical) are used and threshold gradients are neglected. Overpressures in the Opalinus Clay are considered in some cases by using pessimistic values for water fluxes based on elevated hydraulic gradients.

Expected evolution

After resaturation of the repository, the hydrogeological situation will approach a pseudo-steady state. With the EDZ being self-sealed (see Section 5.3.3.1), the seals functioning as designed and overpressures not being effective (neither in the near field, e.g. due to gas generation or convergence, nor in the Opalinus Clay) the situation is about the same as without the repository. The transport paths go vertically upwards and, after having entered the Wedelsandstein, continue in the horizontal direction. The specific vertical fluxes in the Opalinus Clay are again about 2×10^{-14} m s^{-1}.

The temperatures within the Opalinus Clay close to the SF / HLW emplacement tunnels will increase to ~ 90 °C, while regions tens of metres away will reach ~ 60 °C. These peak temperatures are reached within about 1000 years, and a decline to ambient values will occur in about ten thousand years (Johnson et al. 2002). This increase will create a pressure transient, with pore pressures increasing by about 5 MPa (Nagra 2002a). The pressure transient is dissipated before canister breaching and need not be considered.

Alternative evolution

Due to gas generation or tunnel convergence, overpressures in the near field cannot be excluded. Such a situation can lead to enhanced flow through the Opalinus Clay and may lead to flow through the repository tunnel system and ramp / shaft.

Overpressures in the Opalinus Clay may be effective and future glaciations can also lead to additional overpressures. These overpressures are assumed to dissipate due to drainage (the threshold gradient is assumed not to be effective) which may be partially through the tunnel system and shaft / ramp of the repository and will thus have an effect on flow through the Opalinus Clay and may lead to flow through the repository tunnel system and ramp / shaft.

These different cases have been simulated with a hydrodynamic model. Results are summarised in Tab. 5.5-1, which gives flow rates in the Opalinus Clay and for a location in the operations tunnel immediately adjacent to the Pilot facility on the repository side of the tunnel seal (BTH1, see Fig. 4.5.10). The results are briefly discussed below (see also Nagra 2002a for a detailed discussion of flows in this (BTH1) and other locations).

Overpressures in the near field due to gas generation or tunnel convergence are modelled by an imposed overpressure of 7.5 MPa in the near field which drives both transient and steady-state displacement of mobile porewater (cases RLU1 and RLU2, respectively). This 7.5 MPa overpressure is considered to be an upper limit because at that overpressure gas release due to two-phase flow and/or pathway dilation will be effective, the latter also being effective for porewater release. As an alternative, an overpressure of 4 MPa for a limited duration (20 000 years) has also been considered (case RLU4).

For the cases with overpressures in the near field as compared to the undisturbed case, the flow pathways from some of the locations of the repository are no longer vertically upwards but are partially horizontal; this leads to significantly longer migration path lengths in the Opalinus Clay (e.g. for ILW more than 200 m, see Nagra (2002a)). The fluxes at the top of the Opalinus Clay are more variable but still low at the top of the Opalinus Clay (approx. 2×10^{-13} m s^{-1} or lower). However, for these cases a much more pronounced flow through the tunnels is observed (specific flow rates of approx. 10^{-11} or 10^{-12} m s^{-1} compared to approx. 10^{-13} m s^{-1} for the cases without an overpressured near field).

With overpressures in the Opalinus Clay and drainage of porewater both upwards through the top of the Opalinus Clay as well as through the repository tunnel system (case RLU5), a water divide exists in the Opalinus Clay (i.e. in this case, there are increased water flow rates at the top of Opalinus Clay, but not in the rock immediately surrounding waste emplacement tunnels) and a slight increase in vertical flux at the top of the Opalinus Clay results (approx. 5×10^{-14} m s^{-1}), but with low fluxes through the tunnels.

For some of the above discussed cases the impact of a more permeable backfill for the SF / HLW tunnels as well as for their seals has also been investigated. However, this has only a marginal effect on flow and on the hydrodynamic situation in general.

Tab. 5.5-1: Summary of results of modelling cases examining the transient and steady-state hydraulic flow (values in m s^{-1}) impacts of a repository in Opalinus Clay (Nagra 2002a)

Modelling case (Nagra 2002a)	**Typical flow through Opalinus Clay [m s^{-1}]**	**Typical flow through operations tunnel (BTH1) [m s^{-1}]**	**Key characteristics**
RF8	2×10^{-14}	-	case without repository, no overpressures in Opalinus Clay (head difference between Sandsteinkeuper and Wedelsandstein)
RLU0	2×10^{-14}	6×10^{-13}	no overpressures, seals/EDZ as expected
RLU01	2×10^{-14}	7×10^{-13}	no overpressures, leaky EDZ/seal
RLU1	2×10^{-13}	9×10^{-13}	overpressured NF (7.5 MPa) / steady-state
RLU2	2×10^{-13}	5×10^{-11}	overpressured NF (7.5 MPa) / steady-state, leaky EDZ/seal
RLU4	1×10^{-13}	5×10^{-12}	overpressured NF (4 MPa) / transient
RLU51	3×10^{-16}	7×10^{-13}	overpressures in Opalinus Clay, but $K = 10^{-16}$ m s^{-1}
RLU52	3×10^{-16}	8×10^{-13}	overpressures in Opalinus Clay, but $K = 10^{-16}$ m s^{-1}, leaky EDZ/seal
RLU5	5×10^{-14}	5×10^{-13}	overpressures in Opalinus Clay

5.5.3.2 Radionuclide transport in the Opalinus Clay

Based on evidence from studies of Opalinus Clay in many locations, preferential flow through fractures in deep Opalinus Clay is not expected to occur (see Section 4.2.5). The relative contributions of diffusion and advection/dispersion to the overall radionuclide transport rate in the undisturbed Opalinus Clay are dependent on the hydraulic, mechanical and chemical properties of the Opalinus Clay (averaged hydraulic conductivity, diffusion coefficients, hydraulic and mechanical properties of inhomogeneities, consolidation, osmotic efficiency, etc.) and on the driving forces (concentration gradient, gas pressure buildup, tunnel convergence, compaction, anomalous pressures, Onsager coupled processes etc.).

Expected evolution

It is expected that long-term large-scale radionuclide transport in the host rock will be dominated by diffusion, and that advection/dispersion in the clay matrix will play only a minor role. Due to the low osmotic efficiency (ca. 0.1), the contribution of off-diagonal Onsager processes to radionuclide transport is negligible (Soler 1999). If the overburden is reduced to less than 200 m, fracture flow may begin to contribute to transport (see Section 4.2.5). This, however, would occur only after more than a million years.

Sorption is a key factor controlling radionuclide retardation in the far field. Concerning the scientific understanding of diffusion processes in combination with chemical retardation, the comments made for bentonite in Section 5.3.5 are also valid. Anion exclusion is experimentally observed in Opalinus Clay for Cl^- and I^- and the corresponding diffusivities have been determined in the laboratory (van Loon et al. 2002 and 2003). Diffusion experiments have also been carried out for tritiated water (HTO) (see Section 4.2.5). In addition, studies of the diffusivity of tracers in Opalinus Clay have permitted estimation of the diffusion rates of radionuclides (Nagra 2002a). Diffusion studies in the laboratory and field experiments give consistent values for different spatial and temporal scales (Section 4.2.5). An important parameter affecting diffusion is temperature. The dependence of temperature on diffusion constants can be described by an Arrhenius-type behaviour (Van Loon et al. 2003). Extrapolation of these data to host rock conditions (i.e. $T \approx 40$ °C) leads to expected effective diffusivities of about 10^{-11} m^2 s^{-1} for neutral species and 10^{-12} m^2 s^{-1} for anions. The experimentally determined low D_e values indicate anion exclusion effects. For cations very few diffusion data exist to date. Preliminary data for cations interacting with the clay via ion exchange suggest somewhat higher effective diffusion coefficients for these species (Van Loon et al. 2003). This is in agreement with some literature diffusion data for bentonite but the reason is not yet fully understood (Section 5.3.5).

As for bentonite, sorption values were derived from batch sorption measurements on dilute systems using a method described by Bradbury & Baeyens (2003b). When lab measurements were not available for a given radioelement, data was derived from sorption studies performed on similar clays using chemical analogues. In contrast to the situation for bentonite, virtually no diffusion measurements exist for Opalinus Clay or similar claystones for strongly sorbing radionuclides. In analogy to bentonite (Section 5.3.5), the effective diffusivity of water (HTO) and the total porosity were used to derive apparent diffusivities for all species except for anions, for which a lower effective diffusivity and a smaller accessible porosity were assumed based on values measured for chloride (Van Loon et al. 2002 and 2003).

Reference sorption values, diffusion coefficients and accessible porosities are given in Appendix 2, Tab. A2.8. Their derivation is documented in Bradbury & Baeyens (2003b). The reference sorption values derived from measurements were not truncated at 5 m^3 kg^{-1}, in contrast to, e.g. Kristallin-1 (Nagra 1994a), where this was done because of limitations of sensitivity of the measurements, thus actual values well above 10 $m^3\ kg^{-1}$ are retained in the current data set (e.g. for actinides). The uncertainties in these values are also given in Appendix 2 (Tab. A2.8) and are discussed by Bradbury & Baeyens (2003b).

The possibility that organic complexants naturally present in Opalinus Clay might reduce sorption of radionuclides has been studied by Glaus et al. (2001), who found no impact of organic matter on the sorption of Ni, Eu and Th on Opalinus Clay.

5.5.3.3 Possible deviations from expected transport conditions in the far field

Effect of fracture flow on radionuclide transport in the Opalinus Clay

Fractures may exist in the Opalinus Clay. However, with self-sealing being effective and the existing compressive tectonic regime, no enhanced flow is expected. This is also confirmed by experiments at Mont Terri that showed that artificially created pathways formed in the Opalinus Clay did not lead to significant subsequent changes in permeability (Nagra 2002a). However, due to existing uncertainties some small increase in transmissivity cannot be completely ruled out, leading to some advective transport with matrix diffusion perpendicular to the direction of flow. However, transmissivities are expected to be limited to $< 10^{-10}\ m^2\ s^{-1}$ (Nagra 2002a). For this reason, situations with transmissivities of $10^{-10}\ m^2\ s^{-1}$ and $10^{-9}\ m^2\ s^{-1}$ are introduced only as "what if?" cases.

Zero sorption of ^{129}I on Opalinus Clay

The low sorption coefficient of ^{129}I and its importance in dose calculations suggests that a "what if?" case be considered in which the sorption of ^{129}I on Opalinus Clay (and bentonite) is set to zero, even though a measurable sorption of ^{129}I is consistently determined for both these materials.

Radionuclide solubility limitations in the far field

Radionuclide concentrations in the near field will remain higher than in the far field for prolonged periods of time. Concentrations of radionuclides may reach their solubility limits within and in the vicinity of the failed canisters. Within the Opalinus Clay, however, concentrations of most radionuclides will be far below the solubility limits. Nevertheless, incorporation of many radionuclides into minerals are likely to occur. Since the formation of solid solutions are difficult to quantify at present, such processes are conservatively neglected in safety assessment modelling.

Effect of high pH plume on sorption of radionuclides

In the vicinity of the ILW repository, sorption may be affected by the high pH plume generated by the cementitious backfill in the emplacement tunnels. The rate of high pH front propagation is small. The propagation rate, however, may be faster than the migration of many strongly sorbing radionuclides, but the ultimate extent of propagation will be small (maximum ~ 4 m) due to the limited amount of degradable cement in the ILW repository and the low diffusivity of Opalinus Clay. The principal pH-induced effects include changes in the composition of the porewater and changes in mineralogy (Mäder 2003), which may influence sorption of radionuclides, although retention properties of alteration products are also good (Bradbury & Baeyens 1997 and 2003c).

Flow through the ramp and shaft

The operations and construction tunnel and ramp will be backfilled with a bentonite / sand mixture ($K = 5 \times 10^{-11}$ m s^{-1}) and sealed at various locations with highly compacted bentonite ($K = 10^{-13}$ m s^{-1}) (Section 4.5.3.4). The shaft will also be filled with highly compacted bentonite. This is expected to minimise the significance of the ramp and shaft as preferential transport paths, even in the case of SF / HLW emplacement tunnel convergence due to compaction of the bentonite, which will produce a hydraulic gradient that could drive water flow into the surrounding rock and along the tunnels, ramp and shaft. Nonetheless, seals may be less effective than expected, allowing some flow along the tunnels, ramp and shaft (including the more hydraulically conductive EDZ (see Sections 5.3.3.1 and 5.5.1), which may extend from 1.2 to 3 tunnel radii (Nagra 2002a), depending on the tunnel diameter). Overpressuring of the near field due to gas generation may have a similar effect. The increase in Darcy flow is small in both cases, but is considered in the development of assessment cases.

Effect of redox conditions in the far field

The large redox-buffering capacity of Opalinus Clay and the steel SF / HLW canister material would appear to rule out a redox front penetrating entirely through the bentonite. If this were hypothetically to occur, there would be lower solubility limits for some radionuclides in the Opalinus Clay (reducing conditions) than in the near field (oxidising conditions) for some radionuclides, and the difference would have to be taken into account in safety assessment calculations.

The formation, during the operational phase, of Fe(III) oxides by oxidation of pyrite in the EDZ would increase radionuclide sorption.

Colloid-facilitated transport

Colloids exist in Opalinus Clay porewater but are expected to be virtually immobile, due to the small pore size (1 to 100 nm) (Gaboriau & Seron 2000) and due to charged surfaces of clay platelets (Voegelin & Kretzschmar 2002) and moderate ionic strength of porewater. The absence of transport in fractures and the effective filtering of colloids suggests that colloid transport through Opalinus Clay can be ruled out (Voegelin & Kretzschmar 2002).

5.6 Effects of inadvertent human impacts on repository evolution

As discussed throughout Chapters 4 and 5, the repository location and design concept have been selected with consideration for passive safety under the range of expected conditions, as well as for conditions that might result from unlikely disturbances to the host rock or EBS. Further considerations in site selection, repository design and performance assessment involve the consequences of future human behaviour and their implications for repository integrity. Two main areas must be considered, involving potential inadvertent human intrusion and possible abandonment of a repository during the monitoring phase.

Two types of intrusion can be imagined. The first, intentional human intrusion, is outside the scope of the assessment, as intruders can reasonably be considered to have an understanding of repository contents and, thus, the consequences of intrusion. It is generally agreed that deep geological disposal of radioactive wastes in rock with no economic potential diminishes the likelihood of the second category, unintentional intrusion, but that the probability of its occurrence is a question of speculation and is difficult to estimate reliably. It is possible, nonetheless, to minimise the consequences of such an eventuality by selection of a suitable site and repository design concept and by keeping records of the site and by marking its location. The latter approaches should ensure that no intrusion would occur for, at a minimum, some hundreds of years (or some time at which knowledge of the location is assumed to be lost).

The consequences of unintentional intrusion are reduced by radioactive decay during the time until intrusion might occur, by use of robust engineered barriers and by certain inherent attributes of the host rock. In particular, the use of thick-walled canisters for SF / HLW, the resistance of SF / HLW to dissolution, the compartmentalisation of the system (axial separation (3 m) of canisters along tunnels, with intervening bentonite, and spacing of tunnels 40 m apart), and the low permeability of bentonite and Opalinus Clay can clearly be seen to minimise the quantities of nuclides that could be mobilised should intrusion occur. Several specific factors are important considerations for developing potential intrusion scenarios related to drilling of an exploratory borehole into the repository horizon, including:

- A borehole drilled into Opalinus Clay would have to be supported by a liner or it would collapse because of the plasticity of the rock when saturated. A significant hydraulic driving force in the borehole would exist only if the borehole penetrated to the aquifer below the host rock.
- The very low permeability of the Opalinus Clay and bentonite (in the case of SF / HLW tunnels) would limit the water flux into a borehole (assuming leakage of the borehole casing).
- Steel canisters corrode very slowly, thus are unlikely to be breached by a direct hit until a substantial fraction of the wall thickness is corroded (~ 100 000 years).
- For ILW emplacement tunnels, the connected pore space increases the potential impact of a borehole intrusion, although water fluxes are limited by the low permeability of Opalinus Clay.
- In all cases, solubility limits and sorption on repository materials greatly reduce the pore-water concentrations of most radionuclides.

The possibility of abandonment of the repository is also considered. Such a situation might occur in the event of war or other societal breakdown. Abandonment without sealing of canisters in their emplacement tunnels does not appear realistic, as backfilling is done concurrently with emplacement, and additional backfilling to secure the waste could be rapidly performed. During the potentially much longer monitoring phase, however, the possibility of such an event cannot be completely dismissed. The impact of this unlikely situation is minimised by the design concept, which envisions immediate sealing of all emplacement tunnels upon waste emplacement, as well as sealing of operations tunnels (see Section 4.5.3.4), thus only the observation tunnels of the pilot repository and the ramp are still open.

5.7 Summary of the evolution of the repository system

5.7.1 Summary of independent evidence for the long-term performance of natural and engineered barriers – natural analogue and field studies

In various sections of Chapter 4 and 5, a range of evidence has been presented for the long-term performance of disposal system components. A key part of this evidence involves information from studies of the behaviour of engineered barrier materials and host rock over periods greatly exceeding the time frame of laboratory studies. The results of such natural analogue and related studies (e.g. studies of archaeological artefacts, long-term field investigations, or studies of natural systems) are summarised here.

Opalinus Clay

The suitability of Opalinus Clay as a host rock for radioactive waste disposal is based principally on its low permeability and lack of hydraulically active fractures. Considerable direct and indirect evidence for these characteristics comes from several areas, including:

- **Evidence for the low permeability of Opalinus Clay over geological time** – Isotopic analysis of porewater in Opalinus Clay and adjacent sediments clearly illustrates that for at least the past million years, groundwater and solute transport within the Opalinus Clay has been dominated by diffusion (Fig. 4.2-14). Similar findings are reported by Vitart et al. (1999) for the "couche silteuse" at Marcoule (southern France). Based on the derived advection/diffusion rates, the average travel time for radionuclides that experience no sorption to move from the repository through 40 m of Opalinus Clay are of the order of hundreds of thousands of years (see also the discussion in Section 6.6.2).
- **Evidence for the absence of hydraulically transmissive features in, and evidence for the overall low permeability of Opalinus Clay over large domains of rock** – Studies of a number of road and rail tunnels penetrating more than 6 km of Opalinus Clay, indicate that no water inflows could be detected in tunnel sections passing through the Opalinus Clay where the thickness of the overburden was more than 200 m (see Section 4.2.5, Nagra 2002a). Slight water inflows have been observed in a few cases with less than 200 m overburden and where the Opalinus Clay is highly deformed (in the Folded Jura). In most cases, however, and even in locations of relatively high deformation and faults, there is no observable water inflow to the tunnels. Therefeore, in the potential siting region, with about 600 m of overburden and where the Opalinus Clay is far less deformed, the intersection of repository tunnels with naturally occurring water-conducting features with transmissivities higher than about 10^{-10} m^2 s^{-1} is considered unrealistic (lower transmissivity features would not give rise to detectable inflows on tunnel walls and therefore cannot currently be excluded based on this argument alone). The absence of water-conducting features, even in deformed and faulted Opalinus Clay, provides indirect evidence for an effective self-sealing capability (Gautschi 2001, Nagra 2002a). It is noted that the occurrence of fractured Opalinus Clay in surface deposits in northern Switzerland (e.g. clay quarries) is fully consistent with the above interpretation, as such fracturing is expected to result when off-loading of confining stresses occurs.

Bentonite

Bentonite has been selected as a backfill material for SF / HLW emplacement tunnels and for repository shaft and tunnel seals, principally because of its low hydraulic conductivity, swelling capacity and plasticity. Demonstration of the longevity of bentonite has come from a large number of natural analogue studies (Miller et al. 2000).

Pusch et al. (1998) have shown that the Kinekulle bentonite, which formed 450 million years ago, but was heated to ~ 100 – 150 °C for several thousand years about 300 million years ago, experienced 20 to 40 % conversion to illite, but still retains a plasticity and swelling capacity similar to that of high density bentonite. Some cementation as a result of the formation of silica precipitates occurred in this material, as has been observed in laboratory studies of thermally altered MX-80 bentonite (Pusch et al. 2002). This analogue represents an extreme example, because the K^+ concentration in the porewater, required for illitisation to occur, considerably exceeds that of Opalinus Clay, thus substantially less alteration would be expected for the bentonite surrounding the SF / HLW canisters. Nonetheless, the retention of plasticity and some swelling capacity (and thus self-sealing capability) argues for good performance of bentonite over geological time, despite the occurrence of high temperatures in the inner half of the bentonite for a limited period of time.

A study of illitisation in the Murakami deposit indicates that 40 % alteration to illite occurred over 3 million years for clay exposed to temperatures of 240 °C decreasing to 100 °C and 0 % conversion to illite occurred for temperatures of 160 °C decreasing to 100 °C (Kamai et al. 1992).

Canister corrosion

Carbon steel was selected as a canister material because there is long industrial experience with its fabrication, it has high strength, and it has a relatively low and predictable corrosion rate in anoxic environments. Although native Fe is rarely found in natural environments, cast iron and mild steels have been in use for over two thousand years, thus there is evidence for the rate of corrosion of these materials over a time frame comparable to that of the design lifetime of canisters (Miller et al. 1994).

Johnson & Francis (1980) examined iron and iron-alloy archaeological artefacts and derived average corrosion rates of 0.1 to 10 $\mu m\ a^{-1}$. The highest rates were for materials exposed to oxidising saline environments. The results compare favourably to the rate of 0.1 to 1 $\mu m\ a^{-1}$ for anoxic conditions measured in laboratory studies.

Pitting corrosion occurs only in the case of oxidising conditions, thus is relevant to assessment of performance only for the duration of such an oxidising period (some decades after repository closure). Little relevant information on pitting corrosion comes from studies of archaeological artefacts. However, field studies over decades have provided valuable information. Romanoff (1957) showed in a study of steel samples buried for up to 18 years, that the degree of localisation decreases as corrosion progresses, thus initially large pitting factors (~ 100) decrease to ~ 10 when general corrosion has progressed to a depth of 0.3 mm. Based on these findings, pitting is likely to cause < 1 cm of corrosion during the oxidising phase of the repository (Johnson & King 2003).

Spent fuel

The behaviour of uraninite (UO_2) deposits and their associated radioelements in reducing geological environments provides valuable supporting evidence that containment of nuclides within spent fuel and immobilisation within the near field are important safety functions for a repository in Opalinus Clay. Although a small percentage of the inventory of radionuclides such as ^{129}I, ^{14}C and ^{36}Cl can be expected to be released rapidly from spent fuel and not be retained effectively in the near field, effective isolation of the remaining inventory of radionuclides can be inferred from relevant natural analogue studies.

Uranium deposits such as Cigar Lake (Cramer & Smellie 1994), in which an illitic clay layer surrounding the ore maintains a relatively low hydraulic conductivity (< 10^{-9} m s^{-1}), are illustrative of the retention potential of uraninite. Combined with the reducing capacity of minerals such as siderite and pyrite, which maintain redox conditions within the uraninite stability field, these conditions resulted in effectively complete immobilisation of uranium within the ore zone for 1300 million years. The extent of oxidation of UO_2 to higher oxides has been shown to be very limited (Miller et al. 1994). The behaviour of ^{99}Tc and ^{239}Pu at Cigar Lake, arising from natural fission and neutron capture respectively, has also been examined. The results show that, although small fractions of the Tc and Pu are released from the uraninite, the released material is retained within the deposit (Curtis et al. 1999). The repository for spent fuel in Opalinus Clay will incorporate many features similar to those at Cigar Lake, including clay barriers (bentonite and Opalinus Clay) with a hydraulic conductivity significantly lower than at Cigar Lake (i.e. $< 10^{-9}$ m s^{-1}), as well as large quantities of iron (the canister) and pyrite (~ 1 wt % in Opalinus Clay). These features combine to make a strong case for effective overall retention of radionuclides within spent fuel and the near field.

Extensive studies of the Oklo natural reactor have revealed that more than 90 % of the uranium has remained in the same place since criticality and that there is considerable retention of many radioelements within the reactor core (Blanc 1996). Although there was some uranium dissolution and remobilisation during reactor operation, it should be borne in mind that the temperatures at which this occurred far exceed those expected in a spent fuel repository (i.e. 300 to 400 °C at Oklo as opposed to < 100 °C at the time of canister breaching).

Evidence of a significant impact of α-radiolysis of water on the extent and rate of oxidation of uraninite has not been found. Studies performed at Cigar Lake (discussed in Miller et al. 2000) suggest that high radiolytic yield models (see Johnson & Smith 2000) overpredict the extent of radiolysis-induced oxidation of the uraninite in this deposit. A study by Milodowski et al. (2002), examining reaction fronts of uranium silicates with intergrown sulphides and arsenides found no evidence of oxidation of the latter phases over a period exceeding 170 million years.

Solubilities of radionuclides

Evidence for solubility control of radionuclide concentrations comes from natural analogue studies, including those performed at Cigar Lake, Poços de Caldas, and Maqarin. These indicate low groundwater concentrations of uranium and thorium which have been explained by solubility control (Miller et al. 2000). The nature of the solubility-limiting solid phases in such studies is often poorly known, thus it is often assumed that poorly crystalline pure phase assemblages control solubilities (as is assumed in the present study). In many cases this assumption is found to result in reasonable agreement between calculated and measured concentrations (Bruno et al. 1997). Trace elements may precipitate as solid solutions rather than as pure phases which further decreases dissolved concentrations.

The geochemical conditions of the Cigar Lake uranium ore body and surrounding clay halo are similar to the ones expected for the high-level waste near field in terms of pH and Eh conditions. The concentration levels observed in the Cigar Lake groundwater (Cramer & Smellie 1994) are for U 10^{-9} to 10^{-7} M, for Th $< 10^{-10}$ to 2×10^{-9} M, for Pb $< 4 \times 10^{-9}$ to 10^{-7} M and for Ra 10^{-14} to 5×10^{-13} M.

The fate of trace metals was assessed in hyperalkaline systems, such as Oman and Maqarin (Miller et al. 2000), which are considered as useful analogues for a cementitious repository. In general, very low concentrations (often below detection limit) of uranium, thorium, selenium and other trace metals were encountered. Geochemical modelling assuming solubility control by pure minerals generally overpredicted concentration levels.

5.7.2 Summary of expected disposal system evolution and possible deviations

The key features and phenomena relevant to the safety of the disposal system are summarised in Tab. 5.7-1, along with summary statements regarding the expected evolution and uncertainties and possible deviations. Three categories of phenomena/evolutions can be defined:

1. those relating to the normal evolution and deviations of the reference disposal system;

2. those related to behaviour of system design alternatives; and
3. those related to speculative phenomena (involving human behaviour or the evolution of the surface environment).

The phenomena and evolutions provide the basis for the development of assessment cases described in Chapters 6 and 7.

The range of influence of the possible deviations are summarised in Fig. 5.7-1. The expected evolution scenario as outlined in the present chapter is represented by the green box across the top of the diagram. In this scenario, the clay barrier (i.e. the bentonite and Opalinus Clay together) is largely undisturbed by the phenomena occurring, and provides effective isolation of the wastes by limiting mass transport of radionuclides over very long periods of time.

The possible deviations from the expected evolution for all significant phenomena that influence repository performance are listed in the column on the left. For each phenomenon, the bar represents the domain over which there may be an influence. For example, alternative climates influence only the biosphere, thus the bar extends through this domain only. The colour codings represent transport through the host rock (black) and the tunnel and seal system (gray), with red bars representing "what if?" cases.

5.8 How the selected disposal system meets the objectives of providing security and long-term safety

The reference disposal system is sited and designed in accordance with the objectives outlined in Section 2.6. The broad objectives are to ensure that the disposal system provides safety and robustness, as well as reduced likelihood and consequences of potential human intrusion. Such a system must perform several key safety functions including isolation of waste from the environment, long-term confinement of radionuclides, allowing many isotopes to decay to insignificance, and attenuation of releases to the environment in the far future. The background required to confirm the satisfactory performance of these functions over the long term is given in the preceding sections. The performance analysis is carried out in Chapter 7. The principles and objectives related to siting and design are reiterated below, with reference to the attributes of the proposed disposal system described in the previous sections.

5.8.1 Overall disposal principles

- **Passive safety and security** – Disposal in Opalinus Clay should provide long-term passive safety as well as security of fissile material. Long-term passive safety is ensured by the detailed objectives described below. The security principles as outlined in BfE (2001) would be applied in developing the design of the surface and underground facilities. The handling of SF in the surface facilities and repository would involve application of full safeguards implemented according to the IAEA standards (IAEA 1998). The placement of the wastes deep underground ensures their isolation from threats such as terrorism or war. The security of the wastes during waste emplacement and after repository closure can be monitored, e.g. by advanced geophysical techniques, to detect any attempts to divert sensitive nuclear material (e.g. Seidel et al. 2001).

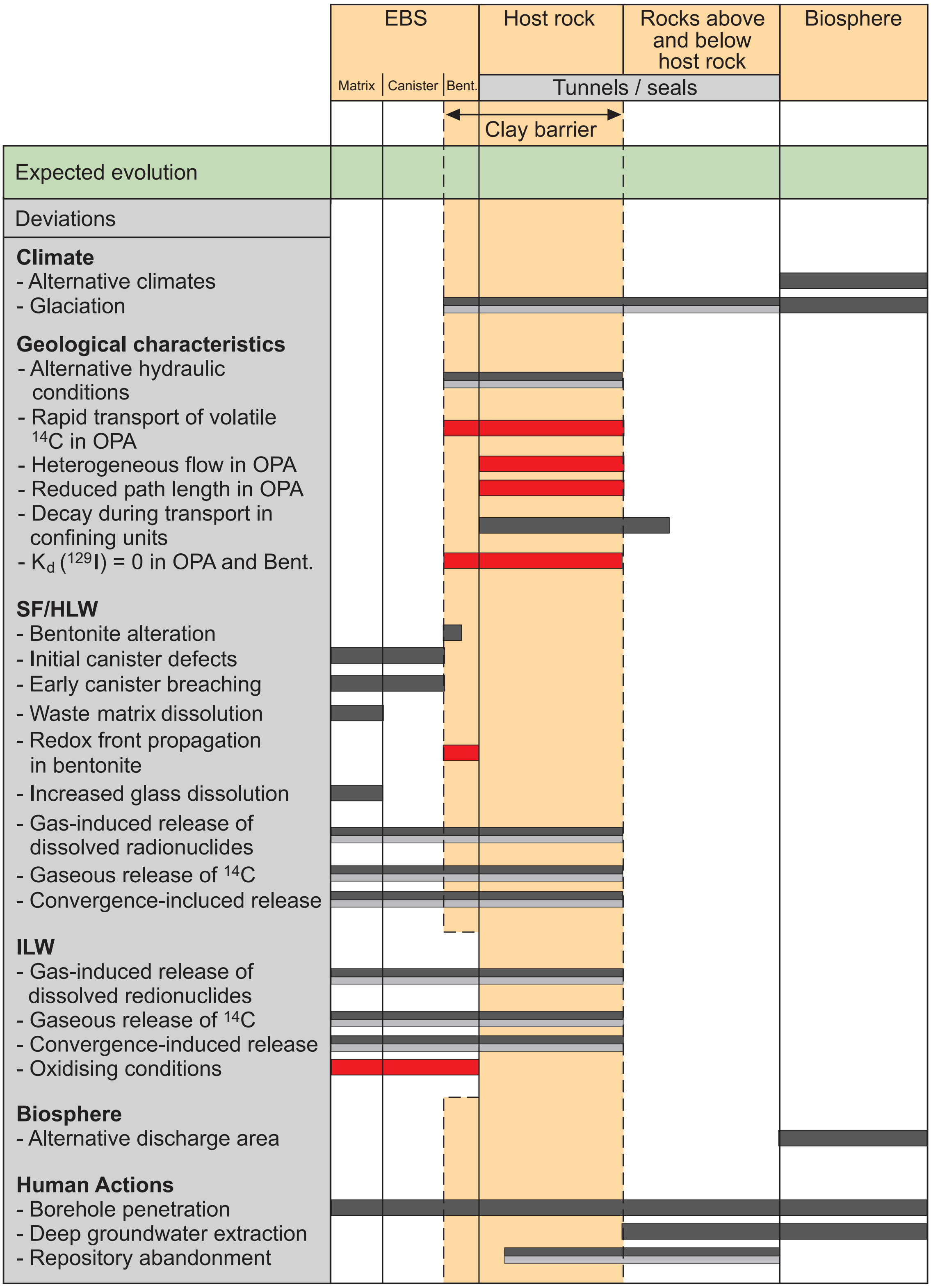

Fig. 5.7-1: Range of influence of possible deviations from the expected evolution of the disposal system

Black bars – transport through host rock, gray bars – transport through tunnels/ramp/shaft and seal system, red bars – "what if?" cases.

Tab. 5.7-1: Key safety-relevant features and phenomena associated with disposal system evolution

SAFETY-RELEVANT FEATURES, PHENOMENA & EVOLUTIONS			
System component	**Key phenomena**	**Expected evolution**	**Uncertainties and possible deviations**
Repository	Repository layout, total waste inventory and design	Reference design parameters, waste inventory for 60 years power plant operation (current power plants only)	Alternative layout for larger waste inventory (300 GWa(e) power production)
	EBS for SF/HLW/ILW - design parameters	Reference design parameters	See design alternatives
Spent Fuel	Radioactive inventories and decay processes (heat and radiation)	Inventories of radionuclides, half lives and decay heat well defined; three canister inventories defined (BWR UO_2, PWR UO_2 and PWR MOX/UO_2) – average burnup 48 GWd/t_{IHM}	Different canister types incorporate IRF variability and variability in burnups, possible increase in burnup up to 75 GWd/t_{IHM}
	Cladding failure and corrosion	Possible short-term failure (cracking), partially failed cladding slows release; slow corrosion-rate controlled release of activation products from Zircaloy, preferential release of some organic ^{14}C from oxide film of cladding	
	Leaching of IRF	IRF dissolves upon canister breaching	Limited information on the IRF of MOX and high burnup UO_2 fuel
	Dissolution of SF matrix	Extremely slow dissolution rate (solubility-controlled) because of reducing capacity of H_2	Oxidising conditions at fuel surface from α-radiolysis (dissolution rate decreases proportionally with α-activity). Preferential release of organic ^{14}C from fuel matrix. "What if?" case assuming high oxidant yield.
	Criticality	Avoided by design basis (geometry, burnup credit)	
HLW Glass	Radioactive inventories and decay processes (heat and radiation)	Inventories of radionuclides, half-lives and decay heat well defined; two HLW glass types defined (BNFL and COGEMA)	
	Dissolution of glass	Different rates for each glass type based on laboratory experiments	Long-term extrapolation uncertain – rates 100 × higher or 20 × lower

Tab. 5.7-1: (Cont.)

SAFETY-RELEVANT FEATURES, PHENOMENA & EVOLUTIONS			
System component	**Key phenomena**	**Expected evolution**	**Uncertainties and possible deviations**
Steel SF/HLW canisters	Failure mechanisms	Brief oxic corrosion phase, limited pitting and microbial corrosion, slow anaerobic corrosion, no SCC – 10 000 a lifetime	Unlikely shorter-term failure (e.g. uncertainties regarding SCC) leading to loss of containment in 1000 a. Residual transport resistance not considered.
	Gas generation	Anaerobic corrosion producing H_2 gas continuously at a low rate	Uncertainty in anaerobic corrosion rate, possible gas-induced porewater displacement (from void space in SF canisters)
	Corrosion products – sorption and redox effects	Sorption of radionuclides onto corrosion products (difficult to quantify); corrosion contributes to reducing conditions	
	Corrosion products – volume expansion effects	Small effect on canister load prior to 10 000 a canister lifetime	
SF/HLW	Precipitation of radionuclides	Solubility limits in the near field (canister/bentonite porewater) for many radionuclides, reducing conditions, co-precipitation of Ra with Ba and Sr	Uncertainties in solubilities. Solubilities also defined for "what if?" case of oxidising conditions in near field
Bentonite	Resaturation	Slow resaturation, delaying start of corrosion and radionuclide release	Time to resaturation is uncertain (~ 100 to 100's of years)
	Radionuclide transport	Diffusion only; good sorption; chemical stability (large amount of material, low water flow rate at outer boundary, mineralogical similarity to Opalinus Clay); colloid filtration	Uncertainty in diffusivity and sorption on bentonite
	Thermal alteration	Degradation mitigated by mixing of UO_2/MOX fuel assemblies to keep temperatures down; little impact on swelling or plasticity	Some cementation and increased diffusivity near canister
	Additional transport processes (Onsager)	Impact likely small; driving forces small beyond 1000 a	

Tab. 5.7-1: (Cont.)

SAFETY-RELEVANT FEATURES, PHENOMENA & EVOLUTIONS			
System component	**Key phenomena**	**Expected evolution**	**Uncertainties and possible devi ations**
Bentonite	Redox front penetration (radiolytic oxidation)	Oxidants scavenged by H_2 and Fe	Evidence appears to eliminate redox front penetration, "what if?" case only
	Gas transport	Release to rock by pathway dilation; negligible water expulsion	
	Microbial effects	Lack of viability of microbes in dense bentonite, no significant effects	
Bentonite/ EDZ	Enhanced hydraulic flow along EDZ	Approximately 10 times higher long-term hydraulic conductivity, lack of connectivity of fractures. Connected EDZ limited by bentonite seals at emplacement tunnel ends.	Hypothetical EDZ hydraulic conductivity of 10^{-10} m s^{-1}
	Gas transport	Pathway dilation along EDZ and into host rock, capillary leakage into rock. Alteration of groundwater transport properties of bentonite, EDZ not expected.	
	Tunnel convergence and compaction of bentonite	Compaction most likely completed before canister breaching (concurrent with saturation), will be limited due to bentonite swelling	Long-term very slow further compaction
ILW	Radioactive inventories and decay processes (heat and radiation)	Inventories of radionuclides, half-lives and decay heat well defined (cemented waste option)	Inventory option (high force compacted waste)
	Breaching of waste containers; dissolution/ leaching of radionuclides	Potential breaching and radionuclide release begins after > 100 years, as a result of slow resaturation; release rates from cemented waste and bitumen uncertain; very slow release from Zircaloy because of low corrosion rate, with some rapid release from oxide film	Longer-term containment by some steel canisters

Tab. 5.7-1: (Cont.)

SAFETY-RELEVANT FEATURES, PHENOMENA & EVOLUTIONS			
System component	**Key phenomena**	**Expected evolution**	**Uncertainties and possible deviations**
ILW	Precipitation of radionuclides	Solubility limits in the near field (cementitious waste/mortar porewater) for many radionuclides, reducing conditions	Uncertainties in solubilities
	Reduced sorption, increased solubility of radionuclides due to complexants in ILW	Separate wastes with high complexant contents into separate emplacement tunnel (ILW-2)	
	Radionuclide transport in ILW tunnels	Low water flow rate (diffusion-dominated transport), good sorption	Uncertainties in sorption. "What if?" case for redox front formation from compacted hulls radiolysis.
	Gas generation	Corrosion of metals and biodegradation of cellulose and other organics	Uncertainties in rates of gas generation
	Displacement of water by gas	Some porewater displaced from containers and mortar into rock	Water displaced along tunnels/EDZ if seals less effective
	Tunnel convergence and compaction of waste/mortar with water displacement	Effect mitigated by choice of materials (mortar), concurrent with saturation	Porewater displacement due to compaction of void space in waste packages after full resaturation
	Resaturation time	Long resaturation time (hundreds of years)	
Mortar/EDZ interface	High pH plume migration into host rock	Porosity reduction in rock, self-sealing (conservatively neglected), depth of reaction front very limited	4 m maximum disturbance of Opalinus Clay based on mass balance
Tunnels/ramp/ shaft seals	Radionuclide transport by advection in tunnel backfill/EDZ	Effective sealing, no preferential transport because of limited EDZ conductivity, effective seals, relatively low gradient	Less effective seals, preferential transport in operations tunnel backfill, EDZ of tunnels, ramp and shaft (EDZ hydraulic conductivity of up to 10^{-10} m s^{-1})
	Transport of gas containing volatile ^{14}C along tunnels, ramp and shaft	Volatile ^{14}C not expected; if it forms it may move through pathways in host rock and EDZ, seals effective	Possibility that gas containing volatile ^{14}C is transported along tunnels, ramp and shaft – seals less effective

Tab. 5.7-1: (Cont.)

SAFETY-RELEVANT FEATURES, PHENOMENA & EVOLUTIONS			
System component	**Key phenomena**	**Expected evolution**	**Uncertainties and possible deviations**
Tunnel plug/ operations tunnel backfill	Gas buildup and transport in ILW emplacement tunnels	Gas storage potential in operations tunnel backfill not considered	Effects of gas buildup and transport in ILW emplacement tunnels mitigated by axial pathway dilation through EDZ and past tunnel plug
Host rock	Length of vertical transport path in Opalinus Clay above and below emplacement tunnels	Typical path length of ~ 50 m, minimum path length of 40 m	"What if?"case of reduced path length
	Advective flow in Opalinus Clay	Flow rate of ~ 10^{-14} m s^{-1}, thus diffusion-dominated solute transport occurs	± 10 fold uncertainty; gas pressure buildup, tunnel convergence, transient increase of overpressure by glacial load all may influence flow. "What if?" case for higher flows.
	Geochemical retardation of radionuclide transport	Retardation of most radionuclides by sorption processes; effective colloid filtration	Uncertainties in sorption coefficients. "What if?" case for $K_d = 0$ for ^{129}I.
	Gas migration	Diffusion of dissolved gas, 2-phase flow and gas pathway dilation; low production rates and stress conditions favour slow horizontal propagation of 2-phase flow (homogeneous gas flow) and dilatant gas pathways, self-sealing of pathways. Slow concurrent transport of volatile ^{14}C.	Rapid volatile ^{14}C transport along a continuous gas pathway – "what if?" case
	Heterogeneous flow	Host rock has little heterogeneity (layering and no hydraulically active discontinuities). Self-sealing is effective.	Undetected discontinuities (although impact negligible because of self-sealing). Evidence appears to eliminate heterogeneous flow, but "what if?" case is considered for pathways with increased transmissivities.

Tab. 5.7-1: (Cont.)

SAFETY-RELEVANT FEATURES, PHENOMENA & EVOLUTIONS			
System component	**Key phenomena**	**Expected evolution**	**Uncertainties and possible deviations**
Geosphere	Neo-tectonic activity	Choice of site with low neo-tectonic activity (low uplift/erosion rates, low seismic activity, no magmatic activity). No reactivation of discontinuities.	
	Retardation in local and regional aquifers	Not well characterised, therefore conservatively neglected	Retardation in confining units (vertical path) and local aquifers (horizontal path)
	Natural resources	Choice of site with no viable natural resources	

SAFETY-RELEVANT PHENOMENA/EVOLUTIONS RELATED TO DESIGN BASIS AND ALTERNATIVES			
System component	**Key phenomena**	**Assumed reference case**	**Design option assumption and residual uncertainties**
Spent fuel	Definition of inventory for disposal	Current power plants operating for 60 years	Increased inventory of SF (300 GWa (e) scenario)
SF/HLW canisters	Potential canister breaching processes	Steel canisters for SF/HLW – 10 000 a design lifetime	Cu canister > 10^5 a lifetime; 1 in 1000 canisters fails immediately (quality assurance), residual transport resistance (pinhole)
	H_2 gas generation	Steel canisters for SF/HLW – continuous H_2 gas production	Selection of Cu, Ni alloy or Ti as external shell canister material, significant gas generation occurs only for failed canisters. Use of alternative insert with negligible gas production.
Bentonite/ EDZ	Alteration by concrete tunnel liner	Avoided by design basis (no concrete liner required in SF/HLW emplacement tunnels)	SF/HLW tunnels may require thin concrete or polymer liner. Small effects expected.
ILW	Inventory definition	Cemented waste option	High force compacted waste option (radionuclide inventory similar to cemented waste option) → possibility for localised radiolysis

Tab. 5.7-1: (Cont.)

SPECULATIVE PHENOMENA/EVOLUTIONS			
System component	**Key phenomena**	**Preferred assumption**	**Alternative assumptions**
Surface environment	Geomorpho-logical evolution	Eroding river valley section, such as present-day Rhine valley below Rhine Falls, and exfiltration in Quaternary gravel at valley bottom assumed (Reference Biosphere)	River valley section with net sedimentation; wetlands; deep groundwater exfiltration to spring at valley sides
	Climatic evolution	"Icehouse" climate regime; present-day climate assumed in the Reference Biosphere	Alternative climates (wet climate, dry climate, periglacial climate)
	Future human behaviour	Unknown, therefore present-day diet and local production of food-stuff is assumed (Reference Biosphere)	Alternative human behaviour during periglacial climate; deep well in Malm aquifer
Host rock	Inadvertant borehole penetration of repository	Assuming current drilling technology: Borehole sealed or collapses and self-seals, canisters not vulnerable (even if partly corroded)	Supported borehole is unsealed, penetrates repository to aquifer below, leakage of near field porewater into borehole.
Tunnels/ramp/ shaft seals	Repository abandoned without backfilling of ramp	Impacts minimised by design basis. Seals, including sealed operations tunnels, isolate emplacement tunnels containing waste.	

5.8.2 Principles related to both site and design

- **Multiple passive barriers** – The barriers include the solidified waste, the canisters, the backfill of bentonite or cementitious material, the Opalinus Clay and its geological confining units and the sealing system for the tunnels, ramp and shaft and are illustrated for the reference disposal system in Figs. 4.4-2 to 4.4-4.
- **Multiple safety functions provided by the barrier system** – The safety functions are isolation from the environment, long-term confinement of radionuclides and their decay, and attenuation of releases to the environment.
- **Stability and longevity of the barrier system** – The Opalinus Clay is 180 million years old and, at the proposed disposal site, is in a stable tectonic setting. The engineered barriers are constructed from well understood materials such as steel, swelling clays and concrete that are compatible with each other in the selected disposal configuration. Evidence for the good long-term performance of these materials comes from the archaeological and geological record as well as from laboratory, field and modelling studies.

- **Avoidance of and insensitivity to detrimental phenomena** – The quiet geological setting favours the avoidance of detrimental phenomena. The plasticity, fine pore structure and self-sealing properties of Opalinus Clay lead to a low sensitivity to disturbances. This is complemented by the use of bentonite clay for backfill, which provides a material that can withstand elevated temperatures and retain its plasticity and hydraulic sealing capacity.
- **Reduced likelihood and consequences of human intrusion** – Measures would be taken to preserve information regarding the location of the disposal site to prevent future inadvertent human intrusion. The lack of resource conflicts in the siting area reduces the likelihood of such intrusion. The isolation of SF and HLW in massive metal canisters and the compartmentalisation of the repository system (separation of canisters and of waste emplacement tunnels) reduces the potential impact of inadvertent human intrusion.

5.8.3 Principles relevant to repository siting

- **Stability –** The potential site is tectonically stable (e.g. with a low incidence of faulting) and has a low rate of uplift and associated erosion.
- **Favourable host rock properties** – The Opalinus Clay has exceptional hydraulic isolation capacity and geochemical properties that favour good performance of the engineered barriers and effective retardation of migrating radionuclides. Furthermore, experience shows that the host rock has engineering properties suitable for constructing, operating and closing the repository.
- **Explorability** – A good understanding of the Opalinus Clay is favoured by its structural simplicity and amenability to characterisation from the surface by borehole and seismic techniques.
- **Predictability** – The well-understood history of the geological setting and the absence of any geologically active zones in the siting area favour the predictability of future evolution.

5.8.4 Principles relevant to repository design and implementation

- **Confinement and attenuation** – The EBS both isolates the wastes and provides desirable properties when canister integrity is lost, i.e. radionuclides are generally released slowly from the waste forms (SF, HLW) and their concentrations in solution are reduced by geochemical processes.
- **The period of complete containment for SF and HLW** – The canister design ensures complete containment of the radionuclides in SF and HLW for a period of more than a thousand years. This provides safety over the period during which the radiotoxicity of the wastes is highest, and in which transient phenomena such as elevated temperatures around the repository due to radiogenic heating and the resaturation of the repository make radionuclide transport more difficult to predict in detail.
- **Redundancy (insensitivity to detrimental phenomena and uncertainties)** – A considerable degree of conservatism is included in the design of some engineered components. For example, the wall thickness of steel canisters for SF and HLW and the thickness of the bentonite backfill are conservatively selected. The impact of detrimental phenomena and uncertainties is also reduced by the fact that multiple phenomena contribute to the safety functions, and by a degree of redundancy in the design of some engineered components. Furthermore, the materials selected for the EBS are specifically chosen because their long-term behaviour can be reliably predicted.

- **Reliability of implementation and reliance on proven technology and materials** – The site and repository design favour safety in that the engineering technology proposed is largely available and many of the design principles and concepts have been satisfactorily demonstrated.
- **Flexibility of implementation** – Several design variants have been evaluated, both to deal with uncertainties in defining the initial conditions (e.g. the exact quantities of wastes) and in evaluating the long-term performance of the system (e.g. alternative materials to reduce gas generation). Alternative areas and host rocks potentially available for siting a repository are discussed in Chapter 1.
- **Cautious, step-wise approach to implementation, with incorporation of monitoring** – The design of the repository, and the evaluation of safety, is based on the concept of monitored long-term geological disposal (see Section 2.4.4). Monitoring of the pilot facility is possible during and after waste emplacement. Retrievability of the wastes, either during the operational stage or after final closure of the facility is feasible with existing technology and is discussed in Nagra (2002b).
- **Reliability of closure of the repository** – The method of waste emplacement, with each emplacement tunnel backfilled and sealed as soon as waste is emplaced, ensures that the repository can be sealed relatively quickly.

6 The Safety Concept and the Identification of Assessment Cases

6.1 Aims and structure of this chapter

The starting point for this chapter is the description of the expected / likely evolution of the barrier system, together with uncertainties (alternative evolutions) and alternative design / system options, given in Chapters 4 and 5. The aims of the present chapter are:

- to describe in qualitative terms how the barrier system is expected to provide safety, i.e. a description of the safety concept,
- to determine, by quantitative analysis (modelling), the fate of radionuclides for the expected evolution of the barrier system,
- to analyse the sensitivity of the system to various assumptions regarding key phenomena, and
- to identify a number of representative assessment cases by which the effects of key uncertainties and design / system options on system performance discussed in Chapters 4 and 5 can be illustrated.

The sensitivity analyses identify whether there are any sudden or complex changes in performance as parameters and model assumptions are varied, and assist both in the identification of assessment cases, so that they focus mainly on uncertainties to which the system or system components are most sensitive, and in understanding of the outcome of the analyses of assessment cases in Chapter 7. The information from the sensitivity analyses is also used to guide the scientific and design work to be carried out in future stages of repository planning and development (Chapter 8). The role of the present chapter in the making of the safety case is indicated in Fig. 6.1-1.

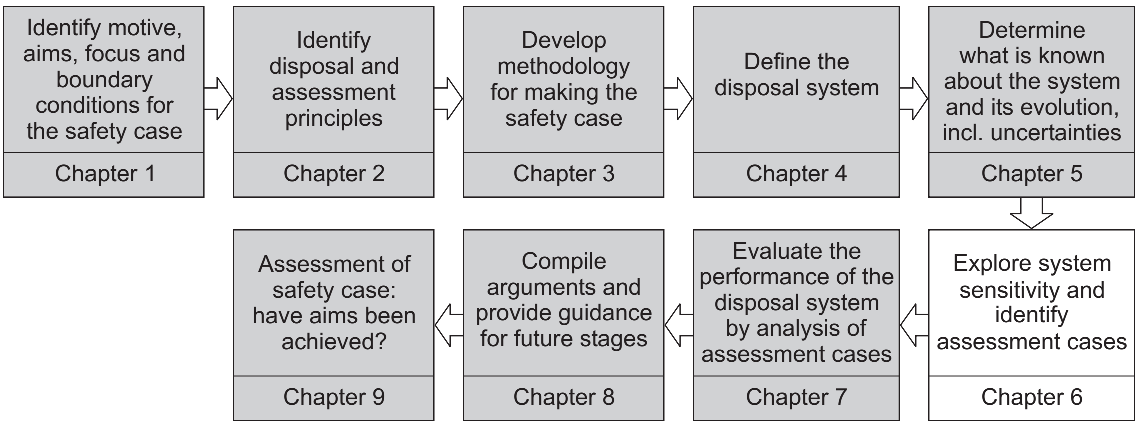

Fig. 6.1-1: The role of the present chapter in the sequence of tasks involved in developing the safety case

Fig. 6.1-2 shows in greater detail how these aims are accomplished. Section 6.2 discusses qualitatively how key features of the barrier system, the "pillars of safety", ensure that the various safety functions defined in Section 2.6.2 are provided. These key features were shown in Chapter 5 to be both well understood and insensitive to various perturbing phenomena that cannot be ruled out. They are thus expected to ensure adequate levels of safety in all reasonably foreseeable circumstances. This is the safety concept for the barrier system. The barrier system does not, however, necessarily preclude some eventual release of radionuclides to the surface environment. Quantitative modelling is thus required to show that the performance of the barrier system is adequate; this is the topic of Section 6.3. The starting point is the expected characteristics and expected broad evolutionary path of the barrier system, which, together with certain model assumptions and parameters, defines a Reference Case. The fate of radionuclides in the Reference Case is examined in depth in Section 6.4, with release rates discussed in Section 6.5 and an examination of the attenuation of release rates by the buffer surrounding the SF and HLW canisters and by the Opalinus Clay discussed in Section 6.6. In Section 6.7, sensitivity analyses, including probabilistic analyses, are discussed that indicate the extent to which the characteristics and evolution of the barrier system can be perturbed before overall performance and the performance of individual system components are significantly affected. The sensitivity study looks also at alternative scenarios. Section 6.8 discusses the treatment of various uncertainties and design / system options in the safety assessment, and identifies specific assessment cases that are described in detail and analysed in Chapter 7. Key messages from this chapter are summarised in Section 6.9.

6.2 How the system provides safety: The Safety Concept

The information presented in Chapters 4 and 5 allows a qualitative description to be made of how the system provides safety.

The barrier system performs a number of functions relevant to long-term security and safety. These safety functions, as defined in Section 2.6.2, are:

(i) *Isolation from the human environment*

The safety and security of the waste, including fissile material, is ensured by placing it in a repository located deep underground, with all access routes backfilled and sealed, thus isolating it from the human environment and reducing the likelihood of any undesirable intrusion and misapplication of the materials. Furthermore, the absence of any currently recognised and economically viable natural resources and the lack of conflict with future infrastructure projects that can be conceived at present reduces the likelihood of inadvertent human intrusion. Finally, appropriate siting ensures that the site is not prone to disruptive events and to processes unfavourable to long-term stability.

(ii) *Long-term confinement and radioactive decay within the disposal system*

Much of the activity initially present decays while the wastes are totally contained within the primary waste containers, particularly in the case of SF and HLW, for which the high integrity steel canisters are expected to remain unbreached for at least 10 000 years. Even after the canisters are breached, the stability of the SF and HLW waste forms in the expected environment, the slowness of groundwater flow and a range of geochemical immobilisation and retardation processes ensure that radionuclides continue to be largely confined within the engineered barrier system and the immediately surrounding rock, so that further radioactive decay takes place.

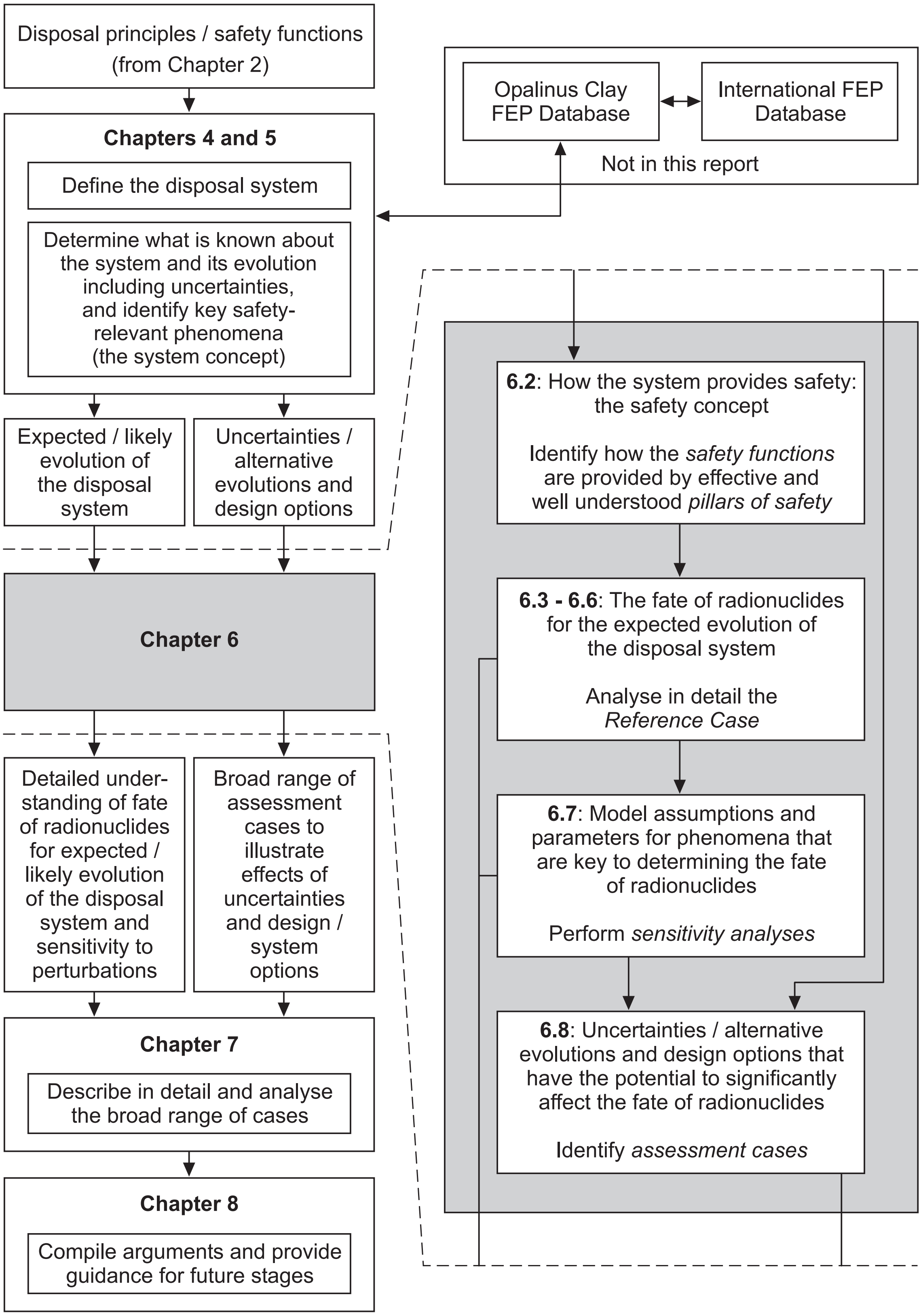

Fig. 6.1-2: The content of the present chapter in the context of developing the safety case

(iii) *Attenuation of releases to the environment*

Although complete confinement cannot be provided over all relevant times for all radionuclides, release rates of radionuclides from the waste forms are low, particularly from the stable SF and HLW waste forms. Furthermore, a number of processes attenuate releases during transport towards the surface environment, and limit the concentrations of radionuclides in that environment. These include radioactive decay during slow transport through the barrier provided by the host rock and the spreading of released radionuclides in time and space by, for example, diffusion, hydrodynamic dispersion and dilution.

Features of the barrier system that are key to providing the safety functions are thus:

- the *deep underground location of the repository*, in a setting that is unlikely to attract human intrusion and is not prone to disruptive geological events and to processes unfavourable to long-term stability,
- the *host rock*, which has a low hydraulic conductivity, a fine, homogeneous pore structure and a self-sealing capacity, thus providing a strong barrier to radionuclide transport and a suitable environment for the engineered barrier system[84],
- a *chemical environment* that provides a range of geochemical immobilisation and retardation processes, favours the long-term stability of the engineered barriers, and is itself stable due to a range of chemical buffering reactions,
- *the bentonite buffer (for SF and HLW)* as a well-defined interface between the canisters and the host rock, with similar properties as the host rock, that ensures that the effects of the presence of the emplacement tunnels and the heat-producing waste on the host rock are minimal, and that provides a strong barrier to radionuclide transport and a suitable environment for the canisters and the waste forms,
- *SF and HLW waste forms* that are stable in the expected environment, and
- *SF and HLW canisters* that are mechanically strong and corrosion resistant in the expected environment and provide absolute containment for a considerable period of time.

Because of the key contributions that these features make to the safety functions, the good level of scientific understanding that is available and their insensitivity to perturbations, as discussed in Chapter 5, they are termed the "pillars of safety" of the barrier system. The pillars of safety ensure that the safety functions provide adequate levels of safety for all realistically conceivable possibilities for the characteristics and evolution of the system. This is the safety concept for the barrier system.

6.3 Modelling the barrier system

6.3.1 The need for modelling

Quantitative modelling is required in order to evaluate the performance of system components, and the performance of the barrier system as a whole. Furthermore, the safety functions that the barrier system provides do not necessarily preclude some eventual release of radionuclides to the surface environment, and quantitative modelling is used to test the adequacy of the levels of safety provided. These models are based on conceptualisations of relevant features of the barrier

[84] The backfilled and sealed access tunnel system is designed to complement the favourable properties of the host rock, isolating the engineered barriers from the surface environment and avoiding the possibility of preferential transport pathways for radionuclides that bypass the host rock.

system and of the events and processes that affect their evolution and the fate of radionuclides. They also take account of processes (particularly dilution) that occur outside the barrier system, including dilution in the surface environment itself.

The starting point is to define and analyse a Reference Case. The Reference Case is obtained by defining a Reference Scenario, conceptualising safety-relevant phenomena (Table 5.7-1) that are relevant in this scenario and assigning a reference set of parameters to the resulting models (Fig. 3.7-3). Note that the Reference Case plays an important role both in Chapters 6 and 7. The topics relevant to its definition are, however, discussed in detail only in Chapter 6 (Sections 6.3.2, 6.3.3 and 6.3.4).

6.3.2 The broad evolution of the barrier system in the Reference Case: The Reference Scenario

In the Reference Scenario, the pillars of safety are assumed to operate broadly as expected (see also Tab. 5.7-1). Briefly:

- *The deep underground location of the repository* is assumed to be maintained over several million years, isolating the waste from the surface environment.
- *The host rock* is assumed to maintain its safety-relevant properties over several million years and these properties are not significantly perturbed by the presence of the repository (e.g. by gas generated within the repository), by geological and climatic events and processes and by any future human activities. The low hydraulic conductivity and the fine, homogeneous pore structure of the host rock, as well as the backfilling and sealing of the access tunnel and shaft, ensure that transport of radionuclides through the near field and host rock is dominated by aqueous diffusion. The sealed access tunnel and shaft are assumed at no time to provide preferential transport pathways, although some limited radionuclide transport along these features may occur.
- *The favourable chemical environment*, which provides a range of geochemical immobilisation and retardation processes, is assumed to be maintained over several million years.
- *The bentonite buffer (for SF and HLW),* which provides a well-defined interface between the canisters and the host rock, with similar properties as the host rock, is assumed to maintain its favourable properties over several million years.
- *The SF and HLW waste forms* continue to retain most radionuclides after canister breaching. Some ILW components, such as hulls and ends from reprocessing, are also stable and will retain radionuclides, although no credit is taken for this in the reference conceptualisation (see below).
- *The SF and HLW canisters* provide an initial period of complete containment (although the possibility of one or more initial canister defects cannot be excluded). For all waste types, the repository and its surroundings are assumed to be fully resaturated by the time porewater comes into contact with the wastes and porewater is assumed to have reached chemical equilibrium.

In addition, any radionuclides released from the host rock migrate through overlying and underlying sediments and are diluted in regional aquifers and in the surface environment.

Fig. 6.3-1 illustrates the approximate timescales over which the principal phenomena provided by the pillars of safety[85] that contribute directly and positively to the safety functions are expected to operate. The figure shows that, at earlier times, complete containment is ensured by the SF and HLW canisters. At later times, a range of physical and chemical processes contribute to continued safety. This is illustrated by the results of numerous calculations throughout the remainder of this report.

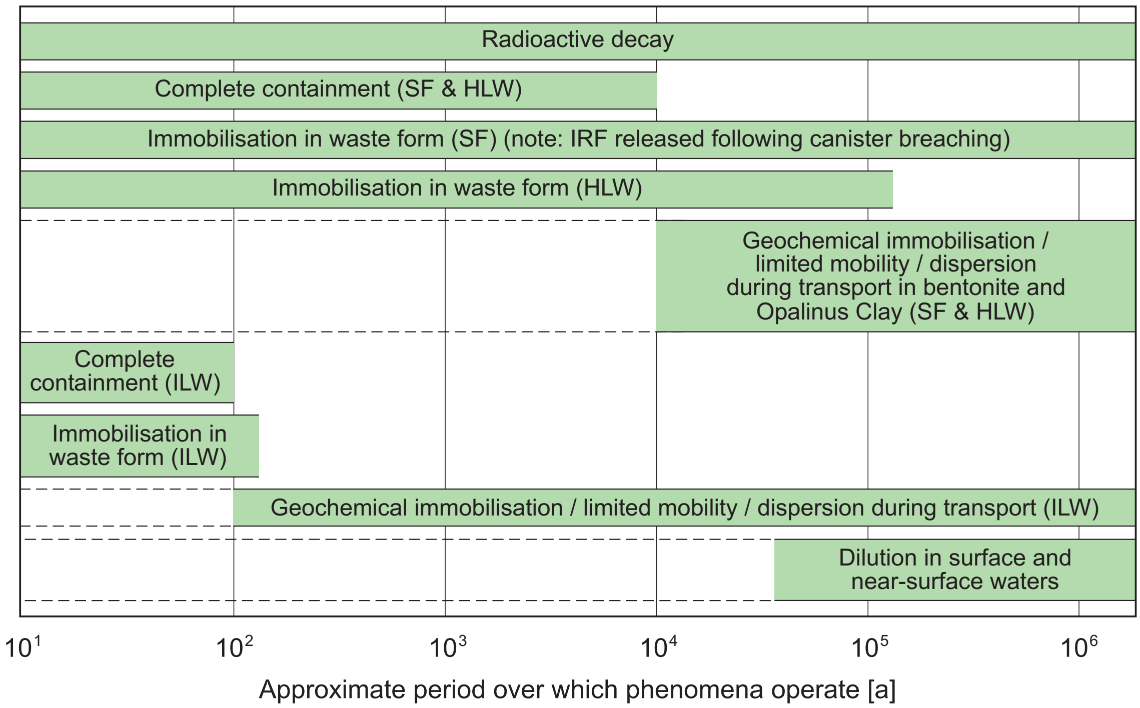

Fig. 6.3-1: The approximate timescales over which various phenomena that contribute positively to the safety functions are expected to operate

Dashed lines indicate that phenomena would operate earlier if canisters were breached sooner than expected or if early releases to the surface environment were to occur.

The following sections describe how key phenomena relevant to the Reference Scenario are conceptualised in order to model the Reference Case. Conceptual assumptions are based on the scientific understanding presented in Chapters 4 and 5 and are, at this stage, mostly stated without further justification. These assumptions are, however, revisited in Section 6.8 in the discussion of uncertainty (see Tab. 6.8-1). It should also be noted that many detailed aspects of the conceptualisations, including geometrical simplifications, are not described in the present report. The reader is referred to the comprehensive descriptions of the models, codes and data given in Nagra (2002c).

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85 Dilution in surface and near-surface waters is also included due to its impact on evaluations of dose, although the surface environment is not a pillar of safety and is not considered to be a part of the barrier system.

6.3.3 Conceptualisation of key phenomena in the Reference Case: The Reference Conceptualisation

The assumptions made and parameter values used in the Reference Case are generally based on the expected evolution of the system components outlined in Tab. 5.7-1, but some pessimistic or conservative conceptual assumptions and parameters are also used (see also Section 6.8.2).

Pilot facility inventory

- The concept of monitored long-term geological disposal (EKRA) involves the disposal of a small but representative amount of the wastes in separate emplacement tunnels (the pilot facility) for monitoring purposes. The radionuclide inventory contained in the pilot facility is not, however, analysed separately, but is added to the model inventories for the main facility[86].

SF

- The instant release fraction (IRF) is released from SF immediately upon canister breaching.
- Other radionuclides in the SF are released congruently with fuel matrix dissolution, at a time-dependent rate proportional to the α activity of the fuel (this is probably less realistic than the alternative assumption of solubility-limited fuel dissolution, and is pessimistic if the proportionality constant is chosen appropriately; see Section 6.8).
- All ^{14}C originating from the SF, including the IRF, is assumed to be in inorganic form. All ^{14}C originating from the Zircaloy cladding, including an IRF from the cladding, is assumed to be in organic form. The overall IRF thus contains a mixture of inorganic and organic ^{14}C.
- For Zircaloy cladding, a 20 % IRF of ^{14}C is assumed. Other radionuclides in the cladding are released congruently with cladding corrosion, which is assumed to occur at a constant rate.

Reference data for SF are given in Tabs. A2.1.1 to A2.1.3 and Tabs. A2.2-1 to A2.2-4 of Appendix 2.

HLW

- Vitrified HLW is solidified in thin stainless steel fabrication flasks, which are then placed in massive carbon steel canisters. The flasks are assigned no barrier function.
- During cooling of the glass, cracks form so that the surface area at the time of emplacement is greater than that of original glass blocks.
- Following canister breaching, radionuclides in vitrified HLW are released congruently with glass dissolution at a constant rate per unit surface area of the waste.
- Glass fragments are conceptualised as a number of equal sized spheres, with a total volume equal to the total volume of glass, and a total surface area that accounts for the cracking of the glass. The surface area of the spheres decreases with time as they dissolve.

Reference data for HLW are given in Tabs. A2.1.4, A2.1.5 and A2.3 of Appendix 2.

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86 This simplification is justified by the argument that in most assessment cases, the major pathway of radionuclide release is by vertical transport through the host rock, and that, therefore, allocating a small amount of waste outside the main facility, while preserving similar near field conditions, has a negligible effect on the overall radionuclide release rates.

Steel SF / HLW canisters

- All SF and HLW canisters are assumed to be breached simultaneously at a reference time of 10 000 years following waste emplacement. No initially defective canisters are present.
- The SF and HLW canisters are conservatively assumed to provide no physical barrier to water ingress or radionuclide release after they are breached.
- No credit is taken for the integrity of the SF Zircaloy cladding (i.e. breaching of the cladding is assumed to occur immediately after canister failure, even though the Zircaloy corrosion rate is low) and the HLW fabrication flasks.

Bentonite (SF / HLW)

- Radionuclides are assumed to be released to a well-mixed volume of water that corresponds to the void space within the canisters and is in contact with the inner boundary of the bentonite.
- The bentonite is assumed to be homogeneous in its transport properties. In particular, the thermally altered zone in the bentonite around the waste packages is assumed to be of negligible extent and any radiolytic oxidants formed near the wastes are assumed not to migrate significantly into the bentonite buffer.
- Reducing conditions are assumed to prevail within and around the repository (except at SF surfaces). Solubility limits shared between isotopes of the same element and appropriate to reducing conditions constrain aqueous radionuclide concentrations. A radionuclide precipitates if the concentration of the corresponding element, summed over all isotopes, exceeds the solubility limit, and redissolution occurs if concentrations fall.
- The element concentrations used to evaluate whether solubility limits are exceeded are obtained by summing the concentrations of all isotopes originating from the waste. The background concentrations of isotopes originating elsewhere are conservatively ignored.
- In the case of radium, since the necessary data are available and because of the potential significance to safety of ^{226}Ra due to the very long half life of its parent ^{238}U, co-precipitation of radium with inactive isotopes of chemically similar elements is taken into account. For all other elements, however, immobilisation by co-precipitation is neglected, as is sorption of radionuclides on the corrosion products of the canisters and waste forms.
- The transport mechanism for solutes in the bentonite buffer is aqueous diffusion, described by Fick's law and retarded by linear, equilibrium sorption, described by an element-dependent sorption coefficient (K_d).
- Transport of radionuclides dissolved in water mediated by gas and/or tunnel convergence is considered to be negligible.
- Any radionuclide-bearing colloids are assumed to be immobile in the bentonite and are not considered.

Safety-relevant parameters for the bentonite near field for SF and HLW are given in Tabs. A2.4 and A2.6 of Appendix 2.

ILW

- For ILW, release of radionuclides is assumed not to begin until a reference time of 100 years after waste emplacement due to incomplete resaturation at earlier times and due to immobilisation in the waste form.
- For times beyond 100 years, immobilisation of radionuclides in the waste form (e.g. activation products in hulls and ends from reprocessing) is conservatively neglected.
- By the time release begins, radionuclides in ILW are assumed to have migrated to the surrounding cementitious backfill, where they are uniformly mixed with porewater and partitioned between aqueous, sorbed and precipitated phases.
- Linear, equilibrium sorption is assumed, described by an element-dependent sorption coefficient (K_d).
- Solubility limits constrain radionuclide concentrations, with precipitation occurring if the solubility limits of the corresponding elements, summed over all isotopes, are exceeded, and redissolution occurring if concentrations fall.
- Radionuclide transport mediated by gas and/or tunnel convergence is considered to be negligible.

Safety-relevant parameters for the cementitious ILW near field are given in Tabs. A2.1.6, A2.1.7, A2.5 and A2.7 of Appendix 2.

Host rock

- The host rock consists of the Opalinus Clay and the Murchisonae Beds in Opalinus Clay facies (although, in later sections of this chapter, the term "Opalinus Clay" is taken to include the Murchisonae Beds and is used in preference to the term "host rock"). The transport barriers provided by the confining units and the regional aquifers are conservatively neglected.
- The host rock is assumed to be homogeneous in its transport properties, with no discontinuities with significant transmissivities.
- Radionuclides are transported by diffusion described by Fick's law, and (very slow) advection described by Darcy's law.
- Both transport processes are retarded by linear, equilibrium sorption, described by an element-dependent sorption coefficient (K_d).
- Advection is driven by the currently observed pressure difference between lower and upper confining units. Glacial cycling and the currently observed overpressures within the host rock are assumed to have negligible effects on advective transport.
- The impact of gas on transport is assumed to be insignificant.
- Colloids are assumed to have no impact on radionuclide transport, on account of the fine pore structure of the host rock.
- Uplift and erosion are assumed to have negligible effects on the hydraulic properties of the host rock over the time period of interest.

Safety-relevant parameters for the host rock are given in Tabs. A2.8 and A2.9 of Appendix 2.

Tunnels / ramp / shaft

- It is assumed that no radionuclides are transported along the tunnels, ramp and shaft.

Overlying and underlying sedimentary layers (confining units)

- Radionuclides that are released from upper and lower boundaries of the Opalinus Clay are assumed to be transferred instantaneously to the biosphere.
- Transport times through the sedimentary layers overlying and underlying the host rock, including the confining units that in reality are expected to contribute significantly to retention, are conservatively neglected.
- Dilution in deep regional aquifers is small compared to that assumed to occur in shallow aquifers and in the surface environment, and is neglected.

The surface environment

The model of the surface environment used in the present chapter and in Chapter 7 is described in detail in Nagra (2003b). Key assumptions are:

- The surface environment consists of five main compartments in which instant mixing of radionuclides is assumed to occur. These compartments are the Quaternary aquifer, a deep soil layer, a top soil layer, surface water and aquatic sediment.
- Transport of radionuclides in the surface environment occurs as a result of movements of water (radionuclides in solution) and movements of solid materials (radionuclides sorbed onto the solid phase).
- The characteristics of the surface environment, at the time when future radionuclide exfiltration is expected to occur, are derived using the present-day Rhine valley just below the Rhine Falls as the reference model area. The investigated model area has a length of 3.5 km along the river Rhine and a width of 0.65 km. Exfiltration of deep groundwaters, possibly conveying dissolved radionuclides, is assumed to occur in the Quaternary aquifer.
- Present-day geomorphological, climatic and hydrological conditions are taken as the basis for the modelling of future biosphere conditions.
- Human inhabitants of the region obtain all their dietary requirements from local sources. A present-day diet is considered, i.e. vegetables, grain, fruit, milk, meat, eggs and fish are consumed. Drinking water for humans is taken from wells in the Quaternary aquifer.

As applied in Chapter 7, the biosphere compartment model is transient, in that it explicitly takes account of the timescales of various phenomena resulting in the transfer of radionuclides between compartments. In the present chapter, however, it is assumed that these timescales are short in comparison to the timescales over which releases from the Opalinus Clay vary significantly. This allows simple *biosphere dose conversion factors* (BDCFs) to be derived, which are multiplied by releases from the Opalinus Clay in order to calculate doses. The simplified approach is justified because the emphasis of the present chapter is on the behaviour of the barrier system, which is not considered to include the surface environment.

BDCFs are obtained by the following procedure. For each radionuclide j, a constant flux of 1 Bq a^{-1} is input into the transient compartment model, with the Reference Case conceptual assumptions and parameters, and the model is run until a steady state is reached. The BDCF for radionuclide j is the ratio of the corresponding steady-state annual dose [mSv a^{-1}], summed over all daughter radionuclides, to the input value of 1 Bq a^{-1}. Doses calculated by multiplying the release from the Opalinus Clay of a particular radionuclide by its BDCF thus include the contributions of all its daughters. Note that radionuclides released from Opalinus Clay that are members of decay chains are treated separately; i.e. they may appear in addition to their precursors in the figures shown in the present chapter. Reference-case BDCFs are given in Tab. A2.11 of Appendix 2.

6.3.4 Mathematical models, computer codes and the timescales over which they are applied

The conceptualisation of the key phenomena described above forms the basis of a set of governing equations, with accompanying initial conditions and boundary conditions, that are solved using the computer codes STMAN for the near field and PICNIC for the host rock, together with biosphere dose conversion factors (BDCFs) for the surface environment calculated with the code TAME. The equations, initial conditions, boundary conditions, computer codes and BDCFs are described in Nagra (2002c).

The reference model chain of STMAN-PICNIC is used to model the radionuclide release and migration of a range of radionuclides that are judged to be safety relevant, and the doses to which these give rise. The selection of safety-relevant radionuclides is explained in Nagra (2002c).

After more than a few million years the assumptions underlying the conceptualisation of some key phenomena may no longer hold. In spite of this, in the present chapter the results of STMAN-PICNIC calculations are evaluated over a period up to 100 million years, in order to illustrate the behaviour of the model system (rather than the actual system) at these distant future times. To emphasise that the results at these late times should be interpreted with caution, a shading scheme is used in graphs that present times that are longer than a million years. In Chapter 7, a cut-off time of 10^7 years is applied to all calculational results.

6.4 The fate of radionuclides within the barrier system in the Reference Case

6.4.1 The decrease in radiotoxicity with time

Radionuclides both decay, and, in some cases, are created by ingrowth. The half lives of different radionuclides differ widely. As a result, their relative importance in terms of radiotoxicity varies with time. Overall, however, the radiotoxicity of the wastes, expressed in terms of the radiotoxicity index (RTI) as defined in Appendix 3, decreases with time. This is illustrated in Fig. 6.4-1, which also shows the contributions of different radionuclides[87] to the RTI. In this figure, the sum of the RTI for the non-safety-relevant radionuclides is also shown (dotted lines). It can be seen that for both SF and HLW, the non-safety-relevant nuclides have largely decayed well before the time of canister failure (10^4 a), while for ILW they are below the lower bound of the figure in the chosen representation. Only safety-relevant nuclides are considered in the remainder of this report.

[87] The selection criteria for plotting an individual radionuclide are: (i) The nuclide has to give rise to a dose exceeding 10^{-9} mSv a^{-1} in the Reference Case (Chapter 7); or (ii) the nuclide has to exceed 10 % of the sum total in at least one time interval in the plot under consideration.

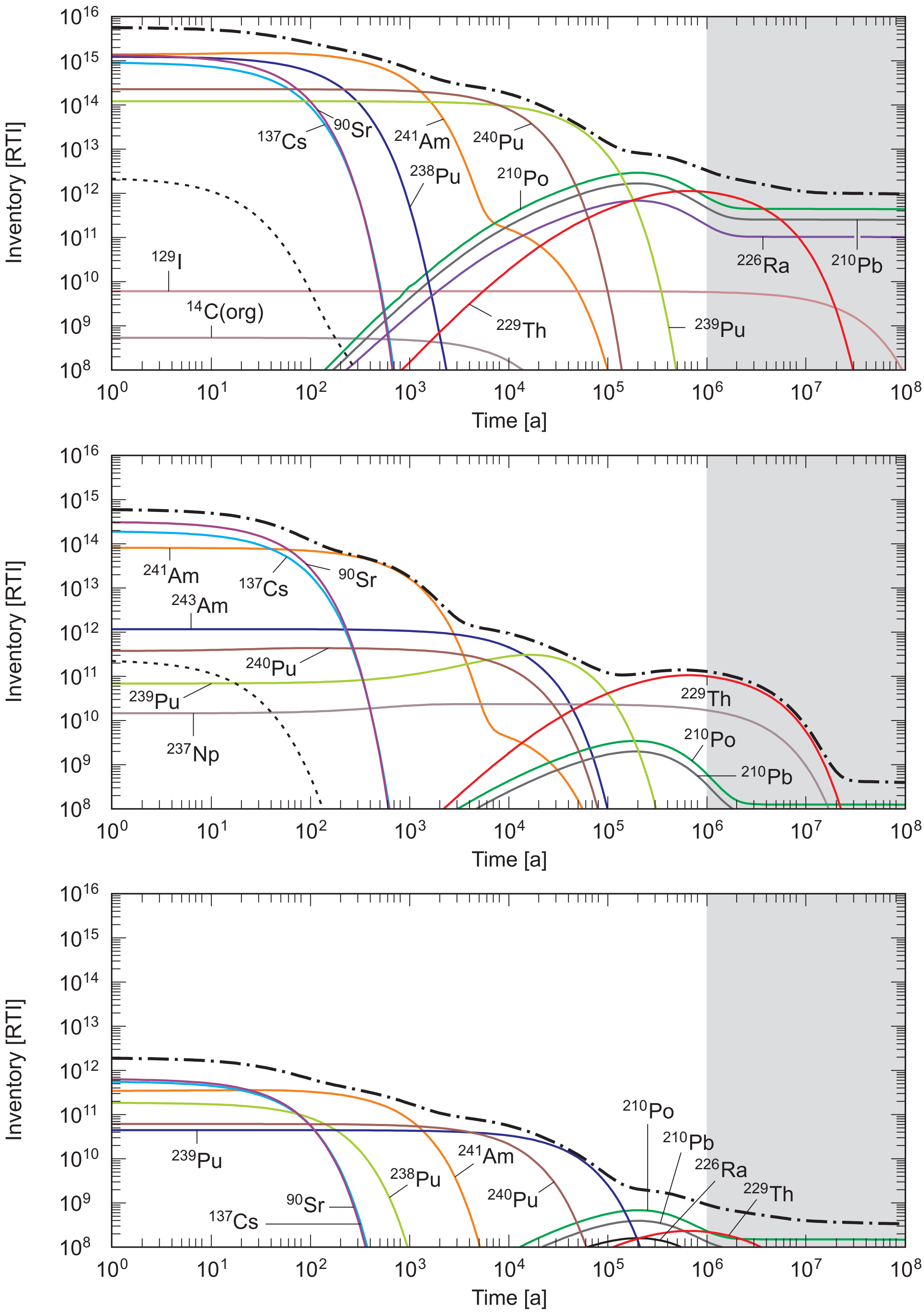


Fig. 6.4-1: The contribution of different radionuclides to the waste inventory, expressed as a radiotoxicity index (RTI), and its variation with time for SF (upper figure), HLW (middle figure) and ILW (lower figure)

The dotted line represents the sum of non-safety relevant radionuclides (see text).

