OPERA

Collection and analysis of current knowledge on salt-based repositories

OPERA-PU-NRG221A

Radioactive substances and ionising radiation are used in medicine, industry, agriculture, research, education and electricity generation. This generates radioactive waste. In the Netherlands, this waste is collected, treated and stored by COVRA (Centrale Organisatie Voor Radioactief Afval). After interim storage for a period of at least 100 years radioactive waste is intended for disposal. There is a world-wide scientific and technical consensus that geological disposal represents the safest long-term option for radioactive waste.

Geological disposal is emplacement of radioactive waste in deep underground formations. The goal of geological disposal is long-term isolation of radioactive waste from our living environment in order to avoid exposure of future generations to ionising radiation from the waste. OPERA (OnderzoeksProgramma Eindberging Radioactief Afval) is the Dutch research programme on geological disposal of radioactive waste.

Within OPERA, researchers of different organisations in different areas of expertise will cooperate on the initial, conditional Safety Cases for the host rocks Boom Clay and Zechstein rock salt. As the radioactive waste disposal process in the Netherlands is at an early, conceptual phase and the previous research programme has ended more than a decade ago, in OPERA a first preliminary or initial safety case will be developed to structure the research necessary for the eventual development of a repository in the Netherlands. The safety case is conditional since only the long-term safety of a generic repository will be assessed. OPERA is financed by the Dutch Ministry of Economic Affairs and the public limited liability company Electriciteits-Produktiemaatschappij Zuid-Nederland (EPZ) and coordinated by COVRA. Further details on OPERA and its outcomes can be accessed at www.covra.nl.

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Summary

The present report describes the results of an evaluation of the aspects of the Safety Case for the final disposal of radioactive waste in rock salt. The evaluation comprises an overview of the present knowledge of the safety and feasibility of a final disposal facility in rock salt in the Netherlands. Where possible, recommendations are provided for the further development of the Safety Case in rock salt in the Netherlands.

The evaluation is performed on the basis of available national (ICK, OPLA, CORA), and international information about the final disposal in rock salt. In that area, the German and US programmes are the most mature ones.

Separate sections at the end of every chapter, dealing with a particular aspect of the Safety Case, summarize and evaluate the main issues. Additionally, recommendations to proceed to a next phase of the Dutch Salt Safety Case, have been provided.

The evaluation of all the aspects shows that there is abundant information available about the final disposal of radioactive waste in rock salt. Most of the knowledge has been gained in the US and German programmes, although the previous Dutch programmes also provided significant contributions to important technical and methodological aspects for assessing the safety of salt-based repositories.

In the Netherlands there is at present little activity with regard to the development of a disposal facility in rock salt. Additionally, the characteristics of the radioactive waste intended for disposal have changed since the CORA programme. This implies that the previously considered facility concepts for disposal in rock salt have not yet been adapted to the newly considered waste types such as spent fuel from research reactors and depleted uranium. As a consequence, the integration of safety arguments related to the salt Safety Case is for a part based on generic considerations that may not all be applicable to the Dutch context, and that does not yet take into account all relevant aspects.

The main recommendation to proceed further with the development of the Salt Safety Case in the Netherlands is to outline a final disposal facility in rock salt, taking into account the most recent waste characteristics and an up-to-date safety concept including the definition of safety functions for the disposal of radioactive waste in rock salt.

A more detailed overview of recommendations to proceed with the Salt Safety Case in the Netherlands is provided in the second Delivarable of the OSSC project, OPERA-PU-NRG221B.

Samenvatting

Dit rapport beschrijft de resultaten van een evaluatie van de diverse onderdelen van de Safety Case voor eindberging in steenzout. De evaluatie omvat een overzicht van de actuele kennis op het gebied van de veiligheid en haalbaarheid van een eindbergingsfaciliteit voor radioactief afval in steenzout in Nederland. Waar mogelijk zijn aanbevelingen gegeven voor de verdere ontwikkeling van de Safety Case voor de eindberging van radioactief afval in steenzout in Nederland.

De evaluatie is verricht op basis van de beschikbare nationale (ICK, OPLA, CORA) en internationale informatie over eindberging in steenzout. Op dit gebied zijn de programma's in Duitsland en USA het verst ontwikkeld.

Separate secties aan het einde van elk hoofdstuk waarin een onderdeel van de Safety Case wordt behandeld geven een samenvatting en evaluatie van de belangrijkste bevindingen. Daarnaast zijn aanbevelingen gegeven voor vervolgstappen voor de Nederlandse Safety Case voor de eindberging van radioactief afval in steenzout in Nederland.

Uit de evaluatie blijkt dat er een grote hoeveelheid informatie beschikbaar is op het gebied van de eindberging van radioactief afval in steenzout. De Amerikaanse en Duitse programma's hebben hiertoe de grootste bijdragen geleverd, hoewel ook de voorgaande Nederlandse programma's een significant aandeel hadden in de ontwikkeling van technische en methodologische aspecten.

In Nederland is op dit moment weinig activiteit gaande op het gebied van de ontwikkeling van een eindbergingsfaciliteit voor radioactief afval in steenzout. Daarnaast zijn de karakteristieken van het radioactieve afval bedoeld voor eindberging veranderd sinds het CORA programma. Hierdoor zijn voorheen beschouwde eindbergingsconcepten niet toereikend om de nieuw beschouwde afvaltypen, zoals gebruikte splijtstof van onderzoeksreactoren en verarmd uranium, te ontvangen. Als gevolg daarvan is de integratie van de argumenten met betrekking tot de veiligheid van een eindbergingsfaciliteit in steenzout voor een deel gebaseerd op algemene overwegingen die niet allemaal van toepassing zijn op de Nederlandse context.

De belangrijkste aanbeveling voor de verdere ontwikkeling van de Safety Case voor de eindberging van radioactief afval in Nederland is het vastleggen van een ontwerp van een faciliteit in steenzout. Dit moet gedaan worden met inachtneming van de meest recente afvalkarakteristieken en een actueel veiligheidsconcept waarbij nog nader te definiëren veiligheidsfuncties worden beschouwd.

Een gedetailleerd overzicht van de aanbevelingen voor een volgende iteratie van de Safety Case voor eindberging in zout in Nederland is weergegeven in het tweede rapport van het OSSC project, OPERA-PU-NRG221B.

1. Introduction

1.1. Background

Radioactive waste is generated in all phases of the nuclear fuel cycle and as a result of the use of radioactive materials in industrial, medical, research and military applications. In principle, all such waste must be managed safely. The most hazardous and long-lived waste, such as spent nuclear fuel and high-level waste from the reprocessing of spent fuel, must be contained and isolated from humans and the environment for very long time periods, i.e. hundreds of thousands of years. Less hazardous and short-lived radioactive waste should in principle also be isolated, but depending on the radiological characteristics the period of isolation may be in the order of several decades up to several thousands of years.

Geological disposal is the currently favoured radioactive waste management end-point providing long-term security and safety (NEA, 2013; p.11), and has been judged and demonstrated to be technically feasible (NEA, 2008a; p.14). The final disposal of radioactive waste in engineered facilities, or repositories, in suitable deep geological formations, is presently being investigated and developed in various countries. Several countries have well-developed programmes on geological disposal such as Belgium, Finland, France, Germany, Sweden, and Switzerland. The only country presently operating a deep geological disposal facility for (transuranic defence) waste is United States (Hart, 2014a).

Netherlands is one of the countries with a disposal programme at its initial stage. The main reasons for that are related to the Dutch waste management policy of extended surface storage, and the relatively small amounts of radioactive waste intended for disposal.

Despite the stage of a disposal programme, the development of geological disposal facilities for radioactive waste will take place over extended periods of time. At various stages in the lifecycle of these facilities, from siting to final release from regulatory control, decisions are needed to proceed through the lifecycle and move towards the next stage. These decisions are supported by safety assessments and evidence from in-situ monitoring and analogues which, in the end, demonstrate that a repository will be safe in the long term.

The complete set of arguments and analyses used to justify the conclusion that a specific repository system will be safe is referred to as Safety Case. A Safety Case includes, in principle, a presentation of evidence that all relevant regulatory safety criteria, which may either be national or regional, can be met. It includes also a series of documents that set out the national or regional context, describe the system design and its safety functions, illustrate the performance, present the evidence that supports the arguments and analyses, and that discuss the significance of any uncertainties or open questions in the context of decision making for further repository development.

An additional important function of the Safety Case is to provide a platform for informed discussion whereby interested parties can assess their own levels of confidence in a project, determine any reservations they may have about the project at a given planning and development stage, and identify the issues that may be a cause for concern or on which further work may be required.

The main objective of the OPERA research programme is to provide tools and data for the development of Safety Cases for national repository concepts for radioactive waste disposals in two host rocks present in the Netherlands, rock salt and Boom Clay (Verhoef, 2011a; p. 6). Within the OPERA context, the Safety Case has been explained as a collection of arguments in support of the long-term safety of the repository (Verhoef, 2011a; p. 5). Although the OPERA research programme is primarily focused on the disposal concept in

Boom Clay, part of the management strategy in the Netherlands is also to develop and maintain the knowledge on the disposal of radioactive waste in rock salt.

There is no fixed recipe for the compilation and maintenance of a Safety Case for the disposal of radioactive waste, since the purpose and context of the Safety Case may differ from country to country. However, it is generally agreed that a Safety Case should constitute all relevant safety aspects of a repository, and that it evolves as time progresses towards the implementation and subsequent closure of the site.

Examples of general schemes that would be applicable to a Safety Case are illustrated in the figures below (NEA, 2004; p.19) (IAEA, 2012; p.17).

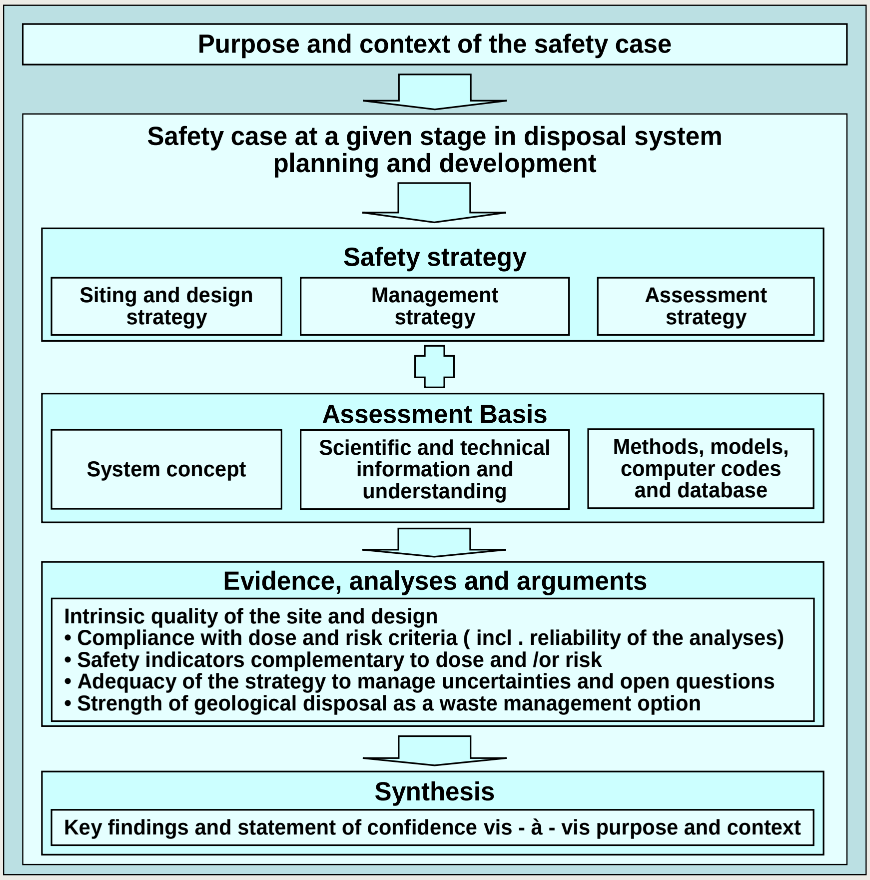

Figure 1-1 Representation of the different stages of a Safety Case (NEA, 2004)

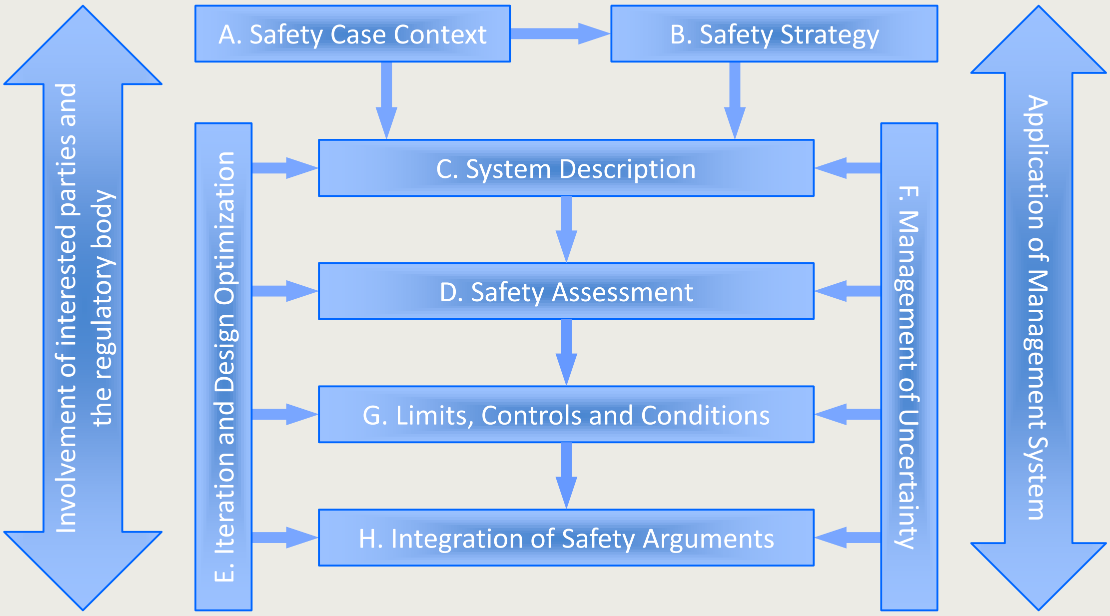

Figure 1-2 Components of the safety case; application of the management system and the process for interaction with the regulatory body and interested parties (IAEA, 2012)

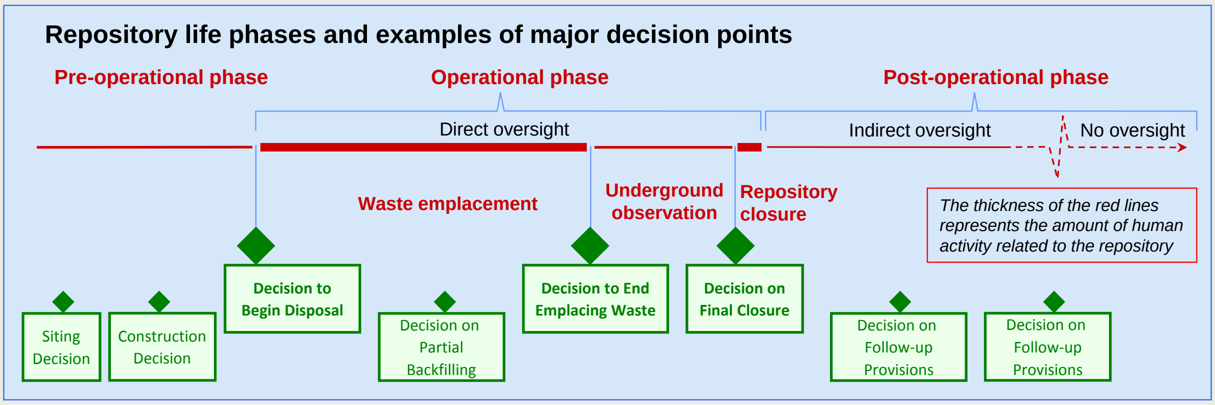

Figure 1-3 Phases of a disposal facility

As time progresses and the implementation of a disposal facility evolves (see Figure 1-3; NEA, 2012; p.10), the Safety Case develops accordingly. This also brings about that decisions have to be taken to move from one phase of the implantation to a next one. As part of the IAEA project PRISM it is recognized that these decisions will be taken under joint responsibilities of the involved parties, which may change during the disposal programme (IAEA, 2012b). This process is indicated in the following figure.

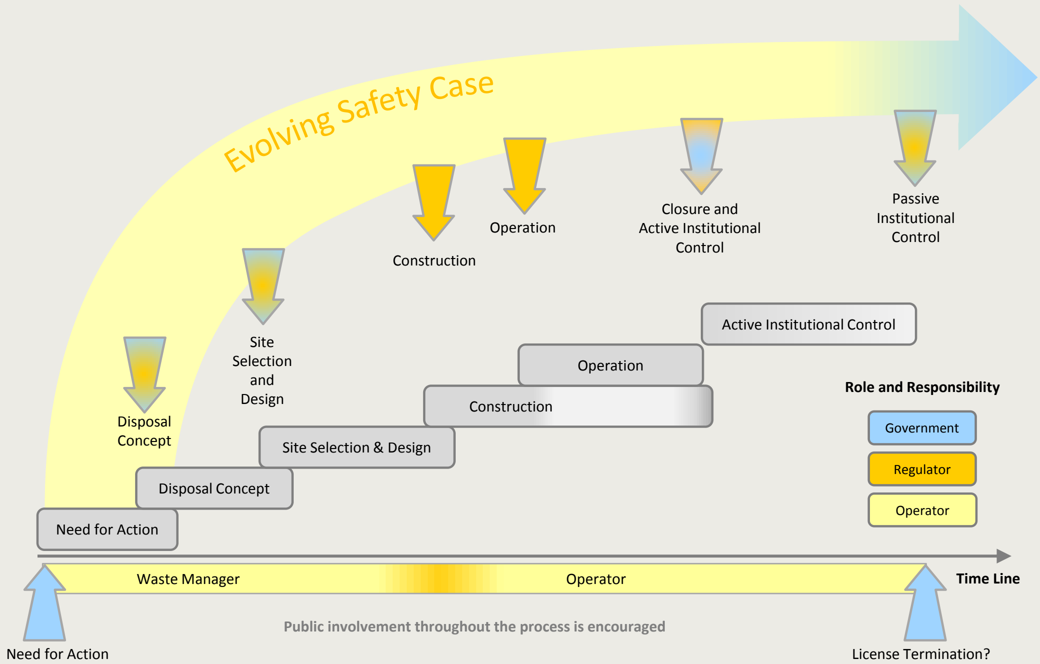

Figure 1-4 Evolution of the different stages within a Safety Case

The concept of the Safety Case is a more or less accepted methodology by countries seriously developing programmes for the final disposal of spent fuel and/or HLW in deep geologic facilities as a feasible means to provide the safe, secure isolation from the biosphere. The nature and status of the programmes vary among countries, but in generally there seems to be convergence of thinking on the challenging task of siting a deep geologic repository intended to isolate long-lived radioactive waste over tens of thousands to hundreds of thousands of years. The collective international experience offers an abundance of information on the development and implementation of a Safety Case for geologic disposal programmes. An overview of several international Safety Cases has been compiled as part of the OPERA OSCAR, resulting in a summary of recent developments of and experiences gained in national repository programmes from a representative sample of countries, i.e. Belgium, Finland, France, Germany, Sweden, Switzerland, and United States (Hart, 2014a).

In the Netherlands much effort has been devoted during the last 40 years to the geologic disposal of radioactive waste in rock salt, for example in the framework of the ICK[1] (ICK, 1979), OPLA[2] (OPLA, 1989), OPLA-1A (RGD, 1993) and CORA[3] (CORA, 2001) programmes. Additional work has been done in several EU Framework projects like EVEREST, BAMBUS, PAMINA, and THERESA. In these programmes performance assessments and detailed analyses have been accomplished for generic repository designs in rock salt. The results of all these programmes have however not yet been integrated according to the recently developed and generally accepted methodology of the Safety Case by NEA (NEA, 2008) and IAEA (IAEA, 2012).

This report describes the results of the research proposed for OPERA Task 2.2.1: *Evaluation of current knowledge for building the Safety Case*, as part of OPERA Work Package 2.2: *Repository design in rock salt*. The OPERA Salt Safety Case (OSSC) project intends to provide the OPERA programme with a first Safety Case of the geological disposal in the Dutch context. This Safety Case is based on a review of the state of the art on geological

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[1] Interdepartementale Commissie Kernenergie (Interdepartmental Nuclear Energy Commission)
[2] Commissie Opberging op Land (Commission on Onshore Disposal)
[3] Commissie Opberging Radioactief Afval (Commission on Disposal of Radioactive Waste)

disposal of radioactive waste in rock salt in the context of a Safety Case and a critical evaluation of the existing national and international knowledge base. The OSSC project assesses the possible gaps in the existing knowledge in relation to the Safety Case and provides recommendations, which could guide future activities in accordance with the radioactive waste management strategy in the Netherlands.

1.2. Objectives

The OPERA Salt Safety Case (OSSC) project aims:

i) to assess the current knowledge base concerning the safety and feasibility of the geologic disposal of radioactive waste in a rock salt formation in the Netherlands,

ii) to process the current knowledge according to the methodology of the Safety Case for deep geological disposal,

iii) to identify knowledge gaps in the Dutch Salt Safety Case, and

iv) to provide recommendations for further development of the Safety Case for rock salt in the Dutch context.

1.3. Realization

The present report summarizes the current knowledge base concerning the safety and feasibility of the geologic disposal of radioactive waste in rock salt formations in the Netherlands. International experiences on relevant feasibility studies and the waste disposal in rock salt have been accounted.

In addition, evaluations of the most relevant aspects of the salt Safety Case have been provided at the end of each chapter, and recommendations have been given for the main issues in the salt Safety Case which need further development within the Dutch context.

1.4. Explanation contents

The objective of the present report is to present and evaluate the various major elements of the salt Safety Case within the Dutch context.

Chapter 2, The Safety Case Context, addresses the context in which the radioactive waste is produced and should be disposed of. It addresses the nuclear profile and the waste management strategy of the Netherlands. It also addresses the legal national and international commitments and guidance.

Chapter 3 concentrates on the safety strategy in the Netherlands, although several aspects of the Dutch strategy will be compared with those that have been implemented in Germany and the US, as these countries are also working on geological disposal in rock salt formations. The comparison only relates to aspects that are typical of the safety concepts and associated aspects.

Chapter 4 presents an overview of the characteristics of the waste that has been collected and should be disposed of, designs of facilities in which the waste was foreseen to be disposed, properties of the salt formation wherein the facility is constructed, the surrounding and overlying sediments on the salt formations and the biosphere.

Chapter 5 provides an overview of the safety assessment methodologies applied in the previous Dutch studies VEOS and PROSA, compared with the methodologies applied in CORA and PAMINA. Additionally, more recent views and developments on safety assessment methodologies have been identified.

Chapter 6 reviews and compares the safety arguments as stated in OPLA and CORA with current insights in process understanding of relevant safety functions of the disposal design. Information provided in the documentation from Germany and US will be taken into account, too.

Chapter 7 provides a short conclusive statement.

Appendix 1 provides an evaluation of features, events, and processes, FEPs, relevant for salt, by comparing the FEP catalogues developed in PROSA, in Germany and US.

Appendix 2 gives criteria for the selection of a salt formation in which a GDF may be constructed, as formulated in 1975 by the sub commission RAS (Radioactieve Afvalstoffen) of ICK, the Interdepartmental Nuclear Energy Commission.

Appendix 3 provides quantitative criteria for determining the number of runs to be performed in a probabilistic safety assessment in order to obtain reliable results.

2. Safety Case Context

2.1. Objective and Scope

The Safety Case Context is an overarching element of the Safety Case, mainly including a country's nuclear profile, types of waste to be disposed of, and legal framework. The following sections provide a summary and an evaluation of these aspects within the Dutch boundary conditions of geological waste disposal in rock salt.

2.2. Nuclear profile of the Netherlands

The Netherlands has a small nuclear programme, with only one 483 MWe nuclear power plant in operation, the Borssele NPP (a PWR, Siemens/KWU design), producing about 4% of the Dutch electrical power consumption. The Dodewaard BWR, a 60 MWe GE design, is in a shut-down state since 1997 and already in an advanced stage of decommissioning (safe enclosure) (MinEZ, 2013; p. 17).

There are presently two operating research reactors in the Netherlands, the largest one is the 45 MWth High Flux Reactor (HFR) in Petten, operated by Nuclear Research & consultancy Group (NRG) and supplying 70% of the European demand for medical radio-isotopes. In addition, the 2 MWth Hoger Onderwijs Reactor (HOR), located at the premises of the Delft Technical University, is used for the generation of neutrons for research purposes. The HOR is planned to be upgraded with extra facilities (MinEZ, 2013; p. 41). Other Dutch research reactors which have been operated in the past are mentioned in Table 2-1 (Dodd, 2000; modified from Table 6).

In the past, some or all of the spent fuel from the research reactors has been returned, in accordance with international policy and agreements, to the country from which it originated. The long-term management of the returned fuel is then the responsibility of the supplier country. Nowadays, the spent fuel from the HFR and HOR is stored in COVRA's HABOG facility for high level waste (MinEZ, 2013; p. 18).

Table 2-1 Test and research reactors in the Netherlands.

Reactor	Type	Owner	Criticality Date	Initial Design/Max Licensed Power (kW)	Shutdown Date
BARN	Pool	ITAL	04-1963	100	01-01-1980
ATHENE	Argonaut	TUE	06-02-1969	10	1971
KSTR	Suspension	KEMA	22-05-1974	1000	18-05-1977
LFR	Argonaut	NRG	28-09-1960	10 / 30	2010
HFR	Tank	European Commission	09-11-1961	20000 / 45000	2020*
HOR	Pool	TUD	25-04-1963	200 / 3000	-

* Present outlook, possibly replaced by the PALLAS reactor

Other nuclear facilities in the Netherlands are the following (MinEZ, 2013; p. 18):

- Nuclear research facilities and laboratories in Delft (Technical University) and in Petten (Nuclear Research & consultancy Group, NRG and the EU Joint Research Centre, the JRC).
- Facilities related to the enrichment of uranium, owned and operated by Urenco Netherlands (uranium enrichment) and Enrichment Technology Netherlands (ET-NL,

development and production of centrifuge technology), both in the Eastern part of the country in Almelo. The licensed capacity currently is 6200 tSW/a.
- The COVRA interim radioactive waste storage facility, located In the South-West of the country, in the municipality of Borsele. It has facilities for the storage of conditioned low, intermediate and high level waste.

At present there are no plans to build new nuclear power plants in the Netherlands, primarily for unstable economic reasons and related uncertainties.

On the other hand, a new research reactor, PALLAS, is under consideration in order to replace the HFR in Petten. Plans for PALLAS were initiated by NRG, the current License Holder and operator of the HFR. At present there is no start-up date foreseen for the PALLAS reactor.

The Technical University of Delft has launched a project to upgrade its research reactor, the Project Oyster (MinEZ, 2013; p. 43). The project is jointly financed by the university and the national government.

Spent fuel from any new-built research reactors will also be transferred to COVRA for long-term surface storage.

2.3. Types of waste to be disposed of

The Dutch policy on spent fuel management is that the decision on whether or not to reprocess spent fuel is primarily a decision to be made by the operator of an NPP. In the early days the NPP operators have decided in favour of reprocessing the spent fuel for economic reasons, reuse of plutonium and reduction of the waste volume. Reprocessing contracts have been concluded for all spent fuel generated by the currently operating Borssele NPP until its end of operation, which is foreseen in 2033. A new treaty was signed by the Republic of France and the Kingdom of the Netherlands on April 20, 2012, regulating for Dutch spent fuel (SF) produced after 2015, its receipt by Areva in France, its reprocessing and the return of vitrified high-level wastes (HLW) from reprocessing to the Netherlands before 31 December 2052 (MinEZ, 2013; p. 20).

In addition to the vitrified HLW, also spent fuels are presently being produced in the Netherlands. The spent fuel intended for final disposal consists of conditioned spent fuel from the Dutch research reactors and spent uranium targets from the production of molybdenum at the premises of NRG, Petten, by the company Mallincrodt.

In (Verhoef, 2011a), an outline of the waste inventory and some of its general features are given. In the disposal layout, four waste disposal sections are distinguished (Verhoef, 2011a; Fig. 5-2), partly named after the Dutch waste classes:
- spent fuel (SF);
- vitrified high level waste HLW;
- non-heat generating high level waste (HLW);
- low and intermediate level waste (LILW), including (Technically Enhanced) Naturally Occurring Radioactive Materials ((TE)NORM).

Unlike in many other countries, where near-surface disposal for LILW is a common practice, in the Netherlands this type of waste, including (TE)NORM waste, is intended for deep geological disposal. Additional information about the different types of radioactive waste intended for final disposal is provided in Section 4.3.

2.4. Legal framework

The treatment of radioactive waste is determined for a large part by a country’s legislative and regulatory framework. The legal framework in the Netherlands related to nuclear activities is explained thoroughly in Section 7.1 *Legislative and regulatory framework* of (MinEZ, 2013). The present report gives a summary overview of the Dutch legal framework.

2.4.1. Dutch legal framework

The legal framework in the Netherlands with respect to nuclear installations can be displayed as a hierarchical structure, as depicted in Figure 2-1.

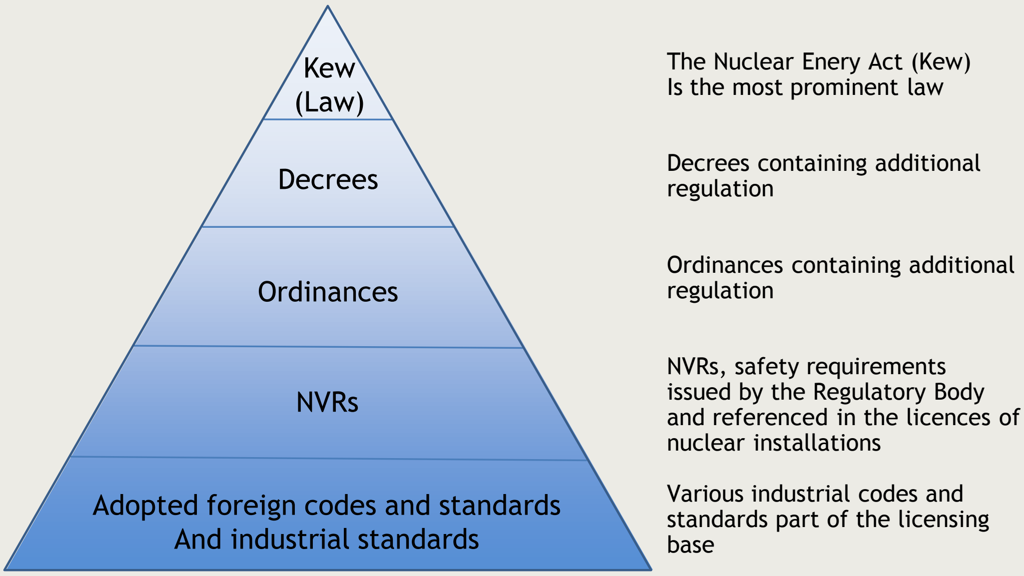

Figure 2-1 Simplified representation of the hierarchy of Dutch the legal framework.

The following are the main laws to which nuclear installations are subject in the Netherlands (MinEZ, 2013; p. 48):

- the Nuclear Energy Act (1963, as amended 2006); (Kew, 1963);
- the Environmental Protection Act (1979, as amended 2002); (Wm; 1979);
- General Administrative Law Act (1992, as amended 2003); (Awb; 1992).
- The Act on Liability for Nuclear Accidents ('Wet Aansprakelijkheid Kernongevallen', WAKO);
- The Water Act ('Waterwet, Ww);
- Environmental Permitting Act ('Wet algemene bepalingen omgevingsrecht', Wabo).

The basic legislation governing nuclear activities is contained in the Nuclear Energy Act (Kew). The Nuclear Energy Act was designed (1) to regulate the use of nuclear energy and ionising radiation techniques, and (2) to lay down rules for the protection of the public and workers against the associated risks. In practice, however, the law has been developed almost entirely to set out the basic rules on nuclear energy, to make provisions for radiation protection, designate the various competent authorities, and outline their responsibilities.

The Environmental Protection Act (Wm), in conjunction with the Environmental Impact Assessment Decree, stipulates (in compliance with EU Council Directive 97/11/EC – De Boer, 1997) that an Environmental Impact Assessment must be presented when an application is submitted for a license for a nuclear installation. In the case of non-nuclear installations, this Act regulates all environmental issues (e.g. chemical substances, stench and noise); in the case of nuclear installations, the Nuclear Energy Act takes precedence and regulates both conventional and non-conventional environmental issues.

The General Administrative Law Act (Awb) sets out the procedure for obtaining a license, and also describes the role played by the general public in this procedure (i.e. objections and appeals).

A number of decrees have also been issued containing additional regulations and these continue to be updated in the light of ongoing developments. The most important of these in relation to the safety aspects of nuclear installations and radioactive materials are:

- the Nuclear Installations, Fissionable Materials and Ores Decree (Bkse) regulates all activities (including licensing) that involve fissionable materials and nuclear installations;
- the Radiation Protection Decree (Bs) regulates the protection of the public and workers against the hazards of all ionising radiation, and establishes a licensing system for the use of radioactive materials and ionising radiation emitting devices, and prescribes general rules for their use;
- the Transport of Fissionable Materials, Ores, and radioactive Substances Decree (Bvser) deals with the import, export and inland transport of fissionable materials, ores and radioactive substances by means of a reporting and licensing system.

In addition to the above-mentioned laws and decrees several Ordinances ('Ministeriële Regelingen, MR') and Nuclear Safety Rules (NVRs, Regulations and guides issued by the Regulatory Body) are applicable. An overview is provided in Section 7.1 of (MinEZ, 2013).

2.4.2. Institutional framework

All nuclear facilities in the Netherlands operate under licence which is awarded after a safety assessment has been carried out. Licences are granted by the Regulatory Body (RB) under the Nuclear Energy Act (Kew, 1963).

Regulatory Body

The (RB) is the authority designated by the government as having legal authority for conducting the regulatory process. This includes the issuing of licences, and thereby regulating nuclear, radiation, radioactive waste and transport safety, nuclear security and safeguards (MinEZ, 2014; Section 4.0.a).

Until the end of 2014, the RB constituted of several entities and the legal foundation of the RB was founded in several ministerial decisions detailing the mandates of these entities. The separate entities that formed the RB operated with working agreements under the responsibility of the minister of Economic Affairs.

The Dutch Council of Ministers decided on January 24, 2014 that the expertise in the area of nuclear safety and most of the expertise on radiation protection will be brought together in a single new administrative Authority, positioned at the ministry of Infrastructure and the Environment. The RB, i.e. the Authority for Nuclear Safety and Radiation Protection (Autoriteit Nucleaire Veiligheid en Stralingsbescherming, or ANVS), is responsible for regulating the nuclear sector and radiation protection as an Independent Administrative Authority (in Dutch a ZBO –Zelfstandig Bestuursorgaan). The ANVS as an organization will fall under the responsibility of the Minister of Infrastructure and the Environment (I&M). The structure of the ANVS is depicted in the following figure[4].

