TECHNICAL REPORT

96-05

SR 95

Template for safety reports with descriptive example

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SR 95

Template for safety reports

with descriptive example

FOREWORD

This report provides a template for future safety reports on long- term safety in support of important decisions and permit applications in connection with the construction of a deep repository system.

The template aims at providing a uniform structure for describing long-term safety, after the repository has been closed and sealed. The availability of such a structure will simplify both preparation and review of the safety reports, and make it possible to follow how safety assessments are influenced by the progressively more detailed body of data that emerges.

A separate section containing "descriptive examples" has been appended to the template. This section illustrates what the different chapters of the template should contain. For chapters that are independent of the siting and design of the deep repository – e.g. the chapter on safety goals – the text is a preliminary version of the text in coming reports. System- and site-specific chapters are exemplified with material from ongoing or previous work. The assumptions or conditions presented may very well change during the work of preparing future safety reports.

The chapters in a safety report that are supposed to describe **applied** methods and quantifying calculations are utilized in the "descriptive example" for a presentation of **available** methods and calculation tools/models. The applicability and quality of the methods and models are discussed and illustrative calculations are presented. The contents of these chapters are based on work currently in progress and constitute a general review of the state of the art.

The purpose of the "descriptive example" is to clarify the contents of the template. The aggregate description does not meet the requirements on completeness and consistency between the different parts that must be met by a safety report. The first complete safety report will be compiled in support of an application for siting and erection of an encapsulation plant.

Stockholm, December 1995

SWEDISH NUCLEAR FUEL
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Template for safety reports

TEMPLATE FOR SAFETY REPORTS

Index

Template for safety reports

1 BACKGROUND AND GOALS

Background

SKB's overall planning for the management and disposal of Sweden's nuclear waste is described in SKB RD&D-Programme 95 /1-5/. There it is explained that the process for accomplishing this includes a number of important permit applications and decisions, each of which requires supporting material containing safety assessments for operation and post-closure containment. To permit a rational handling of this supporting material, a standardized outline or template for future safety reports is presented here. The template is concerned solely with giving an account of the safety and risks associated with the long period of passive containment (the post-closure phase) that commences after the waste has been emplaced and the repository closed and sealed. It does not cover safety in connection with construction and operation.

The next safety reports to be prepared will be in support of the permit application for the encapsulation plant, SR-I, and the deep repository, SR-D. After this, safety reports will be compiled in preparation for initial operation – operating stage 1 – and regular operation – operating stage 2 – of the handling system and deep repository, and in preparation for future closure.

Safety reports serve as a basis for a series of decision steps in the implementation of the deep disposal scheme. The various reports must therefore have a fundamental continuity and uniformity. The reports can, however, present safety judgements based on different underlying facts, and the decisions are sometimes of differing character. This may lead to differences in how the assessments are carried out and how the safety evaluation is reported. For example, the safety report for SR-I will have to examine the safety-related importance of the variation in site characteristics for different sites in Sweden, and the report for SR-D will need to clarify any safety differences between the two sites that have been characterized by means of site investigations. In the subsequent reports, the safety-related importance of a progressively expanding body of geoscientific background data will have to be demonstrated.

Safety reports must always satisfy three fundamental requirements:

- The purpose of the decision that is to be based on the safety report must be clearly defined, and the executed safety evaluation must be relevant to this purpose.
- Assumptions, analyses and results (incl. uncertainties) must be reported to such an extent and in such a way that an independent review can be carried out.
- It must be demonstrated that safety is adequate in relation to given acceptance criteria or that the safety potential is sufficient to permit a transition to the next development phase.

Goals

In order to create a continuity in safety reporting and a uniformity between reports, this template has been developed for giving an account of long-term safety.

The template is also intended to facilitate the preparation and review of safety reports, and to simplify comparisons between how, for example, an expanded body of data influences the assessment of safety potential and uncertainty.

The template is presented as a synopsis for future safety reports. To further clarify the structure of the template, the proposed outline is exemplified with chapter texts. Since SR 95 does not itself constitute a safety report, however, this text does not give an account of any specific safety assessment. The chapters are instead utilized for a description of the underlying premises and the status of the work as a platform for future safety assessments.

Another purpose of the descriptive example is to complement the knowledge overview that is provided in SKB RD&D-Programme 95 with a detailed survey of the methods and modelling tools that SKB has at its disposal for the upcoming safety assessments.

The chosen outline is discussed below, after which the template is described in the form of a synopsis. In the exemplifying text, the different chapters begin with the synopsis text followed by comments on how the particular chapter in the descriptive exemplification differs from this general synopsis.

2 STRUCTURE OF THE TEMPLATE

Each safety report is organized in main parts as follows:

- Premises and purpose
- Description of the deep repository system
- Evolution of the repository system with time
- Evaluation and conclusions

A basis for the subdivision of the main parts into chapters is to facilitate the "reuse" of chapters that don't change from one safety report to the next, and to attempt to concentrate the needs for changes to a few chapters when there have been changes in underlying premises, methodology or design specifications.

Premises and purposes

This main part should clarify what phase of development the work with the repository system is currently in and the specific decision for which the safety report is supposed to provide support. The overall safety goals and acceptance criteria for the activities, as well as any stage requirements, should be set forth.

A retrospect should be provided with reference to previously conducted performance and safety assessments of importance. An account of the safety report's scope or specific premises and delimitations should be given, along with a description of the assessment methodology employed, particularly if any special simplifications or unfavourable conditions have been introduced.

System description

The system description includes chapters for

- quantities and characteristics of the radioactive waste,
- design, general layout and dimensions of the deep repository,
- materials and dimensions for the engineered barriers around the waste, and
- site-specific conditions with regard to the geoscientific characteristics of the facility site and the surrounding biosphere.

In addition, there is a chapter describing how the layout and other design parameters of the facility have been adapted to exploit the safety potential of the site. This chapter is separated from the general system description, since it will probably be necessary to make progressive site-specific adaptations during both the site investigation and construction phase in order to be able to effectively exploit the potential of the site for good safety.

The safety account is based on the specific system defined by the system description. It does not describe work carried out in previous stages in order to optimize the design and dimensions of the repository via performance assessments, sensitivity and variation studies and safety assessments.

Analysis of the evolution of the repository system

This main part provides a detailed description of the methodology employed to assess the safety of the repository, the calculations made and the models and data used for this purpose. The division is divided into chapter-by-chapter accounts of:

- general review of the evolution of the repository system – choice of scenarios
- analysis of the intended performance of the repository and its delimitations
- analysis of radionuclide transport
- detailed qualitative and quantitative assessment of selected scenarios.

The first three chapters also contain accounts of methods and tools that are used in the analyses. The account of the analyses that clarify the normal performance of the repository, to isolate and contain the waste, is separated from the account of the analysis of radionuclide solubility and transport in the event of defective canisters. This is done in order to clarify the differences in how the primary safety function (isolation) and the secondary function (low solubility and high retardation) are achieved. Processes that are common to both functions are dealt with only in the one chapter.

The body of data used for certain safety assessments can be so extensive that it should be broken out to a separate chapter to increase readability. This chapter should then include both measurement data, including uncertainties, and methods for translating them to parameters in the safety assessment.

Evaluation and conclusions

The results of quantitative and qualitative assessments of the performance of the repository must be evaluated with regard to relevance and uncertainty and be compared with safety goals and acceptance criteria. The material must also be judged with respect to the decision for which the safety report constitutes supporting material. This material and the summarizing conclusions of the safety judgements are to be presented in the last two chapters of the report.

3 SYNOPSIS

PREMISES AND PURPOSE

Chapter 1 Introduction

The introduction summarizes the premises of the work of managing and disposing of radioactive waste in Sweden. Thereafter, the programme for this work and how far it has progressed are described.

This is followed by an account of the purpose of the report, the decision that needs to be taken and how long-term safety influences this decision. The chapter concludes with a description of how the report is organized and certain reading tips.

Chapter 2 Safety goals and acceptance criteria

Chapter two describes the safety goals that serve as a basis for SKB's work with the final disposal of the Swedish radioactive waste.

The chapter begins with a summary of international rules and regulations that influence safety goals and acceptance criteria. Only regulations that have a direct bearing on the post-closure phase are included here.

Thereafter a brief overview is presented of the Swedish laws and ordinances that define the safety-related premises for the work. The section then deals with specific regulations for the deep disposal of long-lived radioactive waste in the bedrock.

Subsequent sections discuss the practical application of the rules and regulations. They deal, for example, with the implications of the concept of "a holistic view of radiation protection", collective dose and dose to other organisms than man. They also discuss how safety goals and acceptance criteria can best be translated into different numerical values for repository performance. The numerical values are intended to serve as key ratios or performance indexes that can be employed in a practical manner in different phases of the repository's development or during different stages of the work of building the repository. They should be relevant to important safety functions and simple to relate to formal safety requirements.

Safety-related factors that are utilized in the siting procedure must be selected so that they can be examined in the investigations that are planned during the siting process. Priorities and balance with regard to safety under anticipated normal conditions and under less normal conditions are discussed.

The chapter concludes with a brief resumé of norm systems and practice for non-radioactive hazardous substances.

Chapter 3 Methodology description

Chapter 3 presents SKB's methodology for the safety evaluation.

In an introductory section, the historic background of previous design studies and performance and safety assessments associated with them is to be presented together with an overview of how the work has led up to the general design of the system. It must, however, be clearly shown that the safety assessment that is being reported evaluates only the safety of the selected and presented system. Options for alternative designs are presented to the extent there is a need to give an account of remaining freedom for design changes or possibilities for adaptation to site-specific conditions.

This is followed by a discussion of the safety-related evaluation required for the decisions in question, and the focus and delimitations of the safety report this leads to. The effect of the delimitations on the safety assessments is discussed, after which the procedure for the performance and safety assessments is presented. The section also will provide an overview of how safety in the post-closure phase is related to the handling of the waste and to the design construction and control of the repository and the safety barriers. The section is concluded with a recapitulation of radiological safety goals and acceptance requirements in accordance with chapter 2 and a presentation of the specific numerical values that are utilized in the present safety report.

Subsequent section will give an account of the methods that have been utilized to systematically analyze the conditions in the deep repository that are important for performance and safety, as well as the events and processes that can influence the evolution of the repository with time. Concepts such as process system, normal scenario, reference scenario etc. are defined. Possibilities for covering reasonable evolutionary pathways for the repository with different scenarios and calculation cases are discussed. Similarly, possibilities for quantifying the different scenarios and how this quantification is dependent on the availability of information and the current stage of the work are discussed on a general level.

The next section examines the uncertainties that enter in during the course of carrying out the safety assessment. Possibilities for quantifying or limiting these uncertainties are discussed, as are the possibilities of integrating them in a total uncertainty estimate.

A concluding section describes the quality procedures that have been applied in the execution of the particular safety assessment.

DESCRIPTION OF THE REPOSITORY SYSTEM

Chapter 4 Spent nuclear fuel and other long-lived waste

Chapter 4 gives a description of the spent nuclear fuel and other waste types that will be deposited in the deep repository. The chapter is divided into three sections covering spent nuclear fuel, other long-lived radioactive waste and other toxic waste. The description pertains to the physical and chemical form of the different waste types, as well as quantities and contents of important radionuclides.

The section on spent fuel also contains a discussion of the structure of the fuel: cracks in fuel pellets; the size distribution, surface area and porosity of the fuel fragments; and the properties of the fuel-clad gap. In addition, the distribution of fission and activation products inside the fuel and in the structural elements encapsulated in the canister is discussed.

Models for and calculations of nuclide inventories and decay heats are presented, for both spent fuel and other long-lived waste, as are the selection criteria used to determine which radionuclides are to be included in the analysis.

Chapter 5 Design of the repository system

Chapter 5 describes the design of the repository system and the engineered barriers. The description is based on the background material available in the current design stage submitted in support of the safety assessment that is being presented. The chapter presents the design and materials of the deep repository, as well as quality requirements (impurities) and dimensions for the barriers included in the repository system. Similarly, methods for construction and inspection that may be utilized and their possible effects on the host rock are discussed. Performance studies or previous safety assessments that have served as a basis for choice of dimensions or design are summarized. "Free" parameters that can be utilized for site adaptation or optimization are reported separately.

Against the background of the major safety functions for the various barriers, a review is made of the principles for the design of the repository, after which the different parts of the underground facility and their design/layout are presented. This is followed by a presentation of the method of excavation of deposition positions and the fabrication/design of the canister, bentonite buffer and tunnel backfill. Plugging of tunnels and shafts and the final closure and sealing of the repository are discussed.

Chapter 6 Properties of the repository site

This chapter describes the geoscientific properties of the selected repository site that are of potential importance for the long-term safety of a deep repository in crystalline rock.

Based on site data and geoscientific understanding, a site model is devised representing the structural characteristics of the site with a bearing on long-term safety. This model can be said to be a composite of all the geological, geophysical, geochemical and geohydrological information that has been collected.

The chapter is organized in sections that describe geology, groundwater chemistry, geohydrology and the transport properties of the rock.

The composite site model is associated with uncertainties, which are discussed in each section and summarized in a concluding section. The consequences of these uncertainties in the site description for the results of the safety assessment are discussed later in the report, however. The description of the properties of the site is based on present-day conditions. Time-dependent changes are discussed in the relevant sections.

Chapter 7 Placement and progressive construction of the repository

The repository design in chapter 5 is of a general nature, i.e. independent of any detailed knowledge concerning the specific properties of the site. In practice, the design of a repository always has to be adapted to the specific site. This chapter describes the site adaptations that are relevant to the safety report. These adaptations may have to do with how different repository sections are situated in relation to each other, how tunnels/shafts and deposition areas are located, or how deep down in the rock different facility sections are located. Adaptations may also be necessary with regard to fracture zones or zones of weakness, rock quality, groundwater flow paths or biosphere recipients for deep groundwater.

The chapter also discusses the possibilities for adaptation during the course of construction, alternative design and the residual design freedom that exists for future optimization.

Chapter 8 The biosphere

The biosphere on the selected repository site is described in this chapter, with a focus on factors of importance for the release and migration of radionuclides.

Based on data from the geoscientific and hydrological characterization of the site, groundwater recipients and transport pathways in the vicinity of the repository are described. Similarly, the site- and region-specific exploitation of the natural environment is discussed, as well as transfer pathways to man. Site-specific transport models, if any, are described.

A specific characterization of the biosphere may have to be done as a basis for the analyses of possible radiological consequences for other organisms than man.

The uncertainties are discussed, mainly in view of the high potential of the ecosystems for change with time, and with reference to the site-specific factors than can limit this changeability.

EVOLUTION OF THE REPOSITORY SYSTEM WITH TIME

Chapter 9 Analysis – repository system, scenarios

The chapter begins with an overview of the repository and its performance. This is followed by a review of the repository's barriers, the time-dependent processes that may occur in the repository or between the repository and its surroundings, and the resultant evolution of the repository over time.

The scenario methodology applied in the safety assessment is presented. The resultant interaction matrices and a documentation outline are presented and discussed.

The choice of scenarios and how the scenarios are dealt with in the safety assessment are gone through. Finally, uncertainties and completeness in the scenario work are discussed.

Chapter 10 Analysis – performance of the repository

The analysis of the performance of the repository system is divided into two chapters. Chapter 10 describes the processes and conditions that control (or can change) the primary safety function of the barriers – to isolate the spent nuclear fuel from the groundwater. Chapter 11 describes the processes and conditions that control the repository's secondary safety function – to limit the dissolution of radionuclides in a damaged canister, and to retard their transport to the biosphere.

This chapter describes the anticipated performance of the different repository subsystems in maintaining the isolation, and processes that could influence this performance. The chapter is divided into the following sections:

- thermal development
- rock performance
- canister performance
- buffer performance
- dissolution processes in the canister

Functional requirements, essential processes, methods for analyzing and quantifying the processes, and parameter limitations introduced to ensure the desired function or to simplify the analyses are presented in the sections.

Certain processes and conditions are essential for both the isolation and the retardation functions. One example is the movement of the groundwater and its ability to transport dissolved substances or colloids. Such processes are only dealt with in the one chapter, and references are made between the chapters.

Chapter 11 Models and calculation methods for radionuclide transport

This chapter deals with the performance of the repository parts when it comes to dissolution and transport of radionuclides from a damaged canister and the processes and conditions that can influence this performance.

The chapter is divided into the following sections:

- groundwater movements
- transport in the near field
- transport in surrounding bedrock
- transport of radionuclides in the biosphere and dose calculation.

Essential processes and the parameters and calculation models used to analyze the processes are described in each section. Comments are made on the quality and applicability of the models.

The chapter concludes with an explanation of the strategy that is employed for the calculation chain (from radionuclide release to radiation dose in the environment) that quantifies the consequences of defective canisters. The strategy is chosen based on the purposes of the safety report. General questions are dealt with here, such as the structure of the calculation chain, model couplings, the degree of probability in the analysis, handling of long periods of time, etc. Certain strategic choices for individual models are also discussed here, for example choice of fuel dissolution model, the set of nuclides included in the calculations, conceptualization of near and far field, description of the lateral extent of the repository system in the rock, how site-specific models have been designed, etc.

Chapter 12 Scenarios/calculation cases

This chapter presents the results of the analysis of chosen scenarios or calculation cases. Separate sections contain qualitative descriptions or calculation results from main cases, variations and sensitivity analyses of:

- normal scenario
- canister defect scenario
- glaciation
- earthquakes
- effects of human activities
- materials left behind in the deep repository
- other scenarios

EVALUATION AND CONCLUSIONS

Chapter 13 Integration of results and uncertainty

This chapter presents the evaluation and weighing-together of the judgements and calculations made during the course of the safety assessments which is carried out in order to obtain a composite safety picture. The evaluation/ weighing-together should be related to the purpose at hand and rate the robustness of the safety evaluations with respect to the uncertainty of the underlying data.

Chapter 14 Conclusions

This chapter summarizes the overall safety assessment that has been carried out.

Descriptive example

DESCRIPTIVE EXAMPLE

Index

1 INTRODUCTION

The introduction summarizes the premises of the work of managing and disposing of radioactive waste in Sweden. Thereafter, the programme for this work and how far it has progressed are described.

This is followed by an account of the purpose of the report, the decisions that need to be taken and how long-term safety influences this decision. The chapter concludes with a description of how the report is organized and certain reading tips.

In this report, *section 1.3 comprises a description of the purpose of SR 95.*

1.1 BACKGROUND

The owners of nuclear power plants in Sweden have assigned SKB the task of preparing proposals for how and where the radioactive waste generated at their plants is to be disposed of. Essential parts of the waste repository system are already in operation. The parts that do not yet exist are facilities for encapsulation of spent fuel etc. and for final disposal of long-lived waste, particularly spent nuclear fuel.

The existing system has been developed and built up on the basis of proposals of the Aka Committee in the mid-1970s as well as the research and development work initiated with the KBS project during the latter half of the 1970s. Proposals and alternative options have since been considered and investigated by both public authorities and the nuclear power industry in extensive R&D activities throughout the 1980s. In this way, important questions relating to encapsulation and final disposal have been thoroughly studied. Similar work has been and is being carried out in most countries with major nuclear power programmes.

The work that has been done over the past 15 years or so in Sweden and other countries has led to a broad consensus among international experts that methods exist to implement a final disposal of high-level waste and spent nuclear fuel, and that methods also exist to analyze the long-term safety of such a disposal scheme. RD&D-Programme 92 with supplements /1-1, 2/ presented a prioritized system design and a plan for designating candidate sites for the construction of the repository, for characterizing those sites and for adapting the repository to local conditions. After extensive expert and regulatory review of the programme /1-3, 4/, the Government has accepted the proposed scheme in all its essential respects.

1.2 PLANNING

The progressive construction of an encapsulation plant and a deep repository encompasses a large number of measures on which decisions are made in a step-by-step process. Figure 1.2-1 shows a schematic logic diagram of the deep repository programme. The measures that concern the deep repository span a period of about sixty years or longer from the start of feasibility studies up to completed closure of the repository.

Assessments of long-term safety are carried out and reported in the form of safety reports prior to each important decision-making occasion in the programme. Such decision-making occasions (for example important choices between alternative pathways for the future development, or measures that entail firm commitments to a given site) have also been coupled via legislation to regulatory review and licensing. From the start of repository design and siting up to the issuance of a permit to seal a filled deep repository, a series of safety reports will be produced.

The two decision-making occasions that come first in the programme are a permit for an encapsulation plant for spent nuclear fuel, and a permit for the start of detailed geoscientific site investigations for a deep repository.

Different safety assessments in the programme are based on different bodies of data and therefore differ from each other. Since the purpose of the safety report is to show the safety related foundation for an increasingly firm commitment to a system design and a site for the deep repository, however, it is essential even in early reports to evaluate and substantiate the potential for achieving total safety of the repository. To facilitate follow-up of how the safety assessment of a site is affected by a growing body of information, SKB has deemed it

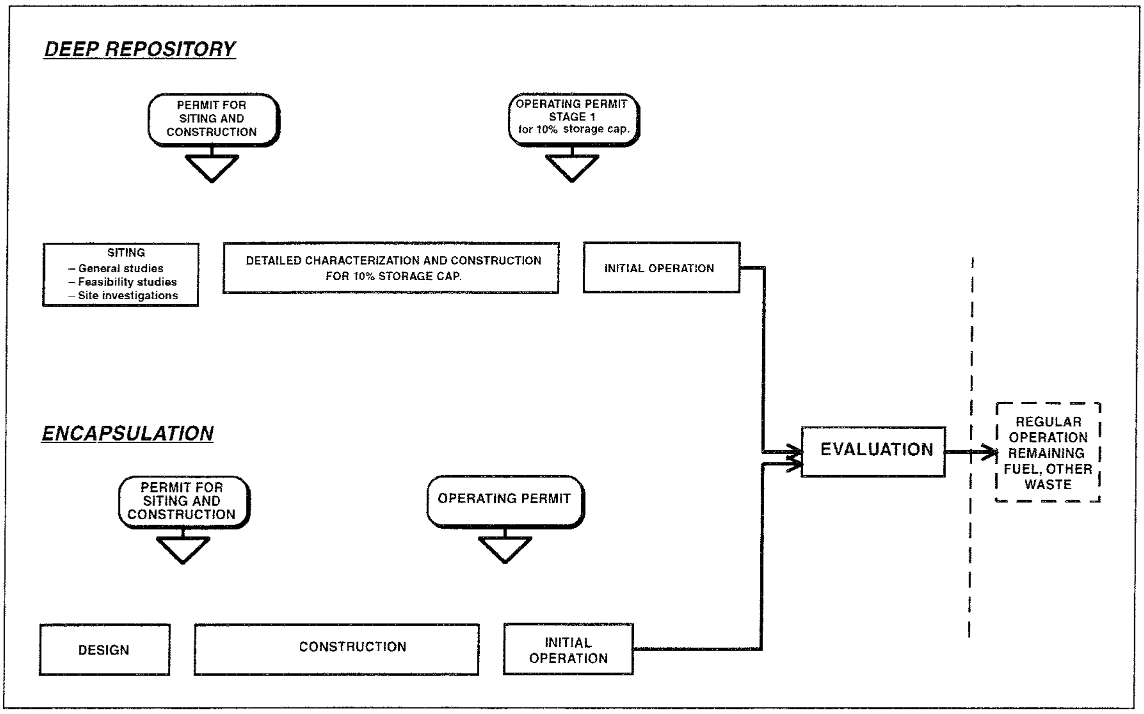


Figure 1.2-1. Diagram of the major steps in the deep repository programme and the requisite permits.

expedient to establish a template for the outline of future safety reports. This report, SR 95, offers a proposal for such a template for future safety reports for the deep repository, and the present part, which comprises a descriptive example, has been structured in accordance with this template.

1.3 PURPOSE OF THE REPORT

Goal

The primary goal of SR 95 is to establish a template for the outline and content of future safety reports. The purpose of the template is to:

- simplify the preparation of future safety reports,
- facilitate a follow-up of how safety assessments are progressively broadened and deepened in different phases of the work,
- simplify the review of the safety reports.

In order to clarify the contents of the template, a descriptive exemplification is also presented, in accordance with the proposed chapter headings. Each chapter has a text which either shows what the report is intended to cover or discusses alternatives to this.

Since SR 95 is being published in conjunction with RD&D-Programme 95 /1-5/, a secondary goal has also been to demonstrate, via the exemplifying text, in greater detail:

- how future safety assessments will be carried out within SKB,
- the methods that have been developed to provide systematics and traceability in the work, and
- the calculation models which SKB currently has at its disposal to quantify the performance and safety of the repository.

1.4 OUTLINE OF THE REPORT

In accordance with the proposed template, the descriptive example contains four main divisions:

- premises and purpose of the safety report
- description of the deep repository system,
- evolution of the repository system with time,
- evaluation and conclusions.

Each chapter begins with a paragraph in italics where the intentions of the section and its contents are presented. Comments are also made here on the specific content of SR 95 that differs from the template, for example in connection with the general reviews of the current level of knowledge and state of the art of safety assessment methods. Thereafter follows, in normal typeface, an illustrative example of what the text in the section may look like for a particular permit application.

Preparations are under way for an application to site and build an encapsulation plant. The illustrative text in SR 95 will therefore be based to a large extent on material that is in the process of being compiled for the safety report for the encapsulation plant, SR-I, or for the safety report next in line – a permit for the Deep Repository and for detailed geoscientific investigations of a potential site, SR-D. At the present time the material is incomplete and the premises that are given may change.

Sections that are independent of specific site data and detailed design have a degree of detail close to that planned for coming safety reports for permit applications. Sections with a strong coupling to site investigations or design of barriers have been utilized in SR 95 to shed light on important issues and to discuss work methodology and approaches for safety assessments. This is done with illustrative material from previous reports, with recently published material for the copper/steel canister or with material from Äspö.

In coming safety reports, chapter 3 (Method description) and chapters 9–11 (Analysis – repository system, scenarios; Analysis – performance of the repository; Models and calculation methods for radionuclide transport) will describe the methods and models for analysis and quantification that have actually been utilized. In the present document, these chapters are used to give a general account of available methods in the safety assessment and their applicability.

In many cases, references are made to work reports (Arbetsrapporter). These often give accounts of work being done in preparation for SR-I. Judgements and evaluations in these reports may be revised during the course of the work.

2 SAFETY GOALS AND ACCEPTANCE CRITERIA

Chapter two describes the safety goals that serve as a basis for SKB's work with the final disposal of the Swedish radioactive waste.

The chapter begins with a summary of international rules and regulations that influence safety goals and acceptance criteria. Only regulations that have a direct bearing on the post-closure phase are included here.

Thereafter a brief overview is presented of the Swedish laws and ordinances that define the safety-related premises for the work. The section then deals with specific regulations for the deep disposal of long-lived radioactive waste in the bedrock.

Subsequent sections discuss the practical application of the rules and regulations. They deal, for example, with the implications of the concept of "a holistic view of radiation protection", collective dose and dose to other organisms than man. They also discuss how safety goals and acceptance criteria can best be translated into different numerical values for repository performance. The numerical values are intended to serve as key ratios or performance indexes that can be employed in a practical manner in different phases of the repository's development or during different stages of the work of building the repository. They should be relevant to important safety functions and simple to relate to formal safety requirements.

Safety-related factors that are utilized in the siting procedure must be selected so that they can be examined in the investigations that are planned during the siting process. Priorities and balance with regard to safety under anticipated normal conditions and under less normal conditions are discussed.

The chapter concludes with a brief resumé of norm systems and practice for non-radioactive hazardous substances.

***In this report**, the review of Swedish regulations for the deep repository is based on two guiding documents: the basic criteria formulated by the Nordic radiation protection and nuclear safety authorities in 1993, and the considerations presented in SSI report 95-02. The section will be revised when formal directives have been issued.*

The section on a holistic view of radiation protection constitutes a survey of the overall situation with regard to both the different facilities that exist in the nuclear fuel cycle and the different time aspects associated with the possible consequences. A practical application of the concept "holistic view" will require a continued dialogue with the authorities regarding which numerical values and methods are suitable for examining and establishing a balance

- *between different facility sections,*
- *between normal operation and the risk of accidents, and*
- *between exposure risks during different time periods.*

This section also needs to be revised to serve as a good guide for interpreting the safety goals.

2.1 INTRODUCTION

The goals for a safety assessment and the criteria against which the results of the assessment are to be judged are determined to a high degree by regulatory standards and criteria.

The development of rules and criteria in nuclear technology is carried out in close cooperation between international organizations. A summary of international rules and regulations that influence safety goals and acceptance criteria for a deep repository is therefore given in section 2.2 as a background to the presentation of Swedish regulatory criteria in section 2.3. Rules which Sweden has undertaken to observe upon joining the EU, as well as international and regional agreements which Sweden has ratified, are also presented here.

Swedish regulatory criteria concerning nuclear facilities, in which category a deep repository is also included, are very well-developed on many points. The criteria primarily concern the operation of nuclear facilities. A deep repository differs from other facilities in that its safety not only has to be demonstrated for the operating period, but also for a very long post-closure period. Another difference is that the safety level for which the facility is designed cannot be established on the basis of operating experience, but must be based on forecasts.

Detailed regulations are less well-developed when it comes to long-term safety. At present there is a joint criteria document from the Nordic authorities and a preliminary draft of a coming directive from the National Radiation Protection Institute. These documents are presented in greater detail in section 2.3.

Regardless of how future national and international regulatory criteria in the area are formulated, they must be translated into practical applications. Section 2.4 deals with how safety goals and acceptance criteria can be translated into different numerical values for the safe performance of the repository. Collective dose to other organisms than man is also discussed here.

In section 2.5, the long-term radiation dose burden from a repository is compared with the burden that arises in other steps in nuclear energy production. The section attempts to provide a holistic view of radiation protection.

Finally, section 2.6 summarizes the regulation systems and practice for the hazardous chemical substances that can occur in a deep repository.

2.2 INTERNATIONAL RULES AND RECOMMENDATIONS

2.2.1 Introduction

Radiation protection matters are dealt with by a number of international bodies. International rules and recommendations often underlie national legislation. An overview of international cooperation on safety goals and acceptance criteria for a deep repository is provided in this section. Rules are also presented here which Sweden has undertaken to follow as a member of the EU, as well as international and regional agreements which Sweden has ratified.

2.2.2 IAEA – the UN's International Atomic Energy Agency

The IAEA publishes recommendations and guides in matters relating to nuclear power. They are adopted and published after extensive circulation for comment. The publications are divided into categories in accordance with a hierarchical system. At the top are the Safety Fundamentals, which deal with overall goals and principles. Then follow several levels with increasingly detailed guides and recommendations. The recommendations are often incorporated as requirements by Swedish authorities in conjunction with licensing.

A number of recommendations exist that are applicable to a disposal facility. Many of the criteria that are contained in the IAEA publication "Criteria for Underground Disposal of Solid Radioactive Wastes" /2.2-1/ are also found in the document from the Nordic safety authorities discussed in section 2.3.

The IAEA also has a safeguards programme for fissile material. The purpose is to control the spread of e.g. plutonium. Spent nuclear fuel is also included in the programme. Recommendations are expected to come for the handling of waste in a deep repository. Among other things, the question of at which stage in the disposal process surveillance should cease will be considered.

The IAEA has a special programme for Radioactive Waste Safety Standards, RADWASS. Here there is a committee that works with disposal.

2.2.3 European Union

When Sweden became a member of the European Union, it also ratified the Euratom Treaty. The treaty regulates the mutual obligations between member states in nuclear-related matters. This means that standards in the radiation protection area that are agreed on within the EU are binding for Sweden as well.

The ICRP's (see below) recommendations, which have been applied by SSI for some time, will be incorporated in "Basic Safety Standards" for radiation protection which the EU plans to publish.

The Euratom Treaty also provides that a member country shall inform the European Commission regarding planned measures for the disposal of radioactive waste.

2.2.4 ICRP, International Commission on Radiological Protection

The ICRP was founded in 1928 and publishes general recommendations and rules for radiation protection in connection with the use of radioactive materials. The resumé journal "Annals of the ICRP" has been published since 1950. Different topics are dealt with in "ICRP Publications". Many of these are relevant to waste management and to safety matters pertaining to a deep repository. The ICRP's recommendations have comprised an important background for the Swedish and Nordic recommendations for high-level long-lived waste that are presented in section 2.3.

2.2.5 Nordic cooperation

Information exchange and joint research efforts in the Nordic countries are channelled through Nordic Nuclear Safety Research, NKS. NKS is financed through the Nordic countries' safety authorities and by companies and institutions active within the fields of nuclear power, nuclear research and radiation protection.

NKS conducts research projects and studies within areas of common interest. Representatives of nuclear power utilities, regulatory authorities and research institutions participate in the projects for the purpose of improving understanding and coordination between the Nordic countries. Areas that lie outside of pure nuclear technology, but which can be of essential importance for e.g. the deep repository, are also dealt with. An example is methods for preservation of information for the future /2.2-2/. This study is one of several within the NKS programme for waste and decommissioning.

2.3 SWEDISH AND NORDIC AUTHORITIES' CRITERIA FOR DEEP DISPOSAL OF LONG-LIVED WASTE

2.3.1 Introduction

Nuclear activities are regulated in Sweden by a number of laws, ordinances and directives. The most important laws are the Act Concerning the Management of Natural Resources, the Act on Nuclear Activities and the Radiation Protection Act.

Based above all on the Act on Nuclear Activities and the Radiation Protection Act, there are a number of ordinances and regulatory directives that regulate nuclear activities in detail. The regulatory rules mainly apply to the operation of facilities.

As mentioned above, a deep repository differs from other facilities in that its safety must be demonstrated not only for the operating period, but also for a very long period thereafter. Ordinances and directives for nuclear facilities applied in the past therefore need to be supplemented with regulatory criteria for the long-term performance of the facility. The requirements in impending legislation in this area harmonize with the requirements that are made on the operation of nuclear facilities today.

Among the regulatory directives for **operation** of nuclear facilities, the following in particular are applicable to the operation of a deep repository:

- Controls of releases of radioactive substances to air and water during operation, SSI FS 1991:5.
- Regulations concerning maximum permissible radiation doses for personnel, SSI FS 1989:1.
- Regulations concerning radiation protection for personnel, SSI FS 1994:2.
- Regulations concerning the plant radiation protection supervisor, SSI FS 1994:1.

Certain fundamental principles for long-term safety as well can also be gleaned from these regulatory criteria. However, the rules and regulations need to be supplemented with further details and practical criteria for safety assessments. As far as long-term safety is concerned, there are at present (October 1995) two more detailed documents for Swedish conditions:

- An initial draft of a forthcoming directive from the National Radiation Protection Institute: "The National Radiation Protection Institute's protection criteria for the management and disposal of spent nuclear fuel" /2.3-1/.
- A joint criteria document from the Nordic countries' radiation protection and nuclear safety authorities: "Disposal of High Level Radioactive Waste, Consideration of Some Basic Criteria" /2.3-2/.

The Swedish Nuclear Power Inspectorate, SKI, has also published its own statutes since 1994. Work is currently under way on a statute providing guidelines for safety assessments. The contents are expected to specify what a safety assessment should contain and how it should be quality-assured and documented.

The rest of section 2.3 discusses the Radiation Protection Institute's initial draft and the joint Nordic criteria document.

2.3.2 Overall objectives

The Nordic criteria document formulates three objectives for the design of the deep disposal of high-level waste.

A **general objective** is to protect human health and the environment and to limit the burden placed on future generations.

Regarding **long-term safety**, the document says that the risks to human health and the effects on the environment from waste disposal at any time in the future shall be low and not greater than would be currently acceptable. The judgement of the acceptability of a disposal option shall be based on radiological impact irrespective of any national boundaries.

As far as **the burden on future generations** is concerned, it shall be limited by implementing, at an appropriate time, a safe disposal option which does not rely on long-term institutional control or remedial actions as a necessary safety factor.

The SSI document also expresses these goals, directly or indirectly.

2.3.3 Radiation protection requirements

The radiation protection principles and requirements that are generally applied in both the Nordic criteria document and the SSI document are based on the fundamental principles for protection against ionizing radiation set up by the ICRP /2.3-3/. These principles are:

a Justification of practice.

b Optimization of radiation protection.

c Individual protection by dose limits.

Justification

Both documents state that point a is not applicable to a deep repository. The disposal of waste is a necessary consequence of the operation of nuclear power plants.

Optimization of protection

Point b corresponds to what is called the ALARA principle, in other words radiation exposure shall be "As Low As Reasonably Achievable". Another way of expressing this is that all measures that are defensible, taking economic and social factors into account, should be adopted to reduce the total radiation dose burden.

The principle is applicable to both the operational phase and the post-closure phase. The principle can be applied to the post-closure phase in the context of decisions concerning the fundamental design of the repository. By establishing the principles for the performance of the repository but retaining options for alternative designs, degrees of freedom are preserved which can eventually be used to optimize construction and operation in order to best fulfil the above criterion.

Dose limits, Nordic recommendations

Point c is applicable both during the operational phase and in the longer term. The principle can be applied in the long term by comparing calculated individual doses with accepted limit values.

As far as dose limits are concerned, the Nordic recommendations divide the time into the first 10,000 years or so and the time thereafter.

For **the first 10,000 years or so** it is stated that individual radiation doses that can be expected under normal conditions due to releases from the repository should not exceed 0.1 mSv/y.

Furthermore, the probability and consequences of unexpected seriously disruptive events should be studied, discussed and presented in qualitative terms. When possible, they should also be assessed in quantitative terms. The annual probability, as well as the consequence expressed as individual dose, should then be estimated. These factors are multiplied to obtain a risk, which may not exceed the risk at 0.1 mSv/y.

The dose 0.1 mSv/y derives from the ICRP's recommendation that the radiation dose to the population caused by activities with ionizing radiation should not exceed 1 mSv/y. Since an individual can be exposed to radiation from several different sources, each individual cause should, according to Nordic recommendations, only contribute 10% of the total acceptable dose.

Dose calculations presume knowledge of the biosphere on the repository site. They also require knowledge of various societal activities, including man's exploitation of the natural environment. In the future, uncertainties concerning the biosphere and society may eventually become so great that use of doses as a measure of health risks will become meaningless. Other numerical values than radiation doses must be used for time periods when substantial changes in, above all, the environment can be expected to have occurred. Examples of such changes are land uplift and climate changes, in particular an expected ice age.

For **times beyond about 10,000 years**, the Nordic criteria document imposes the requirement that radionuclides released from the repository may not "lead to any significant changes in the radiation environment. This implies that the inflows of the disposed radionuclides into the biosphere, averaged over long time periods, shall be low in comparison with the respective inflows of natural alpha emitters."

Dose limits, SSI criteria

Rules for dose calculations on different timescales are also discussed in SSI's draft edition. Protection criteria for personnel and members of the public are presented. The timescale is divided into:

a the first 1,000 years after closure,

b the time after the first 1,000 years up to the next ice age, and

c the time after the next ice age.

For the periods a) and b), quantitative assessments will be required, including calculated doses to critical groups of people under different assumptions. "Estimates of individual doses for the time up to 10,000 years from closure should be reported as best estimates, i.e. neither under- nor over-estimated, with indication of estimated margins of error." The time span of 1,000 years between a) and b) is justified by the fact that important short-lived nuclides such as Cs-137 and Sr-90 will have decayed after about 1,000 years. More rigorous demands will therefore be made on the figures for the initial 1,000 years.

For period c), only qualitative estimates are needed to give a picture of what might happen with the repository. "This discussion shall include consideration of the risk of elevated releases. Comparisons in figures, e.g. regarding possible leakage of radionuclides, can provide indications of how good the protection is as a background for the qualitative assessment."

Furthermore, it is stated: "For very long times, where long-lived radionuclides dominate, the protection shall, wherever possible, follow society's fundamental protection philosophy for stable substances that can cause cancer or chromosomal damage, for example certain heavy metals."

The National Radiation Protection Institute observes that future radiation doses in some calculation cases may exceed stipulated limits. "For such cases, the probability of the chain of events leading up to the radiation dose mentioned shall be discussed." If the probability is judged to be low, the higher dose may in some cases be accepted by SSI.

The SSI document also refers to relevant regulations for **collective doses** to personnel and the public from nuclear activities. SSI FS 1994:2 gives a collective dose limit of 2 manSv/(y · GWa) for personnel, SSI FS 1991:5 gives 5 manSv/(y · GWa) to the public. The collective doses are supposed to be summed over 500 years. The limit values apply to the sum of all steps in energy production, including waste management and the initial phase following closure of the repository. Revisions of these statutes will show how collective dose calculations should be performed for a deep repository.

For **individual dose**, the same dose limits shall apply to personnel as for other activities (SSI FS 1989:1) and 0.1 mSv/y (SSI FS 1991:5) to the public (critical group).

Environmental protection

For the protection of the environment, the National Radiation Protection Institute's draft edition also requires analysis of transport and accumulation in various media in ecosystems. The basic principle is that biological diversity must be preserved. Quantitative requirements may be made similar to the nuclide inflow limits that are stipulated in the Nordic criteria document, or radiation dose to indicator organisms, if it is not obvious that environmental impact is relatively small.

2.3.4 Requirements and recommendations for repository design and safety reporting

The joint Nordic criteria also deal with the design of the repository and safety reporting.

Design

Regarding repository design, it is stated: "The long-term safety of waste disposal shall be based on passive multiple barriers so that

- deficiencies in one of the barriers do not substantially impair the overall performance of the repository system
- realistic geological changes are likely to only partly affect the system of barriers."

As a complement to this multibarrier principle, a number of recommendations are given for suitable repository design and site selection. The recommendations deal with the factors that are currently considered to be the most important today for achieving a safe deep repository, provided that it is built in crystalline rock.

The recommendations pertain to the disposal scheme described in KBS-3 (see chapter 5) and deal with how the requirements on optimization of protection and radiation dose limits that are discussed above can be met. Certain recommendations are site-specific and are included in the criteria for the siting process, while others are system-specific and have to do with the performance of the engineered barriers.

Safety reporting

Principles are set forth for how safety should be reported in order to permit a good evaluation. These principles require safety assessments based on qualitative judgements and quantitative results from calculation models that are validated as far as practicable. Furthermore, requirements are made on a quality assurance programme for complying with the design bases and pertinent regulations.

2.4 PRACTICAL APPLICATIONS

Different types of regulatory criteria for safety assurance for a deep repository must be concretized in the form of actual calculation cases, numerical values and performance indexes. This section presents the numerical values which SKB plans to use in reporting on the performance and radiological safety of the repository. The plans are in part a consequence of the recommendations and criteria presented in section 2.3.

The repository's safety will also be demonstrated through a discussion of confidence in the fact that those radionuclides that can give high doses are contained and will decay without reaching the biosphere and man. The barriers in which different potentially dose-giving nuclides decay will be described for different cases. Margins and reserves for the safety of these barriers will also be demonstrated.

Doses to man

For the time period from the sealing of the repository up to the beginning of an assumed glaciation (in about 10,000 years), the dose commitment to individuals in the critical group will be calculated and reported in Sv/y under probable conditions in a site-specific biosphere. The same calculations will also be performed, where possible, for less probable scenarios. Where possible, the probability that these scenarios will occur will be estimated. The doses will mainly be compared with the limit value of 0.1 mSv/y.

Population doses integrated over 500 years regionally and globally will also be calculated. These doses will be compared with the announced revised limit values from SSI.

Outflows of radionuclides from the repository

For the time after 10,000 years, inflows of alpha-emitting nuclides will be calculated in Bq/y to the biosphere. The inflows will be compared with natural inflows as well as with limit values, if such exist. To make it possible to compare the calculated releases during the two periods, an indicative dose will also be calculated based on a weighting of released nuclides with the dose factors for a standard biosphere, see further section 11.5.

Doses in the natural environment

The maximum radiation dose in sensitive biotopes or to specific organisms will be calculated or estimated and compared with natural conditions to show that protection of the natural environment is ensured.

Estimates of uncertainties will accompany all reported results.

2.5 HOLISTIC VIEW OF RADIATION PROTECTION

2.5.1 Introduction

Deep geological disposal of the waste from nuclear power generation may make some contribution to the radiation burden on man and the environment. To be able to evaluate this possible contribution it is necessary to take a holistic view of radiation protection, taking into account doses caused by different steps in energy production.

Individual and/or collective doses from the mining of uranium ore, production of uranium, transport, fuel fabrication, operation, interim storage, waste treatment and deep disposal should be reported. The contributions from other activities should be expressed as dose per unit of energy produced. A discussion of the timescales for the radiation risks from the different links in the chain should also be included. Furthermore, the probability and consequences of accidents in different production steps should be estimated. The probability and consequences of possible future evolutionary pathways for a deep repository and similar facilities for long-term storage of radioactive waste should also be discussed.

The results of such a review can then be related to the radiation environment in which man lives today.

One form of integrating the radiological risks from different activities is expressed in Swedish legislation concerning releases during normal operation of nuclear facilities. The dose limit to the critical group in the surrounding area is set there at 0.1 mSv/y, regardless of the number of facilities on a given site.

For many of the activities that are discussed below, the radiological impact can be divided into a) contributions from the operational part of the activity, and b) a long-term passive post- closure phase. Mining is one example of this. First the personnel are exposed to radiation during the operational phase, and then the mining waste poses a long-term radiation risk. Often, measures that entail a higher dose during the operational phase can lead to a reduction of the dose from the post-closure phase. A holistic view of the activities presupposes that a suitable way can be found to balance the risks in these two phases.

2.5.2 Radiological risks in the nuclear fuel cycle

The fact that nuclear power reactors are currently licensed to operate is in itself a justification of the necessary supporting activities for energy production – uranium mining, reactor fuel fabrication and management and disposal of the spent nuclear fuel and other waste arising from energy production. Nuclear energy production entails the following activities:

- uranium mining,
- isotope enrichment,
- fuel fabrication,
- nuclear energy generation,
- management, including possible treatment and conditioning, of radioactive waste, and
- transportation and storage.

These activities are often geographically and chronologically sharply separated and are of a very diverse industrial character. Due to the differences in the activities, the radiological consequences of normal operation and possible accidents are also of a very diverse character. These differences concern:

- the radionuclides that are involved,
- the exposure conditions for operating personnel and surrounding population during normal operation,
- the types of accidents that can occur and the potential risks they can give rise to for the personnel and the surrounding population,
- the chronological distribution of exposure and risk of accidents etc.

Uranium mining

In uranium mining, it is primarily the waste from the mining operations that poses a radiation risk. Production of one tonne generates around 50 tonnes of solid waste. This waste is managed differently depending on where the mining takes place, which means that the risks to which society is exposed also vary. The difference lies in how the waste heaps are covered over. In Ranstad, Sweden, for example, covering and other measures led to a reduction of the releases by a factor of 400 /2.5-1/.

The risk to society stems from the fact that the ore after mining has a larger surface area exposed to groundwater and air. This leads to a faster release of both radionuclides (the daughter products of the uranium isotopes) and other chemical substances such as lead, cadmium, mercury and nitric acid. These substances are transported via suspended matter in the air and via leaching by surface water, possibly to groundwater. The contamination process is also extended over a very long time – up to millions of years. UNSCEAR has estimated the collective dose to personnel in uranium mines to be 150 manSv/GWa /2.5-2/. The dose comes primarily from radon.

Isotope enrichment

Isotope enrichment is a process that produces waste in the form of depleted uranium, which is mainly a chemical risk. In considering the long-run risks, it may also be necessary to take into account the daughter products that are formed in the depleted uranium. A small release of uranium daughters formed during the interval between concentration of the ore at the mine and isotope enrichment also occurs. This interval can sometimes be several years /2.5-2/.

Fuel fabrication

Fuel fabrication is a relatively risk-free process, radiologically speaking. The risks that do exist are in connection with particulate releases, which can marginally affect the environmental dose /2.5-2/.

Energy production

Numerous different radionuclides are produced during energy production. Releases during normal operation are carefully monitored. Larger releases can occur in the event of accidents, but the doses from Swedish reactors will probably never be sufficient to cause acute injury.

Very small quantities of long-lived nuclides are released during normal operation. Since, furthermore, waste management is dealt with separately, the time-scale for releases during normal operation is relatively short, about 40 years. C-14 is an exception, since it is expected to give rise to a few manSv globally during a relatively long time span, about 10,000 years. It is estimated that tritium and Kr-85 will contribute 0.1 manSv/GWa. In the longer term I-129, Ra-226 and Np-237 give collective doses on the order of mmanSv/(y · GWa) /2.5-2/.

The accidents at Windscale, Three Mile Island and Chernobyl are estimated to have contributed 2,000, 40 and 600,000 manSv, respectively /2.5-2/.

Operational waste

In Sweden, waste from nuclear power plant operation is temporarily stored at the plants. The operational waste is conditioned either for final disposal on the site (shallow land burial) or for shipment to the final repository for radioactive operational waste, SFR, for disposal. Any releases from the handling outside SFR are dealt with under energy production. The safety assessments for SFR indicate that the risks of releases are very low and the consequences very small /2.5-3/.

Long-lived waste

Long-lived waste (spent nuclear fuel and reactor internals) are temporarily stored in the central interim storage facility for spent nuclear fuel, CLAB. This step results in relatively small quantities of operational waste, which is dealt with in the same way as the waste from energy production. The further management of the long-lived waste, i.e. encapsulation and in particular deep disposal, is discussed in detail in all of this report. Estimates from UNSCEAR of collective dose from long-lived waste are used for the comparisons made in this section /2.5-2/.

Reprocessing

Reprocessing can be included in the management of spent nuclear fuel. Reprocessing is not a part of the waste management system in Sweden, but is nevertheless being included for comparison. It is estimated that the waste's entire content of Kr-85 and 70–80% of its content of C-14 are released during reprocessing /2.5-1/. This is estimated to contribute 50 manSv globally during a period of about 10,000 years. Reprocessing also produces long-lived waste.

Transportation

The total collective dose for transportation has been estimated at 1.3 mmanSv/TWh, of which accidents account for 10^{-7} manSv/TWh /2.5-4/.

Summary of collective doses

Collective doses from different steps in the nuclear fuel cycle are summarized in Table 2.5-1. To put the results in better perspective, the population doses in the table should be compared with the expected natural annual mean dose to the same populations.

It is also of interest to compare the values with the doses entailed by alternative energy production. Coal-fired energy is estimated to give about 20 manSv/GWa, for example. Releases of Cs-137 from combustion of biofuel are estimated to be about 500 times higher than from nuclear power operation, counted per unit of energy produced.

Table 2.5-1. Summary of collective doses in the nuclear fuel cycle

	Timescale (years)	Global collective dose (manSv/GWa) UNSCEAR
Uranium mining	1,000,000	150
Enrichment	100	
Fuel fabrication	100	0.003
Energy generation	100	1.3
Operational waste	1,000	0.5
Long-lived waste	1,000,000	0.05
Reprocessing	100	50

Permissible collective doses for nuclear energy production

SSI FS 91:5 gives a permissible collective dose for nuclear energy production of 5 manSv per year and GW of installed capacity.

If, on average, 2 manSv per year has been used up during 30 years of operation for 10 GW, this would allow a margin of $3 \cdot 30 \cdot 10$ manSv or a total of 900 manSv that could be spread out over about 10,000 years. On average that would mean about 0.09 manSv/y.

Finally, it can be concluded that there is no clear-cut definition of the term "collective dose" today. A description of the problem is provided in /2.5-5/. One of the factors that needs to be discussed is the time for which the dose is to be summed. In the data from UNSCEAR in Table 2.5-1, the alternative 10,000 years has been used.

2.5.3 Application of holistic view to radiation protection

A possible point of departure for the continued discussions of how a holistic view should be applied within the field of radioactive waste is given below:

1 Management and disposal shall not give a significantly larger dose contribution than other parts of the nuclear fuel cycle – either as individual dose or as collective dose to the public or personnel employed in the activity.

2 The collective doses shall, in the normal case, fall within the guidelines laid down for the nuclear fuel cycle totally for a 500-year period at any given time.

3 The measures included in management and preparations for final disposal of the waste shall be designed to provide a reasonable balance between the risks of dose contribution (irradiation) as a result of conditioning and preparations on the one hand, and the future possible risk that the measure is intended to reduce on the other.

It is not reasonable to take a relatively greater present risk to eliminate a much smaller future risk.

The balance between the risks can be determined by means of quantitative or qualitative analyses.

2.6 CHEMICAL RISKS

The chemical substances left behind in a deep repository may pose a non-radiological risk as well. This is particularly the case for substances belonging to the following categories:

- Heavy metals
- Aromatics and chlorinated aromatics
- Polyaromatic hydrocarbons, PAHs
- Phenols
- Phthalates and surfactants
- Chlorinated hydrocarbons
- Cyanide, fluoride and other salts

To obtain a general idea of the rules and guidelines that exist now and that may be introduced in the future, an inventory has been made of national limit values prescribed by the National Food Administration and the Swedish Environmental Protection Agency, as well as by EU directives and other countries' guidelines /2.6-1/.

The limit values that have been set are based on the existence of a threshold value, which can be defined in different ways:

- concentrations below which no effects have been observed
- the lowest concentration at which a harmful effect has been demonstrated, or
- the concentration for which the risk of cancer is equivalent to an acceptable lifetime risk (the value of one in a million is often used).

Limit values are obtained by dividing the threshold value by a safety factor that is dependent on the type of risk.

For water there are limit values for drinking water from wells and surface water, and discharge limits for water discharged directly to a recipient or via a sewage treatment plant. Limit values and guidelines are not comprehensive, but different types of values are in relatively good agreement and normally vary less than one order of magnitude.

There are no fixed rules at present for levels of contaminants in the ground and in sediments, with the exception of regulatory criteria for soil in The Netherlands. The aim there has been to preserve the multiple functionality of the soil without causing any unacceptable risk to man.

For waterborne discharges to bodies of water, it is possible to compile a set of limit values. It is reasonable to assume that a discharge that lies below these limit values for water can never create an unacceptable contamination situation for soil or sediment.

Hazardous chemical substances are discussed further in section 4.4. Such substances are assumed to be primarily present in the waste that is to be emplaced in the repository sections for "other long-lived waste".

3 METHODOLOGY DESCRIPTION

Chapter 3 presents SKB's methodology for the safety evaluation.

In an introductory section, the historic background of previous design studies and performance and safety assessments associated with them is to be presented together with an overview of how the work has led up to the general design of the system. It must, however, be clearly shown that the safety assessment that is presented evaluates only the safety of the selected and presented system. Options for alternative designs are presented to the extent there is a need to give an account of remaining freedom for design changes or possibilities for adaptation to site-specific conditions.

This is followed by a discussion of the safety-related evaluation required for the decisions in question, and the focus and delimitations of the safety report this leads to. The effect of the delimitations on the safety assessments is discussed, after which the procedure for the performance and safety assessments is presented. The section also will provide an overview of how safety in the post-closure phase is related to the handling of the waste and to the design construction and control of the repository and the safety barriers. The section is concluded with a recapitulation of radiological safety goals and acceptance requirements in accordance with chapter 2 and a presentation of the specific numerical values that are utilized in the present safety report.

Subsequent section will give an account of the methods that have been utilized to systematically analyze the conditions in the deep repository that are important for performance and safety, as well as the events and processes that can influence the evolution of the repository with time. Concepts such as process system, normal scenario, reference scenario etc. are defined. Possibilities for covering reasonable evolutionary pathways for the repository with different scenarios and calculation cases are discussed. Similarly, possibilities for quantifying the different scenarios and how this quantification is dependent on the availability of information and the current stage of the work are discussed on a general level.

The next section examines the uncertainties that enter in during the course of carrying out the safety assessment. Possibilities for quantifying or limiting these uncertainties are discussed, as are the possibilities of integrating them in a total uncertainty estimate.

A concluding section describes the quality procedures that have been applied in the execution of the particular safety assessment.

***In this report**, the three levels of safety functions that build up the safety of the deep repository and the fundamental procedure followed in carrying out a safety assessment are presented in section 3.1. The purpose of the two safety reports next in line, SR-I and SR-D, is then discussed in section 3.2, along with the preliminary scope and limitations of these reports.*

Section 3.3 gives an account of the development of scenario methodology that is currently being pursued within SKB. An application of this methodology to the system design described in SR 95 is described in chapter 9.

Section 3.4 discusses the uncertainties inherent in a safety assessment. The chapter comprises a status report on the work of systematically examining all uncertainties. The intention is to show how different uncertainties enter the analysis, examine ways to quantify or delimit them, and discuss the possibilities for integrating the different types of uncertainties.

All of the afore-mentioned sections in SR 95 thus constitute more general surveys of the safety assessment methodology than the corresponding sections in coming safety reports.

3.1 GENERAL ABOUT PERFORMANCE AND SAFETY ASSESSMENTS

To achieve the desired safety during the construction of a deep repository, during the operating phase and during the long-term containment phase (post-closure), requirements are made on the performance of the repository and its components. The composite or integrated performance of all the repository's components must together provide adequate safety.

In order to achieve long-term safety, the repository system is designed to **isolate** the spent nuclear fuel from the biosphere. This isolation is achieved by encapsulating the spent nuclear fuel in tightly sealed canisters which are deposited deep in the crystalline bedrock on a selected repository site. If this isolation should be breached, the repository has the added function of **retaining** the radionuclides and **retarding** their transport. Furthermore, transport pathways and dilution conditions in the biosphere can be influenced by siting so that any radionuclides that escape will only reach man in very small quantities.

The materials used in the repository have been selected in such a way that experience from nature could be utilized to support the assessment of their long-term stability and the safe performance of the repository. For the same reason, the thermal and chemical disturbance which the repository is allowed to cause in its surroundings is limited. The safety philosophy for the deep repository is based on the multibarrier principle, i.e. safety must not be dependent on the satisfactory performance of a single barrier.

Safety functions

The safety functions are influenced by site selection and layout and by the design and dimensions of the engineered barriers. The safety functions can be divided up into three levels:

Level 1 – Isolation

As long as the waste is isolated, the radionuclides can decay without coming into contact with man and his environment.

Level 2 – Retardation

If the isolation is breached, the quantity of radionuclides that can reach the biosphere is limited by:

- very slow dissolution of the spent fuel,
- sorption and very slow transport of radionuclides in the near field,
- sorption and slow transport of radionuclides in the bedrock.

Level 3 – Favourable recipient conditions

The transport pathways along which any released radionuclides can reach man are controlled to a great extent by the conditions at the locations where the deep groundwater first reaches the biosphere (conditions such as dilution, water use, land use and other exploitation of natural resources). A favourable recipient means that the radiation dose to man and the environment is limited. The recipient and the transport pathways are, however, influenced by natural changes in the biosphere.

Safety assessments

The performance of the repository under both anticipated and less probable conditions is analyzed by means of performance and safety assessments. In a phase where system development and the site selection process are under way, continuous evaluations of safety and performance are required as guidance for designing the systems, for selecting the disposal site and for arranging the repository so that the natural barriers against radionuclide transport on the site are effectively exploited. Such recurrent assessments of suitably delimited barriers or subsystems under typical ambient conditions are called **performance assessments**.

To get a whole picture of the overall performance of the repository system, the performance assessments must be integrated to cover the entire system. The possibilities of various external influences must also be analyzed. Finally, the performance of the repository must be compared with the safety goals. These integrated assessments are termed **safety assessments** and are presented in safety reports.

The point of departure for the safety assessments is the understanding of the performance of the system that has been obtained in the performance assessments or in previous safety assessments. They are focused on clarifying effects of the total repository on man and his environment and on how different designs and dimensions affect the integrated safety of the repository.

Many decisions concerning the direction of the work etc. in the development of a repository system must be taken in the face of uncertainty. The analysis work must therefore not only quantify the performance of the repository, but also clarify the uncertainties in the assessment and how these uncertainties affect the conclusions. Investigation of the uncertainties in the assessment, and of confidence in the fact that the purpose of the assessment can be achieved taking into account the uncertainties, is an activity that in practice cannot be separated from the rest of the analysis work.

Regardless of whether the assessments are carried out for all or parts of the system, or whether they are carried out at an early or late phase of development, they must be done in a systematic fashion. A schematic presentation of the systematics is shown in Figure 3.1-1. A complete assessment includes:

- Definition of the purpose of the assessment.
- Description of given assumptions for the assessment, i.e. types and quantities of radioactive waste, the repository system and its dimensions, and the location and surroundings of the repository.
- Description of the scope and delimitations of the assessment, and of safety goals and acceptance criteria.
- Qualitative description of the processes which together describe the performance of the repository under different conditions. Choice of both the probable and less probable or improbable conditions for which the system/facility is to be assessed (scenarios).
- Detailed description of the time-dependent processes which are essential to the intended performance of the repository in different scenarios.
- Choice of analysis method for each scenario. Possible choice of calculation models for quantitative analyses.
- Execution of quantitative or qualitative analyses for the chosen scenarios.

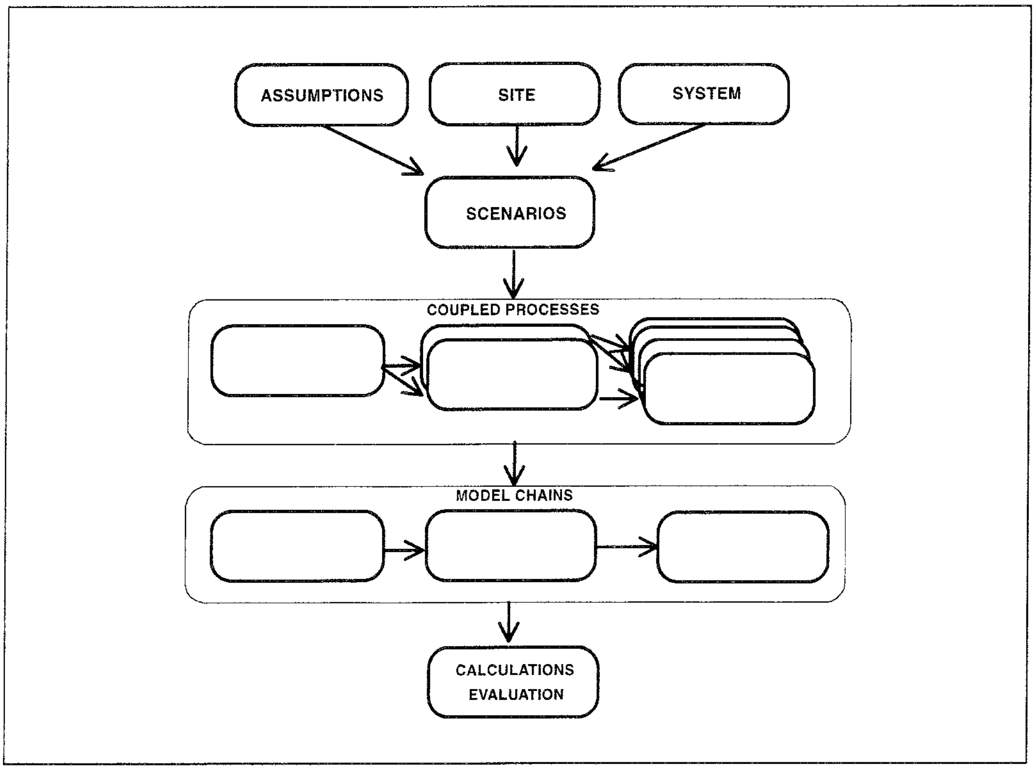


Figure 3.1-1. *Schematic procedure for the execution of a safety assessment. Scenarios describe the possible evolution of the repository with time. The processes that influence safety are often coupled and therefore require large model chains to be set up.*

- Discussion of the uncertainties in the qualitative and quantitative analyses made and an assessment of whether the purpose of the overall assessment can be achieved, taking into account the uncertainties.

To carry out the work in a systematic and traceable fashion, methods for the different steps have been discussed and developed in Sweden, in cooperation with the major nuclear power countries and in international cooperation. The more important of the methods for scenario handling are discussed further in section 3.3.

3.2 SCOPE AND DELIMITATIONS OF THE ASSESSMENT

A number of safety reports will be submitted in support of decisions and permit applications at various steps in the work of building a deep repository in Sweden. Table 3.2-1 is taken from SKB's RD&D-Programme 95 /3.2-1/.

Each of the safety reports mentioned in the table will have a principal focus or specific delimitations that can differ in certain respects from other reports. In order to prevent misunderstandings from arising in connection with comparisons between the reports or with the report template, it is important that such delimitations be defined and the reasons for them given.

A general delimitation is that each safety report deals with a given defined system design or layout. Sensitivity analyses can demonstrate the importance of minor changes in certain dimensions or properties in the vicinity of the result, but an individual analysis cannot be used for an optimization of the repository. Other general delimitations or focuses may be concerned with the time spans to be covered by the assessments or the specific safety assurance requirements set forth in directives etc. These have been discussed in chapter 2.

Table 3.2-1. Forthcoming safety assessments.

Safety assessment	Background material for decision on permit for: Encapsulation plant	Deep repository
Encapsulation plant SR-I	Siting Construction	
Deep repository SR-D		Siting Construction, incl. detailed geoscientific investigations and some construction
Initial operation (Stage 1)	Start encapsulation – spent fuel Initial operation – stage 1	Initial operation – Stage 1 – deposition, progressive excavation of deposition tunnels
Regular operation (Stage 2)	Supplementary extension Regular operation – stage 2	Regular operation – Stage 2 – deposition, progressive excavation of deposition tunnels
Decommissioning	Decommissioning	Possible supervised storage, closure

Safety assessments that are carried out in different phases of system design, site selection and excavation may thus differ in scope and have different delimitations. Specific delimitations with respect to differences in the underlying data may, for example, occasion the modelling of rock volumes of different sizes. Delimitations and focuses stemming from the fact that assessments are done in early phases can warrant limiting the scope of the assessment to simplified or conservative descriptions or models. Complex models intended to be used in later phases can, however, also be utilized earlier in order to permit an analysis of the dependence of different models on data availability.

Site-specific conditions may entail that the geometric delimitations in the modelling of groundwater or radionuclide transport are chosen differently on the candidate sites.

However, the clearest difference between different safety reports probably stems from the fact that the decisions to be made in different phases are different. To shed light on the dissimilarities occasioned by differences in the decisions to be taken and the need to adjust the scope of the report to these decisions, the focus and delimitations of the two reports next in line concerning the long-term safety of the repository are discussed below.

Safety report for permit for Encapsulation Plant, SR-I

The report deals with the canister's long-term safety function, to isolate the spent fuel from groundwater and to retard the transport of radionuclides should this isolation be breached. Safety is evaluated under both anticipated and reasonably unfavourable conditions in the repository. An account of long-term safety for other waste types admitted to the deep repository will be given in connection with permit applications for the deep repository. SR-I will also describe what environmental conditions were assumed to exist when the functional requirements for the canister were determined. These will be compared with the conditions that exist in the bedrock being considered to host the deep repository.

Since the siting process is not expected to have come to the stage that data from candidate sites is available, the safety assessments will be based on typical conditions in Swedish crystalline bedrock. Geochemical conditions in the deep groundwater will be selected with reference to the general knowledge of groundwater in Swedish crystalline rock and with reference to SKB's experience from previous study sites as well as Stripa and Äspö. The safety- and construction-related siting factors that were presented in the supplement to RD&D-Programme 92 /3.2-2/ will be described in greater detail. The sensitivity of the safety assessment to variations in these factors will be elucidated.

Focus and delimitations for SR-I:

- The assessment pertains to the deep repository for spent nuclear fuel, not the repository for other long-lived waste.
- Canister data are taken from the designing of the encapsulation plant.
- Buffer and near-field data are taken from the designing of the deep repository.

- The far field is exemplified with data from one of SKB's investigated sites; the range of variation of essential data in Sweden is reported.
- The biosphere is integrated in the safety assessment via typical biosphere recipients.
- The safety-related importance of the variation in geodata is covered via the range of variation of the parameters.

Safety report for permit for Deep Repository and detailed characterization, SR-D

The report is supposed to present a site-specific assessment of the safety of a deep repository situated on the candidate site recommended for detailed characterization. Only sites with good safety-related potential will be considered for siting.

An initial review and comparison of candidate sites takes place in the site characterization work and associated performance assessments. If the sites are equivalent, they are ranked on the basis of other siting factors described in the supplement to RD&D-Programme 92.

The SR-D safety report is supposed to demonstrate an understanding of the role of the site for long-term safety, i.e. to provide favourable conditions with regard to mechanical stability and chemical environment for the canister and buffer, and to limit the transport of radionuclides to the biosphere.

Focus and delimitations of SR-D:

- The assessment pertains to both the repository for spent fuel and the repository for other long-lived waste.
- The assessment focuses on site-specific conditions and groundwater movements on the candidate site.
- The repository design and layout are made site-specific.
- Biosphere recipients and transport pathways to man are chosen site-specifically.
- The safety-related significance of uncertainties or alternative interpretations of the geological structure of the candidate site is examined and the site's potential for safe final disposal is evaluated taking into account remaining uncertainties.

3.3 SCENARIO METHODOLOGY

3.3.1 Introduction

The deep repository consists of a chain of barriers – engineered, i.e. man-made, and natural. The function of the barriers is to isolate the radioactive waste (see also chapter 9). Besides isolating the waste, the barriers are designed so that dissolution and transport of radionuclides is prevented and/or retarded. The deep repository is designed to isolate the radioactive waste until its radiotoxicity has declined to a level that is comparable to the radiotoxicity of a natural uranium deposit.

The purpose of the safety assessment is to determine whether the repository system meets the stipulated criteria for all realistic states of the system. This requires tools/models to simulate and analyze the behaviour of the repository. An important part of the safety assessment is to compile information on the characteristics and performance of the repository and to identify the system states that should be simulated and analyzed. The methodology used for this purpose is called scenario methodology.

A scenario is a description of a hypothetical future situation. The concept of a scenario includes both a course of events emanating from a set of specified premises and the future situation this course of events leads to. The situations can be described here as states in the repository's barriers and flow systems and in the biosphere. The states should be such that they could somehow affect the performance of the barriers and/or the transport of radionuclides to man. A scenario may also have its starting point in unrealistic assumptions, such as the exclusion of a barrier or a process. The purpose of assessing such a scenario is to demonstrate the robustness of the system or its sensitivity to uncertainties in barrier performance and processes.

The chosen scenarios should together provide a reasonably comprehensive picture of the possible evolutionary pathways of the system. To identify these scenarios it is necessary to adopt a systematic approach, a scenario methodology. The choice of scenarios should be made on the basis of a systematic description of the system and its evolution with time. An important part of the scenario methodology is to offer a procedure for developing such a systematic description.

It is not deemed possible for scenario selection (screening) to be done on purely objective grounds. It must include some measure of subjective judgements in the form of expert opinions. The scenario methodology must handle and document these opinions in a systematic manner.

The entire process of developing a description, scenario selection and describing and justifying the selected scenarios must be documented. A documentation plan is also included in the scenario methodology.

Scenario methodology is still under development. So far development has taken place by trying out a number of methods. An initial attempt to identify scenarios using fault tree analysis was made in 1989 jointly by SKB and SKI /3.3-1/. Since then, the influence diagram method /3.3-2/ and the RES method /3.3-3 – 3.3-5/ have been tried within scenario methodology. During the course of the work, shortcomings and merits have been identified and lessons have been learned from the experience /3.3-6/. Some of the central concepts in the scenario methodology employed today are a legacy from previously tried methods.

3.3.2 Important concepts

FEP

A difficult question, which must be partially answered by means of the scenario methodology and its application, is whether present-day knowledge is sufficient for describing the performance of the repository. Have all the possible evolutionary pathways of the repository really been considered? Have all the features, events and processes in the repository been taken into account? To answer these questions, an attempt has been made to identify the features, events and processes (FEPs) that influence repository performance and might possibly occur now or in the future.

Databases of FEPs have been built up both in Sweden and internationally. Identification of FEPs, couplings between them and the large and complex quantity of information contained in the FEP databases have dominated the scenario methods hitherto employed.

Process System, PS

A need to distinguish those FEPs that describe the characteristics and chronological evolution of the repository from those that describe external events or initial conditions was identified at an early stage. The concept of a Process System, PS, was introduced to accomplish this sorting task, with the following definition:

> The Process System is the **organized** assembly of all phenomena (FEPs) required for the description of barrier performance and radionuclide behaviour in a repository and its environment, and that can be predicted with at least some degree of determinism, given an assembly of external conditions.

A PS is thus an integrated description of the repository's central features and processes. General conditions in the development of a PS provide an understanding of how the system works under various circumstances.

When a description of a PS has been formulated, the FEPs that do not belong to the PS are regarded as external events or phenomena that can affect the PS. Such external FEPs can thus be said to describe the premises for a given scenario.

Scenario

Against the background of what has been said above about the process system, the following is a natural definition of a scenario:

> A scenario is defined by a set of external conditions which will influence processes in a PS. The external conditions determine how the processes in the PS are to be combined and modelled in describing the evolution of the scenario and evaluating its consequences.

The central part of the scenario, the description of the hypothetical future, comprises the premises for its evolution. For this reason, when speaking of the selection of scenarios for a safety assessment, what is often meant is the selection of premises for scenarios. In the following, the term "scenario selection" will often be used in the sense "selection of premises for scenarios".

Reference scenario

Before a description of a PS can be formulated, certain premises have to be established. The concept of a reference scenario has been introduced for this purpose. A reference scenario is a scenario that has been chosen to have for the sake of comparison. In SKB's current methodology, the reference scenario has generic premises. It provides a generic description of a PS. The particular features of many other scenarios can be illustrated within such a description of a PS.

A need to portray PS features and processes in a schematic and illustrative fashion and to link them together in cause/effect relationships was identified early. Different types of diagrams were tried for this purpose and the interaction matrices of the RES method were considered to be best suited to the purpose /3.3-6/. The RES method has therefore had a great influence on the scenario methodology that is employed today by SKB. The RES methodology and the structure of an interaction matrix are presented briefly in the next section. The concluding section describes the structure of the scenario methodology that is employed today.

3.3.3 RES methodology

RES is an acronym for Rock Engineering System. RES is described as an objective-based top-down method. By "objective-based" is meant that the analysis shall be carried out against the background of a defined objective. By "top-down" is meant that the problem is first regarded as a whole and is then broken down into its constituents. This section provides a very brief description of the RES method. Sections 9.3 and 9.4 describe how it has been applied within the scenario methodology.

The basic tool in the RES method is the interaction matrix. The diagonal elements in the interaction matrix consist of the concepts or physical variables that are judged to be important for the problem to be analyzed. The other elements describe the interactions between the diagonal elements. The interaction direction is clockwise, see Figure 3.3-1.

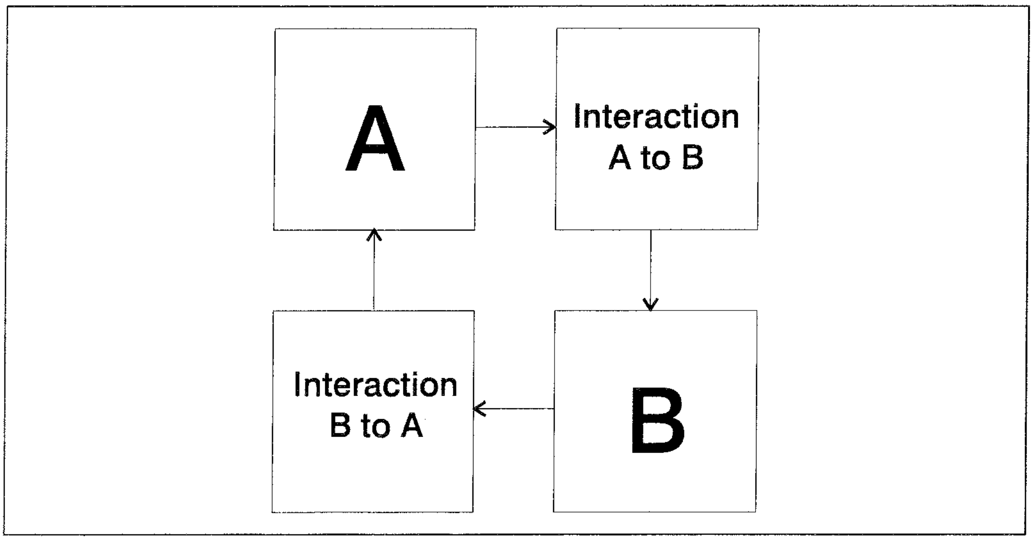


Figure 3.3-1. *Interaction matrix with the diagonal elements A and B and the interactions between them.*

An interaction between two diagonal elements is called binary. If the interaction matrix consists of more than two diagonal elements, a collection of binary interactions may sometimes exist in such a way that chains of interactions, called interaction pathways, can be built up. The chains may sometimes be closed. This requires binary interactions to occur on both sides of the diagonal, see Figure 3.3-2.