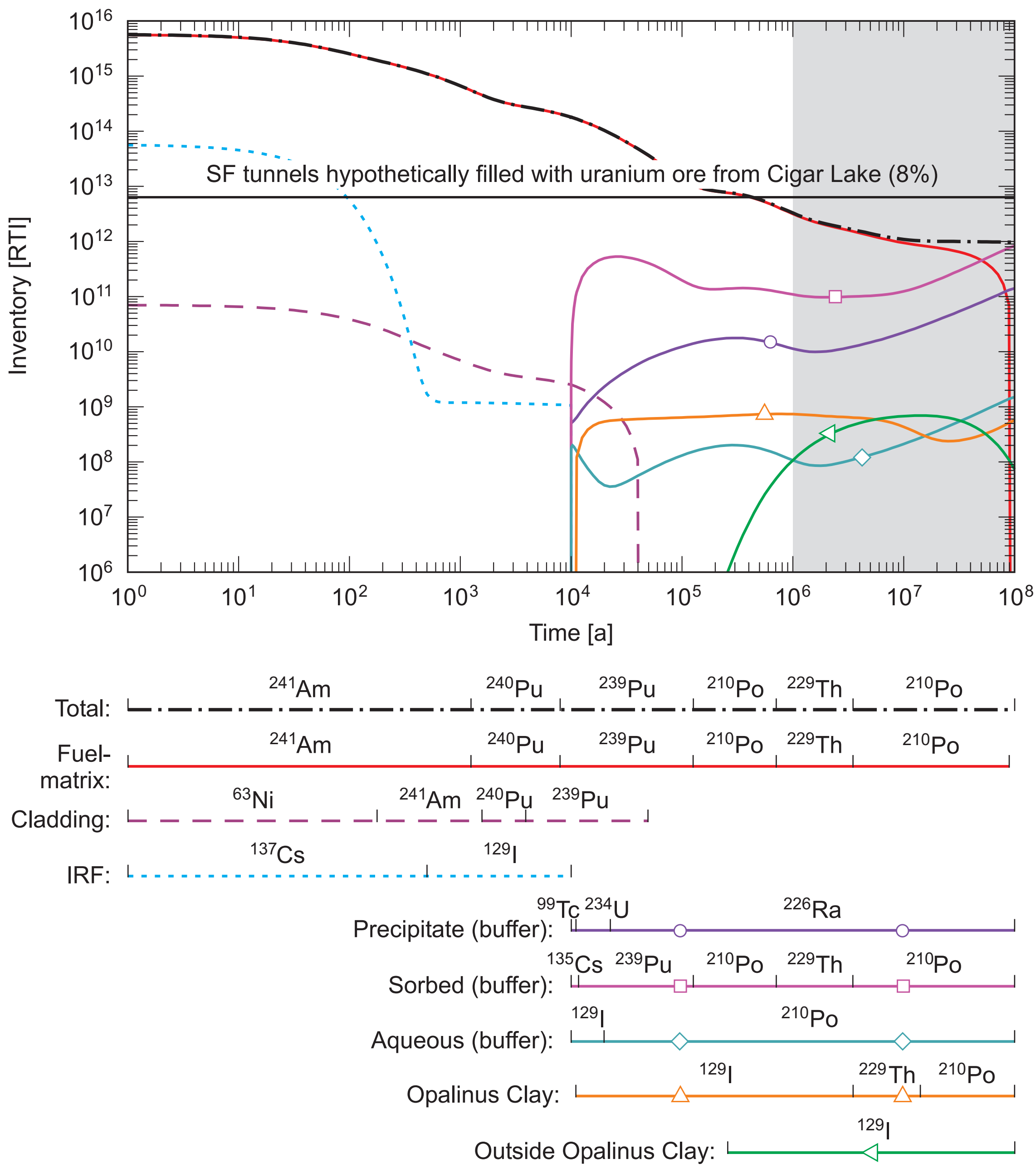


Fig. 6.4-2: The evolution and distribution of RTI from SF in the different components of the near field, the Opalinus Clay and regions outside the upper and lower boundaries of the Opalinus Clay

The bars beneath the graph indicate the radionuclides that make the highest contribution to radiotoxicity at any particular time and in any particular part of the system.

The Reference Case results presented in the following sections illustrate how the barrier system retains radionuclides, especially by immobilisation in the SF and HLW waste forms and geochemical immobilisation within and around the repository, while their associated radiotoxicity declines. It also illustrates that transport processes conveying radionuclides towards the biosphere are very slow. Long periods of retention and the slowness of transport processes mean that substantial radioactive decay takes place for the majority of radionuclides before any eventual release to the biosphere can occur.

6.4.2 The distribution of radiotoxicity as a function of time

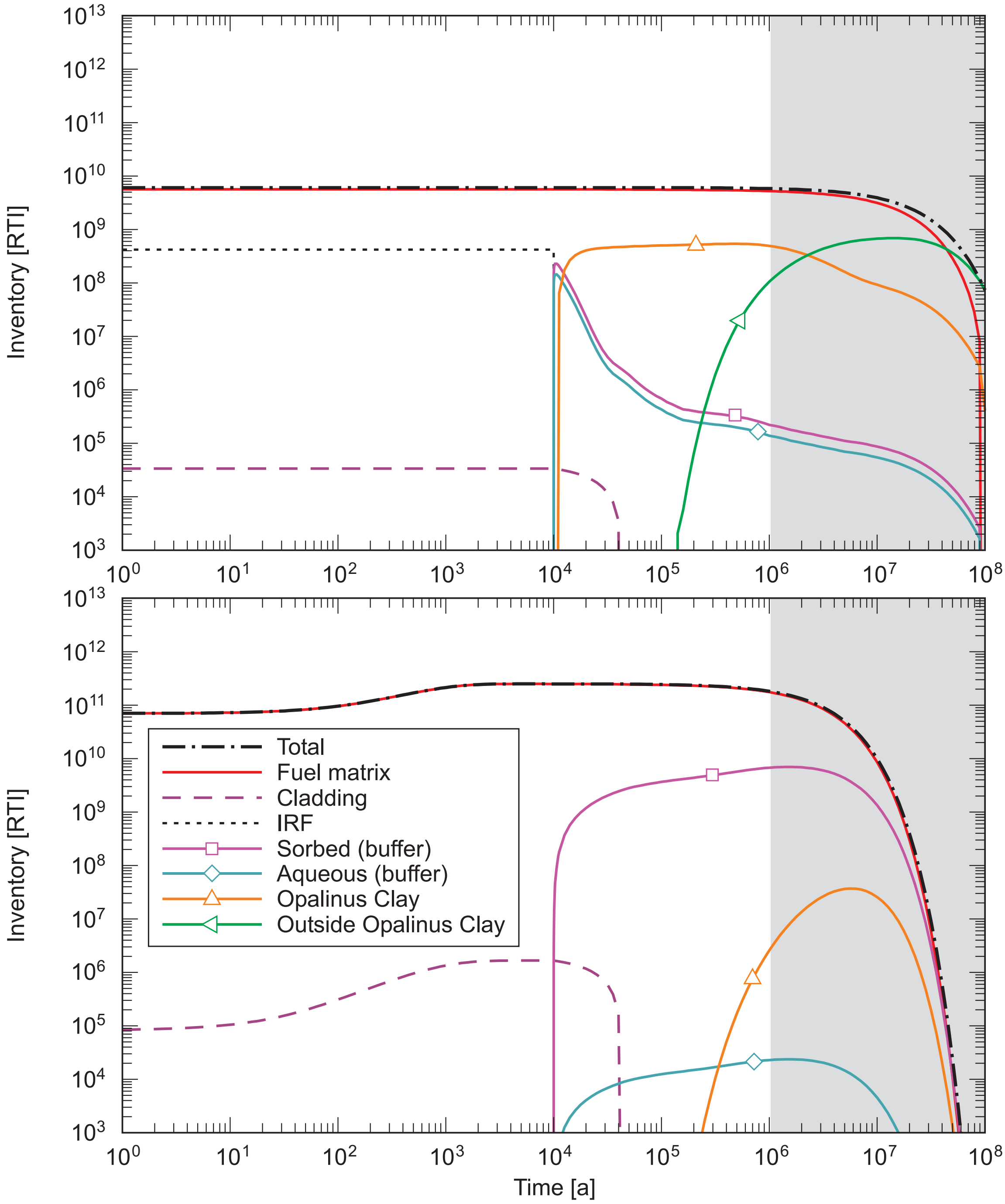
Spent fuel

The distribution of radiotoxicity at different times between the different components of the barrier system, and also the biosphere, is illustrated for SF in Fig. 6.4-2, which shows that the total radiotoxicity, which is dominated by the actinides, is contained almost entirely within the SF matrix for more than 10 million years. After about 400 000 years, the radiological significance of the SF in terms of RTI is similar to that of a uranium ore body with the same volume as the SF disposal tunnels. Only at the very longest times is most radiotoxicity no longer contained within the fuel matrix, and, even then, it is mainly contained in the near field, sorbed in the bentonite buffer.

The RTI of radionuclides that have migrated into the Opalinus Clay, which is dominated by the daughters of ^{238}U, is still increasing, even after 10^8 years. This can be explained in terms of the very low corrosion rate of the fuel matrix, the very long half life of ^{238}U (4.5×10^9 years), and the strong sorption of uranium compared to that of some of its daughters (e.g. ^{226}Ra) within the bentonite and Opalinus Clay. The very small RTI outside the Opalinus Clay host rock, i.e. in overlying and underlying sedimentary layers and in the surface environment, is dominated by the long-lived and poorly sorbing ^{129}I. This rises to a maximum at about 10 million years and then decreases again due to the radioactive decay of ^{129}I, which has a half life of 16 million years.

The maximum RTI outside the Opalinus Clay occurs later than the maximum biosphere dose (which is also dominated by ^{129}I and occurs at about a million years, see Fig. 6.5-1). The RTI that is present outside the Opalinus Clay continues to rise as radionuclides are released from the barrier system, even after the rate of release passes its maximum, until it is eventually attenuated by radioactive decay. The RTI outside the Opalinus Clay is based on the cumulative total activity released from the barrier system, whereas the maximum doses are based on the part of that activity which at any one time is present in the compartments of the local biosphere model representing the surface environment near to the discharge area. Individual doses in the wider environment can, however, be expected to be several orders of magnitude below those estimated in the local biosphere due to the diluting effect of mixing potentially radionuclide-bearing water and sediments from the discharge area with water and sediments from up-stream and down-stream of the discharge area.

Fig. 6.4-3 shows the evolution and distribution of radiotoxicity originating from SF for two long-lived radionuclides with contrasting sorption properties, namely ^{129}I (a low sorbing anion) and ^{237}Np (a strongly sorbing actinide). The figure shows that the RTIs of both radionuclides are contained predominantly in the fuel matrix for more than 10 million years. A significant part (a few percent) of the RTI of ^{129}I is, however, contained initially in the grain boundaries of the fuel matrix, in fuel pellet cracks and in the gap between the fuel and the cladding. Upon canister breaching, this instant release fraction (IRF) enters solution, and is then transported through the clay barriers (i.e. the bentonite and the Opalinus Clay), with retardation by weak sorption, before finally reaching the overlying and underlying sedimentary layers and the biosphere. Even though the maximum RTI outside the Opalinus Clay does not occur until more than 10 million years after emplacement, there is little decay of ^{129}I inventory by this time, due to its very long half life. By contrast, most of the ^{237}Np in SF has decayed before it can migrate into the Opalinus Clay; thus only an insignificant proportion of the ^{237}Np inventory is ever present in the Opalinus Clay, and the amount outside the Opalinus Clay is too small to appear in the figure.



Note: The increase in RTI of ^{237}Np over the period from 100 to 1000 years after emplacement is due to radioactive ingrowth from parent radionuclides in the chain.

Fig. 6.4-3: The evolution and distribution of RTI from SF due to ^{129}I (upper figure) and to ^{237}Np (lower figure)

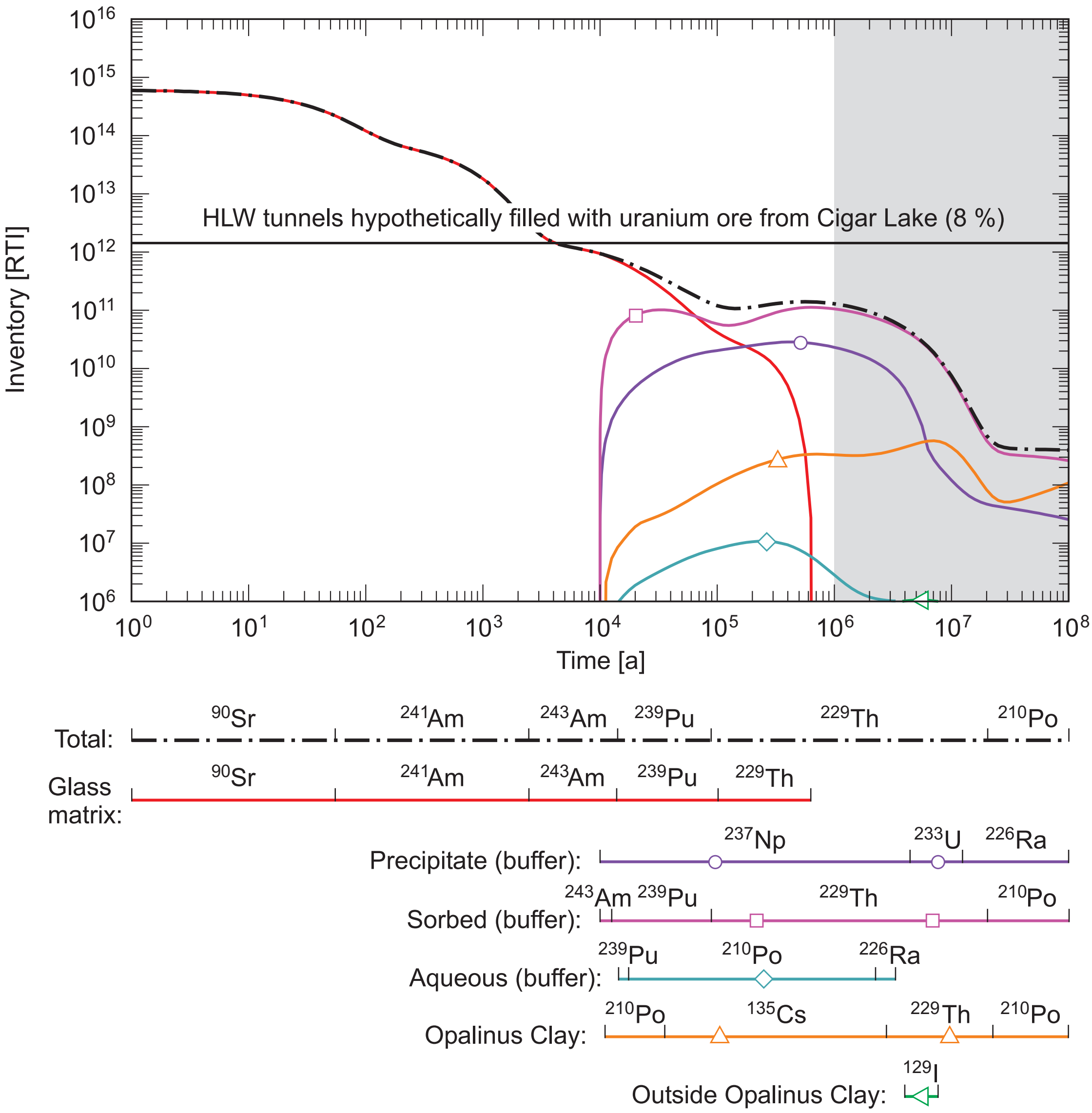


Fig. 6.4-4: The evolution and distribution of RTI from HLW in the different components of the near field, the Opalinus Clay and regions outside the upper and lower boundaries of the Opalinus Clay

The bars beneath the graph indicate the radionuclides that make the highest contribution to radiotoxicity at any particular time and in any particular part of the system.

HLW and ILW

Fig. 6.4-4 and 6.4-5 show the evolution and distribution of radiotoxicity originating from HLW and ILW. Again, radiotoxicity is contained predominantly in the repository near field. As is the case for SF, the main contributors to the total radiotoxicity are the actinides. In the case of HLW, radiotoxicity is initially contained predominantly in the glass matrix, and later sorbed or precipitated in the bentonite buffer. Eventually, after about 10^7 years, about 10 % of the total radiotoxicity is contained in the Opalinus Clay, but the RTI has by then declined due to radioactive decay by about five orders of magnitude. In the case of ILW, radiotoxicity is predominantly present as sorbed and precipitated phases in the cement buffer. The RTI of ILW declines by more than three orders of magnitude over a million years. For both HLW and ILW, the small portion of the RTI that is present in overlying and underlying sediments and in the biosphere reaches a maximum after a few million years, and is dominated again by ^{129}I.

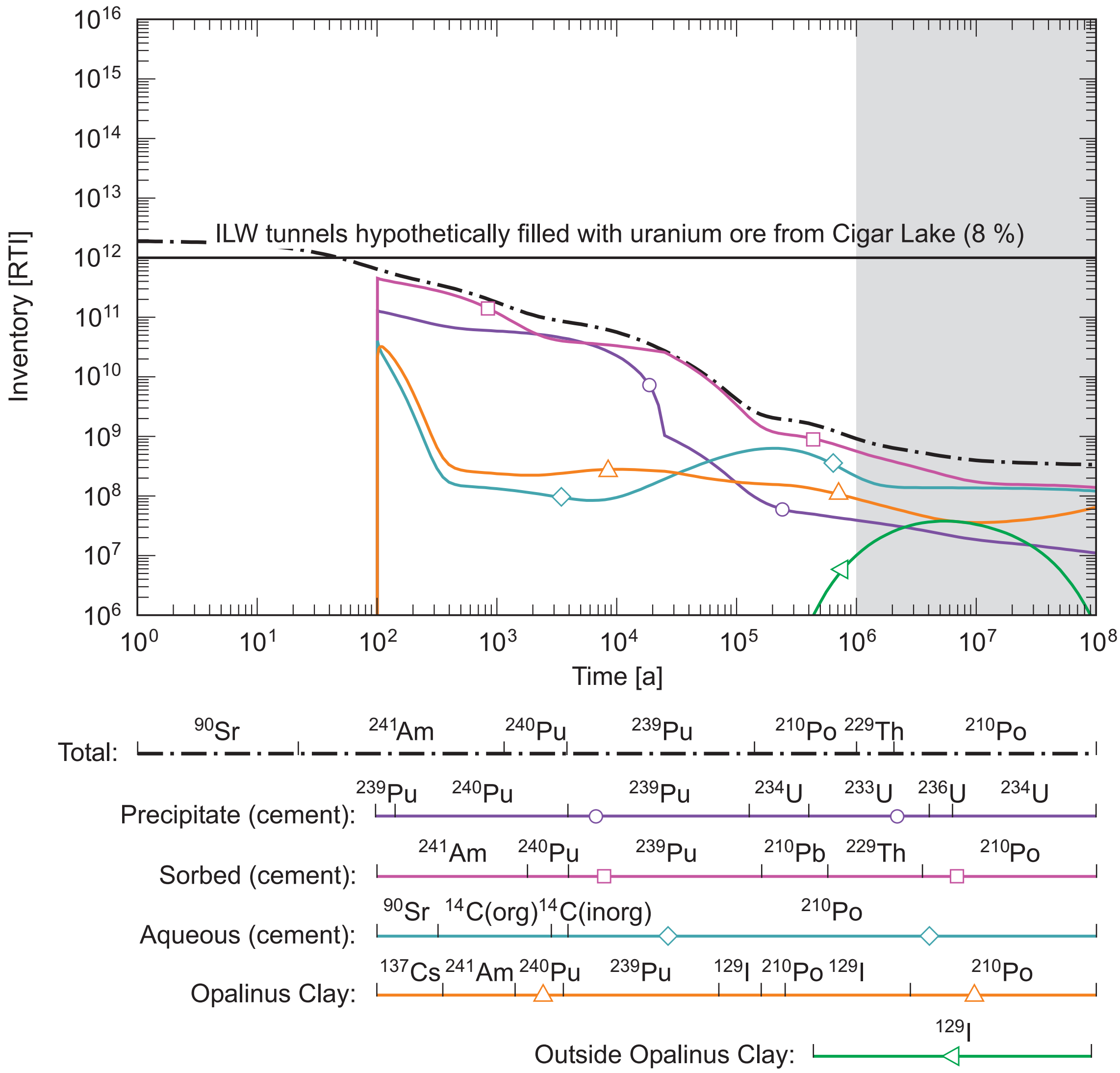


Fig. 6.4-5: The evolution and distribution of RTI from ILW in the different components of the near field, the Opalinus Clay and regions outside the upper and lower boundaries of the Opalinus Clay

The bars beneath the graph indicate the radionuclides that make the highest contribution to radiotoxicity at any particular time and in any particular part of the system.

6.4.3 Extent of decay within the system components

Tab. 6.4-1 shows the proportions of the RTI due to various example radionuclides that decay within different parts of the barrier system for SF, HLW and ILW. Integration time is 10^8 years. The vast majority of individual nuclides decay before penetrating beyond the Opalinus Clay, except for a few low-sorbing and long-lived nuclides, such as ^{129}I, ^{36}Cl and ^{79}Se, which have been deliberately selected for inclusion in Tab. 6.4-1.

Tab. 6.4-1: The proportions of the RTI due to key radionuclides that decay within different parts of the barrier system for SF, HLW and ILW

Nuclide	**Proportion of radionuclide inventory originating in spent fuel that decays in:**				
	waste matrix	**waste matrix + immediate surroundings + precipitates**	**waste matrix + immediate surroundings + precipitates + the buffer**	**waste matrix + immediate surroundings + precipitates + the buffer + the Opalinus Clay**	**outside the Opalinus Clay**
^{14}C (organic)	0.88	0.88	0.90	1.00	4.5×10^{-5}
^{36}Cl	0.41	0.41	0.41	0.89	0.11
^{59}Ni	0.22	0.56	0.95	1.00	0.00
^{79}Se	0.89	0.99	0.99	0.99	6.8×10^{-3}
^{99}Tc	0.96	0.96	1.00	1.00	0.00
^{129}I	0.69	0.69	0.69	0.72	0.27
Nuclide	**Proportion of radionuclide inventory originating in HLW that decays in:**				
	waste matrix	**waste matrix + immediate surroundings + precipitates**	**waste matrix + immediate surroundings + precipitates + the buffer**	**waste matrix + immediate surroundings + precipitates + the buffer + the Opalinus Clay**	**outside the Opalinus Clay**
^{59}Ni	0.62	0.62	0.91	1.00	0.00
^{79}Se	0.07	1.00	1.00	1.00	2.4×10^{-3}
^{99}Tc	0.28	0.93	1.00	1.00	0.00
^{129}I	0.01	0.01	0.01	0.10	0.89
Nuclide	**Proportion of radionuclide inventory originating in ILW that decays in:**				
	waste matrix and cement buffer		**waste matrix and cement buffer + the Opalinus Clay**		**outside the Opalinus Clay**
^{14}C (organic)	0.52		1.00		2.0×10^{-4}
^{36}Cl	0.16		0.84		0.16
^{79}Se	0.22		0.57		0.43
^{99}Tc	0.00		1.00		0.00
^{129}I	0.00		0.10		0.89

Notes: The "immediate surroundings" include a small volume of fluid that is assumed to exist between the waste and the clay barriers.

Integration time is 10^8 years; i.e. for long-lived nuclides the sum of the values in the last two columns may be less than 1.00.

6.5 Radionuclide release rates in the Reference Case

6.5.1 Release rates from the different waste forms

The previous section has shown that the majority of radionuclides decay within the barrier system and never reach the biosphere. A small fraction is, however, released to the biosphere at very low rates. Release rates for the three different waste forms in the Reference Case are presented below. Releases are expressed in terms of doses. In the cases of release from the waste form into the bentonite buffer, and the release from the buffer into the Opalinus Clay, these are hypothetical doses that are obtained by multiplying the releases by the biosphere dose conversion factors for the Reference Case as discussed in Section 6.3.3 in the context of the surface environment.

Spent fuel

Fig. 6.5-1 shows the release rates of radionuclides from SF into the bentonite buffer, from the buffer into the Opalinus Clay, and from the Opalinus Clay, via the overlying and underlying sedimentary layers, to the biosphere, following canister breaching at 10 000 years (the barrier effect of the overlying and underlying sedimentary layers is conservatively neglected – see Section 6.3.3).

The upper figure shows that immediately following canister breaching, releases from SF are dominated by ^{129}I, a part of which is released as an instant release fraction (IRF). The IRF is visible as a spike at 10 000 years in the upper figure. Immediately after that, releases from SF to the buffer are dominated by ^{230}Th (a member of the ^{238}U chain) and, at later times, by ^{231}Pa (a member of the ^{235}U chain).

A comparison of the three figures shows that releases of nuclides of the actinide chains are delayed, and release rates diminished, during transport through the buffer, and even more so during transport through the Opalinus Clay. The actinide chains make no significant contribution to releases to the biosphere.

The lower figure, and a comparison of this with the middle figure, shows that:

- release rates of radionuclides that penetrate the buffer are substantially attenuated during transport through the host rock,
- releases to the biosphere are dominated by the long-lived, highly soluble and low sorbing radionuclide ^{129}I, with smaller contributions from ^{36}Cl, organic ^{14}C (which is assumed to be non-sorbing) and ^{79}Se,
- releases of these radionuclides are also delayed, and release rates diminished during transport through the clay barriers, although to a much lesser extent than for the nuclides of the actinide chains,
- the maximum release to the biosphere, which does not occur until about one million years have elapsed, gives a dose of less than 10^{-4} mSv a^{-1}, which is more than three orders of magnitude below the Swiss regulatory guideline.

The attenuation of radionuclide releases during transport through the buffer and host rock is discussed in more detail in Section 6.6.

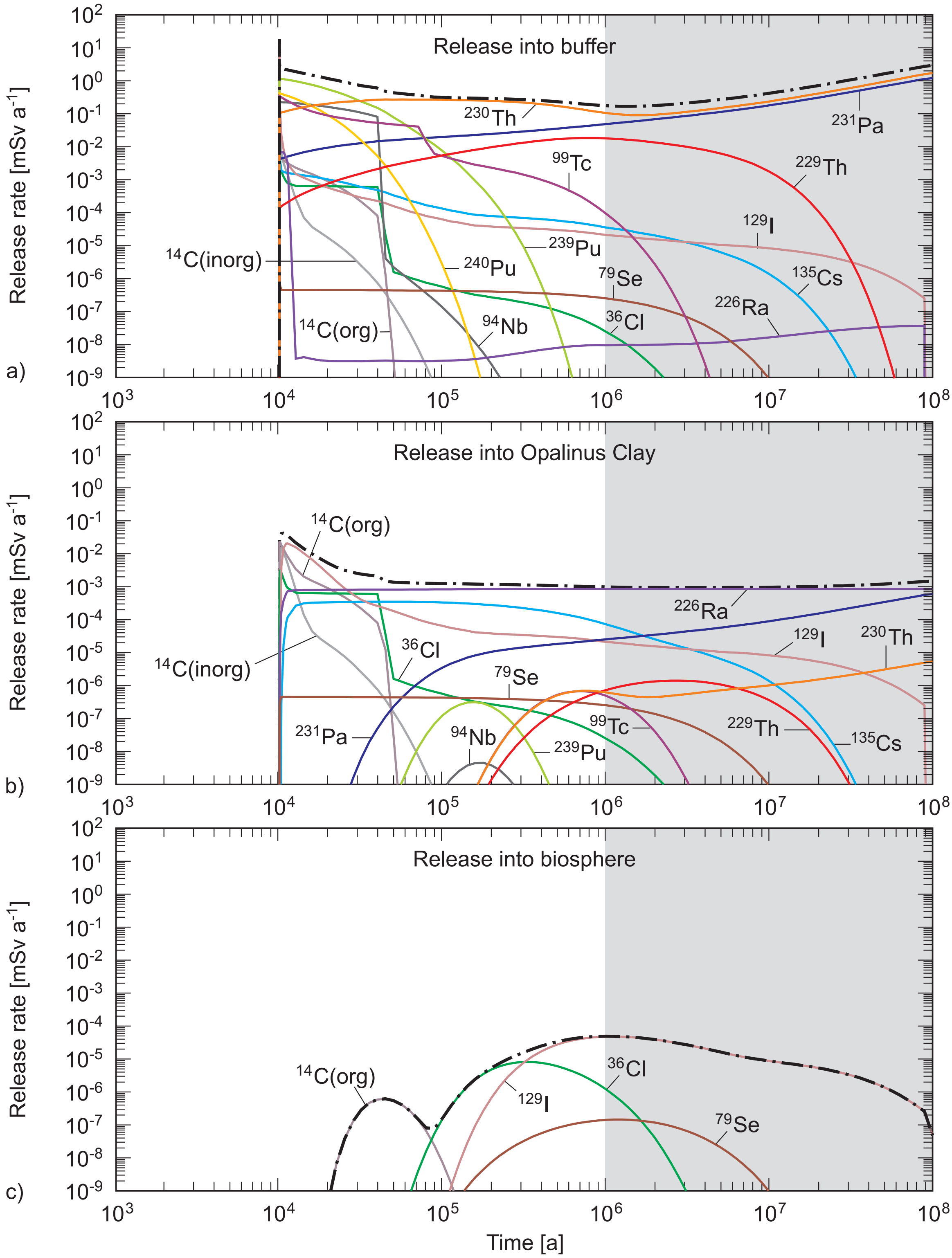


Fig. 6.5-1: The releases of radionuclides from SF

(a) into the bentonite buffer (upper figure), (b) from the buffer into the Opalinus Clay (middle figure), (c) from the Opalinus Clay to the biosphere (lower figure), expressed in terms of a hypothetical dose as defined in the main text.

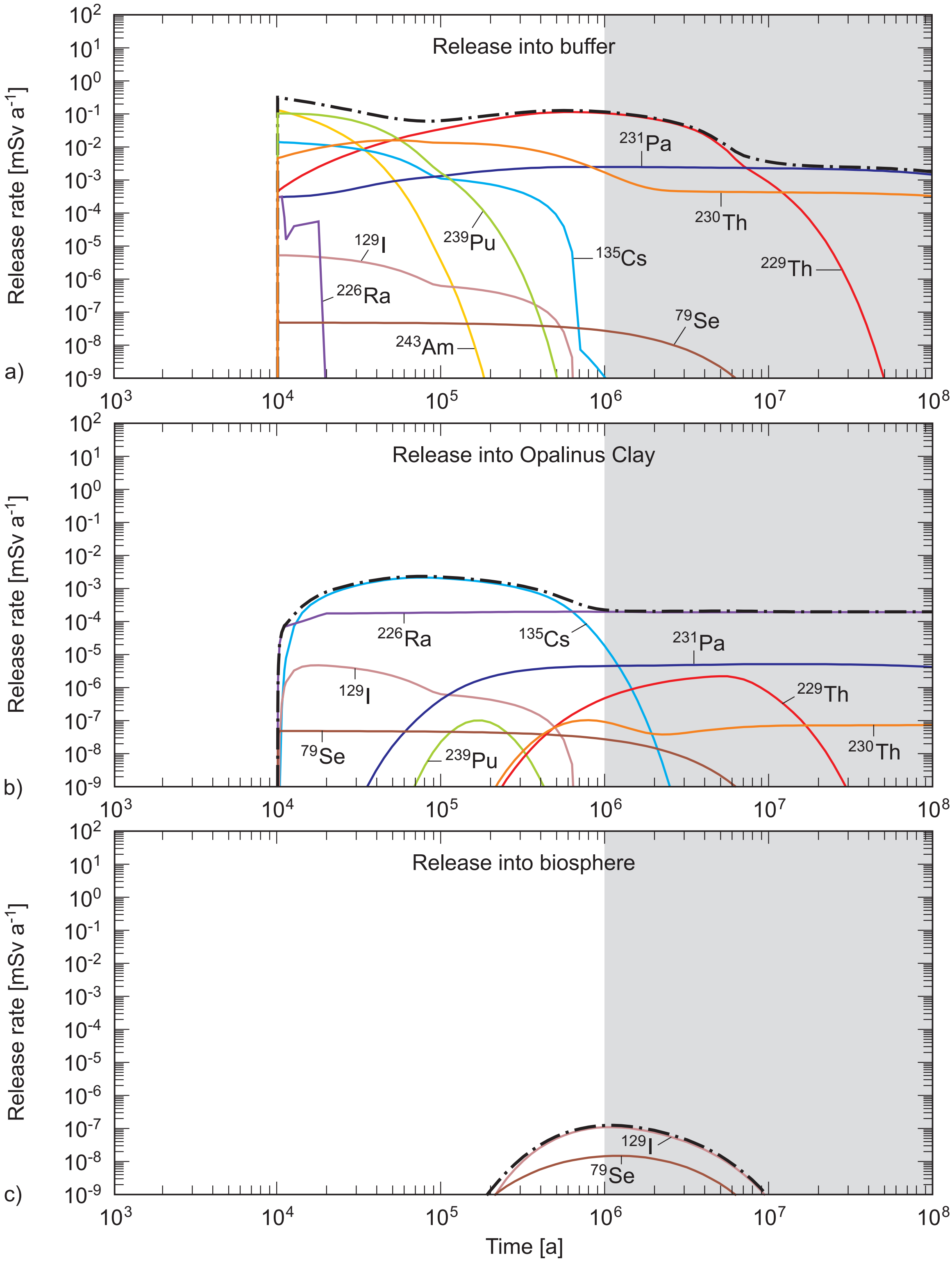


Fig. 6.5-2: The releases of radionuclides from HLW

(a) into the bentonite buffer (upper figure), (b) from the buffer into the Opalinus Clay (middle figure), (c) from the Opalinus Clay to the biosphere (lower figure), expressed in terms of a hypothetical dose as defined in the main text.

HLW

Fig. 6.5-2 shows the evaluated release rates from one barrier system component to another and from the barrier system to the biosphere in the case of HLW.

The most significant difference between releases due to HLW and those due to SF is that, in the case of HLW, there is no instant release "pulse" of radionuclides such as ^{129}I upon canister breaching.

^{135}Cs and the daughters of ^{238}U dominate releases from the buffer to the Opalinus Clay. As in the case of SF, only a very small number of radionuclides penetrate the Opalinus Clay, notably ^{129}I and ^{79}Se. The inventory of ^{129}I, as well as that of ^{79}Se, is, however, very much smaller in the case of HLW; yet it is again ^{129}I that dominates the dose maximum, which occurs again after about a million years. The dose maximum is about 10^{-7} mSv a^{-1} and is thus six orders of magnitude below the Swiss regulatory guideline.

ILW

Fig. 6.5-3 shows the release rates from ILW to the Opalinus Clay following the assumed 100 year period of containment after waste emplacement (mainly due to incomplete resaturation), and from the Opalinus Clay to the overlying and underlying sedimentary layers (instantaneous transport from these sediments to the biosphere assumed).

Radionuclide releases to the Opalinus Clay are dominated, at earlier times, by ^{90}Sr and later by ^{14}C, ^{129}I, ^{94}Nb and finally by the daughters of ^{238}U and ^{235}U. Once again, only a very small number of radionuclides penetrate the Opalinus Clay. ^{129}I makes the greatest contribution to dose, but organic ^{14}C (which is assumed to be non-sorbing), ^{36}Cl and ^{79}Se also contribute. The dose maximum, which again occurs after about a million years, is between 10^{-6} and 10^{-5} mSv a^{-1} and is thus more than four orders of magnitude below the Swiss regulatory guideline.

6.5.2 Overall comments on radionuclide release rates

The Reference Case shows that, irrespective of the waste form, only a few radionuclides, namely ^{129}I, ^{36}Cl, ^{79}Se and organic ^{14}C, penetrate the clay barriers to such an extent that they contribute to calculated doses, and these doses are three to six orders of magnitude below the Swiss regulatory guideline.

The calculations show that the maximum dose due to SF is the highest, followed by those due to ILW and finally HLW, in spite of the fact that the initial radiotoxicity of HLW is about two orders of magnitude higher than that of ILW. This is largely due to the fact that the highly mobile, long-lived radionuclide ^{129}I is present in significant amounts in SF and ILW, whereas it is present in much smaller amounts in HLW.

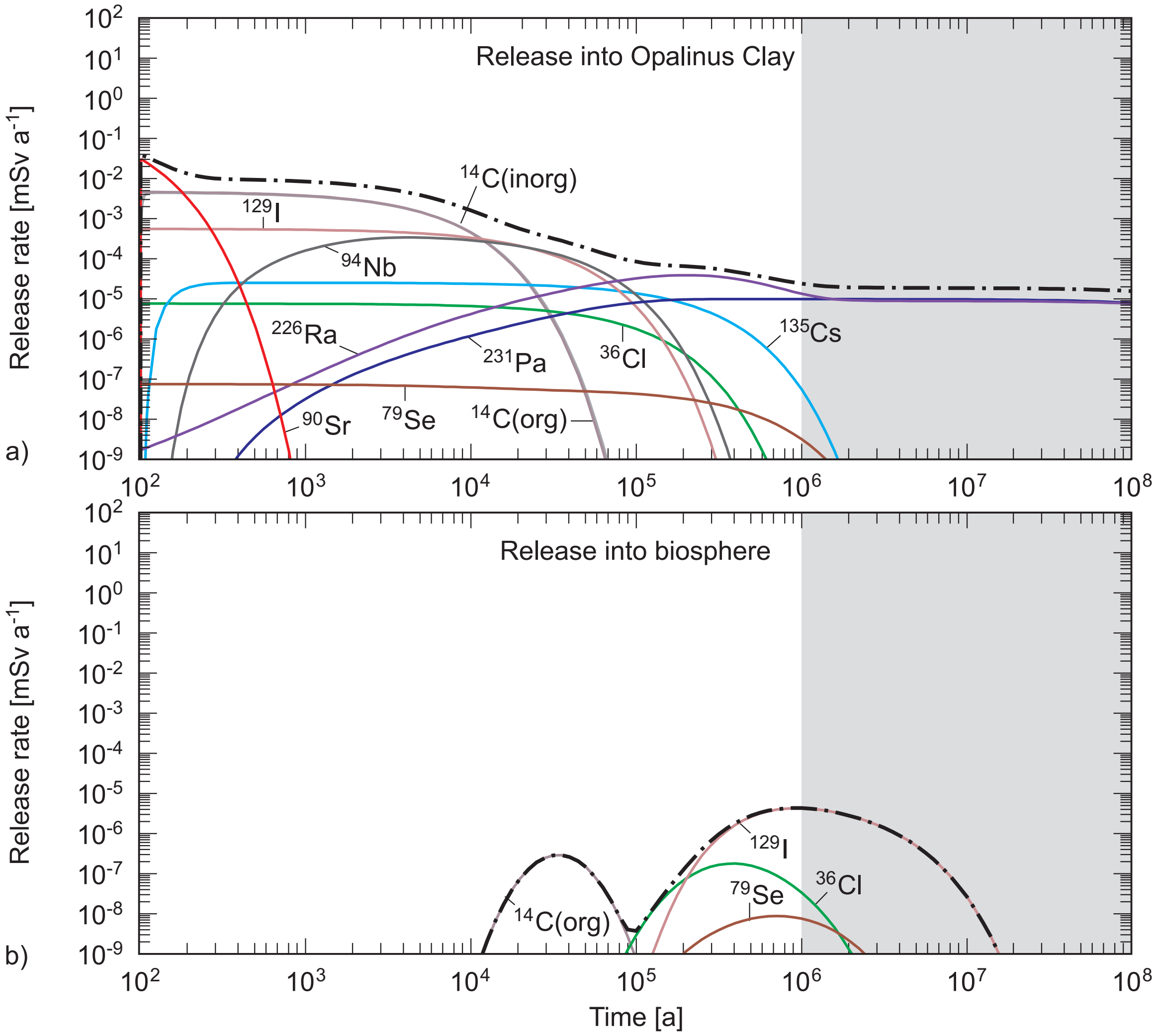


Fig. 6.5-3: The releases of radionuclides from ILW

(a) into the Opalinus Clay (upper figure), (b) from the Opalinus Clay to the biosphere (lower figure), expressed in terms of a hypothetical dose as defined in the main text.

6.6 Detailed examination of the attenuation of releases during transport through the buffer and host rock

6.6.1 Use of insight models and the reference model chain

In the previous section, it was shown that release rates of most radionuclides are substantially attenuated by radioactive decay and by the spreading of releases in time and space during transport through the buffer surrounding the SF and HLW packages and during transport through the Opalinus Clay. This leads to releases to the biosphere that are dominated by the long-lived, highly soluble and low sorbing radionuclides.

In the following sections, a simplified insight modelling approach is used to illustrate in more detail how slow transport, accompanied by radioactive decay, attenuates the releases of individual radionuclides. In the case of ^{129}I, the reference model chain is then used to illustrate the effectiveness of the spreading in time of the instant release fraction (IRF) that occurs during transport as a mechanism for reducing the maximum dose due to this long-lived, highly soluble and low sorbing radionuclide.

6.6.2 Decay during transport through the buffer and Opalinus Clay

Radionuclides that enter solution are transported towards the surface environment through the buffer and Opalinus Clay by aqueous diffusion and, in the case of the Opalinus Clay, a small amount of advection. Diffusion is, however, the dominant transport process under the expected hydraulic conditions (see Sections 4.2.5, 5.4.5 and 5.5.3). If the time required for a radionuclide to diffuse across the clay barriers is significantly longer than its half life, then it will decay substantially during transport. The radionuclide itself will then not contribute significantly to dose, although the possible further migration and release to the biosphere of any more mobile or longer lived daughter radionuclides must still be considered.

The time required for a radionuclide to diffuse across the clay varies between migrating species due to differences in the degree of sorption that they undergo, differences in the amount of accessible porosity, and differences in their diffusion coefficients. Furthermore, due to the random nature of the movement of diffusing particles, different particles of the same species take different amounts of time to diffuse across the clay, giving rise to a spreading in time during diffusive transport. It is, however, possible to define a typical timescale, t_d [a], that characterises diffusion across the clay for a particular species. This timescale is a function of the diffusion coefficient, the diffusion-accessible porosity and the degree of sorption of the diffusing species, and of the thickness of the clay[88].

The ratio $t_d/t_{1/2}$ is plotted in Fig. 6.6-1 for the different radionuclides considered in the safety assessment, with t_d calculated using Reference Case parameter values. The figure shows those radionuclides for which this ratio is greater than 30 (those in the shaded area), and those for which it is less. Most radionuclides have a ratio $t_d/t_{1/2}$ of greater than 1000, and thus will decay to insignificance during transport. Radionuclides for which $t_d/t_{1/2}$ is greater than 1000 are not shown explicitly in the figure due to their large number. Only a few radionuclides are expected to diffuse across the Opalinus Clay without significant decay, namely the low sorbing, long-lived anions ^{36}Cl, ^{79}Se and ^{129}I. These radionuclides will eventually diffuse across the Opalinus Clay, and will do so over a time period that is similar to or (in the case of ^{129}I) less than their half lives. ^{129}I is discussed further in Section 6.6.3. The more highly sorbing and long-lived actinides ^{238}U and ^{232}Th, as well as the low sorbing but shorter lived organic ^{14}C and ^{41}Ca, require a period that is about 100 times their respective half lives to diffuse across the clay barriers, and so will decay substantially (although, even with an attenuation by decay during transport of more than 4 orders of magnitude, organic ^{14}C is still visible in the dose curves for SF and ILW: see Fig. 6.5-1 and 6.5-3). Furthermore, ^{238}U and ^{232}Th require hundreds of millions of years or more to diffuse across the Opalinus Clay. Such time periods are orders of magnitude longer than the million-year timescale of primary interest in the safety assessment. ^{36}Cl, ^{79}Se, ^{129}I, ^{41}Ca and organic ^{14}C, on the other hand, diffuse across the Opalinus Clay relatively quickly due to their low sorption, although a few hundreds of thousands of years are still required for full breakthrough.

The degree of decay of different radionuclides during transport through the Opalinus Clay and, in the case of SF and HLW, the buffer, can be further illustrated using the concept of "barrier efficiency". The barrier efficiency is a measure of decay during transport on the assumptions (i), that radioactive ingrowth can be neglected and (ii) that a steady state has been reached. A

[88] Briefly, $t_d = L^2R/D$, where R is a retention coefficient due to sorption, D [m^2 a^{-1}] is the pore diffusion coefficient and L [m] is the transport distance through the Opalinus Clay.

barrier efficiency of *one* thus corresponds to the ideal situation of complete decay during transport. A barrier efficiency of *zero* indicates that no significant decay occurs. The evaluated efficiency increases with the length of the transport path considered, as greater transport times give rise to more decay.

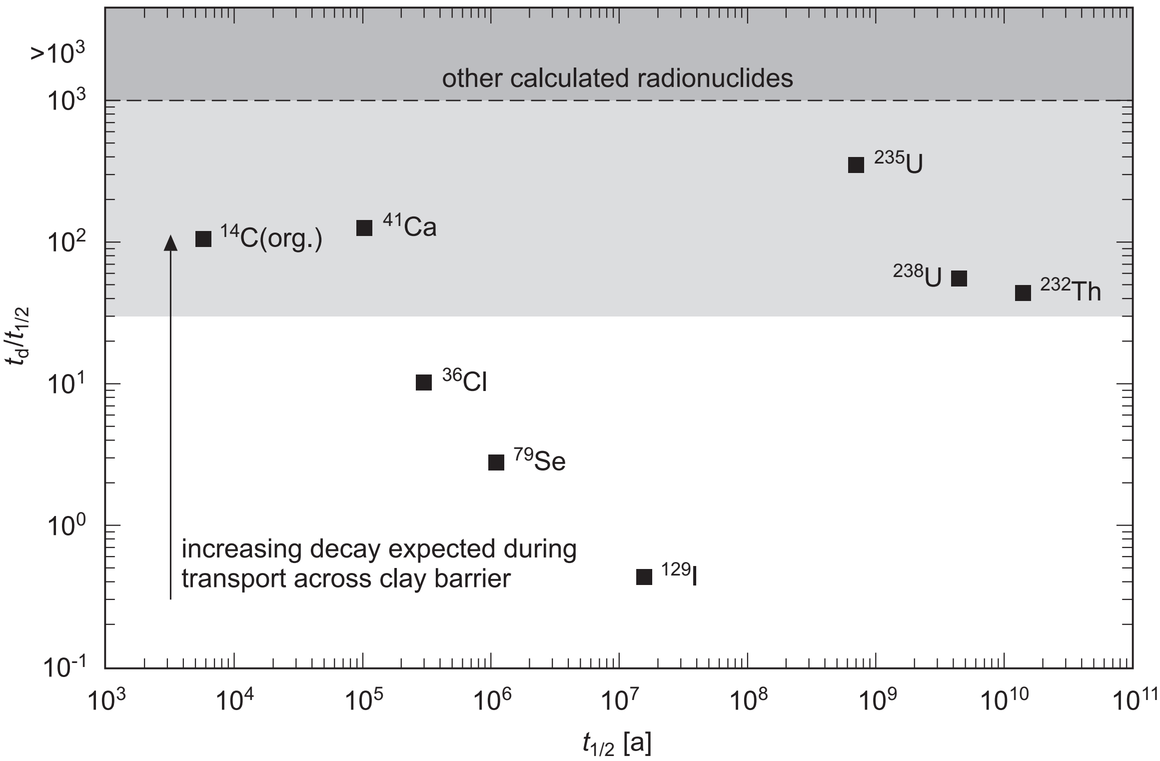


Fig. 6.6-1: $t_d/t_{1/2}$ (the ratio of the timescale for diffusive transport across the Opalinus Clay to half life) vs. $t_{1/2}$ for the radionuclides considered in the safety assessment

Radionuclides for which $t_d/t_{1/2}$ is greater than 1000 are not shown explicitly in the figure because a very large number are in this category and their contribution to dose is negligible.

The barrier efficiency at different positions within the two-medium buffer / host rock system (i.e. the barrier efficiency considering only that portion of the transport path between the waste and the position under consideration) can be evaluated analytically, as described in Nagra (2002c), although geometrical simplifications are again made. Fig. 6.6-2 shows the distance into the buffer and host rock at which the barrier efficiency reaches 99 % and 99.99 % for the various radionuclides considered in the safety assessment that have half lives greater than 100 years (radionuclides with shorter half lives, such as ^{137}Cs and ^{90}Sr, decay almost entirely within the SF and HLW canisters and, even in the case of earlier than expected canister breaching, only penetrate a small distance into the buffer). The near field geometry for SF and HLW disposal is assumed, with only radial diffusive transport across the bentonite buffer considered, as indicated in the sketch in the lower part of the figure. 1 D diffusive and advective / dispersive transport is considered in the host rock, which is again modelled as a semi-infinite, homogeneous medium, the first 40 m of which represent the Opalinus Clay (40 m is assumed to be the shortest distance from the outer boundary of the buffer to the edge of the Opalinus Clay).

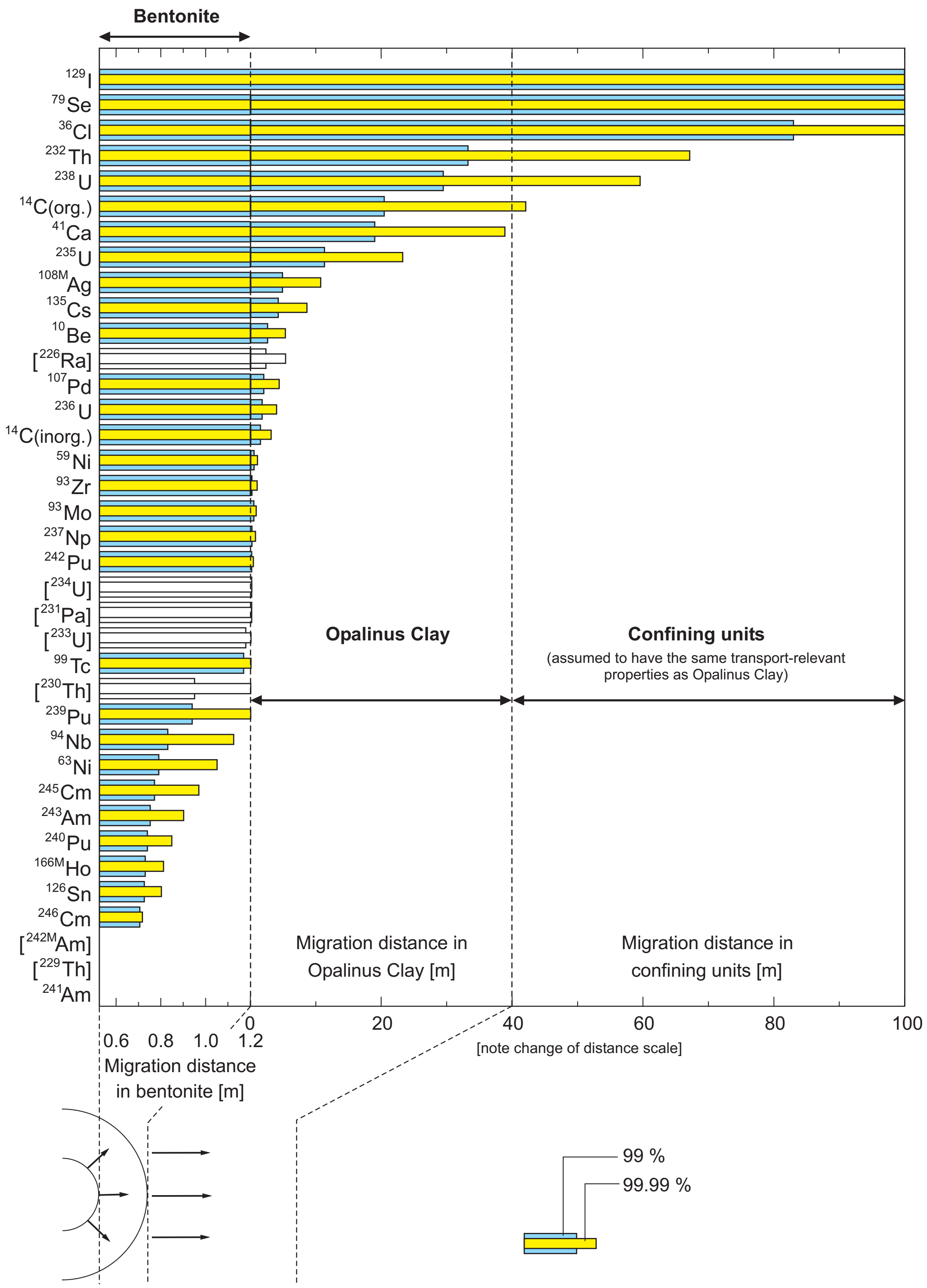


Fig. 6.6-2: Distance into the bentonite buffer and host rock at which, if a steady state is reached, radionuclide transport rates are attenuated by 99 % (broad bars), and 99.99 % (narrow bars), due to decay during transport

Radionuclides such as ^{226}Ra, which are affected by ingrowth during transport, are shown in the figure in parentheses and by non-shaded bars, in order to emphasise that the assumption that ingrowth may be neglected does not hold for these radionuclides.

Fig. 6.6-2 shows again that slow transport, together with decay, prevent most radionuclides from penetrating the Opalinus Clay, even given unlimited time. It must also be emphasised that the distances do not necessarily represent penetration distances that will arise within the million year period of primary interest in the safety assessment, due to the very long times required for a steady state to be reached. This is particularly true for isotopes of the more strongly sorbing elements, including U and Th. Fig. 6.6-3 shows the very limited distance into the Opalinus Clay that isotopes of U[89] and Th migrate in one million years for Reference Case values of the sorption coefficient (K_d), and also for the lower limits (pessimistic values) given in Tab. A2.8 of Appendix 2.

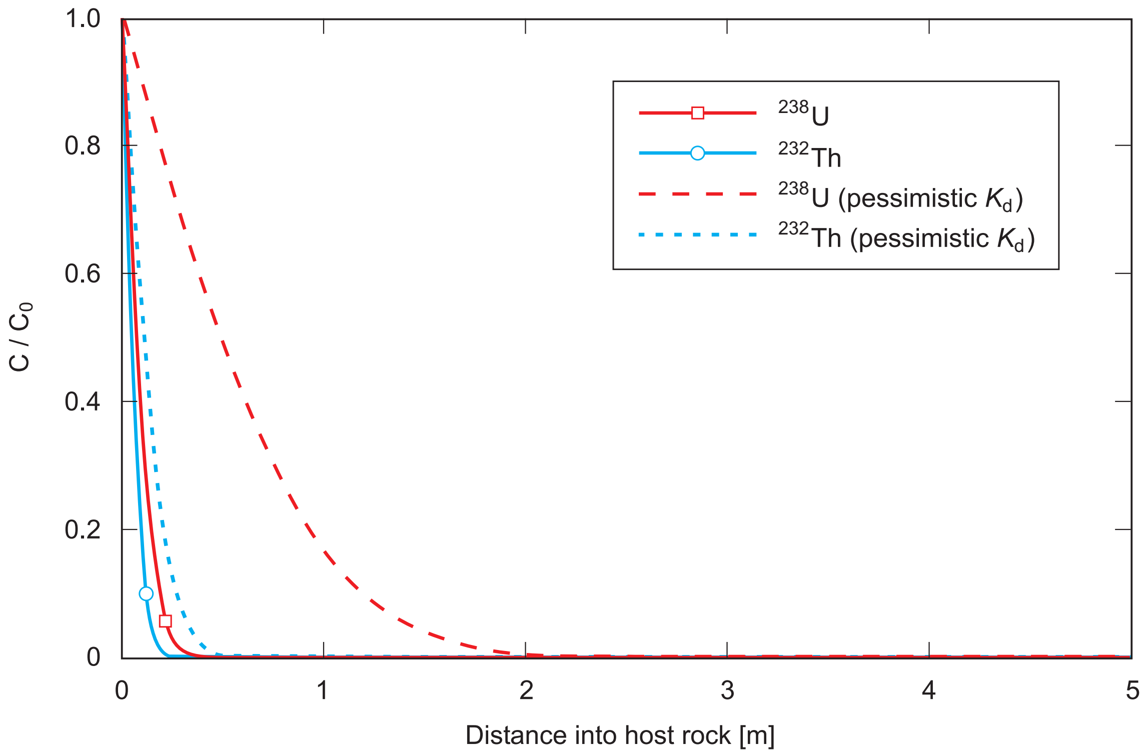


Fig. 6.6-3: Profiles of the concentration of very long-lived isotopes of U and Th within the host rock, represented as a semi-infinite medium, after one million years

The profiles are generated using both Reference Case values and pessimistic values of the sorption coefficient (K_d).

89 The natural concentration of ^{238}U in the groundwater, which is not taken into account in this assessment when calculating diffusive transport, is similar to the Reference Case solubility limit of U in the near field. Based on these data, therefore, ^{238}U from the groundwater would diffuse into the buffer across its outer boundary at the same time as ^{238}U diffuses from the fuel across its inner boundary. Ultimately (although at times beyond those of primary interest to the assessment) there would be no concentration gradient of ^{238}U to drive an overall diffusive flux in either direction.

Fig. 6.6-4 shows the barrier efficiency at different positions within the clay barriers for some example radionuclides. ^{129}I and ^{79}Se are taken as examples of radionuclides that penetrate the clay barriers with little attenuation. Other radionuclides include organic ^{14}C, ^{135}Cs, ^{237}Np, ^{93}Mo and ^{93}Zr.

Fig. 6.6-4 is consistent with the reference model chain results (Figs. 6.5-1, 6.5-2): of the example nuclides, only ^{129}I and ^{79}Se show relatively little decay during transport through 40 m of Opalinus Clay. In the case of SF (Fig. 6.5-1), organic ^{14}C decays substantially (by about 4 orders of magnitude) during transport through Opalinus Clay.

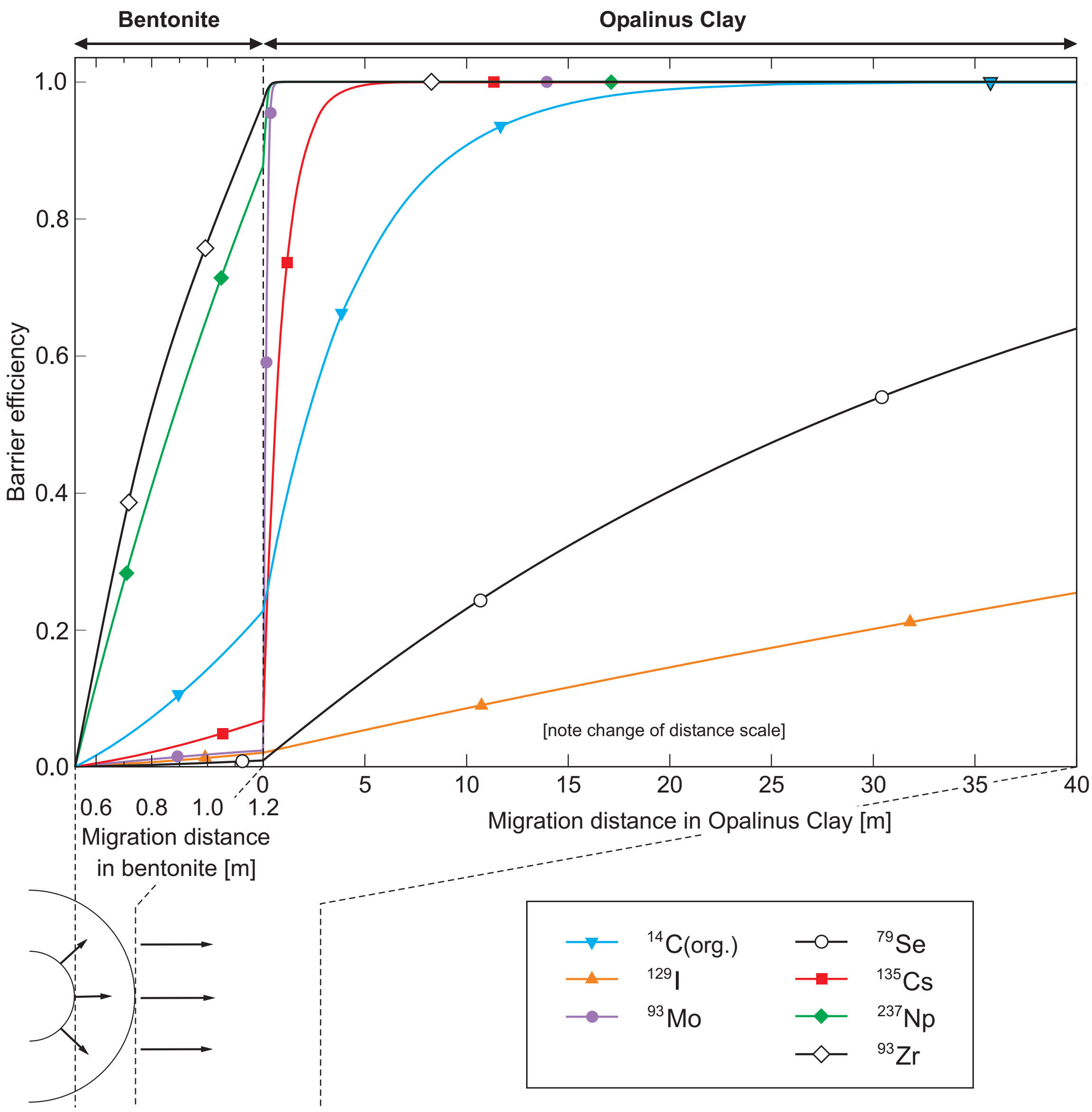


Fig. 6.6-4: The efficiency of the clay barriers as a function of radionuclide migration distance

6.6.3 The spreading in time of the instant release fraction

In Section 6.6.2, it is shown that the decay of ^{129}I during transport through the buffer and Opalinus Clay is small. The release rate of ^{129}I to the biosphere is, nevertheless, attenuated with respect to that from the waste form due principally to the "spreading in time" of the instant release fraction (IRF) that occurs during diffusion-dominated transport.

The release rates of ^{129}I at different points within the barrier system in the Reference Case are shown in Fig. 6.6-5. The figure shows the high ^{129}I release rate from SF that occurs upon canister breaching due to the IRF and the lower but prolonged release rate from the slowly dissolving fuel matrix. The release rate from the corroding cladding is also shown, but is more than three orders of magnitude lower than the release rate from the fuel matrix at the time of canister breaching.

The figure illustrates the spreading of the IRF as it migrates across the bentonite buffer and Opalinus Clay. The IRF pulse is first spread in time during transport through the buffer, giving rise to a prolonged release curve to the Opalinus Clay. Further spreading in time results in a maximum release rate from the Opalinus Clay that is about two orders of magnitude lower than the maximum release rate from the buffer. In spite of this spreading, the effects of the IRF are still evident in the form of the dose curve due to releases from the Opalinus Clay. The magnitude of the IRF, and the degree to which it is spread during transport, are thus important factors in determining the dose maximum due to ^{129}I from SF in the Reference Case.

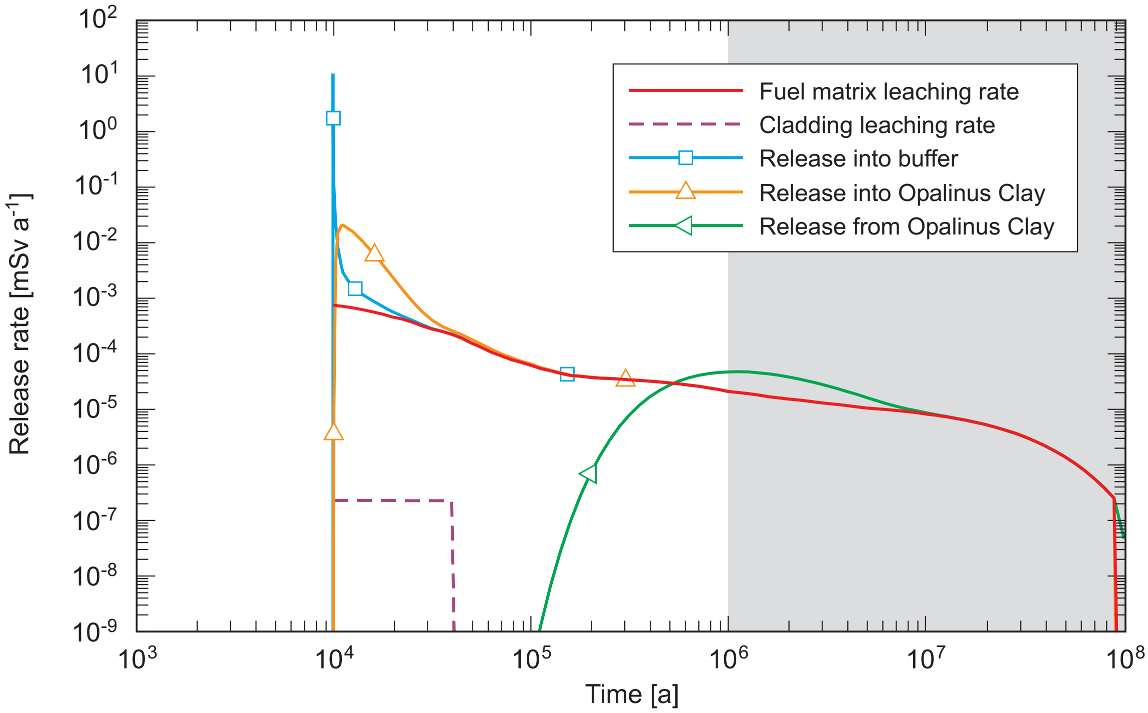


Fig. 6.6-5: The release rate of ^{129}I at different locations within the barrier system, expressed in terms of a hypothetical dose as defined in the main text

6.6.4 Summary of the performance of the barrier system in the Reference Case

From the discussions of the Reference Case in this and previous sections, it can be concluded that, for this particular conceptualisation and set of parameters:

- most radiotoxicity decays within the SF and HLW waste forms and within the surrounding bentonite buffer, or, for ILW, within the cementitious buffer,
- most radionuclides that are released from the wastes decay to insignificance during transport through the clay barriers, and
- those radionuclides that penetrate the clay barriers without substantial decay are dispersed in space and time, and can reach the biosphere only in concentrations that do not give rise to safety concerns.

Sensitivity to perturbations expressed by changes in parameter values and the effect of alternative model assumptions are examined in Section 6.7.

6.7 Sensitivity analyses

6.7.1 Approach to sensitivity analysis

The broad evolution of the barrier system and the conceptualisation of key phenomena in the Reference Case (the Reference Scenario and Reference Conceptualisation) are described in Section 6.3. In this section, sensitivity analyses are described which examine changes in the performance of the barrier system as a result of adopting alternative assumptions and parameter values. The results of the sensitivity analyses assist in determining which uncertainties are relevant to safety, and thus need to be addressed when defining the assessment cases (see Section 6.8.1). They also indicate whether or not the system is well behaved with respect to parameter variations: i.e. whether performance varies smoothly as parameter values are varied, or displays more sudden changes. The sensitivity of a well-behaved system to parameter variations can be illustrated with a smaller number of assessment cases than a system displaying complex behaviour. Sensitivity analyses also provide system understanding, and are therefore a basis for understanding the results of the assessment cases analysed in Chapter 7.

Alternative assumptions regarding the broad evolution of the barrier system and the conceptualisation of key phenomena are considered in Section 6.7.2, using simplified insight models. Specific issues addressed are:

- the effects of the presence of discontinuities in the Opalinus Clay,
- the effects of gas on the transport of radionuclides that may be present as volatile species, and
- the effects of differences in transport properties between the backfilled and sealed ramp / shaft and the surrounding Opalinus Clay.

Sensitivity to individual parameter variations, using the Reference Case as a starting point, is then considered in Section 6.7.3, using the reference model chain, together with simplified insight models to examine specific issues. Specific parameters addressed are:

- the SF / HLW canister breaching time,
- the rate of groundwater flow through the Opalinus Clay,
- the degree of radionuclide sorption in the Opalinus Clay, and

- parameters describing the surface environment.

Finally, in Section 6.7.4, a probabilistic sensitivity analysis is performed using the reference model chain in which the input parameters are varied stochastically in order to investigate the effect of varying several parameters simultaneously.

6.7.2 Sensitivity to alternative assumptions regarding the broad evolution of the barrier system and the conceptualisation of key phenomena

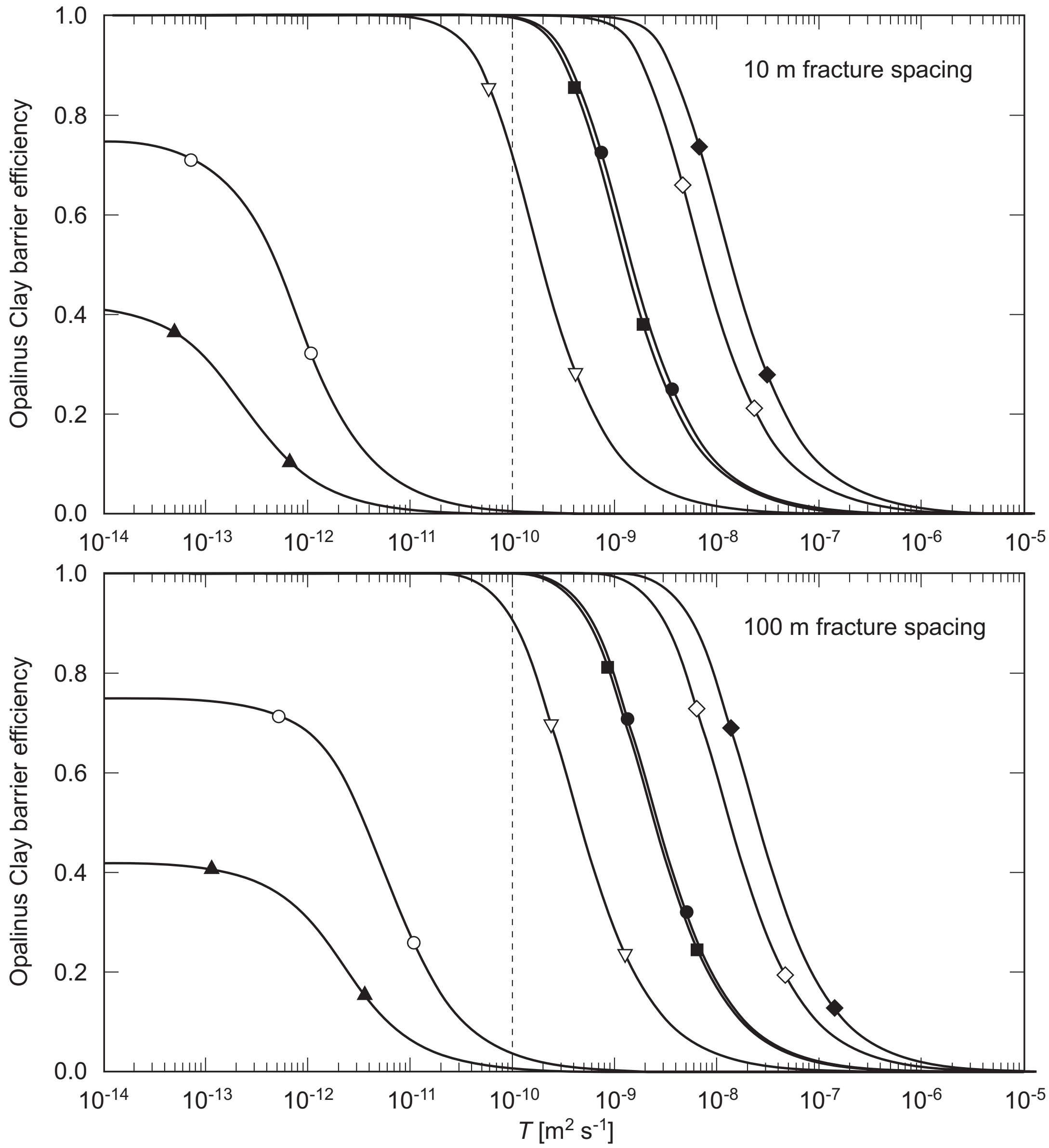
6.7.2.1 The presence of discontinuities in the Opalinus Clay

In the Reference Conceptualisation, it is assumed that the host rock contains no discontinuities with significant transmissivities. In the following discussion, however, it is hypothetically assumed that the Opalinus Clay is intersected by one or more discontinuities, with the potential to provide pathways for advective transport. The presence of sufficiently transmissive discontinuities would mean that, for radionuclides transported along these features, the degree of decay during transport would be less than in the Reference Case. As in the case of a higher groundwater flow rate through the bulk of the clay (Section 6.7.3), radionuclides that decay substantially during transport in the Reference Case may then potentially break through the Opalinus Clay.

The key characteristic of any discontinuities with respect to advective radionuclide transport is their transmissivity. Fig. 6.7-1 shows the barrier efficiency of the Opalinus Clay as a function of transmissivity for various example radionuclides (the same example radionuclides as in Fig. 6.6-4), assuming that the clay is intersected by regularly spaced vertical discontinuities with either 10 m (upper figure) or 100 m (lower figure) spacings. Matrix diffusion into the Opalinus Clay perpendicular to the direction of advection is taken into account, as well as diffusion within the matrix parallel to the direction of advection. The analytical solution for this system is given in Nagra (2002c). As the transmissivity tends to zero, the barrier efficiency tends towards a limiting value determined by diffusive transport through the bulk of the Opalinus Clay, and decreases as the transmissivity of the discontinuities increases.

For ^{129}I and ^{79}Se, the presence of very low transmissivity fractures ($\approx 10^{-12}$ m^2 s^{-1}) is sufficient for the barrier efficiency to begin to decline from its value for homogeneous clay. This does not, however, have a very significant effect on releases to the biosphere, since the barrier efficiency for these radionuclides is in any case low (even a reduction of the barrier efficiency for ^{79}Se from 0.7 to *zero* would, for example, only increase releases by a factor of $1 / (1 - 0.7) \approx 3$). Significant decay of ^{135}Cs, ^{237}Np, ^{93}Zr and ^{93}Mo is expected during transport along discontinuities with transmissivities of 10^{-10} m^2 s^{-1} or less, for fracture spacings in the range investigated. As discussed in Chapter 5, no discontinuities with transmissivities above about 10^{-10} m^2 s^{-1} are expected to exist as permanent features in the Opalinus Clay.

The figure indicates that, among the example radionuclides, only organic ^{14}C might be significantly less attenuated by decay during transport due to the presence of discontinuities with transmissivities in the order of 10^{-10} m^2 s^{-1} or less.



	Efficiency for T = 0		Efficiency for T = 0
─▽─ ^{14}C(org.)	1.00	─■─ ^{135}Cs	1.00
─▲─ ^{129}I	0.42	─◆─ ^{237}Np	1.00
─●─ ^{93}Mo	1.00	─◇─ ^{93}Zr	1.00
─○─ ^{79}Se	0.75		

Fig. 6.7-1: The efficiency of the Opalinus Clay transport barrier as a function of the transmissivity of discontinuities, *T*, assuming these to have spacings of either 10 m (upper figure) or 100 m (lower figure) throughout the Opalinus Clay

The expected situation is that no discontinuities with transmissivities above about 10^{-10} m^2 s^{-1} (the vertical dotted line) exist as permanent features in the Opalinus Clay (Chapter 5). Diffusion, but not advection, within the Opalinus Clay matrix parallel to the direction of advection in the discontinuities is taken into account. Thus, as the transmissivity tends to zero, the barrier efficiency tends towards a limiting value determined by diffusive transport through the bulk of the Opalinus Clay.

6.7.2.2 The effects of gas on the transport of radionuclides that may be present as volatile species

In the Reference Conceptualisation, it is assumed that radionuclides are released to solution, and subsequently migrate by aqueous diffusion and advection. Some radionuclides, however, and in particular ^{14}C, have the potential to form volatile species that could become mixed with gas generated within the repository and migrate in the gas phase through the Opalinus Clay and confining units, and hence to the surface environment.

In a simple insight model described below (for more details, see Nagra 2002c), a steady state model is considered for the calculation of drinking water doses related to the release of ^{14}C, assumed to be in the form of methane. The gas migration path from the repository to the surface environment includes a relatively large "storage volume" provided by different parts of the repository system. Following a period of complete containment (assumed to last 10 000 years for both SF and ILW[90]), ^{14}C instantly released from SF and ILW is mixed with repository-generated gas within the storage volume. In the framework of this simple insight model, the rate at which ^{14}C reaches the surface environment depends on:

- the rate of generation of gas within the repository (principally hydrogen generated by corrosion of the SF / HLW steel canisters and gases generated by ILW)[91],
- the magnitude of the storage volume due to gas-filled pore space in the emplacement tunnels, especially in the cementitious backfill in the ILW tunnels, in the EDZ, in the access tunnel system, in the Opalinus Clay (layer of gas-filled porosity in the vicinity of emplacement tunnels), and in the Wedelsandstein Formation.

In the time period following containment, the rate of gas generation in the repository is assumed to be balanced at all times by the rate of gas transported through the low-permeability formations of the upper confining units. Upon reaching the Malm aquifer, the radioactive methane is assumed to dissolve completely and to be transported instantaneously into the Quaternary aquifer, which is used as a source of drinking water. Degassing and atmospheric dilution of methane is conservatively neglected in the model calculations.

The maximal drinking water dose evaluated using this simple insight model is shown in Fig. 6.7-2 as a function of the gas storage volume. A range of gas storage volumes is considered (1000 to 10^6 m^3) that is wider than the range of values actually expected on the basis of more detailed calculations presented in Nagra (2002c) (6 000 to about 50 000 m^3). The drinking water dose rates are shown for two different steel corrosion rates, as the principal source of repository generated gas is by corrosion of SF / HLW steel canisters.

The figure shows how the maximal drinking water dose decreases with increasing gas storage volume and decreasing steel corrosion rate (which controls the generation of gas in the repository). Doses of about 0.001 mSv a^{-1} or less are obtained for storage volumes of more than 6 000 m^3 and for corrosion rates of less than 1 µm a^{-1} (anaerobic corrosion of the SF / HLW steel canisters in the repository environment is expected to proceed at a rate of ~ 1 µm a^{-1}; see Chapter 5). The magnitude of the storage volume provided by the Wedelsandstein Formation and other components of the barrier system is discussed in the framework of a more detailed model, which is discussed in Nagra (2002c) and summarised in Chapter 7.

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[90] For both SF and ILW, it takes several ten thousand years until breakthrough of gas through the Opalinus Clay to the Wedelsandstein occurs (Nagra 2002c).

[91] The formation rate of volatile radioactive species is many orders of magnitude lower than the generation rate of non-radioactive gases.

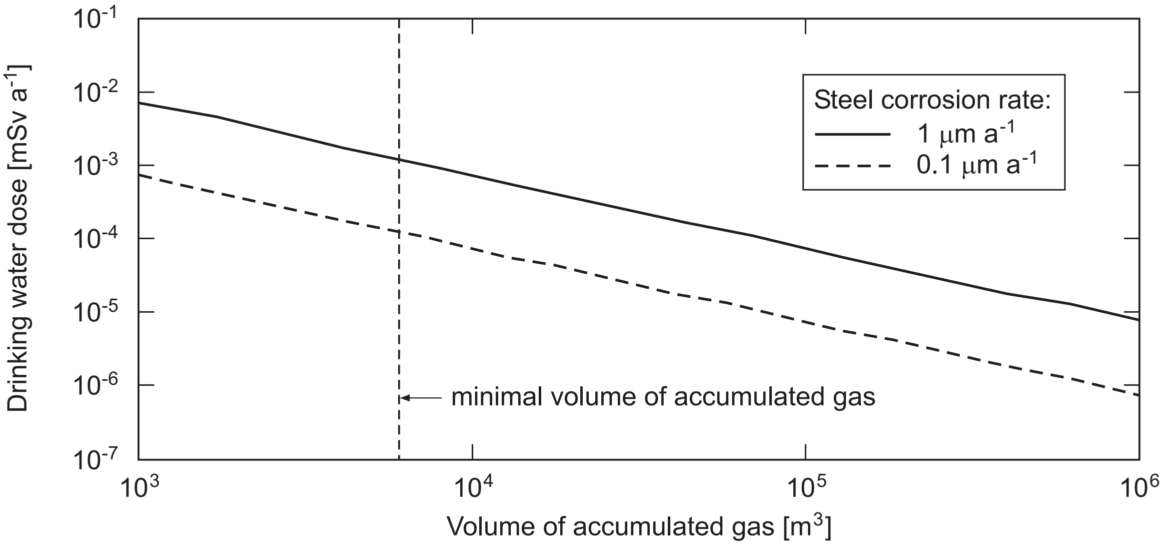

Fig. 6.7-2: Drinking water dose, evaluated using a simple insight model, as a function of accumulated gas and steel corrosion rate

Steel corrosion is the principal source of repository-generated gas.

The effects of differences in transport properties between the backfilled and sealed ramp / shaft and the surrounding Opalinus Clay

In the Reference Conceptualisation, it is assumed that no radionuclides are transported along the tunnels, ramp and shaft. In this section, the barrier efficiencies of the backfilled ramp and shaft and their surrounding excavation-disturbed zones (EDZs) are evaluated separately. The model used to evaluate the barrier efficiencies of these components is described in Nagra (2002c), and involves treating the EDZs as equivalent porous media.

In the case of the ramp, which is backfilled with a 30 % / 70 % bentonite / sand mixture[92] with an assumed hydraulic conductivity of 5×10^{-11} m s^{-1}, water flow and advective / diffusive transport occur predominantly through the backfill material, and flow and radionuclide transport through the EDZ, with an assumed hydraulic conductivity of 10^{-12} m s^{-1}, can be neglected under the assumption that the liner provides no preferential flowpath. Conversely, in the case of the shaft, which is backfilled with pure bentonite with an assumed hydraulic conductivity of 10^{-13} m s^{-1}, water flow and advective / diffusive transport occur predominantly through the EDZ and flow and transport through the backfill can be neglected. Both the ramp and the shaft are lined with cementitious material, but the liner is removed where the sealing plugs are emplaced, and so this is also assumed not to provide a significant pathway for flow and transport.

Fig. 6.7-3 shows the barrier efficiency of the EDZ surrounding the shaft (top) and of the backfilled ramp (bottom) as a function of flow rate, showing how the barrier efficiency for the various example radionuclides decreases with increasing flow rate. For comparison, a hydraulic analysis (Nagra 2002a) yields similar flow rates through the shaft EDZ and through the ramp of about 10^{-3} m^3 a^{-1}.