[4] http://www.autoriteitnvs.nl/anvs/organisatie last accessed on 14 January 2014

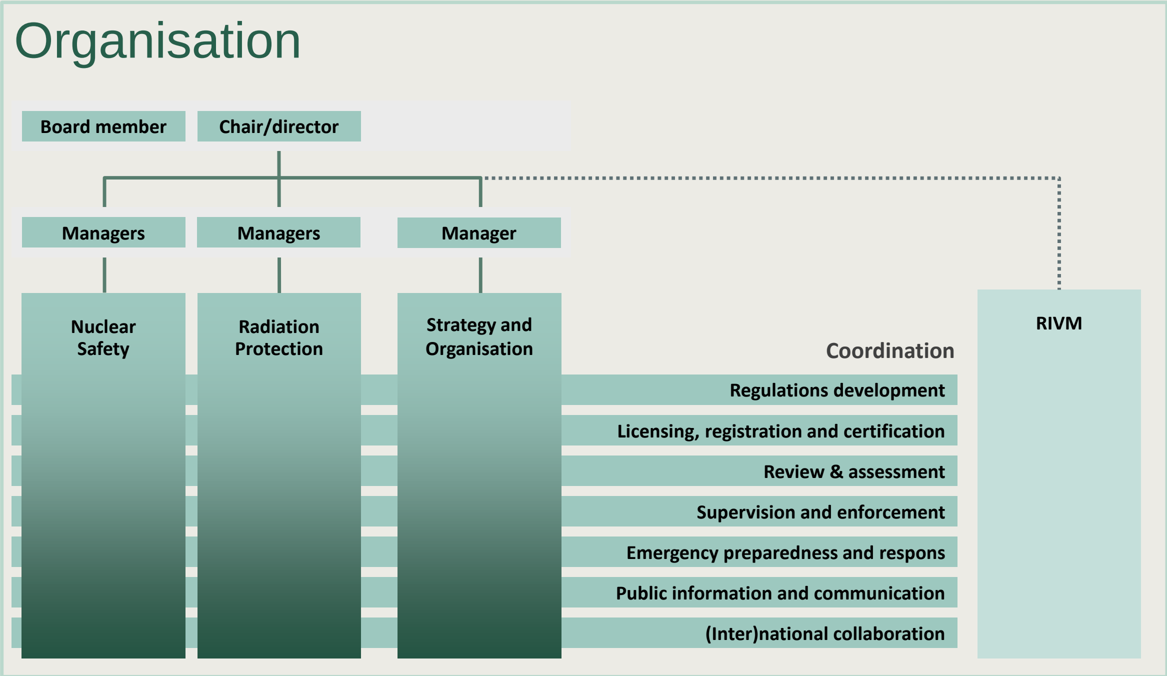

Figure 2-2 Structure of the new RB, the ANVS

The new RB will meet international standards, including those published by the International Atomic Energy Agency (IAEA). The new authority will prepare and draft legislation, develop safety and security requirements and requirements for radiation protection, issue licenses and permits, carry out inspection, safety assessment and enforcement, regulatory research and development and provide information to the public. The ANVS will also be jointly responsible (with the local and regional authorities and the national crisis organization) for emergency preparedness and response in the event of incidents which could result in the release of radiation.

Technical Support Organisations

It is considered one of the basic policies of the RB to have the core disciplines available in-house, while the remaining work is subcontracted to third parties (like RIVM) or other organizations. The RB has a budget at its disposal for contracting external specialists of Technical Support Organisations (TSOs) or other consultancy organizations. Support is provided by foreign TSOs and national and international consultancy organizations. Some major supporting organizations are:

- The Nuclear Research & consultancy Group (NRG) in Petten and Arnhem provides consultancy & educational services to government and industry. NRG also is a Licence Holder (LH). NRG currently has a framework contract with the RB.
- GRS, Germany. The *Gesellschaft für Anlagen- und Reaktorsicherheit* is one of the large German TSOs supporting the German national regulator. In the Netherlands GRS evaluates safety cases and provides other types of consultancy to the RB. In addition GRS provides associated education and training for governmental and commercial organizations. GRS currently has two major framework contracts with the RB.
- In its inspection activities the RB is supported by one independent notified body, Lloyd's Register, for inspections on nuclear equipment under pressure. These inspections are done on behalf of the RB and are paid for by the LHs.

It must be noted that the Dutch institutional framework may be subject to change and shifts of responsibilities. In the past this occurred on several occasions, primarily as a result of changes in the political arena after national elections, and the associated shifts of views and opinions on how to organize the implementation of the basic laws. That said, the information provided in this section should be revisited on a regular basis.

2.4.3. System of licensing

The system of licensing in the Netherlands is explained in Section 7.2 of (MinEZ, 2013). Only the main themes are mentioned in the following.

The Dutch Nuclear Energy Act stipulates (in Article 15, sub b) that a licence must be obtained to construct, commission, operate, modify or decommission a nuclear power plant. Similarly, the Act states (in Article 15, sub a) that a licence is required to import, export, possess or dispose of fissionable material.

The procedures to obtain a licence under the Nuclear Energy Act (and other acts), follow the guidelines specified in the General Administrative Act (Awb). These procedures allow for public involvement in the licensing process. Any stakeholder is entitled to express his views regarding a proposed activity. The Regulatory Body takes notice of all views expressed and respond to them with founded reasoning. If the reply is not satisfactory, the RB can be challenged in court.

In addition to the Nuclear Energy Act, several types of regulation may apply to a nuclear facility and the activities conducted in it and/or supporting it. Therefore often several authorities are involved in the licencing procedures, sometimes at several levels in the governmental organisation.

At present it cannot be foreseen how the laws and responsibilities will change in the next decades, taking into account the Dutch strategy of the long-term surface storage of radioactive waste. As time progresses, and the prospects of an implementation of a deep geological disposal facility will come closer it is mandatory to develop regulatory aspects in that sense.

2.4.4. Legislation and geological disposal

The Dutch policy on radioactive waste management is mainly based on three policy documents:

- the 1984 *Radioactive Waste Policy in The Netherlands; An outline of the Government's position* (VROM, 1984);
- the 1993 *Cabinet Position on Underground Disposal* (VROM, 1993); and
- the 2002 *Radioactive Waste Management; Policy Perspective 2002 – 2010* (VROM, 2002).

The main line of the Dutch policy is to isolate, control, and monitor radioactive waste in above ground structures for at least a hundred years, after which geological disposal is foreseen. During the period of interim storage all necessary technical, economical, and social arrangements are to be made in such a way that geological disposal can really be implemented afterwards. This involves a clear choice for the ownership of the waste, developing appropriate financing schemes, resolving outstanding technical issues, preserving the expertise and knowledge, gaining public understanding of the waste management issues and building public support (MinEZ, 2011; p. 143). A further detailing of the Dutch policy is expected from the national radioactive waste plan scheduled by the Government in 2014 (MinEZ, 2013; p. 52), which is based on the EURATOM council directive 2011/70 from 2011 (EU, 2011).

A recent letter by the MinEZ explained the outlines of the Dutch national programme of radioactive waste and spent fuel (Kamp, 2013). Topics that were mentioned to be investigated in more detail concern:

- International cooperation in the possible realization of a final disposal facility;
- The boundary conditions for which radioactive wastes may be imported to COVRA;
- The possibility to deviate from the presently selected timeline of surface storage at COVRA if future developments of innovations will give rise to that;
- The possibility to flexibly treat the selected option of long-term management. This aspect relates especially to the consideration of surface disposal of low and intermediate level radioactive waste, in the framework of international cooperation.

The last mentioned aspect opens up an important deviation from the 1984 *Policy Document* (VROM, 1984), in which geological disposal of all radioactive wastes is foreseen.

2.4.5. International commitments and guidance

The Nuclear Energy Act and the associated decrees are fully in compliance with the relevant Euratom Directive laying down the basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionising radiation. This Directive (96/29/Euratom; EU, 1996) is incorporated in the relevant Dutch regulations.

In addition to the 'Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management' (CNS), the Netherlands is party to many other Treaties and Conventions related to the use of nuclear technology and materials, such as the 'Convention on Physical Protection of Nuclear Material and Nuclear Installations', the 'Paris Convention on Third Party Liability in the Field of Nuclear Energy' and the 'Brussels Convention' supplementary to the 'Paris Convention', and the 'Joint Protocol Relating to the Application of the Vienna Convention and the Paris Convention' (MinEZ, 2013; p. 53).

In 2011, the Netherlands has brought Council Directive 2009/71/EURATOM of 25 June 2009 on nuclear safety into force. The Netherlands is drafting the required 'National Programme' according to the definition provided by this Directive.

2.5. Financial Considerations

As in other European countries, in the Netherlands the funding for the costs for waste management is regulated by law. The main principle is that the producers of nuclear waste are responsible for the costs of radioactive waste management, viz. the polluter pays principle (MinEZ, 2013; p.83). In the Netherlands, COVRA is the responsible organization for providing adequate financial resources in order to ensure the deep geological disposal facility. The estimation of radioactive waste generation is updated by COVRA every five years in order to assess and/or update the necessary financial resources for future geological disposal. A detailed waste-fee system has been developed to charge radioactive waste generators for the waste they transfer to COVRA. The money is stored at an account at the Ministry of Finance and guaranteed by the State (MinEZ, 2011; p.49).

2.6. Evaluation

- At present no major changes are foreseen for the Dutch nuclear profile: except the elaboration of the PALLAS research reactor.
- Since the 1984 *Policy Document* (VROM 1984) geological disposal of **all** Dutch radioactive wastes is foreseen. In the framework of international cooperation it might be possible to flexibly treat the selected option of long-term management (Kamp, 2013). The possibility, especially related to the consideration of surface disposal of low and intermediate level radioactive waste, opens up an important deviation from the *Policy Document*.

- The Nuclear Energy Act and the associated decrees are fully in compliance with the relevant Euratom Directive 96/29, laying down the basic safety standards for the protection of the health of workers and the general public against the dangers arising from ionising radiation.
- The Netherlands is well involved in international activities and parties concerning the use of nuclear technology and materials.
- It cannot be excluded that in the future the Dutch institutional framework may be subject to change and shifts of responsibilities. In the past this occurred on several occasions, primarily as a result of changes in the political arena after national elections, and the associated shifts of views and opinions on how to organize the implementation of the basic laws. That said, the information provided here as well as in (MinEZ, 2013) should be updated on a regular basis.
- At present it cannot be foreseen how the laws and responsibilities will change in the next decades, taking into account the Dutch strategy of the long-term surface storage of radioactive waste. As time progresses, and the prospects of an implementation of a deep geological disposal facility will come closer it is mandatory to develop regulatory aspects in that sense.
- In the CORA final report the possible risk was acknowledged concerning the uncertainty for the society to manage a long-term surface storage facility for a period of several hundred years, or to be able to finally dispose the waste (CORA, 2001; p. 38).

3. Safety Strategy

3.1. Objective and Scope

A safety strategy is a high-level approach for achieving safe disposal of radioactive waste, given a specific context as described in the previous Chapter. For the Netherlands, this strategy should provide for a systematic process for developing, testing and documenting the understanding of a disposal facility and building and maintaining the necessary knowledge and competences through successive research programmes.

The present chapter concentrates on the safety strategy in the Netherlands, although several aspects of the Dutch strategy will be compared with those that have been implemented in Germany and the US, as these countries are also working on geological disposal in rock salt layers. The comparison only relates to aspects that are typical of the safety concepts and associated aspects. The overall safety strategies of Germany and the US are recently described in detail in (Bollingerfehr, 2013) and (US DOE; 2011; Section A.5.1) respectively.

3.2. Basis of the Dutch waste management strategy

The 1984 Governmental policy plan, together with the policy statement on retrievability, form the basis for the Dutch strategy principles, which can be summarized in the following five points (Haverkate, 2002; p.19):

1. Radiation protection;
2. Isolation, control, and surveillance;
3. Central organisation for managing radioactive wastes;
4. Onshore long-term retrievable disposal;
5. Ongoing research in finding acceptable waste management solutions.

These items have been translated into a safety strategy, which has been summarized in the OPERA Research Plan (Verhoef, 2011a), the OPERA Meerjarenplan (Verhoef, 2011b), and in the recently published Safety Strategy document (Verhoef, 2014), and elucidated in the following sections.

3.3. Present views on the Dutch waste management strategy

In the Netherlands, the development of a geological disposal facility for radioactive waste will take place over a century. Based on the present nuclear programme it requires at least 100 years to collect sufficient radioactive waste to make a disposal facility economically viable. Extended surface storage and therefore ongoing radioactive decay is also beneficial in relation to potential thermal effects which can complicate the post-closure Safety Case of heat-generating High Level Waste. In addition, a long period of interim storage allows time to inform stakeholders about the implementation of a geological disposal facility.

At various stages in the lifecycle of the geological disposal facility, decisions are needed to proceed through the lifecycle and move towards the next stage. These decisions are to be supported by a safety case. It is the responsibility of the implementer of a geological disposal facility to develop and maintain a safety case (IAEA,2011; p.19).

Figure 3-1 shows the common elements in the decision-making processes on geological disposal in different countries and the key stakeholders (Verhoef, 2014; Figure 1). Added on the left is the planned timing for the Netherlands that follows from the current Dutch policy. Indicated on the right is the nature of the decision that has to be taken e.g. the element ‘Operation’ is driven by two decisions: the decision to “Operate” taken by the operator and the decision of issuing a license for operation taken by the regulator.

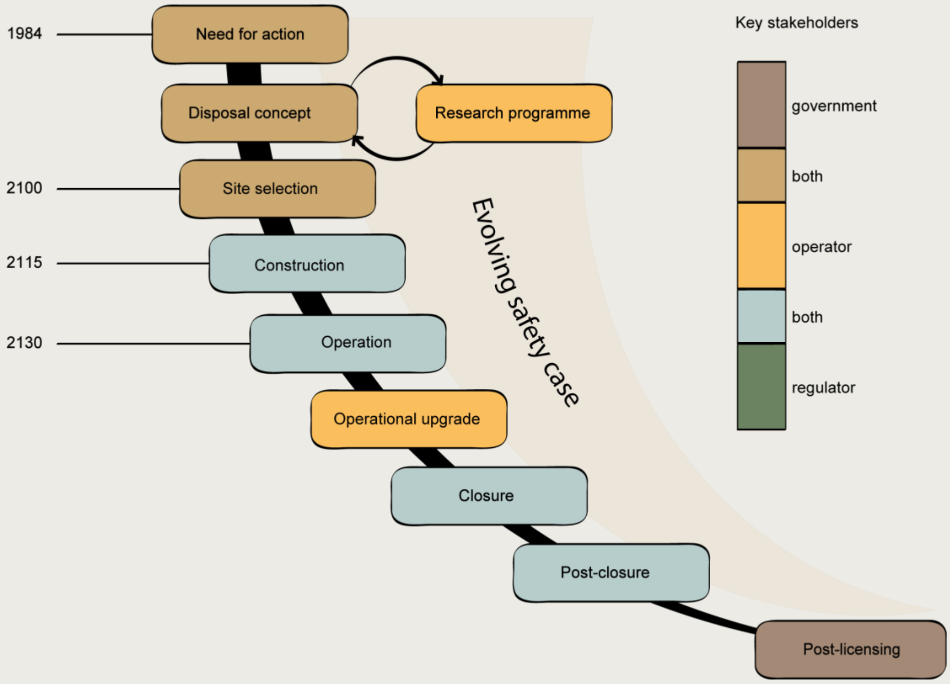

Figure 3-1 **Common elements in the decision-making processes on geological disposal identified in an IAEA meeting and added on the left the timeline for the Netherlands.**

The goal is to progressively develop the safety case in order to provide a sufficient basis for a decision to move forward to the next phase in 2100: the selection of a host rock for final disposal, as indicated in Figure 3-2 below (Verhoef, 2011c; p.6). From present day until 2100, preliminary Safety Cases relying on rather generic assumptions about the properties of the host rock are expected to be compiled at the end of each research programme. Around the turn of the century, sufficient confidence should be acquired to support the decision for site selection.

Decision process	Research on final disposal: National - rock salt	National - Boom Clay	Multi-national
1984 – principal decision on interim storage and final disposal	Feasibility of retrievable final disposal in rock salt and Boom Clay formations in the Netherlands		
1993 – requisite of retrievability	Feasibility of retrievable final disposal in rock salt and Boom Clay formations in the Netherlands		
2002 – confirmation of feasibility of retrievable final disposal in rock salt and Boom Clay formations in the Netherlands -transfer of stock from COVRA to government; start of 100-year period of surface storage	Development of Safety Cases (SC-0: long-term safety) for final disposal in rock salt and Boom Clay in the Netherlands, update of cost estimate and exploration of societal aspects[1]		Feasibility study on multi-national final disposal facility; elaboration of implementation strategy
2015 – OPERA Final Report 2025 – Safety Case 1 2035 – Safety Case 2 2045 – Safety Case 3 2055 – Safety Case 4 2065 – Safety Case 5	Elaboration of final disposal concepts and possible implementation strategies (long-term safety, technical feasibility, operational safety, cost, monitoring, societal support, etc.); evolutionary development of Safety cases (SC-x)		In this period the research on national final disposal will be complementary to that on multi-national final disposal
2080 – selection of national or multi-national final disposal[1]	Detailed comparison possible options: prolonged surface storage, final disposal in the Netherlands in rock salt and Boom Clay formations; elaboration of potential site selection processes		
2100 – Selection of host rock for final disposal	Preparative studies, surface exploration, basic engineering of potential final disposal concepts in selected host rock type		
2115 – Selection of site for final disposal	Underground in-situ exploration at selected site, detailed engineering of the facility		
2130 – Start of operation of disposal facility			

Figure 3-2 Possible stepwise decision steps on the final disposal of Dutch radioactive waste under the assumption that surface storage will not continue after 100 years and a location for a national final disposal facility in the Netherlands is sought.

The following sections elucidate the Dutch safety strategy of the final disposal of radioactive waste in more detail.

3.4. Boundary conditions

The boundary conditions to provide a general orientation for long-term research programmes are derived from the relevant international and national regulatory framework (IAEA, EU (Euratom), ICRP) and national policy. Research activities are required on the basis of Article 8 of the EC Directive on radioactive waste as a way to obtain, maintain and develop necessary expertise and skills for the management of radioactive waste (EU,2011:p.54). The boundary conditions for geological disposal of radioactive waste in the Netherlands have been outlined in (Verhoef, 2014; Chapter 2). A summary of these boundary consitions is given below.

- *The ICM criteria (isolate, control and monitor) form the basis of the radioactive waste management policy.* Since 1984, the policy in the Netherlands is that all hazardous and radioactive waste must be isolated, controlled and monitored.
- *Radioactive waste is stored above ground for a period of at least 100 years.* In 1984 the decision-in-principle has been taken to dispose of the Dutch radioactive waste in a geological disposal facility after interim storage above ground for a period of at least 100 years.
- *A single organisation has been established for management of all steps of the radioactive waste management process.* The policy in the Netherlands is that all radioactive waste[1] produced in the Netherlands is managed by a single organisation, namely COVRA.
- *In addition to a national geological disposal facility (GDF), the option of a multinational GDF is not excluded.* The option of sharing a geological disposal facility (GDF) with one or more countries is also being considered in the Dutch 'dual track' policy.
- *All radioactive waste is intended to be disposed of in a single, deep GDF operating in 2130.* For the Dutch national disposal option, no separate GDFs for Low and Intermediate Level Waste (LILW) and High Level Waste (HLW) are presently envisaged, due the relatively small waste volume expected to be collected in a period of 100 years.
- *The GDF has to be designed, operated and closed such that the process is reversible and the waste is retrievable.* Retrieval of waste, if deemed necessary for whatever reason, would be possible for decades up to more than a century after closing the geological disposal facility, leaving the possibility to future generations to apply other management techniques, if available (EL&I,2011a:p.19).
- *Both rock salt and clay formations are being considered as potential host rocks for geological disposal in the Netherlands.* Due to the adopted long-term surface storage strategy in the Netherlands there is no immediate urgency to select between the potentially suitable host rock salt or clay formations.
- *Specific regulatory criteria for the siting or the performance of a geological disposal facility have not yet been defined.* Due to the adopted long-term surface storage strategy in the Netherlands there is no immediate urgency to develop specific regulatory criteria for the geological disposal of radioactive waste.
- *The public has to be given the necessary opportunities to participate effectively in the decision-making process regarding radioactive waste.* Although this boundary condition is essential for the implementation of a disposal programme, this topic is not treated in the present report.

On the basis of the boundary conditions, presently adopted in the Netherlands, strategic measures have been engaged. The following section elucidates these strategic choices.

3.5. Strategic Choices

Strategic choices are high-level preferences constrained by the boundary conditions and made on the basis of existing knowledge and understanding that aim to further define the requirements for a GDF in the Netherlands (Verhoef, 2014; Chapter 3). The strategic choices formulated in (Verhoef, 2014) are focused on Boom Clay as a host rock. In the following paragraphs these strategic choices have slightly been re-arranged and adapted to rock salt as a host rock.

- *The GDF will be constructed at sufficient depth to take into account the impact of surface phenomena*. The host rock and geological environment should provide an effective containment of the emplaced waste and isolation from the biosphere. The depth of GDF should be sufficient to take into account impact of surface phenomena such as ice ages.
- *The GDF will be constructed within a Tertiary Clay formation or Zechstein rock salt formation*. The geological conditions in the Netherlands, with large salt formations in the Northern part of the country and clay formations at varying depth in the whole country, are in principle favourable from the perspective of disposal of radioactive waste.
- *The materials and implementation procedures should not unduly perturb the safety functions of the host formation, or of any other component*. The design, construction, operation and closure of the GDF should minimize perturbation of the host rock in order to preserve its safety functions. And ensure stability and predictable behaviour of the disposal system over long periods of time.
- *In the case of heat-generating waste, the engineered barriers will be designed to provide complete containment of the wastes at least through the thermal phase*. In case of heat-generating waste the waste package should contain the radionuclides at leaste during the 'thermal phase' during which the temperature in the surrounding host rock is increased.
- *Waste types will be divided into groups to be emplaced in separate sections of the geological disposal facility*. The foreseen physical separation of different waste types is intended to prevent or minimize the influence of the products generated by degradation of waste matrices and packages on other types of waste.
- *The different disposal galleries and sections, and the geological disposal facility as a whole, will be closed (access routes backfilled and sealed) following a progressive, step-wise closure procedure*. During the period after waste emplacement, monitoring is needed as well as regular maintenance of access ways and emplacement/retrieval systems. On a regular basis it could be decided whether to extend the post-operational phase, retrieve the waste, or to close the facility. This decision process could be guided by a legally established procedure, which must be transferred from government to government or even over generations.
- *Geological disposal planning will assume that surveillance and monitoring will continue for as long as deemed necessary*. Retrievability requires long-term arrangements for maintenance, data management, monitoring and supervision e.g. systems to facilitate retrieval of waste. These arrangements should be continued until adequate assurance has been obtained concerning the safety of the geological disposal of waste.
- *There are preferences for using shielded wastes packages that minimise operations and consequent operational radiation doses in the underground*. For disposal in rock salt, any shielding in the HLW disposal package may be removed after emplacement of the waste. In that case it has to be ensured that any possible radiation damage to the surrounding rock salt is excluded.
- *There are preferences for materials and implementation procedures for which broad experience and knowledge already exists*. Processes and procedures for the development, construction and operation of the GDF need to be robust, i.e. simple, reliable, and effective. Above ground construction, assemblage and quality assurance of waste packages are preferred if feasible, as well as the application of proven technologies in related fields of work, such as mining, tunnel construction (concrete support) and the oil and gas industry.

3.6. Siting strategy

Since the early 1980s, siting radioactive waste repositories has proven to be a complicated issue in every country. However, lessons have been learned in the last decade from successful national programmes and advanced processes have emerged that address this aspect. A central element of successful siting strategies is to establish the cooperation with local communities that may be interested in hosting a disposal facility and who wish to become actively involved in its development.

Considering the present stage in the decision-making process in the Netherlands, the efforts within OPERA on the development of a siting strategy will be limited to the aspect of building up public and technical confidence in the technical feasibility and radiological safety of radioactive waste disposal in an early stage (Verhoef, 2011a; p. 7).

Within OPERA efforts will be devoted to investigate how to involve stakeholders and assessing what determines the level of public trust and confidence, especially in OPERA WP1.2, “Political requirements and societal expectations”. The results of that work were not yet available at the time of compiling the present report.

3.7. Safety concept

3.7.1. General approach on the safety concept

The safety concept for geological disposal is the understanding of why the disposal system is safe, irrespective of identified uncertainties and detrimental phenomena; i.e. why it is expected to be robust. It includes a description of the roles of the natural and engineered barriers and the safety functions that these are expected to provide in different time frames, and why the disposal system is expected to be safe. (NEA, 2012; p. 30, Section 17.3).

The disposal system should rely on the intended safety functions taking into account a range of features and associated processes that vary in their effectiveness and in the level of the available scientific and technical understanding. The safety concept is the main starting point to define the technical and scientific requirements for the disposal system and its specific components. Other boundary conditions might be considered, like for example specific requirements from the regulator or other stakeholders, or considerations on reversibility and retrievability (IGD-TP, 2011; p. 10).

On a more generic level the safety concept includes a consideration of fundamentals like the concept of multiple barriers, safety functions, robustness, and passive safety (IAEA, 2012c; Section 5.3). As the development of a programme for the disposal of radioactive waste develops, the safety concept emerges into a disposal concept (IG-DTP, 2011; p. 10), followed by a site-specific concept. During subsequent iteration processes, estimates of performance are made and an understanding is developed of which elements of the disposal system actually provide safety under various conditions, thus refining the disposal concept. In addition, based on the design requirements specific objectives and strategic measures are derived.

The following sections provide information about the safety concepts that have been developed in Netherlands, Germany and US, in relation to geological disposal in salt. The implementation of the various safety concepts, if applicable, is discussed in Chapter 4 (System Description).

3.7.2. The Dutch Safety Concept

In essence the Dutch safety concept for the geological disposal of radioactive waste in Boom Clay relies on the consideration of multiple barriers and safety functions that can be attributed to the subsequent barriers (Verhoef, 2011; Section 4.1). This safety concept would in principle also be applicable for the disposal in salt-based repositories. For salt-based repository concepts, developed in the past in the Netherlands, multiple barriers have been considered as well, unlike the concept of safety functions. The primary reason for not having developed safety functions for salt-based repositories in the Dutch context is that the consideration of safety functions has emerged only after the previous Dutch national disposal programmes OPLA and CORA programme have ended.

The following sections provide an overview of the multi-barrier concept and safety functions in the framework of disposal in a salt-based repository in the Dutch context.

Multiple barriers

The objective of geological disposal is to isolate radioactive waste from the biosphere until the radioactivity of the waste has decayed to sufficiently low levels (for instance the natural radiation level of uranium ore). This requires long-term isolation can be achieved by a system of multiple barriers (e.g. Verhoef, 2011a; Section 4.1).

The principle of a multi-barrier system was already recognized in the OPLA (see Figure 3-3; CORA, 1989; p. 34) and PROSA studies. For the derivation of categories of scenarios the OPLA study considered events and processes that could lead to a disruption of the system of barriers (CORA, 1989; p. 35).

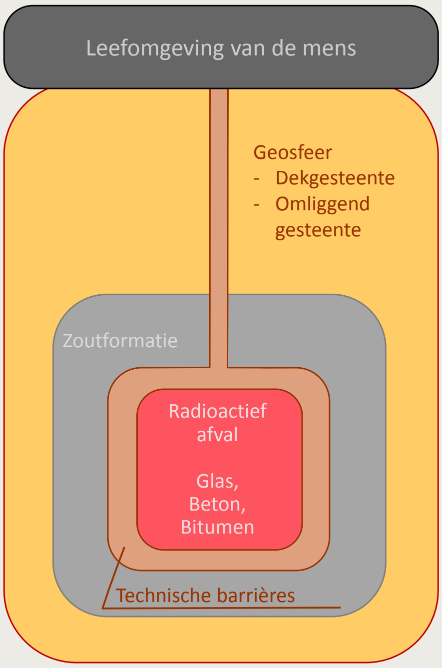

Figure 3-3 Principle of a multi-barrier system in salt formations (OPLA).

The PROSA study also considered a multiple-barrier system for salt-based repositories (Prij, 1993; p. 0.11):

- waste form and package
- the engineered barriers;
- the isolation shield of salt around the repository;
- the overburden on top of the salt formations.

This concept was applied in the derivation of scenarios, amongst others by defining the state of the barrier: i) present and ii) by-passed. Section 5.3 provides more details about the PROSA safety assessment methodology including the application of the multi-barrier concept and the derivation of scenarios.

The geological disposal concept in Boom Clay, adopted in OPERA, also relies on a sequence of complementary and/or redundant barriers (defence-in-depth). These barriers can be natural (geological) and man-made (engineered) and can be subdivided into the following subsystems (Verhoef, 2011a; p. 8; see also Figure 3-4):

- The near-field – including:
  - waste packages (waste matrix, container, overpack if used)
  - additional engineered barriers (buffer materials if used, seals, cap or cover). The zone disturbed during the excavations, the excavation disturbed zone, EDZ, is not a barrier in itself, but in safety assessments it is usually considered as a separate "section" of the near field;
- The far-field - the host rock and surrounding geological formations (or overburden);
- The biosphere - the physical media (atmosphere, soil, sediments, and surface waters) and the living organisms (including humans) that interact with them.

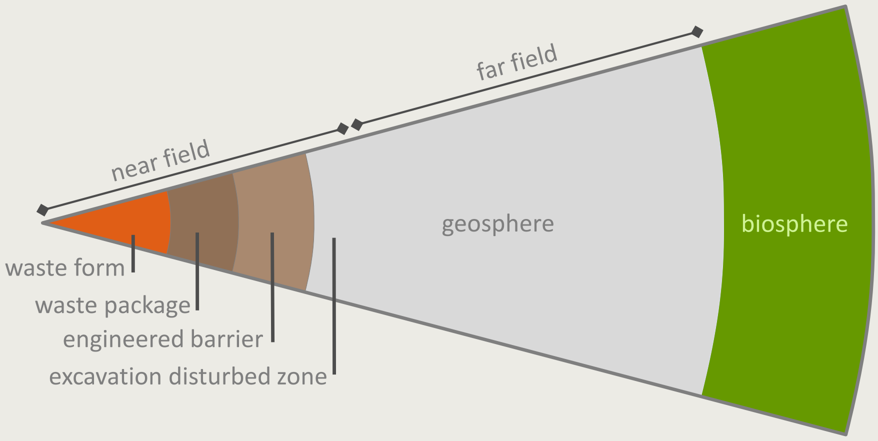

Figure 3-4 Compartments of a design of a repository concept

The near field comprises seals, backfills, and plugs and supporting materials like concrete lining for the case where the host rock cannot prevent collapse of the excavated volume during the time required for waste emplacement waste [5]. Considering the Dutch requirement of retrievability of waste, a longer period of stability needs to be provided by the engineered structures.

The far field comprises the host rock that is not disturbed during the excavation of volumes and the geological media surrounding the host rock. Another term for the far field is "geosphere". Within OPERA, this region is also labelled the "geological environment".

In principle, the multi-barrier systems adopted in PROSA, and later on in CORA (Grupa, 2000; Section 2.1), and in OPERA do not differ significantly.

Safety Functions

In the OPLA and PROSA reports, no specific safety functions were explicitly mentioned to characterize the above-mentioned barriers of the multi-barrier system[6]. The OPLA Final Report only mentions that the geological formation surrounding the disposal facility can be conceived as a multi-barrier system against the dispersion of the disposed radionuclides (OPLA, 1989; p. 34).

The PROSA study assumes that the subsequent barriers of the disposal system can have two possible states: i) present and ii) by-passed. More elaborate functions of the barriers were however not considered in the PROSA study. It must be noted however that a degeneration of a barrier is taken into account by parameter variation.

[5] In rock salt such a concrete lining is not necessary

[6] This does not imply that the functions of the barriers were not accounted for. For each container, seal, backfill and dam mathematical descriptions have been applied to describe the chemical and physical properties in general and for the release and transport of radionuclides through these components for the scenarios considered.

In the CORA programme only in several occasions the barriers were linked to features that may be interpreted as safety functions, although they were not elucidated as such. For example, for the METRO-I concept in salt it was mentioned that rock salt has some favourable properties to act as a host rock for final disposal, i.e. that rock salt is a hydraulic barrier due to its very low permeability (Grupa, 2000; p. 13). However the term “safety function” was not mentioned in the CORA reports.

Retrievability

In the Netherlands, the national policy prescribes that all radioactive waste will be stored above ground in engineered structures allowing retrieval at all times for a period of at least 100 years. After this period of long-term storage, geological disposal is foreseen (Verhoef, 2011a; p. 7).

The retrievability of waste is an important prerequisite for the geologic disposal in the Netherlands. Whereas no explicit legislation or guidelines for waste retrieval has been developed, the general concept is discussed internationally and worked out in last decade to greater detail by developing principles like ‘retrievability’, ‘reversibility’, geological disposal as a ‘staged process’ and the utilization of ‘pilot facilities’ (e.g. NEA, 2011). Although some of these concepts are already integrated in general terms within the OPERA disposal concept for a disposal in Boom Clay, the ‘retrievability’ as an essential aspect of the Dutch policy on radioactive waste disposal needs to be worked out in greater detail. This aspect is under development as part of the OPERA project ENGAGED (ECN, 2012), covering OPERA Task 1.2.3.

The approaches for developing a concept, selecting a site, implementing practical engineering solutions, and monitoring should take into account arrangements to ensure the reversibility of disposal operations and the retrievability of waste packages. Specific provisions should be made by the implementer for possible waste retrieval in which case a dedicated safety assessment for retrievability should be established. This could include, for example, provisions to ensure that the disposal facility has sufficient temporary storage capacity to retrieve numerous waste packages and to allow for safe temporary or permanent storage of waste emplacement.

3.7.3. The German safety concept

In the course of the German R&D project ISIBEL a safety concept for the disposal of HLW in a domal salt structure was developed (Buhmann, 2008). The basic idea is to focus on the systematic demonstration of the **safe containment** of the waste within the salt dome. The critical barriers for safe containment are the rock salt, the shaft seals and the drift seals. Any void volume in emplacement areas is to be backfilled with crushed salt which will be naturally compacted by convergence. During compaction, the porosity and permeability of the crushed salt decreases until, in the long run, it has the same barrier properties as rock salt. This safety concept was upgraded and described in more detail in the R&D project VSG (Mönig, 2012). Based on the safety principles set out in the German “*Safety Requirements governing the final disposal of heat-generating*[7] *radioactive waste*” (BMU, 2010)[8], and on existing knowledge concerning the processes that could impair the safety of the repository, as well as the (Gorleben) site properties, three **guiding principles** have been defined as follows:

- the radioactive waste must be contained as widely as possible in the containment providing rock zone (CRZ[9]),

[7] In Germany radioactive waste is divided into heat-generating radioactive waste and waste with negligible heat generation. Heat-generating waste is generally comprised of spent fuel elements and high-level radioactive waste (fission product solutions) from the reprocessing of spent fuel elements.

[8] In the following referred to as “Safety Requirements”

[9] The English translation of the German Safety Requirements use the expression „isolating rock zone“ and defines this zone as the part of the repository system which, in conjunction with the technical seals ensure

- the containment shall be effective immediately post-closure and it must be provided by the repository system permanently and maintenance-free, and
- the immediate and permanent containment shall be accomplished by preventing or limiting intrusion of brine to the waste forms.

The geological barrier should provide the permanent containment of the radioactive waste. The rock salt within the CRZ is essential for the containment as it is practically impermeable to solutions. Accordingly the integrity of the rock salt within the CRZ must be ensured.For a further development of the safety concept the guiding principles were reinforced by the definition of two further **design requirements** whilst a third additional design requirement stems from the regulatory requirement to avoid criticality in the repository:

- **Containment**: The emplaced waste canisters shall be enclosed quickly and as tightly as possible by the salt;
- **Performance of CRZ**: During the demonstration period of one million years, the CRZ shall remain intact and its barrier function shall not be impaired by internal or external processes and effects; and
- **Subcriticality:** Subcriticality must be guaranteed in all phases of the repository evolution.