The RES method consists of the following steps:

- Formulation of problems, objectives and premises for the analysis.
- Choice of concepts or physical variables to constitute diagonal elements, plus their units of measure. If the diagonal element cannot be described with an absolute measure, the aspects of the element that are modelled are described. If the diagonal element in itself represents several variables with the same name and/or units of measure, this is described.
- Determination of interactions between the diagonal elements.
- Determination of how important each interaction is with regard to the formulated objective and the process it represents. The importance or significance of the interaction is determined by assigning it a numerical code as follows:
  4 critical interaction
  3 strong interaction
  2 medium interaction
  1 weak interaction
  0 no interaction
- Establishment of mathematical relationship between the diagonal elements, if possible.

If quantitative relationships can be established between the diagonal elements, the matrices can be analyzed mathematically. Within the scenario method that is employed today it has been deemed unrealistic to establish quantitative relationships between all diagonal elements. For a description of how the analysis

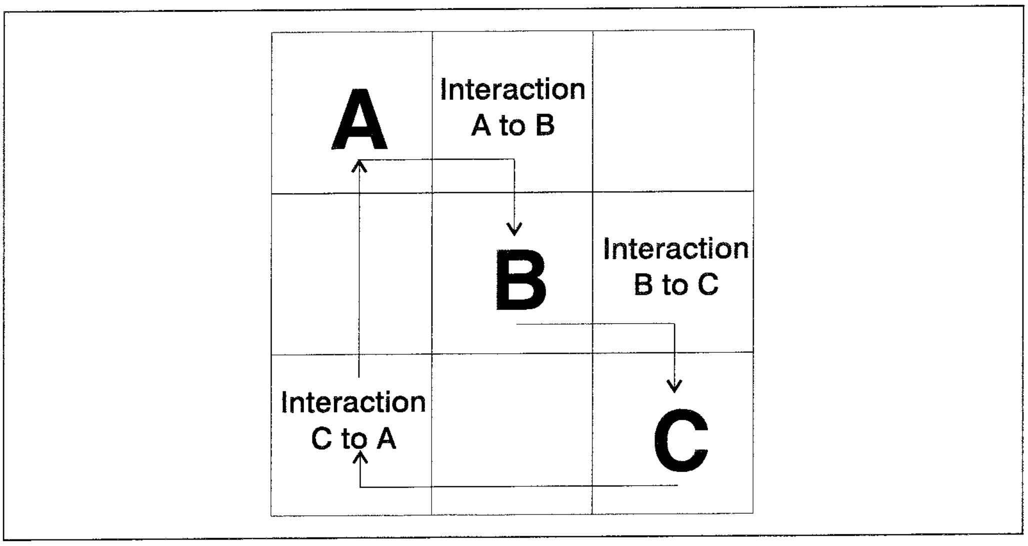


Figure 3.3-2. *Chain of interactions that form a closed cycle.*

is done in such a case, the reader is therefore referred to /3.3-3 – 3.3-5/. When quantitative relationships cannot be established, the significance or degree of importance of the interactions is utilized in the analysis. The importance of the diagonal elements for the system is analyzed via the line and column sums of the interaction codes. The line sum is a measure of the degree to which the diagonal element influences the rest of the system. The column sum is a measure of how sensitive the diagonal element is to perturbations from the rest of the system.

The scenario development is described by starting from a special system state. A perturbation influences the matrix which describes the system state e.g. by change of a diagonal element. The interactions that emanate from the diagonal element are re-evaluated. The impact on affected diagonal elements is analyzed. The interactions that emanate from the affected diagonal elements are re-evaluated, and so on. The most important, critical interaction pathways are identified as those that consist solely of critical interactions. The most critical interaction pathways are considered to be those that can be linked to a loop of critical interactions, Figure 3.3-3. The new system state, in particular the critical interactions, is described more fully.

The RES method with its interaction matrices is a very powerful tool for identifying and describing FEPs and their links to each other. However, in its original form the method lacks definitions, methods and notation for describing the repository as a system composed of system parts which, through their properties and performance, support the performance of the system, and how the system evolves with time. The performance assessment is done today by experts who look for premises that can affect the system's performance.

A	2		4	4
4	**B**	3	1	2
2	1	**C**	4	1
	4		**D**	3
2		2	3	**F**

Figure 3.3-3. *Very critical chain of interactions in an RES matrix.*

3.3.4 Scenario methodology applied today

SKB's scenario methodology as it has been developed up to today includes the following steps:

- A systematic description of features, events and processes that can influence the performance of the repository and their coupling to each other by the construction of a number of interaction matrices.
- A method for documentation of the interaction matrices, including coupling to existing FEP database.
- A plan for how expert opinions are to be handled in connection with the documentation of the interaction matrices.
- Selection of premises for scenarios.
- Description and justification of scenarios.
- Qualitative analysis with the aid of the interaction matrix. Documentation of how the interaction matrices change for different scenarios.

Construction of the interaction matrices has two main purposes:

1 to show that no FEPs have been overlooked, and

2 to identify important interactions and interaction pathways.

Furthermore, the process of constructing the matrices is considered to lead to a better understanding of the system and to facilitate communication between different experts.

In conjunction with the construction of interaction matrices, the diagonal elements and the interactions are described and documented. The documentation includes a brief description, couplings to some existing FEP database and description of whether, and if so how, the diagonal element or the interaction can be modelled in the quantitative evaluation within the safety assessment. Furthermore, the person or persons responsible for the description and their qualifications are given.

Interaction matrices for PSs cannot be constructed until external premises have been chosen. The idea is that the matrices should be able to be used for a large number of scenarios, so the premises for the matrices are chosen to be as widely applicable as possible. The intention is also that the matrices should depict the normal function of the system. The matrices describe PSs for a reference scenario. The definitions of the significance of the interactions that are provided within the RES method must be clarified. The interactions that are given the highest significance, i.e. critical interactions, are ones that are to be handled in the quantitative evaluation of the repository, i.e. to be included in some way in the chain of calculation models that are used in the quantitative evaluation. The interactions that are given the lowest significance are ones that can be ignored. The intermediate steps, strong and medium interaction, are ones that must be taken into consideration but do not necessarily have to be included in the quantitative evaluation.

Scenario selection cannot be done directly from the interaction matrices. The interaction matrices are used to describe how the diagonal elements change and the significance of the interactions changes when one set of premises is exchanged for another. The assessment of the system's performance and evolution is carried out entirely by experts within the scenario methodology employed today. The choice of premises that can influence the system, i.e. scenarios, is also done by experts. The interaction matrices act as checklists for the experts once the scenario selection has been made and to point out the interactions that are important in the selected scenario. The interaction matrix can indirectly support the scenario selection, since the process of constructing it provides knowledge of the system.

Chapter 9 describes how the method is applied in practice. It should be pointed out once again that the scenario methodology is under development. The result of the application of the RES method to the repository system will be scrutinized and evaluated.

3.4 UNCERTAINTY AND VALIDITY

3.4.1 Introduction

Confidence in a safety assessment is a product of confidence in a series of both qualitative and quantitative factors that are included in or comprise premises for the assessment. These include everything from overall qualitative aspects regarding how well science today is capable of describing important processes in the performance of the repository to detailed quantitative uncertainties, for example in values of different parameters that comprise input data in a model calculation.

In this section a systematic survey is made of different types of uncertainty and how they enter into the different steps of the safety assessment. The survey is of a general nature. More detailed discussions of uncertainties linked with different repository parts or processes are held in those sections of the safety report that deal with that particular part or process. The section concludes with a discussion of how the uncertainties can be weighed together to give an integrated picture of the confidence in a safety assessment and its results. A more detailed discussion of uncertainties in the safety assessment and methods for handling them is held in /3.4-1/. Like scenario methodology, methods for handling uncertainties are under development.

3.4.2 Uncertainties in the premises of the assessment

The premises for a safety assessment consists of a body of knowledge gathered from various scientific disciplines relevant to the understanding of the evolution of the repository together with a description of the repository system.

Uncertainties in the description of the repository system

The description of the repository system can be divided into

- a description (characterization and quantities) of the waste to be disposed of,
- a given repository design, i.e. a description of the system of barriers that is supposed to prevent radionuclide release and migration, and

- a description of a specific site on which the repository is intended to be located.

A detailed description of the repository system containing numerous items of information that are used as input data in the safety assessment.

The characteristics and quantity of the waste are determined by, among other things, the scope of the Swedish nuclear power programme, both historically and in the future. The characterization contains information on fuel type, degree of burnup, chemical composition, etc. Uncertainties in the description are discussed in chapter 4. With today's premises for the Swedish nuclear power programme, the quantitative uncertainties can be assumed to decrease with time. Furthermore, these uncertainties can often be dealt with by conventional statistical methods. The uncertainties in the description of the physical and chemical properties of the fuel are of a more qualitative nature and more difficult to treat. The description serves in the safety assessment as a basis for a calculation of a detailed radionuclide inventory and for modelling of changes of the fuel properties with time, dissolution, etc.

The repository design with its different barriers (fuel matrix, canister, buffer, near-field rock) is described in detail mainly in chapter 5. This description includes a number of parameters, all of which are associated with uncertainties. The uncertainties do not stem from different design-related options but rather from the degree of certainty with which different parameters in a chosen design can be determined. The uncertainties related to the repository design are also for the most part quantitative in nature and can often be dealt with by statistical methods. The uncertainties are often determined by the quality achieved in different production and inspection processes. The properties of the copper canister and the bentonite buffer are examples of such parameters.

The description of the **specific site** on which the repository is intended to be located consists of **a geosphere part and a biosphere part**. As far as the geosphere description is concerned (chapter 6), the analysis is based not only on data from geological investigations but also an **interpretation** of these data, resulting in a description of the geosphere. Such a description includes e.g. large structural elements in the rock, major individual fractures, fracture zones, rock type distributions and topography. The uncertainties in some of these factors can be considerable and difficult to characterize. Besides the fact that available geological data are associated with uncertainties that result in uncertainties in a given interpretation, alternative interpretations are sometimes possible for one and the same set of geological set of data. The evolution of the geosphere with time in the form of, for example, rock movements also has to be described. The geosphere description with its uncertainties serves in the safety assessment as a basis for a mathematical model of the geosphere which is used to calculate groundwater flow and radionuclide transport.

A characterization of the biosphere on the particular repository site (chapter 8) covers groundwater recipients, structure and fluxes in the ecological system as well as man's utilization of the natural environment. Changes in the biosphere with time also need to be estimated. The characterization of the biosphere with its uncertainties serves in the safety assessment as a basis for a mathematical model of transport and flux of radionuclides in the biosphere.

Quality of the knowledge base

Knowledge from a large number of scientific disciplines is used in a safety assessment. This knowledge can be said to be one of the basic prerequisites for an assessment. The principal fields include different branches of physics, chemistry, biology, geology and mathematics.

The scope of the scientific knowledge and the varying degree of maturity of different scientific fields thus affect the quality of the safety assessment. This is naturally difficult to express in a precise way. But we always have to be aware of the fact that there are deficiencies in our present-day knowledge, and this constitutes an element of uncertainty in the assessment. It manifests itself, for example, in the discussion of completeness in the following scenario description.

3.4.3 Qualitative description, scenarios

Based on the premises described in preceding sections, a general and qualitative summary of the features, events and processes that influence the evolution of the repository in different ways is made in the safety assessment. In the methodology applied today in SKB's safety assessments, this information is assembled into a process system, which can be schematically depicted in matrix form. In its basic form, the process system describes the evolution of the repository under general and probable circumstances. The designations normal or reference scenario are used for this purpose.

Other scenarios are defined by choosing other, less probable circumstances for the description, see further chapter 9. This leads to a modification of the process system and thus to a change in the description of the expected performance of the repository. In other words, a scenario can be said to be an integrated picture of the evolution of the repository, given a defined set of circumstances. The description is obtained by studying the process system after the circumstances have been defined. In a safety assessment, a selection of scenarios is made that is supposed to provide a reasonable coverage of the repository's different evolutionary pathways.

Uncertainties in the qualitative description and the scenario selection

As far as the process system is concerned, the question to be faced concerns the completeness of the description. Have all relevant processes been included? Have the processes and their interdependencies been correctly understood and described? This question ties in with the discussion of the level of maturity reached by different scientific disciplines which was described among the premises for the safety assessment.

This kind of qualitative uncertainty concerning the completeness of the process system can be partially reduced by adopting a methodical approach in the work. Furthermore, results achieved with different methods can be compared. It can also be valuable if results from different independent expert groups, working with the same or different methods, can be compared.

The choice of scenarios made by expert groups is also associated with uncertainties of a qualitative nature. Does the selection of scenarios give a comprehensive picture of possible evolutionary pathways for the repository? A deeper discussion of this is held in section 9.5.

3.4.4 Modellings

Within the framework of the selected scenarios, a number of calculation cases are selected to analyze repository performance quantitatively. Relevant parts of the process system are modelled mathematically. The first step in the formulation of a mathematical model can be said to be a conceptualization, i.e. an identification and description of the scientific laws and concepts on which the model is based. Conceptualization also includes concepts of a more mathematical nature, for example linearizations of non-linear processes and other mathematical simplifications or abstractions.

Since the current understanding of the processes to be modelled is seldom complete, there is an element of uncertainty in the conceptualization. This can mainly be handled by means of **model validation**, i.e. the predictions made by the model are compared with the part of reality (experimental data) it is supposed to represent. Model validation in different forms is the principal method used for handling conceptual uncertainty, for example by discriminating between alternative conceptual models.

For this purpose, the conceptual model first has to be translated into a mathematical model, which is usually expressed in some form of computer code. There is an element of uncertainty in this process as well, which can be handled by means of **verification** in the form of different tests to make sure that the computer model calculates correctly.

Conceptual uncertainty can sometimes be handled by utilizing concepts that are unrealistically unfavourable (pessimistic) with respect to the safety of the repository. For example, phenomena that have favourable effects on the safety of the repository can be neglected. An example of this is that the barrier function of the Zircaloy cladding tubes that surround each fuel rod can be neglected. In this way, conceptual uncertainty concerning the processes of dissolution of Zircaloy can be circumvented. Such a **conservative treatment** does not actually reduce the uncertainty of the conceptualization, but it does reduce uncertainty for the purpose of the safety assessment – to show under what circumstances the repository is safe.

The models that are used in the safety assessment are verified and validated. The results are described in validation documents with a set format. The validity of different models is discussed more concretely in chapter 11.

3.4.5 Calculations, input data

The identified calculation cases are carried out with the use of verified models, often with several models linked in chains, see section 12.3 for an example. Input data for the models is largely obtained from the description of the repository system that comprises the premise for the analysis. These data are subject to different types of uncertainty. Two principal sources of uncertainty are limited measurement accuracy and variability.

The accuracy of the value of a parameter is limited by the sophistication of the methods available to measure the parameter in question, the accuracy of the measuring instruments, etc. By **variability** is meant the variation of the value of a given parameter in space or time. Even if there is very great accuracy in the determination at a given point in space or time, there is uncertainty as to the value of the parameter at other points. Naturally, an uncertainty can be caused by a combination of lack of accuracy and variability.

To the extent uncertainties in the input data to a model can be quantified in the form of stochastic distributions, it may also be possible to express the calculation result as a stochastic distribution, i.e. uncertainties in input data are reflected quantitatively in uncertainties in the calculation result. If such a **probabilistic treatment** cannot be carried out to its full extent, it is often possible to specify the result as an interval whose breadth is related to the uncertainties in the input data.

The importance of the uncertainty of a given parameter for the calculation result can also be studied by **sensitivity analysis**, i.e. by allowing a parameter to vary around a given value and studying the resulting variation in the calculation result.

Sensitivity analysis is closely related to **variation analysis**, where a parameter is allowed to assume a series of different, not necessarily close, values and the effect on the result is studied.

Numerical uncertainties can also be handled by assigning a value that is unfavourable from a safety viewpoint to a parameter that is associated with uncertainties. This is a way of ensuring that the predictions of the safety assessment regarding repository performance will not be overly optimistic, at the same time as the uncertainty is handled. However, this may give rise to the problem that the final result of the calculation is so heavily influenced by "unfavourable" values that the influence of other factors on repository safety becomes unclear.

3.4.6 Integration

A safety assessment thus contains both qualitative and quantitative uncertainties. The weighing-together or integration of uncertainties in the results of the assessment therefore requires a discussion concerning the qualitative uncertainties as well as a presentation of numerical uncertainties in the calculation results.

Uncertainties must always be discussed in the light of the purpose of the assessment. The degree of uncertainty that can be accepted in the result of a safety assessment of a deep repository is in part dependent on at which of the different phases of planning, design, engineering, construction, commissioning and sealing the safety assessment enters in. A greater uncertainty in the description of the site of the deep repository can, for example, be accepted in early planning stages when the site has not been as thoroughly investigated or even finally established. However, the uncertainties in each phase must be shown in a suitable manner to be of such a nature and magnitude that they do not prevent the work on the repository from proceeding.

The **qualitative uncertainties** in the assessment can be presented by means of a discussion of the ability of the employed methodology to provide a comprehensive picture of the process system and of the possible evolutionary pathways of the repository. Evaluations of the level of maturity of relevant scientific disciplines also enter into such discussions. The choice of scenarios and calculation cases also needs to be justified by argumentation, as does the degree of validation that has been possible to achieve for the calculation models used. Ultimately, however, it is difficult to obtain a strict evaluation of the qualitative uncertainties. It will, for example, never be possible to prove the completeness of the process system in a strict sense. The ambition in the discussion of the uncertainties must instead be to show that the knowledge is sufficient for the purpose of the assessment.

With the discussion of qualitative uncertainties as a background, **quantitative uncertainties** can then be dealt with in the evaluation of the assessment. Given certain numerical uncertainties in the input data to the calculations, it is possible, as described above, to express quantitative uncertainties in the calculation results. The final result of the calculations is generally a measure that can be compared with some established acceptance criterion. If the uncertainties in the calculation result can be expressed in the form of a stochastic distribution, a confidence interval for the result can be calculated.

Accordingly, a properly conducted and balanced discussion of the different types of uncertainties should provide a good idea of the confidence in the safety assessment and its results, both qualitatively and quantitatively.

4 SPENT NUCLEAR FUEL AND OTHER LONG-LIVED WASTE

Chapter 4 gives a description of the spent nuclear fuel and other waste types that will be deposited in the deep repository. The chapter is divided into three sections covering spent nuclear fuel, other long-lived radioactive waste and other toxic waste. The description pertains to the physical and chemical form of the different waste types, as well as quantities and contents of important radionuclides.

The section on spent fuel also contains a discussion of the structure of the fuel: cracks in fuel pellets; the size distribution, surface area and porosity of the fuel fragments; and the properties of the fuel-clad gap. In addition, the distribution of fission and activation products inside the fuel and in the structural elements encapsulated in the canister is discussed.

Models for and calculations of nuclide inventories and decay heats are presented, for both spent fuel and other long-lived waste, as are the selection criteria used to determine which radionuclides are to be included in the analysis.

***In this report**, the reported fuel quantities and burnups have been taken from PLAN 94. Equivalent information for other long-lived waste and other toxic material is based on the pre-studies of these waste types being conducted within SKB. The material may be affected by coming energy decisions.*

4.1 INTRODUCTION

Spent nuclear fuel and long-lived low- and intermediate-level waste are the two waste categories that will be disposed of in a deep repository. Quantities, compositions and other data of importance for an assessment of the long-term safety of a deep repository are discussed in the following for the two waste categories. The toxic or polluting properties of materials disposed of in the deep repository that are not related to radioactivity are described in brief at the end of the chapter.

4.2 SPENT NUCLEAR FUEL

Spent nuclear fuel comprises most of the waste that is to be disposed of in a deep repository. Data on the quantities and composition of the radionuclides in the waste are needed in order to carry out a safety assessment. Information on the chemical and physical structure of the waste is also needed for a safety assessment.

The step-by-step procedure for arriving at the necessary data from such sources as operating data for the Swedish nuclear power programme is shown in the following.

The appearance of the fuel assemblies is touched upon in several places in the following sections. Figure 4.2-1 shows a picture of a fuel assembly.

4.2.1 Quantities and burnups

Each year, SKB submits figures on actual and expected production of energy and waste in the Swedish nuclear power programme in an annual PLAN report. The figures are based on a number of fundamental facts and assumptions concerning the reactors' operating time and availability. The figures for fuel quantities and fuel burnup in SR 95 are taken from PLAN 94 /4.2-1/.

In PLAN 94, the total fuel quantities are calculated for three different alternatives for the Swedish nuclear power programme, Table 4.2-1. Total energy production is estimated to be 2000 TWh if all reactors are shut down in 2010, 1610 TWh for an operating period of 25 years and 2620 TWh for 40 years of operation.

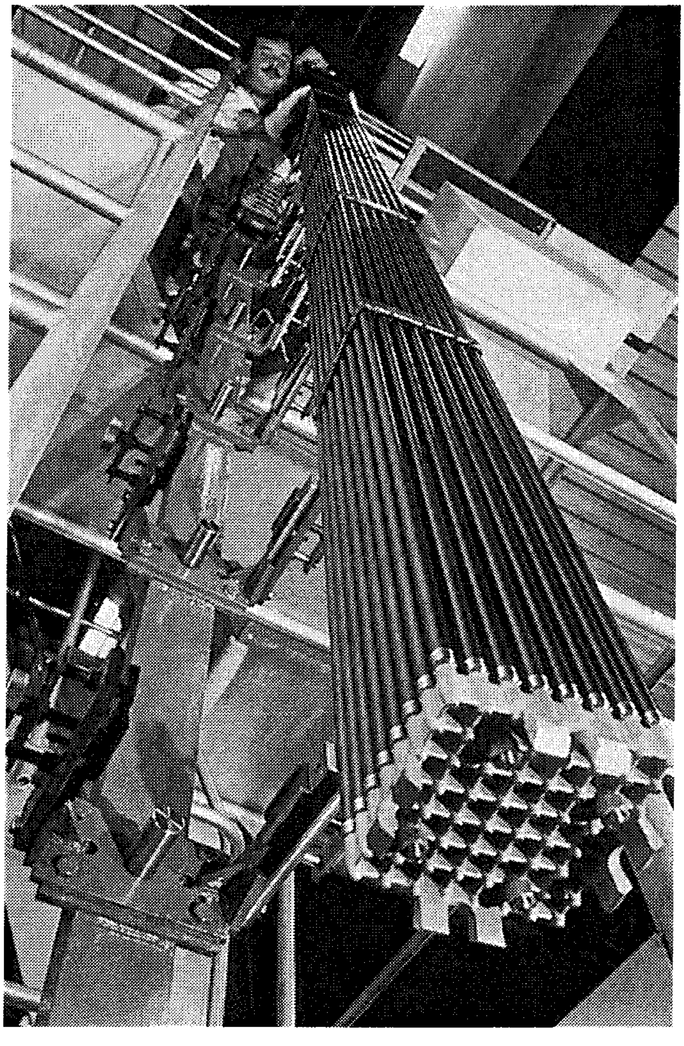


Figure 4.2-1. *Fuel assembly of type SVEA-64.*

Table 4.2.1. Expected fuel quantities for three operating alternatives

Case	Quantity of uranium (tonnes) discharged through 1993	Total
Operation until 2010	2,800	7,800
Operation for 25 years	2,800	6,500
Operation for 40 years	2,800	9,900

In the case with operation until 2010, approximately 5,900 tonnes of the waste is of the BWR type and 1,800 of the PWR type. All the operating alternatives include 20 tonnes of waste from the closed heavy water reactor in Ågesta and 23 tonnes of spent MOX fuel.

The estimates of future additional quantities are based on a average burnup of 38 MWd/kgU for BWRs and 41 MWd/kgU for PWRs.

4.2.2 Physical and chemical structure of the fuel

The physical and chemical structure of the fuel comprises the basis for modelling of fuel dissolution in a safety assessment. The material in this section is mainly taken from reference 4.2-2.

Physical structure

During startup and shutdown of a reactor, the fuel is subjected to thermal stresses that cause the fuel pellets to crack. Most of the fuel exists as fragments larger than 2 mm in PWRs and 4 mm in BWRs. Figure 4.2-2 shows a fuel pellet that has cracked during operation.

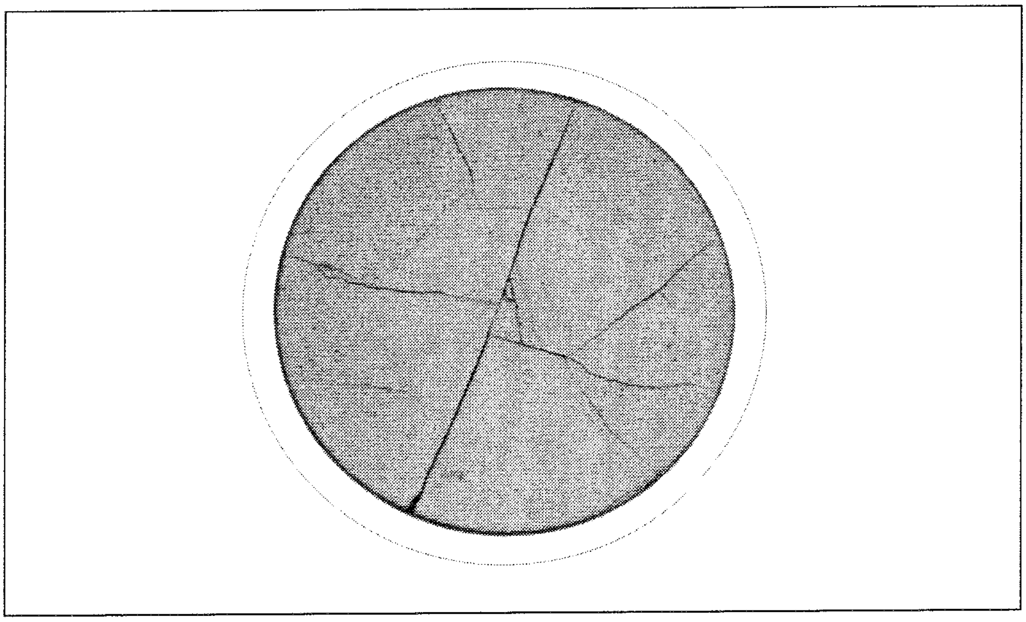


Figure 4.2-2. *Fuel pellet that has cracked during operation. Burnup: 49 MWd/kgU. Magnification 6 x.*

The specific surface area of the spent fuel is dependent on the size of the fuel fragments. Very small fragments (7–20 μm) have specific surface areas of around 3,000 cm^2/g (BET), while larger fragments have areas of 50–100 cm^2/g.

The cladding tubes or "cans" in the fuel assemblies are a barrier to radionuclide transport in a repository. The frequency of damage to the cladding tubes is therefore of interest. The portion of rods with clad damages has so far been very low. Approximately 500 damaged rods of a total of about 1 million used have been reported so far.

Rim effect

A rim zone forms at the tip of the fuel pellets in all light water reactor fuels. This zone is typically 20–40 μm thick. Burnup in the rim zone is approximately 25 percent higher than the average burnup, at an average burnup of 40 MWd/kgU. The α-activity, porosity and proportion of heavier actinides are also higher in this zone than in the inner parts of the pellet.

Chemical form of the nuclides in the fuel

Radioactive nuclides in the form of fission products and actinides are formed in the fuel by fission and neutron capture. The chemical form these nuclides will have in the fuel is determined by diverse chemical processes, by thermal diffusion and by mass transport processes caused by thermal gradients during reactor operation.

The fission products can be roughly divided into four groups according to their chemical form in the fuel, see Table 4.2-2. However, it is not always clear which group a given nuclide belongs to, in particular it can be difficult to decide whether a nuclide should be included in group 2 or group 3.

Table 4.2-2. Classification of fission products according to their chemical form in the fuel.

Group		Element
1.	Fission gases and other volatile fission products	Kr, Xe, Br and I
2.	Fission products that form metallic precipitates	Mo, Tc, Ru, Rh, Pd, Ag, Cd, In, Sn, Sb and Te
3.	Fission products that form oxide precipitates	Rb, Cs, Ba, Zr, Nb, Mo and Te
4.	Fission products that are dissolved as oxides in the fuel matrix	Sr, Zr, Nb, the lanthanides: Y, La, Ce, Pr, Nd, Pm, Sm and Eu

The actinides are present as dissolved oxides in the matrix in the same way as the fission products in group 4.

4.2.3 Definition of typical fuel

A typical fuel is chosen for the calculations of radionuclide inventory and decay heat in a safety assessment. The properties of the typical fuel are supposed to be representative of the fuel compositions and burnups described in section 4.2.1. The fuel quantity in the calculations is supposed to correspond to the amount that is held by the canister used in the repository.

The typical fuel in SR 95 consists of twelve BWR assemblies of type Svea 64 with a burnup of 38 MWd/kgU taken out of service in 1985. The closure date for the repository, i.e. the starting point for the reporting of the calculated radionuclide inventory, has been set at 2050.

The typical fuel in SR 95 is based on the same assumptions as for SKB 91, with the difference that the number of fuel assemblies has been set at twelve, compared to eight in SKB 91 /4.2-3/. The difference is due to the fact that another canister design is used in SR 95.

4.2.4 Radionuclide inventory and decay heat

The radionuclide inventory and decay heat for the fuel in the repository are calculated by means of a computer program. The properties of the typical fuel as regards quantities, burnups and cladding material etc. are used as input data in the calculations.

The calculations of radionuclide inventory and decay heat for the typical fuel as defined above have been performed using the established computer programs CASMO /4.2-4/ and ORIGEN2 /4.2-5/. Cross-sectional data from CASMO have been used as input data to ORIGEN2.

Parts of fuel assemblies and rod cladding made of Zircaloy, stainless steel and Inconel/Incoloy have been included for the calculation of activation products. On the other hand, the Zircaloy boxes that surround the BWR assemblies have been included. Details concerning the calculations are given in /4.2-3, 4.2-6/.

Figure 4.2-3 shows the decline of the decay heat.

For a given quantity of energy produced by light-water reactors, the total decay heat and the aggregate fission product inventory is independent of burnup.

The composition and structure of the fuel can, however, be affected by the burnup. Since future burnups may differ from the value assumed for the typical fuel, calculations have also been performed for a number of other burnups. As is evident from Figure 4.2-4, the decay heat during the time periods of interest is roughly proportional to the burnup and will therefore be more or less the same for a given decay heat.

4.2.5 Selection of nuclides for the safety assessment

At the time of disposal in the repository, the spent nuclear fuel contains a hundred or so different radionuclides /4.2-3/. Many of them are of negligible importance for the safety of the repository. One method for selection of radionuclides for a safety assessment is presented below.

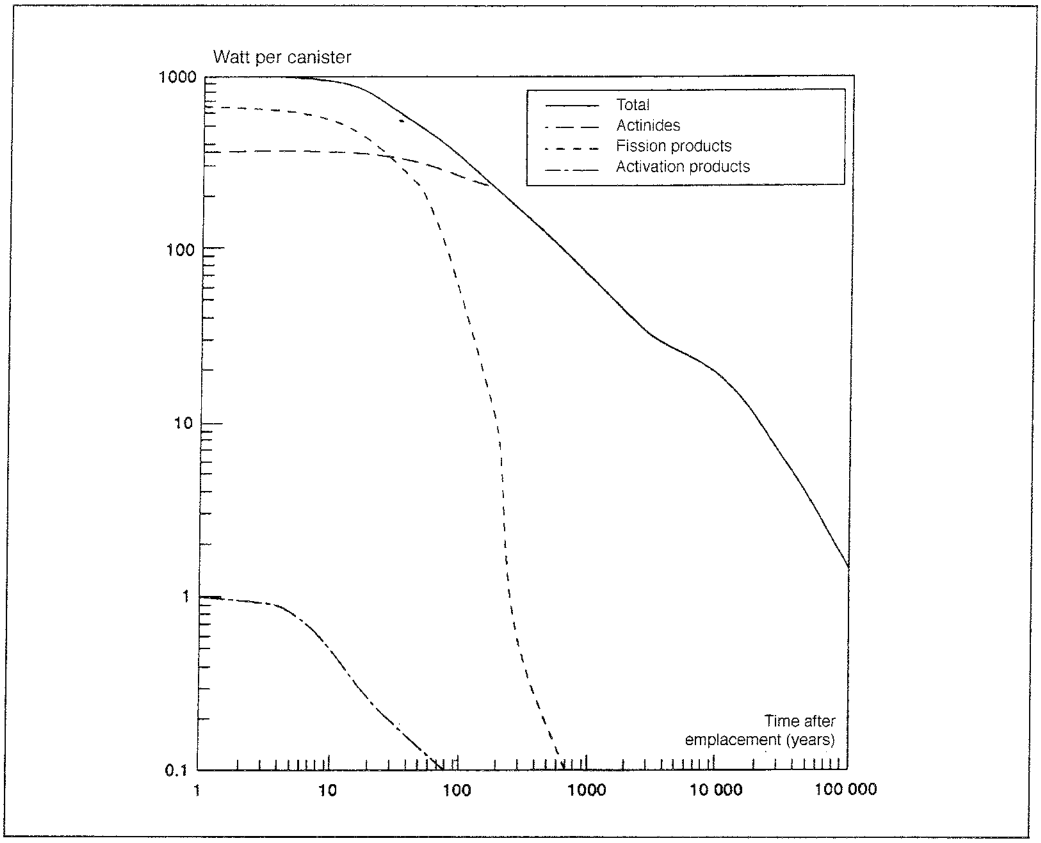


Figure 4.2-3. *Decay heat in the reference canister as a function of time after deposition.*

Three factors influence the importance of a radionuclide for safety:

- quantity,
- dose factor and
- mobility in the repository system.

These factors can vary between different nuclides by many orders of magnitude. The quantity of a given nuclide in the repository at a given point in time and its dose factor are relatively easy to determine. The mobility of the radionuclide, on the other hand, is more complicated to calculate, since it is a combination of many factors such as dissolution processes in the fuel and diffusion and sorption in different media.

The nuclides are divided into two groups for the purpose of selection. One group consists of actinides and actinide daughters, the other of fission products and activation products with light nuclei. The reason for the division is that the difference in mobility between lighter and heavier nuclides can be considerable, whereas the differences within a group are more limited.

For the fission and activation products, the potential toxicity (dose factor $\times$ quantity) is calculated for each nuclide. The calculation is done from the assumed date of closure of the repository, i.e. the year 2050, and a million years ahead in time. If the toxicity of a given nuclide at any time exceeds 0.1‰ of the total toxicity of all the fission and activation products at the same time, the nuclide is included in the assessment. The figure 0.1‰ is chosen to cover differences in mobility. The aggregate effect of the factors that determine

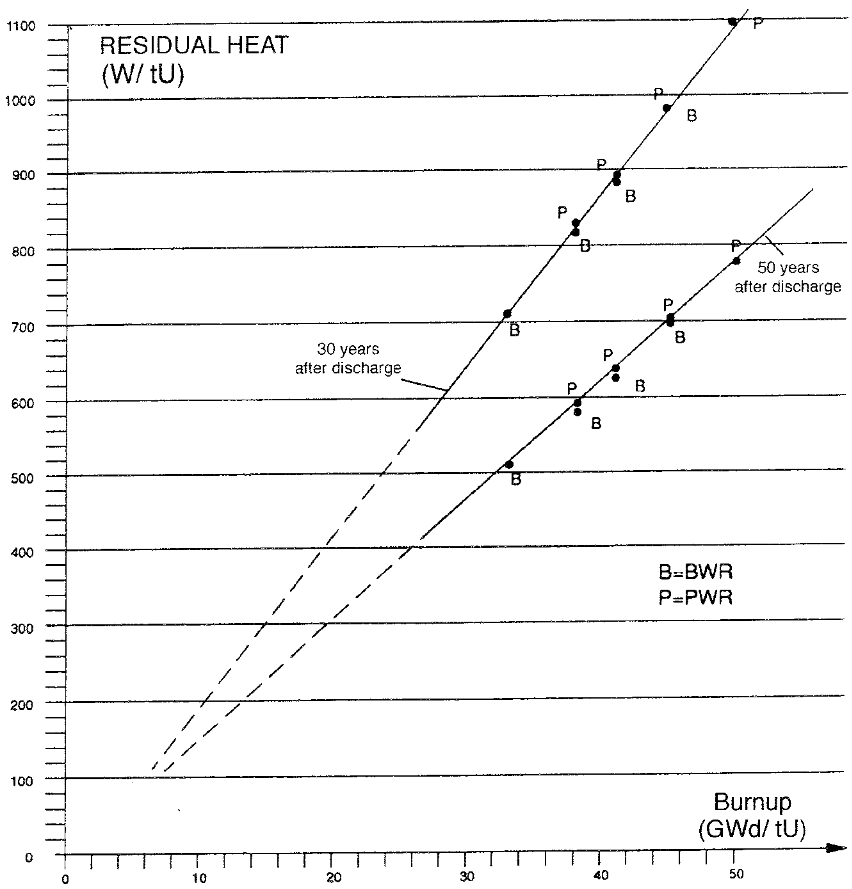


Figure 4.2-4. *Decay heat of the fuel assemblies as a function of burnup for two different time periods after discharge from the reactor.*

mobility is estimated to differ less than by a factor of 10^4 between different light nuclides.

For actinides and actinide daughters, all four decay chains are included in the assessment. The nuclide with the highest mass in each chain is determined by means of the same procedure as for the fission and activation products. Then all daughters in the chain are included (certain ones only in the biosphere).

This selection method is intended to be used for spent fuel, but should also be able to be applied to other waste types.

Tables 4.2-3 and 4.2-4 show the results of the application of the method to the radionuclide inventory described in section 4.2.4. Figure 4.2-5 shows the relative potential toxicity of fission/activation products at different times.

At short times, Cs-137 dominates the potential toxicity of the inventory. The nuclides Co-60, Cd-113m and Eu-154 have shorter half-lives and are already present in much smaller quantities than Cs-137 at the time of disposal. Nor do they have appreciably higher mobility than Cs-137 and can therefore be excluded from the assessment, despite the fact that they exceed 0.1‰ of the total potential toxicity at the shortest times (~10 years).

Kr-85 also has a very short half-life, but can be of importance in special scenarios because it occurs in gas form.

Table 4.2-3. Important fission and activation products

Nuclide	T ½ (year)	Nuclide	T ½ (year)
C-14	5,730	Pd-107	$6.5 \cdot 10^6$
C1-36	$3 \cdot 10^5$	Ag-108 m	127
Co-60	5.3	Cd-113 m	14.6
Ni-59	$7.5 \cdot 10^4$	Sn-126	$1.0 \cdot 10^5$
Ni-63	100	I-129	$1.6 \cdot 10^7$
Se-79	$6.5 \cdot 10^4$	Cs-135	$2.0 \cdot 10^6$
Kr-85	10.8	Cs-137	30.2
Sr-90	28.5	Sm-151	93
Zr-93	$1.5 \cdot 10^6$	Eu-154	8.8
Nb-94	$2.0 \cdot 10^4$	Ho-166 m	1,200
Tc-99	$2.1 \cdot 10^5$		

Table 4.2-4. Parent nuclides and important daughters in the decay chains of the heavy nuclides

Chain	4N	4N+1	4N+2	4N+3
Parent	Cm-244	Cm-245	Cm-246/Am-242m	Cm-243/Am-243
Daughters	Pu-240	Am-241	Pu-242/Cm-242	Pu-239
	U-236	Np-237	U-238/Pu-238	U-235
	Th-232	U-233	U-234	Pa-231
		Th-229	Th-230	
			Ra-226	

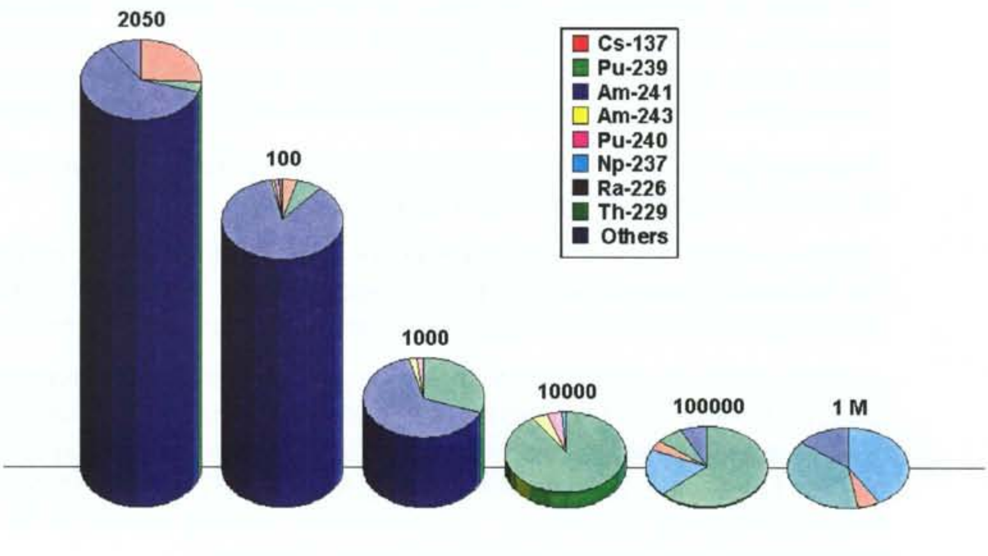


Figure 4.2-5. *The relative toxicity of the fuel at different times is indicated by the height of the cylinders. The share represented by dominant nuclides is indicated on the top surface.*

4.3 OTHER LONG-LIVED WASTE

Long-lived low- and intermediate-level waste is deposited in a separate part of the deep repository. The quantities are comparatively small and the principal source is research and scrapped parts of the reactor that have been situated in or near the fuel core, i.e. core components and reactor internals. Core components are temporarily stored at CLAB and the research waste is stored and conditioned at Studsvik.

This waste will be deposited in three specifically designed parts of the deep repository, see section 5.2. The total volume of waste is estimated to amount to about 25,000 m^3 (the sum of the outer volume of the waste packages). Strictly speaking, only a portion of the waste belongs to the category long-lived waste. The rest is operational waste and decommissioning waste from CLAB and the encapsulation plant. The portion of the waste that could be accepted by SFR comprises approximately 50% of the total volume.

A compilation of quantities, composition and properties of this waste was recently published /4.3-1/. The ambition was to make as good an estimate as possible based on the information available plus calculations and consultation with the waste producers. Better data on waste composition and contents is continuously being obtained, as so-called "type descriptions" are produced, following the same practice as for the SFR waste.

The published report on the waste was used as a basis for an initial performance assessment of the repository barriers /4.3-2/.

Selection of nuclides for the safety assessment

The time schedules for repository design and safety assessment of other long-lived waste are based on the fact that an account must be given in support of the permit application for the deep repository, i.e. in around 2001. The safety assessment of other long-lived waste is therefore not discussed in SR 95.

4.4 CHEMOTOXIC WASTE

Inventories of chemotoxic substances have been done for both canisters of spent fuel /4.4-1/ and other long-lived waste /4.3-1/. Spent fuel contains uranium, but also smaller quantities of other elements produced by the nuclear reactions. Many of these chemical elements can be said to be toxic or harmful to the environment, aside from their radioactivity. Examples of such elements are silver, barium, cadmium, antimony and selenium. Harmful metals such as chromium and nickel are present in the metallic parts of the fuel assemblies, and even the copper of which the canisters are made can be said to possess some toxicity.

Target values given for the highest permissible concentration of a substance in drinking water have been used as a measure of toxicity /4.4-1, 4.3-1/. Uranium is the dominant component in this respect, partly because there is comparatively much uranium in a canister, and partly because it is a relatively toxic element.

There are chemotoxic metals in long-lived low- and intermediate-level waste as well. Such waste also contains a small fraction of organic compounds that can be hazardous to health and the environment. Examples of toxic metals are large quantities of chromium and nickel in the reactor internals of steel that are to be disposed of. Other metals in the waste are lead, copper, cadmium and beryllium. Lead is incorporated in some of the packagings as a radiation shield. The estimates of quantities that have been made are preliminary.

The repository protects the surrounding environment from the chemotoxic substances contained in the waste as well /4.4-2, 4.3-2/.

5 DESIGN OF THE REPOSITORY SYSTEM

Chapter 5 describes the design of the repository system and the engineered barriers. The description is based on the background material available in the current design stage submitted in support of the safety assessment that is being presented. The chapter presents the design and materials of the deep repository, as well as quality requirements (impurities) and dimensions for the barriers included in the repository system. Similarly, methods for construction and inspection that may be utilized and their possible effects on the host rock are discussed. Performance studies or previous safety assessments that have served as a basis for choice of dimensions or design are summarized. "Free" parameters that can be utilized for site adaptation or optimization are reported separately.

Against the background of the major safety functions for the various barriers, a review is made of the principles for the design of the repository, after which the different parts of the underground facility and their design/layout are presented. This is followed by a presentation of the method of excavation of deposition positions and the fabrication/design of the canister, bentonite buffer and tunnel backfill. Plugging of tunnels and shafts and the final closure and sealing of the repository are discussed.

***In this report**, the currently (September 1995) applicable design of the underground portion of the deep repository is described in Chapter 5. The description illustrates how the material will be structured. Details in the design may change during the development work and with the choice of work methods. Plugging of tunnels and shafts has not been discussed.*

5.1 INTRODUCTION

The design of the deep repository is based on the KBS-3 concept. The repository consists of two main parts at a depth of about 500 m in the Swedish crystalline bedrock.

One main part is a repository for the high-level spent nuclear fuel, see Figure 5.1-1. This waste is enclosed in copper canisters. The canisters consist of a steel inner container or insert that provides mechanical stability and a copper outer container or shell that provides corrosion protection. The canisters are deposited one-by-one in bored deposition holes in the bottom of a tunnel system. Each canister is surrounded by a layer of bentonite clay. The clay holds the canisters in place and isolates them from the groundwater in the surrounding rock. The clay also retards the transport of different substances to and from the canister.

The other main part is a repository for other types of long-lived waste that are deposited in caverns excavated in the rock. Physical and chemical barriers of concrete and bentonite limit the waterborne transport of the toxic material.

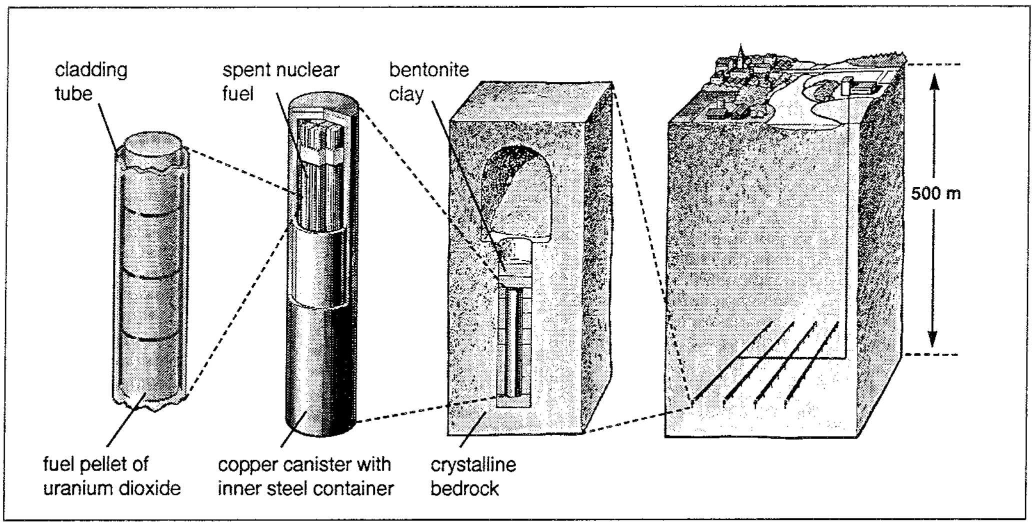


Figure 5.1-1. *Repository part for high-level waste.*

This repository part resembles the existing final repository for radioactive operational waste, SFR.

After the waste has been deposited, tunnels and other spaces are backfilled.

This chapter begins by describing the layout of the repository system with an emphasis on the most important underground portions. Then the canister with bentonite buffer and backfill material are treated in greater detail. The presentation is based on the requirements made on different parts of the system and shows what design the requirements have led to. Performance assessments of different sub-systems are discussed in Chapters 10 and 11.

5.2 DESIGN AND LAYOUT OF THE REPOSITORY SYSTEM

The design and layout of rock caverns, tunnels, deposition positions etc. in the repository system is based on that presented in the KBS-3 report.

Several alternative solutions for the entire deep repository system placed at "mine depth", i.e. between 400 and 700 m below the surface, have previously been studied /5.2-1, 2/. In these comparisons, the KBS-3 method has been judged to be superior. This design is therefore the reference concept for the ongoing work of repository design.

Details in the design and choice of work methods change during the course of the development work. However, the modified solutions always meet at least the same functional requirements as previous designs. The description given below is based on the current design at this time (September 1995).

A drawing of the different parts of the deep repository is shown in Figure 5.2-1. The repository consists of:

- surface facility,
- access shaft or ramp,

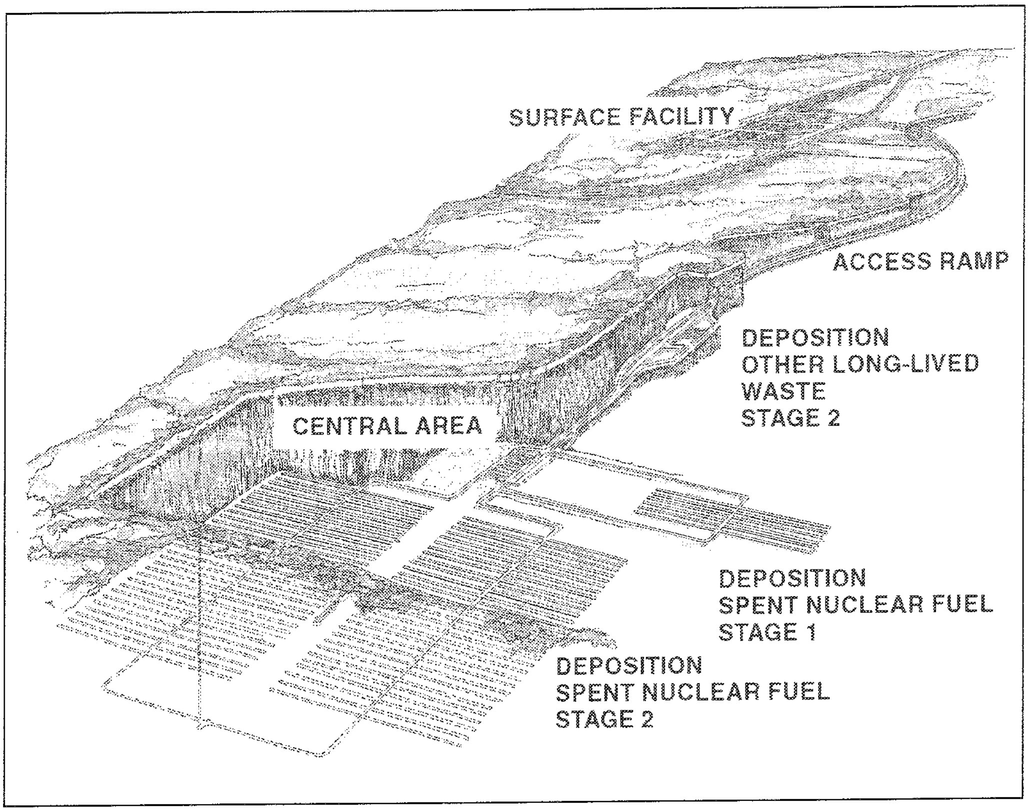


Figure 5.2-1. *Schematic drawing of the deep repository with access ramp.*

- central area under ground,
- area for deposition of spent nuclear fuel, stage 1,
- area for deposition of spent nuclear fuel, stage 2, and
- area for deposition of other long-lived waste.

The deep repository for spent nuclear fuel in accordance with KBS-3 is situated on one or two levels at a depth of about 500 m /5.2-2, 3/. The area for other long-lived waste is designed in accordance with the principles for the final repository for radioactive operational waste (SFR) /5.2-4/. It is situated at such a distance from the repository for spent nuclear fuel that the large quantities of concrete in the units for other long-lived waste do not disturb the chemical conditions in the area for spent nuclear fuel.

The following text describes the parts of the repository that are of the greatest interest for the safety of the repository, namely the underground section intended for spent nuclear fuel and the one intended for other long-lived waste.

5.2.1 Section for spent nuclear fuel

General

The repository for spent nuclear fuel consists of parallel deposition tunnels with holes bored in the bottom. Canisters with the spent nuclear fuel are emplaced in the holes, together with a surrounding clay buffer.

Figure 5.2-2 shows a cross-section through such a canister position. The dimensions are determined by the size of the canister, by the room needed for rock works, operation and deposition, and by the desired performance and safety after closure /5.2-5/. In practice, certain deviations will be made from the indicated dimensions as a consequence of the choice of e.g. rock excavation methods. Narrower tolerances can be achieved within moderate limits at the expense of more complex and costly technology.

The deposition tunnels are connected by tunnels for transport, communication, ventilation and utility lines. These tunnels are connected to a central area under ground and via a communications shaft/tunnel with the surface.

The decay heat in the deposited waste will lead to a heating of the repository. The location of the deposition tunnels, as well as the spacing between the canisters in the tunnels, is determined by the requirement to limit the temperature in the clay buffer. The location will therefore be influenced by the thermal properties of the local host rock /5.2-6/. Finally, local conditions determine how large a portion of the maximum number of canister positions can be utilized.

The above design assumes that all tunnels are bored or blasted using conventional methods and that the deposition holes are fullface-bored. The canisters are simultaneously tilted and lowered into the deposition hole, so that the height in the deposition tunnel can be lower than the length of the canister. These premises have not been changed since SKB 91 /5.2-7/.

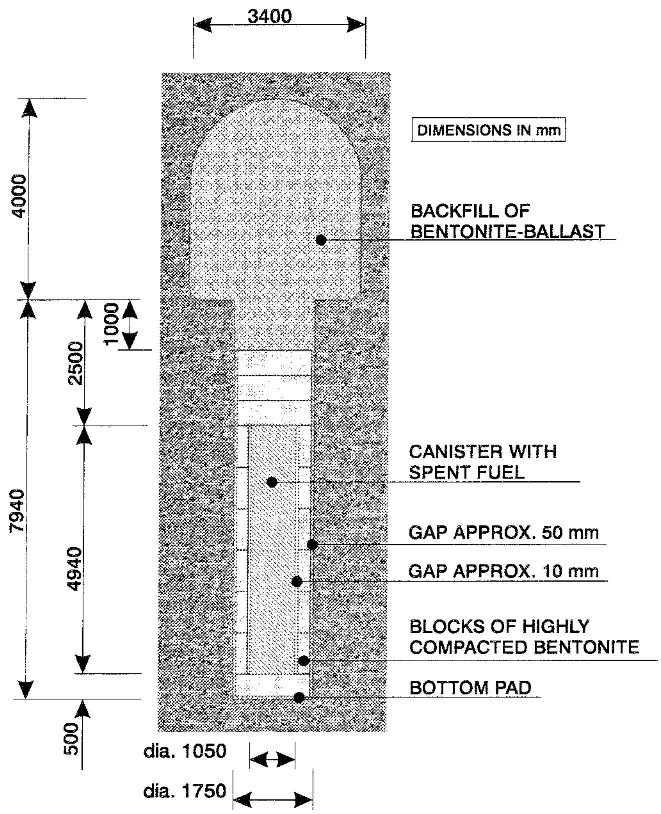


Figure 5.2-2. *Cross-section through deposition hole.*

Dimensions of deposition tunnel

The space requirement for lowering the canister into the deposition hole sets **the tunnel height at 4.0 m**. The deposition machine requires a **tunnel width of 3.4 m**. These dimensions apply for the given canister and the stipulated diameter of the deposition hole. Furthermore, only free-blowing ventilation in the tunnel is assumed.

Dimensions of deposition hole

The *distance between the bottom of the tunnel and the top of the canister* is determined by

- location of canister outside excavation-disturbed zone,
- allowance for importance of zone of stress redistribution,
- allowance for bentonite swelling depending on backfill material in tunnel,
- allowance for influence on bentonite of concrete on tunnel bottom (if any),
- allowance for possible grouting in rock under tunnel,
- allowance for possible bolting for rails on tunnel bottom,
- resistance to radionuclide transport, and
- radiation protection permitting human presence in tunnel.

The requirement that the canister be placed outside the excavation-disturbed zone sets the **distance at 2.5 m**. To achieve low hydraulic conductivity around the canister and furthermore keep the imperviousness of the buffer in the upper part of the hole from decreasing too much, **bentonite backfill is emplaced above the canister to a height of 1.5 m. A mixture of bentonite and aggregate is then used to a height of 1.0 m.**

The *thickness of the buffer underneath the canister* is determined by

- influence of the base pad on the bentonite if concrete is used,
- bearing capacity of the bentonite,
- swelling capacity of the bentonite,
- hydraulic conductivity of the base pad,
- resistance of the base pad to nuclide transport, and
- heat transport through the bottom pad.

To allow for the possibility of a concrete bottom pad and to ensure that the barrier to nuclide transport has at least the same capacity as in other parts of the buffer, the **thickness of the buffer underneath the canister is set at 0.5 m**.

The *diameter of the hole* is the sum of the diameters of the canister and the buffer. The thickness of the bentonite buffer is in turn determined by the desired mechanical, chemical and hydraulic properties and the desired capacity for gas migration.

The permissible heat output of the canister is limited by the temperature rise that can be tolerated in the buffer. A thicker buffer will thus reduce the amount of spent fuel that can be loaded into each canister, which increases costs. In view of this and the requirement on good buffer capacity against nuclide transport, the **bentonite barrier is made 0.35 m thick**, which leads to a hole diameter **of 1.75 m** for the assumed (section 5.3) canister size.

Distance between canisters and between deposition tunnels

The distance (spacing) between deposition holes is determined by

- quantity of fuel per canister,
- thickness of the buffer,
- permissible temperature rise in the buffer,
- ambient rock temperature,
- thermal diffusivity (ability to dissipate heat) of the buffer,
- thermal diffusivity of the rock,
- distance (spacing) between deposition tunnels,
- permissible thermal load per unit horizontal surface area,
- strength of the rock against thermally induced stresses, and
- requirement on limited hydraulic connection between deposition holes.

The design basis factor /5.2-8/ is the temperature rise in the bentonite buffer. For the given canister, thermal data for ordinary granite, 18°C *in situ* in the rock and 80°C maximum temperature in the bentonite, the **canister distance is 6.0 m and the tunnel distance 40 m**. This is an economical optimization between canister distance and tunnel distance. Low ambient temperature, higher permissible temperature in the bentonite and higher quartz content than normal in the rock are examples of some factors that lead to a smaller distance between canisters and/or between tunnels.

Deposition positions

It is not, however, likely that all the positions in the pattern stipulated above can be utilized. Various local properties in the bedrock can, for example, result in poor conditions for secure long-term isolation of the waste in certain positions.

The factors that have been identified as being of importance in deciding whether a canister position is to be utilized or not are /5.2-9/:

- lithology (composition of the rock mass),
- inflow of water to the deposition hole,
- stability from the viewpoint of construction,
- long-term stability and
- thermal properties of the rock.

These factors can be studied both on a regional scale and for individual canister positions. A span of acceptable values can be determined for each factor.

By assessing and setting bounds on these factors and then applying them to the disposal site, a figure is arrived at for degree of utilization of the rock volume. It is assumed for the time being that 90 percent of all positions can be utilized for deposition.

5.2.2 Section for other long-lived waste

General

The repository for other long-lived waste consists of parallel rock caverns as in SFR. The tunnels up to the rock caverns are utilized for deposition of decommissioning waste. The area is situated at the necessary distance from the repository for spent nuclear fuel and in a way that is practical from the viewpoint of layout and transportation.

The repository section for other long-lived waste consists in turn of three parts that differ in the type of waste to be deposited:

1 Operational waste from the central interim storage facility for spent nuclear fuel (CLAB), the Encapsulation Plant and Studsvik.

2 Core components and reactor internals.

3 Decommissioning waste from CLAB and the Encapsulation Plant.

All material is enclosed in concrete moulds or in metal drums that are comparable to the packages deposited today in SFR. The repository is also designed like SFR.

In each rock cavern (see below), different barriers are built around the deposited packages. Allowance is made for the isolating function that is desired after closure and the degree of backfill and plugging that is deemed necessary /5.2-4/.

Design requirements and materials for backfilling are discussed in greater detail in section 5.4.

Section for operational waste from CLAB, Encapsulation Plant and Studsvik

The rock cavern is similar in design to the silo in SFR today. Walls and roof are lined with shotcrete. The concrete structure with pits for the waste packages rests on a base slab of compacted bentonite and rock aggregate. This in turn rests on a drainage layer and a concrete pad, see Figure 5.2-3. The spaces between the waste packages and the concrete walls in the shafts are filled out with a porous concrete. A concrete lid is placed on top. The space between the rock wall and the outer concrete wall is filled with bentonite up to the level of the concrete lid, after which the entire cavern is covered with a layer of crushed rock and a layer of bentonite and rock aggregate. Concrete slabs are placed on top of this, and the rock cavern is filled up to the roof with crushed rock.

The tunnel into the rock cavern is sealed with a plug that forms a tight seal against an exposed rock wall. This can be done with a mixture of bentonite and rock aggregate or with pure bentonite in the same way as in deposition tunnels, see section 5.4. The disturbed zone is left as is.

Section for core components and reactor internals

The design of the rock cavern is similar to that of the concrete mould repository in SFR. The walls and roof are shotcreted. The floor is covered with

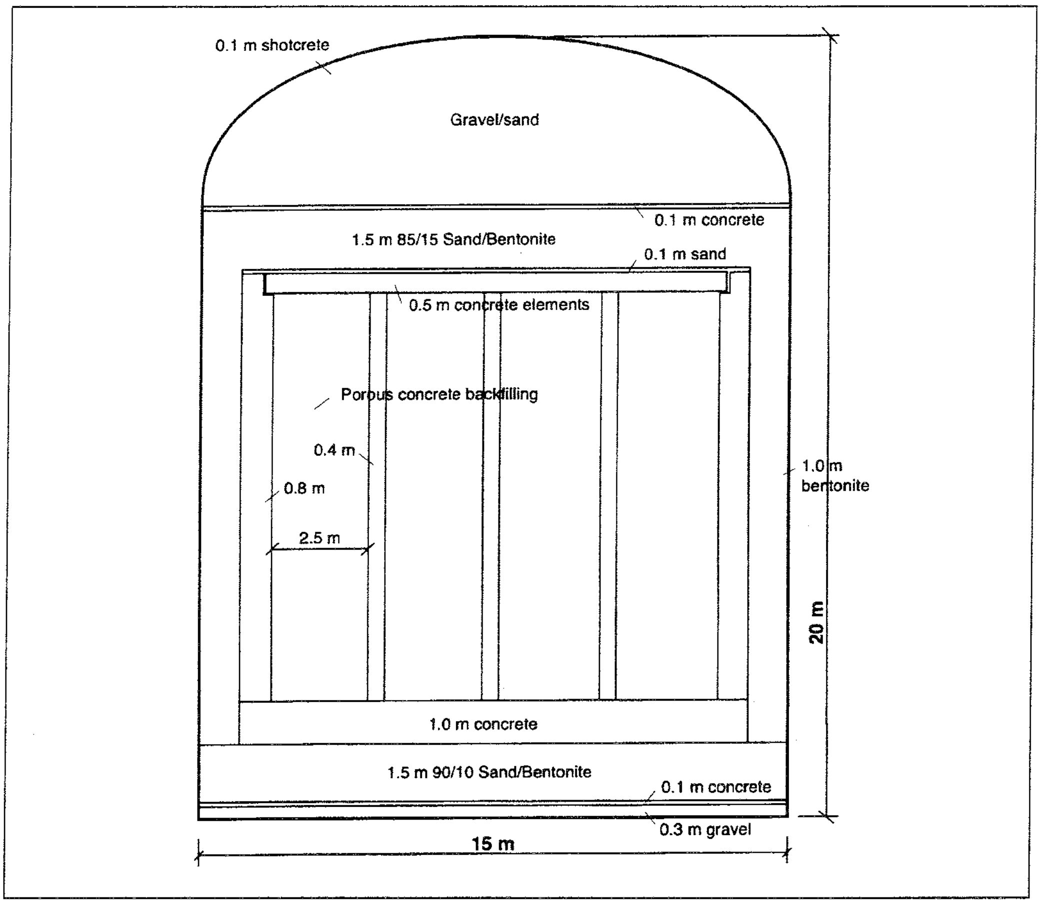


Figure 5.2-3. Cross-section of rock cavern for operational waste from CLAB, the Encapsulation Plant and Studsvik.

a drainage layer on which the floor, exterior walls and partition walls of the concrete structure rest, see Figure 5.2-4. When all moulds have been emplaced, a lid of concrete slabs is put on and the remaining volumes outside the concrete structure are backfilled with crushed rock. The spaces between the packages and between the packages and the concrete walls inside the concrete structure are not backfilled.

The tunnel from the transport tunnel into the rock cavern is backfilled with a mixture of bentonite and rock aggregate. Plugs against any water-conducting discontinuities are built as needed, so that transport pathways for nuclides are determined by the structure of the rock and not by the presence of tunnels and rock caverns.

Section for decommissioning waste

This section consists of the tunnel system up to the rock caverns within the area for other long-lived waste. The walls and roof are shotcreted. A concrete floor is poured on a drainage base layer. The decommissioning waste is enclosed in metal containers. All empty spaces between the containers and between the containers and the rock walls are backfilled with crushed rock. Finally, the tunnel system is plugged by backfilling with bentonite and rock aggregate so that a seal is obtained against the rock walls. To the extent shotcrete is needed in the section for rock reinforcement purposes, the rock wall is cleaned to ensure good adhesion.

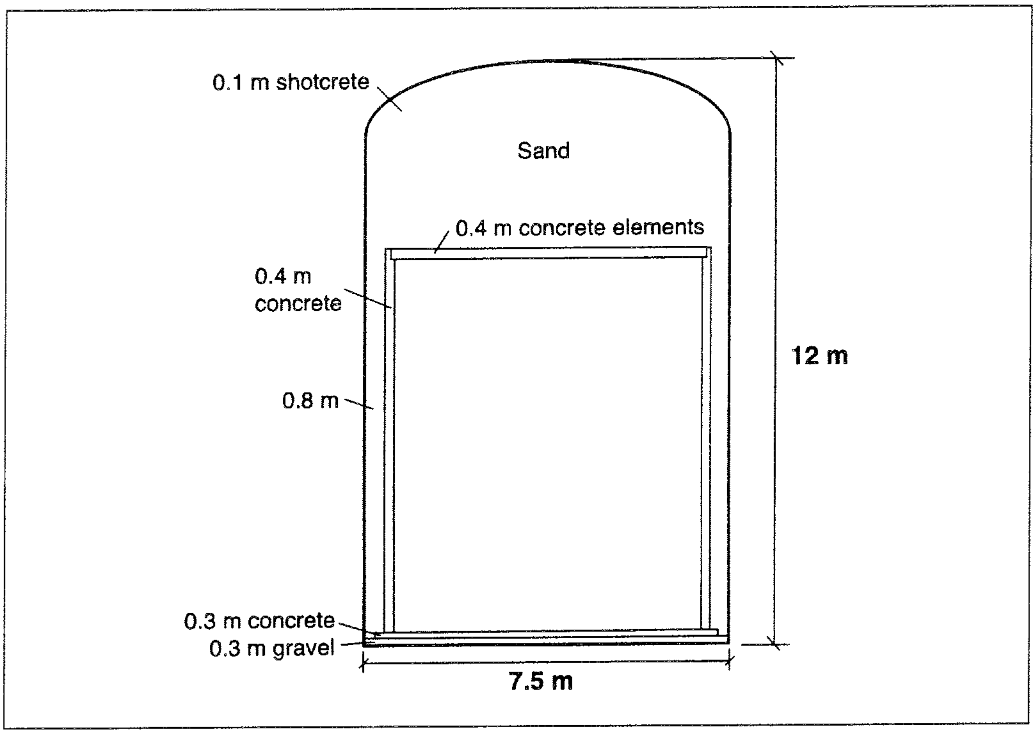


Figure 5.2-4. Cross-section of rock cavern for core components and reactor internals.

5.3 CANISTER

5.3.1 Introduction

The canister is a fundamental engineered barrier in the repository system. It has to satisfy two primary functional requirements in order to provide the necessary isolation in the deep repository:

1. The canister must retain its integrity over a long time, which in turn imposes requirements on
   - initial integrity,
   - corrosion resistance, and
   - strength.
2. The canister must not have any harmful effect on the other barriers in the deep repository, which imposes requirements on
   - choice of material that does not adversely affect the buffer and rock,
   - limitation of heat and radiation dose in the near field,
   - design so that the fuel remains subcritical even if water enters the canister, and
   - limitation of the bottom pressure on the bentonite.

The canister's dimensions and materials are described below. Important fabrication methods are described briefly, along with post-fabrication inspection.

Since the design of the canister has not been finally decided, alternative designs are briefly discussed.

The performance of the canister in the repository system, given the following premises, is assessed in Chapter 10.

5.3.2 Design

The reference canister consists of two components: a cast insert and a copper shell /5.3-1/. The cast insert, with individual channels for the fuel assemblies, lends the canister mechanical strength to resist external pressures in the deep repository and handling loads. The copper shell, which is 50 mm thick, lends the canister its corrosion resistance, see Figure 5.3-1.

All design details have not yet been finalized. They may be modified in response to requirements made on the fabrication and encapsulation process. One possible design is shown in Figure 5.3-2. This design serves as a basis for SR 95. In this alternative, the canister consists of an insert of cast steel fabricated in two parts, each with half the total length, and welded together in the middle. The outer copper shell can be fabricated either as a seamless tube by extrusion or from two plates that have been formed into tube halves and welded together by two longitudinal electron beam welds. The lid and bottom can be fabricated by working from thick plate or can be forged and machined to their final shape. The total weight of the filled canister fabricated in this way is about 24.5 tonnes. The insert accounts for about 13.5 tonnes and the copper shell for 7.5 tonnes.

Figure 5.3-1. *The canister's copper shell.*

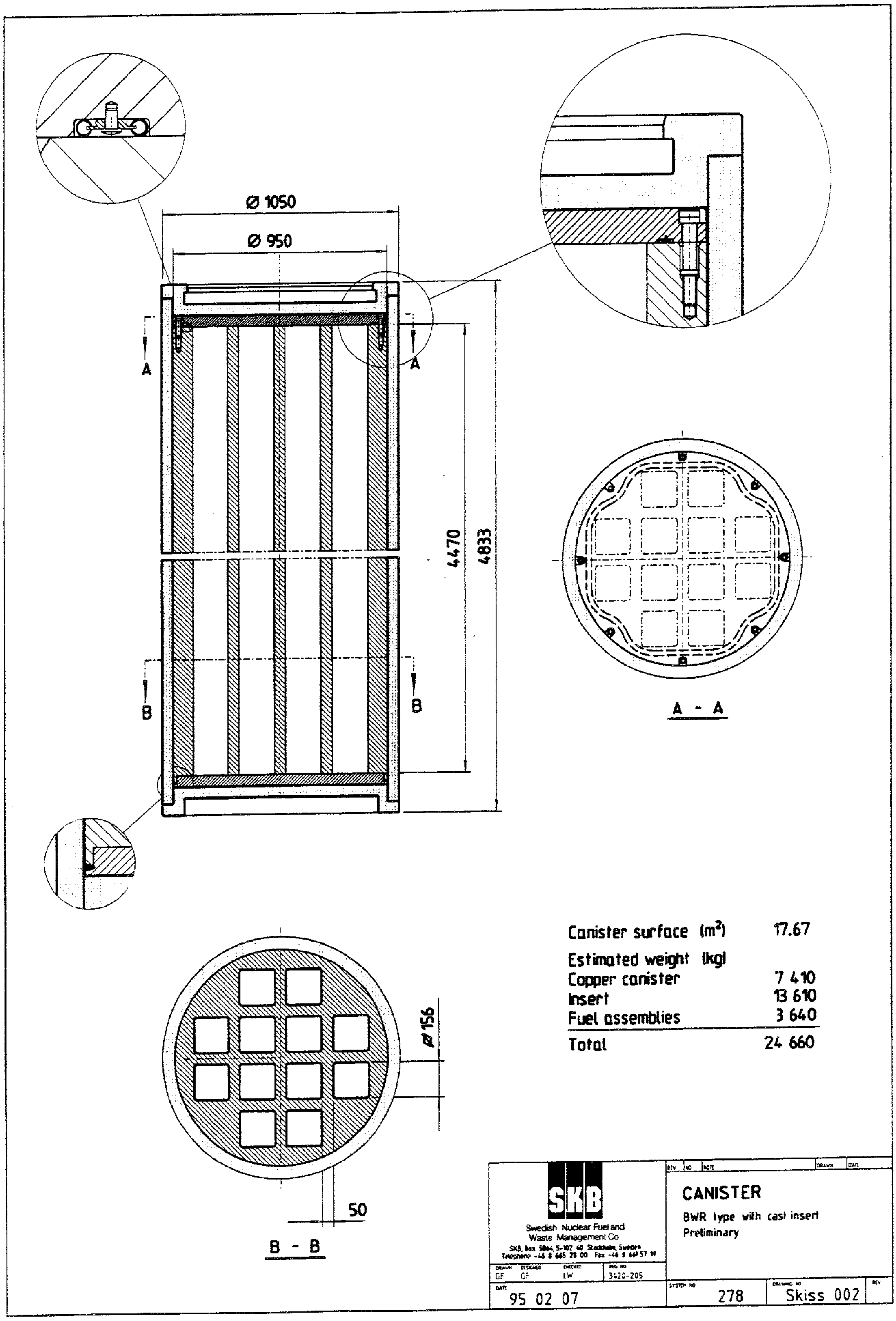


Figure 5.3-2. *One possible design of the canister's shell and insert.*

5.3.3 Quality inspection

Fabrication and sealing of the canister will be inspected and verified by means of nondestructive testing. The details around the testing programme have not been finalized. It is assumed that the stock material will be checked by ultrasonic testing and the welds by a combination of radiography and ultrasonic testing. The welds will also be inspected by methods to ensure that the copper shell does not have any minor surface-breaking defects. Such defects would not be of importance for the canister's mechanical integrity, but could serve as starting points for crevice corrosion. In the same way, it is assumed that the sealing weld on the lid will be inspected by radiography, ultrasonic examination and some method for detecting surface-breaking cracks.

5.3.4 Alternative designs

Since a final canister choice has not yet been made, alternative designs may be considered. All of these designs feature a corrosion barrier of copper and an inner pressure-bearing component to provide mechanical strength. Alternative designs and materials may be considered for the inner component. The most likely materials for cast alternatives are spheroidal graphite iron and bronze. A self-supporting steel tube has been examined as an alternative to a cast insert.

5.4 BUFFER AND BACKFILL

The deposited canisters will be surrounded by a buffer intended to keep the canister in place, isolate the canister from groundwater and greatly retard the transport of radionuclides. Furthermore, tunnels, rock caverns and shafts in the repository will be backfilled in such a way that groundwater flow is prevented and chemical changes are impeded and delayed.

These somewhat similar functional requirements on buffer and backfill also lead to similar considerations for material choices.

Functional requirements, material selection and method for application of buffer and backfill are described in the following. A performance assessment for the buffer, based on the premises described below, is presented in Chapter 10.

5.4.1 Functional requirements

Buffer

The choice of buffer material and the design of the buffer around the canister are based on the requirements that the buffer shall

- bear the canister,
- prevent groundwater flow,
- permit hydrogen gas to escape,
- dissipate heat from the fuel,
- remain in place for a long time,
- resist chemical transformation for a long time, and
- filter colloids.

Furthermore, the buffer must not jeopardize the prospects for the canister and the rock to meet their functional requirements. In addition, there is the economic requirement that the availability of the type of material to be used shall be good.

Different types of swelling clays with a high smectite content have good prospects for meeting both the technical and the economic requirements. Such a material also possesses a number of other favourable properties, such as

- limitation of transport of corrodants up to the surface of the canister (diffusion transport),
- sorption of radionuclides,
- filtration of micro-organisms, and
- stabilization of the walls of the deposition hole.

The different functions are controlled by a number of measurable properties of the buffer:

- hydraulic conductivity,
- swelling pressure,
- swelling capacity,
- shear strength,
- rheological properties,
- pore volume,
- diffusion and sorption properties, and
- thermal conductivity.