92 Except for seals, where pure bentonite is used.

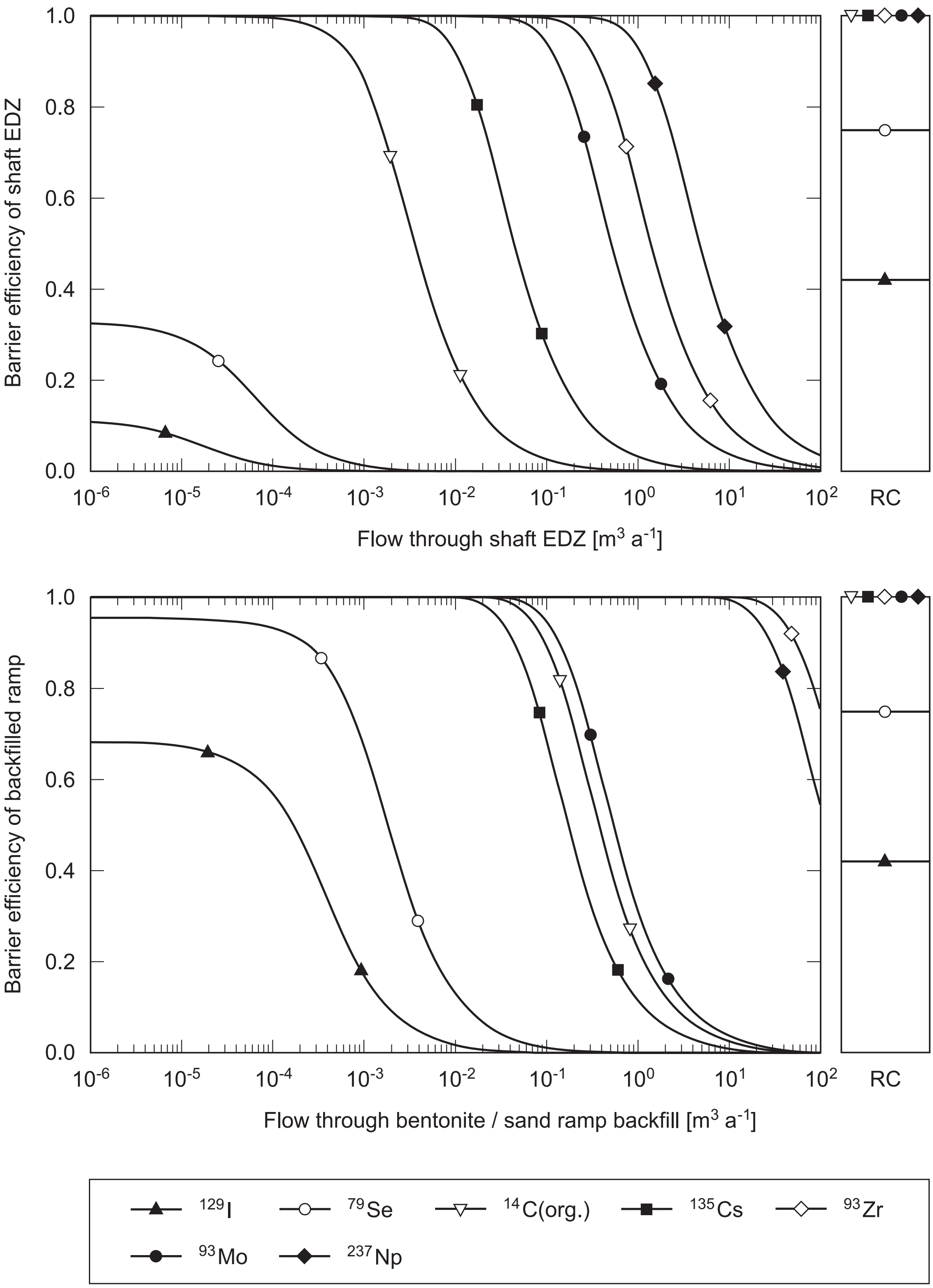

Fig. 6.7-3: The barrier efficiencies of the shaft EDZ (upper figure) and the bentonite / sand backfilled ramp (lower figure) as a function of water flow rate along these potential radionuclide transport pathways

RC (right-hand side of figure) denotes Reference Case values.

For ^{129}I and ^{79}Se (as in the case of the undisturbed Opalinus Clay) the barrier efficiency of the shaft EDZ is low, even with no flow along the pathways. Other radionuclides, however, decay substantially during diffusion-dominated transport. As the flow rate along the shaft EDZ is increased above about 10^{-4} m^3 a^{-1}, and advection becomes more significant, the barrier efficiency for organic ^{14}C begins to fall. As it is increased above about 10^{-2} m^3 a^{-1}, the barrier efficiencies of ^{135}Cs starts to fall significantly. Substantial decay of ^{93}Mo, ^{93}Zr and ^{237}Np occurs until the flow rate along the shaft EDZ is increased to 0.1 m^3 a^{-1} or higher. In the case of the ramp backfill, flow rates above about 0.1 m^3 a^{-1} are required before the barrier efficiencies of organic ^{14}C, ^{135}Cs and ^{93}Mo start to fall. Substantial decay of ^{93}Zr and ^{237}Np occurs until the flow rate along the ramp backfill is increased to about 10 m^3 a^{-1}.

6.7.3 Sensitivity to parameter variations around the Reference Case

6.7.3.1 SF / HLW canister breaching time

Fig. 6.7-4, which is calculated using the reference model chain, shows the effects on the maximum dose and its time of occurrence of varying the SF / HLW canister breaching time independently: i.e. while keeping other parameters held fixed at their Reference Case values. The figure demonstrates the insensitivity of the dose maximum to canister breaching time over a wide range. The dose maximum is dominated by ^{129}I for SF and HLW for all the breaching times considered.

Some sensitivity to canister breaching time might be expected to arise in the case of SF, due to the fact that a long-lived canister favours longevity of the SF matrix. In the conceptualisation of fuel matrix dissolution in the Reference Case, the rate of dissolution is controlled by the presence of radiolytic oxidants, which are not generated until the canister is breached and water contacts the fuel surfaces. A long canister lifetime results in a lower radiation field at the time of breaching, due to decay of the radioactivity of the waste. It thus favours low concentrations of radiolytic oxidants and low fuel matrix dissolution rates[93]. Fig. 6.7-5 shows the proportion of the SF matrix that is dissolved as a function of time, for different canister breaching times. The canister breaching time has only a small effect on the proportion dissolved by, say, 10^6 years, decreasing from 6.6 % for a 10 year canister breaching time, to 6.3 %, 5.6 %, 4.5 % and 2.7 % for 100 years, 1000 years, 10 000 years and 100 000 years breaching times, respectively. The canister breaching time therefore has little effect on the proportion of radionuclides that decay within the SF. ^{129}I provides the greatest contribution to the dose maxima for all canister breaching times considered in the sensitivity analysis (Fig. 6.7-4) and, as noted earlier, the dose maximum is dominated by the IRF in the Reference Case. The proportion of the SF matrix dissolved by 10^6 years is lower than, say, the 9 % ^{129}I instant release fraction assumed for the reference BWR UO_2 fuel, even in the case of early canister breaching shortly after emplacement. This indicates that the dose maxima, due to SF, continue to be dominated by the ^{129}I IRF, irrespective of the time of canister breaching.

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93 Corrosion of the fuel cladding and dissolution of HLW are assumed to proceed at a constant rate per unit area following canister breaching, irrespective of the canister breaching time.

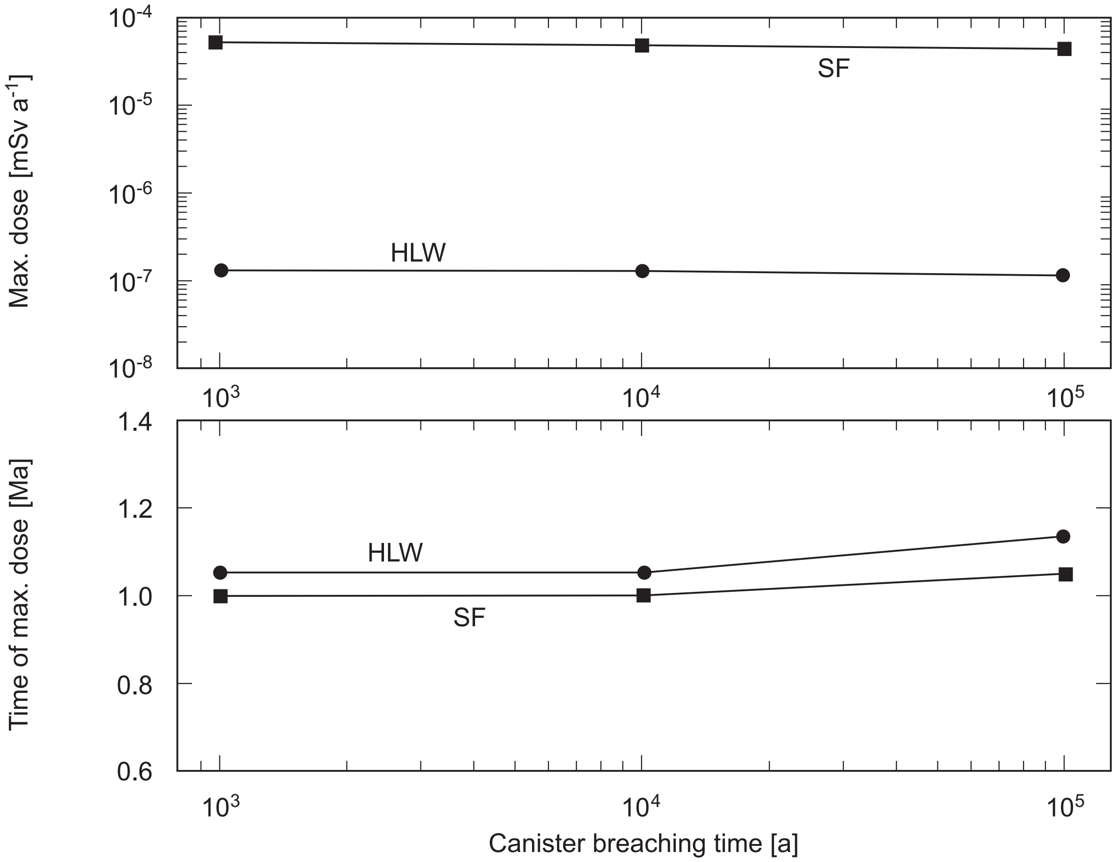

Fig. 6.7-4: Results of dose calculations for SF and HLW for a range of canister breaching times

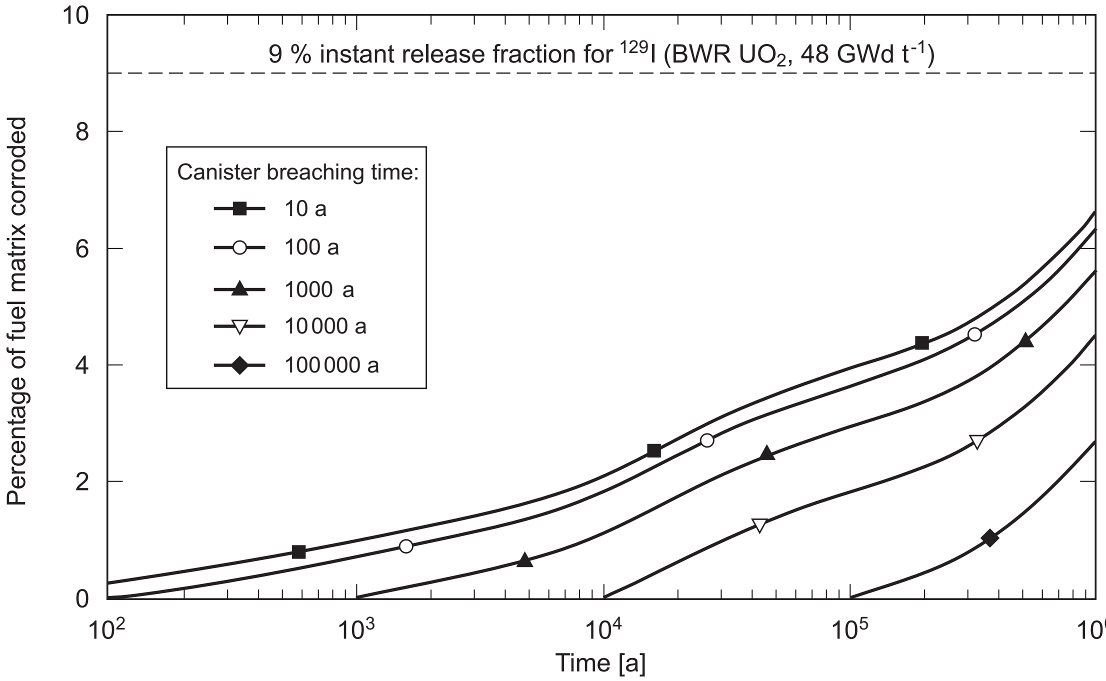

Fig. 6.7-5: The proportion of the SF matrix that is corroded as a function of time for different canister breaching times

These discussions should not be taken to indicate that the canister serves no function. The evidence supporting a prolonged period of complete containment before canister breaching means that compliance with the guidance provided by the Swiss regulatory authorities regarding the need for complete containment of the radionuclides from SF and HLW within the repository during the initial period of high hazard potential[94] can readily be demonstrated. It also makes the modelling of radionuclide transport processes simpler and more reliable, since it avoids the necessity of modelling these processes during the period when the SF and HLW near field is only partly saturated and subject to the transient effects of the thermal and radiation field from the wastes.

6.7.3.2 Groundwater flow through the Opalinus Clay

A number of phenomena could potentially lead to increased water flow through the Opalinus Clay, as discussed in Chapter 5 and summarised in Tab. 5.7-1. Here the objective is to analyse the effects of such increased water flows on radionuclide release and transport. If the assumed rate of groundwater flow through the Opalinus Clay is increased above the Reference Case value, and advection becomes a more significant transport process, two factors affect the resulting release rates:

- the "spreading in time" of near field releases that occurs during diffusion-dominated transport through the bentonite is likely to be reduced, which, in the case of SF, may significantly affect the dose due to the instant release fraction (IRF) of ^{129}I, and,
- the degree of decay during transport will be reduced as transport times become less, and radionuclides may break through the Opalinus Clay that decay substantially during transport in the Reference Case.

Fig. 6.7-6 shows the effects of variations in the specific groundwater flow rate through the Opalinus Clay on the release rates of ^{129}I from SF across the outer boundary of the buffer and across the outer boundary of the Opalinus Clay, calculated using the reference model chain. It is again emphasised that the range of parameter values investigated (groundwater flow rate in this case) extends beyond the range of possibilities based on the discussions in Chapter 5. Release rates are again expressed as doses or, in the case of releases across the outer boundary of the buffer, hypothetical doses. The Reference Case specific flow rate is 2×10^{-14} m s^{-1}.

When the specific flow rate through the Opalinus Clay is low (as in the Reference Case), the result is a broad, spread-out dose curve due to the spreading in time that occurs during transport through the buffer and Opalinus Clay. A comparison of the upper and middle parts of the figure shows the additional spreading that transport in the Opalinus Clay produces. As the flow rate is increased, the transport time through the Opalinus Clay decreases and the amount of spreading is reduced, making the dose curves narrower, with higher maxima. The hypothetical doses from the buffer releases also, but to a lesser extent, become narrower and higher, even though there is no advective transport assumed to occur in the bentonite. This is because a higher rate of groundwater flow reduces the ^{129}I concentration at the bentonite / Opalinus Clay interface and thus increases the concentration gradient, and the rate of diffusion, across the bentonite buffer.

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[94] According to HSK & KSA (1993): "In the case of high-level waste disposal, there is a particularly high hazard potential during the initial phase (around 1000 years). During this phase complete containment of the radionuclides within the repository should be aimed at."

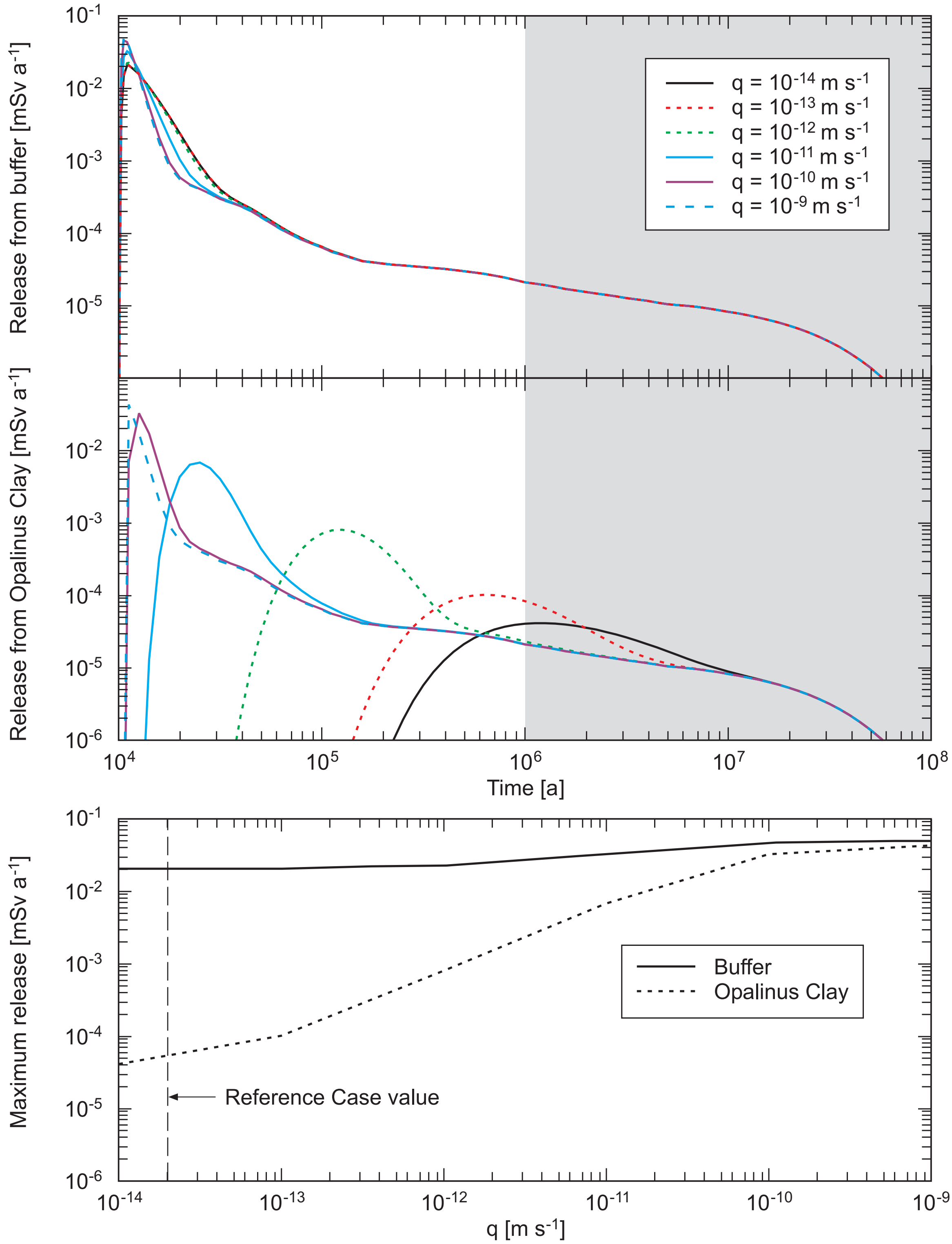

Fig. 6.7-6: Radionuclide release, for SF, from the bentonite buffer (upper figure) and from the Opalinus Clay (middle figure) due to ^{129}I for different values of specific groundwater flow through the Opalinus Clay

Maximum releases from the buffer and from the Opalinus Clay are presented as functions of specific groundwater flow in the lower figure. Releases are expressed in terms of a hypothetical dose as explained in the main text.

The maximum releases from the buffer and from the Opalinus Clay, expressed in terms of hypothetical dose, are presented as functions of specific groundwater flow rate in the lowest figure. The maxima corresponding to releases from the buffer and those corresponding to releases from the Opalinus Clay both increase, and start to converge, as the flow rate is increased. Eventually (at a specific flow rate of about 10^{-10} m s^{-1} or higher) the dose maxima become insensitive to further increases in this parameter, and virtually no spreading occurs during transport through the Opalinus Clay. At these relatively high flow rates, radionuclides are advected away from the bentonite / Opalinus Clay interface so rapidly that there is essentially a zero concentration boundary condition here, irrespective of the exact value of the flow rate.

Considering now the issue of additional radionuclides breaking through the Opalinus Clay, Fig. 6.7-7 shows the distance into the buffer and host rock at which the barrier efficiency reaches 99 % for various radionuclides considered in the safety assessment, and compares the Reference Case flow rate through the Opalinus Clay with the cases in which the flow rate is increased one hundred and one thousand-fold[95] (compare also with Fig. 6.6-2).

The figure shows that such increases in flow rate would mean that ^{232}Th, ^{238}U, ^{235}U, ^{41}Ca and organic ^{14}C, as well as ^{36}Cl, ^{79}Se and ^{129}I, could eventually penetrate the Opalinus Clay without substantial decay. Other radionuclides still do not penetrate the Opalinus Clay to a significant degree. Furthermore, U and Th isotopes still require extremely long periods of time to migrate across the Opalinus Clay. Even in the case of a thousand-fold increase in flow rate, these isotopes require more than 10^9 years to migrate through 40 m of Opalinus Clay, compared to about 10^4 years for ^{36}Cl, ^{79}Se, ^{129}I, ^{41}Ca and organic ^{14}C. Thus, a thousand-fold increase in flow rate could potentially lead to significantly increased breakthrough of organic ^{14}C and ^{41}Ca, whereas other radionuclides will either decay during transport, or will be retained almost indefinitely within the clay.

In Fig. 6.7-8, the combined barrier efficiency of the bentonite and Opalinus Clay (the clay barriers) is plotted against the specific groundwater flow rate through the Opalinus Clay, denoted by q, using the same example radionuclides as in Fig. 6.6-4. The Reference Case value of q is shown by the vertical dotted line. As the groundwater flow rate tends to zero, the barrier efficiency tends towards a limiting value determined by diffusive transport through the clay barriers. The figure shows that, for these example radionuclides, the effect of advection on the barrier efficiency is generally small in the Reference Case. For most of the radionuclides, the barrier efficiency is close to one both in the Reference Case and in the limiting case of zero flow. For ^{129}I and ^{79}Se, the barrier efficiencies are significantly less than one in the Reference Case, and somewhat higher in the limiting case of zero flow. As expected from the discussion of Fig. 6.7-7, the barrier efficiency for organic ^{14}C drops significantly below one if the flow rate through the Opalinus Clay is increased by two to three orders of magnitude or more. Larger increases in groundwater flow (about four orders of magnitude) are needed before decay in the Opalinus Clay becomes ineffective for ^{135}Cs and ^{93}Mo, and for the barrier efficiency of ^{93}Zr and ^{237}Np to start to decline.

For sufficiently large groundwater flow rates through the Opalinus Clay, the barrier efficiency tends towards another limiting value determined by diffusive transport through the bentonite buffer alone, which is always assumed to be impermeable. For the example radionuclides shown in the figure, however, with the exception of ^{237}Np and ^{93}Zr, this limiting value is close to zero.

95 Both are outside the range of possible flow rates expected on the basis of the discussions in Chapter 5.

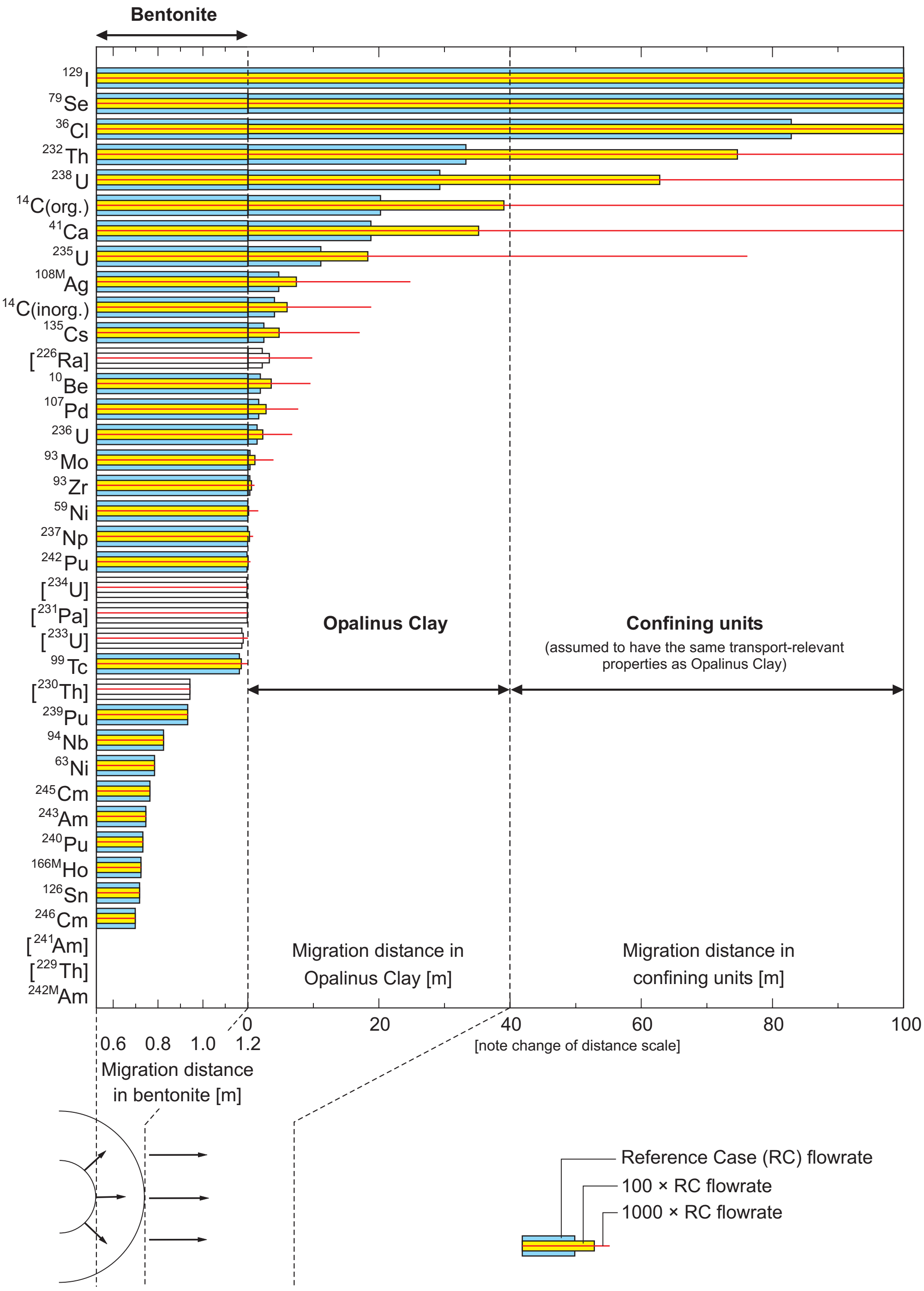

Fig. 6.7-7: Distance into the bentonite buffer and host rock at which, if a steady state is reached, radionuclide transport rates are attenuated by 99 % due to decay, assuming the Reference Case flow rate through the Opalinus Clay (broad bars), as well as 100-fold and 1000-fold increases in specific flow rate (medium width bars and narrow bars, respectively)

Note: As in Fig. 6.6-2, radionuclides such as ^{226}Ra, which are affected by ingrowth during transport, are shown in the figure in parentheses and by non-shaded bars, in order to emphasise that the assumption that ingrowth may be neglected does not hold for these radionuclides.

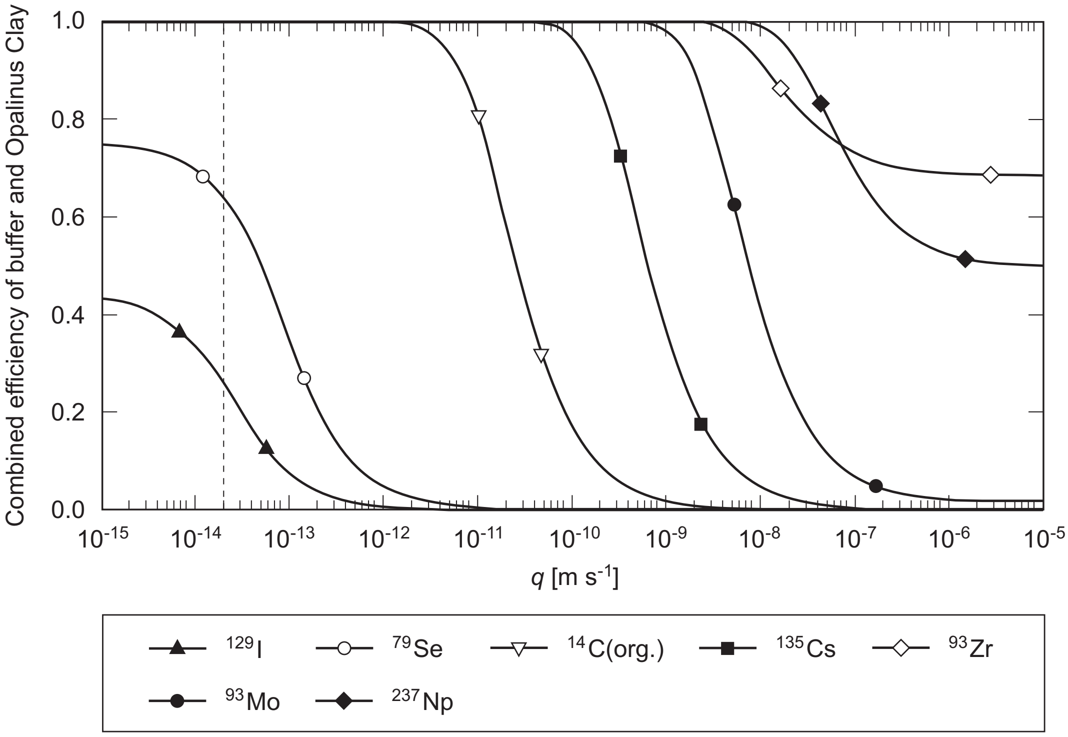

Fig. 6.7-8: The combined efficiency of the bentonite and Opalinus Clay as a function of q, the specific groundwater flow rate through the Opalinus Clay

The Reference Case value of q is shown as a vertical dotted line.

6.7.3.3 Sorption during transport through the Opalinus Clay

The discussions up to now have shown that the Opalinus Clay provides a highly effective barrier to radionuclide transport for all but a few longer-lived and less well sorbing radionuclides. Fig. 6.7-9 illustrates the sensitivity of this transport barrier to sorption. It shows radionuclides on scatter plots of half life ($t_{1/2}$) vs. K_d. The graphs are divided into regions in which the barrier efficiency of the Opalinus Clay is less than 0.99, greater than 0.99 and less than 0.9999 and greater than 0.9999. Anions and other species are plotted separately (in the upper and lower parts of the figure, respectively). Anions differ from other species in their diffusion coefficient and in the amount of porosity that is available to them for diffusion, and therefore also differ in the relationship between barrier efficiency, $t_{1/2}$ and K_d. In addition to the Reference Case values of K_d, the pessimistic values are also shown (Tab. A2.8 of Appendix 2).

The figure shows that radionuclides with half lives of less than about 10 000 years, including organic and inorganic ^{14}C, ^{93}Mo and ^{226}Ra, would decay significantly during transport (> 99.99 %), even if no account were taken of their sorption. Other longer lived radionuclides would be transported through the Opalinus Clay with little decay were it not for the sorption that they undergo. Nevertheless, with the exception of ^{41}Ca, all radionuclides that decay significantly during transport (> 99.99 %) in the Reference Case would require reductions in K_d of more than an order of magnitude (and, for example, for ^{59}Ni, ^{94}Nb and ^{231}Pa, several orders of magnitude) before the degree of decay during steady-state transport becomes less than 99.99 %. If the pessimistic K_d values are used, ^{235}U and ^{238}U decay less than 99 % during steady-state transport (the K_d of uranium is 20 m^3 kg^{-1} in the Reference Case, whereas the pessimistic value is 0.5 m^3 kg^{-1}). Referring to Fig. 6.6-3 and the associated discussion, however, the steady state is not expected to be achieved for uranium for many millions of years.

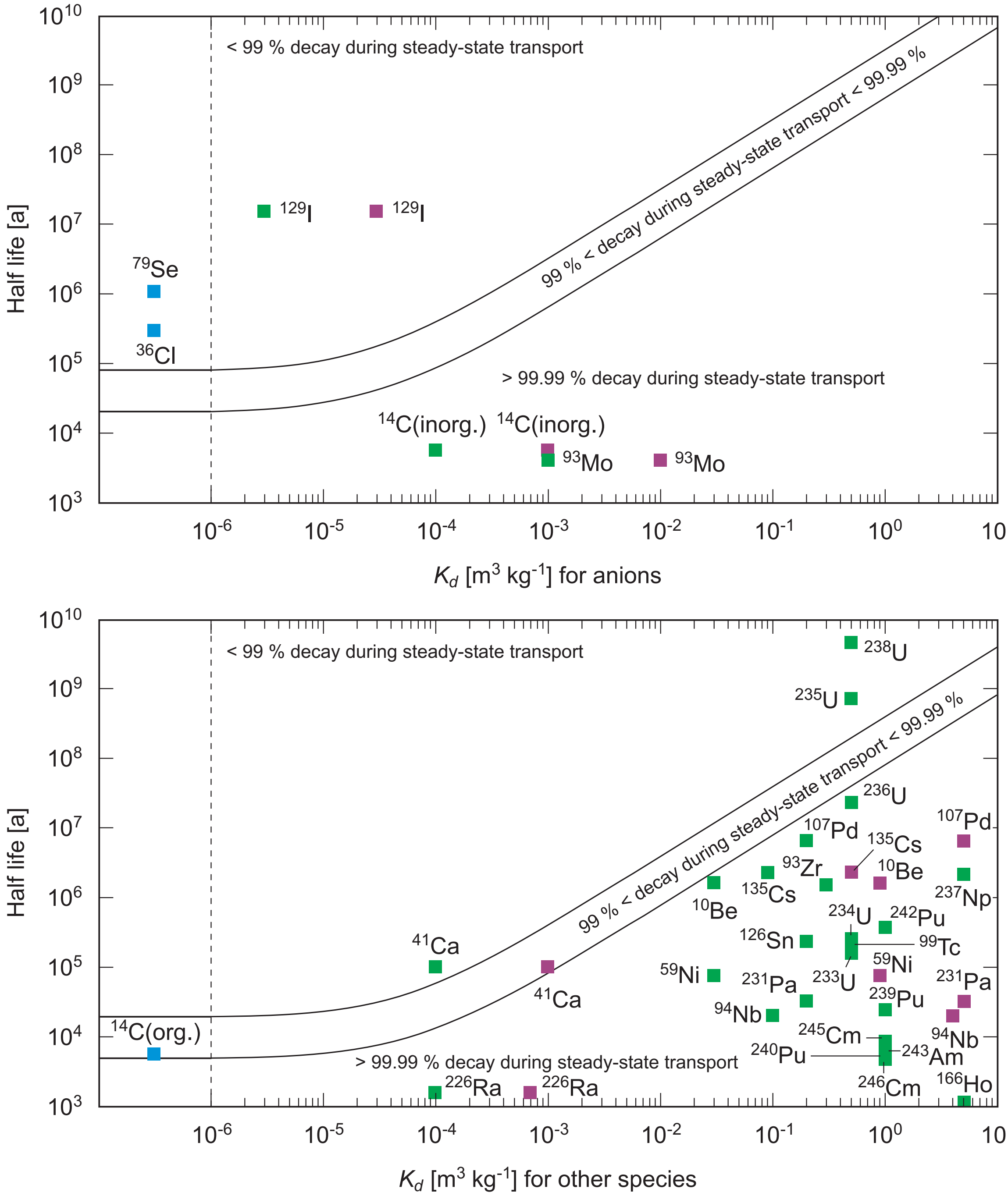

Fig. 6.7-9: Scatter plots of half life vs. Reference Case K_d, showing regions in which the barrier efficiency of the Opalinus Clay is less than 99 %, between 99 and 99.99 % and greater than 99.99 %

Anions and other species are shown in the upper and lower plots, respectively. Magenta points indicate Reference Case values of the sorption coefficient (K_d). Green points correspond to pessimistic values. Blue points are used where the Reference Case and pessimistic values are the same.

6.7.3.4 Sensitivity to assumptions regarding the surface environment

The assumptions used to model the characteristics and evolution of the surface environment may have significant effects on the degree of dilution of radionuclides, and thus on dose. The main uncertainties in the modelling of the surface environment are related to climate, geomorphology at the site of exfiltration and human behaviour. Different (stylised) possibilities for the characteristics and evolution of the surface environment are considered, as discussed in Chapter 2 and 3. Speculations regarding possible changes in the patterns of human behaviour in the future are outside the scope of the present safety assessment, in accordance with regulatory advice.

The sensitivity to assumptions regarding the surface environment is illustrated in Fig. 6.7-10. The quantity on the vertical axis of the figure is the ratio of the individual dose to an exposed individual to the corresponding release rate of radionuclides to the biosphere, termed biosphere dose conversion factor (BDCFs, see Section 6.3.3). This factor is given for a range of safety-relevant radionuclides and biosphere cases. The presented set of radionuclides, a sub-set of all safety-relevant radionuclides[96], is selected by expert judgement based on a consideration of their properties (half-lives, geochemical behaviour, etc.) and the relative importance of their contribution to dose. The range of biosphere cases includes:

- The combination of the reference area and the present-day climate, i.e. Reference Case assumptions.
- Alternative areas: the characteristics of the surface environment based on a sedimentation area and on an area of wetlands.
- Alternative climates: drier / warmer and wetter / warmer than present-day climate.
- Drinking water (Reference Case dilution): i.e. drinking water doses, based on the assumption that drinking water is obtained from a shallow Quaternary aquifer in the reference area, with an assumed dilution rate of 10^6 m^3 a^{-1}.
- Drinking water (spring at valley side): i.e. drinking water doses, based on the assumption that drinking water is obtained from a spring at the side of a larger river valley, with an assumed discharge of 60 l min^{-1} (3.2×10^4 m^3 a^{-1}) and a capture efficiency of 10 %[97].
- Drinking water (Malm aquifer): i.e. doses are drinking water doses, based on the assumption that drinking water is obtained from a deep well, with an extraction rate of 300 l min^{-1} (1.6×10^5 m^3 a^{-1}), which represents a minimal rate for a viable exploitation of a drinking water well, and an assumed capture efficiency of 10 %.

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[96] The procedure for the selection of safety-relevant radionuclides is discussed in NAGRA (2002c).

[97] The capture efficiency is the fraction of radionuclides released from the repository that is captured by a spring or well.

Fig. 6.7-10 indicates that the variability in BDCFs for individual radionuclides is, with a few exceptions, less than three orders of magnitude. This is to a large extent related to variability in the degree of dilution in the surface environment. For most radionuclides, the highest BDCFs are obtained for the biosphere case considering the effects of a dry climate, whereas the Reference Case BDCFs are typically near the middle of the variability range. The lowest BDCFs are obtained for the cases involving high dilution (wetland) or ingestion of drinking water only. It is worthwhile noting that for ^{129}I, which determines the dose maximum in most cases (see Chapter 7), the BDCF for the dry climate case is only about one order of magnitude higher than that for the Reference Case. For other important radionuclides, such as ^{14}C, ^{79}Se and ^{36}Cl, the BDCF is increased by up to a factor of 40, when comparing the dry climate case and the Reference Case. Drinking water BDCFs depend on the ratio of the capture efficiency to the dilution rate. For this reason, the drinking water BDCFs for the reference biosphere area and the deep well in the Malm aquifer are similar, whereas the drinking water BDCFs for the spring at a valley side are higher by about a factor of 5.

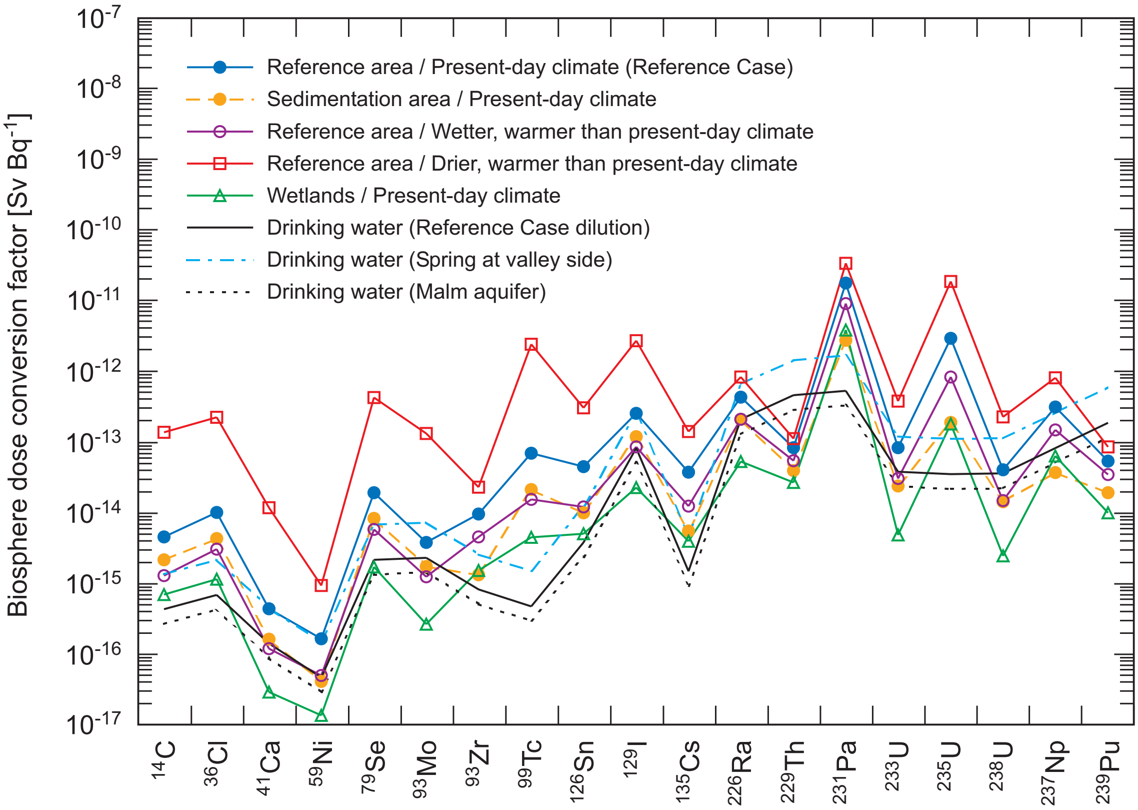

Fig. 6.7-10: Biosphere dose conversion factors for various cases related to uncertainty in conditions in the surface environment

6.7.4 Probabilistic safety and sensitivity analysis

Methodology

In order to investigate the sensitivity of calculated doses to reference model chain input parameters in a comprehensive manner, including the effects of varying all parameters at the same time, the number of individual model chain runs that have to be performed is very large. A systematic methodology is required to select parameter combinations and run the models. This methodology is probabilistic safety / sensitivity analysis (PSA).

In PSA, input parameters for individual runs are selected from probability density functions (PDFs), some of which may be correlated, and a large number of runs are carried out, in order to obtain a distribution of results.

The PSA described below considers sensitivity to parameter variations in the Reference Case near field and geosphere, and does not examine the sensitivity to parameter variations in the biosphere, in accordance with the approach to the treatment of the biosphere discussed in Chapter 2. Doses are calculated using the steady-state Reference Case biosphere dose conversion factors given in Tab. A2.11 of Appendix 2.

Probability density functions

PDFs are selected using several approaches. For some parameters (e.g. solubilities and sorption), the optimistic and pessimistic values in Appendix 2 define the limiting values of the PDF. In some other cases they are based on a critical review of the uncertainties associated with deriving the Reference Case value used in the deterministic calculations. In a small number of cases, expert judgement is applied according to the principle that the PDF should be sufficiently broad, but compatible with general scientific understanding; i.e. they should be physically plausible and chemically plausible. Note that this restriction was deliberately not applied when defining parameter ranges for some of the insight calculations presented in the previous sections, where the aim often included illustrating where a given barrier to radionuclide transport would break down, e.g. for an increasing specific groundwater flow rate (e.g. Fig. 6.7-8).

The principles applied for deriving PDFs are discussed in Appendix 1, and the PDFs and key other data used in the calculations are given in Appendix 2.

Statistical analysis and presentation of results

The probabilistic calculations discussed above result in a large number[98] of dose curves for each waste type. There are numerous possibilities as to how these results can be analysed and presented. The type of analysis selected should be guided by the particular questions or topics that need to be addressed. Tab. 6.7-1 lists typical questions, possibilities for analyses of PSA results that are helpful in addressing these questions, and the benefit of the analyses.

Tab. 6.7-1: Possibilities for analysing and presenting results of PSA calculations

Topic	**Analysis**	**Benefit**
Testing the effect of parameter combinations	Plot of dose curves showing extreme realisations	Confidence in safety and robustness of system
Building system understanding	Analysis of extreme realisations (highest / lowest max. dose)	Confidence in system understanding (important parameter combinations)
Sensitivity study	Scatter plots: max. dose vs. p_i	Confidence in system understanding (relative importance of parameter p_i)
Sensitivity study	Slope: dlog(max.dose)/dlog(p_i)	Confidence in system understanding (relative importance of parameter p_i)
Sensitivity study	Correlation coefficient for parameter p_i with respect to max. dose	Confidence in system understanding (relative importance of parameter p_i , weighted with input PDF for p_i)
Compliance with regulatory criteria (discussed in Chapter 7)	Probability of exceeding a maximal dose, as a function of that dose (complementary cumulative density function, CCDF)	Confidence that even unfavourable, but theoretically possible, parameter combinations do not result in unacceptable doses

98 1000 realisations for each waste type in the present analysis.

In the following paragraphs, the key results from PSA are briefly discussed according to the waste type.

Results for SF

Fig. 6.7-11 shows dose as a function of time for a number of different realisations, including those that gave the highest (sample 164) and the lowest (sample 151) dose maxima out of 1000 realisations. The Base Case is included to allow an easy comparison. The bars beneath the graph indicate the radionuclides that make the highest contribution to dose at any particular time. Key findings are:

- Fig. 6.7-11 shows that of the sampled 1000 realisations, none resulted in a dose curve that was not at least two orders of magnitude below the Swiss regulatory guideline of 0.1 mSv a^{-1}.
- The realisation that gave the highest dose maximum is sample 164. The dose maximum is determined by ^{129}I. This result can be explained by a combination of high groundwater flow rate (~ 9 × the Reference Case value), low sorption constant for ^{129}I in the Opalinus Clay (~ 1/10 × the Reference Case value) and a high IRF for ^{129}I (12 %).
- The realisation that gave the lowest dose maximum is sample 151. The dose maximum is again determined by ^{129}I. This can be explained by a combination of low groundwater flow rate (~ 0.14 × the Reference Case value), a high sorption constant for ^{129}I in the Opalinus Clay (~ 13 × the Reference Case value) and a long transport distance in the host rock (59 m).

These findings are compatible with an analysis of the correlation coefficients between peak dose and the various parameters investigated, which indicates that sensitive parameters, listed according to decreasing absolute values of the correlation coefficient, are:

- the sorption coefficient of ^{129}I in Opalinus Clay,
- the rate of groundwater flow through the Opalinus Clay,
- the instant release fraction for ^{129}I,
- the diffusion coefficient in the Opalinus Clay for anions (^{129}I),
- the length of the geosphere pathway,

All other parameters are significantly less sensitive. However, in order to judge the significance of these parameters for choosing assessment cases, one also needs to take into account the reliability of the corresponding data.

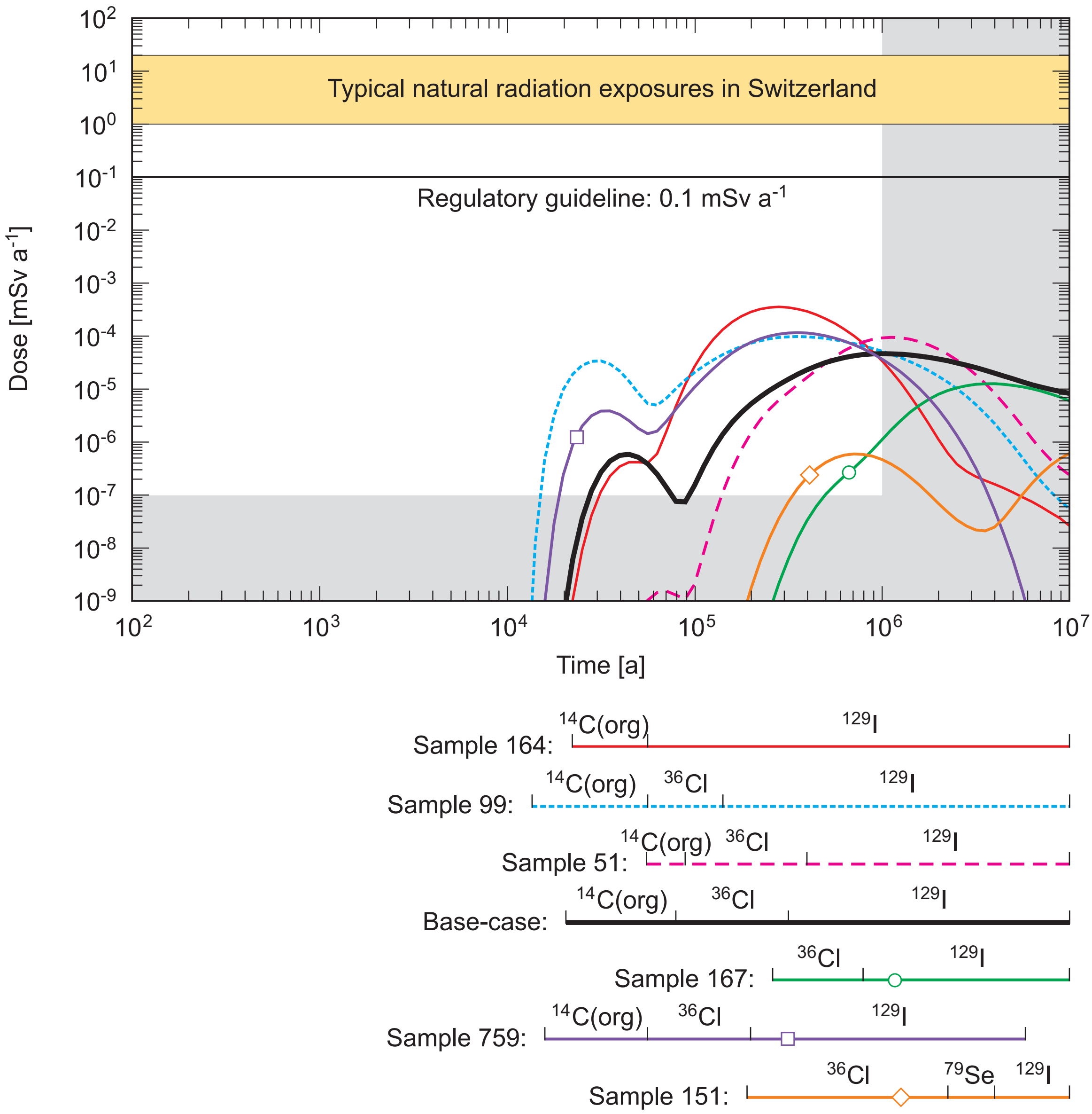

Fig. 6.7-11: Dose as a function of time, for SF, for a number of different realisations, including those that gave the highest (sample 164) and the lowest (sample 151) dose maximum out of 1000 realisations

The Base Case is included to allow an easy comparison. The other curves represent realisations that give the highest dose maximum or the lowest dose maximum in the period between 10^4 and 10^5 years or 10^6 and 10^7 years, respectively. The bars beneath the graph indicate the radionuclides that make the highest contribution to dose at any particular time.

Results for HLW

Fig. 6.7-12 shows dose as a function of time for a number of different realisations, including those that gave the highest (sample 107) and the lowest (sample 813) dose maxima out of 1000 realisations. The Base Case is included to allow an easy comparison. The bars beneath the graph indicate the radionuclides that make the highest contribution to dose at any particular time. Key findings are:

- Fig. 6.7-12 shows that of the sampled 1000 realisations, none resulted in a dose curve that was not at least three orders of magnitude below the Swiss regulatory guideline of 0.1 mSv a^{-1}.
- The realisation that gave the highest dose maximum is sample 107. The dose maximum is determined by ^{79}Se. This result can be explained by a combination of a high solubility for ^{79}Se (2000 × the Reference Case value) and a high groundwater flow rate (~ 9 × the Reference Case value).
- The realisation that gave the lowest dose maximum is sample 813. The dose maximum is determined by ^{79}Se. This can be explained by a combination of a low groundwater flow rate (~ 0.1 × the Reference Case value), high sorption constant for ^{129}I in the Opalinus Clay (~ 13 × the Reference Case value), a long transport distance in the host rock (58 m) and a low diffusion constant (~ 0.7 × the Reference Case value) in the Opalinus Clay.

These findings are compatible with an analysis of the correlation coefficients between peak dose and the various parameters investigated, which indicates that sensitive parameters, listed according to decreasing absolute values of the correlation coefficient, are:

- the solubility of ^{79}Se in the near field,
- the sorption coefficient of ^{129}I in Opalinus Clay,
- the rate of groundwater flow through the Opalinus Clay,
- the diffusion coefficient in the Opalinus Clay,
- the length of the geosphere pathway.

All other parameters are significantly less sensitive. However, in order to judge the significance of these parameters for choosing assessment cases, one also needs to take into account the reliability of the corresponding data.

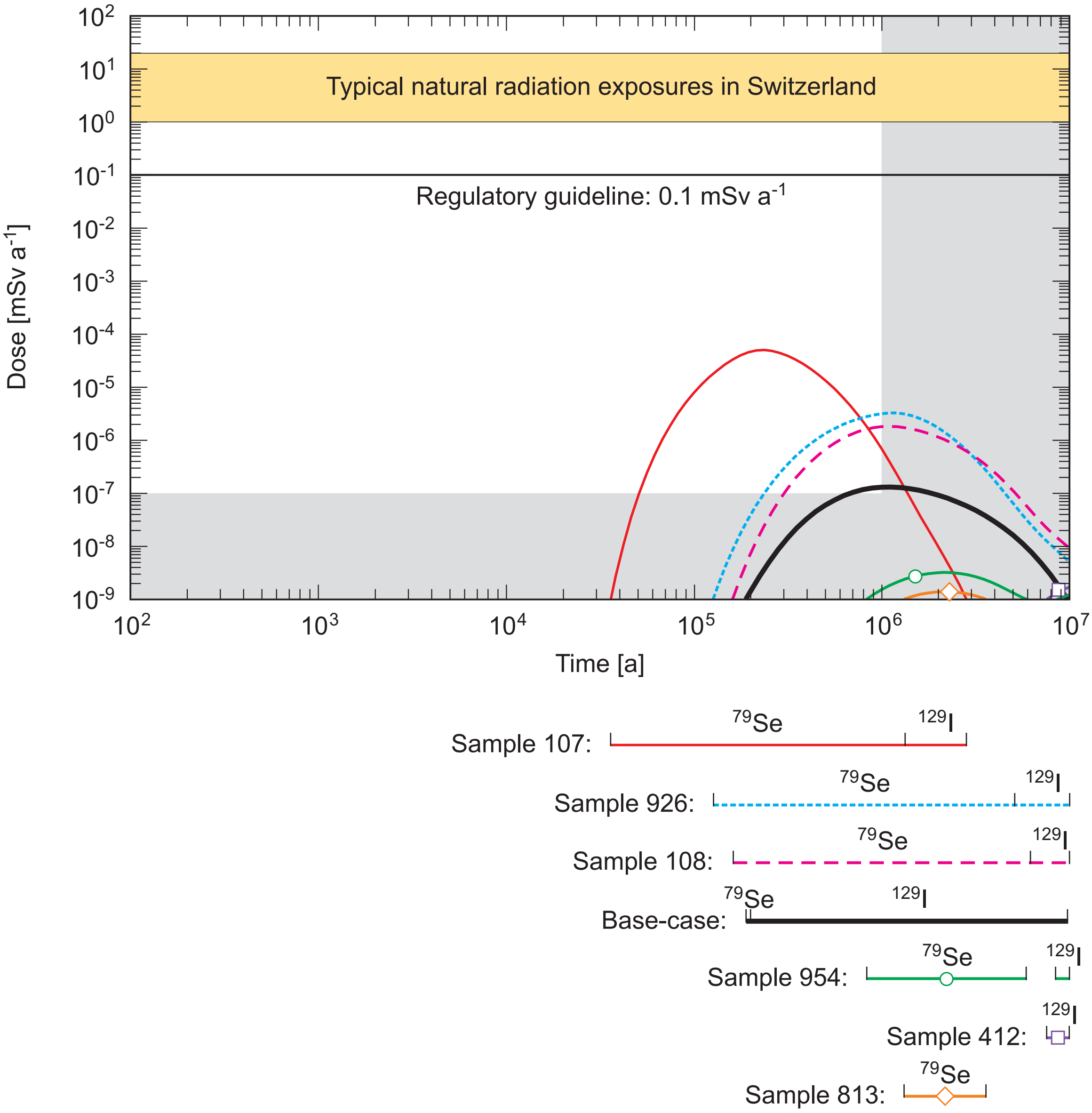

Fig. 6.7-12: Dose as a function of time, for HLW, for a number of different realisations, including those that gave the highest (sample 107) and the lowest (sample 813) dose maximum out of 1000 realisations

The Base Case is included to allow an easy comparison. The other curves represent realisations that give the highest dose maximum or the lowest dose maximum in the period between 10^4 and 10^5 years or 10^6 and 10^7 years, respectively. The bars beneath the graph indicate the radionuclides that make the highest contribution to dose at any particular time.

Results for ILW

Fig. 6.7-13 shows dose as a function of time for a number of different realisations, including those that gave the highest (sample 512) and the lowest (sample 368) dose maxima out of 1000 realisations. The Base Case is included to allow an easy comparison. The bars beneath the graph indicate the radionuclides that make the highest contribution to dose at any particular time. Key findings are:

- Fig. 6.7-13 shows that of the sampled 1000 realisations, none resulted in a dose curve that was not at least three orders of magnitude below the Swiss regulatory guideline of 0.1 mSv a^{-1}.
- The realisation that gave the highest dose maximum is sample 512. The dose maximum is determined by ^{129}I. This result can be explained by a combination of a low sorption constant for ^{129}I (~ 1/10 × the Reference Case value) and a high groundwater flow rate (~ 9 × the Reference Case value).
- The realisation that gave the lowest dose maximum is sample 368. The dose maximum is determined by ^{129}I. This result can be explained by a combination of a low diffusion constant (~ 0.5 × the Reference Case value) in the Opalinus Clay, low groundwater flow rate (~ 0.2 × the Reference Case value), a long transport distance in the host rock (58 m) and a high sorption constant for ^{129}I in the Opalinus Clay (~ 13 × the Reference Case value).

These findings are compatible with an analysis of the correlation coefficients between peak dose and the various parameters investigated, which indicates that sensitive parameters, listed according to decreasing absolute values of the correlation coefficient, are:

- the sorption coefficient of ^{129}I in Opalinus Clay,
- the rate of groundwater flow through the Opalinus Clay,
- the diffusion coefficient in the Opalinus Clay,
- the length of the geosphere pathway,

All other parameters are significantly less sensitive. However, in order to judge the significance of these parameters for choosing assessment cases, one also needs to take into account the reliability of the corresponding data.

Summary of findings

The results of the PSA indicate that, for parameter variations around the Reference Case within the region of parameter space investigated, the system is well behaved, with no sudden or complex changes in performance as parameters are varied.

The findings discussed above for SF, HLW and ILW can largely be explained by the observation from the previous sections of Chapter 6 that, for all waste forms, releases to the biosphere are dominated by a few long-lived, highly soluble and low sorbing radionuclides, particularly ^{129}I, ^{36}Cl, ^{79}Se and organic ^{14}C (Figs. 6.7-11, 6.7-12, 6.7-13). These radionuclides are highly soluble and have low sorption coefficients across the range of geochemical conditions covered by the probabilistic sensitivity analysis, although the solubility limits and sorption coefficients of other radionuclides vary considerably.

The rates at which ^{129}I, ^{36}Cl, ^{79}Se and organic ^{14}C are released to the biosphere are a function of the rate at which they are released from the waste forms, and the degree of spreading in time that occurs during transport. Significant decay during transport is not expected except in the case of ^{14}C (see Section 6.6 and earlier parts of 6.7).

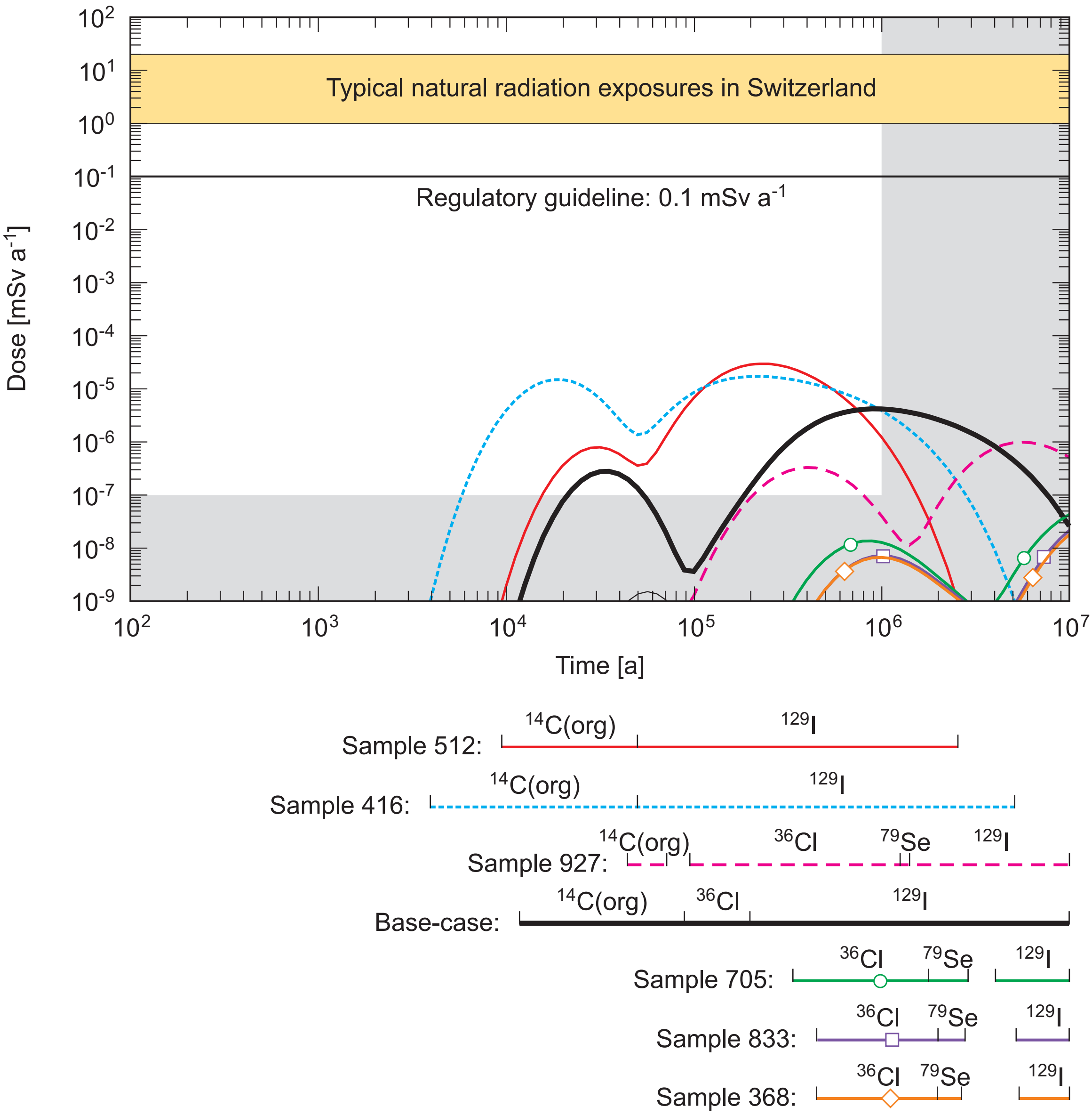

Fig. 6.7-13: Dose as a function of time, for ILW, for a number of different realisations, including those that gave the highest (sample 512) and the lowest (sample 368) dose maximum out of 1000 realisations

The Base Case is included to allow an easy comparison. The other curves represent realisations that give the highest dose maximum or the lowest dose maximum in the period between 10^4 and 10^5 years or 10^6 and 10^7 years, respectively. The bars beneath the graph indicate the radionuclides that make the highest contribution to dose at any particular time.

The degree of spreading in time that occurs during diffusion-dominated transport is clearly a function of the diffusion coefficients in the bentonite buffer (for SF and HLW) and in the Opalinus Clay. The transport path length through the bentonite buffer around the canisters is, however, much less than that through the Opalinus Clay layer (of the order of 1 m, compared to

40 m). Thus, much less sensitivity to the diffusion coefficients in the bentonite than to those in the Opalinus Clay is observed. The degree of spreading that occurs is also a function of the groundwater flow rate that is assumed for the Opalinus Clay, as discussed in Section 6.7.3.2.

The typical timescales characterising diffusive transport across 40 m of Opalinus Clay are greater than the 10 000 years expected period of complete containment in SF and HLW high integrity canisters. Even for ^{129}I, ^{36}Cl, ^{79}Se and other low sorbing radionuclides, the typical timescale for diffusive transport is around a million years[99] (although the beginning of break-through of these nuclides will be much earlier due to the nature of diffusion). This explains the relative insensitivity to canister breaching time. If, for example, the canisters were to be breached earlier than expected, radionuclides that would have decayed during a 10 000 years period of complete containment would in any case decay during transport across the Opalinus Clay.

6.8 Treatment of uncertainty and the identification of assessment cases

6.8.1 Use of expert judgement in the identification of cases

As described in Chapter 3, the impact of various uncertainties on the level of safety provided by the barrier system is illustrated by means of a broad range of representative assessment cases, the analysis of which is described in Chapter 7. The identification of individual cases is a matter of expert judgement, guided by:

- the understanding of the system and its evolution described in Chapters 4 and 5, and
- the understanding of the fate of radionuclides in the Reference Case, and sensitivity to various conceptual assumptions and parameter variations, described in the earlier sections of the present chapter.

The fact that the system is well behaved with respect to parameter variations and shows insensitivity to some conceptual assumptions means that the number of cases can be kept within manageable bounds. Furthermore, some potentially detrimental phenomena, such as the possibility of criticality, can be adequately dealt with by means of side calculations and more qualitative arguments (see Chapter 5).

Efforts have been taken to develop consensus among scientific specialists as to what cases are consistent with scientific understanding. Specifically, a series of meetings with highly qualified scientific specialists from various disciplines (notably geology, hydrogeology, geochemistry, geomorphology, and climatology) were conducted in which assessment cases were presented, with their corresponding conceptualisations, with the aim of eliciting comments (in the form of either agreement or "constructive disagreement").

6.8.2 Relevance and treatment of specific uncertainties

All potentially relevant uncertainties identified in the course of deriving the system concept as discussed in Chapters 4 and 5 are considered, and their effects qualitatively assessed (see Tab. 5.7-1 for a summary). Many of these uncertainties are small and / or their consequences are

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[99] The timescale t_d was defined in Section 6.6.2 by the equation $t_d = L^2 R / D$, where R is a retention coefficient due to sorption, D [m^2 a^{-1}] is the pore diffusion coefficient and L [m] is the transport distance through the Opalinus Clay. In the absence of sorption, $R = 1$. In the Reference Case, $D = 1.7 \times 10^{-11}$ m^2 s^{-1} for anions and 8.3×10^{-11} for other species and $L = 40$ m. This gives values of t_d of about 3 million years for non-sorbing anions and about 600 000 years for other non-sorbing species.

minimised by the selection of the site and the design. Other uncertainties have turned out to be of low relevance in terms of their potential to perturb overall system performance or the evolution of the pillars of safety. Uncertainties that fall into this category are not considered further in defining the assessment cases evaluated in Chapter 7. Others are judged to be potentially important and are taken into account in defining the Reference Case and a broad range of alternative cases.

Tab. 6.8-1 summarises the significance and treatment in assessment cases of uncertainties and design / system options associated with key groupings of FEPs ("Super-FEPs"), with the uncertainties and possible deviations identified in Chapter 5 (Tab. 5.7-1). As explained in Chapter 3, some uncertainties are treated in all the cases that they affect using pessimistic or conservative conceptual assumptions and parameters, whereas others are treated by defining one or more alternative cases, in addition to the Reference Case, in order to explore their radiological consequences. Tab. 6.8-1 indicates how the various uncertainties are treated in both the Reference Case and in any alternative cases that address their consequences. The assessment cases are described only briefly in this table. More complete descriptions are given in Chapter 7 and in the report Nagra (2002c). The numbering scheme for the cases shown in Tab. 6.8-1 refers to Tab. 6.8-2 to facilitate tracing of individual cases.

6.8.3 Grouping of assessment cases

The assessment cases from Tab. 6.8-1 are divided into a number of groups in Tab. 6.8-2 according to the issues or types of uncertainty that they address. The top-level groups (first column in Tab. 6.8-2) correspond to alternative scenarios and explore scenario uncertainty. Each of these groups is further divided according to alternative conceptualisations, which explore conceptual uncertainty (second column in Tab. 6.8-2). Finally, a specific conceptualisation may be evaluated using different parameter sets, thereby exploring parameter uncertainty (third column in Tab. 6.8-2). For this grouping the scheme depicted in Fig. 5.7-1 is being used. This scheme illustrates which of the uncertainties has the potential to significantly affect the disposal system and thus gives a good indication on whether a specific case can be treated as a variant to the Reference Scenario or whether it can lead to a rather different system behaviour and thus requires to be treated within a different scenario. In the following the grouping of assessment cases into different scenarios is briefly discussed.

Reference scenario

The reference scenario comprises assessment cases that are characterised by the release of radionuclides dissolved in groundwater through a system of homogeneous clay barriers of very low permeability, assuming that the system evolves broadly as expected. These cases explore the range of possibilities arising from particular uncertainties affecting the barrier system, where this range can be bounded with reasonable confidence on the basis of available scientific understanding. Despite the broad spectrum of cases considered within the reference scenario, the overall system behaviour is always dominated by the homogeneous low permeability clay barriers.

Alternative scenarios

The alternative scenarios are characterised by a fundamentally different behaviour of the system. They include unlikely but still possible evolutions of the system in which the very slow release of dissolved radionuclides through the clay barriers is severely changed. In the alternative scenarios different release pathways than those through the homogeneous low permeability clay barriers or different mechanisms than the slow advective-diffusive transport of dissolved radionuclides dominate at least a part of the system or some of the radionuclides. Alternative scenarios also address different types of uncertainty (e.g. related to future human actions) which cannot be quantified in the same manner as the uncertainties addressed in the Reference Scenario. Nevertheless, their potential effects still need to be assessed. The following alternative scenarios are considered:

- **Release of volatile radionuclides along gas pathways** – This scenario is distinct from the Reference Scenario because by assuming a continuous gas pathway through the clay barriers with transport of volatile radionuclides along this pathway, the low permeability clay barriers will be bypassed for these radionuclides. Although the transport in a gas pathway without complete dissolution in the porewater is considered to be very unlikely, it cannot be completely ruled out and is evaluated separately for volatile ^{14}C using a different model chain. The results of this separate analysis allow a well-supported evaluation of the relative importance of the release of volatile ^{14}C along gas pathways to be made.
- **Release of radionuclides affected by human actions** – This scenario is distinct from the Reference Scenario because uncertainties related to future human actions are of a different quality compared to other uncertainties which can be bounded with reasonable confidence (e.g. those related to canister lifetime); i.e. any statement about possible future human actions is highly speculative, as discussed in Chapter 2. This scenario is evaluated by defining a few typical (stylised) conceptualisations and assessment cases. It also includes two conceptualisations (borehole penetration, abandoned repository) in which at least for a part of the system the proper functioning of the homogeneous low permeability clay barriers is significantly impaired.

"What if?" cases

"What if?" cases form a separate class of cases because they are set up to test the robustness of the disposal system, while, in contrast to the Reference Scenario and the alternative scenarios, it is acknowledged that they are outside the range of possibilities supported by scientific evidence. To limit the number of cases, they are restricted to those that test the effects of perturbations to key properties of the pillars of safety. However, these cases are not intended to be exhaustive.

Cases exploring the consequences of design or system options

This group of cases is distinct from the other scenarios because the associated uncertainties are of a different quality compared to other uncertainties in that, in principle, the implementer has full control over the selection of the design or system option. A number of design or system options are evaluated in order to assess the effects of alternative design variants (e.g. copper canister) and to demonstrate flexibility to accommodate possible future requirements (e.g. increased waste arisings, different waste specifications).

Illustration of the effects of uncertainties in the biosphere on calculated doses

As discussed in Chapter 2, the uncertainties related to the biosphere are of a different quality compared to other uncertainties, in that, similarly as in the case of future human actions, any statement about them is highly speculative. These uncertainties are also different in that they do not affect the isolation and retention capability of the repository system; they only affect the fate of any radionuclides that are released into the environment. They are evaluated separately from those related to the EBS and the geosphere by analysing a broad range of credible biosphere situations using calculated release rates for the Reference Case near field and geosphere conceptualisations and parameters.

Tab. 6.8-1: The significance and treatment in assessment cases of uncertainties and design / system options associated with specific Super-FEPs

Reserve FEPs and other FEPs that are conservatively omitted in analysing the assessment cases are indicated in italics (see also Section 6.8.4). The cases are assigned numbers and are presented in sequence in Table 6.8-2. The term PDF indicates that a probability density function is defined for probabilistic assessment calculations (see Appendix 2, Table A2.13).

SF				
Super-FEP	**Associated uncertainties and design / system options**	**Significance of uncertainties**	**Treatment of uncertainties in assessment cases (case numbers in parentheses)***	
			Reference Case (1.1a)	**Alternative cases**
Quantities and burnup of fuel and associated radionuclide inventories, including the instant release fraction (IRF)	The possibility for increased nuclear power production	Larger amounts of waste possible	Waste inventory based on 192 GWa(e) scenario, corresponding to 60 a lifetime	Waste inventory based on 300 GWa(e) scenario (5.1a)
	The current trend to higher burnups	A higher average burnup results in a higher IRF – analyses of Reference Case indicate that radionuclides released in the IRF dominate the calculated dose maximum	Burnup of 48 GWd/t_{IHM} assumed (likely / expected)	Parameter variation with respect to Reference Case – increased burnups up to 75 GWd/t_{IHM} (1.1b)
	Chemical state of ^{14}C released from SF	If ^{14}C released is organic, reduced geochemical retardation occurs	^{14}C released in inorganic form	^{14}C released in organic form (1.1k)
	The limited information available on the IRF of high burnup UO_2 fuel and MOX fuel	See above	See above	See above (1.1b) (PDF for IRF)
Corrosion of cladding (see also "The creation of gas pathways", below)	Whether preferential release of organic ^{14}C from the cladding occurs; corrosion rate of cladding	Affects instant release fraction (IRF), time-dependent release rate and degree of geochemical retardation	Constant rate of corrosion assumed, but with some organic ^{14}C assigned to the IRF (pessimistic)	"What if?" case with 10 × higher corrosion rate (4.9a) (PDF for cladding corrosion rate)
Breaching of cladding	The possibility of early fracturing of the cladding, preventing an extended period of complete containment following canister breaching	Sensitivity analyses indicate that the period of complete containment for canisters has small effects on overall system performance, within the range investigated. This is expected to apply to cladding as well	Possibility of extended period of complete containment following canister breaching not included (pessimistic)	No alternative cases defined

* PDF refers to probability distribution functions that have been defined for some parameters

Tab. 6.8-1: (Cont.)

SF				
Dissolution of fuel matrix	Whether the rate of dissolution is controlled by the rate of production of radiolytic oxidants or (if reducing conditions prevail at the fuel surface) by the solubility of U(IV)	If, as expected, reducing conditions prevail at the fuel surface, then the fuel dissolution rate, and radionuclide release rates from the fuel matrix, will be very low, but the IRF will be unaffected	Rate of dissolution is controlled by the rate of production of radiolytic oxidants (conservative, given the pessimistically chosen parameters for this model – see below)	Alternative conceptualisation within the Reference Scenario – rate of dissolution controlled by diffusion and the solubility of U(IV) (1.2a)
	If the rate of dissolution is controlled by the rate of production of radiolytic oxidants, the proportion of the oxidants that is available for reaction at the fuel surface (not recombined, e.g. with radiolytic reductants)	Affects the rate of release of radionuclides from the fuel matrix, but the IRF will be unaffected	Proportionality is assumed between the production rate of oxidants and the dissolution rate of the fuel, with a proportionality constant pessimistically chosen on the basis of wide ranging experimental data and observations from nature	"What if?" case of fuel dissolution rate increased 10 (4.3a) and 100 fold (4.3b) with respect to Reference Case (PDF for fuel matrix dissolution rate)
Criticality	Whether or not it can be completely ruled out	Ruled out by design and supporting calculations	Not considered (discussion of calculation results in Chapter 5)	

HLW glass				
Super-FEP	**Associated uncertainties**	**Significance of uncertainties**	**Treatment of uncertainties in assessment cases (case numbers in parentheses)**	
			Reference Case (1.1a)	**Alternative cases**
Quantities of glass and associated radionuclide inventories	Low – compositions specified by the two reprocessors	Low – unless deviation from specified compositions is large	Inventory based on specified compositions (likely / expected)	No alternative cases defined
Dissolution rate of glass	Long-term rates based on extrapolation of much shorter-term experiments. Creation of additional surfaces for dissolution by fracturing of glass.	Although sensitivity of dose to dissolution rate is expected to be small, a specific assessment case with an increased dissolution rate is analysed because the glass matrix is an important barrier	Long-term rates taken directly from experiments (likely / expected). Fracturing represented in terms of equivalent spheres (see Section 6.3.3).	Pessimistic case with glass dissolution rate increased ~ 100 fold (1.1e) (PDF for glass dissolution rate)

Tab. 6.8-1: (Cont.)

SF / HLW canisters				
Super-FEP	**Associated uncertainties and design / system options**	**Significance of uncertainties and design / system options**	**Treatment of uncertainties and design / system options in assessment cases (case numbers in parentheses)**	
			Reference Case (1.1a)	**Alternative cases**
Breaching of steel canisters	The conditions under which mechanical failure of corroded canisters will occur; possibility of localised corrosion processes	Sensitivity analyses indicate small effects on overall performance, although performance of canisters (a pillar of safety) is affected	Canisters breached at 10 000 a (pessimistic assumption)	Parameter variation with respect to Reference Case – further reduced canister lifetime of 1000 a (1.1c)
	The remote possibility of breaching prior to 1000 a, and the possibility of initially defective steel canisters	Early failure has low probability (1 in 1000 canisters), thus early releases would be small	See above	No alternative cases defined (see, however, "canister material", below)
	Distribution of canister breaching times	Gives rise to attenuation of releases due to spreading in time	All canisters breached simultaneously (conservative). *The spreading of radionuclide releases in time due to the fact that SF/HLW canisters would not be breached simultaneously is omitted (conservative)*	No alternative cases defined
Gas generation by steel canister corrosion	See "The creation of gas pathways", below			
Canister material	Design option of a copper canister with steel insert for SF (or alternative material to mitigate / avoid gas production)	Expected to give an extended period of complete containment and to largely eliminate gas generation in the case of SF	Steel canisters	Copper / steel canisters, which are breached simultaneously at 100 000 a (5.3a), or a case in which one canister has an initial pinhole defect, with full breaching at 100 000 a (5.3b/c)

Tab. 6.8-1: (Cont.)

SF/HLW Near field				
Super-FEP	**Associated uncertainties and design / system options**	**Significance of uncertainties and design / system options**	**Treatment of uncertainties and design / system options in assessment cases (case numbers in parentheses)**	
			Reference Case (1.1a)	**Alternative cases**
The long resaturation time of the repository and its surroundings	The rate of the resaturation process	Delays commencement of corrosion and dissolution processes – likely to be of low importance for SF/ HLW except in the case of earlier than expected canister breaching	*SF/HLW parts of repository are conservatively assumed to be resaturated at time of repository closure (reserve FEP)*	No alternative cases defined
Geochemical immobilisation and retardation in the near field	Thermodynamic data and water chemistry	Relevant to the evaluation of solubility limits and sorption coefficients, and thus to retention and decay in the near field	Solubility limits and sorption coefficients in the near field based on realistic near field geochemical dataset	Parameter variations for solubilities and sorption coefficients in the near field based on pessimistic geochemical dataset (1.1d) and for pessimistic sorption coefficients in the geosphere (1.1i) (PDF for solubilities and K_d values)
	Colloid filtration by bentonite	Small pore size eliminates colloid transport	No colloid transport	No alternative cases defined
	The extent of co-precipitation of radionuclides with secondary minerals derived from SF and glass dissolution and canister corrosion	Co-precipitation could significantly enhance retention and decay in the near field, but the necessary data for modelling is unavailable for most radionuclides	Co-precipitation of radium is taken into account. *Co-precipitation of other radionuclides is conservatively omitted (reserve FEP)*	No alternative cases defined
	The extent of sorption of radionuclides on canister corrosion products	Could significantly enhance retention and decay in the near field, but the necessary data for modelling is unavailable	*Sorption of radionuclides on canister corrosion products conservatively omitted (reserve FEP)*	No alternative cases defined
	The natural concentrations of isotopes in bentonite porewater	Could reduce the effective solubilities of some radionuclides	*Natural concentrations of isotopes conservatively neglected when evaluating whether solubility limits are exceeded and precipitation occurs (reserve FEP)*	No alternative cases defined

Tab. 6.8-1: (Cont.)

SF/HLW Near field				
Super-FEP	**Associated uncertainties and design / system options**	**Significance of uncertainties and design / system options**	**Treatment of uncertainties and design / system options in assessment cases (case numbers in parentheses)**	
			Reference Case (1.1a)	**Alternative cases**
Migration of radiolytic oxidants generated at SF surfaces into bentonite buffer	Degree to which these oxidants are scavenged by reductants in the system	Scoping calculations indicate that only a small proportion of the bentonite would be affected and sensitivity analyses indicate that degradation of properties of a small part of the bentonite is insignificant with respect to overall performance	Oxidants consumed by steel – no migration into the bentonite (likely / expected)	"What if?" case (4.4a) of radiolytic oxidants affecting the bentonite, bounded by a redox front
Thermal alteration of the bentonite buffer adjacent to the SF/ HLW canisters	Existence / extent of any thermally altered region	Some detrimental effects on transport-relevant properties within affected region cannot be excluded; sensitivity analyses indicate that degradation of properties of a small part of the bentonite is insignificant with respect to overall performance	No significant alteration occurs (likely / expected)	Alternative conceptualisation within the Reference Scenario – inner half of bentonite thermally altered, with pore diffusion coefficient (D_p) set equal to that of free water; no effects on sorption (1.3a)
Transport resistances provided by internal spaces (fractures) within the waste forms, by the breached SF/ HLW canisters and by corrosion products	The magnitude and time-dependence of these resistances	Increases radionuclide transport times, but likely to be a small effect compared to the overall transport resistance of the clay barrier (bentonite buffer and Opalinus Clay)	*Transport resistances conservatively omitted except for Cu canister cases (reserve FEP) (5.3b/c)*	No alternative cases defined
Tunnel liner	Design option of concrete or polymer liners for emplacement tunnels in case increased tunnel support required	Effects on bentonite expected to be small	No liner required (likely / expected)	No alternative cases defined
Hydraulic transport characteristics of bentonite	Compaction of bentonite by tunnel convergence	Not significant, as convergence expected to be concurrent with resaturation	Decreased bentonite thickness due to tunnel convergence	No alternative cases defined (PDF for D_e)
Gas transport characteristics of bentonite	Uncertainties in corrosion rate	Water transport unaffected by gas buildup and transport, except in the case of gas-induced porewater displacement from void space in canisters		Alternative conceptualisation within the reference scenario (1.8a)

Tab. 6.8-1: (Cont.)