To derive a site-specific safety concept these design requirements are then used to derive specific objectives and to determine strategic measures which embrace design specifications, for example with respect to the mine position in the salt dome, and technical provisions. Typically, each strategic measure supports a number of specific objectives. The strategic measures together meet the objectives of the safety concept. The principle types of correlation between design requirements, specific objectives and measures are schematically shown in Figure 3-5 (Bollingerfehr, 2013; p.20).

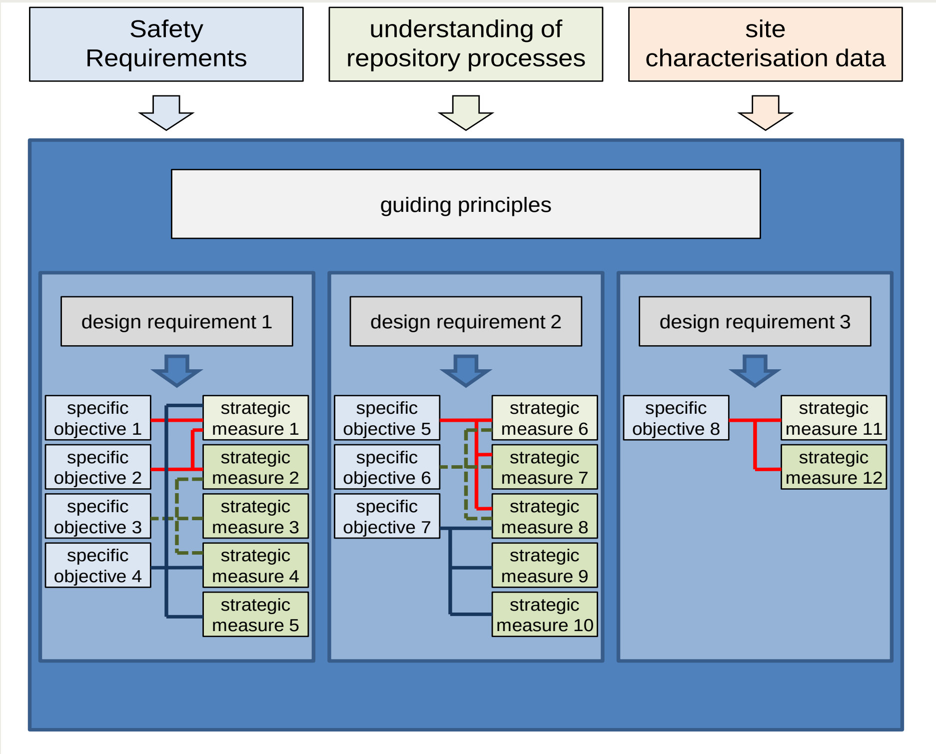

Figure 3-5 Principle approach to derive specific objectives and strategic measures.

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the containment of the waste. Since this zone refers explicitly to the safety function “containment”, the term “containment providing rock zone” is used in this report.

This approach allows mapping of the general stipulations of the German Safety Requirements to objectives and measures for the safety concept of a given site to be shown in a transparent way.

The safety concept based on the containment of the radioactive waste in the CRZ is the important contribution to a defence in depth concept and for the robustness of the repository system: The safety functions available outside the CRZ are not taken into account in the German safety concept.

3.7.4. The WIPP Safety Concept

In the United States the terms “safety concept” and “safety functions” are not commonly used. The term utilized in the US that is perhaps the most appropriately related to the safety concept is “performance goal”. In the US, an assessment of repository safety after closure addresses the ability of a site and repository facility to meet safety standards and to provide for the safety functions of the engineered and/or geological components.

The US regulation 10 CFR 63.113 states for example that the geologic repository must include multiple barriers, consisting of both natural barriers and an engineered barrier system. In addition the engineered barrier system must be designed so that, working in combination with natural barriers, releases of radionuclides into the accessible environment and radiological exposures to the reasonably maximally exposed individual are within the limits specified by US regulation (US-NRC, 2013).

In addition to these statements about a multiple barrier system, US regulation 40 CFR 191[10] stipulates so-called “Containment requirements”, which refer to the likelihood of exceeding future dose quantities.

Moreover, the WIPP repository is required to have disposal system barriers to meet specific regulatory requirements of 40 CFR § 194.44(a)[11] which “prevent or substantially delay the movement of water or radionuclides toward the accessible environment”.

The DOE has proposed shaft seals, borehole plugs, and panel closures as engineered barriers in the CCA. However, the Environmental Protection Agency (EPA) concluded in its certification decision, that “only MgO meets the regulatory definition of an engineered barrier” (US DOE, 2004; pp. 3-14 to 3-15). While shaft seals, panel closures, and borehole plugs are not considered engineered barriers, they are important physical elements of the WIPP disposal system (elucidated in Section 4.2.5 of the present report: “*Designs considered in US*”).

Magnesium oxide, MgO, is used in the WIPP to meet the requirements for multiple natural and engineered barriers. It acts as an engineered barrier by decreasing actinide solubilities through the consumption of essentially all carbon dioxide possibly produced by microbial activity. Since microbial activity is an uncertain process, the MgO engineered barrier reduces uncertainty in the repository chemical conditions (US-DOE, 2014; p. 44-3).

An important component of the WIPP facility, and a crucial aspect of the salt repository Safety Case, is the shaft sealing system, to which the following basic functions have been attributed (Hansen, 2011; p. 21) (US DOE, 2004; p. 3-17):

- Limit waste constituents reaching regulatory boundaries
- Restrict formation water flow through the seal system
- Use materials possessing mechanical and chemical compatibility
- Protect against structural failure of system components
- Limit subsidence and prevent accidental entry
- Utilize available construction methods and materials

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[10] 40 CFR 191.13 - Containment requirements; http://www.law.cornell.edu/cfr/text/40/191.13
[11] 40 CFR 194.44 - Engineered barriers; http://www.law.cornell.edu/cfr/text/40/194.44

To ensure that waste shafts inhibit radionuclide release, the DOE has taken a “defense in depth” approach in the sealing systems, which incorporate compacted crushed salt and multiple engineered materials to reduce the systems’ permeability to approach permeability values of unexcavated, intact salt (US DOE, 2004; p. 3-3). Shaft sealing will begin at the end of the WIPP’s operational period as part of its final facility closure process (US DOE, 2004; pp. 3-4 and 3-5).

The functions allocated to the shaft were assessed as part of a study to investigate the feasibility and utility of developing a defensible safety case for disposal of United States Department of Energy (U.S. DOE) high-level waste (HLW) and DOE spent nuclear fuel (SNF) in a conceptual deep geologic repository that is assumed to be located in a bedded salt formation of the Delaware Basin (MacKinnon, 2012; p. iii). It is expected that the shaft seal system for a repository containing DOE HLW and SNF will meet requirements associated with the repository system performance (MacKinnon, 2012; p. 16).

The WIPP safety concept has been translated into a design of the actual repository, which is elucidated in more detail in Section 4.2.5 “*Designs considered in US*” of the present report.

3.8. Safety criteria

The IAEA SSG-23 provides guidance on the topic of safety criteria that need to be established in a Safety Case for the geological disposal or radioactive waste (IAEA, 2012; p. 37,38).

Safe operation of a disposal facility requires the establishment of safety criteria that have to be complied with, for example limits on the waste categories and inventory to be disposed of. These limits and conditions are derived from a combination of specific regulatory requirements and site-specific safety assessment considerations, in particular, from the operational and long-term assessments.

The limits and conditions are not constrained to radiological aspects. They could also be related to chemical and physical characteristics of the waste, as well as specifications for the design of the engineered barriers.

To provide evidence that the disposal facility is performing as expected and that system components are fulfilling their safety functions, monitoring programmes should be developed and implemented, including establishing background levels and measuring potential releases to environmental media (e.g. soil, surface water, ground water and atmosphere). Such monitoring needs to be undertaken systematically throughout the lifecycle of the disposal facility. On the basis of monitoring data, system performance should be reviewed and if necessary corrective actions should be considered.

A key consideration for any repository operator or regulator is to ensure that the wastes disposed to any repository are consistent with the safety case for the facility. This is achieved by setting waste acceptance criteria that include requirements derived from the safety case. The criteria may cover a range of different characteristics of the waste e.g.:

- limits on the categories of waste acceptable for disposal;
- constraints on the total quantity or concentration of certain radionuclides;
- limits on certain chemical materials that might enhance contaminant transport;
- limits on voidage;
- requirements for the use of certain sorts of container;
- specifications for waste conditioning;
- controls on physical properties of the waste.

In the following sections several of these aspects will be assessed for the geological disposal of radioactive waste in rock salt. As it is presently not foreseen that a disposal facility in rock salt will be implemented in the Netherlands within a manageable time frame, the focus of the present report is on the long-term safety more than on the operational safety.

One important use of quantitative assessment results is for comparison with safety criteria; in particular with dose and risk limits or constraints. In addition, complementary safety and performance indicators can be used for the evaluation and appraisal of the results of calculations. The following sections elucidate the various types of indicators.

3.8.1. Types of indicators and terminology

For a comprehensive summary of the different types of indicators is referenced to (Rosca-Bocancea, 2013), treating the development of safety and performance indicators in geological disposal. Only the main concepts will be mentioned in the following.

Key to the safety assessment is the post-closure radiological impact. This requires analysis of the long-term evolution of a disposal system and its components, quantification of the performance of the engineered barriers and evaluation of radiological dose and/or associated risk as end-points of the assessment. The calculated dose and/or risk are compared to regulatory defined limits in order to demonstrate the 'safety' of the system (e.g. IAEA, 2012; p. 46).

Calculated dose and risk are examples of *indicators*. Within the concept of a safety case, an indicator is a characteristic or consequence of a disposal system that can be measured or calculated and eventually compared to rigid or more loosely defined measures or 'yardsticks' in order to formulate such arguments (IAEA, 2003; p. 5).

A general distinction that should be made is between *safety indicators* and *performance indicators,* following the recommendations of the IAEA made in 2003. In addition, sometimes *safety function indicators* are distinguished related to explicit safety functions of individual repository components. These safety function indicators can be considered as a special case and application of performance indicators.

Safety indicators

A recently developed definition of the safety indicators was proposed in the EC project PAMINA (Becker, 2009; p. 9):

> *"A safety indicator is a quantity, calculable by means of suitable models, that provides a measure for the total system performance with respect to a specific safety aspect, in comparison with a reference value quantifying a global or local level that can be proven, or is at least commonly considered, to be safe."*

Usually the following types of safety indicators are identified:

- '*dose-rate*' related indicators (individual dose rate, collective dose rate, dose rate to animals and plants);
- '*risk*' related indicators (individual risk, societal risk);
- '*concentration*' related indicators (concentration in groundwater, concentration in biosphere water, concentration in soil, concentration in air);
- '*flux*' related indicators (radiotoxicity release).

Safety indicators must be compared with independent quantities, known as *reference values*, which represent some minimum measure of safety that is generally considered to be acceptable (NEA, 2012; p. 12). However, the derivation of appropriate reference values is identified as one of the major difficulties in the use of safety indicators, (Becker, 2009; p. 70), (NEA, 2012; p. 5). There is only a limited number of universally applicable reference values that may be used in all Safety Cases, e.g. such as internationally agreed drinking water standards (NEA, 2012; p. 5). In most cases site-specific reference data are used, because these provide the most relevant situational context.

Performance indicators

The EC's PAMINA project built on the outcome of the FP5 project SPIN and recommended slightly different definition of the term (Becker, 2009; p. 10):

> *"A performance indicator is a quantity, calculable by means of appropriate models, that provides a measure for the performance of a system component, several components or the whole system."*

Because performance indicators provide a measure of the behaviour of an individual repository component or sub-system, they are usually more concept or site-specific than safety indicators. Performance indicators may be compared with independent quantities, known as indicator criteria, although these are not essential for their application and often, no meaningful indicator criteria may be available for comparison with a performance indicator. Where these are available, indicator criteria may be derived from independent modelling, laboratory studies or, occasionally, natural analogue studies (e.g. to provide a measure of long-term metal corrosion rates) (NEA, 2012; p. 5).

Performance indicators can be divided into groups based on their nature and the information they provide. Each group of indicators can be applied to one or more relevant compartments of a repository concept. It should be noted that these groups are not independent if the status of one of the barriers could have a significant impact on the flux of radionuclides across it and, consequently, the content of radionuclides in the compartments on either side. As consequence, a certain redundancy in information exists. Examples of performance indicators are (NEA, 2012; Table 2):

- Activity in compartments
- Radiotoxicity inventory in compartments
- Radiotoxicity flux from compartment
- Integrated radiotoxicity flux from compartments
- Radiotoxicity flux from the geosphere
- Radiotoxicity concentration in biosphere water

Safety Function Indicators

In some national disposal concepts, as well as in the OPERA Research Plan (Verhoef et al., 2011), safety functions are defined that can be attributed to individual repository components. In (NEA, 2012; p. 5), it is mentioned that some organisations define a set of indicators that can be related to these safety functions. These indicators are called *Safety function indicators* or *Performance indicators based on safety functions*. These are measurable or calculable properties that indicate the extent to which the system components achieve their safety function. Safety function indicators are usually compared with indicator criteria which define the quantitative limits (maximum or minimum conditions) that are the boundary conditions under which the matching safety function may be maintained. These are generally derived from independent studies.

3.8.2. Reference values and safety criteria

A difficulty with the use of safety indicators lies in the derivation of appropriate *reference values*. There is a trend, however, towards using site-specific reference values, such as local or regional groundwater concentrations, because these are often considered to provide the most relevant situational context. Several national repository development organisations anticipate deriving site-specific reference values from characterisation studies, once sites have been chosen.

Reference values for safety indicators and indicator criteria for performance indicators may be derived from a number of sources. National regulations typically provide the formal limits and constraints that are compared to the primary safety indicators of dose or risk. Apart from the regulatory requirements, the following sources for reference values and indicator criteria for comparison with complementary indicators can be identified (IAEA, 2003; p. 6):

- Safety recommendations from international organisations that may relate to radiological safety or broader health and environmental safety.
- The principle that the repository should not significantly perturb the radiological or chemical conditions naturally present in the environment. Corresponding yardsticks can be derived from observations (measurements) of radionuclide concentrations and fluxes in nature averaged over both global and local scales.
- Consideration of the physical processes by which the safety functions of the disposal system are provided. Such yardsticks are directly derived from system understanding, which may be developed from modelling and laboratory studies.
- The results of calculations conducted in performance assessment. Typical values for specific quantities that may be used as yardsticks can be derived by such calculations (e.g. a critical minimum container lifetime).
- Societal values or expectations. The subjective perception of different risks in human society can influence the derivation of yardsticks from other sources (e.g. by using safety margins to meet the safety culture in different countries) but also stakeholder expectations may derive minimum performance requirements as part of negotiations for a local community hosting a repository.

3.8.3. Waste Acceptance Criteria

An important aspect of the compliance with safety criteria relates to the acceptance of radioactive wastes, and any accompanying chemically hazardous wastes into salt-based repositories. IAEA guidelines mention that:

"*waste packages and unpackaged waste accepted for emplacement in a disposal facility shall conform to criteria that are fully consistent with, and are derived from, the safety case for the disposal facility in operation and after closure*" (IAEA, 2011; Requirement 20).

Examples of possible parameters for waste acceptance criteria include the characteristics and performance requirements of the waste packages and the unpackaged waste to be disposed of, such as the radionuclide content or activity limits, the heat output and the properties of the waste form and packaging.

As part of Requirement 20 it is also stated that modelling and/or testing of the behaviour of waste forms has to be undertaken to ensure the physical and chemical stability of the different waste packages and unpackaged waste under the conditions expected in the disposal facility, and to ensure their adequate performance in the event of anticipated operational occurrences or accidents.

In addition, quality control of waste packages has to be undertaken and is achieved mainly on the basis of records, preconditioning testing (e.g. of containers) and control of the conditioning process. Post-conditioning testing and the need for corrective measures have to be limited as far as practicable.

The topic of Waste Acceptance is presently only applicable at the WIPP site, see Section 3.8.6.4 for more details.

3.8.4. Approach followed in Netherlands

3.8.4.1. Safety and design requirements

An overview of the safety requirements for nuclear facilities in the Netherlands is provided in the Annex of (Verhoef, 2014).

The provisions of the European radiation protection criteria and standards as established in Council Directive 96/26/Euratom (EU, 1996) have been implemented into the Dutch legislation, in particular in the Decree on radiation protection (SZW, 2001) of the Nuclear Energy Act. All nuclear facilities in the Netherlands consequently require a permit in accordance with the Nuclear Energy Act. The relevant radiation protection criteria and standards for such permit can be found in the Decree on radiation protection which has been in force since March 1, 2002. This includes among others a legal location limit of 0.1 mSv effective dose per year for people outside the nuclear facilities as described in Article 48.1 of that Decree (SZW, 2011; p. 21).

As the basis for a permit, the radiation dose that members of the public receive as a consequence of (possible) emissions of radioactive material (into the atmosphere or surface water) or as a consequence of external irradiation have to be calculated.

3.8.4.2. Safety and performance indicators

The results of the VEOS safety assessment concerned the dose rates and estimates of the probability of occurrence. In addition to these safety indicators other indicators have been calculated as well to explain and analyse relevant processes (Prij, 1989; Chapter 5), for example:

- Salt dome internal rise rate
- Subrosion rate
- Brine migration velocity
- Temperature distributionsin salt due to the heat-generating containers
- Stress and strain near excavations
- Groundwater flow rates

The PROSA study (Prij, 1993) did not make a clear distinction between safety indicators and performance / safety function indicators. For parameters of importance in the dose calculations and with large uncertainties, probability density functions were selected to serva as a basis for the uncertainty/sensitivity analyses.

The CORA safety assessment (see also Section 5.4) concentrated on the neglection scenario, assuming an open flow path through the flooded galleries to the groundwater system. In addition to the main indicators considered in the Netherlands, i.e. dose rate and risk, there has been a focus on the self-sealing behaviour and “closure times” of plugs and sealing materials of boreholes of salt-based repositories (Grupa, 2000; Poley, 2000a). The calculated indicators were porosity and permeability.

3.8.5. Approach followed in Germany

The approach for dealing with safety and performance indicators in Germany has been described in different reports – Bollingerfehr (2013), Noseck (2012), Becker (2009) – and is summarized in the following sections.

3.8.5.1. Safety and design requirements

The German Safety Requirements stipulate the following about limits and indicators (BMU, 2010; Section 6).

The Radiological Protection Ordinance does not contain any criteria for assessing the protection of future generations and the environment from ionising radiation. In the case of a planned, operated and decommissioned final repository observing these Safety Requirements, all essential measures are taken to protect future generations and the environment from ionising radiation, which means that as a rule further evidence is no longer necessary. Internationally there is agreement that calculated or estimated risks or doses may only be interpreted as indicators of the level of protection being striven for with the final disposal. The following assessment criteria apply for these indicators:

- The integrity of the isolating rock zone (cf. Section 3.7.3 of this report) is crucial for protection from damage caused by ionising radiation during the post-closure phase. The radioactive waste must be isolated in this rock zone in such a way that it remains in situ and, at best, only minimal quantities of substances are able to exit this rock zone. Additional radiation exposure should only be able to occur in a limited area so that as small a number of people in a generation as possible can be affected.
- For the post-closure phase, evidence must be provided that for probable developments through the release of radionuclides from the emplaced radioactive waste, an additional effective dose in the range of only 10 microsieverts per year can occur for individuals. Individuals with today's life expectancy and with a lifetime of exposure are to be taken considered.
- For a less probable evolution in the post-closure phase, evidence must be provided that the additional effective dose caused by the release of radionuclides from the emplaced radioactive waste does not exceed 0.1 millisieverts per year for the individuals affected. Here too, individuals with today's life expectancy and with a lifetime of exposure are to be considered. For these types of evolutions, higher releases of radioactive substances are admissible because the occurrence of such evolutions is less probable.
- For improbable future evolutions, reasonable risks or reasonable radiation exposures have not been quantified. However, where such future evolutions may lead to high radiation exposure, it is necessary to investigate, within the context of optimisation, whether it is possible to reduce such effects with a reasonable input. However, this must not impair optimisation in relation to other possible scenarios.
- For scenarios associated with unintentional penetration of the isolating rock zone, reasonable risks or reasonable radiation exposure have not been quantified.

3.8.5.2. Radiological safety indicators

Following the requirements stipulated in the previous section, the following two radiological safety indicators are acknowledged:

- the effective dose in the biosphere, and
- a radiological indicator, which is based on the release of radionuclides from the CRZ.

The calculation of effective dose has been applied for many years and the application scheme is straightforward. Criteria for the additional effective dose are specified in the German Safety Requirements (BMU, 2010; Section 6).

To implement the specifications in the Safety Requirements for a radiological indicator the RGI (Radiologischer Geringfügigkeits-Index (index of marginal radiological impact)) was developed in the ISIBEL project (Mönig, 2012; Section 4.6.1). The calculation of the RGI is based on a stylised calculational scheme. It is assumed that the total radionuclide flux released from the CRZ (S_i) is diluted in the annual water consumption (W) of one adult individual. The calculation does not consider how the radionuclides are transported from the boundary of the CRZ to the water body used by the individual considered. To determine the radiological consequences of the radionuclide concentration in the water body, a biosphere model equivalent to the calculation of the effective dose in the biosphere is applied. The calculated exposures are normalized in order to highlight the fact that this calculation is an indicator for safety and not a prognosis of future exposures. By means of nuclide-specific dose conversion factors DKF_i [(Sv/a)/(Bq/m^3)] and the reference value for a minor release K_{RGI} [Sv/a], this results in the following equation for the indicator *RGI* (Mönig , 2012; p. 46):

$$RGI = \frac{\sum_i S_i \cdot DKF_i}{W \cdot K_{RGI}}.$$

Thus RGI is equal to the total collective dose of consuming the total radionuclide flux by the population devided by ($W \cdot K_{RGI}$). In ISIBEL, the application was set up for the assessment of a radionuclide release in the liquid phase. The concept of the RGI can easily be expanded to assess both the gaseous and liquid pathways of radionuclide migration. At the moment there are no specifications in the Safety Requirements regarding consideration of the gaseous phase and no generally accepted calculational scheme exists.

3.8.5.3. Containment of radionuclides in the CRZ

The assessment of the containment of radionuclides is based on the simplified radiological statement defined in the German Safety Requirements (BMU, 2010). The qualitative and quantitative evaluation of fluid and radionuclide transport processes yields a staged assessment (Figure 3-6). Complete containment is provided when no radionuclides are released from the CRZ. For a repository in salt this is true if no contact between any intruding solution and the waste occurs and when no radionuclides are released into the gas phase. (RGI= 0, stage 1 in Figure 3-6).

If radionuclides are released from the CRZ, safe containment has to be demonstrated. For this purpose the RGI is applied, for which the calculation scheme is elucidated in (Bollingerfehr, 2013; Section 5.9.1).

The calculation of the RGI results in an index which indicates whether the released radionuclides cause any significant harm for human health. If the RGI is below 1, a safe containment of the radionuclides within the CRZ is demonstrated (stage 2). If the RGI is above 1, the radionuclide release from CRZ is not insignificant (stage 3). This does not mean that the repository system is not safe, but an additional assessment, especially the calculation of the effective dose in the biosphere, is required in order to identify whether the consequences of the analysed scenario can be considered to meet the criteria of the Safety Requirements. If not (stage 4) the defined repository system is not suitable.

Considering the above, the parameter RGI can be regarded as a quantitative measure of the safety function “containment” in the CRZ.

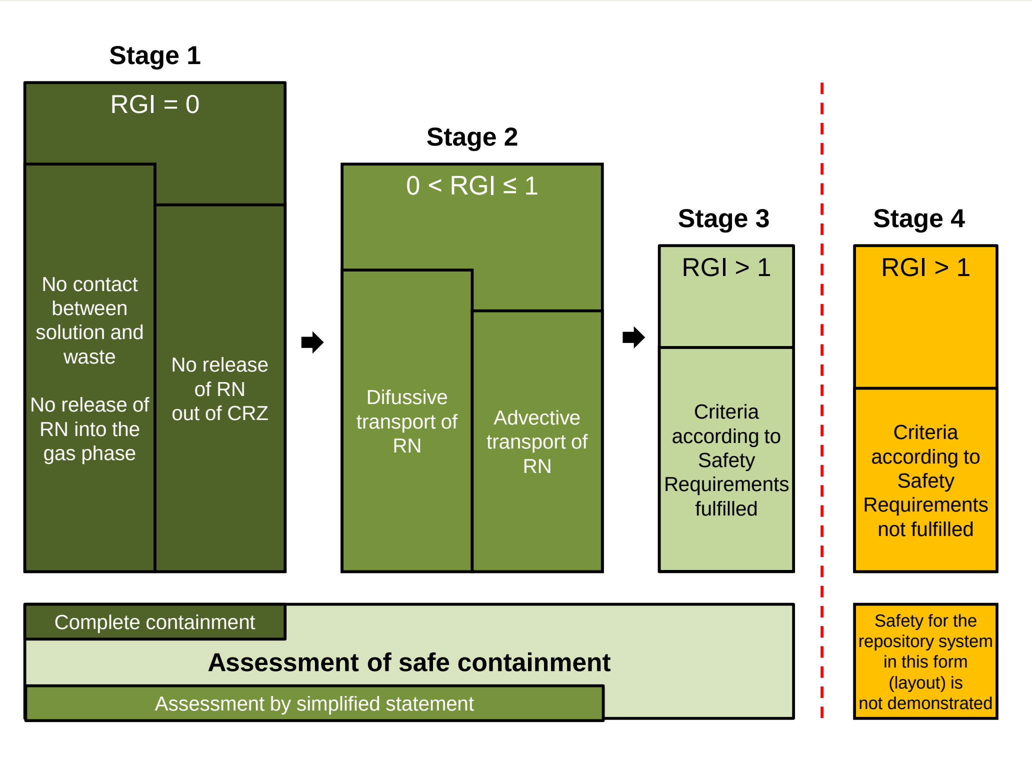

Figure 3-6 **Staged approach for the long-term safety assessment.**

3.8.6. Approach followed in the US

3.8.6.1. Requirements

Protection and isolation requirements are defined by the US EPA for WIPP (US DOE, 2004) in three parts. Subpart A addresses radiation doses during management and storage of the WIPP. Subparts B and C establish standards for long-term (over a 10,000 year period) disposal of radioactive waste, specifically for the following disposal system aspects: (1) isolation of radionuclides per the containment requirements of the disposal system; (2) radiation exposure protection for individuals for a period of 10,000 years; and (3) groundwater protection from radioactive contamination, also for a period of 10,000 years (US DOE, 2004; p. 1-6).

Requirements in 40 CFR §191.13 include active and passive institutional controls for mitigation or prevention of potential human disturbance for an extended period of time, multiple natural and engineered barriers, and other measures intended to enhance confidence in the disposal system performance (US DOE, 2004; p. 1-6). Active institutional controls include controlling access to the site, maintenance, monitoring, and clean-up; passive institutional controls include permanent markers as notification devices and archives (US DOE, 2004; p. 1-10).

3.8.6.2. Performance confirmation at WIPP

The basic US requirements have been implemented in the WIPP Performance Confirmation programme, which is extensively described in (Hansen, 2011).

The WIPP monitoring spectrum includes different categories of monitoring that apply to a disposal system, which in the relevant documentation are termed environmental monitoring, operations monitoring, and performance confirmation defined as follows (Hansen, 2011; Section 3.2):

- Environmental monitoring includes sampling and evaluation of air, surface water, groundwater, sediments, soils, and biota for radioactive contaminants. This type of monitoring determines public and environmental impact of the site. Comparisons are then possible between baseline data gathered before site operations and data generated during disposal operations.
- Operations monitoring is defined here as monitoring activities used to comply with regulatory requirements for general siting, facility operations, and decommissioning. These requirements are identified in existing regulations, state agreements or organizational agreements.
- Performance Confirmation constitutes a programme of tests, experiments, and analyses that is conducted to evaluate the adequacy of the information used to demonstrate compliance with the site specific preclosure and postclosure performance objectives. In the WIPP case, some performance confirmation monitoring started during initial site characterization.

Consequently, performance confirmation is distinct from the many other monitoring practices involved with environmental permits and repository operation. The WIPP documents refer to performance confirmation as “compliance” monitoring. Periodic review of the monitored parameters is necessary to meet the intent of the EPA’s assurance requirements applicable to WIPP, 40 CFR 191.14(b):

Disposal systems shall be monitored after disposal to detect substantial and detrimental deviations from expected performance. This monitoring shall be done with techniques that do not jeopardize the isolation of the wastes and shall be conducted until there are no significant concerns to be addressed by further monitoring.

3.8.6.3. Reference values

As part of the US WIPP programme, reference values of measured parameters, the so-called Compliance Monitoring Parameters, or “COMPs”, are indicated as Compliance Baseline values. These values are derived iteratively by evaluating monitoring data against PA expectations (Wagner, 2010; p. iv). In addition to the Compliance Baseline values, Trigger Values (TVs) are defined that are used in the compliance monitoring programme as an indicator of conditions that may require further actions should a compliance monitoring programme parameter's TV be exceeded (Wagner, 2010; p. iv).

As the Waste Isolation Pilot Plant (WIPP) project knowledge advances with the maturing monitoring programme, the basis for TVs may also change. Ten years of compliance monitoring results, performance assessment (PA) improvements and new PA results have indicated that some of the original monitoring parameter TVs must be updated to align them with expected conditions predicted or assumed in the latest baseline PA. Therefore, on a regular basis TV reports are being revised to account for these conditions and assign new TV s where needed. The procedures to obtain COMPs, as well as the presently adopted reference values and TVs are reported in (Wagner, 2010).

3.8.6.4. Waste acceptance criteria

The waste acceptance criteria (WAC) applicable to the transportation, storage, and disposal of contact-handled (CH) and remote-handled (RH) transuranic (TRU) waste at the Waste Isolation Pilot Plant (WIPP) are described in (US DOE, 2013), and are subject to regular updates. These criteria serve as the U.S. Department of Energy's (DOE’s) primary directive to ensure that CH and RH TRU waste is managed and disposed of in a manner that protects human health and safety and the environment.

In addition to these requirements and criteria, the WAC also include the subject of waste characterization relating to a determination of whether the waste is hazardous, and protocols to be used in determining compliance with the physical and chemical property criteria of the waste, including a determination of the radiological properties of CH waste. The WIPP requires radiological characterization data to:

- track the WIPP radionuclide inventory, by isotopic activity and mass, for a set of selected radionuclides
- demonstrate that each payload container disposed of at the WIPP contains TRU waste as specified, and
- verify that applicable transportation and facility limits on individual payload containers and assemblies for FGE (Fissile Gram Equivalent), PE-Ci (Plutonium-239 equivalent curie), and decay heat are not exceeded.

At present there is a programme underway in the US to assess the feasibility to also accept, besides defence waste, commercial spent fuel (Used Nuclear Fuel, UNF) and HLW at a salt-based repository. This idea has emerged as one of the waste management options after ceasing the activities performed in the Yucca Mountain project and the DOE forming the Blue Ribbon Commission on America's Nuclear Future in 2010. In (Hansen, 2011) an overview is provided about all aspects that are needed to finally emplace heat generating UNF and HLW in a salt based repository. Much of the required information is based on the experiences gained at the WIPP site, but there have been quite a number of issues identified that need to be addressed in order to come to an acceptable Safety Case for heat-generating waste.

For example, as presently there is currently no performance standard for disposal of HLW in salt in the US, consideration of HLW disposal in salt would require changes to the US legal framework (Hansen, 2011; p. 4). In addition, HLW disposal in salt would place specific requirements on PA models that are intended to demonstrate compliance with regulatory performance objectives. It also have to be assessed to see whether regulations specific to the retrievability of waste would be met by the existing technologies available for HLW disposal in salt. Another major activity would comprise in situ testing which is vital for proof-of-principle demonstrations.

3.9. Managing of uncertainties

Confidence in long-term safety can be achieved by the implementation of the safety approach and demonstration that the repository will perform as expected. Demonstrating confidence demands that uncertainties be identified and managed to the degree necessary to ensure safety.

The approach to the management of uncertainties is a part of the safety strategy (IGSC, 2011; p. 61) (NEA, 2013; p. 8). Within a step-by-step approach to the disposal facility development, information about uncertainties and perspectives on how they can be managed form an important input for the decisions to be taken at each step. How much uncertainty can be accepted at a given step depends on the decisions to be taken at that step and the potential effects of remaining uncertainties on that decision. A key issue in the Safety Case is inclusion of a register of significant uncertainties and a management process for assessing and, where appropriate, avoiding, mitigating, or reducing them.

3.9.1. Sources of uncertainty

Sources of uncertainty are categorized into the following five groups (IAEA, 2012b; Section 9.3):

- Data and parameter uncertainty, in terms of inputs, spatial and temporal variability;

- Model uncertainty, in terms of conceptual and mathematical model development;
- Future (scenario), in terms of the near-field geosphere and biosphere;
- Resource uncertainty, in terms of financial, human, technological, etc., and
- Contextual uncertainty, in terms of regulations/laws, stakeholders, etc.

Historically, especially in the context of a safety assessment, it is only the first three categories of uncertainty that are normally considered and these general categories are reflected in recent safety assessment and safety case reports (e.g. NEA, 2012; p. 40).

In the broader context of a Safety Case, uncertainties not specifically addressed in a safety assessment are an important consideration, and in the case of waste disposal, can often be a major factor in decision-making. As a consequence, resource and contextual uncertainties related to conditions and factors outside the calculational safety assessment framework should be identified as two separate sources of uncertainties to manage. Such uncertainties often prove to have more influence on the lifecycle of a disposal facility than the calculational safety assessment uncertainties.

3.9.2. Approach to managing uncertainties

Some uncertainties can be reduced by methods including additional site characterisation, design studies, fabrication and other demonstration tests, experiments both in the laboratory and in underground test facilities (NEA, 2013; p. 25). As a programme matures, studies will increasingly focus on key safety-relevant uncertainties and the specific data and measurements needed to increase confidence in system safety. For example, in situ experiments of radionuclide migration may improve confidence in the migration models or allow their improvement. In some cases, uncertainty can be managed by seeking multiple lines of evidence for particular assessment assumptions or parameters, including, for example, evidence from natural analogues to support the longevity of engineered materials. In other cases, it may be preferable to avoid the sources of uncertainty or mitigate their effects by modifications to the location or design of the repository.

The importance of addressing uncertainties in safety assessment is reflected in para. 4.59 of IAEA GSR-4 (IAEA, 2009), which states that "*Uncertainties in the safety analysis have to be characterized with respect to their source, nature and degree, using quantitative methods, professional judgement or both*." IAEA GSR-4 further requires that "*Uncertainties that may have implications for the outcome of the safety analysis and for decisions made on that basis are to be addressed in uncertainty and sensitivity analyses*."

Strategies of treating uncertainties within the safety assessment are well established. Generally, these fall into one or more of the following five strategies (NEA, 2013; Section 8.2):

- Demonstrating that the uncertainty is irrelevant to the safety assessment;
- Addressing the uncertainty explicitly – for example through a probabilistic approach or through a series of sensitivity studies;
- Bounding the uncertainty – for example by making a number of simplifying assumptions taking a conservative view, i.e. assumptions are made such that the calculated safety indicators such as dose rate or radiological risk will be overestimated;
- Ruling out the uncertain event or process – for example ruling out uncertain events on the basis of very low probability or because should the event happen, there will be more serious consequences elsewhere;
- Using an agreed stylised approach to avoid addressing the uncertainty explicitly – for example, biosphere uncertainties and uncertainties regarding future human behaviour patterns may be addressed used a stylised "reference person" and an agreement that the assessment should be based on present day conditions and technologies.