Backfill

Different mixtures of bentonite and aggregate are planned to be used for backfilling of tunnels, rock caverns and shafts. The required function of these materials is to:

- counteract swelling of the bentonite out of the deposition hole,
- limit the flow of water in the deposition tunnel, and
- resist chemical transformation for a long period of time.

In cases where water transport to the near-field rock is limited, the requirement on low hydraulic conductivity in the tunnel will be meaningless. In these cases the requirement on mechanical stability will be most important.

The economic requirement is to look for the cheapest mixture that meets the functional requirements. This requirement assumes importance in the choice of material in that pure bentonite is expensive and the functional requirements can be met with lean bentonite mixtures. If requirements are made only on mechanical stability, unmixed aggregate can be considered.

The properties that determine the performance of the material and that can be measured are the same for bentonite/aggregate mixtures as for pure bentonite. An additional property of importance for backfill materials is compressibility, which determines how swelling-out from the deposition holes can be limited.

5.4.2 Material selection

Buffer

A suitable buffer material is a smectite-containing clay. However, the swelling performance of these clays could in the long run be jeopardized by conversion of smectite to illite. A study of different buffer materials /5.4-1/ has shown that the original content of smectite should be at least 50 percent in order that the performance of the buffer can be guaranteed. The study is based on a model for degradation of clay in the form of conversion of smectite to illite /5.4-2, 3/. The study shows that only the smectite types montmorillonite and saponite with sodium as the primary absorbed ion should be considered. Such clays, bentonites, are available in large quantities in a number of countries.

Even higher original smectite contents are very valuable for providing effective self-healing and homogenization of the clay. Bentonites with a montmorillonite content of 70–90 percent and with sodium as the dominant absorbed cation are commercially available in the USA, Greece and Italy.

The content of sulphur minerals and organic substances should be low, regardless of smectite content.

Backfill

The preliminary choice of backfill material is a mixture of 10-20% bentonite and the remainder aggregate that is deposited and compacted in place. After water saturation, the backfill will have a hydraulic conductivity of less than 10^{-9} m/s, which is on a par with that of very impervious rock. For environmental and economic reasons, use of the rock that is excavated in the deep repository is recommended, after it is crushed to a suitable grain size. The properties of such material have been investigated and found to be comparable to the properties of quartz sand as aggregate /5.4-4/.

Alternative backfill materials are rock aggregate or glacial till without bentonite, but their hydraulic conductivity will be higher after compaction than that of hydraulically impervious rock /5.4-5/.

5.4.3 Application method

Buffer

Precompacted magnum blocks are applied as segments or whole rings of full diameter. These segments or rings are pressed to a high degree of water saturation. After deposition of the canister, the gaps are filled with water with a low electrolyte content so that a high pore water pressure is reached quickly.

Backfill

The backfill material is deposited in horizontal layers and compacted with a vibratory roller or in inclined layers at the top against the roof, where compaction is done with a vibratory plate mounted on a moveable arm or similar equipment.

Compaction of blocks with a weight of about 20 kg has been done with good results /5.4-6/. Such blocks could possibly be used in the upper part of the tunnel against the roof.

6 PROPERTIES OF THE REPOSITORY SITE

This chapter describes the geoscientific properties of the selected repository site that are of potential importance for the long-term safety of a deep repository in crystalline rock.

Based on site data and geoscientific understanding, a site model is devised representing the structural characteristics of the site with a bearing on long-term safety. This model can be said to be a composite of all the geological, geophysical, geochemical and geohydrological information that has been collected.

The chapter is organized in sections that describe geology, groundwater chemistry, geohydrology and the transport properties of the rock.

The composite site model is associated with uncertainties, which are discussed in each section and summarized in a concluding section. The consequences of these uncertainties in the site description for the results of the safety assessment are discussed later in the report, however. The description of the properties of the site is based on present-day conditions. Time-dependent changes are discussed in the relevant sections.

***In this report**, the site description in Chapter 6 is based on geoscientific data from the Äspö Hard Rock Laboratory (HRL). This does not means that the Äspö HRL will be used for the final disposal of radioactive waste. Äspö data have been used due to the fact that they comprise an unusually extensive, coherent geoscientific database for a specific site in crystalline rock. The chapter compiles all the data that have been gathered up to the end of 1994. It should be noted that the description in this chapter is supplemented in section 10.3, where the performance of the rock is described. Rock performance is discussed there in general terms, but also particularly for Äspö.*

6.1 GENERAL ABOUT THE SITE

6.1.1 Introduction

A number of performance and safety assessments for the deep repository will be carried out by SKB during the next few years. An important question to study in these assessments is the performance of the rock barrier in a deep repository. Site-specific data for potential deep repository sites will, however, not be available until site investigations have been carried out. In the present report, site data have therefore been taken from the Äspö HRL in Småland in southern Sweden, which has perhaps the most extensive of all available geoscientific databases for a specific site today.

6.1.2 Äspö

SKB has built one of Europe's largest full-scale facilities for field tests within geology, rock mechanics, geohydrology and groundwater chemistry on the island of Äspö off the coast of Oskarshamn: the Äspö Hard Rock Laboratory, see Figure 6.1-1. It consists of a 3,600 m long tunnel down to a depth of 460 m, see Figure 6.1-2. Conventional drill-and-blast technique was employed for the first 3,200 m of the tunnel, while the last 400 m were fullface-bored with a tunnel boring machine (TBM).

Äspö was chosen because it offers varied and undisturbed experimental conditions within a limited area. Furthermore, its nearness to the nuclear power station on the Simpevarp Peninsula offers good infrastructural facilities. Äspö has hydrological conditions, rock and fracture zones of varying character. The bedrock in the area, much of it Småland granite, is more than 1,700 million years old. The uninhabited island of Äspö is flat, about two km^2 in size and situated on the coast. The composition of the groundwater is typical for Swedish crystalline bedrock along the Baltic Sea. The surrounding sea offers relatively well-defined boundary conditions.

6.1.3 Overview of completed investigations and available data

The pre-investigations aimed at finding a suitable site for an underground rock laboratory started in 1986 in the Simpevarp area in the municipality of Oskarshamn. The island of Äspö was identified a couple of years later, and the pre-investigations proceeded up until 1990.

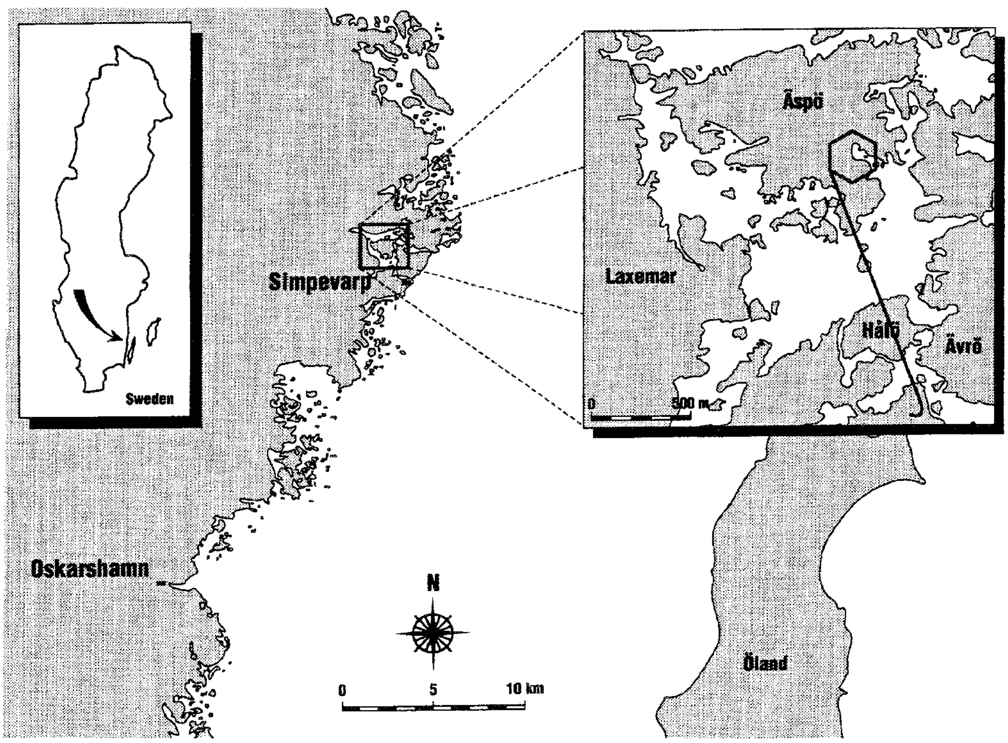


Figure 6.1-1. The Äspö HRL on the southeast coast of Sweden.

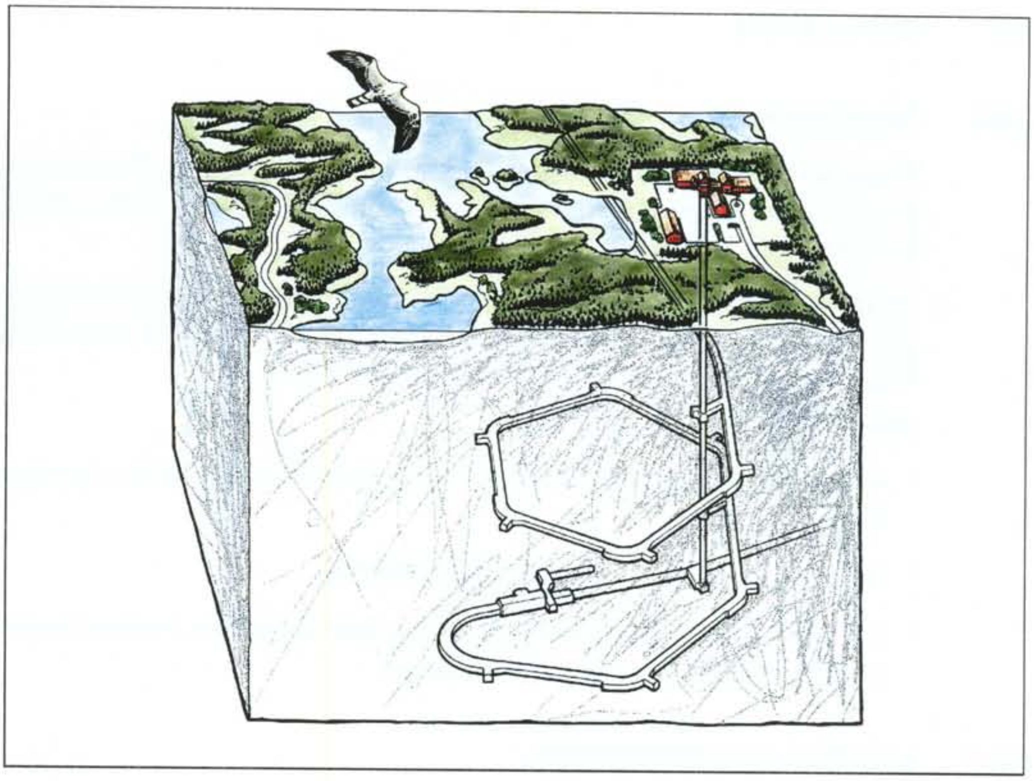


Figure 6.1-2. Design of the Äspö HRL. The length of the tunnel is 3,600 m down to a depth of 460 m. The last part of the tunnel was fullface-bored with a TBM.

This phase of the work was aimed at:

- finding a suitable site for the laboratory,
- describing the natural conditions in the rock, both regionally and locally,
- predicting the changes that would occur during construction.

The results of geological, geophysical, hydrological and geohydrochemical investigations during this phase have been reported in eight technical reports /6.1-1 – 6.1-8/. The following sections will summarize these investigations.

The construction phase began in 1990 and was concluded at the beginning of 1995 with the completion of the tunnel. This phase was aimed at:

- checking the prediction models that had been set up for geology, rock mechanics, geohydrology and geohydrochemistry for Äspö during the pre-investigation phase,
- developing methodology for detailed underground characterization,
- expanding the database for properties of granitic bedrock to improve models for description of flow and transport in fractured rock.

At present the experimental phase of the Äspö HRL is under way. A large number of experiments on different scales and within different disciplines will be carried out during the coming years.

The illustrative calculations that are presented later in this report are based on the structural model of Äspö that was available in the autumn of 1994. This means that the investigations of various kinds that were carried out during the last stage of the construction phase are not taken into account.

6.2 GEOLOGY

6.2.1 Introduction

The most important safety-related role of the rock is to provide mechanical and chemical properties so that the long-term performance of the engineered barriers is not jeopardized.

As far as mechanical stability is concerned, the repository will be situated in parts of the rock that do not comprise zones of fractured rock in which future fault movements of significance could be released.

Favourable conditions are:

- rock stresses and thermal conductivity properties normal for Swedish bedrock,
- homogenous and easily interpreted bedrock,
- access to rock blocks with few fracture zones and low fracture frequency surrounded by clear zones of weakness.

6.2.2 Location and topography

The landscape in the Äspö area is characterized by flat land forms that are interrupted by marked fracture valleys running principally in the north-south, east-west and north-west directions. The area dips slight eastward from 20–30 m above sea level in the west to sea level on the Simpevarp Peninsula and the islands of Äspö and Ävrö in the Misterhult archipelago.

The soil cover, usually glacial till, is for the most part thin and large areas consist of exposed rock in the form of gently dome-shaped more or less contiguous outcroppings. Local elevation differences are as a rule less than 5 m. Depressions between projecting rock outcroppings are usually covered by forest or mire land.

6.2.3 Completed geological investigations

Very extensive geological investigations have been undertaken for the Äspö HRL. In the pre-investigation stage, investigations were carried out within an approximately 900 km^2 large area. The goal was to design a geological model and to make geological predictions for the rock volume in which the Äspö HRL was to be built. Continuous documentation in conjunction with the blasting of the tunnel has contributed to increased knowledge of the bedrock in the Äspö area. This has in turn permitted updating of the geological model /6.2-1, 2/.

The geological investigations for the Äspö tunnel were conducted in three stages during the pre-investigation phase during the period 1986–1990. A summary account of the pre- investigation results is provided in a number of reports /6.2-3, 4, 5, 6/.

In the first stage, aerogeophysical and gravimetric investigations were carried out within a 25x35 km^2 large area west of Simpevarp /6.2-7/. Together with lineament and satellite picture studies, general bedrock mapping and structural geology studies /6.2-8/, these investigations provided the basis for an initial geological model of the Äspö area.

In the second stage, the investigations were concentrated to Äspö-Ävrö-Laxemar. Geophysical ground measurements /6.2-9/, detailed bedrock mapping /6.2-10/ and introductory drillings /6.2-11/ contributed towards a deeper understanding of the geological structure of the area.

In the third stage, the investigations were concentrated to Äspö. The purpose was now to describe in detail the rock volume where the Äspö HRL was going to be built, see Figure 6.2-1. The investigations included detailed studies of rock type distribution, fractures and other structures along exposed rock surfaces on the island /6.2-12, 13/. Detailed magnetic and electrical ground measurements supplemented the geological studies /6.2-14/ and an extensive drilling programme gave information on conditions at greater depth in the rock volume. Radar measurements /6.2-15, 16/ and seismic measurements /6.2-17/ in the cored boreholes helped in orienting the major fracture zones and establishing a structural model for the Äspö HRL /6.2-5/.

The investigations have continued in conjunction with the blasting of the tunnel for the purpose of updating the primary geological model. In the tunnel, continuous documentation has been made of e.g. rock types and structures. Cored boreholes have been drilled to locate and characterize major fracture zones more precisely. Radar and seismic measurements have been performed to orient water-bearing structures.

6.2.4 Composition of the bedrock

The bedrock in the Äspö area is dominated completely by granitic rock types of the Småland granite type. This granite occurs in different variants with regard to colour and grain size. A more basic variant goes under the designation of Äspö diorite /6.2-18/.

As inclusions in the large matrix of Småland granite there are larger or smaller plutons of basic rock types (gabbro and diorite). Smaller schlieren or lenses of a fine-grained basic rock type – designated greenstone – also occur widely in the Småland granite. In a similar manner, small residues of old volcanic rock types can also occur in the granite. A very characteristic feature of the regional bedrock in the Äspö area is the coarse-grained and typical massive granites that occur in large rounded plutons in Götemar and the Uthammar area, north and south of Äspö, respectively. These granites are younger than the Småland granite and have pushed up through it in the form of round domes called diapirs. Magnetic and gravimetric data indicate that these diapirs have a depth of about 5 km /6.2-7/.

Fine-grained red granite, and to a lesser extent pegmatite, are common features in the rock mass in the form of narrow dikes and irregular schlieren. The fine-grained granite also occurs in the form of irregular slightly larger intrusions in Småland granite and Äspö diorite.

6.2.5 Geological development and structural geology pattern

The Äspö area has undergone a tectonic evolution in several phases. Deformations have occurred both in the plastic stage in the form of folding – presumably several phases – and in the brittle, hard rock mass in the form of

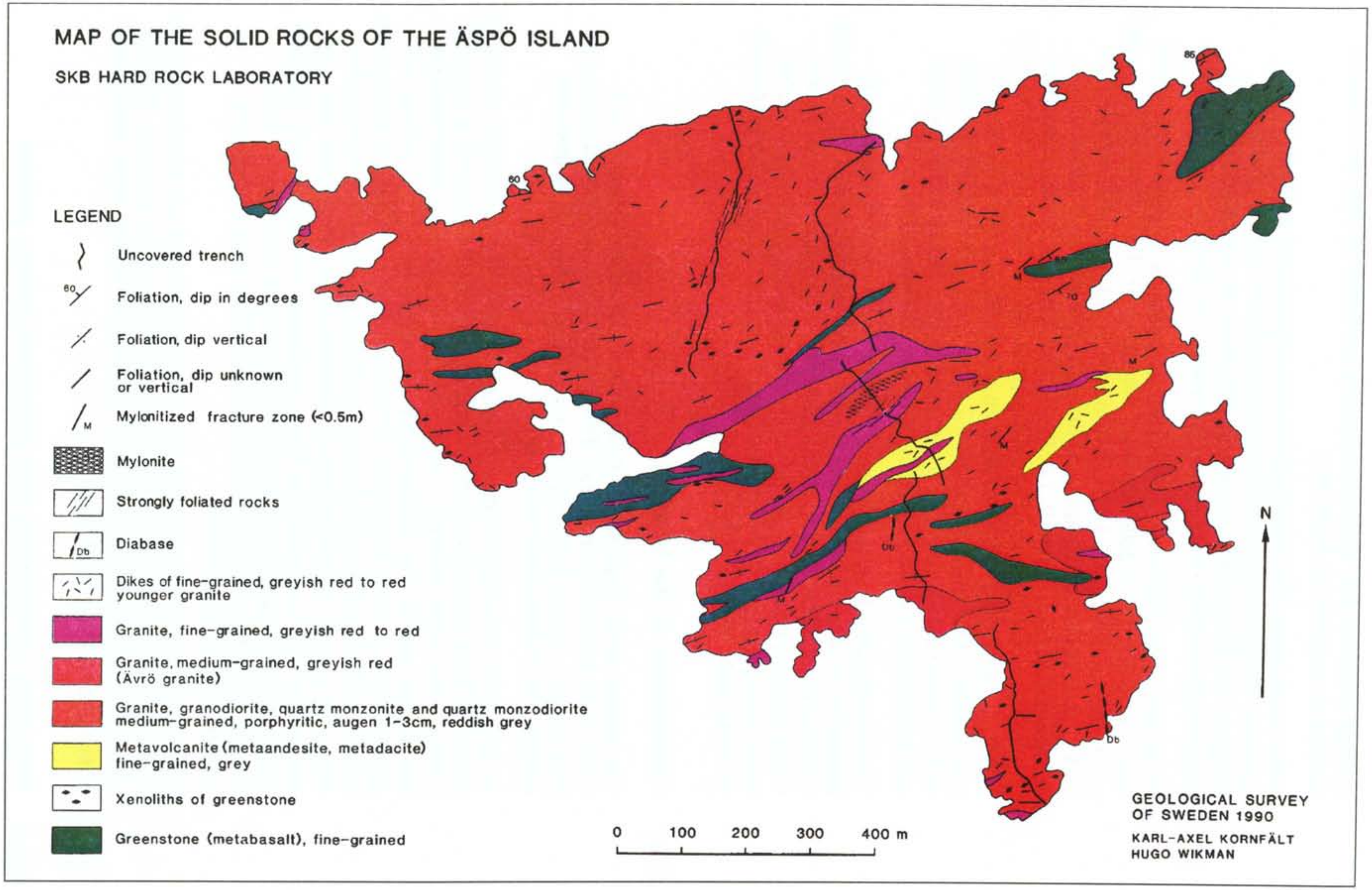


Figure 6.2-1. Bedrock map of Äspö.

fracturing and faulting. The early folding phase that can be traced today in the form of varying degree of foliation in the rock mass is uniformly oriented approximately ENE with mainly steep dips. This structure direction is completely dominant within the area and seems to a large extent to have had a controlling influence on the deformation occurring in later phases /6.2-8, 6.2-12/.

Lineaments and structures in the Äspö area have a particularly prominent orientation in the E- W and N-S directions. Structures in the NW and NE directions are also common, however. The geophysical measurements gave very clear indications of a structural pattern with the above orientation, where the magnetic measurements in particular indicated that the structures can be many km long and up to 100–200 m wide /6.2-7/. Geophysical ground measurements carried out at a later time have shown that these regional aerial indications generally correspond to fracture zones no more than around ten metres wide /6.2-9/. Oxidation (conversion of magnetite to non-magnetic haematite) has taken place around these fracture zones within an often fairly broad area, which explains the size of the aeromagnetic indication. An example of a regional structure with an approximate NE orientation is the Äspö shear zone.

The deformation of the brittle rock mass in different phases can be followed and interpreted via studies of fractures and fracture-filling material in the bedrock. Four dominant fracture directions can be distinguished, namely: E-W, N60°W, N-S and N60°E /6.2-13/. Of these fractures – which were mapped on the rock surface – about 85% have a steep dip (70–90°). Open fractures are most frequent in the direction N55°W, which agrees well with the value of the largest horizontal principal stress measured in boreholes on Äspö /6.2-19/. Quartz-filled fractures are primarily oriented N-S and E-W, and the very characteristic reddish (iron-oxide- filled) fractures appear to coincide with the direction of the open fractures at N55°W.

In the entire Äspö area, the fine-grained granite and the fine-grained greenstones are more highly fractured than the Småland granite and the Äspö diorite.

6.2.6 Geological model of the Äspö area

The geological descriptive model that is presented here is the result of a continuous updating of the primary models that were presented in conjunction with the pre-investigations for the Äspö HRL.

A regional structure-geology description encompasses mainly large regional structures, see Figure 6.2-2. These structures have been defined as having in general a longitudinal extent on the order of 10 km or more and a width of 100–300 m. The majority of these structures are only indicated by aerogeophysical and lineament studies. In some cases, supplementary studies in the form of geophysical ground surveys and drilling have contributed to more detailed knowledge on the character of the structures. This of course particularly applies to zones passed in the Äspö tunnel /6.2-20/.

As is evident from Figures 6.2-2 and 6.4-1, there is one system of structures with a N-S and E-W orientation and another where the directions NW-SE and NE-SW are the most prominent. The N-S structures are judged to be predominantly subvertical and to have a more open character, while the E-W structures can probably vary in dip from virtually vertical to medium-steep towards the

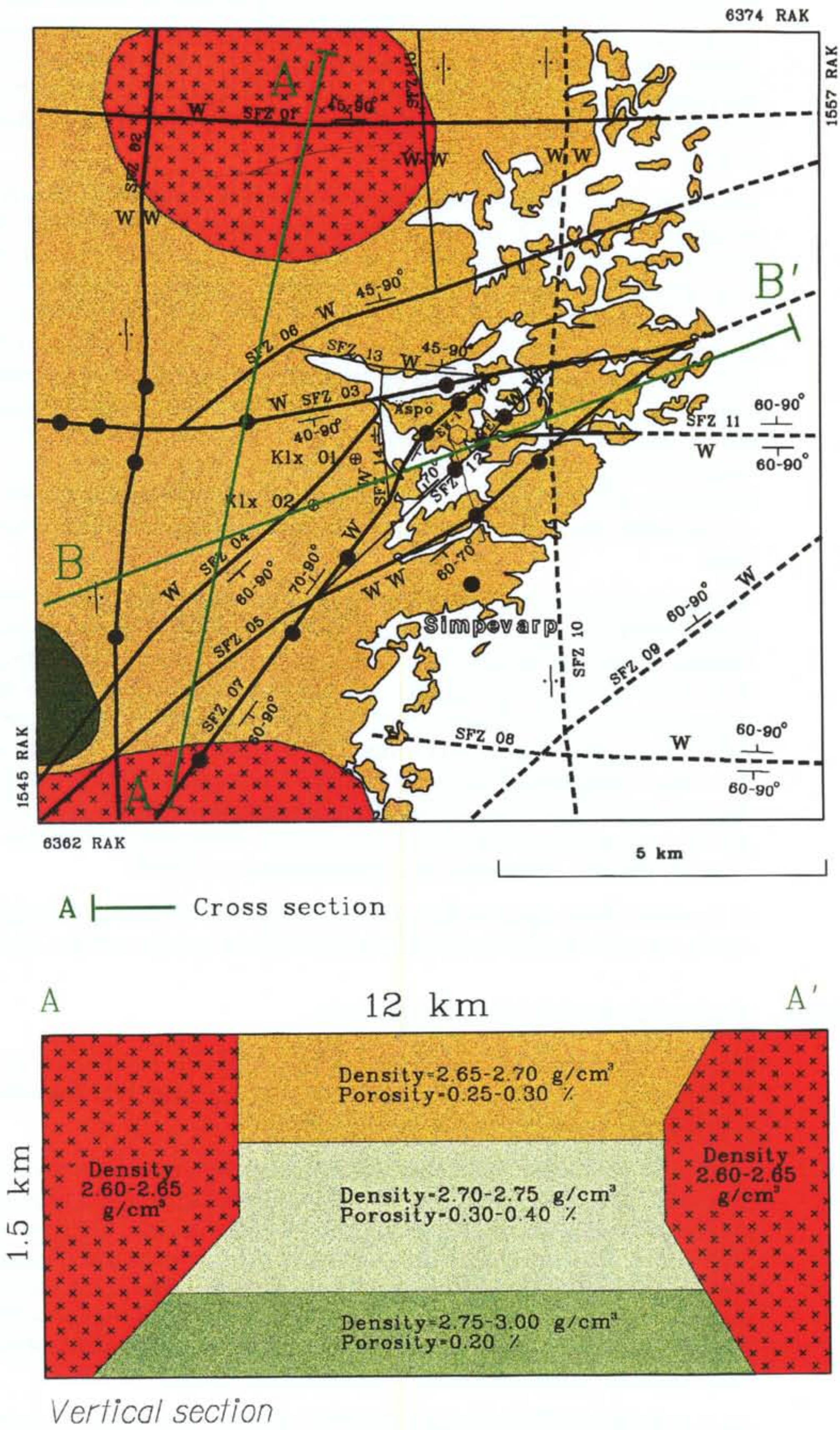


Figure 6.2-2. *Structural model and lithological model of the Äspö area on a regional scale, 12x12x1.5 km^3. See also explanation in Figure 6.2-3.*

SSE. The latter often exhibit a distinct fault character. The NW structures coincide in direction with that of the principal stress and have been found to be hydraulically conductive.

In conjunction with the pre-investigations, a "swarm" of narrow, up to a few metres wide, nearly vertical fracture zones with a predominantly NNW or NNE direction was observed. Structures of this type have been found in the tunnel to be the most water-conducting, along with the more northwesterly oriented zones.

The lithological model presented in Figure 6.2-3 is to be regarded as an interpretation of the rock type distribution in the Äspö area at a given investigation phase. It is based on geophysical data, mapping of the surface rock and the in-depth information that has been obtained by coring and studies in the Äspö tunnel. This highly schematic model is divided into three horizontal elements that represent a relatively heterogeneous rock mass with granite (Småland granite)–granodiorite with inclusions of volcanites and dikes of fine-grained granite. A transition to more basic varieties (Äspö diorite) is assumed to take place with increasing depth.

6.2.7 Rock mechanics

For assessment of the mechanical stability of the rock on Äspö, rock stress measurements were performed in three boreholes down to a depth of 950 m in conjunction with the pre- investigations. These measurements were supplemented by rock-mechanics tests of drill cores in the laboratory /6.2-21/.

Rock stress measurements were performed periodically at different levels during the blasting of the Äspö tunnel. These measurements were also combined with laboratory tests of rock- mechanics parameters.

In summary it can be said that:

- the greatest principal stress is relatively high, 30–35 MPa at repository depth, and increases with depth,
- the orientation of the greatest principal stress is subhorizontal in the NW-SE direction,
- the average ratio between maximum and intermediate principal stresses is 1.9 but can vary up to 3.

No rock stress phenomena of the rock burst type were noted down to a depth of 460 m in the tunnel.

6.2.8 Uncertainty of the model and time-dependent changes

The uncertainty of the geological model varies to a great degree depending on the input parameters. For example, for the "first-order structures" in the regional structural model, the uncertainty regarding the lateral geographic location of the structures is estimated to be between 100–200 m when only aerogeophysical and lineament indications are available. In cases where the position of the structures has been verified by ground geophysics, geological field observations and drilling, the uncertainty naturally decreases considerably (5–20 m).

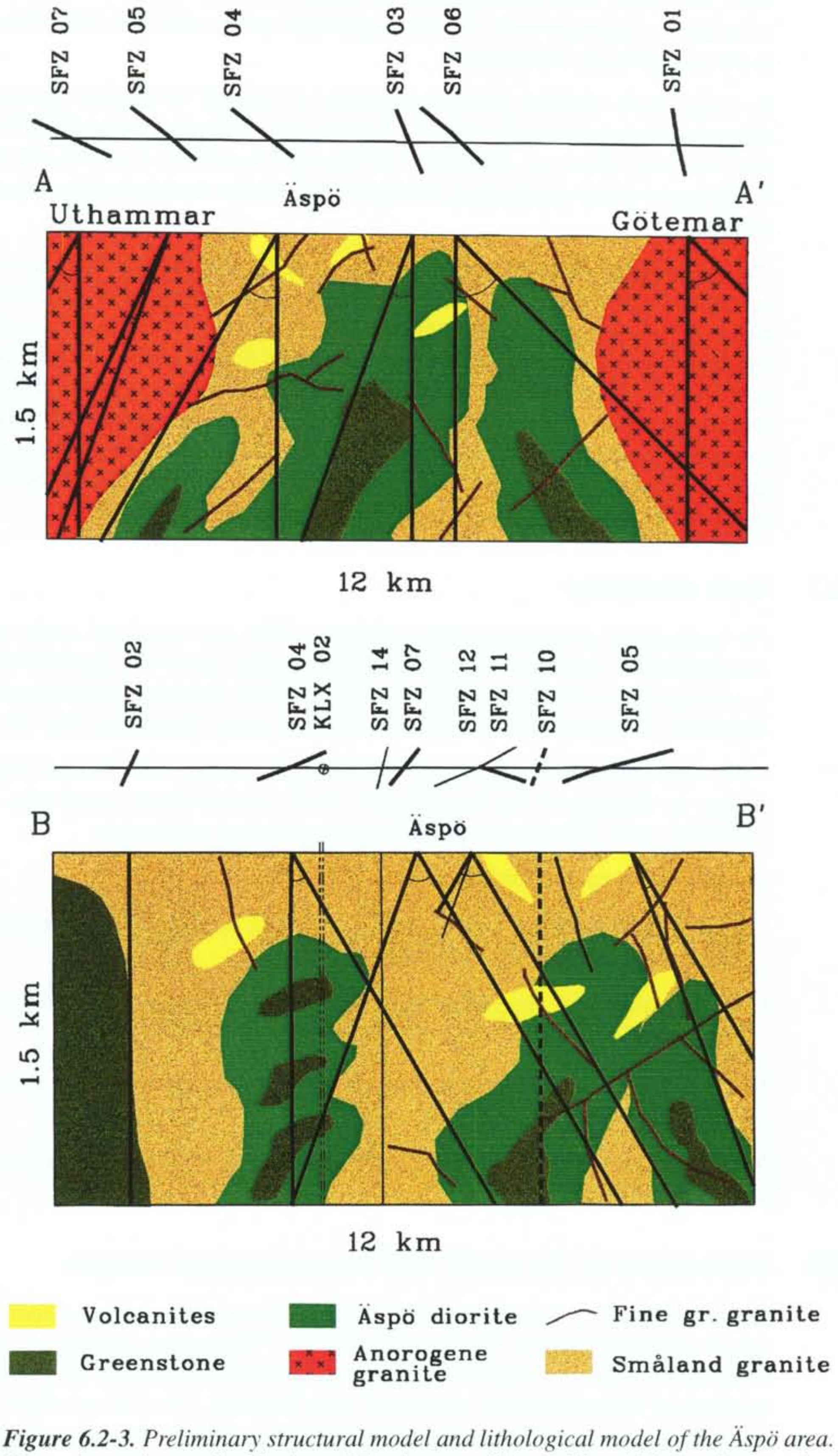


Figure 6.2-3. *Preliminary structural model and lithological model of the Äspö area. Shows the rock type distribution within the area. The position of the vertical section is shown by figure 6.2-2.*

The uncertainty in the dip and character of the zones is very great (20–30°) if only geophysical data are available. Geological field observations, and above all drilling, increase the certainty considerably but are often based on individual sections through the zone. A relatively high uncertainty with regard to dip and character therefore almost always exists for a major structure, unless very extensive investigations have been conducted.

The reliability of the lithological model is dependent on the degree of exposure of the rock and on geophysical measurements and coring work. Under favourable circumstances, it is possible to obtain a good reliability of the lithological model as regards the percentage distribution of the principal rock types. On the other hand, uncertainty is great with regard to the spatial distribution of smaller individual rock type units, such as greenstone lenses or irregular schlieren of fine-grained granite.

As far as time-dependent changes in the rock mass are concerned, changes in the present-day stress field in the form of increased or decreased horizontal stresses or rotation can cause movements in existing fracture zones. However, it is not possible to predict the direction of the stresses that could affect the Äspö area in the future. All regional structures with a large regional extent are therefore to be regarded as potential zones of movement. Displacements take place chiefly in zones with a low shear strength and a less advantageous orientation in relation to the stress field in question /6.2-5/.

6.3 GROUNDWATER CHEMISTRY

6.3.1 Introduction

The groundwater in the deep repository should be stably chemically reducing and possess properties which contribute towards

- preservation of the properties of the bentonite,
- low corrosion rate for the canister material,
- low dissolution rate of the fuel,
- low mobility and good sorption properties for the radionuclides.

By and large, groundwater chemistry conditions are expected to be favourable on most sites. At a depth of 100–1,000 m in rock with granitic composition and mineralogy and with reducing conditions, the chemical conditions will scarcely deviate more from site to site than they vary within one and the same site.

This section provides an overview of the groundwater chemistry conditions on Äspö based on the investigations that have been carried out within the Äspö project. A reference water representative of conditions at a depth of 500 m has been selected from the reported results. Uncertainties and variations are discussed in section 6.3.6, and section 6.3.4 is devoted to redox conditions.

6.3.2 Geochemical investigations on Äspö

Most of the boreholes on Äspö were sampled during the pre-investigation phase /6.2-5, 6.1-7/. Scope of analysis and sampling technique varied widely from a restrictive procedure that only included the main components in samples taken during drilling, to a very inclusive one that also included redox-sensitive trace elements, stable and radiogenic isotopes, dissolved gas, microbes and colloids. During the tunnelling phase, sampling and analysis were carried out in accordance with an established programme /6.3-2/. The results are compiled in /6.3-3/. The results of the more inclusive analyses are published in /6.3-1/.

6.3.3 Groundwater chemistry model of Äspö

The rock underneath Äspö contains water of varying origin. The most important identified classes are modern water, Baltic Sea water, sediment water, glacial water and old salt water. In identifying these end members, it has also been possible to trace their hydrochemical evolution.

During the most recent deglaciation, meltwater was transported through the rock mass down to a maximum depth of between 400 and 600 m. During subsequent fresh and salt water stages of the Baltic Sea, the water chemistry was altered dramatically on at least one occasion. Saline Littorina water displaced the lighter meltwater during the period that followed from about 7,000 years ago. The result of this is that the meltwater fraction is higher in less pervious rock volumes than in conductive zones, even though the salt content is almost the same. As far as conditions for nuclide transport and chemical stability are concerned, it is important to note that the water below a depth of about 500 m has not been significantly affected by events that have taken place since the most recent ice age. The water has thus been stagnant for on the order of 10,000 years or more. On the surface, infiltrated groundwater has washed out the salt water down to a depth of about 40–50 m during the 3–4,000 years that have passed since Äspö rose up out of the sea. A schematic illustration of the hydrochemical conditions is shown in Figure 6.3-1.

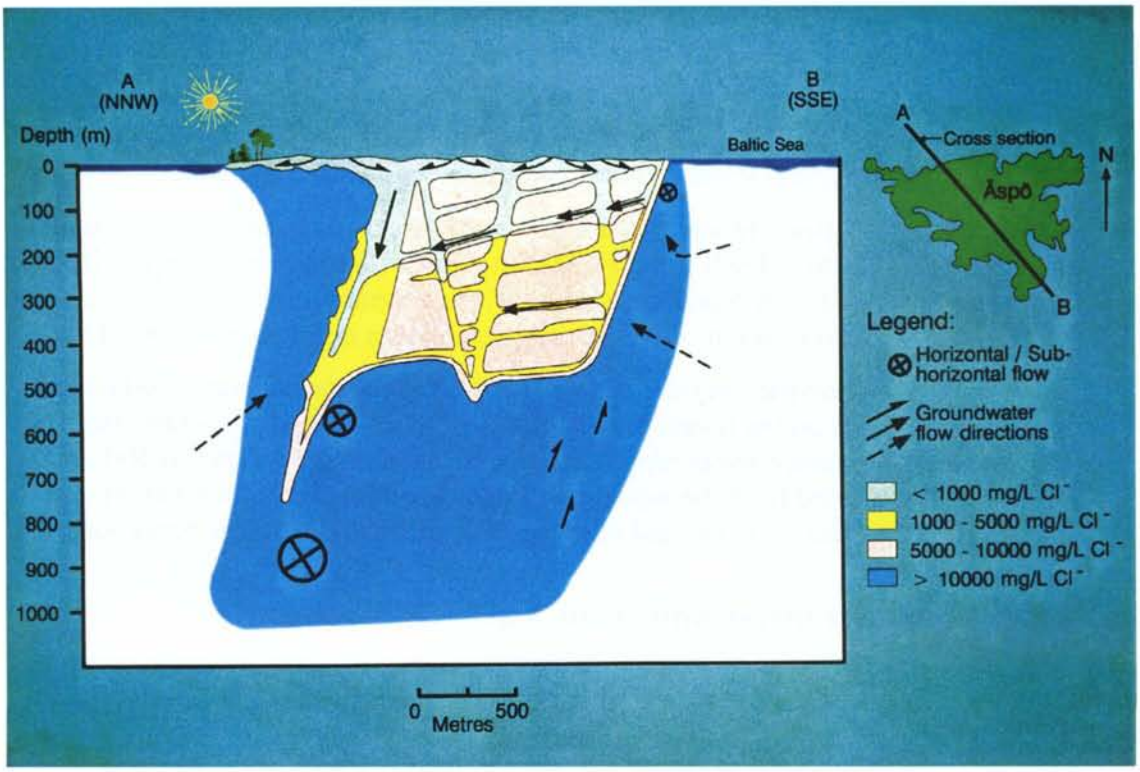


Figure 6.3-1. Groundwater chemistry model for Äspö.

Bacterial sulphate reduction occurs on a large scale in the tunnel section that runs underneath the sea from the Simpevarp Peninsula out to Äspö. The process is probably associated with the presence of bottom sediments rich in organic material. According to calculations, the sulphate reduction encountered there has produced a sulphite quantity equivalent to about 100 mg/l, which is more than two orders of magnitude greater than the sulphide quantity that normally occurs in the groundwater, see Table 6.3-1. There are sulphate-reducing bacteria at greater depths beneath Äspö, but no sign of any large-scale sulphate reduction.

A co-interpretation of hydrological, chemical and biological data shows that it is probably the presence of about 40% or more of old Baltic Sea water, which has been in contact with bottom sediments, that is responsible for the fact that this process has occurred on such a large scale. Chloride concentrations of 4,000–6,000 mg/l and TOC (total organic carbon) concentrations of 10 mg/l correlate positively with high hydrogen carbonate concentrations, low sulphate concentrations and the presence of sulphate-reducing bacteria.

6.3.4 Redox conditions

The redox potential (Eh value) is a measure of how oxidizing or reducing the conditions are. In the repository context, the water should be so reducing that uranium occurs in its tetravalent (= insoluble) form.

Virtually all of the examined groundwater samples have been found to be reducing. However, it is a combination of both chemical reactions in the bedrock and biological processes that determines the redox conditions. In a recently concluded experiment on Äspö, it was found that bacteria are primarily responsible for consuming the dissolved oxygen that is drawn down with inflowing groundwater towards the tunnel. In this context, it is the high content of organic matter in the water that contributes to the bacterial process. In other words, the redox buffer does not consist solely of reducing minerals in the rock. Through bacterial processes, dissolved organic matter will also contribute.

6.3.5 Reference water for Äspö

A reference water is supposed to be characteristic of the site and represent present-day geohydrochemical conditions at repository depth, i.e. around 500 m. A reference water for Äspö has been chosen, namely the water from borehole KAS02, at a depth of 530–535 m. The composition of the water is shown in Table 6.3-1. The uncertainties in these values can be estimated in the following way:

The chloride, bromide, iodide, sulphate, sodium, calcium, strontium and lithium concentrations vary with the total salinity. Since these substances are, with a few exceptions, the main components, they are also the ones that determine the ion balance in the water. The uncertainties in the concentrations lie within ±10%.

The calcium, silicate and fluoride concentrations are in equilibrium with the minerals in the rock and thereby relatively independent of the variations in salinity. Fluoride can always be determined with good accuracy, ±5% or better, while the potassium and silicate analyses are uncertain in salt water, ±50%.

The magnesium concentrations can be determined with high precision. However, they vary widely between different sampling points, without exhibiting any correlation with the salinity. The uncertainty has been estimated at ±50%.

The magnesium, iron and sulphide concentrations reflect the redox conditions. The analyses are reliable, but the concentrations vary. The uncertainty is therefore estimated to be ±50%, 90% and 90%, respectively, for these elements.

The hydrogen carbonate concentration can be determined as alkalinity with great precision. Due to differences between sampling points, the uncertainty caused by the variability is estimated to be ±50%.

The nitrogen compounds nitrate, nitrite and ammonium and the phosphorus concentration lie close to the detection limit for the analysis methods. The uncertainty is estimated to be ±100%.

pH ± 0.1 and Eh ± 25mV.

Table 6.3-1. Chemical composition of reference water for the deep repository on Äspö. This is taken from borehole KAS02. All concentrations are given in mg/l except pH and Eh. The uncertainty in the values is given in parentheses. The uncertainty is a combination of measurement uncertainty and uncertainty in the interpretation model.

Component	Concentration	Component	Concentration
pH	8.3 (0.1)	Eh	–300 (25) mV
sodium	2100 (200)	hydrogen carbonate	10 (5)
potassium	8.1 (4)	fluoride	1.6 (0.1)
calcium	1900 (200)	chloride	6400 (600)
magnesium	42 (20)	bromide	42 (4)
strontium	35 (4)	iodide	0.5 (0.1)
lithium	1.0 (0.1)	sulphate	550 (60)
manganese	0.29 (0.15)	sulphide	0.18 (0.16)
iron	0.23 (0.2)	phosphate	0.009 (0.009)
silicate	4.1 (2)	nitrate	0.04 (0.04)
ammonium	0.03 (0.03)	nitrite	0.003 (0.003)
TOC	1.0		

6.3.6 Uncertainty and variations in time

The contents of a reference water for Äspö were described in section 6.3.5, including the uncertainties in the concentrations of the components. In certain cases it is a combination of measurement uncertainty and conceptual uncertainty in the interpretation model. In using these data for a general description of a repository site, the following should be observed:

Each groundwater chemistry sampling is associated with some risk of contamination. This risk can be minimized by careful selection of sampling method and equipment. Nevertheless, it is impossible to completely rule out disturbances in the form of contaminants which in many cases derive from the actual drilling of the deep cored boreholes. Among the parameters in Table 6.3-1, it is mainly TOC (total organic carbon) which is uncertain for this reason.

The redox- and pH-sensitive parameters pH, Fe, sulphide concentration and carbonate concentration have a greater relative uncertainty than other main components because they can be affected by pressure changes and atmospheric oxygen in connection with sampling. Measurements at depth in the rock have sometimes given pH values that have differed by up to one pH unit from measurements on the ground surface. The difference has often been around 0.3 pH unit.

Salt water at great depth in the bedrock, as at a depth of 500 m in Äspö, has a character that indicates that it moves very slowly. The interpretation on Äspö is that the conditions that have prevailed since the last deglaciation, about 12,000 years ago, have only given rise to changes down to a depth of 500 m. Near the ground surface, down to a depth of about 50 m, the salt water has been washed out completely during the last 3,000 years. At a depth of 500 m, a change to fresh water might possibly take place in conjunction with the deglaciation phase following a future ice age.

6.4 GEOHYDROLOGY

6.4.1 Introduction

Since the groundwater in the rock is virtually the only transport pathway for radioactive substances to escape from the repository, all conditions that have to do with the transport of solutes (dissolved substances) with the groundwater are of potential importance.

The most important factors are:

- The groundwater flow at repository level, which is of importance for canister life, the out-transport rate for the radioactive substances and possibly for the dissolution of the fuel.
- The transport time for solutes from the repository to the biosphere.

The geohydrological investigations for the Äspö HRL have been very large in scope. The goal in the pre-investigation phase was to devise a geohydrological model and to make geohydrological predictions for the rock volume in which the Äspö HRL was to be built. In conjunction with the blasting of the tunnel, continuous documentation and supplementary investigations at tunnel level have contributed towards greater knowledge of the bedrock in the Äspö area. This has in turn permitted an updating of the geohydrological model.

6.4.2 Geohydrological investigations

Geohydrological investigations have been carried out in the Äspö area to identify principal hydraulic conductors in the rock, their hydraulic properties, and boundary conditions for the hydrological system /6.4-1, 2/. Among others, the following types of investigations have been conducted:

- Collection of hydrometeorological data through continuous recording
- Measurement of the flow distribution along boreholes during pumping (flow logging)

- Injection tests in boreholes
- Groundwater flow measurements in sealed-off borehole sections (dilution measurements)
- Hydraulic interference tests (test pumpings)
- Tracer experiments
- Flow measurements in tunnel

A compilation of available geohydrological data from Äspö is given in /6.4-3/.

6.4.3 Groundwater table and hydrometeorology

Groundwater and pressure head levels in borehole sections are recorded continuously on Äspö, during 1994 in about 60 boreholes /6.4-4/. The purpose of these recordings is to document groundwater conditions before, during and after tunnelling, and to measure responses caused by other hydraulic events such as cross-hole tests. Natural level variations are caused by meteorological factors but also by tidal effects. Other hydrometeorological data that are collected include precipitation, sea water level, air temperature and potential evapotranspiration, i.e. the theoretical evapotranspiration from a surface completely covered by a homogeneous vegetation surface.

Äspö consists of several relatively small drainage basins, on the order of 10,000 m^2, and no major surface watercourses drain water from the island. The mean value for precipitation on Äspö is 650 mm/y. The mean value for potential evapotranspiration is 616 mm/y, and the calculated actual evapotranspiration is 490 mm/y /6.4-5/. Estimates of the average groundwater recharge to the deeper hydraulic system in the rock varies between a few mm/y to 125 mm/y /6.4-6, 7, 11/. In general it can be expected that the groundwater recharge is concentrated within areas with major hydraulic conductors that are exposed on the surface or that have contact with permeable soil layers.

6.4.4 Hydraulic properties of the bedrock

The hydraulic material properties of the bedrock vary considerably /6.4-1/. Most water is transported in large hydraulic conductors, see Figure 6.4-1. Dominant hydraulic structures are determined by the overall structural geology pattern. Based on the pre-investigation results, the Äspö area is considered to consist of three areas, geologically speaking /6.4-3/. These areas run as parallel bands in the north-easterly direction, see section 6.2. The middle band consists of the Äspö shear zone. There are good hydraulic conductors, particularly in the border zone between the areas, such as for example NE-1 and EW-1. Good hydraulic conductors also occur in the north-northwesterly direction, and it can be assumed that they partially connect the NE zones and the EW zones hydraulically.

The permeability of the rock is interpreted to a large extent from measurements in boreholes, where hydraulic transmissivity is estimated along the length of the borehole. In water injection tests, transmissivity values are interpreted from pressure head responses between packers. This has been done at Äspö with a packer distance of 3 and 30 metres. In flow logging, transmissivity is interpreted from inflow measurements along the borehole at the same time as the entire borehole is pumped relatively near the surface. Interpreted values for

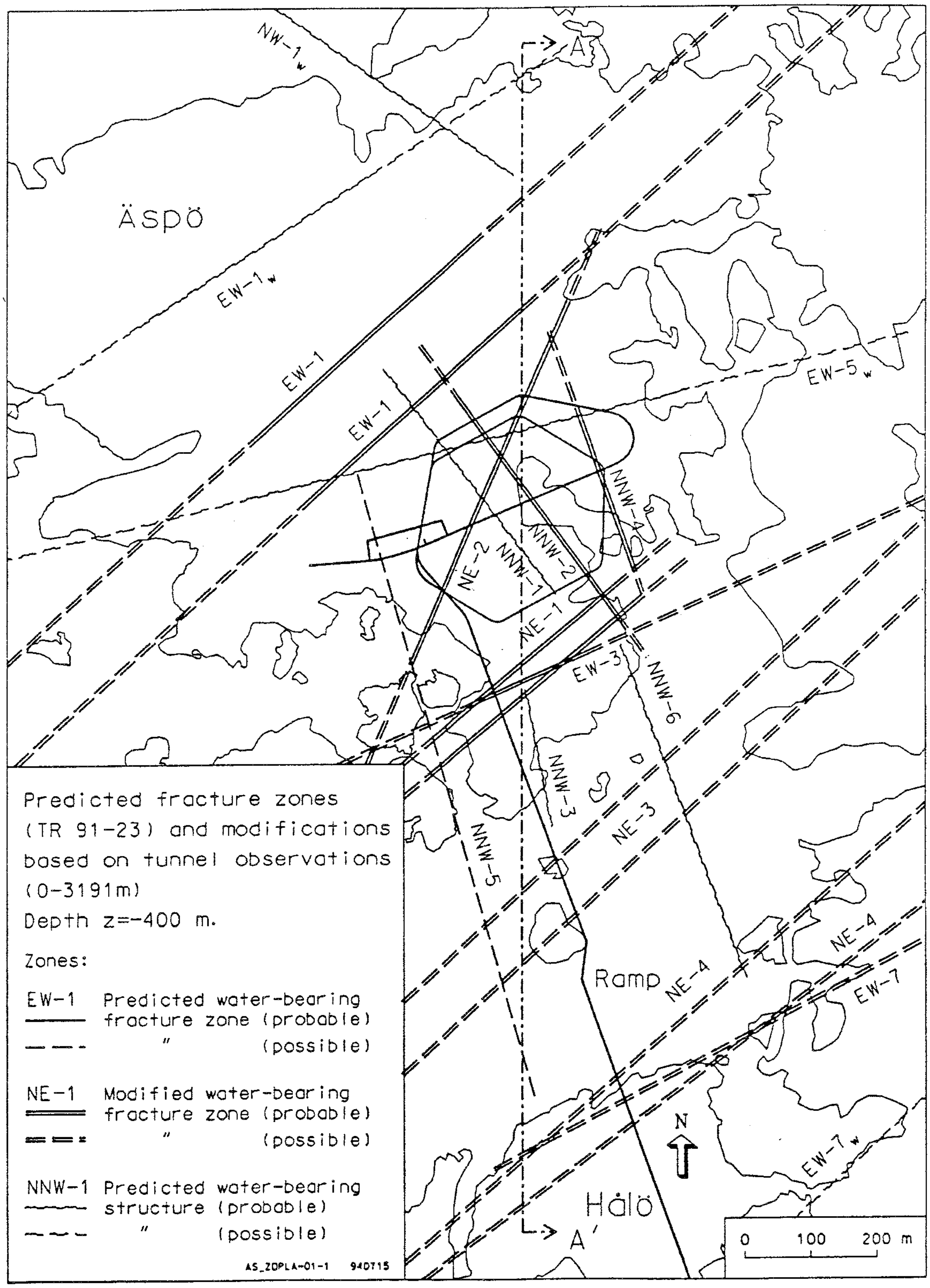


Figure 6.4-1. *Interpreted fracture zones on Äspö intersecting a horizontal plane at a depth of 400 metres /6.4-3/. Interpretation model from autumn 1994.*

transmissivity in zones are available from hydraulic interference tests. The transmissivity values in the zones are as a rule on the order of 10^{-5} – 10^{-4} m^2/s /6.4-3/. Interpreted transmissivity values from single-hole tests in the rock outside the zones are several orders of magnitude lower and exhibit large spatial variations /6.4-3/. In certain cases, there are also interpreted values for storage coefficients. Flow porosities have been interpreted above all from tracer tests. The LPT2 test gave porosity values in the range 0.0002–0.001 for fracture zones /6.4-8/.

6.4.5 Boundaries in the groundwater flow system and discharge areas

The local topography is judged to dominate the relatively superficial flow system (about 50–100 m deep), while the flow at greater depths is generally influenced more by regional boundary conditions and the large-scale flow geometry (fracture zone structure). However, in the event of major hydraulic disturbances such as tunnelling, this pattern can be expected to be altered so that superficial groundwater is drained to greater depth. This process is being followed by means of hydrogeochemical measurements in sealed-off borehole sections /6.4-9/. The groundwater table for a groundwater reservoir in the fractured rock corresponds to an integrated piezometric pressure head in the geohydrological system. The groundwater table on Äspö has been drawn down by about 90 m at most by the construction of the tunnel /6.4-2/.

The horizontal extent of the groundwater that passes at a depth of about 500 m is relatively difficult to determine. Even if the large-scale regional hydrological system proceeds from inflow on the mainland (the Småland highland) to outflow in the Baltic Sea, the recharge and discharge areas for a repository at a depth of 500 m will probably be determined by the pattern of hydraulic conductors in the immediate environment /6.4-10/. Dominant persistent vertical structures with good hydraulic conductivity probably serve as hydraulic boundaries for a repository area. In practice, this entails including an area that is judged to be "sufficiently large" for model studies of the hydraulic system. The results of model calculations indicate that the hydraulic structures within a distance of about 800 m from the central portions of the Äspö tunnel are the ones that mainly control the flow /6.4-11/. Figure 6.4-2 shows the area that is used for model calculations of the groundwater flow within the Äspö project, together with surface watercourses and surface water divides in the area.

6.4.6 Geohydrological model of Äspö

The predicted hydraulic structural model from the pre-investigations /6.4-12/ has thus far only been modified to a small extent since the tunnelling work was completed and new information was obtained. Strike and dip were modified for certain zones, and a zone in the north-northwesterly direction was added to the model /6.4-2/. Quite a bit of interpretation work remains, however. Interpreted transmissivity values for conductive structures, see Figure 6.4-1 based on results from the pre-investigations as well as results obtained during the tunnelling work, are given in Table 6.4-1.

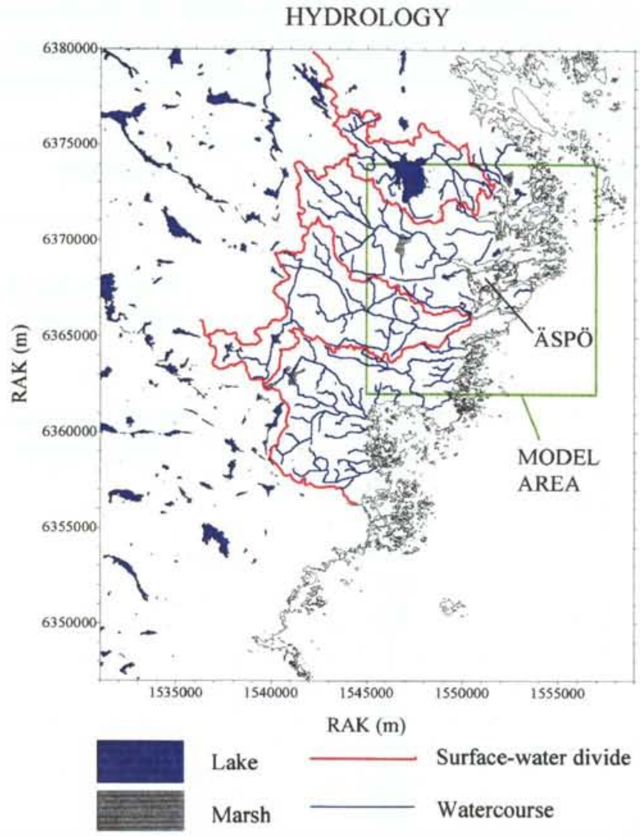


Figure 6.4-2. *Water divides and watercourses in the Simpevarp area /6.4-3/.*

6.4.7 Uncertainty and time-dependent changes

Geohydrological uncertainties are largely found in the determination of the geohydrological structure, and not least in the relationship between geological indications and water-bearing structures /6.4-13/. As a rule, cross-hole tests are required in order to determine hydraulic connectivity and properties in principal hydraulic conductors with certainty /6.4-14/. Uncertainties in interpreted material properties from hydraulic tests in general are well- documented /6.4-15/ and apply to data from Äspö as well. For example, Cooper's and Jacob's method is used for the recovery phase to interpret injection tests on Äspö /6.4-16/, while several alternative interpretation models are also possible. Further, there are generally not enough data available on material properties in large hydraulic conductors to give an idea of the spatial variability in the zones /6.4-17/.

An example of long-term natural changes that occur with time in the hydrological system is the continuous process of postglacial isostatic land uplift. Laxemar, on the mainland just west of Äspö, rose up out of the sea about 7,000 years ago while Äspö started to rise out of the sea about 3,800 years ago. This means that the hydraulic boundary conditions for the area are time-dependent and that the natural flow conditions in the area are of a transient nature. Short-term natural variations mainly consist of seasonal and tidal effects, which are noticeable in most measurement sections /6.4-4/.

Table 6.4-1. Interpreted transmissivity values in fracture zones. Transmissivity values in parentheses indicate insufficient underlying data. The table is given in a more detailed form in /6.4-3/. The location of the fracture zones is shown in Figure 6.4-1.

Fracture zone	Transmissivity $[m^2/s]*10^{-5}$	Comments
EW-1	2.0	
EW-3	(2.0)	0–200 m depth
EW-3	0.05	>200 m depth
EW-5	2	
EW-7	0.6	
NE-1	45	0–300 m depth
NE-1	20	>300 m depth
NE-2	0.5	
NE-3	43	
NE-4	34	
NW-1	0.7	
NNW-1	2.0	
NNW-2	7	
NNW-3	2	
NNW-4	14	
NNW-5	(5)	
NNW-6	(5)	
NNW-7	2.7	

6.5 TRANSPORT PROPERTIES IN THE ROCK

6.5.1 Introduction

The transport of dissolved radionuclides is influenced to a high degree by their chemical properties. Sorption of dissolved radionuclides on mineral surfaces in the rock prevents, or at least retards, their transport away from the repository. This allows time for the radionuclides to decay in the rock.

Dissolved radionuclides are sorbed not only on the surfaces of the fractures in which the groundwater flows. By diffusing into the rock matrix's interconnected system of microfissures with virtually stagnant water, the radionuclides escape the flow of water and are exposed to a much larger available sorption area. Diffusion into the microfissures and sorption there makes the dominant contribution to radionuclide retention.

The properties in the rock that are of importance for nuclide transport are discussed in this section. Results from Äspö are given special attention.

6.5.2 Sorption and matrix diffusion

Sorption encompasses a variety of different mechanisms /6.5-1/. Some are partially irreversible. The strength of the sorption is highly dependent on the charge of the ions, hydrolysis, and the possible presence of complexes with strong complexing agents. It is therefore essential to know the groundwater's pH, redox conditions and content of complexing agents such as humic and fulvic acids.

Ion exchange is an important sorption mechanism for e.g. Cs^+ and Sr^{2+}. The salinity of the water is therefore also of great importance. High salinity reduces the sorption of Cs^+ and Sr^{2+}.

The minerals that comprise the actual substrate for sorption have different capacities for sorbing radionuclides. Certain minerals are, for example, good ion exchangers, while others are not. Distribution coefficients, K_d, are used to quantify the sorption so that it can be used for transport modelling. They are determined experimentally in the laboratory by using minerals and water compositions that are typical for the repository site and then trying to vary the important parameters, such as pH, ionic strength and concentration of radionuclides. The K_d values that are to be used in the safety assessment are chosen so that the retardation in radionuclide transport is not overestimated /6.5-2/. Complexing with humic and fulvic acids can reduce the sorption of some of the radionuclides. The size of the reduction depends on how the radionuclide behaves as a dissolved ion and the content of humic substances in the groundwater. The chosen K_d values are compensated for this /6.5-2/.

Even if the K_d values are chosen conservatively for the safety assessment, they can be compared with those obtained from field tests in a natural environment. A number of K_d values have so far been reported for Äspö /6.5-3/.

Matrix diffusion of any dissolved radionuclides could take place to the microfissure structure in the rock surrounding the large fractures and zones. This has a significant retarding effect on both non-sorbing and sorbing radionuclides. The most important parameters that determine diffusion into the rock matrix are the specific area available for sorption, the diffusion coefficient and the diffusion porosity. There is still no good method for determining the specific area, but investigations are currently under way within the Äspö programme.

6.5.3 Colloids

Sorbing radionuclides could in principle be transported with the water if they adhered to colloidal particles in the groundwater /6.5-8/. The median concentration of colloids is less than 0.05 mg/l. They consist of inorganic particles, for example silicon, iron hydroxide and clay, and can sorb radionuclides. If the uptake of radionuclides on colloidal particles is reversible, it is of no consequence for radionuclide transport, since the nuclide is then turned over to the rock somewhere along the flow path. If, on the other hand, the nuclide should become stuck irreversibly, the nuclide will be transported with the particle and, at worst, not be retarded at all by sorption in the rock. Laboratory tests verify that radionuclides really can form colloids, and that the sorption on the mineral colloids is largely reversible. The strength of the sorption is roughly equivalent to measured K_d values for corresponding minerals and solutes /6.5-4/.

In summary, it can be concluded that radionuclides in the groundwater can occur as colloids and that the possibility cannot be entirely ruled out that a small portion will be irreversibly bound to mobile natural colloidal particles. However, calculations show that even for such an extreme case the consequences are of no importance for safety. The evaluation is summarized in /6.5-5/.

6.5.4 Bacteria

Bacteria can influence a number of parameters of importance for the isolation of radioactive waste, e.g. migration, solubility and gas formation /6.5-6/. The bacteria can be of advantage, for example by contributing to the chemical reduction of oxygen and radionuclides, or disadvantage, for example by reducing sulphate to sulphide.

Careful analyses of the groundwater show that bacteria also exist at great depth. All species have not been identified, but methane bacteria and sulphate-reducing bacteria have been found /6.5-6/. The environment is poor in nutrients. Laboratory tests show that bacteria can take up radionuclides. In principle, radionuclides could accompany bacteria in the same way as they accompany other colloidal particles in the groundwater. However, the concentrations of microbes are very low; for example, in the deep groundwater on Äspö, concentrations between a couple of million and ten thousand bacteria per ml have been measured of /6.5-7/.

The importance of bacteria transport for safety has been analyzed in the same way as for inorganic colloids /6.5-5/. The conclusions are the same, that it is of no importance for the far field barrier.

6.5.5 Uncertainty and time-dependent changes

The concentration of colloids in deep groundwaters could theoretically increase on the retreat of the ice sheet following an ice age. However, new studies show that the already low levels will persist even in this case /6.5-8/.

In summary, it can be concluded that laboratory tests and field tests show that the sorption of radionuclides on mineral surfaces and diffusion into the microfissures in the rock are robust retardation mechanisms that will not be appreciably affected by any possible future changes in the chemical composition of the groundwater.

6.6 UNCERTAINTIES IN THE SITE MODEL

6.6.1 Introduction

The description of a site – whether it be a geohydrological, geochemical or geological description – is associated with uncertainties. Since site investigations are by necessity always limited in scope, such uncertainties will always exist.

6.6.2 Representativeness of site characteristics

The applicability of the geological model to Swedish crystalline basement

The regional geological model for the Äspö area can in the main be said to be representative of conditions in Swedish crystalline basement rock in general, particularly as far as granitic sections are concerned. In terms of lithology, different types of granite dominate completely, and in terms of structural geology, the granite mass can be seen to be chopped up into sub-units by major structures spaced at a distance of 3–5 km. There is greater lithological heterogeneity and a slightly higher fracture frequency on the island of Äspö where the HRL has been built than in surrounding parts of the Äspö area, giving wider opportunities for different experiments.

The applicability of the geohydrochemical model to Swedish crystalline basement

The geohydrochemical conditions in Swedish crystalline basement rock were investigated in conjunction with SKB's study site investigations in the early 1980s, as well as at Stripa, Finnsjön, Laxemar and on Äspö. In all of these investigations, the analyses have been focused on the components deemed to be of the greatest importance for the long-term safety of the repository.

The Äspö area represents a hydrochemical situation that can be considered to be characteristic of the Baltic Sea coast, see section 6.3. Similar conditions have been encountered at Finnsjön and in SFR. Further away from the coast one can expect to find non-saline (0–1,000 mg/l Cl^-) groundwater even at great depth, i.e. further down than 500 m.

Carbonate content, iron content and sulphate content can vary locally as a consequence of bacterial processes, which can occur within a limited area on a site. On the other hand, redox conditions appear to be nearly identical on all investigated sites, Eh -200 –> -400 mV, i.e. reducing conditions. The pH value is usually in the range 6–9.

Concentrations of particulate and organic matter vary widely between the sampling points, but are very similar within the investigated sites. On the other hand, a general decrease in the levels with depth can be observed.

The applicability of the geohydrological model to Swedish crystalline basement

If saline groundwater occurs in the bedrock, this reflects the fact that the groundwater is slow-moving or even stagnant. These conditions exist on Äspö and this can be regarded as typical of a coastal location along the Baltic Sea. Existing discharge areas, sea or sea bays, are naturally also typical of a coastal location.

The frequency and hydraulic properties of the set of fracture zones that has been identified on Äspö does not differ substantially from what can be expected on another site in Sweden. As far as the hydraulic conductivity of the rock mass is concerned, SKB has often found a decrease with depth in the study site investigations. This is not as clear for Äspö.

Flat fracture zones have been identified on several sites in Sweden, e.g. Finnsjön, and their importance for the groundwater situation is considerable /6.6-1/. It has furthermore been asserted that they should occur with a certain frequency in Swedish bedrock /6.6-2/. No flat zones have been encountered in the Äspö area.

The narrow NW-NNW structures, which have proved to be hydraulically important on Äspö, probably have their counterpart in many other granite plutons in the Swedish crystalline basement.

6.6.3 Integrated site model

A number of models have been set up for the bedrock around Äspö. They cover e.g. rock types, geological structures, groundwater chemistry, geohydrology and mechanical stability. The models developed on the basis of the pre-investigations were presented in /6.6-3/. To test the models, detailed predictions were also made of the expected data that would be collected while the facility was being built. The final evaluation of the reliability in the Äspö investigations will be reported for the most part during 1996, but based on preliminary conclusions the geological, geochemical and geohydrological models for Äspö have been presented in sections 6.2 to 6.4.

Within the Äspö project, the close linkage between the different subject areas has been essential for developing a well-integrated site model for Äspö. The site model is the point of departure for a site-specific safety assessment and provides necessary information for describing the natural barrier in the repository system.

The importance of the site or the site model for long-term safety has been the subject of many studies, such as the SKB 91 safety assessment.

6.6.4 Uncertainty in the site description

The description of a site is associated with uncertainties, whether it be geohydrology, geochemistry or geology in crystalline rock. This has been discussed for each subject area in sections 6.2 to 6.4. This means there is also room for alternative interpretations of e.g. the fracture zone geometry. The Äspö project has worked with a single descriptive model ever since the start of the project in 1986. This site model, which covers approximately 1 km^3 of rock around the laboratory, has been continuously updated as the pre-investigations and the investigations during construction have proceeded. In this way, confidence in the model has gradually increased as the uncertainty interval has decreased. An important tool for building confidence is carrying out large-scale pumping tests in the area and measuring responses in the available boreholes. Two such long-term pumping tests have been performed so far.

A special classification of fracture zones is used within the Äspö project. The fracture zones are designated "certain, probable or possible". This provides some basis for performing the variation analyses of the importance of the fracture zones for the description of e.g. nuclide transport. A special difficulty in the characterization of the rock is presented by the relatively water-conducting fracture zones in the NW-NNW direction mentioned in previous sections. They are narrow and difficult to detect. It should therefore be assumed that additional zones in this direction may exist with similar hydraulic properties, and the effects of such zones should be examined in the analysis of nuclide transport.

A descriptive site model of the Äspö area has been developed on a larger geometric scale as well ($12 \cdot 12 \cdot 1.5$ km^3). Here the uncertainty is naturally greater with regard to, for example, the characteristics of fracture zones and rock types at depth. The classification of fracture zones mentioned above is not as detailed, and the dip of zones is given with a large uncertainty interval.

6.6.5 The potential importance of uncertainty for long-term safety

The uncertainty in the site model is of importance for the description of the natural barrier in the deep repository system. Uncertainties regarding the existence of water-bearing structures and their direction etc. with depth mean that many flow paths from the repository to the biosphere are possible. However, estimates of uncertainty intervals within the framework of the site description make it possible to define different variation cases in the safety assessment. These cases can be used to study the consequences of the uncertainties. Many variation cases of this type were carried out in SKB 91 /6.6-1/, and in particular the influence of flat structures was studied. They were found to be of great importance for the groundwater flow around a deep repository.

A complete safety assessment should contain an account of consequences of uncertainties in the site description. In SR 95, however, only illustrative calculations are given for a base case.

Various studies have been conducted in recent years to determine the importance of the site information – the field database – in describing relevant measures for the far-field barrier and above all how the uncertainty in the results is affected by a gradually growing quantity of information. Some of these studies have been limited to geohydrological modelling and borehole tests, e.g. /6.6-4/, while others have been broader in scope and have studied dose releases in the biosphere from a waste repository /6.6-5/.

Furthermore, a more concrete project has been carried out to show how the description of Äspö has changed from 1990, when the pre-investigation phase was concluded, to 1995, when the construction phase of the Äspö HRL was concluded /6.6-6/. The aim has been solely to describe Äspö with relevant measures for the far field under natural conditions and not how, for example, the interpretation of different fracture zones or groundwater salinities have changed during the course of the Äspö project. Concretely, the project has shed light on how much better the far field for a planned deep repository can be described in a safety assessment once you get down into the rock. The following were chosen as relevant measures for the far field:

- water transport times from a theoretical repository level in the rock to the biosphere,
- location of the discharge area, and
- groundwater flows in fracture zones where dilution measurements under natural conditions are available.

The conclusions are closely linked to these measures.

It should be pointed out that this evaluation may have been carried out too early, since the Äspö project has not yet finally updated its geological, geochemical and geohydrological models. The work is based on preliminary interpretation models, and these will probably change before the construction phase can be regarded as concluded. However, the preliminary results obtained suggest that the exhaustive pre-investigations that have been conducted on Äspö in themselves give a very good picture of geohydrological conditions around a deep repository. The picture of the existence and horizontal extent of fracture zones and the hydraulic properties of the zones has not yet changed dramatically during the construction of the laboratory, which also directly affects the description of the far zone under natural conditions. The exact position of fracture zones is not so important for the performance of the rock barrier from a safety point of view. The study also shows that flow paths and the position of discharge areas are more sensitive measures for the far field with respect to different interpretation models. The dominant transport pathways are the same and have only been changed to the extent that the zones emerge on the surface differently.

7 PLACEMENT AND PROGRESSIVE CONSTRUCTION OF THE REPOSITORY

The repository design in Chapter 5 is of a general nature, i.e. independent of any detailed knowledge concerning the specific properties of the site. In practice, the design of a repository always has to be adapted to the specific site. This chapter describes the site adaptations that are relevant to the safety report. These adaptations may have to do with how different repository sections are situated in relation to each other, how tunnels/shafts and deposition areas are located, or how deep down in the rock different facility sections are located. Adaptations may also be necessary with regard to fracture zones or zones of weakness, rock quality, groundwater flow paths or biosphere recipients for deep groundwater.

The chapter also discusses the possibilities for adaptation during the course of construction, alternative design and the residual design freedom that exists for future optimization.

***In this report**, the adaptation of the repository to the site, i.e. to conditions on Äspö, has only been partially realized.*

7.1 INFLUENCE OF SITE ON REPOSITORY LAYOUT

7.1.1 General

The description of the repository that is given in Chapter 5 consists of a detailed description of the size and shape of tunnels, deposition holes and rock caverns, plus a rough layout of their location relative to each other. Shapes and sizes are only to a small extent dependent on the properties of the site, while the layout of their relative location is highly site-dependent.

7.1.2 Relative location of repository sections

The location of different repository sections is determined firstly by taking into account rock and building conditions. An important question is how and where the passage from one rock block to another is to take place. A fundamental requirement in this respect is that stable tunnels and shafts can be built. Furthermore, water seepage must not be excessive.

Within the rock blocks for deposition of spent nuclear fuel, the layout will be determined by the temperature and thermal conductivity of the rock, so that the bentonite buffer is not exposed to a higher temperature than allowed, see further section 7.2.

7.1.3 Shape and size of rock chambers

The site dependence of the shape and size of the rock chambers is primarily associated with rock stresses and rock strength. Rock strength is in turn determined by the strength properties of the rock type and its structure of discontinuities. The stresses at a depth of 500 m in Swedish bedrock are not generally of such a magnitude that they exceed the rock strength in tunnels, deposition holes and rock caverns with the geometries chosen in the design described in Chapter 5. The likelihood of adjustments is deemed to be greatest in the case of the large rock caverns in the central area and in the area for other long-lived waste.

7.1.4 Experience from layout of tunnels and rock caverns in the Äspö HRL

A design based on data from Äspö is described in /7.1.1/. The rock conditions there have not given reason for any change of the size and shape of the rock caverns. The layout has been determined with regard to the uneven composition of the rock mass, the local stress field and the orientation of water-conducting discontinuities. In this special case, account has also been taken of the existing tunnel system and the possibility of connecting the central area under ground with the surface facility.

The result, based on present-day knowledge of the conditions, is that the most suitable repository level is deemed to be at a depth of 450 m. The investigated rock volume on Äspö is, however, not sufficient to accommodate the entire deep repository; it would only provide room for the central area and Deposition Area I, i.e. about 10% of the number of canisters. The predominant constraints are steep discontinuities running in a northeasterly direction which divide the rock mass into rock blocks of insufficient size. These steep zones would make it difficult to position the deep repository on this site. Figure 7.1-1 shows the rock blocks that are bounded by discontinuities that may not occur within deposition blocks. Figure 7.1-2 shows how these blocks can be utilized for a smaller repository, taking into account the properties of the rock in the blocks.

7.2 SITE-SPECIFIC THERMAL PROPERTIES

7.2.1 General

The bentonite barrier must not be exposed to a higher temperature than about 130°C. Otherwise serious dissolution and conversion of bentonite to non-swelling clay can occur. In practice, a safety margin needs to be provided to this temperature, and the work thus far has been aimed at keeping the temperature below 100°C at atmospheric pressure. With regard to the margin for uncertainties in data and calculations, a maximum temperature of 80°C has been assumed in the work with the layout.