ILW				
Super-FEP	**Associated uncertainties and design / system options**	**Significance of uncertainties and design / system options**	**Treatment of uncertainties and design / system options in assessment cases (case numbers in parentheses)**	
			Reference Case (1.1a)	**Alternative cases**
Quantities of waste and associated radionuclide inventories	Uncertainty in inventories low, but two different waste specifications considered	Low – unless deviation from specified compositions is large	Inventory based on cemented waste option	Inventory based on high force compaction option (5.2a)
Breaching of ILW steel drums and emplacement containers	The timing of drum / container breaching	Affects period of total containment	*Complete containment by drums / containers omitted (reserve FEP)* – release from ILW delayed until 100 a after emplacement due to slow resaturation of near field	No alternative cases defined
Corrosion / dissolution of ILW	Rate of corrosion / dissolution of waste matrix and rate of release of radionuclides	The delayed release of radionuclides due to the slow corrosion rate of ILW metallic materials (e.g. hulls and ends) is potentially significant	Immediate release of all radionuclides in ILW into near field pore water 100 a after emplacement of wastes. *The delayed release of radionuclides due to the slow corrosion rate of ILW metallic materials (e.g. hulls and ends) is conservatively neglected (reserve FEP).*	No alternative cases defined
Immobilisation and retardation in the ILW near field	Thermodynamic data and water chemistry	Relevant to the evaluation of solubility limits and sorption coefficients, and thus to retention and decay in the near field	Solubility limits and sorption coefficients in the ILW near field based on realistic dataset. *Sorption of nuclides on canister corrosion products conservatively neglected (reserve FEP).*	Parameter variations with respect to Reference Case – solubility limits and sorption coefficients in the near field based on pessimistic geochemical dataset (1.1d, 1.1i) (PDF for solubilities in near field and K_d values in near and far field) "What if?" case for redox front from compacted hulls (4.4a)

Tab. 6.8-1: (Cont.)

ILW				
Super-FEP	**Associated uncertainties and design / system options**	**Significance of uncertainties and design / system options**	**Treatment of uncertainties and design / system options in assessment cases (case numbers in parentheses)**	
			Reference Case (1.1a)	**Alternative cases**
Gas generation	See "The creation of gas pathways", below			
Hydraulic and gas transport characteristics of the ILW near field	Inflow rate of water; displacement of water by gas into rock and along tunnels	Influences rate of gas pressure buildup and timing of gas pathway formation in Opalinus Clay		Alternative conceptualisation within the reference scenario (1.8b)
Compaction of waste/mortar	Timing of convergence arising from void reduction of breached, corroded containers			Alternative conceptualisation dealing with rapid and very slow convergence (1.7a/b)
The long resaturation time of the ILW tunnels	The rate of the resaturation process	Delays commencement of corrosion and dissolution processes	Release from ILW commences at 100 a after emplacement. *The long resaturation time of the repository and its surroundings, which delays the commencement of corrosion and dissolution processes is conservatively neglected (reserve FEP).*	No alternative cases defined

Tab. 6.8-1: (Cont.)

Tunnels / ramp / shaft and seals				
Super-FEP	**Associated uncertainties**	**Significance of uncertainties**	**Treatment of uncertainties in assessment cases (case numbers in parentheses)**	
			Reference Case (1.1a)	**Alternative cases**
Low groundwater flowrate along the sealed tunnels / ramp / shaft, and through the surrounding excavation disturbed zones (EDZs)	The hydraulic conductivity of the EDZs with respect to that of the surrounding undisturbed rock	Relatively high hydraulic conductivity of the EDZ could allow dissolved radionuclides to migrate along the tunnels / ramp / shaft, driven by the natural hydraulic gradient	The self-sealing capacity of the host rock means that any EDZ with enhanced hydraulic conductivity is a transient feature, and can be neglected (likely / expected)	Alternative conceptualisation within the Reference Scenario – release of radionuclides affected by the ramp / shaft and their surrounding EDZs (1.6a/b)
	The driving force for flow along the EDZs provided by tunnel convergence	As above, but with tunnel convergence providing the driving force for migration along the tunnels / ramp / shaft	As above	Alternative conceptualisation within the Reference Scenario – convergence-induced release of radionuclides affected by the ramp / shaft and their surrounding EDZs, two different modelling approaches (1.7a/b)
	The driving force for flow along the EDZs provided by near field gas, as well as the rate of near field gas production (see also "The migration of repository-induced gas", below)	As above, but with near field gas pressure providing the driving force for migration along the tunnels / ramp / shaft	As above	Alternative conceptualisation within the Reference Scenario – gas induced release of dissolved radionuclides affected by the ramp / shaft and their surrounding EDZs, two different waste forms (SF and ILW) and for each waste form, two rates of water flow (1.8a/b)
The seals and the surrounding rock (see also "The creation of gas pathways", below)	The possibility of sealed zones being bypassed by relatively permeable EDZs	The bypassing of seals could allow dissolved radionuclides to migrate preferentially along the tunnels / ramp / shaft, driven by near field gas pressure	Seals remain effective (likely / expected)	"What if?" cases (4.5a/b) of gas-induced release of dissolved radionuclides from ILW along ramp only, two different rates of water flow

Tab. 6.8-1: (Cont.)

Opalinus Clay and confining units				
Super-FEP	**Associated uncertainties**	**Significance of uncertainties**	**Treatment of uncertainties in assessment cases (case numbers in parentheses)**	
			Reference Case (1.1a)	**Alternative cases**
Low groundwater flowrate through undisturbed Opalinus Clay (see also "Low groundwater flowrate along the sealed tunnel / ramp / shaft, and through the surrounding excavation disturbed zones", above)	The groundwater flowrate through the Opalinus Clay, which is affected by uncertainties of about an order of magnitude	Sensitivity analyses indicate that releases from Opalinus Clay of ^{129}I and ^{79}Se are affected by uncertainties in groundwater flow rate	Groundwater flowrate of 2×10^{-14} m s^{-1} (likely / expected)	Parameter variations with respect to Reference Case – groundwater flow rate in Opalinus Clay increased 10-fold (1.1f) and decreased 10-fold due to threshold hydraulic gradient (1.1g) (PDF for ground-water flow rate) "What if?"case (4.1a) 100-fold increase with respect to Reference Case (RC) "What if?" case for no advection in rock (4.8a) "What if?"case (4.7a/b/c) – RC flow rate, 10-fold increase, 100-fold increase combined with pessi-mistic near field/ geosphere geo-chemical data set
	The effects of gas pressure buildup in the near field, tunnel convergence and ice loads on the hydraulic gradient	See above	The effects of gas pressure buildup in the near field, tunnel convergence and ice loads are insignificant (likely / expected)	Alternative conceptualisation within the Reference Scenario – glacially induced flow in the Opalinus Clay (1.4a)
	The eventual increase of hydraulic conductivity caused by erosion of the overburden	Ruled out within the one million year period of primary interest in the safety assessment	Not considered (qualitative discussion in Chapter 5)	
Length of vertical transport path from emplacement tunnels to overlying and underlying formations	Variability in path length	Relevant to transport time through Opalinus Clay and to degree of dispersion	40 m path length (minimum value)	"What if?" case based on 30 m path length (4.11a) (PDF for transport path length)

Tab. 6.8-1: (Cont.)

Opalinus Clay and confining units				
Super-FEP	**Associated uncertainties**	**Significance of uncertainties**	**Treatment of uncertainties in assessment cases (case numbers in parentheses)**	
			Reference Case (1.1a)	**Alternative cases**
Geochemical immobilisation and retardation in the Opalinus Clay and confining units	Sorption mechanisms and groundwater composition, diffusivity	Relevant to the evaluation of diffusion rates and sorption coefficients, and thus to retention and decay in the Opalinus Clay and confining units	Sorption coefficients and diffusivities in the Opalinus Clay based on realistic geochemical dataset	Sorption coefficients in the Opalinus Clay based on pessimistic geochemical dataset (1.1h), pessimistic solubility limits and sorption coefficients in the near field (1.1i) and pessimistic geosphere diffusion constants (1.1j) (PDF for K_d values and diffusion constants in Opalinus Clay) "What if?" case for $K_d = 0$ for ^{129}I in bentonite and Opalinus Clay (4.10a)
	The effectiveness of long-term immobilisation processes (precipitation / co-precipitation) in the geosphere	Could increase time available for decay during transport to the surface environment	*Long-term immobilisation processes omitted (reserve FEP – conservative, except if a change in geochemical conditions leads to remobilisation of previously immobilised radionuclides)*	No alternative cases defined
Homogeneity	The possibility of low transmissivity (i.e. 10^{-10} m^2 s^{-1} or less) undetected discontinuities in the Opalinus Clay	Although not totally excluded on the basis of current scientific under-standing (Chapter 5), these are shown by the sensitivity analyses to have negligible impact on the performance of the Opalinus Clay as a transport barrier, except possibly in the case of organic ^{14}C	No discontinuities with significant transmissivities are present in the Opalinus Clay (likely / expected)	"What if?" cases (4.2a/b/c) of a discontinuity of trans-missivity 10^{-10} m^2 s^{-1} intersecting the repository, variants according to whether ILW or SF/HLW parts of the repository affected and, for SF/HLW, the number of canisters affected
	The possibility of higher-transmissivity discontinuities, faults and repository-induced fractures	Ruled out because of the self-sealing capacity of the clay	Not considered	"What if?" cases for discontinuity with transmissivity 10^{-9} m^2 s^{-1} intersecting SF/HLW near field (4.2d/e) and ILW near field (4.2f)

Tab. 6.8-1: (Cont.)

Opalinus Clay and confining units				
Super-FEP	**Associated uncertainties**	**Significance of uncertainties**	**Treatment of uncertainties in assessment cases (case numbers in parentheses)**	
			Reference Case (1.1a)	**Alternative cases**
Migration of high-pH plume from ILW backfill into Opalinus Clay	The depth of migration, and associated physical and chemical changes in the clay	The maximum possible depth of migration, based on mass balance, is about 4 m (Section 5.4.4). Even if it is hypothetically assumed that this results in a complete loss of geochemical retardation over this distance, the sensitivity analyses show that the impact on the performance of the Opalinus Clay as a transport barrier is small.	Not considered (qualitative discussion in Section 5.4.4)	
Radionuclide transport through the confining units and regional aquifers	Transport-relevant properties of the confining units and regional aquifers	Slow transport could increase time available for decay during transport to the surface environment	Decay during transport through the confining units / aquifers omitted (conservative)	Alternative conceptualisation within the Reference Scenario – decay during vertical transport through the confining units (1.5a) or horizontal transport in local aquifers (1.5b) taken into account

Tab. 6.8-1: (Cont.)

The barrier system (general) Note: uncertainties that significantly affect the broad path of evolution of the barrier system give rise to Alternative Scenarios				
Super-FEP	**Associated uncertainties**	**Significance of uncertainties**	**Treatment of uncertainties in assessment cases (case numbers in parentheses)**	
			Reference Case (1.1a)	**Alternative cases**
The migration of repository induced gas	The possibility that gas pathways are created through host rock and the uncertainty in gas permeability that might allow the transport of ^{14}C in a volatile form with flowing gas	Affects rate of transport of ^{14}C released by SF and ILW which has the potential to form volatile species	Gas pathways unimportant (likely / expected)	Alternative conceptualisation within scenario addressing transport of radionuclides as volatile species – "tight seals" (2.1a/b/c)
	The possibility that the EDZs would allow the tunnels / ramp / shaft to provide gas pathways and the uncertainties in gas permeability	Affects properties of pathways	See above	Alternative conceptualisation within scenario addressing transport of radionuclides as volatile species – "leaky seals" (2.2a/b/c)
	The rate of gas generation by corrosion of the SF/HLW canisters	Affects rate of transport along gas pathways	See above	No alternative cases defined
	Rapid transport of radionuclides as volatile species through continuous gas path	Affects rate of transport along gas pathways	See above	"What if?" case (4.6a-c) in which ^{14}C is transported unretarded through the host rock
	The rate of generation of gas by ILW	Affects rate of transport along gas pathways	See above	No alternative cases defined
	Extent of preferential release of organic ^{14}C from SF cladding	Could affect release of radionuclides with the potential to form volatile species along gas pathways	See above	Some ^{14}C assigned to the IRF (all affected cases)

Tab. 6.8-1: (Cont.)

The barrier system (general) Note: uncertainties that significantly affect the broad path of evolution of the barrier system give rise to Alternative Scenarios				
Super-FEP	**Associated uncertainties**	**Significance of uncertainties**	**Treatment of uncertainties in assessment cases (case numbers in parentheses)**	
			Reference Case (1.1a)	**Alternative cases**
Future human actions	Possibility of future drilling of a borehole inadvertently affecting the barrier system	It is likely that any borehole would either be sealed, or collapse and self-seal, and so would not provide a radionuclide transport path. Most significant if unsealed and supported borehole penetrates the repository.	No borehole is drilled in the future that affects the barrier system	Borehole penetration of the repository is considered as one conceptualisation of the scenario addressing future human actions, with and without a SF canister being directly hit (3.1a-g)
	Possibility of future extraction of drinking water from a deep well drilled into the Malm aquifer and the production rate of such a well should it be created	Could lead to ingestion of radionuclides with less dilution than if they had reached the surface environment	No deep well for drinking water is created in the Malm aquifer	Drinking water extraction from the Malm aquifer, with different capture efficiencies of the well (3.2a/b)
	Possibility of the repository being abandoned before it is backfilled and sealed	Could lead to preferen-tial transport of radionuclides through collapsed access tunnels	The repository is successfully backfilled and sealed as planned	Preferential transport of radionuclides from repository occurs through collapsed tunnels (3.3a)

The surface environment				
Super-FEP	**Associated uncertainties**	**Significance of uncertainties**	**Treatment of uncertainties in assessment cases (case numbers in parentheses)**	
			Reference Case (1.1a)	**Alternative cases**
Climatic evolution	The nature and timing of climate change	Affects dilution in the surface environment and the exposure pathways that are relevant. It also affects geomorphological evolution (see below)	Present-day climatic conditions are assumed to persist for all calculated times (stylised conceptualisation) – also referred to as (6.2a)	Alternative stylised conceptualisations in which a dry warm climate (6.2b), humid warm climate (6.2c) and periglacial climate (6.2d) are assumed to persist for all calculated times
Geomorpho-logical evolution	Properties of the discharge area for groundwater conveying radionuclides	Affects radionuclide transport / uptake in the surface environment	Discharge to section of the Rhine valley where the groundwater table is at some distance from the soil (likely / expected)	Discharge where the groundwater table is close to the soil – either to a sedimentation area (6.1b) or to wetland (6.1c). Discharge to a spring at the side of a valley (6.1d)

Tab. 6.8-2: List of scenarios, "what if?" cases, design and system options and illustration of effects of biosphere uncertainty with associated conceptualisations and parameter variations that define the different assessment cases, structured according to the categories of uncertainty that they address

References to sections in Chapter 7 where the scenario under consideration is discussed are given in brackets.

Alternative scenarios addressing scenario uncertainty	Alternative conceptualisations addressing conceptual uncertainty	Parameter variations addressing parameter uncertainty
1. Reference scenario	1.1 Reference Conceptualisation	1.1a Reference Case (RC)
Release of dissolved radionuclides (7.4)		1.1b Variability in canister inventory
		1.1c Reduced canister lifetime
		1.1d Pessimistic near field geochemical dataset
		1.1e Increased glass dissolution rate in HLW
		1.1f Increased water flow rate in geosphere (10-fold increase)
		1.1g Decreased water flow rate in geosphere (10-fold decrease)
		1.1h Pessimistic geosphere sorption constants
		1.1i Pessimistic near-field and geosphere geochemical dataset
		1.1j Pessimistic geosphere diffusion constants
		1.1k Pessimistic treatment of ^{14}C (organic) in SF
	1.2 Solubility-limited dissolution of SF	1.2a Base Case only
	1.3 Bentonite thermal alteration	1.3a Base Case only
	1.4 Glacially-induced flow in the Opalinus Clay	1.4a Base Case only
	1.5 Additional barrier provided by confining units	1.5a Vertical transport through confining units
		1.5b Horizontal transport in local aquifers
	1.6 Radionuclide release affected by ramp / shaft	1.6a Base Case
		1.6b Increased hydraulic conductivity of EDZ (100-fold increase)
	1.7 Convergence-induced release affected by ramp (ILW)	1.7a Steady-state hydraulics
		1.7b Water pulse
	1.8 Gas-induced release of dissolved radionuclides affected by ramp / shaft	1.8a Base Case
		1.8b Increased water flow rate in ILW
2. Alternative Scenario 1 Release of volatile radionuclides along gas pathways (7.5)	2.1 Release of ^{14}C from SF and ILW as volatile species in the gas phase not affected by ramp / shaft ("tight seals")	2.1a-c Three different gas permeability values
	2.2 Release of ^{14}C from SF and ILW as volatile species in the gas phase affected by ramp / shaft ("leaky seals")	2.2a-c Three different gas permeability values
3. Alternative Scenario 2 Release of radionuclides affected by human actions (7.6)	3.1 Borehole penetration	3.1a-d Penetration of SF/HLW/ILW emplacement tunnel (parameter variations related to the water flow rate through the borehole and the number of canisters affected)
		3.1e-g Direct hit of a SF canister (parameter variations related to the water flow rate through the borehole)
	3.2 Deep groundwater extraction from Malm aquifer (production of well as dilution)	3.2a/b Two different degrees of plume capture efficiency (10 %, 100 %)
	3.3 Abandoned repository	3.3a Base Case only

Tab. 6.8-2: (Cont.)

Alternative scenarios addressing scenario uncertainty	**Alternative conceptualisations addressing conceptual uncertainty**	**Parameter variations addressing parameter uncertainty**
4. "What if?" cases to investigate robustness of the disposal system (7.7)	4.1 High water flow rate	4.1a Increased water flow rate in geosphere (100-fold increase)
	4.2 Transport along transmissive discontinuities	4.2a/b Number of SF/HLW canisters affected ($T = 10^{-10}$ m^2 s^{-1})
		4.2c ILW ($T = 10^{-10}$ m^2 s^{-1})
		4.2d/e Number of SF/HLW canisters affected ($T = 10^{-9}$ m^2 s^{-1})
		4.2f ILW ($T = 10^{-9}$ m^2 s^{-1})
	4.3 SF: Increased fuel dissolution rate	4.3a 10-fold increase with respect to RC
		4.3b 100-fold increase with respect to RC
	4.4 Redox front (SF / ILW compacted hulls)	4.4a Base Case only
	4.5 ILW: Gas-induced release of dissolved radionuclides through the ramp only	4.5a/b Two different water flow rates
	4.6 Unretarded transport of ^{14}C released as volatile species through host rock; retardation in confining units taken into account	4.6a-c Three different gas permeability values
	4.7 Poor near field and pessimistic near field / geosphere geochemical dataset	4.7a RC flow rate
		4.7b 10-fold increase of flow rate
		4.7c 100-fold increase of flow rate
	4.8 No advection in geosphere (diffusive transport only)	4.8a Base Case only
	4.9 SF: Increased cladding corrosion rate	4.9a 10-fold increase with respect to RC
	4.10 Kd(I) for NF and geosphere = 0	4.10a Base Case only
	4.11 Decreased transport distance in Opalinus Clay (30 m)	4.11a Base Case only
5. Design and system options (7.8)	5.1 Increased waste arisings (300 GWa(e))	5.1a Base Case only
	5.2 ILW high force compacted waste option	5.2a Base Case only
	5.3 SF canister with Cu shell	5.3a Canister breaching at 10^5 years
		5.3b Initial defect (small initial pinhole, full breaching at 10^5 years)
		5.3c Initial defect (large initial pinhole, full breaching at 10^5 years)
6. Illustration of effects of biosphere uncertainty (7.9)	6.1 Reference and alternative geomorphology	6.1a Reference area (RC)
		6.1b Sedimentation area
		6.1c Wetland
		6.1d Exfiltration to spring located at valley side
	6.2 Reference and alternative climates	6.2a Present-day climate (RC)
		6.2b Drier/warmer than present-day climate
		6.2c Wetter/warmer than present-day climate
		6.2d Periglacial climate

6.8.4 Conservative omissions and reserve FEPs

A conservative approach in some cases involves deliberately omitting FEPs that are favourable to safety. The reserve FEPs are those that are currently omitted on the grounds of conservatism, but good prospects for improved scientific understanding, models and data means that they may be mobilised at a later stage of the repository programme. FEPs that are conservatively omitted in defining assessment cases, including reserve FEPs, are listed in Tab. 6.8-3.

Tab. 6.8-3: FEPs that are conservatively omitted in defining the assessment cases, including reserve FEPs

It should be noted that the distinction between reserve FEPs and other FEPs that are treated conservatively is a matter of judgement, and some of the latter may in fact turn out to be reserve FEPs that can be mobilised in the future.

Reserve FEPs
The co-precipitation of radionuclides with secondary minerals derived from SF, glass and canister corrosion (except for co-precipitation of radium).
Sorption of radionuclides on canister corrosion products.
Natural concentrations of isotopes in solution in bentonite porewater, which could reduce the effective solubilities of some radionuclides.
Long-term immobilisation processes (precipitation / co-precipitation) in the geosphere.
The long resaturation time of the repository and its surroundings, which delays the commencement of corrosion and dissolution processes (likely to be of negligible importance for SF/HLW except in the case of earlier than expected canister breaching).
The delayed release of radionuclides, due to the slow corrosion rate of ILW metallic materials (e.g. hulls and ends), as well as a period of complete containment by ILW steel drums and emplacement containers.
Irreversible sorption of radionuclides in the near field or in the geosphere (e.g. surface mineralisation).
Degassing of volatile $^{14}CH_4$ in the biosphere.
Other FEPs that are treated conservatively
A period of complete containment provided by the SF Zircaloy cladding following canister breaching (conservatively omitted in all cases).
The conditions under which mechanical failure of the corroded canisters will occur (the Reference Case breaching time of 10^4 years errs on the side of pessimism).
The spreading of radionuclide releases in time due to the fact that SF/HLW canisters would not be breached simultaneously (it is conservatively assumed that all canisters are breached simultaneously, except in cases addressing initial defects in the copper / steel canister design option).
The transport resistance provided by internal spaces (fractures) within the waste forms, by the breached SF/HLW canisters and by corrosion products (conservatively omitted in all cases).
The spreading of radionuclide release in space and time due to the lateral extent of the repository and the three-dimensional nature of diffusive transport (transport paths from the repository to the biosphere are assumed to be one-dimensional in all cases).
The barrier efficiency of regional aquifers (conservatively omitted in all cases).

6.9 Key messages from this chapter

How the system provides safety: The safety concept

The barrier system performs a number of functions relevant to long-term security and safety. These safety functions are:

- isolation from the human environment,
- long-term confinement and radioactive decay within the barrier system, and
- attenuation of releases to the environment.

The system possesses certain "pillars of safety", which are well understood features of the barrier system that are insensitive to perturbing phenomena and make key contributions to the safety functions. These are:

- *the deep underground location of the repository*, in a setting that is unlikely to attract human intrusion and is not prone to disruptive geological events and to processes unfavourable to long-term stability;
- *the host rock*, which has a low hydraulic conductivity, a fine, homogeneous pore structure and a self-sealing capacity, and thus provides a strong barrier to radionuclide transport and a suitable environment for the engineered barrier system;
- *a chemical environment* that provides a range of geochemical immobilisation and retardation processes, favours the long-term stability of the engineered barriers, and is itself stable due to a range of chemical buffering reactions;
- *the bentonite buffer (for SF and HLW)* as a well-defined interface between the canisters and the host rock, with similar properties as the host rock, that ensures that the effects of the presence of the emplacement tunnels and the heat-producing waste on the host rock are minimal, and that provides a strong barrier to radionuclide transport and a suitable environment for the canisters and the waste forms;
- *SF and HLW waste forms* that are stable in the expected environment;
- *SF and HLW canisters* that are mechanically strong and corrosion resistant in the expected environment and provide absolute containment for a considerable period of time.

Although the overall performance of the barrier system is insensitive to canister breaching time, the canister, nevertheless, ensures that compliance with the guidance provided by the Swiss regulatory authorities regarding the need for complete containment of the radionuclides from SF and HLW within the repository during the initial period of high hazard potential can readily be demonstrated. It also makes the modelling of radionuclide transport processes simpler and more reliable, since it avoids the necessity of modelling the early, post-closure period characterised by complex, transient phenomena.

The pillars of safety ensure that the safety functions provide adequate levels of safety in all reasonably foreseeable circumstances. This is the safety concept.

The performance of the barrier system

The safety functions do not necessarily preclude some eventual release of radionuclides to the surface environment, and quantitative modelling of radionuclide transport must be carried out in order to test the adequacy of the performance of the barrier system. The starting point is a Reference Case, based on the likely / expected broad evolutionary path of the barrier system, together with some pessimistic or conservative model assumptions and parameters. An analysis of the Reference Case shows that, for all waste forms:

- the barrier system retains most radionuclides for considerable periods, particularly by immobilisation in the SF and HLW waste forms, by geochemical immobilisation within and around the repository and by the slowness of transport processes in the Opalinus Clay,
- although these processes do not entirely prevent releases to the biosphere, they do allow time for substantial radioactive decay of the majority of radionuclides to occur, and
- only a few radionuclides, namely ^{129}I, ^{36}Cl, ^{79}Se and organic ^{14}C penetrate the clay barriers to an extent whereby the calculated doses exceed 10^{-9} mSv a^{-1}, and these doses are three to six orders of magnitude below the Swiss regulatory guideline.

Of the three waste forms, SF gives the highest evaluated dose maximum. The magnitude of the instant release fraction of ^{129}I and the degree to which it is spread during transport are important factors in determining the magnitude of this dose maximum.

Sensitivity studies

The sensitivity studies show for the individual components of the repository system in the range of the expected parameters a behaviour as expected. This means that for small to moderate deviations of parameter values, system performance will not be significantly affected. Even if larger changes are made, system performance is in general still good. The probabilistic analyses confirm these results; i.e. the system is well behaved and the results are as expected and no complex patterns are observed. This means that the definition and selection of assessment cases can be made by expert judgement based on system understanding. However, it is considered worthwhile that also in the analysis of the assessment cases in Chapter 7 the deterministic calculations are complemented by probabilistic analyses to ensure that no unfavourable parameter combinations are overlooked for key conceptualisations.

Identification of assessment cases

The identification of individual cases is a matter of expert judgement, guided by:

- the understanding of the system and its evolution described in Chapters 4 and 5, and
- the understanding of the fate of radionuclides in the Reference Case, and sensitivity to various conceptual assumptions and parameter variations, described in the earlier sections of the present chapter.

Assessment cases are defined that:

- address the release of radionuclides dissolved in groundwater through a system of homogeneous clay barriers of very low permeability, assuming that the system evolves broadly as expected (the Reference Scenario), exploring the range of possibilities arising from particular uncertainties affecting the barrier system where this range can be bounded with reasonable confidence on the basis of available scientific understanding,
- explore the consequences of the release of radionuclides as volatile species in the gas phase,
- look at different (stylised) possibilities for the release of radionuclides affected by human actions,

- test the robustness of the barrier system ("what if?" cases),
- consider design / system options, and
- deal with different (stylised) possibilities for the characteristics and evolution of the surface environment (the biosphere).

The assessment cases are described in detail and analysed in Chapter 7.

7 Evaluation of the Performance of the Disposal System

7.1 Aims and structure of this chapter

The aim of the present chapter is to evaluate the performance of the disposal system by quantitative analysis of the assessment cases identified in Chapter 6 (Tab. 6.8-2). The broad range of assessment cases and their quantitative assessment illustrates the level of safety provided by the disposal system, taking into account the impact of various potentially detrimental phenomena and uncertainties. Fig. 7.1-1 places the role of the present chapter in the context of the making of the safety case.

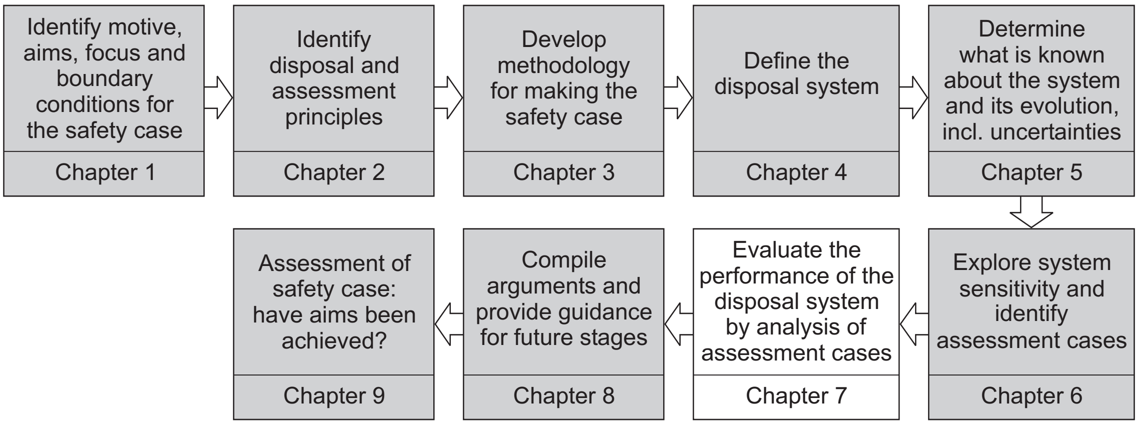

Fig. 7.1-1: The role of the present chapter in the sequence of tasks involved in developing the safety case

The assessment cases are classified according to the broad behaviour of the system, the different classes of uncertainties as well as the design and system options, as discussed in Chapter 6. Chapter 7 is structured accordingly: Section 7.4 addresses the assessment cases analysed in the framework of the Reference Scenario. The Reference Conceptualisation and the application of the reference dataset define the Reference Case. A number of parameter variations of the Reference Case, addressing parameter uncertainty, are analysed. Conceptual uncertainty within the Reference Scenario is investigated by a number of alternative conceptualisations. For each alternative conceptualisation, a Base Case is defined, using realistic or, for some parameters, pessimistic data. In addition, a number of parameter variations are defined for those phenomena where significant parameter uncertainty exists.

Sections 7.5 and 7.6 describe alternative scenarios related to the release of ^{14}C from SF and ILW as volatile species along gas pathways and related to the release of dissolved radionuclides affected by human actions. In both scenarios, a number of alternative conceptualisations and, where significant parameter uncertainty exists, parameter variations are analysed.

Section 7.7 addresses phenomena that are outside the range of possibilities supported by scientific evidence, but involve perturbations to key properties of the pillars of safety. As explained in Chapter 3, this group of "what if?" cases serves to test and illustrate the robustness of the system, but is not intended to be exhaustive.

Design and system options are addressed in Section 7.8.

The effects of uncertainty in the biosphere with respect to geomorphologic and climatic evolutions on calculated doses are illustrated in Section 7.9.

The chapter closes with a summary of results for all investigated assessment cases in Section 7.10, which also contains a comprehensive table with all results.

7.2 Conventions for the presentation of results

The results presented in this chapter are used to assess the performance of the disposal system in the context of regulatory guidelines. The results are presented as annual individual dose[100] to an average member of the critical group due to radionuclides originating from SF, HLW and ILW, the latter being the sum of the waste groups ILW-1 and ILW-2. The radionuclide inventory contained in the pilot facility is included in the corresponding model inventory for SF / HLW / ILW of the main facility. A description of the conventions used to indicate the limitations of the output from the models is given below. In particular, the timescales and ranges of radiological impact require definition so that they are used in context and do not give a misleading impression of the precision in the calculated results. In Chapter 7, the results of certain assessment cases (e.g. all cases involving the release of ^{14}C as volatile species) only include contributions from those release paths that characterise the assessment case under consideration. This approach is considered to help the reader to form a clearer view of the consequences of specific assessment cases. Such cases are marked in the summary table of results (Tab. 7.10-1).

Time is measured from the end of waste emplacement. In Chapter 2, it is argued that the timescale over which the repository system has to provide well-functioning barriers against radionuclide transport is of the order of one million years. This is also the timescale beyond which significant geological changes cannot be ruled out. This is reflected in the dose curves shown in the current chapter by dark background shading of the time interval between 10^6 and 10^7 years.

Natural radiation exposures in Switzerland lie in the range 1 to 20 mSv a^{-1} (BAG 2001), where the higher end of the range is due to unusually high exposures to radon daughters corresponding to the 95-percentile value. This range is shown for comparison with the results of the calculations and with the regulatory guideline.

The lower limit of the dose range shown with a white background is set at 10^{-7} mSv a^{-1}, which is far below the level of insignificant dose set at 0.01 mSv a^{-1} by IAEA (1996). A dose of 10^{-7} mSv a^{-1} to an individual corresponds to a risk of death due to radiation-induced fatal cancer of the exposed person of 5×10^{-12} a^{-1}. This estimate is based on the ICRP recommended risk factor for fatal cancer of 0.05 Sv^{-1} (ICRP 1991).

Many results in the current safety assessment show radiological impacts even below this level. In order to illustrate the behaviour of the models representing the disposal system, the lower limit on the plots has been extended to 10^{-9} mSv a^{-1}. However, such numbers have no radiological meaning. This is reflected in the dose curves shown in the current chapter by dark background shading of the dose interval between 10^{-9} mSv a^{-1} and 10^{-7} mSv a^{-1}.

7.3 Mathematical models and computer codes

The reference model chain STMAN-PICNIC-TAME is used to model the radionuclide release and migration of a range of radionuclides that are judged to be "safety relevant", and the doses

[100] The *annual individual dose*, or short *dose*, is defined in Appendix 5.

to which these give rise. The selection of safety-relevant radionuclides is discussed in Nagra (2002c).

The codes are applied not only to the Reference Case, but also to most of the alternative cases addressed in this chapter. For a few cases, however, alternative conceptualisations are considered that do not fall within the scope of these codes, and alternative codes (e.g. FRAC3DVS) or analytical solutions are employed.

In the following sections, the conceptual models and data are discussed in some detail, and results of the analyses are presented. A full description of conceptual and mathematical models, computer codes and data is given in Nagra (2002c).

7.4 The Reference Scenario

7.4.1 Description of the scenario

In the Reference Scenario the pillars of safety are assumed to operate broadly as expected. In this scenario, the principal phenomena that contribute directly and positively to the performance of the disposal system at different times are

- radioactive decay,
- a period of complete containment by the SF and HLW canisters,
- immobilisation in the waste forms,
- geochemical immobilisation and retardation of radionuclides released from the waste forms,
- limited mobility of dissolved radionuclides and dispersion during transport through the clay barriers, and
- dilution in regional aquifers and in the surface environment.

The Reference Scenario includes the Reference Conceptualisation and a number of alternative conceptualisations where conceptual uncertainty exists (see Fig. 3.7-3). The Reference Case is the assessment case based on the Reference Conceptualisation and the reference dataset. The Reference Case is based, in general, on likely or expected possibilities, currently preferred hypotheses, the reference design / system and a reference, stylised conceptualisation of the biosphere. Parameter variations of the Reference Case are made where significant parameter uncertainty exists. A detailed description of the Reference Scenario, including the Reference Conceptualisation and the Reference Case, is given in Chapter 6.

The results for the Reference Case are discussed in Section 7.4.2.1. The reasons for and the results of the investigated parameter variations made with respect to the Reference Case are presented in Section 7.4.2.2.

In Sections 7.4.3 to 7.4.9, the alternative conceptualisations considered within the framework of the Reference Scenario are discussed in some detail. Each alternative conceptualisation contains a Base Case, defined by the most plausible dataset, and a number of parameter variations where significant parameter uncertainty exists. The presentation of the alternative conceptualisations includes a short introduction which highlights the main differences between the investigated cases and the Reference Case, a discussion of the main conceptual assumptions for the Base Case and the parameter variations, the data used and the results obtained. More details on the models, codes and data used in the different assessment cases can be found in Nagra (2002c).

In all except a few assessment cases, the effectiveness of the confining units as a barrier to radionuclide transport is conservatively neglected[101]. Once radionuclides enter the confining units they are assumed to be transported instantaneously to the exfiltration area where dilution takes place by mixing with flowing water in the Quaternary aquifer[102].

The calculated summed dose maxima (summed over all safety-relevant radionuclides) for the Reference Scenario are summarised in Fig. 7.4-2 (Reference Case and corresponding parameter variations) and in Fig. 7.4-8 (alternative conceptualisations of the Reference Scenario and corresponding parameter variations).

7.4.2 The Reference Conceptualisation

7.4.2.1 The Reference Case

Conceptual assumptions

The Reference Case is based on the assumption that the likely / expected broad evolutionary path of the disposal system is followed (the Reference Scenario), and also on a number of assumptions regarding the conceptualisation for modelling purposes of key FEPs associated with the various system components (the Reference Conceptualisation), together with a reference set of parameters. In the definition of the Reference Conceptualisation and reference parameter set, some pessimistic or conservative conceptual assumptions and parameter values are used.

The conceptual assumptions and parameter values used in the Reference Case near field, geosphere and biosphere are discussed in Chapter 6. To convert calculated near field or geosphere releases into dose, steady-state biosphere dose conversion factors were used in Chapter 6. In contrast to this simplified treatment of the biosphere, in Chapter 7 transient dose calculations are performed using the biosphere code TAME, with time-dependent source terms (Nagra 2002c and 2003b).

The key parameters of the Reference Conceptualisation and their treatment within the Reference Case are summarised in Tab. 7.4-1.

Results for the Reference Case

The calculated doses for SF, HLW and ILW in the Reference Case are shown in Fig. 7.4-1 as a function of time. In the case of SF, the dose curves rise at 2×10^4 years, i.e. about ten thousand years after the time of canister breaching (10^4 years).

[101] The exceptions relate to conceptualisations where the transport pathway in the geosphere model calculations is refined with the purpose of quantifying the effectiveness of the upper and lower confining units as a barrier to radionuclide transport (Sections 7.4.6, 7.5 and 7.7.7).

[102] In the assessment case considering drinking water extraction from a deep well in the Malm aquifer, dilution takes place in the Malm aquifer only, not in the Quaternary aquifer (see Section 7.6.3).

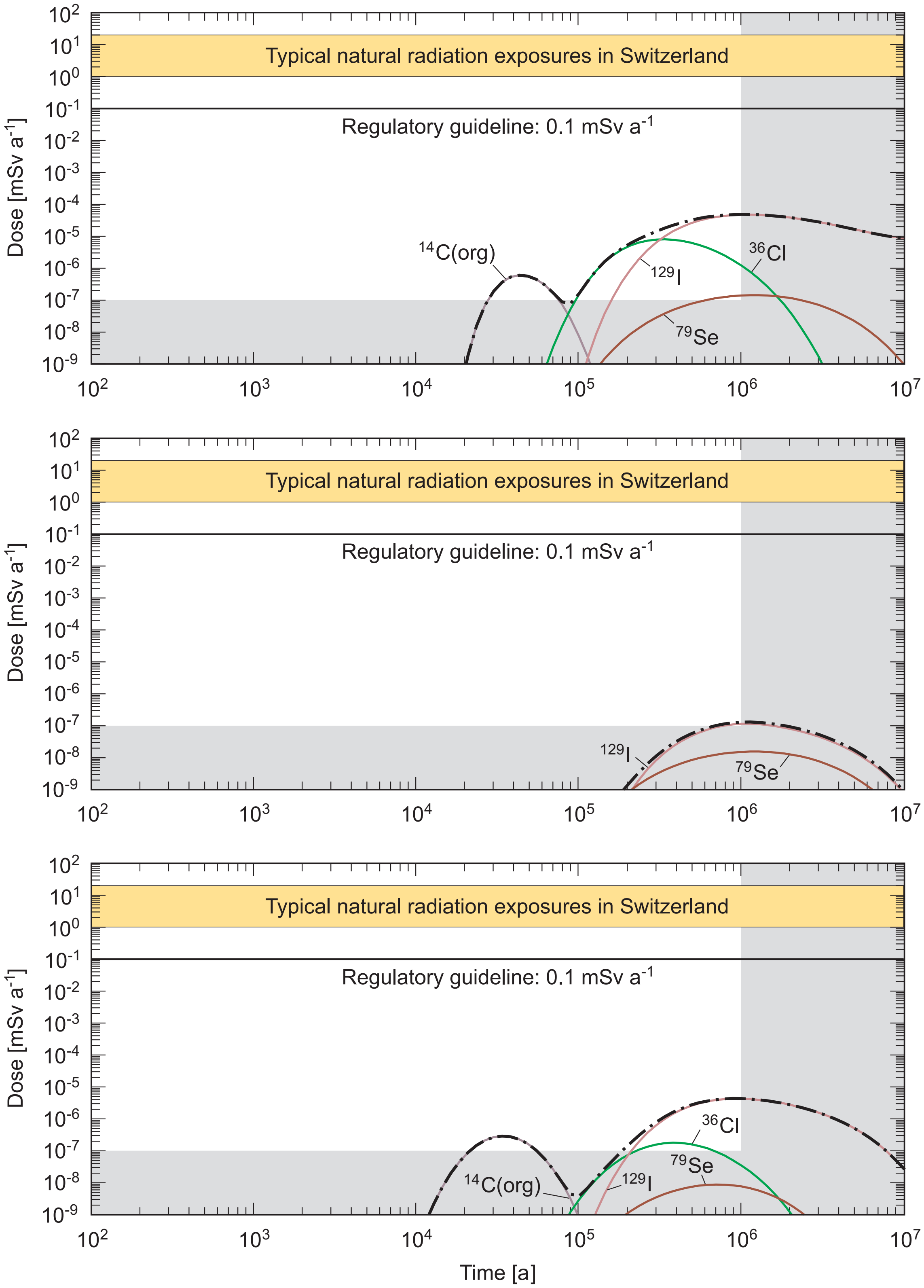

Fig. 7.4-1: Doses for the Reference Case as a function of time

Upper figure: SF, middle figure: HLW, lower figure: ILW (sum of waste groups ILW-1 and ILW-2).

The summed dose maximum is 4.8×10^{-5} mSv a^{-1} for SF, occurs at 1.0×10^{6} years and is dominated by the instant release fraction of ^{129}I and, to a lesser extent, by ^{36}Cl. For HLW, the summed dose maximum is 1.3×10^{-7} mSv a^{-1}; it occurs at 1.1×10^{6} years and is dominated by the fission products ^{129}I and ^{79}Se. In the case of ILW, the summed dose maximum is 4.3×10^{-6} mSv a^{-1}, occurs at 8.9×10^{5} years and is dominated by ^{129}I and ^{36}Cl from ILW-1. The anions ^{129}I, ^{36}Cl and ^{79}Se are transported more slowly in Opalinus Clay than organic ^{14}C because of anion exclusion effects; consequently the dose maximum due to ^{14}C arises earlier than those due to ^{129}I, ^{36}Cl and ^{79}Se. The results show that the summed dose maxima of the Reference Case are more than 3 (SF), nearly 6 (HLW) and more than 4 (ILW) orders of magnitude below the regulatory guideline of 0.1 mSv a^{-1}. The total dose maximum, summed over all radionuclides and the three waste types SF, HLW and ILW, is given in Table 8.2-2.

7.4.2.2 Deterministic parameter variations to the Reference Case

Conceptual assumptions

Tab. 7.4-1 lists the key parameters of the Reference Conceptualisation and the sources of uncertainty and variability. The treatment of key parameters within the Reference Case is discussed. Significant parameter uncertainty or variability is addressed by parameter variations of the Reference Case, within the framework of the Reference Conceptualisation.

Variability in canister inventory

The single canister summed dose maxima calculated for the various reference canisters are in the range $1.6 - 2.8 \times 10^{-8}$ mSv a^{-1} (SF) and $1.6 - 2.1 \times 10^{-10}$ mSv a^{-1} (HLW), see Tab. 7.4-2. As an illustration of the effect of uncertainty in the SF composition, various hypothetical canister loadings have been investigated. The resulting summed dose maxima for a single SF canister are in the range $3.3 - 5.2 \times 10^{-8}$ mSv a^{-1}. The highest dose is obtained for a canister containing 3 PWR UO_2-55 assemblies plus 1 PWR UO_2-75 assembly. The differences in dose are mainly due to differences in inventory and IRF.

Reduced canister lifetime

A reduced SF / HLW canister lifetime of 1 000 years does not significantly affect the radionuclide release rate to the biosphere, because the transport time through the clay barriers is significantly longer than the containment time and because the summed dose maximum is dominated by the instant release fraction of ^{129}I. As discussed in Section 6.7.3, the release rate of ^{129}I is only little affected by changes in the containment time. The summed dose maxima are, therefore, only slightly increased with respect to those of the Reference Case.

Pessimistic near field geochemical dataset

Using pessimistic sorption constants and solubility limits for the near field, the summed dose maximum of SF is less than 10 % higher than in the Reference Case. The dose maximum is dominated by ^{129}I, which is only slightly affected by the changes in the geochemical data base (a bentonite sorption constant of 5×10^{-5} m^3 kg^{-1} instead of the reference value of 5×10^{-4} m^3 kg^{-1})[103]. For HLW, the summed dose maximum, which is now dominated by ^{79}Se, is

[103] Note that in a special case, the sorption values for iodine in bentonite, cement and Opalinus Clay are set to zero ("what if?" case, see Section 7.7.11).

9.9×10^{-6} mSv a^{-1}, about 2 orders of magnitude higher than the Reference Case dose. This is due to the pessimistic assumption regarding the solubility limitation of ^{79}Se (a bentonite solubility of 10^{-5} mol L^{-1} is used instead of the reference value of 5×10^{-9} mol L^{-1}). For ILW, the summed dose maximum is 4.3×10^{-6} mSv a^{-1}, which is identical to the Reference Case dose, for similar reasons as in the case of SF.

Tab. 7.4-1: Key parameters of the Reference Conceptualisation that are subject to uncertainties/variability and treatment of key parameters within the Reference Case and within parameter variations

Parameter	Source of uncertainty/ variability	Reference Case	Parameter variations
Canister inventory	The burnup and type of SF (affecting instant release fraction and fuel dissolution)	Three reference SF canister loadings have been defined: - 9 BWR UO_2 fuel assemblies - 4 PWR UO_2 fuel assemblies - 3 PWR UO_2 assemblies plus 1 PWR MOX fuel assembly, all with a burnup of 48 GWd/t_{IHM}. For HLW, two reference canister loadings exist: COGEMA and BNFL. In the Reference Case, there are 935 canisters with BWR UO_2 fuel, 680 with PWR UO_2 fuel, 450 with PWR UO_2/MOX fuel, 460 with COGEMA glass and 270 with BNFL glass (rounded numbers).	A number of hypothetical SF canister loadings are considered in order to illustrate the impact of higher burnups and fuel type: - 4 PWR UO_2 assemblies, three of which have a burnup of 55 GWd/t_{IHM} and the fourth 55 GWd/t_{IHM} (case 1), 65 GWd/t_{IHM} (case 2) and 75 GWd/t_{IHM} (case 3) - PWR UO_2/MOX fuel, containing three UO_2 assemblies with a burnup of 48 GWd/t_{IHM} plus 1 MOX assembly with a burnup of 65 GWd/t_{IHM}.
Canister lifetime	Corrosion modes (e.g. uncertainties regarding stress corrosion cracking) and start of corrosion	All SF/HLW canisters are assumed to be breached simultaneously after 10^4 a	All SF/HLW canisters are hypothetically assumed to be breached simultaneously after 10^3 a
Near field geochemical data	The chemical processes affecting sorption constants and solubility limits for some elements in the near field	The Reference Case near field geochemical data is based on realistic assumptions with regard to chemical processes affecting sorption constants and solubility limits	The effects of more pessimistic sorption constants and solubility limits are considered
HLW dissolution rate	The corrosion rate of the glass is principally determined by temperature and solution composition. Long-term extrapolation of laboratory data leads to significant uncertainties.	The glass dissolution rates for HLW are 5.5×10^{-4} kg m^{-2} a^{-1} and 7.3×10^{-5} kg m^{-2} a^{-1} for BNFL and COGEMA glasses, respectively (Section 5.3.4.6)	An extremely pessimistic glass dissolution rate of 4.0×10^{-2} kg m^{-2} a^{-1} is used for both BNFL and COGEMA glasses
Water flow rate in the Opalinus Clay	There is some uncertainty in the measured hydraulic conductivity and gradient	The Reference Case flow rate is based on an upwardly directed Darcy flow of 2×10^{-14} m s^{-1}	As parameter variations, the vertical Darcy flow rate in the Opalinus Clay is assumed to be increased and decreased by a factor of 10
Opalinus Clay sorption parameters	The chemical processes affecting sorption constants for some elements in the host rock	The Reference Case is based on a realistic set of sorption constants	The effects of more pessimistic sorption constants are considered
Opalinus Clay diffusion constant	The processes controlling diffusion for some elements in the host rock	The Reference Case is based on a realistic set of effective diffusion constants	The effects of more pessimistic effective diffusion constants are considered
Organic carbon	The speciation of ^{14}C released from SF, affecting the geochemical behaviour of ^{14}C	^{14}C released from cladding is considered to be organic, whereas ^{14}C released from the waste matrix is assumed to be inorganic	All ^{14}C released from SF is assumed to be organic

Tab. 7.4-2: Summed dose maxima for a single canister, containing various reference and alternative (hypothetical) canister loadings for SF (reactor type – burnup in GWd t_{IHM}^{-1}) and HLW

Canister loading		**Summed dose maximum for a single canister [mSv a^{-1}]**
Reference SF canisters	9 BWR UO_2-48	2.8×10^{-8}
	4 PWR UO_2-48	1.6×10^{-8}
	3 PWR UO_2-48 + 1 PWR MOX-48	2.5×10^{-8}
Alternative SF canisters (hypothetical loadings)	4 PWR UO_2 -55	3.6×10^{-8}
	3 PWR UO_2-55 + 1 PWR UO_2-65	4.4×10^{-8}
	3 PWR UO_2-55 + 1 PWR UO_2-75	5.2×10^{-8}
	3 PWR UO_2-48 + 1 PWR MOX-65	3.3×10^{-8}
Reference HLW canisters	COGEMA	1.6×10^{-10}
	BNFL	2.1×10^{-10}

Increased glass dissolution rate in HLW

The results for an extremely pessimistic value of the glass dissolution rate (about 100-fold increase) show that the summed dose maximum of HLW is only slightly higher than the dose of the Reference Case. This is due to the fact that many key elements are solubility limited and that the summed dose maximum of HLW is primarily controlled by the effectiveness of the Opalinus Clay as a barrier to radionuclide transport.

Increased water flow rate in geosphere

Because radionuclide transport at such low flow rates is strongly affected by diffusion, a 10-fold increase in water flow rate does not cause a linear increase in the total transport rate. The summed dose maxima are 1.9×10^{-4} mSv a^{-1} (SF), 4.3×10^{-7} mSv a^{-1} (HLW) and 1.8×10^{-5} mSv a^{-1} (ILW), about a factor of 3-4 higher than in the Reference Case.

Decreased water flow rate in geosphere

In the case with a 10-fold reduced Darcy velocity, radionuclide transport is dominated by diffusion. The resulting summed dose maxima are only marginally decreased with respect to the Reference Case (by less than a factor of 2).

Pessimistic geosphere sorption constants

The results for the pessimistic geosphere sorption database show that the summed dose maxima for SF, HLW and ILW are increased by about a factor of 2 compared to the Reference Case dose. This is because the dominating radioelements remain the same within the sorption ranges considered and are either weakly sorbing (I) or non-sorbing (Se, Cl) in the reference sorption dataset for the Opalinus Clay.

Pessimistic near field and geosphere geochemical dataset

Using pessimistic sorption constants and solubility limits for the near field, combined with pessimistic sorption constants for the geosphere, the summed dose maxima are 1.0×10^{-4} mSv a^{-1} (SF), 1.0×10^{-5} mSv a^{-1} (HLW) and 8.7×10^{-6} mSv a^{-1} (ILW). For the reasons discussed above, the doses for SF and ILW are increased by about a factor of 2, whereas for HLW, the dose is increased by 2 orders of magnitude compared to the Reference Case.

Pessimistic geosphere diffusion constants

Using pessimistic effective diffusion constants for the geosphere, the summed dose maxima are 3.5×10^{-4} mSv a^{-1} (SF), 2.8×10^{-7} mSv a^{-1} (HLW) and 1.7×10^{-4} mSv a^{-1} (ILW). These maximal doses are a factor of 7, 2 and 40 higher than the Reference Case dose maxima, respectively. In the case of SF and ILW, this increase is due to the contribution of organic ^{14}C, the dose maximum of which arises significantly earlier than in the Reference Case.

Pessimistic treatment of ^{14}C (organic) in spent fuel

Assuming all ^{14}C in spent fuel to be organic, the summed dose maximum, which is due to ^{129}I, is unchanged with respect to the Reference Case. However, the early peak in the dose curve due to ^{14}C is slightly increased.

Results for the parameter variations to the Reference Case

The range of summed dose maxima calculated for the parameter variations of the Reference Case are shown in Fig. 7.4-2, together with the summed dose maxima for the Reference Case.

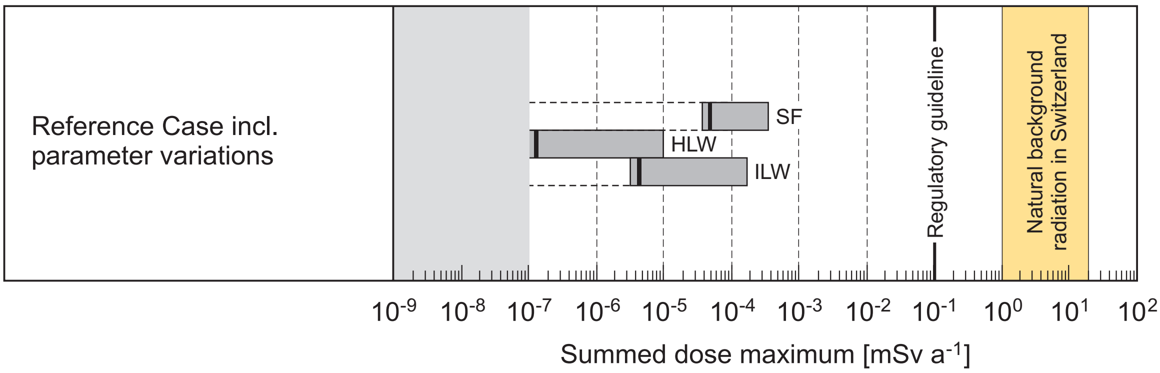

Fig. 7.4-2: Summed dose maxima for SF, HLW and ILW for the Reference Case (bold lines) and ranges of summed dose maxima associated with the different parameter variations considered

7.4.2.3 Probabilistic parameter variations to the Reference Case

As discussed in Section 6.7.4, probabilistic analyses are performed in addition to the deterministic analyses in order to investigate the sensitivity to reference model chain input parameters in a comprehensive manner. This specifically includes scoping the effects of varying all parameters at the same time to further enhance confidence that no critical parameter combination has been overlooked.

Fig. 7.4-3a shows the complementary cumulative density function (CCDF[104]) of the dose for the probabilistic analysis of the Reference Conceptualisation. The input PDFs are given in Appendix 2 (Tab. A2.13). Note that in contrast to the deterministic Reference Case parameter variations, the PDFs also contain increased cladding corrosion rates and that for all parameters varied, values more optimistic than reference values are also included. Moreover, in the probabilistic calculations, the upper truncation for the effective diffusion constant in the Opalinus Clay has been set to 3 times the Reference Case value; whereas in the deterministic calculations, the pessimistic value is increased by a factor of 3 (anions) and 10 (non-anions) with respect to the Reference Case.

In Fig. 7.4-3b, the evolution of the median, the 95th percentile and the highest / lowest dose maxima of all samples are given.

The results for all three waste types are well below the Swiss regulatory guideline[105].

7.4.3 Solubility-limited dissolution of spent fuel

Main difference to Reference Case

In the Reference Case, a radiolytic model is used to calculate the dissolution rate of spent fuel. In this alternative conceptualisation, the effect of solubility limitation on spent fuel dissolution is explicitly taken into account. Solubility limitation in the bentonite porewater significantly reduces the rate of fuel matrix dissolution, because the dissolution rate is controlled by the rate of transport of U(IV) through the clay barriers (bentonite, Opalinus Clay) rather than by radiolytic oxidation at the fuel surface. Transport of radionuclides of the instant release fraction and dissolution of cladding are not affected by this alternative dissolution model.

Conceptual assumptions

The calculations for spent fuel are performed in two steps:

- In a first step, the near field release of U(IV) is calculated, assuming instant dissolution of the fuel matrix, and considering the solubility limitation in the pore space adjacent to the fuel matrix, sorption and diffusive transport in the bentonite and radioactive decay.
- The time dependent near field release rate of U(IV) calculated in step 1 is then used as the effective dissolution rate of the fuel matrix in a second calculation, considering shared solubility limitation in the pore space adjacent to the fuel matrix, sorption and diffusive transport in the bentonite and radioactive decay for all safety-relevant radionuclides. The treatment of the cladding and gap inventory is identical as in the Reference Case.

No parameter variations are performed.

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[104] The CCDF is the probability of exceeding a given dose, as a function of that dose.

[105] The regulatory constraint at low probabilities in the figures showing the results of the probabilistic analyses in the present chapter is derived from HSK's Protection Objective 2 (risk limit of 10^{-6} a^{-1}, see Chapter 2), using the risk factor for fatal cancer of 0.05 Sv^{-1} recommended by the ICRP (ICRP 1991). Thus for a probability of, e.g., 10^{-2} of receiving an annual dose the corresponding dose limit is 2 mSv a^{-1}.

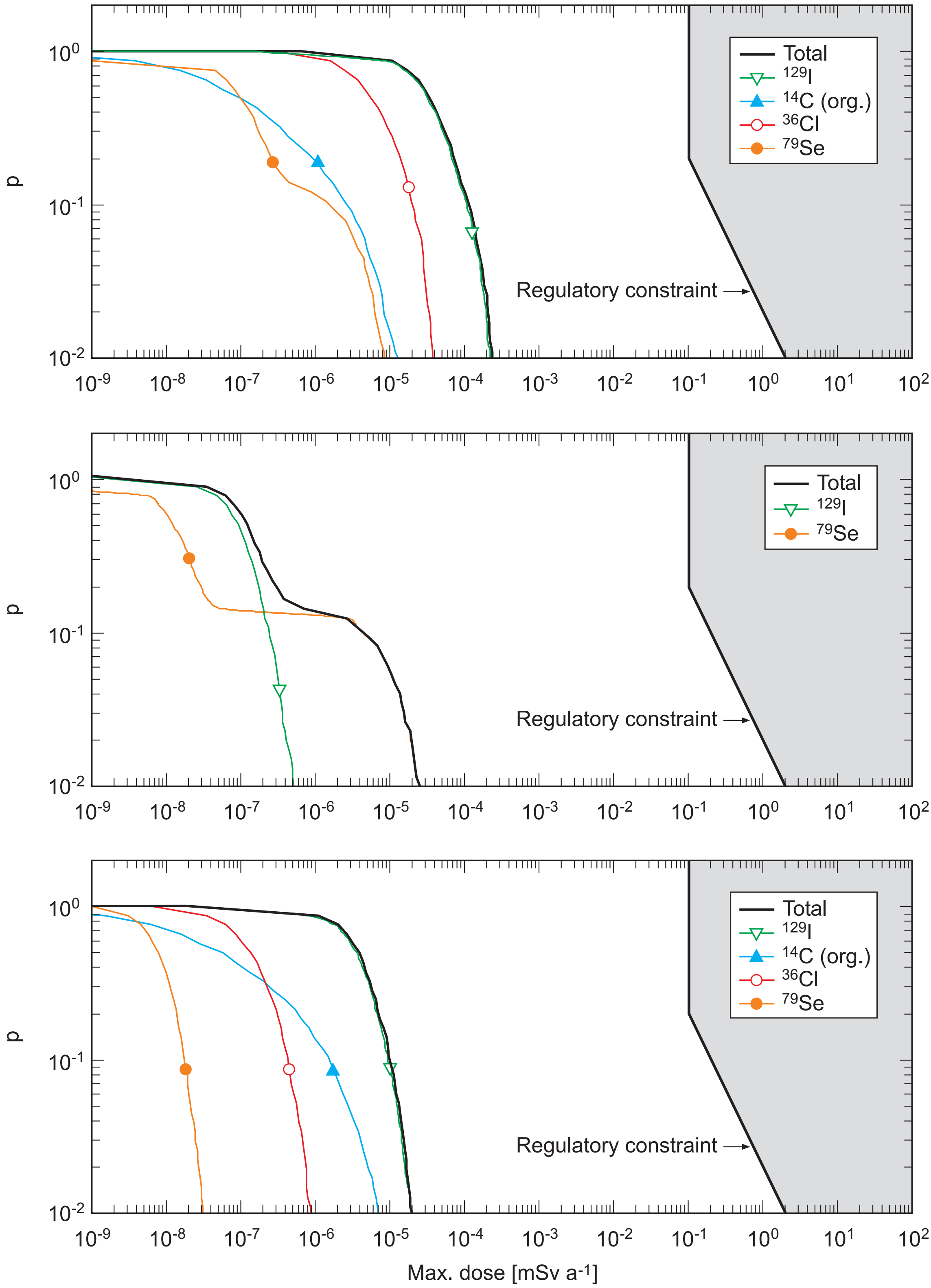

Fig. 7.4-3a: Probabilistic analysis of the Reference Conceptualisation – CCDFs for key radionuclides and for the sum over all safety-relevant radionuclides

Top: SF, middle: HLW, bottom: ILW

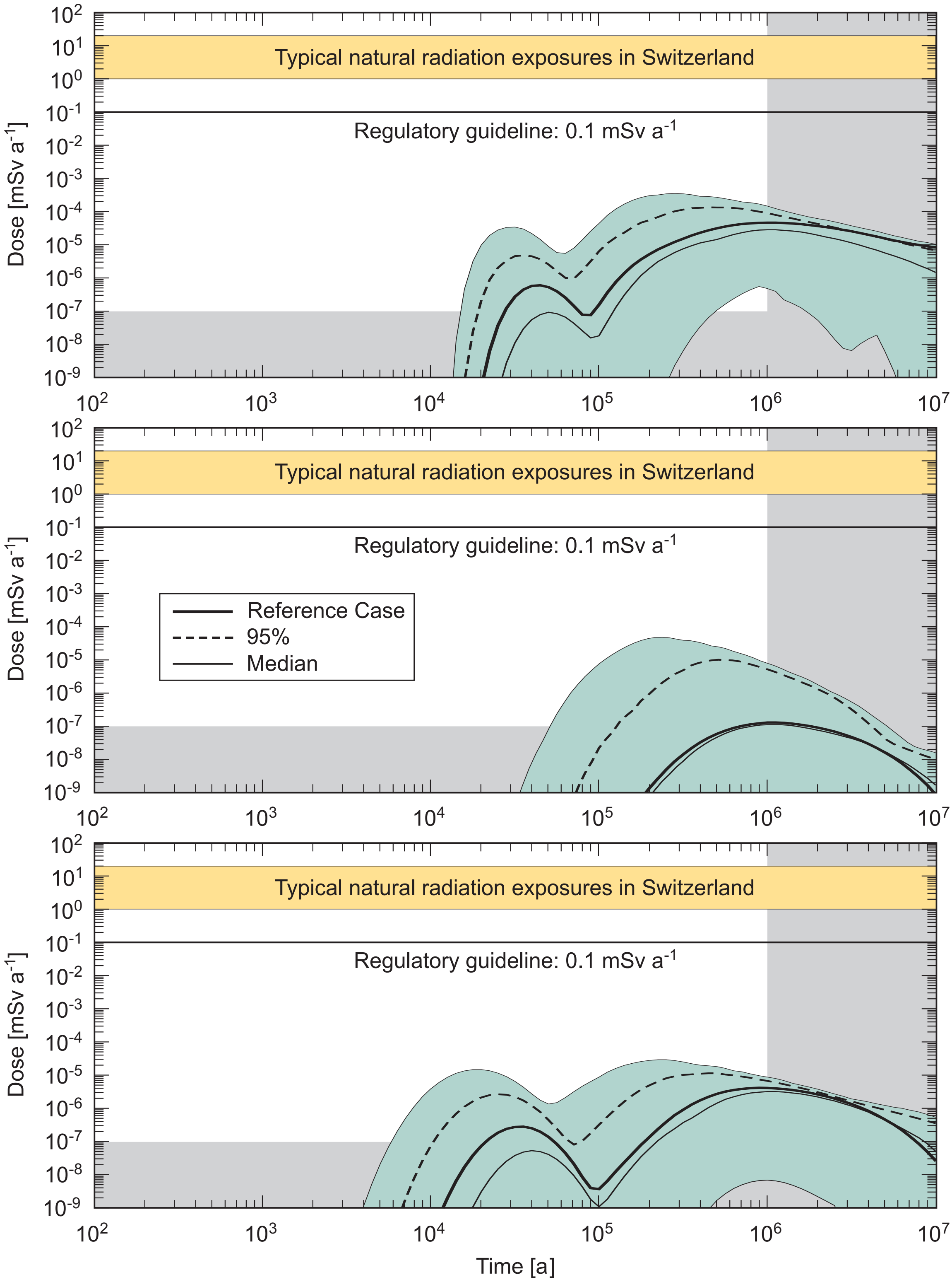

Fig. 7.4-3b: Probabilistic analysis of the Reference Conceptualisation – Evolution of the median, the 95th percentile and the highest / lowest dose maxima of all samples

For comparison the deterministic results for the Reference Case are also included. Top: SF, middle: HLW, bottom: ILW.

Results

The analysis shows that the summed dose maximum for spent fuel is only slightly reduced compared to the Reference Case. This is due to the fact that the dose maximum is dominated by the instant release fraction of ^{129}I, which is not affected by the solubility-limited fuel matrix dissolution rate. The dose due to the radionuclide fraction initially contained in the fuel matrix is greatly reduced compared to the Reference Case.

7.4.4 Bentonite thermal alteration

Main difference to Reference Case

This conceptualisation differs from the Reference Case in that the inner part of the bentonite buffer around SF / HLW canisters is assumed to be thermally degraded due to heat generation by the wastes.

Conceptual assumptions

Increased temperatures above 125 °C due to radioactive decay in spent fuel canisters are limited to the inner half of the bentonite buffer (Section 5.3.2). In this conceptualisation, it is pessimistically assumed that these increased temperatures lead to thermal degradation of the inner half of the bentonite. Thermal degradation is assumed to enhance radionuclide diffusion, but the sorption properties of bentonite are taken to be unaffected by thermal effects. The outer half of the bentonite functions according to design, i.e. its swelling pressure is fully developed, colloids are filtered by its micro-porous structure, and transport is diffusion dominated.

For this conceptualisation, the pore diffusion constant in the inner half of bentonite is pessimistically assumed to correspond to that of free water. Radionuclide sorption in the inner half of the bentonite buffer is taken to be the same as in the unperturbed outer half. No parameter variations are considered.

Results

The analysis shows that the summed dose maxima for SF / HLW is only slightly affected by whether or not the inner part of the bentonite buffer is thermally degraded.

7.4.5 Glacially-induced flow in the Opalinus Clay

Main difference to Reference Case

In the Reference Case, the cycling between glacial and interglacial periods is considered to have an insignificant effect on safety, because small fluctuations of the water flow rate in the Opalinus Clay have little effect on release of radionuclides. To test this assumption, the present conceptualisation investigates the effect of repeated compaction and drainage of the Opalinus Clay induced by ice loads during the next one million years.

Conceptual assumptions

In the course of the next one million years, a periodic series of 10 glaciations is assumed to occur (with an assumed frequency of one glaciation every 10^5 years), starting at 50 000 years from today. The duration of each glaciation is taken to be 20 000 years, with an assumed ice shield thickness of 200 m for the first 8 glaciations and 400 m for the last 2 glaciations. Periodic compaction/drainage and elastic rebound of the clay barriers (Opalinus Clay and bentonite) occurs, leading to spatial and temporal changes in the Darcy velocity in the clay barriers. The clay barriers are assumed to remain unperturbed, i.e. no fracturing occurs before, during or after ice loading. The hydraulic conductivity is assumed to be 2×10^{-14} m s^{-1} for the Opalinus Clay and 10^{-13} m s^{-1} for bentonite. The specific storage coefficient is 10^{-5} m^{-1} for Opalinus Clay and 2×10^{-4} m^{-1} for bentonite. The analysis is conducted for spent fuel only and is limited to those radionuclides that dominate the summed dose maximum of the Reference Case. The source term for radionuclide release from the spent fuel canisters to the bentonite is taken to be identical to the Reference Case. For comparison purposes with the calculated dose for the Reference Case, the reference biosphere model is employed, although biosphere conditions will be drastically changed during glaciations (very low population density and agricultural production rates, sparse vegetation, lower dilution rates). Alternative climatic conditions with the potential to affect the calculated doses are considered in a separate conceptualisation (see Section 7.9.3).

These calculations are carried out using the code FRAC3DVS (Nagra 2002c). No further parameter variations are performed.

Results

The analysis of glacially-induced flow in the Opalinus Clay shows that the doses are only slightly affected by compaction and drainage of the Opalinus Clay during glaciations. The dose maximum for spent fuel is shown to be about a factor of 1.6 higher than the Reference Case dose.

7.4.6 Additional barrier provided by confining units

Main difference to Reference Case

In the Reference Conceptualisation, the geological barrier is conservatively restricted to a transport path segment through the unperturbed host rock with a length of 40 m. The transport barriers provided by the confining units and the regional aquifers are neglected. In contrast to this simplified representation of the geological barrier, the present conceptualisation investigates the effectiveness of the confining units as an additional barrier to radionuclide transport (see Fig. 4.2-10 in Section 4.2.4).

Conceptual assumptions

A prolonged transport pathway is considered, with the purpose of quantifying the effectiveness of the confining units as a barrier to radionuclide transport. The conceptual assumptions are illustrated in Fig. 7.4-4. Two possibilities are envisaged:

- Vertical transport upward/downward through Opalinus Clay (40 m each), followed by vertical transport through the entire upper (100 m) and lower (160 m) confining units; this case represents the possibility that the local aquifers (Wedelsandstein, Sandsteinkeuper) are not hydraulically connected over long distances and, therefore, do not convey radionuclides with flowing groundwater. Exfiltration occurs via the overlying Malm aquifer and via the underlying Muschelkalk aquifer.

- Vertical transport upward/downward through Opalinus Clay (40 m each) and Lias/upper Keuper below the Opalinus Clay (60 m), followed by lateral transport in the local aquifers (Wedelsandstein, Sandsteinkeuper) within the confining units. In the upper confining units, additional retardation is provided by advective/dispersive transport in a segment of 25 km length within the fractured Wedelsandstein Formation and by considering diffusion from the fractures into the intact rock matrix. In the lower confining units, advective/dispersive transport takes place in a segment of 15 km length within the Sandsteinkeuper, which is treated as an equivalent porous medium. The Darcy flows within the local aquifers are derived from observed hydraulic heads and conductivities and are 5×10^{-13} m s^{-1} for the Wedelsandstein and 10^{-10} m s^{-1} for the Sandsteinkeuper.

Biosphere conditions in the Rhine valley just below the Rhine Falls (Reference Case) and further downstream are similar. The biosphere model is, therefore, taken to be identical to the Reference Case for all exfiltration areas described above.

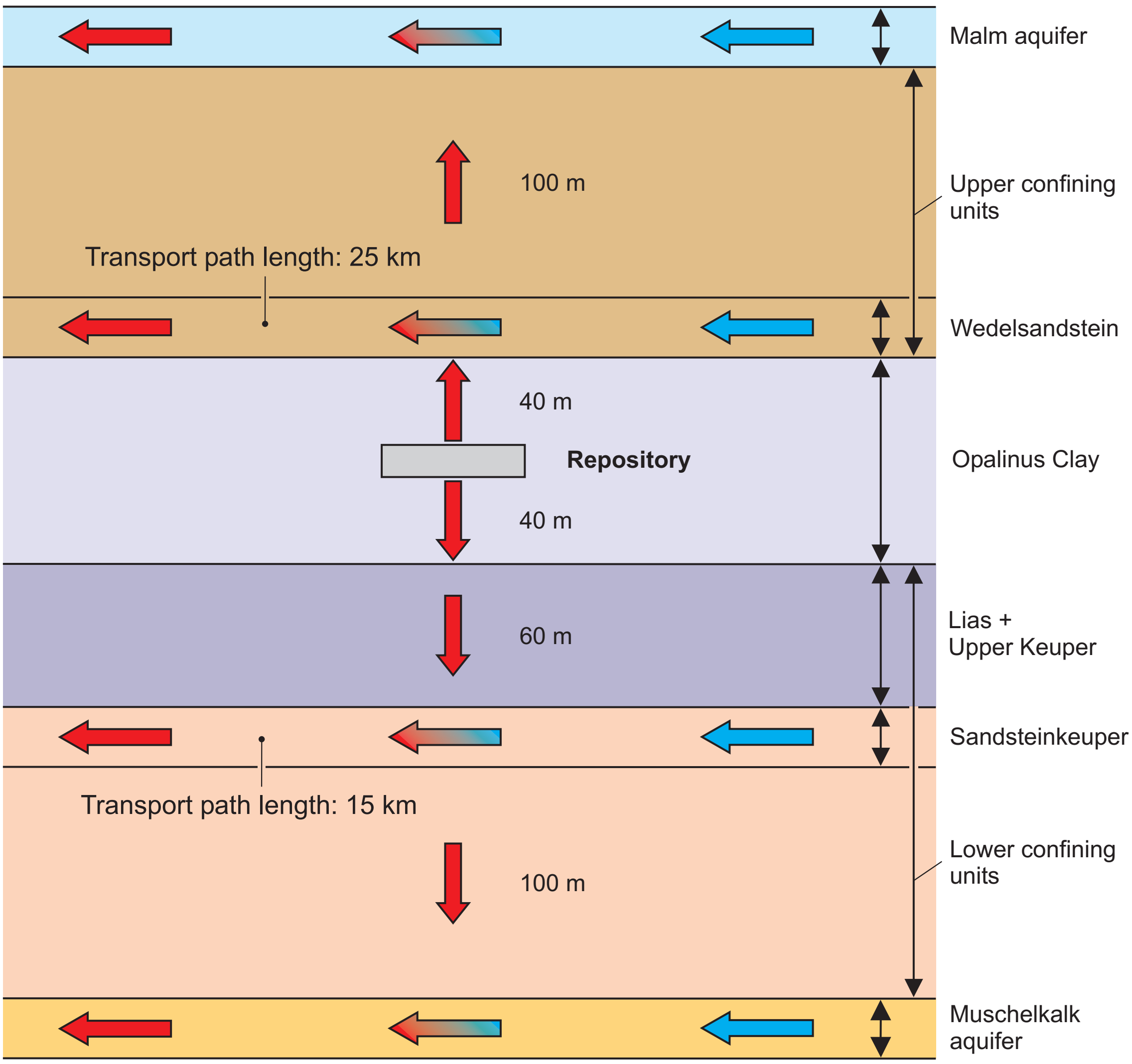

Fig. 7.4-4: Illustration of the radionuclide transport pathways in the conceptualisation considering the additional barriers provided by the confining units

Base Case and variant

In the Base Case, vertical transport through Opalinus Clay, followed by vertical transport through the entire upper and lower confining units is considered. In a variant, vertical transport through Opalinus Clay and Lias/Upper Keuper below the Opalinus Clay, followed by lateral transport in the local aquifers within the confining units is investigated.

In both cases, the two transport pathways from the Opalinus Clay via the upper and lower confining units to the biosphere are treated in a single model calculation using the network capability of the code PICNIC.

Results

For the Base Case considering the additional barrier for vertical transport in the upper and lower confining units, the summed dose maxima are 2.3×10^{-5} mSv a^{-1} (SF), 5.3×10^{-8} mSv a^{-1} (HLW) and 1.8×10^{-6} mSv a^{-1} (ILW). The dose maxima occur at about 3×10^{6} a, compared to about 1×10^{6} a in the Reference Case.

For the additional barrier due to horizontal transport within the local aquifers in the upper and lower confining units, the summed dose maxima are 3.6×10^{-6} mSv a^{-1} (SF), 6.9×10^{-9} mSv a^{-1} (HLW) and 2.4×10^{-7} mSv a^{-1} (ILW), i.e. more than an order of magnitude lower compared to the Reference Case. The times of maximal dose are delayed by about a factor of 7.

7.4.7 Radionuclide release affected by ramp / shaft

Main difference to Reference Case

One of the basic assumptions in the Reference Conceptualisation is that transport along the access tunnels is negligible compared with transport through the host rock. In this alternative conceptualisation, the validity of this assumption is tested by calculating the radionuclide release through host rock and access tunnel system in parallel.

Conceptual assumptions

The calculations are performed in three distinct steps, as described in detail in Nagra (2002c):

- In a first step, the water flow rates in the repository system are calculated using a simplified resistor network model that is compared with the results of a 3 D hydrodynamic finite element model (Nagra 2002a).
- In a second step, the near field release rates are calculated, taking into account the hydraulic boundary conditions calculated in the first step.
- In a third step, the simultaneous radionuclide release through host rock and access tunnel system is calculated based on the source term and the flow field calculated in the previous steps.

As in the Reference Conceptualisation, the effectiveness of the confining units as a barrier to radionuclide transport is conservatively neglected. Once the radionuclides enter the confining units they are assumed to be instantaneously transported to the exfiltration area where dilution takes place by mixing with flowing groundwater in the Quaternary aquifer.

Calculation of water flow rates in the repository system

The water fluxes in the repository are calculated analytically by means of a steady-state resistor network model, assuming sealing zones to be located as indicated in Fig. 7.4-5a. Inflow of water to the tunnels and flow of water along the tunnels is driven by the hydraulic head difference between the Sandsteinkeuper and the Wedelsandstein Formation.

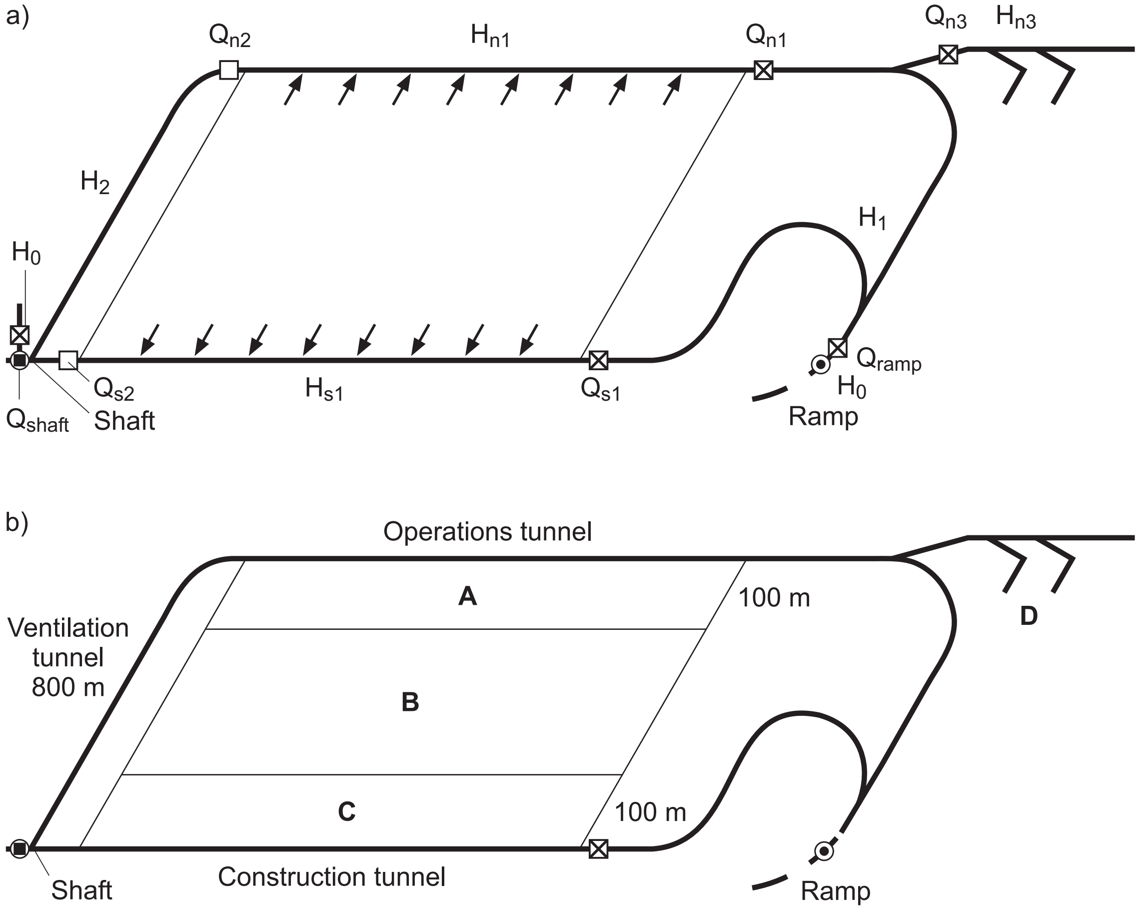

Fig. 7.4-5: a) Resistor network model for hydraulic calculations of water flow through the tunnel system, b) repository domains for modelling radionuclide transport through the host rock and the ramp / shaft

H: hydraulic heads in tunnel segments between sealing zones; Q: water fluxes through sealing zones (squares with crosses) and control points (empty squares); A/B/C: SF / HLW repository; D: ILW repository.

The repository is assumed to be fully backfilled and all sealing zones are functioning as designed. The hydraulic conductivities of Opalinus Clay, EDZ, sealing zones (bentonite) and tunnel backfill (bentonite / sand mixture) are assumed to be 2×10^{-14}, 1×10^{-12}, 1×10^{-13} and 5×10^{-11} m s^{-1}, respectively. This means that water flow through the sealing zones takes place predominantly through the EDZ (under the conservative assumption that the EDZ is not removed prior to emplacement of the sealing material), whereas water flow in tunnel sections backfilled with a bentonite / sand mixture is assumed to occur mainly in the tunnel backfill. The head difference between the Sandsteinkeuper and the Wedelsandstein Formation corresponds to a hydraulic gradient of 1 m m^{-1}.

In Nagra (2002c), it is shown that the water fluxes in the tunnel system are sensitive to the hydraulic conductivity of the EDZ. These water fluxes increase with the conductivity of the EDZ, but level off at a conductivity of about 1×10^{-10} m s^{-1}, because of the limited water inflow from the Opalinus Clay. Thus, in a parameter variation, an increased hydraulic conductivity of the EDZ of 1×10^{-10} m s^{-1} is considered.

The results of the resistor network model are compared with the results from a 3 D hydrodynamic finite element model (Section 5.5.3.1) at several different locations in the repository. The analysis is based on the case RLU0[106]. The comparison for the Base Case shows that the water flow rates calculated by the two models are in fair agreement (for more details see Nagra 2002c).