Additional information of the various sources of uncertainty and approaches to handle uncertainties is provided in (IAEA, 2012b; Section 9.3), (NEA, 2012; Chapter 8), (Becker, 2013), (NEA, 2013). In the present document various types of uncertainties are discussed in the appropriate sections.

3.9.3. Dutch approach to manage uncertainties

The PROSA (PRObabilistic Safety Assessment) in itself was a methodology to include and characterize uncertainties in the safety assessment. In PROSA the determination of the radiological effects on humans was also based on the systematics in the scenario selection and the handling of the uncertainties. (Prij, 1993; p. 1.1). In the PROSA method all relevant aspects of the repository development were considered, where uncertain parameters were described by probability density functions.

The determination of the radiological consequences of geologically disposed radioactive waste required a careful consideration of the evolution of the waste, the waste repository, the salt formation and the surrounding rocks including the fluids. This evolution is influenced by a wide variety of Features, Events and Processes (FEPs) some of which are interrelated and time dependent. The PROSA study applied a list of FEPs to define of scenario type of analysis followed by a probabilistic consequence analysis.

In PROSA the uncertainties were carefully considered, but they were restricted to the safety assessment itself. As already mentioned in the previous section a modern Safety Case should consider also other aspects of uncertainties. In the Netherlands this broadening of sources of uncertainties has also been recognized recently, viz in the Dutch Safety Strategy document OPERA-PG-COV014 (Verhoef, 2014; p. 11): "*Thus, as far as reasonably possible, events and processes that could be detrimental to isolation and containment, as well as sources of uncertainty that would hamper the evaluation of how the systems evolve over time, are avoided or reduced in magnitude, likelihood or impact by means of siting or design choices.*"

In addition it is mentioned that one of the specific OPERA objectives is to identify and conduct research to reduce uncertainties (Verhoef, 2014; p. 12). That objective is formulated in the definition of the Work Packages and Tasks defined in the OPERA Research Plan (Verhoef, 2011b).

3.10. Demonstrability

By definition the Safety Case and supporting safety assessment provide the basis for demonstration of safety of a disposal facility for radioactive waste and for its licensing (e.g. IAEA, 2013; p. 2). As a consequence, demonstrability of safety related features and aspects is an important strategic aspect of the Safety Case (IAEA, 2013; p. 25).

For disposal concepts the demonstration of safety may be ensured by means of assessment, testing or other physical demonstration of functionalities of the disposal system. Demonstration of safety may also be undertaken by physical demonstration in mock-up facilities or on the site of the disposal facility, either at the surface or underground.

Once a site has been identified and an initial engineering concept defined, the decisions may involve more detailed planning of the scope of surface and sub-surface investigations, including demonstrations of the engineering feasibility of key elements, choices between design variants and the optimisation of the underground layout.

In the past numerous tests, large-scale experiments, assessments and large-scale demonstrations have been performed to address and verify the safety of a salt-based repository. For example the German programme ISIBEL (summarized in Buhmann, 2008) and VSG[12] provide a wealth of information about these issues. Also the WIPP programme in the US elaborated a large library of information about demonstrating safety (e.g. Sevougian, 2013 and references therein).

The following sections provide examples of existing safety demonstration concepts in Germany and US respectively.

3.10.1. Safety Demonstration – The Netherlands

At present no consolidated design of a repository in rock salt in the Netherlands has been established. However, for several generic disposal concepts safety has been demonstrated on the basis of several detailed studies, scenario analyses and safety assessments, e.g. in the early eighties (Hamstra, 1981), in the VEOS study as part of the OPLA programme (OPLA, 1989), the PROSA study (Prij, 1993), and in the CORA programme (CORA, 2001).

The OPLA programme mainly focused on the technical and scientific feasibility of final disposal of radioactive waste in rock salt formations, thereby looking at different mining techniques and disposal concepts (e.g. salt mines, deep boreholes). Several aspects of safety were demonstrated in OPLA:

- Scenario analysis;
- Thermal and mechanical aspects of rock salt;
- Transport of radionuclides through the subsurface;
- Radiation damage in NaCl;
- Mining engineering;
- Feasibility testing, participation in experiments in the German Asse mine.

In relation to the last mentioned topic it is worthwhile to note that in the eighties and nineties ECN participated in several demonstration projects in the Asse mine, e.g. the "HAW Project". Within that project all aspects of the management of a disposal mine in salt for high-active heat-generating waste were aimed to be demonstrated (Vons, 1988).

The PROSA project mainly focused on demonstrating the long-term safety of geological disposal in rock salt by means of scenario analyses and safety assessment calculations.

Within the CORA programme also several aspects of safety of a salt-based repository were demonstrated, e.g.

- Assessment of a concept for the retrievable disposal in salt (METRO-I);
- Assessment of borehole concepts for the retrievable disposal in salt;
- Assessment of the behaviour of backfill material (crushed salt, BAMBUS Project);
- Radiation damage in NaCl;
- Additional safety assessments under conditions of glaciation and changes in the stress field.

Summarising, there has been substantial valuable information generated within several Dutch projects on the demonstration of safety related issues of the geological disposal in salt formations. However, the information is scattered in terms of time (several decades of research), scope (no consolidated repository design), and availability (many reports are not yet available in digital format). Much of the available information taking into account the recent developments of conceptualising the Safety Case of geological disposal has been summarized in (Hart, 2015).

[12] http://www.grs.de/endlagersicherheit/gorleben/ergebnisse (last accessed 25 August 2014)

3.10.2. Safety Demonstration concept – example from Germany

The fundamentals of the German safety demonstration concept are specified in the German Safety Requirements (BMU, 2010). A first implementation of such a concept was developed in ISIBEL (Buhmann, 2008) and later refined in project VSG (Mönig, 2012). In the ISIBEL project the concept focused on the demonstration of the long-term safe containment of the waste by demonstrating the integrity of the geotechnical barriers and the geological barrier. An evaluation of radionuclide release was carried out for evolutions of the repository system for which an impairment of the barrier integrity, and therefore the development of a continuous pathway for radionuclides, could not be excluded. Whether these evolutions are probable or less probable, or whether they can be excluded from further consideration, is then covered by the scenario analysis (Buhmann, 2008).

According to this concept the decisive elements are:

- the demonstration of the integrity of the geological barrier,
- the demonstration of the integrity of the geotechnical barriers,
- the scenario analysis, and
- the evaluation of release scenarios.

These elements were complemented by concepts on how to consider subcriticality, non-radiological protection goals and operational safety in the safety demonstration. Additionally, the safety demonstration concept was supported by reports on how to deal with uncertainties and how to evaluate release scenarios.

The safety concept and the elements of the safety demonstration were further refined in the project VSG (Mönig, 2012). The key elements of the safety demonstration concept are schematically shown in Figure 3-7.

On the basis of a comprehensive handling of uncertainties, particularly the handling of scenario uncertainties by a scenario analysis, the containment of the waste is evaluated for all scenarios that need to be considered in order to cover uncertainties regarding the future evolution of the repository system. This evaluation includes the assessment of:

- the permanence of the CRZ,
- the integrity of the geological barrier for probable scenarios,
- the integrity of the geotechnical barriers for probable scenarios, and
- the releases of radionuclides from the CRZ employing a suitable radiological safety indicator for probable and less probable scenarios.

If the safe containment of the radionuclides in the CRZ can be demonstrated, this assessment is extended by evaluating subcriticality, non-radiological protection goals and operational safety. The safety demonstration concept now also includes an assessment of human intrusion as required by the German Safety Requirements, the results being used for optimisation of the repository concept. These additional elements can be regarded as stand-alone analyses and are represented as single (blue) columns in Figure 3-7. The red elements form the safety analysis, the core of the safety assessment.

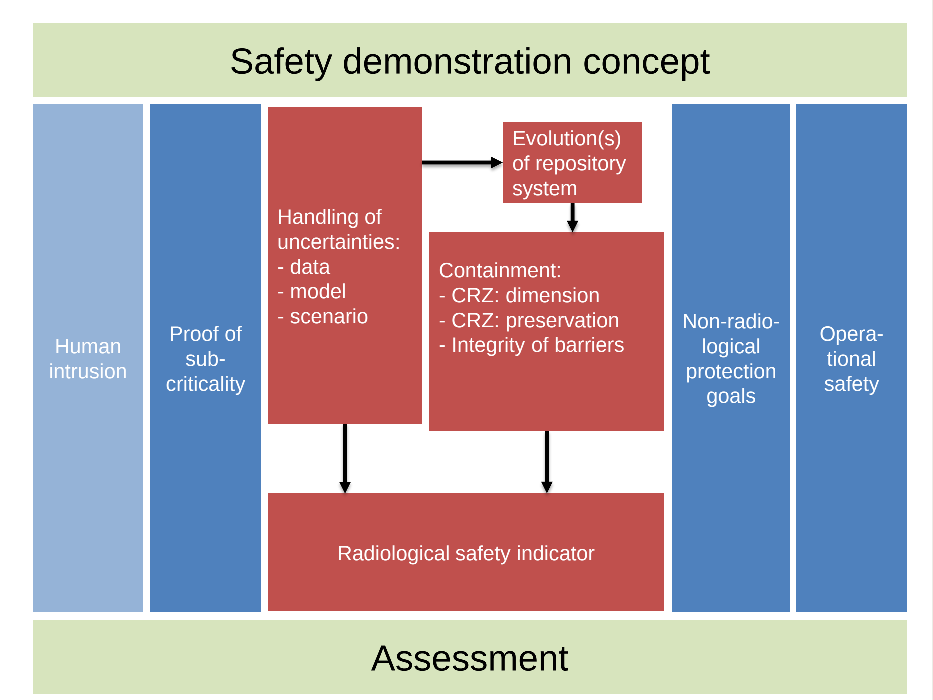

Figure 3-7 Principle Elements of the German safety demonstration concept, red parts are elements of the safety analysis, red and blue are elements of the safety assessment.

The safety demonstration concept was developed for salt, but the basic elements in Figure 3-7 must be assessed for any host rock type considered for a HLW repository in Germany.

3.10.3. Safety Demonstration – the WIPP

The Waste Isolation Pilot Plant (WIPP) is the DOE (United States Department of Energy) waste disposal facility designed to safely isolate non heat-generating defence-related transuranic (TRU) waste. From commencement of operations in March 1999 through May 2011, the DOE has emplaced over 74,500 m^3 of defence-related TRU wastes in the New Mexico-based repository (NEA, 2011).

Recently, a programme has been established to investigate the feasibility and utility of developing a defensible Safety Case for disposal of heat-generating U.S. DOE high-level waste (HLW) and DOE spent nuclear fuel (SNF) in a conceptual deep geologic repository that is assumed to be located in a bedded salt formation of the Delaware Basin. Lessons learned from siting and operating the WIPP facility can be used to support the development of an HLW/SNF disposal facility in salt, particularly since the original design concepts and siting requirements for WIPP were based on the intent to dispose of HLW in addition to TRU waste (MacKinnon, 2012; p. 9).

Lessons learned from DOE‘s experience on the WIPP and Yucca Mountain Project (YMP), and collaborations with the German salt repository programme, are applied and add confidence to the conclusion that a defensible initial Safety Case can be developed at the present time using the available technical basis. This experience includes many key aspects of repository development, operations, and safety assessment, including repository and seal system design, pre-closure safety analysis, and application of performance assessment (PA) methodology. It was also demonstrated that, even if the eventual site of a DOE repository is located outside of the Delaware Basin (New Mexico), but still in bedded salt, the relevance of the WIPP experience and other technical bases would nonetheless be significant (MacKinnon, 2012; p. 1).

It was concluded in (MacKinnon, 2012; p. 22) that the potential benefits of developing a Safety Case for the disposal of heat-generating waste in salt layers include leveraging previous investments in WIPP to reduce future new repository costs, enhancing the ability to effectively plan for a repository and its licensing, and possibly expediting a schedule for a repository. A Safety Case provides the necessary structure for organising and synthesising existing salt repository science and identifying any issues and gaps pertaining to safe disposal of DOE HLW and DOE SNF in bedded salt. The Safety Case synthesis will help DOE to plan its future R&D activities for investigating salt disposal using a risk-informed approach that prioritizes test activities that include laboratory, field, and underground investigations.

This example elucidates that the demonstration of safety does not necessarily imply to build the Safety Case for the disposal of radioactive waste in a salt-based repository from scratch, but that experiences gained in other projects provide many tools and entries for safety demonstration purposes.

3.11. Monitoring

The geological disposal of radioactive waste is envisaged as a staged process that will take many decades to implement. During the long period over which a repository will be sited, constructed, operated and closed, future operators, and current and future generations will need to make decisions about how, when and if to implement various steps in the development of the repository system. Decisions at each stage of repository implementation can be supported by information provided by monitoring results (e.g. IAEA, 2012; p. 8).

In the 1984 Policy Document it is mentioned that that all hazardous and radioactive waste must be isolated, controlled and monitored (VROM,1984; p. 10). On the basis of that requirement a statement has been formulated in (Verhoef, 2014; p. 11) about monitoring and geological disposal in the Netherlands:

> *8. Geological disposal planning will assume that surveillance and monitoring will continue for as long as deemed necessary.*
>
> *Although waste retrievability allows future generations to make their own choices, it is dependent upon the technical ability and preparedness of society to keep the facility and site accessible for surveillance and monitoring over a long period, even after closure. Retrievability requires long-term arrangements for maintenance, data management, monitoring and supervision e.g. systems to facilitate retrieval of waste. Post-closure surveillance and monitoring is assumed to be continued until adequate assurance has been obtained concerning the safety of the geological disposal of waste. It important to understand, however, that the post-closure performance and safety does not depend in any way on the ability to continue monitoring.*

The subject of monitoring in deep geological disposal was recently studied extensively in the EU-FP7 project MoDeRn, “Monitoring Developments for Safe Repository Operation and Staged Closure” (White, 2013). Monitoring is a broad subject, and monitoring within a radioactive waste management programme can encompass many different objectives and activities. These objectives and activities include technical and non-technical aspects, such as monitoring changes in the inventory, changes in waste treatment and conditioning practices, and changes in the societal context. Repository monitoring is a more narrow discipline and is related to monitoring the features, events and processes (FEPs) affecting the behaviour of a geological disposal facility.

A key objective of a monitoring programme is to provide information to support the Safety Case of geological disposal (White, 2013; p. 2):

- During the early stages of repository implementation, monitoring can strengthen the understanding of system behaviour used in developing the post-closure safety case and to allow further testing of models of long-term behaviour.
- During the operation of the repository, monitoring can be undertaken to demonstrate that the assumptions in the safety case are valid and to check compliance with licence conditions.

Given the present status of the Dutch geological disposal programme, there is no urgent need to initiate activities related to monitoring at a geological disposal facility. However, since monitoring is considered relevant in the Dutch safety strategy (see Sections 3.4 and 3.5), it is important to develop monitoring strategies and programmes for performance confirmation of the repository. Included in the strategies and programmes are also the developments of tools in support of decision-making from the recorded monitoring data. These topics are subject of the MoDeRn (White, 2013) and recently commenced Modern2020[13] projects, both partially funded by the European Commission and the Dutch Ministry of Economic Affairs, in which NRG was respectively is an active participant.

At present the topic of monitoring in or at salt-based repositories is only applicable to the WIPP site, which is currently in operation in the US. Details about monitoring activities at the WIPP are described in Section 3.8.6.

3.12. Evaluation

At present the Dutch safety strategy for the disposal of radioactive waste is being elaborated (Verhoef, 2014), as part of the process of implementing Council Directive 2011/70/EURATOM of 19 July 2011 'Establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste' (EU, 2011). The Netherlands is drafting the required 'Nationaal Programma' according to the definition provided by this Directive, and a Safety Strategy document establishing the Dutch strategy for the final disposal of radioactive waste.

In the Netherlands both rock salt and clay formations are being considered as potential host rocks for geological disposal. Both host rocks in principle require a similar safety strategy, which is translated into a safety concept.

In the previously executed programmes and projects OPLA, PROSA and CORA, the Dutch safety concept for the geological disposal of radioactive waste in rock salt has been worked on. However, since 2001 no systematic activities have been performed on this topic in the Netherlands.

On the other hand, internationally there has been substantial progress on the development of the Safety Case of geological disposal in general (e.g. NEA, 2004; NEA, 2013; IAEA, 2012), and in relation to the disposal in rock salt / Zechstein, notably in the US (e.g. MacKinnon, 2012), and in Germany (Bollingerfehr, 2013). As a consequence, the Dutch safety concept for the geological disposal in rock salt needs to be upgraded. Notably the detailing of the multi-barrier system, and the designation and effectuation of the appropriate safety functions to each of the barriers, also in relation to the Dutch retrievability requirement, needs to be established.

In order to quantitatively assess the performance of a disposal system the application of indicators is imperative. The application of safety indicators like dose rate (exposure) and risk were and still are mandatory for judging the post-closure, long-term safety of the

[13] http://cordis.europa.eu/project/rcn/196921_en.html, last accessed 30 June 2015

system. The calculation of additional safety indicators like flux-related and concentration-related indicators may contribute to the overall understanding of the disposal system. An imperative of the use of safety indicators is the designation of appropriate reference values, against which the values of the safety indicators should be validated. However, a difficulty with the use of safety indicators lies in the derivation of appropriate reference values.

Except for dose rate and risk the derivation of reference values has however not been established in the Netherlands, unlike in other countries, and should be considered in the future.

Although at present no consolidated design of a repository in rock salt in the Netherlands has been established, for several generic disposal concepts safety has been demonstrated on the basis of several detailed studies, scenario analyses and safety assessment. However, the information is scattered in terms of time (several decades of research), scope (no consolidated repository design), and availability (many reports are not yet available in digital format). Although the present report summarizes many of these aspects, it was not feasible, or even possible, within the OSSC project to retrieve all experimental details or performed safety assessmewnts, and to digitize the available reports. This aspect of information consolidation and quality assurance should be dealt with in a next phase of the Dutch salt Safety Case.

Taking into account the above-mentioned topics it is recommended to critically review the Dutch safety concept of final disposal in rock salt, both in the light of recent international developments and the most recently adopted Dutch strategy.

4. System Description

4.1. Objective and Scope

The Safety Case component “System Description” presents the characteristics of the waste to be disposed of, the designs of the facility in which the waste is disposed, the salt formation wherein the facility is constructed, the surrounding and overlying sediments on the salt formations and the biosphere. The boundary conditions and constraints of the Safety Case component “System Description” are embedded in and provided by the “Safety Case Context” (Chapter 2) and the “Safety Strategy” (Chapter 3).

It must be realized that until now only generic designs have been considered in the Netherlands, in generic salt formations without site specific information.

4.2. Facility Designs

4.2.1. Early design studies

The radioactive waste related research start in 1971 at the then Reactor Centrum Nederland and was focussed on the salt formations in the northern and eastern part of the country (RCN, 1972). Important elements for the choice of salt formations were the good mining properties, the almost negligible amount of water, the good heat conduction properties and the self-healing capacity of salt. Natural processes recognised to be able to disturb the isolation of the waste are subrosion[14] and diapirism[15]. Based on generic safety evaluation and of conceptual designs as illustrated in Figure 4-1 (Hamstra, 1981; p.108[16]) it was concluded that the risk associated with these natural processes could be made sufficiently low by selecting a suitable site (ICK, 1979). Based on the findings of the research carried out a rather detailed list of criteria for the selection of a salt formation has been formulated in (ICK, 1979).

It was also recognized that the isolation of the waste could be disturbed by interactions between the salt and the waste; to be more precise due to the heat of the heat generation of a portion of the HLW resulting from the spent fuel of the NPP’s. The first approach to cope with this problem was the formulation of criteria for the temperature rises keeping the temperatures below the dehydration temperature of some hydrate minerals such as polyhalite, carnallite and bischofite. In (Hamstra, 1979; p. 25) the following limits were considered to be sufficient:

- ✓ 60°C for the salt in the 200 m thick isolation shield around the repository;
- ✓ 100°C as the maximum global salt temperature in the repository;
- ✓ 150°C as the maximum local temperature in the small area around the boreholes;
- ✓ 60°C as the maximum temperature in the galleries during the operational period (no ventilation taken into account).

Hamstra et al also showed (Hamstra, 1979; pp. 17-26) that the layout of the repository for all the radioactive waste resulting from 40 years operation of nuclear power plants with a capacity of 25,000 MWe in a medium sized salt dome can be made such that these temperature limits can be kept.

[14] Subrosion is the dissolution of the salt formation by the groundwater.

[15] Diapirism is a slow rise of the salt formation caused by density differences between salt and surrounding sediments.

[16] The mine was designed for hosting the waste resulting from NPP’s with a total capacity of 1000 GWyear (31,5 EJ) (Hamstra, 1981; p.213).

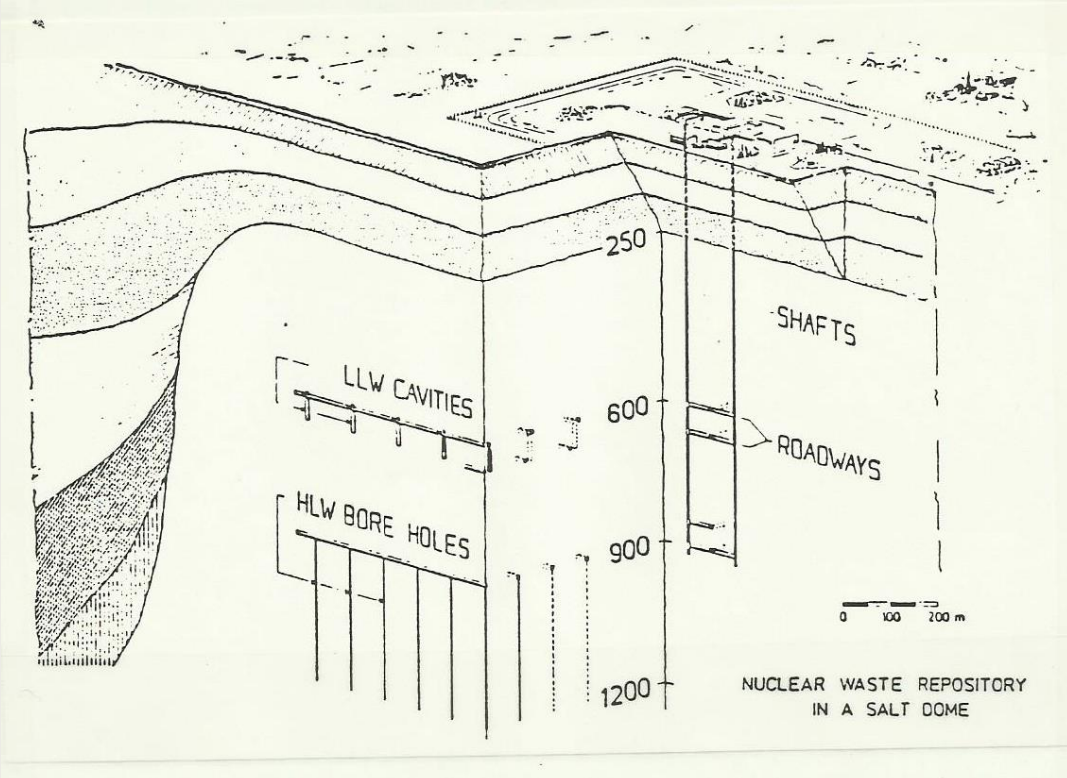

Figure 4-1 Artist's impression of a nuclear waste repository (1981).

Based on this work and that of the IFCE (IFCE, 1980) the Dutch Government reached the conclusion that "the method of disposal of radioactive waste in deep-lying stable geological formations, without unacceptable risks to persons and their environment, is in principle available". This statement of principle was to be validated as soon as possible by means of field studies in five salt domes located in the northern and eastern part of the Netherlands. In addition, the results of those field studies would have to be available before 1981, the beginning of the government organised Public Debate on Energy Policy. However, local and regional governments were opposed to test drillings required for these investigations. On February 25, 1980 the Lower House passed the Lansink motion on this matter, which resulted in these test drillings being deferred until the end of the above-mentioned Public Debate.

This Public Debate has been designed with a view to record all expressions of opinions concerning future energy policy, and assessing the consequences of those opinions. In its final report published in 1984, the Steering Group for the Public Debate stated that, with respect to the disposal of radioactive waste, there were widespread doubts about selection criteria for salt domes formulated by ICK. The Steering Group also found that there was a large measure of reservation towards test drillings in the northern provinces of the Netherlands. "Although final solutions for high-level waste disposal were not yet applied in any country whatsoever", the Steering Group found that "in circles of expert and official organisations such as IAEA and OECD and the European Community there was a growing confidence that is was possible and feasible for the problem of radioactive waste to be solved within 10 to 20 years, in a safe and affordable way" The Steering Group recommended that rigorous efforts should be made towards that solution. On the basis of this recommendation the government formed a scientific committee, the committee OPLA (OPberging te LAnd; Dutch for disposal onshore), with the task to define and execute a research programme on the disposal of radioactive waste.

In 1984 the OPLA committee launched its research proposal setting forth how research into a potential geological radioactive waste repository should be arranged in order to arrive at a sound decision. This research programme covers three consecutive stages.

Phase 1 covered tentative research, comprising laboratory investigations, office studies and participation in foreign projects. *Phase 2* covered research in support of the models and analyses developed in phase 1, requiring e.g. tentative field explorations such as soil drilling for groundwater studies and the like. *Phase 3* covered tentative field explorations more specifically concentrated on the appropriate geological formation.

The first phase of this research programme, planned to be finalised in 1988, was approved by the Parliament in 1985. The final report of this phase became available in 1989 (Prij, 1989). It was concluded that about 20 sites should be considered to be used for final and safe disposal of the radioactive waste. A further selection with respect to field studies to verify the assumptions made in phase 1 could not be made with the existing data and analytical tools. So it was recommended to extend the Phase 1 research with additional work aiming at the selection of some sites for further field study. This research, OPLA phase 1A, started in July 1990 and was finished in 1993.

4.2.2. Designs considered in OPLA

In OPLA three generic salt formations were taken as possible host rocks. Schematic pictures are given below in Figure 4-2 (OPLA, 1989; p. 34).

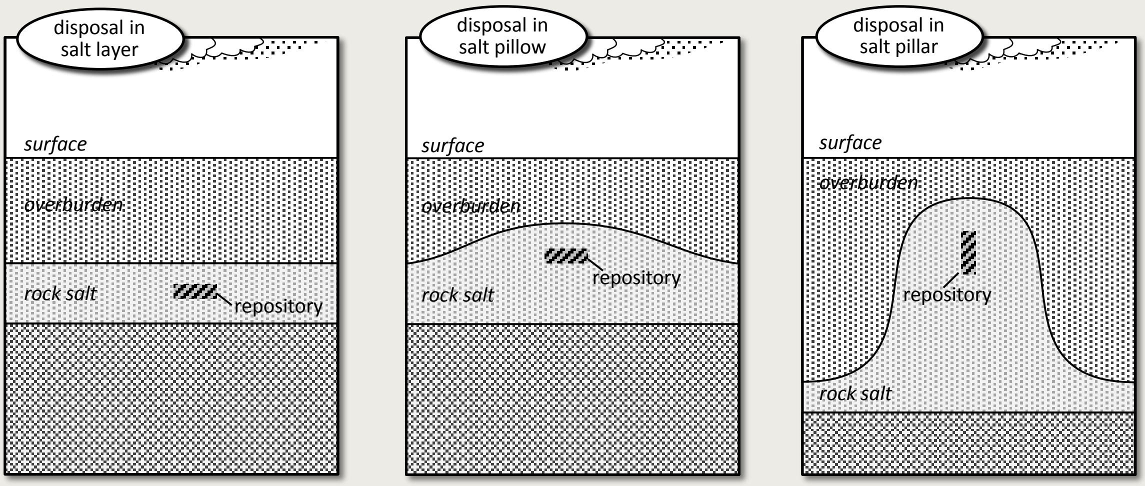

Figure 4-2 Generic salt formations considered in OPLA (1989).

The disposal technique studied in the OPLA research programme are:

- a conventional mine. The mine is made with standard mining techniques and consists of boreholes for all heat-generating waste and chambers or caverns for the remaining MLW and LLW. The mine can be made in a salt dome and in a salt pillow.
- deep boreholes and caverns. The deep boreholes are mined from the surface and meant for the heat generating HLW. The caverns, leached from the surface, are meant for all remaining waste. This disposal technique can be applied in all type of formations.

This is schematically indicated in Figure 4-3 (OPLA, 1989; p. 21).

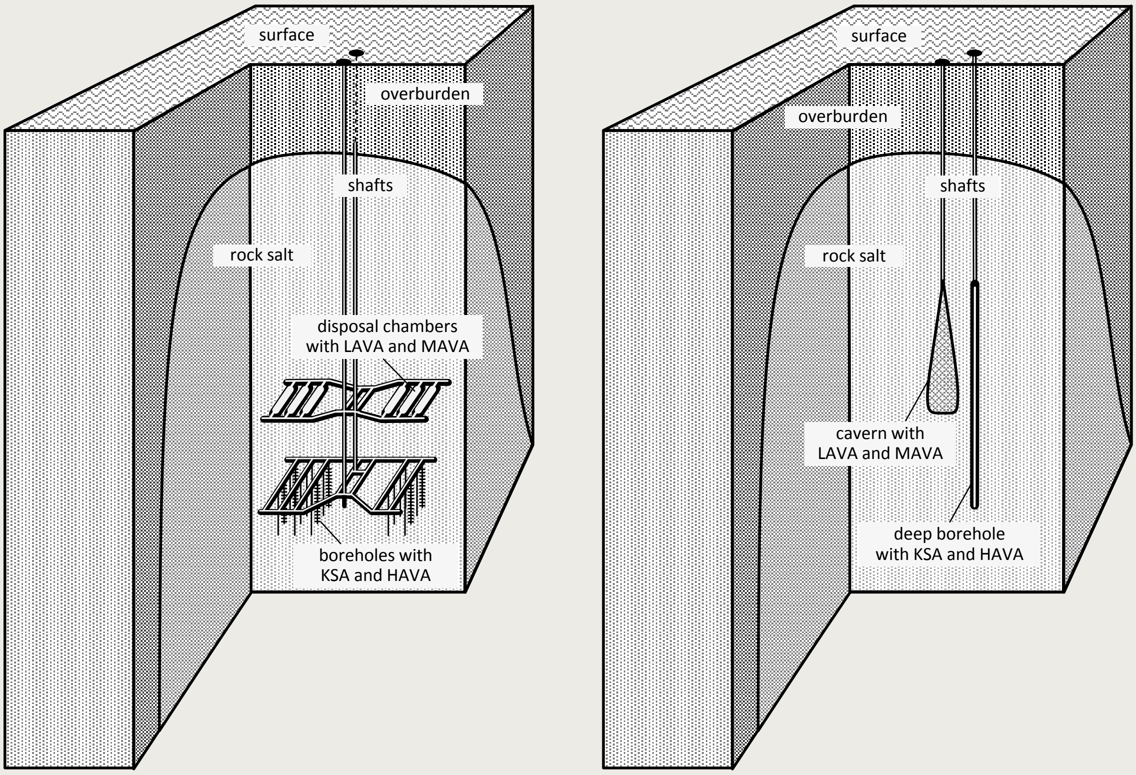

Figure 4-3 Schematic view of the disposal techniques studied in OPLA.

To support the design work an extensive research programme was carried out consisting out of experimental as well as theoretical work. The experiments were performed in the former salt mine Asse near Braunschweig (Germany). As this salt mine is situated in the same geological area as the salt domes in the Netherlands it can be expected that the behaviour of the rock salt is similar. As more than 98% of the radio nuclide inventory is concentrated in the deep dry drilled bore holes priority was given tot to this HLW area of the disposal facility. The first experiment was the dry drilling experiment of a vertical borehole aiming at the demonstration of the dry drilling technique. The successful dry drilling indicated the attractiveness and feasibility of an optimal use of the vertical dimension of the host rock for disposal purposes in a salt dome. The borehole itself offered an excellent possibility for in situ experiments. As vitrified HLW also produces heat from radioactive decay research was carried out on the effects of this heat on the mechanical behaviour of rock salt. This work (Prij, 1991b) was focussed on:

- The development of calculation models for the thermo-mechanical behaviour of rock salt
- Validation of models by means of experiments
- Derivation of constitutive relations for rock salt
- Analyses of the thermo-mechanical behaviour under repository relevant conditions
- Implications for the post closure evaluation.

The layout of the disposal mine was determined with help of the calculational tools developed.

Another important part of the research was on the effect of the radiation on the rock salt (Hartog, 1988). Although some uncertainty remains this research did not lead to changes in the design.

In PROSA the disposal concept consisted of a conventional mine made with standard mining techniques, and consisting of:

- A central field with the drifts from the shafts;
- A chamber wing for the non-heat producing wastes;
- A borehole wing for the heat producing high level wastes.

The PROSA disposal concept was intended for all categories of waste, viz. HLW, MLW and LLW. The heat producing waste (HLW + MLW) would be placed into bereholes whereas the remaining medium and low activity waste would be loaded into chambers. The boreholes for the HLW would be drilled from the level of the galleries in the two wings of the mine as shown in Figure 4-4 (Prij, 1993; Fig. 5.1). The access to these wings would be via two pairs of flank galleries from (the shafts in) the central area of the mine.

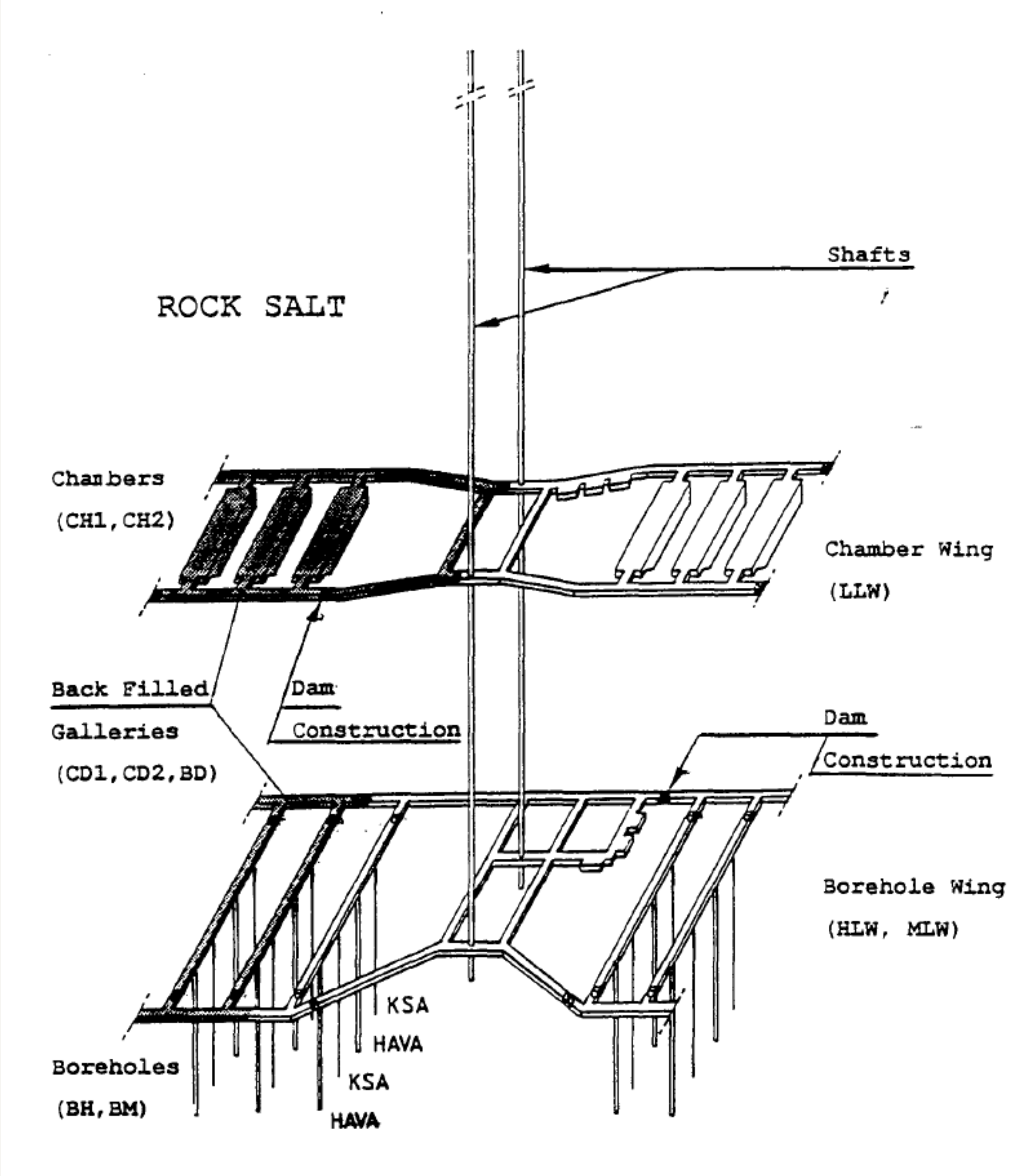

Figure 4-4 Schematic view of the disposal facility studied in PROSA.