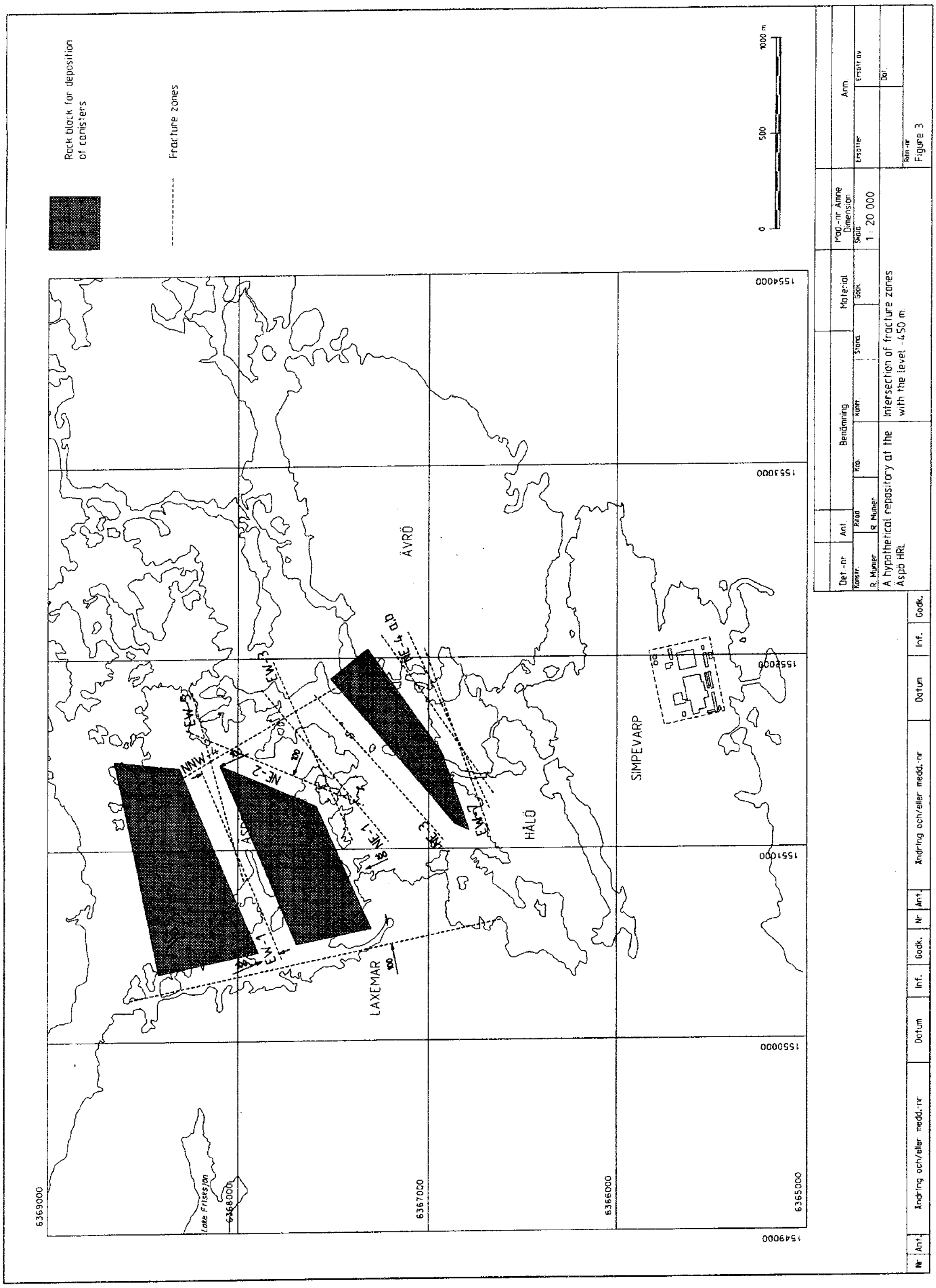


Figure 7.1-1. Block-bounding discontinuities at the 450 m level.

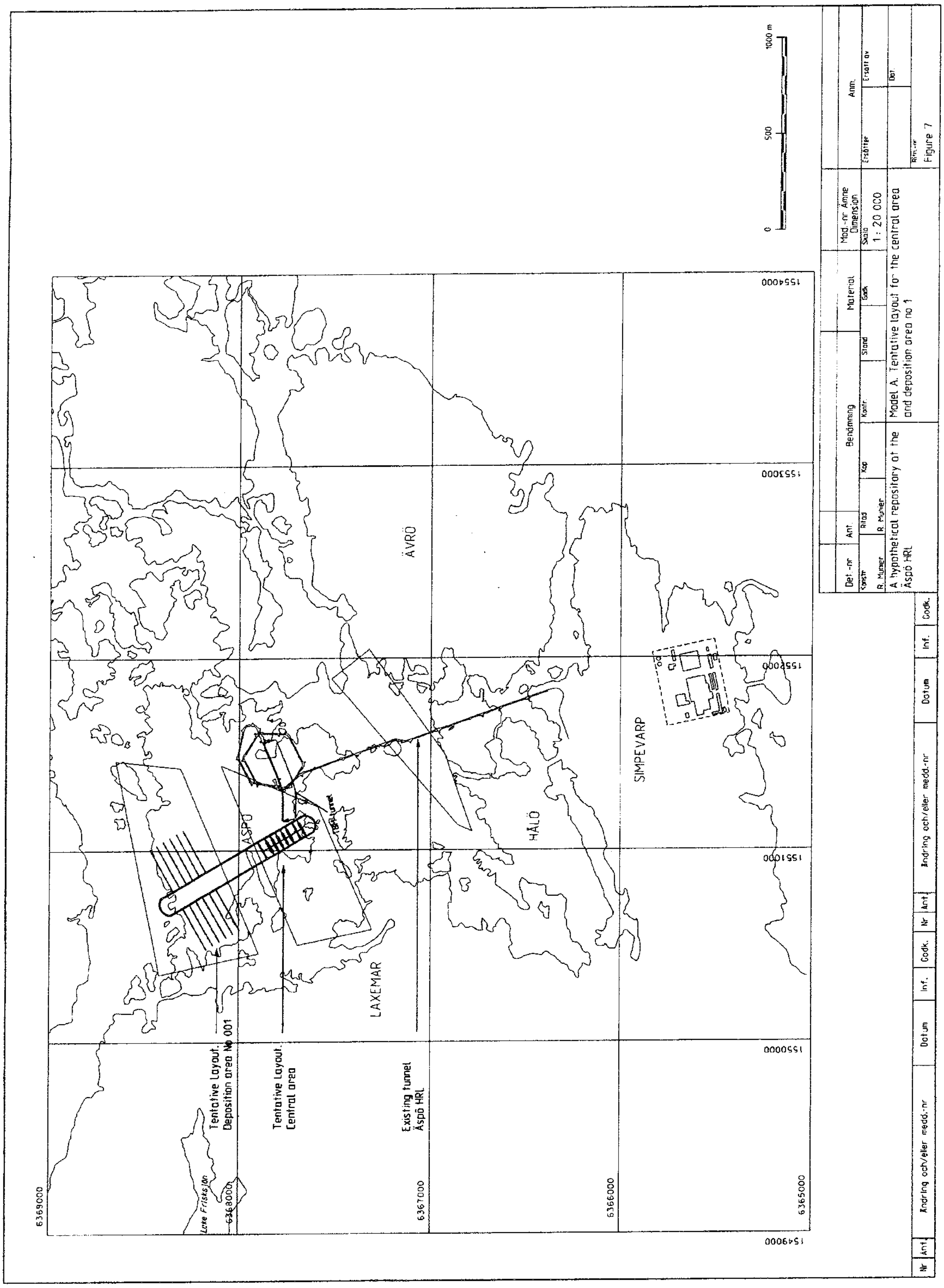


Figure 7.1-2. *A hypothetical repository at a depth of 450 m. Deposition tunnels have been oriented with regard to the direction of maximum principal stress, water-conducting zones and available space /7.1-1/.*

7.2.2 Thermal properties of the rock

The parameters that describe the thermal properties of the rock are its original temperature at repository level and its thermal diffusivity. Diffusivity is in turn determined by the density, thermal conductivity and specific heat of the rock.

All of these parameters are normally simple to measure on rock samples or in boreholes. It must also be determined how representative the samples are of the entire rock mass, and how big an influence different inhomogeneities in the rock have on the global properties.

7.2.3 Thermal properties of the bentonite buffer

The thermal properties of the buffer are determined by its thermal diffusivity, which is in turn determined by its density, thermal conductivity and specific heat. All of these quantities are dependent on the water saturation of the bentonite. A lower water saturation means lower thermal conductivity.

The thermal properties of the bentonite buffer are similarly easy to measure, and laboratory determinations have been done that show with great accuracy parameter values at different water saturations. The difficulty here is to determine the water saturation cycle undergone by the buffer around a canister. This cycle is dependent above all on the hydraulic conditions that prevail in the rock around the deposition hole.

If water is in ample supply, the process of water saturation should only take ten years at the most. Calculations of the thermal development of the repository can then be performed for water-saturated bentonite. If the supply is moderate but still sufficient to provide a flow in towards the deposition hole, the water content will not fall below the original value. The calculations can then be based on the values for the water saturation of the bentonite at the time of deposition. If, on the other hand, the water supply is poor, there is a risk that the bentonite will dry out, resulting in very poor thermal conductivity.

The lower the coefficient of thermal conductivity of the bentonite is, the less heat output can be allowed in each canister at the time of deposition so that a given temperature in the bentonite will not be exceeded. A lower thermal conductivity therefore leads to a greater number of canisters, which means a larger horizontal extent of the repository.

7.2.4 Experience from the work in the Äspö HRL

Given a certain repository layout, the thermal conductivity and temperature of the rock determine what heat generation can be allowed in the disposal canisters.

The different rock types at Äspö have average thermal conductivities ranging from just below 3.0 W/mK up to about 3.7 W/mK. The original temperature of the rock is about 17°C.

The generic layout described in section 5.2 is based on a thermal conductivity of 3.0 W/mK and a temperature of 18°C. In other words, there is room for a slightly higher heat generation per canister at Äspö, although this has not been utilized in the repository design presented in /7.1-1/.

The importance of the thermal conductivity of the rock is explored in /7.2-1/. For the same canister, an increase from 3.0 to 3.7 W/mK means that heat generation can increase by about 10% or that the temperature rise on the surface of the canister will be about 10% lower.

A deviation in rock temperature of 1°C from the value in the generic layout means that the temperature on the canister surface can be allowed to increase another 1°C. This corresponds to about a 1.5% greater heat generation in the canisters.

The structure of the rock in terms of the fracture patterns that are of importance for the conduction of water into the deposition holes has not been studied in detail on Äspö. The judgement that can be made today is that the quantity of water that is present from the start in the buffer and in the gaps in the deposition hole will remain in the deposition hole instead of being driven out into the near-field rock /7.2-2/. In other words, the buffer will not dry out, and in fact its water ratio will increase until full water saturation is reached. This means that the thermal conductivity of the buffer can be assumed to be 1.0 W/mK from the beginning /7.2-3/ and then to increase. At the same time, good thermal contact can be assumed to exist between canister and buffer and between buffer and rock due to the ever-present water contact.

7.3 EFFECTS ON PROPERTIES OF NEAR-FIELD ROCK

7.3.1 General

The rock in walls, roofs and floors around shafts, tunnels and rock caverns will be affected by the actual blasting procedure and the boring of deposition holes. In the vicinity of canisters with heat-generating spent fuel, such effects will also occur as a consequence of the heating and cooling. The effects are dependent on the original geological conditions, the excavation method used and the thermal load.

The properties that are expected to be obtained in the near-field rock, assuming shaft boring of the deposition holes and conventional drilling and blasting (careful blasting) of the deposition tunnels, are described below. The effects of heat generation in deposited canisters are also described briefly.

7.3.2 Influence of shaft boring of deposition holes

Shaft boring of the 1.75 m diameter deposition holes has a very limited influence on the surrounding rock. The new fractures that form will probably not extend more than 100 mm into the wall rock /7.3-1/. To this must be added the elasto-plastic change undergone by the near-field rock as a consequence of the stress redistribution. Experimental boring of 1.52 m diameter holes at Olkiluoto /7.3-2/ shows that a change in the form of increased porosity in the rock can be discerned in to a depth of about 20 mm from the hole wall. In Grimsel, Switzerland, cores were taken in the wall of a TBM tunnel at a depth of 1,000 m. Changes were noted in these cores in to a depth of 10–30 mm from the tunnel wall /7.3-3/.

The conclusions that have been drawn thus far are thus that a small zone is formed nearest the rock wall that can have an elevated water conductivity in the height direction of the hole. This is good for the water saturation of the buffer, since an even distribution of water along the entire height of the deposition hole could be assumed.

7.3.3 Influence of blasting of tunnels

Blasting of the deposition tunnels, as well as other tunnels, causes much greater damages in the rock than mechanized excavation methods such as full-face boring. The blasting damage experiment in the Äspö ramp /7.3-4/, as well as in the ZEDEX tunnel /7.3-5/, verifies the preliminary estimates made in the Stripa Project /7.3-6/ and the damages determined on theoretical grounds /7.3-7/. The conclusions from the experiment on the Äspö ramp indicate that it is necessary to allow for a heavily affected zone of 0.3 m in the walls and roof and 1.0–1.5 m in the floor. The stress redistribution leads to changes, albeit of a less serious nature further out from the tunnel. The judgement is made in /7.3-8/ that hydraulic conductivity increases 100–1000 times in the blast-disturbed zone (on average 100 times in a 1 m thick zone) and about 10 times in the stress-redistributed zone outside (one tunnel diameter out). However, few measurements have been made to verify this assessment. Some results instead suggest that hydraulically interconnected structures extend only a distance of a metre or so /7.3-9/.

7.3.4 Influence of heating

Heating affects the rock by increasing the stresses. However, the increase is not so great that the strength of the homogeneous rock is exceeded at a depth of 400–700 m. Since the bentonite in the deposition hole exerts a pressure on the rock wall, there is little possibility of movements along the fracture plane. After cooling, the thermally induced stresses will have dispersed, and it is assumed that the deposition holes will have resumed the original properties of the near-field rock.

7.4 BUILDING MATERIALS

7.4.1 Consumption of building materials

Underground activities will result in various materials being left after plugging and sealing. These include both deliberately emplaced materials such as canisters and bentonite, cement and iron for grouting, shotcrete, rock bolting and other reinforcement, and residues of materials used to carry out the work, propel machines, etc. These residues can be left as a result of small, continuous releases or large, uncontrolled ones. An estimate of material quantities needed for grouting and rock reinforcement under Äspö conditions is shown in Table 7.4-1 /7.4-1/.

Cement and iron are used here as general categories. Their chemical composition may vary, to some extent in response to wishes to minimize certain substances.

7.4.2 Other materials that may be left in the deep repository

Work is under way to quantify how large quantities of other materials than cement and iron may be left, based on existing knowledge concerning what substances are used under ground and how they are handled. Estimates of the material balance between what is taken down under ground and what is taken up to the surface again are also weighed into the equation. Since no measurements are available from work sites, the data are uncertain. A preliminary estimate of different quantities is given in /7.4-2/.

Table 7.4-1. A rough estimate of material quantities for grouting and rock reinforcement in the deep repository under Äspö conditions. Only the cement and iron content of the building materials used is given. Reinforcement measures are designed for a longevity of 50 years.

Building area	Cement (tonnes)	Iron (tonnes)
Conventional tunnelling		
Access ramp	1,700	100
Central area	900	100
Area for other long-lived waste	1,000	50
Area for spent nuclear fuel – Stage 1		
– Driveway	1,600	70
– Deposition tunnels	200	20
Area for spent nuclear fuel – Stage 2		
– Driveways	3,300	150
– Deposition tunnels	2,000	200
TOTAL, ROUNDED OFF	11,000	700
Tunnelling with TBM		
Access ramp	800	50
Central area	900	80
Area for other long-lived waste	1,000	50
Area for spent nuclear fuel – Stage 1		
– Driveway	700	30
– Deposition tunnels	100	5
Area for spent nuclear fuel – Stage 2		
– Driveways	1,400	60
– Deposition tunnels	1,000	30
TOTAL, ROUNDED OFF	6,000	300

8 THE BIOSPHERE

The biosphere on the selected repository site is described in this chapter, with a focus on factors of importance for the release and migration of radionuclides.

Based on data from the geoscientific and hydrological characterization of the site, groundwater recipients and transport pathways in the vicinity of the repository are described. Similarly, the site- and region-specific exploitation of the natural environment is discussed, as well as transfer pathways to man. Site-specific transport models, if any, are described.

A specific characterization of the biosphere may have to be done as a basis for the analyses of possible radiological consequences for other organisms than man.

The uncertainties are discussed, mainly in view of the high potential of the ecosystems for change with time, and with reference to the site-specific factors than can limit this changeability.

***In this report**, Chapter 8 comprises a general description of the role of the biosphere in the transport of radionuclides from the bedrock to man's environment. Land uplift and other changes in the biosphere take place during relatively brief time spans, and today's site- specific conditions need not be representative of the biosphere in the event of a release near the ground surface. Instead, Chapter 8 comprises a review of all possible relevant recipient types.*

The initial recipient of the deep groundwater in the biosphere determines to a high degree what the transport pathways will be for any escaping radionuclides. Normally a well is regarded as the most unfavourable scenario, and for this reason it is given a more quantified description in the text. No corresponding analysis is performed of the importance of the recipients for other organisms than man.

8.1 INTRODUCTION

The purpose of the safety assessment can be said to be to show that the consequences for man and the environment of reasonable future evolutions of a deep repository lie within acceptable limits. To assess these consequences, it is necessary to describe the biosphere around the repository site in question. The description should contain information on the recipients for deep groundwater in the biosphere and on the local ecosystems. Man's exploitation of the natural environment should also be included in the description.

Since the biosphere is expected to undergo changes during the time dealt with by the assessment, it is necessary to analyze not only the present-day biosphere on the repository site but also possible future variations. For this reason, this chapter will be more general in its nature than the geological description of the repository site in Chapter 6.

Different biosphere recipients for deep groundwater and man's exploitation of nature are described in general terms in the following. Brief discussions of species and habitats worthy of protection and of time-dependent changes of the biosphere conclude the chapter.

8.2 RECIPIENTS IN THE BIOSPHERE

The radionuclides that may be released from a deep repository will be transported by the groundwater and may eventually come into contact with the biosphere via watercourses, lakes, seas and wells. The groundwater can also push up through sediments, peat bogs or soils if these are discharge areas. In rare cases, radionuclides can be transported through the rock via gas.

Any release of radionuclides from a repository will probably not start until many years after closure. Transport from the repository to the biosphere will then also take a long time. This means that any releases in the biosphere will take place in ecosystems thousands of years into the future. The release of radionuclides can be assumed to proceed over a very long period of time, during which that the landscape will change further. Lake sediments, for example, may come into use as agricultural land. These changes may be due to climate change (change of sea level, eutrophication etc.) or to human impact such as drainage or dredging of lakes and watercourses.

The transport at the transition between geosphere and biosphere is often difficult to model, since chemical and other conditions in the water and surrounding media may change dramatically within a small transition area. A low-level release of radionuclides may pass a zone where the nuclides precipitate and are accumulated during a long period. If the properties of this zone change, the radionuclides may be released during a shorter period. An example is the enrichment of uranium at the bottom of peat bogs, which can theoretically lead to a large release during a short period if the peat is burned, tilled or eroded.

The consequences of a release are highly dependent on whether and how the radionuclides reach man. This can take place via a drinking water well, a watercourse, a lake, sea water or agricultural products and other foods.

Watercourses

Watercourses such as ditches, creeks, brooks, rivers and streams are examples of recipients where the radionuclides are diluted relatively quickly before they reach man. The processes that are important are largely the same as for a well, see below, with the difference that irrigation is of relatively greater importance for the watercourses. Dredging of watercourses comprises an important link between groundwater and agricultural land. Consumption of fish and shellfish as well as bathing are also possible exposure pathways.

Lakes

In lakes the radionuclides will be further diluted before they reach man. Fish and shellfish, other types of aquaculture and bathing are of greater importance in lakes than in watercourses. External exposure on beaches is another exposure pathway.

Cultivation of the bottom sediments in lakes after land uplift, eutrophication or drying-out has been investigated /8.2-1/ with the result that this evolution could give up to 10 times greater dose compared with if the lake were not to change.

Seas

If the radionuclides reach the biosphere via the world oceans or the Baltic Sea, the doses to the critical group will only be a few tenths of a percent of those obtained from equivalent inland releases. The dose will in these cases mainly be obtained from fish. The main reasons for the big difference in dose are the dissimilarities in exposure pathways and the very long residence times in sediments, which allow the nuclides either to be removed from the systems or to decay. More rapid dilution in the seas also contributes to the difference.

Wells

A drinking water well acts as a short circuit between the geosphere and the biosphere. The travel times are shortened in the rest of the geosphere, and dilution can be considerably less than in large water volumes such as lakes, watercourses or near-surface groundwater reservoirs. A well is a recipient whose utilization can result in comparatively large dose conversion factors. For this reason, different situations around wells are discussed in somewhat greater detail below. Wells are usually treated as special scenarios in safety assessments.

The dose to man from radionuclides in a well is affected more by how the well's water is used than by how much water is pumped up. A large well can be expected to draw a slightly larger portion of a possible release from a repository, but this is normally compensated for by the fact that only a small portion of the water is then used for drinking water. The water withdrawal rate can vary from a m^3 or so per year to several thousand m^3 per year.

Groundwater flows at a depth of about 500 m in the rock are on the order of 3 litres/($m^2 \cdot y$). The surface area of a disposal canister is about 9 m^2, which means that the smallest volume that could reasonably contain the entire quantity of leaking radionuclides from **one** canister is a few tens of litres/year. The equivalent volume from the entire repository with 5,300 canisters on an area of 1 km^2 is about 3,000 m^3/y.

An extremely unfavourable case would be a small well (a few m^3/y) that is used solely for drinking water and that receives all nuclides from a canister that happens to be defective and furthermore leaks during the very years the well exists. This is the only case that could be foreseen where all the nuclides released from a canister would be consumed in drinking water.

In order for the nuclides released from several canisters to get into the water from the same well, so much water would have to be pumped up that it would most likely be used for other purposes than drinking water.

It is evident from the above that it is probable that all groundwater from the repository area around **a single canister** could reach one and the same well, even if this well is relatively small. However, it is not realistic that the entire quantity of radionuclides from a release from a number of canisters spread over the entire repository will get into one and the same well. A well that catches groundwater from the entire repository would have to have a withdrawal rate of several thousand m^3/y of only the water that runs through the repository area, which is highly unrealistic. A specific scenario, "stylized well", is described in /8.2-2/.

Atmosphere

Gas releases from the repository will, when they reach the biosphere, be distributed relatively rapidly around the northern hemisphere. The travel times from the repository to the atmosphere can be assumed to be greater than the time for further distribution around the hemisphere.

8.3 MAN'S EXPLOITATION OF THE NATURAL ENVIRONMENT

Naturally, a number of scenarios can be imagined for man's future exploitation of the natural environment on a given site. A general scenario that gives relatively large dose factors consists of a small farm with a high percentage of local production. In this scenario, man has a large impact on the natural environment through agriculture and forestry. The choice of parameters which in the model describe consumption, irrigation, livestock farming etc. are normally based on the **present-day biosphere**. Even though there are likely to be considerable changes from today's situation by the time the radionuclides reach the biosphere, this is the only system that can be described with reasonable precision.

As far as other scenarios are concerned, it is ultimately only our imagination that sets the limit for how we conceive that man might exploit and impact the natural environment. The borderline between normal activities and special scenarios may in some cases be difficult to draw. Examples of such special cases are city building, development of agriculture, forestry and aquaculture.

8.4 SPECIES AND HABITATS WORTH PROTECTING

The measures that are adopted to protect human beings far into the future should be in balance with what is done to protect other forms of life in our world. What is important is not to protect individuals, but rather populations and species, and above all the function of the ecosystems. Ecosystems are the habitats for man and other species, which are necessary for their survival in the future.

8.5 TIME-DEPENDENT CHANGES

In connection with safety assessments, the time horizon for attempts to predict changes in the biosphere is usually set at the start of the next ice age, i.e. between 5,000 and 10,000 years from now.

Before the next ice age we can expect such events to occur as land uplift, changes in sea level and climate, various anthropogenic environmental effects and use of new technology. It can often be difficult to predict in general terms what effects these events may have in the biosphere of relevance to the transport and intake of nuclides, and above all the risks that may be associated with this. With good knowledge of the conditions on a specific site, on the other hand, it is often possible to make relatively far-reaching predictions regarding time-dependent changes in the biosphere and to analyze their consequences.

9 ANALYSIS – REPOSITORY SYSTEM, SCENARIOS

The chapter begins with an overview of the repository and its performance. This is followed by a review of the repository's barriers, the time-dependent processes that may occur in the repository or between the repository and its surroundings, and the resultant evolution of the repository over time.

The scenario methodology applied in the safety assessment is presented. The resultant interaction matrices and a documentation outline are presented and discussed.

The choice of scenarios and how the scenarios are dealt with in the safety assessment are gone through. Finally, uncertainties and completeness in the scenario work are discussed.

***In this report**, Chapter 9 is a status report on the scenario work that is being done for SR-I. The review of the different steps in the work is relatively detailed to show what kinds of questions crop up in the practical work. The RES methodology which is presented in section 3 has been utilized to create interaction matrices for the near field, the far field and the biosphere. The near-field matrix has been detailed in a fuel matrix and a buffer matrix. These matrices will be revised as the premises are adjusted. The biosphere matrix is the result of the international cooperation within BIOMOVS2. In section 9.4, a number of scenarios have been grouped in tables according to the character of the initiating event. These tables may also be revised during the course of the SR-I analyses.*

9.1 INTENDED FUNCTION OF THE REPOSITORY

The repository is supposed to isolate the radioactive waste during the time its radioactivity is elevated above that of its surroundings. To achieve a high level of safety, the system is composed of a number of independent barriers, the "multibarrier" principle. According to the multibarrier principle, the safety of the entire system shall not be dependent on the satisfactory performance of **one** single barrier. The barriers possess such properties that they prevent in various ways the radionuclides from reaching the biosphere and man. The barriers prevent dissolution, contain the radionuclides, and buffer, filter and retard the transport of radionuclides. The barriers can be divided into engineered, i.e. man-made, and natural. Together, the barriers comprise a redundant system for isolating the waste during the time its radioactivity is elevated.

In safety assessments, the system being analyzed is usually divided into near field, far field and biosphere. The near field includes the engineered barriers. They are the fuel, the canister with steel insert and copper shell, and the bentonite buffer. Tunnel fills and the part of the rock that has been affected by the construction of the repository also belong to the near field. The far field

THE REPOSITORY SYSTEM

A. THE BIOSPHERE

- Radionuclides are transmitted to humans via recipients for deep groundwater and local ecosystems.
- Dilution conditions, capacity of recipient to buffer, store or accumulate radionuclides, and land and water use influence the radiation dose to man and the environment.
- The radiation dose can be limited by selecting a site with favourable conditions.

B. THE ROCK

- The rock gives the engineered barriers a stable environment both chemically and mechanically.
- If the engineered barriers have been damaged, the rock:
  - retards transport of radionuclides via slow water flow and thereby long transit times,
  - retains radionuclides by acting as a filter and buffer.

C. THE BUFFER

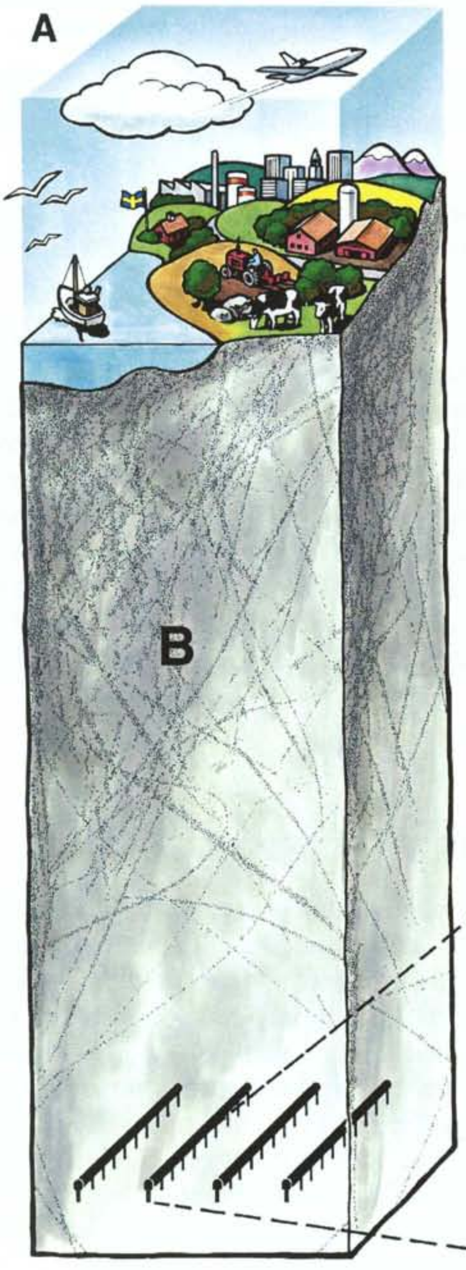
- The buffer of bentonite clay acts as a mechanical and chemical buffer, a sealing layer and a filter.
- Due to its rheological properties, the bentonite acts as a buffer against mechanical stresses.
- The chemical buffering properties of the bentonite make conditions around the canister less corrosive.
- Low hydraulic conductivity retards water transport so that corrodants are hindered from reaching the canister and radionuclides from leaving it.
- Due to the material properties of the bentonite, particles and solutes are captured via filtration and sorption.

D. THE CANISTER

- The steel insert lends mechanical strength to the canister.
- Loads in the form of stresses and shear forces are taken up by the steel insert so that the fuel assemblies and copper shell remain intact.
- The copper shell keeps radionuclides in the canister and water out.

E. THE FUEL

- The fuel hinders the escape of radionuclides thanks to its low solubility in water and low corrosion rate.
- The radionuclides are tightly bound in the fuel structure and therefore difficult to dissolve.



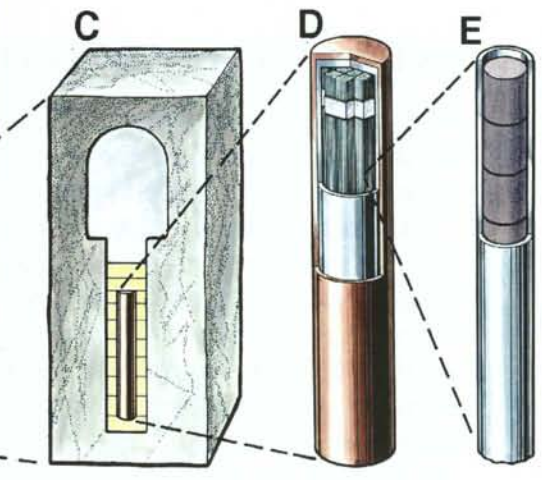


Figure 9.1-1. *The different parts of the analyzed system. The function of the different barriers and the influence of the biosphere on the transmission of radionuclides to human beings.*

includes the rest of the rock. The biosphere is the living environment on the surface in the repository's surroundings. The different parts, their barrier functions, their evolution with time, and the influence of the biosphere on the transmission of radionuclides to human beings are described below. A summary description is also given in Figure 9.1-1.

9.1.1 Near-field barriers, their function and evolution

The fuel

The fuel's barrier function is to bind the radionuclides in itself. The fuel acts as a barrier thanks to its low solubility in water and its low corrosion rate. The radionuclides are tightly bound in the fuel structure and therefore difficult to dissolve. The dissolution rate in groundwater of the composition normally present at a depth of several hundred metres in the rock is very low. Only a portion of the readily soluble substances that are present on the surface of the fuel and in the fuel-clad gap will be dissolved rapidly in the event of exposure to groundwater. On the whole, the fuel comprises a very effective barrier to the dissolution of radioactive substances.

At the time of deposition, the fuel has very high radioactivity. Its radioactivity and thereby its "toxicity" and decay heat decline relatively rapidly to begin with, but more and more slowly with time. Section 4.2 describes how the deposited fuel changes with time.

Helium is formed by alpha disintegration in the fuel rods. During the post-closure period, helium pressure will build up at a very slow rate. Calculations show that the pressure will not be able to damage the canister during the time the fuel should be kept isolated.

If the fuel should come into contact with groundwater, the dissolution rate is very slow. Fuel dissolution is affected by many factors such as redox potential, groundwater composition, radiation, temperature and interaction with other engineered barriers. The dissolution processes are described in section 10.6.

The steel insert

The function of the steel insert is to lend mechanical strength to the canister. The loads to which the canister might be exposed in the rock in the form of stresses and shear forces are taken up by the steel insert so that the fuel assemblies and the surrounding copper shell remain intact. The factors that have been taken into consideration in designing the steel canister are discussed in section 5.3.

Since the steel insert is not in contact with groundwater, its degradation is negligible and its function remains intact. In order for the groundwater to reach the steel insert, it must pass through the copper shell. If groundwater should come into contact with the steel insert, the insert will be subjected to various types of corrosive attack. Corrosion of the steel insert under different conditions is described in section 10.4.4.

The copper shell

The function of the copper shell is to keep the radionuclides in and the groundwater out. As long as the copper shell remains intact, no transport of radionuclides, in any form, can take place. If water should enter the canister, it will create conditions leading to corrosion of the steel insert, dissolution of the fuel and transport of radionuclides.

In the chemical environment in which the canister is emplaced, copper possesses very good corrosion properties. The copper shell corrodes so slowly that the canister is expected to remain intact throughout the long period of time during which the waste will have elevated radioactivity in relation to its surroundings. Furthermore, the ductility of the copper material permits strain in conjunction with creep or temperature movements without the copper shell cracking. See section 5.3 for the design requirements on the copper shell.

The corrosion rate of the copper shell is dependent on the quantity of corrosive substances dissolved in the water around the canister. The corrosive substances in the water are primarily dissolved oxygen and, for the reducing conditions that prevail in the repository, dissolved sulphide. The groundwater chemistry in the vicinity of the canister is described in section 10.5.4. The different corrosive attacks to which the copper shell could conceivably be exposed in the repository are dealt with in section 10.4.3. Besides groundwater chemistry, the time to corrosion penetration is dependent on the purity and structure of the copper, possible fabrication defects and prevailing temperature and pressure conditions.

The buffer – bentonite

The buffer has several functions: It acts as a mechanical and chemical buffer, a sealing layer and a filter. Due to its rheological properties, the bentonite acts as a buffer against mechanical stresses. The chemical buffering properties of the bentonite make conditions around the canister less corrosive. Due to its low hydraulic conductivity, the bentonite prevents water transport. Corrodants are prevented from reaching the canister, and radionuclides are prevented from leaving it. Thanks to the large surface area and many narrow channels the bentonite has by virtue of its porosity, particles and dissolved substances are captured via filtration and sorption.

After the bentonite has been emplaced in the deposition holes, it is rewetted and the material swells. Since the space is limited, a swelling pressure arises. The swelling pressure leads to self-healing and homogenization and prevents the creation of water-conducting passages. This is described in sections 10.5.2 and 10.5.3.

To keep the canister in place, the type of bentonite to be used in the final repository has good bearing capacity, see also section 10.5.3.

Bentonite is the geological designation of naturally occurring clays that have been formed from volcanic ash. Bentonites are rich in swelling clay minerals called smectites, see also section 5.4. Due to its swelling capacity, buffering properties and rheological properties, the smectite accounts for much of the function of the buffer. In the presence of potassium, especially at high temperatures, bentonite can undergo a structural alteration called illitization, whereby the clay mineral smectite is converted to illite. Illite has little swelling capacity and thereby poorer properties from the viewpoint of nuclear waste disposal. The conversion of smectite to illite is described in section 10.5.3.

Another temperature-dependent process that can degrade the performance of the buffer is cementation. What is meant by cementation and how it affects the properties of the buffer are described in section 10.5.3.

Thus, high temperature can, via different processes, contribute to deteriorated buffer performance. The temperature of the buffer can be reduced by reducing the quantity of fuel in the canister. Section 5.4 explains what allowances have been made for temperature in designing the buffer. The thermal development of the repository is dealt with in section 10.2.

Backfilled tunnels and the disturbed zone

When the repository has been built and the waste emplaced, the tunnels are backfilled with a mixture of quartz sand and/or crushed rock and bentonite. This is described in section 5.4. The disturbed zone, or excavation-disturbed zone (EDZ), is the part of the rock that has been affected by the construction of the repository. The tunnel backfill and disturbed zone have barrier functions similar to those of the rock. They can be regarded as a geometrically restricted portion of the rock with properties similar to those of the rock, but which differ from the rock in terms of the characteristic parameters that describe these properties (e.g. hydraulic conductivity).

When the repository has been closed, the groundwater seeps back into the backfilled tunnels. The bentonite in the bentonite-sand mixture swells, a swelling pressure arises and the original stress conditions in the rock are largely restored. The presence of foreign materials in the repository can influence the groundwater chemistry. This is described in section 10.3.3.

9.1.2 The far field, its function and evolution

The primary function of the rock is to give the engineered barriers a stable environment both chemically and mechanically. Another important safety-related function of the rock is to retain or retard the transport of radionuclides if the engineered barriers have been damaged. The rock acts as a filter and a buffer. The rock surrounds the engineered barriers. When the repository has been closed and sealed, it can only be affected via the rock or the water moving in the rock.

The rock contains fracture systems in different scales from macroscopic (metre-wide, many kilometres long) down to microscopic (intercrystalline, tenths of a micron). Large fracture zones surround rock blocks with small fractures. The large fracture zones arose when the rock was formed and in response to stresses to which they have been subjected over billions of years. The waste is emplaced in rock blocks surrounded by major fracture systems. The fracture zones comprise zones of weakness where external loads and movements are taken up. They thereby contribute to a mechanically stable environment for the engineered barriers.

Water moves in the fracture systems in the rock. In the small fractures and at the depths at which the waste is emplaced, the water flow is very slow. By virtue of chemical processes during its passage, the rock acts as a buffer.

Particles and dissolved substances are retained by the rock via filtration and sorption. All in all, the rock contributes to a chemically stable environment for the engineered barriers at the same time as it prevents dangerous radionuclides from reaching the biosphere via filtration and sorption.

The composition and flow of the groundwater are affected by human activities (agriculture, pollution, large construction projects, well-drilling, tunnelling etc.) as well as climate change on the surface. The properties of the rock that prevent radionuclides from reaching the surface also prevent pollution on the surface from getting down into the repository. The disposal depth and the usually occurring rock type contribute towards reducing the risk of disturbance by drilling or construction activities.

Stress conditions and fracture systems in the rock are affected locally when the repository is built. After backfilling of the repository and when the groundwater flows back, the original conditions are largely restored. Those parts of the rock where the waste is deposited are expected to remain very stable over time.

The performance and evolution of the rock are described in greater detail in section 10.3.

9.1.3 The biosphere and its evolution

The transmission of radionuclides from the repository (via near field and far field) to human beings, animals and plants finally takes place in the biosphere. The description of the biosphere includes information on e.g. recipients for deep groundwater and local ecosystems. The repository site, and thereby which recipient could receive radionuclides, is of importance for the radiation dose to which humans, animals and plants are exposed. Some factors that affect the dose are dilution conditions, the recipient's capacity to buffer, store or accumulate radionuclides, and land and water use. By choosing a site with favourable conditions, the radiation dose to man and the environment can be limited.

It will take a long time for any release of radionuclides to reach the biosphere. During this time the biosphere will change. It is often difficult to predict the exact changes and when they will occur. Based on a good knowledge of a specific site, however, relatively far-reaching predictions can be made concerning the evolution of the site up until the next ice age.

One way to handle uncertainty in the biosphere description is to make a number of stylized biospheres to illustrate how different conditions affect the expected radionuclide doses. As far as man's exploitation of the natural environment for his food and water supply is concerned, it is assumed that the same conditions prevail as today.

The biosphere and its evolution are described in Chapter 8. The importance of the biosphere within the repository system is dealt with in section 11.5.

9.2 SCENARIO METHODOLOGY

The scenario methodology and its purpose are described in section 3.4. In this chapter, the application of the methodology will be presented. Three central concepts in the scenario methodology are FEP, process system and of course scenario. Their meaning is repeated below.

FEP

The features, events and processes that influence repository performance and that might possibly occur now or in the future.

Process system, PS

The Process System is the organized assembly of all phenomena (FEPs) required for the description of barrier performance and radionuclide behaviour in a repository and its environment, and that can be predicted with at least some degree of determinism, given an assembly of external conditions.

A systematic description of a PS is given in the interaction matrices that are set up for the analyzed system.

Scenario

A scenario is defined by a set of external conditions which will influence processes in a PS. The external conditions determine how the processes in the PS are to be combined and modelled in describing the evolution of the scenario and evaluating its consequences.

A scenario is a description of a hypothetical future sequence of events. The term "scenario" includes both an evolution starting from a set of defined premises and the future situation to which the evolution leads. Within the scenario methodology employed, the central feature of a given scenario is the set of premises for the sequence of events. When speaking of selecting scenarios for a safety assessment, what is often mean is the selection of premises for scenarios. In the following, the term "scenario selection" will often be used in the sense of "selection of premises for scenarios".

To be able to give as comprehensive a picture as possible of the repository and how it will react to perturbations, all features, events and processes that influence the repository in any way must be identified, described and placed in their context. This is done by constructing a number of interaction matrices (RES matrices) over the near field, the far field and the biosphere. Besides providing a systematic description of features, events and processes that can influence the performance of the repository and their coupling to each other, the interaction matrices comprise a visualization of how processes and features are related. This picture, or model, of the different parts of the analyzed system, and the documentation associated with it, provide a tool that can be used by experts and researchers in different scientific fields to describe different scenarios. The construction of the interaction matrices also contributes towards an increased understanding of the analyzed system.

Scenario selection, or the selection of premises for different scenarios, is done by experts. Scenario selection is done on the basis of the understanding that has been built up concerning the function of the system parts, the function of the whole system and the processes and features of the repository and their relationships with each other.

9.3 INTERACTION MATRICES

The first step in constructing an interaction matrix is to formulate the problem and the goal of the analysis. The next step is to discuss which concepts or physical variables are essential and should therefore be described. They are then placed as diagonal elements in an interaction matrix, see Figure 3.3-1. In SKB's application, a diagonal element can describe a system premise, a system part, characteristics of a system part, a process or a physical variable. The number of diagonal elements should not be too great in order that the interaction matrix should not become unwieldy.

If a system is complex from some viewpoint, a choice must be made between selecting diagonal elements of a general or a detailed nature. General diagonal elements are often of the type system premise or system part. Examples taken from the far-field matrix are: design/layout (system premise) and buffer/backfill/source, disturbed zone and biosphere (system parts). Examples of diagonal elements of a more detailed nature are: natural fracture system (characteristic of a system part), gas generation/gas transport (process) and temperature/heat (physical variable). These examples are also taken from the far-field matrix. Observe that the classification into different types of diagonal elements is more or less subjective. It has been done to illustrate what factors should be considered in selecting and defining diagonal elements.

If general diagonal elements of the type system parts or system premises are chosen, the whole system can be shown in a matrix. This provides a good system overview and overall system understanding. The disadvantage is that each box in the matrix is overloaded with information. There is a risk that essential details will not be given sufficient attention and that internal processes will be overlooked. The interactions between general diagonal elements are often not purely binary, but often consist of multiple interactions. The interaction boxes will in these cases contain descriptions of several interactions, processes or mechanisms.

If detailed diagonal elements are chosen, many sub-matrices are needed. The descriptions of the system's features and processes become detailed at the expense of a general system overview. Even when the diagonal elements are detailed, the interaction boxes can contain descriptions of several interactions, processes or mechanisms.

In the construction of interaction matrices for the final repository of radioactive waste, the choice of degree of detailing and a natural classification into engineered barriers, rock and biosphere has led to a subdivision of the system into near field, far field and biosphere. The near field is described by the three interaction matrices near field, fuel and buffer/backfill. The far field and the biosphere are described in their own matrices. The relationship between the different matrices is shown in Figure 9.3-1.

Before the different interaction matrices can be constructed, the premises or external conditions that apply to each matrix must be defined. The idea is that

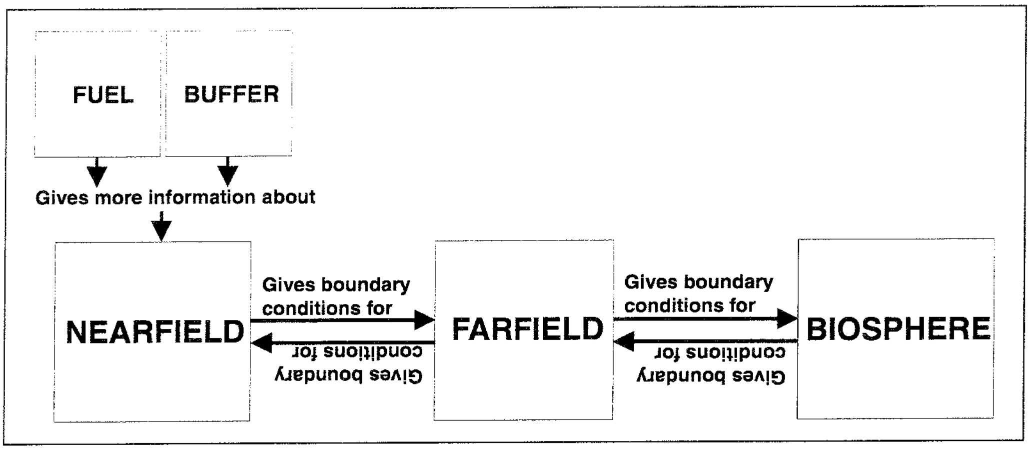


Figure 9.3-1. The different interaction matrices and their coupling to each other.

the matrices should be able to be used for analysis of all scenarios, so the premises for the matrices are chosen to be as broadly applicable as possible.

When problems, the goals of the analysis and premises have been established, the work of constructing the interaction matrices can begin. Important system premises, system parts, characteristics of system parts, processes or physical variables are placed as diagonal elements. The other matrix elements describe the interactions between the diagonal elements, see Figure 3.3-1. An objective is that the definitions of the diagonal elements should apply to all scenarios. Only the "value" of the diagonal elements, the interactions and their importance should have to be changed and re-evaluated when analyzing different scenarios.

It is important to think through the definitions of the diagonal elements carefully. What is it that is being described, and in what areas is the diagonal element valid? An example of this is groundwater movement in the far-field matrix. What is being described is the magnitude, direction and distribution of the groundwater movement. Groundwater movements in tunnels, deposition holes, the disturbed zone and the rest of the rock are dealt with. Correct, carefully thought-out definitions of the diagonal elements are a prerequisite in order for the definitions of the interactions to be consistent.

When the diagonal elements have been chosen and defined, all binary interactions are analyzed. In this analysis, the definitions of the diagonal elements may need to be revised. A group of experts and/or specialists assigns each binary interaction a significance. Clear definitions have to be given of what is meant by the different significance levels. A four-point scale has been defined for the near-field and far-field matrices. For the near-field and biosphere matrices a five-point scale has been used. The different significance levels and their definitions are presented in connection with the different interaction matrices in the fold-out appendices 1–5.

At present a thorough review and documentation of the matrices is under way. In connection with this, the near-field and biosphere matrices will probably be assigned the same significance scale as the other matrices.

Each interaction element in the matrix can represent several interactions, processes or mechanisms. This is often true if the diagonal element is of a general nature, but can also happen with more detailed diagonal elements.

For the diagonal element causing the interaction, the exact driving force behind the interaction, the process or the mechanism is defined. For the affected diagonal element, what is affected is defined in a similar manner. If the significance of the interactions varies, the highest occurring code is shown in the figure above the interaction matrix. In the text within the interaction box, each treated interaction is given a name or a very short description.

As the matrices are constructed they are also documented. Routines and methods for how this work should be done have been worked out /9.3-1/. The work has been done with the far-field matrix as an example. Documentation has therefore been finished for the far field /9.3-1/. The work of documentation remains to be done for the other matrices.

For each box in the matrix, the documentation should include the following information:

- position in the matrix
- which matrix, name and version, the element is included in
- name or title of the element
- type of element, i.e. diagonal element or interaction
- a brief written description of the element
- references to literature where the element is described
- reference to different FEP databases
- description of how the element is modelled in the safety assessment
- if the element is an interaction:
  - significance
  - explanation (motivation) for significance
- who has made the description and their qualifications

Examples of documentation sheets are shown in Figure 9.3-2.

Element number: 06.08 **Revision date:** 95-11-30
Interaction matrix: FAR-FIELD1 **Version:** A
Element name:
6.8 Density affects groundwater head

Element type: Interaction **Number of interactions:** 1

Recordnumber: 107 **Total number of records:** 219

Description:
The density affects groundwater head (gradient). It therefore affects the modelling of the groundwater flow.

Priority: **Priority date:**
○ 0=White ○ 1=Green ○ 2=Yellow ● 3=Red 1995-06-12

Motivation:

Density is a parameter in the basic equation, (Hydraulic head, $\phi = p_{tot}/\rho g + z$). Density differences = driving forces.

Group identification:
SKB: T Eng, LO Ericsson,
L Morén, O Olsson, A Ström,
P Wikberg.
Kemakta: K Skagius & M Wiborgh.

Expertise:
● Experts
○ General Know how
○ Limited

SKB FEP reference:
Groundwater chemistry, far-field
Groundwater flow

Treatment of interaction in Performance Assessment

Interaction: **6.8 Density affects groundwater head**
Treatment:
☐ PA prerequisities **Date:** 95-08-15
☐ Assumptions
☒ Modelling **By:** A Ström (SKB)

PA prerequisities:

Assumptions:

Modelling application:
Density affects the groundwater head and is included in the description of groundwater flow. Part of the SKB PA model chain for radionuclide migration.

Model A name: HYDRASTAR 1.4
Model A reference: User's Guide, SKB AR 94-14

Model B name: NAMMU 6.2
Model B reference: Validity Document, SKB AR 95-11

Spec modelling assumptions:

Figure 9.3-2. *Documentation of an element in the far-field matrix.*

9.3.1 The near-field matrices

Three matrices have been constructed for the near field: one for the entire near field and two more detailed ones that describe the fuel and the buffer/backfill.

The near-field matrix

The matrix the describes the entire near field is called "Near Field 1" and is shown in Appendix 1 (fold-out). The matrix is the result of a first attempt to construct an interaction matrix. It has not been documented according to established routines. Experience from the work with the far-field matrix will be drawn upon when it is documented. This means that the matrix will probably be revised.

The premises for the matrix are:

- It describes a PS.
- The matrix is based on the previous copper/steel canister /9.3-2/ and not the canister described in Chapter 5. The only difference is that the new canister lacks filler material.
- The rock in itself is not counted as a part of the near field, but the fracturing of the rock is taken into account.
- In the original matrix the canister is intact, which means that no interactions with radionuclide transport are described. Canister damage is included as a change of state in the diagonal element "Cu canister". The case of canister damage must be dealt with as a scenario where interactions and chains of interactions are considered and documented. Since there are several conceivable scenarios leading to canister damage, and since processes that have to do with radionuclide transport are very important for the safety assessment, it would be better if the matrix assumed that there are radionuclides accessible for transport. Then interactions that describe nuclide transport could be described within the matrix and documented in accordance with established routines.
- The diagonal elements "Water flow/chemistry" and "Pressure" include the ice-age scenario.
- The repository has been closed and saturated with water.
- The water saturation process is not taken into account.

The boundary conditions for the matrix are given by the diagonal elements Fuel Rod Complex (element 1.1), Water Movement and Chemistry (element 8.8), Fracturing in Rock (element 9.9), Pressure (element 10.10) and Construction of Repository and Emplacement of Canisters (element 11.11). The factors that have been deemed important when describing processes in the near field have been selected as diagonal elements. The selected diagonal elements, their positions in the matrix and definitions are given in Table 9.3-1. Some documentation of the matrix is also provided in /9.3-3/.

Table 9.3-1. The diagonal elements in the interaction matrix "Near Field 1", their indexes and definitions.

Diagonal element	Position	Definition
Fuel rod complex	1.1	All radionuclides, the fuel itself, the Zircaloy and the metal parts of the fuel assemblies.
Filler and void	2.2	No filler material is being considered today. This element represents the void in the canister.
Steel inner canister (insert)	3.3	
Steel-copper gap	4.4	For fabrication reasons there will be a space between the steel insert and the copper shell. This gap will be about one millimetre at deposition. However, the copper shell is expected to creep on the steel, and the gap is expected to disappear after a few thousand years.
Copper outer canister (shell)	5.5	
Temperature	6.6	The whole near field will be subjected to elevated temperatures, so the definition is not fixed in space.
Buffer/backfill	7.7	The bentonite in the deposition holes with impurities and the backfill material in the tunnels.
Water movement and chemistry	8.8	The movements and composition of the water in rock and buffer
Fracturing in rock	9.9	The natural fractures in the rock and those formed during construction of the repository.
Pressure	10.10	The pressure in the system: the hydrostatic pressure, the lithostatic pressure and any pressure changes.
Construction and emplacement	11.11	Repository design, construction, emplacement (deposition) of canisters and closure. Reinforcements and forgotten materials.

The fuel

The fuel and the inner near field have been studied separately. The matrix is shown in Appendix 2 (fold-out). The processes illustrated in this matrix should also be found in the near-field matrix. However, the "Fuel 1" matrix has a higher degree of detailing and more comprehensive process descriptions.

The premises for the matrix are:

- It describes a PS.
- The matrix is based on the previous copper/steel canister /9.3-2/ and not the canister described in Chapter 5. The only difference is that the new canister lacks filler material.
- The copper shell has been damaged and is allowing water to enter.
- The repository has been closed and backfilled.
- Water-saturated conditions prevail.

The external boundary condition is given by the canister/buffer boundary. The factors that have been deemed important when describing fuel processes have been selected as diagonal elements. The selected diagonal elements, their positions in the matrix and definitions are given in Table 9.3-2. Documentation of this matrix is not yet available.

Table 9.3-2. The diagonal elements in the interaction matrix "Fuel 1", their indexes and definitions.

Diagonal element	Position	Definition
UO_2 matrix	1.1	The physical structure of the fuel. Uranium as a radionuclide is found in 2.2
Matrix-bound elements	2.2	The radionuclides that were **originally** in the fuel matrix. In this definition they are not fixed in space, but may be anywhere in the system.
"Segregated" elements	3.3	The nuclides that were **originally** on the surface and in the grain boundaries of the fuel, plus activation products in Zircaloy and the structural parts of the fuel assemblies. This diagonal element is a little unusual since it is partially fixed in space (the original position of the radionuclides is an FEP) and partially not fixed (like those in 2.2).
Radiation	4.4	
Temperature	5.5	The system is assumed to be stead-state. No transient thermal processes are taken into account.
Zircaloy and other metal parts	6.6	The radionuclide content is in 3.3.
Water	7.7	The water inside the canister.
Filler	8.8	Non-metallic original filler in the canister. Not included in present-day canister design.
Canister materials and their degradation products	9.9	Metallic canister materials and their degradation products, i.e. corrosion products and hydrogen gas.
Bentonite	10.10	The outer boundary in the system, defined as clay material and pore water. The pore water is separate from the water in 7.7.

The buffer matrix

The behaviour of the buffer has been studied in greater detail in a separate interaction matrix. The constructed matrix is named "Buffer 1" and is shown in Appendix 3 (fold-out). The premises for the matrix are:

- It describes a PS
- The purpose of this matrix differs from the others in the sense that it does not treat the processes that are included in the calculation chain, but merely describes the behaviour of the buffer material.

- The copper shell is intact. All radionuclide transport processes are treated in the near-field matrix. An exception is made for corrosion gases, however, which are treated in the matrix.
- The repository has been closed and backfilled.
- The restoration phase, i.e. the transient process when the repository returns to natural conditions, is taken into account.

The internal boundary condition for the matrix is given by the diagonal elements Fuel (element 1.1) and Canister (element 2.2). It is difficult to give a clear definition of the external boundary, but it can be said to lie in the elements Groundwater Hydrology (element 8.8), Groundwater Chemistry (element 9.9), Near Field Rock (element 10.10) and Site/Layout (element 13.13). The factors that have been deemed important when describing fuel processes have been selected as diagonal elements. The selected diagonal elements, their positions in the matrix and definitions are given in Table 9.3-3. Complete documentation of this matrix is not yet available.

Table 9.3-3. The diagonal elements in the interaction matrix "Buffer 1", their indexes and definitions.

Diagonal element	Position	Definition
Fuel	1.1	
Canister	2.2	The canister is assumed to be intact.
Smectite	3.3	The clay material without pore water and impurities. The physical dimensions are included.
Pore water	4.4	The water inside the buffer material, in physical contact with the canister, impurities and groundwater, but not with rock or reinforcements.
"Minerals"	5.5	The impurities in the buffer.
Gas	6.6	All gas phases in the system: entrapped air, radiolysis gases and corrosion gases. Some identified processes are only relevant for a corroding canister (in contrast to definition 2.2).
Temperature	7.7	The system is assumed to be stead-state. No transient thermal processes are taken into account.
Groundwater hydrology	8.8	
Groundwater chemistry	9.9	There is a sharp boundary between groundwater and pore water at the buffer/rock boundary.
Near-field rock	10.10	The rock that affects, or is affected by, the other diagonal elements.
Reinforcements	11.11	Construction materials in the repository and forgotten materials.
Backfill	12.12	Filler material in tunnels.
Site, layout, design	13.13	Engineering and material selection are included here.

9.3.2 The far-field matrix

The constructed matrix is named "Far field 1". The matrix is shown in Appendix 4 (fold-out). The premises for the matrix are:

- It describes a PS.
- The part of the rock that is affected by the construction of the repository (the excavation- disturbed zone) belongs chiefly to the near field.
- The boundary between the far field and the biosphere depends on how the passage of radionuclides has been modelled and how site-specific the biosphere modelling has been done.
- There are radionuclides accessible in the far field, which assumes canister damage.
- The diagonal element "Biosphere" includes the scenarios: external influence on water chemistry, minor climate change (not ice age) and ice age.
- The repository has been closed and backfilled.
- The restoration phase, i.e. the transient process when the repository returns to natural conditions, is taken into account.

The boundary conditions for the matrix are given by the diagonal elements Construction/Layout (element 1.1), Buffer/Backfill/Source (element 2.2) and Biosphere (element 13.13). The factors that have been deemed important when describing fuel processes have been selected as diagonal elements. The selected diagonal elements, their positions in the matrix and definitions are given in Table 9.3-4. The definitions in Table 9.3-4 are abbreviated versions of those given in the documentation for the matrix /9.3-1/.

The far-field matrix shows that groundwater movements, groundwater chemistry, the natural fracture system, the rock matrix and mineralogy as well as the disturbed zone (EDZ = Excavation-Disturbed Zone) are of the greatest importance for the transport of radionuclides and other solutes. Of these elements, groundwater movements is the most sensitive to changes in other diagonal elements. Groundwater movements is thus often affected by perturbations in the system. The different interactions are described in the documentation /9.3-1/ on the matrix. The way certain interactions have been treated in the safety assessment models is also described.

9.3.3 The biosphere matrix

The interaction matrix for the biosphere has been processed within BIOMOVS II, an international cooperation forum for Biospheric Model Validation. The first matrix was developed by an expert group in Sept. '94 /9.3-4/. It was then revised in May '95 by a group with experts from Ciemat, AEA, NIREX, IPCN, ANDRA, NAGRA, ENRESA, SKB and Interra. The matrix, which was given the name "Biosphere 1", is shown in Appendix 5 (fold-out). Complete documentation is not yet available for this matrix.

Background information and boundary conditions such as climate, purpose of the analysis, repository type etc. are treated in a general matrix /Fritz vDorp from NAGRA/. The premises include:

- The release takes place far in the future, so the present-day situation on the site may not exist at the time of release.
- Release of radionuclides takes place through groundwater to an inland location.
- The release continues with almost constant flow for a long time compared with the time scale for transport and concentration in the biosphere, i.e. about 10,000 years.
- The purpose of the analysis is to calculate the annual individual dose to the critical group.

Table 9.3-4. The diagonal elements in the interaction matrix "Far Field 1", their indexes and definitions.

Diagonal element	Position	Definition
Construction/Layout	1.1	Construction and layout of the repository. The element defines boundary conditions for the far field.
Buffer/Backfill/Source	2.2	The canisters, the buffer (bentonite) around the canisters and the backfill material in the tunnels. The element defines boundary conditions for the far field.
EDZ	3.3	The part of the rock that has been affected by the construction of tunnels and deposition holes.
Rock matrix/Mineralogy	4.4	The unaffected rock and its mineralogy.
Natural fracture system	5.5	The natural fracture system in the rock, fracture mineralogy, different kinds of fracture systems including fracture zones, and the mechanical properties of the fractures.
Groundwater chemistry	6.6	The groundwater chemistry in the EDZ and in the rest of the far-field rock.
Groundwater movement	7.7	All kinds of groundwater movements, both in the EDZ and in the rest of the far-field rock.
Groundwater pressure	8.8	The total groundwater pressure.
Temperature/Heat	9.9	Temperature and heat in the EDZ and in the rest of the far-field rock.
Rock stresses	10.10	The total rock stresses in the EDZ and in the rest of the far-field rock.
Gas generation/Gas transport	11.11	Gases that have been generated naturally or due to the waste, include the gases in the EDZ and in the rest of the far-field rock.
Transport of radionuclides	12.12	Transport of radionuclides in tunnels, the EDZ and in the rest of the far-field rock.
Biosphere	13.13	Describes the conditions above the repository, climate, vegetation, wells, topography etc. The element defines boundary conditions for the far field.

The selected diagonal elements and their definitions are shown in Table 9.3-5. When the interactions in the matrix were checked against an existing FEP list, it was found that fairly many new interactions had been added.

The interactions are ranked with regard to radionuclide transport as well as with regard to their influence on the actual process system. For example, the atmosphere influences the surface soil to a high degree through the process of rain, but its importance for nuclide transport is small. The level of knowledge often varies widely and must be given simultaneously.

Table 9.3-5. The diagonal elements in the interaction matrix "Biosphere 1", their indexes and definitions.

Diagonal element	Position	Definition
Source term	1.1	Radionuclide flow to the biosphere. Represents all the other matrices.
Permanent saturated zone	2.2	Soil and sediments below the groundwater table, including the groundwater there.
Surface water	3.3	Water in seas, lakes, rivers and surface runoff.
Sediment	4.4	Sediments in lakes and rivers, including pore water.
Variable saturated zone	5.5	The zone between the surface soil and the lowest groundwater table. Can be saturated at certain times.
Surface soil	6.6	Uppermost layer of the soil that is tilled and where most of the plants' roots are.
Atmosphere	7.7	The air that humans and animals breathe, including dust and aerosols.
Flora	8.8	All plants, including terrestrial and aquatic/marine plants, fungi and agricultural products.
Fauna	9.9	All animals, including terrestrial and aquatic/marine animals and agricultural products.
Human activities	10.10	Agriculture etc.
Dose to critical group	11.1	The objective of the analysis.

9.4 INITIATING EVENTS AND SCENARIOS

As mentioned previously, a scenario is a description of a hypothetical future sequence of events (evolution). The description includes both a sequence of events emanating from a set of specified premises and the future situation this sequence of events leads to. The methodology that has been chosen for a systematic description of the system is such that the premises for the sequence of events become the central aspect of the scenario description. Identifying a scenario thus entails identifying premises for the scenario. The premises for a scenario are often called initiating events.

The total body of knowledge regarding the behaviour and processes of the system parts, the behaviour of the whole repository and the features and processes of the repository and their relationship with each other constitutes the basis for scenario selection. Based on this knowledge, judgements are made of which conditions could jeopardize the performance of the repository and possible events or external conditions that could lead to these critical conditions.

9.4.1 Selection of scenarios

When the repository is closed and sealed it can be affected via

A initial conditions
B the rock
C the groundwater

A Examples of initial conditions that can influence the performance of the repository are:

- design and/or material defects in the engineered barriers (fuel, canister with steel insert and copper shell or buffer)
- faults that have to do with the construction of the repository, such as forgotten material, changes in the stresses and fracture system in the rock, faulty emplacement of engineered barriers or backfilling of tunnels
- sabotage

B Through changed stress conditions in the rock, the performance of the repository can be affected directly due to mechanical damage to the engineered barriers, or indirectly due to changes in the permeability of the rock and/or its ability to act as a buffer and a filter. Examples of loads that can change the stress conditions in the rock are ice load, earthquake and construction. By forcing their way through the rock, people can – deliberately or accidentally, with good or malicious intentions – get at the waste or affect the barriers.

C Changes in the biosphere can reach and affect the barriers through the groundwater. Examples of changes in the biosphere are climate changes, air and water pollution and man's exploitation of the natural environment.

Based on these performance-related considerations and knowledge of the repository's processes and their relationship with each other, a first scenario selection is made, i.e. a first evaluation of the scenarios and their consequences. Some scenarios can be dismissed from further analysis. One reason for not analyzing a scenario is a low probability that the initiating events will occur. Scenarios can also be dismissed due to the fact that the consequences can be said with certainty to be negligible or that the repository has been designed to withstand the scenario. Another reason for not analyzing a scenario is if the initiating event in itself causes so much damage that the impact of a radionuclide release can be neglected. The scenarios that are left are to be analyzed in greater detail.

Depending on the character of the scenario, the analysis can be done in different ways. One method for analyzing a scenario is through qualitative discussion. This discussion is often supported by rough calculations and/or investigation results from detailed studies within delimited areas. For certain types of scenarios, the analysis is performed as a conventional risk analysis. Risk is a weighing-together of the probability of an accidental event and the consequences of the event. In a risk analysis, both an assessment of how probable an accidental event is and a description of its consequences are done. The analysis can also be carried out by simulating the scenario using a computer model chain. Descriptions of the system's parts and their essential features and processes comprise the basis for the computer models. The consequences of the scenario are calculated using the computer models.

In all cases, interaction matrices can be used as a support for the analysis. The scenario can be set up on the diagonal elements of the matrix and an analysis of the interactions in the matrix can be made. Those interactions that are of particular importance within the scenario are identified. In those cases where the importance of the interactions changes due to the scenario, this is handled in a systematic fashion by documentation and description of how the change is handled in the continued analysis. So far, however, the matrices have not been used in this way.

A number of possible scenarios, their initiating events and possible consequences are presented very briefly below in tabular form. A proposal is also made for how the scenario can be analyzed. The scenarios are divided into those caused by unfavourable initial conditions (Table 9.4-1), those that affect the repository through the rock (Table 9.4-2) and those that affect the repository through the groundwater (Table 9.4-3).

Table 9.4-1. Examples of scenarios whose initiating events consist of unfavourable initial conditions. Some possible consequences are identified in the column headed "Consequences". The consequences are described in more detail as a result of the analysis.

Scenario	Initiating events	Consequences	Analysis method
Early canister damage	Fabrication defect in copper shell	Possibility for water to come into contact with the fuel. Dissolution and dispersal of radionuclides. Corrosion of steel canister. Criticality.	Calculation in computer model chain.
Defective buffer	Impure bentonite. Faulty construction of barrier. Too rapid or poor rewetting of barrier.	Elevated conductivity. Poorer thermal conductivity resulting in illitization, cementation. Poor bearing capacity, canister comes into contact with rock. Swelling pressure fails to develop. Degraded mechanical and chemical buffering capacity.	Discussion supported by rough calculation. Parameter variation in computer model chain.
Faulty emplacement of canister	Uneven bottom of deposition hole and/or faulty bentonite bottom pad.	Canister comes into contact with rock.	Discussion supported by rough calculation.
Forgotten material	Spills of various kinds not recovered. Smoke from fire. Too much grout and/or liquefying agent in cement.	Chemicals that can affect buffer and corrosion of copper shell present in repository.	Discussion supported by separate investigation and rough calculation.
Micro-organisms	High humidity, slow rate of deposition and/or poor refilling with water.	Production of corrodants.	Discussion supported by rough calculation.
Rock works	Works in one part of the repository lead to induced movements in a deposited and backfilled part.	Impact on fracture system in rock. Locally altered conductivity and/or collapse.	Discussion supported by rough calculation. Parameter variation in computer model chain.
Rock burst	Risk of rock burst and area is not abandoned.	Most of rock affected by construction. Locally altered conductivity.	Discussion. Parameter variation in computer model chain.
Sabotage	Impact on copper canister. Impact on buffer. Impact on rock. Impact on chemical environment of canister.	Degraded isolation of waste.	Discussion.

Table 9.4-2. Examples of scenarios initiated by changes in the rock. Some possible consequences are identified in the column headed "Consequences". The consequences are described in more detail as a result of the analysis.

Scenario	Initiating events	Consequences	Analysis method
Seismicity	Earthquake after ice age. Earthquake due to plate movements. Earthquake due to human activity.	Altered rock stresses. Movements in rock. Altered conditions for groundwater flow.	Discussion supported by separate study.
Ice age	Climate change	Altered conditions for groundwater flow. Altered stresses in rock. Decomposition of top layer of rock.	Discussion supported by separate study.
Stripping of rock cover	Surface mine above repository	Altered conditions for groundwater flow. Altered rock stresses. In extreme cases, short-circuiting of barriers.	Discussion Risk analysis
Tunnelling	A tunnel is excavated into the repository. Sabotage	Altered conditions for groundwater flow. Altered rock stresses. Short-circuiting of barriers.	Discussion Risk analysis
Rock drilling	A well is drilled into the repository. A borehole is drilled near or through the waste. Sabotage	Altered conditions for groundwater flow. Short-circuiting of barriers.	Discussion Risk analysis

Table 9.4-3. Examples of scenarios initiated by changes in the groundwater. Some possible consequences are identified in the column headed "Consequences". The consequences are described in more detail as a result of the analysis.

Scenario	Initiating events	Consequences	Analysis method
External impact on water chemistry	Land use, e.g. waste facility, intensive agriculture	Altered density and viscosity. Altered groundwater chemistry. More corrosive water. Water that affects buffer performance. Greater capacity to transport radionuclides.	Discussion supported by separate studies and rough calculation. Parameter variation in computer model chain.
Minor climate changes (not ice age)	Natural climate changes. Global warming, human impact	Altered flow. Altered density and viscosity. Altered groundwater chemistry.	Discussion supported by separate studies and rough calculation. Parameter variation in computer model chain.
Ice age	Climate change	Altered flow. Altered density and viscosity. Altered groundwater chemistry.	Discussion supported by separate study.

9.4.2 Analyzed scenarios

Examples of how different scenarios can be analyzed via qualitative discussion supported by separate studies and rough calculations, simulation and calculation in a computer model chain, and conventional risk analysis are given in this report.

For qualitative discussion, a scenario is selected where the analyzed system performs as intended. All barriers work and evolve normally. This scenario is called the normal scenario. The normal scenario is dealt with in section 12.2.

For simulation and calculation in a computer model chain, a scenario with early canister damage is selected. In this scenario it is assumed that a portion of the copper shells on the canisters have fabrication defects in the form of initial holes. This scenario is called the type defect scenario and is dealt with in section 12.3.

The scenarios whose initiating events are a consequence of human activities are usually termed future human action (FHA) scenarios. What is meant by FHA scenarios and the kind of reasoning that can be applied to them is clarified by qualitative discussions. The actual procedure followed in analysis of FHA scenarios can vary depending on the type of initiating event. As an example, a special case with rock drilling is treated where the analysis has taken the form of a risk analysis. This is presented in section 12.4.

Finally, section 12.5 shows how the sequence of events during an ice age can be handled in the safety assessment. The parts of SKB's paleogeohydrological research programme that deal with climate change and evolution and the retreat of an ice sheet as a consequence of them is presented. The way in which climate and ice can affect rock and groundwater flow is also described. The section does not contain a consequence analysis, but merely describes how the sequence of events during a glaciation cycle leads to different external conditions for the repository.

9.5 UNCERTAINTY AND COMPLETENESS

As is evident from section 3.4, the following questions regarding uncertainty and completeness must be addressed in the scenario work:

- completeness of the description of the repository system
- completeness of the scenario selection

The selected scenarios should provide a comprehensive picture of the possible and relevant evolutionary pathways of the repository system. In order for this picture to be complete, it should be based on an analysis of a complete description of the repository system.

The description and analysis of the system and the selection of scenarios are done by expert groups. No one part can be more complete than the aggregate knowledge possessed by the experts who carry out the work. The word "aggregate" should be underscored, since it can be assumed that an integration of results from several scientific disciplines will lead to a more complete picture than the separate results alone. Knowledge is changeable and becomes broader and deeper with time. Answering the question of whether the description of the repository system and the choice of scenarios are complete is not possible in the strict sense. What can be done is to make an assessment of whether the present-day state of knowledge is sufficient to carry out the desired description, analysis and scenario selection.

To solve the completeness problem, a large quantity of knowledge of various kinds needs to be assembled and placed in its context. Research results from various academic disciplines, within both the natural sciences and the humanities, must be gathered and coordinated. Doing this requires a systematic approach, a methodology, what we call in this context a scenario methodology.

The scenario methodology should offer systematic methods to:

- describe the repository system,
- analyze the described system, and
- select a set of scenarios based on the description and analysis.

Answering the question of whether the selected scenarios cover the realistically possible evolutionary pathways of the repository can be said to be equivalent to answering the question of whether the scenario methodology and its application meet the stipulated requirements.

A complete description of the repository system should include barriers, flow systems and the transport of radionuclides, as well as the biosphere and the transmission of radionuclides to humans, animals and plants. Furthermore, the description must be able to handle the evolution of the repository with time, its evolution as a consequence of given initial conditions, and its evolution as a consequence of external influence. Today this description is done by constructing a number of interaction matrices of the system.

Answering the question of whether the formulated description is adequate, or complete, is naturally difficult. To gather the knowledge available today, a great deal of work has been devoted to identifying all features, events and processes (FEPs) that could conceivably influence a repository system and its performance and that could conceivably exist now or in the future. Databases of FEPs have been built up both in Sweden and internationally. Demonstrating that all FEPs from such databases are included, or can be handled, within the formulated description is one way to demonstrate the completeness of the description.

Another way to determine whether the description is adequate or complete is to answer the question of whether the methodology for formulating it meets stipulated requirements. Can the desired description of the repository system be formulated by following the instructions in the methodology and using its symbols? Having a methodology is, of course, not enough; the application of the methodology must also be correct and complete.

The above procedure can also be used for determining whether the analysis of the system and the choice of scenarios based on this analysis are satisfactory. Can the employed methodology and its application be said to meet the stipulated requirements? Does the scenario methodology offer a systematic procedure for analyzing the described repository system? Do clear instructions exist for how scenario selection is to be done? Has the methodology been applied in a correct and complete manner?

As mentioned previously, the scenario methodology is under development. Assessments of whether the current methodology and its application meet stipulated requirements remain to be made.

10 ANALYSIS – PERFORMANCE OF THE REPOSITORY

The analysis of the performance of the repository system is divided into two chapters. Chapter 10 describes the processes and conditions that control (or can change) the primary safety function of the barriers – to isolate the spent nuclear fuel from the groundwater. Chapter 11 describes the processes and conditions that control the repository's secondary safety function – to limit the dissolution of radionuclides in a damaged canister, and to retard their transport to the biosphere.

This chapter describes the anticipated performance of the different repository subsystems in maintaining the isolation, and processes that could influence this performance. The chapter is divided into the following sections:

- *thermal development*
- *rock performance*
- *canister performance*
- *buffer performance*
- *dissolution processes in the canister*

Functional requirements, essential processes, methods for analyzing and quantifying the processes, and parameter limitations introduced to ensure the desired function or to simplify the analyses are presented in the sections.

Certain processes and conditions are essential for both the isolation and the retardation functions. One example is the movement of the groundwater and its ability to transport dissolved substances or colloids. Such processes are only dealt with in the one chapter, and references are made between the chapters.

***In this chapter**, the systematic review of the performance of the repository parts in relation to the functional requirements has been replaced by a review of essential processes in different parts of the repository. The discussion here centres on the processes and models that are available to analyze and quantify performance, and the background data and parameters that are required for this quantification. In this work it has been found that those sections that present the background data in both chapters 10 and 11 can be very long. One way to improve readability is to break them out to a separate data chapter.*

10.1 INTRODUCTION

The most important parts of repository performance are discussed in this chapter. It is an in-depth analysis of the central features and processes of the repository that emerged in the description of the evolution of the repository in Chapter 9.

The analyses focus on the ability of the repository parts to uphold the isolation of the radioactive waste, and processes that may threaten this isolation.

The following sections deal with the thermal development of the repository, the performance of the rock, the buffer and the canister, dissolution processes in the canister, and the chemical speciation of the radionuclides.

10.2 THERMAL DEVELOPMENT

10.2.1 Introduction

The presentation focuses on essential mechanisms, modelling of heat transport and a brief description of important temperature calculations performed previously.