Calculation of radionuclide release from the near field

In the case of SF / HLW, the vertical water flow rate per canister, driven by the hydraulic head difference between the Sandsteinkeuper and the Wedelsandstein Formation, dominates over the horizontal flow rate along the axis of the emplacement tunnel, except for the tunnel sections in the vicinity of the operations/construction tunnels (see Nagra 2002c). This means that for the majority of the canisters, the near field release term is not affected by the axial flow rate. Only at the ends of the emplacement tunnels, i.e. within some tens of meters from both tunnel ends, is the axial water flow rate in the same order of magnitude as the vertical water flow rate. It can be shown that, even for extremely unfavourable conditions, the axial flow rate is not higher than 10 times the vertical flux. Even for these extreme conditions, the transport of radionuclides predominantly takes place upwards through the Opalinus Clay for the major part of the SF / HLW facility.

For this reason, the SF / HLW near field is split up into three domains, labelled A, B and C in Fig. 7.4-5b. Within segment B, the near field SF / HLW release terms are taken to be identical to the Reference Case. The calculation of the near field release terms for domains A and C is also based on the Reference Case, but with a 10-fold increased groundwater flow rate. By similar arguments, the near field release term for ILW (domain D) is also modelled in the same way as for the Reference Case, but with a 10-fold increased groundwater flow rate.

Calculation of radionuclide transport through host rock and access tunnel system

For the calculation of radionuclide transport affected by the ramp / shaft, the waste is split up into four domains, as shown in Fig. 7.4-5b. The inventory of domain B (corresponding to 75 % of the inventory of the SF / HLW repository) is released through the Opalinus Clay only; the corresponding geosphere model calculations are performed as in the Reference Case. For domains A/C (both SF and HLW) and D (ILW), radionuclide transport occurs both through the host rock and through the ramp / shaft. For each of the waste domains A/C/D, the flux of radionuclides is, therefore, split up between host rock and access tunnel system, according to the effective water flows involved (Nagra 2002c).

Retardation in the various segments in the operations / construction / ventilation tunnels is explicitly considered, assuming advection, dispersion, diffusion, decay and sorption in the bulk of the tunnel backfill (mixture of 30 % bentonite and 70 % quartz sand).

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[106] The case RLU0 involves vertical groundwater movement driven by the hydraulic head difference between the Sandsteinkeuper and the Wedelsandstein formation. without consideringoverpressures in near field and in the geosphere (see Nagra 2002a).

Results

The contributions of the radionuclide pathways through Opalinus Clay, shaft and ramp to the dose curves are shown in Fig. 7.4-6 (Base Case). The summed dose maxima are 4.7×10^{-5} mSv a^{-1} (SF), 1.3×10^{-7} mSv a^{-1} (HLW) and 4.1×10^{-6} mSv a^{-1} (ILW). These doses are nearly identical to those calculated for the Reference Case, the reason being that the dose curves for all three waste streams are dominated by the transport pathway through the Opalinus Clay, which is modelled in an identical way as in the Reference Case[107].

In the case of SF / HLW, the contribution of the shaft to the dose rises later than the contribution from the Opalinus Clay and shows a lower summed dose maximum by several orders of magnitude. The contribution of the ramp arises at much later times (several million years) and is significantly lower than the contribution of the shaft (for HLW, the contributions from the ramp and shaft are below the lower boundary of Fig. 7.4-6). In the case of ILW, no release through the shaft takes place and the summed dose maximum related to the ramp is more than 4 orders of magnitude lower and arises much later than the dose maximum related to the release of radionuclides through the Opalinus Clay. These results can be explained by the long transport pathways and the relatively low flow in the backfilled tunnels.

The summed dose maxima for the parameter variation considering a 100-fold increased hydraulic conductivity of the EDZ of 10^{-10} m s^{-1} are 4.6×10^{-5} mSv a^{-1} (SF), 1.2×10^{-7} mSv a^{-1} (HLW) and 3.9×10^{-6} mSv a^{-1} (ILW). These dose maxima are slightly lower compared to the Base Case doses, because even more radionuclides are deviated along the tunnel system, in comparison with the Base Case, which lowers the dose at the time of its maximum.

7.4.8 Convergence-induced release affected by ramp / shaft

Main difference to Reference Case

In the Reference Case for SF / HLW, tunnel convergence induced by time-dependent deformation of the Opalinus Clay is considered to be completed before canister breaching, with no effect on radionuclide release. In the Reference Case for ILW, tunnel convergence is assumed to occur during repository construction, with little or no deformations of the tunnel cross-section occurring after full resaturation of the emplacement tunnels and the start of radionuclide release. The current conceptualisation differs from the Reference Case in that alternative assumptions regarding the extent and duration of tunnel convergence are considered, that may lead to enhanced water flow either through the host rock or through the access tunnel system.

Conceptual assumptions

Due to time-dependent deformation of the Opalinus Clay, the bentonite buffer of the SF / HLW emplacement tunnels may be compacted and its porosity reduced. There is significant uncertainty in the degree of compaction and the timescales involved. The range of uncertainty is considered to be bounded by the following limiting cases:

1. The swelling pressure of bentonite at emplacement density (2 – 4 MPa) is sufficient to prevent significant tunnel convergence and compaction of bentonite. This view is supported by the evidence that an anisotropic stress field is observed in the Opalinus Clay (with an

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[107] The dose maxima are actually slightly lower compared to the Reference Case doses, because part of the radionuclides are deviated along the tunnel system, decreasing the dose at the time of its maximum and increasing it at some other time.This effect is similar to macroscopic dispersion that is observed if, at a macroscopic scale, different transport paths result in a spread of transport times and, therefore, cause concentration lowering at the downstream boundary.

anisotropy factor of approximately 1.4 – 1.5) which indicates that time-dependent deformation of the Opalinus Clay is not an efficient process for completely removing stress heterogeneities or anisotropies, such as those induced by the repository. This case is not modelled in the current safety assessment.

2. The rock surrounding the bentonite will deform, compacting the bentonite and increasing its swelling pressure until it is balanced by the external stress field, and, at the same time, reducing the bentonite porosity from 45 % to about 36 % and the tunnel radius from 1.25 m to 1.15 m. The time period over which these processes take place may be in the order of

   a) a few decades to hundreds of years (if full bentonite compaction develops in parallel with near field saturation); this is assumed in the Reference Case for SF / HLW,

   b) tens of thousands to hundreds of thousands of years (long-term hydromechanical evolution of near field conditions); this is considered in the present conceptualisation.

In cases 1 and 2a), no effects on radionuclide release are expected because bentonite compaction is either absent (case 1) or is fully developed before canister breaching (case 2a). In case 2b, there seems to be a possibility that tunnel convergence leads to the displacement of contaminated water from the near field either along the EDZ of the emplacement tunnels or through the Opalinus Clay. Squeezing of porewater through the Opalinus Clay is possible if the hydraulic gradient increases. For a hydraulic conductivity of the Opalinus Clay of 2×10^{-14} m s^{-1} and an assumed duration of bentonite compaction of 10^4 (10^5) years, an increase of the hydraulic gradient in the order of 1.5 (0.15) m m^{-1}, i.e. 2.5 (1.15) times the Reference Case value, is sufficient for the displacement of convergence-induced water through the Opalinus Clay. As shown in Chapter 6 (Fig. 6.7-6 and 6.7-8) and also by the parameter variation to the Reference Case considering a 10-fold increased water flow, such small increases of Darcy flow in the Opalinus Clay have no safety-relevant impact.

Even if an initially defective canister were present, negligible water displacement is expected during tunnel convergence in case 2a, due to the prevailing unsaturated conditions in the near field. Therefore, convergence-induced release of water-borne radionuclides from the SF / HLW repository is considered to be negligible.

In the case of the ILW repository, little convergence is expected after the end of waste emplacement due to the strength of the aggregates contained in the cementitious materials (concrete, mortar). There is, however, some uncertainty related to the compaction of void volumes present in the waste containers. In the present conceptualisation, it is assumed that the corroded waste containers will be partially compacted, leading to tunnel convergence and water displacement (Section 5.4.3). The total reduction in void volume per unit length of ILW tunnel is estimated to be 0.6 m^3 m^{-1}, resulting in a maximal cumulative water displacement of about 100 m^3. Tunnel convergence is assumed to take place within 1 000 years following waste emplacement, in parallel with tunnel saturation. This leads to a convergence-induced water flux of approximately 0.1 m^3 a^{-1}. It can be shown that the displacement of water will occur predominantly through the host rock if the seals operate as expected (Nagra 2002c). The associated hydraulic gradient between the repository and the Wedelsandstein is about an order of magnitude higher than in the Reference Case.

The modelling of simultaneous release of radionuclides through the host rock and the access tunnel system is performed in the same way as in Section 7.4.7. The near field release rate for ILW-1 is assumed to be a pulse of dissolved radionuclides. Transport properties of host rock and access tunnel system are identical to the case treated in Section 7.4.7.

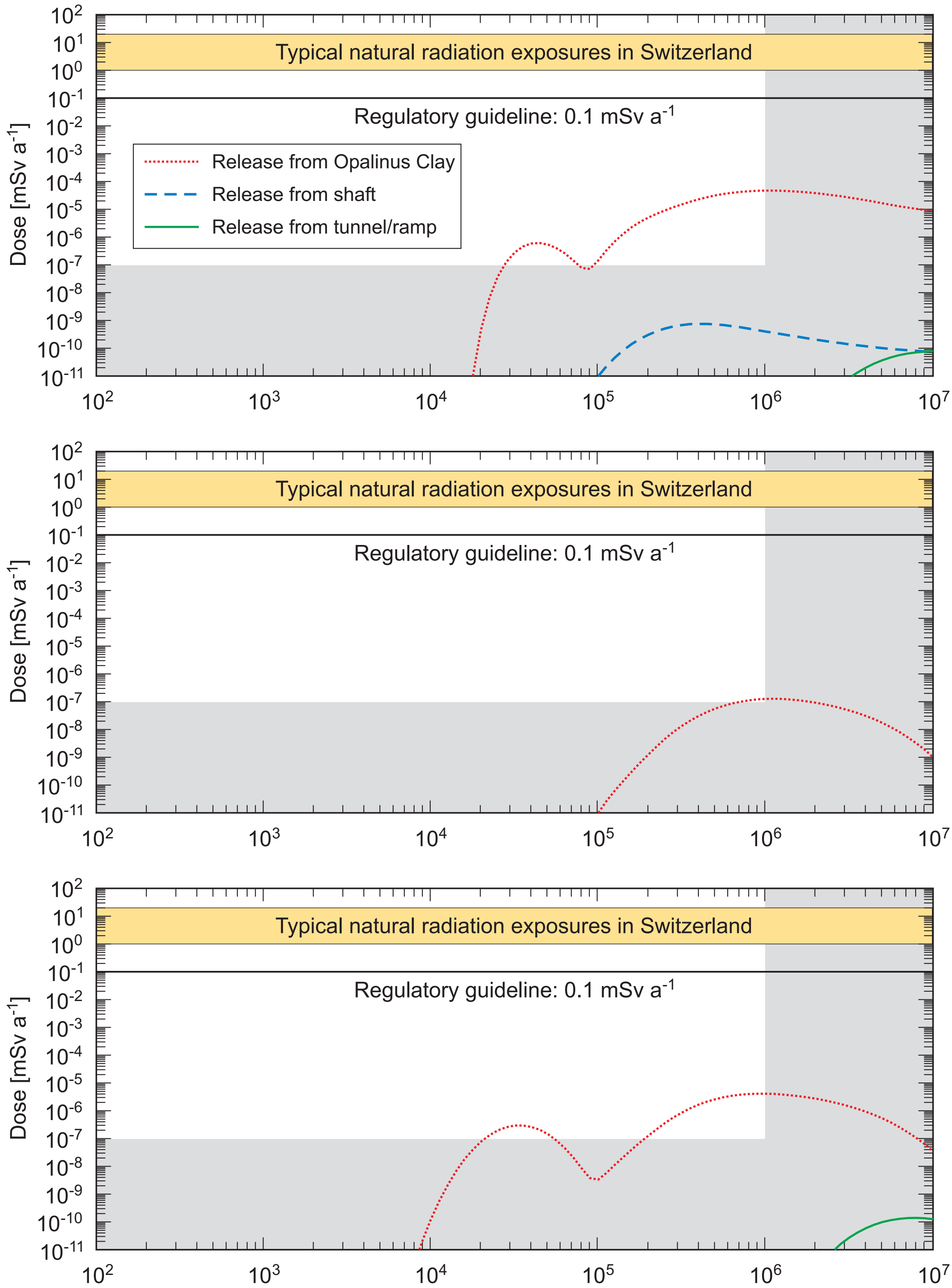

Fig. 7.4-6: Dose as a function of time for the assessment case considering radionuclide release affected by the ramp / shaft (Base Case)

Upper figure: SF, middle figure: HLW, lower figure: ILW.

Base Case and variant

In the case of SF / HLW, the effect of tunnel convergence on radionuclide release is negligible, as discussed above. No calculations have been performed.

In the Base Case for ILW-1, a pulse of water containing dissolved radionuclides is assumed to be released due to tunnel convergence with subsequent radionuclide transport both through the host rock and through the access tunnel system, assuming steady-state water flow rates. The total amount of radionuclide-containing water displaced (about 100 m^3) corresponds to approximately 3 % of the pore volume present in the ILW-1 tunnels. It is therefore assumed that 3 % of the radionuclide inventory dissolved in the porewater is released within a time span of 1 000 years, starting at 100 years and ending at 1100 years, at a water flow rate of 0.1 m^3 a^{-1}. The remainder of the radionuclides is released and transported as in the Reference Case.

In a variant for ILW-1, a pulse of water containing dissolved radionuclides is assumed to be released due to tunnel convergence with subsequent transport through the host rock only, but with transient water flow rates. The same near field source term is used as in the Base Case, with the remainder of the radionuclides being released and transported as in the Reference Case.

Results

In the Base Cases for SF / HLW the effect of tunnel convergence on safety is considered to be negligible, consequently no calculations have been performed for SF / HLW. The summed dose maxima are expected to be close to those of the Reference Case.

In the Base Case for ILW-1, the summed dose maximum for steady-state flow rates is 4.3×10^{-6} mSv a^{-1}, which is identical to the Reference Case dose. In a parameter variation for ILW-1, the transient water flow rates through the geosphere lead to a summed dose maximum of 3.2×10^{-6} mSv a^{-1}, which is slightly lower than that for the Reference Case due to differences in the modelling approach used (use of FRAC3DVS instead of PICNIC, see Nagra 2002c).

7.4.9 Gas-induced release of dissolved radionuclides affected by ramp / shaft

Main difference to Reference Case

In the Reference Case, little or no displacement of contaminated porewater is assumed to occur as a consequence of gas pressure build-up following gas generation in the emplacement tunnels. The present conceptualisation differs from the Reference Case in that the possibility of accelerated release of dissolved radionuclides through the host rock and through the access tunnel system is considered for SF and for ILW (gas-induced displacement of contaminated porewater).

Conceptual assumptions

Gas generation from anaerobic metal corrosion and, in the case of ILW, from microbial degradation of organic material leads to the generation of hydrogen gas, carbon dioxide and methane. In the case of SF, steel corrosion takes place upon contact with water at a rate of 1 μm a^{-1} or less. In the case of ILW, the total gas generation rate has been estimated to be roughly 700 m^3 (STP) a^{-1}, for the first few years after the onset of gas generation, declining rapidly to 100 m^3 (STP) a^{-1} after 10 years and gradually to 10 m^3 (STP) a^{-1} after 10 000 years (Fig. 5.4-2). Initially, the generated gas dissolves in the near field porewater until the gas solubility is reached. Thereafter, a free gas phase is formed and gas pressure starts to build up.

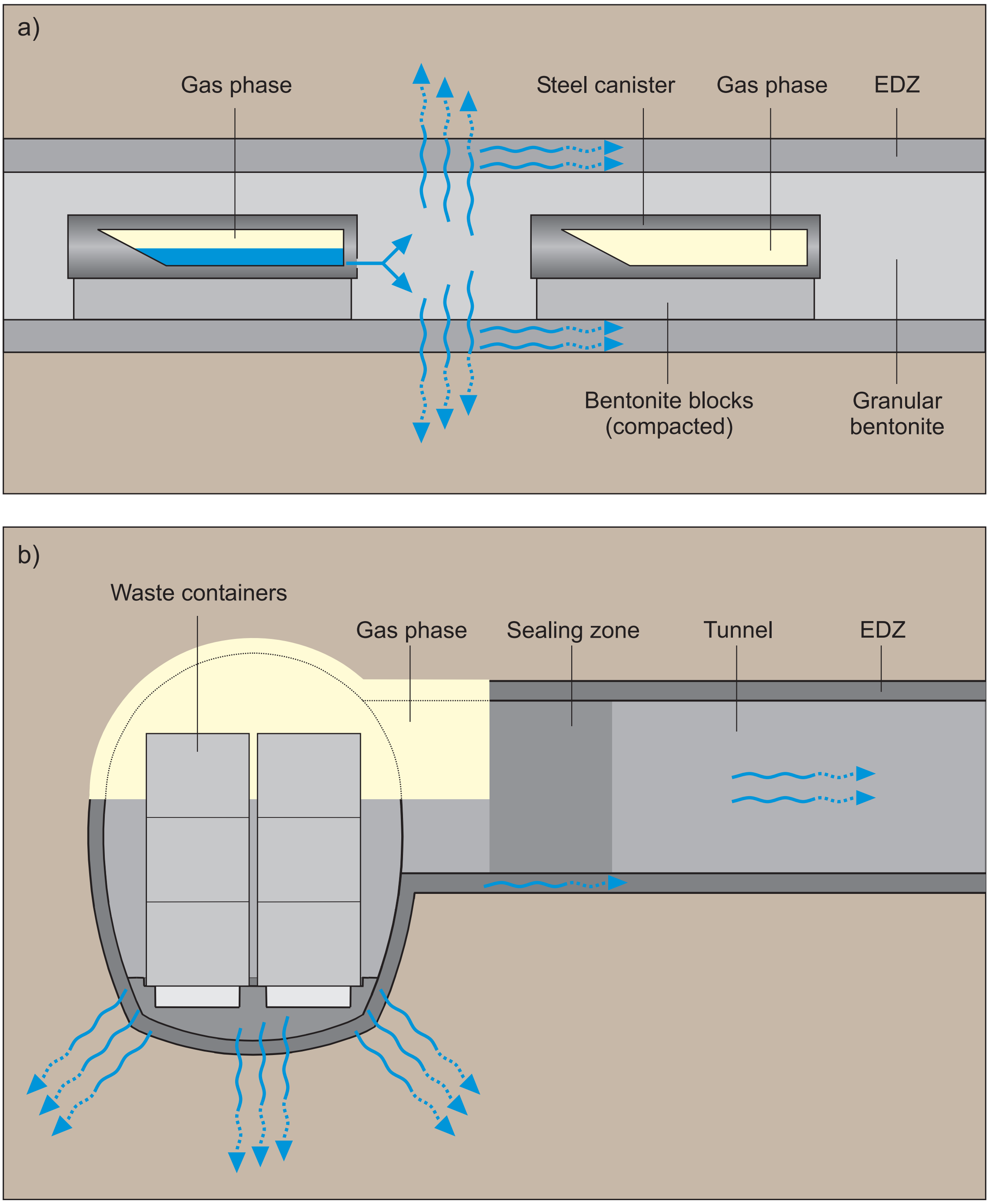

Fig. 7.4-7: Scheme with the conceptual model for gas-induced release of dissolved radionuclides through the Opalinus Clay and through the access tunnel system: a) SF, b) ILW

After breaching of the SF canisters, the present assessment case considers the possibility that accelerated radionuclide release could arise from hydrogen gas production due to corrosion of internal canister surfaces. This could expel water from the canister if the defect were in an unfavourable location (e.g. the underside of the canister, see Fig. 7.4-7). In reality, a canister is likely to be breached at several locations, or it may already be pressurised with hydrogen gas diffusing into the canister before full breaching. Both of these processes significantly reduce the maximum amount of water displaced. This is taken into account by assuming that only 10 % of all canisters are subject to water displacement from the canister interior. In all other canisters, gas may escape without displacing significant amounts of water. Furthermore, the inflow and outflow of water may be limited by the hydraulic resistivity of the pinholes, but this effect is neglected. Instead, it is pessimistically assumed that the entire canister void volume of 0.7 m^3 is filled with water before the internal gas generation starts. Assuming a uniform probability density function for the location of a single pinhole around the canister circumference, the mean water volume displaced per canister is 0.35 m^3.

This effect is potentially significant for SF, since the instant release fraction of the inventory would be available for expulsion by gas, whereas HLW radionuclides would be immobilised almost entirely within the glass matrix.

In the case of ILW, gas-induced displacement of water through the host rock and access tunnel system may take place. The corresponding water fluxes are estimated as follows:

According to Section 5.5.2.2, it takes about 5000 – 10 000 years for the pressure to reach values in the order of 10 MPa where significant porewater displacement from the ILW emplacement tunnels into the formation starts to occur. By then, the gas generation rate is reduced to roughly 10 m^3 (STP) a^{-1}, corresponding to 0.1 m^3 a^{-1} at a pressure of 10 MPa. Assuming a volume balance between generated gas and displaced water, the rate of water displacement is thus 0.1 m^3 a^{-1}. More detailed calculations show that the rate of water displacement is 0.05 m^3 a^{-1} (Nagra 2002c and 2003a), maintained for a time period of about 30 000 years. The cumulative amount of displaced water is 1500 m^3, which roughly corresponds to 50 % of the available pore space in the ILW repository. In a pessimistic variant, it is assumed that an amount of water corresponding to 100 % of the available pore space is displaced within 10 000 years at a rate of 0.3 m^3 a^{-1}.

Gas-induced displacement of contaminated water from the SF and ILW repository takes place through the host rock and through the access tunnel system. The corresponding water fluxes and branching ratios for radionuclide transport are calculated in an analogous way as in the assessment case considering radionuclide release affected by the ramp / shaft (Section 7.4.7).

Base Case and parameter variations

In the Base Case for SF, it is assumed that the entire instant release fraction is expelled by gas into the Opalinus Clay and the access tunnel system over a 1 000 years period following canister breaching (corrosion rate 1 μm a^{-1}). The total water flow rate from 200 SF canisters (approximately 10 % of all canisters) is 0.07 m^3 a^{-1}.

In the Base Case for ILW, a pulse release, starting at 10 000 years and lasting for 30 000 years, with a mean water flow rate of 0.05 m^3 a^{-1} is assumed, conveying 50 % of the total dissolved radionuclide inventory of the ILW repository. In a parameter variation, a mean water flow rate of 0.3 m^3 a^{-1} is assumed, starting at 1 000 years and lasting 10 000 years, conveying 100 % of the total mobile radionuclide inventory of the ILW repository.

In all cases, the dose contribution of the remaining radionuclides is neglected.

Results

For the Base Case for SF, the calculated summed dose maximum is 4.0×10^{-6} mSv a^{-1} for a release pulse duration of 1 000 years. This peak arises at about 0.9 million years.

In the case of gas-induced porewater displacement from the ILW repository through the Opalinus Clay and, in parallel, through the operations tunnel and ramp, the calculated summed dose maximum is estimated to be 4.2×10^{-6} mSv a^{-1} for a water flow rate of 0.05 m^3 a^{-1} conveying 50 % of the total dissolved inventory and 4.6×10^{-5} mSv a^{-1} for a water flow rate of 0.3 m^3 a^{-1} conveying 100 % of the total dissolved inventory. These peaks arise at about 0.3 and 0.08 million years, respectively.

7.4.10 Summary of results of Reference Scenario

The results for the different conceptualisations of the Reference Scenario and of the parameter variations are summarised in Fig. 7.4-8.

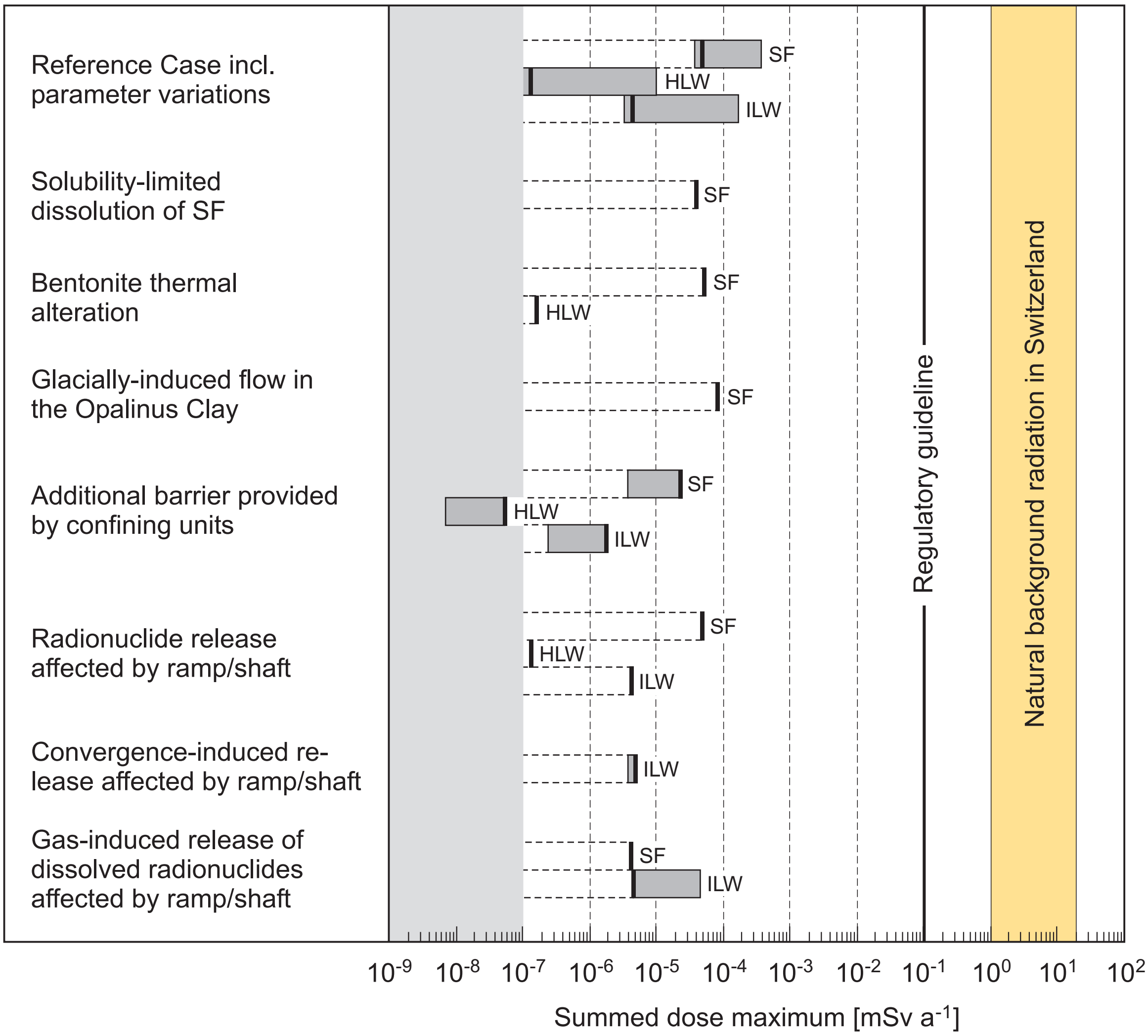

Fig. 7.4-8: Summed dose maxima and ranges for SF, HLW and ILW for the various conceptualisations and parameter variations of the Reference Scenario considering the release of dissolved radionuclides (Base Cases marked by bold lines)

7.5 Release of radionuclides as volatile species along gas pathways

7.5.1 Description of the scenario

In this scenario, it is assumed that radionuclides present in the waste as potentially volatile species (^{14}C in methane) are transported from the near field via gas pathways through the Opalinus Clay or through the access tunnel system to the overlying Wedelsandstein Formation. Gas accumulates in pore spaces in the near field, in and around the gas pathways created in the Opalinus Clay, in the access tunnel system and in the Wedelsandstein Formation. Volatile radioactive species are diluted in the gas-filled pore space with non-radioactive gas generated by anaerobic metal corrosion and microbial degradation. Eventually, transport of volatile radioactive species takes place from the Wedelsandstein Formation by gas diffusion through the low-permeability upper confining units to the Malm aquifer. Radionuclide transport through the Malm aquifer to the biosphere is assumed to be instantaneous. The dissolved radioactive volatile species are diluted in the flowing groundwater of the Quaternary aquifer. Degassing of volatile species is pessimistically assumed not to occur.

As in the Reference Scenario, the principal phenomena that contribute directly and positively to the performance of the disposal system at different times are:

- radioactive decay,
- a period of complete containment by the SF / HLW canisters,
- immobilisation in the waste forms,
- delayed release through Opalinus Clay and upper confining units,
- dilution in regional aquifers and in the surface environment.

The calculated summed dose maxima of the present scenario are compared in Fig. 7.5-1 with the doses for the Reference Case.

7.5.2 Release of ^{14}C as volatile species in the gas phase not affected by ramp / shaft ("tight seals")

Main difference to Reference Case

In the Reference Case, ^{14}C, the only radionuclide with the potential to form a volatile species, is assumed to dissolve in the porewater and to be transported through the Opalinus Clay by advection / diffusion. In the present conceptualisation, the fate of ^{14}C is investigated under the assumption that gas pathways exist through the Opalinus Clay, which can lead to an accelerated release of volatile ^{14}C.

Conceptual assumptions

^{14}C is present in the structural materials of SF. Such structural materials are present both in SF and in ILW. It is pessimistically assumed that all of the ^{14}C inventory contained in these materials is converted to methane. For HLW, the ^{14}C is inorganic, thus volatile release is not considered.

In this conceptualisation, part of the volatile ^{14}C inventory is released instantly from the structural material upon contact with water, the remainder being released congruently with cladding dissolution. The released volatile radionuclides are mixed with non-radioactive gases (predominantly hydrogen gas from anaerobic metal corrosion). In the course of time the generated gas is accumulated in the pore space at different locations in the repository system, including emplacement tunnel backfill, excavation-disturbed zone, Opalinus Clay and Wedelsandstein Formation. The sealing zones are assumed to be completely tight and no transport occurs through the access tunnel system. On its way from the waste through the near field and geosphere, a significant fraction of the volatile radionuclides are dissolved in the porewater and diffuse away. In the present conceptualisation it is assumed that continuous gas pathways through the Opalinus Clay will evolve, which eventually lead to gas breakthrough to the overlying Wedelsandstein Formation. The breakthrough time is dependent on the cumulative amount of gas previously lost to the Opalinus Clay by gas leakage and diffusion. After breakthrough, gas is assumed to accumulate in the Wedelsandstein Formation (predominantly non-radioactive gases from SF / HLW, and to a lesser extent from ILW). As a result, the volatile ^{14}C is completely confined within the geological environment for an extended period of time, causing significant decay of ^{14}C. The release rate from the Wedelsandstein Formation to the Malm aquifer is controlled by gas diffusion through the low-permeability formations of the upper confining units. Complete dissolution of all volatile species in the Malm aquifer and instantaneous aqueous transport to the reference biosphere area are assumed to occur. Dilution takes place in the flowing groundwater within the Quaternary aquifer. Further dilution in the atmosphere following degassing of volatile ^{14}C is conservatively neglected.

Base Case and parameter variations

In all calculations, a steel corrosion rate of 1 µm a^{-1} is assumed. The gas permeability in the Opalinus Clay is varied. In the Base Case, a gas permeability of the rock matrix of 10^{-23} m^2 is used for the calculation of the loss of gas from tunnel walls and from dilated pathways, if the corresponding threshold is exceeded. In the framework of two parameter variations, the effects of a changed gas permeability in the matrix of 10^{-22} m^2 and 0 m^2 are investigated, reflecting uncertainty in the gas transport properties of Opalinus Clay.

In all cases, the dose contribution of non-volatile radionuclides is not taken into account.

Results

Simplified semi-analytical calculations for the gaseous release of volatile ^{14}C have been performed. The results for the Base Case of the conceptualisation "tight seals" show that the drinking water dose maxima due to volatile ^{14}C are 5.8×10^{-7} mSv a^{-1} (SF) and 1.0×10^{-7} mSv a^{-1} (ILW), arising at about 4×10^4 years.

For the parameter variations considering a 10-fold increased gas permeability of 10^{-22} m^2, the dose maxima are 3.4×10^{-6} mSv a^{-1} (SF) and 4.3×10^{-7} mSv a^{-1} (ILW). For a zero gas permeability, the doses are comparable to the doses of the Base Case.

7.5.3 Release of ^{14}C as volatile species in the gas phase affected by ramp / shaft ("leaky seals")

Main difference to Reference Case

In the Reference Case, volatile ^{14}C is assumed to dissolve in the porewater and to diffuse through the Opalinus Clay. In the present conceptualisation, the fate of ^{14}C is investigated under the assumption that gas pathways exist through the access tunnel system, which lead to an accelerated release of volatile ^{14}C.

Conceptual assumptions

In contrast to the case considering tight seals (Section 7.5.2), in the present conceptualisation it is assumed that transport of volatile ^{14}C occurs from the near field through gas pathways in the access tunnel system to the Wedelsandstein Formation. As a result, gas accumulates in the near field (including EDZ), access tunnel system and Wedelsandstein Formation. No continuous gas pathways are assumed to be formed in the Opalinus Clay. All other conceptual assumptions and parameter values are identical to the model considering tight seals.

Base Case and parameter variations

In all calculations, a steel corrosion rate of 1 $\mu m\ a^{-1}$ is assumed. The gas permeability in the Opalinus Clay is varied to calculate the loss of gas from the emplacement tunnels if the gas entry pressure for 2-phase flow is exceeded. In the Base Case, a gas permeability of 10^{-23} m^2 is used. In the framework of two parameter variations, the effects of a changed gas permeability of 10^{-22} m^2 and 0 m^2 are investigated, reflecting uncertainty in the gas transport properties of Opalinus Clay.

In all cases, the dose contribution of non-volatile radionuclides is not taken into account.

Results

The drinking water dose maxima for the Base Case for the conceptualisation "leaky seals" are 3.7×10^{-5} mSv a^{-1} (SF) and 7.2×10^{-6} mSv a^{-1} (ILW), arising at about twelve thousand years.

For a 10-fold increased gas permeability of 10^{-22} m^2, the dose maximum for SF and ILW are slightly lower than the Base Case doses, reflecting the effects of gas leakage and decay of ^{14}C in the Opalinus Clay matrix before gas breakthrough by way of the access tunnel system to the Wedelsandstein Formation takes place. For zero gas permeability, the doses are nearly identical to the Base Case doses.

7.5.4 Summary of results of the Scenario "Release of radionuclides as volatile species along gas pathways"

The results for the different conceptualisations of the Scenario "Release of radionuclides as volatile species along gas pathways" and of the parameter variations are summarised in Fig. 7.5-1.

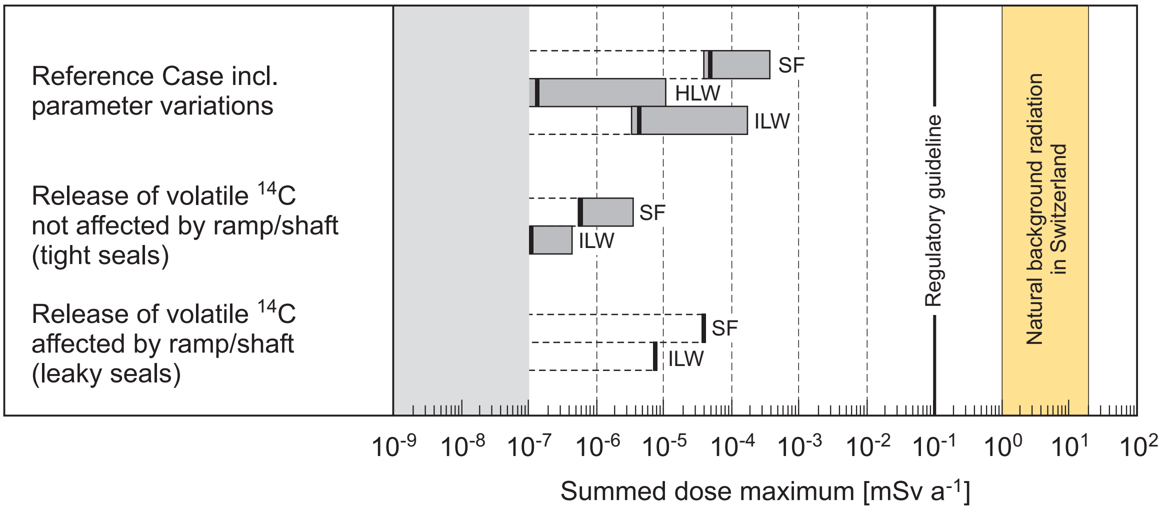

Fig. 7.5-1: Dose maxima and ranges for SF and ILW for the various conceptualisations and parameter variations of the scenario considering a release of volatile ^{14}C along gas pathways. (Base Cases marked by bold lines).

The contribution of doses due to other pathways are not included. The Reference Case results with parameter variations are also shown to allow an easy comparison.

7.6 Release of radionuclides affected by human actions

7.6.1 Description of the scenario

In this scenario, the possible radiological consequences of future human actions related to exploratory activity, exploitation of deep groundwater or the abandonment of the repository before final backfilling and sealing are assessed. For example, one or more boreholes may be drilled in the vicinity of the repository, either in the course of some future exploratory activity, or for the extraction of deep groundwater from the overlying Malm aquifer for hydrothermal purposes or as mineral water for drinking.

There is general consensus that a possibility exists that records of a geological repository may be lost, or at least neglected, within a period of several hundred years after closure, and thus this type of scenario must be addressed in a safety assessment[108]. The uncertainties involved are, to a large extent, non-quantifiable and irreducible. Thus, the model calculations presented in this scenario serve for illustrative purposes.

Due to the compartmentalisation of the repository, a borehole penetration will only cause localised perturbations of the repository, but does not undermine the pillars of safety for the major part of the repository. For example, a borehole penetrating a SF / HLW emplacement tunnel has only a small effect on the transport of radionuclides originating from neighbouring emplacement tunnels or from other parts of the disposal system (ILW, pilot facility), because the tunnel separation is roughly equal to the vertical transport path length in the Opalinus Clay.

[108] In accordance with regulatory advice, deliberate human intrusion is not discussed in the present report (see Chapter 2).

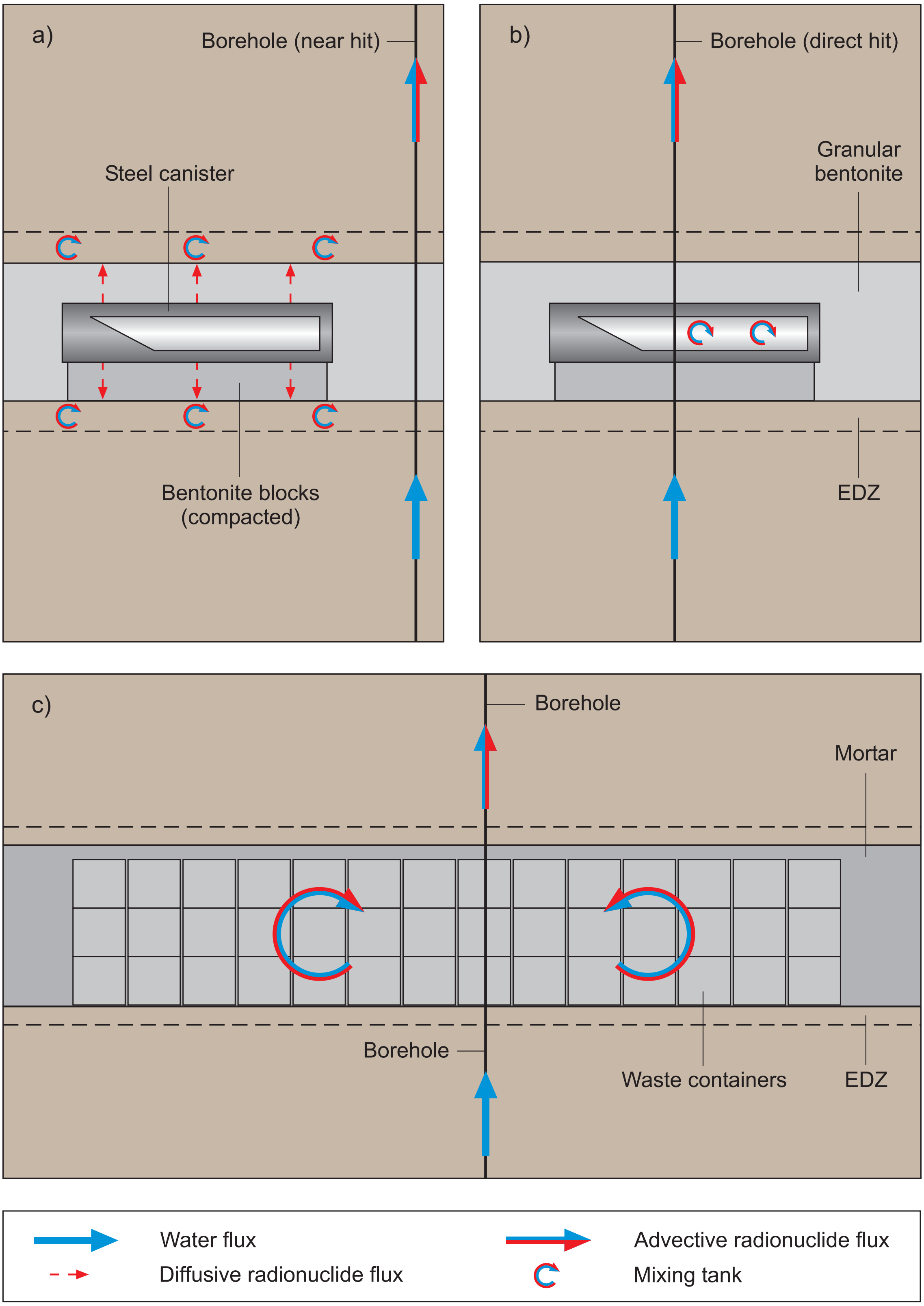

Fig. 7.6-1: Scheme of investigated variants of borehole penetration: a) SF / HLW (near hit), b) SF (direct hit), c) ILW-1 (direct hit)

As in the Reference Scenario, the principal phenomena that contribute directly and positively to the performance of the disposal system at different times are:

- radioactive decay,
- immobilisation in the waste forms,
- geochemical immobilisation and retardation of radionuclides released from the waste forms in the bulk of the clay barriers (where they are not affected by a borehole),
- limited mobility of dissolved radionuclides in the bulk of the clay barriers (where they are not affected by a borehole), and dispersion during transport, and
- dilution in regional aquifers and in the surface environment.

The calculated summed dose maxima of the present scenario are compared in Fig. 7.6-2 with those for the Reference Case.

7.6.2 Borehole penetration into the repository

Main difference to Reference Case

The Reference Case is based on the assumption that there will be no future boreholes within or near the repository affecting the safety of the repository. The present conceptualisation addresses the effects of a hypothetical borehole penetrating one of the emplacement tunnels, or, in the most extreme case, a spent fuel canister.

Conceptual assumptions

It is assumed that in the course of some future exploratory activity a borehole may penetrate the repository. Besides the exposure of the drilling personnel, long-term safety may be affected due to the fact that a direct pathway from the repository (or, in the most extreme case, from a spent fuel canister) to the biosphere will exist for some time period. The borehole is assumed to penetrate all the way through the Opalinus Clay and to be abandoned without proper sealing and without removing the casing. Initially, the casing may be regarded as being impermeable to water inflow from the formation, preventing radionuclides to be released to the borehole. As corrosion progresses, the casing becomes leaky and radionuclides may penetrate and migrate in the borehole. When the casing is completely corroded, the borehole collapses and finally self-seals due to swelling and disintegration of the unloaded Opalinus Clay in the vicinity of the borehole. Experiments show that the hydraulic conductivity of boreholes with sedimented debris of claystone may be in the order of 10^{-7} to 10^{-8} m s^{-1} or even lower (Nagra 2002e). In the model calculations, it is pessimistically assumed that complete self-sealing of the borehole does not occur and that a steady-state water flux through the borehole is established.

Water flow in the borehole is driven by the hydraulic gradient between the local aquifers (directed upwards from the Sandsteinkeuper to the Wedelsandstein Formation). No overpressure is assumed in the Opalinus Clay. Thus, the flow rate is primarily determined by the hydraulic conductivity of the borehole, which is assumed to be time-independent. A wide range of steady-state flow rates along the borehole is considered, ranging from 10^{-4} m^3 a^{-1} to 1 m^3 a^{-1}, assuming unlimited availability of water in the penetrated Sandsteinkeuper and hydraulic conductivities of the borehole of 10^{-10} to 10^{-6} m s^{-1}.

Transport of radionuclides upwards through the borehole into the Malm aquifer, and from there to the reference biosphere area is conservatively assumed to be instantaneous; i.e. no retention is considered.

Three cases are considered (Fig. 7.6-1):

- the borehole penetrates a SF / HLW emplacement tunnel between two canisters,
- the borehole penetrates a SF canister (direct hit),
- the borehole penetrates an ILW-1 emplacement tunnel (direct hit).

Base Case and parameter variations

Penetration of a SF / HLW emplacement tunnel between two canisters (near-hit)

The borehole is assumed to penetrate a SF / HLW emplacement tunnel midway between 2 canisters, shortly after canister failure (see Fig. 7.6-1). Two cases are considered: i) The near field release of 2 canisters is discharged directly into the borehole (Base Case); ii) the near field release of 4 canisters is discharged directly into the borehole. Radionuclide retardation along the emplacement tunnel is conservatively neglected. In both cases, the radionuclides from all the other canisters are assumed to be released to the unperturbed clay matrix, as in the Reference Case, but their dose contribution is not taken into account in the present case.

For the SF / HLW source term, a reference near field is assumed, with the exception of a changed boundary condition at the interface near field / geosphere using a range of flow rates (Base Case: 10^{-2} m^3 a^{-1}, parameter variations of 10^{-4} and 1 m^3 a^{-1}). These water fluxes are shared between the affected canisters. This boundary condition realistically represents the hydraulic conditions for the two SF / HLW canisters in the vicinity of the borehole, whereas it is considered to be pessimistic for the canisters further away from the borehole (case of 4 affected canisters). Solubility limitations and sorption constants for reducing conditions are employed in the calculations. Radionuclides released from the near field are directly transported to the Reference Case biosphere.

Penetration of a spent fuel canister (direct hit)

The most severe case is that of a borehole drilled directly into a spent fuel canister. As long as the steel canister walls are entirely or partially intact, it is not considered possible, with current drilling technology, to drill through the walls without detection. A direct hit is, therefore, not to be expected before about 10^5 years, i.e. before a substantial part of the canister wall has been corroded away. In the model calculations, a mixed PWR-48 canister is assumed to be penetrated after 10^5 years. It is conservatively assumed that no release of radionuclides takes place before the perforation of the canister, and, after perforation, the release of dissolved radionuclides occurs directly from the canister to the reference biosphere area. The bentonite is assumed not to function as a diffusion barrier (no sorption) and solubility limitations are neglected. For the hydraulic boundary condition between near field and host rock, a range of water flow rates are used (Base Case: 10^{-2} m^3 a^{-1}, parameter variations of 10^{-4} and 1 m^3 a^{-1}). The remainder of the SF / HLW repository is assumed not to be affected by the borehole and its dose contribution is neglected in the present case.

Penetration of an ILW-1 emplacement tunnel (direct hit)

The borehole is assumed to penetrate one of the two ILW-1 emplacement tunnels 500 years after the end of emplacement, i.e. when the information record on the repository may have been lost. Due to the connected pore space of the cementitious materials, the entire emplacement tunnel is affected by the borehole, corresponding to half of the ILW-1 inventory. The dose contribution of the remaining radionuclides from the other emplacement tunnels is not taken into account in the present case.

In the calculation of the ILW source term, a loss of the diffusion barrier is postulated. As in the Reference Case, instantaneous mixing of radionuclides within the entire emplacement tunnel is assumed and solubility limitations and sorption constants for reducing conditions are employed in the calculations. The release of radionuclides from the near field is assumed to take place by advection through the borehole only, at a rate controlled by the flow rate in the borehole. Advective and diffusive transport of radionuclides through the Opalinus Clay is neglected. The water flux in the borehole is varied over a broad range: 10^{-2} m^3 a^{-1} (Base Case) and 10^{-4} and 1 m^3 a^{-1} (parameter variations). Radionuclides released from the near field are directly transported to the Reference Case biosphere without considering any retention.

Results

In the case of a borehole drilled into a SF emplacement tunnel midway between two canisters and shortly after canister breaching (10^4 a), the summed dose maximum is 3.3×10^{-5} mSv a^{-1} for the Base Case, with only 2 canisters affected by the borehole and a borehole water flow rate of 10^{-2} m^3 a^{-1}. For the investigated parameter variations, the summed dose maxima are 4.4×10^{-5} mSv a^{-1} (4 canisters affected by the borehole), 7.0×10^{-7} and 7.6×10^{-5} mSv a^{-1} (borehole water flow rates of 10^{-4} and 1 m^3 a^{-1}, respectively). For water flows in the order of 10^{-2} m^3 a^{-1} or lower, the resulting dose depends almost linearly on the flow rate. Significantly larger flow rates in the order of 1 m^3 a^{-1} cause an efficient reduction in concentration, and the dose levels off with increasing flow.

The corresponding summed dose maxima for HLW are 5.3×10^{-6} mSv a^{-1} (Base Case) and 2.0×10^{-7} to 1.3×10^{-5} mSv a^{-1} for the parameter variations (see Fig. 7.6-2).

A direct hit of a single SF canister (mixed PWR-48) by a borehole at 10^5 years after waste emplacement yields a summed dose maximum of 1.6×10^{-3} mSv a^{-1}, arising shortly after 10^5 years. For decreased and increased water flow rates in the borehole, the summed dose maxima are 1.3×10^{-4} and 2.5×10^{-3} mSv a^{-1}.

In the case of a borehole drilled directly into a ILW-1 emplacement tunnel, the summed dose maximum is 1.6×10^{-4} mSv a^{-1} (Base Case with a borehole water flow rate of 10^{-2} m^3 a^{-1}), and ranges from 1.6×10^{-6} to 1.4×10^{-2} mSv a^{-1} (for borehole water flow rates of 10^{-4} to 1 m^3 a^{-1}, respectively). Here, the dose maxima depend linearly on the water flow rate and occur some 700 to 1400 years after the time of borehole penetration.

7.6.3 Deep groundwater extraction from Malm aquifer

Main difference to Reference Case

This conceptualisation differs from the Reference Case in that radionuclide release to the biosphere is not assumed to take place in the Quaternary aquifer but in a deep well in the Malm aquifer used for extraction of drinking water.

Conceptual assumptions

Radionuclides are assumed to be released from the upper confining units into the catchment area of a deep well located in the Malm aquifer, where they are mixed with uncontaminated water from the Malm aquifer. The pumping rate is taken to be 300 $l\ min^{-1}$ ($1.6 \times 10^5\ m^3\ a^{-1}$), which represents a minimal rate for a viable exploitation of a drinking water well. For comparison, the lowest pumping rate from the Malm aquifer in a range of hydrogeothermal boreholes in southern Germany is 160 $l\ min^{-1}$, but that particular borehole is only used as a sampling point. All other boreholes have substantially larger pumping rates (Bertleff et al. 1988).

There is significant uncertainty related to the capture efficiency of the deep well considered in this assessment case, i.e. the fraction of radionuclides released from the repository to the Malm aquifer that are captured by the well. Because the source of radionuclides released to the Malm aquifer is distributed over an area as large as the repository area (plane source resulting from predominantly vertical release through the Opalinus Clay and upper confining units), the well is likely to catch only a small fraction of all radionuclides released. In an extremely pessimistic variant, a capture efficiency of 100 % is assumed.

The assumed pumping rate corresponds to a dilution that is about an order of magnitude lower than in the Reference Case biosphere. Radionuclide transport in the near field and geosphere is assumed to be identical to the Reference Case.

Base Case and parameter variation

In the Base Case, it is assumed that only 10 % of the radionuclides released from the repository to the Malm aquifer are captured by the well. The dose contribution of the remaining radionuclides is not taken into account in the present conceptualisation. In a parameter variation, the capture efficiency is conservatively taken to be 100 %.

Results

The summed drinking water dose maxima for the Base Case are 9.6×10^{-6} mSv a^{-1} (SF), 2.4×10^{-8} mSv a^{-1} (HLW) and 8.6×10^{-7} mSv a^{-1} (ILW), all of which are about a factor of 5 lower than the Reference Case dose. This can be explained qualitatively by the 10-fold lower dilution capacity of the well, its capture efficiency of 10 % and by the fact that the contribution of drinking water to the summed dose maxima of the Reference Case is in the order of 10 % only.

The resulting summed drinking water dose maxima for the considered parameter variation with a pessimistic capture efficiency of 100 % are 9.6×10^{-5} mSv a^{-1} (SF), 2.4×10^{-7} mSv a^{-1} (HLW) and 8.6×10^{-6} mSv a^{-1} (ILW).

7.6.4 Abandoned repository

Main difference to Reference Case

This conceptualisation differs from the Reference Case in that the repository is assumed to be abandoned in the observational period without proper backfilling/sealing of the access tunnel system. The emplacement tunnels and part of the operations/construction tunnels are, however, assumed to be fully backfilled/sealed during or shortly after waste emplacement.

Conceptual assumptions

After the abandonment of the repository, it is assumed that no maintenance activity of the open part of the access tunnel system will be undertaken (see Fig. 4.5-9). As a result of saturation and creep of the Opalinus Clay, the initially open lined tunnels collapse. This causes a triangular shaped progressive damage zone above the abandoned tunnels, with debris of rock and liner falling down into the tunnel.

The minimal transport distance (within properly backfilled/sealed tunnel segments) between any waste package, emplaced in the main facility or pilot facility, and the abandoned part of the tunnel is 100 m (Fig. 4.5-9). This minimal transport distance provides an efficient barrier against water flows (hydraulic barrier) and radionuclide migration (transport barrier), so that most radionuclides decay before they reach the abandoned part of the access tunnel system. Any non-decayed radionuclides arriving in the abandoned tunnel segments are conservatively assumed to be instantaneously released to the Wedelsandstein Formation. The model calculations of radionuclide transport are performed in the same way as in Section 7.4.7 (case related to the release of radionuclides affected by ramp/shaft), except for the conservative neglect of the transport barrier provided by the abandoned tunnel sections. Thus, the backfilled/sealed tunnel segments and the EDZ of the shaft are modelled as equivalent porous media, where radionuclide transport occurs at a rate controlled by the water flows in the tunnel system (calculated below) and by sorption. Once the radionuclides arrive in the Wedelsandstein Formation, instantaneous transport to the Reference Case biosphere is assumed.

The water fluxes in the abandoned SF / HLW / ILW repository are calculated analytically by means of a steady-state resistor network model, which is very similar to the model employed in Section 7.4.7 (Fig. 7.4-5). The major conceptual difference is that the inflow of water from the Opalinus Clay to the higher permeability collapsed tunnel segments is explicitly taken into account (for details see Nagra 2002c).

The results of the resistor network model are compared with the results from a 3 D hydrodynamic finite element model, for a hydraulic conductivity of the access tunnel segments of 10^{-8} m s^{-1} (Nagra 2002a). The numerical analysis is based on the case RLU02[109]. The comparison shows that the water flow rates calculated by the two models are in fair agreement. It is further demonstrated that for hydraulic conductivities of the access tunnel segments above 10^{-9} m s^{-1}, the water fluxes in the tunnel system are insensitive to any further increase of the conductivity (Nagra 2002c).

Results

In the conceptualisation related to the abandonment of the repository (no proper sealing / backfilling of part of the access tunnel system), the summed dose maxima are 4.7×10^{-5} mSv a^{-1} (SF), 1.3×10^{-7} mSv a^{-1} (HLW) and 3.6×10^{-6} mSv a^{-1} (ILW). Even under such unfavourable conditions as those assumed in the present conceptualisation, the performance of the repository is shown to be good. This is due to the significant residual radionuclide barrier effectiveness of the sealed tunnel segments.

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[109] The case RLU02 involves vertical groundwater movement driven by the hydraulic head difference between the Sandsteinkeuper and the Wedelsandstein formation and with higher permeability tunnel sections due to an abandonment of the repository (no consideration of overpressures in near field and geosphere).

7.6.5 Summary of results of the Scenario "Release of radionuclides affected by human actions"

The results for the different conceptualisations of the Scenario "Release of radionuclides affected by human actions" and of the parameter variations are summarised in Fig. 7.6-2.

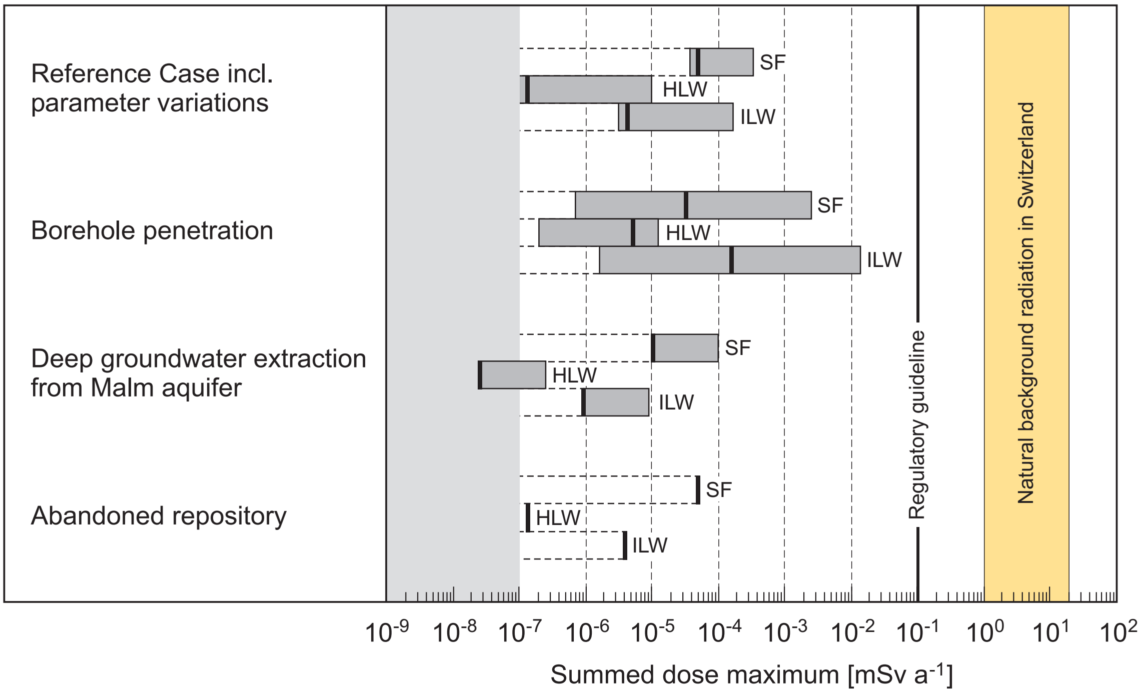

Fig. 7.6-2: Summed dose maxima and ranges for SF, HLW and ILW for the various conceptualisations and parameter variations of the scenario considering a release of radionuclides affected by human actions (Base Cases marked by bold lines)

The Reference Case results with parameter variations are also shown to allow an easy comparison.

7.7 "What if?" cases

7.7.1 Introduction

As discussed in Chapter 3, a "what if?" case is an assessment case addressing phenomena that are outside the range of possibilities supported by scientific evidence, but involve assumed perturbations to the key properties of the pillars of safety. "What if?" cases should be regarded as illustrations of the robustness of the disposal system under extreme conditions (see also Appendix 1), but the selection is not intended to be exhaustive.

The following "what if?" cases are considered in the present chapter:

- high water flow rate in the geosphere (100-fold increase),
- transport along transmissive discontinuities in the host rock,
- redox front penetration within the near field,
- increased fuel dissolution (10-fold and 100-fold increase),

- gas-induced release of dissolved radionuclides from ILW repository through ramp only,
- unretarded instantaneous transport of ^{14}C released as volatile species through the host rock,
- combination of poor near field performance, pessimistic near field geochemical dataset, pessimistic geosphere geochemical dataset and enhanced water flow in the geosphere,
- no advection in geosphere (diffusive transport only),
- increased cladding corrosion (spent fuel),
- zero sorption for iodine in near field and geosphere,
- decreased transport distance in Opalinus Clay.

The calculated summed dose maxima for the "what if?" cases are compared in Fig. 7.7-7 with those for the Reference Case.

7.7.2 High water flow rate in the geosphere

7.7.2.1 Deterministic analysis

Main difference to Reference Case

This conceptualisation differs from the Reference Case in that the upwards directed water flow rate in the Opalinus Clay is assumed to be hypothetically increased by a factor of 100. Such an increase is not supported by scientific evidence. The present assessment case is, therefore, regarded as a "what if?" case.

Results

As shown in Fig. 7.7-1, the summed dose maxima are 1.9×10^{-3} mSv a^{-1} (SF), 2.2×10^{-6} mSv a^{-1} (HLW) and 1.5×10^{-4} mSv a^{-1} (ILW), arising at about 6×10^{4} years (SF, ILW) and 1×10^{5} years (HLW). These doses are about a factor of 20-40 higher than the Reference Case doses. This reflects the fact that, in the case of the reference water flow rate in the Opalinus Clay, radionuclide transport is controlled by diffusion and advection, whereas at higher flow rates, radionuclide transport is dominated by advection and the doses increase linearly with the flow rate in the Opalinus Clay. This is also confirmed by insight calculations presented in Chapter 6.

7.7.2.2 Probabilistic analysis

The present "what if?" case has also been analysed probabilistically. The water flow rate in the Opalinus Clay has been set to 2×10^{-12} m s^{-1} (i.e. a 100-fold increase with respect to the Reference Case), while all other parameters have been sampled from their PDFs as in the probabilistic analysis of the Reference Conceptualisation (see Appendix 2, Tab. A2.13).

Fig. 7.7-2 shows the complementary cumulative density function of the dose for the present "what if?" case. The results for all three waste types are well below the Swiss regulatory guideline.

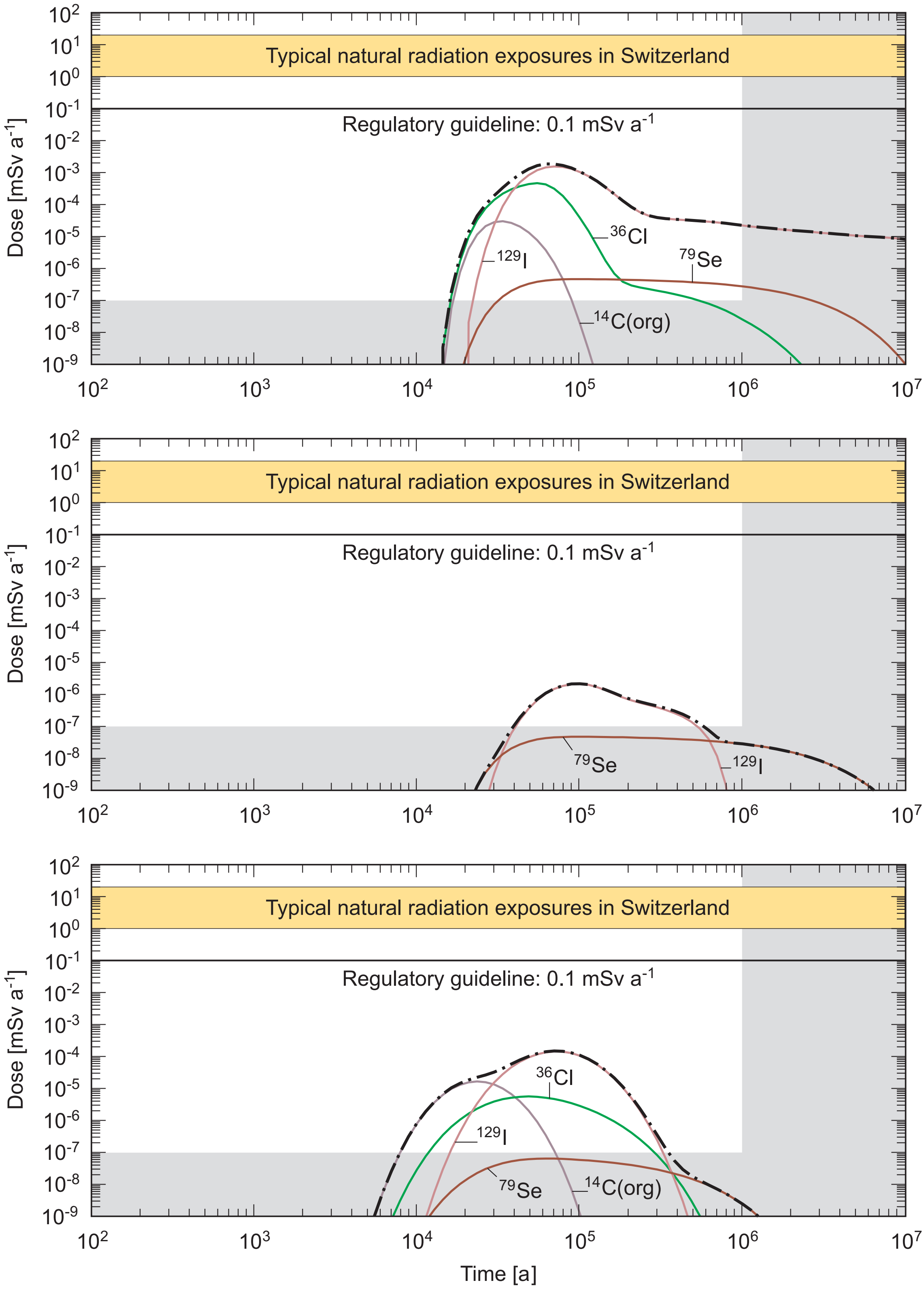

Fig. 7.7-1: Doses as a function of time for the "what if?" case "high water flow rate in the geosphere" (100 × Reference Case value)

Top: SF, middle: HLW, bottom: ILW.

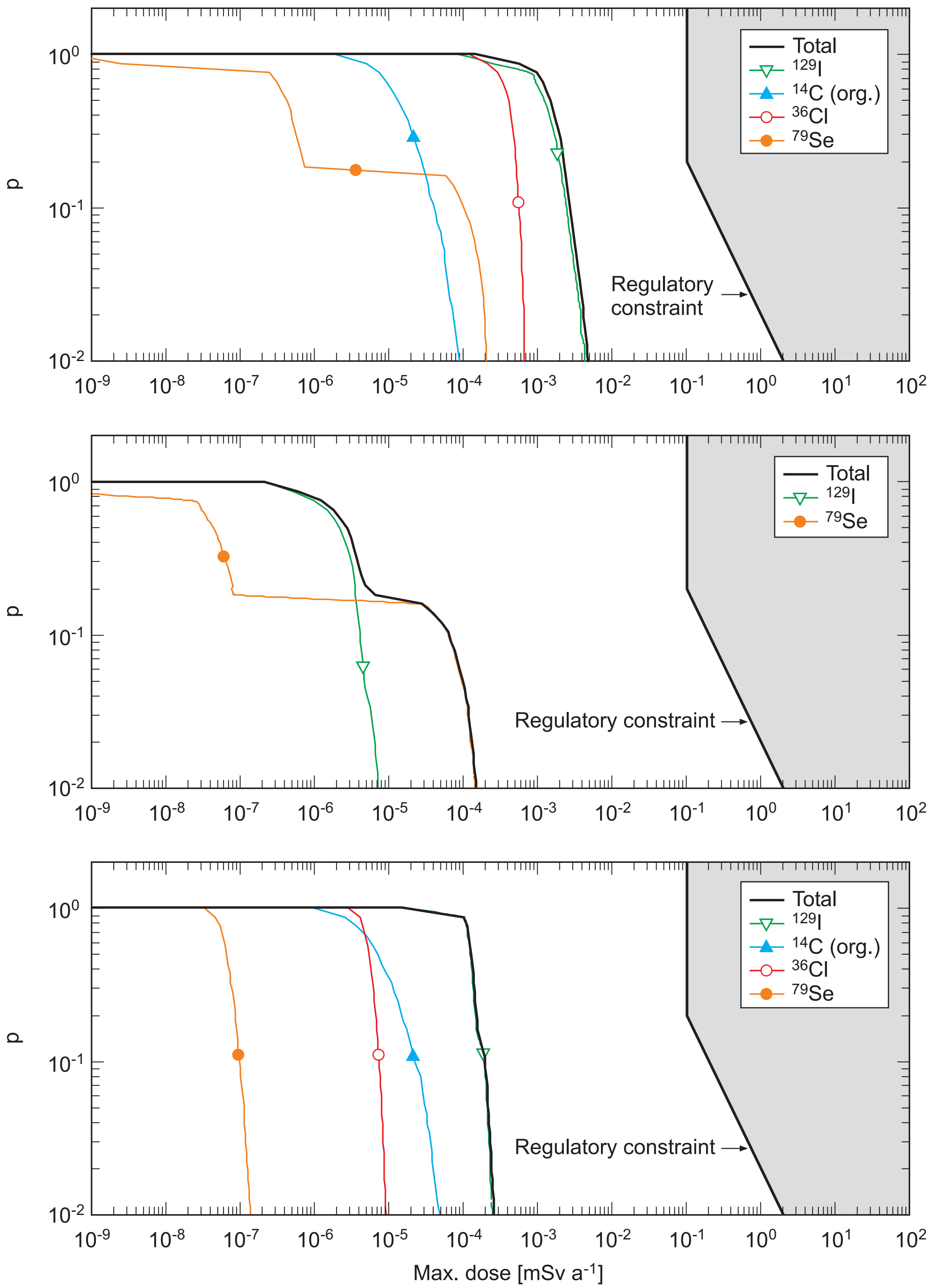

Fig. 7.7-2: Probabilistic analysis of the "what if?" case "high water flow rate in the geosphere" (100 × Reference Case value)

CCDFs for key radionuclides and for the sum over all safety-relevant radionuclides. Top: SF, middle: HLW, bottom: ILW.

7.7.3 Transport along transmissive discontinuities

7.7.3.1 Deterministic analysis

Main difference to Reference Case

In the Reference Case, any discontinuities present in the Opalinus Clay are considered not to be significantly hydraulically different from the intact Opalinus Clay rock matrix. In the present conceptualisation a small number of permeable discontinuities are hypothetically postulated to be present in the Opalinus Clay. These permeable features result in advective transport of radionuclides predominantly along the plane of the discontinuities, with matrix diffusion into the rock matrix. As discussed in Sections 5.2.2 and 5.5.3.2, no credible mechanism has been identified that could lead to discontinuities with enhanced hydraulic transmissivity. The present assessment case is, therefore, regarded as a "what if?" case.

Base Case and parameter variations

In the Base Case for this conceptualisation, a single water-conducting discontinuity with a transmissivity of 10^{-10} m^2 s^{-1} is assumed to be present in the host rock and a hydraulic gradient of 1 m m^{-1} is assumed.

In the case of SF / HLW, the discontinuity is assumed to affect all 27 emplacement tunnels. Near field release of 27 canisters (one canister from each intersected emplacement tunnel: 22 PWR mixed UO_2/MOX canisters + 5 BNFL HLW canisters) occurs along the discontinuity. In a parameter variation, the effect of two discontinuities intersecting all emplacement tunnels is investigated. Each discontinuity is assumed to convey the near field release of two canisters from each intersected emplacement tunnel, which results in the release from 108 canisters being transported along the two discontinuities. In further parameter variations, the hydraulic transmissivity of the discontinuities in both of the above cases is taken to be 10^{-9} m^2 s^{-1}.

In the case of ILW, a single discontinuity with a transmissivity of 10^{-10} m^2 s^{-1} is assumed to pass through all emplacement tunnels of ILW-1 and ILW-2. The axial transport of radionuclides within the emplacement tunnel to the discontinuity is assumed to occur without retardation. The near field release of the entire ILW part of the repository is pessimistically assumed to be discharged into the discontinuity, i.e. no release through the Opalinus Clay matrix is assumed. In a parameter variation, the hydraulic transmissivity of the discontinuity is taken to be 10^{-9} m^2 s^{-1}.

For the SF / HLW source terms, a reference near field is assumed, with the exception of a zero concentration boundary condition imposed at the interface between near field and host rock (which maximises the release of radionuclides from the near field) for the canisters affected by the transmissive discontinuity. For ILW, the water flow rate conveyed by the discontinuity is explicitly taken into account in the boundary condition for the near field calculations.

Results

Assuming that a single discontinuity intersects all emplacement tunnels (affecting one canister per SF / HLW tunnel, being equivalent to approximately 1 % of the tunnel length), the summed dose maxima are 1.3×10^{-4} mSv a^{-1} (SF) and 1.3×10^{-7} mSv a^{-1} (HLW), as shown by Fig. 7.7-3. If two discontinuities are present, each affecting 2 SF canisters per emplacement tunnel (4 % of tunnel length), then the summed dose maxima are 5.3×10^{-4} mSv a^{-1} (SF) and

1.3×10^{-7} mSv a^{-1} (HLW). Note that in the Base Case for SF, the maximal dose occurs shortly after canister breaching and is dominated by the contribution from the discontinuities (Fig. 7.7-3). In the case of HLW, the maximal dose occurs at approximately one million years and is dominated by the contribution from the release of radionuclides through the Opalinus Clay matrix, for both the Base Case and the parameter variation.

For the cases with an increased hydraulic transmissivity of 10^{-9} m^2 s^{-1}, the summed dose maxima are 6.5×10^{-4} mSv a^{-1} and 2.7×10^{-3} mSv a^{-1} (SF) and 4.0×10^{-6} mSv a^{-1} and 1.6×10^{-5} mSv a^{-1} (HLW), respectively. In all these cases the dose contribution from the discontinuities dominates over the contribution from the Opalinus Clay matrix.

The summed dose maxima for a discontinuity of transmissivity 10^{-10} m^2 s^{-1} and 10^{-9} m^2 s^{-1} affecting the entire ILW repository are 1.4×10^{-4} mSv a^{-1} and 1.1×10^{-2} mSv a^{-1}, respectively.

7.7.3.2 Probabilistic analysis

The present "what if?" case has also been analysed probabilistically. The transmissivity of the discontinuity has been set to 10^{-10} m^2 s^{-1}, while all other parameters have been sampled from their PDFs as in the probabilistic analysis of the Reference Conceptualisation (see Appendix 2, Tab. A2.13).

Fig. 7.7-4 shows the complementary cumulative density function of the dose for the present "what if?" case. The results for all three waste types are well below the Swiss regulatory guideline.

7.7.4 Increased fuel matrix dissolution in spent fuel

Main difference to Reference Case

Fuel matrix dissolution in the Reference Case is assumed to be governed by radiolytic oxidation. In the present conceptualisation the dissolution rate for spent fuel is hypothetically increased by factors of 10 and 100, both of which are considered to be outside the range of scientific evidence.

Results

The analysis of a 10-fold increase in spent fuel dissolution rates shows that the summed dose maximum for spent fuel is 1.9×10^{-4} mSv a^{-1}, which is about a factor of 4 higher than the dose for the Reference Case. For a 100-fold increase in spent fuel dissolution rates, the summed dose maximum for spent fuel is 5.0×10^{-4} mSv a^{-1}, about an order of magnitude higher than the Reference Case dose. These results show that the dose does not linearly increase with the fuel dissolution rate. Rather, the dose is controlled by a variety of parameters, such as the instant release fractions, solubility limitation etc.

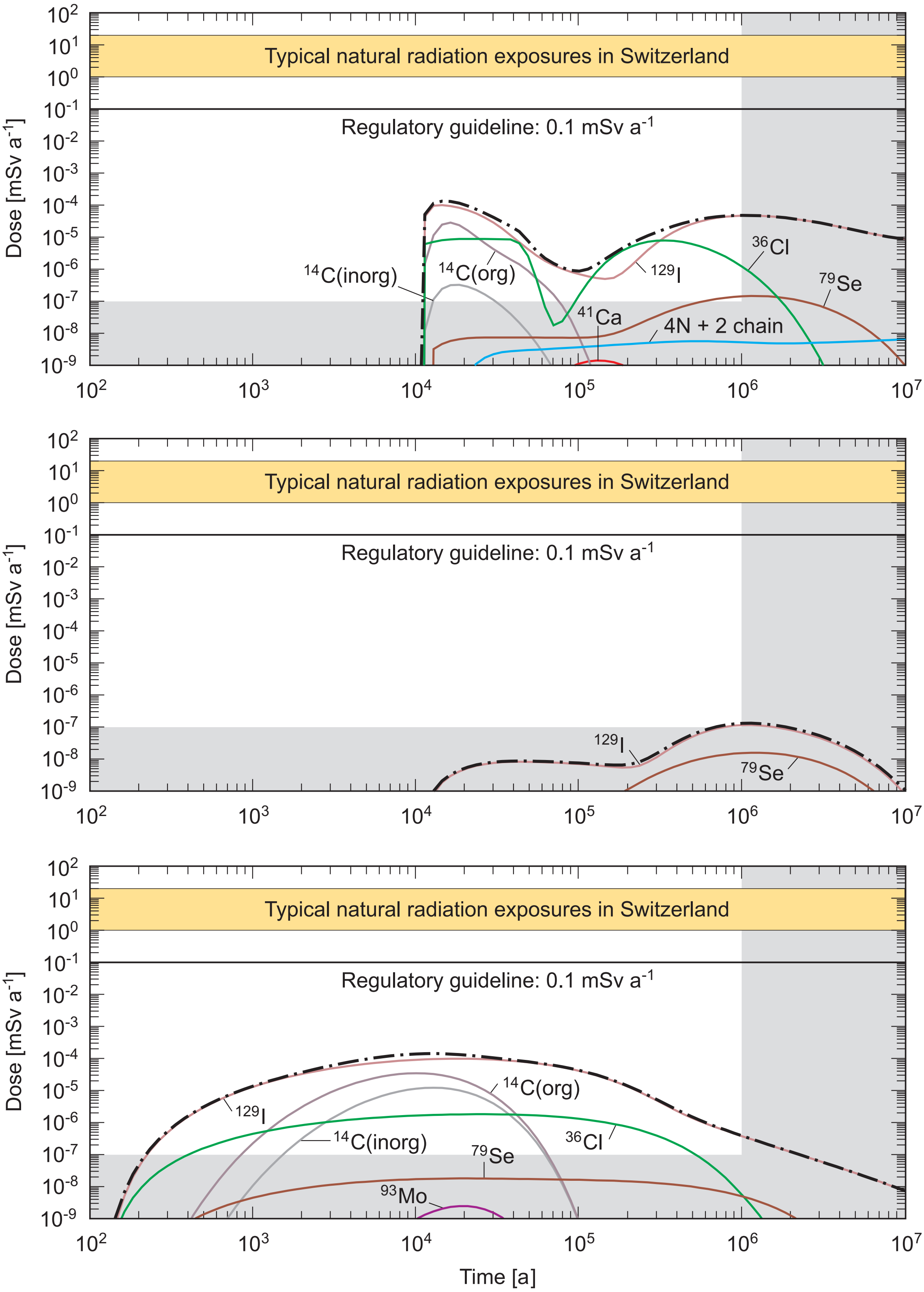

Fig. 7.7-3: Doses as a function of time for the "what if?" case "transport along transmissive discontinuities" (one discontinuity with a hydraulic transmissivity of 10^{-10} m^2 s^{-1})

Top: SF, middle: HLW, bottom: ILW.

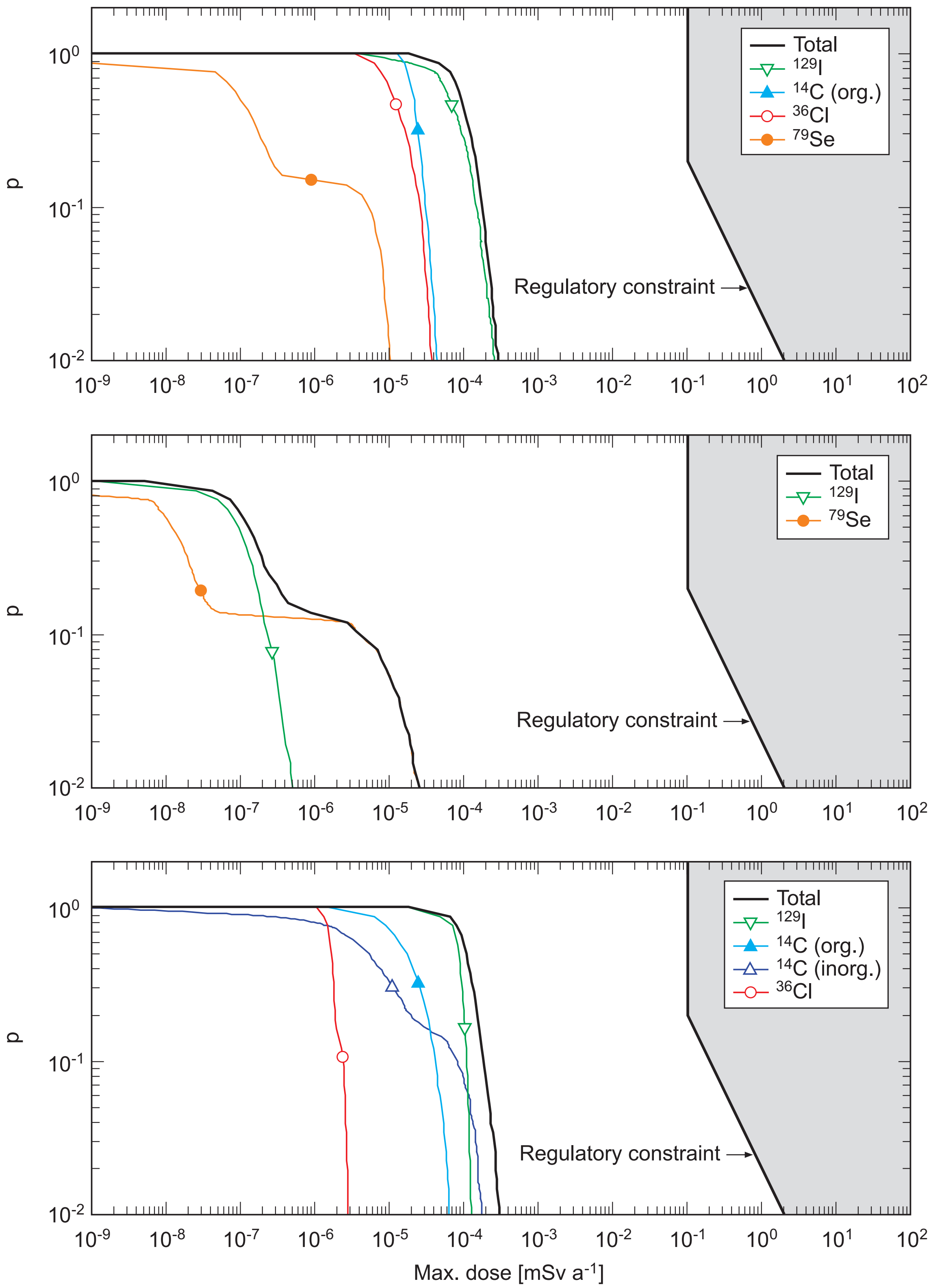

Fig. 7.7-4: Probabilistic analysis of the "what if?" case "transport along transmissive discontinuities" (one discontinuity with a hydraulic transmissivity of 10^{-10} m^2 s^{-1})

CCDFs for key radionuclides and for the sum over all safety-relevant radionuclides. Top: SF, middle: HLW, bottom: ILW.