In 1993 the policy directive of the Dutch Government decrees that deep underground disposal of highly toxic waste, such as radioactive waste, would only be permitted if that waste remains retrievable for a long time. Consequently OPLA phase 2 and 3 were not started but a new research programme, called CORA, was started which focussed on the technical feasibility of long-term retrievability of the waste. The final report was presented in 2001 (CORA, 2001).

4.2.3. Designs considered in CORA

As part of the programme a disposal concept has been developed, that take retrievability explicitly into account: the TORAD-B (Poley, 2000a, see Figure 4-5) and METRO-I design (see Figure 4-6; Vate, 2011; p.13).

There are two design aspects to the implementation of lengthy retrievability:

1) The 'compartments' in the facility where the waste is disposed of, the so-called disposal cells, have to be designed in such a way that the waste canister can be retrieved relatively easily. The retrieval of the canister from the disposal cell should not be much more complicated than the emplacement of the canister into the disposal cell.
2) The access shafts and access galleries cannot be sealed and closed without losing some of the potential to retrieve the waste. Once these galleries and shafts are closed and sealed, the only way to retrieve the waste is to build a new mining facility.

The TORAD-B design comprises a steel liner for enhanced stability and to facilitate the retrieval of emplaced waste canisters.

In the METRO-I concept the disposal of the waste is a long-term operation that is carried out in a number of stages. The decision to move from one stage to the next can be deferred as required. Once the chosen waste disposal strategy is trusted sufficiently, one may decide to advance to the next stage. The designers of the METRO-I disposal concept assume that a phased disposal strategy provides 'passive safety', i.e. even in the case of unexpected events during the operational phase, the facility as a whole will evolve into a safe condition without a need for human actions. The design considered in CORA consist of short horizontal boreholes (Heijdra, 1997), drilled from a gallery see Figure 4-7 (Vate, 2001; Figure 2). A schematic of the emplacement and retrieval of HLW canisters is depicted in Figure 4-7 (Grupa, 2000; Figuur 3.1 and Figuur 5.1).

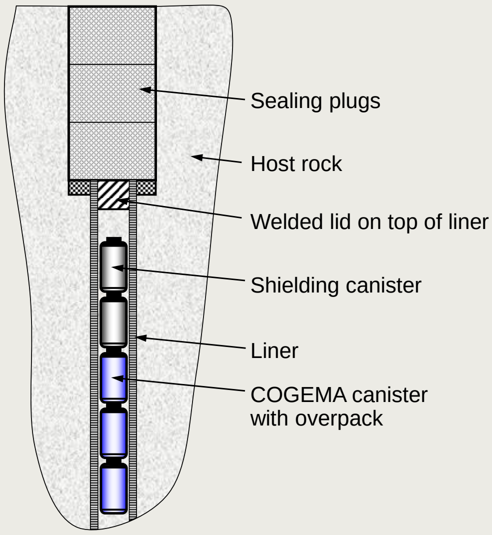

Figure 4-5 Schematic view of the retrievable design with vertical boreholes considered in CORA.

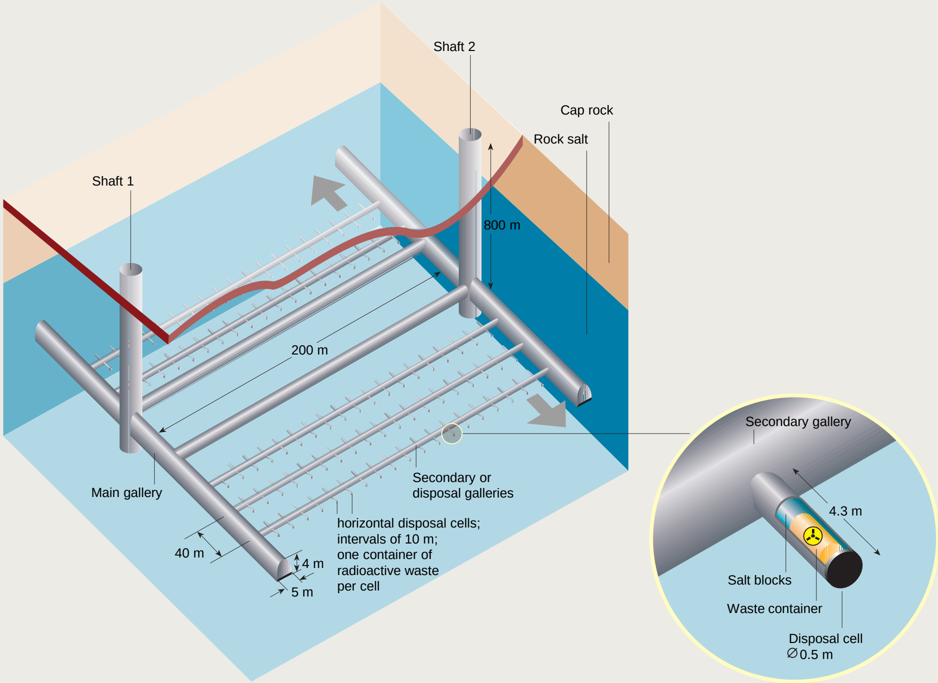

Figure 4-6 Lay-out of the METRO-I disposal concept.

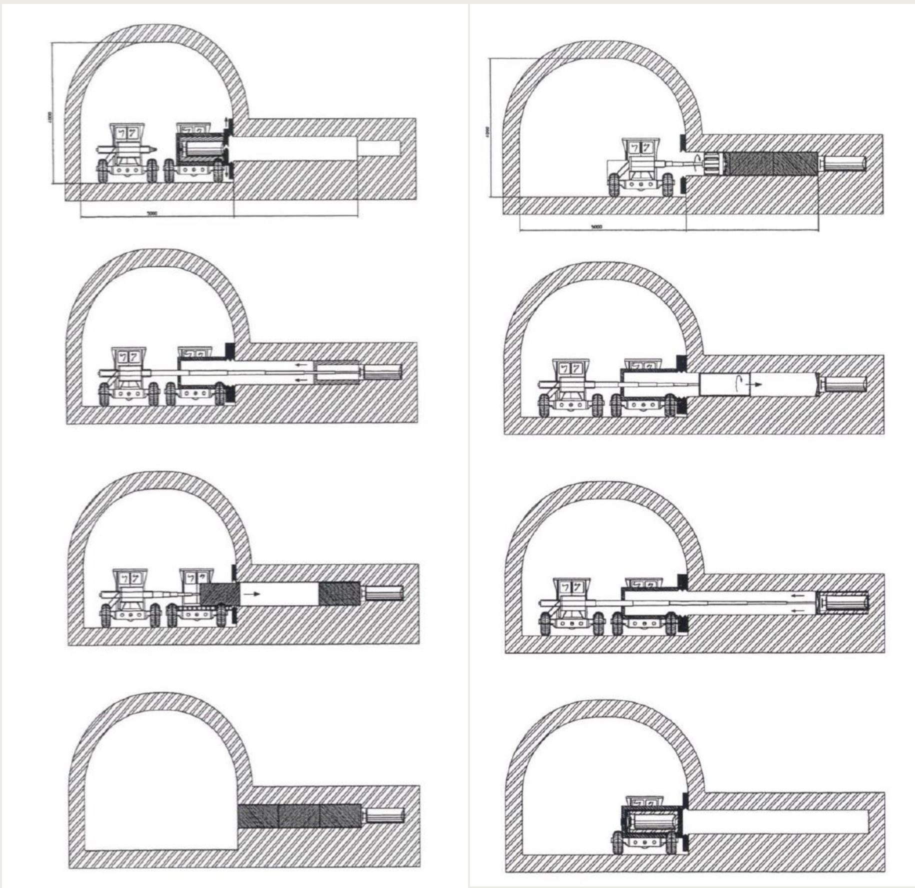

Figure 4-7 Schematic view of the retrievable disposal design considered in CORA. Left the emplacement, and right the retrieval of an HLW canister.

The consequences for the safety of the METRO-I disposal concept have also been analysed in CORA-4 (Grupa, 2000) see also Section 5.4.

4.2.4. Designs considered in Germany

According to the general approach of the safety concept there are two main design requirements. The first is to enclose the emplaced waste canisters quickly and tightly as possible within the containment providing rock zone (CRZ, see Bollingerfehr, 2013; p. 13) and the second states that the CRZ must remain intact and that its barrier function is not impaired by internal or external processes and events. Two variants for the design of the repository in the Gorleben salt formation were developed (Bollingerfehr, 2013; p. 38).

Variant 1: Emplacement of all heat generating radioactive waste in POLLUX® and CASTOR® casks in horizontal drifts. In addition – for comparison only – the emplacement of all heat-generating radioactive waste in transport and storage casks (CASTOR®) in horizontal boreholes was considered.

Variant 2: Emplacement of all heat-generating radioactive waste in different types of retrievable canisters in long vertical boreholes. As an option, the emplacement of a certain amount of radioactive waste with negligible heat generation in a separate area of the salt dome in horizontal emplacement chambers was considered, see Figure 4-8.

Details of the design were determined with thermo-mechanical analyses.

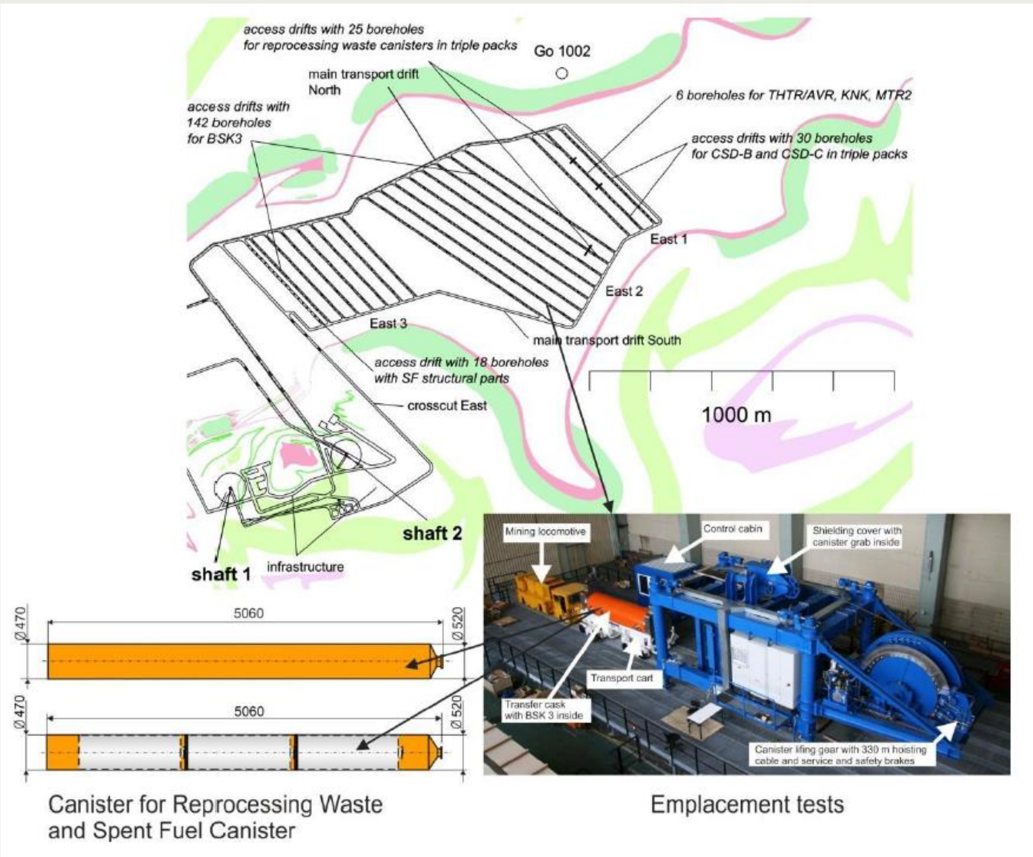

Figure 4-8 Repository design variant 2: Emplacement of all heat-generating waste in lined vertical boreholes (adjusted to the assumed geological structure).

4.2.5. Designs considered in US

Understanding the general performance criteria for a geologic waste disposal system, scientists at Oak Ridge National Laboratories (ORNL) emphasized the importance of the following features for a waste disposal site in salt (ORNL, 1973 as cited in US DOE, 2004; p. 1-13):

- the ability of salt to creep and eventually encapsulate waste;
- purity of the salt in order to minimize unfavourable or complicating properties;
- isolation from aquifers, reducing possible contact with circulating groundwater;
- tectonic stability, ensuring that isolation of the waste is secure long-term;
- minimal presence of boreholes, reducing the possibility that these could become conduits for release of waste or dissolution; and
- minimal resource development activity, effectively minimising disruption possibilities.

This broad understanding of key features required for a secure waste disposal system created a conceptual model that linked those features with factors that affect the system (ibid.). The construction of this disposal system, the Waste Isolation Pilot Plant (WIPP) construction began in the early 1980s following a site selection study which concluded with the current selection near Carlsbad, based on extensive geotechnical research and supplemental testing. The specific area for the WIPP was selected with this conceptual model in mind in that the bedded salt is continuous throughout a large geographic area. In addition, the repository horizon is relatively pure as no anhydrite or polyhalite interbeds nor major clay seams are present (US DOE, 2004; p. 1-11).

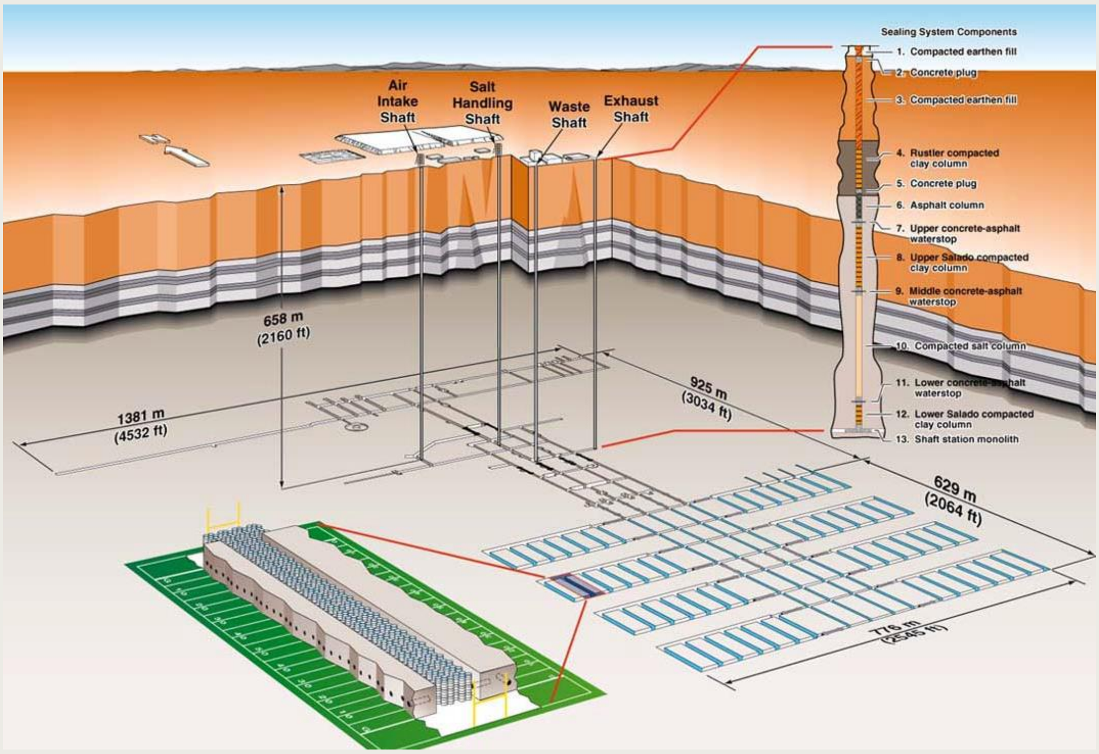

Figure 4-9 Layout of the Waste Isolation Pilot Plant.

The WIPP site has the following favourable geological characteristics (US DOE, 2004; p. 1-2).

- The host rock formation will eventually encapsulate the waste because of the formation's plastic behaviour.

- Dissolution effects are minimized and predictable.
- Repository excavation is relatively easy.
- Development of resources in the future is thought to be minimal and predictable.
- The host rock formation is both structurally and lithologically uncomplicated.

The site also exhibits favourable hydrological characteristics (US DOE, 2004; p. 1-2) such as:

- impermeable host rock containing little interstitial brine;
- minimal and predictable groundwater flow;
- no permanent surface waters
- minimal opportunities for developing potable groundwater: and
- low probability that the host rock will be affected by long-term climate changes.

WIPP also provided a location where the use of federal lands was maximized, where no drill holes existed, and where impacts on potash deposits were minimized. In addition the area also avoided endangered species and critical habitats and is in an area of seismic stability (US DOE, 2004; p. 1-2).

Transuranic (TRU) radioactive waste is disposed of 655 m (2,150 feet) below ground at the Waste Isolation Pilot Plant (WIPP) in south-eastern New Mexico (Figure 4-9: Camphouse, 2011; Figure 1). The disposal site is located in Permian salt beds which will spread (or “creep”), eventually surrounding the waste. WIPP is located in the Permian Salado Formation of the Delaware Basin.

The WIPP’s disposal area contains eight “panels,” (see Figure 4-9), each consisting of seven rooms, along with adjacent “access drifts” and crosscuts. The DOE proposed the following engineered barriers for the disposal system: shaft seals, panel closures, magnesium oxide (MgO) to surround the waste, and borehole plugs. However, the EPA concluded in its certification decision, that “only MgO meets the regulatory definition of an engineered barrier” (US DOE, 2004; pp. 3-14 to 3-15).

The DOE will close each panel with a closure system (See Figure 4-10) designed for operational protection of workers, as well as to protect the public and the environment from the waste (US DOE, 2004; p. 3-3), rather than for long-term performance of the repository (US DOE, 2004; pp. 3-22 and 3-24).

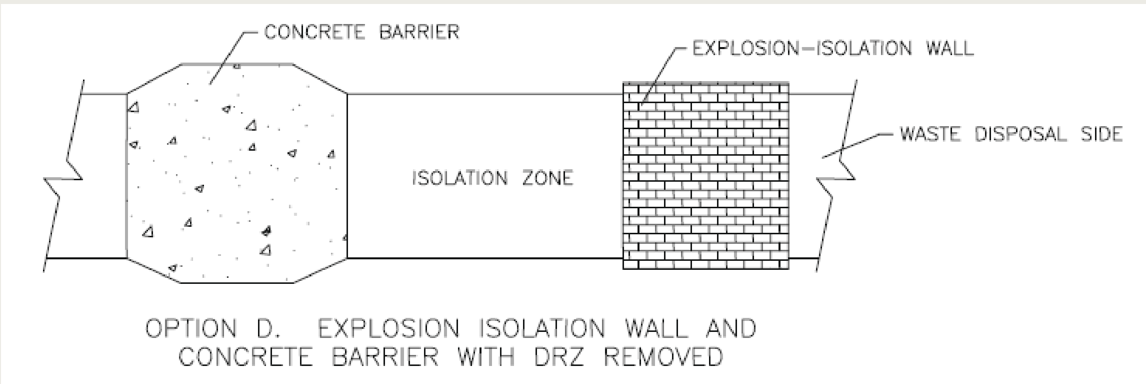

Figure 4-10 Panel Closure (US DOE, 2004; p. 3-22).

The third type of barrier chemically conditions brine that may reach the waste to reduce solubility of the radionuclides (US DOE, 2004; p. 3-3). Magnesium oxide (MgO) decreases the solubility of the actinide elements of TRU waste by consuming carbon dioxide produced from microbial decomposition of cellulose, plastic and rubber in emplaced CH waste (US DOE, 2011; p. 2-20).

The final type of barrier, borehole plugs, are designed to limit the volume of water reaching the repository from overlying water-bearing zones in order to reduce potential migration of contaminants (in contaminated brine) toward the accessible environment (US Doe, 2004; p. 3-24). This is generally accomplished through the use of cement (ibid.).

The property for the WIPP facility was transferred in 1992 from the U.S. Department of the Interior to the DOE. Lands within and surrounding the WIPP site are administered under a multiple land-use policy, consistent with the WIPP mission. The site will remain under federal control during operations, planned to last 35 years. WIPP began receiving and emplacing CH (“Contact Handled”) waste in March 1999 and RH (“Remote Handled”) waste in January 2007.

4.3. Waste Characteristics

For performing a safety assessment the characteristics of the Dutch radioactive waste intended for disposal has to be defined, In general the waste characteristics depend on:

- The amount of waste, e.g. from existing and possibly future NPPs, industry, hospitals;
- The type of waste, e.g. spent fuel, high-level waste, intermediate and low level waste;
- The adopted waste strategy, e.g. reprocessing versus direct disposal, long-term surface storage.

The present section provides a summary overview of the waste characteristics that have been applied in the past as source term in the various safety assessments performed for salt-based disposal concepts in the Netherlands. In addition, the waste characteristics determined in the framework of OPERA are explained shortly. Finally, an overview of open issues concerning the waste characteristics in relation to the final disposal in salt-based repository is provided.

4.3.1. Waste characteristics applied in previous studies

For the VEOS study as part of the OPLA programme (OPLA, 1989), the waste characteristics, including the radionuclide inventories of the then considered waste types have been determined in detail (Heijboer, 1988). The following table provides an overview of the various types of waste that were considered in VEOS (Heijboer, 1988; pp. 138,139). The radionuclide inventories of each of these waste types/canisters have been listed in Tables 1 to 11 of (Heijboer, 1988).

Table 4-1 Overview of waste types considered in VEOS.

Type of waste/canister	Description
KSA, vitrified, nuclear waste	Highly radioactive, heat generating waste, resulting from the reprocessing of spent fuel from nuclear power plants.
HAVA-200 canisters	These contain the waste from the glass ovens used in the vitrification process.
HAVA-1430 canisters	These contain the concrete cast cladding and end caps.
HAVA-220 canisters	These contain the bituminized waste originating from the separation of uranium and plutonium, and the further purification of both elements.

Type of waste/canister	Description
HAVA-2200 canisters	These contain bituminized waste from the purification of water from the receival basins of spent nuclear fuel in the form of ion exchangers, filters, and evaparatien residues.
MAVA-1200 canisters	These contain the technological alpha waste cast in concrete consisting of spent material and parts from the re-processing.
LAVA-665 canisters	These, like the MAVA-1200 canisters, also contain technological waste cast in concrete, though with a lower nuclide content and negligible heat generation.

In VEOS the total radionuclide inventory of the disposal mine was determined taking into account the *summed* inventories of the different waste types (Table 4-1), and for the following three so-called “waste strategies” (Slagter, 1988; Section 4.2):

- Strategy A, which was based on the radioactive waste from the Borssele and Dodewaard NPPs (installed capacity 500 MWe, operational period of 30 years), reprocessing of the spent fuel elements, and a surface interim storage of the waste for 50 years.
- Strategy B, where the waste generation from the Borssele and Dodewaard NPPs is supplemented by that from 3000 MWe nuclear power generation, and 105 years collection of radioactive waste from hospitals and laboratories. For this waste strategy final disposal was foreseen between 2080 and 2090 after 50 years of interim storage.
- Strategy C, where the waste generation from NPPs equals that of Strategy B, supplemented by the collection of waste from hospitals and laboratories during 55 years, and a foreseen phased disposal between 2000 and 2040 after 10 years of interim storage.

Notable for the VEOS waste characteristics are:

- The distinction of the different waste types, viz KSA, HAVA, MAVA, and LAVA, and their definitions are outdated and no longer apply;
- Spent fuel from the research reactors has not been considered in VEOS, because prior to 1988 that waste fraction was returned to the US;
- The waste strategies B and C include radioactive waste from NPPs that in retrospect have not been built, and therefore the resulting radionuclide inventories are considerably larger than is foreseen at present.

Section 5.2 of the present report describes the results of the VEOS safety assessment.

The PROSA study adopted the same waste types and radionuclide inventories for each waste type as elaborated in VEOS (Prij, 1993; Section 6.2.3). In PROSA the total radionuclide inventory of the disposal mine was determined taking into account the *summed* inventories of the different waste types (see Table 4-1), and for the following two so-called “waste strategies”:

- Strategy A, which was based on the radioactive waste from the Borssele and Dodewaard NPPs (installed capacity 500 MWe, operational period of 30 years), reprocessing of the spent fuel elements, and a surface interim storage of the waste for 50 years.
- Strategy B, where the waste generation from the Borssele and Dodewaard NPPs is supplemented by that from 3000 MWe nuclear power generation, and 105 years collection of radioactive waste from hospitals and laboratories. For this waste

strategy final disposal was foreseen between 2080 and 2090 after 50 years of interim storage.

Noteable for the PROSA waste characteristics are:

- The distinction of the different waste types, viz KSA, HAVA, MAVA, and LAVA, and their definitions are presently outdated and no longer apply;
- Similar to VEOS, spent fuel from the research reactors has not been considered in PROSA;
- The waste strategy C includes radioactive waste from NPPs that in retrospect have not been built, and therefore the resulting radionuclide inventories are considerably larger than is foreseen at present.

Section 5.3 of the present report describes the results of the PROSA safety assessment.

As part of the CORA programme (CORA, 2001) a safety assessment was performed for the METRO-I disposal concept (Grupa, 2000; Chapter 4; see also Section 4.2.3 of the present report). The radionuclide inventory of that safety assessment consisted of the summed inventory of 300 "COGEMA" containers with vitrified HLW. The inventory of a single container was obtained from the VEOS database, viz the "W11-1 KSA canister" (Heijboer, 1988; Tabel 2).

Noteable for the CORA waste characteristics are:

- Only a single waste fraction, i.e. vitrified HLW, was considered;
- Only 300 HLW canisters were taken into account, resulting in too low an inventory compared to the present expected inventory (see Section 4.3.3);

Section 5.4 of the present report describes the results of the CORA safety assessment.

For the EU FP6 PAMINA project NRG has performed a probabilistic safety assessment for a generic disposal concept in rock salt, including the Torad-B borehole design (Schröder, 2009). As in CORA, the PAMINA safety assessment assumed a radionuclide inventory consisting of the summed inventory of 300 "COGEMA" containers with vitrified HLW. The inventory of a single container was again obtained from the VEOS database, viz the "W11-1 KSA canister" (Heijboer, 1988; Tabel 2).

Since the PAMINA waste characteristics were identical to the CORA waste characteristics, the same notes apply:

- Only a single waste fraction, i.e. vitrified HLW, was considered in the respective safety assessments;
- Only 300 HLW canisters were taken into account, representing less than half the amount of the presently expected inventory (see Section 4.3.3). In PAMINA this was justified since the scope of the PAMINA exercise was not to perform a complete safety assessment, but to test uncertainty and sensitivity analysis methods.

As mentioned above the classification of radioactive waste as distinguished in the VEOS and PROSA studies is currently no longer applied. In the 1990's and 2000's alternative waste classification schemes were developed and adopted by the international community. The following section provides a summary overview of the presently adopted waste classification.

4.3.2. Waste Classification

The most recent waste classification scheme developed by IAEA (IAEA, 2009) covers all types of radioactive waste and provides a generic linkage with disposal options for all types of waste. That scheme is based on considerations of long-term safety, and thus, by implication, disposal of the waste. Six classes of waste have been derived and used as the basis for the classification scheme (IAEA, 2009; p. 5). The IAEA classification also covers radioactive waste having such low levels of activity concentration that it is not required to be managed or regulated as radioactive waste.

The waste classification scheme for The Netherlands is not based on a law or regulation but since 1985 it is common practice to use this classification scheme. Roughly in the Dutch waste classification scheme there are three waste categories, namely Low and Intermediate Level Waste (LILW), non-heat generating High Level Waste (HLW non-heat-generating) and heat generating High Level Waste (HLW heat-generating). Three groups of LILW can be distinguished: LILW, LILW (NORM) and LILW (Depleted Uranium, DU).

The comparison of the definitions of waste classes in both the Dutch and the IAEA's waste classification schemes led to the matrix presented in Table 4-2 (Hart, 2014b; p.7). The radioactive waste categories that are not required to be managed or regulated as radioactive waste have not been considered in this comparison. The table shows that in the Netherlands no distinction is made between short-lived and long-lived LILW as defined by the IAEA Safety Guide on Classification (IAEA, 2009; p. 5). The reason is that shallow land burial is not applicable for the Netherlands. All categories of waste will be disposed of in a deep geologic repository in the future; due to the small amounts of radioactive waste, no separate disposal facilities for LILW and HLW are envisaged (EL&I, 2011; p. 23).

Table 4-2 Comparison of the Dutch and IAEA's radioactive waste classification schemes.

	(IAEA) Distribution %		
Waste Class Name (NL)	**LILW-SL**	**LILW-LL**	**HLW**
LILW	90	10	
LILW-NORM	100		
LILW, DU		100	
HLW, non-heat generating		100	
HLW, heat generating			100

4.3.3. Current inventory of the OPERA reference database

As part of the OPERA programme, Task 1.1.1 *Definition of radionuclide inventory and matrix composition* has defined an up-to-date 'source term', both in terms of the radioactive inventory and the waste matrix. The 'base' radionuclide inventory and the waste matrix were compiled taking into account the presently adopted Dutch waste strategy, i.e. reprocessing of spent fuel from NPPs, no new nuclear power plants, and a surface storage period until 2130. In addition to the 'base' inventories, radionuclide inventories have been determined for alternative nuclear fuel cycle scenarios, in order to estimate the consequences of additional nuclear power plants in the Netherlands, and alternative fuel cycles, e.g. no reprocessing of spent fuel, application of Gen IV reactor types.

The following paragraphs provide more detailed information about the different waste types that are currently considered to be disposed. The expected total number of containers of each type of waste to be disposed of in each waste disposal section is summarized in (Verhoef et al., 2014; Appendix, Table A-1 to A-3).

Spent fuel
Spent fuel comprises conditioned spent fuel from the HOR (Hogere Onderzoeks Reactor) in Delft and the HFR (Hoge Flux Reactor) in Petten, and spent uranium targets from molybdenum production (EL&I, 2011; p. 24). The very small amounts of spent fuel from the dismantled LFR (Lage Flux Reactor) in Petten are insignificant compared to those from the HFR and HOR; see (Hart, 2014b; Section 2). To be noted is that also spent fuel of a future replacement of the HFR has to be considered. That reactor is currently indicated as PALLAS.

Vitrified HLW
Vitrified high-level waste consists of the radioactive waste from the reprocessing of spent fuel from the two nuclear power reactors in the Netherlands (Borssele and Dodewaard).

Non heat-generating high-level waste
Non-heat-generating HLW mainly consist of the residues from the reprocessing of spent fuel, other than the vitrified HLW. It also includes waste from research on reactor fuel, the production of isotopes, and waste resulting from the decommissioning of the nuclear power plants in the Netherlands (Verhoef, 2011a; p. 9).

Low- and intermediate-level waste
Low- and intermediate-level waste LILW arises from activities with radioisotopes in among others industry, research institutes and hospitals. It includes lightly contaminated materials, such as tissues, plastic -, metal - or glass objects, or cloth. In addition, drums with waste conditioned in cement, originating from the nuclear power plants, contribute to the amount of LILW.

(TE)NORM
The only fraction of (TE)NORM that is foreseen for geological disposal (Verhoef, 2011a) is depleted uranium (DU) originating from the uranium enrichment facility of URENCO (EL&I, 2011; p. 22). The DU is presently stored in DV-70 containers. For the purpose of OPERA it is assumed that the DU will be immobilized in concrete (1:1) and finally disposed of in KONRAD type II containers.

4.3.4. Waste characteristics - evaluation

The waste categories indicated in the previous sections include a variety of radionuclides which together define a source term of the radiological safety assessment calculations, to be performed within OPERA's WP7 *Scenario development and performance assessment*.

Compared to the radionuclide inventories considered in the previous studies performed in VEOS, PROSA, CORA, and PAMINA, the following considerations apply:

- Unlike in the past, the spent fuel from research reactors, including the future replacement of the HFR (PALLAS) is part of the radioactive waste inventory. The nuclide inventory of the spent fuel elements represents a considerable part of the total nuclide inventory (Hart, 2014b; Table 6-2).
- Unlike in the past, depleted uranium (DU) currently adds to the radioactive waste inventory, foreseen to be disposed in a future geological disposal facility. Although DU represents, in radiological terms, a relatively small fraction of the total inventory (Hart, 2014b; Table 6-2), it comprises a significant volume (Meeussen, 2014; Section 3.2).

- The distinction of the different waste types KSA, HAVA, MAVA, and LAVA and their definitions , as practiced in VEOS and PROSA, no longer apply.
- In the previous studies VEOS and PROSA consideration was given to the so-called Waste Strategies B and C, which included radioactive waste from NPPs that in retrospect have not been built, and therefore the assumed radionuclide inventories from NPPs were considerably larger than is foreseen at present.
- In the CORA analyses and in the PAMINA probabilistic safety assessment only a single waste fraction, i.e. vitrified HLW, was considered, and only 300 HLW canisters were taken into account, resulting in too low an inventory compared to the present expected inventory.

As a conclusion it can be stated that previously performed safety assessments for salt-based repositories, in VEOS, PROSA, CORA, and PAMINA, applied waste characteristics and radionuclide inventories which differ considerably from the presently foreseen inventories. As a consequence the quantitative results from the safety assessments performed in the past are not representative for a future geological disposal facility in rock salt under the present waste inventory.

It is therefore recommended to update previously performed safety assessments, taking into account the presently foreseen nuclide inventories and waste fractions, and for a disposal concept that is adapted to these waste types and amounts.

4.4. Engineered barriers

The safety of a disposal facility depends primarily on the favourable characteristics or properties of natural barriers as well as the man-made engineered barriers. Engineered barriers are part of the multi-barrier system (cf. Section 3.7.1) and may include the waste form itself, waste container, borehole backfill, borehole plugs, backfilled galleries, dams, and backfilled shafts (shaft seals). Also concrete or steel lining constructions can be part of the engineered barriers of a disposal design.

In the following sections only borehole and gallery plugs, and shaft seals will be discussed separately, since these features are specifically related to salt-based repositories. In addition, in existing salt-based disposal facilities lining of transport galleries is usually not considered. Issues related to backfill are treated in Section 4.4.

4.4.1. Borehole plugs

After emplacement of radioactive waste in a disposal cell or cavity, the disposal cells will be closed by means of a plug, usually manufactured from pre-compacted crushed salt (e.g. Heijdra, 1996; Section 3.1). In addition, the transport galleries and other underground cavities will be backfilled with granular salt. After backfilling and closure of the facility, the compacted crushed salt plug as well as the backfill holds a porosity higher than of the surrounding rock salt. The increased porosity forms a potential pathway for any brine that, if present, may reach the waste canisters and for contaminated brine to leave the disposal galleries.