The deposition of canisters with spent fuel in the deep repository entails the emplacement of a powerful and long-lasting heat source in the rock mass. The heating of both the near field around the canister and the surrounding rock can affect barrier performance in several ways. The groundwater movements are affected by the thermal gradient that is created, but also by possible changes in the rock's fracture structure, caused by the temperature increase and the subsequent cooling, which causes stress redistributions and movements in the bedrock. The temperature in the immediate vicinity of the canister rises rapidly and within a few days reaches a level that may be of importance for certain chemical or physical processes. Limiting this temperature rise imposes requirements on the canister's dimensions and on the distance between canisters in the repository.

All in all, this means that studies of the heat transport in and around the deep repository comprise an important element in both safety assessments and repository design/layout. They have therefore been included in the development work for the deep repository right from the start.

10.2.2 General about heat transport in the deep repository

Heat transport can take place by conduction, radiation and convection. In the rock mass – as in all solid, homogeneous materials – heat transport takes place solely by conduction. In the fractured bedrock and in porous material, all three modes of transport can occur. For practical reasons, however, heat transport in these latter materials is also regarded as taking place solely by conduction, whereby the coefficient of thermal conductivity is determined so that it includes the aggregate effect of all three mechanisms.

Heat transport can also take place via groundwater flow or via ventilation of rock caverns and tunnels, as long as these remain open.

Modelling of the deep repository is done primarily with a view towards calculating heat transport by conduction. This calculation is performed with the general heat conduction equation, which is described in all standard works on heat, for example in /10.2-1/. To be solved, the equation must be adjusted to the boundary and initial conditions that have been set up. Similarly, material constants and thermal parameters in the equation must be chosen, including thermal conductivity and heat capacity. It is especially important to take the heat transfer between different materials into consideration.

"Temperature" is found as a diagonal element in several of the interaction matrices presented in Appendices 1–4. Interactions between temperature and different repository parts and other features can thus be studied here.

10.2.3 Modelling of heat transport

Analytical methods can only be applied in very simple configurations in the calculation of heat propagation. Such a simple case may be heat propagation from a single canister. Normally, the heat conduction equation is solved by means of some numerical method. Since the heat input gradually declines in accordance with the decay curve for the fuel, time-dependent calculations are performed.

In modelling the deep repository as a whole, the geometric scope of the model is bounded by any planes of symmetry and, normally, the assumed location of the ground surface, as well as by planes located at an adequate distance from the front of the heat wave for the time span covered by the calculation or which otherwise have a negligible effect on the calculation result. The boundary planes define the boundary conditions which best agree with the actual heat flow, e.g. adiabatic condition at the planes of symmetry (insulated), given temperature conditions or coefficient of heat transfer for the interface with the ground surface, etc.

The geometric modelling varies in degree of detail depending on the purpose of the calculation. It may be to conduct studies of the interaction between repositories on different levels, thermomechanical effects within areas at a given distance from the repository, e.g. at the ground surface, or, in the extreme case, studies involving future ice ages.

The initial conditions, material constants and thermal parameters assumed in the model contain several uncertainties. The choice of parameter values is made from case to case with varying conservatism, depending on the purpose of the calculation in question.

The initial temperature distribution over the model takes the form of an assumed temperature at repository level on which a straight-line temperature gradient is superimposed (temperature increases with depth), see section 6.2. Other thermal parameters are taken from the studies and field investigations that are continuously being conducted. The buffer material around the canister, compacted bentonite, is of particular importance in this respect. Its properties are of great importance for the maximum temperature at the canister surface and thereby comprise one of the crucial factors for the distance (spacing) between the canisters and thereby the horizontal extent of the repository. For a more detailed description of relevant thermal parameters, see section 7.2.

The thermal energy input, given as the source density of the heat-generating elements in the model, derives from the fuel's decay heat and is taken into account in the model by insertion of the exponential functions for the decay.

The limitations in the modelling relate primarily to the difficulty of describing the bedrock with all its inhomogeneities, fracture structures, etc. At present, this is being handled by assigning suitable values to the input parameters. An important step in the analysis is to study the relative importance of possible ranges of variation of these values by means of variation studies. This means that the limitation lies not in which processes can be handled by the numerical models, but rather in the input data that must be specified.

10.2.4 Available calculation tools

Heat propagation calculations for the deep repository are performed today by means of FEM analysis. There are several general codes permitting advanced heat flow calculations on the market. The two shown in Table 10.2-1 are used within SKB's activities. They are also used in many other areas of activity with good results.

Calculation accuracy, based on the chosen parameters, is primarily dependent on chosen element type and element subdivision in modelling. Modelling is progressively refined until the accepted accuracy is achieved. Calibrations take the form of introductory pilot studies.

Table 10.2-1. Numerical models for description of temperature effects around a deep repository in fractured rock.

ANSYS	ANSYS is a general FEM program with a directory that includes all existing element types. The program has well-developed pre- and post-processors with many options for integrated operation with other databases and formats. A more detailed description is provided in /10.2-2/. ANSYS is used for linear and non-linear, thermal, statistical, dynamic, magnetic and acoustic analyses. The element directory includes composite scale and solid elements. The program furthermore does an automatic error estimate where numerical model accuracy is estimated.
SOLVIA	SOLVIA is also a general FEM program with largely the same features as ANSYS. The program is particularly suitable for non-linear analyses. A more detailed description is provided in /10.2-3/. SOLVIA is a further refined version of the well-known program ADINA.

10.2.5 Results of temperature calculations

No new temperature calculations have been done for the canister type that is presented in SR 95, see section 5.3. Previous calculations show that the maximum temperature on the canister surface lies below 100°C for all times /10.2-4/. For the currently selected canister and the selected reference fuel, the temperature is not expected to appreciably exceed 80°C. The maximum will occur around 10 years after deposition, after which the temperature will fall again. The important parameters are the thermal conductivity of the buffer and the thermal properties of the rock, plus handling of contacts between canister/buffer and buffer/rock. Updated temperature calculations will be presented for future safety assessments.

Other, more large-scale temperature calculations have been carried out to obtain the temperature distribution in the rock around the deep repository for a number of hypothetical heat outputs from the fuel /10.2-5/. The temperature in the central parts of the repository reaches a maximum after 100–400 years, after which the temperature declines. After around 400 years, the heating front has reached the ground surface. The mechanical effects of the heat output at ground level have also been studied. In summary, the results in /10.2-5/ show that from a global thermo-mechanical viewpoint, it is possible to dispose of nuclear fuel according to the KBS-3 concept with an initial heat load of up to 10 W/m^2 of repository area.

Other large-scale calculations show that the temperature increase in the repository does not appreciably affect the flow of water around the repository /10.2-6/. This conclusion is, however, dependent upon which repository layout is chosen and on the selected site.

10.3 ROCK PERFORMANCE

10.3.1 Introduction

The performance of the rock in a deep repository, and in particular the interaction between the deep repository and the rock, is discussed in this section. The section should be regarded as a link between the description of the repository site in Chapter 6 via the application of the scenario methodology for the far field in section 9.3.2 to the results that are discussed further on in this report. It is vital that the performance description in this section be linked to the review of the process system for the far field that was done in section 9.3.2 and that is depicted in a fold-out appendix at the end of the report. The text therefore contains references within square brackets [X.Y], where X.Y indicates the intended interaction in the interaction matrix. The description in this section does not cover all interactions, but together with the description in Chapter 11 the most essential ones, including the application of the interaction matrix in /10.3-1/, have been taken into consideration.

10.3.2 Rock performance in a deep repository

The deep repository will be sited in rock where it is possible to avoid major future rock movements at the canister positions [5.1, 10.1]. After closure, the rock around the repository will become saturated with groundwater [1.8], [2.8], [7.8], [8.7]. The decay heat from the fuel raises the temperature in the rock nearest the deposition holes [2.9], which subsequently slowly declines [4.9], [7.9]. Elevated temperature will prevail for thousands of years. The moderate temperature increase scarcely gives rise to any increased rock stresses in the near field [9.10]. One effect of the temperature increase is that the aperture width of the fractures in the rock nearest the deposition hole will decrease [9.4] –> [4.5], only to increase again when the temperature falls. The temperature rise may also create local convection cells [9.8], [9.7], [8.7] and changes in the groundwater chemistry [9.6], but these processes do not affect the performance of the repository. In view of the thermal effects, a distributed repository layout is used [2.1].

Some central properties of the bedrock, see Figure 10.3-1, are essential to guarantee the performance and long-term radiological safety of the deep repository. There are obvious couplings between these properties:

- a long-lasting *mechanical stability* [10.1], [10.2], [10.3], [10.4], [10.5],
- a *chemically stable environment* with a groundwater which does not contribute to corrosion of the canister material, and which ensures low solubility and high retardation of the outward transport of the radionuclides in the waste [6.1], [6.2], [6.3], [6.4], [6.5],
- a *slow groundwater* flux which limits the transport of radionuclides and substances having an adverse effect on the waste products and the backfill material [7.2], [7.12].

Besides the above central geoscientific properties, it is essential to minimize the risks of future intrusion. The deposition depth and the usually occurring rock type contribute towards reducing these risks.

Mechanical stability

With reference to the above properties, the Swedish bedrock generally offers an environment that is well-suited to guaranteeing the performance of the repository. The Swedish crystalline basement is a part of the Baltic Shield. Most of the Swedish crystalline basement has an age of about 1,700 million years.

A compilation has been made of the different tectonic regimes to which the Baltic Shield has been subjected during the past 1,200 million years /10.3-2/. The review focuses on different "load cases" with the associated stress fields that have prevailed. The present-day surface of the crystalline basement exhibits fracture patterns and heterogeneities that have been formed and evolved during many hundreds of millions of years. Analyses show that large-scale stresses have caused the fractured bedrock to be deformed along already existing fractures and zones. Any active loads during the coming 100,000-year period will thus probably reactivate old zones and fractures in the earth's

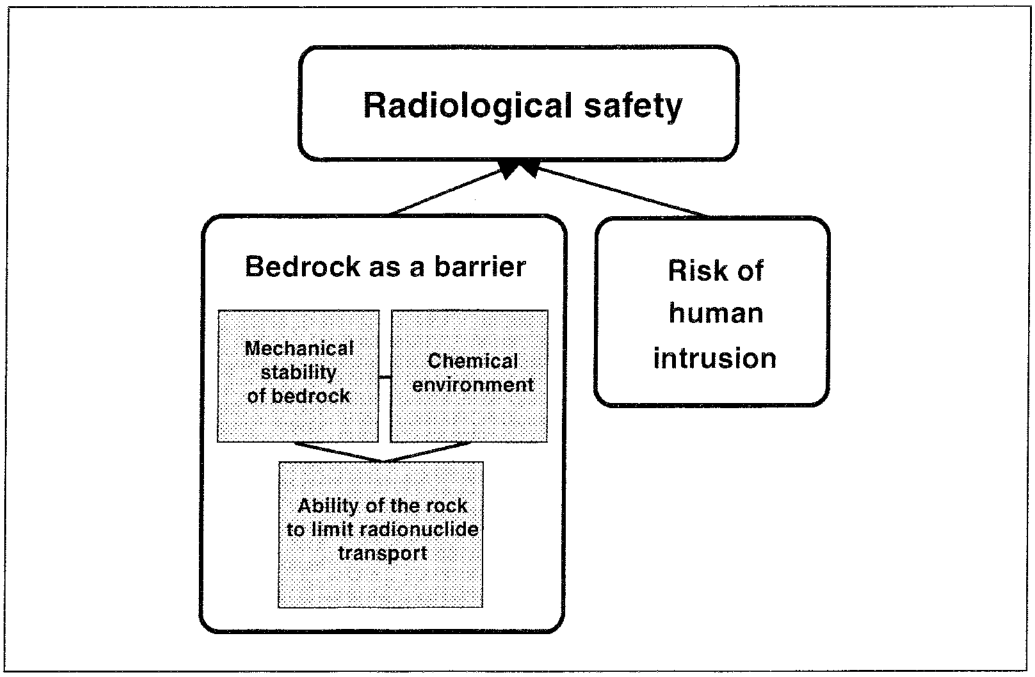


Figure 10.3-1. *Properties and conditions in the bedrock that are of importance for the long-term radiological safety of a deep repository.*

lithosphere on a regional scale. The present-day tectonic situation with a relatively passive response to the Mid-Atlantic ocean floor spreading is a calm period for the Baltic Shield.

Chemical environment

Groundwater from Äspö can be mentioned as an example of the chemical environment at repository depth. It can be described as a mixture [7.6] of modern infiltrated fresh water [13.6], pore water in bottom sediments, glacial meltwater [13.8], modern Baltic Sea water and old, very saline water. Results of investigations from Äspö and Laxemar show that the chemical conditions at great depth, 1,000 m, have presumably been stable on a time scale of 100,000 years or more.

Groundwater flux

The ability of the rock to limit transport of different substances is a composite property which has to do with groundwater flows and flow paths, among other things. These are in turn determined by the permeability [3.7], [5.7] and inhomogeneity of the rock, and the topography [13.8], water balance [13.6], [13.7], etc. of the area in question. If water transport takes place without the influence of retention mechanisms, it is regarded as non-reactive and involves only non-sorbing substances [7.12]. The non-reactive transport process includes advection, kinematic dispersion [7.12] and molecular diffusion [5.12]. The ability of the rock to retard nuclides has to do with the diffusion and sorption properties of the bedrock. They are determined by e.g. the groundwater chemistry [6.12], the mineralogical composition of the rock [4.12], the character of the fracture pattern [5.12] and the availability of sorption surfaces [3.12], [5.12]. The mechanisms that are of importance for groundwater movements and radionuclide transport are detailed in Chapter 11, where modelling aspects are also touched upon.

The impact of an ice age on groundwater has been analyzed in the Äspö laboratory. The groundwater has been affected down to a depth of about 500 m by conditions since the most recent ice age. The earlier stages of the evolution of the Baltic Sea can be traced in the composition of the water. The work has reinforced previous indications that the water at great depth, at Äspö below 500 m, is insignificantly affected by postglacial events and is therefore to be regarded as stagnant on a 10,000-year time scale /10.3-3/.

10.3.3 Effects of the deep repository in the rock

Hydrogeological effects

The construction of the tunnels and shafts in the deep repository will create great disturbances in the groundwater system around the repository [1.8]. These disturbances can to some extent be limited by grouting of water-conducting fracture zones and fractures around the repository. After closure, the situation returns to water-saturated conditions in the rock [2.8]. The time this takes depends on the hydraulic conductivity of the rock [5.7] and the properties of the disturbed zone around tunnels and shafts [3.7]. Separate studies have been made to study the resaturation process, and most of the evidence suggests that it takes at least one year. Finally, it can be observed that the chosen buffer material penetrates into fractures that intersect the deposition holes, which in turn affects the groundwater movements [2.5], [5.7].

Geochemical effects

Chemical changes in the groundwater can occur for many reasons, e.g.:

- mixing of groundwaters with different properties [7.6], or
- chemical reactions due to changes in pressure [8.6], temperature [9.6], pH or Eh.

During construction, a number of foreign substances are introduced which influence the environment in the rock and are conducted down into the repository [1.6]. The chemical composition of the groundwater is also affected by the presence of the bentonite clay. The bentonite in particular will be a source for the formation of colloids [2.6]. The existence of the disturbed zone also influences the groundwater chemistry [3.6].

The repository for other long-lived waste should be situated so that it does not influence the groundwater chemistry at the repository for spent fuel.

Mechanical effects

The part of the rock that is affected directly by the construction work or indirectly by the stress redistributions during construction is called the disturbed zone, or EDZ (Excavation Disturbed Zone) [1.3], [3.1], [10.1], [10.3]. The buffer and the backfill material can penetrate into the EDZ and affect its properties [2.3]. The importance of the EDZ for the migration of radionuclides from the deep repository must be evaluated in the safety assessment [3.7], [3.12], see e.g. /10.3-10/.

The design and size of the tunnels, shafts and rock caverns in the repository affect the stress conditions in nearby rock [1.10].

Thermal effects

The thermal effects were described briefly in the introduction and have furthermore been dealt with in section 10.2.

Effects of gas

During the construction of the deep repository, air and blasting gases are introduced into the rock [1.11]. Furthermore, the decrease in groundwater pressure will lead to the release of gases dissolved in the groundwater [1.8] –> [8.11]. Rock reinforcements may corrode and generate hydrogen gas [1.11]. Hydrogen may also be formed by corrosion of the canister's steel insert, if the copper shell has been penetrated [2.11]. In addition, it is conceivable for gases to be generated by chemical or bacterial action [6.11]. The fracture system in the rock comprises transport pathways for gas [6.11]. The effects of the gas on radionuclide transport must be analyzed [11.12], [11.13]. The presence of both gas and groundwater in the fracture system in the rock means that two-phase flow with interaction between gas, water and fracture system can arise [11.7]. The existence of gas affects the groundwater chemistry [11.6] and the conductivity of the fractures [11.5]. Groundwater pressure and temperature affect the gas transport [8.11], [9.11].

10.3.4 Changes with time

Meteorological changes and topographical variations

The groundwater levels in Swedish bedrock largely follow the topography of the ground surface. In valleys there are small differences between the groundwater level and the ground surface, while there may be a difference of a metre or so in high areas. A certain seasonal variation can also be noted. The groundwater level in the rock controls the pressure gradients, which in turn control groundwater movements in the rock. The existing variations in water levels give a negligible uncertainty in a model for groundwater movements in comparison with other factors such as vertical boundary conditions and hydraulic structures /10.3-4/.

During the time up to the next ice age, the topography may be affected by land uplift and erosion. Land uplift is completely dominant for the rock types in question. The present-day rate of land uplift in the Äspö area is 1 mm/y. Erosion in connection with glaciations is discussed in the next section.

Changes in topography can affect groundwater movements and thereby the transport of radionuclides. Furthermore, the character of the discharge areas can change, especially in coastal areas. The latter is of importance for the biosphere description.

Effects of topographical variations are not dealt with in the model chain which SKB uses for illustrative calculations in SR 95. However, a special analysis has been done to ascertain the effect of future land uplift at Äspö /10.3-5/. In approximately 2,000 years, the sea bays around Äspö are expected to become closed off from the surrounding sea, leading to a different biosphere description. The effects this will have on groundwater movements are examined in /10.3-5/.

Climate change and glaciation

The most recent ice age in Scandinavia started approximately 100,000 years ago. At its maximum extent about 18,000 years ago, the continental ice sheet covered all of Scandinavia and northern Germany. The ice cover over Småland was then approximately 2,000–2,500 m thick. The weight of the ice caused the bedrock to be pushed down about 500 m. The sea level in the oceans was then around 20 m below today's level. The ice cover melted away relatively quickly, and about 12,500 years ago the continental ice sheet retreated from Småland. Due to the previous depression, this region was then covered by sea. The region was subsequently covered alternately by seawater and fresh water until 3,800 years ago, when the process of postglacial land uplift left Äspö as dry land. The former highest coastline lies around 90 m above today's sea level.

The general effects of the colder climate are vast areas with permafrost in the ground, altered precipitation conditions, growing glaciers and greatly lowered sea level. During a glaciation equivalent to the last one, the ground surface will be eroded by several metres and large quantities of soil will be redistributed.

During the deglaciation phase, the oceans rise once again. Ice stream material is deposited in the form of long ridges or delta formations in the landscape. The groundwater flux in the upper parts of the rock can be affected during a transition period. The earth's crust strives to resume its original form, and increased seismic activity can be expected immediately after a land area has become ice-free. Earthquakes are triggered by movements in already existing and very old, large zones of weakness in the bedrock.

Based on our knowledge of the extent of the Weichsel glaciation during different stages and indirect climate data from pollen and isotope analyses, SKB has had a time-dependent glaciation model developed /10.3-6/. The model is able to handle coupled processes with:

- temperature changes in ice and bedrock,
- growth of inland ice sheet and flow in landscape with actual topography,
- generalized mechanical influence on the lithosphere,
- meltwater flows in different stages of a glaciation cycle

A simple calculation program for groundwater flow has also been attached to the model for the purpose of being able to handle and describe hydrogeological changes accompanying permafrost, glaciation and deglaciation. Of particular interest are e.g. pressure and gradient changes under the ice and in the bedrock. The groundwater model also makes it possible to follow the flow of percolating meltwater in simplified fashion.

Seismicity

Seismicity can be said to be a sign of more or less instantaneous bedrock movements in a geological time perspective. More than 95% of all earthquakes take place at the boundaries of the continental plates. Approximately one million earthquakes with a magnitude of more than 2 on the Richter scale occur annually in the world. Of these, around 10 quakes occur in Sweden. In other words, our country is a seismically inactive area. The largest earthquakes occurring in Sweden reach a magnitude of 5. There are indications that more powerful earthquakes can occur in Sweden in conjunction with glaciations and deglaciations.

The Swedish earthquakes are primarily concentrated in two areas. One area extends from Lake Vänern down to the west coast. The other area follows the coast along the Gulf of Bothnia towards Tornedalen and northern Lapland. Most of the Swedish quakes occur deep down in the bedrock, i.e. the epicentre of the earthquake lies 10–20 km below the surface and the movements there are small. Calculations indicate displacement sums of about 10 mm at magnitudes of around 5 /10.3-7/. The movement takes place as a reactivation in an existing fault structure and within a radius of about 900 m down at the epicentre of the earthquake.

As far as the effects of earthquakes on underground facilities are concerned, the mechanical stresses on such facilities are generally less than for facilities on the ground surface. Many observations, above all from mines, support this contention /10.3-8/.

For the performance of a deep repository in Swedish crystalline basement, the mechanical effects of seismicity are probably subordinated to geohydraulic and hydrochemical influences, which in turn are found to have very limited effects /10.3-9/.

10.4 CANISTER PERFORMANCE

The performance of the canister and the processes than can affect it are dealt with in this section. The mechanical properties of the canister, the effects of radiation inside and outside the canister, and copper and iron corrosion are taken up for discussion. The section concludes with a description of how natural analogues can be used to understand the performance of the canister in a deep repository.

The canister is described in greater detail in section 5.3. It consists of two components, a cast steel insert (inner canister) and a copper shell (outer canister). The performance of the canister in the repository is also described in brief in section 9.1. Appendix 1 contains the interaction matrices that describe the features and processes of the canister.

10.4.1 Mechanical properties of the canister

The canisters are emplaced, embedded in bentonite clay, at a depth of 400–700 metres in the rock. They are designed to withstand an evenly distributed maximum load of 7 MPa hydrostatic pressure and approximately 10 MPa pressure from bentonite and rock. The customary safety margins are incorporated in the design requirements. The canisters must also be able to withstand uneven load and increased hydrostatic pressure. Furthermore, the ductility of the copper material must permit strain in conjunction with creep or temperature movements without the copper shell cracking. Strength calculations show that the reference canister collapses at an external pressure of about 80 MPa /10.4-1/.

The canisters must be fabricated, sealed and inspected with methods that guarantee that less than 0.1% of the finished canisters will contain defects that could lead to early canister failure.

10.4.2 Effects of radiation inside and outside the canister

Radiolysis products on the outside of the canister can contribute to corrosion attack on the copper shell. The canister must provide such radiation protection that corrosion caused by radiolysis products is small compared with the corrosion caused by the oxygen from the operating period. This condition gives a permissible surface dose rate of approximately 500 mGy/h, see also section 10.4.3.

The radiation dose in the void between the fuel and the steel insert is initially approximately 300 Gy/h. In the presence of water and air, this radiation can lead to radiolytic formation of nitrate. Nitrate could cause stress corrosion failure of the steel insert. The fuel is temporarily stored submerged in water and will therefore be dried before it is placed in the canister. The air in the canister void will be replaced with inert gas. Stress corrosion on the steel insert will be further investigated, see also section 10.4.4.

10.4.3 Copper corrosion under different conditions

The corrosion properties of copper have been thoroughly studied, and a summarizing assessment of the state of knowledge is given in /10.4-2/. The conclusions are that the copper shell is so resistant to both general and local corrosion that it is deemed unlikely that corrosion will be the factor that limits the lifetime of the canister in the repository. How extensive the corrosion attacks will be is determined by the chemical environment in the immediate near field of the canister. Of great importance are the composition and redox conditions of the bentonite pore water.

Oxygen will be introduced into the repository during its operating period. It is estimated that the oxygen will remain in the deep repository up to a few hundred years after closure /10.4-4/. The oxygen is consumed in reactions with copper or minerals in the near field. When the oxygen from the operating period has been consumed, the repository will be oxygen-free.

When the oxygen is consumed, sulphides will be the dominant corrodant. The corrosion rate will then be dependent on the availability of sulphides. The sulphide ion acts as a corrodant by forming copper sulphide, thereby reducing the free copper ion concentration. Sulphide is present as an impurity in the buffer material. When this sulphide has been exhausted, a small continuous supply comes from the groundwater.

The following table shows the expected corrosion depth after 100,000 years. The corrosion has been calculated for a probable case and for a case based on pessimistic assumptions /10.4-3/.

General corrosion	**Probable case**	**Pessimistic case**
Initial phase in presence of oxyggen	0.5 μm	5 μm
Oxygen-free long-term phase	5 μm	0.4 mm
Pitting		
Initial phase in presence of oxygen	250 μm	2.5 mm
Oxygen-free long-term phase	10 μm	2 mm
Total corrosion depth	~ 270 μm	~ 5 mm

As mentioned previously, radiolysis contributes to corrosion of the copper shell. The canister shields off some of the radiation that can cause radiolysis. Calculations /10.4-5/ show that at the planned thicknesses of the insert and shell, the contribution to corrosion from radiolytically produced oxidants will be negligible.

Micro-organisms in the repository can reduce sulphate to sulphide and thereby contribute to corrosion of the copper shell, so-called microbial corrosion. Previously, it was believed that the availability of organic matter in the near field limited the growth of micro-organisms. More recent research results indicate that the availability of organic matter does not limit bacterial growth. Further studies of the growth of sulphate-reducing bacteria in compacted bentonite have therefore been initiated and are still going on. Results obtained thus far show that bacteria cannot survive at bentonite densities above 1,500 kg/m^3 /10.4-14/, see further section 10.5.4.

To obtain a copper with good resistance to stress corrosion cracking (SCC), a high-purity oxygen-free copper grade was originally suggested, since the likelihood of SCC was then judged to be non-existent /10.4-6/. More recent research results have shown that SCC cannot be ruled out on these grounds, but that SCC is unlikely under the conditions prevailing in the repository /10.4-7/. The latter results have been confirmed in several studies, which show that the tendency towards SCC in the repository environment is small /10.4-8/.

10.4.4 Iron or steel corrosion under different conditions

As mentioned previously, the radiation inside the canister in the presence of water and air can lead to radiolytic formation of nitrate, which could cause stress corrosion failure of the steel insert. This process and its consequences have not been fully investigated. To limit the supply of water, the fuel is dried and the air in the canister is replaced with inert gas. If a canister initially contains 0.01% air, calculations show that about 0.02 g of nitric acid will be formed /10.4-9/.

The consequences of corrosion of the steel insert after penetration of the copper shell have been explored in recent years. The following phenomena have been investigated:

- corrosion rate, corrosion mechanism and generation of hydrogen gas (the consequences of hydrogen gas production are dealt with in section 10.5.3)
- pressure buildup caused by the growth of corrosion products
- galvanic effects

Both experimental studies and theoretical modellings have been carried out. The results of these studies are found in /10.4-10/ and /10.4-11/. The main conclusions from the studies are:

- Experiments show that the corrosion rate for steel under oxygen-free conditions is dependent on pH and ionic strength. Lower pH and higher ionic strength increase the rate. The corrosion rate is in principle independent of other factors. For example, experiments show that the corrosion rate in a humid anoxic atmosphere is the same as for a specimen immersed in water. The corrosion rate is controlled by the formation of a protective film of magnetite on the steel. Under the conditions prevailing in the repository, the corrosion rate is estimated to lie in the range 0.1–1 μm/y. This is equivalent to a maximum hydrogen gas production of about 1.6 m^3/y (NTP).
- Calculations show that in the event of a defect in the copper shell, corrosion will take place over a large portion of the steel surface. The corrosion products formed should therefore be uniformly distributed in the canister, which means that will not give rise to such stresses in the copper shell that the defect in the shell will be widened.
- Galvanic coupling between the copper shell and the steel insert can only occur if there is a continuous solution between the metals. This can only arise if water enters the gap between the copper shell and the steel insert.

In the presence of oxygen, a considerable galvanic corrosion of the steel could take place. The low transport rate for oxygen through the bentonite prevents any significant galvanic corrosion during the oxygenated period of the repository, however. The copper-steel coupling can increase the corrosion rate of the steel even under anoxic conditions. Under these conditions the increase in corrosion rate is only by a factor of 2. Experiments also show that the magnetic film is reaction-controlling, i.e. galvanic effects will not influence the corrosion rate of the steel once the film has formed.

If the steel corrodes at the maximum corrosion rate, 1 μm/y, and corrosion occurs over the entire steel surface, both inside and outside, it will take more than ten thousand years for the steel to lose its bearing capacity. By that time, all voids inside the canister will be filled with corrosion products, magnetite. Since the canister if filled with magnetite, it will not collapse, despite the fact that the steel insert has lost its bearing capacity.

10.4.5 Natural analogues

Copper occurs in metallic form in nature. Those analogues that are most valuable for corrosion estimates are, however, archaeological finds. One example is a cannon from the Swedish warship Kronan /10.4-12/. The cannon has been buried with its muzzle down in the sediment since the ship sank in 1676. The high copper content of the cannon and the montmorillonite content of the sediment make the cannon a close analogue to the canisters. Chemical analyses show that copper has diffused approximately 4 cm into the clay. The corrosion was uniform over the entire surface of the cannon. The corrosion rate could be estimated at $1.5 \cdot 10^{-5}$ mm/y.

Metallic iron and steel occur very sparsely in nature. Archaeological artefacts such as the Roman nails at Inchtuthil /10.4-13/ show hat iron also corrodes very slowly if it is not exposed to oxygen.

10.5 BUFFER PERFORMANCE

10.5.1 Introduction

The performance of the buffer in the final repository is discussed in this section. The section begins with a description of the initial water saturation process undergone by the bentonite buffer.

Then follows a review of the processes and features that affect buffer performance: thermal stability, radiation, smectite/illite conversion, hydraulic conductivity, swelling properties, gas transport capacity, and the importance of cement. The natural analogues that have been studied to get an idea of the long-term properties of the buffer are also mentioned here.

The way in which the bentonite influences the chemical conditions in the near field is then described. The buffer's impact on the groundwater chemistry and effects of micro-organisms, organic matter and colloids in the bentonite are discussed.

The section concludes with a discussion of the mass transport properties of the buffer material. Diffusion in, and sorption on, the buffer material are discussed here.

The description concerns the specific buffer material that has been selected for the final repository. It has a high content of Na-montmorillonite and goes under the technical designation MX-80. A description of the chosen buffer is given in section 5.4. Appendix 3 presents the interaction matrix in which the features and processes of the buffer have been studied.

10.5.2 The unsaturated period

Boundary conditions

In order to obtain as high a water saturation as possible at an early stage, the bentonite blocks are compacted with the highest degree of water saturation possible. The gaps between canister and bentonite block and between bentonite block and rock wall are filled manually with water with a low ionic strength. The bentonite volumes are designed so that the buffer in swelled-out condition has a density of about 2.0 t/m^3.

After closure, the groundwater moves back into the repository. It is assumed that the hydraulic gradient that develops leads to a movement of water in towards the deposition hole. The total water content of the buffer can therefore not be lower than its original value. Furthermore, it is assumed that the rock zone nearest the buffer is sufficiently fractured to distribute the water evenly along the entire height of the deposition hole.

Saturation process

When the gaps between bentonite block and canister and between bentonite block and rock wall are filled with water, the hydration process begins. It is driven by negative capillary pressure in the buffer. The negative capillary pressure is greater than the swelling pressure developed by the buffer at full water saturation /10.5-1/. The negative pressure is therefore also greater for the clay particles that have higher density. It is therefore these particles that become saturated and swell first. The rate of the water saturation process is determined by the negative pressure in the bentonite, the groundwater pressure and the hydraulic conductivity of the near-field rock and the buffer.

When full water saturation has been obtained, water and clay particles are redistributed so that the dense clay particles expand and thin layers of bentonite are released to the gradually shrinking pores. After a while, a relatively homogeneous microstructure is obtained in which clay gels of low porosity connect expanded, but still dense, clay particles with each other, see Figure 10.5-1 /10.5-2/.

The heat generation also causes thermal stresses which break up the expanded clay particles and lead to a more homogeneous microstructure. When the water supply is unlimited and evenly distributed along the entire height of the deposition hole, the water saturation process is estimated to take on the order of 10 years. The subsequent homogenization process is estimated to take an equally long time.

10.5.3 Processes that affect buffer performance after water saturation

Bearing capacity

An important function of the buffer is to keep the canister in place. Calculations /10.5-3/ show that the canister does not subside more than a centimetre or so during one million years at a buffer density of about 2 t/m^3.

Thermal stability

On heating of the buffer, two counteracting physical processes take place:

- A partial dehydration, which entails a reduction in the number of interlamellar hydrate layers. This leads to an increase in the pore space between the lamellae.
- A thermally induced disintegration leading to a microstructural homogenization.

If the temperature is below 130°C and the density is higher than about 1.8 t/m^3 in the water-saturated state, the physical properties of the buffer are largely retained /10.5-4/.

Temperature is also of importance for conversion of smectite to illite. This requires availability of potassium, however, see below under the account of conversion to hydrated mica.

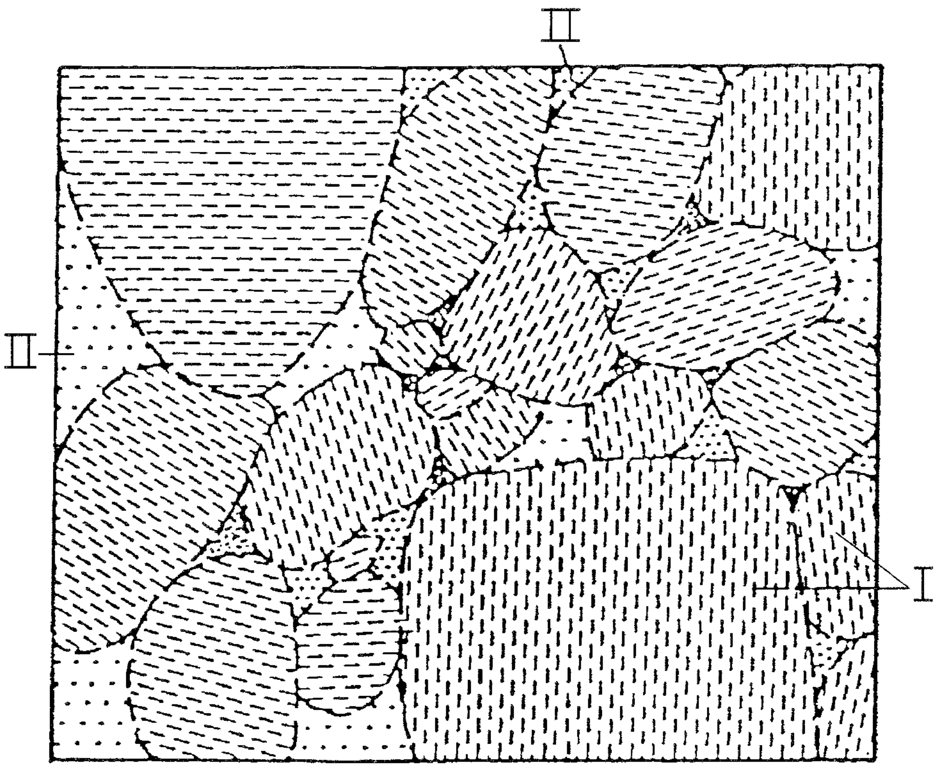


Figure 10.5-1. *Schematic illustration of the microstructure of homogenized, compacted bentonite. I) Dense, expanded clay particles, II) Clay gels formed in pores between clay particles.*

Influence of radiation

The buffer can be exposed to radiation from an intact canister or from released radionuclides that migrate from a defective canister. In the first case, only γ-radiation is involved, in the second case also α-radiation.

In experiments, bentonite has been heated to 130°C and exposed to γ-radiation doses that simulate the entire radiation dose to which the bentonite is exposed during the decay of the fuel /10.5-5/. Comparisons between the irradiated specimen and a specimen that has only been heated showed that the mineralogical changes that occurred were the same for both specimens. The microstructure of the bentonite was also as expected in both specimens, and the cation exchange capacity remained unchanged. Investigation of the physical properties of the specimens confirmed the uniform change that had been noted in mineralogical and chemical composition. The conclusion is that the γ-radiation from intact canisters to which the buffer is exposed in a deep repository does not affect its performance.

If water enters a canister and dissolves the fuel, the bentonite may also be exposed to α-radiation during transport of dissolved nuclides through the buffer. Positively charged ions move in the bentonite by cation exchange, which means that radionuclides are sorbed and expose the crystal lattice of the bentonite to radiation. Experiments with montmorillonite saturated with aqueous solutions containing Pu and Am led to a conversion of the montmorillonite to non-swelling mineral. For the scope of this process, see /10.5-6/.

Conversion of smectite to hydrated mica

If potassium ions are present, a structural conversion of smectite takes place to non-expandable hydrated mica, illite /10.5-7/. Model calculations have been performed of the kinetics of the conversion of smectite to hydrated mica. With the limited availability of potassium present in the repository, the model shows that the conversion in the buffer should not pose any problem on a 100,000-year time scale if the temperature is below about 130°C /10.5-8/. Geological examples in Sweden, Norway, Italy and Japan and laboratory experiments confirm the validity of the model.

Conversion of Na-smectite to Ca-smectite

A cation exchange is expected to take place in montmorillonite from Na to Ca, since it has a higher affinity for Ca than for Na. This is expected to lead to a change of the structure with three hydrate layers. According to a recently developed chemical model /10.5-9/, approximately 20% of the absorption sites in dense bentonite are occupied by Ca in groundwater of Äspö quality, and that percentage increases very slowly. After approximately 150,000 years, the percentage of Ca is estimated to be around 40%. The rate of exchange slows down after that and Na can be expected to constitute the dominant cation for a very long time.

Hydraulic conductivity

The hydraulic conductivity of Na-bentonite is illustrated in Figure 10.5-2, which shows the hydraulic conductivity of MX-80 Na-bentonite as a function

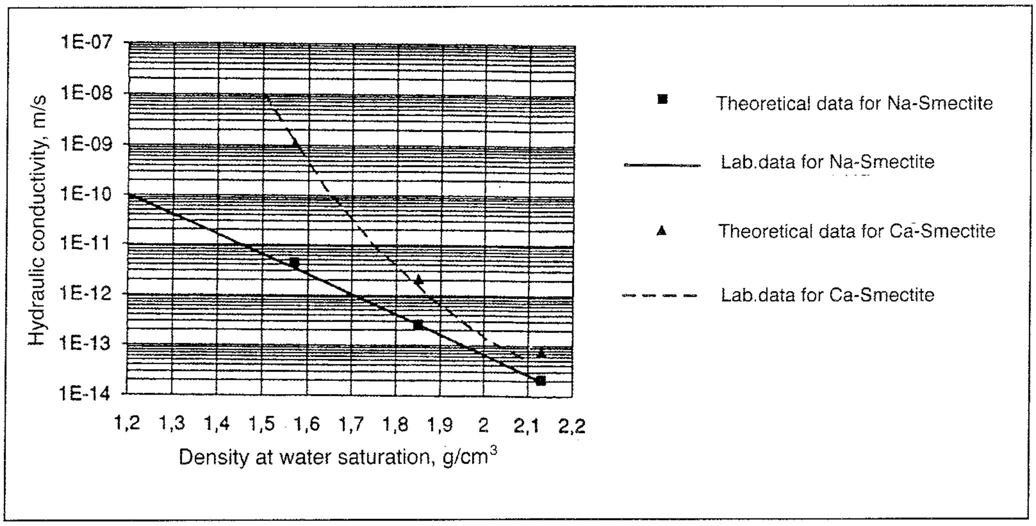


***Figure 10.5-2.** Hydraulic conductivity as a function of density after water saturation.*

of the density of the water-saturated bentonite. In comparison, the same relationship is shown for Moosburg Ca-bentonite. Within the range where the actual density of the bentonite in the deep repository is expected to lie, i.e. between 1.95 and 2.05 t/m^3, there is little difference between Na- and Ca-bentonite.

The conclusion of this relationship is that the hydraulic conductivity of the bentonite is not dependent on the rate of the process of cation exchange from Na to Ca.

Swelling pressure

The swelling pressure of the buffer is influenced greatly by the density of the buffer and the salinity of the groundwater during water saturation. The latter is particularly true at low buffer densities. If the density in the saturated state is over 1.6–1.7 t/m^3, the swelling pressure is only marginally affected by the salinity of the pore water /10.5-10/. With a buffer density of around 2.0 t/m^3 in the saturated state, the density is so high that the salinity of the groundwater does not pose any problem.

Gas transport

The mechanism for significant gas transport in water-saturated, homogenized bentonite is believed to be that one or more channels are opened through interconnected, gel-filled spaces between the denser bentonite particles /10.5-11/. Laboratory experiments with nitrogen gas and helium gas have shown that gas passage occurs at a given "critical" pressure. This pressure is dependent on the hydrostatic pressure and the bentonite's swelling pressure. On average, the "critical" pressure is equal to the hydrostatic pressure plus a pressure that is between 60 and 90% of the bentonite's swelling pressure. The gas transport capacity becomes as great as needed to allow entrapped gas to pass through the channels that are opened in the buffer.

The quantity of water that accompanies the gas is small and has been measured to be only 0.01–1% of the pore water volume, which means that the buffer material remains water-saturated.

Natural analogues

There are a large number of bentonite deposits all around the world. They exhibit variations as regards exposure to temperature and different cations. The bentonite grades that are being considered for the deep repository have all preserved their unique properties as regards swelling capacity and low hydraulic conductivity for millions to hundreds of millions of years. Other bentonite grades have undergone various processes that have led with time to an alteration that has varied with the conditions on the site. In Sweden, bentonites from Gotland and Västergötland have been studied. They all have an age of about 400 million years /10.5-12, 13/. The results of these studies have above all been utilized to verify the model that is used for conversion to hydrated mica.

10.5.4 Influence of the buffer on the chemical conditions in the repository

10.5.4.1 Influence of the buffer on groundwater chemistry

The chemical composition of the pore water in the buffer will be a result of equilibration between the groundwater and the buffer material. The copper canister has very low solubility and will not affect the chemistry of the water. Besides smectite, the buffer material also contains a certain amount of impurities, see section 5.4. The impurities that have the greatest influence on the pore water chemistry are $CaCO_3$, $CaSO_4$ and NaCl (pH) as well as pyrite, FeS_2 (Eh).

Oxygen consumption of impurities

Oxidation of pyrite is the dominant process for consumption of residual oxygen after closure of the repository. The time it takes for the repository to return to anoxic conditions has been estimated to be on the order of 10 to 100 years /10.5-14/.

pH

Readily soluble impurities such as gypsum/anhydrite ($CaSO_4$) and table salt (NaCl) can be of great importance for the pH of the pore water. A surface chemistry model for bentonite has been developed /10.5-9/. Sensitivity studies have then been made /10.5-15/ to assess the importance of different concentrations of these impurities. The calculations have been performed with a relatively fresh groundwater, but the result is expected to be the same for more saline water. The following general conclusions can be drawn from this study:

- The pH of the pore water in compacted bentonite lies between 6.8 and 9.3.
- In compacted bentonite, the pH is primarily buffered by the acid/base equilibrium in the clay mineral. This is in contrast to conditions with lower bentonite/water ratios, where the pH is often higher, since it is buffered by carbonate in the buffer.

- The quantity of readily soluble impurities therefore has a crucial effect on the pH in the pore water. The more impurities, the lower the pH.
- Relatively large variations in pH can be expected depending on the amount of ions for cation exchange. For example, an Na-bentonite with little impurities initially has a pH of around 9, while a pure Ca-bentonite is around 7.

The amount of impurities in the bentonite will decline with time as a result of washing-out to the groundwater. This will naturally affect the pH of the pore water. When all impurities are gone, the pore water will have a pH very close to that of the groundwater.

10.5.4.2 Effects of micro-organisms, organic matter and colloids

Importance of bacteria for oxygen consumption

Bacteria can participate in reactions in which oxygen entrapped in the repository is consumed, i.e. reactions with pyrite and ferrominerals in bentonite, backfill and rock. This is desirable, since oxygen can also lead to corrosion of the canister's copper shell. Oxygen-consuming oxidation reactions also take place spontaneously without the presence of bacteria.

Importance of bacteria for oxygen corrosion of copper

There are reports in the literature of bacteria forming a film on the inside of copper water supply pipes and contributing to the corrosion of the copper pipes. However, the process requires the presence of oxygen as an oxidant, and furthermore the reaction can take place spontaneously without bacteria. The influence of bacteria can therefore be disregarded in assessments of copper corrosion.

Importance of bacteria for sulphide corrosion of copper

Sulphide corrosion takes place without bacteria. All that is needed is an aqueous solution with sulphide. Bacteria can contribute by reducing sulphate to sulphide. This is a reaction that would otherwise not take place in the repository, since it is strongly kinetically inhibited. Since there is considerably more sulphate than sulphide in bentonite and groundwater, this is a process that cannot be ignored.

However, sulphate-reducing bacteria also need a nutrient that can act as a reducing agent. Normally it is a question of organic compounds. The concentration of organic compounds in the groundwater is low, up to about 2 mg/l TOC. Furthermore, it is difficult for bacteria to degrade /10.5-16/ the organic matter in bentonite. Even with the conservative assumption that all the organic matter in the bentonite is available to be used for sulphate reduction by bacteria, the scope of reduction is still small due to the fact that the quantity of organic matter is limited.

There are reports that bacteria can also use hydrogen gas to bring about reduction of sulphate. This is deemed to be of subordinate importance, however. Sulphate-reducing bacteria, hydrogen gas transport and survival of bacteria in compacted bentonite buffer are dealt with in /10.5-17/.

Transformation to colloids

Clay particles can occur as colloids, but sodium bentonite forms a stable gel in contact with the groundwater. In order for clay particles to be released as colloids from the gel, the water must be free of dissolved cations, such as sodium and calcium /10.5-18/. Calcium cations are better than sodium cations when it comes to stabilizing the gel, since they have twice as high a charge. Groundwater generally has more than 20 mg/l of calcium cations and even more sodium cations, which stabilize the clay. The clay also contains its own calcium and sodium cations, which means that considerable leaching-out is required before the surroundings are depleted of cations. This is even true under the assumption that extremely cation-poor water reaches the buffer.

10.5.5 Mass transport properties of the buffer material

Diffusion in clay

Water-saturated compacted clay has a very low hydraulic conductivity, see section 10.5.3. Radionuclides and other substances dissolved in water can therefore only be transported by diffusion in the bentonite buffer. The water-filled pore spaces in compacted bentonite are very constricted. This means that solutes with a high molecular weight, in particular colloidal particles, are very limited in their mobility /10.5-19/. Diffusivities, D_e, of between $3 \cdot 10^{-8}$ and $6 \cdot 10^{-8}$ m^2/s have, for example, been measured for organic anions with molecular weights in the range 290–30,000 amu /10.5-20/.

If water comes into contact with spent fuel, some radionuclides go into solution. This solution also contains particles. Dissolved radionuclides in the form of colloidal particles will be filtered out by the bentonite. This is an important property of the buffer. If actinides such as plutonium and americium could be released as colloids, their thermodynamic solubility would no longer be limiting for the release of these substances. Besides the fact that size and charge are an obstacle to the transport of these particles, they have a strong tendency to sorption on solid materials at the pH values between 7 and 9 that exist in the bentonite's pore water /10.5-21/.

Sorption in clay

Dissolved substances in ionic form, for example, can diffuse through bentonite. However, many radionuclides are sorbed very strongly on the surface of the clay particles. This means that even some fairly long-lived substances have time to decay to insignificant levels while still in the buffer. The time from canister penetration up to penetration of the buffer barrier (if ever) is described as the transient state. Due to the fact that diffusion in general is so extremely slow, most laboratory measurements pertain to this state. In this way, the parameters that describe the steady state are also covered, while the transient state can be said to be the best described state. In the case of a few substances that move relatively fast such as Cs, Sr and I, direct measurements of sorption in the steady state are made nowadays.

Diffusion and sorption data

Table 10.5-1 presents the diffusion constants and K_d values that are used to calculate transport of radionuclides in bentonite. The same constants are used in SKB 91 /10.5-22/. However, new, not-yet-published measurements of Sr and Cs indicate that the diffusivity D_e is more than 50 times lower than is given in the table /10.5-23/.

Table 10.5-1. Element-specific diffusion and distribution coefficients in bentonite.

Element	D_e m^2/y	K_d m^3/kg
C	$3.2 \cdot 10^{-3}$	0
Cl	$7.9 \cdot 10^{-5}$	0
Ni	$3.2 \cdot 10^{-3}$	0.5
Se	$3.2 \cdot 10^{-3}$	0.003
Sr	$7.9 \cdot 10^{-1}$	0.01
Zr	$3.2 \cdot 10^{-3}$	2
Nb	$3.2 \cdot 10^{-3}$	0.2
Tc ox	$7.9 \cdot 10^{-5}$	0
Tc red	$3.2 \cdot 10^{-3}$	0,1
Pd	$3.2 \cdot 10^{-3}$	0.01
Sn	$3.2 \cdot 10^{-3}$	3
I	$7.9 \cdot 10^{-5}$	0
Cs	$7.9 \cdot 10^{-1}$	0.05
Sm	$3.2 \cdot 10^{-3}$	1
Np	$3.2 \cdot 10^{-3}$	3
Pu	$3.2 \cdot 10^{-3}$	50
Am	$3.2 \cdot 10^{-3}$	3

10.6 DISSOLUTION PROCESSES IN THE CANISTER

10.6.1 Introduction

On penetration of the disposal canister's copper shell, the inside of the canister will come into contact with water. The water will interact with the inner steel canister, subsequently reaching metal parts in the fuel assemblies and the fuel itself. This chapter discusses the dissolution of the fuel and the release of radionuclides to which this leads.

To describe fuel dissolution, the chemical environment at penetration of the copper shell is first described, in particular the composition of the water in and around the canister. Dissolution/corrosion of the fuel and of metal parts in the fuel assemblies is then described in two sections.

10.6.2 Hydrochemical environment

The hydrochemical environment in a canister where water has penetrated is determined by the natural composition of the groundwater, and by the influence of water from other repository sections, the inner steel canister and the fuel. The natural composition of the groundwater is described in section 6.3.

Influence of other repository sections

The groundwater at the canister has passed through the bentonite. The water composition has then been affected, mainly by an increase of carbonate and sulphate concentrations and by changes in pH. These effects are dealt with in section 10.5. It is probable that impurities from the bentonite will be of crucial importance for the near-field environment if a canister failure occurs within 100,000 years from the closure of the repository. In the case of later canister penetration, the natural composition of the water will assume greater importance. Other repository sections will also affect the composition of the water, but this is deemed to be of minor importance.

Influence of the inner steel canister

In the event of a penetrating defect in the copper canister, the groundwater will come into contact with the inner steel canister. The interaction of the water with iron in the steel leads to corrosion and influence on the redox potential. In a canister with a defective copper shell, water will be in contact with metallic iron. The reaction between iron and water

$$3Fe + 4H_2O \rightarrow Fe_3O_4 + H_2(g)$$

reaches equilibrium when the partial pressure of hydrogen is ~1000 atm. This means that equilibrium will not be reached inside the canister as long as metallic iron is present. The corrosion rate for iron under anaerobic conditions is estimated to lie in the range 0.1–1 μm/y, which means that metallic iron can remain for tens of thousands of years.

The transformation of Fe(0) to Fe(II) in the above reaction will drive the redox potential inside the canister to very low values.

Influence of the fuel

The fuel influences the environment above all through alpha-induced radiolysis of water, which in turn affects the redox potential. Radiolysis mainly gives rise to H_2, O_2 and H_2O_2. H_2 is the most inert of these, which means that H_2 can be transported out of the canister without reaction. The net effect of the radiolysis is thereby an increased redox potential. This effect is considerably less than the corrosion effect, but can still be of importance since it occurs on the surface of the fuel. For thermodynamic reasons, the Zircaloy cladding tubes could also be of importance for the redox conditions, but it is believed that the thin layer of zirconium oxide on the metal will effectively prevent all reactions.

Summary

Impurities from the bentonite will probably have a considerable effect on the environment for at least 100,000 years after closure. The conditions will in any case be strongly reducing, partly because the undisturbed, natural environment in the near field is reducing, and partly because iron corrosion contributes further to this. The pH will lie in the neutral to weakly alkaline range. The ionic strength will always lie in an interval that is favourable for the performance of the repository.

10.6.3 Dissolution of fuel

The spent fuel acts as an engineered barrier to the dispersal of radionuclides in the repository by virtue of its low solubility in water and low corrosion rate. On penetration of the disposal canister, escape of radionuclides from the repository will be limited by, among other things, the slow rate of dissolution of the fuel.

10.6.3.1 Mechanisms for release

The release of radionuclides from spent fuel in contact with water is the result of two principal mechanisms:

- Release of segregated radionuclides, i.e. nuclides from the fuel-clad gap in the fuel rods, nuclides in grain boundaries and nuclides in segregated phases
- Release of radionuclides due to dissolution or transformation of the uranium dioxide matrix

These mechanisms are in turn dependent on numerous different factors such as burnup, water composition, local redox conditions, temperature etc. Normally, however, redox conditions are of the greatest importance.

Release of segregated nuclides

Approximately 0.5% of the total fuel mass consists of substances that are soluble in the fuel matrix of uranium dioxide /10.6-1/. It is metallic fission products such as Tc, Mo, Pd and Ru which normally form **microscopic metallic inclusions in the grain boundaries of the fuel.** Other insoluble fission products are Cs, I and noble gases, which are found as **bubbles in the grain boundaries.** A fraction of these elements, all of which are volatile during reactor operation, have been released from the matrix and are present in the **fuel-clad gap or in cracks in the fuel.** This fraction is released in a short time if the fuel comes into contact with water and will therefore dominate the radioactive release following a canister failure. The fraction of liberated Cs and I is comparable to the release of fission gas during reactor operation. A typical value for immediate Cs release is ~1% of the caesium inventory in a canister /10.6-2/.

Release due to matrix dissolution/transformation

Fission products and actinides in the fuel matrix comprise the predominant portion of the fuel's radioactivity /10.6-1/. Nuclides that lie embedded in the fuel matrix are protected against direct dissolution in water. They can only be released if the matrix is dissolved or transformed. In an environment where UO_2 is thermodynamically stable, the release of radionuclides will be limited by the solubility of UO_2. Even under mildly oxidized conditions where UO_2 is oxidized to the composition $UO_{2.33}$, solubility will be limiting, since this oxidation does not lead to a break-up of the crystal lattice. If the oxidation goes further, on the other hand, the crystal lattice will be broken up. The release of radionuclides will then be determined by the transformation rate, even though the uranium concentration in solution may still be low.

10.6.3.2 Factors that influence matrix dissolution

The dissolution of the uranium dioxide matrix is affected by several factors. The most important are redox potential, groundwater composition, radiation, temperature and interaction with other engineered barriers /10.6-3/.

Redox potential

Under anoxic reducing conditions, uranium dioxide is stable and the dissolution of the fuel matrix is controlled by the solubility of uranium in groundwater. Under these conditions, the solubility of **pure** uranium dioxide is very low, on the order of 10^{-9} mol/dm^3. For fuel, typical concentrations under anoxic conditions are around $2 \cdot 10^{-7}$ mol/dm^3. If oxygen is present, the solubility of uranium increases by several orders of magnitude. The reason for this is that uranium dioxide is then no longer stable, but is oxidized to U(VI), which dissolves as uranyl complex.

The redox conditions are determined by the availability of oxidants in the groundwater, by radiolytic disintegration of the groundwater and by the availability of other redox-sensitive species in the near field.

Groundwater composition

Under oxidizing conditions, the concentration of dissolved UO_2^{2+} is determined by hydrolysis and complexation. These factors also influence the dissolution kinetics. This means that pH and complexing anions such as carbonate and phosphate are of great importance.

Radiation

The most important effect of radiation is the formation of oxidants by radiolytic disintegration of the groundwater. Radiolysis produces equal quantities of oxidizing and reducing species, so the total redox situation is unaffected. However, the relatively higher reactivity of the oxidizing radiolysis products can lead to net oxidation of the fuel, leading to an increased dissolution rate.

Temperature

Depending on when canister penetration takes place, the temperature in the near field may be between 80 and 15°C. In this range, the influence of the temperature on the dissolution rate is of subordinate importance.

Interaction with other engineered barriers

Metallic iron and copper, corrosion products of these metals and bentonite are present in the near field. These materials can influence the dissolution of the fuel by controlling the redox conditions in the near field. They can also retard releases from the near field by sorption of radionuclides. Sorption might also lead to increased dissolution, since it may then take a longer time to reach equilibrium concentrations.

10.6.3.3 Used model for fuel dissolution

A model for fuel dissolution must have a term that describes the release of nuclides due to matrix dissolution and a term for release from the fuel-clad gap and from cracks in the fuel. It is also desirable to describe the dissolution of any radionuclide content in grain boundaries.

The release of nuclides in gaps can exhibit considerable variation, depending on the irradiation history of the fuel. It takes place rapidly and can be regarded as instantaneous in comparison with other processes. After this phase, a period with release from grain boundaries may dominate before the slow matrix dissolution takes over as the rate-limiting process.

The structure of the fuel dissolution models that are currently used in SKB's safety assessments is described below.

Gap and crack inventory

The modelling of this phase is relatively straightforward, providing operating data on fission gas release are available for the fuel. Typical values lie around 1% of the total inventory for Cs and I, but can in extreme cases be slightly higher. In SKB's model, it is assumed that this inventory is released immediately when water comes into contact with the fuel.

Grain boundary inventory

The modelling of this phase is considerably more difficult. It is very uncertain whether there are any nuclides at all in the grain boundaries, and if so how much and how rapidly they are released. Since the uncertainties are so great, a conservative assumption is made in the model that the grain boundary inventory is released in the same way as the gap and crack inventory, i.e. immediately upon water contact.

A special case of grain boundary inventory where reliable data are available is metallic inclusions of Mo, Tc, Ru, Rh and Pd, which are preferentially present in the grain boundaries. This inventory is modelled as solubility-limited with the individual metals as solubility-limiting phases.

Matrix transformation

In deep natural groundwaters and in groundwater in contact with the repository's engineered barrier, pH and redox conditions are always such that UO_2 is thermodynamically stable. This means that it should be possible to model the fuel as solubility-limited with UO_2 as the solubility-limiting phase. The radiation from the fuel can, however, disintegrate water into oxidizing and reducing species. Of these, the oxidizing species are more reactive and could give rise to local oxidizing conditions near the fuel surface. If this oxidation is sufficiently powerful to oxidize UO_2 to a composition higher than $UO_{2.33}$, the crystal lattice will be broken up and a solubility-limited model can no longer be applied. This means that there are two alternative models for how the matrix transformation is to be described.

An oxidation model was used in SKB 91 /10.6-4/ where it was assumed that the transformation rate was proportional to the α-dose rate in the fuel. The proportionality constant was taken from measured strontium release from fuel leaching tests under oxidizing conditions. This gives a very conservative value of the dissolution rate. If a value for strontium release from tests under anaerobic (more realistic) conditions were to be used instead, the longevity of the fuel would increase by many orders of magnitude.

In the alternative model, the oxidation of the UO_2 matrix never goes further than to $UO_{2.33}$ and the release of radionuclides is proportional to the product of matrix solubility and water flux in the canister.

In experiments under anaerobic conditions, where the radiolysis products O_2, H_2 and H_2O_2 have been measured during leaching, it has been found that the oxygen is consumed in the system without this leading to equivalent quantities of U(VI) being detected. This suggests that the oxidation with radiolysis products has not proceeded as far as to $UO_{2.33}$ on the surface of the fuel, which means that the alternative model would be applicable. It is not yet fully understood how the oxidation proceeds. The background data for the model are therefore incomplete, and further experimental information is needed before it can be formulated and applied. The model in SKB 91 will therefore be used, even though the experimental data that exist show that it gives a very conservative value of the dissolution rate. The model probably doesn't describe the oxidation process correctly either.

10.6.4 Release of nuclides from metal parts

Besides the fuel itself, the fuel assemblies in a light-water reactor consist of a number of structural parts of different materials. Like the fuel itself, these parts are exposed to the neutron flux that keeps the chain reaction going during reactor operation. This means that neutrons are captured in the atomic nuclei of the structural materials, which can lead to the formation of radionuclides.

The primary structural materials are the stainless steels SIS 2331, 2333 and 2352 and AISI 304, and the alloys Zircaloy-2 and -4 and Inconel X-750 and X-718. The most important radionuclide in the stainless steels is Co-60, while the most important radionuclides in Inconel are Ni-59 and Ni-63 and in Zircaloy Zr-93 and C-14.

The activity is relatively evenly distributed in all of these materials. The release can be modelled as congruent with the corrosion of the metal in question. The corrosion rate is judged to be very low for all materials under repository conditions, but the body of data is insufficient. Steel and Inconel are therefore modelled conservatively with immediate release of the entire inventory. General corrosion of Zircaloy is judged to be such a slow process that the inventory of C-14 there is not released at all. Zr-93 is very long-lived, and the possibility can therefore not be ruled out that it will be released from the Zircaloy on a longer time scale. However, the weathering product ZrO_2 has very low solubility and will be formed from Zr-93 together with stable zirconium. This has led to the judgement that no Zr-93 will be released from the Zircaloy.

10.7 SOLUBILITY OF THE RADIONUCLIDES

10.7.1 Introduction

Owing to the very low water flux inside a canister, many of the radioelements will precipitate as secondary minerals if they are released from the fuel matrix. Solubility limitations are a very important barrier function, since the release of radionuclides from the near field is often directly proportional to solubility. It is likely that the radionuclides would precipitate as mixed solid phases, which results in very low solubilities for trace substances with low inventories. Since knowledge concerning co-precipitation processes is limited, it is assumed that only pure phases precipitate. This means that the calculated solubilities are always conservative.

Another factor that is of great importance for solubility is the degree of crystallinity. A purely crystalline phase has much lower solubility than an amorphous one, particularly in the case of actinides. Normally, amorphous phases are the first ones to precipitate, but they crystallize with time and the solubility of the substance declines. On the time scales that apply to a deep repository, it could be assumed that all secondary minerals achieve a high degree of crystallinity. This is difficult to prove conclusively, however, so normally only amorphous solubility-limiting phases are used.

Reducing conditions normally prevail inside the canister (see 10.6.2), although more oxidizing conditions can occur locally on the fuel surface, resulting in neutral to weakly alkaline pH values. Solubility calculations should therefore be done for low Eh values and pH values between 6 and 10. Solubilities for oxidizing conditions are also given in this section, however.

An element-by-element overview of anticipated speciation, solubility-limiting phases and comparisons with natural systems follows below /10.7-1/. The treatment is incomplete, however; no solubility calculations have been done for one thing. The overview is rather a presentation of the factual material that is available in the area.

10.7.2 Fission and activation products

Carbon

C-14 is an activation product from impurities of both nitrogen and oxygen and is present in roughly equal amounts in the fuel and in the Zircaloy. Carbon should normally occur as carbonate under repository conditions. This could theoretically mean that calcite or some other carbonate mineral would be a solubility-limiting phase. In the near field, however, this is not of very great importance, since the carbonate concentrations in solution can be high. Carbon could also occur as methane, which has very high solubility. In the assessment it is assumed that carbon lacks solubility limitation under all conditions.

Chlorine

Cl-36 is an activation product from chlorine impurities in the fuel. Chlorine will occur as chloride ions in solution. It lacks solubility limitation under all conditions.

Nickel

Ni-59 and Ni-63 are activation products of nickel and occur in structural parts of stainless steel and Inconel. Nickel is bivalent in aqueous solution and forms strong carbonate complexes. Nickel silicates, e.g. $NiSiO_3$, have very low solubilities, but they cannot be formed at the low temperatures that exist in a repository. Nickel can also form relatively poorly soluble sulphide minerals, either together with iron and copper or as pure phases. It is, however, uncertain to rely on sulphide being available in the repository. The solubility-limiting phase that can be used in the safety assessment will be NiO, $Ni(OH)_2$ or $NiCO_3$. These phases all have relatively high solubilities in the neutral pH range and the solubility of nickel is therefore assumed to be of no importance.

Selenium

Se-79 is a fission product. Under reducing conditions, selenium occurs as selenide in aqueous solution, H_2Se or HSe^-. The solubility-limiting phase under repository conditions will be ferroselite, $FeSe_2$, or some other metal selenide. Ferroselite occurs in roll-front uranium deposits, which suggests that it can precipitate in contact with bivalent iron. The solubility of ferroselite and other metal selenides is very low. An uncertainty in this solubility assumption is the kinematics of the reduction reaction. If selenium is present in the fuel with the oxidation numbers IV or VI, it is possible that the reduction to -II goes very slowly in the absence of microbial activity. Selenium is probably already present in reduced form in the fuel, however.

Under oxidizing conditions, selenium occurs with the oxidation numbers IV and VI. In aqueous solution, it occurs as $HSeO_3^-$, SeO_3^{2-} and SeO_4^{2-}. The solubility-limiting phases here will be some selenite or selenate minerals, e.g. $MgSeO_3$, but the solubility of these phases is too high to be of any practical importance.

Krypton

Kr-85 is a fission product. Krypton is an inert gas and forms no solid phases.

Strontium

Sr-90 is a fission product and one of the more dominant radionuclides on closure of the repository. Its half-life is only 28.5 years, however, which means that its activity declines rapidly. Strontium only occurs with the oxidation number II. In aqueous solution Sr^{2+} dominates, but a small fraction of carbonate complex can occur at high carbonate levels. Solubility-limiting phases could be strontianite and celestite. The solubilities of these minerals are relatively high, however, and this, in combination with a high specific activity, means that the solubility limitation lacks importance.

Zirconium

Zr-93 occurs preferentially as a fission product, but also to a relatively high degree as an activation product in the Zircaloy cladding. In nature, zirconium always occurs with the oxidation number IV. In aqueous solution $Zr(OH)_5^-$ dominates. The most common Zr minerals are zircon, $ZrSiO_4$, and baddeleyite, ZrO_2. A film of ZrO_2 forms very rapidly if metallic Zr comes into contact with water. This phase is also used as solubility-limiting in the assessment. The chemistry of zirconium is independent of the oxidation conditions.

Niobium

Nb-94 is an activation product of niobium in the structural parts of Inconel. PWR assemblies contain more than ten times as much Nb-94 per tonne of uranium as BWR assemblies. Niobium almost always occurs with the oxidation number V in natural systems. In aqueous solution $Nb(OH)_5$ dominates. In nature, niobium occurs preferentially as oxides. Niobite, $(Fe, Mn)(Nb, Ta)_2O_6$, occurs as a continuous solid solution with tantalite. Nb_2O_5 is chosen as the solubility-limiting phase in the safety assessment.

Technetium

Tc-99 is a fission product. All technetium isotopes are unstable, so the nuclide does not occur in nature. Technetium has the oxidation number VII under oxidizing conditions and normally IV under reducing ones. The dominant speciation in aqueous solution is TcO_4^- and $TcO(OH)_2$. A large fraction of the technetium in the fuel occurs as metal in separate phases together with Mo-Ru-Rh-Pd. Two different solid phases are used in the assessment: Tc(s) if it is thermodynamically stable and otherwise TcO_2(am). Technetium has very high solubility under oxidizing conditions.

Palladium

Pd-107 is a fission product. Palladium usually occurs with the oxidation number II in nature, but elemental palladium is not unusual. Palladium forms weak complexes with "hard" ligands such as carbonate, sulphate and phosphate. The dominant species, under relevant conditions, is therefore $Pd(OH)_2$. Like technetium, palladium occurs in metallic inclusions in the fuel. Pd(s) has a very large stability range and is therefore chosen as the solubility-limiting phase, but the solubility of PdO is used as a conservative variation at high redox potentials.

Silver

Ag-108m is an activation product of impurities of silver in the fuel. The inventory of silver in the fuel is very low. This, in combination with the fact that

Ag-108m has a half-life of 127 years, which gives a high specific activity, means that a solubility limitation for silver is of no importance in the safety assessment. A reasonable choice of solubility-limiting phase would be Ag(s), since it is likely that silver occurs in metallic form together with Mo-Tc-Ru-Rh-Pd.

Tin

Sn-126 is a fission product. Tin occurs with the oxidation numbers II and IV in nature, depending on the redox conditions. Under repository conditions, Sn(IV) will dominate with $Sn(OH)_4$ in solution and SnO_2(am) as the solubility-limiting phase. This applies regardless of the redox conditions.

Iodine

I-129 is a fission product. Under all conceivable repository conditions, -I is the stable oxidation state for iodine. However, an oxidation to I_2 or IO_3^- is theoretically possible. Iodine can form poorly soluble solid phases with silver, but also with copper and lead. However, it is difficult to demonstrate conclusively that these phases will form, so in the assessment it is assumed that iodine does not have any solubility limitation.

Caesium

Cs-135 and Cs-137 are two important fission products. Disintegration of Cs-137 dominates the activity in the repository at closure. Caesium is an alkali metal and will under all conditions have the oxidation number I. In aqueous solution, Cs^+ dominates. Caesium has high solubility, and in the assessment it will not be assumed to be solubility-limited under any conditions.

Samarium

Sm-151 is a fission product. Samarium normally occurs in nature with the oxidation number III, but can also be found in bivalent form. In aqueous solution carbonate complexation dominates, but Sm^{3+} is also of importance. Solid phosphate phases have high solubilities, but they do no form on precipitation at low temperatures. $Sm(OH)CO_3$(s) has a wide stability range and is used as the solubility-limiting phase in the analysis. At high carbonate concentrations, however, $Sm_2(CO_3)_2$(s) must be taken into account. This applies regardless of the redox conditions.

Holmium

Hm-166m is formed both by neutron activation of dysprosium impurities in the fuel and fission (activation of other fission products). The low inventory of Hm-166m makes it one of the least important radionuclides. Holmium occurs with the oxidation number III in nature. In contrast to samarium, holmium belongs to the heavy lanthanides. These elements have stronger carbonate complexes and it is less clear which solid phases, with considerably higher solubilities, are formed. In the assessment, $Ho_2(CO)_3$ is assumed to be the solubility-limiting phase. This applies regardless of the redox conditions.

10.7.3 Heavy nuclides

Solubility limitations are only interesting for those nuclides that have relatively long half-lives and are thereby present in relatively large quantities. The waste also contains isotopes of the heavy nuclides with such short half-lives that they do not contribute to chemical transport from repository to biosphere. Due to ingrowth, however, they will be of importance in the biosphere and are therefore dealt with in the biosphere sections.

Radium

Ra-226 is part of the decay chain of U-238. Radium always has the oxidation number II. Radium does not form any strong complexes and occurs as Ra^{2+} in aqueous solution. The radium phase with the lowest solubility is $RaSO^4$. Despite its low solubility, it scarcely contributes to nuclide retention due to the high specific activity of Ra-226. This phase is nevertheless conservatively chosen as solubility-limiting in the assessment. Radium can be a dominant nuclide in certain scenarios, and sensitivity studies should therefore be done to evaluate the importance of the well-documented co-precipitation in solid solutions with barium and strontium.

Thorium

The thorium isotopes Th-229, Th-230 and Th-232 occur as daughters in the decay chains of the actinides. Thorium occurs solely with the oxidation number IV. Thorium is a strong complexing agent. In nature hydroxide complexes are dominant, but phosphate, sulphate and fluoride complexes can also be important. This is also applied in the safety assessment. ThO_2 is a reasonable choice as the solid phase.

Protactinium

Pa-231 is a daughter in the decay chain of U-235. In water relevant to repository conditions, V is the only stable oxidation number for protactinium. In aqueous solution it occurs as PaO_2^+. Pa_2O_5 with low crystallinity is used as the solubility-limiting phase.

Uranium

The fuel consists for the most part of U-238. Some U-235 is also left. U-233, U-234 and U-236 occur as daughters in the actinide chains. Uranium can assume all oxidation numbers between II and VI, but under repository conditions only IV and VI are of importance. At reducing conditions, U(IV) dominates. It primarily forms hydroxide complex, $U(OH)_4$, in the aqueous phase and the fuel itself is the solubility-limiting phase. As mentioned previously, it is possible that more oxidizing conditions will exist on the surface of the fuel. Then U(VI) would be formed and would dominate in solution. Its solubility is higher than that of U(IV), especially in the presence of carbonate, with which it forms strong complexes.

At the moderately oxidizing conditions that are expected in the repository, some uranium oxide with a lower oxygen content than $UO_{2.33}$ will be solubility-limiting. If the oxidation proceeds further, $UO_{2.66}$ may be limiting. Under strongly oxidizing conditions, which are unreasonable in the repository but occur in nature, a weathering process of the uranium begins. U(VI) oxides are formed first in this process, but they are then transformed to U(VI) silicates with low solubility. Phosphate may be of importance for the chemistry of uranium if the $[PO_4^{3-}]/[CO_3^{2-}]$ ratio is greater than 0.1. This is taken into account in the assessment.

Neptunium

The only neptunium isotope of interest is Np-237. IV and V are the relevant oxidation numbers for neptunium under repository conditions. Under reducing conditions, neptunium occurs in the tetravalent state and forms, like uranium, hydroxide complex in the aqueous phase. In the assessment, $Np(OH)_4(am)$ is assumed to be the solubility-limiting phase. This phase crystallizes with time to NpO_2 and its solubility declines by several orders of magnitude. This process is difficult to model and is therefore neglected, which means that the solubility of neptunium is overestimated. Under moderately oxidizing conditions, Np(V) is formed and NpO_2^+ becomes the dominant species in the aqueous phase, but solubility is still determined by $Np(OH)_4(am)$. Under strongly oxidizing conditions, the most stable solid phases are $NaNpO_2CO_3$ and NpO_2OH.

Plutonium

Pu-239 and Pu-240 are two of the most important radionuclides due to their relatively large inventory, but Pu-238, Pu-241 and Pu-242 must also be taken into account in the assessment. Plutonium can have any oxidation number between III and VI under repository conditions, with some reservation for VI. As for uranium and neptunium, Pu(IV) is the expected state, with $Pu(OH)_4$ in solution and $Pu(OH)_4(am)$ as the solubility-limiting phase. $Pu(OH)_4(am)$ is solubility-limiting at a very wide range of redox potentials. In solution, Pu^{3+} can dominate at low potentials and PuO_2^+ at high ones. The line or reasoning regarding the crystallinity of the solid phase for neptunium also applies to plutonium.

Americium

Am-241 dominates the α-activity in the repository for the first thousand years. Am-243 is also a very important nuclide. Am-242m is also taken into account in the assessment, but its inventory is very small. For americium, III is the only oxidation number that has to be taken into account. Americium principally forms the carbonate complex $AmCO_3^+$ in solution, but Am^{3+} also occurs. The normal solubility-limiting phase is $Am(OH)CO_3$, but $Am_2(CO_3)_2$ can be formed at high carbonate concentrations and $Am(OH)_3$ at low ones.

Curium

The curium isotopes Cm-242, Cm-243, Cm-244 and Cm-246 are taken into account in the assessment, but none of them is of any great importance, since their inventories are relatively small.

What was said about the chemistry of americium also applies to curium.

10.7.4 Calculations of radionuclide solubility

Tools

Calculations of radionuclide solubility are carried out using thermodynamic equilibrium programs, such as EQ3/6 or PHREEQE. These programs calculate equilibrium solubilities and speciation for a given system. The programs have been verified against each other and also validated against laboratory experiments /10.7-2/. The greatest uncertainty when an equilibrium program is used is the understanding of the systems that are being modelled: Does equilibrium prevail? What solid phases occur? Are the kinetics of importance? There are also large uncertainties in the thermodynamic data for certain elements.

Calculation procedure

Since precipitation of secondary phases is a very important retardation mechanism for radionuclides and the hydrochemical conditions inside the canister are so uncertain, a large number of solubility calculations would have to be done in order to cover all conceivable cases. For most radioelements, however, only a few components in the water are of importance for solubility. pH is of some importance for many, redox potential and carbonate concentration for a few, while all other components can be regarded as being without importance. But this must be proven by calculations. This is not done within the framework of SR 95, however. The assessment is intended as an illustration, and the same values of the element solubilities as in SKB 91 /10.7-3/ are used. An extensive solubility study will be carried out in preparation for the next safety assessment. Table 10.7-1 contains the values that are used in SR 95. For the sake of comparison, it also gives the values used in Kristallin-I /10.7-4/ and the concentrations in seawater. The differences between SKB 91 (SR 95) and Kristallin-I can for the most part be explained by the degree of conservatism in the assumptions.