7.7.5 Redox front penetration in the near field

7.7.5.1 Deterministic analysis

Main difference to Reference Case

In the Reference Case, reducing conditions are assumed throughout the SF / HLW / ILW near field and host rock. In the present conceptualisation for SF and ILW (but not for HLW where the generation of radiolytic oxidants is negligible), a redox front is hypothetically assumed to migrate throughout the near field, affecting sorption and solubility limitation in the near field, as well as the dissolution of the spent fuel matrix and cladding, both of which are increased by a factor of 10. For ILW-1, the high force compacted waste option is considered, because of its increased concentrations of α emitters (fuel debris), which could result in an increased concentration of radiolytic oxidants compared to the cemented waste option (see Section 7.8.3). The calculations for ILW-2 are identical to the Reference Case. In the Opalinus Clay, reducing conditions prevail, but solubility limitation in the Opalinus Clay is conservatively neglected. No parameter variations are considered.

Results

Considering oxidising geochemical conditions in the near field and reference geosphere conditions, the summed dose maxima are 1.9×10^{-4} mSv a^{-1} (SF) and 4.3×10^{-6} mSv a^{-1} (ILW). The doses for SF are shown in Fig. 7.7-5 as a function of time. Due to the increased spent fuel matrix and cladding dissolution rate, the dose for SF is increased by about a factor of 4 compared to the Reference Case dose, whereas the dose for ILW is identical to the Reference Case dose because it is dominated by ^{129}I, which is not sensitive to redox conditions.

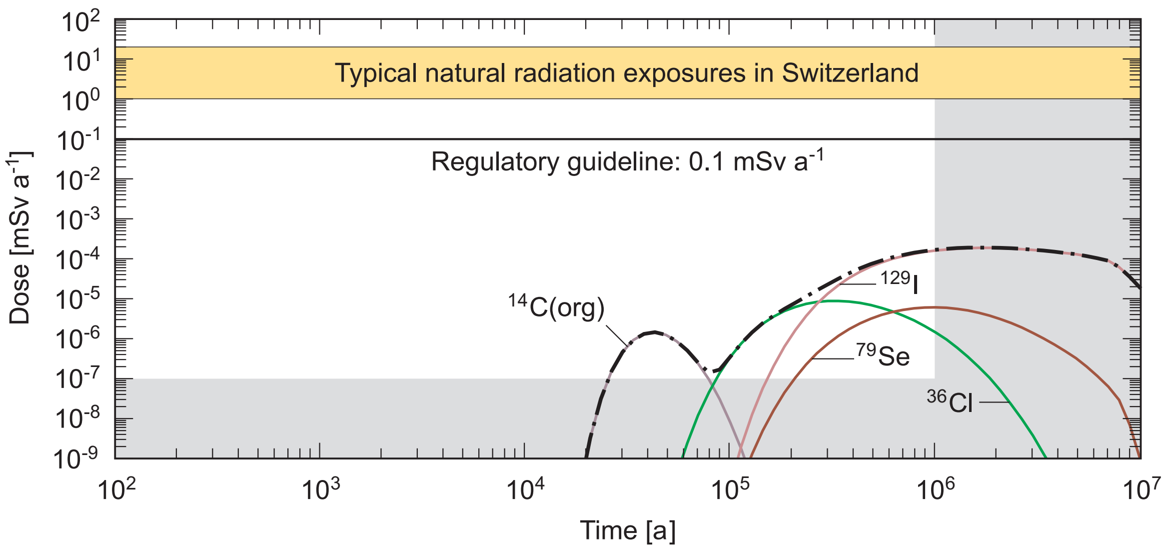

Fig. 7.7-5: Doses as a function of time for the "what if?" case "redox front penetration in the near field" for spent fuel

7.7.5.2 Probabilistic analysis

For spent fuel, the present "what if?" case has also been analysed probabilistically. The near field sorption and solubility limit parameters for the redox-sensitive elements have been set to their values for oxidising conditions and the spent fuel and cladding dissolution rate have been increased by a factor of 10, while all other parameters have been sampled from their PDFs as in the probabilistic analysis of the Reference Conceptualisation (see Appendix 2, Tab. A2.13). Fig. 7.7-6 shows the complementary cumulative density function of the dose for the present "what if?" case. Again, the results are well below the Swiss regulatory guideline.

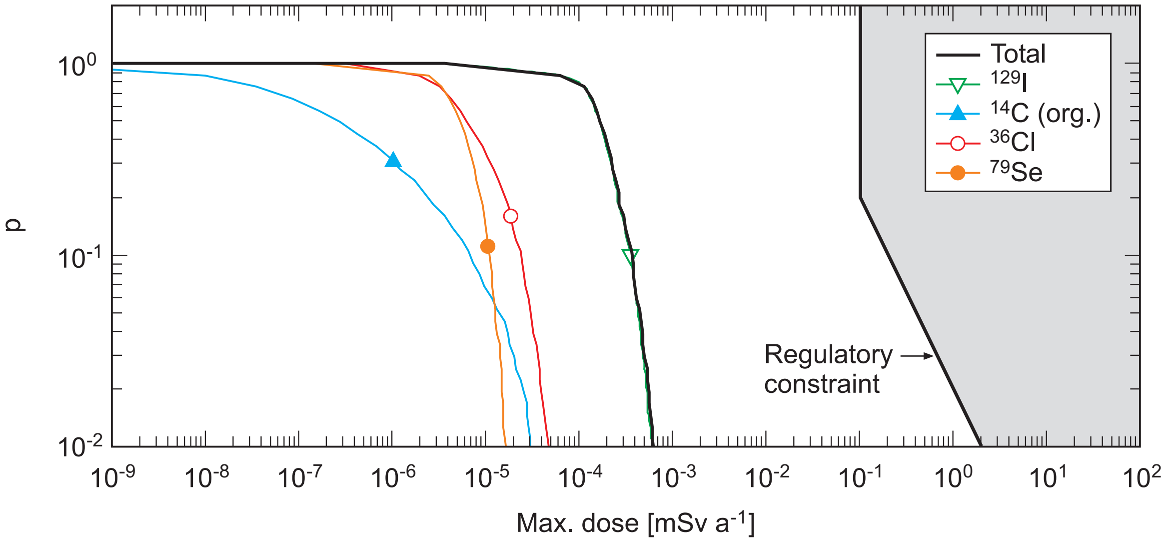

Fig. 7.7-6: Probabilistic analysis of the "what if?" case "redox front penetration in the near field" for spent fuel

CCDFs for key radionuclides and for the sum over all safety-relevant radionuclides.

7.7.6 Gas-induced release of dissolved radionuclides from ILW through the ramp only

Main difference to Reference Case

In the Reference Case, little or no displacement of contaminated porewater is assumed to occur along the emplacement tunnels and the ramp/shaft as a consequence of gas pressure build-up following gas generation. This conceptualisation for ILW differs from the Reference Case in that the possibility of accelerated release of dissolved radionuclides through the operations tunnel to the ramp is considered (gas-induced displacement of contaminated porewater). The host rock is postulated to be impermeable.

Conceptual assumptions

In the present conceptualisation, the possibility of accelerated release of radionuclides from the ILW repository along the ramp due to gas generation is considered. The model calculations are based on the assumption that transport of radionuclides takes place along the operations tunnel and ramp exclusively, i.e. that neither water is displaced in nor radionuclide transport occurs through the host rock. As in the case previously discussed in Section 7.4.9, the rate of gas-induced water displacement is assumed to be balanced by the volume of gas generated at repository pressure (approximately 10 MPa). This results in a water flow rate of 0.05 $m^3\ a^{-1}$ maintained for 30 000 years. The cumulative amount of water displaced is 1 500 m^3, corresponding to 50 % of the total pore space in the ILW emplacement tunnels. The near field source term and the transport pathway through the access tunnel system are identical to the conceptualisation of gas-induced release of radionuclides considered in Section 7.4.9. In both cases, the calculated dose considers only those radionuclides that are released through the tunnel.

Base Case and parameter variations

In the base case for ILW, a pulse release lasting from 10 000 to 40 000 years and a mean water flow rate of 0.05 m^3 a^{-1} is assumed, conveying 50 % of the total mobile radionuclide inventory in the ILW repository through the operations tunnel to the ramp. In a parameter variation, a pulse release lasting from 1 000 to 11 000 years and a mean water flow rate of 0.3 m^3 a^{-1} is assumed, conveying 100 % of the total mobile radionuclide inventory to the ramp.

Results

The summed dose maxima for ILW are 1.1×10^{-5} (Base Case) and 3.4×10^{-4} mSv a^{-1} (parameter variation), arising at 2.1×10^{5} and 1.8×10^{4} years, respectively.

7.7.7 Unretarded transport of ^{14}C as volatile species through the host rock

Main difference to Reference Case

In the Reference Case, organic and inorganic ^{14}C is assumed to dissolve in the porewater and to be transported through the Opalinus Clay by advection/diffusion. In the present conceptualisation, organic ^{14}C is assumed to be volatile and - after the available pore space in the near field has been filled with gas - to escape rapidly to the Wedelsandstein Formation, due to the postulated existence of a continuous gas pathway leading to instantaneous release through the Opalinus Clay. Gas accumulation in the Wedelsandstein Formation is taken into account.

Conceptual assumptions

In contrast to the cases considering tight and leaky seals (Sections 7.5.2 and 7.5.3), in the present conceptualisation it is assumed that transport of volatile ^{14}C occurs rapidly from the near field to the Wedelsandstein Formation: After filling the available pore space in the near field with gas, a continuous gas pathway is instantaneously formed. The instantaneous formation of such a gas pathway is not supported by scientific evidence, but would have an impact on the pillars of safety because it involves an instantaneous bypass of the Opalinus Clay. For this reason, the present assessment case is classified as a "what if?" case.

Gas accumulation takes place in the pore space within the near field (including EDZ) and in the directly surrounding Opalinus Clay (accessible by 2-phase flow), and, after the creation of a vertical continuous gas pathway, in the Wedelsandstein Formation, but neither in horizontal pathways in the Opalinus Clay (no pathway dilation) nor in the access tunnel system. All other conceptual assumptions and parameter values are identical to the models discussed in Sections 7.5.2 and 7.5.3.

Base Case and parameter variation

In all calculations, a steel corrosion rate of 1 μm a^{-1} is assumed. The gas permeability in the Opalinus Clay is varied. In the Base Case, a gas permeability of 10^{-23} m^2 is used. In the framework of two parameter variations, the effects of a changed gas permeability of

10^{-22} m^2 and 0 m^2 are investigated, reflecting uncertainty in the gas transport properties of Opalinus Clay.

In all cases, the dose contribution of non-volatile radionuclides is not taken into account.

Results

The summed drinking water dose maxima for the Base Case of the conceptualisation "unretarded transport of ^{14}C as volatile species through Opalinus Clay" are 4.9×10^{-5} mSv a^{-1} (SF) and 9.7×10^{-6} mSv a^{-1} (ILW), arising at about ten thousand years.

For a 10-fold increased gas permeability of 10^{-22} m^2, the summed dose maximum for SF and ILW are slightly lower than the Base Case doses, reflecting the effects of gas leakage and decay of ^{14}C in the Opalinus Clay before gas breakthrough to the Wedelsandstein Formation takes place. For zero gas permeability, the doses are nearly identical to the Base Case doses.

7.7.8 Poor near field performance, pessimistic near field and geosphere geochemical datasets combined with increased water flow in the host rock

Main difference to Reference Case

The objective of the present conceptualisation is to test the robustness of the repository system by collectively considering a number of pessimistic assumptions: Short canister life-time, high waste dissolution rates, pessimistic near field and geosphere geochemical datasets, increased water flow rates in the geosphere.

Conceptual assumptions

In the framework of the present "what if?" case, pessimistic assumptions are considered simultaneously for several pillars of safety. It is hypothetically assumed that the SF / HLW canister lifetime is reduced to 100 years and that the fuel and cladding dissolution rates are increased by a factor of 10 in comparison with the Reference Case. In the case of ILW, the containment time is reduced from 100 years to zero. At the same time, pessimistic near field and geosphere geochemical data (solubility limits, sorption constants) are used and in two parameter variations the geosphere water flow rate is increased with respect to its reference value.

Base Case and parameter variations

In the Base Case for this conceptualisation, the reference water flow rate in the geosphere is used. In two separate parameter variations, a 10-fold and 100-fold increase of the water flow rate is considered.

Results

The summed dose maxima for the Base Case of the conceptualisation "Poor near field and pessimistic near field/geosphere geochemical datasets" and for the two parameter variations for SF are 4.3×10^{-4} mSv a^{-1} (Base Case), 1.5×10^{-3} mSv a^{-1} (10-fold increased flow rate), 1.0×10^{-2} mSv a^{-1} (100-fold increased flow rate), respectively.

In the case of HLW, the resulting dose maxima are 1.1×10^{-5} mSv a^{-1} (Base Case), 4.4×10^{-5} mSv a^{-1} (10-fold increased flow rate), 9.6×10^{-5} mSv a^{-1} (100-fold increased flow rate). For ILW, the resulting dose maxima are 8.7×10^{-6} mSv a^{-1} (Base Case), 3.5×10^{-5} mSv a^{-1} (10-fold increased flow rate), 2.6×10^{-4} mSv a^{-1} (100-fold increased flow rate).

These dose maxima arise at about 5 - 6 $\times 10^5$ years (Base Case), 2 - 3 $\times 10^5$ years (10-fold increased flow rate) and 3 - 8 $\times 10^4$ years (100-fold increased flow rate).

7.7.9 No advection in the geosphere (diffusive transport only)

Main difference to Reference Case

In the present conceptualisation the Darcy flow rate in the geosphere is set to zero, so that the only radionuclide transport mechanism in the near field and the geosphere is diffusion.

Results

For the transport by diffusion only, the analysis shows that the summed dose maxima are 1.8×10^{-5} mSv a^{-1} (SF), 4.8×10^{-8} mSv a^{-1} (HLW) and 1.1×10^{-6} mSv a^{-1} (ILW). These doses are about a factor 3 (SF / HLW) and 4 (ILW) below the doses for the Reference Case.

7.7.10 Increased cladding corrosion rate in spent fuel

Main difference to Reference Case

In the present conceptualisation the cladding corrosion rate for spent fuel is hypothetically increased by a factor of 10 with respect to the Reference Case, which is considered to be outside the range of scientific evidence ("what if?" case).

Results

The analysis of a 10-fold increase in spent fuel cladding corrosion rate shows that the summed dose maximum is 4.8×10^{-5} mSv a^{-1}, which is identical to the dose for the Reference Case. This result is explained by the fact that the summed dose maximum of spent fuel is dominated by the instant release fraction of ^{129}I which is not affected by the cladding corrosion rate.

7.7.11 Zero sorption for iodine in near field and geosphere

Main difference to Reference Case

In contrast to the Reference Case, where the assumed sorption values (K_d) for iodine in bentonite, cement and Opalinus Clay are 5×10^{-4}, 1×10^{-3} and 3×10^{-5} m^3 kg^{-1}, in the present conceptualisation the sorption values for iodine are set to zero both in the near field and in the geosphere. This is outside the range of scientific evidence (“what if?” case).

Results

For zero sorption values for iodine in bentonite, cement and Opalinus Clay, the summed dose maxima are 1.1×10^{-4} mSv a^{-1} (SF), 2.5×10^{-7} mSv a^{-1} (HLW) and 9.7×10^{-6} mSv a^{-1} (ILW). These dose maxima are about a factor of two higher and occur earlier than the dose maxima for the Reference Case. This can be explained by the lower retention for iodine in Opalinus Clay compared to the Reference Case.

7.7.12 Decreased transport distance in Opalinus Clay

Main difference to Reference Case

In the Reference Case, the transport distance in the Opalinus Clay is assumed to be at least 40 m. To test the robustness of the disposal system, the transport distance is reduced to 30 m in the present conceptualisation, which is outside the range of scientific evidence for all waste packages emplaced ("what if?").

Results

For the case of a decreased transport distance in Opalinus Clay, the summed dose maxima are 8.1×10^{-5} mSv a^{-1} (SF), 2.1×10^{-7} mSv a^{-1} (HLW) and 7.3×10^{-6} mSv a^{-1} (ILW), i.e. less than a factor of 2 increased in comparison to the Reference Case doses.

7.7.13 Summary of results of the "What if?" cases

The results for the different "What if?" cases are summarised in Fig. 7.7-7.

7.8 Design and system options

7.8.1 Introduction

In this section, various design and system options related to the amounts and types of wastes and to an alternative canister design (steel canister with copper shell) are presented. The calculated summed dose maxima of the considered design and system options are compared in Fig. 7.8-1 with the doses for the Reference Case.

7.8.2 Increased waste arisings

Main difference to Reference Case

The Reference Case is based on a 192 GWa(e) scenario for Switzerland, assuming a lifetime of 60 years for the existing nuclear power plants. The present alternative conceptualisation serves for demonstrating the flexibility of the repository system with respect to increased amounts of radioactive waste, corresponding to a 300 GWa(e) scenario.

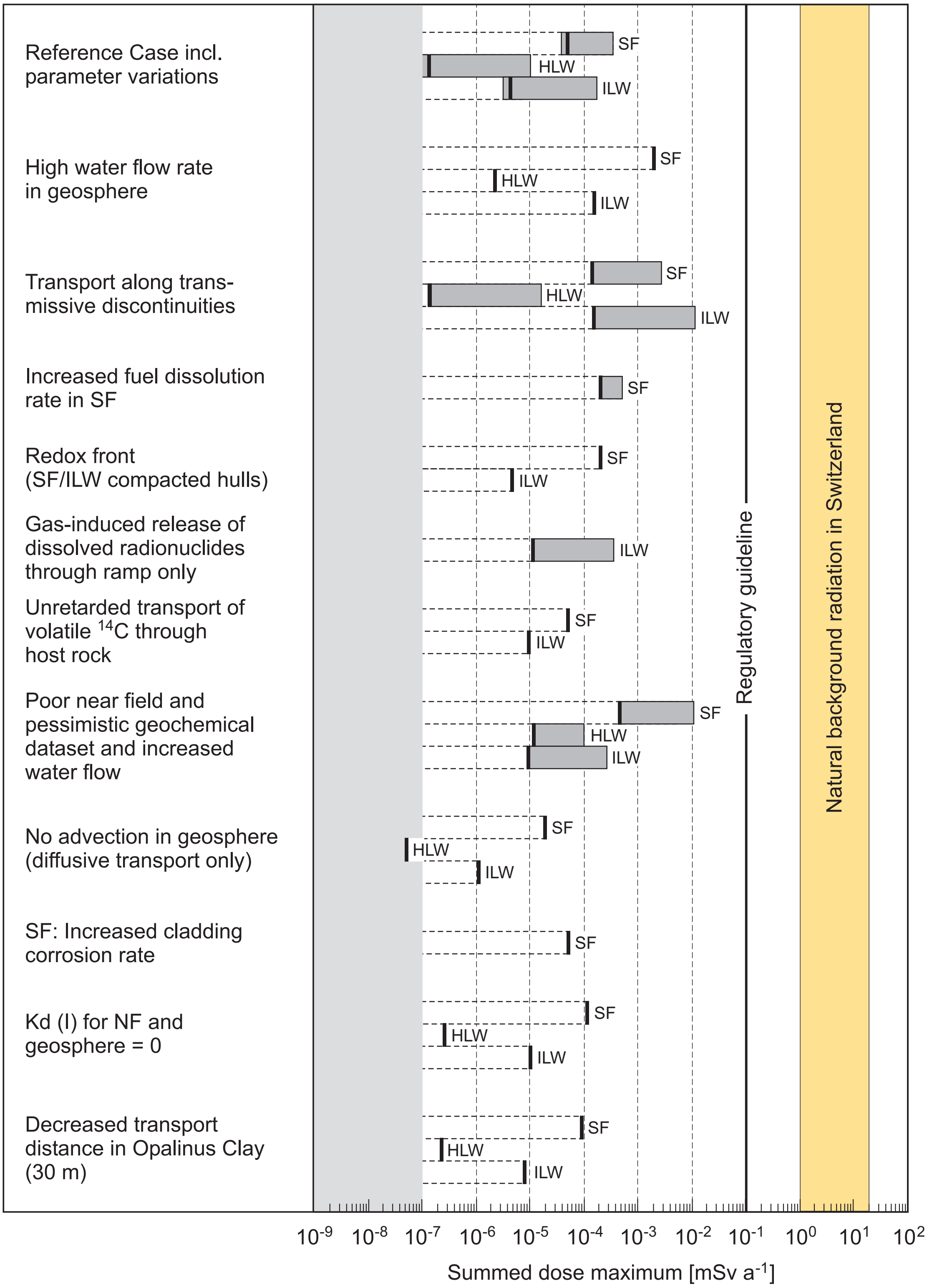

Fig. 7.7-7: Summed dose maxima and ranges for SF, HLW and ILW for the various conceptualisations and parameter variations considered as "what if?" cases

Base Cases are marked by bold lines. The Reference Case results with parameter variations are also shown to allow an easy comparison.

Conceptual assumptions

The assumed amount of waste arising as SF and HLW in the 192 GWa(e) scenario for Switzerland is 3217 t_{IHM} and 292 t of glass, representing reprocessing of 1195 t_{IHM} of SF (Reference Case). In the case of the 300 GWa(e) scenario, the extra SF is assumed to be directly disposed of without reprocessing, yielding 5576 t_{IHM} as SF, which is a factor of about 1.7 more than in the case of the 192 GWa(e) scenario (Tab. 4.5-1). The mass of HLW is identical in the two scenarios.

No parameter variations are performed.

Results

The summed dose maximum for a 300 GWa(e) scenario directly scales with the mass of SF to be disposed of (number of SF canisters) and is, therefore, a factor of 1.7 higher compared to a 192 GWa(e) scenario. Doses for HLW and ILW are not affected (no additional reprocessing).

7.8.3 ILW high force compacted waste option

Main difference to Reference Case

This conceptualisation differs from the Reference Case in that an alternative waste conditioning method for part of the ILW is considered, in which the hulls and end-pieces and ILW technological wastes of the COGEMA are treated by high-force compaction.

The ILW-1 inventory for the high force compacted waste option is given in Nagra (2002c). The calculations for ILW-2 are identical to the Reference Case. No parameter variations are performed.

Results

The summed dose maximum for the high force compacted waste option for the ILW-1 inventory is 4.3×10^{-6} mSv a^{-1}, which is identical to the dose obtained for the cemented waste option for the ILW inventory. In both cases, the contribution of ILW-2 is included in the dose.

7.8.4 Spent fuel canister with a copper shell

Main difference to Reference Case

The reference design concept involves cast steel canisters for SF / HLW (Section 4.5.3). In the present conceptualisation, a design option for SF with a cast steel insert and a copper shell is considered.

Conceptual assumptions

This conceptualisation is based on the assumption that all spent fuel canisters consist of an inner container of cast iron and a shell of copper (design option). The cast iron insert provides mechanical stability and the copper shell protects against corrosion in the repository environment. The design lifetime of such a composite canister is 100 000 years. Simultaneous breaching of all spent fuel canisters is assumed after 100 000 years, neglecting the residual transport resistance of the breached canisters (Base Case).

There is a remote possibility that a few spent fuel canisters will have small defects that escape detection during inspection and which could affect canister integrity (e.g. through-wall or near through-wall welding defects). Such failures are considered to be extremely unlikely. In a variant it is assumed that a single spent fuel canister (containing 3 PWR UO_2-48 assemblies and 1 PWR MOX-48 assembly) in the repository fails prematurely as a result of an undetected manufacturing defect (see Section 5.3.4.4). It is pessimistically assumed that the defect allows water ingress and radionuclide release instantaneously upon repository closure, and that resaturation of the repository and its surroundings has already occurred by this time.

Base Case and parameter variations

In the Base Case for this conceptualisation, it is assumed that all spent fuel canisters are breached simultaneously at 100 000 years. Alternatively, it is assumed that there is a single defect in one spent fuel canister, having a cross-sectional area of 4 mm^2. Full breaching of the initially defective canister as well as of all other canisters is assumed to take place at 100 000 years. In a further parameter variation the size of the defect is increased to 50 mm^2.

Results

For a spent fuel canister lifetime of 100 000 years, the breakthrough time to the biosphere is correspondingly enlarged by 100 000 years. The summed dose maximum is reduced by only about 10 % when compared to the reference spent fuel canister lifetime of 10 000 years, because the dose maximum is due to the long-lived ^{129}I.

The analysis of an initial defect in the composite canister results in summed dose maxima for a single canister of 1.5×10^{-9} mSv a^{-1} (small pinhole) and 4.9×10^{-9} mSv a^{-1} (large pinhole) which is a factor of 17 and 5 lower, respectively, than the dose obtained for a single canister with a lifetime of 10 000 years in the Reference Case. This dose reduction is due the residual transport resistance of the pinhole which causes a significant spreading in time of the instant release fraction. The times of occurrence of the summed dose maxima of all three cases are nearly identical to the Reference Case.

7.8.5 Summary of results for the design and system options

The results for the different design and system options are summarised in Fig. 7.8-1.

7.9 Illustration of effects of biosphere uncertainty on calculated dose

7.9.1 Introduction

As discussed in Chapter 2, the uncertainties related to the biosphere are of a different quality compared to other uncertainties, in that, similarly as in the case of future human actions, any statement about them is highly speculative. These uncertainties are also different in that they do not affect the isolation and retention capability of the repository system; they only affect the fate of any radionuclides that are released into the environment. They are evaluated separately from those related to the EBS and the geosphere by analysing a broad range of credible biosphere situations using the Reference Case near field and geosphere conceptualisations and parameters.

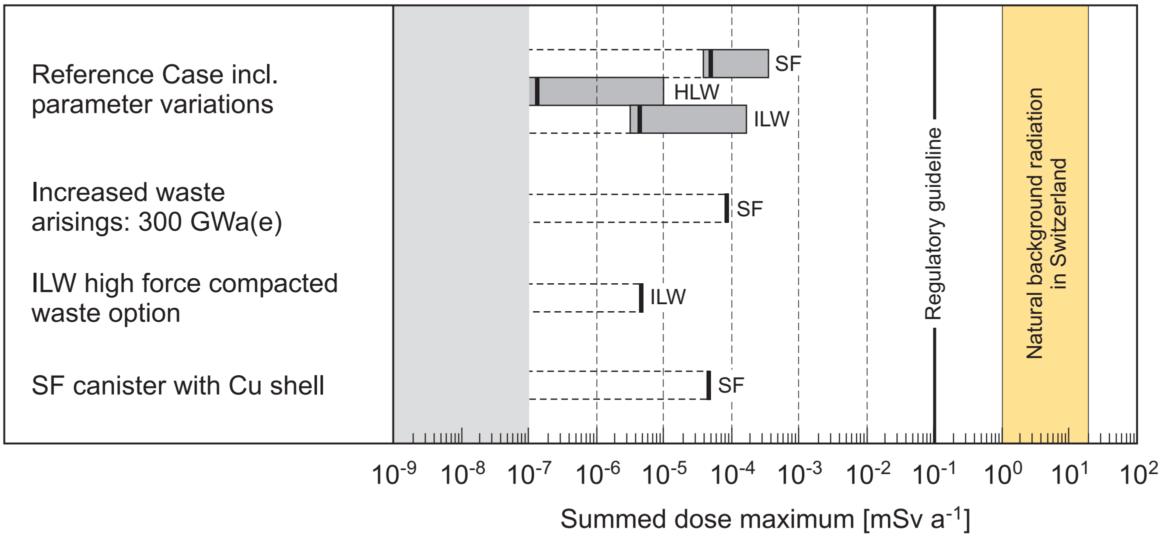

Fig. 7.8-1: Summed dose maxima and ranges for SF, HLW and ILW for the various conceptualisations and parameter variations considered as design and system options (Base Cases marked by bold lines)

The Reference Case results with parameter variations are also shown to allow an easy comparison.

A variety of such stylised representations for the characteristics and evolution of the surface environment are considered to illustrate the effect of biosphere uncertainty on calculated doses. Special emphasis is put on alternative geomorphological and climatic conditions. Human behaviour and diet are assumed to be identical to the Reference Case, i.e. no specific parameter variations are performed addressing alternative human habits and diets, except in the assessment case addressing the periglacial climate state.

In all stylised representations of the biosphere, including the Reference Case biosphere, five main compartments are considered, in which instant mixing of radionuclides is assumed to occur. These compartments are the Quaternary aquifer, a deep soil layer, a top soil layer, surface water and aquatic sediment. Transport of radionuclides in the surface environment occurs as a result of movements of water (radionuclides in solution) and movements of solid materials (radionuclides sorbed onto the solid phase).

The dose calculations are performed using the computer code TAME, resulting in time-dependent dose for all safety-relevant radionuclides. The calculated summed dose maxima are compared in Fig. 7.9-1 with the doses for the Reference Case.

In Chapter 6, the effects of uncertainty related to the biosphere conditions are further illustrated by means of nuclide-specific biosphere dose conversion factors (dose per unit activity release into biosphere calculated for steady-state conditions). In Fig. 6.7-10, these factors are shown for a range of different radionuclides and biosphere cases.

7.9.2 Alternative geomorphology

Main difference to Reference Case

In the Reference Case, radionuclide exfiltration is assumed to take place in the Quaternary aquifer within a section of a river valley that is characterised by net erosion, similar to the present-day Rhine valley just below the Rhine Falls. The present conceptualisation addresses the effects of alternative geomorphological conditions on radionuclide transport in the biosphere. These include a section of a river valley where net deposition takes place (sedimentation area), an area of wetlands and a spring located at the valley side where deep groundwater exfiltrates that is used for drinking water purposes.

Conceptual assumptions

Different geomorphological conditions are investigated, with the aim of illustrating the effects of uncertainty in the evolution of the surface structure and hydrological conditions on radionuclide transfer and uptake in the biosphere and on the resulting dose. In all variants considered, climatic conditions and radionuclide release rates at the geosphere / biosphere interface are assumed to be identical to the Reference Case.

In the Base Case, reference geomorphological conditions are considered. This case is identical to the Reference Case.

In a first variant, it is assumed that exfiltration occurs in an equivalent situation as for a larger side valley south of the present-day Rhine (e.g. Thur). The investigated area is characterised by net deposition (sedimentation area), i.e. significant accumulation of solid material transported from upstream sections of the river to the model area is assumed to occur. The top of the water table is close to the surface (a few meters), allowing flooding of agricultural land and capillary rise in soils. Human habits and exposure pathways are taken to be identical to those in the Reference Case.

In a second variant, the effects of exfiltration of radionuclides to a river with subsequent flooding in an area of wetlands is investigated. The biosphere area is assumed to consist of open water (small river), marsh and agricultural land. Frequent flooding of the agricultural land is considered to occur, with an assumed seepage rate from the top soil into the deeper soil strata of 1.0 m a^{-1}. Strong radionuclide sorption in the soils is assumed due to the relatively high organic content. The agricultural productivity depends on various factors, including type and water content of soils, human activities (active drainage, use of fertilisers, etc.). In wetter areas, straw production and pasture land is predominant and the productivity is low while in more desaturated areas a wider range of agricultural production with higher yields is possible, as in the Reference Case. In this conceptualisation, the agricultural practices are restricted to cattle raising and milk production.

In a further variant, the effects of exfiltration of deep groundwater from the Malm aquifer to a spring located at the valley side are investigated. Although such a situation can be ruled out for present-day hydrogeological conditions, it cannot be completely excluded in the long term. The spring is assumed to be used for the extraction of drinking water only. The production rate is taken to be 10^{-3} m^3 s^{-1}, a conceivable value for natural discharge of Malm groundwater from the south in the area of interest. It is assumed that 100 % of the discharged water is deep groundwater, i.e. no mixing with near-surface (meteoric) water is conservatively assumed. The capture efficiency (the fraction of radionuclides released from the repository to the Malm aquifer that is captured by the spring) is assumed to be 10 %. The dose is calculated for the drinking water exposure pathway only, other exposure pathways are considered to be negligible.

Results

The case considering reference geomorphological conditions is identical to the Reference Case, the doses are thus identical.

The analysis of radionuclide exfiltration into a sedimentation area, located in a larger tributary valley south of the Rhine, shows that the summed dose maxima are 2.3×10^{-5} mSv a^{-1} (SF), 6.0×10^{-8} mSv a^{-1} (HLW) and 2.0×10^{-6} mSv a^{-1} (ILW). Due to a higher dilution rate by sedimentation of uncontaminated solid materials on the top soil of the tributary valley, compared to the case of the Rhine valley, the doses are approximately a factor of 2 lower than in the Reference Case.

For the variant considering radionuclide release to an area of wetlands, the summed dose maxima are 4.4×10^{-6} mSv a^{-1} (SF), 1.2×10^{-8} mSv a^{-1} (HLW) and 3.9×10^{-7} mSv a^{-1} (ILW). These doses are about an order of magnitude lower than the Reference Case doses, because radionuclide exfiltration is assumed to occur into the river, leading to significant dilution.

For the variant related to exfiltration of deep groundwater from the Malm aquifer to a spring located at the valley side (capture efficiency 10 %), the summed dose maxima are 4.9×10^{-5} mSv a^{-1} (SF), 1.2×10^{-7} mSv a^{-1} (HLW) and 4.4×10^{-6} mSv a^{-1} (ILW). These values are nearly identical with the doses of the Reference Case.

7.9.3 Alternative climates

Main difference to Reference Case

The conceptualisation of the Reference Case is based on present-day climatic conditions that are typical for interglacial conditions. The present conceptualisation illustrates the effects that alternative climatic conditions may have on the radionuclide transport in the biosphere. The illustration includes a wet climate (wetter / warmer compared to present-day conditions) and a dry climate (drier / warmer than present-day conditions) as well as a periglacial climate.

Conceptual assumptions

As discussed in Section 5.2.1, a continuation of the glacial/interglacial cycling is very likely to take place in the next one million years. For all variants discussed here, radionuclide exfiltration to the Reference Case biosphere area is considered, based on Reference Case release rates from the near field via geosphere to the biosphere. The values for climate related parameters are summarised in Tab. 7.9-1. The water flow rates between the different compartments of the biosphere model area are calculated from these values based on mass balance considerations (for more details, see Nagra 2002c and 2003b).

Tab. 7.9-1: Climate-related parameters used in the modelling of the biosphere (parameter values for Reference Case biosphere area)

Parameter	Present-day climate	Wet climate	Dry climate	Periglacial climate
Precipitation rate (rainfall) [m a^{-1}]	1	2	0.5	low
Evapotranspiration rate [m a^{-1}]	0.6	1	1	low
Irrigation rate [m a^{-1}]	0.5	0.25	0.6	0

In a variant, a wet climate is investigated, in which rainfall is increased by a factor of 2, evapotranspiration is increased from 0.6 m a^{-1} to 1.0 m a^{-1}, and irrigation is reduced by a factor of 2 (Tab. 7.9-1).

In a further variant, a dry climate is considered, characterised by decreased rainfall, increased evapotranspiration and increased irrigation.

Low temperatures, low precipitation rates and permafrost are some of the characteristic features of the periglacial climate state. The corresponding radiological consequences are evaluated semi-quantitatively (Nagra 2002c). The diet of locally produced food is derived from natural and semi-natural environments (no agricultural activity, diet based on berries, mushrooms, fish and reindeer), in which activity accumulation is significantly higher compared to present-day conditions. However, in such environments a larger area is required for food production, leading to lower average activity concentrations in the food.

Base Case and parameter variations

The Base Case for this conceptualisation represents the present-day climate, which is characteristic for interglacial climatic conditions, and which is taken to be identical to the Reference Case. In the framework of parameter variations, the radiological effects of a wet climate and a dry climate are investigated, both of which may occur during interglacial periods. In a further variant, a semi-quantitative dose estimate is performed for the periglacial climate. All of these climatic variants are investigated for the geomorphological situation prevailing in the reference biosphere area.

Results

It is found that the highest doses are obtained for the dry climate, because dilution in the biosphere is significantly reduced during dry periods. For this climate state, the summed dose maxima are 5.3×10^{-4} mSv a^{-1} (SF), 1.6×10^{-6} mSv a^{-1} (HLW) and 4.6×10^{-5} mSv a^{-1} (ILW), which is about an order of magnitude higher in comparison with the Reference Case.

For a wet climate state, the summed dose maxima are 1.6×10^{-5} mSv a^{-1} (SF), 4.3×10^{-8} mSv a^{-1} (HLW) and 1.4×10^{-6} mSv a^{-1} (ILW), which corresponds to a decrease by about a factor of 3 in comparison with the Reference Case.

In the case of a periglacial climate state, the estimated range of doses is in the order of 1 % to 100 % of the doses obtained for the Reference Case.

7.9.4 Summary of results of the illustration of the effects of biosphere uncertainty

The results of the illustration of the effects of biosphere uncertainty on calculated dose are summarised in Fig. 7.9-1.

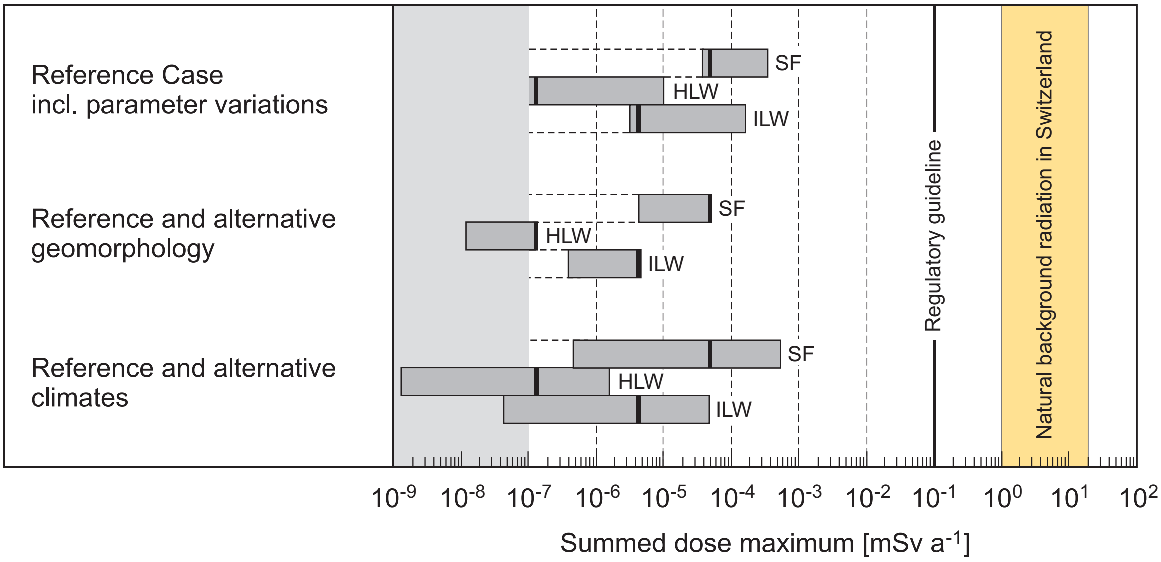

Fig. 7.9-1: Summed dose maxima and ranges for SF, HLW and ILW for the various conceptualisations and parameter variations considered as an illustration of the effects of biosphere uncertainty on calculated doses (Base Cases marked by bold lines)

The Reference Case results with parameter variations are also shown to allow an easy comparison.

7.10 Summary and evaluation of results

In Chapter 7, the results for the radiological impact of the repository system have been presented for those assessment cases requiring a quantitative analysis, organised according to Tab. 6.8-2.

The starting point is the Reference Case, based on:

(i) the Reference Conceptualisation of the Reference Scenario, which envisages a repository with a near field evolving according to the design functions of the engineered barriers, a geosphere based on the current understanding of the geological environment and a biosphere based on present-day geomorphological, hydrogeological and climatic conditions, with conservative assumptions regarding human behaviour and diet,

(ii) a set of physico/chemical parameters. Parameter variations to the Reference Case have been investigated where significant data uncertainty exists.

Alternative conceptualisations of the Reference Scenario address phenomena in near field and geosphere where uncertainty exists on their actual importance for the reference radionuclide release pathway. Data uncertainty within alternative conceptualisations is investigated by parameter variations.

The effects of uncertainty in the future evolution of the system is explored by means of alternative scenarios. As in the Reference Scenario, different conceptualisations and parameter variations are considered in the alternative scenarios.

A category of "what if?" cases has been introduced addressing phenomena that are outside the range of possibilities supported by scientific evidence, but involve perturbations to key properties of the pillars of safety. This group of cases serves to illustrate the robustness of the repository system, but is not intended to be exhaustive.

Design and system options are evaluated separately, because they address conceptualisations where flexibility, rather than uncertainty, exists in the characteristics of the repository system.

The sensitivity of radionuclide transport in the biosphere is illustrated by a number of assessment cases related to alternative geomorphological and climatic conditions. In the framework of this group of cases, the focus is on illustrating the effects of biosphere uncertainty on calculated doses using different (stylised) possibilities for the characteristics and evolution of the surface environment.

The results of the safety assessment, in terms of summed dose maxima for SF, HLW and ILW, are summarised in Tab. 7.10-1 (dose values) and graphically illustrated in Fig. 7.10-1 (dose ranges). Fig. 7.10-2 shows a different representation of the results. For each waste type, all cases analysed are represented in a scatter plot showing the summed dose maximum versus the time of occurrence of that dose. The symbols indicate to which scenario (according to the classification of Table 7.10-1) a given case belongs.

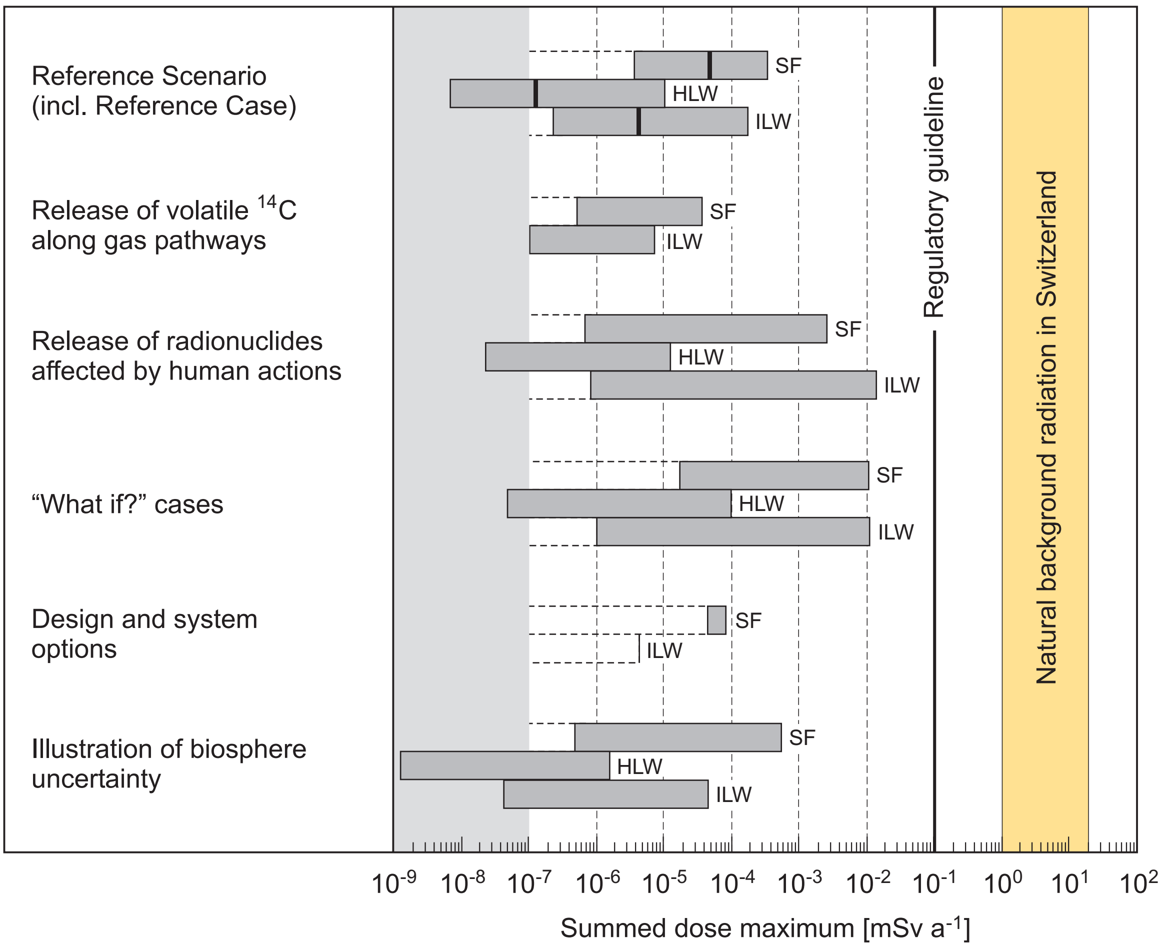

Fig. 7.10-1: Range of summed dose maxima for SF / HLW / ILW, spanned by all cases analysed for a given scenario

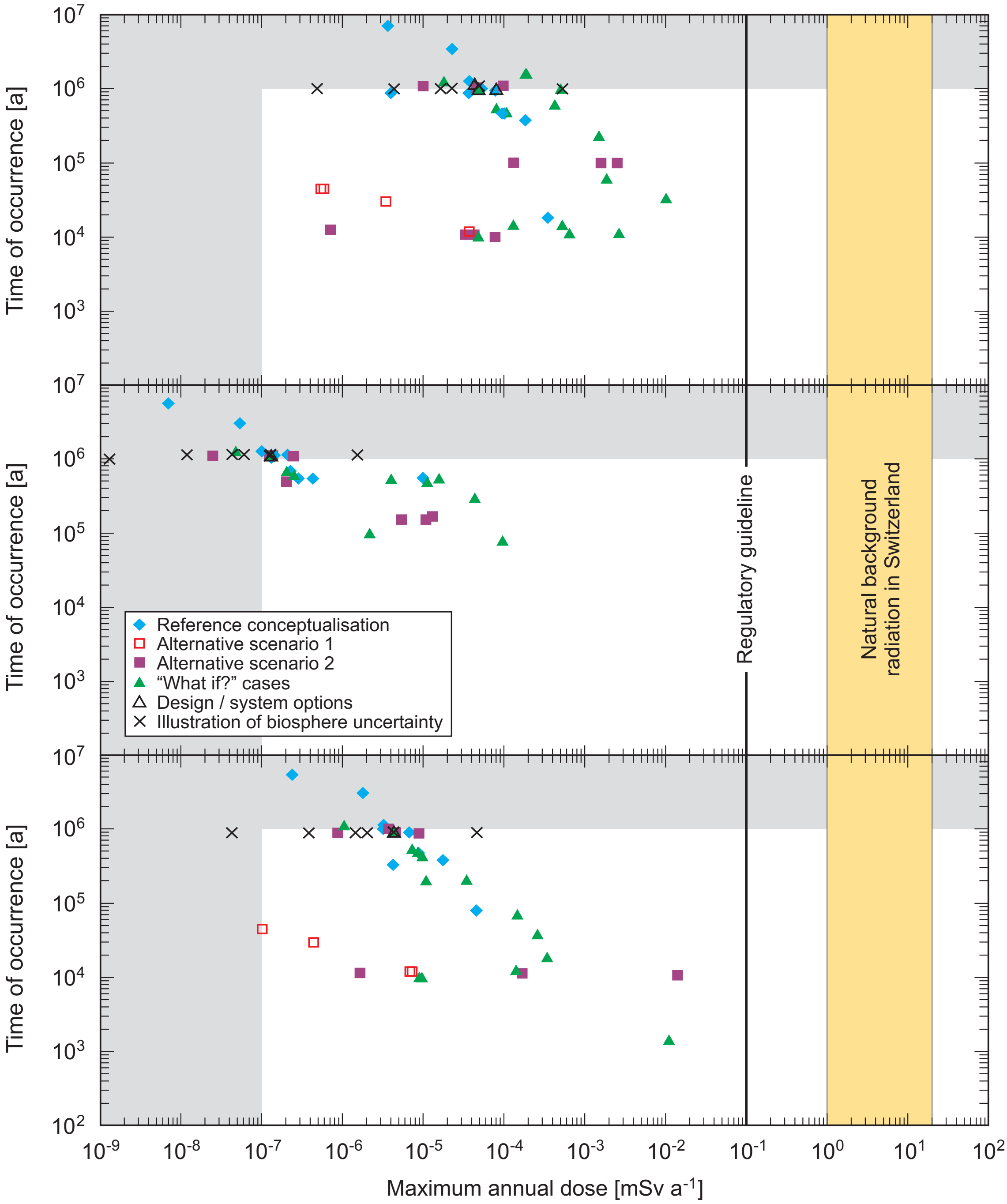

Fig. 7.10-2: Plot showing the summed dose maximum vs. the time of occurrence of that dose for all cases analysed

Top: SF, middle: HLW, bottom: ILW. The symbols indicate to which scenario a given case belongs. Note that while cases belonging to the reference conceptualisation and to the alternative scenario 1 (dose due to release of volatile ^{14}C along gas pathways) are considered to be realistic possibilities, those belonging to the alternative scenario 2 (human actions) and the "what if?" cases are in an entirely different category. The "human actions" cases should be seen as illustrative stylised conceptualisations, while the "what if?" cases are outside the range of possibilities supported by scientific evidence, but are included as examples to illustrate the robustness of the system behaviour under extreme conditions.

Tab. 7.10-1: Summary of summed dose maxima for the various scenarios, "what if?" cases, design and system options and illustration of biosphere uncertainty with associated conceptualisations and parameter variations that define the different assessment cases

Scenario	Conceptualisation	Parameter variation		Summed dose maximum [mSv a^{-1}] SF	HLW	ILW
1. Reference Scenario Release of dissolved radionuclides	**1.1 Reference Conceptualisation**	1.1a Reference Case (RC)		4.8×10^{-5}	1.3×10^{-7}	4.3×10^{-6}
		1.1b Variability in canister inventory[1]	9 BWR UO_2-48[2]	2.8×10^{-8}	-	-
			4 PWR UO_2-48[2]	1.6×10^{-8}	-	-
			3 PWR UO_2-48 + 1 PWR MOX-48[2]	2.5×10^{-8}	-	-
			4 PWR UO_2-55	3.6×10^{-8}	-	-
			3 PWR UO_2-55 + 1 PWR UO_2-65	4.4×10^{-8}	-	-
			3 PWR UO_2-55 + 1 PWR UO_2-75	5.2×10^{-8}	-	-
			3 PWR UO_2-48 + 1 PWR MOX-65	3.3×10^{-8}	-	-
			COGEMA[2]	-	1.6×10^{-10}	-
			BNFL[2]	-	2.1×10^{-10}	-
		1.1c Reduced canister lifetime		5.2×10^{-5}	1.3×10^{-7}	-
		1.1d Pessimistic near field geochemical dataset		5.1×10^{-5}	9.9×10^{-6}	4.3×10^{-6}
		1.1e Increased glass dissolution rate in HLW		-	1.3×10^{-7}	-
		1.1f Increased water flow rate in geosphere (10-fold increase)		1.9×10^{-4}	4.3×10^{-7}	1.8×10^{-5}
		1.1g Decreased water flow rate in geosphere (10-fold decrease)		3.7×10^{-5}	1.0×10^{-7}	3.3×10^{-6}
		1.1h Pessimistic geosphere sorption constants		9.4×10^{-5}	2.3×10^{-7}	8.7×10^{-6}
		1.1i Pessimistic near field and geosphere geochemical dataset		1.0×10^{-4}	1.0×10^{-5}	8.7×10^{-6}
		1.1j Pessimistic geosphere diffusion constants		3.5×10^{-4}	2.8×10^{-7}	1.7×10^{-4}
		1.1k Pessimistic treatment of ^{14}C (organic) in SF		4.8×10^{-5}	-	-
	1.2 Solubility-limited dissolution of SF	1.2a Base Case only		3.7×10^{-5}	-	-
	1.3 Bentonite thermal alteration	1.3a Base Case only		4.9×10^{-5}	1.5×10^{-7}	-
	1.4 Glacially-induced flow in the Opalinus Clay	1.4a Base Case only		7.7×10^{-5}	-	-
	1.5 Additional barrier provided by confining units	1.5a Vertical transport through confining units		2.3×10^{-5}	5.3×10^{-8}	1.8×10^{-6}
		1.5b Horizontal transport in local aquifers		3.6×10^{-6}	6.9×10^{-9}	2.4×10^{-7}
	1.6 Radionuclide release affected by ramp / shaft	1.6a Base Case		4.7×10^{-5}	1.3×10^{-7}	4.1×10^{-6}
		1.6b Increased hydraulic conductivity of EDZ (100-fold increase)		4.6×10^{-5}	1.2×10^{-7}	3.9×10^{-6}
	1.7 Convergence-induced release affected by ramp (ILW)	1.7a Steady-state hydraulics		-	-	4.3×10^{-6}
		1.7b Water pulse		-	-	3.2×10^{-6}
	1.8 Gas-induced release of dissolved radionuclides affected by ramp / shaft[3]	1.8a Base Case (ILW: 50 %, 0.05 $m^3 a^{-1}$)		4.0×10^{-6}	-	4.2×10^{-6}
		1.8b Increased water flow rate in ILW (100 %, 0.3 $m^3 a^{-1}$)		-	-	4.6×10^{-5}

Tab. 7.10-1: (Cont.)

Scenario	Conceptualisation	Parameter variation		Summed dose maximum [mSv a^{-1}] SF	HLW	ILW
2. Alternative Scenario 1 Release of volatile radionuclides along gas pathways	2.1 Release of ^{14}C from SF and ILW as volatile species in the gas phase not affected by ramp / shaft ("tight seals")[3]	Gas permeability [m^2]	2.1a 10^{-23}	5.8×10^{-7}	-	1.0×10^{-7}
			2.1b 10^{-22}	3.4×10^{-6}	-	4.3×10^{-7}
			2.1c 0	5.2×10^{-7}	-	1.0×10^{-7}
	2.2 Release of ^{14}C from SF and ILW as volatile species in the gas phase affected by ramp / shaft ("leaky seals")[3]	Gas permeability [m^2]	2.2a 10^{-23}	3.7×10^{-5}	-	7.2×10^{-6}
			2.2b 10^{-22}	3.6×10^{-5}	-	6.7×10^{-6}
			2.2c 0	3.7×10^{-5}	-	7.2×10^{-6}
3. Alternative Scenario 2 Release of radionuclides affected by human actions	3.1 Borehole penetration[3]	3.1a Near hit, 2 canisters / 1 ILW-1 tunnel affected		3.3×10^{-5}	5.3×10^{-6}	1.6×10^{-4}
		3.1b Near hit, 4 canisters affected		4.4×10^{-5}	1.1×10^{-5}	-
		3.1c Near hit, 2 canisters / 1 ILW-1 tunnel affected, increased water flow rate		7.6×10^{-5}	1.3×10^{-5}	1.4×10^{-2}
		3.1d Near hit, 2 canisters / 1 ILW-1 tunnel affected, decreased water flow rate		7.0×10^{-7}	2.0×10^{-7}	1.6×10^{-6}
		3.1e Direct hit[1]		1.6×10^{-3}	-	-
		3.1f Direct hit[1], increased water flow rate		2.5×10^{-3}	-	-
		3.1g Direct hit[1], decreased water flow rate		1.3×10^{-4}	-	-
	3.2 Deep groundwater extraction from Malm aquifer (production of well as dilution)[3]	Plume capture efficiency	3.2a 10 %	9.6×10^{-6}	2.4×10^{-8}	8.6×10^{-7}
			3.2b 100 %	9.6×10^{-5}	2.4×10^{-7}	8.6×10^{-6}
	3.3 Abandoned repository	3.3a Base Case only		4.7×10^{-5}	1.3×10^{-7}	3.6×10^{-6}
4. "What if?" cases to investigate robustness of the disposal system	4.1 High water flow rate in geosphere	4.1a Increased water flow rate in geosphere (100-fold increase)		1.9×10^{-3}	2.2×10^{-6}	1.5×10^{-4}
	4.2 Transport along transmissive discontinuities	4.2a/c 1 discontinuity ($T = 10^{-10}$ m^2 s^{-1}) affecting 27 SF/HLW canisters and entire ILW repository		1.3×10^{-4}	1.3×10^{-7}	1.4×10^{-4}
		4.2b 2 discontinuities ($T = 10^{-10}$ m^2 s^{-1}) affecting 108 SF/HLW canisters		5.3×10^{-4}	1.3×10^{-7}	-
		4.2d/f 1 discontinuity ($T = 10^{-9}$ m^2 s^{-1}) affecting 27 SF/HLW canisters and entire ILW repository		6.5×10^{-4}	4.0×10^{-6}	1.1×10^{-2}
		4.2e 2 discontinuities ($T = 10^{-9}$ m^2 s^{-1}) affecting 108 SF/HLW canisters		2.7×10^{-3}	1.6×10^{-5}	-
	4.3 SF: Increased fuel dissolution rate	4.3a 10-fold increase		1.9×10^{-4}	-	-
		4.3b 100-fold increase		5.0×10^{-4}	-	-
	4.4 Redox front (SF/ILW compacted hulls)	4.4a Base Case only		1.9×10^{-4}	-	4.3×10^{-6}
	4.5 ILW: Gas-induced release of dissolved radionuclides through the ramp only[3]	Water flow rate	4.5a 50 %, 0.05 m^3 a^{-1}	-	-	1.1×10^{-5}
			4.5b 100 %, 0.3 m^3 a^{-1}	-	-	3.4×10^{-4}
	4.6 Unretarded transport of ^{14}C from SF and ILW released as volatile species through host rock; retardation in confining units taken into account[3]	Gas permeability [m^2]	4.6a 10^{-23}	4.9×10^{-5}	-	9.7×10^{-6}
			4.6b 10^{-22}	4.7×10^{-5}	-	9.0×10^{-6}
			4.6c 0	4.9×10^{-5}	-	9.7×10^{-6}

Tab. 7.10-1: (Cont.)

Scenario	Conceptualisation	Parameter variation		Summed dose maximum [mSv a^{-1}] SF	HLW	ILW
4. "What if?" cases to investigate robustness of the disposal system	4.7 Poor near field and pessimistic near field/geosphere geochemical dataset	Water flow rate in geosphere	4.7a RC flow rate	4.3×10^{-4}	1.1×10^{-5}	8.7×10^{-6}
			4.7b 10-fold increase	1.5×10^{-3}	4.4×10^{-5}	3.5×10^{-5}
			4.7c 100-fold increase	1.0×10^{-2}	9.6×10^{-5}	2.6×10^{-4}
	4.8 No advection in geosphere (diffusive transport only)	4.8a Base Case only		1.8×10^{-5}	4.8×10^{-8}	1.1×10^{-6}
	4.9 SF: Increased cladding corrosion rate	4.9a Base Case only		4.8×10^{-5}	-	-
	4.10 Kd(I) for NF and geosphere = 0	4.10a Base Case only		1.1×10^{-4}	2.5×10^{-7}	9.7×10^{-6}
	4.11 Decreased transport distance in Opalinus Clay (30 m)	4.11a Base Case only		8.1×10^{-5}	2.1×10^{-7}	7.3×10^{-6}
5. Design and System Options	5.1 Increased waste arisings (300 GWa(e))	5.1a Base Case only		8.1×10^{-5}	-	-
	5.2 ILW high force compacted waste option	5.2a Base Case only		-	-	4.3×10^{-6}
	5.3 SF canister with Cu shell [3]	5.3a Canister breaching at 10^5 a		4.4×10^{-5}	-	-
		5.3b Initial defect (small initial pinhole, full breaching at 10^5 a) [1]		1.5×10^{-9}	-	-
		5.3c Initial defect (Large initial pinhole, full breaching at 10^5 a) [1]		4.9×10^{-9}	-	-
6. Illustration of effects of biosphere uncertainty	6.1 Reference and alternative geomorphology	6.1a Reference area (RC)		4.8×10^{-5}	1.3×10^{-7}	4.3×10^{-6}
		6.1b Sedimentation area		2.3×10^{-5}	6.0×10^{-8}	2.0×10^{-6}
		6.1c Wetland		4.4×10^{-6}	1.2×10^{-8}	3.9×10^{-7}
		6.1d Exfiltration to spring located at valley side [3]		4.9×10^{-5}	1.2×10^{-7}	4.4×10^{-6}
	6.2 Reference and alternative climates	6.2a Present-day climate (RC)		4.8×10^{-5}	1.3×10^{-7}	4.3×10^{-6}
		6.2b Drier/warmer than present-day climate		5.3×10^{-4}	1.6×10^{-6}	4.6×10^{-5}
		6.2c Wetter/warmer than present-day climate		1.6×10^{-5}	4.3×10^{-8}	1.4×10^{-6}
		6.2d Periglacial climate		4.8×10^{-7} to 4.8×10^{-5}	1.3×10^{-9} to 1.3×10^{-7}	4.3×10^{-8} to 4.3×10^{-6}

[1] Summed dose maximum for a single canister

[2] Reference canister loadings

[3] Summed dose maxima include only the contributions from the release paths that characterise the case under consideration

8 The Safety Case: Main Arguments and Results

8.1 Aims and structure of the chapter

This chapter presents the main arguments and results that make up the safety case for the proposed SF / HLW / ILW repository in the Opalinus Clay of the Zürcher Weinland, and, as indicated in Fig. 8.1-1, represents the final step in compiling the safety case as described in Chapter 3.

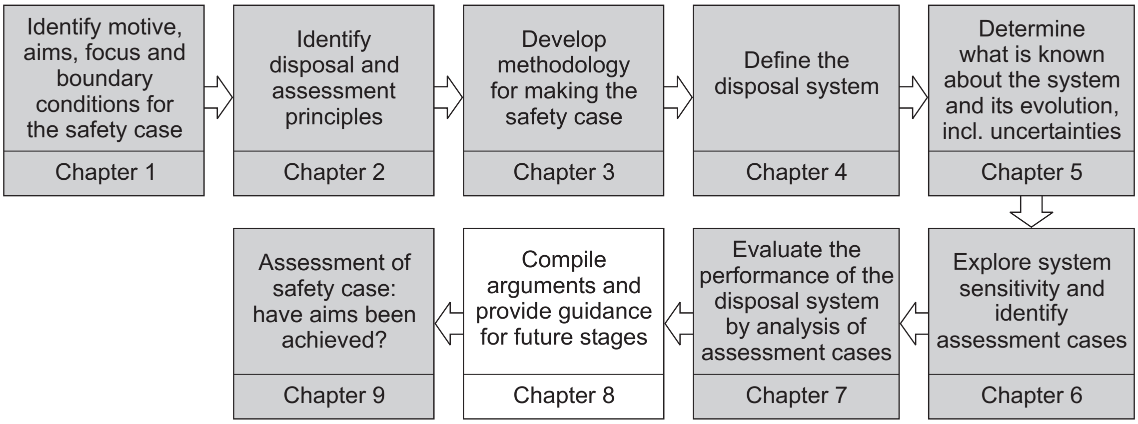

Fig. 8.1-1: The role of the present chapter in the sequence of tasks involved in developing the safety case

The main lines of argument are given in Section 8.2, starting with an overview and then going on to describe each line of argument in more detail. Section 8.3 describes the ways in which observations of natural systems and other information provide evidence for the potential effectiveness of geological disposal, as well as support for key model assumptions and parameter values used in the safety assessment. Finally, guidance from the safety assessment for future stages of repository planning and development is discussed in Section 8.4.

The messages from this chapter form the basis of the conclusions drawn in Chapter 9.

8.2 The lines of argument

8.2.1 Overview

In accordance with the methodology described in Chapter 3, the safety case for the proposed repository includes arguments for:

- the strength of geological disposal as a waste management option,
- the safety and robustness of the chosen disposal system,
- the reduced likelihood and consequences of human intrusion,
- the strength of the stepwise repository implementation process,
- the good scientific understanding that is available and relevant to the chosen disposal system and its evolution,

- the adequacy of the methodology and the models, codes and databases that have been used to assess radiological consequences, and
- *multiple arguments for safety* that:
  - demonstrate safety and compliance with regulatory protection objectives,
  - use indicators of safety that are complementary to those of dose and risk and that show that radionuclide releases and concentrations due to the repository are well below those due to natural radionuclides in the environment,
  - indicate that the actual performance of the disposal system will, in reality, be more favourable than that evaluated in quantitative analyses (the existence of reserve FEPs), and
  - no issues have been identified that have the potential to compromise safety.

The analyses and lines of arguments are discussed individually in the following sections.

8.2.2 The strength of geological disposal as a waste management option

Radioactive waste needs to be managed in a way that ensures the safety of humans and the protection of the environment, as well as providing security from malicious intervention, now and in the future. According to current understanding, geological disposal is the only waste management option that offers long-term passive safety. Placing the waste in a deep rock formation favours security in that it reduces the possibility of irresponsible interference. Furthermore, the feasibility of safe geological disposal is supported by:

- **The existence of suitable rock formations** – In Switzerland and elsewhere, deep rock formations exist in which events and processes that might convey radionuclides to the surface environment are either absent, or extremely rare or slow.
- **Safety assessments conducted world-wide** – The findings of integrated safety assessments conducted world-wide for a wide range of sites and designs support the possibility of safe geological disposal.
- **Observations of natural systems** – Indirect support for safety also comes from observations of natural systems, including the longevity of uranium ore deposits in many different geological environments around the world. This includes the observed retention of most of the radioactive inventory of the Oklo natural reactor over a period of ~ 2 billion years. Furthermore, there is ample evidence of the importance of the natural processes of solubility control, sorption and diffusion in attenuating concentrations of species dissolved in porewater.
- **Characteristics of surface facilities versus geological disposal** – Radioactive waste can be stored for a time in surface facilities. The safety of these facilities is, however, dependent on continued societal stability, which is subject to uncertainties that are far greater than those associated with the evolution of conditions deep underground in geological formations that would be suitable to host a repository. As a long-term waste management option, deep geological disposal has the positive attribute that, if the site and design are chosen appropriately, societal stability, allowing for government and regulatory control, is not a prerequisite for long-term safety. No burden is placed on future generations to maintain and control a disposal site once the facility has been closed, although such control is certainly possible if society so wishes.

8.2.3 The safety and robustness of the chosen disposal system

The proposed SF / HLW / ILW repository in the Opalinus Clay of the Zürcher Weinland is chosen to provide *safety and robustness*, in accordance with the *system objectives and disposal principles* described in Chapter 2. In particular, the *safety functions* of isolation from the human environment, long-term confinement and radioactive decay within the disposal system and attenuation of releases to the environment are provided by features of the system that are well understood and insensitive to perturbations. These features, termed the *pillars of safety*, are:

- the deep underground location of the repository, in a setting that is unlikely to attract human intrusion and is not prone to disruptive geological events and to processes unfavourable to long-term stability,
- the host rock, which has a low hydraulic conductivity, a fine, homogeneous pore structure and a self-sealing capacity, thus providing a strong barrier to radionuclide transport and a suitable environment for the engineered barrier system,
- a chemical environment that provides a range of geochemical immobilisation and retardation processes, favours the long-term stability of the engineered barriers, and is itself stable due to a range of chemical buffering reactions,
- the bentonite buffer (for SF and HLW) as a well-defined interface between the canisters and the host rock, with similar properties as the host rock, that ensures that the effects of the presence of the emplacement tunnels and the heat-producing waste on the host rock are minimal, and that provides a strong barrier to radionuclide transport and a suitable environment for the canisters and the waste forms,
- SF and HLW waste forms that are stable in the expected environment, and
- SF and HLW canisters that are mechanically strong and corrosion resistant in the expected environment and provide absolute containment for a considerable period of time.

More specifically, the safety and robustness of the disposal system is achieved by:

- **Multiple phenomena contributing to the safety functions** – The system chosen ensures that the set of passive barriers provides multiple phenomena that contribute to the safety functions "isolation from the human environment", "long-term-confinement" and "attenuation of releases". This significantly contributes to the robustness of the system: uncertainty in the detailed performance of some phenomena is compensated by existence of other phenomena which together ensure the operation of the safety functions for all realistically conceivable possibilities for the characteristics and evolution of the system. Specifically, it is the longevity of the SF / HLW canisters that provides complete containment in the early phase when radiotoxicity is highest, the corrosion resistant waste matrices for SF and HLW that retain most of the radionuclides, the bentonite with its favourable chemical properties (leading to low solubilities and strong sorption for many radionuclides) and its very low permeability providing an excellent transport barrier, and, most important, the very low permeability host rock in a stable geological environment that retains nearly all radionuclides that may eventually be released from the EBS. In Chapter 6 it has been shown that these individual components very strongly contribute to one or more of the safety functions and that they complement each other very well even if a specific component would be less effective than expected.
- **The long-term stability of the Opalinus Clay and the repository** – The long-term stability of the Opalinus Clay and the repository is favoured by the stable and protected environment provided by the geological setting. In particular, the low uplift rate, low water flow, favourable geochemical conditions, sufficient distance of the repository from geologically

active zones, absence of potentially detrimental geological events, and resistance of overlying sediments to erosion and the sealed and backfilled access tunnels all contribute to long-term stability of the disposal system. Chemical buffering reactions occur throughout the disposal system and ensure a stable chemical environment. The clay barrier (the Opalinus Clay and bentonite) is stable with respect to physical and chemical alteration in such an environment. The long-term physical integrity of the clay barrier is further favoured by the plasticity and self-sealing capacity of the clay materials. The mechanical and thermal buffering provided by the bentonite protects the SF and HLW canisters from small rock movements and protects the host rock from radiogenic heat. Radiation shielding provided by the canisters prevents significantly enhanced corrosion at the outer surfaces of the canisters due to radiolysis in the surrounding porewater when radiation is still high. The stability of the high-integrity canisters, the fuel cladding and the waste matrices under conditions expected in the repository (chemical environment, mechanical loads, radiation fields etc.) are favoured by their resistance to corrosion, dissolution and radiation damage. Furthermore, with a minimum burnup of spent fuel, criticality can be ruled out.

- **Avoidance of uncertainties and detrimental phenomena** – Uncertainties and detrimental phenomena with the potential to affect the pillars of safety significantly have, to a large extent, been avoided by the choice of site and design. The initial characteristics of the geological setting and engineered materials are well known and are predictable, in the sense that any changes in their characteristics will be the result of slow and well-understood processes. In addition, there are numerous specific features of the site and design that reduce the effects of detrimental phenomena and uncertainties. The siting of the repository under a deep and erosion-resistant sedimentary cover in a region with a low rate of uplift counters the effects of surface environmental changes, and an appropriate distance between SF / HLW emplacement tunnels and ILW tunnels counters possible effects on the SF / HLW part of the repository due to an alkaline plume originating from the ILW tunnels. Flexibility in the implementation of the repository also contributes to countering currently existing uncertainties. There is, for example, the option of using a composite copper / steel canister for spent fuel to greatly reduce gas production.

Overall, the pillars of safety ensure that the system is robust, with a high level of confidence that the safety functions will operate broadly as expected for all reasonably foreseeable circumstances. The adequacy of safety that the system provides has been tested in the safety assessment with a broad spectrum of cases as documented in Chapter 7.

8.2.4 The reduced likelihood and consequences of inadvertent human intrusion

The reduction of likelihood of human intrusion and the mitigation of degradation of system performance in the case of intrusion is achieved by:

- **Preservation of information** – Measures will be taken to ensure that information regarding the purpose, location, design and contents of the repository are preserved so that future generations are made aware of the consequences of actions they may choose to take that might affect the operation of the disposal system. Because closure of the repository is still tens of years away, no specific proposals have yet been prepared for ensuring preservation of information; ample time is available to develop the necessary steps.

- **Avoidance of resource conflicts** – Because of the absence of viable natural resources in the area, the potential site for a repository for SF / HLW / ILW in the Opalinus Clay of the Zürcher Weinland described in this report meets the requirement that a repository should be sited such that any foreseeable future resource conflict is avoided to minimise the likelihood of future inadvertent human intrusion.

- **Compartmentalisation and solidification of wastes** – The design of the repository ensures that in the case of intrusion only a small part of the repository is affected, as shown in Chapter 7 (assessment cases 3.1). This is achieved by compartmentalisation, i.e. each SF / HLW waste package is completely surrounded by massive amounts of bentonite and thus forms an isolated compartment, with no shortcut to the next one. The limited size of the ILW emplacement tunnels has a similar effect. Solidification of the wastes ensures that only the small fraction of radionuclides released from the waste form that are in solution can be transported to the surface instantaneously.

8.2.5 The strength of the stepwise repository implementation process

The stepwise repository implementation process, as discussed in Chapter 1, contains several elements that ensure successful progress. These include:

- **Sufficiency of a limited information basis at a given stage** – At the current stage of the Swiss HLW programme, not all the details of the repository system need to be fixed in detail and, therefore, the given, but still limited information basis is considered to be sufficient to demonstrate feasibility of disposal and to provide a platform for discussion and a foundation for decision-making on how to proceed with the programme. The siting area of the Zürcher Weinland with the Opalinus Clay as host rock, with its proven good explorability, good predictability and quiet geology which provides ample space for the suggested repository, is sufficiently well understood for these purposes.
- **Reliance on understood and reliably characterised components** – The proposed site has been characterised reliably by surface-based techniques. It is argued that this is sufficient for the current, early stage of the repository project and in view of the good explorability of the site. The system of engineered barriers relies on well understood materials. This, together with a sufficiently well understood site, provides adequate predictability of the overall system. Construction of the repository and the emplacement of the EBS is feasible with proven technology without excessive demands (Nagra 2002b).
- **Involvement of stakeholders and the opportunities for feedback** – The stepwise approach requires the development of a safety report at each milestone. The current safety report and the other documents of Project *Entsorgungsnachweis*, together with the review provided by the authorities and the different advisory committees, will provide an excellent platform for interaction with all stakeholders that will provide feedback on how to proceed.
- **Possibilities for modification** – The stepwise repository implementation process provides flexibility with respect to new findings. The siting area of the Zürcher Weinland has reserves in space that allow for optimal allocation of the emplacement tunnels, and the availability of design options will allow the adaptation to specific future needs. This includes the possibility to choose an optimal design and placement of the surface facilities to consider environmental issues and requirements of the local population. The design of the repository, conforming to the concept of monitored long-term geological disposal as proposed by EKRA, also allows the reversal of decisions in the course of the implementation process, including the retrieval of emplaced wastes.

  In addition, as explained in Chapter 1, there is flexibility regarding siting. Besides the proposed siting area of the Zürcher Weinland, alternatives exist for the Opalinus Clay host rock with the regions "Nördlich Lägern" and "Jurasüdfuss-Bözberg". Furthermore, for the host rock option "crystalline basement", the region "Mettauer Tal", which was characterised by 2 D seismics, has been identified as an area for further investigations if this were found to be necessary in the future. Finally, for the reserve host rock option "Lower Freshwater Molasse" preferred regions have been identified.

- **Possibilities for monitoring** – The design of the proposed repository conforms to the concept of monitored long-term geological disposal (EKRA 2000) and provides opportunities for monitoring and review and possible reversal of decisions in the course of the implementation process.

8.2.6 The understanding of the system and its evolution

Scientific understanding of the geological setting is discussed at length in Nagra (2002a). Chapters 4 and 5 of the present report summarise the scientific understanding of the geological setting and the EBS. As mentioned above, the initial characteristics of the geological setting and engineered materials are known, or can be determined without undue difficulties, and are largely predictable.

Understanding of the long-term behaviour of the system is supported by wide ranging information, including that from laboratory and field experiments and from observations of natural systems. The most important elements are summarised in Tab. 8.2-1.

Overall, a reliable database for safety assessment is available, in spite of the current early stage of the repository planning process. In the case of the Opalinus Clay, remaining uncertainties mainly relate to the potentially perturbing effects of the repository on the geological setting. In the case of the EBS, they mainly relate to the long-term evolution of materials and their interaction. Those uncertainties that are relevant to the radiological consequences of the repository have been taken into account in the safety assessment.

8.2.7 The safety assessment methodology and the models, codes and databases that are available to assess radiological consequences

General

The safety assessment has been carried out to test whether adequate levels of safety are to be expected based on what is known about the proposed SF / HLW / ILW repository in the Opalinus Clay of the Zürcher Weinland. In the safety assessment, the disposal system is analysed in order to evaluate its performance and that of its components and to test compliance with Swiss regulatory criteria. In addition to considering its most likely characteristics and path of evolution, various uncertainties and design and system options are considered, leading to alternative evolutionary paths that are incorporated into a wide range of assessment cases. The cases are analysed to provide quantitative estimates of possible radiological consequences, which are discussed in Section 8.2.8.

Adherence to assessment principles

The methodology for conducting the safety assessment accords with the assessment principles described in Chapter 2. In particular, the methodology is designed to provide a safety case that is robust, transparent and, as far as possible, complete, with a documentation that is traceable[110]. Completeness and robustness imply that the range of assessment cases is adequate to illustrate the effects of various sources of uncertainty and design and system options. Models and databases that are used to analyse the cases are, as far as possible, validated, and key computer codes verified. These issues are discussed further, below.

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[110] This includes documents in addition to the present report, see Fig. 1.4-1 in Chapter 1.

Tab. 8.2-1: Understanding of the disposal system and its evolution: evidence and arguments for safety-relevant characteristics and the selection of parameter values that provide the foundation for the safety case (based on detailed discussion in Chapters 4 and 5 and in Nagra (2002a))

Disposal system component	**Issue**	**Evidence / arguments for properties at waste emplacement**	**Evidence / arguments for expected evolution after waste emplacement**
Host rock	Geometry (domain size and thickness)	Proven exploration methodology (3-D seismics) that is calibrated with Benken borehole data; respect distance from major faults (faults with displacements > 10 m, and evidence of activity in Tertiary)	No changes (stable situation)
	Absence of significant faults and discontinuities	High resolution of 3-D seismics	No significant fault movements (no reactivation of existing faults, no formation of new faults) expected as there is negligible neo-tectonic activity in the potential siting area
	Low permeability environment (including faults)	General geological understanding (pore size distribution); experimental evidence from Benken borehole and measurements at Mont Terri, comparison with other argillaceous rocks, no evidence for leakage of faults in tunnels and deep bore-holes if overburden is > 200 m (all faults tested had low permeability)	Porewater profiles (chemistry, isotopes) at Benken and Mont Terri (evidence from long-term processes)
	Tightness of discontinuities	Evidence from observations in boreholes and from road and rail tunnel mapping (see above), general scientific understanding, slightly compressive regime	Evidence from observations in boreholes and from road and rail tunnel mapping, general scientific understanding, long-term stability of slightly compressive regime
	Self-sealing capacity	Field studies at Mont Terri (EDZ and gas-frac self-sealing experiments) and laboratory studies; existing faults have low permeability (see above)	Field studies at Mont Terri and laboratory studies; palaeohydrogeological aspects (absence of significant mineral veins in faults in Opalinus Clay), no anomalies in porewater profiles in faulted section

Tab. 8.2-1: (Cont.)

Disposal system component	**Issue**	**Evidence / arguments for properties at waste emplacement**	**Evidence / arguments for expected evolution after waste emplacement**
Host rock	Diffusion-dominated transport regime	Porewater profiles in Benken and Mont Terri (significant advective transport can be excluded), diffusion properties derived from lab experiments (Benken, Mont Terri samples), in situ field experiments (Mont Terri), comparison with other argillaceous rocks	Extrapolation of diffusion rates derived from profiles, long-term stability of host rock
	Low influence of EDZ on transport	Properties irrelevant at waste emplacement, geochemical alteration (oxidation, salt accumulation) insignificant	Self-sealing of resaturated disturbed material under applied stress, studies of EDZ evolution at Mont Terri (see above: self-sealing experiments), geochemical alteration (high-pH plume) provides favourable sorption properties
	Stable geochemical environment (redox control, sorption)	Properties irrelevant at waste emplacement (short-term oxidising environment will disappear during resaturation), strong redox-buffering capacity of host rock and engineered barriers	Favourable mineralogy, large buffering capacity (pyrite and organic carbon content of host rock), negligible long-term changes of mobile components (diffusive), no indication of past disturbances (e.g. by glaciation), general scientific understanding
	Gas transport	Properties irrelevant at waste emplacement	Experimental studies (lab and field), oil and gas industry empirical experience and studies
	Lack of detrimental effects of increased temperatures	Properties irrelevant at waste emplacement	Projected maximum temperature in Opalinus Clay (95 °C, short-term) has roughly occurred previously in the formation (85 ± 5 °C, long-term), natural analogues (no detrimental effects on natural bentonite at temperatures below about 130 °C)
	Long-term stability (including the geological setting)	Low uplift and erosion rates from several independent data bases (geomorphology, basin model, geodesy); quiet neo-tectonics from studies in Zürcher Weinland (low seismicity, no active faults detected)	Extrapolation of low uplift and erosion rates, long-term stability of stress regime

Tab. 8.2-1: (Cont.)

Disposal system component	Issue	Evidence / arguments for properties at waste emplacement	Evidence / arguments for expected evolution after waste emplacement
Waste inventory	Nuclide inventory	Validated codes and data bases; model inventory developed and uncertainties evaluated	Validity/accuracy of radioactive decay calculations
Fuel matrix	Dissolution	Experimental evidence for low matrix dissolution rates under reducing conditions	Natural analogue (Cigar Lake), evidence for long-term stability of uraninite
	Instant release fraction	Experimental evidence (up to moderate burnups)	Low fission product and Pb loss rates from Oklo uraninite (as well as negligible actinide losses)
	Radiolysis effects	Experimental evidence for small radiolytic effects	Natural analogue (absence of significant radiolytic effects in Cigar Lake, Oklo and other ancient uranium ores)
Glass matrix	Corrosion	Experimental evidence for low dissolution rates	Natural analogues
Canister (steel)	Corrosion	Experimental evidence for low corrosion rates	Low corrosion rates of steel / iron derived from archaeological artifacts and small degree of pitting derived from long-term field studies
Canister (copper)	Corrosion	Experimental evidence for low corrosion rates	Low corrosion rates derived from natural analogues and archaeological artifacts
Bentonite	Long-term stability (limited illitisation)	Natural analogues, extensive experimental evidence	Natural analogues (no detrimental effects on natural bentonite at temperatures below about 130 ^oC)
	Retention of swelling properties and plasticity if alteration occurs	Experimental evidence	Natural analogues (e.g. Kinekulle bentonite)
	Low permeability	Extensive experimental evidence	Natural analogues
	Sorption	General scientific understanding; experimental evidence	Natural analogues
	Low solubilities for key elements	General scientific understanding (thermodynamics); experimental evidence; observations in nature	Thermodynamics, observations in nature

Adequacy of consideration and treatment of uncertainty

Measures have been taken to ensure that all relevant uncertainties are considered in defining the range of assessment cases and, furthermore, that no potentially detrimental phenomena have been overlooked. Measures include the systematic approach to development of the system concept (Chapters 4 and 5), the screening and abstraction of information, the use of FEP catalogues, plausibility checks and comparison with simplified calculations, comparison with safety assessments conducted abroad and with earlier safety assessments conducted in the Swiss programme, and internal and external reviews.