The convergence process leads to compaction of the plugs sealing the disposal cells and the granular backfill, resulting in a decrease of the porosity in time. After some time the porosity of the compacted salt plugs and the backfill in the open spaces has decreased to such an extent that these engineered barriers have become almost impermeable. The porosity value where the crushed salt is assumed to become impermeable is also referred to as the *threshold porosity* and was determined experimentally in the *NF-PRO* project at about 1%, the average porosity of (impermeable) intact (dense) rock salt (Zhang, 2006;

p.11). Other reported values for the percolation threshold porosity are between 1% (Grupa, 2000; p.95) and 5% (Poley, 2007).

It is still unclear whether compacted crushed rock salt will eventually become impermeable like pristine rock salt or that its permeability only decreases to undetectable low values. A comparison of the results of the experiments and modelling efforts performed in the NF-PRO project showed that, during the compaction process of granular salt, there is some mechanism that inhibits full closure of pores. Possible explanations include the anisotropic nature of surface energy in NaCl, which may lead to stable (permanent) wetting of crystal interfaces, or to inhibition of mass transfer by the presence of vapour bubbles in the intergranular brine phase. Additional efforts would be needed to improve the agreement between the experiments and modelling of the permeability reduction at low porosity during compaction (Grupa, 2000; p.96).

In the Netherlands the sealing properties of borehole plugs were investigated in the EU-FP4 Framework project BAMBUS-II (Grupa, 2003) and PAMINA (Schröder, 2009). Both in the BAMBUS-II and PAMINA studies the focus was on the (calculated) compaction behaviour of compacted salt sealing plugs of boreholes, under dry and wet (flooding) conditions.

The calculational exercise performed in the PAMINA project lead to the conclusion that, due to convergence by the surrounding host rock, a disposal cell sealing plug would become impermeable after somewhat less than 800 years after its emplacement. Such a long time frame makes it unfeasible to experimentally verify this closure behaviour.

In case a salt-based disposal concept would adopt the sealing of disposal cells using pre-compacted salt plugs, additional efforts would be needed to confirm the time for the plugs to reach the threshold porosity.

4.4.2. Gallery seals

In the past gallery seals, also referred to as drift seals, have not explicitly been designed, and only limitedly considered in the Netherlands. The OPLA-1 and CORA studies consider a sealing system consisting of “dam constructions”, which have not been specified in more detail (OPLA, 1989; p.66).

In the German disposal facility design drift seals are considered (Bollingerfehr, 2013; Section 4.3.4.2). In addition to backfilling, drift seals (engineered barriers) will be located close to the shaft filling station and infrastructure areas at selected positions in all drifts connected to the shaft at the exploration level and at the emplacement level. This will ensure that potential fluid pathways to the shaft will be sealed and the heat-generating radioactive waste will be separated from the waste with negligible heat generation.

Calculations related to the safety demonstration were carried out for the drift seals. In all relevant scenarios, the stability and tightness of the drift seal could be demonstrated.

4.4.3. Shaft seal

A crucial part of a geological disposal facility is the shaft seal, as this feature is intended to finally isolate the repository from the biosphere.

In the past shaft seals have not explicitly been designed, and only limitedly considered in the Netherlands. The OPLA-1 and CORA studies consider a shaft sealing system consisting of crushed salt, that would be compacted by external forces of the overburden. At some point in time the compaction would result in seal properties similar to those of rock salt (OPLA, 1989; p.66) (CORA, 2001; Section 4.7.3).

However, from extensive analyses performed in Germany and US it appeared that filling a shaft with crushed salt only would not be sufficient to isolate the repository from the biosphere both in the short term and in the long term. One reason is that the timing of complete "closure" of the compacted crushed salt, where the crushed salt properties would be similar to those of rock salt, would not be immediately after its emplacement.

The shaft seal systems proposed in Germany and US are summarized below.

4.4.3.1. Shaft seal system - Germany

The proposed shaft seal system in Germany, consisting of two shafts, is adopted to the conditions prevailing at the Gorleben site (Bollingerfehr, 2013; Section 4.3.4.3). Both shafts will be sealed by shaft seals which are engineered barriers consisting of several components comprising sealing elements, abutments and pore storage. The components and material were selected in accordance with the geologic environment along the shaft length and the composition of brines that might intrude from the overburden. The concept took into account the detailed stratigraphic situation along the shaft length, the existing shaft accesses at the ex-ploration level and the planned emplacement level and the composition of potentially intruding brine and its potential timing.

Figure 4-11 shows the functional elements of the shaft seal. Each element has its specific function. For example, the lower "sealing element" separates the emplacement level from the overlying shaft and secondly it seals an anhydrite layer with higher hydraulic conductivity called the "Gorleben Bank". The successive shaft layers are also adapted to the neighbouring geological features of the Gorleben Bank. The uppermost element of the shaft seal below the shaft foundation is a filter layer. In the overburden formation the shaft is backfilled conventionally.

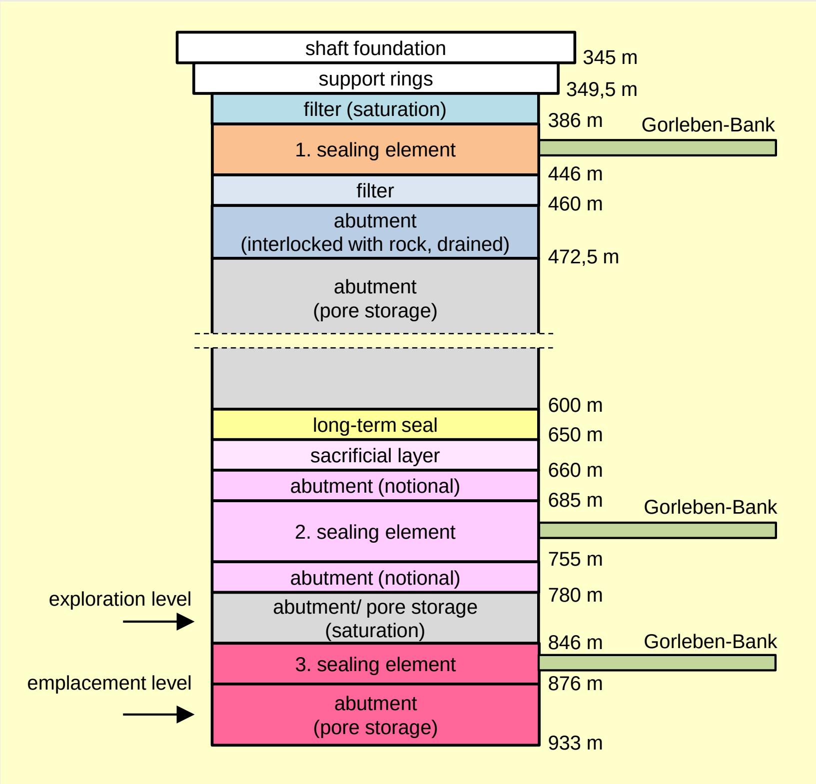

Figure 4-11 Sketch of the functional model of the shaft seal - German design.

The design of the shaft seal was investigated by preliminary design calculations. Geochemical calculations were carried out to analyse the influence of water/brine ingress from the top of the shaft through the sealing elements in order to confirm that the selected materials are suitable. The geochemical calculations were followed by preliminary mechanical calculations in order to determine the length of each abutment and to assess potential settling effects in order to avoid damage to the sealing elements. These analyses included an assessment of the shaft seal being capable of retaining any intruding brine for such a period of time that the compacted crushed salt backfill would be capable of preventing the transport pathway further on.

4.4.3.2. Shaft seal system – US

The WIPP shaft sealing system system (Hansen, 2011; Section 2.3) is designed to limit entry of water and release of contaminants through the existing shafts after decommissioning. The design approach applied redundancy to functional elements and specifies multiple, common, low-permeability materials to reduce uncertainty in performance. The system comprises 13 elements that completely fill the shafts with engineered materials possessing high density and low permeability.

From laboratory and field measurements of component properties and performance it was concluded that an acceptable seal system can be designed and constructed using existing technology, and that the seal system can meet requirements associated with repository system performance. These goals would be met by using a set of guidelines that incorporates seal performance issues. These guidelines were formalized as design guidance for the shaft seal system:

- Limit waste constituents reaching regulatory boundaries
- Restrict formation water flow through the seal system
- Use materials possessing mechanical and chemical compatibility
- Protect against structural failure of system components
- Limit subsidence and prevent accidental entry
- Utilize available construction methods and materials.

The WIPP the shaft seal system would include the following components, see Figure 4 13 (Dennis, 2011; Figure G2-5):

- Shaft Station Monolith—The base of the shaft will be sealed with salt-saturated, Portland cement-based concrete.
- Clay Column—A sodium bentonite compacted clay component is placed on top of the mass of concrete. Alternative construction methods including block placement and dynamic compaction are viable. Clay columns effectively limit formation water movement from the time they are placed. The stiffness or swelling pressure associated with the clay column is sufficient to promote healing of fractures in the surrounding rock near the bottom of the shafts, thus effectively removing the proximal damage zone as a potential pathway.
- Salt Column—A crushed salt element for the shaft seal system has been thoroughly evaluated for salt application. The performance has been established from the microstructural scale to the full construction scale. Very low permeability is attained by the crushed salt in laboratory time, which is an excellent basis for concluding the crushed salt element will become impermeable in a matter of decades.
- Asphalt Column—Asphalt is a widely used construction material with properties considered desirable for sealing applications. Asphalt is readily adhesive, highly waterproof, and durable. Furthermore, it is a plastic substance that provides controlled flexibility to mixtures of mineral aggregates with which it is usually combined. It is highly resistant to most acids, salts, and alkalis. A number of asphalts and asphalt

mixes are available that cover a wide range of viscoelastic properties and which can be tailored to design requirements.

- Earthen Fill—The upper shaft is filled with locally available earthen fill. Most of the fill is dynamically compacted (e.g., by the method used to construct the compacted clay column) to a density approximating the surrounding lithologies. The uppermost earthen fill is compacted with a sheep-foot roller or vibratory plate compactor.

The seal material specifications, construction methods, rock mechanical analyses, and fluid flow evaluations already developed and approved remain applicable. This design concept proved to the regulators and stakeholders that the salt repository will be totally isolated after closure.

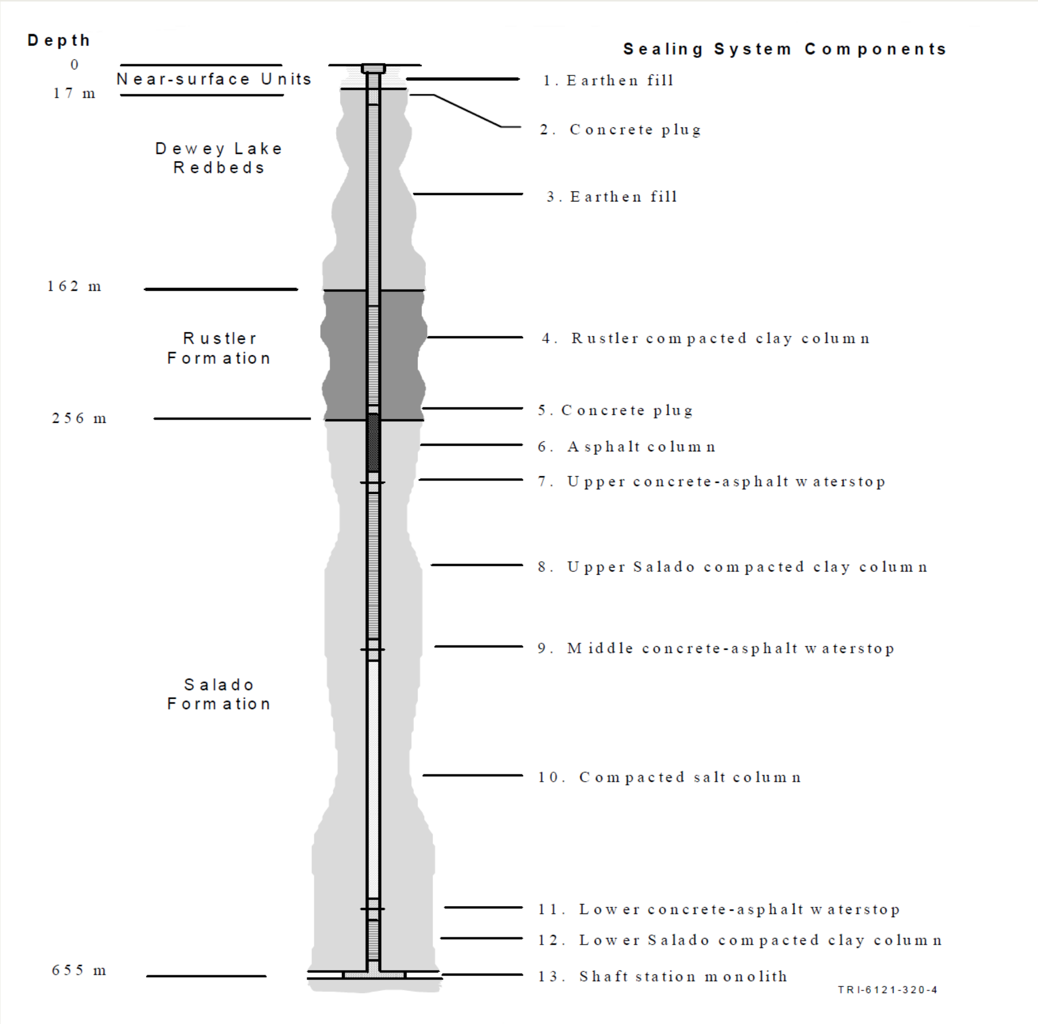

Figure 4-12 Sketch of the WIPP sealing system components - US design.

4.4.4. EU FP7 project DOPAS

A major effort on the design and performance of seals and plug is executed in the recently started EU-FP7 project DOPAS (Hansen, 2012), in which NRG also participates. DOPAS aims to improve the adequacy and consistency regarding industrial feasibility of plugs and seals, the measurement of their characteristics, the control of their behavior over time in repository conditions and also their hydraulic performance acceptable with respect to the safety objectives. The project addresses the design basis, reference designs and strategies to demonstrate the compliance of the reference designs to the design basis, for plugs and seals in geological disposal facilities. The DOPAS project is derived from the IGD-TP's Strategic Research Agenda that points out the topic of “plug and seals” as a first priority issue for joint European RTD projects.

Amongst the different types of plugs and seals under experimentation or under planned experimentation in DOPAS the large-scale deep shaft seal experiment in salt dome environment (ELSA large scale demonstration experiment) is relevant for the salt safety case.

4.4.5. Engineered barriers - evaluation

- In principle, disposal cells can be sealed using pre-compacted salt plugs. After their emplacement the salt plugs will be compacted further as a result of convergence by the surrounding host rock and finally will become impermeable.
- In case a salt-based disposal concept would adopt the sealing of disposal cells using pre-compacted salt plugs, additional efforts would be needed to confirm the time for the plugs to reach the threshold porosity.
- For a well designed shaft seal, adaption of the successive layers of the seal system to the neighbouring geological environment is crucial. This implies that a shaft seal is a site-specific feature of the disposal system, and can only be detailed if a site has been established.
- Especially in Germany significant and relevant information is available concerning the design and performance of gallery seals (drift seals, dams) in salt-based repositories. Therefore, there is at present no urgent need to perform similar activities in the Netherlands.
- NRGs participation in the EU-FP7 project DOPAS is relevant for enhancing the knowledge in the Netherlands with respect to the design and performance of plugs and seals in salt-based repositories.

4.5. Salt host rock

4.5.1. Introduction

Rock salt (halite) has been considered as a candidate host rock for radioactive waste emplacement for a long time. There are more than one hundred years of experience in salt mining in several countries, which shows that underground structures can be constructed in a stable way. In ambient conditions rock salt is practically impermeable to gases and liquids. In addition, rock salt shows good thermal conductivity and favourable deformation behaviour. Few other rocks are as thermally conductive or as close to truly viscous as natural rock salt (Warren, 2006). On account of these features cavities excavated in rock salt are sealed by compaction as time progresses and the waste is tightly enclosed in the rock.

Rock salt is the main component of the evaporation cycle and is originally precipitated from a saturated surface or near-surface brine. The rock salt itself consists of a variety of saline layers. Each of these layers has specific characteristics, e.g. density, thermal conductivity, expansion coefficient, or fluid content. An important feature of rock salt is its mechanical behaviour, i.e. the potential deformation of the material under stress. In an isotropic stress field no deformation will occur. The behaviour of salt in anisotropic stress fields is one of the most important processes to be taken into account for repositories in salt and will be discussed in the following sections. It is also the basis for the development of salt domes (diapirism). The primary driving force for salt tectonics is differential loading, which may be induced by gravitational forces, by forced displacement of one boundary of a salt body relative to another, or by a thermal gradient. Salt will move only if driving forces exceed the resistance to flow. In order for a salt diapir to intrude into its overburden, any rock previously occupying that space must be removed or displaced. Emplacement may occur by extension, erosion, or uplift of the overburden or by overthrusting of the salt (Hudec, 2007).

According to the geological evolution two main types of salt formations can be distinguished: bedded salt formations and domal salt formations, see Figure 4-13 (ENC, 2014). The characteristics of bedded salt and domal salt are quite different. Due to its uprise **domal salt** is relatively pure and homogeneous. The lateral extent of domes is limited, however, and therefore the dome margins delimit the area useful for a repository. Subrosion by groundwater produces a cap rock at the top of the salt formation and, in some locations, sides of the domes. Cap rock may have low permeability and armor the dome against dissolution or, it may be permeable. **Bedded salt** is less pure than domal salt. It is interbedded with limestone, dolomite, anhydrite, polyhalite and fine-grained siliciclastic beds, which can provide pathways for advective transport. Salt beds are typically continuous over large areas.

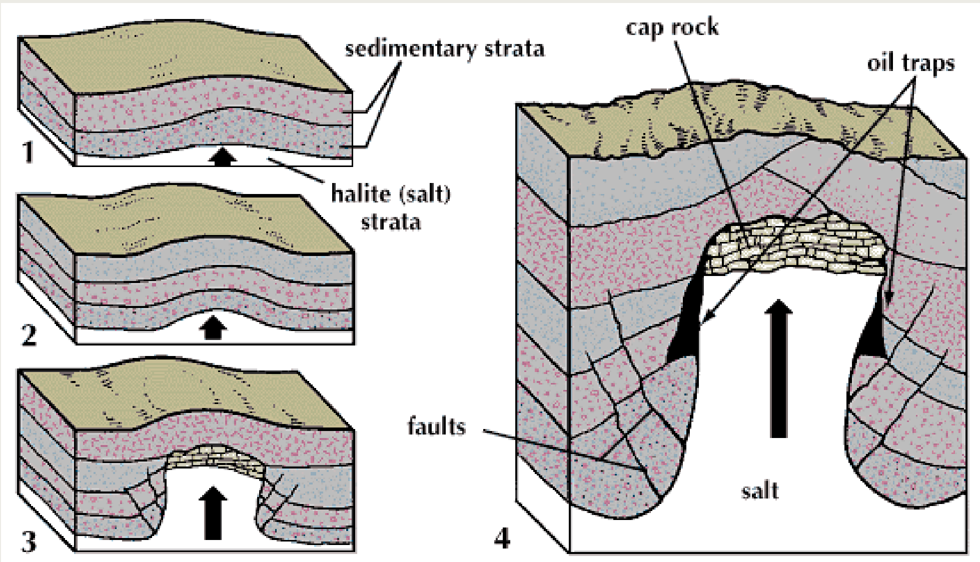

Figure 4-13 Formation of a salt dome; upper left: bedded structure; right: salt dome.

In a safety assessment for a repository site, the geological history and the resulting properties of the salt formations must be well known for calculations of the thermo-mechanical behaviour. In Germany, a comprehensive overview of the geological situation including characteristics of a salt dome under consideration has been given by the geological survey, BGR (Bornemann, 2008). In the U.S. the Salado formation is a well-known example for a bedded salt deposit (Sandia, 1992). The situation in the Netherlands is described in Section 4.5.4.

In a safety assessment, also appointed as total system performance assessment, the interaction of all relevant effects potentially affecting the safety must be taken into account. This holds for thermal, hydraulic, mechanical and chemical (THMC) interactions, which will be basically described in Section 4.6. Highest effort is laid on mechanical and hydraulic (MH) effects, because these are essential for the enclosure of waste and potential release of radionuclides from a repository. Thermal (T) effects are implicitly involved in all processes and should be taken into account if applicable. Chemical (C) effects are treated separately. As further discussed in Section 4.6.3.4 the combined modelling of chemical effects with THM processes is a challenging task.

The excavation of a mine influences the stress field and the mechanical processes in the salt host rock, as will be introduced in Section 4.6.1. After the emplacement of the waste, additional engineered barriers will be put into place in the excavated volumes. The relevant engineered barriers for the containment of the radioactive waste are the seals, and crushed salt serving as backfill of any void volume in emplacement areas. After its emplacement the backfill will be compacted by convergence as a result of pressure forces exerted by the overburden. During compaction, the porosity and permeability of the crushed salt decreases until, in the long run, it has the similar barrier properties as rock salt. At that time, the salt has become impermeable for liquids and gases. Thus a discussion of the properties of salt must include the properties of rock salt and crushed rock salt.

Much of the knowledge about rock salt properties has been derived from exploration of the Gorleben and Asse sites in Germany, and of the Waste isolation Pilot Plant (WIPP) in the U.S. Extensive summaries of earlier studies, performed in the last decades of the 20th century and encompassing several thousand pages, have been provided by BGR, i.e. the “Salzmechanik” volumes (Fahland, 2013). The latest state of the art of discussion has been presented and discussed during the International Conferences TIMODAZ-THERESA in Luxembourg (Li, 2012), SaltMech6 in Hannover (Schulze, 2007), and SaltMech7 in Paris (Bérest, 2012). The influence of disturbed rock zones on performance assessments has been discussed in (Davies, 2005).

In a disposal concept in rock salt, the rock salt has not only an important role as barrier, the convergence behaviour of the rock salt is also of relevance for the proper function of the engineered barrier system (EBS). Therefore in the following section, the geomechnical behaviour of rock salt as host rock (Section 4.6.2) will be discussed. Crushed rock salt as used in the EBS will be treated in Section 4.6.3. Section 4.6.4 gives an overview of the constitutive modelling of (coupled) thermal-hydraulic-mechanical-chemical processes, including the aspects of computer codes available for appropriate calculations.

As chemical aspects are not included in most of the actual investigations, in Section 4.6.5 a detailed discussion of geochemical conditions and modelling is given.

4.5.2. Previous geological studies aimed at deep disposal of radioactive waste in salt

During the past decades, extensive studies have been made of geological disposal of radioactive waste in salt in the Netherlands. The OPLA-programme consisted of two phases OPLA-1 (1984-1989) and OPLA-1A (1989-1993) which focussed on disposal in salt in the onshore part of the Netherlands.

Geologically, report OPLA-12 is the most elaborate (RGD, 1988) and focusses on 38 salt locations (19 salt domes, 15 salt pillows and 4 salt layers). OPLA-13 investigated geophysical methods related to exploration of subsurface salt occurrences (DGV-TNO, 1987).

OPLA-1 was followed by OPLA-1A, which aimed at narrowing down uncertainties discovered in OPLA-1. The setup of OPLA-1A is difficult to recover, but geologically speaking the work done by Rijks Geologische Dienst (RGD) seems most relevant for the present study (RGD, 1993). In a series of sub-studies an evaluation was made of spatial data (a), salt movement (b), caprock formation and subrosion (c), fluvial and subglacial erosion (d) and the geological barrier model (e). Further, during OPLA-1A another study focussed on generating an overview of the applicability of a number of geophysical techniques or a combination of such techniques, for geohydrological exploration of shallow salt bodies and their overburden (TNO, 1993).

The OPLA-programme was followed by the CORA-programme (1996-2000) which evaluated the obligation set by the Dutch government in 1993, stating that geological disposal should be retrievable. Besides a focus on salt, this programme also focussed on disposal in the Boom Clay (Rupel Clay Member) in the Dutch subsurface. Less attention was given to the geological aspects of salt, which was covered in the OPLA-programme. All programmes have final reports summarising the results of the sub-programmes:

- OPLA (1989): Onderzoek naar geologische opberging van radioactief afval in Nederland – Eindrapport Fase 1. Commissie Opberging te Land (OPLA), Ministerie van Economische Zaken, 130 p.;
- RGD (1993): Evaluatie van de Nederlandse zoutvoorkomens en hun nevengesteente voor de berging van radioactief afval - Overzicht van de resultaten – Eindrapport van geologisch onderzoek in het project GEO-1A, een onderdeel van het nationale Programma van Onderzoek 0PLA, Fase 1A. RGD rapport 30.012/ER, Ministerie van Economische Zaken, 116 p.;
- CORA (2001): Terugneembare berging, een begaanbaar pad? Onderzoek naar de mogelijkheden van terugneembare berging van radioactief afval in Nederland – Eindrapport. Commissie Opberging Radioactief Afval (CORA), Ministerie van Economische Zaken, 110 p.

4.5.3. General information

Salt accumulates when salt-rich lake or sea water evaporates. The thickest salt occurrences on earth are of marine origin. In the Netherlands salt deposits mainly occur in Permian and Triassic intervals, but here we focus on salt of Permian age (260-254 million years old) which attains greatest thicknesses and belongs to the Zechstein Group.

Deposition of salt is different from deposition of clastic sediments (sand, silt and clay), because it is the result of evaporation. As mentioned in Section 4.6.2.1 (Table 4-6) number of main salt types is distinguished and salt has several specific characteristics:

- Clastic sediments accumulate at rates of roughly 6-9 cm/1000 years, but salt accumulates at 10-20 cm/year;
- The density of clastic sediments increases with depth due to burial, but the density of salt remains the same due to the crystalline structure. Because of this, salt is generally heavier than the surrounding rocks in the upper 500 m of the subsurface. Below 500 m, rock salt is lighter than the surrounding rocks, which makes that the rock salt rises, while other rocks sink;
- Salt can deform plastically when under pressure in the subsurface, without faulting;
- Salt is virtually impermeable for oil, gas and water. The porosity of salt lies around 1% and the permeability is around 10^{-22}-10^{-19} m^2;
- Salt has a higher thermal conductivity than most other rock types.

Movement of salt occurs as a result of the characteristics described above and is called *halokinesis*. Halokinesis is triggered by movement of an underlying fault when the salt layer has a minimum thickness of at least several hundreds of metres. During the first stages of salt movement, a salt pillow is made by lateral movement. A salt pillow does not pierce the overlying sediments. In the next stage a pillow evolves into a salt diapir or salt dome, in which vertical salt movement occurs and overlying layers are pierced.

4.5.4. Salt formations in the Netherlands

The Zechstein Group consists of various rock types, including clay, carbonate, anhydrite and rock salt (Figure 4-14). These rock types are not available as separate maps and therefore we present maps of the Zechstein Group as a whole.

The following maps are provided:

- Depth top Zechstein Group in the Netherlands, Figure 4-15
- Depth base Zechstein Group in the Netherlands, Figure 4-16
- Thickness Zechstein Group in the Netherlands, Figure 4-17
- Salt domes Zechstein Group, Figure 4-18
- Salt extraction licences in the Netherlands, Figure 4-19
- Zechstein top between 500-1500 m, Figure 4-20

The depth maps of the top and base of the Zechstein Group have been constructed based on existing, but recently updated data (NITG-TNO, 2004; p. 31; Kombrink et al., 2012; p. 427), which are based on interpreted seismic data (2D and 3D) and borehole data. The thickness of the Zechstein Group was constructed by calculating the thickness between the base of the overlying deposits (i.e. top Zechstein Group) and the base of the Zechstein Group.

The salt domes (Figure 4-18) have been extracted from the thickness map, by assuming a minimum thickness of rock salt in a salt dome of 1300 m. Also indicated on this map are locations where salt is present within 1500 m below the surface and thicker than 300 m. Figure 4-19 shows the three areas with salt extraction licences in the Netherlands, with superimposed on that the salt domes from Figure 4-18. From these locations more information on the composition and structure of the salt may be obtained, resulting from the mining activities.

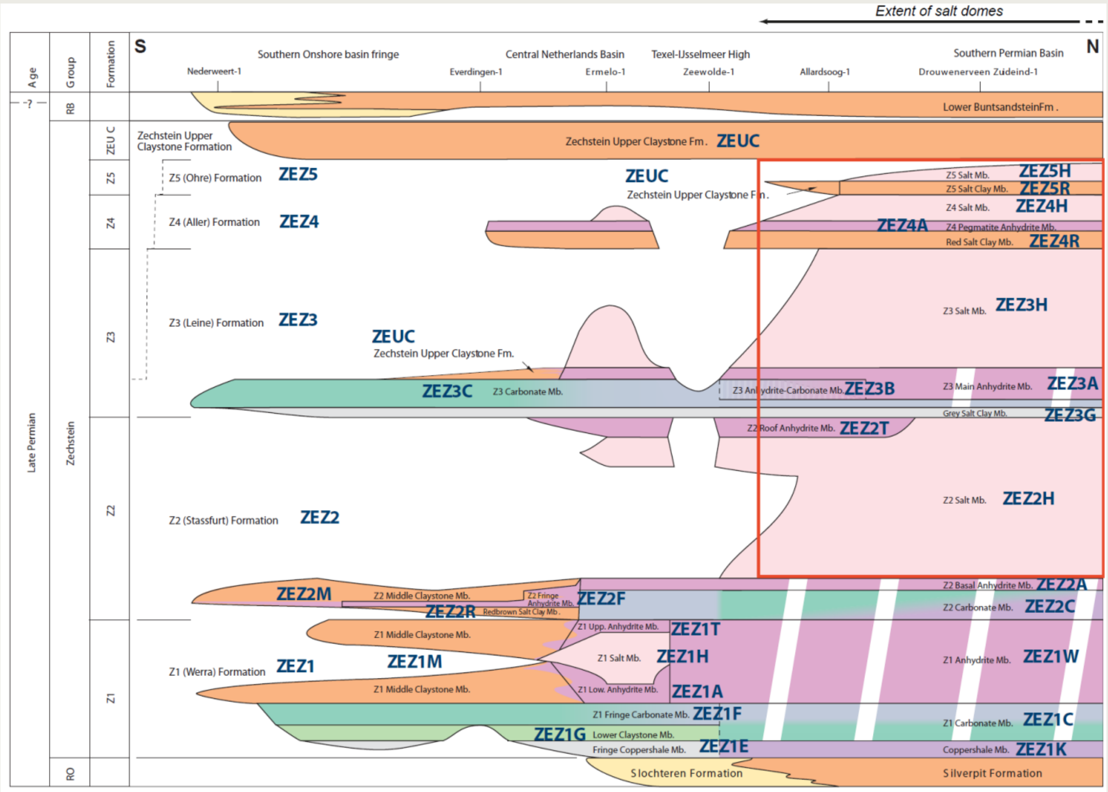

Figure 4-14 Lithostratigraphy of the Zechstein Group in the Netherlands (Van Adrichem Boogaert and Kouwe, 1993-1997).

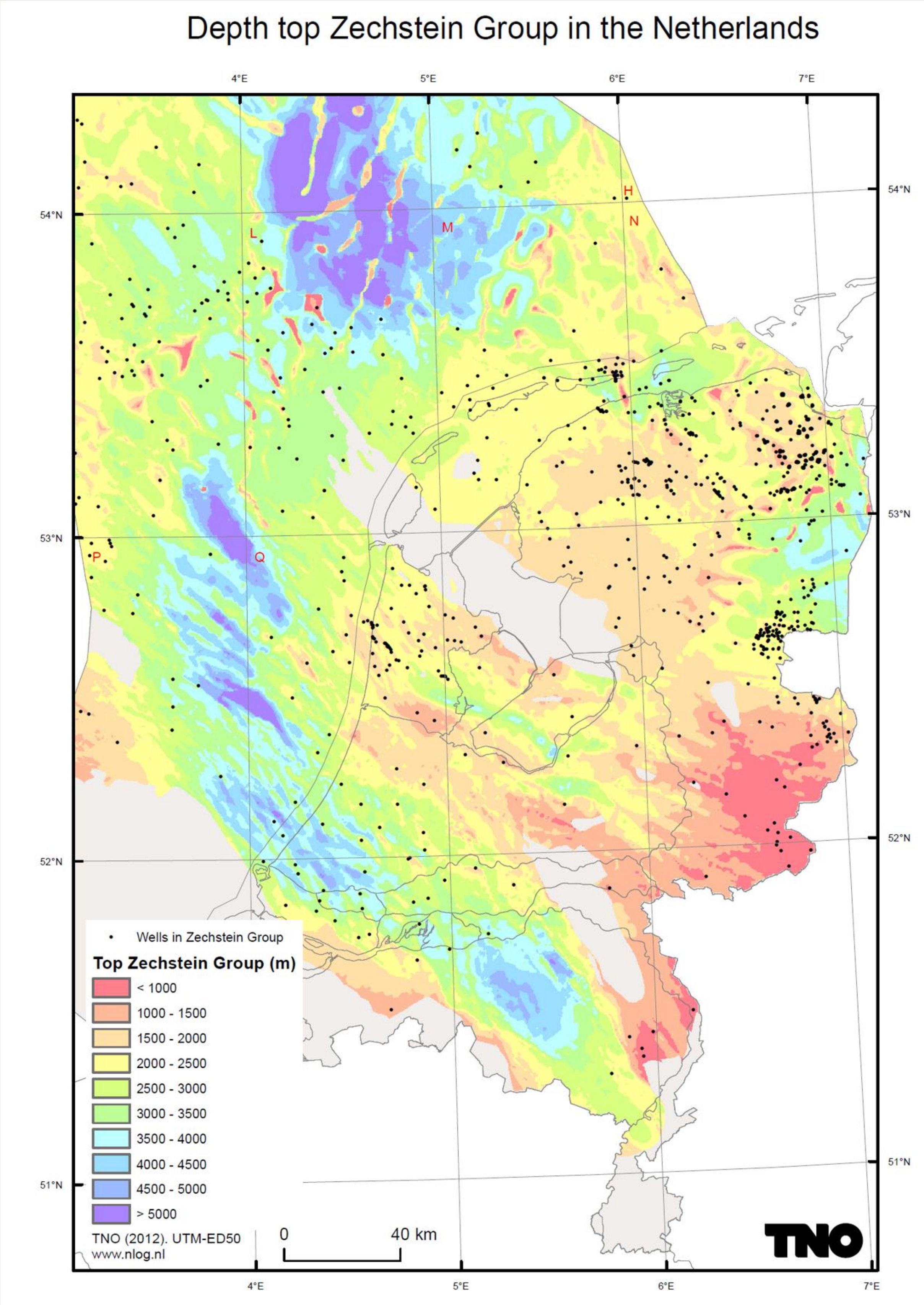

Figure 4-15 Depth top Zechstein Group in the Netherlands (2013).