Table 10.7-1. Element solubilities in mol/l for the reference cases in SR-95 (SKB 91) and Kristallin-I, and concentrations of the elements in seawater. Elements in boldface are those treated in this assessment.

Element	SR-95 (SKB 91)	Kristallin-I	Seawater
Nickel		High	$2 \cdot 10^{-8}$
Selenium	$\mathbf{10^{-20}}$	$\mathbf{10^{-8}}$	$\mathbf{8 \cdot 10^{-8}}$
Zirconium	$2 \cdot 10^{-11}$	$5 \cdot 10^{-9}$	$1\text{-}30 \cdot 10^{-13}$
Niobium	$1 \cdot 10^{-5}$	10^{-3}	10^{-13}
Technetium	$\mathbf{2 \cdot 10^{-8}}$	$\mathbf{10^{-7}}$	**–**
Palladium	$\mathbf{3 \cdot 10^{-7}}$	$\mathbf{<10^{-11}}$	$\mathbf{\sim 10^{-14}}$
Silver	–	–	$1 \cdot 10^{-11}$
Tin	$3 \cdot 10^{-8}$	10^{-5}	$7 \cdot 10^{-11}$
Samarium	$1 \cdot 10^{-5}$	10^{-5}	$1 \cdot 10^{-14}$
Holmium	–	–	$2 \cdot 10^{-13}$
Radium	$\mathbf{1 \cdot 10^{-6}}$	$\mathbf{10^{-10}}$	$\mathbf{1 \cdot 10^{-13}}$
Thorium	$\mathbf{2 \cdot 10^{-10}}$	$\mathbf{5 \cdot 10^{-9}}$	$\mathbf{2 \cdot 10^{-12}}$
Protaktinium	$\mathbf{3 \cdot 10^{-7}}$	$\mathbf{10^{-10}}$	**–**
Uranium	$\mathbf{2 \cdot 10^{-7}}$	$\mathbf{10^{-7}}$	$\mathbf{1.4 \cdot 10^{-8}}$
Neptunium	$\mathbf{2 \cdot 10^{-9}}$	$\mathbf{10^{-10}}$	–
Plutonium	$\mathbf{2 \cdot 10^{-8}}$	$\mathbf{10^{-8}}$	–
Americium	$\mathbf{2 \cdot 10^{-8}}$	$\mathbf{10^{-5}}$	–
Curium	$\mathbf{2 \cdot 10^{-8}}$	$\mathbf{6 \cdot 10^{-8}}$	–

Carbon, Chlorine, Krypton, Strontium, Iodine and **Caesium** have solubilities that are so high that their solubility limitations are of no importance in the assessment.

11 MODELS AND CALCULATION METHODS FOR RADIONUCLIDE TRANSPORT

This chapter deals with the performance of the repository parts when it comes to dissolution and transport of radionuclides from a damaged canister and the processes and conditions that can influence this performance.

The chapter is divided into the following sections:

- *groundwater movements*
- *transport in the near field*
- *transport in surrounding bedrock*
- *transport of radionuclides in the biosphere and dose calculation.*

Essential processes and the parameters and calculation models used to analyze the processes are described in each section. Comments are made on the quality and applicability of the models.

The chapter concludes with an explanation of the strategy that is employed for the calculation chain (from radionuclide release to radiation dose in the environment) that quantifies the consequences of defective canisters. The strategy is chosen based on the purposes of the safety report. General questions are dealt with here, such as the structure of the calculation chain, model couplings, the degree of probabilism in the analysis, handling of long periods of time, etc. Certain strategic choices for individual models are also discussed here, for example choice of fuel dissolution model, the set of nuclides included in the calculations, conceptualization of near and far field, description of the lateral extent of the repository system in the rock, how site-specific models have been designed, etc.

In this report, *Chapter 11 constitutes a review of essential processes and the calculation models for transport of radionuclides which are available to SKB today. If certain processes are modelled on the basis of several conceptually different modes of description, these models and their structure are described. The parameter requirements of the calculation models and the derivation of the parameters from available data are very important. However, these are not systematically examined in this report.*

Validity documents have been prepared for some of the bigger models. This is a part of an ongoing effort to systematically report experience of the quality and applicability of the models. The chapter concludes with a description of the calculation strategy that has been chosen to illustrate the use of coupled model chains. This section can either be the last section in Chapter 11 or the first section in Chapter 12.

11.1 GENERAL ABOUT MODELLING

The concept *model* is used frequently in this chapter. Assumptions, simplifications and relationships that are used to describe a chemical or physical process are embodied in a *conceptual model.* The conceptual model is thus a description of how the geometric conditions, the structure and the constituent processes are represented /11.1-1/. The conceptual model is the point of departure for the *mathematical model,* which contains equations that are solved analytically or numerically. The complexity of the studied processes makes it necessary to use *numerical models* in the safety assessment. An interconnected chain of calculations models, linked together by the computer program PROPER, is used in SR 95 for the illustrative calculations of radionuclide transport for the canister defect scenario, see section 12.3.

The safety assessment, however, contains modelling in other contexts than just analysis of radionuclide transport. The models and calculation tools that are used for this are not described in this report in the same detail as the models for radionuclide transport.

Within the framework of a safety assessment, modelling is carried out in the following contexts.

- **Modelling of radionuclide transport in the calculation chain or in parts of the calculation chain.** For this purpose, SKB uses a chain of models for groundwater movements, near field, far field and biosphere. The calculation tools used are described in detail in sections 11.2 to 11.5. Alternatives to these models are also described.
- **Modelling to obtain input data or boundary conditions for the above calculation models.** An example of such analysis is calculations of radionuclide content performed with the program ORIGEN2, see Chapter 4. Another example is FracMan/MAFIC, with whose help it is possible, based on fracture properties, to obtain effective parameter values for the numerical model on the scale chosen to describe groundwater movements.
- **Modelling to improve understanding of individual processes, to verify the performance of individual parts of the repository system, and to interpret experimental data,** from the laboratory or from field tests. This includes analyses such as estimating hydraulic conductivity based on field data from water injection tests in individual boreholes or the use of surface complexation models to improve understanding of sorption mechanisms. These analyses can be used to support the choice of parameters in the calculation chain. Another example is analysis of the effects of future ice ages on the groundwater situation. Several of these analyses were discussed in Chapter 10.

The modelling strategy that is presented in section 11.6 sets the frames for what is included in the calculation chain for SR 95. The level of complexity for the calculation chain is not self-evident. Many of the models used in the calculation chain could very well be moved to "modelling of input data and boundary conditions" and vice versa, much depending on the purpose of the safety assessment and on the strategy chosen for the assessment.

11.2 GROUNDWATER MOVEMENTS

11.2.1 Introduction

The most important safety-related function of the rock is to provide long-term stable chemical and mechanical conditions for the engineered barriers. Another important function of the rock is to comprise a safety barrier by absorbing and retaining any radioactive materials that have been released so that their transport will be slow. An analysis of radionuclide transport from the repository to the biosphere, based on site-specific data, is included in the safety assessment.

Since the groundwater in the rock is virtually the only transport pathway for radioactive materials from the repository, all conditions that have to do with the transport of solutes with the groundwater are of potential importance.

The most important factors are:

- The groundwater flow at repository level, of importance for the life of the canister, the out-transport rate of radioactive materials and possibly for the dissolution of the fuel.
- The travel time for solutes from the repository to the biosphere.

See further in /11.2.1/.

11.2.2 General about groundwater movements

The water flux at great depths in fractured bedrock generally takes place very slowly, but the flow is unevenly distributed. Only a portion of all fractures conduct water, and the flow in these fractures is unevenly distributed, since it is restricted to channels in the fracture plane.

The driving force for groundwater movements in the rock depends on spatial differences in groundwater pressure, temperature and chemical conditions. Further, the properties of the fracture system are of importance: What degree of fracturing occurs in the rock? What do individual fractures look like and how are the fractures connected with each other? Fracture frequency and rock stresses interact. Groundwater movements are also influenced by the actual repository system, e.g. by the backfill material in the sealed tunnel and by the properties of the excavation-disturbed zone. All of these couplings and interdependencies are shown in the interaction matrix that has been constructed for the far field, see the fold-out appendix at the back of the report. A detailed discussion of this far-field matrix and couplings with the diagonal element "Groundwater movement" are presented in /11.2-2/.

Calculating groundwater movements in a rock volume when knowledge is available on all physical and chemical properties and the fracture geometry of the volume is a well-known problem. In principle, a coupled thermo-hydromechanical problem has to be solved. Simplifications of various kinds can be done for many of the problems, however. Groundwater pressure is generally the predominant driving force, which enables considerable simplifications to be made in problem solving. In other situations, there are large variations in salinity and/or temperature which influence the density. Here a coupled problem must be solved where both groundwater flow and transport of salt/heat are taken into consideration.

Even if the fundamental physical and chemical processes that control groundwater movements and transport of solutes in groundwater are known, predictions of the far-field barrier involve perhaps the greatest uncertainties in the entire safety assessment. This is due to the fact that only limited site investigations are available, so the quantity of information simply does not cover the entire rock volume to be predicted. Furthermore, the information does not permit a clear discrimination between different conceptual descriptions of nuclide dispersal.

In modelling groundwater movements within the safety assessment, it is therefore of great importance also to shed light on what these uncertainties mean for the assessment of the performance of the rock barrier.

11.2.3 Modelling of groundwater movements

The models that are used in the safety assessment for assessment of radionuclide transport from the repository and through these barriers are often simplified. The conservative simplifications must, however, be based on a structured analysis of all the assumptions that are made, and in many cases on separate analyses which verify the correctness of the simplification. The separate analyses of e.g. groundwater movements are often made using much more complex methodology than is used in the actual performance or safety assessment. The modelling strategy, section 11.6, defines what is included in the actual model chain for calculation cases reported in SR 95.

Predictive modelling of water flow and transport in fractured rock is relatively complicated compared with in geological porous media. To describe the groundwater flow in the bedrock, it is necessary to represent the heterogeneous nature of the rock in the models and to take into account the scale on which the calculation problem is being regarded. Different types of approaches, conceptual models, are used. They can be said to represent different idealizations of how the groundwater flows. Three different approaches are shown in Figure 11.2-1. There are other possible subdivisions of models for groundwater flow and transport.

Based on the chosen structure, the three methods can be described as follows.

Discrete Fracture Network (DFN) Modelling is intuitively appealing because the primary flow paths are assumed to be represented by a network of interconnnected fracture planes. The model is built up on the basis of a statistical description of the geometric and hydraulic properties of the individual fractures. This requires data that give distributions of the position, length, orientation and transmissivity of existing fractures. Different fracture populations can thereafter be simulated based on these statistics. Individual fractures with known properties can be explicitly included in the model.

Stochastic Continuum (SC) modelling is based on the assumption that properties in the rock can be described in terms of physical parameters (hydraulic conductivity, storage coefficient, etc.) which are assumed to vary in the rock in accordance with spatial distribution functions. The properties of the fractures enter implicitly into the modelling through the utilization of suitable effective parameters. The stochastic approach contains several geostatistical elements. The advantages are that proven methodology can be used and that modelling can be carried out on a regional scale (km scale). Classical or deterministic

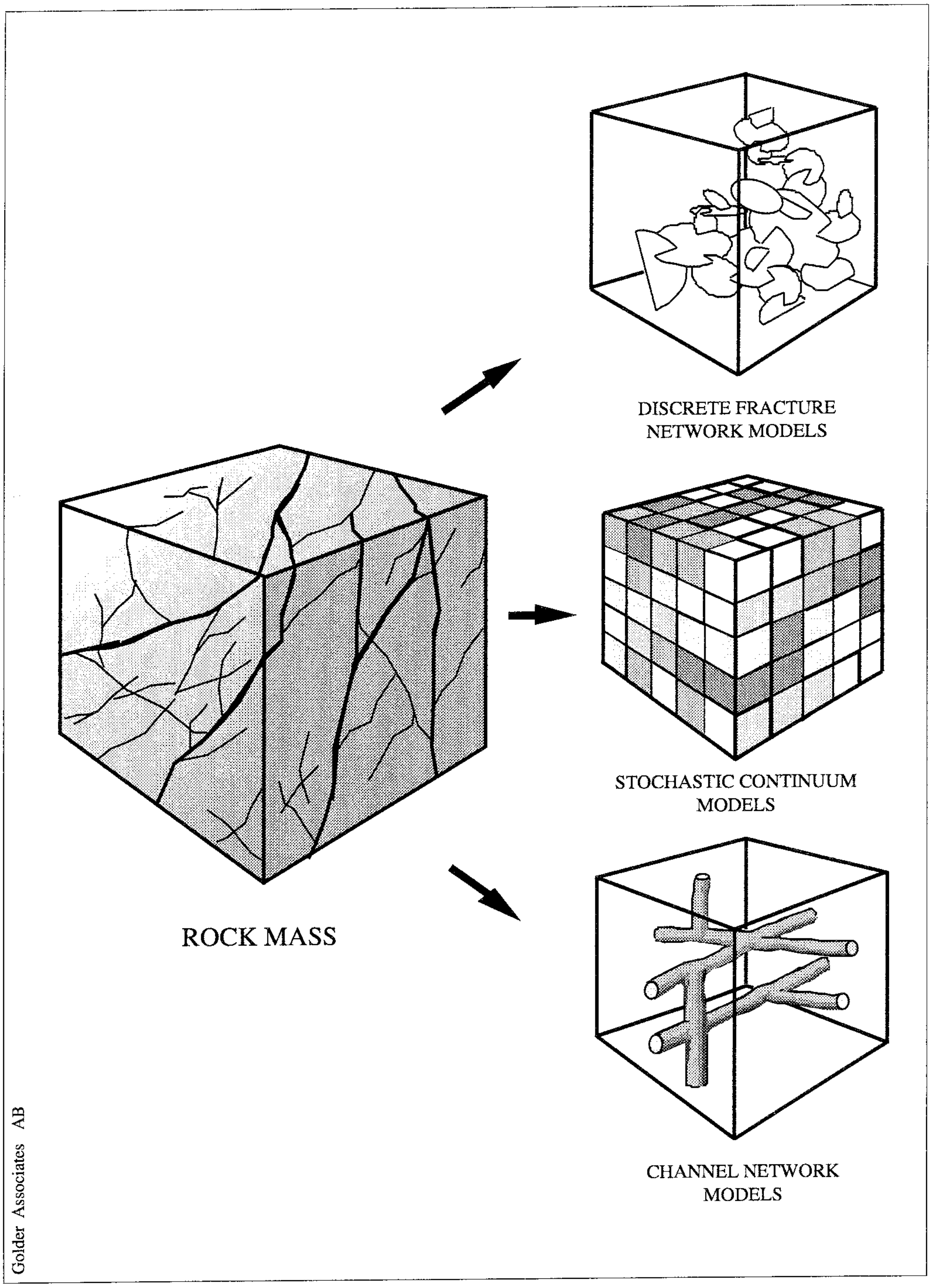


Figure 11-2.1. *Three common approaches to modelling groundwater flow and groundwater transport in fractured bedrock. The figure is taken from /11.2-3/.*

continuum modelling can be said to constitute a special case of SC modelling, in the classification that has been chosen here, since the hypotheses describe the same processes and lead in principle to the same equation system for the flow problem. Stochastic continuum modelling is used to better represent the heterogeneous nature of the rock and to illustrate the importance of this variability.

Finally, *Channel Network (CN) modelling* represents the flow in fractured rock as limited, discrete and for the most part one-dimensional flow paths, channels, which intersect each other at certain intervals. The approach is based on observations in the field, mainly from tunnels, where groundwater often occurs as flows along channels in fractures in the rock /11.2-4/. Many parameter values can, it is believed, be obtained from water injection tests in individual boreholes. Measurements of channel widths are necessary. Field measurements in tunnels are, however, associated with uncertainties due to the disturbances in the rock caused by the tunnelling work.

A more detailed description of these basic concepts is provided in various SKB reports, e.g. /11.2-3/ and /11.2-5/. Further details on the three approaches, with a special emphasis on the data requirements for the models, are provided in /11.2-6, 7/ and /11.2-4/.

The choice of model for a performance or safety assessment is entirely dependent on the purpose of the particular assessment, which geometric scale is to be studied and available data. A large part of the safety assessment has to do with how to handle uncertainties, which has been touched upon in section 3.4. These uncertainties are of various kinds, including uncertainty as to whether the chosen conceptual model is an accurate description of reality. The different conceptual models described above can, quite simply, give rise to different results. It is obviously necessary to reduce or at least have control over this source of uncertainty.

Regardless of which model concept is chosen, there are a number of common central issues which must be taken into consideration in groundwater modelling:

- The models contain *parameters* that can be obtained from field data from the ground surface, boreholes and tunnels. The derivation of these parameters is central and is described in /11.2-4, 6, 7/.
- The *boundary conditions* have a great influence on the results. It is primarily a question of whether the dominant contribution to groundwater movements at repository depth comes from regional groundwater systems or from local topography in combination with steeply dipping water-conducting structures /11.2-8/. One of the driving forces for the groundwater flow is differences in water density. When modelling also takes into account density differences in the groundwater caused by salinity variations, this must also be dealt with in the choice of boundary conditions.
- The way in which dominant water-conducting *structures* are handled in modelling differs depending on the model concept. A common feature, however, is that the representation of the spatial variability in the numerical models causes conceptual as well as numerical difficulties.
- It is of great importance to make judgements within the safety assessment of the influence of rapid transport pathways, *channels*, on the retention capacity of the rock barrier. The models must therefore be able to represent this mechanism.
- Input data to the modellings are obtained from different *field investigations in boreholes*. These are performed on a given geometric scale. It may be uncertain which rock volume the actual tests represent. Numerical ground-

water modelling with continuum models requires that the results of hydraulic tests in the field be adjusted, scaled, to the chosen calculation scale for the numerical model. Fracture network models can be used to derive the necessary effective parameters. This scaling requires extra care when modelling is done in fractured rock /11.2-9, 10/.

- *Cross-hole tests* on an investigation site provide more large-scale information than single-hole hydraulic measurements are able to contribute. It should therefore be possible to utilize steady-state and transient pressure measurements more systematically in conjunction with modelling. A proposed method for automatic calibration of groundwater models has therefore been thoroughly tested /11.2-11/. Studies show that the method has a potential to reduce the uncertainty in predictions of far-field transport by means of a systematic utilization of pressure data. Practical limitations may, however, prevent the method from being used yet in the calculation chain for radionuclide transport.

Finally, it can be mentioned that /11.2-2/ not only contains the interaction matrix that has been developed for the far field, including all interactions. It also presents an initial explanation of how the different interactions are handled in a safety assessment, and in particular how they are analyzed.

11.2.4 Available calculation tools

SKB has considered it necessary to develop and/or apply different model concepts in parallel as long as it is not possible to find the very "best" model. Different models are also justified on different geometric scales. Groundwater movements are often studied on different scales, from a regional km scale to 100 m blocks around the actual repository in the rock, and finally down to the rock around a single canister. At present there are also limitations on what physical reality the different models can represent. Thus, the calculations tools complement each other.

An important part of the work with the calculation tools is to build up confidence in their ability to produce accurate results. The concepts of verification and validation have been discussed in general terms in section 3.4. Many international projects have been devoted to this, for example HYDROCOIN /11.2-12/ and Stripa /11.2-13/. At present, work is under way within a Task Force in the framework of the Äspö Project. Field results from tests at Äspö are being utilized for parallel modelling performed by different groups with different tools. Each task is being evaluated by a group of international composition /11.2-14/.

Work is under way on the preparation of validity documents for the calculation tools intended to be utilized for modelling in the safety assessment. Such a document is available for NAMMU /11.2-15/.

The following presentation of available calculation tools follows the structure used in the preceding section:

Fracture network models

Commercially available computer programs have been developed by Golder Associates, FracMan/MAFIC /11.2-16/, and AEA Technology, NAPSAC /11.2-17/. The international Stripa Project contributed strongly to the development of these calculation programs and to their application with data from fractured rock, the Stripa Mine /11.2-18/. FracMan/MAFIC is described in Table 11.2-1.

Continuum models

In recognition of the inhomogeneities in the rock and spatial variations in its hydraulic properties, as well as the fact that these properties are measured in a limited number of points, a number of tools have recently been developed that are based on geostatistical methods and subsequent stochastic simulation. This development is proceeding in parallel with the development of computing capacity in the computer field. In recent years, SKB has developed and used a calculation tool called HYDRASTAR /11.2-19/. HYDRASTAR is furthermore adapted to PROPER, see section 11.6.1, which makes it possible to use the model to analyze radionuclide transport in a chain of models. HYDRASTAR has been utilized for the calculations of groundwater movements that are included in the illustrative calculation chain for SR 95.

The more classically used porous medium models also belong under this heading. They could be said to constitute a special case of the heading. Models of this type are used a great deal for analysis of groundwater movements, and above all for large-scale simulation. A large number of commercial calculation programs are available. SKB currently has access to NAMMU, developed by AEA Technology /11.2-20/. Another example is PHOENICS, which has been used intensively within the Äspö Project.

Channel network models

Field observations suggest that flow in fractured rock can take place in a few, small channels along fracture planes and fracture intersections. This has initiated a development of channel network models. Development of CHAN3D is currently being pursued at KTH (the Royal Institute of Technology in Stockholm) /11.2-22/. The channel model was above all developed for transport modelling of radionuclides in the far field and is therefore described in section 11.4.

Table 11.2-1 gives a description of the most commonly used numerical models for description of groundwater movements in fractured rock.

Table 11.2-1. An overview of calculation tools used by SKB for modelling of groundwater movements in fractured rock.

FracMan/ MAFIC	FracMan is a graphic software package for interactive analysis of geological systems dominated by discrete fractures. Analysis of the hydrological and mechanical properties of such systems requires a good understanding of the importance of the three-dimensional geometry for the properties of the system on different scales. The FracMan software package offers rational management of fracture data. The package includes a number of modules for data analysis and statistical/geological inference. The results can then be transferred to a numerical equation solver (MAFIC) for simulation of steady-state and/or transient groundwater flow and mass transport.
HYDRASTAR	HYDRASTAR is a finite difference model for stochastic simulation of groundwater flow. Simulation is currently being done under the assumption of constant density of the groundwater. Transient problems can be studied. After geostatistical analysis that identifies the spatial correlation structure of hydraulic data, the Turning Bands algorithm is used to generate realizations of the distribution for hydraulic conductivity, each of which statistically agrees with the identified one. These realizations can then be fitted to interpreted values of hydraulic conductivity from borehole tests. Fracture zones identified in the field can be included in the analysis. For further description with an emphasis on verification, see /11.2-23/.
NAMMU	NAMMU is used for modelling of groundwater flow and transport through porous media. NAMMU can be used to model a large number of different flow and transport phenomena. These include e.g. coupled groundwater flow and heat transport, saturated and unsaturated groundwater conditions, coupled groundwater flow, transport of solutes, etc. The tool has its own support organization which develops new versions of the program, among other things /11.2-15/. There is a user group of international composition.
PHOENICS	PHOENICS is a general equation solver for fluid mechanics problems /11.2-24/ which makes use of the finite volume method. The tool has mainly been used in SKB's applications to shed light on the effect of density variations on groundwater movements. Density contrasts arise due to thermal effects or due to the presence of saline groundwater, which is often encountered deep down in the bedrock. The potential of PHOENICS for use in safety assessments is discussed in /11.2-25/.
CHAN3D	CHAN3D generates a stochastic network of water-conducting channels and not individual fractures. The network is made up of a rectangular network with potential connections. Individual channel segments are assumed to have constant conductance, volume and flow-wetted surface area. Hydrodynamic dispersion is neglected within each channel.

11.3 NUCLIDE TRANSPORT, NEAR FIELD

11.3.1 Introduction

The concept of "near field" is difficult to define. One possible definition includes the engineered barriers plus that part of the rock that is affected by the construction of the repository. In this definition, however, it is difficult to draw a borderline between "affected" and "unaffected" rock. In the modelling of nuclide transport in the near field, a more practical definition is used instead where the borderline between near field and far field is drawn in the boundary between buffer/backfill and rock.

The near field consists of three primary barriers: **canister, buffer and fuel.** The primary function of the canister is to completely contain all radioactive materials (radionuclides) for very long periods of time. Even a damaged canister can greatly restrict the leakage of radionuclides. The primary function of the buffer is to protect the canister from flowing groundwater, but it also has a mechanical and a chemical function. Certain radionuclides sorb strongly on the buffer material, which means that the buffer is also a barrier to nuclide transport. The fuel in itself also has a very good ability to limit the dispersal (transport) of radionuclides from a defective canister due to the fact that the fuel matrix has very low solubility in the water present in the repository. Even if the matrix is dissolved, many radioelements will precipitate as secondary phases and thereby limit the leakage. The performance of the near-field barrier is discussed thoroughly in Chapter 10.

11.3.2 Transport of nuclides in the near field

In all probability it will take many millions of years before any canister in the repository leaks out radionuclides, but since the possibility that this might happen earlier cannot be ruled out entirely, the consequences of leaking canisters must be analyzed. Three processes are of importance for nuclide dispersal in the near field:

- The mechanism of canister penetration
- The dissolution of the uranium dioxide matrix
- The out-transport of radionuclides to the flowing water in the rock

The first two are dealt with in sections 10.4 and 10.6, the last below.

The three processes apply to the radionuclides that can be expected to be transported with the groundwater. The inner steel canister will generate hydrogen gas in the event of a canister failure, which means that nuclide transport in the gas phase can be an important transport pathway for certain nuclides. This is dealt with separately in section 11.3.4.

The transport of radionuclides in the near field can be divided into several steps:

- From the fuel pellet to the canister defect
- Through the canister defect
- Through the buffer and out to a fracture

Transport from the fuel pellet to the canister defect

To reach a defect in the canister, the radionuclides must first be transported across the gap between the fuel and the Zircaloy cladding, and then from there to the defect in the copper shell, see figure in section 4.1. This presumes that the steel canister is also defective.

Zircaloy is very resistant to general corrosion, and the proportion of initially damaged cladding tubes is very low, which means that the Zircaloy cladding could be an effective barrier to nuclide transport. The properties of Zircaloy when it comes to crevice corrosion, hydrogen-induced cracking and stress corrosion cracking are incompletely understood, however. It is therefore difficult to evaluate Zircaloy as a barrier over extended periods of time. But it is clear that defects in the cladding tubes will be limited in size, since the general corrosion rate is so low. This is only significant for mass transport if the total area of defects in the Zircaloy is less than the area of the defect in the copper shell (the smallest hole determines the rate of mass transport). For rapidly released, non-solubility-limited nuclides, the distance from the fuel pellet to the defect in the Zircaloy is also of importance for the mass transport.

Transport through a defect in the canister

Damage to the canister can take different forms depending on its cause:

- Damage caused by unsuccessful welding can hardly cause a through-wall hole. In the safety assessment, however, initial damages in welds are evaluated with through-wall holes of limited size;
- If a canister penetration is caused by general corrosion, a larger part of the canister wall may be lost, since the corrosion attack takes place fairly evenly over the surface;
- If the canister is damaged by mechanical stress, the appearance of the defect will depend on the nature of the stress: a small movement will cause a small defect, and so on.

Two types of defects are assumed in the modelling of canister damage. Either the damage is very limited, e.g. an initial weld defect, in which case the canister wall offers high transport resistance. Or the canister wall is gone entirely. The importance of the size of the defect is illustrated by Figure 11.3-1, which shows how Q_{eq} is dependent on the size of the hole and the local flow at the repository level. Q_{eq} is an inverted measure of the diffusion resistance in the barriers, including the water outside the buffer, and corresponds to a kind of hypothetical water flow which leaves the near field with the saturation concentration of the solubility-limited radioelements. This calculation only takes into account the resistance caused by the limited area of the hole. The radial diffusion through a little hole can also be limiting for the mass transport, particularly if the canister wall is thick.

Transport through the buffer and out to a fracture

If radionuclides get through a defect in the canister, they will reach the stagnant water in the buffer. Here mass transport is assumed to take place solely by diffusion, see the discussion in section 10.5. Numerous radionuclides sorb very strongly on the surface of the clay particles in the buffer material. This allows even some relatively long-lived nuclides to decay to insignificant levels during the transient phase of the outward diffusion, even if canister failure should occur early. The radionuclides are transported to the flowing water in the rock, either to a fracture which intersects the deposition hole (radial diffusion), to the excavation-disturbed zone (EDZ) below the deposition tunnel, or directly to the tunnel (axial diffusion). Whether the axial diffusion goes to the disturbed zone or the tunnel depends primarily on the tunnelling method and the tunnel fill material used. Those nuclides that do not decay during the transient phase

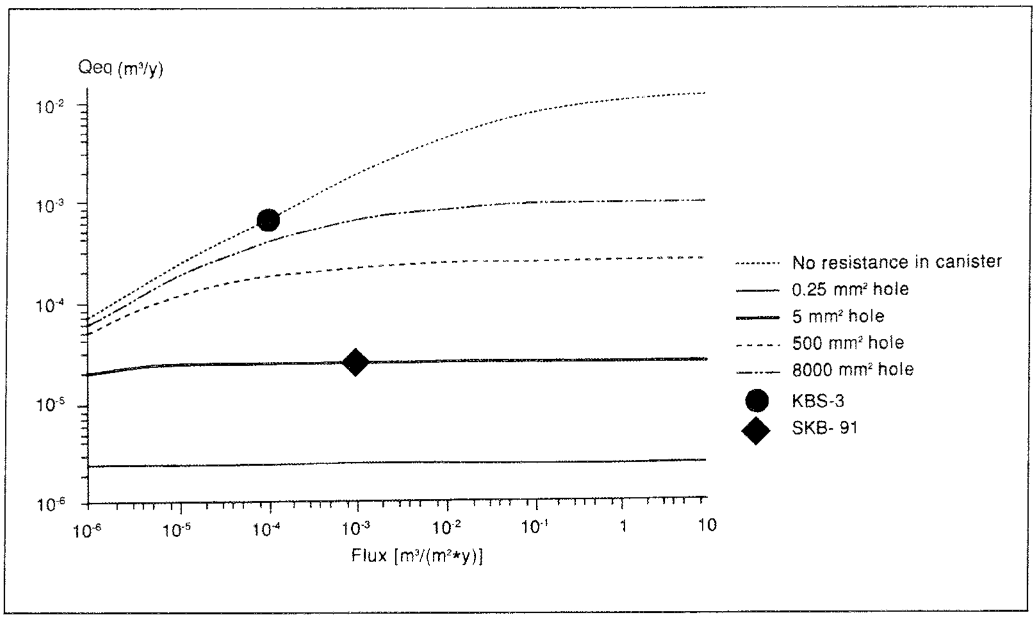


Figure 11.3-1. *Q_{eq} as a function of the flow on the repository level.*

will reach the flowing water in the rock. However, the flow rate in the rock is so low that a concentration profile is built up in the water, which contributes to the transport resistance.

11.3.3 Calculation tools for nuclide transport in the near field

SKB has developed two different computer programs for modelling of nuclide transport in the near field: Tullgarn and NUCTRAN. A description of them follows. There are also other programs that are used by other organizations. A number of them are described in /11.3-1/. The way Tullgarn and NUCTRAN are used in SR 95 is described in section 12.3.

Tullgarn and TULL22

Tullgarn /11.3-2/ is a further development of the near-field transport program NEAR21, which was based on the model concepts for the near field that were used by SKB in KBS-3. TULL22 is a version of Tullgarn intended for calculations in model chains under the administration program PROPER. All features described here can be found in both Tullgarn and TULL22.

The process which Tullgarn handles are:

- Radioactive chain decay;
- Three mechanisms for canister penetration:
  - Initial damage;
  - Internal pressure caused by helium from α-decay.
  - Corrosion. The quantity of copper which corrodes due to free oxygen present in deposition holes and tunnels is given as an input datum. Tullgarn then calculates the corrosion caused by sulphide minerals in

the bentonite, and when these have been consumed, the corrosion caused by sulphide in the groundwater.

It is also possible to predetermine the canister penetration time.

- Two models for fuel dissolution:
  - Radiolytic dissolution with effective G value, expressed as the number of transformed molecules of UO_2 per 100 eV, counted on the entire α activity.
  - Solubility-limited dissolution with uranium dioxide solubility;
- Dissolution/precipitation of secondary phases and split solubilities between different isotopes of the same element;

The transport calculations are performed with a resistance network model /11.3-3/, where the concept for mass transport is based on a comparison of Fick's law with Ohm's law for electric circuits. The transport resistances in the near field are described as coupled resistors, see Figure 11.3-2.

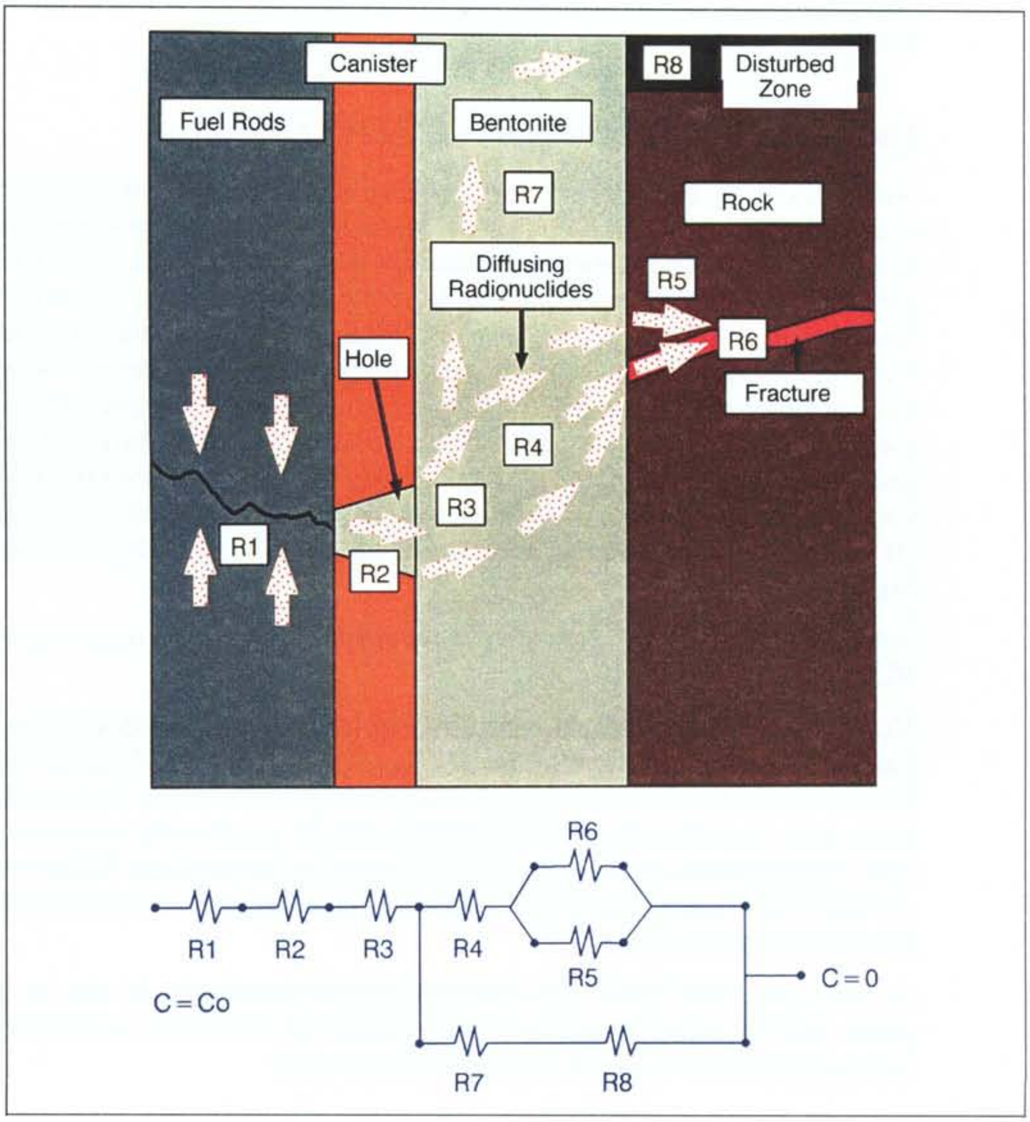


Figure 11.3-2. *Resistance network model for the transport resistance in the near field. Rx designates the different transport resistances.*

Tullgarn calculates the steady-state transport of radionuclides from the fuel surface through the hole in the canister, where R3 is the resistance caused by the limited area of the hole and R2 is the diffusion resistance in the hole, via diffusion through the buffer (R4) to a fracture in the rock (R6), or axial diffusion (R7) to the disturbed zone (R8). The program can also calculate diffusion through the rock matrix to a fracture if the mouth of the fracture should be sealed with bentonite (R5). The transport resistance inside the canister is completely neglected. The out-leakage of the gap and grain boundary inventory is modelled differently depending on the type of canister damage. If the canister has an initial defect, the gap and grain boundary inventory is dissolved in the canister's void volume and is then released from there with Q_{eq}. In the cases when the transport resistance of the canister wall is neglected, the inventory is dissolved in the buffer's water volume and is then released with Q_{eq} for that case. Tullgarn does not take any account of the transient phase of the outward diffusion of nuclides after canister penetration, which gives pessimistic results for certain nuclides.

A distinguishing characteristic of Tullgarn/TULL22 is their ability to model a complex geometry and the speed of the calculation program, which makes it possible to execute many realizations of the coupled model chain, see section 12.1.1.

The compartment model: NUCTRAN/COMP23

NUCTRAN was originally developed to be able to calculate transient processes in complex geometries /11.3-4/. In the model, the different parts of the near field – such as the hole in the canister wall, the canister's inner volume, the buffer, the tunnel backfill, fractures in the deposition hole, etc. – are described as a number of compartments. This process simulates the discretization that is done in a finite-difference or integrated finite-difference model for three-dimensional problems. The big difference is that a compartment model uses much fewer cells or compartments. So as not to lose accuracy in the calculations due to the coarse discretization, analytical solutions are used to define sizes and shapes of compartments in sensitive areas. Examples of sensitive areas are the hole in the canister wall and the mouth of the fracture towards the deposition hole.

Figure 11.3-3 shows the geometry of the near field and how it is described in NUCTRAN.

NUCTRAN can also calculate the effect of the canister's insides and void as well as the importance of filler material and damages in the Zircaloy tubes. It is also capable of modelling a growing hole in the canister. On the other hand, there is no function for calculating canister penetration (corrosion or other mechanisms), so the penetration time must be specified. Today's version of NUCTRAN cannot calculate solubilities split between different isotopes of the same element either.

In COMP23, NUCTRAN has been modified so that the model can be used together with the administration program PROPER. COMP23 and NUCTRAN are identical when it comes to the calculation part.

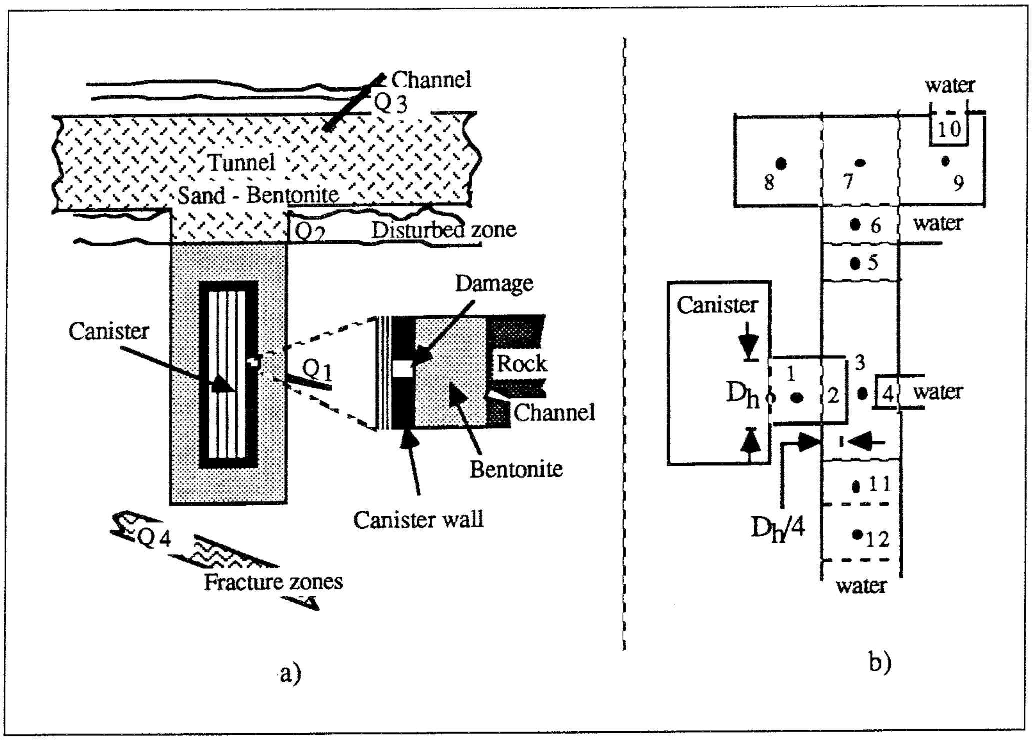


Figure 11.3-3. *Illustration of how the geometry of the near field is described in NUCTRAN.*

a) Schematic illustration of the geometry of the near field

b) Discretization of the system in a) into compartments

Verification and validation of NUCTRAN and Tullgarn

A complete validation of the representation of all the coupled transport pathways in Tullgarn and NUCTRAN is not possible to do. Comparisons with more advanced numerical models of how the individual processes are described in the programs have, however, been made with good results. There is a validity document for NUCTRAN /11.3-5/ and a similar one is planned for Tullgarn.

11.3.4 Radionuclide transport with gas

Hydrogen that is formed during the corrosion of the steel container will be transported out from the near field in the gas phase, see section 10.5. This gas phase could transport certain radionuclides (in practice only Kr-85 and C-14). Transport in the gas phase means that all transport resistance in the near field is lost. The only limitation is the time it takes to build up sufficient gas pressure to penetrate through the buffer. Nuclide transport with gas cannot be treated in existing calculation models for nuclide transport, but separate calculations of the gas transport are done for SR 95 in section 12.3.4.

11.4 NUCLIDE TRANSPORT, FAR FIELD

11.4.1 Introduction

The rock in a deep repository constitutes a safety barrier by sorbing and retaining any released radioactive materials so that their transport is slow. A sufficiently slow transport means that the nuclides have time to decay completely or at least be greatly reduced by radioactive disintegration. An analysis of radionuclide transport from the repository to the biosphere, based on site-specific data, is included in the safety assessment. The travel time for radionuclides from the repository is influenced by the following factors:

- Groundwater velocity and its distribution, which is in turn dependent on the conductivity, flow porosity and fracture pattern of the rock as well as the hydraulic gradient.
- The transport distance, the flow path, from repository to biosphere.
- Diffusion to areas with stagnant water and to micropores in the rock, so-called matrix diffusion.
- Sorption and precipitation on mineral surfaces.
- The chemical properties of the radionuclides, formation of organic complexes and colloids.

The fracture structure of the rock is also important for the retardation of the transport. A high surface/volume ratio in the fractures is favourable, especially together with a strong chemical sorption capacity of the rock type, see further in /11.4-1/. Transport properties with respect to a real site are discussed in section 6.5.

11.4.2 Transport of nuclides in the rock

The mechanisms that influence transport of radionuclides in groundwater are:

- *Advection* with the groundwater. Advection is the name given to processes where solutes are transported solely by the movement of the groundwater.
- Hydrodynamic *dispersion* is a "mixing phenomenon" which is dependent on differences in velocity in the flow within a fracture and on differences in velocity between different fractures. In connection with transport in fractured rock, this dispersion is dominated by velocity variations between different flow paths.
- Molecular *diffusion*. In this transport mechanism, solutes dissolved in the water move from areas of high concentration to areas of low concentration. Molecular diffusion is generally regarded as being subordinate to the effect of dispersion in non-reactive transport. The process is particularly important in matrix diffusion into the microfissures in the rock.
- Chemical and physical *retention*, e.g. sorption, matrix diffusion and radioactive disintegration (decay).

The subdivision of transport into an advective and a dispersive component is relatively arbitrary. The advective portion describes the average movement, while the dispersive portion takes into account the effects of heterogeneity in the rock.

For many nuclides, transport through the rock will be greatly limited by sorption on the fracture surfaces in the rock. Diffusion into the system of interconnected microfissures in the rock, known as matrix diffusion, can be considerable, since the water there is virtually stagnant compared with the water moving in the fractures. Matrix diffusion increases the surface area that is available for sorption, but even weakly sorbing radionuclides are retarded by being caught in volumes with stagnant water where only diffusion-mediated transport is possible. The presence of an interconnected system of microfissures in granitic rock has been verified by field tests /11.4-2/.

Figure 11.4-1 illustrates the mechanisms that influence the transport of nuclides from a repository to the biosphere. A more detailed discussion of transport processes in fractured rock can be found in e.g. /11.4-3/ and /11.4-26/.

Laboratory tests show that colloids can adsorb and transport radionuclides, at least under special conditions /11.4-5/. However, deep groundwaters have such low concentrations of colloidal particles that they cannot contribute to radionuclide transport to any appreciable extent. In other words, it has been shown that the natural concentration of colloidal particles in the groundwater does not have any safety-related importance /11.4-15/.

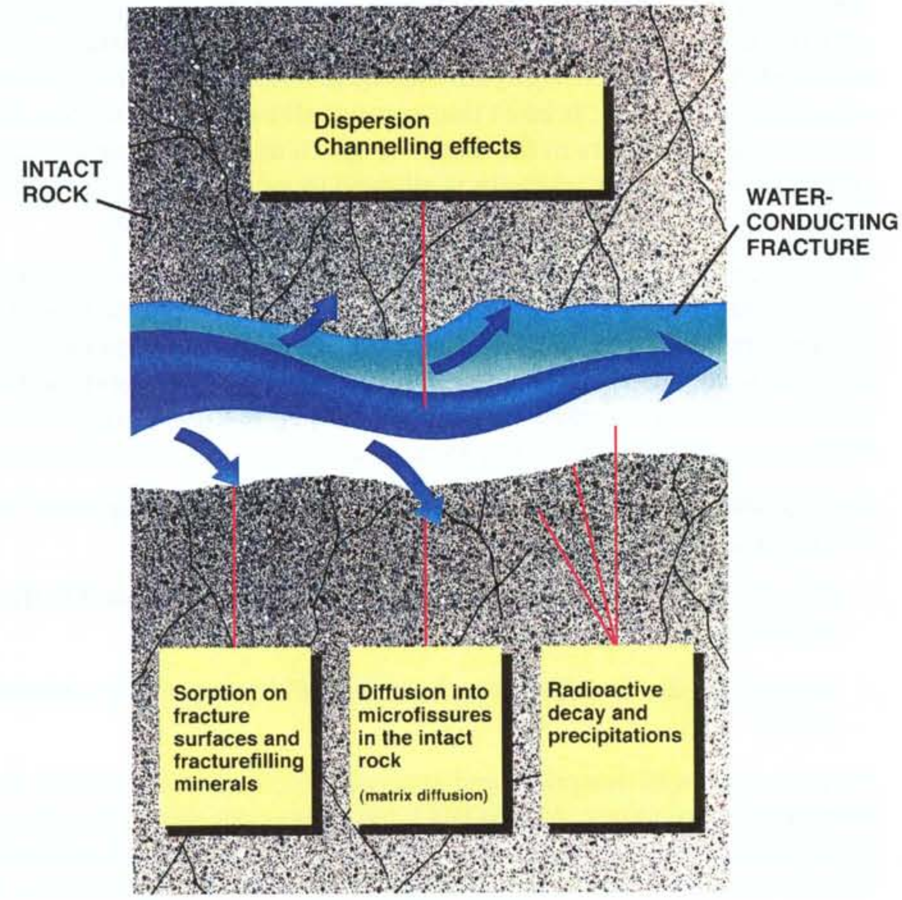


Figure 11.4-1. *Transport of a solute in rock along a flow path. Illustration of the mechanisms that influence transport of radionuclides in fractured rock.*

Gas transport of nuclides is also a possible migration mechanism. Transport with naturally occurring gases and transport with hydrogen gas generated by corrosion of the steel container are conceivable alternatives. The latter mechanism in particular is important in the safety assessment. Certain radioactive nuclides in the waste, such as C-14 and Kr-85, can be transported in the gas phase. The phenomenon has been studied and it has been found that once the gas has passed the bentonite, there are enough transport pathways for it to continue up to the ground surface /11.4-6/.

All couplings and dependencies between repository sections and various transport processes of importance for long-term safety are shown in the interaction matrix that has been constructed for the far field, see fold-out Appendix 4. There is a detailed discussion of this far-field matrix in /11.4-7/, and couplings to the diagonal element "Transport of radionuclides" are presented.

11.4.3 Modelling of nuclide transport

Modelling of flow and transport should preferably be coupled, but for practical reasons advection is often modelled in the safety assessment by means of a groundwater model (as is dispersion, for certain model concepts), while diffusion, dispersion and retention are modelled by means of a nuclide transport model. The flow model provides information to the transport model in the form of lengths of transport pathways or water travel times.

The stream tube concept entails one-dimensional modelling and can be based on particle tracking in a geohydrological model. This method is used throughout in SKB 91 /11.4-8/. A stream tube model is also used for the illustrative calculations in SR 95. It takes into account all essential processes, but requires that several parameters in the model be given as effective parameters for each stream tube. The flow intensity is allowed to vary along the flow path, however. The conceptual model is described in detail in /11.4-9/.

The principal reason for the choice of the stream tube concept is that advectively driven transport dominates in fractured rock. A detailed discussion of the stream tube concept is found in /11.4-10/, with a special emphasis on the coupling between the hydrology model and the far-field transport model, the choice of effective parameters for stream tubes and modelling of dispersion. Different suggestions for refinement of the concept are also discussed.

Among alternative approaches for analysis of nuclide transport in the rock, the following can be mentioned:

- two- or three-dimensional transport model directly coupled to the geohydrological model, and
- transport in discrete fracture networks (DFNs), especially channel networks (CNs).

Furthermore, both numerical and analytical methods are available for solving the transport equations. At GEOVAL '94, it was concluded that the geosphere should preferably be characterized statistically and analyzed stochastically. This does not necessarily entail extensive Monte Carlo simulations, however. Analytical methods are available, and development is under way /11.4-13/.

To summarize, the transport processes are dealt with in the following ways in different numerical models:

- *Advection* with the groundwater:
  The mean velocity of the groundwater is often modelled as the Darcy velocity divided by the flow porosity. Flow porosity is that proportion of the rock that is occupied by the flowing groundwater, and it is less than the total porosity.

- Hydrodynamic *dispersion:*
  Dispersion is actually a concept that is set apart from other transport mechanisms. What differentiates different model concepts (DFN, CN, SC etc.) is often how dispersion is treated in the analysis. Dispersion is usually modelled with a diffusion term proportional to the velocity of the groundwater, and transversal dispersion is often much less than longitudinal dispersion. With this procedure, the mass flow of a substance along flow paths in the rock is given by the product of the concentration gradient, the velocity of the groundwater and a coefficient called longitudinal dispersion length. The latter parameter can often be very uncertain to estimate due to the difficulty of conducting tracer tests in fractured rock over suitable length scales. The scale dependence of the dispersion coefficient that has been observed in the field is sometimes simulated by the use of a constant, known as the Peclet number. The Peclet number represents the ratio between a characteristic time for dispersive transport and a characteristic time for advective transport. The parameter is included in the advection-dispersion formulation for particle transport in rock that is often used in safety assessments. For a detailed discussion of this subject, see /11.4-9, 4/ or /11.4-11/.

- Molecular *diffusion:*
  Described by Fick's law /11.4-3/.

- *Retention:*
  The sorption models are often based on the assumption that actual kinematic behaviour can be simplified and modelled with a linear equilibrium model. For this purpose, distribution coefficients, K_d values, are assigned to each nuclide. The K_d values are determined in laboratory tests and represent the ratio between the concentration of the nuclide in rock and in groundwater. The equilibrium model applies if the concentrations are small and if the time scale for sorption is much smaller than the time scale for transport with advection and dispersion. Conservative coefficients are assigned so as not to overestimate sorption if changes occur in the water chemistry /11.4-12/. Sorption can also be described with surface complexation models. These models are general and well-founded but require a large quantity of thermodynamic data for the sorbing substance. They also require surface complexation constants, which are difficult to measure for all conceivable combinations of nuclides and minerals. The results of different experiments with surface complexation constants have therefore not been used to replace the sorption coefficients (K_d values), but to increase understanding of the sorption mechanisms and determine what they are dependent on, i.e. how reliable the sorption can be considered to be /11.4-12/. A double-porosity description of the fractured medium is used for calculations of matrix diffusion. An exchange term between the two media is included in the calculations. Trans-

port in the fractures is dominated by advection, while transport between the fractures and the rock matrix is dominated by diffusion /11.4-9/. Finally, radioactive decay, including chain decay, should naturally be taken into account in modelling of nuclide transport.

As was discussed in the preceding section, the natural concentrations of colloids in deep groundwaters are so low /11.4-14/ that they are not deemed to be of any safety-related importance /11.4-15/. SKB has not found reason to develop a special transport model for this mechanism.

As far as modelling of nuclide transport with the hydrogen gas that is generated by corrosion of the steel container in the canister is concerned, this transport presumably takes place so rapidly that it is not necessary to develop a specific model /11.4-6/. A short-circuit between near field and biosphere is assumed so as not to underestimate the effects. An example of such an analysis is given in section 12.3.4.

Field tests are being conducted to test the models for radionuclide transport. Tests with sorbing radionuclides show that the parameters that are chosen do not lead to overestimation of retardation due to sorption. Even when rapid flows and high-conductive zones or very short migration distances are used in the tests, it is difficult to get substances that are more sorbing than e.g. Sr^{2+} to go through. Even if the tests go on for months and years, transport of Cs^{+} and similar substances sometimes cannot be detected /11.4-16, 17, 18/.

Finally, it can be mentioned that /11.4-7/ contains an initial review of how the different interactions are handled in a safety assessment and how they are analyzed.

11.4.4 Available calculation tools

SKB has considered it necessary to develop and/or apply different model concepts in parallel as long as it is not possible to find the very "best" model.

An important part of the work with the calculation tools is to build up confidence in their ability to produce accurate results, i.e. verification and validation. SKB has participated in many international projects devoted to this purpose: for example, INTRACOIN /11.4-19/, Stripa /11.4-20/ and INTRAVAL /11.4-21/. As mentioned above, work is also being pursued within the framework of the Äspö Project in the Task Force. This work has strengthened the view that the calculation tools describe relevant processes with sufficient accuracy.

A list of the calculation tools which SKB has available for future safety assessments follows below.

Coupled flow-transport models

Models are available that directly couple the flow description in the rock to the description of nuclide transport. However, their practical limitations have often been too great for them to be used in safety assessments. This is discussed in /11.4-10/.

Simplified transport models

Analytical solutions of the transport equations often require considerable simplifications in the description of different processes, but can nevertheless serve as checks of results obtained from numerical solutions.

Simplified one-dimensions transport models for use in safety assessments are often based on the stream tube concept. Such a model takes the essential processes into account but at the same time requires that many parameters be averaged, since they have to be given as constants for each stream tube. This is true of the Peclet number, flow-wetted surface area, K_d values, etc. The flow intensity and the cross-sectional area can, however, vary along the flow path.

A listing of available numerical models for description of transport in fractured rock is given in Table 11.4-1. A brief description of the calculation programs is also given.

Table 11.4-1. The table gives an overview of the modelling tools that can be used in safety assessments for description of transport of radionuclides in fractured rock.

NAMMU	NAMMU is a calculation program for modelling of groundwater flow and transport through porous media. NAMMU can be used to model a large number of different flow and transport phenomena. These include e.g. coupled groundwater flow and heat transport, saturated and unsaturated groundwater conditions, coupled groundwater flow and transport of solutes. The transport portion of NAMMU contains advective and dispersive transport, linear sorption and chain decay. A validity document has been prepared for NAMMU /11.4-22/.
PHOENICS/ PARTRACK	PHOENICS is a general equation solver for fluid mechanics problems which makes use of the finite volume method. The use of PHOENICS in safety assessment contexts is discussed in /11.4-23/. The program code PHOENICS/ PARTRACK is being used for groundwater and transport modelling within the Äspö Project. The method in PARTRACK indirectly describes the processes that influence the transport of nuclides in fractured rock. Improvements are needed when it comes to, for example, handling of chain decay and methodology for relating field data for radionuclide retention to parameter values in PARTRACK /11.4-23/.
CHAN3D	The Division of Chemical Engineering at KTH (Royal Institute of Technology, Stockholm) has developed the Channel Network Model (CHAN3D) /11.4-24/. The model generates a stochastic network of water-conducting channels and not individual fractures. The network is made up of a rectangular network with potential connections. Individual channel segments are assumed to have constant conductance, volume and flow-wetted surface area. Hydrodynamic dispersion is neglected within each channel. The transport portion is simulated by particle tracking in the network. Matrix diffusion is included.
FARF31	FARF31 is used to describe the transport equations for a single stream tube based on a double-porosity description of the fractured medium /11.4-25/. The processes included in the model are advective and dispersive (Fickian) transport, one-dimensional matrix diffusion and matrix sorption, plus chain decay. The most important parameters are the travel time for the groundwater from the repository to the surface, the Peclet number (which determines the dispersive contribution), the matrix sorption coefficients (K_d) for the different radioelements, and the specific area per unit volume of rock which is available for matrix diffusion. The model can handle arbitrary boundary conditions.

11.5 RADIOECOLOGY AND DOSE CALCULATION

11.5.1 Introduction

In order to report consequences that can be compared quantitatively with given acceptance criteria for a deep repository, the flow of nuclides from the geosphere must be converted to individual dose to critical group, collective dose or radiation level to biota. This is done in "biosphere models", which describe the flow of nuclides through the natural environment and calculate doses via different exposure pathways, e.g. soil – grass – cow – milk – man.

11.5.2 Time scale

The processes that are attributed to the biosphere include many different transport pathways from groundwater to exposure and intake of radionuclides. These transport pathways depend on how society works with regard to land use, irrigation, distribution of water and food, etc. Climate and weather also influence the transport pathways in the long and short term. In other words, uncertainty will increase with the time. Reasonable predictions within an uncertainty factor of 10 are limited to a few hundred years ahead /11.5-1/.

The properties of a site that influence the dispersal of radionuclides from a final repository change more rapidly in the biosphere than in the geosphere. During the first few hundred years after closure, the conditions can be considered to be relatively well known. For longer time spans, it is necessary to consider how land and water might conceivably change and what kind of societal activities can be expected on the site. Since a deep repository will preferentially be sited in an area with a flat topography, the possibility cannot be excluded that the area will, at least during some period, be used for agricultural purposes if it is situated above sea level.

The geographic situation can give an indication of how intensive societal activities will be with respect to water and food distribution. If the future climate can be predicted, this can provide frames for agricultural production.

The various transport pathways in the biosphere will constantly change, depending on changes in eating habits, introduction of new crops, new farming methods etc. Certain exposure pathways are relatively insensitive to change, however. One example is consumption of drinking water. This water may of course come from a bottle, from a large water works (consider the situation in southern Uppland county, where 2 million people in an area 60 x 30 km drink water from the same waterworks) or from a small well. This makes a considerable difference in dilution, but the dose calculation is the same, i.e. the dose depends solely on the nuclide concentration in the water.

Variation and uncertainty in the biosphere are thus difficult to quantify on a time scale greater than 1,000 years. This is due in part to conceptual uncertainties, and in part to uncertainty and variation in the model parameters. Studies /11.5-2/ indicate that the biosphere's contribution to the uncertainties in dose calculations in a safety assessment varies between 2 and 5 orders of magnitude, depending on the radionuclide. A more site-specific assessment involves smaller uncertainties.

11.5.3 Calculation models

A rough subdivision is made in calculation models for the biosphere into the ecological model and the exposure pathways to man.

The ecological model mainly includes natural processes such as groundwater transport up through sedimentary layers, soils and sediments, surface water transport, etc., but also human influences such as irrigation. The ecological situation to be dealt with is described in the form of a conceptual model in terms of "compartments" and "transfer factors", normally called the compartment model. Flows between these compartments can be calculated in the form of transfer factors, i.e. the fraction of the content of a compartment that is transferred to the next compartment per unit time. This gives a system of linear differential equations whose solution describes the transport well. The results of this part consist of time-dependent nuclide concentrations in household, lake or sea water, soils (cropland, meadow, garden) or air.

The exposure pathways are dealt with one at a time by multiplication of the nuclide concentration in a given box by a number of factors that describe uptake fraction, concentration and finally dose per Bq. Finally, the exposure pathways are summed to a total dose per nuclide and sometimes even to a total dose.

The calculation of these two parts has been combined in the program BIOPATH, which has been used for several decades to calculate the biosphere dose factors in different cases /11.5-3/. Ample experience in the use of the program has been gained thanks to the validation exercises that have been carried out. The calculation program can also be considered to be well-verified. In other words, the uncertainty in the model results reflects the uncertainty in how the model has been built up (conceptualized and discretized), which processes are included and the parameter values in these processes.

11.5.4 Calculation strategy

Previously, biosphere modelling has primarily been used to get a quantitative comparison of different repository alternatives by describing an unfavourable but not improbable common case for all alternatives /11.5-4/. A set of isotope-specific weight factors, based on the dose in Sv which the release of one Bq of an isotope gives rise to if it is released into a typical biosphere, is applied to the release of radionuclides.

To give a realistic picture of what consequences a release could lead to, what is known, or can be assumed on good grounds, about the specific site should be taken into account. The uncertainties that exist concerning e.g. how future societies will develop must not be exaggerated. SKB will therefore avoid only applying standardized biospheres, e.g. self-sufficient farm.

General calculation strategy

The information that is needed for site-specific biosphere modelling is:

A Where (geographic situation) the radionuclides enter the biosphere

B When this happens (how far in the future)

C What chemical form the nuclides have at the release point

D The present-day size of the recipients (site description)

E Probable future changes in the size of the recipients

F Current activities (agriculture etc.) and habits (diet etc.)

G Probable future changes in activities and habits

H Dose factors for external exposure and intake

Since transport modelling in the far field is done with a stream tube model, information can be obtained from this on A and B. This model also includes information on C.

A site characterization gives information on D and F and premises for estimating E and G. The complete result will be an estimate in time and space of the probability that a certain recipient class will exist at a given place at a given point in time.

The uncertainty in the estimation of these probabilities in the future evolution of the biosphere cannot, however, be represented site-specifically for a given point in time.

By applying the calculation strategy for estimating the dose which corresponds to a given release in a given recipient, a set of dose factors is obtained for each recipient class. These dose factors should be calculated with a critical group specific for each nuclide and recipient class. The uncertainties in the dose factor can also be described as a time-dependent function.

11.5.5 Dose calculation

A complete dose calculation presumes that all exposure pathways are taken into consideration. In practice, it is necessary to concentrate on those that can be expected to give significant contributions, e.g. intake of water, milk, meat, green vegetables, root vegetables, grain, fish and soil, plus external irradiation from the ground, see Figure 11.5-1.

The quantification of these exposure pathways is based on today's pattern of consumption. Since this gives a good picture in most cases, the calculations have not been complicated in this descriptive example with time-dependent exposure pathways.

Critical groups have not been defined specifically for each exposure pathway; instead, one critical group per biosphere/recipient class has been used in the calculations. The difference is of minor importance, since there is normally one exposure pathway that dominates for a biosphere/recipient class. The critical group should reflect the composition of the population, which means doses to children should also be calculated.

The dose that is calculated is the "committed equivalent dose", which for most substances and radiation types can be translated to a risk of cancer. The ICRP's weighted dose factors have been used, including the most recent ICRP publication.

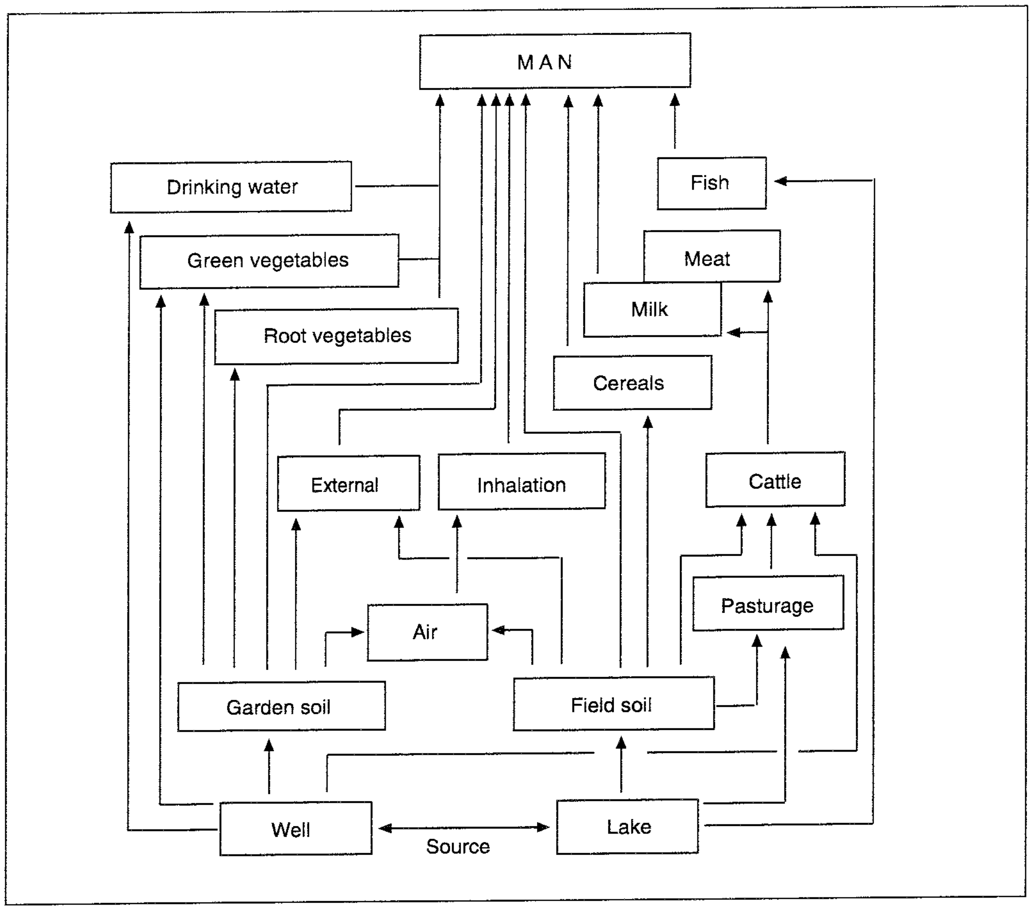


Figure 11.5-1. Exposure pathways for critical group.

11.6 STRATEGY FOR MODELLING OF RADIONUCLIDE TRANSPORT

11.6.1 Introduction

As is evident from the previous sections in this chapter, there are alternative possibilities for modelling radionuclide transport from a deep repository. A carefully thought-out modelling strategy is therefore required for the execution of the calculations in a safety assessment. An account is given in this section of the aspects that have been considered in the modelling of radionuclide transport in SR 95. Even more long-term strategic choices, for example working with a modularized chain, are described.

11.6.2 Modularized calculation chain

The principal calculation case analyzed in this report, the type defect scenario, is made up of a chain of submodels, see Figure 11.6-1. Results of calculations early in the chain comprise input data to subsequent models. First in the chain lies a geohydrological model, which is followed by calculation models for near field, far field and biosphere. This **modularized structure**, i.e. subdivision into several submodels, provides great flexibility when it comes to depicting

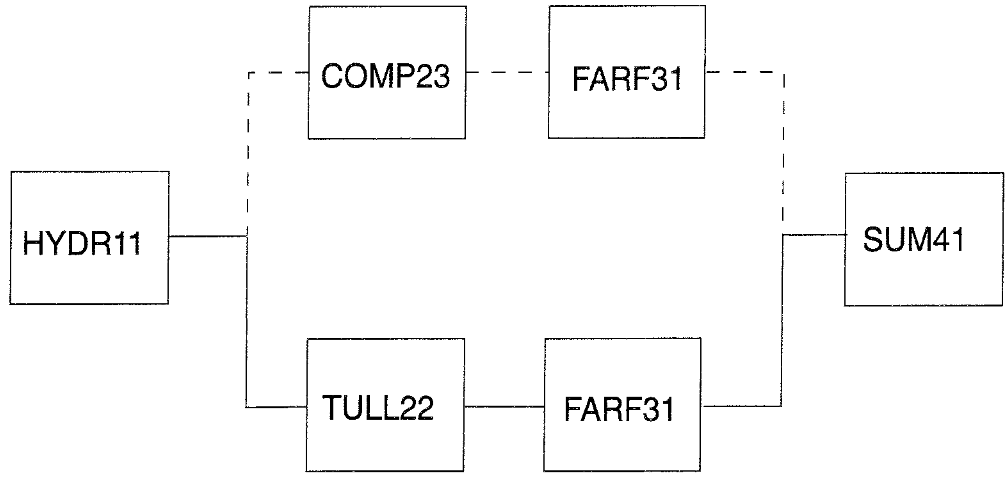


Figure 11.6-1. Model chain for type defect scenario. The geohydrological calculations are handled by the module HYDR11, the near-field transport by TULL22. Far-field transport is calculated in FARF31 and the biosphere is modelled with separately calculated dose factors which are finally included in the calculation via summation modules (simplified in the figure). In future work, the near-field calculations are planned to be done with the module COMP23 for canisters with initial damages, and with TULL22 for initially intact canisters.

various repository designs. This can be done on a high level by combining different submodels without making any changes in the fundamental computer codes. The modularization also makes it easy to study the interim results of the different submodels. A special administration system called PROPER is used to administer the chain calculations.

11.6.3 Model chain, processes in and outside the chain

Processes that are calculated outside the chain and then weighed into the calculations

In the calculation cases that are carried out with a model chain, certain processes are taken into account which are not calculated in the actual chain.

An example of this is the **nuclide inventory**, which is calculated separately based on data on fuel type, burnup, etc., see section 4.2. The results of this calculation comprise input data to the near-field model in the calculation chain. In principle, the inventory calculation could also be included in the chain. However, this calculation is very extensive, and since it is not covered by any probabilistic treatment, it has been placed outside the calculation chain and calculated "once and for all".

Another example of a process that is not included in the chain is **the temperature evolution of the repository**. This is also calculated in a separate model, section 10.2. The other calculation models do not contain any temperature dependencies, but the models are only valid within a limited temperature range (which may be different for different parts of the repository system). The results of the temperature calculation are used to ensure that the temperature of the repository does not lie outside this range at any time.

Processes excluded from the calculation chain

All the processes dealt with in the interaction matrices for a given scenario are not taken up in the corresponding calculation case. These may be processes that are of negligible importance in view of the purpose of the analysis or processes that are handled by means of conservative assumptions and therefore do not have to be modelled in detail.

Radionuclide transport via colloids is an example of a process that is deemed to be of negligible importance for the safety of the repository and can therefore be neglected, see section 11.4. Corrosive dissolution of the cladding tubes of Zircaloy in the fuel assemblies is an example of a process that is handled conservatively. In reality, the cladding tubes constitute a barrier in the repository system. By conservatively assuming that they are initially completely dissolved, the problem of modelling the corrosion process is circumvented.

11.6.4 Degree of probabilism in the analysis

A central decision in the modelling strategy is the choice between probabilistic and non-probabilistic calculation. In a probabilistic calculation, input data can be given in the form of statistical distributions instead of as fixed values. In this way, uncertainties in input data can be treated quantitatively in the calculations. The results as well are then obtained in the form of statistical distributions, in contrast to non-probabilistic calculations, where input data and results are fixed values. The choice of type of calculation is therefore mainly determined by the strategy for handling of uncertainties in the analysis. Probabilistic calculations may require more computing power and a compromise may therefore sometimes have to be struck between the degree of probabilism in the calculation and the level of detailing in constituent models.

The PROPER administration system permits probabilistic calculations /11.6-1/. In the actual model chain for the type defect scenario, there are probabilistic treatments of the spatial variability of the conductivity field in the geohydrological model and in the random selection of canisters with initial damages. Initial canister damage could conceivably occur as a consequence of defects in fabrication or materials. It is reasonable to conceive of such defects as occurring randomly. It is therefore also reasonable in the model to choose repository positions with damaged canisters randomly.

Due to the limited knowledge of uncertainties and variabilities for most input data, a more elaborate probabilistic treatment does not appear meaningful today. More detailed knowledge of the nature of the modelled processes may often be required before it is reasonable to carry out a comprehensive probabilistic analysis of the uncertainties in input data. For this reason, no extensive probabilistic treatment has been done in the illustrative calculations in SR 95.

More elaborate statistical treatments can be carried out when more reliable information regarding uncertainty for input data to different models is available. However, a compromise must always be made with regard to the purpose of the analysis. If, for example, the purpose is to shed light on the importance of uncertainties in certain factors in the repository design, an excessively probabilistic treatment of other factors may obscure the effects of the uncertainties one wishes to shed light on.

11.6.5 Choice of geohydrological calculation model

As has been noted in section 11.2, predictive modelling of water flow and transport in rock is complicated. In order to model the rock, its heterogeneous nature must be represented in the models.

This report with illustrative calculations of geohydrological conditions at Äspö uses HYDRASTAR as the calculation model. The model takes into account the spatial variability of the rock with regard to its hydraulic properties. The possibility of including repository tunnels in the model and the existing coupling to near-field and far-field transport models also motivate the choice.

HYDRASTAR is not, however, able at the present time to model the effects on the water movements of varying density in the groundwater. Such effects can be expected at Äspö, where the salinity rises with the depth, see section 6.3. These effects must be analyzed in detail in a complete safety assessment. Conservative calculations are used in SR 95 as an illustrative example of the calculation chain.

11.6.6 Distributed repository model

The geohydrological model that is used, HYDRASTAR, permits, in combination with the modularized structure of the calculation chain, a distributed model of the repository to be devised. This means that the hydrogeological "environment" can be permitted to vary between different repository canisters or groups of canisters. The distributed calculation model is much more realistic than a non-distributed model, where a typical hydrogeological environment is modelled for **one** canister, which then has to represent all canisters in the repository. This is particularly true when detailed knowledge exists of the hydrogeological conditions on a repository site.

11.6.7 Conceptualization of the near field

The near field in a repository of the KBS-3 type is geometrically relatively complex and there are many conceivable transport pathways for radionuclides out from the repository. Both of the models that are available for safety assessments, COMP23 and TULL22 (see section 11.3) can model several transport pathways. This is necessary in a realistic transport model. In general, models for transport in the near field are a compromise between simplification of the geometry and accuracy in the equation solution. The development of the models that are used by SKB has been focused on representing the near-field geometry as well as possible. This permits the execution of variation cases with different near-field designs. Models with exact solutions of the transport equations often require rough simplifications of the premises, e.g. spherical canister or zero nuclide concentration in the flowing water. The accuracy of COMP23 has been tested by comparisons with more exact models, and the results show that even if very few compartments are used, equivalent results are obtained /11.6-2/. COMP23 is a more realistic transport model than TULL22, since it can describe transient processes. However, it still lacks an α oxidation model for the fuel, handling of split solubilities and modelling of canister penetration. Moreover, the calculations are considerably more involved, which requires greater computer capacity.

Due to the fact that COMP23 is still in the development stage, the chain calculations in SR 95 are carried out with TULL22. The intention is to use COMP23 in future assessments for modelling of early canister damages, where there is a great need for a good description of the transient transport through the buffer. TULL22 will be used for canister penetration that occurs after such a long time that the transient portion of the transport is of less importance.

11.6.8 Choice of fuel model

Available conceptual models for release of nuclides and dissolution/transformation of the fuel matrix were discussed in section 10.7. Today, assigned IRFs (instant-release fractions) are used for nuclides in gaps, grain boundaries and metal parts, i.e. this fraction of the inventory is assumed to be released immediately when water comes into contact with the fuel. Three conceptual models are available today for describing matrix transformation (which was discussed in section 10.6):

- Thermodynamic solubility, i.e. the fuel dissolves at the rate UO_2 dissolves,
- Kinetically controlled dissolution, i.e. the dissolution rate is proportional to the α-dose rate,
- Immediate dissolution, i.e. all fuel dissolves immediately when it comes into contact with water. This is a very conservative and highly simplified model.