Uncertainties in the initial characteristics of the disposal system are generally relatively small (with the exception of some alternative system or design options). Key uncertainties that are illustrated by the assessment cases concern mainly the rates of processes affecting the evolution of the EBS and a range of phenomena that may perturb the geological setting. Sensitivity analyses over broad ranges of parameter values with consideration of various phenomena provided a good understanding of the behaviour of the system with respect to perturbations and the extent to which deviations from the likely / expected characteristics and evolution of the disposal system affect overall performance and the performance of individual system components. They provided insight into the robustness of the system chosen, guided the definition of alternative assessment cases and assisted in the interpretation of results.

Additional system understanding is provided by probabilistic calculations. Experts in scientific fields relevant to a particular set of parameters have been asked to assign ranges of values to various parameters, taking into account all relevant information. The ranges incorporate the expected or most likely value, a pessimistic value (that is, when it is applied within an assessment model, it will over-predict radiological consequences), and an optimistic value. These are provided in the form of probability density functions (PDFs), as discussed in Appendix 1. Probabilistic analyses around the Reference Case provide an indication of compliance with regulatory criteria taking into account the combined effects of uncertainties and provide assurance that no unfavourable combinations of parameter values exist that can compromise safety (see Section 8.2.8.).

In addition, and in agreement with the precautionary principle, a range of "what if?" cases have been analysed in order to test the robustness of the system. In a "what if?" case, a particular conceptual assumption or parameter value is assumed that lies outside the range of possibilities supported by scientific evidence. In order to limit the number of such cases, they are restricted to those that test the effect of perturbations to key properties of the pillars of safety. It was not the aim to derive a comprehensive list of all conceivable "what if?" cases, but rather to select and analyse a few typical cases in order to illustrate system behaviour under extreme conditions. The results of these cases are presented in Section 8.2.8.

The inclusion of uncertainties in specific cases is, to some extent, guided by the deterministic and probabilistic sensitivity analyses reported in Chapter 6, which enhance understanding of the behaviour of the system, sometimes under extreme or hypothetical conditions, and provide indications of how sensitive system performance is to particular uncertainties. Nevertheless, the selection of cases remains a matter of expert judgement. Efforts have been taken to develop consensus among scientific specialists as to what cases are consistent with scientific understanding. Specifically, a series of meetings with highly qualified scientific specialists from various disciplines (notably geology, hydrogeology, geochemistry, geomorphology, climatology and numerical modelling in safety assessment) were conducted in which assessment cases were presented, with their corresponding conceptualisations, with the aim of eliciting comments (in the form of either agreement or "constructive disagreement").

Validation and verification

The models, codes and databases used to analyse the assessment cases are the result of considerable research and development work, carried out to support this and former safety assessments. Reliability implies both accuracy of the codes (i.e. verification, or freedom from errors) and applicability of models, codes and databases for their intended purposes (i.e. validation). Regarding accuracy, tests have been carried out to verify the principal codes used to analyse assessment cases (Nagra 2002c) and a procedure established to record modifications in a traceable manner. Regarding applicability, measures include[111]:

- a systematic and transparent approach to model development and consideration of alternative conceptual models,
- use of laboratory and field tests as well as general scientific understanding, experience from other safety assessments and observations from nature to test models and databases, and to ensure their consistency, as far as possible, with all relevant information,
- use of natural analogues and palaeohydrogeological models to evaluate uncertainties arising from the temporal scales of concern, and
- use of expert judgement and critical internal and external review.

In most instances, uncertainties that are not investigated by defining specific assessment cases are treated using a pessimistic or conservative approach (see Chapter 6). The aim of this approach is to ensure that the near field and geosphere models, codes and databases err, if at all, on the side of pessimism, while achieving as much realism as possible given the existing levels of uncertainty. In the cases of modelling solute transport by diffusion and advection in the near field and in the host rock, the models, codes and data used are relatively realistic. In the treatment of other safety-relevant phenomena, such as geochemical immobilisation, isolation in canisters and a number of perturbing phenomena, conceptual models are simpler, with uncertainty handled via conservative assumptions, sometimes involving the neglect of poorly understood, impact-reducing processes, and/or pessimistic choices of parameter values. Iterative model refinement has, however, been undertaken where these were judged to be overly simple or conservative.

As discussed in Chapter 2, there are irreducible and unquantifiable uncertainties associated with the treatment of the biosphere and future human actions, and a stylised approach is adopted to deal with these. For example, for the purpose of the assessment, possible future human actions that may affect the repository are constrained to those that are possible with present-day technology or moderate developments thereof.

8.2.8 Multiple arguments for safety

8.2.8.1 Compliance with regulatory protection objectives

As discussed in Chapter 2, HSK-R-21 (HSK & KSA 1993) gives three specific Protection Objectives that a repository should be shown to satisfy.

[111] All of these measures are, in broad terms, concerned with "validation", defined in the Swiss programme (HSK & KSA 1993) as: "Providing confidence that a computer code used in safety analysis is applicable for the specific repository system". Confidence in a computer code implies confidence in the underlying model assumptions and data.

Protection Objective 1: *The release of radionuclides from a sealed repository subsequent upon processes and events reasonably expected to happen shall at no time give rise to individual doses which exceed 0.1 mSv per year.*

The present safety assessment considers a number of different cases describing a range of possible evolutions of the disposal system. For all of these cases, radiological consequences are evaluated quantitatively in terms of individual doses, and compared to the dose limit specified in Protection Objective 1.

The annual dose for the Reference Case, summed over all waste groups, is presented in Fig. 8.2-1. The maximum dose, which is due to ^{129}I from spent fuel, occurs at about one million years and is more than three orders of magnitude below the Swiss regulatory guideline and more than two orders of magnitude below the "level of insignificant dose" set at 0.01 mSv a^{-1} by the IAEA (1996).

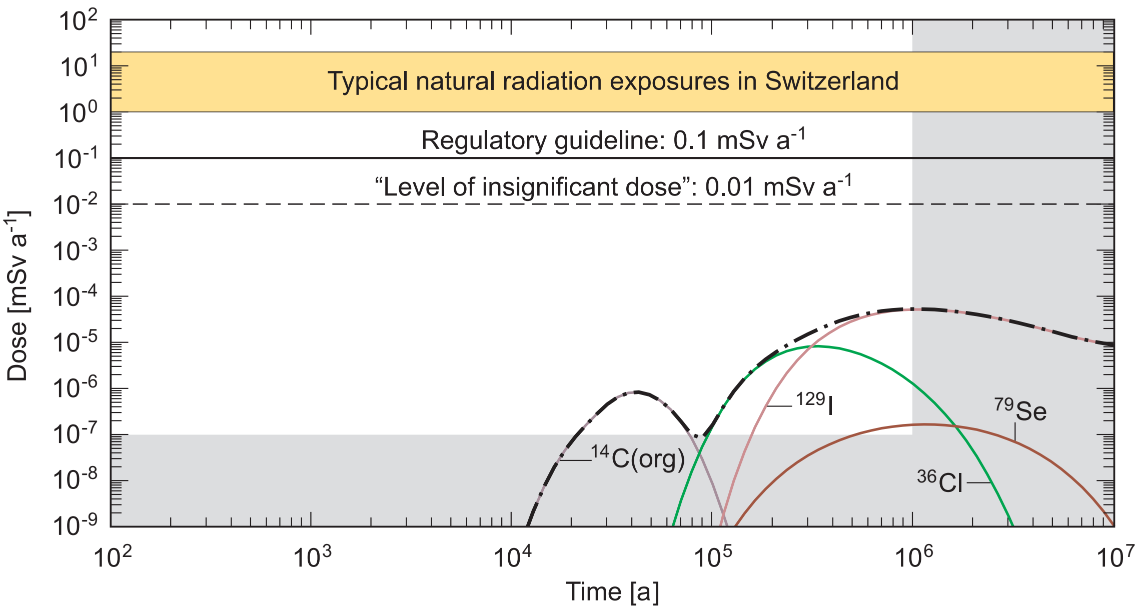

Fig. 8.2-1: Total dose for the Reference Case as a function of time (sum of the three waste types SF, HLW and ILW)

The maximum annual doses for all cases analysed, summed over all radionuclides and all waste groups, are summarised in Fig. 8.2-2 and in Tab. 8.2-2. In all of the assessment cases considered, dose maxima are below the Swiss regulatory guideline. In many cases, the calculated maxima are several orders of magnitude below the regulatory guideline. Note that in contrast to Chapter 7, in Chapter 8 the results include the contributions from all release pathways, except for the cases involving release of ^{14}C as volatile species (see also Tab. 8.2-2).

Fig. 8.2-3 shows a different representation of the results. All cases analysed are represented in a scatter plot showing the maximal dose versus the time of occurrence of that dose. The symbols indicate to which scenario (according to the classification of Tab. 8.2-2) a given case belongs.

Protection Objective 2: *The individual radiological risk of fatality from a sealed repository subsequent upon unlikely processes and events not taken into consideration in Protection Objective 1 shall, at no time, exceed one in a million per year.*

In addition to the deterministic analyses of the assessment cases, complementary probabilistic analyses have been performed for the Reference Conceptualisation and for three classes of "what if?" cases, thereby taking into account the combined effects of uncertainties. The results are given in terms of CCDFs (Complementary Cumulative Density Functions) and are in all cases well below the regulatory constraint derived from Protection Objective 2, as shown in Figs. 7.4-3a, 7.7-2, 7.7-4 and 7.7-6.

Protection Objective 3: *After a repository has been sealed, no further measures shall be necessary to ensure safety. A repository must be designed in such a way that it can be sealed within a few years.*

The proposed repository is consistent in its design with the requirement that, at any time during a possible extended monitoring phase, it could be sealed within a few years, and that it does not rely for safety on any further measures after it has been sealed; i.e. the requirements of Protection Objective 3.

The concept of monitored long-term geological disposal has been implemented, and design measures allowing for monitoring and easy retrievability will not impair long-term safety. This is the reason for backfilling and sealing of the emplacement tunnels immediately after waste emplacement. In this case, even abandoning the repository without proper backfilling and sealing of the access ramp does not give rise to doses above the regulatory guideline (see assessment case 3.3a).

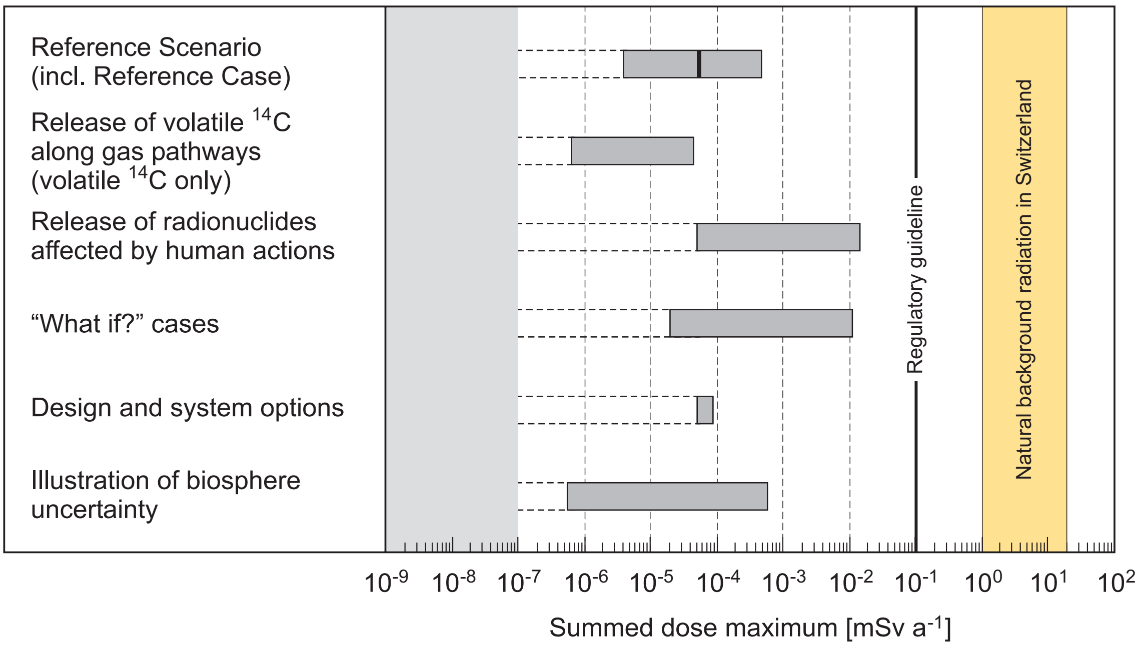

Fig. 8.2-2: Summary of maximum annual doses, summed over all radionuclides and the three waste types SF, HLW and ILW, for the various categories of assessment cases (includes all assessment cases shown in Tab. 8.2-2)

Tab. 8.2-2: Summary of maximum annual doses, summed over all radionuclides* and the three waste types SF, HLW and ILW, for the various scenarios, "what if?" cases, design and system options and illustration of biosphere uncertainty with associated conceptualisations and parameter variations that define the different assessment cases

* The only exceptions are the cases 2.1, 2.2 and 4.6 related to the release of ^{14}C as volatile species, where only doses due to ^{14}C are given.

Scenario	Conceptualisation	Parameter variation	Maximum annual dose	
			Dose [mSv a^{-1}]	Time of occurrence [a]
1. Reference Scenario Release of dissolved radionuclides	**1.1 Reference Conceptualisation**	1.1a Reference Case (RC)	5.3×10^{-5}	1.0×10^{6}
		1.1b Variability in canister inventory	See table 7.10-1	
		1.1c Reduced canister lifetime	5.7×10^{-5}	1.0×10^{6}
		1.1d Pessimistic near field geochemical dataset	6.3×10^{-5}	8.9×10^{5}
		1.1e Increased glass dissolution rate in HLW	5.3×10^{-5}	1.0×10^{6}
		1.1f Increased water flowrate in geosphere (10-fold increase)	2.1×10^{-4}	3.8×10^{5}
		1.1g Decreased water flowrate in geosphere (10-fold decrease)	4.1×10^{-5}	1.1×10^{6}
		1.1h Pessimistic geosphere sorption constants	1.0×10^{-4}	4.8×10^{5}
		1.1i Pessimistic near field and geosphere geochemical dataset	1.2×10^{-4}	4.8×10^{5}
		1.1j Pessimistic geosphere diffusion constants	4.5×10^{-4}	1.8×10^{4}
		1.1k Pessimistic treatment of ^{14}C (organic) in SF	5.3×10^{-5}	1.0×10^{6}
	1.2 Solubility-limited dissolution of SF	1.2a Base Case only	4.1×10^{-5}	8.9×10^{5}
	1.3 Bentonite thermal alteration	1.3a Base Case only	5.3×10^{-5}	1.0×10^{6}
	1.4 Glacially-induced flow in the Opalinus Clay	1.4a Base Case only	8.4×10^{-5}	9.5×10^{5}
	1.5 Additional barrier provided by confining units	1.5a Vertical transport through confining units	2.5×10^{-5}	3.4×10^{6}
		1.5b Horizontal transport in local aquifers	3.8×10^{-6}	7.0×10^{6}
	1.6 Radionuclide release affected by ramp/shaft	1.6a Base Case	5.1×10^{-5}	1.0×10^{6}
		1.6b Increased hydraulic conductivity of EDZ (100-fold increase)	5.0×10^{-5}	1.0×10^{6}
	1.7 Convergence-induced release affected by ramp (ILW)	1.7a Steady-state hydraulics	5.3×10^{-5}	1.0×10^{6}
		1.7b Water pulse	5.2×10^{-5}	1.0×10^{6}
	1.8 Gas-induced release of dissolved radionuclides affected by ramp / shaft	1.8a Base Case (ILW: 50 %, 0.05 $m^3 a^{-1}$)	2.3×10^{-5}	1.3×10^{6}
		1.8b Increased water flow rate in ILW (100 %, 0.3 $m^3 a^{-1}$)	4.6×10^{-5}	7.9×10^{4}

Tab. 8.2-2: (Cont.)

Scenario	Conceptualisation	Parameter variation		Maximum annual dose	
				Dose [mSv a^{-1}]	Time of occurrence [a]
2. Alternative Scenario 1 Release of volatile radio-nuclides along gas pathways (volatile ^{14}C only)	2.1 Release of ^{14}C from SF and ILW as volatile species in the gas phase not affected by ramp / shaft ("tight seals")	Gas permeability [m^2]	2.1a (10^{-23})	6.8×10^{-7}	4.5×10^{4}
			2.1b (10^{-22})	3.9×10^{-6}	3.0×10^{4}
			2.1c (0)	6.2×10^{-7}	4.5×10^{4}
	2.2 Release of ^{14}C from SF and ILW as volatile species in the gas phase affected by ramp / shaft ("leaky seals")	Gas permeability [m^2]	2.2a (10^{-23})	4.4×10^{-5}	1.2×10^{4}
			2.2b (10^{-22})	4.2×10^{-5}	1.2×10^{4}
			2.2c (0)	4.4×10^{-5}	1.2×10^{4}
3. Alternative Scenario 2 Release of radionuclides affected by human actions	3.1 Borehole penetration	3.1a Near hit, 2 canisters / 1 ILW-1 tunnel affected		1.9×10^{-4}	1.1×10^{4}
		3.1b Near hit, 4 canisters affected		2.1×10^{-4}	1.1×10^{4}
		3.1c Near hit, 2 canisters / 1 ILW-1 tunnel affected, increased water flow rate		1.4×10^{-2}	1.1×10^{4}
		3.1d Near hit, 2 canisters / 1 ILW-1 tunnel affected, decreased water flow rate		5.1×10^{-5}	1.0×10^{6}
		3.1e Direct hit		1.6×10^{-3}	1.0×10^{5}
		3.1f Direct hit, increased water flow rate		2.5×10^{-3}	1.0×10^{5}
		3.1g Direct hit, decreased water flow rate		1.3×10^{-4}	1.0×10^{5}
	3.2 Deep groundwater extraction from Malm aquifer (production of well as dilution)	Plume capture efficiency	3.2a (10 %)	5.8×10^{-5}	1.0×10^{6}
			3.2b (100 %)	1.1×10^{-4}	1.1×10^{6}
	3.3 Abandoned repository	3.3a Base Case only		5.0×10^{-5}	1.0×10^{6}
4. "What if?" cases to investigate robustness of the disposal system	4.1 High water flow rate in geosphere	4.1a Increased water flow rate in geosphere (100-fold increase)		2.0×10^{-3}	6.2×10^{4}
	4.2 Transport along transmissive discontinuities	4.2a/c 1 discontinuity ($T = 10^{-10}$ m^2 s^{-1}) affecting 27 SF/HLW canisters and entire ILW repository		2.7×10^{-4}	1.4×10^{4}
		4.2b 2 discontinuities ($T = 10^{-10}$ m^2 s^{-1}) affecting 108 SF/HLW canisters		6.7×10^{-4}	1.4×10^{4}
		4.2d/f 1 discontinuity ($T = 10^{-9}$ m^2 s^{-1}) affecting 27 SF/HLW canisters and entire ILW repository		1.1×10^{-2}	1.4×10^{3}
		4.2e 2 discontinuities ($T = 10^{-9}$ m^2 s^{-1}) affecting 108 SF/HLW canisters		1.1×10^{-2}	1.4×10^{3}
	4.3 SF: Increased fuel dissolution rate	4.3a 10-fold increase		1.9×10^{-4}	1.6×10^{6}
		4.3b 100-fold increase		5.1×10^{-4}	1.0×10^{6}
	4.4 Redox front (SF/ILW compacted hulls)	4.4a Base Case only		1.9×10^{-4}	1.6×10^{6}
	4.5 ILW: Gas-induced release of dissolved radionuclides through the ramp only	Water flow rate	4.5a (50 %, 0.05 m^3 a^{-1})	5.3×10^{-5}	1.0×10^{6}
			4.5b (100 %, 0.3 m^3 a^{-1})	3.4×10^{-4}	1.8×10^{4}

Tab. 8.2-2: (Cont.)

Scenario	Conceptualisation	Parameter variation		Maximum annual dose	
				Dose [mSv a^{-1}]	Time of occurrence [a]
4. "What if?" cases to investigate robustness of the disposal system	4.6 Unretarded transport of ^{14}C from SF and ILW released as volatile species through host rock; retardation in confining units taken into account (volatile ^{14}C only)	Gas permeability [m^2]	4.6a (10^{-23})	5.8×10^{-5}	1.0×10^{4}
			4.6b (10^{-22})	5.5×10^{-5}	1.0×10^{4}
			4.6c (0)	5.8×10^{-5}	1.0×10^{4}
	4.7 Poor near field and pessimistic near field/ geosphere geochemical dataset	Water flow rate in geosphere	4.7a RC flow rate	4.5×10^{-4}	6.2×10^{5}
			4.7b 10-fold increase	1.6×10^{-3}	2.3×10^{5}
			4.7c 100-fold increase	1.1×10^{-2}	3.4×10^{4}
	4.8 No advection in geosphere (diffusive transport only)	4.8a Base Case only		1.9×10^{-5}	1.3×10^{6}
	4.9 SF: Increased cladding corrosion rate	4.9a Base Case only		5.3×10^{-5}	1.0×10^{6}
	4.10 Kd(I) for NF and geosphere = 0	4.10a Base Case only		1.2×10^{-4}	4.8×10^{5}
	4.11 Decreased transport distance in Opalinus Clay (30 m)	4.11a Base Case only		8.8×10^{-5}	5.5×10^{5}
5. Design and System Options	5.1 Increased waste arisings (300 GWa(e))	5.1a Base Case only		8.6×10^{-5}	1.0×10^{6}
	5.2 ILW high force compacted waste option	5.2a Base Case only		5.3×10^{-5}	1.0×10^{6}
	5.3 SF canister with Cu shell	5.3a Canister breaching at 10^{5} a		4.8×10^{-5}	1.1×10^{6}
		5.3b Initial defect (small initial pinhole, full breaching at 10^{5} a) [1]		5.3×10^{-5}	1.0×10^{6}
		5.3c Initial defect (Large initial pinhole, full breaching at 10^{5} a) [1]		5.3×10^{-5}	1.0×10^{6}
6. Illustration of effects of biosphere uncertainty	6.1 Reference and alternative geomorphology	6.1a Reference area (RC)		5.3×10^{-5}	1.0×10^{6}
		6.1b Sedimentation area		2.5×10^{-5}	1.0×10^{6}
		6.1c Wetland		4.8×10^{-6}	1.0×10^{6}
		6.1d Exfiltration to spring located at valley side		1.0×10^{-4}	1.0×10^{6}
	6.2 Reference and alternative climates	6.2a Present-day climate (RC)		5.3×10^{-5}	1.0×10^{6}
		6.2b Drier/warmer than present-day climate		5.8×10^{-4}	1.0×10^{6}
		6.2c Wetter/warmer than present-day climate		1.8×10^{-5}	1.0×10^{6}
		6.2d Periglacial climate		5.3×10^{-7} to 5.3×10^{-5}	1.0×10^{6}

[1] Summed dose maximum for a single canister

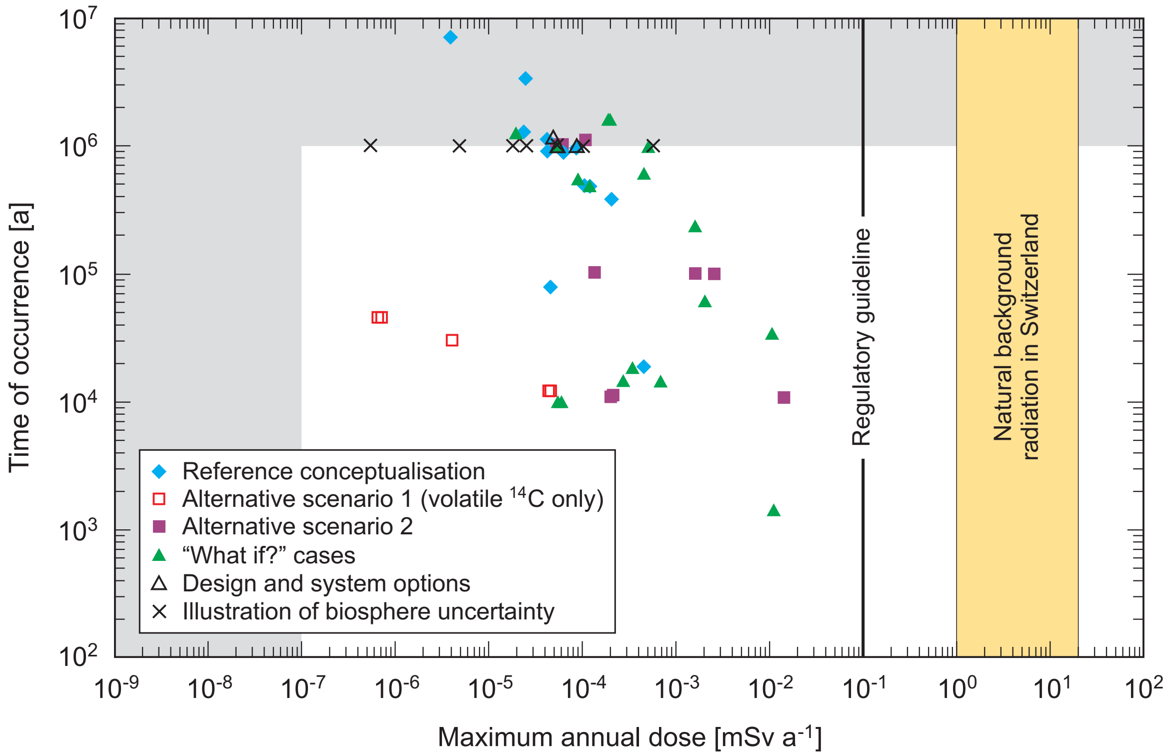

Fig 8.2-3: Scatter plot showing the maximal dose and the time of occurrence of that dose for all cases analysed (Tab. 8.2-2)

The symbols indicate to which scenario a given case belongs. Note that while cases belonging to the reference conceptualisation and to the alternative scenario 1 (dose due to release of volatile ^{14}C along gas pathways) are considered to be realistic possibilities, those belonging to the alternative scenario 2 (human actions) and the "what if?" cases are in an entirely different category. The "human actions" cases should be seen as illustrative stylised conceptualisations, while the "what if?" cases are outside the range of possibilities supported by scientific evidence, but are included as examples to illustrate the robustness of the system behaviour under extreme conditions.

8.2.8.2 Complementary safety indicators

As well as evaluating safety in terms of dose and risk, the safety and performance of the repository system can be assessed using a range of other safety indicators (see, e.g., IAEA 1994b, IAEA 2002b). Here, the following complementary safety indicators are used (see also Appendix 3):

- **radiotoxicity** of the wastes, which is evaluated as a function of time and compared with that of natural materials,
- **radiotoxicity fluxes** due to radionuclides released from the repository in the course of time, which are compared with natural radiotoxicity fluxes in the surface environment,
- **radiotoxicity concentrations** originating from the repository at the top of the Opalinus Clay, as a function of time, compared with natural radiotoxicity concentrations in the Opalinus Clay, and
- **distribution of radiotoxicity** in different components of the repository system, which are evaluated as functions of time, illustrating the fate of radionuclides and, in particular, the degree to which they decay before reaching the surface environment.

Radiotoxicity of the wastes

As discussed in Chapter 2, the *radiotoxicity index (RTI)* is a useful measure of the potential hazard of radioactive material. In Fig. 2.5-1, the RTI of the total radionuclide inventory of the three waste categories to be emplaced in the proposed repository is compared with that of the natural radionuclides contained in 1 km^3 of Opalinus Clay and with that of a volume of natural uranium ore corresponding to the volume of the SF / HLW emplacement tunnels. In the latter case, three uranium concentrations (uranium ore grades) are considered. These are 3 %, which is the average uranium concentration of the small uranium deposit of La Creusa, Switzerland; 8 %, which is a representative value for the Cigar Lake uranium deposit; and 55 %, which is near the upper end of observed concentrations in uranium ore bodies. After one million years, the radiotoxicity of even the most toxic waste, the spent fuel, has dropped to well below that of a volume of natural uranium ore sufficient to fill the SF / HLW emplacement tunnels.

Radiotoxicity fluxes

In Fig. 8.2-4, the reference-case radiotoxicity flux from the repository at the boundary Opalinus Clay - confining units is compared with naturally occurring radiotoxicity fluxes. The figure shows that the maximum radiotoxicity flux from the repository is more than an order of magnitude below the natural radiotoxicity flux in the biosphere aquifer and about seven orders of magnitude below that from natural radionuclides transported by the river Rhine near Basel today. In order to widen the basis for comparison, the natural radiotoxicity fluxes corresponding to (i) the radionuclides transported by the river Thur, (ii) the erosion of the reference biosphere area, and (iii) the annual consumption of mineral water in Switzerland are also given (for details, see Appendix 3).

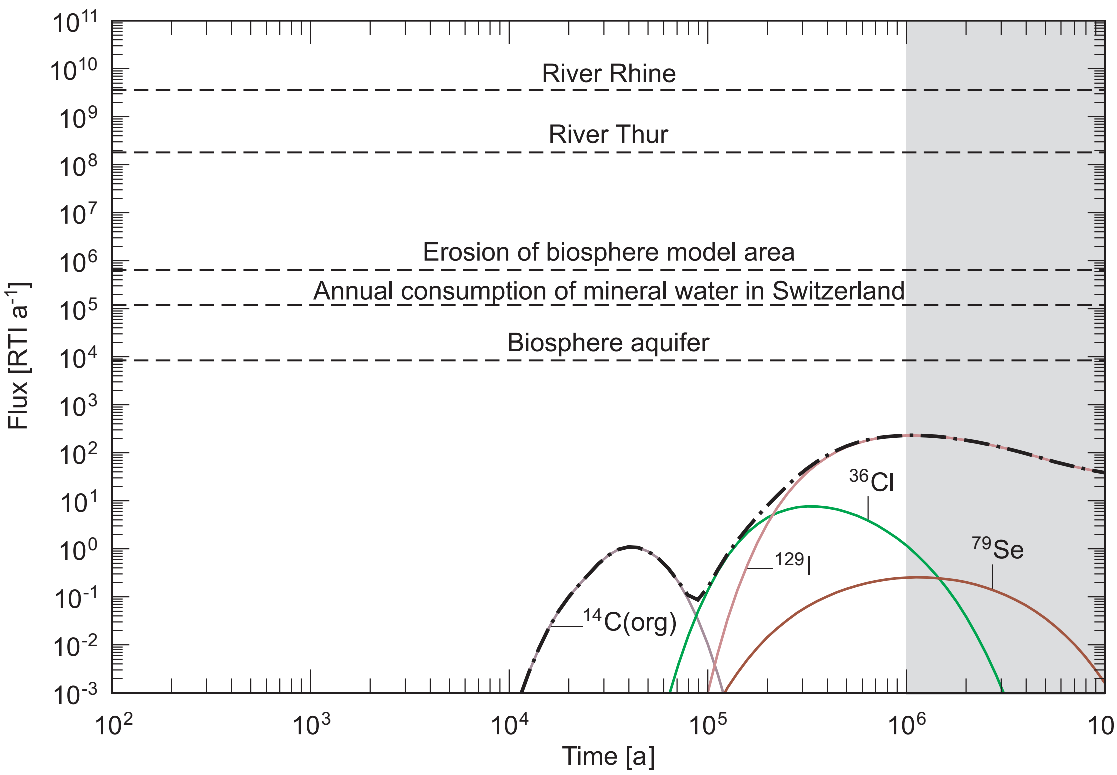

Fig. 8.2-4: Radiotoxicity flux, for the Reference Case, from the repository (summed over all waste types) at the boundary Opalinus Clay – confining units, compared with a range of radiotoxicity fluxes due to naturally occurring radionuclides

Radiotoxicity concentrations

In Fig. 8.2-5, the time-dependent radiotoxicity concentration due to the repository, which is regarded as a planar source, is calculated for a layer of Opalinus Clay that is 1 m thick, covers an area of 1 km^2 (roughly equivalent to the area occupied by the waste emplacement tunnels) and is 40 m away from the repository for the Reference Case. This is compared with the radiotoxicity concentration in Opalinus Clay due to natural radionuclides. The figure shows that the maximum radiotoxicity concentration due to the repository is more than three orders of magnitude below the natural radiotoxicity concentration in Opalinus Clay.

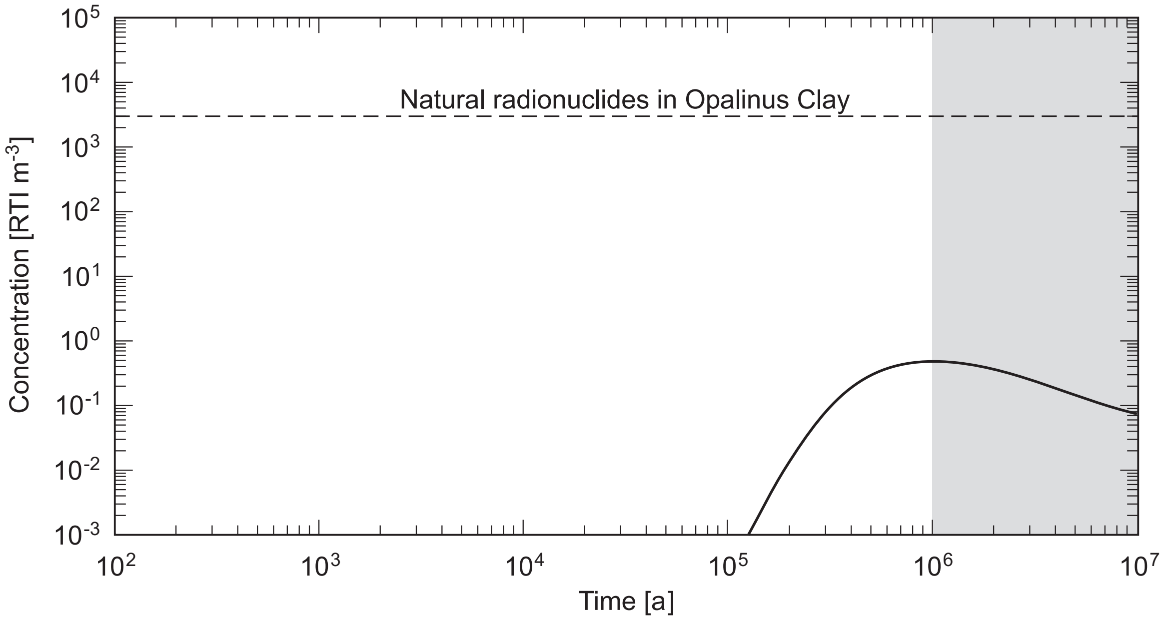

Fig. 8.2-5: Radiotoxicity concentration, for the Reference Case, in a 1 m thick layer at the top of the Opalinus Clay due to the repository (summed over all waste types), compared with that in Opalinus Clay due to naturally occurring radionuclides

Distribution of radiotoxicity

The distribution of radiotoxicity in the various repository system components as a function of time has been discussed in Chapter 6 (Figs. 6.4-2, 6.4-4, 6.4-5). Key findings were:

- **SF** – For times up to 10^7 years and beyond, the largest fraction of the radiotoxicity is contained within the fuel matrix. At 10^7 years, the radiotoxicity present outside the Opalinus Clay is still about three orders of magnitude below that in the fuel matrix.
- **HLW** – For times up to about 10^5 years, the largest fraction of the radiotoxicity is contained within the glass matrix and at later times sorbed in the bentonite buffer. Just before 10^7 years, the radiotoxicity present outside the Opalinus Clay reaches a maximum (barely visible in Fig. 6.4-4), but is still about five orders of magnitude below that sorbed in the buffer.
- **ILW** – For times up to 10^7 years and beyond, the largest fraction of the radiotoxicity is sorbed and/or precipitated in the cementitious near field. Just before 10^7 years, the radio-toxicity present outside the Opalinus Clay reaches a maximum, but is still about an order of magnitude below that sorbed on cement.

- **All three waste types** – The fraction of radiotoxicity released to the biosphere compared to that contained in the wastes at waste emplacement is for all three waste types extremely small.

8.2.8.3 Identification of reserve FEPs

Some FEPs that are considered likely to occur and are beneficial to safety are deliberately (and conservatively) excluded from quantitative analysis because suitable models, codes or databases are unavailable. Such FEPs are termed *reserve FEPs*, since they may be mobilised at a later stage of the waste disposal programme when the necessary models, codes and databases will have been developed. Important reserve FEPs identified in the course of the present safety assessment are:

- the co-precipitation of radionuclides with secondary minerals derived from spent fuel, glass and canister corrosion (except for co-precipitation of radium, which is included in all cases),
- sorption of radionuclides on canister corrosion products,
- natural concentrations of isotopes in solution in bentonite porewater, which could further reduce the effective solubilities of some radionuclides,
- additional retardation in the confining units (analysed within an alternative conceptualisation, but conservatively neglected in all other conceptualisations, including the Reference Conceptualisation),
- irreversible sorption of radionuclides in the near field or in the geosphere (surface mineralisation),
- long-term immobilisation processes (precipitation / co-precipitation) in the geosphere,
- the delayed release of radionuclides, due to the slow corrosion rate of ILW metallic materials (e.g. hulls and ends), as well as a period of complete containment by ILW steel drums and emplacement containers.

The reserve FEPs have the potential, in the future, to provide additional quantitative contributions to the evaluated performance of the disposal system. Even in the current assessment, the presence of these reserve FEPs constitutes, in effect, an additional qualitative argument for safety, since it indicates that the actual performance of the disposal system will, in reality, be more favourable than that evaluated in the analysis of assessment cases.

In addition to the reserve FEPs, there are further reserves due to a number of simplifying pessimistic or conservative assumptions that had to be made for the quantitative analysis of some assessment cases because of the limitations of available codes and / or data. Many of these are expected to be replaced by more realistic ones in future safety assessments, which will yield even lower calculated doses than in the present assessment.

8.2.8.4 Absence of outstanding issues with the potential to compromise safety

The current safety assessment, despite an analysis of a wide range of assessment cases that were derived in a careful and methodical way, has not identified any outstanding issues with the potential to compromise safety.

8.2.8.5 Summary: Adequate consideration of safety-relevant phenomena

Complete isolation of radionuclides and their decay without reaching the surface environment cannot be ensured for all times due to a broad spectrum of phenomena that have been listed and discussed in Chapters 4 and 5. These phenomena are considered in detail in this safety assessment and it is shown that despite these phenomena sufficient safety can be achieved because:

1. For the chosen disposal system many of the phenomena critical to safety are ruled out or their consequences are minimised through proper and rigorous application of the "Disposal Principles" (Chapter 2).
2. For several other phenomena a detailed analysis (qualitative reasoning and quantitative considerations) has shown that for the chosen disposal system their domain of influence is either very limited and/or their effect on radionuclide release is very limited (Chapters 4, 5 and 6).
3. For all other phenomena the possibility could not be excluded that they might have an effect on radionuclide release for the chosen disposal system. Therefore, the potential effects of these phenomena have been analysed in quantitative assessment cases. All of the cases yielded results well below the Swiss regulatory guideline, despite the fact that a number of conservative or pessimistic assumptions had been made in the quantitative analyses (Chapter 7).

8.3 Additional evidence for the effectiveness of deep geological disposal in Opalinus Clay

Indirect support for the possibility of safe geological disposal comes from the longevity of uranium ore deposits in many different geological environments around the world. One well-known example of a uranium ore deposit that is analogous in several ways to the disposal system under consideration in the Opalinus Clay is that of the 1900 million year old Cigar Lake ore body in Canada. An iron-rich zone (analogous to the corroded steel canister) surrounds the uranium ore (analogous to spent fuel). The iron-rich zone is in turn surrounded by a clay-rich zone (analogous to the clay barrier provided by the bentonite buffer and Opalinus Clay). The significant features of the ore body as an analogue are that radionuclides have been retained within and around the deposit for prolonged periods and, despite the fact that the deposit contains the highest grade of natural uranium ore currently known (up to 60 wt %), there is no trace of either radiation or radionuclides from the ore at the ground surface some 450 m above. Indeed, the ore body was only discovered due to the application of geochemical expert knowledge, which indicated the likely environments for uranium ore entrapment.

Another example of effective isolation of radioactive elements comes from studies of the Oklo natural reactors. The rich uranium deposit at Oklo formed about 2000 million years ago and a nuclear chain reaction occurred in several parts of the ore body over a period of several hundred thousand years. As a result, large quantities of actinides and fission products were produced. The conditions in the reactors, including temperatures exceeding 400 ^{o}C and the presence of circulating hydrothermal fluid, were far more extreme than those expected in the repository. Despite these extreme conditions, the uraninite in the deposit retained many of the radionuclides, including most of the actinides such as plutonium and neptunium and the long-lived fission product ^{99}Tc. Some more mobile elements such as iodine were lost from the uraninite, probably while the reactors were operating. While quantitative results are difficult to derive from such a system, the behaviour of the elements broadly matches the results of the safety assessment calculations and provides direct evidence of natural isolation of many radionuclides. Further evidence for the effectiveness of radionuclide retention processes in natural systems is discussed in Section 5.7.1.

8.4 Guidance for future stages of planning and development

At the current stage of the repository programme, with possible implementation still tens of years away, an important role of safety assessment is to provide a platform for discussion of a broad range of topics related to repository development. In addition, the findings from the safety assessment, together with those from the regulatory authorities' review thereof, will provide guidance for future stages of planning and development. One important result of the present safety assessment that can provide such guidance is the identification of the *pillars of safety.* Clearly, future work should focus on phenomena directly connected to the pillars of safety. A further enhancement of understanding of these key phenomena will strengthen future safety cases.

To a large extent, such work has already started. At the Mont Terri rock laboratory and elsewhere, a number of experiments are on-going and studies are under way or planned that investigate key phenomena, including:

- diffusion experiments in Opalinus Clay at Mont Terri and in the laboratory using a variety of radionuclides;
- investigations of self-sealing of fractures within the excavation-disturbed zone of the tunnel and in other locations;
- a heater experiment at Mont Terri to enhance system understanding associated with the transient phase when temperatures are enhanced;
- studies on deformation mechanisms to enhance system understanding;
- studies at Mont Terri on the transient behaviour of the host rock during saturation and desaturation as well as the interaction of the Opalinus Clay with the bentonite buffer;
- investigations regarding the practicalities of using bentonite granules to fill repository tunnels;
- additional experiments looking at gas release through the host rock and seals (under discussion).

Other issues being addressed include those associated with geochemical immobilisation (with a very extensive laboratory programme underway at the Paul Scherrer Institute) and the acquisition of geochemical data for *in situ* conditions.

Furthermore, a number of international collaborations, with active Nagra participation, are under way that focus on the key phenomena linked to the pillars of safety identified in the current safety assessment. Some of these collaborations are within the European Commission's Framework Programmes; others were initiated by waste management agencies interested in enhancing the understanding of specific phenomena.

Finally, the work and progress in other countries will be carefully monitored. This includes the work in France on the Callovo-Oxfordian that is rather similar to the Opalinus Clay, the work in the Boom Clay in Belgium but also the work on engineered barriers, e.g. in Sweden and Finland.

9 Conclusions

9.1 Aims and structure of the chapter

This chapter gives the conclusions of the safety assessment, structured according to the assessment aims given in Chapter 1, which were:

- to determine the suitability of the Opalinus Clay of the Zürcher Weinland as a host rock for a repository for SF / HLW / ILW from the point of view of long-term safety,
- to enhance the understanding of the multiple safety functions that the proposed disposal system provides,
- to assess the robustness of the disposal system with respect to uncertainties and the effects of phenomena that may adversely affect the safety functions, and
- to provide a platform for the discussion of a broad range of topics related to repository development and future studies from the point of view of long-term safety.

Conclusions of the assessment with respect to each of these aims are given in Sections 9.2 to 9.5, respectively. In addition, Section 9.6 summarises the overall project conclusions in view of the overall project aims, which were:

- to demonstrate disposal feasibility of SF, HLW and ILW in the Opalinus Clay of the Zürcher Weinland in order to fulfil the requirements defined by the Federal Council in 1988 in its judgement of Project Gewähr 1985, and
- to provide a platform for discussion and a foundation for decision-making on how to proceed with the Swiss programme for the disposal of SF, HLW and ILW.

9.2 Evaluation of the Opalinus Clay of the Zürcher Weinland as a host rock for the repository from the point of view of long-term safety

Geological disposal is attractive as a waste management option because of the favourable conditions for long-term isolation that can be found deep underground and the long-term predictability of these conditions. To make a convincing case that a chosen site and repository design can provide the high levels of safety demanded by regulations, it is necessary to assess the behaviour of the disposal system based on the observed properties of the system and a sound understanding of how the system can evolve with time. The Opalinus Clay in the siting area in the Zürcher Weinland possesses a number of attributes that intrinsically favour safety and its demonstration. These include very low permeability, good retardation properties, a self-sealing capacity, homogeneity, reasonable constructability and good explorability and predictability.

The safety assessment has evaluated quantitatively the suitability of the siting area in the Zürcher Weinland and the proposed repository design from the point of view of long-term safety. The objective is not to deterministically predict the future behaviour of the system. Instead, the spectrum of possibilities for the characteristics and evolution of the proposed repository in Opalinus Clay has been assessed based on current scientific understanding. The expected, or most likely, characteristics and evolution have been identified, as have possible deviations from these, with emphasis on potentially detrimental phenomena and uncertainties.

Since the system is sited and designed to avoid detrimental phenomena and uncertainties where possible, some deviations have been shown to be either highly unlikely or irrelevant since they do not impair the basic safety functions of the repository. Some deviations do affect the safety functions, but their effects are mitigated by favourable features of the site, by specific design measures and/or by a degree of flexibility in the manner in which the disposal system might be implemented. The fact that multiple phenomena contribute to the safety functions also mitigates the effects of some deviations; poor performance by one safety barrier in the system can be compensated in part by the correct functioning of others.

For the credible scenarios of future system evolution, a wide range of cases has been quantitatively analysed to determine the potential radiological consequences. These analyses use models and data that are judged to be reliable either because they are demonstrably realistic or, where there is significant uncertainty, because they are known to give a pessimistic bias to calculated consequences. These cases represent not only the expected, or most likely, characteristics and evolution of the disposal system, but also possible deviations, with an emphasis on those with the potential to adversely perturb or by-pass the clay barrier. The results of the analyses have shown that the disposal system provides the required level of long-term safety in that they comply with Swiss regulatory criteria. In summary the analyses performed show that for a repository in Opalinus Clay in the siting area of the Zürcher Weinland:

- the security of fissile materials will be assured,
- almost all radionuclides will be retained within the repository and its surroundings for sufficiently long so that they largely decay before they can reach the biosphere,
- any long-lived radionuclides that do reach the biosphere will be present only at very low concentrations and pose no undue hazard to humans or the environment.

In addition, there are a number of phenomena (reserve FEPs) that are expected to be beneficial to safety, but are not included in the current evaluation of assessment cases due to limitations in the available information or modelling tools. The existence of these phenomena constitutes an additional qualitative argument for safety. Furthermore, independent evidence has been used to ensure the adequacy of the range of cases considered and the reliability of the conceptualisations, mathematical models, computer codes and data used for the analyses of these cases.

The findings of the safety assessment support the suitability of the Opalinus Clay in the siting area of the Zürcher Weinland as a host rock for a repository for SF / HLW / ILW and a safety case has been made that shows that safe disposal in this siting area is feasible.

9.3 Understanding of the multiple safety functions that the proposed disposal system provides

The safety assessment has led to a better understanding of the *safety functions* that the disposal system provides. Certain features of the disposal system are key to providing the safety functions; they are termed *pillars of safety.* They are:

- the *deep underground location of the repository*, in a setting that is unlikely to attract human intrusion and is not prone to disruptive geological events and to processes unfavourable to stability,
- the *host rock*, which has a low hydraulic conductivity, a fine, homogeneous pore structure and a self-sealing capacity, thus providing a strong barrier to radionuclide transport and a suitable environment for the engineered barrier system[112],

[112] The backfilled and sealed underground tunnel system is designed to complement the favourable properties of the host rock, isolating the engineered barriers from the surface environment and avoiding the possibility of preferential transport pathways for radionuclides that by-pass the host rock.

- a *chemical environment* that provides a range of geochemical immobilisation and retardation processes, favours the long-term stability of the engineered barriers, and is itself stable due to the phenomena mentioned above and a range of chemical buffering reactions,
- *the bentonite buffer (for SF and HLW)* as a well-defined interface between the canisters and the host rock, with similar properties as the host rock, that ensures that the effects of the presence of the emplacement tunnels and the heat-producing waste on the host rock are minimal, and that provides a strong barrier to radionuclide transport and a suitable environment for the canisters and the waste forms,
- *SF and HLW waste forms* that are stable in the expected environment, and
- *SF and HLW canisters* that are mechanically strong and corrosion resistant in the expected environment, providing complete containment for a considerable period of time.

These pillars of safety, which are critical to the operation of the safety functions and thus to overall system performance, are well understood and reliable. Understanding of the behaviour of individual safety barriers has been enhanced by wide ranging information, including that from laboratory and field experiments, as well as general observations from nature. Understanding of how the barriers work in concert to provide overall safety has been built by examining a wide variety of different assessment cases using both deterministic and probabilistic approaches.

Given the range of evidence that exists to support the understanding, effectiveness and reliability of the pillars of safety, the safety functions provided by the proposed disposal system are judged to be sufficiently well understood to give confidence that they will ensure sufficient safety.

9.4 Robustness of the disposal system

A disposal system founded on a system of passive barriers that provide multiple safety functions in a manner that is well understood is *robust*. In the disposal system considered in the present study, the barriers include the waste forms, canisters, backfill, and the surrounding geological setting, including the Opalinus Clay host rock. The different barriers contribute to the multiple phenomena that comprise the safety functions that ensure safety.

The proposed disposal system has been sited and designed to include attributes that intrinsically favour safety, and that avoid or minimise detrimental phenomena and uncertainties, or mitigate their effects. The safety assessment, in analysing a wide range of cases, has demonstrated the robustness of the disposal system with respect to various detrimental phenomena and uncertainties.

It is also noted that there will be further opportunities for enhancing confidence in the robust performance of the system. Realisation of the repository is still decades ahead and even afterwards there will be no irreversible steps taken. This is because the repository system is based on the concept of monitored long-term geological disposal that has been developed to provide the long-term safety and security of geological disposal, while addressing societal demands that decisions should be made in a cautious stepwise manner, with the possibility of reversal of decisions.

Thus, the disposal system is robustly designed and is robust with respect to uncertainties and to the effects of phenomena that may adversely affect the safety functions. An evaluation of such uncertainties and phenomena has shown that they do not lead to unacceptable consequences.

9.5 The present safety assessment: A platform for discussion and guidance for future stages of repository planning and development

The present safety assessment provides a platform for discussion of a broad range of topics related to repository development and to future studies. Concerning possible future developments, the most important conclusion that can be drawn from the safety assessment described in this report is that, from a technical point of view, it is considered to be justifiable for Nagra to propose, for consideration by the Swiss Government, the siting area of the Zürcher Weinland with the Opalinus Clay as host rock as the focus for further investigations for a repository for SF / HLW / ILW. This point is expanded upon in Section 9.6. At a more detailed level, the safety assessment and its regulatory review will provide a platform for discussion of future studies that can best further enhance confidence in safety, contribute to optimisation of the system and ultimately support future possible licence applications.

The comprehensive safety assessment documented in this report provides a well-founded platform for planning further repository development work and for identifying key scientific issues that would benefit from further investigations. Some issues are already being addressed in ongoing studies.

9.6 Overall conclusions

1. Project *Entsorgungsnachweis* is a response to the request by the Swiss Government for a convincing demonstration of siting feasibility following the Government's review of the earlier Project Gewähr. The work described in the present report has shown that disposal is feasible from a safety point of view for the chosen system in the Opalinus Clay in the siting area in the Zürcher Weinland. Specifically the data and the analyses show that:

   - the reference site has properties that ensure sufficient safety. The safety case provides arguments that the repository is safe: there is sufficient safety for a broad spectrum of cases and the spectrum of analysed cases is broad enough to cover all reasonable possibilities;
   - the system is robust; i.e. remaining uncertainties do not put safety in question;
   - the information basis for the wastes and the engineered barrier system is adequate and draws on more than 20 years of work in Switzerland and wide experience abroad.

   In addition, as is extensively documented in the accompanying report on the facilities, the site properties and the design of the facility allow construction, operation and closure of the repository according to specifications, and thus to safety requirements.

   The information basis for the site is sufficient and the site is sufficiently well understood to support these statements on safety and engineering feasibility. The third report, the geosynthesis, illustrates that:

   - the geometry and structure of the host rock and confining units are well characterised with state-of-the-art 3 D seismics to identify a sufficiently large undisturbed area for allocation of the repository;
   - the host rock and confining units, which have been characterised with the deep borehole at Benken, have favourable properties that ensure long-term safety;
   - relevant processes have been investigated in detail in the underground rock laboratory at Mont Terri and in the laboratory and confirm and complement the findings from the borehole at Benken. Thus, the properties of the site and host rock and their future evolution can be bounded with confidence based on information from an extensive

regional geological programme and the fact that the overall situation of the site is reasonably simple.

2. Project *Entsorgungsnachweis* provides a platform for discussion and a foundation for decision-making on how to proceed with the Swiss HLW programme and to assess the role of the Opalinus Clay of the Zürcher Weinland in this programme. The excellent results obtained from the geological investigations and from the safety assessment in Project *Entsorgungsnachweis* have led Nagra to put forward a proposal, for consideration by the Swiss Government, to focus future work for the waste management option "Disposal of SF / HLW / ILW in Switzerland" on the Opalinus Clay of the Zürcher Weinland[113]. This is justified by the facts that:

   - a systematic screening of potential sedimentary host rocks[114] has indicated that Opalinus Clay has a number of particularly favourable properties, such as tightness, good retardation properties, a self-sealing capacity, good constructability, and good explorability;
   - a systematic screening of situations[114] has indicated that the Zürcher Weinland has a number of favourable properties, such as a low tectonic activity, presence of Opalinus Clay at a suitable depth and with sufficient lateral extent, and the existence of confining units with similar properties as the host rock itself;
   - Project *Entsorgungsnachweis* clearly indicates that, for a reference system in the Opalinus Clay of the Zürcher Weinland
     - a high level of safety can be expected,
     - construction, operation and closure of the repository is feasible,
     - the site has good qualities and offers sufficient flexibility.

   The positive results obtained for this host rock and region does not imply that a safe system could not be implemented in other regions where Opalinus Clay is present, or in other host rocks. However, the technical arguments (based on safety, geological simplicity and predictability) that led to this region being preferred are considered to be plausible and well founded.

3. There are still many steps to be taken before a repository is definitively sited in Switzerland, underground explorations are carried through, final designs are agreed upon, and licensing activities are undertaken. A formal siting decision, which will be a milestone within the general licence process, is expected around the year 2020 at the earliest. This means that ample time is available to continue the investigations and to iterate on the repository design. Therefore, the level of detail documented in Project *Entsorgungsnachweis* is considered to be adequate for the current phase of the programme and sufficient to support the conclusions that disposal of SF / HLW / ILW in Switzerland is feasible (Statement 1 above) and that the choice of the Zürcher Weinland as the focus for future work, with the Opalinus Clay as host rock, is justified (Statement 2 above).

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[113] Disposal abroad is an officially recognised option of the Swiss waste management strategy.

[114] This step-wise approach that lasted several years was done in close interaction, and in agreement, with the authorities and their advisory committees and is documented in several reports (see Chapter 1).

10 References

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Appendix 1 Parameter Values, Probability Density Functions and their Application in Deterministic and Probabilistic Assessment Calculations

A1.1 Values supported by observations and experiments

There are inevitably uncertainties associated with all model parameters that are derived from observations and experiments. This is partly because these observations and experiments are always, to some extent, incomplete and of finite accuracy. It is not, for example, practical to measure every potentially relevant rock property at every point in space, and every measurement has associated measurement errors. Parameter uncertainties also arise because the models in which the parameters are used are always simplifications of reality. Even if, for example, the spatial heterogeneity of a medium is fully characterised, if the model represents the medium as homogeneous, then the averaging of parameters that has to be carried out in order to apply the model is generally a source of uncertainty. A more thorough discussion of uncertainties is presented in Chapter 3.

In the safety assessment, experts in scientific fields relevant to a particular set of parameters have been be asked to assign values to parameters, taking into account all relevant information, that they consider to be:

- expected or most likely, and
- pessimistic, in that, when the parameter value is applied within an assessment model, calculated radiological consequences are towards the high end of the range of possibilities supported by current understanding (Fig. A1.1).

For some key parameters relating to the characteristics and evolution of the disposal system, experts have also been asked to select optimistic[1] values and probability density functions[2] (PDFs). PDFs are assumed to take one of a number of standard forms. These forms include:

- uniform,
- log-uniform,
- triangular,
- log-triangular,
- normal, and
- log-normal.

The first four lie between optimistic and pessimistic bounding values, while normal and log-normal PDFs are generally unbounded. However, the experts can choose to select upper and / or lower cut-off values, beyond which a normal or log-normal PDF is truncated, and assumed to take a zero probability. Finally, it is possible to assign probabilities to a limited number of discrete parameter values, while setting the probabilities of all other values to zero. Fig. A1.1 illustrates some of these alternative distributions.

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[1] When an optimistic parameter value is applied within an assessment model, calculated radiological consequences are towards the low end of the range of possibilities supported by current understanding (Fig. A1.1).

[2] A PDF for a parameter p takes a value $f(p_1)$ if the probability of the parameter value lying between p_1 and $p_1 + \delta p$ is $f(p_1)\delta p$, where δp is a small increment in p.

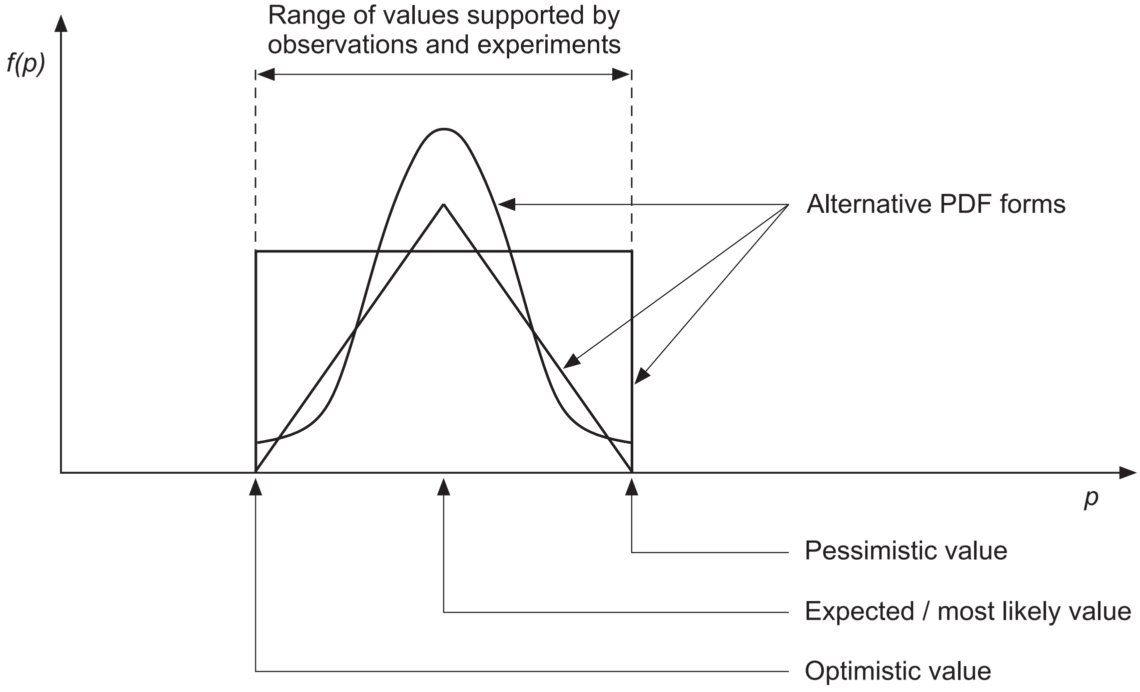

Fig. A1.1: Schematic illustration of PDFs for a parameter *p*

A1.2 "What if?" values

Where possible, support for parameter values and PDFs is sought from a wide range of sources, including laboratory and field experiments and observations from nature, in order to ensure that the range of parameter uncertainty is reliably bounded. In order to test the robustness of the system with respect to parameter uncertainty, a limited number of "what if?" parameter values are considered that lie outside the range of possibilities supported by observations and experiments and for which it is thus meaningless to make a statement about their probability of occurrence. Fig. A1.2 illustrates the relationship between "what if?" values and the range of values supported by observations and experiments.

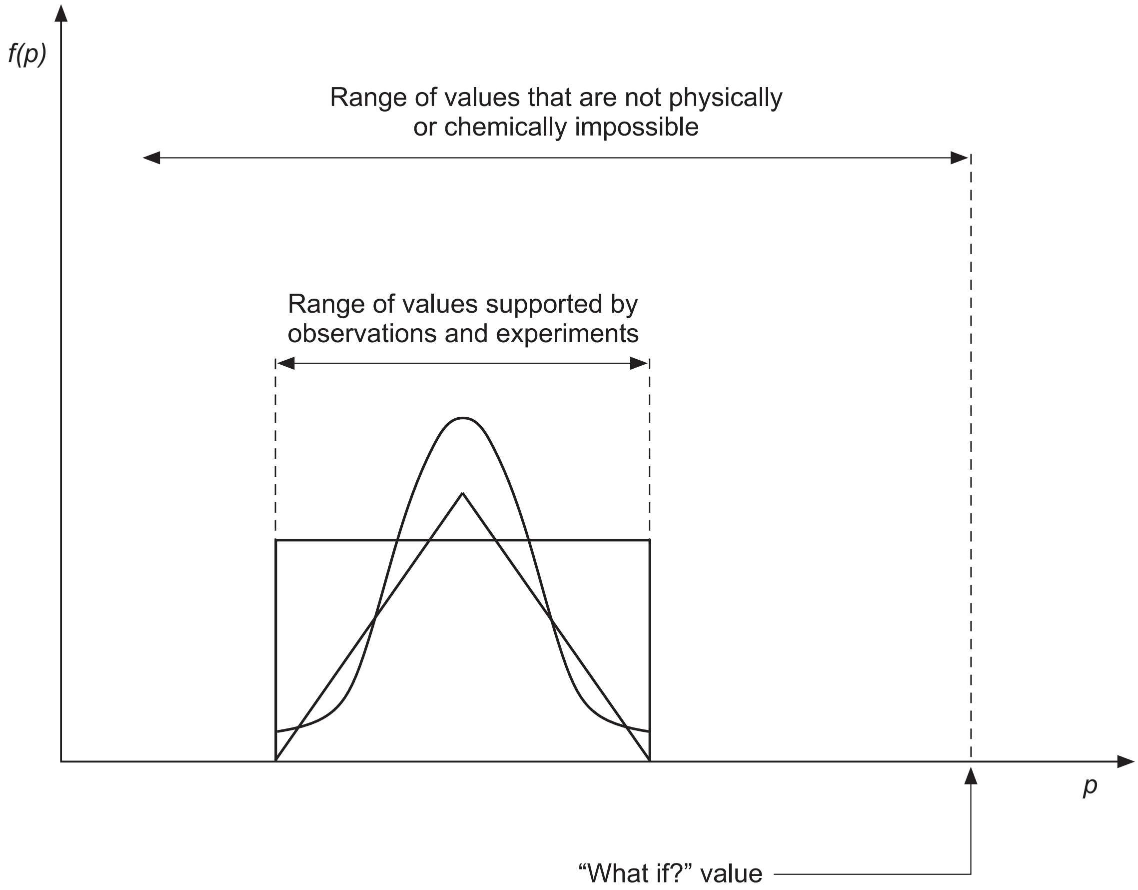

Fig. A1.2: The relationship between "what if?" values and the range of values supported by observations and experiments

Appendix 2 Key Data for Evaluating Assessment Cases

The following tables give data used in performing various safety assessment calculations for the Reference Case. Data sources are discussed in Chapters 4 and 5. The selection of radionuclides included in the tables is based on screening calculations discussed in Nagra (2002c).

A2.1 Radionuclide inventories in reference canisters of SF and HLW and in emplacement tunnels ILW-1 and ILW-2

Tab. A2.1.1: Inventories of radionuclides in a reference canister containing 9 BWR UO_2 fuel assemblies with a burnup of 48 GWd/t_{IHM}, after 40 years decay

Tab. A2.1.2: Inventories of radionuclides in a canister containing 3 PWR UO_2 and 1 MOX fuel assemblies with a burnup of 48 GWd/t_{IHM}, after 40 years decay

Tab. A2.1.3: Inventories of radionuclides in a canister containing 4 PWR UO_2 fuel assemblies with a burnup of 48 GWd/t_{IHM}, after 40 years decay

Tab. A2.1.4: Average radionuclide inventory of a single BNFL HLW glass flask, after 40 years decay

Tab. A2.1.5: Average radionuclide inventory of a single COGEMA HLW glass flask, after 40 years decay

Tab. A2.1.6: Total radionuclide inventory of the ILW-1 disposal tunnels, after 40 years decay

Tab. A2.1.7: Total radionuclide inventory of the ILW-2 disposal tunnel, after 40 years decay

A2.2 Other reference data for assessment calculations for SF, HLW and ILW

Tab. A2.2.1: Instant release fractions (IRF) for spent UO_2 and MOX fuel

Tab. A2.2.2: Dissolution rates of spent UO_2 and MOX fuel

Tab. A2.2.3: Radionuclide release parameters for fuel assembly structural materials

Tab. A2.2.4: Other SF near field parameter values

Tab. A2.3: Reference data for HLW canisters

Tab. A2.4: Solubilities of radionuclides in the SF / HLW near field

Tab. A2.5: Solubilities of radionuclides in the ILW near field

Tab. A2.6: Transport parameters for the SF / HLW near field. Sorption values (K_d), effective diffusion coefficients (D_e) and accessible porosities (e) in compacted bentonite, pH = 7.25, Eh = -194 mV

Tab. A2.7: Sorption values (K_d) in cement for the waste groups ILW-1 and ILW-2

A2.1 Radionuclide inventories in reference canisters of SF and HLW and in emplacement tunnels ILW-1 and ILW-2

Tab. A2.1.1: Inventories of safety-relevant radionuclides in a reference canister containing 9 BWR UO_2 fuel assemblies with a burnup of 48 GWd/t_{IHM}, after 40 years decay

The mass of fuel in a canister is 1.593 t_{IHM}; the total mass of structural materials 0.796 t. Data taken from McGinnes (2002). A dash means activity is less than 1 Bq.

Radionuclide	Fuel [Bq]	Structural materials [Bq]	Radionuclide	Fuel [Bq]	Structural materials [Bq]
^{3}H	7.0×10^{12}	5.4×10^{9}	^{228}Ra	3.2×10^{1}	-
^{10}Be	1.8×10^{7}	2.9×10^{3}	^{227}Ac	3.3×10^{5}	-
$^{14}C_{inorg}$	6.7×10^{10}	0	^{228}Th	1.9×10^{8}	6.8×10^{1}
$^{14}C_{org}$	0	5.7×10^{10}	^{229}Th	1.4×10^{4}	-
^{36}Cl	8.6×10^{8}	1.4×10^{9}	^{230}Th	3.2×10^{7}	1.6×10^{1}
^{41}Ca	2.2×10^{8}	7.0×10^{7}	^{232}Th	4.1×10^{1}	-
^{59}Ni	8.9×10^{8}	1.3×10^{11}	^{231}Pa	7.6×10^{5}	-
^{63}Ni	9.2×10^{10}	1.3×10^{13}	^{232}U	1.8×10^{8}	6.5×10^{1}
^{79}Se	1.6×10^{9}	8.0×10^{3}	^{233}U	6.7×10^{6}	4.6×10^{0}
^{90}Sr	2.2×10^{15}	6.0×10^{9}	^{234}U	9.7×10^{10}	8.3×10^{4}
^{93}Zr	1.5×10^{11}	1.8×10^{10}	^{235}U	9.1×10^{8}	-
^{93m}Nb	1.2×10^{11}	1.4×10^{10}	^{236}U	2.1×10^{10}	6.0×10^{3}
^{94}Nb	2.1×10^{8}	1.3×10^{10}	^{238}U	1.9×10^{10}	2.2×10^{4}
^{93}Mo	8.3×10^{7}	3.3×10^{8}	^{237}Np	3.2×10^{10}	4.6×10^{4}
^{99}Tc	1.1×10^{12}	6.7×10^{7}	^{238}Pu	2.6×10^{14}	6.0×10^{8}
^{107}Pd	1.0×10^{10}	1.2×10^{5}	^{239}Pu	2.2×10^{13}	2.2×10^{8}
^{108m}Ag	1.4×10^{9}	4.3×10^{4}	^{240}Pu	4.0×10^{13}	3.0×10^{8}
^{126}Sn	3.0×10^{10}	3.2×10^{5}	^{241}Pu	1.5×10^{15}	1.9×10^{10}
^{129}I	2.7×10^{9}	1.9×10^{4}	^{242}Pu	1.9×10^{11}	2.9×10^{6}
^{135}Cs	3.7×10^{10}	1.8×10^{5}	^{241}Am	3.0×10^{14}	3.7×10^{9}
^{137}Cs	3.5×10^{15}	1.9×10^{10}	^{242m}Am	5.7×10^{11}	7.1×10^{6}
^{151}Sm	1.6×10^{13}	1.1×10^{8}	^{243}Am	2.2×10^{12}	3.8×10^{7}
^{166m}Ho	2.1×10^{9}	1.0×10^{4}	^{243}Cm	6.7×10^{11}	1.0×10^{7}
^{210}Pb	9.4×10^{4}	-	^{244}Cm	8.0×10^{13}	1.4×10^{9}
^{210}Po	9.1×10^{4}	-	^{245}Cm	5.6×10^{10}	1.0×10^{6}
^{226}Ra	2.9×10^{5}	-	^{246}Cm	1.2×10^{10}	2.6×10^{5}

Tab. A2.1.2: Inventories of safety-relevant radionuclides in a canister containing 3 PWR UO_2 and 1 MOX fuel assemblies with a burnup of 48 GWd/t_{IHM}, after 40 years decay

The mass of fuel in a canister is 1.495 t_{IHM}; the total mass of the structural materials is 0.564 t. Data taken from McGinnes (2002). A dash means activity is less than 1 Bq.

Radionuclide	Fuel [Bq]	Structural materials [Bq]	Radionuclide	Fuel [Bq]	Structural materials [Bq]
^{3}H	6.6×10^{12}	3.0×10^{9}	^{228}Ra	2.4×10^{1}	-
^{10}Be	1.7×10^{7}	4.5×10^{3}	^{227}Ac	3.0×10^{5}	-
$^{14}C_{inorg}$	5.6×10^{10}	0	^{228}Th	1.7×10^{8}	5.1×10^{1}
$^{14}C_{org}$	0	3.2×10^{10}	^{229}Th	1.1×10^{4}	-
^{36}Cl	6.9×10^{8}	7.0×10^{8}	^{230}Th	3.0×10^{7}	1.0×10^{1}
^{41}Ca	1.8×10^{8}	3.4×10^{7}	^{232}Th	3.1×10^{1}	-
^{59}Ni	7.2×10^{8}	1.2×10^{11}	^{231}Pa	7.0×10^{5}	-
^{63}Ni	7.3×10^{10}	1.2×10^{13}	^{232}U	1.7×10^{8}	4.9×10^{1}
^{79}Se	1.4×10^{9}	4.4×10^{3}	^{233}U	5.4×10^{6}	3.0×10^{0}
^{90}Sr	1.9×10^{15}	3.4×10^{9}	^{234}U	9.5×10^{10}	5.1×10^{4}
^{93}Zr	1.3×10^{11}	9.4×10^{9}	^{235}U	8.4×10^{8}	-
^{93m}Nb	1.0×10^{11}	8.0×10^{9}	^{236}U	1.6×10^{10}	3.7×10^{3}
^{94}Nb	1.8×10^{8}	3.6×10^{10}	^{238}U	1.6×10^{10}	1.5×10^{4}
^{93}Mo	7.3×10^{7}	7.1×10^{8}	^{237}Np	2.8×10^{10}	2.9×10^{4}
^{99}Tc	1.0×10^{12}	1.8×10^{8}	^{238}Pu	3.2×10^{14}	3.6×10^{8}
^{107}Pd	1.1×10^{10}	6.5×10^{4}	^{239}Pu	2.9×10^{13}	1.5×10^{8}
^{108m}Ag	1.2×10^{9}	2.2×10^{4}	^{240}Pu	6.6×10^{13}	2.0×10^{8}
^{126}Sn	3.3×10^{10}	1.8×10^{5}	^{241}Pu	2.5×10^{15}	1.2×10^{10}
^{129}I	2.7×10^{9}	1.1×10^{4}	^{242}Pu	3.1×10^{11}	1.5×10^{6}
^{135}Cs	4.4×10^{10}	1.2×10^{5}	^{241}Am	4.9×10^{14}	2.4×10^{9}
^{137}Cs	3.3×10^{15}	1.1×10^{10}	^{242m}Am	2.3×10^{12}	5.6×10^{6}
^{151}Sm	1.9×10^{13}	7.0×10^{7}	^{243}Am	4.0×10^{12}	2.0×10^{7}
^{166m}Ho	2.0×10^{9}	5.2×10^{3}	^{243}Cm	2.0×10^{12}	5.5×10^{6}
^{210}Pb	8.3×10^{4}	-	^{244}Cm	1.8×10^{14}	7.1×10^{8}
^{210}Po	8.0×10^{4}	-	^{245}Cm	1.8×10^{11}	4.8×10^{5}
^{226}Ra	2.6×10^{5}	-	^{246}Cm	3.3×10^{10}	1.2×10^{5}

Tab. A2.1.3: Inventories of safety-relevant radionuclides in a canister containing 4 PWR UO_2 fuel assemblies with a burnup of 48 GWd/t_{IHM}, after 40 years decay

The mass of fuel in a canister is 1.556 t_{IHM}; the mass of structural materials is 0.589 t. Data taken from McGinnes (2002). A dash means activity is less than 1 Bq.

Radionuclide	Fuel [Bq]	Structural materials [Bq]	Radionuclide	Fuel [Bq]	Structural materials [Bq]
^{3}H	6.9×10^{12}	3.1×10^{9}	^{228}Ra	3.1×10^{1}	-
^{10}Be	1.7×10^{7}	4.5×10^{3}	^{227}Ac	3.9×10^{5}	-
$^{14}C_{inorg}$	6.2×10^{10}	0	^{228}Th	2.0×10^{8}	4.7×10^{1}
$^{14}C_{org}$	0	3.6×10^{10}	^{229}Th	1.4×10^{4}	-
^{36}Cl	8.1×10^{8}	8.2×10^{8}	^{230}Th	3.4×10^{7}	1.1×10^{1}
^{41}Ca	2.2×10^{8}	4.0×10^{7}	^{232}Th	4.0×10^{1}	-
^{59}Ni	8.4×10^{8}	1.2×10^{11}	^{231}Pa	9.0×10^{5}	-
^{63}Ni	8.6×10^{10}	1.3×10^{13}	^{232}U	2.0×10^{8}	4.5×10^{1}
^{79}Se	1.6×10^{9}	4.8×10^{3}	^{233}U	6.5×10^{6}	3.0×10^{0}
^{90}Sr	2.2×10^{15}	3.7×10^{9}	^{234}U	1.0×10^{11}	5.4×10^{4}
^{93}Zr	1.5×10^{11}	1.0×10^{10}	^{235}U	1.1×10^{9}	-
^{93m}Nb	1.2×10^{11}	8.7×10^{9}	^{236}U	2.0×10^{10}	3.9×10^{3}
^{94}Nb	1.9×10^{8}	3.7×10^{10}	^{238}U	1.7×10^{10}	1.6×10^{4}
^{93}Mo	7.6×10^{7}	7.3×10^{8}	^{237}Np	3.1×10^{10}	2.8×10^{4}
^{99}Tc	1.1×10^{12}	1.9×10^{8}	^{238}Pu	2.3×10^{14}	3.9×10^{8}
^{107}Pd	9.6×10^{9}	7.2×10^{4}	^{239}Pu	2.2×10^{13}	1.4×10^{8}
^{108m}Ag	1.3×10^{9}	2.5×10^{4}	^{240}Pu	3.6×10^{13}	2.0×10^{8}
^{126}Sn	3.0×10^{10}	2.0×10^{5}	^{241}Pu	1.5×10^{15}	1.2×10^{10}
^{129}I	2.6×10^{9}	1.2×10^{4}	^{242}Pu	1.7×10^{11}	1.7×10^{6}
^{135}Cs	3.9×10^{10}	1.2×10^{5}	^{241}Am	2.8×10^{14}	2.3×10^{9}
^{137}Cs	3.4×10^{15}	1.2×10^{10}	^{242m}Am	6.2×10^{11}	5.0×10^{6}
^{151}Sm	1.6×10^{13}	6.7×10^{7}	^{243}Am	2.0×10^{12}	2.3×10^{7}
^{166m}Ho	1.9×10^{9}	6.2×10^{3}	^{243}Cm	6.5×10^{11}	6.1×10^{6}
^{210}Pb	1.0×10^{5}	-	^{244}Cm	7.5×10^{13}	8.7×10^{8}
^{210}Po	9.8×10^{4}	-	^{245}Cm	5.0×10^{10}	5.9×10^{5}
^{226}Ra	3.1×10^{5}	-	^{246}Cm	1.2×10^{10}	1.5×10^{5}

Tab. A2.1.4: Average safety-relevant radionuclide content of a single BNFL HLW glass flask, after 40 years decay

Data taken from McGinnes (2002). A dash means activity is less than 1 Bq.

Radionuclide	Activity [Bq]	Radionuclide	Activity [Bq]
$^{14}C_{inorg}$	7.1×10^{7}	^{229}Th	6.3×10^{3}
^{59}Ni	6.7×10^{8}	^{230}Th	1.1×10^{7}
^{63}Ni	7.1×10^{10}	^{231}Pa	2.0×10^{6}
^{79}Se	1.6×10^{9}	^{232}Th	-
^{90}Sr	1.6×10^{15}	^{232}U	2.3×10^{5}
^{93}Zr	1.3×10^{11}	^{233}U	3.5×10^{6}
^{93}Mo	5.0×10^{7}	^{234}U	9.9×10^{7}
^{93m}Nb	1.1×10^{11}	^{235}U	3.0×10^{5}
^{94}Nb	9.9×10^{6}	^{236}U	3.7×10^{6}
^{99}Tc	9.4×10^{11}	^{238}U	6.0×10^{6}
^{107}Pd	7.5×10^{9}	^{237}Np	2.0×10^{10}
^{108m}Ag	4.7×10^{8}	^{238}Pu	7.7×10^{11}
^{126}Sn	5.1×10^{10}	^{239}Pu	6.7×10^{10}
^{129}I	2.1×10^{6}	^{240}Pu	2.3×10^{11}
^{135}Cs	3.9×10^{10}	^{241}Pu	2.2×10^{12}
^{137}Cs	2.3×10^{15}	^{242}Pu	3.5×10^{8}
^{151}Sm	1.7×10^{13}	^{241}Am	1.0×10^{14}
^{166m}Ho	1.5×10^{8}	^{242m}Am	1.2×10^{12}
^{210}Pb	8.1×10^{4}	^{243}Am	8.8×10^{11}
^{210}Po	7.9×10^{4}	^{243}Cm	2.9×10^{11}
^{226}Ra	1.9×10^{5}	^{244}Cm	1.2×10^{13}
^{227}Ac	1.6×10^{6}	^{245}Cm	6.5×10^{9}
^{228}Ra	-	^{246}Cm	1.2×10^{9}
^{228}Th	2.3×10^{5}		

Tab. A2.1.5: Average safety-relevant radionuclide content of a single COGEMA HLW glass flask, after 40 years decay

Data taken from McGinnes (2002). A dash means activity is less than 1 Bq.

Radionuclide	**Activity [Bq]**	**Radionuclide**	**Activity [Bq]**
$^{14}C_{inorg}$	1.9×10^8	^{229}Th	5.5×10^3
^{59}Ni	2.4×10^9	^{230}Th	3.1×10^6
^{63}Ni	2.6×10^{11}	^{231}Pa	8.3×10^5
^{79}Se	1.2×10^9	^{232}Th	-
^{90}Sr	1.3×10^{15}	^{232}U	1.0×10^6
^{93}Zr	9.4×10^{10}	^{233}U	3.1×10^6
^{93}Mo	5.1×10^7	^{234}U	1.0×10^8
^{93m}Nb	7.7×10^{10}	^{235}U	1.2×10^6
^{94}Nb	6.8×10^8	^{236}U	1.7×10^7
^{99}Tc	8.4×10^{11}	^{238}U	1.9×10^7
^{107}Pd	6.5×10^9	^{237}Np	1.7×10^{10}
^{108m}Ag	5.4×10^8	^{238}Pu	3.1×10^{11}
^{126}Sn	3.7×10^{10}	^{239}Pu	2.1×10^{10}
^{129}I	1.6×10^6	^{240}Pu	1.9×10^{11}
^{135}Cs	2.0×10^{10}	^{241}Pu	8.9×10^{11}
^{137}Cs	1.9×10^{15}	^{242}Pu	1.1×10^8
^{151}Sm	1.4×10^{13}	^{241}Am	3.1×10^{13}
^{166m}Ho	1.8×10^8	^{242m}Am	6.0×10^{11}
^{210}Pb	2.3×10^4	^{243}Am	7.6×10^{11}
^{210}Po	2.2×10^4	^{243}Cm	2.7×10^{11}
^{226}Ra	5.4×10^4	^{244}Cm	1.5×10^{13}
^{227}Ac	6.3×10^5	^{245}Cm	5.9×10^9
^{228}Ra	-	^{246}Cm	1.1×10^9
^{228}Th	1.1×10^6		

Tab. A2.1.6: Total safety-relevant radionuclide inventory of the ILW-1 disposal tunnels, after 40 years decay

Data taken from McGinnes (2002).

Radionuclide	Activity [Bq]	Radionuclide	Activity [Bq]
^{3}H	2.2×10^{15}	^{227}Ac	3.7×10^{5}
$^{14}C_{inorg}$	9.2×10^{12}	^{228}Th	1.6×10^{8}
$^{14}C_{org}$	9.1×10^{12}	^{229}Th	3.5×10^{3}
^{36}Cl	6.3×10^{10}	^{230}Th	1.8×10^{7}
^{60}Co	4.0×10^{14}	^{231}Pa	8.3×10^{5}
^{59}Ni	1.2×10^{14}	^{232}U	1.6×10^{8}
^{63}Ni	1.4×10^{16}	^{233}U	1.8×10^{6}
^{79}Se	1.3×10^{9}	^{234}U	5.5×10^{10}
^{90}Sr	2.1×10^{15}	^{235}U	1.0×10^{9}
^{93}Zr	5.9×10^{12}	^{236}U	7.6×10^{9}
^{93m}Nb	6.0×10^{12}	^{238}U	1.1×10^{10}
^{94}Nb	3.2×10^{13}	^{237}Np	7.4×10^{9}
^{93}Mo	1.4×10^{12}	^{238}Pu	7.6×10^{13}
^{99}Tc	1.6×10^{12}	^{239}Pu	1.7×10^{13}
^{121m}Sn	2.6×10^{14}	^{240}Pu	2.4×10^{13}
^{126}Sn	1.6×10^{10}	^{241}Pu	5.8×10^{14}
^{129}I	4.3×10^{10}	^{242}Pu	7.6×10^{10}
^{135}Cs	1.2×10^{11}	^{241}Am	1.6×10^{14}
^{137}Cs	4.3×10^{15}	^{242m}Am	1.0×10^{11}
^{151}Sm	5.3×10^{13}	^{243}Am	3.6×10^{11}
^{154}Eu	3.1×10^{13}	^{243}Cm	9.2×10^{10}
^{210}Pb	4.7×10^{4}	^{244}Cm	5.4×10^{12}
^{210}Po	4.5×10^{4}	^{245}Cm	1.5×10^{9}
^{226}Ra	1.5×10^{5}	^{246}Cm	3.1×10^{8}

Tab. A2.1.7: Total safety-relevant radionuclide inventory of the ILW-2 disposal tunnel, after 40 years decay

Data taken from McGinnes (2002). A dash means activity is less than 1 Bq.