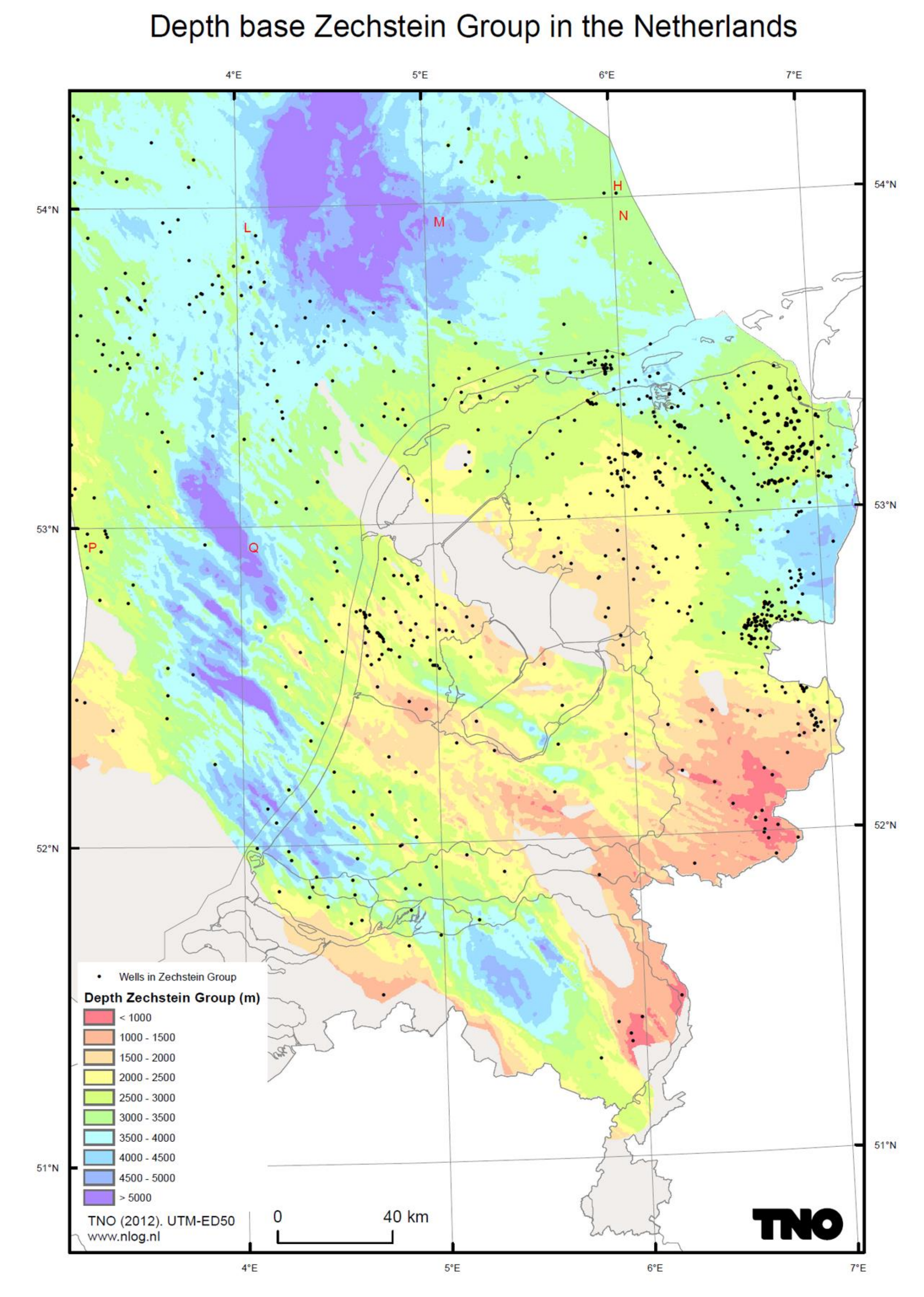

Figure 4-16 Depth base Zechstein Group in the Netherlands (2013).

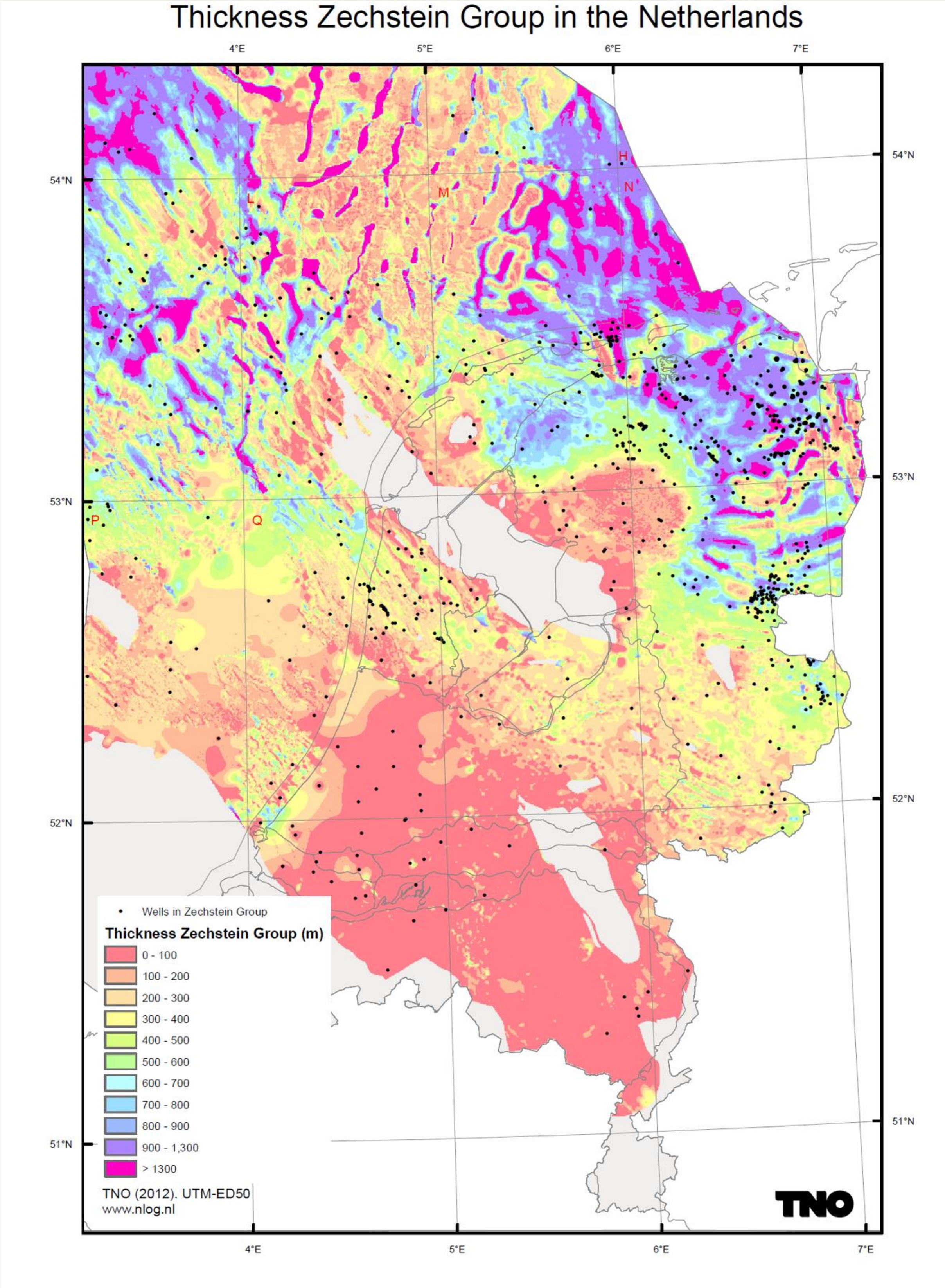

Figure 4-17 Thickness Zechstein Group in the Netherlands (2013).

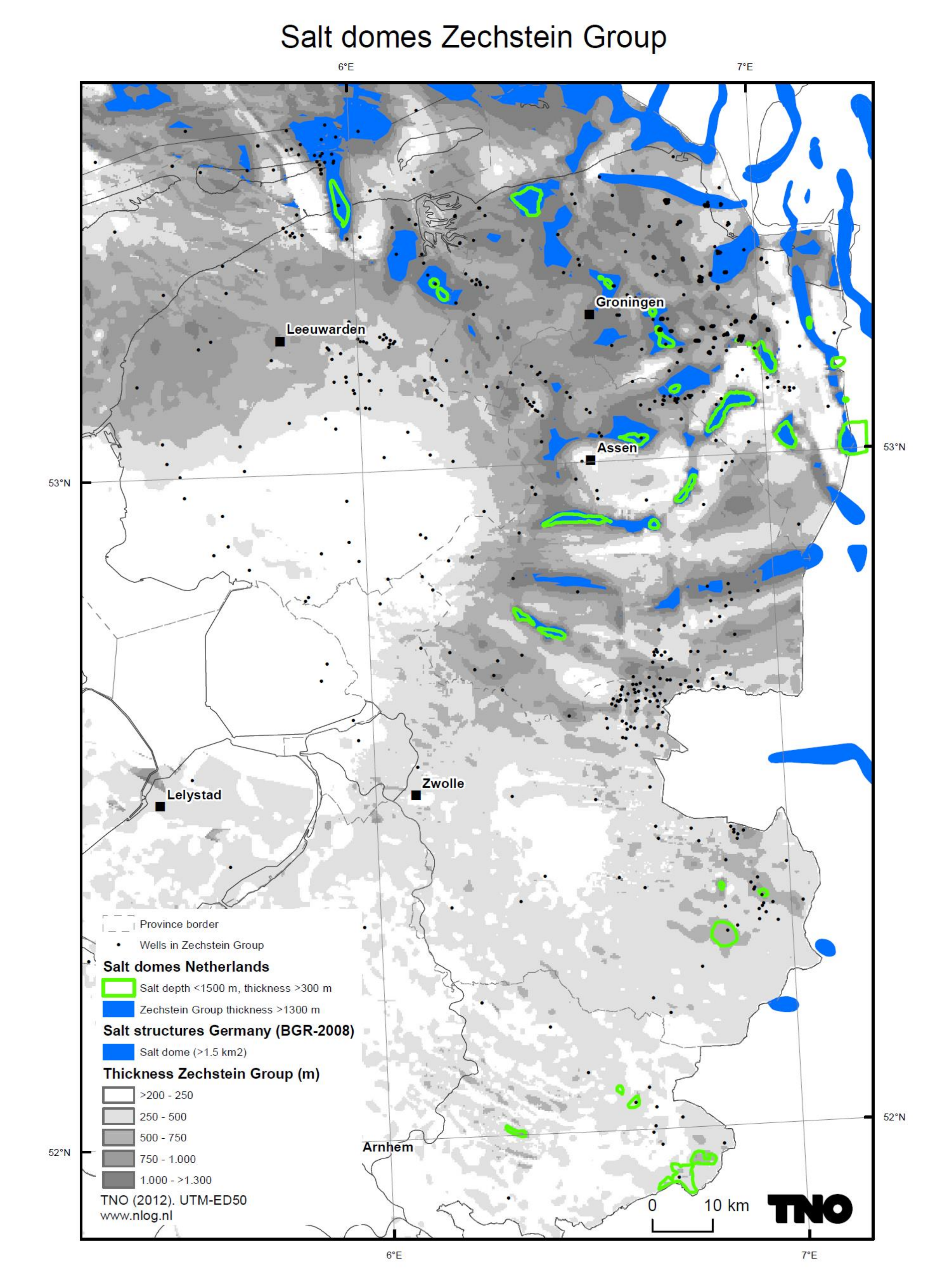

Figure 4-18 Salt domes Zechstein Group.

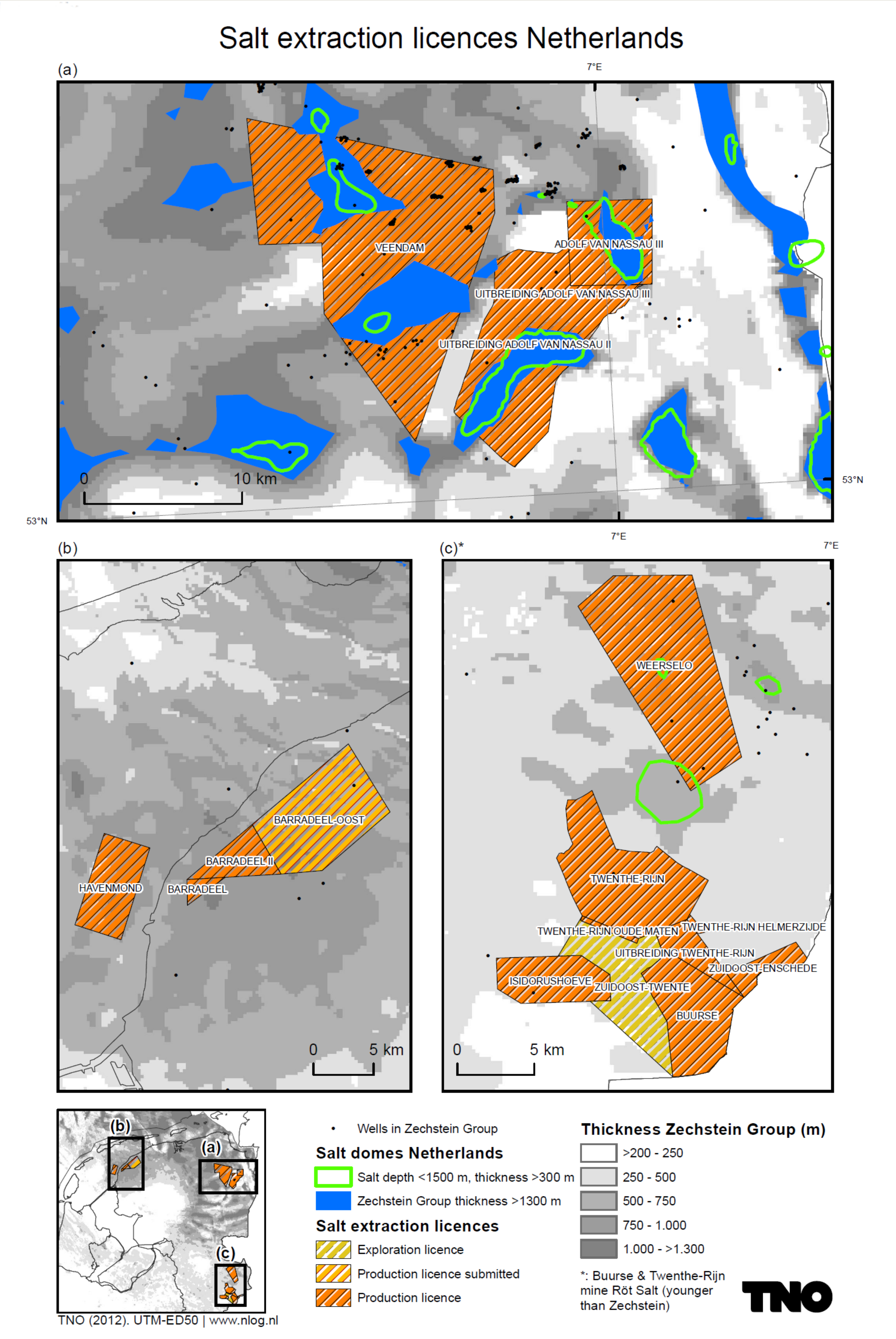

Figure 4-19 Salt extraction licences in the Netherlands.

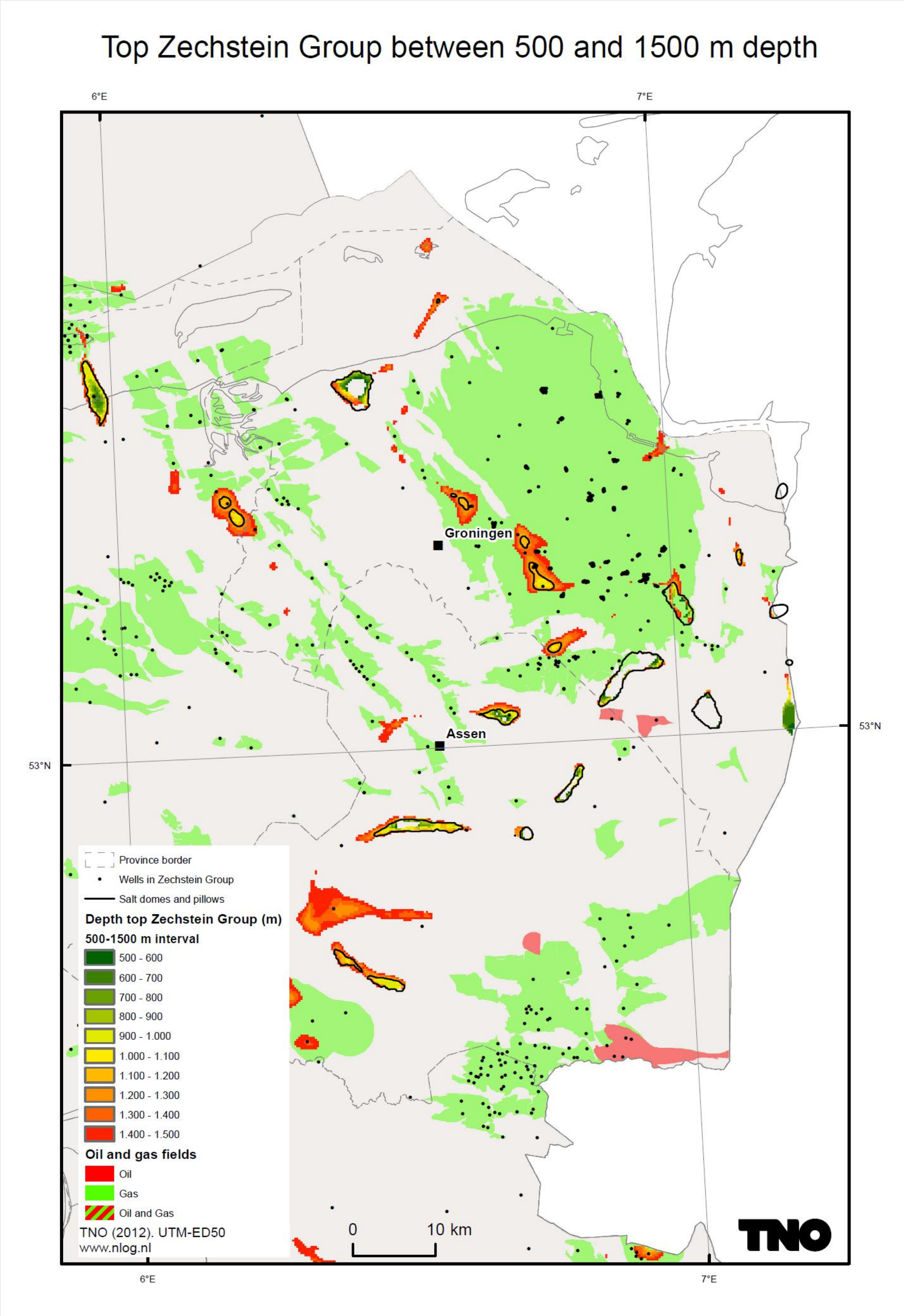

Figure 4-20 Zechstein top between 500 - 1500 m.

4.5.5. Salt domes

With respect to depth and thickness of a disposal location in salt, salt domes are the most obvious target. Rock salt in salt domes generally contains Zechstein Z2 salt (Figure 4-14) in the core and younger salt along the margins of the dome. This is shown to be true for several studied salt domes (Harsveldt, 1980, 1986). In other salt domes, the situation is reverse (Harsveldt, 1980). Salt domes have complex internal geometries and the ductile salt may include brittle inclusions (stringers) consisting of, for example, anhydrite (Van Gent et al., 2011). The internal tectonics of salt domes varies and needs to be addressed separately for each individual structure (Geluk, 1998). Geluk (1998) suggests the following steps when predicting the internal tectonics of a salt structure:

1. Assessing inhomogeneities within the salt. This should be done by using well and 3D seismic data of the dome and its surroundings. Attention should be given to lateral facies variations in the salt and anhydrite;
2. Understanding how inhomogeneities (stringers) will react to deformation;
3. Determining the structural position (bedded salt, pillow salt, salt dome, salt wedge) and shape of the salt structure (elongated, circular, overhangs etc.). This is done using detailed seismic mapping;
4. Unravelling the structural development during geologic history. Was the salt flow symmetrical or asymmetrical? Did the shape of the structure stay the same during geologic history? What were the consequences of the salt-flow pattern for the internal structure?

4.5.6. Aquifers surrounding rock salt

There are basically two configurations in which rock salt can be in touch with aquifers (Figure 4-21). The situation which is probably most relevant for this study, is when a salt dome has grown and intersects present aquifers or reaches/is covered by an aquifer at the top. Fluids may migrate from permeable aquifers through fractures in brittle inclusions into the margin of the salt dome. However, details of this mechanism are unclear and it needs further investigation. The other configuration involves a fault with an offset, juxtaposing salt against a permeable aquifer or fault zone. The 24 aquifers mentioned in Table 4-3 (Van Adrichem Boogaert and Kouwe, 1993-1997) may possibly occur in one of the configurations given above. The list is based on the occurrence of the aquifers in the north and east of the Netherlands, where thick salt layers and domes are present. Their actual configuration with respect to rock salt occurrences has not been checked.

Table 4-3 Aquifers which may occur on top of, next to or below Zechstein Group deposits in the north and northeast of the Netherlands. Aquifers are ordered chronologically from oldest below to youngest at the top.

Code	Full name	Facies	Dominant lithology
NU	Naaldwijk Formation	marine	Sand, very fine to medium grained (105 - 210 µm), clayey of slightly silty, carbonate rich, shells
NU	Eem Formation	marine	Sand, medium to very coarse grained (150 - 420 µm), marine shells
NU	Maassluis Formation	marine	Sand, very fine to medium coarse grained (63 - 300 µm), grey, mainly carbonate rich, conataining marine shells
NU	Oosterhout Formation	marine	Sand, very fine to very coarse (105-420 µm), shell remains and shells, glauconite
NU	Breda Formation	marine	Sand and clay, glauconite
NU	Appelscha Formation	fluvial	Sand and gravel

Code	Full name	Facies	Dominant lithology
NU	Peize Formation	fluvial	Sand, medium coarse to very coarse (210 - 2000 µm), no carbonate, slightly to medium gravelly (fine and medium coarse; 2 - 16 mm)
NU	Urk Formation	fluvial	Sand and gravel
NU	Drente Formation	glacial	Sand and clay
NU	Formatie van Peelo	glacial	Sand and clay
NMVFV	Voort Member	marine	Sands
NMRFV	Vessem Member	marine	Sands
NLFFS	Brussels Sand Member	marine	Sands
NLFFT	Basal Dongen Tuffite	-	Silty tuffite
CKGR	Ommelanden Chalk	marine	Thick succession, predominantly consisting of carbonate rocks.
KNNSF	Friesland Member	marine	Mainly consists of fine- to medium-grained argillaceous, glauconitic sandstones, although locally grading into conglomeratic sandstones
KNNSG	Gildehaus Sandstone Member	marine	Chiefly consists of coarse-grained to conglomeratic sandstones with pebbles of quartz, sandstone and limestone.
KNNSP	Bentheim Sandstone Member	marine	Generally fairly thick sequence of massive sandstones, calcareous, with abundant shell fragments, lignite particles and glauconite grains
RNSOB	Basal Solling Sandstone Member	marine	A light-coloured sandstone interval
RBMH	Hardegsen Formation	fluvial	Comprising several stacked alternations of off-white to pink sandstones and red claystones
RBMDL	Lower Detfurth Sandstone Member	fluvial	East: two distinct sandstone units with an intercalated layer of reddish-brown siltstone or silty claystone
RBMVL	Lower Volpriehausen Sandstone Member	fluvial	This member is a well-defined, pink to grey, (sub-)arkosic sandstone unit
ZE	**Zechstein Group**	**marine**	**SALT LAYERS, PILLOWS, DOMES**
RO	Rotliegend Group	aeolian-fluvial	Sands
DC	Limburg Group	fluvial	Sands

At an abandoned salt cavern at Barradeel (Van Heekeren et al., 2009) measurements have shown that since shut-in, fluids leak off at rates suggesting permeation rather than fracturing of the rock salt. This implies that the pressure in the abandoned cavern at 2500-3000 m depth did not reach levels high enough to fracture the salt. This in turn suggests that in more shallow situations relevant for nuclear waste storage, pressures and temperatures are likely not to cause fracturing of the rock salt either.

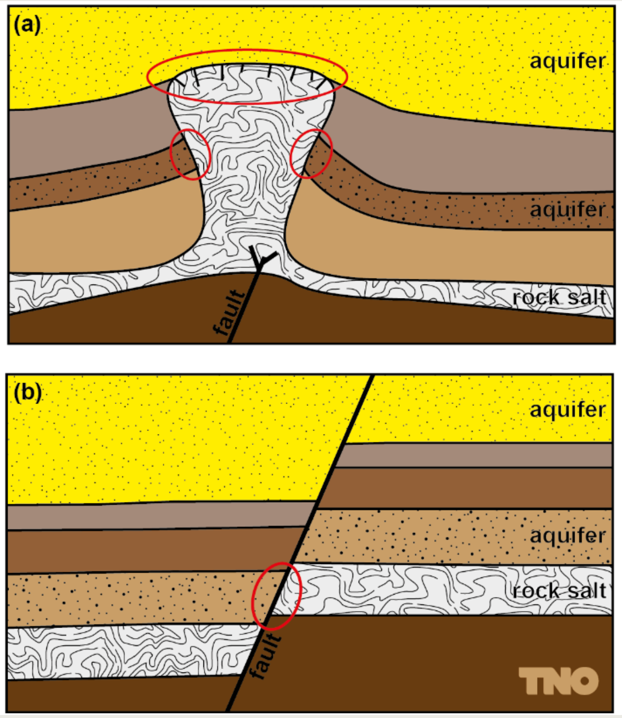

Figure 4-21 **(a) A salt dome which has grown through overlying rock layers. The red circles indicate possible connections between an aquifer and the salt; (b) A salt layer offset by a fault, causing juxtaposition. Connectivity between aquifer and salt may also occur via the fault zone.**

4.5.7. Salt formation in the Netherlands - evaluation

First of all it is important to consider all the extensive studies which have been done with respect to disposal of nuclear waste in rock salt during the 1980's and 1990's. Part of the work done in those studies is still relevant and up-to-date, other parts will have to be revised using the current state-of-art.

Knowledge on salt domes is steadily increasing, due to their use for salt extraction and possible gas storage. With respect to gaps in knowledge on salt-dome properties, attention should be given to:

- Detailed 3D seismic mapping of the shape and internal structure of salt domes;
- Acquiring borehole data (logs and cores) from salt domes;
- Improving knowledge on connectivity between aquifers and salt domes;
- Assessing inhomogeneities within the salt, including encapsulated liquids and gasses;
- Understanding how inhomogeneities (stringers) will react to deformation;
- Determining the structural position (bedded salt, pillow salt, salt dome, salt wedge) and shape of the salt structure (elongated, circular, overhangs etc.);
- Unravelling the structural development during geologic history.

4.6. Safety-relevant processes

4.6.1. Rock salt in the near field/excavation damage zone (EDZ)

The originally undisturbed host formation will be disturbed by excavation processes that are initiated by the construction of a repository. Due to the excavation the rock salt surrounding the excavated areas will be exposed to a geomechanical response. In this section the characteristics of rock salt will be discussed, that are related to the mechanical effects in the vicinity of excavated areas. In Figure 4-22 the different states of rock salt before and after excavation activities are shown (Marschall, 2008; p. 49). It starts with undisturbed rock salt (see Section 4.6.2) and ends with the backfilled state including internal pressure.

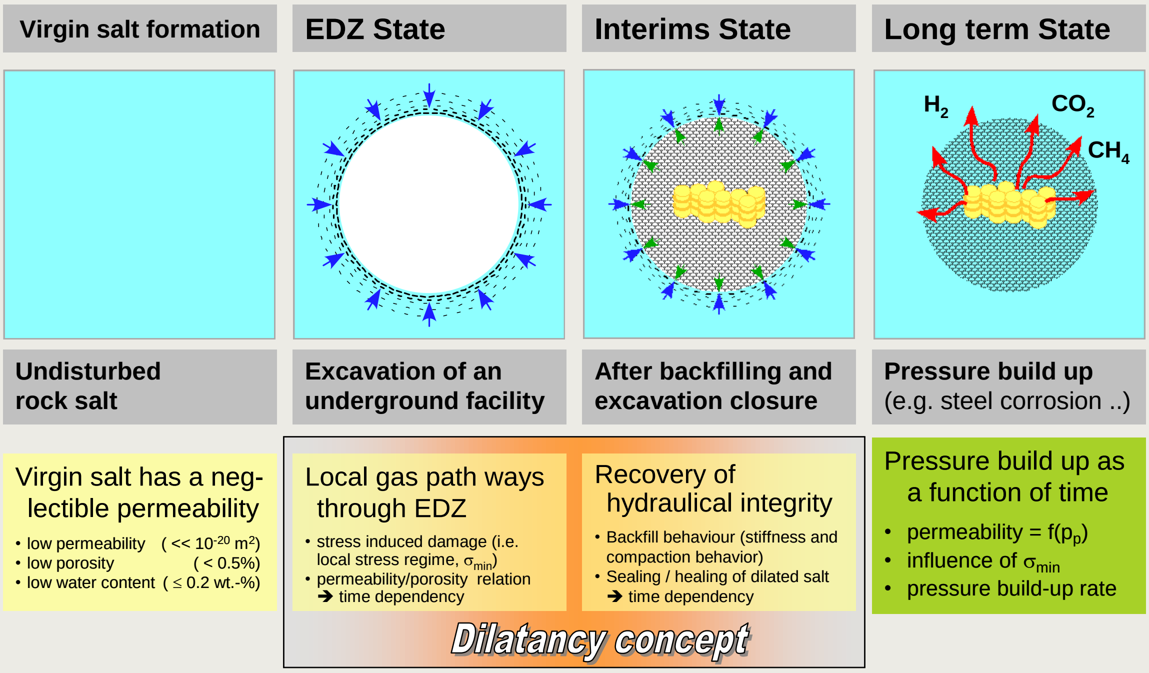

Figure 4-22 Schematic view of different states of rock salt in relation to underground mining.

Disturbed rock zone (DRZ)

A disturbed rock zone is an inevitable feature close to an excavated area in rock salt. It is of high importance for all constructions (seals, backfill) that are placed later in the mine. The DRZ may contribute significantly to flow and transport processes in any part of the mine.

A DRZ evolves as a consequence of the disturbance of the initially static equilibrium of stress in the rock salt. Fissures and cracks can evolve proximal to the excavations. The fissures continue to grow and develop networks around the excavations with preferential orientations parallel to the opening (Hansen, 2011; p. 25). The stress differences will decrease in time, after the open spaces in the excavations have been closed, and the DRZ can heal and its properties may return to those of undisturbed rock salt.

The DRZ is the zone around an excavation that exhibits changes in hydraulic and mechanical properties, although the dimensions are not clearly defined. Typical properties of a DRZ are (1) dilational deformation ranging from microscopic to readily visible, (2) loss of strength evidenced by rib spall, floor heave, roof degradation and collapse, and (3) increased fluid permeability via connected porosity (Hansen, 2011; p. 25).

The relevant properties of the DRZ related to potential flow and transport in the repository are cross-section, porosity and permeability. Many investigations have been performed to characterise these properties, see for instance (Rothfuchs, 2009). Laboratory salt creep measurements have demonstrated, that the damage can be expressed in terms of volumetric strain and principle stress. A variety of theoretical models have been derived to describe this behaviour, see for instance (Schulze, 2007).

If the shear stress diminishes, i.e. at a stage of the development of the repository when the pore spaces are closed, the damage can reverse and a (mechanical) healing process can start. Although the processes and time scales for the healing process are not completely understood, it can be assumed that at elevated temperatures in a repository with heat-generating waste the time for healing of the DRZ is in the order of several hundreds of years (Lux, 2007).

Principles of salt deformation

Rock salt is a material with viscoplastic and elastic behaviour. Considering the temperature and pressure conditions in a repository the elastic behavior is not very distinctive. The viscoplastic behavior means, that rock salt not only receives a permanent deformation after the application of loads but continues to undergo a creep flow as a function of time under the influence of the applied load.

Due to the viscoplastic behaviour of salt, the original in-situ stress condition prior to excavation is isotropic (lithostatic, i.e. $\sigma_1 \approx \sigma_2 \approx \sigma_3$). After excavation, the rock close to the opening experiences deviatoric (shear) states of stress ($\sigma_1 > \sigma_2 > \sigma_3$). These deviatoric states of stress induce viscoplastic flow and deformation. While elastic deformation occurs instantaneously as a response to changes in stress state, the other mechanisms are time-dependent, and are known as creep of salt. Figure 4-23 shows as an example of mechanical effects the strength behaviour for different minimum stresses (Günther, 2007; p. 110).

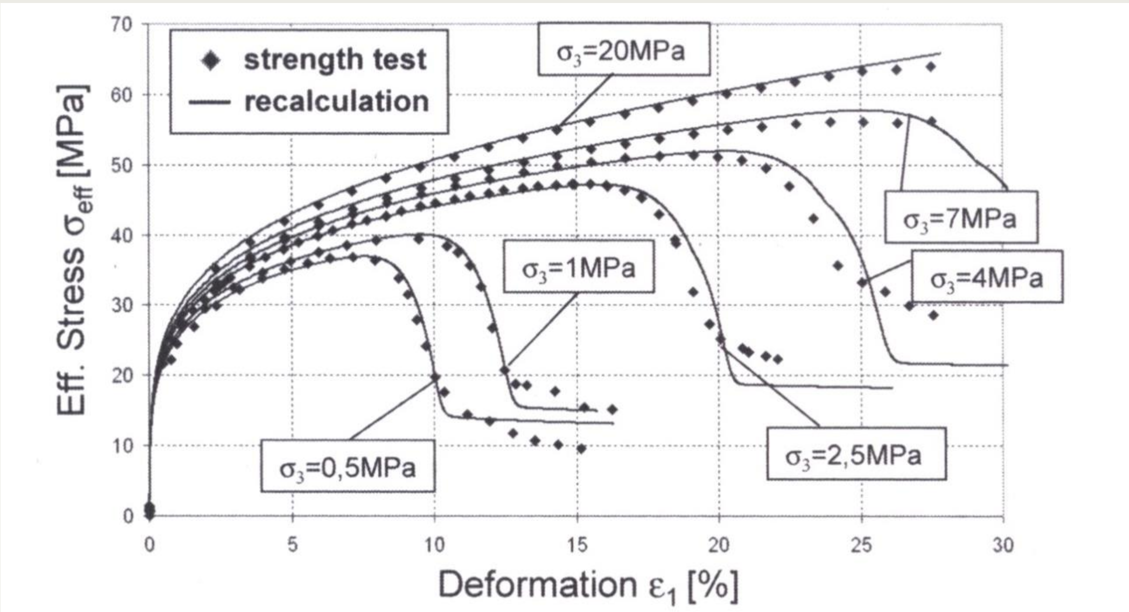

Figure 4-23 Dependence of effective stress on deformation.

Due to the importance of creep, it has been widely investigated in situ and in the Laboratory. Results from U.S. and German salt repository programmes and from pure research programmes have been presented and discussed in a variety of special salt conferences and workshops, e.g. (Wallner, 2007; Bérest, 2012; Li, 2012). Details will be discussed in following sections.

Viscoplastic flow is an incompressible process and does not induce damage to the salt matrix. Damage is related to relatively high deviatoric stresses compared to the mean stress and results in microfracturing. Models for the microfracturing process are under debate.

Fractures in the rock salt can heal after closure of the excavations. Healing means either a recovery of the fractures by mechanical effects or a closure by chemical deposits (crystallization). The mechanical healing starts when the magnitude of the deviatoric stress decreases relative to the mean stress. Healing effects have been observed in laboratory experiments and in situ. The actual models for healing include crystallization. In a salt formation with elevated temperatures, as is expected for disposal of high-level waste, the healing process may be accelerated.

4.6.2. Rock salt in the far-field (undisturbed host rock)

Rock salt consists of sequences of saline deposits. In a bedded structure these layers exist in their natural sequence, while in a domal structure the sequences are uplift and folded by diapirism. The following description of characteristics and processes of rock salt is in general independent of the type of formation. Differences between the rock salt types will be described, if applicable.