The kinetic model that was used in SKB 91 is implemented in the near-field model TULL22, see section 10.6. It is conservative according to today's knowledge, but will nevertheless be used in SR 95.

In COMP23, there is a choice today between using thermodynamic solubility or the model for immediate dissolution. The latter model overestimates the nuclide release by many orders of magnitude.

11.6.9 Choice of nuclides

The nuclides that should be included in a complete safety report are described in section 4.2.5. SR 95 is not intended to be a complete safety report, and the available time for illustrative calculations is limited. The nuclide list has therefore been reduced to simplify the calculations. The nuclides which experience has shown do not contribute to dose and activity releases have been deleted. The nuclides that are found in section 4.2.5 but are not included in the calculations are: The Cm-244 chain, Ni-59, Ni-63, Zr-93, Nb-94, Ag-108m, Sn-126, Sm-151 and Ho-166m.

11.6.10 Conceptualization of the far field

Section 11.6.2 sheds light on the coupling between the near-field and far-field models. As discussed in section 11.3, the near-field description permits a number of different transport pathways out through the engineered barriers. Allowance is made for e.g. the transport resistance in the disturbed zone in the floors of the deposition tunnels and the resistance from bentonite that has penetrated into fractures surrounding the deposition hole. The sum of the activity releases from the near field is then given as input data in a single point to the transport model for the rock. This pertains to a single canister. When the distributed repository model (see section 11.6.6) and the stream tube concept are used, each transport pathway in the far field will receive a point-source near-field release representing a number of canisters. A resolution problem hereby arises between the near-field and far-field descriptions.

The stream tube concept entails one-dimensional modelling and can be based on particle tracking in a large-scale geohydrological calculation model. This method is used throughout SKB's most recent safety report, SKB 91 /11.6-3/. Use of the stream tube concept is also found in other safety assessments in recent years /11.6-4, 5/. The choice of the stream tube concept is primarily warranted by the assumption that advectively driven transport dominates in fractured rock.

Limitations of the stream tube concept are (see further /11.6-6/):

- the coupling between the hydrology and the transport models, i.e. how well can the flow paths of the water in the rock be described with stream tubes,
- the fact that mixing between stream tubes is not allowed, and
- obtaining effective parameters for each individual stream tube.

The stream tube concept requires data in the form of flow paths and their parameters. These can be generated in different ways. An alternative method for generation of stream tubes has been tried where a discrete description of the bedrock is utilized /11.6-7/. Discrete fracture networks in three dimensions are used here for the rock mass, generated with the aid of fracture statistics from a site. The method identifies the most conductive flow pathways through the fracture network with the use of graph theory. The methodology also has the potential to determine the parameters that are necessary for calculations of radionuclide transport in the safety assessment. In other words, this is an alternative to generating pathlines with a regional continuum model, which always entails difficulties with dissolution in the near field around the repository. The idea is to obtain better consistency between flow paths in the calculation model and the geological situation, i.e. the fracture geometry, around individual canisters.

Regardless of whether a discrete or continuum description of the rock is used, it is deemed to be entirely satisfactory to use the stream tube concept for the illustrative calculations that are planned to be done in SR 95.

11.6.11 Site-specific biosphere

Analysis of the location of the discharge points relative to the coastline at Äspö shows that about 2% lie inside the coastline, see section 12.3. Of these, 90% are closer to the coastline than 50 m. Since, for calculation-related reasons, the pathlines stop about 25 m below the surface of the sea, it can be assumed that the nuclides will enter the biosphere via the water in the Baltic Sea. The various recipient classes are thereby reduced to brackish water alone (possibly bay + sea). The expected future course of events before the next ice age will not change this. This means that all releases to the biosphere are assumed to go via the water of the Baltic Sea for the type defect scenario that is described in section 12.3.

For SR 95, the dose factors are taken from the Baltic Sea case in SKB 91 and supplemented with data for curium. No reliable uncertainty analysis is available today for these data.

The SUM41 routine is used, and here the total nuclide flow from all stream tubes is multiplied by the dose factor for each nuclide. Finally, a total dose is summed up. No stream tube-dependent dose factors are required for this simplified case with a single type of recipient (Baltic Sea).

11.6.12 Handling of time-dependent changes in geosphere and biosphere

The description of geosphere and biosphere in the calculation chain relates to present-day conditions. Descriptions of time-dependent changes, e.g. land uplift or changes of man's exploitation of the natural environment, are not included in the models. The main reason for this is the difficulty of making reasonable predictions on the same level of detail as the models otherwise work with.

Certain of the time-dependent changes can be described by means of variation analyses. A land uplift can, for example, be described by a variation of the topography in the hydrology model and a modified biosphere model. This is discussed more thoroughly in section 12.3.4. Other changes can be treated in separate scenarios and calculation cases, for example influence of an ice age, see section 12.5.

12 SCENARIOS/CALCULATION CASES

This chapter presents the results of the analysis of chosen scenarios or calculation cases. Separate sections contain qualitative descriptions or calculation results from main cases, variations and sensitivity analyses of:

- *normal scenario*
- *canister defect scenario*
- *glaciation*
- *earthquakes*
- *effects of human activities*
- *materials left behind in the deep repository*
- *other scenarios*

In this report, *analyses of a limited set of scenarios are presented. The accounts are based on material large parts of which are being reworked for SR-I and which consists primarily of examples of different possible methods for scenario analyses.*

12.1 INTRODUCTION

The scenarios chosen in Chapter 9 are analyzed in this chapter. In a complete safety assessment, the selected scenarios should together provide as comprehensive a picture as possible of the possible evolutionary pathways of the repository. Only a few of the most important scenarios are analyzed in SR 95. Since SR 95 is a template for safety assessments with illustrative examples, the scenarios have been chosen so that different methods for scenario analysis are illustrated.

First, in the normal scenario, the expected evolution of the repository is described. This is a descriptive text with many references to the performance assessments that were presented in Chapter 10.

In the canister defect scenario, the consequences of serious initial defects in a small portion of the canisters in the repository are analyzed. This scenario is analyzed with far-reaching model calculations.

Scenarios caused by human activities are then discussed in general terms. A more detailed analysis of a rock drilling scenario, performed as a standard risk assessment, is also presented.

Glaciation scenarios are not analyzed in SR 95. However, section 12.5 examines conceivable external conditions in the event of a glaciation and how these conditions could affect the geohydrological situation. The section also outlines how a more complete analysis of a glaciation could be carried out.

12.2 NORMAL SCENARIO

12.2.1 Introduction

This section discusses the normal scenario, i.e. the expected evolution of the repository. The description of the normal scenario is important above all for two reasons:

- The normal scenario covers the most probable of the repository's conceivable evolutionary pathways
- The normal scenario is used as a point of departure for the analysis of many other scenarios

The account is relatively brief and is based to a large extent on the premises and performance assessments described in previous chapters.

12.2.2 Premises

The repository should be designed for the primary purpose of isolating the waste. If this isolation should be breached, the design should also ensure that the outward transport of radionuclides is prevented or retarded and that the recipient conditions on the site are favourable.

The repository is designed so that the isolation is not breached in the event of normal repository performance. The design requirements are therefore primarily related to the repository's isolating capacity. These requirements can be summarized in point form for the geological conditions on the site, the canister and the buffer /12.2-1/.

The geological conditions on the site must be good with respect to

- mechanical stability of the rock,
- chemical environment for canister and buffer in groundwater/rock,
- presence and transport of substances corrosive to the canister,
- prevention of future intrusions and alternative uses,
- groundwater conditions.

The canister must be designed and fabricated so that it

- is leaktight at deposition,
- can resist the chemical action of
  - oxygen and other oxidants that are introduced during the repository's construction and operating periods,
  - substances that can normally occur in reducing groundwaters,
- limits the effects resulting from
  - external and internal corrosion caused by radiolysis products,
  - internal corrosion caused by residual oxygen and water,

- can resist mechanical stresses caused by
  - hydrostatic pressure at repository depth,
  - the swelling pressure from the buffer material,
  - extra loads during an ice age,
  - rock movements caused by stress redistributions resulting from the repository's construction.

The buffer must

- completely envelop the canister for a long period of time – remain in the deposition cavity,
- bear the canister centred in the deposition hole,
- prevent groundwater flow and thereby retard the inward transport of corrodants,
- dissipate heat from the canister,
- resist chemical transformation for a long time,
- not jeopardize the abilities of the canister and the rock to fulfil their functional requirements,
- protect the canister by comprising a plastic protection against rock movements.

The design that has been chosen to meet these requirements is described in greater detail and in quantified terms in Chapter 5; the adaptation to a specific site in Chapter 7. The quantity of deposited waste is given in Chapter 4. In the discussion of repository evolution that follows here, it is assumed that the site meets the specified requirements. However, the description does not apply to any specific site. Furthermore, the discussion is restricted to the repository for spent nuclear fuel.

12.2.3 Expected evolution of the system

Temperature

The decay heat from the spent nuclear fuel causes the temperature in the buffer and the rock nearest the deposition holes to rise to a maximum around 10 years after deposition, after which it slowly falls. Calculations have shown that the maximum temperature does not appreciably exceed 80°C. Elevated temperature will prevail for thousands of years. The change of temperature at the repository is described in greater detail in section 10.2.

Rock mass

After closure, the rock around the repository and around the deposition positions will be saturated with groundwater. The time required for this depends on the local groundwater conditions and can vary widely. It is estimated to be at least one year and probably more than 10 years for typical conditions at the depths in question in Swedish bedrock.

The heat generated by the disposal canisters causes the aperture width of the fractures in the rock nearest the deposition hole to decrease, after which it increases again when the temperature declines. The temperature increase can also create local convection cells and marginally increased rock stresses in the near field, but this is not expected to affect the performance of the repository.

A climate change is expected to occur in Sweden in 5,000–10,000 years with regional glaciation in the mountain range and permafrost in large parts of the rest of the country. A glaciation is often analyzed as a separate scenario, see further section 12.5. The probability of canister damages due to a glaciation is highly site-specific.

The geological history and expected long-term evolution of the site will have to be studied in connection with site selection. The disposal canisters will be placed in positions where the probability of rock movements that could lead to canister damages can be judged to be very small. It is therefore believed that the normal evolution will be that the rock will constitute a stable environment for the repository even for very long periods of time. See further section 10.3.

A more detailed analysis cannot be made until the site has been selected.

Buffer and near-field chemistry

The buffer material is given a high water content initially, but it is not completely water-saturated. Gradually the buffer will become saturated with groundwater from the rock. The time required for this depends on, among other things, the process of water saturation of the rock around the deposition hole in question, see above. Water saturation of the buffer is estimated to take tens of years under typical conditions.

The temperature increase of about 80°C in the buffer does not have an adverse effect on its performance. The buffer material is chosen to withstand a temperature of about 130°C without its performance being affected.

The buffer material used, sodium bentonite, consists for the most part of sodium smectite. The presence of calcium ions in compounds and groundwater is expected to lead to a gradual conversion from sodium to calcium smectite. Some of the swelling pressure in the buffer will then be lost, but the other important properties of the material will be retained. The time for this conversion is on the order of 100,000 years.

Conversion of the smectite material in the buffer to hydrated mica could affect its performance. However, such a process takes millions of years under the conditions expected to prevail.

The chemistry in the near field will be determined by the buffer material and its content of impurities. Changes in the natural groundwater chemistry are therefore of less importance. The buffer is expected to give pH values of between 7 and 9, which is favourable for the performance of the other barriers.

The performance of the buffer under different conditions is analyzed more thoroughly in section 10.5.

Canister

The canister must be designed and fabricated so that it is intact and leakproof on deposition. An important question is the size and importance of the defects that cannot be ruled out by means of the inspection methods that will be used. The less probable assumption that serious initial defects are present in the copper shell is analyzed in the canister defect scenario in section 12.3.

The canister will be exposed to corrosion attack from impurities in the groundwater and the buffer material. Under ordinary circumstances, the copper shell is expected to provide sufficient corrosion protection to guarantee the integrity of the canister for millions of years, see further section 10.4.

In future reports, strength calculations will show that the canister can, with high probability, withstand whatever mechanical loads may arise in connection with ice ages, stress redistributions in the rock, seismic events, uneven swelling pressure in the buffer, etc.

Biosphere

The evolution of the biosphere includes the evolution of human society, including changes in man's exploitation of the natural environment. The evolution of the biosphere is highly uncertain to predict, even in what is called a "normal" scenario. However, this evolution is not expected to lead to such far-reaching climate changes, environmental degradation or other kinds of changes that these will have consequences at repository depth.

Nor is it assumed that man will, in the normal course of events, deliberately or inadvertently intrude into the repository. The probability of an intrusion has been reduced by avoiding a site with ore deposits that are or might conceivably be of interest. Different cases of intrusion and environmental impact, although judged to be unlikely, are analyzed as separate scenarios, see section 12.4.

12.2.4 Consequences of the normal scenario

Under normal circumstances, the copper shell is, as noted above, expected to retain its integrity for millions of years. During this time, the radioactivity of the waste will decay considerably. After around 100,000 years, the potential toxicity of the deposited waste will be comparable to the toxicity of the mineral that was originally mined to fabricate the fuel.

Thus, in the normal scenario, no releases take place for very long times. When the integrity of the canister is finally lost, the radiotoxicity of the waste has long since been comparable to naturally occurring deposits of radioactive minerals.

12.3 CANISTER DEFECT SCENARIO

In the canister defect scenario, some of the deposited canisters are assumed to be defective. The defect is defined as an initial hole of a given size, for example through a weld. A release of radionuclides is obtained from the defective canisters. The released radionuclides are retained and retarded by the engineered and natural barriers. Some of the release finally reaches the biosphere.

In SR 95, the canister defect scenario is used to illustrate how a scenario is analyzed by means of extensive model calculations. The premises for the calculations presented here are unrealistic in certain respects. The choice of site, for example, is not realistic in so far as the investigated bedrock at Äspö is not big enough to accommodate the entire planned deep repository. The hydrogeological model description has not fully utilized the available site data. Furthermore, the hydrogeological description of the site is based on data from Äspö, whereas many transport parameters for radionuclides are not yet available for Äspö. These have instead been taken from SKB 91, where data from another site (Finnsjön) were used for the analysis.

The canister defect scenario as it is presented here nevertheless serves as a good illustration of how a quantitative analysis with model calculations is carried out. Even the types of results that are presented coincide well with what could be included in an analysis of a more realistic case.

The premises for the modelling of the canister defect scenario, and the results of the calculations, are presented in the following. A brief discussion then follows of what influence different variations in the premises for the modelling might have on the results.

The models used in the calculation chain do not take into account the hydrogen gas that can be expected to be formed when the inner steel container comes into contact with groundwater. Finally, one effect of this gas generation, namely radionuclide transport in the gas phase, is discussed in section 12.3.4.

12.3.1 Premises for the calculation

The calculations for the canister defect scenario are carried out with a model chain according to Figure 12.3-1. The degree of probabilism in the calculation has been kept low, see section 11.6.4. In order to achieve reasonable precision in the calculations, 100 realizations of the chain have been carried out. Other general aspects that have been considered in connection with the modelling have been discussed in section 11.6.

Repository siting and design

The repository has not yet been sited. Data representative of the area around the Äspö Hard Rock Laboratory, HRL, in the Municipality of Oskarshamn have been used. The principal reason this site has been chosen for the calculation case is that an extensive geoscientific database is available for Äspö. The site is described in greater detail in Chapter 6.

The repository design is based on the KBS-3 concept. The size of the repository in the calculation is about 10 percent of the predicted size of a Swedish deep repository. The modelled repository includes 400 canisters emplaced at a depth of 450 m. The details of the design are described in Chapter 7 and in /12.3-1/.

Figure 12.3-2 shows the repository site with the positions of the canister segments marked. The picture includes the model area for the hydrological calculations with HYDRASTAR, see below. The repository layout has not

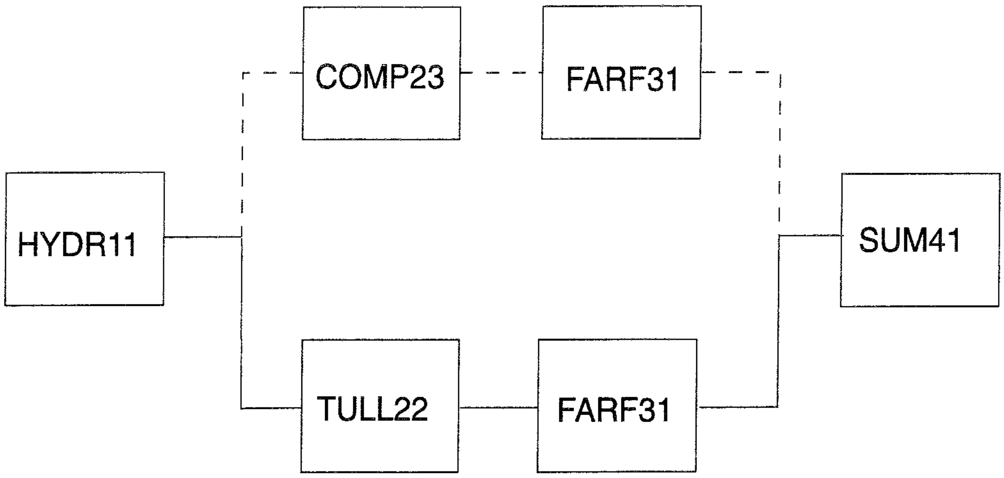


Figure 12.3-1. Model chain for canister defect scenario. The hydrogeology calculations are handled by the module HYDR11, the near-field transport by TULL22. Far-field transport is calculated in FARF31 and the biosphere is modelled with separately calculated dose factors which are finally included in the calculation via summation modules (simplified in the figure). In future work, the near-field calculations are planned to be done with the module COMP23 for canisters with initial damages, and with TULL22 for initially intact canisters.

been optimized in detail with reference to the hydrogeological data on the site. According to the present-day interpretation of the character of the NNW structures, they are allowed to intersect deposition tunnels but not individual deposition holes. The NNW structure has not been avoided in this illustrative example, however. See further Figure 12.3-3a.

Radionuclide inventory

Calculations of the radionuclide inventory based on operational data for the Swedish nuclear power programme are described in section 4.2. Many of the nuclides in the inventory are of negligible importance for safety. A methodical selection of nuclides is therefore made prior to a safety assessment, this is also described in section 4.2. The inventory has been further reduced slightly in the modelling of the canister defect scenario, which is described and justified in section 11.6.9.

Canister, canister damages

The copper-steel canister used in the calculations is described in section 5.3. The probability of an initial canister defect has been set at 0.001 per canister. The defect is assumed to be a hole with a diameter of 5 mm^2 which penetrates the copper canister. It is difficult to make realistic estimates of both the probability and the size of initial canister defects. These sizes are dependent on the fabrication and inspection methods used for the canister, which have not been determined yet. However, it is likely that the probability and above all the size of the defect are greatly overestimated in the stipulated assumptions. Canister data for the calculations is given in Table 12.3-1.

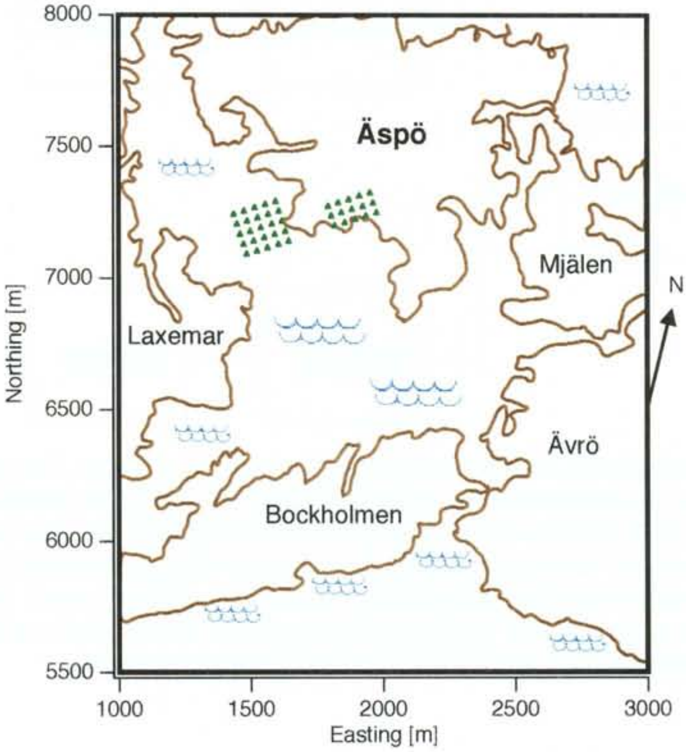


Figure 12.3-2. *Model area for the geohydrological calculations with the 40 repository segments marked with triangles. Each segment represents 10 canisters. The repository is situated at a depth of 450 m. The coordinates are given in the Äspö system.*

For canisters without an initial defect, the case where the barrier function of the canister is lost due to either inner pressure or corrosion is modelled. With the data used for the canister and the near field, it is estimated, based on the model calculations, that initially intact canisters will be destroyed by corrosion after more than 100 million years. It will take even longer for a critical inner pressure to be built up.

Hydrogeology calculations

The hydrogeological calculations are carried out with the programme HYDRASTAR, designated HYDR11 in the model chain, over an area of 2.5x2x2 km^3, see Figure 12.3-2. The hydrogeological model in the calculation is based on the site description in Chapter 6. The repository area is divided in the hydrology calculation into 40 segments, where each segment represents 10 canisters. Each segment corresponds to one stream tube, see section 11.6.10. To reduce the risk of unrealistic boundary effects, the results of calculations in a larger area are used to give boundary conditions to the calculation with HYDR11. The program HYDR11 is described in greater detail in section 11.6.5.

The choice of statistical description has been made after analysis of measurement data from Äspö. The measurement data have been obtained from water injection tests performed in boreholes with a packer straddle interval of 3 and 30 metres. The statistical analysis shows low spatial correlation /11.2-9/. The variance for $^{10}\log K$ was set to 2.3 in the calculations.

Radionuclide transport in the near field

A resistance model, TULL22, is used today for radionuclide transport. In future work, a compartment model, COMP23, is intended to be used to calculate transport from initially damaged canisters and TULL22 for initially intact canisters. The compartment model provides a good description of dynamic processes in the near field, which is desirable with early releases. The resistance model requires less computer power, at the same time as it describes sufficiently well the processes connected with release after long times.

Radionuclide transport in the near field in general, as well as the two models used, is discussed in greater detail in section 11.3. The near-field geometry is described in Chapter 5. Input data for the calculations in the near field are given in the Tables 12.3-1 and 12.3-2.

Radionuclide transport, far field

The modelling of radionuclide transport in the far field is based on the stream tube concept, see under hydrogeology calculations above. The results of the stream tube calculations in the hydrogeology model are used in the FARF31 module to calculate radionuclide transport in the far field. The far-field calculations result in nuclide-specific activity quantities that are released into the biosphere. Nuclide transport in the far field in general is discussed in section 11.4, the model FARF31 in section 11.4.4. Input data for the calculations in the far field are given in Table 12.3-3.

Table 12.3-1. Non-nuclide-specific data used for the near-field model TULL22

Canister height	4.833 m
Inside diameter of copper shell	0.950 m
Outside diameter of copper shell	1.050 m
Diameter of deposition hole	1.75 m
Distance between canister and tunnel floor	2.5 m
Distance between canisters	6 m
Distance between tunnels	25 m
Extent of disturbed zone	1 m
Porosity in disturbed zone near tunnel	10^{-4}
Effective diffusivity in rock	$3.2 \cdot 10^{-6}$ m^2/y
Factor for effective diffusivity in bentonite plug	10
Probability of initial canister damage	0.001
Hole diameter with initial canister damage	2.5 mm
Maximum helium pressure	29 MPa
Canister's void volume	1 m^3
Pitting factor	2
Corrosion depth at deposition	0.07 mm
Concentration of HS^- in groundwater	0.013 mol/m^3
Weight content of sulphur in bentonite	0.13%
Effective diffusivity of HS^- in bentonite	$3.2 \cdot 10^{-3}$ m^2/y
Distance between fractures in deposition holes	0.5 m
Fracture aperture width	0.05 mm
Diffusivity in water	$6.3 \cdot 10^{-2}$ m^2/y
Dry density of bentonite	1,600 kg/m^3
Porosity of bentonite	0.25
Time between canister damage and first release	0

Table 12.3-2. Element-specific data for near field. IRF (Instant Release Fraction) indicates the fraction of the inventory that is assumed to be available for immediate dissolution.

Element	Effective diffusivity in bentonite m^2/y	K_d in bentonite m^3/kg	Solubility mol/m^3	IRF
Am	$3.2 \cdot 10^{-3}$	3	$2 \cdot 10^{-5}$	
Cm	$3.2 \cdot 10^{-3}$	3	$2 \cdot 10^{-5}$	
Pu	$3.2 \cdot 10^{-3}$	50	$2 \cdot 10^{-5}$	
U	$3.2 \cdot 10^{-3}$	3	$2 \cdot 10^{-4}$	
Th	$3.2 \cdot 10^{-3}$	3	$2 \cdot 10^{-7}$	
Ra	0.79	0.5	$1 \cdot 10^{-3}$	
Np	$3.2 \cdot 10^{-3}$	3	$2 \cdot 10^{-6}$	
Pa	0.79	3	$3 \cdot 10^{-4}$	
C	$3.2 \cdot 10^{-3}$	0	high	0.5
Cl	$7.9 \cdot 10^{-5}$	0	high	0.1
Se	$3.2 \cdot 10^{-3}$	0.003	$1 \cdot 10^{-17}$	
Sr	0.79	0.01	high	0.05
Tc	$3.2 \cdot 10^{-3}$	0.1	$2 \cdot 10^{-5}$	
Pd	$3.2 \cdot 10^{-3}$	0.01	$2 \cdot 10^{-3}$	
I	$7.9 \cdot 10^{-5}$	0	high	0.1
Cs	0.79	0.05	high	0.05

Table 12.3-3. Data for far-field modelling.

Peclet number (dispersion)	2
Contact area	1,000 m^2/m^3 water
Effective diffusivity in rock	$3.2 \cdot 10^{-6}$ m^2/y
Diffusion porosity in rock matrix	0.005
Maximum penetration depth in rock	10 m

Element	K_d (m^3/kg) in rock	Element	K_d (m^3/kg) in rock
Cm	0.2	C	0.001
Am	0.2	Cl	0
Pu	0.2	Se	0.001
U	2	Sr	0.015
Th	2	Tc	1
Ra	0.15	Pd	0.001
Np	2	I	0
Pa	1	Cs	0.15

Biosphere modelling

The hydrology calculations show that most of the discharge or outflow of groundwater from the repository area takes place to the Baltic Sea, see section 12.3.2.1. It has therefore been assumed in the biosphere modelling that all release of radionuclides from the repository takes place to the Baltic Sea. The modelling is carried out outside of the model chain and results in nuclide-specific dose conversion factors for the Baltic Sea. The calculations are described in greater detail in section 11.5. The resulting dose conversion factors are given in Table 12.3-4. The calculated releases from the far field are converted to dose to critical group by simple multiplication and summation in a number of summation modules, represented in Figure 12.3-1 by the module SUM41. The consequences of the biosphere modelling of expected land uplift are discussed in section 12.3.3.

Table 12.3-4. Dose conversion factors from the biosphere modelling.

Nuclide	Factor Sv/Bq	Nuclide	Factor Sv/Bq
Pu-242	$3.7 \cdot 10^{-16}$	C-14	$3.5 \cdot 10^{-17}$
U-238	$3.8 \cdot 10^{-16}$	Cl-36	$2.0 \cdot 10^{-18}$
U-234	$4.2 \cdot 10^{-16}$	Se-79	$2.9 \cdot 10^{-16}$
Th-230	$1.9 \cdot 10^{-16}$	Sr-90	$8.2 \cdot 10^{-18}$
Ra-226	$7.2 \cdot 10^{-16}$	Tc-99	$1.8 \cdot 10^{-19}$
Am-241	$7.8 \cdot 10^{-16}$	Pd-107	$2.0 \cdot 10^{-19}$
Np-237	$5.9 \cdot 10^{-16}$	I-129	$4.1 \cdot 10^{-16}$
U-233	$4.3 \cdot 10^{-16}$	Cs-135	$1.1 \cdot 10^{-17}$
Th-229	$2.3 \cdot 10^{-15}$	Cs-137	$6.1 \cdot 10^{-17}$
Pu-239	$4.2 \cdot 10^{-16}$		
U-235	$4.0 \cdot 10^{-16}$		
Pa-231	$5.3 \cdot 10^{-14}$		

12.3.2 Results

Altogether, 100 realizations of the model chain were carried out for the canister defect scenario. The differences between the realizations are caused by the stochastic elements in the calculations. These are found in the calculation of the conductivity field (see below) and in the determination of number and position of initially damaged canisters.

The results of the calculations are presented first in the form of interim results from the hydrogeological calculations. Then the final results from the execution of the model chain are described in the form of dose and release curves for the biosphere.

12.3.2.1 Hydrogeological calculations

This section presents results of the calculations of

- conductivity field,
- stream tubes and discharge areas,
- water travel times and
- groundwater flows at repository level.

Conductivity field

To start with, the hydraulic conductivity of the model area is calculated. The calculation is based on data from hydrogeological investigations of the site. Figure 12.3-3a shows the conductivity in the fracture zones for a slice of the model area at the repository depth of 450 m as it has been calculated by HYDRASTAR. Equivalent results for the uppermost layer of the model area are shown in Figure 12.3-3b. The calculation is stochastic and each realization is an equally probable representation of the actual conductivity field. Interpreted conductivity data from boreholes have been used to guide the field through conditional simulation /11.2-23/.

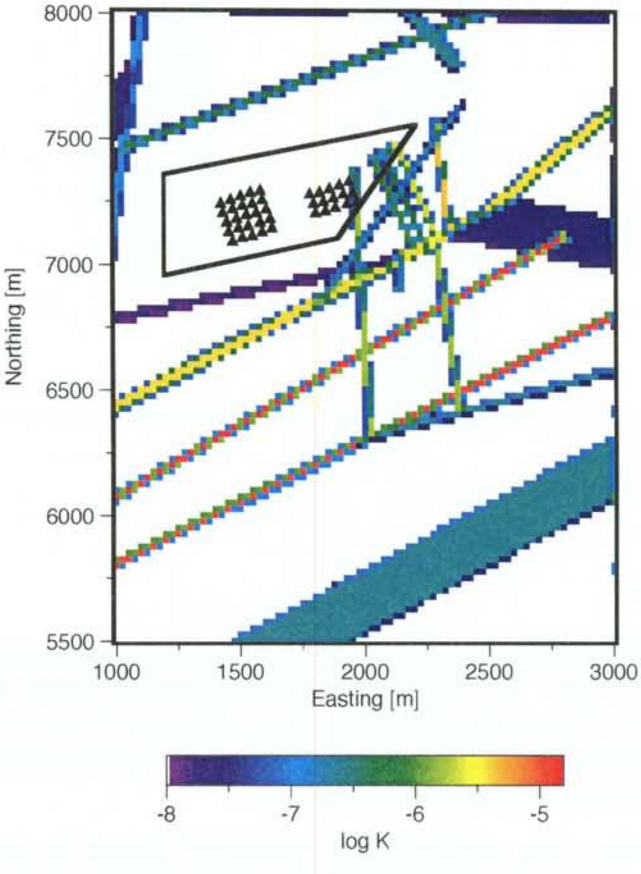


__Figure 12.3-3a.__ The conductivity field in the fracture zones at a depth of 450 m as it has been calculated by HYDR11. The locations of the fracture zones can be compared with the geological structural model given in Figure 6.4-1. Intact rock is shown white in the figure so that the fracture zones will stand out more clearly. Note that some repository segments are intersected by a fracture zone in the NNW direction. This zone is of less hydraulic importance, but would probably have been avoided anyway in a realistic case during placement of the deposition holes.

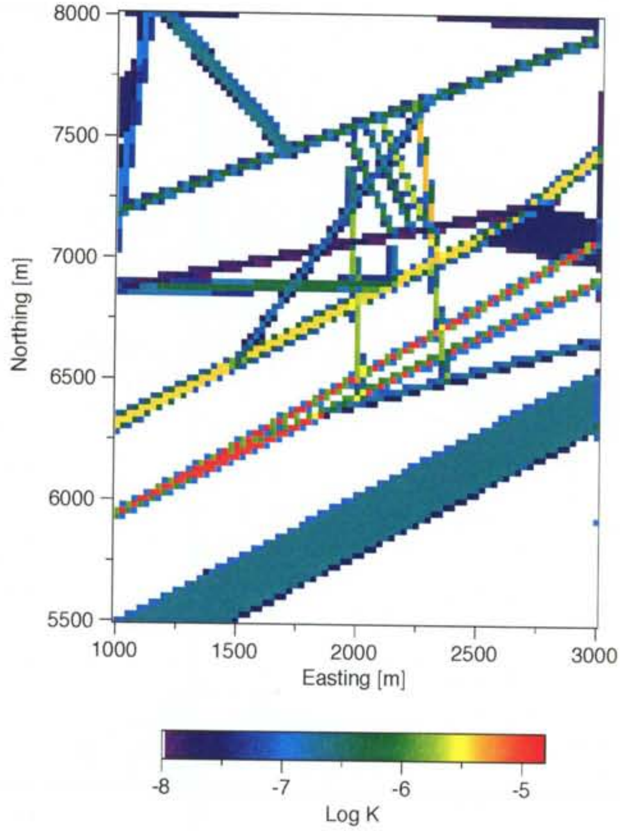


Figure 12.3-3b. The conductivity field in the fracture zones at the surface as it has been calculated by HYDR11. The locations of the fracture zones can be compared with the geological structural model given in Figure 6.4-1. Intact rock is shown white in the figure so that the fracture zones will stand out more clearly.

Pathlines and discharge areas

After the hydraulic conductivity of the rock in the model area has been determined for each realization, pathlines and travel times for the groundwater from the repository can be calculated. The repository area is then divided into 40 segments, each of which constitutes the starting point of a stream tube (see section 11.6.10). Together the stream tubes represent the total water flow from the repository area. Particle tracking is used to map the flow paths, whereby an imaginary particle transported with the water flow in a stream tube is followed from the repository until it leaves the model area. In this way, the flow can be modelled in both time and space. The projection of pathlines from a typical realization is shown in Figure 12.3-4.

The discharge points for particle tracks in the model area in question are shown in Figure 12.3-5. The figure shows the composite result from the 40 pathlines and 100 realizations, i.e. 40x100 discharge points in all. Different outcomes of the stochastic modelling of the conductivity distinguish the realizations from each other. Over 99 percent of the particles have reached the upper model boundary after 10,000 years. The remaining particles have travel times of more than 10,000 years, see below.

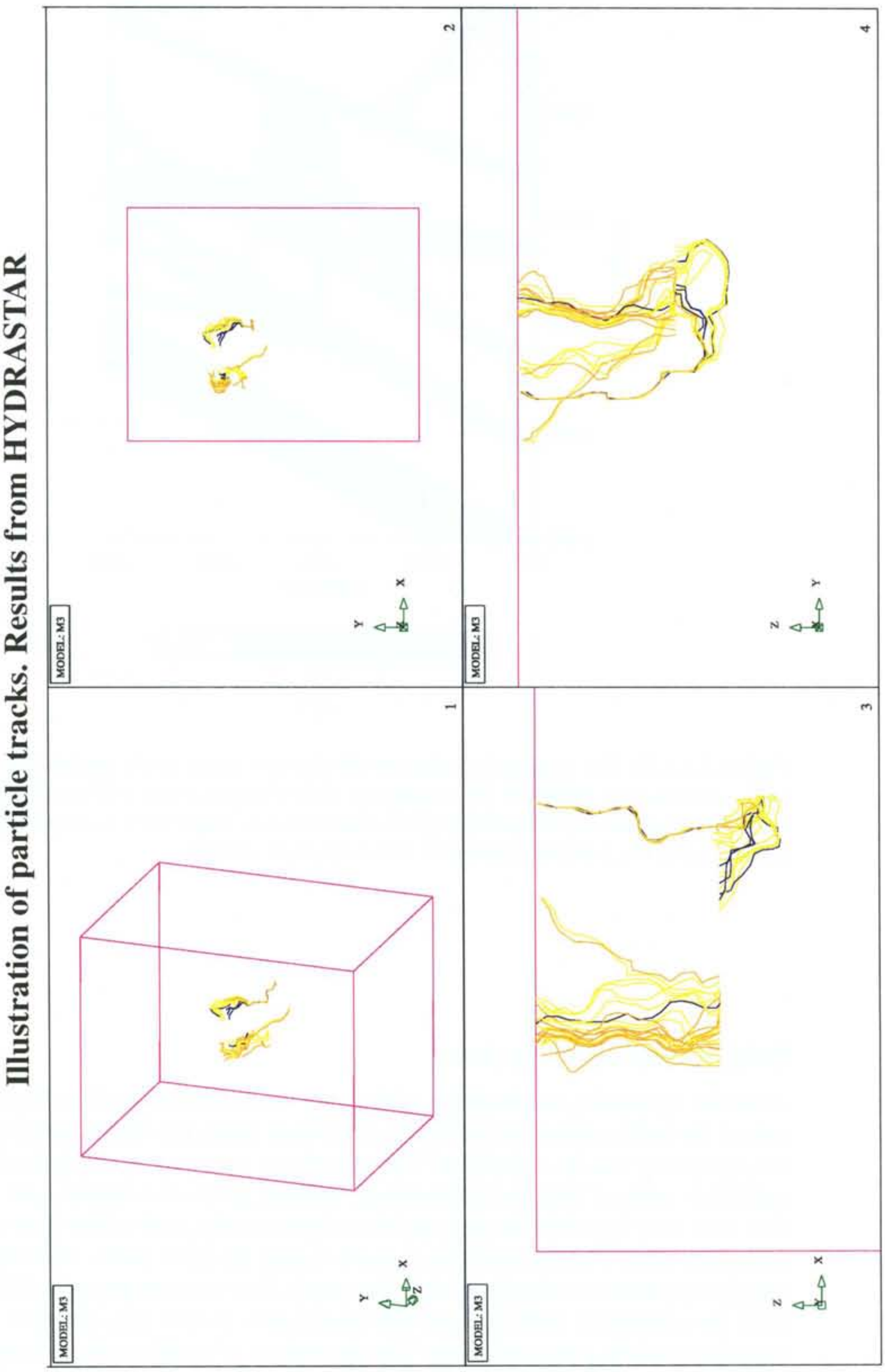


Figure 12.3-4. *Pathlines from the repository for a typical realization. The views are 1) to give a perspective, 2) from above, 3) from the southern side of the block and 4) from the eastern side of the block. See Figure 12.3-2 to orient the block in relation to the geography of Äspö. The colours indicate: red = water travel time <10 years; yellow = 10–100 years; blue = 100–1,000 years; violet = 1,000–10,000 years. The calculations are based on a flow porosity of 10^{-4}.*

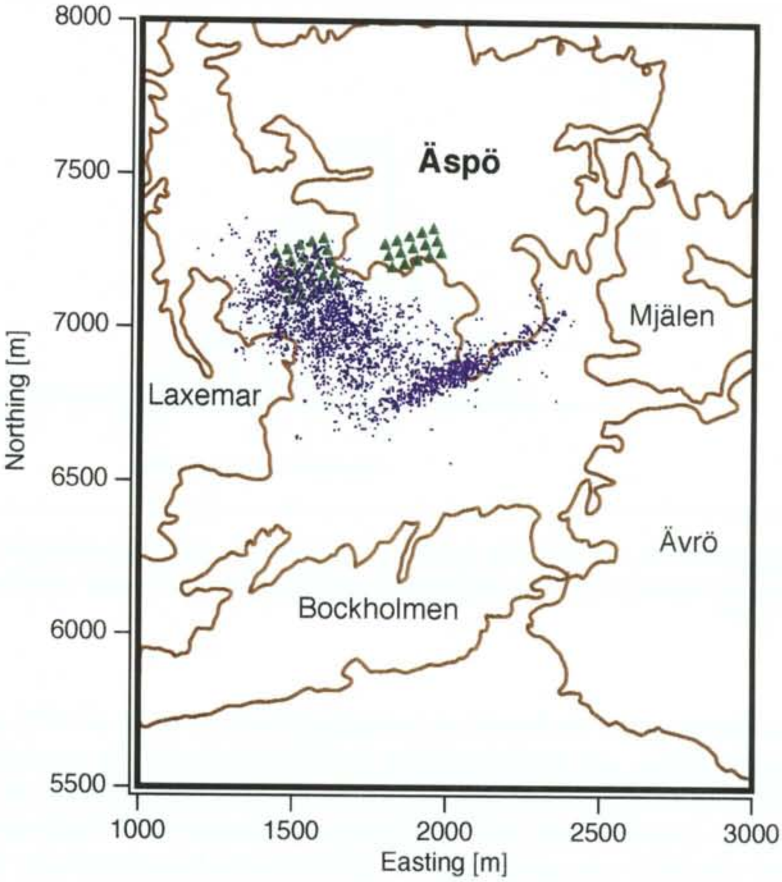


Figure 12.3-5. *Discharge area for the stream tubes from the repository. The picture shows the situation after 10,000 years. Each end position for the 4,000 particle tracks is marked by a point. The overwhelming majority of the stream tubes discharge into the bays of the Baltic Sea. The end points in the modelling are in the uppermost block of the model, 25 m below the surface. Particles with end points under land, within about 100 m of the shoreline, are expected in reality to discharge into the Baltic Sea as well, owing to the hydraulic conditions near the surface. Compare also with the fracture zones in Figure 12.3-3b.*

As is evident from Figure 12.3-5, the overwhelming majority of the stream tubes discharge into bays in the Baltic Sea. This supports the assumption that dose conversion factors for the Baltic Sea can be used for calculation of dose to critical group in today's biosphere. Expected land uplift will make these dose conversion factors unrealistic with time. This is discussed further in section 12.3.3.

Water travel times

The travel time for particles carried by the groundwater flow from the repository to the surface is calculated in HYDRASTAR. The results of 100 realizations of 40 stream tubes each are shown in Figure 12.3-6. The majority of the travel times lie between 10 and 100 years. The travel times then serve as a basis for the modelling of the flow of radionuclides in the far field.

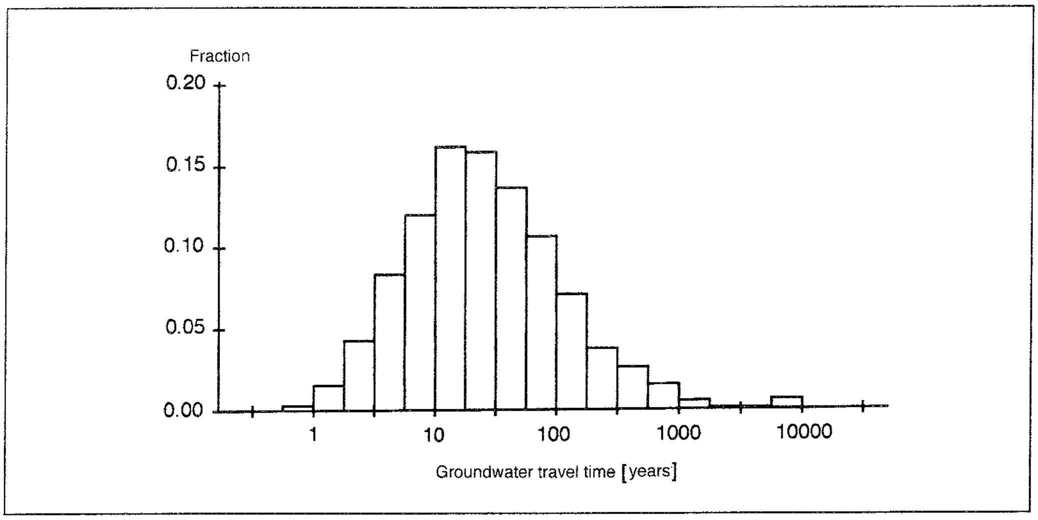


Figure 12.3-6. Distribution of travel times for groundwater transport of particles from the repository to the surface. The calculations are based on a flow porosity of 10^{-4}.

The calculations are based on a constant flow porosity of 10^{-4}, which is a great simplification, see further section 11.2. Furthermore, no account has been taken of the high salinities that exist in the deep groundwater on Äspö. The salinity causes density effects in the groundwater flow. Both of these simplifications are relatively great and give conservative (unrealistically short) travel times.

Groundwater flows at repository level

The groundwater flow at repository level is of importance for the leakage of radionuclides from the near field. These groundwater flows can be calculated with the aid of the conductivity field. Figure 12.3-7 shows the results of 100 realizations of the conductivity field. Typical flows lie around 1 litre/($m^2 \cdot y$)·

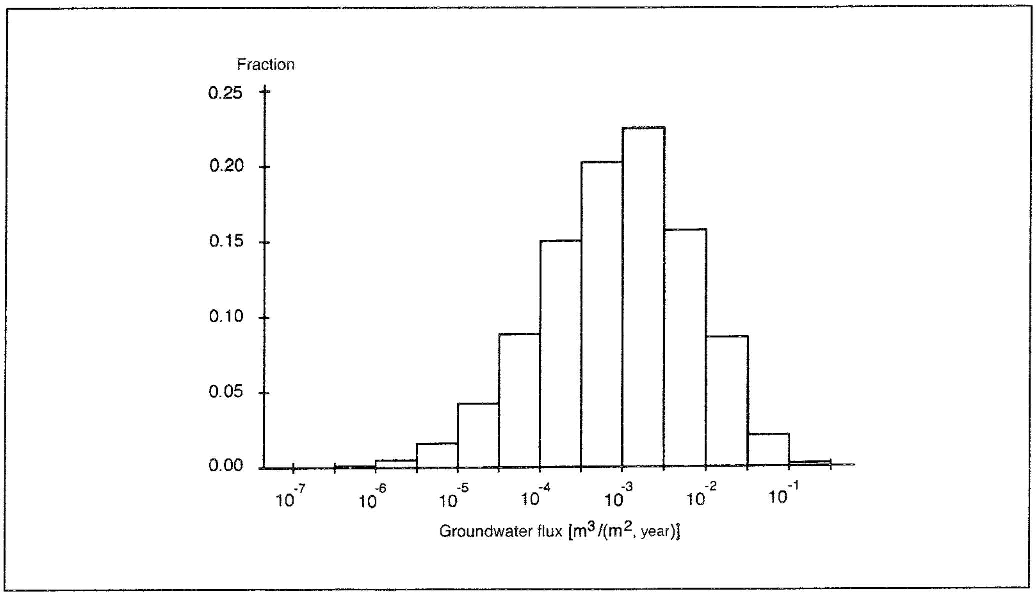


Figure 12.3-7. Distribution of the groundwater flow at repository level as calculated by HYDR11.

12.3.2.2 Dose and release calculations

This section presents results of calculations of

- annual dose as a function of time,
- maximum annual doses and
- releases of fission products and alpha-emitters.

Calculations of releases and doses to the biosphere are performed for each of the 100 realizations of the model chain. These calculations extend up to the time 10^7 years after closure of the repository.

As mentioned, three different causes of canister failure with subsequent leakage of radionuclides are modelled: an initial canister defect, corrosion of the copper canister, and canister failure as a result of inner pressure caused by helium buildup.

With the given premises for the calculations, corrosion is estimated to lead to canister penetration after 10^8 to 10^9 years, i.e. times that lie outside of the calculation. A critical pressure in the canisters is not expected to occur until after considerably longer times. The results of the model chain therefore only reflect releases caused by initial canister damages.

With the given premises (400 canisters, probability of initial canister damage 0.1%), around 2/3 of the realizations will not contain any damaged canister. The result for these realizations will therefore be that no radionuclides are released.

For the third of the realizations that contain one or more damaged canisters, release of radionuclides to the biosphere as a function of time is calculated. Annual dose to critical group (see section 11.5) in the biosphere as a function of time is also calculated.

Annual dose as a function of time

The total annual dose from all radionuclides has been calculated as a function of time. Figure 12.3-8 shows the results of two different realizations. The upper curve reaches its maximum value, about 10^{-6} Sv/y, after about 200 years. The lower curve, which is the result of another realization, has a maximum value of about 10^{-11} Sv/y. Such maximum values have been determined for the dose curves from each of the 100 realizations. The maximum values were sought in the limited time interval 0–10,000 years. The curves were then sorted according to their maximum values. The upper curve is designated the 95th percentile. This means that 95% of the realizations had a lower maximum value than this curve. The lower curve shows the results for the 75th percentile.

The doses are, especially for short times, unrealistically high above all for the following reasons:

- The near-field model used, TULL22, exaggerates releases from the near field for short times. The compartment model that will be used in future assessments for the near field gives more realistic results. See further section 11.3.3.
- The travel times through the far field are, as has already been observed in section 12.3.2.1, unrealistically short.

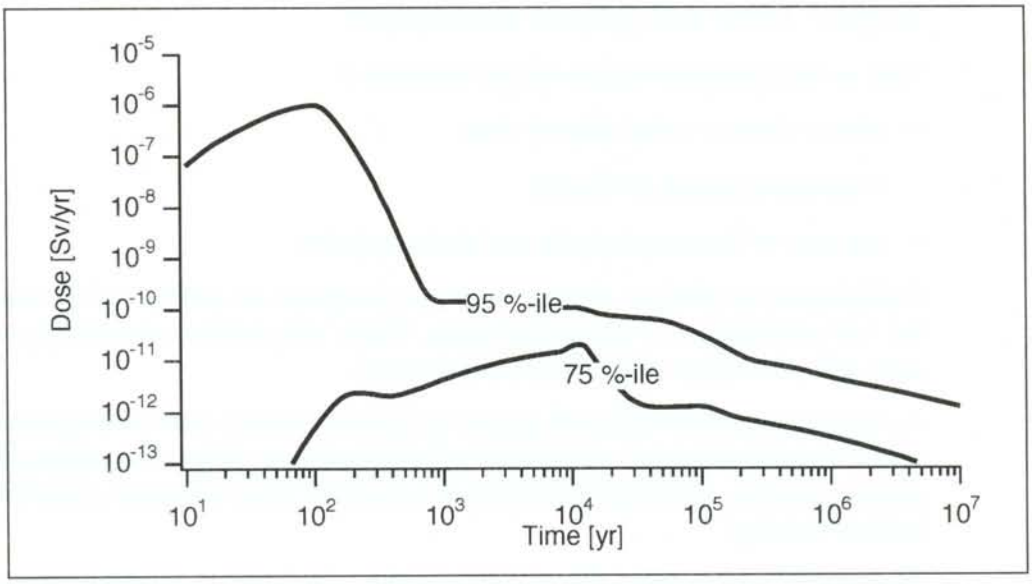


Figure 12.3-8. *Annual dose for critical group as a result of releases from defective canisters in a hypothetical repository underneath Äspö. The figure shows the realizations called the 75th and the 95th percentile, respectively. This means the following: For each realization, the size of the maximum annual dose in the time interval 0–10,000 years is noted. The realization called the 95th percentile has a higher maximum dose in the time interval than 95% of all realizations. The modelled repository size is about 10 percent of the planned deep repository. The result cannot be readily translated to an expected result for a full-size repository.*

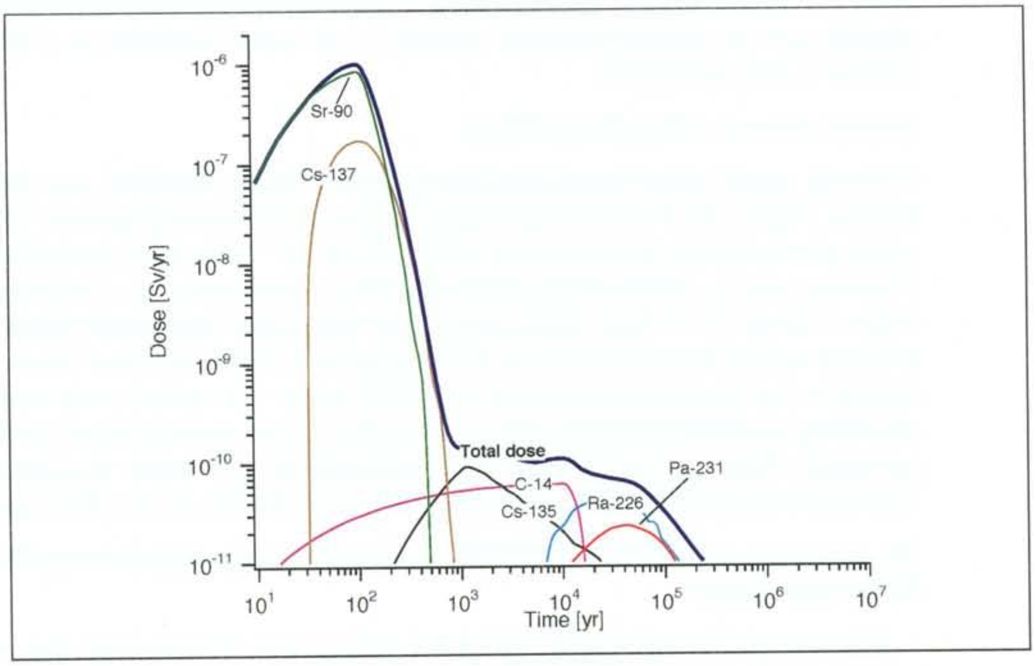


Figure 12.3-9. *Annual dose as a function of time for critical group as a result of releases from defective canisters. The realization shown in the figure is the 95th percentile with respect to maximum dose in the interval 0–10,000 years. The modelled repository size is about 10 percent of the planned deep repository. The result cannot be readily translated to an expected result for a full-size repository.*

Figure 12.3-9 shows the annual dose from the 95th percentile broken down into different radionuclides. The maximum annual dose from this realization is around 10^{-6} Sv/y and occurs after about 200 years. To put the results in perspective, it can be mentioned that the natural background radiation in Sweden is around 10^{-3} Sv/y. The maximum permitted dose burden from nuclear facilities in Sweden today is 10^{-4} Sv/y.

Maximum annual doses

The statistics for maximum doses are shown in Figure 12.3-10. The figure shows the distribution of maximum doses for the 33 realizations that resulted in releases.

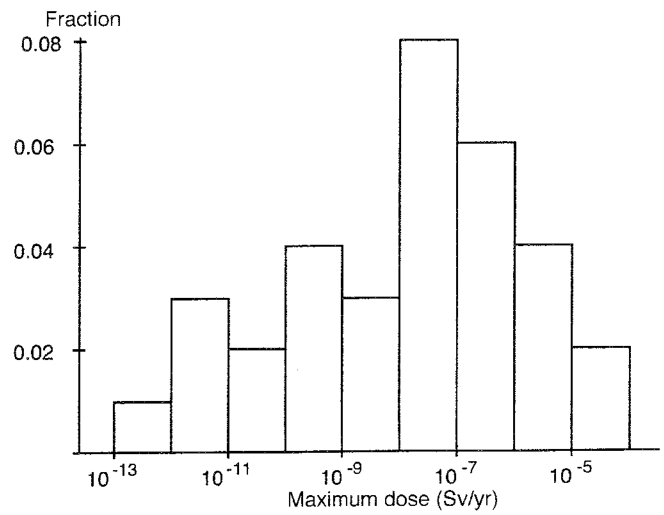


Figure 12.3-10. *Histogram for maximum dose up to 10,000 years for the 33 realizations that resulted in releases. The modelled repository size is about 10 percent of the planned repository. The result cannot be readily translated to an expected result for a full-size repository.*

Release of fission products and alpha-emitters

The uncertainties surrounding the appearance of the biosphere in the future are great after only relatively short (in this context) times. This means that the dose calculations, which are based on a model of the biosphere, also become progressively less certain. To provide a measure of the consequences of the canister defect scenario even after long periods of time, releases of radionuclides are therefore also reported expressed in activity, which is a measure that does not require for its calculation any knowledge of the appearance of the biosphere. The releases can then be compared with the natural flux of radionuclides.

The account is divided into long-lived fission products (beta-emitters) and alpha-emitters. The reason for the division is that a major dividing line runs between the two nuclide groups as far as half-lives and radiotoxicity are concerned. Alpha-emitters generally have longer half-lives and much higher radiotoxicity expressed as dose contribution per disintegration compared with beta-emitters. For times longer than about 10,000 years, it is therefore primarily of interest to study the releases of alpha-emitters to the biosphere.

Figure 12.3-11 shows releases of long-lived fission products (beta-emitters) as a function of time, expressed in Bq/y. The equivalent data for alpha-emitters is shown in Figure 12.3-12.

Both figures show the 95th percentile with respect to maximum releases over 10^7 years. Corresponding statistics for maximum releases over 10^7 years from all realizations are shown in Figures 12.3-13 and 12.3-14, respectively.

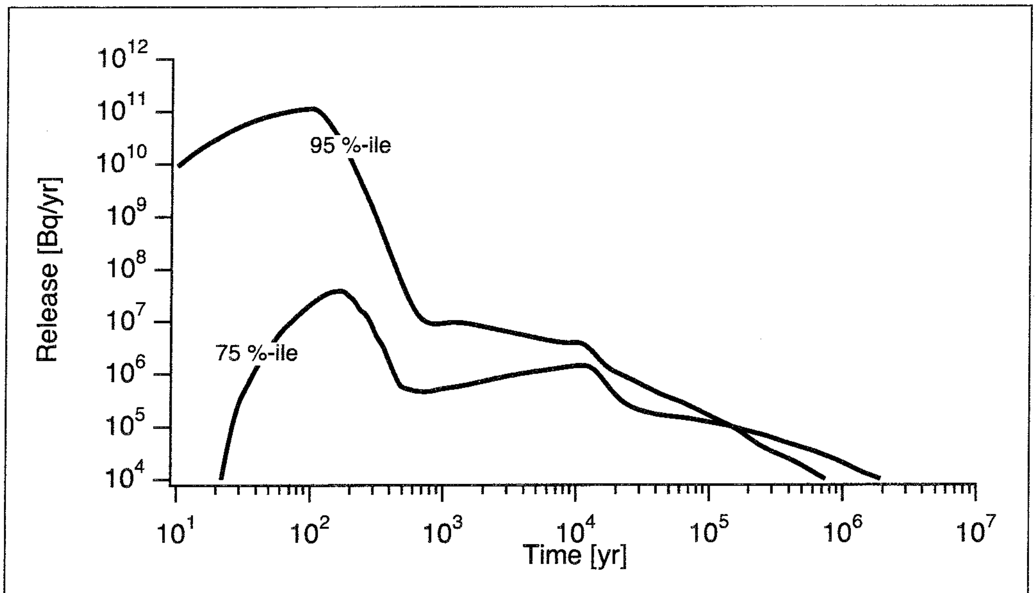


__Figure 12.3-11.__ Releases of fission products as a function of time. The figure shows the realizations corresponding to the 75th and 95th percentiles of the distributions of maximum release in the time interval 0–10^7 years. The modelled repository size is about 10 percent of the planned repository. The result cannot be readily translated to an expected result for a full-size repository.

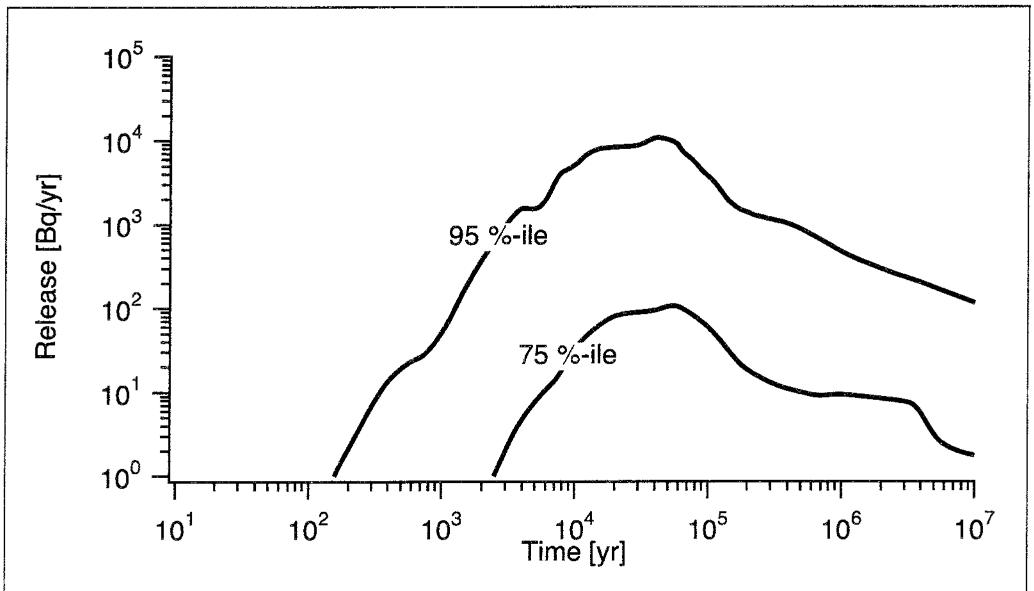


__Figure 12.3-12.__ Releases of alpha-emitters as a function of time. The figure shows the realizations corresponding to the 75th and 95th percentiles of the distributions of maximum release in the time interval 0–10^7 years. The modelled repository size is about 10 percent of the planned repository. The result cannot be readily translated to an expected result for a full-size repository.

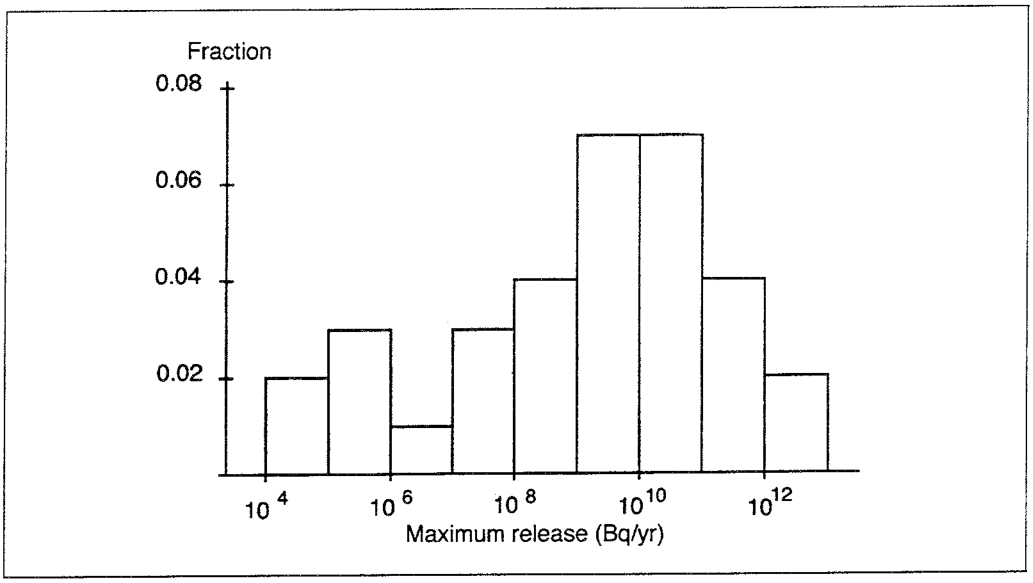


Figure 12.3-13. *Histogram of maximum release over 10^7 years for fission products. The modelled repository size is about 10 percent of the planned repository. The result cannot be readily translated to an expected result for a full-size repository.*

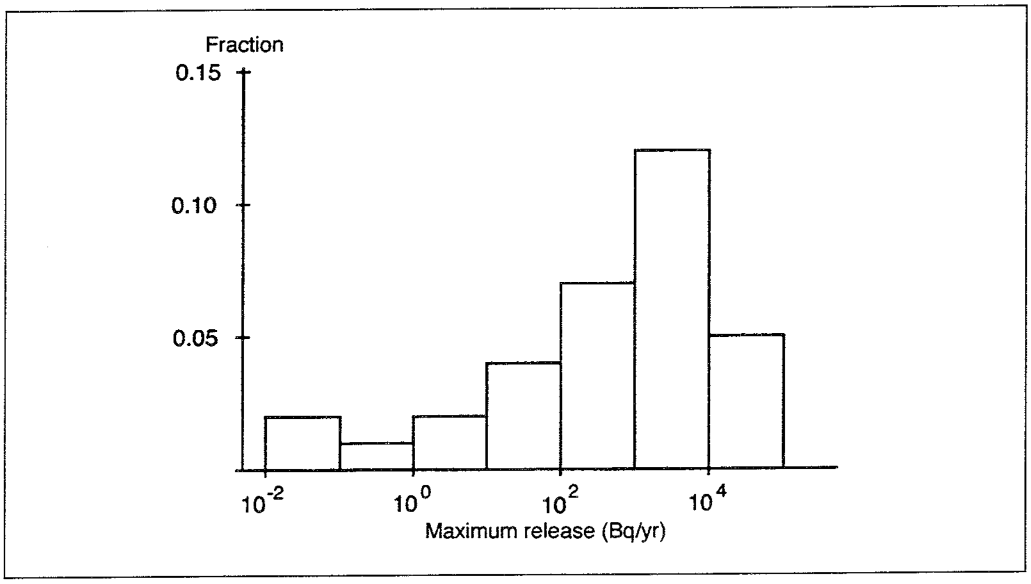


Figure 12.3-14. *Histogram of maximum release over 10^7 years for alpha-emitters. The modelled repository size is about 10 percent of the planned repository. The result cannot be readily translated to an expected result for a full-size repository.*

12.3.3 General variation discussions

The premises for the calculations of radionuclide transport were presented in section 12.3.1. The premises can be chosen in many different ways in an analysis of a canister defect scenario. This includes both choice of values for different parameters and choice of conceptual models. This section discusses briefly how several different variations in the modellings might affect the results of the calculations.

Conceptual variations in the near field

Table 12.3-5 shows what different variations in the conceptual model of the near field would entail. The presentation is qualitative and the consequences are rated in terms such as great, small, negligible, or in the form of powers of ten, etc. The idea for this accounting method is taken from EIS /12.3-2/.

Table 12.3-5. Variations in the premises for modelling of the near field

Premise and nature of premise, i.e. realistic (R), conservative (C) or simplification (S)	Alternative premise or model	Consequence of other model or premise
All canisters contain exactly the same type of fuel (amount, nuclide inventory and nuclide distribution) (S)	Different inventories in the canisters, based on actual expectations	Insignificant
Fuel dissolution is proportional to the α-dose rate, as in SKB 91 (C)	UO_2-solubility-controlled fuel dissolution	Several powers of ten lower leakage of matrix-bound radionuclides, influences the total dose to a lesser extent
Grain boundary inventory is released immediately when canister fails (C)	Kinetic model for grain boundary release	Lower maximum release of Cs and I. Limited importance.
No consideration given to transport resistance in Zircaloy tubes (C)	Model for penetration of and transport resistance in Zircaloy tubes, applicable primarily at short times (years)	Simple model is provided in NUCTRAN/COMP23, powers of ten lower leakage of e.g. Cs-137 and Sr-90
Amorphous solubility-limiting phases for actinides (C)	Crystalline solubility-limiting phases for actinides	Several powers of ten lower leakage of radioelements in question, but these seldom contribute to the total dose
High probability of initial through defects (0.1%) (C)	All canisters are intact for a very long time	This is dealt with in the normal scenario
Immediate water-filling of damaged canister (C)	Modelling of water-filling process	No release of short-lived nuclides
Corrosion gases inside the canister are neglected (C)	Modelling of corrosion processes inside the canister	One possible consequence is: no release of nuclides in aqueous solution
Copper shell offers transport resistance during long periods (R)	Degradation of copper shell	A couple of powers of ten higher release of solubility-limited nuclides, little importance for rapidly released nuclides and total dose
Simplified fracture geometry in deposition holes (parallel planes) (S)	Realistic fracture geometry from field data	Difficult to assess
Only diffusive transport in buffer (R)	Model for cracked buffer or leaning canister	Moderately increased release of nuclides
Simplified model of transport in the near field (compartment or resistor) (S)	Advanced numerical model	Existing models are often conservative, the difference is probably small

Land uplift

Land uplift caused by postglacial isostatic rebound is a known effect that is not taken into account in the reported model chain for the canister defect scenario. The effect could be treated as a variation of the premises for the model chain. What the result of such a variation might be is discussed here.

The rate of land uplift in the Äspö area is about 1 mm/y. This is predicted to result within a couple thousand years in the sea bays around Äspö becoming lakes or wetlands. This situation results in new flow conditions in the geosphere and new discharge areas. The situation therefore needs to be analyzed with new descriptions of both the geosphere and the biosphere.

For the geosphere, geohydrological calculations with NAMMU have been carried out for a situation with a land uplift of two metres /12.3-3/. The results are comparable to those obtained with today's situation. The water travel times are not changed appreciably. The transport pathways appear to be more collected in the fracture zones that occur nearest Äspö. Approximately 80% of the discharge takes place to the former sea bays south of Äspö, which have been transformed into wetlands. A smaller portion, about 10–15%, appears to go to lakes north and south of Äspö.

In the biosphere, other recipients have to be considered and thereby other transport models and exposure pathways. These have not been dealt with at present and quantitative dose conversion factors can therefore not be given. The level of knowledge for this case is relatively good, however, thanks in part to the studies that have been done of how lakes and sea bays age (eutrophy) and become cropland /12.3-4/. A follow-up for land uplift effects is in progress /12.3-5/. Dose factors for lake/cropland can be roughly estimated to be 100 times higher than those for the Baltic Sea.

For wetlands, especially peat, more extreme scenarios such as fertilizing gardens with peat could give individual doses a few orders of magnitude higher. But it is uncertain whether wetlands, and in particular peat, will be present in these former sea bays, since this would require undisturbed conditions for a long period of time.

12.3.4 Radionuclide transport in gas phase

As the inner steel container corrodes, hydrogen gas will be generated and then transported out, see section 10.4. This is not taken into consideration in the available near-field models. One possible effect of hydrogen gas production is discussed in this section, namely transport in the gas phase of radionuclides from the repository to the atmosphere.

The nuclides for which gas transport could have practical importance are C-14 (in CO_2 or CH_4) and Kr-85. Other nuclides such as I-129 and Tc-99 could also be transported with gas, but their concentrations are so low that the effect is without practical importance. To judge the consequences of radionuclide transport with gas, some simple rough calculations are made. The premises for the calculations are:

- Pulse release of 2.5% of the inventory of C-14 and the whole inventory of Kr-85 from **one** canister /12.3-6/. The release is assumed to take place 10 years after deposition. The inventory is calculated to be 38.6 GBq/tU for C-14 and 7.65 TBq/tU for Kr-85 in the year 2060. Each canister contains 1.2 tonnes of uranium, which gives a release of 1.2 GBq of C-14 and 9.2 TBq of Kr-85 with the above assumptions.
- The time for transport through the near and far field is neglected, i.e. the gas is assumed to be released immediately into the atmosphere.
- The release in the atmosphere is similar to releases from reprocessing plants. UNSCEAR describes this case and gives collective doses to the local and regional population of 0.4 mmanSv/GBq for C-14 and 7.4 μmanSv/TBq for Kr-85 /12.3-7/.

With these premises, the gas transport gives a total collective dose of 480 μmanSv for C-14 and 68 μman Sv for Kr-85 **locally and regionally**.

The calculations assume that all gas has been transported out ten years after deposition and that gas transport is not delayed in either the near or the far field. A more realistic assumption is that the travel time is on the order of 100 years. The half-lives of Kr-85 and C-14 are 10.7 and 5730 years, respectively. Neglecting the travel time therefore causes the dose for Kr-85 to be overestimated by several orders of magnitude, while the effect for C-14 is small.

The collective doses to the local and regional population are distributed over a large number of individuals and a long period of time, which means that the annual individual doses will be considerably less. This difference is particularly great for C-14.

The **global** collective doses are 85 and 0.0002 manSv/TBq, respectively, which is 200 and 30 times, respectively, the value for the regional and local population. Natural background radiation from C-14 is about 0.01 mSv/y /12.3-7/.

It would also be of interest to estimate the individual doses to which the gas release gives rise, particularly for C-14. This requires an analysis of the exposure situation for individuals and a dose factor. Application of ICRP's ALI and DAC values for carbon dioxide /12.3-8/ gives a dose factor of 3 nSv/Bq /12.3-9/ for C-14. Individual exposure situations have not been analyzed more closely, however.

12.4 HUMAN-INDUCED SCENARIOS

12.4.1 Introduction

When human-induced scenarios are to be described and analyzed, the following points should be discussed and considered:

- impact, i.e. how repository performance is influenced
- purpose, i.e. whether the impact is intentional or inadvertent

- knowledge, i.e. knowledge on the part of those who impact the repository, both regarding the repository specifically and the general level of knowledge
- intent, i.e. whether those who impact the repository have good or bad intentions

Scenarios caused by man differ from other scenarios in that predictions regarding them cannot be made solely on a scientific basis. In addition to scientific calculations and judgements, the analysis includes speculation on future politico-societal development, ethical and moral considerations and behavioural judgements.

There are a large number of human activities which could conceivably affect the safety of the repository. The list of such activities can never be made complete. Speculation on the future is always more or less subjective. Future generations' knowledge of the waste and their reasons for coming into contact with it cannot be predicted exactly. The scope of future impacts on the rock and/or groundwater is difficult to foresee. In view of this, predictions of future human impact on the repository should be regarded as illustrations of conceivable situations. The choice of human-induced scenarios must be based on knowledge of the repository, its design and function. Assumptions of future conditions are reported in connection with estimations of probabilities, times and consequences.

The repository and its barriers are designed to withstand any stresses to which they can reasonably be expected to be subjected. Human-induced scenarios have been implicitly or explicitly considered in designing the disposal concept. Some examples of requirements where human activities have been considered in connection with repository design are:

- Repository performance shall not be dependent on monitoring or institutional control.
- Monitoring and institutional control shall not be impossible to carry out.
- The waste shall be retrievable.
- The backfill shall be such that it cannot easily be forced.
- Materials (e.g. in the canister) in engineered barriers should not be so rare or valuable that they can induce human intrusion.

Human activities have also been considered in formulating repository site selection criteria. Examples are:

- The repository shall be situated in commonly occurring rock that does not contain valuable minerals.
- The depth shall be great enough to make human intrusion unlikely.

Discussion and analysis of human-induced scenarios takes place against the background of repository design, site selection and the four discussion points: impact, purpose, knowledge and intent.

The work presented is based on /12.5-1, 2/. The work of identifying and analyzing human activities that can impact on the repository system continues.

12.4.2 Discussion points – impact, purpose, knowledge and intent

Impact

Human activities can be divided into the following two categories with respect to impact:

- direct impact via intrusion on the radioactive waste
- indirect impact on the performance of the barriers

If impact is direct, one or more barriers have been completely removed. If impact is indirect, the performance of one or more barriers has been affected.

Examples of human activities that can indirectly affect the repository are global warming, spreading of waste, intensive agriculture and extensive construction activities.

The repository's barriers are affected by altered conditions in the rock, see Table 9.4-2, or in the groundwater, see Table 9.4-3. The scenarios can be analyzed by means of qualitative discussion supported by separate studies and/or rough calculations. In some cases, analysis may employ special calculation cases within the type defect scenario. In scenarios with construction in rock, the premises for the site model (see Chapter 6) may have to be revised. The biosphere models (see section 11.5) may also have to be reworked.

In cases with direct intrusion in the repository, analysis takes the form of a standard risk assessment. Such an analysis consists of two parts. In the first, the probability or expected frequency of impact is estimated. In the second, an assessment of the consequences is made. Risk is a weighing-together of probability and consequences. Risk can be expressed as collective or individual risk. How risks in conjunction with intrusion into the repository should be dealt with is being discussed both in Sweden and internationally /12.4-2/.

Speculative assumptions are made in connection with both estimation of probabilities and assessment of consequences. The assumptions are always more or less subjective. The quantitative results of the analysis are therefore to be regarded as illustrations. It is important to stipulate the grounds for the assumptions and to justify them. What importance the assumptions have for the final results must also be shown. Another way is to avoid speculation about the future, but this is very difficult in practice.

Purpose

Human-induced scenarios can be divided into the following categories with regard to purpose:

- intentional
- inadvertent

Purpose is closely associated with knowledge of the repository and the waste.

If intrusion is intentional, it is assumed that the intruders have all the necessary available information. In general, intentional intrusion in the repository is not dealt with in the safety assessment; however, an exception is made for sabotage. Responsibility for a voluntary action is considered to rest with those who carry out the action. It is their responsibility to analyze the risks involved with their action.

If intrusion is inadvertent, it is assumed that the intruders do not have sufficient knowledge of the repository and its contents. Knowledge of the repository may have been lost entirely. It is also conceivable the intruders have found something in the rock but don't quite know what it is, and/or how it got there, and/or why it is there. Their reason for finding something in the rock may be due to their having some fragments of information on the repository, that the site has been marked or that something has been detected. How the find has been made affects the assumptions concerning expected knowledge of the waste.