Radionuclide	Activity [Bq]	Radionuclide	Activity [Bq]
^{3}H	4.2×10^{10}	^{227}Ac	4.2×10^{4}
$^{14}C_{inorg}$	0	^{228}Th	-
$^{14}C_{org}$	8.0×10^{8}	^{229}Th	-
^{36}Cl	1.6×10^{7}	^{230}Th	2.3×10^{6}
^{60}Co	2.0×10^{10}	^{231}Pa	9.6×10^{4}
^{59}Ni	6.9×10^{8}	^{232}U	-
^{63}Ni	6.9×10^{10}	^{233}U	7.7×10^{4}
^{79}Se	9.2×10^{6}	^{234}U	6.5×10^{9}
^{90}Sr	7.7×10^{12}	^{235}U	1.1×10^{8}
^{93}Zr	1.0×10^{11}	^{236}U	1.2×10^{9}
^{93m}Nb	8.4×10^{10}	^{238}U	1.5×10^{9}
^{94}Nb	6.5×10^{6}	^{237}Np	5.0×10^{8}
^{93}Mo	2.7×10^{6}	^{238}Pu	3.8×10^{12}
^{99}Tc	3.8×10^{9}	^{239}Pu	6.5×10^{11}
^{121m}Sn	4.2×10^{8}	^{240}Pu	7.7×10^{11}
^{126}Sn	3.0×10^{9}	^{241}Pu	3.1×10^{13}
^{129}I	3.8×10^{9}	^{242}Pu	4.2×10^{9}
^{135}Cs	3.0×10^{8}	^{241}Am	6.9×10^{12}
^{137}Cs	3.2×10^{13}	^{242m}Am	1.5×10^{10}
^{151}Sm	2.6×10^{11}	^{243}Am	3.7×10^{10}
^{154}Eu	2.4×10^{11}	^{243}Cm	1.6×10^{9}
^{210}Pb	6.1×10^{3}	^{244}Cm	1.0×10^{11}
^{210}Po	6.1×10^{3}	^{245}Cm	3.7×10^{7}
^{226}Ra	2.0×10^{4}	^{246}Cm	6.5×10^{6}

A2.2 Other reference data for assessment calculations for SF, HLW and ILW

Tab. A2.2.1: IRF values of key radionuclides for BWR and PWR UO_2 fuel and PWR MOX fuel

In all cases, the IRF is applied to the radionuclide inventory present in the fuel matrix. In the case of ^{14}C, there is an additional IRF of 20 % assumed for the inventory of the cladding. Data taken from Johnson & McGinnes (2002).

Nuclide	t ½ [a]	IRF Value [%]			
		BWR UO_2 Fuel (48 GWd/t_{IHM})	**PWR UO_2 Fuel (48 GWd/t_{IHM})**	**PWR UO_2 Fuel (75 GWd/t_{IHM})**	**PWR MOX Fuel (48 GWd/t_{IHM})**
^{3}H*	1.23×10^{1}	1	1	1	1
^{10}Be	1.6×10^{6}	10	10	10	10
^{14}C	5.73×10^{3}	10	10	10	10
^{36}Cl	3.0×10^{5}	13	10	25	15
^{79}Se	1.1×10^{6}	9	4	25	15
^{90}Sr	2.86×10^{1}	1	1	1	1
^{99}Tc	2.1×10^{5}	2	2	17	2
^{107}Pd	6.5×10^{6}	2	2	17	2
^{126}Sn	2.3×10^{5}	9	4	25	15
^{129}I	1.57×10^{7}	9	4	25	15
^{135}Cs	2.3×10^{6}	5	4	25	10
^{137}Cs	3.02×10^{1}	5	4	25	10

* Assessment calculations were performed using an IRF of 2 %.

Tab. A2.2.2: Fractional dissolution rates of spent UO_2 and MOX fuel

Data taken from Johnson & Smith (2000).

Time [a]	**UO_2** **48 GWd/t_{IHM}** **[a^{-1}]**	**MOX** **48 GWd/t_{IHM}** **[a^{-1}]**	**MOX** **65 GWd/t_{IHM}** **[a^{-1}]**	**3 UO_2 + 1 MOX** **(48 GWd/t_{IHM})** **[a^{-1}]**
1.0×10^{3}	2.4×10^{-6}	9.1×10^{-6}	8.6×10^{-6}	3.8×10^{-6}
2.0×10^{3}	1.3×10^{-6}	4.6×10^{-6}	4.4×10^{-6}	2.0×10^{-6}
3.0×10^{3}	8.9×10^{-7}	3.2×10^{-6}	3.1×10^{-6}	1.4×10^{-6}
5.1×10^{3}	6.7×10^{-7}	2.3×10^{-6}	2.3×10^{-6}	1.0×10^{-6}
7.6×10^{3}	5.9×10^{-7}	2.0×10^{-6}	1.9×10^{-6}	8.9×10^{-7}
1.0×10^{4}	5.3×10^{-7}	1.8×10^{-6}	1.7×10^{-6}	7.9×10^{-7}
1.5×10^{4}	4.2×10^{-7}	1.3×10^{-6}	1.3×10^{-6}	6.1×10^{-7}
2.0×10^{4}	3.3×10^{-7}	9.7×10^{-7}	9.3×10^{-7}	4.7×10^{-7}
3.0×10^{4}	2.1×10^{-7}	5.6×10^{-7}	5.3×10^{-7}	2.8×10^{-7}
5.1×10^{4}	1.1×10^{-7}	2.5×10^{-7}	2.3×10^{-7}	1.4×10^{-7}
6.4×10^{4}	7.6×10^{-8}	1.8×10^{-7}	1.7×10^{-7}	9.8×10^{-8}
8.1×10^{4}	5.6×10^{-8}	1.4×10^{-7}	1.3×10^{-7}	7.3×10^{-8}
1.1×10^{5}	4.0×10^{-8}	1.1×10^{-7}	1.0×10^{-7}	5.4×10^{-8}
1.5×10^{5}	3.0×10^{-8}	8.4×10^{-8}	8.3×10^{-8}	4.2×10^{-8}
2.1×10^{5}	2.7×10^{-8}	7.5×10^{-8}	7.4×10^{-8}	3.7×10^{-8}
3.1×10^{5}	2.5×10^{-8}	6.7×10^{-8}	6.6×10^{-8}	3.4×10^{-8}
5.2×10^{5}	2.2×10^{-8}	5.7×10^{-8}	5.5×10^{-8}	2.9×10^{-8}
8.2×10^{5}	1.8×10^{-8}	4.7×10^{-8}	4.5×10^{-8}	2.4×10^{-8}
1.0×10^{6}	1.6×10^{-8}	4.2×10^{-8}	4.0×10^{-8}	2.1×10^{-8}
1.6×10^{6}	1.3×10^{-8}	3.4×10^{-8}	3.2×10^{-8}	1.8×10^{-8}
2.1×10^{6}	1.2×10^{-8}	2.8×10^{-8}	2.7×10^{-8}	1.6×10^{-8}
3.1×10^{6}	1.1×10^{-8}	2.2×10^{-8}	2.1×10^{-8}	1.3×10^{-8}
4.9×10^{6}	1.0×10^{-8}	1.6×10^{-8}	1.5×10^{-8}	1.1×10^{-8}
$>1.0 \times 10^{7}$	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}

Tab. A2.2.3: Radionuclide release parameters for fuel assembly structural materials

Data taken from McGinnes (2002).

Radionuclide	**IRF**	**Release rate** **[a^{-1}]**
^{14}C	20 %	3×10^{-5}
All other nuclides	0	3×10^{-5}

Tab. A2.2.4: Other SF near field parameter values

Parameter	**Reference value**	**Variations**	**Comments / References**
Time of canister failure	10^4 a	10^3 a	Also 1 a for initial canister defect scenario and 10^5 a for Cu canister case (Johnson & King 2003)
Number of canisters, n_{BE}	2065 a) 935 BWR b) 680 PWR c) 450 mixed ($3\ UO_2$ / 1 MOX)	3580 a) 1630 BWR b) 1500 PWR c) 450 mixed	McGinnes (2002)
Canister radius r	0.525 m		Johnson & King (2003)
Canister length L	4.6 m		The canister length of 4.6 m is the weighted average of all PWR and BWR canisters, based on the numbers and types of canisters in McGinnes (2002). Canister lengths for PWR and BWR canisters are from Johnson & King (2003).
Number of pinholes	0	1	(canister defect scenario)
Pinhole area	4 mm^2	50 mm^2	Assumption
Volume for dissolution (thickness, d, of reservoir)	0.041 m		Derived from internal void volume V = 0.65 m^3 of canister loaded with spent fuel. Dissolution is assumed to occur in an annular water reservoir of thickness d, length L and radius r, with: $V = L*2\pi*(r+d/2)*d$
Inner bentonite radius	0.525 m	0.84 m	Variation value based on half-thickness, assuming clay reaching >100 °C is altered and is a poor diffusion barrier (Johnson et al. 2002)
Outer bentonite radius	1.15 m		Radius after tunnel convergence (Section 5.3.3.1)
Bentonite porosity	0.36		Porosity at saturated density of 2150 kg m^{-3} (Section 5.3.3.1)
Bentonite solid density	2760 kg m^{-3}		Nagra (1994a)

Tab. A2.3: Reference HLW near field parameter values

Parameter	**Reference value**	**Variations / Comments**
Time of canister failure	10^4 a	10^3 a (Johnson & King 2003)
Glass corrosion rate		
MW glass (WA-BNF-1)	5.5×10^{-4} kg m^{-2} a^{-1}	4.0×10^{-2} kg m^{-2} a^{-1} (Curti 2003)
SON-68 glass (WA-COG-1)	7.3×10^{-5} kg m^{-2} a^{-1}	4.0×10^{-2} kg m^{-2} a^{-1} (Curti 2003)
Number of canisters, n_{COG} n_{BNFL}	 460 270	McGinnes (2002)
Length of glass blocks, h	752 m (1.03 m for each block, 730 blocks)	McGinnes (2002)
Initial diameter of glass blocks, d_0	0.43 m	McGinnes (2002)
Glass density	2750 kg m^{-3}	McGinnes (2002)
Factor of glass surface area increase due to fracturing	15	Pessimistic assumption, adjusted upwards from value of 12.5 used in Nagra (1994a)
Volume for dissolution, V_R (thickness, d[1], of reservoir)	d = 2.5 cm 24.6 m^3 (0.037 m^3 for each block)	$V_R = h*(d_0/2+d/2)*2\pi*d$
Canister radius	0.47 m	Steag & Motor Columbus (1985)
Inner bentonite radius	0.47 m	0.81 m (half-thickness, assuming alteration)
Outer bentonite radius	1.15 m	Radius after convergence (Section 5.3.3.1)
Bentonite porosity	0.36	(Section 5.3.3.1)
Bentonite solid density	2760 kg m^{-3}	Nagra (1994a)

[1] d represents a 2.5 cm thick annular region surrounding each glass block.

Tab. A2.4: Solubility limits and associated uncertainties for the SF / HLW near field, for Reference Case (pH = 7.25, Eh = -194 mV) and for oxidising conditions ("what if?" case)

Data taken from Berner (2002).

Element	**Reducing conditions**			**Oxidising conditions / "what if?" case**
	Reference Case $[mol\ l^{-1}]$	**Lower limit (optimistic)** $[mol\ l^{-1}]$	**Upper limit (pessimistic)** $[mol\ l^{-1}]$	$[mol\ l^{-1}]$
H	high	high	high	high
Be	1×10^{-6}	1×10^{-6}	high	1×10^{-6}
C_{inorg}	3×10^{-3}	6×10^{-4}	7×10^{-3}	3×10^{-3}
C_{org}	high	high	high	high
Cl	high	high	high	high
Ca	1×10^{-2}	1×10^{-2}	1×10^{-2}	1×10^{-2}
Ni	3×10^{-5}	1×10^{-5}	8×10^{-5}	3×10^{-5}
Se	5×10^{-9}	2×10^{-11}	1×10^{-5}	high
Sr	2×10^{-5}	3×10^{-6}	1×10^{-4}	2×10^{-5}
Zr	2×10^{-9}	3×10^{-11}	2×10^{-9}	2×10^{-9}
Nb	3×10^{-5}	1×10^{-8}	1×10^{-4}	3×10^{-5}
Mo	1×10^{-6}	1×10^{-6}	1×10^{-5}	1×10^{-6}
Tc	4×10^{-9}	1×10^{-9}	1×10^{-8}	high
Pd	5×10^{-8}	1×10^{-10}	2×10^{-7}	5×10^{-8}
Ag	3×10^{-6}	1×10^{-10}	3×10^{-6}	3×10^{-6}
Sn	1×10^{-8}	5×10^{-9}	1×10^{-7}	1×10^{-8}
I	high	high	high	high
Cs	high	high	high	high
Sm	5×10^{-7}	3×10^{-7}	9×10^{-7}	5×10^{-7}
Ho	5×10^{-7}	3×10^{-7}	9×10^{-7}	5×10^{-7}
Pb	2×10^{-6}	2×10^{-8}	8×10^{-5}	2×10^{-6}
Po	high	high	high	high
Ra	2×10^{-11}	4×10^{-12}	5×10^{-8}	2×10^{-11}
Ac	1×10^{-6}	5×10^{-8}	3×10^{-5}	1×10^{-6}
Th	7×10^{-7}	2×10^{-7}	3×10^{-6}	7×10^{-7}
Pa	1×10^{-8}	1×10^{-8}	1×10^{-5}	1×10^{-5}
U	3×10^{-9}	3×10^{-10}	5×10^{-7}	3×10^{-4}
Np	5×10^{-9}	3×10^{-9}	1×10^{-8}	1×10^{-5}
Pu	5×10^{-8}	3×10^{-9}	1×10^{-6}	3×10^{-8}
Am	1×10^{-6}	5×10^{-8}	3×10^{-5}	1×10^{-6}
Cm	1×10^{-6}	5×10^{-8}	3×10^{-5}	1×10^{-6}

Tab. A2.5: Solubility limits and associated uncertainties for the cementitious near field of the waste groups ILW-1 and ILW-2

Data taken from Berner (2003).

Element	ILW-1			ILW-2 [1]		
	Ref. Case [mol l^{-1}]	Lower limit (optimistic) [mol l^{-1}]	Upper limit (pessimistic) [mol l^{-1}]	Ref. case [mol l^{-1}]	Lower limit (optimistic) [mol l^{-1}]	Upper limit (pessimistic) [mol l^{-1}]
H	high	high	high	high	high	high
C_{inorg}	2×10^{-4}	1×10^{-4}	4×10^{-4}	2×10^{-4}	1×10^{-4}	4×10^{-4}
C_{org}	high	high	high	high	high	high
Cl	high	high	high	high	high	high
Co	7×10^{-7}	7×10^{-7}	7×10^{-6}	high	high	high
Ni	3×10^{-7}	1×10^{-8}	8×10^{-6}	high	high	high
Se	1×10^{-5}	7×10^{-6}	7×10^{-4}	high	high	high
Sr	3×10^{-3}	2×10^{-3}	6×10^{-3}	3×10^{-3}	2×10^{-3}	6×10^{-3}
Zr	6×10^{-6}	6×10^{-7}	6×10^{-5}	6×10^{-6}	6×10^{-7}	6×10^{-5}
Nb	high	high	high	high	high	high
Mo	3×10^{-5}	3×10^{-6}	2×10^{-3}	3×10^{-5}	3×10^{-6}	2×10^{-3}
Tc	high	3×10^{-7}	high	high	high	high
Sn	1×10^{-7}	1×10^{-7}	8×10^{-6}	1×10^{-7}	1×10^{-7}	8×10^{-6}
I	high	high	high	high	high	high
Cs	high	high	high	high	high	high
Sm	2×10^{-6}	2×10^{-7}	2×10^{-5}	2×10^{-6}	2×10^{-7}	2×10^{-5}
Eu	2×10^{-6}	2×10^{-7}	2×10^{-5}	2×10^{-6}	2×10^{-7}	2×10^{-5}
Pb	3×10^{-3}	3×10^{-3}	high	3×10^{-3}	3×10^{-3}	high
Po	high	high	high	high	high	high
Ra	1×10^{-5}	1×10^{-6}	2×10^{-5}	1×10^{-5}	1×10^{-6}	2×10^{-5}
Ac	2×10^{-6}	2×10^{-7}	2×10^{-5}	2×10^{-6}	2×10^{-7}	2×10^{-5}
Th	3×10^{-9}	8×10^{-10}	1×10^{-8}	3×10^{-9}	8×10^{-10}	1×10^{-8}
Pa	1×10^{-8}	1×10^{-8}	high	1×10^{-8}	1×10^{-8}	high
U	1×10^{-8}	1×10^{-8}	5×10^{-7}	1×10^{-8}	1×10^{-8}	5×10^{-7}
Np	5×10^{-9}	3×10^{-9}	1×10^{-8}	high	high	high
Pu	4×10^{-11}	1×10^{-11}	1×10^{-10}	6×10^{-11}	2×10^{-11}	6×10^{-10}
Am	2×10^{-9}	3×10^{-10}	1×10^{-8}	2×10^{-9}	3×10^{-10}	1×10^{-8}
Cm	2×10^{-9}	3×10^{-10}	1×10^{-8}	2×10^{-9}	3×10^{-10}	1×10^{-8}

[1] The derived ILW-2 values may also be used as conservative estimates for a "what if?" case that assumes oxidising conditions in ILW-1.

Tab. A2.6: Sorption values (K_d), effective diffusion coefficients (D_e) and accessible porosities (ε) in compacted bentonite: Reference Case (pH = 7.25, Eh = -194 mV) including lower (pessimistic) and upper (optimistic) limits, and "what if?" case for oxidising conditions

Data taken from Bradbury & Baeyens (2003a).

Element	Reducing conditions					Oxidising conditions [1]			Remarks
	K_d			D_e	ε	K_d	D_e	ε	
	Ref. case [m^3 kg^{-1}]	**Lower limit (pessimistic)** [m^3 kg^{-1}]	**Upper limit (optimistic)** [m^3 kg^{-1}]	[m^2 s^{-1}]	[-]	[m^3 kg^{-1}]	[m^2 s^{-1}]	[-]	
H	**0**	0	0	**2×10^{-10}**	**0.36**				
Be	**0.2**	0.009	5	**2×10^{-10}**	**0.36**				
C_{inorg}	**6×10^{-5}**	2×10^{-5}	3×10^{-4}	**3×10^{-12}**	**0.05**				Anion
C_{org}	**0**	0	0	**2×10^{-10}**	**0.36**				
Cl	**0**	0	0	**3×10^{-12}**	**0.05**				Anion
Ca	**0.003**	5×10^{-4}	0.02	**2×10^{-10}**	**0.36**				
Ni	**0.2**	0.009	5	**2×10^{-10}**	**0.36**				
Se	**0**	0	0	**3×10^{-12}**	**0.05**	0.06			Anion
Sr	**0.003**	5×10^{-4}	0.02	**2×10^{-10}**	**0.36**				
Zr	**80**	1	4000	**2×10^{-10}**	**0.36**				
Nb	**30**	1	900	**2×10^{-10}**	**0.36**				
Mo	**0**	0	0	**3×10^{-12}**	**0.05**				Anion
Tc	**60**	0.5	600	**2×10^{-10}**	**0.36**	0	3×10^{-12}	0.05	ox: Anion
Pd	**5**	0.2	100	**2×10^{-10}**	**0.36**				
Ag	**0**	0	0	**2×10^{-10}**	**0.36**				
Sn	**800**	1	10000	**2×10^{-10}**	**0.36**				
I [2]	**5×10^{-4}**	5×10^{-5}	0.005	**3×10^{-12}**	**0.05**				Anion
Cs	**0.1**	0.03	0.3	**2×10^{-10}**	**0.36**				
Sm	**4**	0.1	100	**2×10^{-10}**	**0.36**				
Ho	**4**	0.1	100	**2×10^{-10}**	**0.36**				
Pb	**7**	0.5	100	**2×10^{-10}**	**0.36**				
Po	**0.06**	0.008	0.5	**3×10^{-12}**	**0.05**				Anion
Ra	**0.002**	3×10^{-4}	0.01	**2×10^{-10}**	**0.36**				
Ac	**20**	1	300	**2×10^{-10}**	**0.36**				
Th	**60**	10	200	**2×10^{-10}**	**0.36**				
Pa	**5**	0.2	100	**2×10^{-10}**	**0.36**				
U	**40**	2	400	**2×10^{-10}**	**0.36**	0.01			
Np	**60**	6	600	**2×10^{-10}**	**0.36**	0.01			
Pu	**20**	1	300	**2×10^{-10}**	**0.36**	10			
Am	**20**	1	300	**2×10^{-10}**	**0.36**				
Cm	**20**	1	300	**2×10^{-10}**	**0.36**				

[1] Only given when different from reducing conditions

[2] The possibility of K_d (I) = 0 m^3 kg^{-1} is considered in a "what if?" case

Tab. A2.7: Sorption values (K_d) in cement for the waste groups ILW-1 and ILW-2: Reference Cases and corresponding lower (pessimistic) and upper (optimistic) limits

Data taken from Wieland & Van Loon (2002).

Element	K_d ILW-1			K_d ILW-2 [1]		
	Ref. Case [m³ kg⁻¹]	**Lower limit (pessimistic) [m³ kg⁻¹]**	**Upper limit (optimistic) [m³ kg⁻¹]**	**Ref. Case [m³ kg⁻¹]**	**Lower limit (pessimistic) [m³ kg⁻¹]**	**Upper limit (optimistic) [m³ kg⁻¹]**
H	**1×10^{-4}**	7×10^{-5}	1×10^{-4}	**1×10^{-4}**	7×10^{-5}	1×10^{-4}
C_{inorg}	**0**	0	0	**0**	0	0
C_{org}	**0**	0	0	**0**	0	0
Cl	**0.005**	0.003	0.007	**0.005**	0.003	0.007
Co	**0**	0	0	**0**	0	0
Ni	**0**	0	0	**0**	0	0
Se	**0.03**	0.02	0.04	**0.001**	7×10^{-4}	0.001
Sr	**0.001**	7×10^{-4}	0.001	**0.001**	7×10^{-4}	0.001
Zr	**10**	2	30	**10**	2	30
Nb	**1**	0.7	1	**1**	0.7	1
Mo	**0**	0	0	**0**	0	0
Tc	**0.001**	7×10^{-4}	1	**0.001**	7×10^{-4}	0.001
Sn	**10**	2	30	**10**	2	30
I	**0.001**	7×10^{-4}	0.001	**0.001**	7×10^{-4}	0.001
Cs	**5×10^{-4}**	3×10^{-4}	7×10^{-4}	**5×10^{-4}**	3×10^{-4}	7×10^{-4}
Sm	**80**	20	300	**80**	20	300
Eu	**80**	20	300	**80**	20	300
Pb	**0.5**	0.3	0.7	**0.5**	0.3	0.7
Po	**0**	0	0	**0**	0	0
Ra	**0.05**	0.03	0.07	**0.05**	0.03	0.07
Ac	**80**	20	300	**80**	20	300
Th	**80**	20	300	**80**	20	300
Pa	**0.1**	0.07	0.1	**0.1**	0.07	0.1
U	**2**	1	2	**2**	1	2
Np	**80**	20	300	**0.1**	0.07	0.1
Pu	**80**	20	300	**0.1**	0.07	0.1
Am	**80**	20	300	**80**	20	300
Cm	**80**	20	300	**80**	20	300

[1] The Reference Case values for ILW-2 may also be used for ILW-1 in the "what if?" case that assumes oxidising conditions for ILW-1; this is, however, considered to be a pessimistic assumption.

Tab. A2.8: Sorption values (K_d), effective diffusion coefficients ($D_{e\perp}$) and accessible porosities (ε) in Opalinus Clay

Reference Case (pH = 7.25, Eh = -167 mV) incl. lower (pessimistic) and upper (optimistic) limits for K_d and upper (pessimistic) limits for $D_{e\perp}$ (perpendicular to bedding). In addition, diffusion coefficients parallel to bedding ($D_{e=}$) are given. Data taken from Bradbury & Baeyens (2003b) and Nagra (2002a).

Element	K_d			$D_{e\perp}$		$D_{e=}$	ε	Remarks
	Ref. Case [$m^3\ kg^{-1}$]	Lower limit (pessimistic) [$m^3\ kg^{-1}$]	Upper limit (optimistic) [$m^3\ kg^{-1}$]	Ref. Case [$m^2\ s^{-1}$]	Upper limit (pessimistic) [$m^2\ s^{-1}$]	[$m^2\ s^{-1}$]	[-]	
H	**0**	0	0	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Be	**0.9**	0.03	20	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
C_{inorg}	**0.001**	1×10^{-4}	0.006	$\mathbf{1 \times 10^{-12}}$	3×10^{-12}	5×10^{-12}	**0.06**	Anion
C_{org}	**0**	0	0	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Cl	**0**	0	0	$\mathbf{1 \times 10^{-12}}$	3×10^{-12}	5×10^{-12}	**0.06**	Anion
Ca	**0.001**	1×10^{-4}	0.007	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Co	**0.4**	0.01	20	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Ni	**0.9**	0.03	20	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Se	**0**	0	0	$\mathbf{1 \times 10^{-12}}$	3×10^{-12}	5×10^{-12}	**0.06**	Anion
Sr	**0.001**	1×10^{-4}	0.007	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Zr	**10**	0.3	300	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Nb	**4**	0.1	100	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Mo	**0.01**	0.001	0.2	$\mathbf{1 \times 10^{-12}}$	3×10^{-12}	5×10^{-12}	**0.06**	Anion
Tc	**50**	0.5	500	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Pd	**5**	0.2	100	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Ag	**0**	0	0	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Sn	**100**	0.2	1000	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
I[1]	$\mathbf{3 \times 10^{-5}}$	3×10^{-6}	4×10^{-4}	$\mathbf{1 \times 10^{-12}}$	3×10^{-12}	5×10^{-12}	**0.06**	Anion
Cs	**0.5**	0.09	3	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Sm	**50**	5	600	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Eu	**50**	5	600	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Ho	**50**	5	600	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Pb	**2**	0.02	300	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Po	**0.1**	0.04	0.7	$\mathbf{1 \times 10^{-12}}$	3×10^{-12}	5×10^{-12}	**0.06**	Anion
Ra	$\mathbf{7 \times 10^{-4}}$	1×10^{-4}	0.005	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Ac	**10**	1	200	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Th	**50**	10	200	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Pa	**5**	0.2	100	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
U	**20**	0.5	200	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Np	**50**	5	500	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Pu	**20**	1	300	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Am	**10**	1	200	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	
Cm	**10**	1	200	$\mathbf{1 \times 10^{-11}}$	1×10^{-10}	5×10^{-11}	**0.12**	

[1] The possibility of K_d (I) = 0 $m^3\ kg^{-1}$ is considered in a "what if?" case

Tab. A2.9: Transport parameters in Opalinus Clay – Extract of geodataset used in the modelling of geosphere performance (the full geodataset is given in Nagra 2002a)

Parameter	Unit	Value	
		Reference value	**Alternative value**
Transport path length within host rock	[m]	40	
Hydraulic conductivity:	[m s^{-1}]		*
- perpendicular to bedding		2×10^{-14}	10^{-13}
- parallel to bedding		10^{-13}	5×10^{-13}
Hydraulic gradients (all upwards):	[m m^{-1}]		*
- mean value between Stubensandstein Fm. and Wedelsandstein Fm. at Benken		1	
- mean value between Muschelkalk and Malm aquifers at Benken		0.05	
- mean value between mid Opalinus Clay and Wedelsandstein Fm. (due to anomalous overpressures) at Benken		-	5
Diffusion accessible matrix porosity	[-]	See Table A2.8	-
Effective diffusion constant	[m^2 s^{-1}]	See Table A2.8	See Table A2.8
Exfiltration areas (for present-day conditions)			
- via Malm aquifer		Quaternary gravel south of Rhine falls	
- via Wedelsandstein		Quaternary gravel between Zurzach and Mellikon	
- via Sandsteinkeuper		Quaternary gravel in Klettgau	
- via Muschelkalk aquifer		Quaternary gravel near confluence of Aare and Rhine	

* The hydraulic conductivity, the hydraulic gradient (natural formation pressures) and the resulting water flux are interrelated. Therefore, pessimistic values of these parameters should not be combined: i. e. large gradients are only possible in combination with low hydraulic conductivities, otherwise overpressure would rapidly disappear. The overpressures measured at Benken cannot be reconciled with hydraulic conductivities > 10^{-13} m s^{-1} (Section 4.2.5).

Tab. A2.10: Dose coefficients for inhalation and ingestion

Data taken from StSV (1994) and ICRP (1996).

Nuclide name	Dose coefficient for inhalation [Sv Bq^{-1}]	Dose coefficient for ingestion [Sv Bq^{-1}]	Daughters included in the dose coefficients (neglected daughters in brackets)
Tritium	2.6×10^{-10}	4.2×10^{-11}	none
^{10}Be	3.5×10^{-8}	1.1×10^{-9}	none
^{14}C	5.8×10^{-9}	5.8×10^{-10}	none
^{36}Cl	7.3×10^{-9}	9.3×10^{-10}	none
^{41}Ca	1.8×10^{-10}	1.9×10^{-10}	none
^{60}Co	3.1×10^{-8}	3.4×10^{-9}	none
^{59}Ni	4.4×10^{-10}	6.3×10^{-11}	none
^{63}Ni	1.3×10^{-9}	1.5×10^{-10}	none
^{79}Se	6.8×10^{-9}	2.9×10^{-9}	none
^{90}Sr	1.6×10^{-7}	3.1×10^{-8}	^{90}Y
^{93}Zr	2.5×10^{-8}	1.1×10^{-9}	none
^{93m}Nb	1.8×10^{-9}	1.2×10^{-10}	none
^{94}Nb	4.9×10^{-8}	1.7×10^{-9}	none
^{93}Mo	2.3×10^{-9}	3.1×10^{-9}	none
^{99}Tc	1.3×10^{-8}	6.4×10^{-10}	none
^{107}Pd	5.9×10^{-10}	3.7×10^{-11}	none
^{108m}Ag	3.7×10^{-8}	2.3×10^{-9}	none (^{108}Ag)
^{121m}Sn	4.7×10^{-9}	5.6×10^{-10}	^{121}Sn
^{126}Sn	2.8×10^{-8}	5.1×10^{-9}	^{126m}Sb (^{126}Sb)
^{129}I	3.6×10^{-8}	1.1×10^{-7}	none
^{135}Cs	8.6×10^{-9}	2.0×10^{-9}	none
^{137}Cs	3.9×10^{-8}	1.3×10^{-8}	none (^{137m}Ba)
^{151}Sm	4.0×10^{-9}	9.8×10^{-11}	none
^{154}Eu	5.3×10^{-8}	2.0×10^{-9}	none
^{166m}Ho	1.2×10^{-7}	2.0×10^{-9}	none

Tab. A2.10: (Cont.)

Nuclide name	**Dose coefficient for inhalation [Sv Bq^{-1}]**	**Dose coefficient for ingestion [Sv Bq^{-1}]**	**Daughters included in the dose coefficients (neglected daughters in brackets)**
^{210}Pb	5.7×10^{-6}	6.9×10^{-7}	^{210}Bi (^{206}Hg, ^{206}Tl)
^{210}Po	4.3×10^{-6}	1.2×10^{-6}	none
^{226}Ra	9.5×10^{-6}	2.8×10^{-7}	^{214}Pb, ^{214}Bi (^{218}Po, ^{218}At, ^{214}Po, ^{210}Tl)
^{228}Ra	1.6×10^{-5}	6.9×10^{-7}	^{228}Ac
^{227}Ac	5.7×10^{-4}	1.2×10^{-6}	^{227}Th, ^{223}Ra, ^{215}Po (^{219}At, ^{215}Bi, ^{211}Pb, ^{211}Bi, ^{207}Tl, ^{211}Po)
^{228}Th	4.4×10^{-5}	1.4×10^{-7}	^{224}Ra, ^{212}Pb, ^{212}Bi (^{216}Po, ^{208}Tl, ^{212}Po)
^{229}Th	2.6×10^{-4}	6.1×10^{-7}	^{225}Ra, ^{225}Ac, ^{213}Bi (^{221}Fr, ^{217}At, ^{213}Po, ^{209}Tl, ^{209}Pb)
^{230}Th	1.0×10^{-4}	2.1×10^{-7}	none
^{232}Th	1.1×10^{-4}	2.3×10^{-7}	none
^{231}Pa	1.4×10^{-4}	7.1×10^{-7}	none
^{232}U	3.7×10^{-5}	3.3×10^{-7}	none
^{233}U	9.6×10^{-6}	5.1×10^{-8}	none
^{234}U	9.4×10^{-6}	4.9×10^{-8}	none
^{235}U	8.5×10^{-6}	4.7×10^{-8}	^{231}Th
^{236}U	8.7×10^{-6}	4.7×10^{-8}	none
^{238}U	8.0×10^{-6}	4.8×10^{-8}	^{234}Th (^{234}Pa, ^{234m}Pa)
^{237}Np	5.0×10^{-5}	1.1×10^{-7}	^{233}Pa
^{238}Pu	1.1×10^{-4}	2.3×10^{-7}	none
^{239}Pu	1.2×10^{-4}	2.5×10^{-7}	none
^{240}Pu	1.2×10^{-4}	2.5×10^{-7}	none
^{241}Pu	2.3×10^{-6}	4.8×10^{-9}	none (^{237}U)
^{242}Pu	1.1×10^{-4}	2.4×10^{-7}	none
^{241}Am	9.6×10^{-5}	2.0×10^{-7}	none
^{242m}Am	9.7×10^{-5}	2.0×10^{-7}	^{238}Np, ^{242}Am, ^{242}Cm
^{243}Am	9.6×10^{-5}	2.0×10^{-7}	^{239}Np
^{243}Cm	6.9×10^{-5}	1.5×10^{-7}	none
^{244}Cm	5.7×10^{-5}	1.2×10^{-7}	none
^{245}Cm	9.9×10^{-5}	2.1×10^{-7}	none
^{246}Cm	9.8×10^{-5}	2.1×10^{-7}	none

Tab. A2.11: Biosphere dose conversion factors (BDCFs)

The BDCF for a given radionuclide is the ratio of the corresponding steady-state annual dose [Sv a^{-1}] to the input value of 1 Bq a^{-1}. For each radionuclide, a constant flux of 1 Bq a^{-1} is input into the transient compartment model, with the Reference Case conceptual assumptions and parameters, and the model is run until a steady state is reached (see Chapter 6).

Nuclide name	**Biosphere dose conversion factor [Sv Bq^{-1}]**	**Nuclide name**	**Biosphere dose conversion factor [Sv Bq^{-1}]**
^{3}H	1.68×10^{-16}	^{226}Ra	4.20×10^{-13}
^{10}Be	4.82×10^{-16}	^{227}Ac	1.49×10^{-15}
^{14}C	4.48×10^{-15}	^{228}Ra	8.03×10^{-16}
^{36}Cl	9.86×10^{-15}	^{228}Th	4.54×10^{-18}
^{41}Ca	4.33×10^{-16}	^{229}Th	8.10×10^{-14}
^{59}Ni	1.62×10^{-16}	^{230}Th	5.14×10^{-12}
^{60}Co	3.41×10^{-17}	^{231}Pa	1.72×10^{-11}
^{63}Ni	6.18×10^{-18}	^{232}Th	4.60×10^{-12}
^{79}Se	1.89×10^{-14}	^{232}U	4.78×10^{-15}
^{90}Sr	4.00×10^{-15}	^{233}U	8.14×10^{-14}
^{93}Mo	3.74×10^{-15}	^{234}U	2.56×10^{-13}
^{93m}Nb	1.42×10^{-18}	^{235}U	2.84×10^{-12}
^{93}Zr	9.41×10^{-15}	^{236}U	3.52×10^{-14}
^{94}Nb	2.00×10^{-13}	^{237}Np	3.03×10^{-13}
^{99}Tc	6.75×10^{-14}	^{238}Pu	3.52×10^{-16}
^{107}Pd	9.96×10^{-17}	^{238}U	3.98×10^{-14}
^{108m}Ag	1.20×10^{-14}	^{239}Pu	5.23×10^{-14}
^{121m}Sn	2.07×10^{-17}	^{240}Pu	2.13×10^{-14}
^{126}Sn	4.37×10^{-14}	^{241}Am	1.24×10^{-15}
^{129}I	2.48×10^{-13}	^{241}Pu	4.23×10^{-17}
^{135}Cs	3.64×10^{-14}	^{242m}Am	4.96×10^{-16}
^{137}Cs	1.94×10^{-16}	^{242}Pu	9.04×10^{-14}
^{151}Sm	7.65×10^{-19}	^{243}Am	3.56×10^{-14}
^{154}Eu	4.34×10^{-18}	^{243}Cm	1.34×10^{-16}
^{166m}Ho	1.72×10^{-14}	^{244}Cm	9.38×10^{-17}
^{210}Pb	3.82×10^{-15}	^{245}Cm	5.70×10^{-14}
^{210}Po	9.12×10^{-18}	^{246}Cm	1.77×10^{-14}

Tab. A2.12: Biosphere parameters

Biosphere model area	**Eroding river (reference biosphere area)**			**Sedimen-tation area**	**Wetland**
Climate	Present-day	Wet	Dry	Present-day	Present-day
Surface area [m^2]	2.3×10^6	2.3×10^6	2.3×10^6	2.3×10^6	1.0×10^6
Compartment depths [m]:					
Top soil	0.25	0.25	0.25	0.25	0.25
Deep soil	2.0	2.0	2.0	2.0	2.0
Quaternary aquifer	20	20	20	20	20
Rainfall [m a^{-1}]	1.0	2.0	0.5	1.0	1.0
Evapotranspiration rate [m a^{-1}]	0.6	1.0	1.0	0.6	0.6
Capillary rise [m a^{-1}]	0	0	0	0.1	0
Irrigation from Quaternary aquifer [m a^{-1}]	0.5	0.25	0.6	0.25	0
Flooding/irrigation from river [m a^{-1}]	0	0	0	0.5	1
Erosion rate [mm a^{-1}]	0.1	0.1	0.1	0.6	0
Sedimentation rate [mm a^{-1}]	0	0	0	0.65	0.05
Net erosion rate [mm a^{-1}]	0.1	0.1	0.1	-0.05	-0.05
Water fluxes [m^3 a^{-1}]:					
Uncontaminated inflow to aquifer (from upstream and valley sides)	1.5×10^6	1.8×10^6	1.3×10^6	1.3×10^6	1.3×10^6
Through flow in River	1.2×10^{10}	2.4×10^{10}	6.0×10^9	5.0×10^7	5.0×10^7

Tab. A2.13: Probability distribution functions (PDFs) for probabilistic calculations

For some parameters, correlations between waste types/radionuclides/species (anions, non-anions) have been considered (for details see Nagra 2002c). "What if?" values are included to allow an easy comparison.

SF/HLW Near field	**Parameter values**		
Model parameter	**Reference Case values**	**PDF attributes**	**"What if?" values**
Cladding dissolution rate	3×10^{-5} a^{-1}	**Lognormal** median = Reference Case value, σ(log) = 0.35 upper truncation = 1.5×10^{-4} a^{-1} (2 σ) lower truncation = 6×10^{-6} a^{-1} (2 σ)	3×10^{-4} a^{-1}
IRF	Table A2.2.1 (weighted average)	**Log-uniform** lower cut-off - values for 48 GWd/t_{IHM} PWR fuel (best estimate); upper cut-off - values for 75 GWd/t_{IHM} PWR fuel	-
SF matrix dissolution rate	Table A2.2.2 (UO_2 48 GWd/t_{IHM})	**Log-uniform** maximum = reference rate (t), minimum = solubility limited rate (t)	10 × and 100 × Reference Case value
HLW dissolution rate	Table A2.3	**Triangular** maximum = pessimistic variation values (Table A2.3) minimum = 0.05 × reference value	See highly pessimistic variations (Table A2.3)
Radionuclide solubilities	Table A2.4	**Discrete** Probability (P) = 0.7 for reference value, P = 0.15 for optimistic and pessimistic values	Values for oxidising conditions in Table A2.4
K_d values	Table A2.6	**Discrete** P = 0.7 for reference value, P = 0.15 for optimistic and pessimistic values	Values for oxidising conditions in Table A2.6 "What if?" case for ^{129}I $K_d = 0$
Bentonite D_e	Table A2.6	**Log-normal** median = reference values, σ(log) = 0.3 upper truncation = 2 × reference values (1 σ), lower truncation = 0.1 × reference values (3.3 σ)	-

Tab. A2.13: (Cont.)

ILW Near field	**Parameter values**		
Model parameter	**Reference Case values**	**PDF attributes**	**"What if?" values**
Radionuclide solubilities	Table A2.5	**Discrete** Probability (P) = 0.7 for reference value, P = 0.15 for optimistic and pessimistic values	Values for oxidising conditions in Table A2.5
K_d values	Table A2.7	**Discrete** P = 0.7 for reference value, P = 0.15 for optimistic and pessimistic values	"What if?" case for ^{129}I $K_d = 0$

Opalinus Clay	**Parameter values**		
Model parameter	**Reference Case values**	**PDF attributes**	**"What if?" values**
K_d values	Table A2.8	**Discrete** P = 0.7 for reference value, P = 0.15 for optimistic and pessimistic values	"What if?" case for ^{129}I $K_d = 0$
Transport path length	40 m	**Uniform** minimum = 40 m maximum = 60 m	30 m
D_e	Table A2.8	**Log-normal** median = reference values, σ(log) = 0.24 upper truncation = 3 × reference values (2 σ), lower truncation = 0.5 × reference values (1.26 σ)	-
Darcy velocity	2×10^{-14} m s^{-1}	**Log-normal** median = reference values, σ(log) = 1 upper truncation = 2×10^{-13} m s^{-1} (1 σ), lower truncation = 1×10^{-15} m s^{-1} (1.3 σ)	100-fold increase in flow rate
Transmissivity of hypothetical transmissive discontinuity	-	-	10^{-9} and 10^{-10} m^2 s^{-1}

Appendix 3 Background to the Alternative Safety and Performance Indicators

A3.1 Definition of radiotoxicity

In order to estimate potential health effects resulting from ingestion of radionuclides, their activity A_j [Bq] as well as their dose coefficient for ingestion, F_j [Sv Bq^{-1}], need to be considered. The radiotoxicity of a given amount of radioactive material is often expressed in terms of a "radiotoxicity index" or RTI (see, e.g., Gera & Jacobs 1971, Hamstra 1975, Hedin 1997), which is here defined as the hypothetical dose resulting from ingestion of the activity A_j at a time t, divided by 10^{-4} Sv (derived from the Swiss regulatory annual dose limit):

$$RTI(t) = \Sigma A_j(t) F_j / (10^{-4} \text{ Sv}) \qquad \text{(Eq. A3-1)}$$

The RTI allows a direct comparison of the radiological hazard potential of radioactive waste or radionuclides released from the waste with that of different natural materials. A number of possible comparisons are given in Tab. A3.1.

Tab. A3.1: Possible comparisons of radioactive waste, or radionuclides released from the waste, with natural materials

Radioactivity related to the repository	**Natural material**
Radioactive waste	The uranium which was mined to produce nuclear fuel
	Uranium ores of various grades
	Opalinus Clay
Radionuclides that have migrated into the Opalinus Clay	Opalinus Clay
Radionuclides accumulated in the biosphere	Natural soil

A3.2 Calculation of the radiotoxicity of natural materials

A large number of natural radionuclides exist in the environment (see, e.g., Eisenbud & Gesell 1997, Baertschi & Keil 1992). The major radionuclides in soils and in most environmental materials causing doses to humans are ^{40}K, ^{232}Th, ^{235}U and ^{238}U. In uranium ore, however, the contributions of ^{40}K and ^{232}Th are, in general, negligible.

In the following calculations of the RTI for ^{232}Th, ^{235}U and ^{238}U, radioactive decay products are taken into account, assuming that they are in radiological equilibrium with their precursors. The corresponding dose coefficient are given in Tab. A3.2.

Although the numbers appearing in the tables of Appendix 3 are rounded to two significant digits, all calculations are performed using the full figures. This may lead to small apparent discrepancies between the results shown in the tables and those calculated using the two-digit figures.

Tab. A3.2: Dose coefficients (DCs) for ingestion for adult members of the public

DCs for ^{232}Th, ^{235}U and ^{238}U include radioactive decay products.

Nuclide	DC [Sv Bq^{-1}]
^{40}K	6.2×10^{-9}
^{232}Th	1.1×10^{-6}
^{235}U	2.0×10^{-6}
^{238}U	2.5×10^{-6}

A3.2.1 Natural uranium

It takes about eight tonnes of natural uranium to fabricate one tonne of nuclear fuel (Hedin 1997). In Tab. A3.3 the RTI of eight tonnes of natural uranium is calculated.

Tab. A3.3: Activity and toxicity concentrations of natural uranium

Nuclide	kg element per kg material	Bq precursor per kg material	RTI per kg material	RTI of 8 tonnes uranium
^{235}U	1	5.6×10^{5}	1.1×10^{4}	8.8×10^{7}
^{238}U	1	1.2×10^{7}	3.1×10^{5}	2.5×10^{9}
Total			$\mathbf{3.2 \times 10^{5}}$	$\mathbf{2.6 \times 10^{9}}$

A3.2.2 Uranium ores of various grades

The natural uranium described in Section A3.2.1 can, in principle, be obtained from uranium ores from different locations and different grades. Investigations in Switzerland have shown that there is a range of rocks with higher than average uranium concentrations. An example is the small uranium ore body of La Creusa (VS) containing up to about 3 % of uranium (Gilliéron 1988). The RTI of uranium ore from La Creusa is calculated in Tab. A3.4.

A well-known, large uranium ore body exists in Canada, with uranium grades in the range between about 2 % and 60 % (Cramer & Smellie 1994). Tab. A3.4 presents examples of ores containing 8 % (average grade) and 55 % (near upper end of range) of uranium.

Tab. A3.4: Activity and toxicity concentrations of natural radionuclides in various uranium ores

Nuclide	kg element per kg rock	Bq precursor per kg rock	RTI per kg rock	RTI per m³ rock
Uranium-containing rock, La Creusa (VS), Switzerland (density 2.4×10^3 kg m^{-3})				
^{235}U	3.0×10^{-2}	1.7×10^4	3.3×10^2	8.1×10^5
^{238}U	3.0×10^{-2}	3.7×10^5	9.3×10^3	2.3×10^7
Total			$\mathbf{9.6 \times 10^3}$	$\mathbf{2.3 \times 10^7}$
Cigar Lake uranium ore, 8 % (density 2.8×10^3 kg m^{-3})				
^{235}U	8.0×10^{-2}	4.5×10^4	8.8×10^2	2.5×10^6
^{238}U	8.0×10^{-2}	1.0×10^6	2.5×10^4	6.9×10^7
Total			$\mathbf{2.6 \times 10^4}$	$\mathbf{7.1 \times 10^7}$
Cigar Lake uranium ore, 55 % (density 5.8×10^3 kg m^{-3})				
^{235}U	5.5×10^{-1}	3.1×10^5	6.1×10^3	3.5×10^7
^{238}U	5.5×10^{-1}	6.8×10^6	1.7×10^5	9.9×10^8
Total			$\mathbf{1.8 \times 10^5}$	$\mathbf{1.0 \times 10^9}$

In a number of figures in Chapters 2 and 6, the RTI of waste emplacement tunnels hypothetically filled with natural uranium ores is given. These levels are calculated by multiplying the radiotoxicity indices per m^3 of rock as given in Tab. A3.4 by emplacement tunnel volumes. The tunnel volumes used for these calculations are based on the repository layout as described in Chapter 4 and are given in Tab. A3.5.

Tab. A3.5: Volumes assumed to calculate the RTI of waste emplacement tunnels hypothetically filled with natural uranium ores

Waste type	Tunnel volume [m³]
Spent fuel	8.6×10^4
Vitrified HLW	2.0×10^4
ILW (ILW-1 + ILW-2)	1.3×10^4

A3.2.3 Opalinus Clay

The concentrations of natural radionuclides in Opalinus Clay are given in Nagra (2001). Tab. A3.6 summarises these concentrations and gives the RTI of Opalinus Clay per kg and per m^3 of rock.

Tab. A3.6: Activity and toxicity concentrations of natural radionuclides in Opalinus Clay (Nagra 2001)

Nuclide	kg element per kg rock	Bq precursor per kg rock	RTI per kg rock	RTI per m^3 rock
Opalinus Clay (density 2.4×10^3 kg m^{-3})				
^{40}K	2.0×10^{-2}	6.2×10^2	3.8×10^{-2}	9.3×10^1
^{232}Th	1.2×10^{-5}	4.9×10^1	5.2×10^{-1}	1.3×10^3
^{235}U	2.0×10^{-6}	1.1	2.2×10^{-2}	5.3×10^1
^{238}U	2.0×10^{-6}	2.5×10^1	6.2×10^{-1}	1.5×10^3
Total			**1.2**	**2.9×10^3**

A3.2.4 Radiotoxicity of granite and syenite

The concentrations of natural radionuclides of Böttstein granite (a well-characterised, typical granite in northern Switzerland) are given in Baertschi (1995), and those of Giuv syenite (a rock in the eastern Aar Massif, south-eastern Switzerland) are given in Labhart & Rybach (1971). Tab. A3.7 summarises these concentrations and gives the RTI per kg and per m^3 of rock.

Tab. A3.7: Activity and toxicity concentrations of natural radionuclides in granite and syenite

Nuclide	kg element per kg rock	Bq precursor per kg rock	RTI per kg rock	RTI per m^3 rock
Böttstein granite (density 2.6×10^3 kg m^{-3})				
^{40}K	4.1×10^{-2}	1.3×10^3	7.9×10^{-2}	2.1×10^2
^{232}Th	2.5×10^{-5}	1.0×10^3	1.1	2.9×10^3
^{235}U	1.0×10^{-5}	5.6	1.1×10^{-1}	2.9×10^2
^{238}U	1.0×10^{-5}	1.2×10^2	3.1	8.2×10^3
Total			**4.4**	**1.2×10^4**
Giuv syenite (density 2.7×10^3 kg m^{-3})				
^{40}K	5.9×10^{-2}	1.8×10^3	1.1×10^{-1}	3.0×10^2
^{232}Th	6.6×10^{-5}	2.7×10^2	2.9	7.7×10^3
^{235}U	2.2×10^{-5}	1.2×10^1	2.4×10^{-1}	6.5×10^2
^{238}U	2.2×10^{-5}	2.7×10^2	6.8	1.8×10^4
Total			**1.0×10^1**	**2.7×10^4**

A3.2.5 Radiotoxicity of average soil

Miller et al. (2002) give the activity concentrations of natural radionuclides in an "average soil". The data given in Tab. A3.8 derived from Miller et al. (2002) are roughly comparable to those for Opalinus Clay (Tab. A3.6).

Tab. A3.8: Activity and toxicity concentrations of natural radionuclides in average soil

Nuclide	**kg element per kg soil**	**Bq precursor per kg soil**	**RTI per kg soil**	**RTI per m³ soil**
Average soil (density 1.2×10^3 kg m^{-3})				
^{40}K	1.4×10^{-2}	4.3×10^{2}	2.7×10^{-2}	3.2×10^{1}
^{232}Th	9.0×10^{-6}	3.7×10^{1}	3.9×10^{-1}	4.7×10^{2}
^{235}U	2.0×10^{-6}	1.1	2.2×10^{-2}	2.6×10^{1}
^{238}U	2.0×10^{-6}	2.5×10^{1}	6.2×10^{-1}	7.4×10^{2}
Total			**1.1**	$\mathbf{1.3 \times 10^{3}}$

A3.3 Radiotoxicity of the wastes

The RTI of the wastes is also calculated using equation (Eq. A3-1). However, the nuclide-specific activities $A_j(t)$ are in this case time-dependent due to the shorter half lives of the majority of the radionuclides present in the wastes. The dose coefficients F_j used to calculate the RTI of the wastes are given in Appendix 2, Tab. A2.10. The RTI of SF, HLW and ILW at selected times are given in Tab. A3.9. For a graphical display, see Fig. 2.5-1.

Tab. A3.9: RTI of spent fuel, vitrified HLW and ILW at selected times

Waste type	$\mathbf{10^0}$ **a**	$\mathbf{10^2}$ **a**	$\mathbf{10^4}$ **a**	$\mathbf{10^6}$ **a**	$\mathbf{10^8}$ **a**
Spent fuel	5.6×10^{15}	2.5×10^{15}	1.8×10^{14}	3.3×10^{12}	9.7×10^{11}
Vitrified HLW	6.0×10^{14}	1.2×10^{14}	9.4×10^{11}	1.3×10^{11}	3.9×10^{8}
ILW (ILW-1 + ILW-2)	1.9×10^{12}	6.4×10^{11}	5.6×10^{10}	9.1×10^{8}	3.3×10^{8}

A3.4 Definition of radiotoxicity flux

The annual radiotoxicity flux (RTF) across a given interface (units: RTI a^{-1}) is calculated by replacing $A_j(t)$ in Equation A3-1 by $A_j'(t)$, where $A_j'(t)$ [Bq a^{-1}] is the annual activity flux of nuclide j across that interface at time t:

$$RTF(t) = \Sigma A_j'(t) F_j / (10^{-4} Sv) \quad \text{(Eq. A3-2)}$$

A3.5 Calculation of natural radiotoxicity fluxes

The radiotoxicity flux can be used for a number of comparisons. Some examples are given in Tab. A3.10.

Tab. A3.10: Possible comparisons of radionuclide fluxes originating from the repository with natural radionuclide fluxes

Fluxes originating from the repository	Fluxes of natural radioactivity
Flux of radionuclides from the repository into the geosphere	Flux of natural radionuclides over the repository area due to erosion
Flux of radionuclides from the geosphere into the biosphere	Flux of natural radionuclides dissolved in groundwater within the biosphere aquifer
	Flux of natural radionuclides dissolved in groundwater within typical rivers
	Flux of natural radionuclides due to erosion

The same natural radionuclides are considered for the fluxes as for the concentrations (see Section A3.4). In the calculation of the radiotoxicity of solid materials it was assumed that all radioactive decay products of ^{232}Th, ^{235}U and ^{238}U are in radiological equilibrium with their precursors (i.e. the dose coefficients of Tab. A3.2 are used). The same is assumed to apply to radionuclides in solid material fluxes (erosion and sediments transported by rivers). However, regarding radionuclides dissolved in rivers or aquifers it is assumed that there is no radiological equilibrium. The corresponding dose coefficients are given in Tab. A3.11.

Tab. A3.11: Dose coefficients (DCs) for ingestion of radionuclides dissolved in river or aquifer water for adult members of the public

The DC for ^{235}U does not include any decay products, while the DC for ^{238}U includes that for ^{234}U and the DC for ^{232}Th includes that for ^{228}Th.

Nuclide	DC [Sv Bq^{-1}]
^{40}K	6.2×10^{-9}
^{232}Th	3.0×10^{-7}
^{235}U	4.7×10^{-8}
^{238}U	9.4×10^{-8}

A3.5.1 Rivers Rhine and Thur

There are two possibilities for transport of radionuclides in river water; i.e. radionuclides dissolved in water and radionuclides contained in sediments transported by the river. For the Rhine the water flow at Rheinfelden is assumed (Spreafico et al. 2001). The concentrations of radionuclides dissolved in water are taken from Baertschi & Keil (1992). The suspended sediment concentration in Rhine water is taken from Spreafico et al. (2001) and LHG (1999). The radionuclide concentrations in the suspended sediment are assumed to be the same as those in average soil (see Section A3.2.5).

The fluxes of natural radionuclides in the Thur have been calculated using again data from Spreafico et al. (2001) and LHG (1999). The radionuclide concentrations in the water of the Thur are assumed to be the same as those in the Rhine.

The radionuclide concentrations assumed for the Rhine and the Thur are comparable to the "average" rivers described by Miller et al. (2002) and are given in Tab. A3.12 and A3.13, respectively, together with radiotoxicity concentrations and fluxes.

Tab. A3.12: Concentrations of natural radionuclides and radiotoxicity in the river Rhine and corresponding radiotoxicity flux

Nuclide	kg element per kg water	Bq precursor per kg water	RTI per kg water	RTI per year
Natural radionuclides dissolved in Rhine water water flux at Rheinfelden: 3.2×10^{10} m^3 a^{-1}				
^{235}U	7.0×10^{-10}	3.9×10^{-4}	1.8×10^{-7}	5.9×10^{6}
^{238}U	7.0×10^{-10}	8.7×10^{-3}	8.2×10^{-6}	2.6×10^{8}
Total			$\mathbf{8.4 \times 10^{-6}}$	$\mathbf{2.7 \times 10^{8}}$
Natural radionuclides in sediments transported by Rhine water suspended sediment: 0.1 kg m^{-3}				
^{40}K	1.4×10^{-6}	4.3×10^{-2}	2.7×10^{-6}	8.6×10^{7}
^{232}Th	9.0×10^{-10}	3.7×10^{-3}	3.9×10^{-5}	1.2×10^{9}
^{235}U	2.0×10^{-10}	1.1×10^{-4}	2.2×10^{-6}	7.0×10^{7}
^{238}U	2.0×10^{-10}	2.5×10^{-3}	6.2×10^{-5}	2.0×10^{9}
Total			$\mathbf{1.1 \times 10^{-4}}$	$\mathbf{3.4 \times 10^{9}}$
Natural radionuclides (total) in Rhine water				
Total			$\mathbf{1.1 \times 10^{-4}}$	$\mathbf{3.6 \times 10^{9}}$

Tab. A3.13: Concentrations of natural radionuclides and radiotoxicity in the river Thur and corresponding radiotoxicity flux

Nuclide	kg element per kg water	Bq precursor per kg water	RTI per kg water	RTI per year
Natural radionuclides dissolved in Thur water water flux: 5.0×10^7 m^3 a^{-1}				
^{235}U	7.0×10^{-10}	3.9×10^{-4}	1.8×10^{-7}	9.2×10^3
^{238}U	7.0×10^{-10}	8.7×10^{-3}	8.2×10^{-6}	4.1×10^5
Total			$\mathbf{8.4 \times 10^{-6}}$	$\mathbf{4.2 \times 10^5}$
Natural radionuclides in sediments transported by Thur water suspended sediment: 3.5 kg m^{-3}				
^{40}K	4.9×10^{-5}	1.5	9.4×10^{-5}	4.7×10^6
^{232}Th	3.2×10^{-8}	1.3×10^{-1}	1.4×10^{-3}	6.8×10^7
^{235}U	7.0×10^{-9}	3.9×10^{-3}	7.7×10^{-5}	3.9×10^6
^{238}U	7.0×10^{-9}	8.7×10^{-2}	2.2×10^{-3}	1.1×10^8
Total			$\mathbf{3.7 \times 10^{-3}}$	$\mathbf{1.8 \times 10^8}$
Natural radionuclides (total) in Thur water				
Total			$\mathbf{3.7 \times 10^{-3}}$	$\mathbf{1.8 \times 10^8}$

A3.5.2 Biosphere aquifer

In the reference-case biosphere the local aquifer is a compartment in which major dilution of radionuclides occurs. The flux of natural radionuclides dissolved in this aquifer can be compared with the flux of radionuclides released from the repository into the aquifer. The water flux is assumed to be 1.0×10^6 m^3 a^{-1} which is a typical value for surface aquifers in northern Switzerland along the Rhine and its tributary valleys. The concentrations of natural radionuclides in groundwater used for drinking water are taken from Baertschi & Keil (1992). The concentration of solid material (mainly colloids) in groundwater is neglected in the estimation of the radionuclide fluxes. The relevant data for the biosphere aquifer are summarised in Tab. A3.14.

Tab. A3.14: Concentrations of natural radionuclides and radiotoxicity of water in surface aquifers in northern Switzerland and corresponding radiotoxicity flux

Nuclide	kg element per kg water	Bq precursor per kg water	RTI per kg water	RTI per year
Natural radionuclides in a surface aquifer water flux: 1.0×10^6 m^3 a^{-1}				
^{235}U	6.9×10^{-10}	3.9×10^{-4}	1.8×10^{-7}	1.8×10^2
^{238}U	6.9×10^{-10}	8.6×10^{-3}	8.1×10^{-6}	8.1×10^3
Total			$\mathbf{8.3 \times 10^{-6}}$	$\mathbf{8.3 \times 10^3}$

A3.5.3 Annual consumption of mineral water

Mineral waters contain natural radionuclides at different levels (Bosshard et al. 1992, Baertschi 1995). In Switzerland, the annual consumption of mineral water in the year 2001 was about 7.7×10^5 m^3 a^{-1} (SMS 2002). The concentrations of natural radionuclides and the radiotoxicity of average Swiss mineral water and the radiotoxicity flux corresponding to the above annual consumption is given in Tab. A3.15.

Tab. A3.15: Concentrations of natural radionuclides and radiotoxicity of average Swiss mineral water and the radiotoxicity flux corresponding to an annual production of 7.7×10^5 m^3

Nuclide	**kg element per kg water**	**Bq precursor per kg water**	**RTI per kg water**	**RTI per year**
Natural radionuclides in average Swiss mineral water consumption rate: 7.7×10^5 m^3 a^{-1}				
^{40}K	3.0×10^{-6}	9.3×10^{-2}	5.8×10^{-6}	4.4×10^{3}
^{232}Th	2.5×10^{-10}	1.0×10^{-3}	3.1×10^{-6}	2.4×10^{3}
^{235}U	1.2×10^{-8}	6.7×10^{-3}	3.2×10^{-6}	2.4×10^{3}
^{238}U	1.2×10^{-8}	1.5×10^{-1}	1.4×10^{-4}	1.1×10^{5}
Total			$\mathbf{1.5 \times 10^{-4}}$	$\mathbf{1.2 \times 10^{5}}$

A3.5.4 Erosion of the biosphere model area

In the Reference Case a biosphere model area of 2.3 km^2 and an erosion rate of 0.27 kg m^{-2} a^{-1} (uplift rate of 0.1 mm a^{-1}) are assumed (Nagra 2002c), resulting in an erosion rate of solid material of 6.1 × 10^5 kg a^{-1}. The radionuclide concentrations in soil are assumed to be the same as those presented in Section A3.2.5. The concentrations of natural radionuclides and radiotoxicity in soil are given in Tab. A3.16, as well as the radiotoxicity flux due to erosion.

Tab. A3.16: Concentrations of natural radionuclides and radiotoxicity in soil and radiotoxicity flux due to erosion of soil

Nuclide	**kg element per kg soil**	**Bq precursor per kg soil**	**RTI per kg soil**	**RTI per year**
Average soil erosion rate: 0.27 kg m^{-2} a^{-1}, biosphere area: 2.3 km^2				
^{40}K	1.4 × 10^{-2}	4.3 × 10^2	2.7 × 10^{-2}	1.7 × 10^4
^{232}Th	9.0 × 10^{-6}	3.7 × 10^1	3.9 × 10^{-1}	2.4 × 10^5
^{235}U	2.0 × 10^{-6}	1.1	2.2 × 10^{-2}	1.4 × 10^4
^{238}U	2.0 × 10^{-6}	2.5 × 10^1	6.2 × 10^{-1}	3.8 × 10^5
Total			**1.1**	**6.5 × 10^5**

A3.6 Calculation of the radiotoxicity flux into the biosphere aquifer originating from the repository

The radiotoxicity flux into the biosphere aquifer originating from the repository is also calculated using Equation A3-2. The activity fluxes $A_j'(t)$ are those across the geosphere – biosphere interface for the Reference Case. The dose coefficients F_j are given in Appendix 2, Tab. A2.10.

Appendix 4 Role and Interaction of Different Groups in Developing the Safety Case

A4.1 Groups of personnel contributing to the safety case

Personnel involved in the planning of the SF / HLW / ILW repository have a number of distinct roles in developing the safety case for a repository, and may participate in one or more of the following groups:

- The management group, responsible for overall management of repository planning.
- The science and technology group, responsible for the scientific and technical basis of the safety case.
- The safety assessment group, responsible for the development of a system concept, the safety concept and the approach to safety assessment, as well as the carrying out of the safety assessment and the compilation of the safety case.
- The bias audit group, responsible for ensuring that the scientific basis for safety assessment is complete, fully documented and exploited in an unbiased manner in safety assessment.

There are also personnel whose role it is to review the safety assessment as a whole, and to review individual elements of the project. Ideally, there should be little or no overlap in personnel between these groups. In practice, this is not possible in a small programme, thus in many cases, individuals may participate in the work of more than one group. Nonetheless, the roles of the groups are clearly explained to all participants, so that the appropriate perspective is maintained by each group.

Interaction between these groups must take place, to keep work focused on project goals, to ensure that the knowledge and expertise of available personnel are fully exploited, and to ensure that all personnel understand and support the safety case arguments. At the same time, a degree of independence must be maintained, to avoid unduly biasing the work of one group by the expectations of another. The science and technology group, for example, should be aware of the possibility of new phenomena, even though these may not fit conveniently into existing methods or models used by the safety assessment group.

The roles of the groups, and the types of interactions between them, are summarised in Tab. A4.1, and discussed in more detail in the following sections.

A4.2 Contributions of the different groups to the safety case

A4.2.1 Overview

Fig. A4.1 shows the safety assessment process, and indicates which items or tasks are the responsibility of each of the different groups.

Tab. A4.1: Roles and interaction of different groups

	Management	Science and technology	Safety assessment	Bias audit
Management	Overall management of project	Keep work of groups focused on project goals; maintain overview of role/requirements of different groups; be responsive and accessible		
Science and technology	Keep management informed (incl. any problem areas that need resources focussed on them)	Develop and evaluate scientific basis for safety assessment	Provide safety assessment team with understanding required to develop system concept and tools for analysis; revisit key assumptions	Provide bias audit team with comprehensive view of system understanding (including limitations)
Safety assessment	(as above)	Ensure that science and technology is aware of activities and information requirements of safety assessment team	Formulate and analyse safety assessment cases; assess "independent evidence"; compile safety case	Provide bias audit team with description of cases and concepts and tools used to analyse them
Bias audit	(as above)	Ensure that the scientific basis for safety assessment is adequately documented	Ensure that appropriate use is made by safety assessment of available scientific understanding	Ensure that the scientific basis for safety assessment is complete, adequately documented and exploited in safety assessment

Note: Boxes along the top left to bottom right diagonal summarise the overall roles of the groups. Other boxes summarise the responsibilities of a group in a particular row towards a group in a particular column.

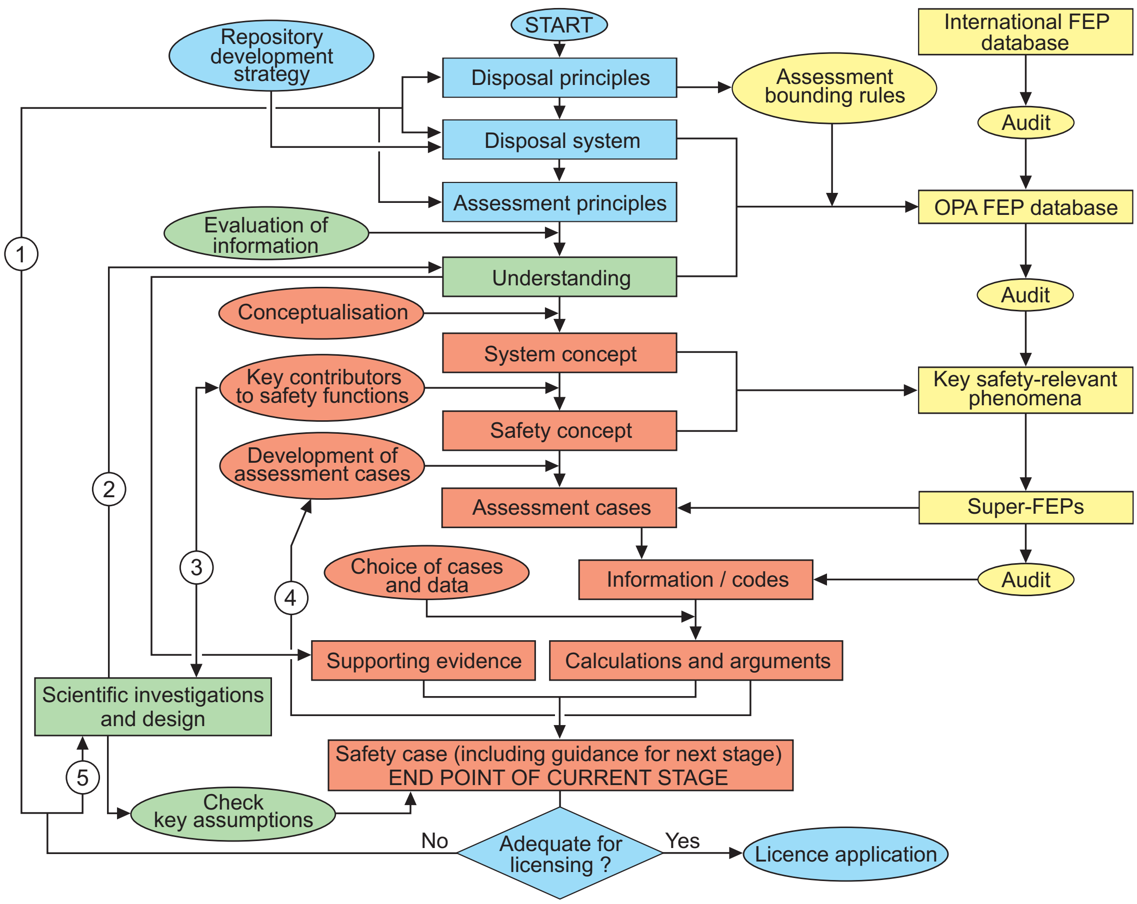

Fig. A4.1: The procedure for constructing the safety case, with the roles of different groups indicated by colours (see text)

A4.2.2 The management group

The role of the management group is indicated in blue in Tab. A4.1 and in Fig. A4.1. The group takes overall responsibility for the selection of a suitable site and system of engineered barriers, and for the development of arguments that will support the eventual licensing of construction, operation and closure of the facility. The group is responsible for keeping the work of the other groups focussed on the project goals and allocating resources to these groups based on an overview of their roles and requirements. It must maintain flexibility in the strategy for repository planning and development, in part so that it can react to findings regarding the significance of any uncertainties or outstanding issues that are identified in developing the safety case. The management group must therefore be accessible and responsive to the concerns of the other groups, and the other groups are responsible for keeping the management group informed of concerns as they arise.

A4.2.3 The science and technology group

The role of this group is indicated in green in Tab. A4.1 and in Fig. A4.1. The group manages the scientific and engineering investigations that lead to the acquisition of information relevant to the disposal system under consideration and evaluates this information (e.g. in the form of geological and engineering syntheses). The understanding and knowledge acquired by this group of project staff and contractors are the primary basis for the development of the system concept and a database of potentially relevant phenomena.

The group also, towards the end of the safety assessment, performs an audit of key assumptions underlying the safety case (*Audit: Check key assumptions* in Fig. A4.1). This audit ensures that the knowledge and expertise of the science and technology group are exploited as fully as possible in developing the safety case and, in particular:

- that the scenarios considered, and the conceptual assumptions and (often simplified) models and data used to analyse assessment cases, are supported (or at least understood) by those with detailed understanding of the underlying phenomena and awareness of available observations and measurements,
- that complementary arguments (supporting evidence) for safety derived directly from scientific investigations and design studies are included in the safety case, and
- that the overall conclusions are consistent with current scientific understanding.

A4.2.4 The safety assessment group

The role of this group is indicated in red in Tab. A4.1 and in Fig. A4.1. This group, together with the science and technology group, ensures that scientific understanding is properly reflected in the development of the system concept and safety concept and that the approach to safety assessment is appropriate. The group is responsible for formulating the assessment cases and identifying the *information and codes* ("assessment capability") required for their evaluation. The adequacy of the information and codes is evaluated by auditing them against a catalogue of relevant phenomena, or FEPs[1] (*Audit super-FEPs vs. information / codes* in Fig. A4.1). If necessary, further information is sought, or additional tools acquired or developed. The group ensures that the inclusion or exclusion of phenomena from safety assessment cases is not unreasonably biased by the availability of particular methods, models and datasets and that the safety assessment is carried out in a reliable and traceable manner. Finally, the group brings together the various arguments and analyses that constitute the safety case.

A4.2.5 The bias audit group

The role of this group is indicated in yellow in Tab. A4.1 and in Fig. A4.1. The group ensures that all relevant scientific understanding is taken into account in the definition of assessment cases. An important aspect of this is the FEP management procedure, which is discussed in some detail in Chapter 3. The procedure is structured to ensure that all reasonably conceivable FEPs are identified, that they are evaluated (or screened out) in the context of the disposal system in Opalinus Clay, that the safety-relevant super-FEPs (SFEPs, or groups of related FEPs) are incorporated into assessment cases and that a clear path exists for the entire process, including audits and checks.

Details of the FEP management procedure are discussed in Section 3.7.5.

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[1] Features, events and processes. As discussed in Chapter 3, FEPs are grouped and screened to form super-FEPs (SFEPs), which form a more convenient basis for evaluation in the safety assessment.

A4.3 The use of expert judgement in decision-making

A4.3.1 Decisions involving the application of expert judgement

The methodology described in Chapter 3 does not consist of a series of unique or predefined instructions, such that any team of safety assessors would arrive at an identical safety case. Rather, it calls on the groups involved in the planning of the SF / HLW / ILW repository to carry out both well-defined procedures and to exercise expert judgement. Although such judgements are based on the subjective opinions of experts, they should nevertheless result in transparent and well-founded decisions. General principles to ensure the proper application of expert judgement are described in the following section.

A4.3.2 Ensuring proper application of expert judgement

General principles to ensure the proper application of expert judgement include:

(i) Experts are asked to take account of the views of the scientific and technical community as a whole, and not simply to present their own personal opinions. An individual expert might, for example, believe that a particular phenomenon is unimportant, whilst being aware that this view is not universally shared. Divergences in opinion must be taken into account in evaluating ranges of uncertainty.

(ii) Experts are made aware of all available information (and its limitations) that is relevant to the judgement they are being asked to make. There are, for example, many phenomena that may occur during the evolution of the repository and these phenomena are often coupled. A judgement as to the relevance of one phenomenon may require information on several other phenomena.

(iii) Experts from the scientific programme are asked to give unbiased judgements. They should not, for example, bias their judgements in the interests of conservatism or disregard information of questionable reliability, although they should comment on the reliability of such information and take account of the quality of different sources of information in arriving at an overall judgement. Possible modifications to the judgements made by the scientific experts regarding conservatism and the inclusion or exclusion of phenomena are the responsibility of other groups involved in safety assessment.

(iv) In order to achieve (i) and (ii), experts are encouraged to interact with others in their own field and in other relevant fields.

(v) Expert judgements are reviewed internally (and, if necessary, modified) to ensure that key project personnel concur with, or are at least aware of, the judgements made that fall within their area of responsibility or expertise.

(vi) Experts are expected to take responsibility for the judgements that they make. This makes it less likely that irrelevant or unimportant issues will be assigned undue importance. This may also be important if clarification or further justification for a judgement is required during the internal (or external) review process.

Appendix 5 Definitions

Alternative Conceptualisations

Credible alternative evolutions of part of the system.

Assessment Capability

Information and tools available for a specific safety assessment, including

- site characterisation data;
- assessment methods;
- models, codes and data.

Assessment Case

An assessment case is a specific conceptualisation of the evolution of the disposal system that is investigated in the safety assessment. It also includes the assignment of parameter values.

Conservatism

The use of conceptual assumptions and parameter choices that over-predict radiological consequences, and are known to lie outside the range of possibilities.

Demonstration of Construction Feasibility[1]

Demonstration that a repository for spent fuel, vitrified high-level waste and long-lived intermediate-level waste of Swiss origin can be constructed, operated and closed in a potential siting area using current technology.

Demonstration of Disposal Feasibility[2]

Demonstration that a safe repository for spent fuel, vitrified high-level waste and long-lived intermediate-level waste of Swiss origin could be implemented in Switzerland. It consists of three elements:

1. Demonstration of Siting Feasibility;
2. Demonstration of Construction Feasibility;
3. Demonstration of Long-Term Safety.

[1] English translation of the German term "Baumachbarkeitsnachweis"

[2] English translation of the German term "Entsorgungsnachweis"

Demonstration of Long-Term Safety[1]

Demonstration that a given design for a repository for spent fuel, vitrified high-level waste and long-lived intermediate-level waste of Swiss origin, located within a potential siting area, meets the applicable standards for long-term safety[2].

Demonstration of Siting Feasibility[3]

Demonstration that at least one region in Switzerland contains a potential siting area where a safe repository for spent fuel, vitrified high-level waste and long-lived intermediate-level waste of Swiss origin can be constructed.

Dose

The quantity calculated by the biosphere model is the annual effective dose to an adult individual, defined as the sum of the weighted dose equivalents in specific organs, integrated over 50 years, from the intake of activity into the body in one year, plus the sum of the weighted dose equivalents from external radiation in one year. For convenience, the term annual individual dose, or simply dose, is used in this report.

Host rock

The host rock consists of the Opalinus Clay and the Murchisonae Beds in Opalinus Clay facies.

Optimism

The use of conceptual assumptions and parameter choices that give rise to calculated radiological consequences that are towards the low end of the range of possibilities supported by current understanding.

Pessimism

The use of conceptual assumptions and parameter choices that give rise to calculated radiological consequences that are towards the high end of the range of possibilities supported by current understanding.

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[1] English translation of the German term "Sicherheitsnachweis"

[2] Regulatory Guideline HSK-R-21

[3] English translation of the German term "Standortnachweis"

Pillars of Safety

The pillars of safety are features of the disposal system that are key to providing the *safety functions:*

- *The deep underground location of the repository*, in a setting that is unlikely to attract human intrusion and is not prone to disruptive geological events and to processes unfavourable to long-term stability;
- *the host rock* which has a low hydraulic conductivity, a fine, homogeneous pore structure and a self-sealing capacity, thus providing a strong barrier to radionuclide transport and a suitable environment for the engineered barrier system;
- *a chemical environment* that provides a range of geochemical immobilisation and retardation processes, favours the long-term stability of the engineered barriers, and is itself stable due to a range of chemical buffering reactions;
- *the bentonite buffer (for SF and HLW)* as a well-defined interface between the canisters and the host rock, with similar properties as the host rock, that ensures that the effects of the presence of the emplacement tunnels and the heat-producing waste on the host rock are minimal, and that provides a strong barrier to radionuclide transport and a suitable environment for the canisters and the waste forms;
- *SF and HLW waste forms* that are stable in the expected environment;
- *SF and HLW canisters* that are mechanically strong and corrosion resistant in the expected environment and provide absolute containment for a considerable period of time.

Reference Case

The Reference Case is the reference scenario, evaluated with the reference conceptualisation and the reference parameters set.

Reference Conceptualisation

That conceptualisation within a scenario which is most consistent with our current understanding but includes conservative and pessimistic assumptions.

Reference Scenario

The assumed future evolution of the repository system which is most consistent with our current understanding.

Reserve FEP

A feature, event or process (FEP) that is considered likely to occur and to be beneficial to safety and which is deliberately excluded from assessment cases, or at least from their analysis, when the level of scientific understanding is insufficient to support quantitative modelling, or when suitable models, codes or databases are unavailable. Such FEPs are termed reserve FEPs, since they may be mobilised at a later stage of repository planning if the level of scientific understanding is sufficiently enhanced, and the necessary models, codes and databases are developed. The existence of reserve FEPs constitutes an additional, qualitative argument for reserves of safety beyond those indicated by the quantitative analysis.

Safety Case

The safety case is the set of arguments and analyses used to justify the conclusion that a specific repository system will be safe. It includes, in particular, a presentation of evidence that all relevant regulatory safety criteria can be met. It includes also a series of documents that describe the system design and safety functions, illustrate the performance, present the evidence that supports the arguments and analyses, and that discuss the significance of any uncertainties or open questions in the context of decision making for further repository development.

Safety Concept

The safety concept is the conceptual understanding[1] outlining why the disposal system is safe. It includes a description of attributes of the disposal system that are of key importance to safety; i.e.

- that contribute directly to preventing or reducing radionuclide releases;
- that avoid detrimental phenomena and uncertainties;
- that mitigate the effects of detrimental phenomena and uncertainties.

Safety Functions

The disposal system performs a number of functions relevant to long-term security and safety. These are termed safety functions; they include:

- *Isolation from the human environment:* The safety and security of the waste, including fissile material, is ensured by placing it in a repository located deep underground, with all access routes backfilled and sealed, thus isolating it from the human environment and reducing the likelihood of any undesirable intrusion and misapplication of the materials. Furthermore, the absence of any currently recognised and economically viable natural resources and the lack of conflict with future infrastructure projects that can be conceived at present reduces the likelihood of inadvertent human intrusion. Finally, appropriate siting ensures that the site is not prone to disruptive events and to processes unfavourable to long-term stability.
- *Long-term confinement and radioactive decay within the disposal system:* Much of the activity initially present decays while the wastes are totally contained within the primary waste containers, particularly in the case of SF and HLW, for which the high integrity steel canisters are expected to remain unbreached for at least 10 000 years. Even after the canisters are breached, the stability of the SF and HLW waste forms in the expected environment, the slowness of groundwater flow and a range of geochemical immobilisation and retardation processes ensure that radionuclides continue to be largely confined within the engineered barrier system and the immediately surrounding rock, so that further radioactive decay takes place.
- *Attenuation of releases to the environment:* Although complete confinement cannot be provided over all relevant times for all radionuclides, release rates of radionuclides from the waste forms are low, particularly from the stable SF and HLW waste forms. Furthermore, a number of processes attenuate releases during transport towards the surface environment, and limit the concentrations of radionuclides in that environment. These include radioactive decay during slow transport through the barrier provided by the host rock and the spreading

[1] based on the system concept

of released radionuclides in time and space by, for example, diffusion, hydrodynamic dispersion and dilution.

Scenario

An assumed future evolution of the repository system. A scenario is characterised by the main phenomena governing system evolution and radionuclide release. A group of assessment cases is being used to evaluate the consequences of a scenario.

System Concept

The system concept is the conceptual model[1] of the disposal system developed for the purpose of assessing long-term safety. It includes a description of the key features of the system, of events, processes and interactions that may affect its evolution, and of the possible paths that its evolution may take.

"What if?" Cases

A "what if?" case is an assessment case set up to test the robustness of the disposal system. "What if?" cases are outside the range of possibilities supported by scientific evidence. To limit the number of cases, they are restricted to those that test the effects of perturbations to key properties of the pillars of safety.

[1] Conceptual model: "A set of qualitative assumptions used to describe a system or subsystem for a given purpose" (OECD/NEA, 1995c)