Inadvertent intrusion in the repository is dealt with in the safety assessment. Observe that inadvertent intrusion can be either direct or indirect.

Knowledge

Both the probability of human impact and its consequences are linked to the knowledge future generations have of the waste and the repository. In a safety assessment, it is not enough to be aware of whether the knowledge is good or non-existent. Who possesses the knowledge, how it is interpreted and how it is used must also be taken into consideration. Methods for conservation of information about the repository are being discussed internationally /12.4-1/. Since the consequences of a possible intrusion depend on when in the future the intrusion takes place, the length of time for which the information might be preserved is also being discussed.

Besides knowledge of the repository and the waste, assumptions must be made concerning the general state of knowledge among the future generation which is conceived as impacting the repository. What knowledge do they possess about radioactive materials and their effects on humans, animals and plants?

Intent

The consequences of an intrusion depend on whether the intruder is acting with malicious or benign intent. It is up to the designer of the repository to build a system that is invulnerable to sabotage.

12.4.3 Example – drilling through a canister

An example of a human-induced scenario is that someone drilling deep boreholes unintentionally drills through a canister. The analysis is performed as a risk assessment. The risk assessment begins with a discussion of the probability of an accidental event, in this case drilling through the canister, occurring. Then the possible consequences of the accidental event, expressed as the expected number of fatalities, are discussed.

In drilling through the canister, it is assumed that the drillers have drilled up to and through the deposited fuel as well. All barriers have been short-circuited and humans are exposed directly to the waste. Only the consequences for the intruders are dealt with here. It is emphasized that this is just an example. The purpose is to illustrate a conceivable case and a possible analysis method. In an actual safety assessment, the existence of one or more drilled-through canisters in the repository can be more interesting. The consequences in the case where the waste is left in the natural environment or somewhere else where humans could be exposed should also be examined.

Assessment based on repository performance and present-day knowledge

Getting at the waste by drilling requires that the technology to drill deep boreholes is known. The drilling technology used can affect the pathways of transfer of radionuclides to man. An analysis should at least cover the consequences given the technology generally used today. In a more comprehensive analysis, the consequences given other known/conceivable drilling methods can also be analyzed. Given that the technology is known, it is assumed that it is applied, and thereby that drilling of deep boreholes occurs.

This example deals with cored drilling, the bore dust is assumed to be removed by water. The drilling method, dimensions of the borehole and drill core are those that are most common today /12.4-2/. Drilling carried out for an exploratory purpose is always cored. Removing the bore dust via water gives a better working environment than letting it be carried away via the air.

Purpose

Only inadvertent intrusion is dealt with. There are a number of possible reasons why someone might accidentally drill through a canister. The repository site may, for example, have been identified via markers, access to information or detection, and this may have aroused curiosity and a desire to investigate further by drilling. Totally random drilling through a canister is also conceivable in connection with, for example, extraction of geothermal energy or a general geological survey.

Knowledge

If the technology for drilling is known, it is assumed that the general knowledge level is roughly equivalent to that of today.

It is deemed likely that retrieved drill cores are examined, and that the radioactive material is identified. After the radioactivity has been identified, it is assumed that the drill cores are handled in a suitable fashion. The example deals only with the consequences for those who perform the drilling.

Intent

In view of the design of the repository and the efforts required to bring up the radioactive waste to the surface, it is deemed highly unlikely that anyone would drill through a canister with malicious intent.

12.4.3.1 Assessment of probabilities of drilling through a canister

The probability of drilling through a canister, given that deep boreholes are being drilled, can be divided into three factors:

- the probability that those who are drilling have no knowledge of the repository and its contents
- the probability that the borehole happens to land within the repository area
- the probability of drilling through a canister if the borehole is within the repository area

The three probabilities are assumed to be independent of each other, only their order of magnitude is estimated.

The probability that the information on the repository has been lost changes with time. How long conservation of information can be taken into account in the safety assessment is being discussed internationally. Time periods of 100–500 years have been mentioned /12.4-1/. The probability that a borehole will happen to land within the repository area can be estimated as ratio of the surface area of the repository to the surface area of Sweden. The probability of drilling through a canister if the borehole is within the repository area can be estimated geometrically. The estimated probabilities are presented in Table 12.4-1.

Table 12.4-1. The estimated probability of drilling through a canister divided into three factors.

Event	Probability
The drillers have no knowledge of the repository and its contents	0 for first 200 years 1 after 500 years Linear increase from 0 to 1 during period between 200 and 500 years
The borehole lands within the repository area	10^{-5}
The borehole penetrates a canister	10^{-2}

12.4.3.2 Estimation of doses and their consequences

Only the consequences for the intruders are dealt with here. The consequences of drilling through a canister are dependent on several factors, e.g.:

- the quantity of waste and its content of radionuclides
- how humans come into contact with the waste
- exposure time
- drilling method
- location of humans in relation to the retrieved waste
- distance of humans to the retrieved waste

Speculative assumptions must be made on each point. The importance of the assumptions should be evaluated and quantified.

The quantity of waste depends on the length and diameter of the borehole and how large a portion of the drill core consists of fuel. The possible quantity is estimated based on the assumed drilling method. The proportion of fuel in the drill core is calculated based on the design of the fuel assemblies and the canister. Radioactivity at the time of drilling is calculated based on the radioactivity content of the fuel when it was placed in the repository and the half-lives of the nuclides /12.4-2/.

Given a specific quantity and type of radionuclides, the radiation dose is dependent on the exposure time and the exposure pathway. Here, external irradiation is assumed to be the only exposure pathway. The exposure time is set at one hour. The radiation is assumed to come from the drill core, the ground surface contaminated by bore dust and contaminated clothing. Which parts of the body are exposed to the radiation is also of importance. In the example, it is assumed that the radiation is evenly distributed over the body. The dose factors used correspond to effective total body doses. Assumptions of a geometric nature are shown in Figure 12.4-1.

Dose from drill core

Of the assumptions made, the distance between the drill core and the exposed individual is of the greatest importance for the radiation dose from the drill core. Variation of this distance can lead to changes in the dose by a factor of 100. Within distances deemed reasonable for longer-term exposure, i.e. one to five metres, the dose varies by a factor of 10. A doubling of the thickness of the drill core causes the dose to increase by a factor of 4. If a two-metre long drill core is taken up in one long portion instead of in two one-metre portions,

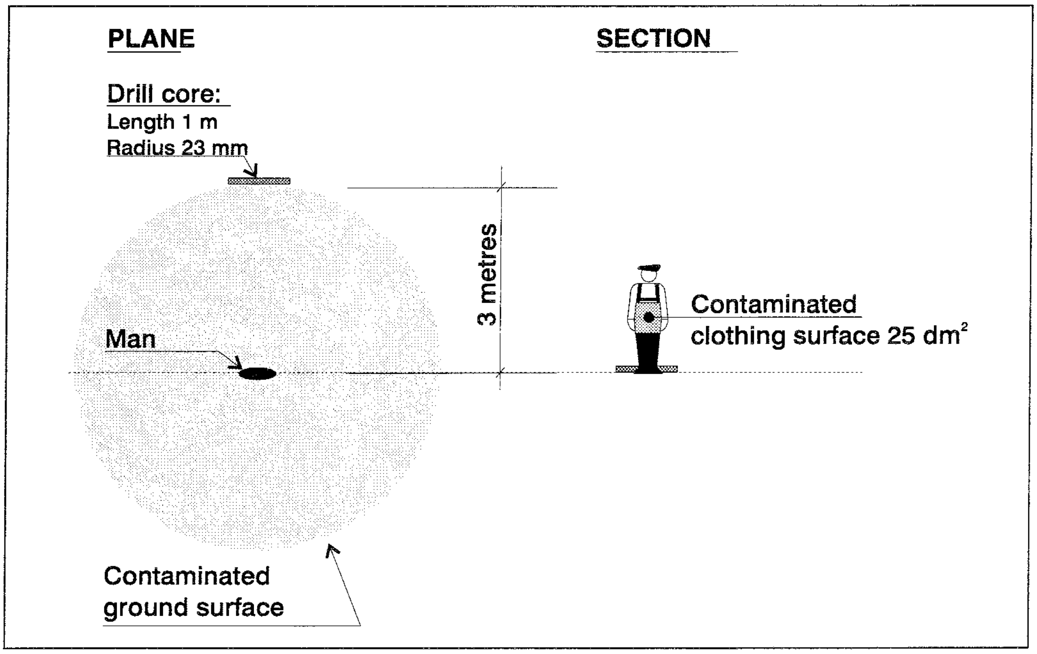


Figure 12.4-1. *The position of the exposed human in relation to the radiation sources. The person is exposed to radiation at a distance of 3 metres from a 1 metre-long drill core with a radius of 23 mm. The ground surface contaminated by bore dust is assumed to be a circle with a radius of 3 m. The person is assumed to be exposed to the radiation at a height of 1 metre to the whole body. The dust contaminated clothing surface is assumed to be 25 dm^2.*

the dose increases by a factor of 2 for a person standing perpendicular to the centre of the core.

Dose from contaminated ground

The quantity of drill dust depends on the gap between the borehole and the drill core. The proportion of fuel in the dust on the ground depends on how large a surface the cuttings have been spread over, and where the person is located in relation to this surface. The dose is greatest if the person is in the middle of and close to the surface. If the surface is assumed to be circular, the dose at a height of 2 metres is about 1/10th of the dose at a height of one centimetre. If the radius of the circle increases from one to five metres, the dose decreases by a factor of 5.

Dose from contaminated clothing

The radiation dose from the person's clothing depends on how much of the dust have adhered to his clothing. The area and the shape of the surface as well as the thickness of the dust layer are factors of importance. All these factors are difficult to predict. It is assumed here that the dust layer is 0.1 mm thick. The radiation at a point about 1 cm from the middle of a circular surface with an area of 25 dm^2 has been calculated. The calculated radiation is assumed to be evenly distributed over the person's body. An idea of how big the surface is in relation to the body is given in Figure 12.4-1.

Contributions of the different radiation sources to the total radiation dose

The contributions of the different radiation sources to the total radiation dose is shown in Figure 12.4-2. The dose has been calculated for three different distances to the retrieved drill core. The radius of the contaminated surface has been set equal to the distance to the drill core.

The dose from the clothing dominates in the chosen case, see Figure 12.4-1. Approximately 82% of the dose comes from the person's clothing, 11% from the core and 7% from the drill dust on the ground. The dose from the core and the drill dust comes exclusively from gamma radiation. Of the dose from the contaminated clothing, a couple of ten-thousandths comes from beta radiation and the rest from gamma radiation.

If the clothing dose is kept unchanged, the total radiation dose increases by a factor of 2 if the distance to the drill core and the radius of the contaminated surface is reduced to one metre. The dose is directly proportional to the exposure time and the quantity of radionuclides. The exposure time has been set equal to 1 hour here. The quantity of radionuclides has been calculated based on the design of the fuel and the canister and the activity when the repository was closed. In view of the importance of the different assumptions, it is subjectively assumed that the estimated dose varies by a factor of around 10. If all changes are of the same kind, i.e. either increases or decreases, it is assumed (also subjectively) that maximum errors on the order of a factor of 100 can arise. The errors concern only the described drilling method and the assumed exposure pathways.

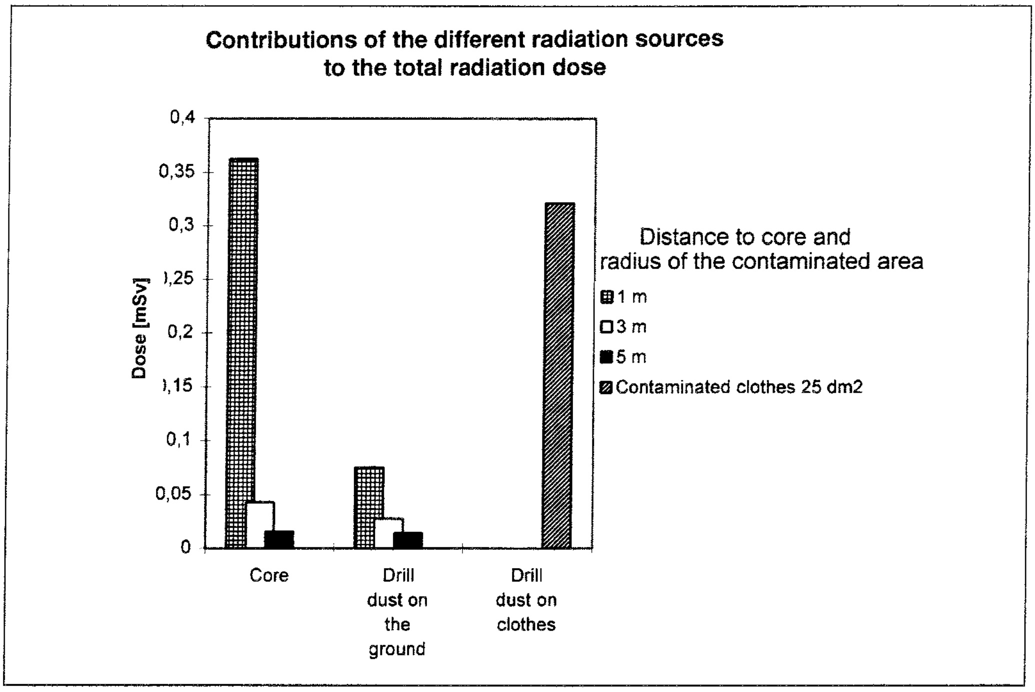


Figure 12.4-2. *The contributions of the different radiation sources to the total dose. The length of the drill core is one metre, the exposure time 1 hour and the waste is taken up 300 years after closure of the repository.*

Influence of age of waste on radiation dose

The radioactivity of the waste declines with time. The rate of decline is greatest in the beginning, becoming slower and slower as time goes on. Figure 12.4-3 shows how the dose in the described case declines with time. For the purpose of illustration, an approximately 10 times higher dose, the internationally accepted annual dose of 1 mSv /12.4-3/, and the annual dose from the repository accepted by the Swedish authorities of 0.1 mSv /2.3-1/ have been plotted as well.

As seen in the figure, the dose resulting from drilling through the canister cannot be neglected until thousands of years after closure of the repository. A weighing-together of the consequence and the probability of the event occurring, i.e. a risk assessment, is necessary.

In a risk assessment, it is not the dose in itself but its consequences in the form of expected number of fatalities that is interesting. The probability of contracting fatal cancer has been estimated to be 0.05 per Sv /12.4-3/. The relationship between dose and probability of contracting cancer is assumed to be linear. The accepted risk would thus be $5 \cdot 10^{-5}$ fatality if the international limit value of 1 mSv is used and $5 \cdot 10^{-6}$ fatality if the Swedish limit value of 0.1 mSv is used. Figure 12.4-4 shows the probability of contracting fatal cancer after one hour's exposure for the person shown in Figure 12.4-1. The probability is depicted as a function of the time between repository closure and intrusion.

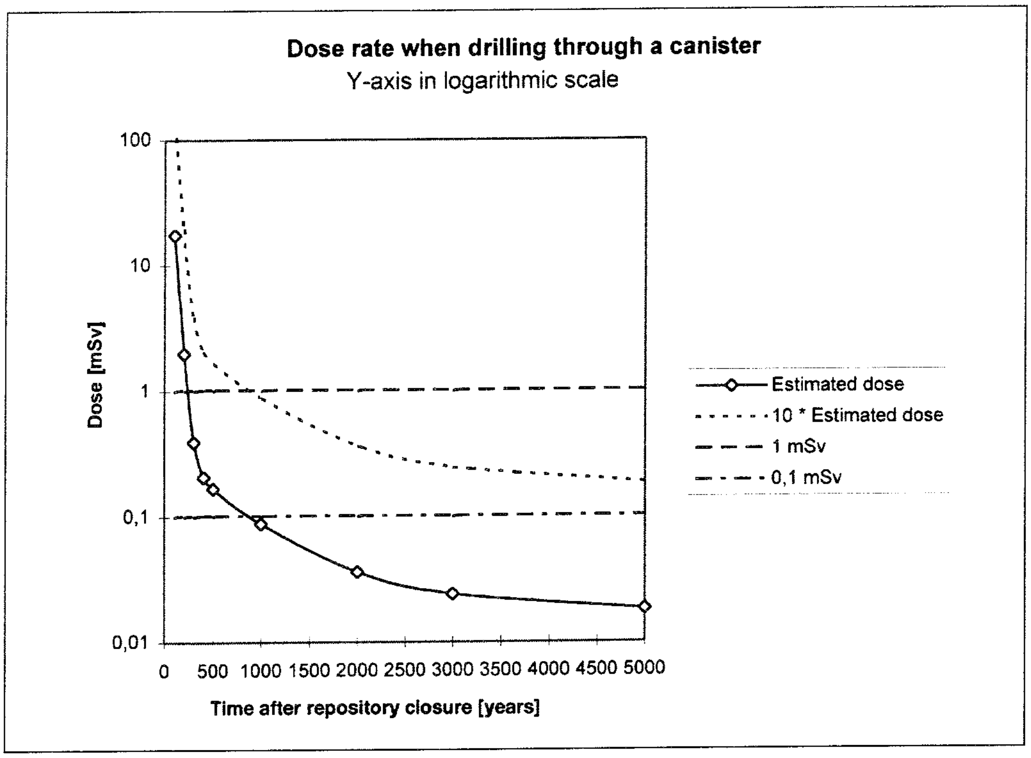


Figure 12.4-3. *Dose as a function of time. The exposure time is 1 hour. The radiation sources and the man's position in relation to them are shown in Figure 12.4-1.*

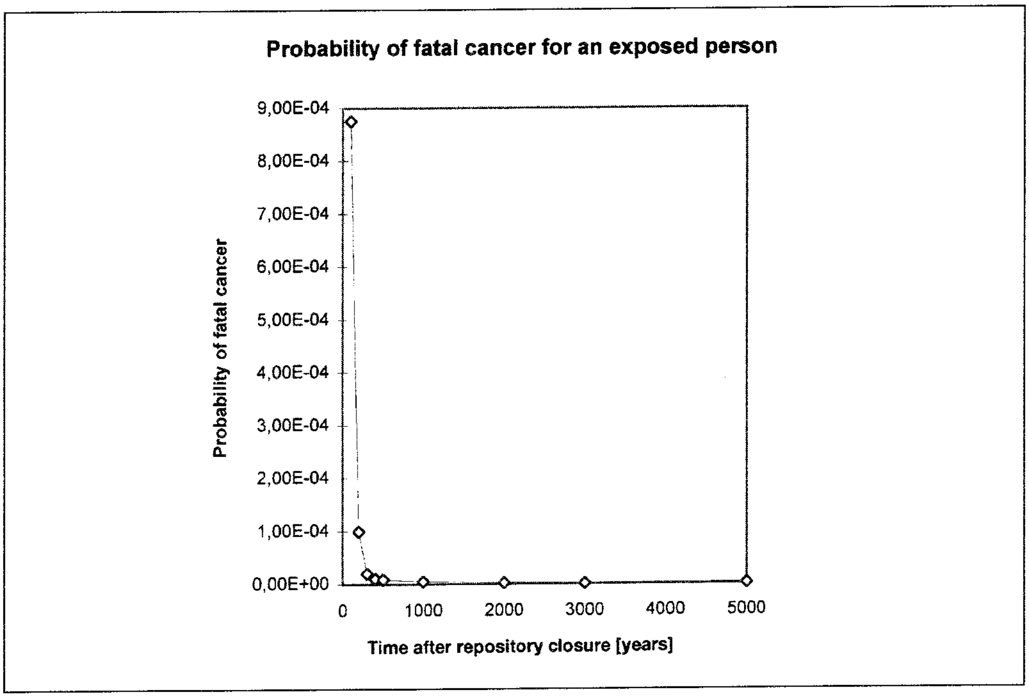


Figure 12.4-4. *Probability of contracting fatal cancer for a person who is exposed to the radiation from the sources illustrated in Figure 12.4-2 for one hour's time. The probability is depicted as a function of the time between repository closure and intrusion.*

12.4.3.3 Assessment of risk

Risk is a product of the probability an event will occur and its consequence in the form of expected number of fatalities. An assessment of the probability of inadvertently drilling through a canister while drilling deep boreholes has been made in section 12.4.3.1. Table 12.4-1 presents the estimated probability divided into three factors. A description of conceivable consequences has been done for a case where a man is exposed during the course of 1 hour to the dose at a distance of 3 metres from a 1 m long drill core with a radius of 23 mm, the dose at a height of one metre from a contaminated circular dust area with a radius of 3 m, and the dose from a contaminated clothing surface of 25 dm^2. Figure 12.4-3 shows how the dose declines as a function of the time to the intrusion. What remains to do is to weigh together probability and consequence in terms of expected number of fatalities. The risk for persons who drill deep boreholes to die due to drilling through a canister containing radioactive waste is shown in Figure 12.4-5. The figure shows the risk per drilled hole as a function of the time between repository closure and intrusion.

The maximum risk per drilled hole is on the order of 10^{-12} fatality and occurs 500 years after closure of the repository, i.e. at a time when knowledge of the repository is assumed to be limited. If a limit value of 0.1 mSv is used, the accepted risk is $5 \cdot 10^{-6}$ fatality. The assumptions made can thus be changed by a factor on the order of 1,000,000 without the limit value being exceeded. The uncertainties in the assumptions made should thereby be considered to be covered.

Annual loss is often spoken of in discussions of risk. In order to make a judgement of the risk on an annual basis, an estimate must be made of how many boreholes a person drills per year. Since speculations about the future should be avoided wherever possible, it is merely concluded that about 1,000,000 boreholes can be drilled per year without the accepted risk being exceeded (given the described premises).

As is evident from Figure 12.4-5, the assumptions that knowledge of the repository has been preserved are of great importance for the risk. To illustrate this, the risk in a case where the positive effects of knowledge preservation have not been assumed is shown in Figure 12.4-6.

At a time 100 years after repository closure, the maximum risk differs from the accepted risk by a factor on the order of 10,000 if the positive effects of knowledge preservation are neglected.

Some conclusions that can be drawn from the assessment are:

- Drilling through a canister is an event that cannot be neglected in view of the estimated consequences.
- The probability that someone will drill through a canister by mistake is judged to be small, on the order of 10^{-7}.
- It is nevertheless good if knowledge of the repository can be preserved so that the probability of inadvertently drilling through a canister can be further reduced.

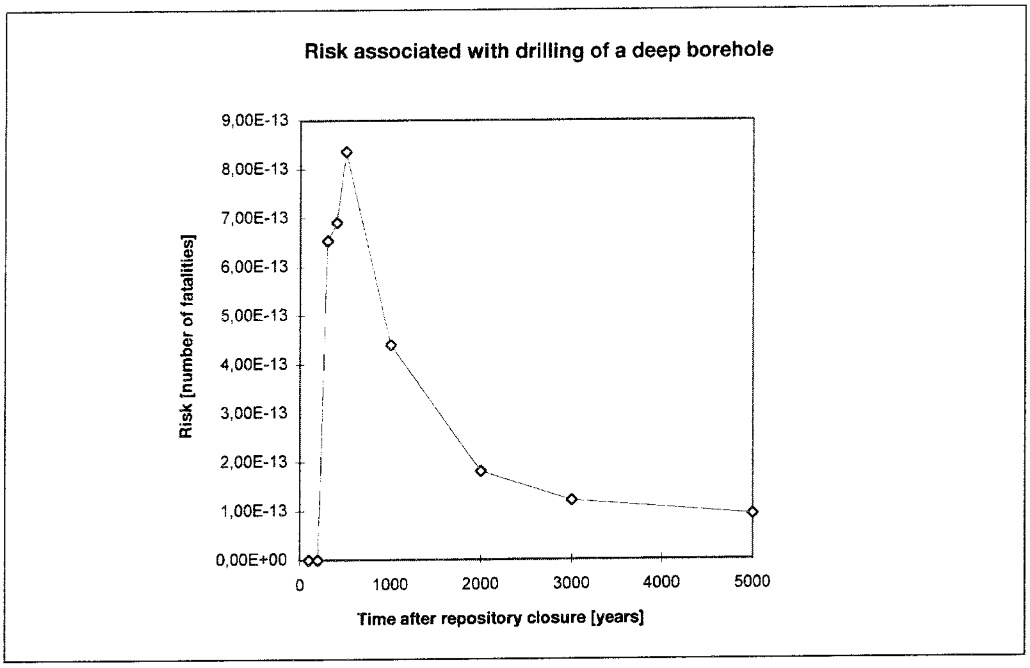


Figure 12.4-5. Risk for persons who drill deep boreholes as a function of time.

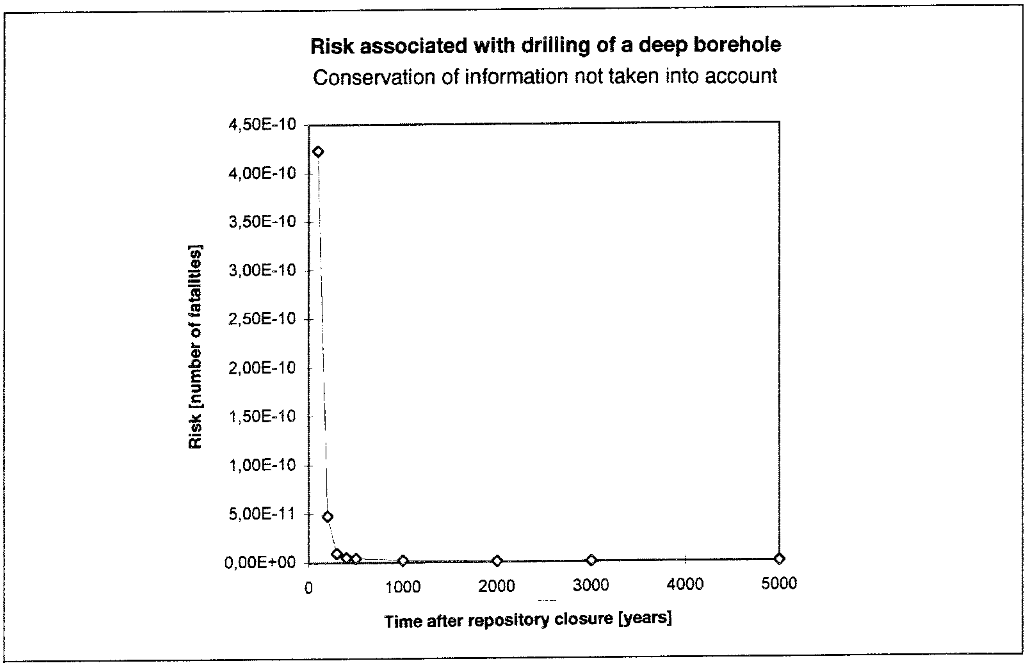


Figure 12.4-6. Risk for persons who drill deep boreholes as a function of time. It is assumed that knowledge of the repository is lost at the same instant the repository is closed.

12.5 GLACIATION SCENARIOS

12.5.1 Background

The radioactivity in the repository will decay during time spans on the time scale of Quaternary geology, i.e. hundreds of thousands of years. The Baltic Shield is believed on scientific grounds to have been subjected to several large-scale glaciations during about 85% of the past 700,000 years. At the time of its maximum extent about 18,000 years ago, the most recent glaciation, the Weichsel, reached all the way down to Poland and northern Germany.

During the next 100,000 years, long cold periods with permafrost and an inland ice sheet are expected in Sweden. This is true regardless of possible climate change due to the greenhouse effect (global warming). During a glacial cycle, the biosphere and geosphere are subjected to a number of far-reaching changes. The changes during the cycle affect the rock and the groundwater flow in different ways and to different extents. The question is: What changes can be expected and how might they affect the performance of the repository? Does any particularly dangerous evolution or situation exist? Could radiological hazards arise for humans, animals and vegetation that lay claim to terrestrial and aquatic areas after the next glaciation? If so, what are the consequences?

The parts of SKB's paleogeohydrological research programme that handle methods and models for formulating glaciation scenarios are described in this section. The section differs from the others in Chapter 12 since it does not contain any analysis of the consequences of the scenario, but describes a methodology that is under development for future safety reports.

12.5.2 Climate change, glaciations and their effects

The forces that have driven climate change in the past are assumed to act in the future as well. Compelling empirical evidence suggests that global climate change is ultimately due to changes in insolation, i.e. incoming solar radiation. These changes are in turn due to changes in the earth's orbit around the sun, according to the Milankovitch theory /12.5-3/. The climate change in northern Europe leads to repeated glaciations.

The primary effects of a colder climate are vast areas with permafrost, altered precipitation patterns, growing glaciers and greatly lowered (100–150 metres) sea levels. During the deglaciation (melting) stage, the oceans rise once again. During the different phases of a glacial cycle, the boundary conditions for groundwater flow change. Ice and meltwater subject the earth's crust to mechanical stresses. During the most recent glaciation, the surface of the ground was eroded several metres and enormous quantities of soil were displaced and restratified. Climate change is thus transmitted via mechanical stresses and altered conditions of groundwater flow to the environs of the repository and the repository system.

Four climate based process domains can be identified during a glacial cycle of the Quaternary Period:

- glacial domain
- permafrost domain
- marine domain
- interglacial domain

Within the different climate based process domains there are regimes and subregimes with different conditions for groundwater flow. The regimes are related to the ice and its extent in time and space. The subregimes are related to environmental factors and ambient conditions. Prevailing climate based process domains, regimes and subregimes determine the conditions for groundwater flow. Viewed from the repository, they can be regarded as external conditions for the groundwater flow. Table 12.5-1 presents the climate based process domains, their regimes and subregimes. Figure 12.5-1 shows the glacial domain and its regimes. The purpose of the figure is to clarify what is meant by the terms climate based process domains, regimes and subregimes. For a more complete description of the conditions within the different climate based process domains, see /12.5-1/.

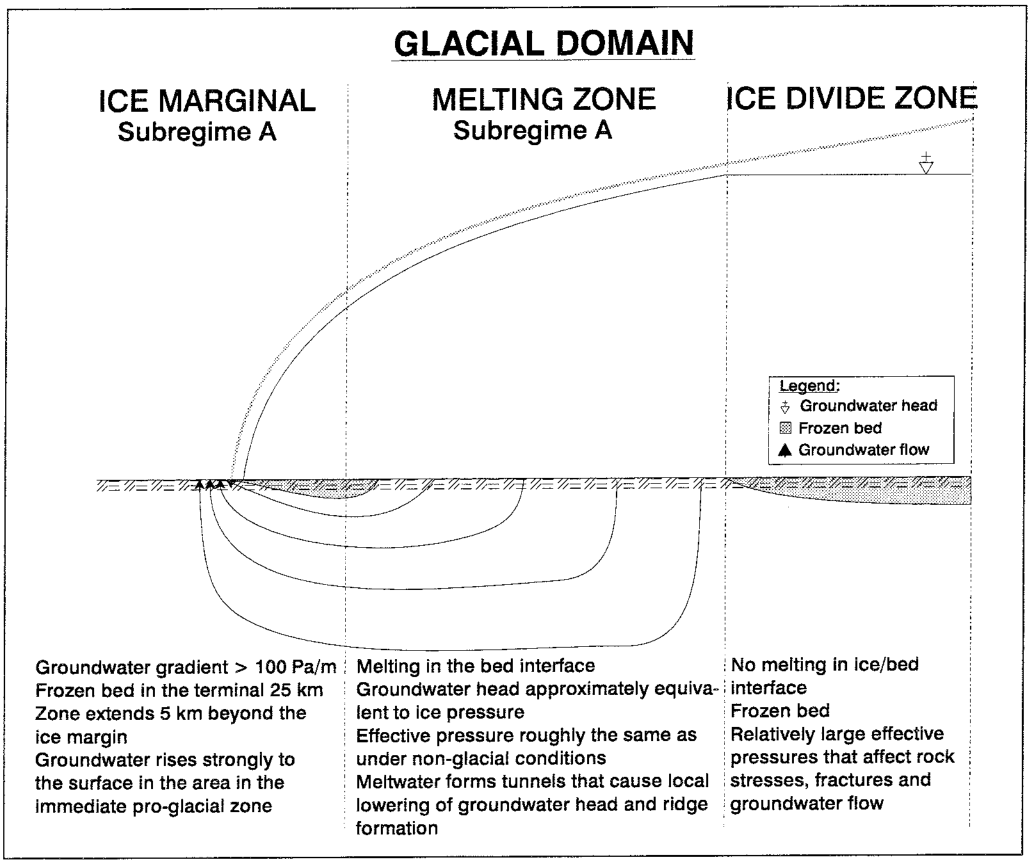


Figure 12.5-1. *The glacial climate based process domain. The boundary conditions for groundwater flow are determined by the presence of the ice. The more exact conditions are determined by prevailing regimes and subregimes.*

Table 12.5-1. The different climate based process domains, their regimes and subregimes. A glaciation cycle can be regarded as a series of climate-controlled process states.

Climate based process domain	Regime	Subregime
Glacial domain	Ice divide zone (shown in Figure 12.5-1)	–
	Melting zone	A No impact of ice tunnel, ridge (shown in Figure 12.5-1) B Impact of ice tunnel, ridge
	Ice marginal:	A No proglacial permafrost and above the marine limit (shown in Figure 12.5-1) B Proglacial permafrost and above marine limit C Below the marine limit
Permafrost	Continuous Discontinuous	
Marine domain	Glaciation stage – global lowering of sea level	A Lowering of sea level by about 70 m during relatively long period, rise of about 40 m during relatively short period before transition to glacial domain B Lowering by about 30 m at roughly the same rate as in A, rise of about 30 m during relatively short period before transition to glacial domain C Little lowering at roughly the same rate as in A, rise of about 30 m during relatively short period prior to transition to glacial domain D More or less no lowering of sea level, rise of about 30 m during relatively short period prior to transition to glacial domain
	Deglaciation stage – global rise of sea level	A About 70 m higher sea level than before glaciation, a period with relatively rapid lowering is followed by a period with slowly rising sea levels B About 150 m higher sea level than before glaciation, a period with rapid lowering of the sea level is followed by lowering at a progressively slower rate C As B, but the sea level lowering proceeds faster D About 300 m higher sea level than before glaciation, a period with very rapid lowering of the sea level is followed by lowering at a slower rate
	Interstadial – little lowering of sea level	
Interglacial domain		

In reality, the evolution of a glacial cycle is the result of a continuous change in climate. To clarify characteristics and important phases of the evolution, a glacial cycle can be viewed as a series of climate based process domains. This approach also simplifies and clarifies the safety assessment procedure.

The geographic extent and scope of a glaciation vary in both time and space. In judging the effects of a glaciation on a deep repository, it is necessary to weigh in not only the different climate based process domains, but also the physiographic and geological settings of the repository site. Interesting geological characteristics are fractures and fracture zones, hydraulic properties, rock stresses and chemical conditions. The physiographic characteristics that are of interest are topography, location in relation to the sea and latitude. The last two factors are crucial in determining the prevailing regime and subregime given a certain climate based process domain. With respect to glaciations, Sweden can be divided into three physiographic zones: mountain zone, hill and plateau zone and coastal plain zone. The different zones, their characteristics and extent are presented in Table 12.5-2 and Figure 12.5-2.

Table 12.5-2. Division of Sweden into zones with respect to glaciations.

Geographic zone	Characteristics
Mountain zone	Relief in excess of 500 m Located in western part of the country In a colder climate, small glaciers coalesce to ice caps which gradually grow to ice sheet size The zone of final ice sheet decay at the end of a glacial period In interglacial periods, steep slopes may create large groundwater gradients Groundwater gradients are reduced during glacial periods
Hill and plateau zone	Relief in the order of 100–200 m Plateaus with a height above sea level of 200 m (today) Extends in over the foot of the mountain zone Except for the northernmost inland part, is not situated below sea level in the postglacial period The slopes are local and have little influence on large-scale groundwater flow patterns
Coastal plain zone	Relief of less than 50 m Situated below sea level in the postglacial period

During a glaciation, the earth's crust is subjected to stresses from a several kilometres thick ice sheet. When the ice has melted, the downwarped crust strives to resume its original form. Increased seismic activity can therefore be expected immediately after a land area has become ice-free. Earthquakes can be triggered by movements in already existing and very old, large faults or fracture zones in the bedrock /12.5-2/. The effects of the earthquakes are examined in special earthquake scenarios.

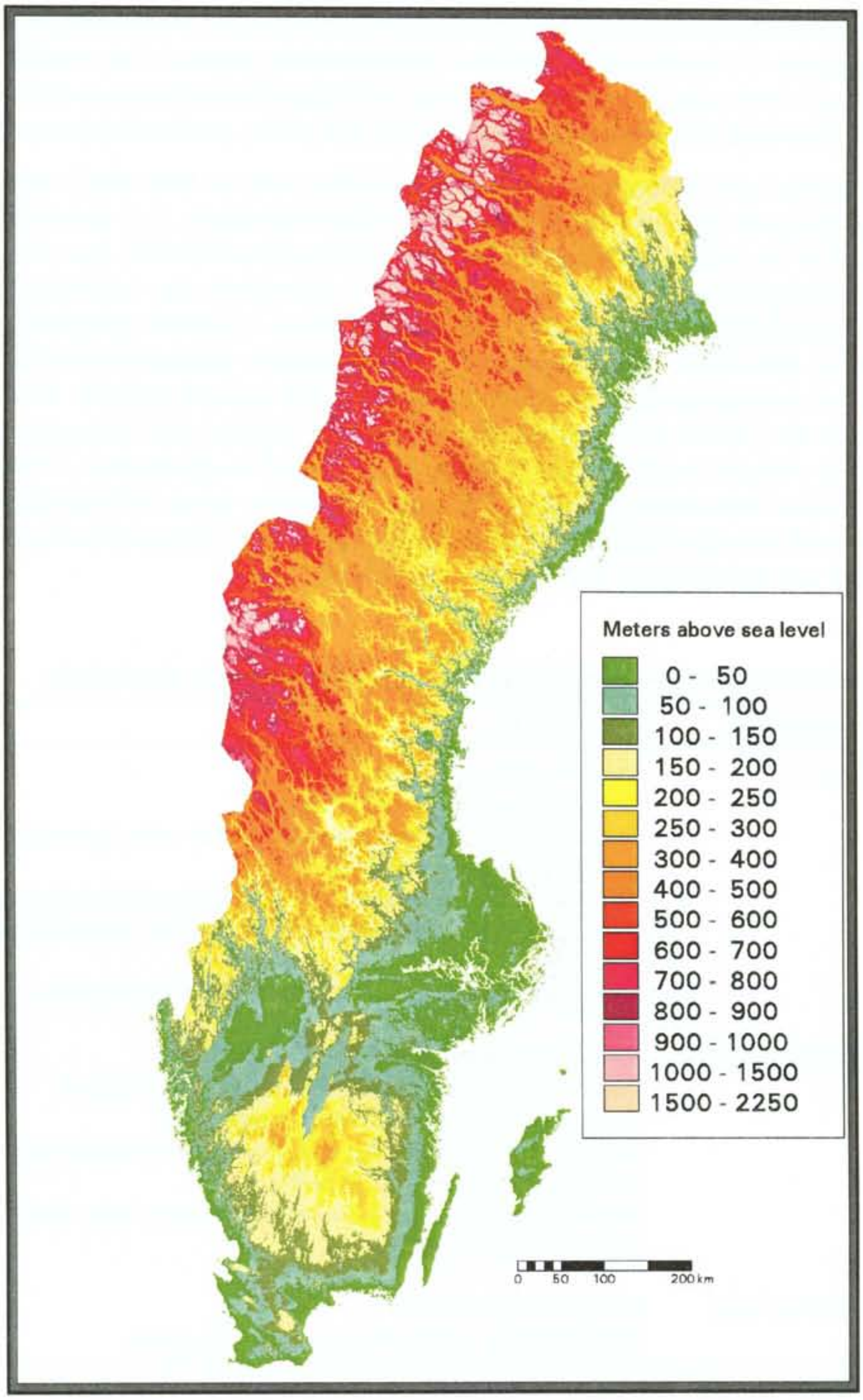


Figure 12.5-2. *With respect to glaciations, Sweden can be divided into three physiographic zones: mountain zone, hill and plateau zone and coastal plain zone. The borderline between the mountain zone and the hill and plateau zone runs at approximately 500–600 m above present-day sea level. The borderline between the hill and plateau zone and the coastal zone runs at the highest coastline during the glaciation cycle. The coastal plain zone corresponds roughly to the green areas on the map. In the far south, however, the borderline between the coastal zone and the high-plateau zone runs at a height of about 50 m above present-day sea level, and in the north at about 300 m. Besides height above sea level, the topography is important in determining the zonal classification, see Table 12.5-2.*

12.5.3 Modelling of glaciations and their effects

The biosphere and geosphere are subjected to great changes during a glaciation cycle. Estimations of the scope of the changes and the mechanisms that transmit them to the repository need to be done in a safety assessment. An important part of the assessment is judging the scope and sequence of the different climate based process domains. SKB has therefore commissioned the development of a glaciation model with which the extent of the ice sheet during a glaciation cycle can be calculated.

The development of the glaciation model comprises part of a paleohydrogeological research programme. The purpose is to identify and increase understanding of thermo-hydro-mechanical and chemically coupled processes, and to develop reliable predictive calculation models for the described processes. The idea is that it should be possible to check modelled and calculated phenomena and data against geological observations.

Input data

The input data to the glaciation model are the climate changes that drive the glaciation cycle, and physiographic and geological boundary conditions. Examples of boundary conditions are the form and topography of the lithosphere and geothermal heat flux.

The first step in describing a future glaciation is to describe how the climate will change. The description of the climate changes can be done with the aid of different methods /12.5-1/. The input data to the glaciation model are thus output data from some form of modelling of climate change.

Time-dependent driving forces for regional climate change have been identified within the paleohydrogeological research programme and related to predictable changes in the earth's orbit around the sun and insolation. Assuming that the relationships found are the same in the future as in the past, future climate changes can be described with the aid of statistical extrapolation. To check the validity of the model, the climate in the past has been described, and the calculated changes have been compared with geological data.

Glaciation model

In greatly simplified terms, it can be said that with knowledge of the driving climate changes, a mass balance can be set up and the quantity of ice that has formed or melted can be calculated. The calculated ice mass is then added or subtracted, depending on the physiographic and geological boundary conditions and the properties of the ice sheet. The glaciation model solves a thermo-mechanically coupled problem. Besides growth of the ice, it calculates the extent of permafrost and the temperature conditions underneath the ice sheet.

Output data

The output data from the glaciation model are temperature zones, extent of ice sheet and meltwater flow. As mentioned previously, the idea is that it should be possible to check the validity of the model against geological observations. The direction of growth of the ice can be verified via erratic transport and striations in the rock. In order to translate other output data into verifiable quantities, it is necessary to model groundwater flow and/or rock stresses and loads. Hydrogeological and rock-mechanics models can therefore be coupled to the glaciation model.

Models coupled to the glaciation model

A simple geohydrological model can be used in order to describe the groundwater flow during a glaciation cycle. The groundwater flow is calculated according to Darcy's law, where the rock is described as a homogeneous porous medium. In general, permeability declines with depth, even though the presence of fracture zones leads to local variations. In the model coupled to the glaciation model, the uppermost 6,000 metres of the earth's crust is divided into three layers with constant hydraulic conductivity. This type of geohydrological modelling is considered to be acceptable on a scale of 100–1,000 km.

With a simple hydrogeological model coupled to the glaciation model, boundary conditions for an area that is studied in greater detail can be calculated. For the Äspö area, the hydraulic boundary conditions will be calculated for a 100x100 km^2 area with Äspö-Laxemar in the middle. Hydrogeological data have been measured within the area. With their help, the boundary conditions for a smaller area (12x12 km^2) can be calculated. A more detailed geohydrological modelling is planned within this smaller area. Calculated data can be compared with geochemical data and models.

12.5.4 Glaciation scenarios

A scenario is a systematic description of a future situation and the sequence of events (evolution) leading up to it. The climate based process domains, regimes and subregimes shown in Table 12.5-1 define the external premises for an ice-age scenario. In assessing the effects of a glaciation, it is also necessary to weigh in physiographic and geological aspects. A complete description of an ice-age scenario is obtained when external conditions and site are combined. In principle, an ice-age scenario can consist of **one** climate based process domain on a site. This is a static approach where the evolution and history of the repository are not included, at least not directly. One way to describe the evolution is to link together several climate based domains to a series and study the repository given them.

Based on the climate based process domains and the site description, hypothetical critical situations or evolutionary pathways can be identified. With the aid of the glaciation model and the models coupled to it, the hypothetical evolutionary pathways can be underpinned, discussed, modified and justified. In this way, glaciation scenarios that should be considered in the performance assessment can be formulated. Site-specific data should be included in the assessment whenever possible. The hydraulic aspects are of central interest, but thermal, mechanical and certain hydrochemical conditions can also be treated.

If site-specific data are lacking or limited, different geographic locations can be discussed generically. Based on the geographic subdivision of Sweden and with the aid of the glaciation model, glaciation scenarios can be formulated as characteristic series of climate based process domains on a given site. Such glaciation scenarios have been described for the following typical sites /12.5-1/:

- a repository within the mountain zone
- a repository within the hill and plateau zone, on the borderline with the mountain zone
- a repository within the hill and plateau zone, on the borderline with the coastal zone
- a repository within the coastal plain zone

It is also possible to illustrate on the repository performance in relation to different phases of the glaciation cycle. This can be done with the aid of the identified climate based process domains.

Based on the selected scenarios, the radiological hazards to humans, animals and vegetation which might occupy land and water after a glaciation are analyzed. The impact of the glaciation on the geosphere, i.e. the far field, is being dealt with in the paleohydrogeological research programme. The glaciation model can also be used to support the description of the biosphere.

13 INTEGRATION OF RESULTS AND UNCERTAINTY

This chapter presents the evaluation and weighing-together of the judgements and calculations made during the course of the safety assessments which is carried out in order to obtain a composite safety picture. The evaluation/ weighing-together should be related to the purpose at hand and rate the robustness of the safety evaluations with respect to the uncertainty of the underlying data.

14 CONCLUSIONS

This chapter summarizes the overall safety assessment that has been carried out.

***In this report**, Chapter 13 and 14 do not have any text. The descriptive text in SR 95 consists of: a) a review of available methods and tools, and b) status reports from ongoing work for SR-I. Thus, the material does not comprise a coherent safety report.*

Principles and methods for handling of uncertainties and confidence have been discussed in Chapter 3.

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FUEL ROD	1.2 Radiolysis air + water Radiation effects	1.3 Radiation effects (n)	1.4	1.5 Radiation effects (n)	1.6 Decay heat	1.7 Radiation effects	1.8	1.9	1.10	1.11
2.1 Surface coating	FILLER/VOID	2.3 Surface coating	2.4	2.5	2.6 Temperature gradient	2.7	2.8	2.9	2.10	2.11
3.1	3.2 Confinement	STEEL CANISTER	3.4 Causes the gap	3.5 Load on canister bottom	3.6 Temperature gradient	3.7	3.8	3.9	3.10	3.11
4.1	4.2	4.3	GAP Fe/Cu	4.5	4.6 Temperature gradient	4.7	4.8	4.9	4.10	4.11
5.1	5.2	5.3 Confinement	5.4 Causes the gap	Cu CANISTER	5.6 Temperature gradient	5.7 Cu - ion excange. Cementation. Pressure	5.8 Changes the natural flow paths	5.9	5.10	5.11 Repository layout
6.1 State of the fuel Pressure	6.2 State of the filler	6.3 Thermal expension	6.4	6.5 Thermal expansion	TEMPERATURE	6.7 Mineral alteration. Change of properties	6.8 Convection cells	6.9 Formation fractures. Change of properties	6.10	6.11 Repository layout
7.1	7.2	7.3	7.4	7.5 Confinement	7.6 Temperature gradient	BUFFER/ BACKFILL	7.8 Decides local hydrology + chemistry	7.9 Intrusion into fractures	7.10 Swelling pressure	7.11 Repository layout
8.1	8.2	8.3	8.4	8.5 Transport of corrodants	8.6 Temperature gradient	8.7 Saturation Erosion Mineral alt. Ion excange	WATER MOVEMENT	8.9 Fracture filling materials dissolution precipitation	8.10	8.11 Positioning of deposition holes
9.1	9.2	9.3	9.4	9.5	9.6	9.7 Large movements may damage canisters	9.8 Fracture system decides water flow	FRACTURING	9.10 Rock movements may give transient load	9.11
10.1	10.2	10.3	10.4	10.5 Creep SCC	10.6	10.7	10.8 Decides the gradient	10.9 Sealing and possible widening of fractures	PRESSURE CONSTANT LOAD	10.11
11.1	11.2	11.3	11.4	11.5 Damage during emplcement	11.6	11.7 Affects properties	11.8 Chemical effects - man made materials	11.9 Fracture injections and plugs	11.10 Repository depth - hydrostatic pressure	CONSTRUC. EMPLACEMENT

Near field 1: It is the repository system according to Figure 9.1-1 that is described. The copper shell is intact. The resaturation phase is not considered. The matrix is not completely documented.

Significance scale

White (0): No interaction

Green (1): Interaction present
– The influence on other parts of the PS and the rest of the repository system can be neglected.

Yellow (2): Interaction present
– The interaction influences the PS or the rest of the repository system to a limited extent and/or under special circumstances.

Orange (3): Interaction present
– Can influence other parameters, should be well documented.

Red (4): Important interaction
– Should be included in model for quantitative evaluation within the safety assessment.

Appendix 1. Interaction matrix, near field 1

Table 1. The diagonal elements, their positions and definitions

Diagonal element	Position	Definition
Fuel rod complex	1.1	All radionuclides, the fuel itself, the Zircaloy and the metal parts of the fuel assemblies.
Filler and void	2.2	No filler material is being considered today. This element represents the void in the canister.
Steel inner canister (insert)	3.3	
Steel-copper gap	4.4	For fabrication reasons there will be a space between the steel insert and the copper shell. This gap will be about one millimetre at deposition. However, the copper shell is expected to creep on the steel, and the gap is expected to disappear after a few thousand years.
Copper outer canister (shell)	5.5	
Temperature	6.6	The whole near field will be subjected to elevated temperatures, so the definition is not fixed in space.
Buffer/backfill	7.7	The bentonite in the deposition holes with impurities and the backfill material in the tunnels.
Water movement and chemistry	8.8	The movements and composition of the water in rock and buffer.
Fracturing in rock	9.9	The natural fractures in the rock and those formed during construction of the repository.
Pressure	10.10	The pressure in the system: the hydrostatic pressure, the lithostatic pressure and any pressure changes.
Construction and emplacement	11.11	Repository design, construction, emplacement (deposition) of canisters and closure. Reinforcements and forgotten materials.

UO_2-MATRIX	1.2 Confinement	1.3	1.4	1.5	1.6 Volyme change	1.7 Oxidant sink	1.8	1.9	1.10
2.1 Allocation Affects	**MATRIX BOUND ELEMENTS**	2.3	2.4 Source	2.5	2.6	2.7 Contamination Particle	2.8 Surface contamination	2.9 Contamination	2.10 Contamination
3.1	3.2	**"SEGREGATED" ELEMENTS**	3.4 Source	3.5	3.6 Surface contamination	3.7 Contamination	3.8 Surface contamination	3.9 Contamination	3.10 Contamination
4.1 Radiation damage (α,n)	4.2 Elemental change	4.3 Elemental change	**RADIATION**	4.5 Source	4.6 Structural damage	4.7 Radiolysis	4.8 Radiation effects	4.9 Radiation effects	4.10 Radiation effects
5.1 Chemical effects	5.2 ΔH, ΔV	5.3 ΔH, ΔV	5.4	**TEMPERATURE**	5.6 ΔV	5.7 Chemical effect Phase	5.8 ΔH, ΔV	5.9 ΔH, ΔV	5.10 Stability
6.1 Confinement	6.2	6.3 Confinement Source	6.4 Shielding	6.5	**ZIRCALOY + OTHER METALL PARTS**	6.7 Contamination Oxidant sink	6.8	6.9	6.10
7.1 Alteration Dissolution	7.2 Precipation of secondary phases + equilibration	7.3 Precipation of secondary phases + equilibration	7.4 Shielding	7.5	7.6 Corrosion	**WATER**	7.8 Corrosion Dissolution	7.9 Corrosion Dissolution	7.10 Transfer of species
8.1	8.2 Sorbtion	8.3 Sorbtion	8.4 Shielding	8.5	8.6 Surface contamination	8.7 Contamination	**FILLER**	8.9 Surface contamination	8.10
9.1	9.2	9.3	9.4 Shielding	9.5	9.6	9.7 Displacement Contaminat	9.8 Confinement	**CANISTER MATERIALS**	9.10 Contamination with Cu-ions
10.1	10.2 Sorbtion Diffusion	10.3 Sorbtion Diffusion	10.4 Shielding	10.5 Heat transfer	10.6	10.7 Transfer of species Colloid	10.8	10.9 Corrosion Confinement	**BENTONITE**

Fuel 1: It is the repository system according to Figure 9.1-1 that is described. The copper shell is defective. There is water in the canister. Water-saturated conditions prevail, the resaturation phase is not taken into account. The matrix is not completely documented.

Significance scale:

White (0): No interaction

Green (1): Interaction present
- Neglible influence on other parts of the PS, defined in this matrix, or in other parts of the repository system.

Yellow (2): Interaction present
- The interaction influences the PS or the rest of the repository system to a limited extent and/or under special circumstances.
- The interaction is important but is principally dealt with in another part of this matrix or in one of the other interaction matrices that describe the repository system.
- The influence on other parts of the PS or the rest of the repository system is uncertain, it is probably negligible but should be further investigated.

Red (3): Important interaction
- Can influence other parts of the PS, defined in this matrix, or in other parts of the repository system.
- Should be included in the safety assessment. The interaction can be a basic premise for the safety assessment, handled by assumptions or by being included in a model in the safety assessment.

Appendix 2. Interaction matrix, fuel 1

Table 1. The diagonal elements, their positions and definitions

Diagonal element	**Position**	**Definition**
UO_2 matrix	1.1	The physical structure of the fuel. Uranium as a radionuclide is found in 2.2.
Matrix-bound elements	2.2	The radionuclides that were **originally** in the fuel matrix. In this definition they are not fixed in space, but may be anywhere in the system.
"Segregated" elements	3.3	The nuclides that were **originally** on the surface and in the grain boundaries of the fuel, plus activation products in Zircaloy and the structural parts of the fuel assemblies. This diagonal element is a little unusual since it is partially fixed in space (the original position of the radionuclides is an FEP) and partially not fixed (like those in 2.2).
Radiation	4.4	
Temperature	5.5	The system is assumed to be stead-state. No transient thermal processes are taken into account.
Zircaloy and other metal parts	6.6	The radionuclide content is in 3.3.
Water	7.7	The water inside the canister.
Filler	8.8	Non-metallic original filler in the canister. Not included in present-day canister design.
Canister materials and their degradation products	9.9	Metallic canister materials and their degradation products, i.e. corrosion products and hydrogen gas.
Bentonite	10.10	The outer boundary in the system, defined as clay material and pore water. The pore water is separate from the water in 7.7.

FUEL	1.2 Radiation effects Dimension	1.3 Radiation effects	1.4 Gamma Radiolysis	1.5 Radiation effects	1.6 Radiation effects	1.7 Temperature increase	1.8	1.9 Radiolysis	1.10 Radiation effects	1.11	1.12 Radiation effects	1.13
2.1 Confinement	CANISTER	2.3 Pressure buildup	2.4 Cu - ion formation	2.5	2.6 Corrosion gas (Hydrogen)	2.7 Δt over the canister wall	2.8 Intersects transport paths	2.9	2.10	2.11	2.12	2.13 Size deposition holes
3.1	3.2 Confinement Swelling pressure. Shear.	SMECTITE	3.4 [illegible]	3.5 Confinement of minerals	3.6 Gas inclusion	3.7 Δt	3.8 Intersects transport paths	3.9 Weak effect on chemistry	3.10 Swelling pressure. Mechanical impact. Sealing.	3.11 Swelling pressure. Mechanical impact	3.12 Swelling pressure	3.13 Size deposition holes
4.1	4.2 Transport of species. Gives corrosion. Creates pressure	4.3 [illegible]	PORE WATER	4.5 Dissolution Precipitation	4.6 Holds gas. Dissolves gas. Pressure	4.7 Δt	4.8 Pressure	4.9 Exchange of species. Transport	4.10 Pressure	4.11 Dissolution	4.12 Pressure	4.13
5.1	5.2	5.3 Contaminant	5.4 Dissolution. Precipitation. Colloid formation.	MINERALS	5.6	5.7 Δt	5.8	5.9 Indirect effect	5.10	5.11	5.12	5.13
6.1	6.2 Pressure. Corrosion	6.3 Pressure. Chemical effect	6.4 Pressure. Dissolution. Transport	6.5 Chemical effect	GAS	6.7 Δt	6.8 Intrusion into fractures gives unsaturated conditions	6.9 Dissolution in groundwater	6.10 Filling of fractures	6.11 Pressure effect	6.12 Pressure effects. Gas saturation	6.13
7.1 Structural alteration	7.2 Δl - expansion. Structure of the material. Internal pressure.	7.3 Transformations. Permeability	7.4 Boiling. Pressure. Viscosity. Reaction speed. Transport due to gradients	7.5 Chemical effects	7.6 Pressure. Solubility. Transport.	TEMPERATURE	7.8 Convection cells. Viscosity	7.9 Reaction speed	7.10 Permeability. Structure of fractures	7.11 Reaction speed. Δl - expansion	7.12 Transformations. Permeability. Convection	7.13 Geometry. Distances deposition holes, tunnels
8.1	8.2	8.3 Erosion. Water uptake	8.4 Pressure. Exchange	8.5 Erosion	8.6 Pressure. Transport of gas	8.7 $\Delta\Delta t$	GROUNDWATER HYDROLOGY	8.9 Groundwater supply. Transport of species.	8.10 Erosion. Fracture width. Sedimentation.	8.11 Erosion. Pressure	8.12 Water transport. Erosion.	8.13 Selection of deposition holes. Direction of tunnels.
9.1	9.2	9.3	9.4 Exchange. Transport	9.5	9.6 Groundwater dissolution in gas.	9.7 (Δt)	9.8 Density. Viscosity.	GROUNDWATER CHEMISTRY	9.10 Dissolution. Precipitation	9.11 Dissolution. Corrosion	9.12 Dissolution. Precipitation. Ion exchange.	9.13
10.1	10.2	10.3 Confinement. Shear. Pressure. Density decrease. Initial space width.	10.4 Pressure effect	10.5	10.6 Capture of gas	10.7 Δt	10.8 Decides hydrology	10.9 Dissolution and precipitation of fracture minerals	NEARFIELD ROCK	10.11 Shear. Deformations	10.12 Confinement	10.13 Respect distance
11.1	11.2	11.3 Confinement	11.4 Contamination	11.5	11.6 Corrosion gases. Inclusions.	11.7 (Δt)	11.8 New flow-paths	11.9 Contamination	11.10 Mechanical strength	REINFORCEMENTS	11.12 Confinement	11.13
12.1	12.2	12.3 Confinement	12.4 Pressure	12.5	12.6 Transport path. Inclusion.	12.7 Δt	12.8 Changes in natural flow paths	12.9 Contamination. Precipitation. Ion exchange.	12.10 Pressure. Support. Effect on fracture systems	12.11 Mechanical load	BACKFILL	12.13
13.1	13.2 Decides canister design	13.3 Decides properties of the smectite	13.4 Initial effect	13.5 Bentonite quality	13.6 Initial condition	13.7 Ambient temperature. Tolerated temperature	13.8 Gradient. Ambient pressure	13.9 Initial condition	13.10 Initial condition. Disturbances at construction.	13.11 According to needs	13.12 Decides initial amounts	SITE LAYOUT

Buffer 1: It is the repository system according to Figure 9.1-1 that is described. The matrix describes only the performance of the buffer material. The copper shell is intact. The resaturation phase is taken into account. The matrix is not completely documented.

Significance scale:

White (0): No interaction

Green (1): Interaction present

- Neglible influence on other parts of the PS, defined in this matrix, or in other parts of the repository system.

Yellow (2): Interaction present

- The interaction influences the PS or the rest of the repository system to a limited extent and/or under special circumstances.
- The interaction is important but is principally dealt with in another part of this matrix or in one of the other interaction matrices that describe the repository system.
- The influence on other parts of the PS or the rest of the repository system is uncertain, it is probably negligible but should be further investigated.

Red (3): Important interaction

- Can influence other parts of the PS, defined in this matrix, or in other parts of the repository system.
- Should be included in the safety assessment. The interaction can be a basic premise for the safety assessment, handled by assumptions or by being included in a model in the safety assessment.

Appendix 3. Interaction matrix, buffer 1

Table 1. The diagonal elements, their positions and definitions

Diagonal element	Position	Definition
Fuel	1.1	
Canister	2.2	The canister is assumed to be intact.
Smectite	3.3	The clay material without pore water and impurities. The physical dimensions are included.
Pore water	4.4	The water inside the buffer material, in physical contact with the canister, impurities and groundwater, but not with rock or reinforcements.
"Minerals"	5.5	The impurities in the buffer.
Gas	6.6	All gas phases in the system: entrapped air, radiolysis gases and corrosion gases. Some identified processes are only relevant for a corroding canister (in contrast to definition 2.2).
Temperature	7.7	The system is assumed to be stead-state. No transient thermal processes are taken into account.
Groundwater hydrology	8.8	
Groundwater chemistry	9.9	There is a sharp boundary between groundwater and pore water at the buffer/rock boundary.
Near-field rock	10.10	The rock that affects, or is affected by, the other diagonal elements.
Reinforcements	11.11	Construction materials in the repository and forgotten materials.
Backfill	12.12	Filler material in tunnels.
Site, layout, design	13.13	Engineering and material selection are included here.

CONSTRUCTION/ LAYOUT	1.2 Excavation method	1.3 Excavation method Grouting Reinforcement	1.4	1.5 Displacement effects	1.6 Construction materials Stray materials	1.7	1.8 Drawdown effects	1.9 Repository depth Ventilation	1.10 Tunnel dimension	1.11 Ventilation Blasting gas Gas source	1.12	1.13 Industrial facility Dumps
2.1 Swelling ability Heat	BUFFER/ BACKFILL/ SOURCE	2.3 Buffer/backfill penetration into EDZ	2.4	2.5 Buffer into intersecting fractures	2.6 Colloid source Groundwater composition	2.7 Changed flow around holes Changed flow in tunnels	2.8 Resaturation	2.9 Heat generation	2.10 Swelling pressure	2.11 Gas source	2.12 Source term	2.13
3.1 Excavation method Amount of reinforcement	3.2 Volume for buffer/backfill swelling Rock fallout	EDZ	3.4	3.5	3.6 Changed ε_r and a_w Colloid and particulate generation	3.7 Changed permeability	3.8	3.9 Modified thermal diffusivity	3.10 Fractures affected	3.11 Indiffusion of air Transport path for gas	3.12 Changed ε_r and a_w Sorption capacity	3.13
4.1 Layout/ construction method	4.2	4.3 Magnitude and geometrical extent	ROCK MATRIX/ MINERALOGY	4.5 Fracture characteristics and infilling mineralisation	4.6 Rock-water interaction	4.7 Matrix K Rock compressibility	4.8	4.9 Thermal properties	4.10 Genesis, tectonic history and rock type	4.11 Radon generation	4.12 Sorption Matrix diffusion	4.13 Land-use Potential human intrusion
5.1 Avoid major zones Constructability	5.2	5.3 Mechanical properties and fracture frequency	5.4	NATURAL FRACTURE SYSTEM	5.6 Dissolution of fracture minerals Colloid generation	5.7 Flow paths Connectivity Fracture aperture Storage capa.	5.8	5.9 Thermal properties	5.10 Stress magnitude and orientation	5.11 Transport path for gas	5.12 Molecular diffusion Surface area Sorption	5.13 Wells
6.1 Depth affected by redoxpot. Construction materials	6.2 TDS, ion exchange, illitisation	6.3 Precipitation/ bacterial growth	6.4 Groundwater rock interaction	6.5 Precipitation and dissolution of fracture minerals	GROUNDWATER CHEMISTRY	6.7 Density and viscosity	6.8 Density affects groundwater head	6.9 Heat conductivity	6.10	6.11 Chemically generated gas Microbially generated gas Clathrates	6.12 Sorptivity and solubility Colloids and bacteria	6.13 Water-use Biotopes
7.1 Canister positioning Construction methods	7.2 Saturation Bentonite erosion	7.3 Erosion	7.4	7.5 Erosion and sedimentation	7.6 Mixing	GROUNDWATER MOVEMENT	7.8 Equalisation of pressures	7.9 Forced heat convection	7.10	7.11 Two-phase flow	7.12 Transport of dissolved gas Sorbing & non-sorbing species Hydrodynamic dispersion	7.13 Recharge and discharge
8.1 Construction methods	8.2	8.3	8.4	8.5	8.6 Solubility	8.7 Driving force due to pressure gradient	GROUNDWATER PRESSURE	8.9	8.10 Effective stress	8.11 Gas solubility Gas law	8.12	8.13 Potential effect on vegetation
9.1	9.2 Temperature in buffer/ backfill	9.3	9.4 Thermal expansion Thermal conductivity	9.5 Permafrost	9.6 Dissolution and precipitation of minerals	9.7 Viscosity	9.8 Density	TEMPERATURE/HEAT	9.10 Thermal expansion	9.11 Gas solubility Gas law	9.12 Kinetic effects	9.13
10.1 Design/layout Construction methods	10.2 Reaction force on swelling pressure Rock fallout	10.3 Mechanical stability Fracture aperture	10.4 Mechanical stability	10.5 Mechanical stability Fracture aperture	10.6	10.7	10.8 Confined aquifers	10.9	ROCK STRESSES	10.11	10.12	10.13 Mechanical stability
11.1 Ventilation problems	11.2	11.3 Opening of fractures Heat conduction	11.4 Fracturing Thermal properties	11.5 Fracture aperture	11.6 pH, Eh affected	11.7 Creation of 2-phase flow conditions	11.8 Capillary forces	11.9 Gas law	11.10	GAS GENERATION AND TRANSPORT	11.12 Colloid sorption on gas bubbles	11.13 Gas release
12.1 Design/ layout	12.2	12.3	12.4	12.5	12.6 Radiolysis Redox front	12.7	12.8	12.9	12.10	12.11	TRANSPORT OF RADIONUCLIDES	12.13 Contamination
13.1 Siting, design/ layout	13.2	13.3	13.4	13.5	13.6 Infiltrating water	13.7 Surface water recharge & percolation	13.8 Land use Climatic & tidal driving force Hydraulic gradients	13.9 Climatic driving forces	13.10 External load Erosion	13.11	13.12	BIOSPHERE

Farfield 1: It is the repository system according to Figure 9.1-1 that is described. The copper shell is defective, radionuclides are accessible outside the canister. The resaturation phase is taken into account. The matrix is not completely documented.

Significance scale:

White (0): No interaction

Green (1): Interaction present
- Neglible influence on other parts of the PS, defined in this matrix, or in other parts of the repository system.

Yellow (2): Interaction present
- The interaction influences the PS, defined in this matrix, or in other parts of the repository system to a limited extent and/or under special circumstances.
- The interaction is important but is principally dealt with in another part of this matrix or in one of the other interaction matrices that describe the repository system.
- The influence on other parts of the PS or the rest of the repository system is uncertain, it is probably negligible but should be further investigated.

Red (3): Important interaction
- Can influence other parts of the PS, defined in this matrix, or in other parts of the repository system.
- Should be included in the safety assessment. The interaction can be a basic premise for the safety assessment, handled by assumptions or by being included in a model in the safety assessment.

Appendix 4. Interaction matrix, far field 1

Table 1. The diagonal elements, their positions and definitions

Diagonal element	Position	Definition
Construction/Layout	1.1	Construction and layout of the repository. The element defines boundary conditions for the far field.
Buffer/Backfill/Source	2.2	The canisters, the buffer (bentonite) around the canisters and the backfill material in the tunnels. The element defines boundary conditions for the far field.
EDZ	3.3	The part of the rock that has been affected by the construction of tunnels and deposition holes.
Rock matrix/Mineralogy	4.4	The unaffected rock and its mineralogy
Natural fracture system	5.5	The natural fracture system in the rock, fracture mineralogy, different kinds of fracture systems including fracture zones, and the mechanical properties of the fractures.
Groundwater chemistry	6.6	The groundwater chemistry in the EDZ and in the rest of the far-field rock.
Groundwater movement	7.7	All kinds of groundwater movements, both in the EDZ and in the rest of the far-field rock.
Groundwater pressure	8.8	The total groundwater pressure.
Temperature/Heat	9.9	Temperature and heat in the EDZ and in the rest of the far-field rock.
Rock stresses	10.10	The total rock stresses in the EDZ and in the rest of the far-field rock.
Gas generation/ Gas transport	11.11	Gases that have been generated naturally or due to the waste, include the gases in the EDZ and in the rest of the far-field rock.
Transport of radionuclides	12.12	Transport of radionuclides in tunnels, the EDZ and in the rest of the far-field rock.
Biosphere	13.13	Describes the conditions above the repository, climate, vegetation, wells, topography etc. The element defines boundary conditions for the far field.

SOURCE TERM	1.2 Contamination	1.3 Contamination	1.4 Contamination	1.5 Contamination	1.6	1.7	1.8	1.9	1.10	1.11
2.1	PERMANENT SATURATED ZONE	2.3 Flow water + solute (Discharge)	2.4 Flow water + solute (Discharge)	2.5 Water transport	2.6 Via irrigation Capillary rise	2.7	2.8 Irrigation Due to human activities	2.9 Drinkink water	2.10 Use of water	2.11 Ingestion Other water uses
3.1	3.2	SURFACE WATER	3.4 Sedimentation. Diffusion. Advection.	3.5 Recharge (through river bank)	3.6 Floadin Diffusion Sedimentation Erosion Irrigation	3.7	3.8 Uptake Irrigation	3.9 Uptake	3.10 Water supply	3.11 Uptake External Inmersion
4.1	4.2 Water + solutes	4.3 Sediment Resuspension	SEDIMENT	4.5	4.6 Conversion Dredging	4.7 Aerosols format Degasing Evaporation Suspension	4.8 Uptake External contamination	4.9 Uptake External contamination	4.10	4.11 External and direct contamination
5.1	5.2 Percolation Solid transport	5.3 Exfiltration Discharge Transport of eroded material	5.4	VARIABLE SATURATED ZONE	5.6 Gas Capillary transfer Soil formation	5.7	5.8 Deep root Species Uptake	5.9 Burrowing Species	5.10 Builders Land use	5.11 External (Digging)
6.1	6.2	6.3 Run off or wash off Transport of eroded material Chemical effects	6.4 Bank collapse	6.5 Infiltration Chemical effects Mass transfer	SURFACE SOIL	6.7 Resuspension Evaporation Degassing Suspension	6.8 Uptake Rain splash	6.9 Soil consuption External contamination Inhalation	6.10 Land uses	6.11 External Dermal absorption
7.1	7.2	7.3 Deposition Precipitation	7.4 Wind erosion Rainfall	7.5	7.6 Wind erosion Deposition Precipitation	ATMOSPHERE AIR	7.8 Deposition Rain Snow Precipitation	7.9 Inhalation Deposition	7.10 Minimal on Weather depending	7.11 Inhalation External Inmersion
8.1	8.2	8.3	8.4 Bioturbation Death	8.5 Deep rooting	8.6 Death Organic biotubation Fertilis in Washout Water extract	8.7 Exhalation Transpiration Canopy Burnin Reduction of wind speed	FLORA	8.9 Ingestion Contamination	8.10	8.11 Ingestion External
9.1	9.2	9.3 Contamination Drums	9.4 Bioturbation	9.5	9.6 Burrowing Excretion Bioturbation Erosion	9.7 Exhalation	9.8 Consumption Fertilising Direct contamination	FAUNA	9.10 Depending on boundary conditions	9.11 Ingestion External
10.1	10.2 Water extraction Pollution Recharge Treatment	10.3 Water recharge Water extraction Treatments Dam and drain	10.4 Dredging Removal	10.5 Pollution Civilengineer (Deep Plowing)	10.6 Agriculture Pollution Forestry Constuction Irrigation	10.7 Pollution Filtration Ventilation	10.8 Recycling Storage Burning Making furniture	10.9 Farming Storage Hunting	HUMAN ACTIVITIES	10.11 Depending on boundary conditions
11.1	11.2	11.3	11.4	11.5	11.6	11.7	11.8	11.9	11.10	DOSE TO CRITICAL GROUP

Biosphere 1: It is the repository system according to Figure 9.1-1 that is described. The matrix assumes release of radionuclides. The matrix is not completely documented.

Significance scale

White (0): No interaction

Green (1): Interaction exists
- The influence on other parts of the PS and the rest of the repository system can be neglected.

Yellow (2): Interaction exists
- The interaction influences the PS or the rest of the repository system to a limited extent and/or under special circumstances.

Orange (3): Interaction exists
- Can influence other parameters, should be well documented.

Red (3): Important interaction
- Should be included in model for quantitative evaluation within the safety assessment.

Appendix 5. Interaction matrix, biosphere 1

Table 1. The diagonal elements, their positions and definitions

Diagonal element	Position	Definition
Source term	1.1	Radionuclide flow to the biosphere. Represents all the other matrices.
Permanent saturated zone	2.2	Soil and sediments below the groundwater table, including the groundwater there.
Surface water	3.3	Water in seas, lakes, rivers and surface runoff.
Sediment	4.4	Sediments in lakes and rivers, including pore water.
Variable saturated zone	5.5	The zone between the surface soil and the lowest groundwater table. Can be saturated at certain times.
Surface soil	6.6	Uppermost layer of the soil that is tilled and where most of the plants' roots are.
Atmosphere	7.7	The air that humans and animals breathe, including dust and aerosols.
Flora	8.8	All plants, including terrestrial and aquatic/marine plants, fungi and agricultural products.
Fauna	9.9	All animals, including terrestrial and aquatic/marine animals and agricultural products.
Human activities	10.10	Agriculture etc.
Dose to critical group	11.11	The objective of the analysis.

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TR 121
KBS Technical Reports 1 – 120
Summaries
Stockholm, May 1979

1979
TR 79-28
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Summaries
Stockholm, March 1980

1980
TR 80-26
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KBS Technical Reports 80-01 – 80-25
Summaries
Stockholm, March 1981

1981
TR 81-17
The KBS Annual Report 1981
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Summaries
Stockholm, April 1982

1982
TR 82-28
The KBS Annual Report 1982
KBS Technical Reports 82-01 – 82-27
Summaries
Stockholm, July 1983

1983
TR 83-77
The KBS Annual Report 1983
KBS Technical Reports 83-01 – 83-76
Summaries
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Stockholm, May 1986

1986
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Including Summaries of Technical Reports Issued during 1986
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1987
TR 87-33
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1988
TR 88-32
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Stockholm, May 1989

1989
TR 89-40
SKB Annual Report 1989
Including Summaries of Technica! Reports Issued during 1989
Stockholm, May 1990

1990
TR 90-46
SKB Annual Report 1990
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Stockholm, May 1991

1991
TR 91-64
SKB Annual Report 1991
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1992
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Karsten Pedersen (editor)
Department of General and Marine Microbiology, The Lundberg Institute, Göteborg University, Göteborg, Sweden
February 1996

TR 96-02
Microbial analysis of the buffer/container experiment at AECL´s Underground Research Laboratory
S Stroes-Gascoyne[1], K Pedersen[2], S Daumas[3], C J Hamon[1], S A Haveman[1], T L Delaney[1], S Ekendahl[2], N Jahromi[2], J Arlinger[2], L Hallbeck[2], K Dekeyser[3]
[1] AECL, Whiteshell Laboratories, Pinawa, Manitoba, Canada
[2] University of Göteborg, Department of General and Marine Microbiology, Göteborg, Sweden
[3] Guigues Recherche Appliquée en Microbiologie (GRAM), Aix-en-Provence, France
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Reduction of Tc (VII) and Np (V) in solution by ferrous iron. A laboratory study of homogeneous and heterogeneous redox processes
Daqing Cui, Trygve E Eriksen
Department of Chemistry, Nuclear Chemistry, Royal Institute of Technology, Stockholm, Sweden
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Jordi Bruno, Lara Duro, Salvador Jordana, Esther Cera
QuantiSci, Barcelona, Spain
February 1996