After excavation of a mine the initial stress field of the undisturbed rock salt is disturbed. Due to deviatoric stress close to the open voids a damaged zone in rock salt develops, consisting of a network of fissures and fractures. Also in the presence of water in rock salt humidity induced cracks can occur (Hessner, 2007). The damaged salt is of higher permeability than the undisturbed salt and a potential pathway for fluids in the repository system. The region of damaged salt is called DRZ (damaged rock zone) or EDZ (excavation damaged zone) and its extension is typically in the range of 0.5 to 1.5 meter. It can heal over time, as will be elucidated in more detail in Section 4.6.2.4.

4.6.2.1. Thermal, hydraulic, mechanical, and chemical properties of the host rock

The knowledge of properties of all THMC processes is the basis for investigations of all saline systems. Recently, in the project VIRTUS (Wieczorek, 2013), the available literature about THMC parameters has been collected and evaluated. The aim of the VIRTUS project was to compile a database of all parameter values that are needed for THMC calculations, including band widths. At present, the information about these parameter values is still widespread and in each organization involved in THMC calculations the actually used values are treated as a proprietary knowledge data base.

As a general characteristic of rock salt, the densities of different rock salt components are well investigated and range from about 1800 kg/m^3 (Carnallite) to 2750 kg/m^3 (Anhydrite). The mean value for Halite is 2170 kg/m^3 and differs for halite of different strata (Leine, Staßfurt, Werra).

The thermal conductivity, heat capacity, and thermal expansion coefficient as well as their temperature dependencies of disturbed rock salt differ from those of undisturbed rock salt due to the higher porosity. The higher porosity is related to a higher permeability of the DRZ (see Table 4-5). The mechanical properties of disturbed rock salt are similar to undisturbed rock salt, but the effect of porosity has to be taken into account, which results in lower densities and changed values for elasticity, plasticity and viscoplasticity. The chemical properties of disturbed rock salt are the same as those of undisturbed rock salt.

In extensive laboratory studies the damage of salt has been investigated mainly in the U.S. and in Europe. It was found, that damage can be described by a stress-invariant dilatancy model, and that there is a constitutive relationship between dilatancy and permeability. The damage is linearly increasing with creep deformation under dilatant conditions and decreasing (or healing) with time in the compressive domain (Schulze, 2007).

The various properties are discussed in somewhat more detail in the following paragraphs.

Thermal properties: Undisturbed rock salt has a relatively high thermal conductivity compared to other types of host rock (hard rock, argillaceous rock) considered for disposal of radioactive waste. It is thus favourable for emplacement of heat-generating (high-active) waste, because the heat is transported relatively fast away from the emplacement sites. For various types of rock the heat conductivity, heat capacity, and thermal expansion coefficient as well as their temperature dependencies salt are adequately known and can be taken into account in safety assessments. Table 2-1 gives examples of best estimate values for thermal properties of various rock salt types (Wieczorek, 2014; Annex).

Table 4-4 Examples of rock salt characteristics from the VIRTUS project.

Parameter	Halite	Potash	Anhydrite	Salt Clay
Thermal conductivity [W/m K] (0 to 400 °C)	6.1 – 2.1	0.6 (0 - 160 °C)	6.1 – 2.1	3.0 – 2.19 (0 - 160 °C)
Heat capacity [J/m² K] (0 to 250 °C)	875 - 912	n.a.	875 - 912	n.a.
Thermal expansion coefficient [1/K] (20 – 160 °C)	$3.9\cdot10^{-5}$ $4.3\cdot10^{-5}$	* $3.3\cdot10^{-5}$	* $4\cdot10^{-5}$	n.a.

* used for VIRTUS calculations. No best estimate value available due to sparce data

n.a. no data available

Hydraulic properties: Undisturbed rock salt is impermeable to fluids. Due to this property, rock salt formations are often used as reservoirs for gas and oil and play an important role for the storage of fluctuating energy supplies (power to gas). From the sedimentation process, rock salt formations can contain considerable amounts of residual liquids: brine or hydrocarbon. In domal salt, the content of liquid inclusions is generally lower than in bedded salt; this can be explained by the uplift and folding process, by which the liquids may be moved out of the rock salt. In (Bracke, 2012) an overview is given about hydrocarbon content in rock salt and the potential processes of slow liquid movement, e.g. by thermo-migration. Besides these processes, a movement of liquids within rock salt is only possible in fractures and fissures, e.g. in damaged zones.

Parts of the rock salt like anhydrite may be of higher permeability to liquids. Although the permeabilities of these parts are in the undisturbed material of comparable size as in halite, the material is less flexible and in the disturbed state may contain higher amounts of fissures. In Table 4-5 some examples of hydraulic properties from (Wieczorek, 2014; Annex) are compiled.

Table 4-5 Hydraulic properties compiled in the VIRTUS project.

Parameter	Halite	Sylvinit	Anhydrite	Salt Clay
permeability (undisturbed) [m²]	$< 10^{-21}$	$< 10^{-21}$	$< 10^{-21}$	$< 10^{-21}$
permeability (DRZ) [m²]	10^{-21} - 10^{-14}	$< 10^{-16}$	$< 10^{-06}$	$< 10^{-18}$
diffusion coefficient [m²/s]	10^{-9} - 10^{-7}	-	-	-
hydraulic dispersion length [m]	0.05 – 0.005	-	-	-

Mechanical properties: Due to its mechanical properties rock salt has the tendency to creep under external mechanical load. The stress field and the creep process are described in Sections 4.6.2.2 and 4.6.2.3, respectively. The values of mechanical parameters

(parameters describing elasticity and viscoplasticity) are generally available for all materials, although some are still under debate, see the discussions on international conferences, e.g. (Cosenza, 1993; Wallner, 2007; Bérest, 2012). The mechanical properties of rock salt are the basis for constitutive models describing the mechanical processes, see Section 4.6.4.

Chemical properties: The mineral composition of a saline system depends on its chemical composition. The different mineral phases found in salt domes are summarized in Table 4-6. A disposal concept in rock salt should be located in halite (NaCl), and inclusions of anhydrite or carnallite should be generally avoided.

Table 4-6 Salt minerals and their chemical composition.

Name	*Chemical composition*
Halite (rock salt)	NaCl
Anhydrite	$CaSO_4$
Gypsum	$CaSO_4 \cdot 2H_2O$
Bischofite	$MgCl_2 \cdot 6H_2O$
Carnalite	$KClMgCl_2 \cdot 6H_2O$
Kainite	$KClMgSO_4 \cdot 2.75H_2O$
Kieserite	$MgSO_4 \cdot H_2O$
Langbeinite	$K_2Mg_2(SO_4)_3$
Polyhalite	$K_2MgCa_2(SO_4)_4 \cdot 2H_2O$
Sodium carbonate	$Na_2CO_3 \cdot 2H_2O$
Sylvite	KCl
Epsomite	$MgSO_4 \cdot 7H_2O$
Hexahydrite	$MgSO_4 \cdot 6H_2O$

Details of the chemical system and chemical reactions of the host rock are described in Section 4.6.5.

4.6.2.2. Stress field

The stress field of undisturbed rock salt is the basis to compare with the stress field in the case of damaged salt. Undisturbed salt at repository depth exhibits an isotropic, lithostatic state of stress. The static equilibrium is disturbed in those regions of the homogeneous salt volume that are involved in underground mining activities, see Section 4.6.3. The disturbed stress field results in deviatoric stress and microfractures, as already mentioned in Section 4.6.1.

4.6.2.3. Creep / plasticity

In an undisturbed salt body with isotropic stress field no small-scale relative movements occur; only a movement of the entire salt body by uplift (diapirism) or other large-scale effects is possible. Viscoplasticity is an intrinsic behaviour of salt and is visible in the case of creep: if an open void is excavated and the stress field becomes anisotropic, the salt tends to creep into the direction of the open void. This will be discussed in more detail in Section 4.6.3, because it describes mainly the convergence process in a backfilled area. The effect of creep on the healing process in a DRZ is described in Section 4.6.2.4.

4.6.2.4. Healing of DRZ

Disturbed or damaged zones in rock salt can heal over time, if the stress field changes from dilatant to compressive conditions. The models for healing of these zones are based on observations and have been investigated in several projects. Presentations at international conferences (e.g. Wallner, 2007 and Bérest, 2012) show the variety of models that have been tested applying different computer codes. Evidence for healing has been observed in laboratory experiments, small-scale tests, and through observations of natural analogues.

The damages in a DRZ are most pronounced close to the surface of the excavations which it is surrounding. With increasing distance from the surface the intensity of damages decreases. The permeability of a DRZ is highest close to the surface and for the calculation of transport processes (flow of fluids through the DRZ) an average value has to be used.

The healing mechanisms include microfracture closure and bonding of fracture surfaces (Pfeifle, 1998). Microfracture closure is a mechanical response to increased compressive stress applied normal to the fracture, while bonding of fracture surfaces occurs either through crystal plasticity, a relatively slow process, or pressure solution and re-deposition, a relatively rapid process (Hansen, 2011; p. 27) (Pluymakers, 2014; p. 1). At elevated temperatures healing is faster than at lower temperatures due to higher creep of salt. Thus in repositories for heat-generating waste with rock temperatures up to some 100 °C, the DRZ will disappear rapidly in the galleries close to the waste[17].

In (Hansen, 2011; e.g. Section 2.4.1.5) some examples of field and laboratory healing tests in the U.S. are described, demonstrating that healing occurs. In the German ALOHA project healing around a rigid enclosure in the Asse mine in Germany has been investigated (Wieczorek, 1999; Wieczorek, 2004), and demonstrated a low permeability of the DRZ relative to the adjacent drift after healing. Similar studies have been performed e.g. in Canada.

In the Netherlands the healing of DRZ's in rock salt based repositories has been studied at the Utrecht University (e.g. Hart, 2009; Houben, 2008). An important conclusion of that work was that the geometry of a crack in a rock salt is not known. Therefore three models, all describing different crack geometries, have been used to predict the permeability evolution over time, for a brine-filled as well as for a 'dry' case (i.e. with a thin brine film on the crack walls). Permeability experiments were performed on dilated salt and showed that after an initial quick permeability rise the permeability decreased with time. Three theoretical models were applied to assess the test results. It was concluded that more research would be needed to be conclusive about the way intersecting cracks heal and to see whether or not the penny-shaped crack model is suitable for natural rock salt.

In the presence of a liquid phase, a DRZ can heal by recrystallization of salt minerals (additionally to healing by compaction, see above). In general, the knowledge of thermodynamic data of these processes is good, but for elevated temperatures above 100 °C more data are necessary. Such high temperatures can be expected close to heat-producing waste, and if the surface storage period prior to disposal is relatively short (i.e. 40 to 60 years). However, for the Dutch situation of elongated surface storage and decreased heat production, such high temperatures are not applicable. On the other hand, even if such high temperatures are not applicable, some data at elevated temperatures are missing.

4.6.2.5. Thermal effects

Thermal effects play a major role for many processes in rock salt, especially for mechanical processes, mainly because heat assists creep without creating fractures. The temperatures close to the emplaced waste and thus also in damaged rock zones can be as high as 200 °C (e.g. Bollingerfehr, 2012,see Section 4.3.3.3), which is much higher than the rock temperature[17]. There will be a thermal pulse from the heat-generating waste; the

[17] It is noted that in the present Dutch context, viz extended surface storage, these elevated temperatures will not be reached for heat-generating waste.

temperature will drop to initial rock temperature after several hundreds of years. At elevated temperatures in a HLW repository the creep rates are drastically higher than at normal rock temperature, which has been demonstrated by many experiments in the laboratory and by field observations, see Section 4.6.3.2.

The physics of plastic deformation of salt is governed by processes at the microscopic scale. Influence of impurities, grain boundaries, potential water contents, etc. must be taken into account. At elevated temperatures a dehydration of the rock salt has been observed, which is favourable for a smaller release of radionuclides from the waste, because water is necessary for most of the mobilization processes.

Thermal effects on hydraulic and chemical processes are taken into account, but are of less importance compared to the mechanical effects, especially in the Dutch situation with the expected mild temperature increases. At elevated temperatures the density and the viscosity of fluids are lower, and evaporation is of higher relevance. Most of the chemical reactions are temperature-dependent. Changes in solubility limits, sorption coefficients, etc. due to temperature changes are well investigated and documented.

4.6.2.6. Effects of radiation

A topic for which considerable and detailed research has been performed, both in Germany and the Netherlands, was on the possible consequences of radiation damage in rock salt (e.g. Hartog, 1988). It has been shown that due to the gamma radiation the NaCl crystals can be disintegrated into (colloidal) sodium and chlorine, and that energy is stored in the damaged crystals. Furthermore, laboratory experiments with very small samples have shown that the stored energy can be released instantaneously resulting in the complete destruction of the sample. Due to uncertain model predictions of the build-up of radiation damage and a lack of unambiguous evidence it cannot yet be excluded that radiation induced stored energy will be released completely in a relatively short time.

The gamma energy will be deposited in the salt in the direct vicinity of the waste. Most of this energy will be converted into heat, whilst a small portion will create defects in the salt crystals, i.e. the disintegration of NaCl into (colloidal) sodium and chlorine. It has been shown that due to this disintegration energy is stored in the damaged crystals which could be released if the Na and Cl recombine. A typical value of 70 J/g is found for the stored energy per mole percent colloidal sodium. If an overpack of 5 cm is applied damage can be avoided (OPLA, 1989; p. 80). If this will not be done it cannot be excluded that damage occurs. The effects however are concentrated in a small area of some metres around the HLW (Prij, 1991; pp. 178-190). Possible cracks would heal in a relatively short time due to creep and recrystallization.

4.6.2.7. Coupled thermal – hydraulic – mechanical – chemical effects

In general, thermal, hydraulic, mechanical, and chemical (THMC) effects as described in the previous section are coupled processes (see Figure 4-25) that together describe the behaviour of the host rock. For the purpose of a safety assessment however, the simultaneous modelling of all coupled processes is usually not necessary. For example, in the case of a mild or absent heat output from the disposed radioactive waste the thermal effects are insignificant. Additionally, as the coupling parameters are not always known adequately, simplifications are necessary. In the description of salt systems primarily the hydraulic and mechanical effects are coupled. Thermal effects are generally always taken into account, although some parameters of temperature dependencies are not known.

Chemical effects are usually treated separately, and the approach used depends on the particular question. Main aspects here are the corrosion/degradation behaviour of the waste container and waste matrix, degradation of cementitious barriers (dams, shaft closure), the solubility of radionuclides in brine, and the general behaviour of rock salt or other minerals of the host rock in the presence of a solution. The coupled modelling of chemical and thermal effects, i.e. the behaviour of chemical reactions at higher temperatures depends on the availability of experimental and thermodynamic data for the elements of interest. For example, NEA-TDB (Thermodynamic Data Base) is the major source for data of the aqueous and solid uranium species (Guillaumont, 2003). At present there is a considerable effort to review the base of thermodynamic data (Moog, 2015) including temperature dependencies. There is only little information available about the coupling of chemical and mechanical effects, although work in this direction is in progress.

In undisturbed salt there is almost no hydraulic system and the coupling is mainly thermal – mechanical. There are many articles regarding the temperature dependent mechanical state and mechanical effects in salt, see for instance the proceedings of the SaltMech conferences (Wallner, 2004; Bérest, 2012).

In damaged salt all THMC effects should be taken into account and a fully coupled description is the aim of all investigations. However, due to its complexity this approach is difficult if not impossible to accomplish and many simplifications must be introduced. The most convenient approach is to treat one of the couplings in detail, e.g. hydraulic-mechanical, and to consider a third coupling in a simplified way. For instance in a geo-mechanical assessment of the structural integrity of shaft seals (Müller-Hoeppe, 2012) the integrity of seal elements has been modelled by creep of salt due to external stress, mechanical back stress from the seal and change of hydraulic system (porosity) by compaction. The chemical environment in the pore space of the seal and the mechanical parameters of the seal elements have been estimated from chemical interaction of brine with solid phases for specified (estimated in advance) chemical systems.

The healing of a DRZ (Section 4.6.2.4) is a typical example of a coupled THM process. There are many articles dealing with this topic, see for instance the SaltMech conferences (Wallner, 2004; Bérest, 2012). Coupling of hydraulic and chemical effects is of high interest for the assessment of transport of contaminants and mainly relevant in systems with large pore volumes in crushed salt, see Section 4.6.3.4.

The THERESA project ("Coupled thermal-hydrological-mechanical-chemical (THMC) processes for application in repository safety assessment") was an international co-operative research project sponsored by the European Commission (EC) under its FP6 programme (Jing, 2010). The overall goal of THERESA was the development of a scientific methodology for evaluating the capabilities of mathematical models and computer codes used for the design, construction, operation, performance assessment (PA), safety assessment (SA), and post-closure monitoring of geological nuclear waste repositories.

The focus of the research was on constitutive models of long-term damage behaviour of rock salt, constitutive models for representing the coupled THM behaviour of bentonite and bentonite-rock interfaces, more specifically the EDZ, and solution of ill-conditioned partial differential equations governing the coupled THM processes of rock salt, bentonite and bentonite-rock interfaces.

4.6.3. Crushed salt

Before closure of a repository, the open excavations will be backfilled. One of the favourite materials for backfilling is crushed salt, which originally has been excavated from the mine and stored for backfilling purpose. Crushed salt as a backfill will be compacted in time due to creep of rock salt surrounding the backfilled excavations (convergence). By this process, the initially high porosity of crushed salt (30% to 40%) will be reduced to low values of around 1% or even less. The initially high permeability will then drop to low values in the range of permeabilities of undisturbed rock salt (Bollingerfehr, 2012; p. 17).

4.6.3.1. *Thermal, hydraulic, mechanical, chemical properties*

Crushed salt is in several aspects different to undisturbed rock salt (cf. Section 4.6.2.1). The chemical properties are in principle the same. However, the global thermal properties are changed due to the high porosity, e.g. the open voids in the backfill can reduce the thermal conductivity if they are filled with gas as an isolator. The consequence of an initially high porosity is a high initial permeability for fluids compared to undisturbed rock salt. There is a relationship between porosity and permeability, see for instance (Prij, 1993; Section 5.2.2, and Müller-Lyda, 1999; Schröder, 2009; Section 3.3.2). Geomechanical properties as the permeability of crushed salt are different to undisturbed rock salt over a wide range of porosity values. At high porosities, the material can be more easily compacted by external stress. There are several constitutive models to describe the mechanical behaviour of crushed salt, see for example contributions in (Wallner, 2007).

4.6.3.2. *Compaction of crushed salt*

Crushed salt is widely used as backfill material in radioactive waste disposal. After backfilling, the crushed salt has high porosities of about 40% and is thus highly permeable for fluids. By creep of the surrounding rock salt (convergence) the backfill will be compacted over time. By compaction the porosity will drop to values in the range of 1% or even lower.

The physics of plastic deformation of salt is governed by processes at the microscopic scale, see Section 4.6.2.3. In addition to undisturbed salt, for crushed salt the processes related to re-crystallization of salt particles at the contact spots of the salt grains have to be considered too.

The process of backfill compaction at low porosities is still not fully understood and under debate (Kröhn, 2009; Kröhn, 2012). For the understanding of contaminant transport in a repository knowledge of the residual porosity in backfilled regions is essential. Relevant aspects are: (1) is there a connected pore space at these porosities?, (1) which permeabilities are to be expected under these porosities?, and (3) what is the size of the pores at the end of the compaction process (volume for dilution and transport of contaminants)?

Compaction of crushed salt has been investigated internationally in a wide range. In Germany, experiments in different laboratories and under different conditions have been performed to investigate the compaction of crushed salt in general and with respect to concrete disposal projects, e.g. (Schulze, 2007; Wieczorek, 2010). In the U.S. experiments have been performed for the WIPP disposal site (Hansen, 1995). In the Netherlands relevant experiments and modelling have been performed at the Utrecht University (Zhang, 2006). The experiments have been performed for a wide range of porosities and external stress. Some experiments ended at porosities in the range of 1%. There is an ongoing debate about the question whether the experimental conditions (high pressure yielding a fast compaction) are adequate for in-situ conditions. Thus, further field investigations are necessary, see below. Another open question is the final porosity that can be reached upon compaction by plasticity: is the observed low porosity of undisturbed rock salt a consequence of sedimentation (bedded salt) or faulting (domal salt) or can it be reached also by creep?

Field investigations on the compaction of crushed salt are reported for the Asse mine and commercial salt mines in Germany (e.g. Brenner, 1999), and for the WIPP disposal site in the U.S. (Hansen, 1995), etc. These investigations are for sites with mainly natural temperature fields, but some of the results are for elevated temperatures as well (Rothfuchs, 2003). These investigations demonstrate that the compaction process at the depth of emplacement fields lasts in the range of several decades to several hundreds of years.

As an example, Figure 4-24 shows the temporal evolution of porosity of backfilled areas in a repository with different amounts of humidity and with different temperatures (Larue, 2013; p. 125).

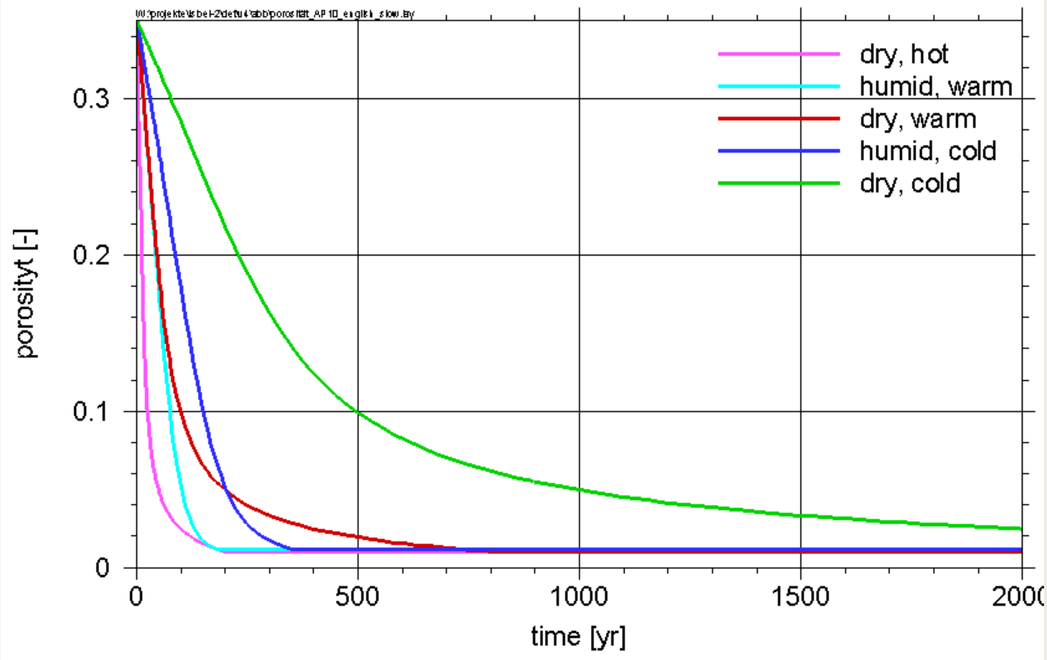

Figure 4-24 Temporal evolution of porosity at different temperatures.

4.6.3.3. Thermal effects

Crushed salt will be used in all parts of a repository with heat-generating waste and can thus be exposed to elevated temperatures. Depending on the type of radioactive waste and the surface storage time, the temperatures close to the disposed waste can be as high as 200 °C (e.g. Bollingerfehr, 2012; Section 4.3.3.3), which is much higher than the ambient rock temperature. In the Dutch context however, with an extended surface storage time, the maximum temperatures reached in the salt will be moderate, although that has to be confirmed.

Temperature plays a major role in many processes in rock salt. The mechanical behaviour is mainly influenced by the temperature dependence of ductility. Usually, in mechanical models the temperature dependency is taken into account by an Arrhenius term. Elevated temperatures may cause dehydration of salt and thus influence creep.

The thermal conductivity of crushed salt is lower than of undisturbed salt. The consequences are higher temperatures for instance for heat-generating containers surrounded by crushed salt compared to the host rock. Other thermal effects as described in Section 4.6.2.5 are also applicable to crushed salt.

4.6.3.4. Coupled thermal – hydraulic – mechanical – chemical effects

In the presence of brine, THMC interactions in crushed salt have to be taken into account to understand the geomechanical consequences, see Section 4.6.2.7. For instance in a recent project (Moog, 2007), the transport of radionuclides in a HLW repository has been modelled in a liquid phase, where the movement of liquid is driven by convergence of the salt (creep), the creep is influenced by back-pressure of the liquid, and the retention of radionuclides is calculated for a chemical environment, which from time to time is calculated with a separate geochemical code.

The compaction of crushed salt (Section 4.6.3.2) is a typical example for coupled THM processes. There are many articles dealing with this topic, see for instance (Wallner, 2007) and (Bérest, 2012). As part of the German TSDE project an attempt was made to predict numerically the porosity development of crushed salt in a temperature field and to compare this to measured values over 8 years. The predicted values show generally higher convergence rates, i.e. faster compaction as compared to the test results (Rothfuchs, 2003; Section 6), indicating that the modelling needs to be further improved.

Coupling of hydraulic and chemical effects is relevant for the assessment of the transport of contaminants through the pores in crushed salt. The pore spaces in crushed salt can contain a liquid or gaseous phase. The chemical equilibrium in this highly saline environment is usually calculated by applying the Pitzer approach for the activity correction (see Section 4.6.5). Depending on the availability of appropriate thermodynamic data for the individual processes (i.e. a large number of necessary Pitzer-constants), the equilibrium composition of the fluid and the equilibrium composition of the solid phase can be estimated. However, rock salt can undergo several conversions until equilibrium is reached (Ostwald phase rule), the added value of equilibrium models for such analyses is therefore questionable.

4.6.4. Constitutive modelling

The THMC behaviour of rock salt can be described by a variety of constitutive models. Related to performance assessments of repositories in salt, the term “constitutive model” is mainly used for the mechanical behaviour of salt. For instance, the mechanical behaviour can be described by models for the transient phase, a stationary phase of creep, and a creep failure phase. At present, chemical effects are rarely taken into account in these models, although there is progress in chemical modelling, see Section 4.6.5. Constitutive models should take into account as far as possible all the relevant processes that are active during the loading and the deformation of rock salt.

Thermal, hydraulic, mechanical, and chemical processes are strongly related under the conditions in a repository system. Figure 4-25 shows the relation of these processes in a simplified sketch (Kuhlman, 2014; Figure 1.3).

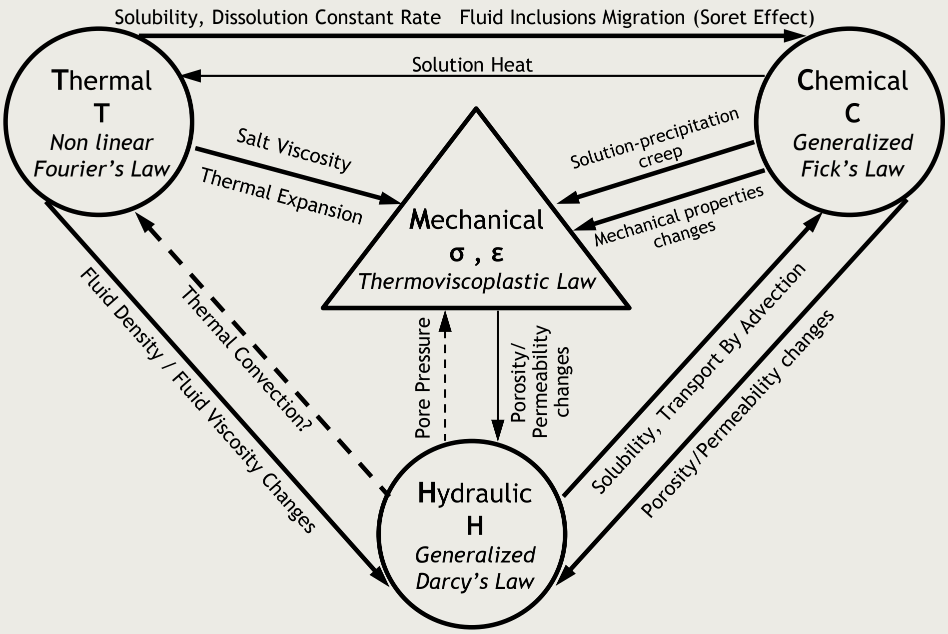

Figure 4-25 Relationship between thermal, hydraulic, mechanical, and chemical processes.

Dashed connections in this figure show processes that may not exist in undisturbed salt, while heavier solid arrows indicate stronger relationships between processes. Each process can influence the potentials and parameters of the other processes. Except under restrictive circumstances, it is not possible to isolate a single process without significant error and loss of physical significance (Kuhlman, 2014; Section 1.2).

The thermal-mechanical (TM) coupling in salt is characterized by the strong dependence of mechanical deformation properties on temperature; higher temperature reduces salt's effective viscosity.

The thermal-hydraulic-mechanical (THM) coupling in salt is further characterized by the even stronger dependence of salt's hydraulic properties on mechanical deformation, along with the differential thermal expansion of salt and brine.

Chemical effects can be superimposed on the THM representation, with additional coupling due to the precipitation and dissolution of salt in the pore space.

4.6.4.1. *Constitutive laws*

Constitutive laws have been proposed by many companies and research institutions and over a long time period. In 2007, an overview of advanced models has been given by (Schulze, 2007). In that paper, the models used in Germany by BGR, IfG, TUC, FZK, and IUB[18] are described. This selection of models refers to a joint project, in which the partners applied their models to test cases and compared the results.

BGR applies the Composite Dilatancy Model (CDM) which describes the total inelastic strain rate as a set of different functions (Hampel, 2007). IfG applies two different models, a strain-hardening model (Günther, 2007) and an elastic-viscoplastic model (Minkley, 2007). TUC applies a model in which the total strain rate is the result of the additive superposition of an elastic part, a viscoplastic part at constant volume, and a damage-induced dilatancy part (Hou, 2003). FZK applies an elastic-viscoplastic model, described in (Pudewills, 2007). IUB applies a modification of the former MDCF model (Multimechanism Deformation Coupled Fracture) (Hauck, 2001).

ECN/NRG has improved several times the model for convergence of crushed/compacted rock salt. Originally based on analytical work the model for convergence has been improved by taking into account the transient behaviour of free convergence and includes the properties for dry as well as brine saturated backfill (Prij, 1993; Section 5.2.3). Therefore both convergence and compaction of backfilled excavations in rock salt show a quite different behaviour under dry and wet conditions and result in orders of magnitude smaller release of radionuclides from a repository. To distinguish this adapted module from the original EMOS model it is called EMOS_ECN (Heijdra, 1995).

As part of the EU Framework project BAMBUS (Bechtold, 1999; Poley, 2000b), NRG applied a consitutive model for the description of the mechanical behavior of dry crushed salt, consisting of four distinguished processes: elastic strain, creep strain, thermal strain, and compaction strain (Bechtold, 1999; Section 5.3.3). This model has successfully been applied to the DEBORA 1/2 experiments.

In the METRO-III project of the CORA programme it was recognized that the compaction of dry crushed salt or plugs is substantially different from the compaction of moist/wet salt (Grupa, 2000; Section 4.4.3). Based on the work of Spiers, a 'Fluid Assisted Diffusional Transfer' (FADT) model was implemented in the EMOS code (EMOS_ECN) to enhance the description of wet crushed salt and plugs under the conditions of a flooded facility. Such conditions may occur e.g in the case of "abandonment" of a facility.

[18] BGR: Bundesanstalt für Geowissenschaften und Rohstoffe; IfG: Institut für Gebirgsmechanik; TUC: Technische Universität Clausthal; FZK: Forschungszentrum Karlsruhe; IUB: Institut für unterirdisches Bauen, Hannover

During the NF-PRO project the so-called ‘Coupled Creep Model’ (CCM) was developed describing the compaction rate of crushed salt (Zhang, 2006; Section 5.3). The CCM is based on mechanistic considerations and incorporates an additional model for the pressure solution creep of compacted crushed salt, but it also includes constants that need to be fitted by the measurement of several model parameters during compaction experiments.

As part of the PAMINA project, several of these model parameters were fitted and implemented in the EMOS code, resulting into the EMOS_ccm2 version (Schröder, 2008). That version of the EMOS code was applied to a probabilistic performance assessment (Schröder, 2009). An important finding of this work was that the rate of the compaction of salt plugs has only a minor effect on the ultimate dose rate to the biosphere (Schröder, 2009; Section 7.2.3). That result implies that putting efforts in enhancing models to describe the mechanical behaviour of salt only slightly affects the results of a long-term safety assessment of the salt-based repository.

Many other models to describe the mechanical behaviour of salt have been presented in the literature. It can be assumed, that the above mentioned models cover all of the other models, although differences in details occur. In the computer code package SIERRA (Argüello, 2012), which is applied in the U.S. programme, a multi-mechanism deformation (MD) creep model is implemented. This model is based on a model originally developed by Munson & Dawson (Munson, 1979).

Results from the above mentioned project show that there are differences in the calculated strain curves compared to measured values and that these differences vary from code to code (Heemann, 2009). Measured temporal evolutions of permeabilities of a gallery can be calculated in principle, but for instance the prediction of creep failure is different for different constitutive models or codes.

One of the outcomes of Schulze’s study was that the constitutive models for the behaviour of the DRZ should be improved to predict creep failure more precisely, to be supported by additional investigations on the evolution of dilatancy affected processes (e.g. for different types of rock salt) (Schulze, 2007).

4.6.4.2. Computer Codes

Computer codes for THMC calculations are applied in all fields of mechanical, hydraulic, and geochemical calculations. Important factors for calculations are the capabilities of computer hardware, which limit the size of computing meshes, the degree of coupling of different models, or the model size. The coupling of models in computer codes is most advanced for hydraulic-mechanical processes. Many aspects of these models have been treated in the 2003 conference (Davies, 2005). The coupling of geochemical processes with hydraulic processes has been attempted for several codes but is at the moment limited to hardware restrictions, because this coupling is time-consuming. The full coupling of all processes in one code is in principal possible but in reality limited due to very long computing times. With ongoing improvement of computing technology full coupling will be a future option.

A review of available computer codes for THMC calculations is given in (Kuhlman, 2014). Among the computer codes that are widely used in the field of thermal-hydraulic-mechanical calculations are: CodeBright, FEHM, TOUGH-FLAC, JIFE, and SIERRA Mechanics. Geochemical codes are addressed in Section 4.6.5.

All of these codes are based on either two- or three-dimensional models of the system and apply finite elements, finite differences, or finite volumes techniques to solve the equations. Most of the codes assume flow in a porous continuum, while some are able to treat also flow in fractured media. The coupling capabilities of the codes differ and depend on the applied techniques. There will be not “one” code that is able to solve all the existing tasks in THMC calculations, but there will be a need to develop specialized codes. This holds until the computer hardware is so much advanced that highest coupled codes can run in an acceptable time.

The codes for hydraulic calculations that have been developed recently take density effects better into account, e.g. d^3f (Schneider, 2012), which even models heat transport (TH(C) coupling). This is necessary for systems in highly saline environment, where the rock salt surface in contact to an aquifer provides the boundary condition for the density of the solution. If transport of contaminants (radionuclides) is considered an advective or diffusive process, transport is calculated in a flow field derived from other hydraulic codes. In these transport codes the coupling with geochemical codes is in development to take retardation (sorption) effects into account.

4.6.5. Chemical conditions / geochemical modelling

With respect to the chemical conditions and processes in the EBS of rock salt and their modelling, four main topics can be distinguished:

- dissolution processes in the host rock or crushed salt in the EBS in the presence of brine;
- corrosion processes of the waste container and waste matrix;
- corrosion processes of cementitious barriers;
- solubility/precipitation of dissolved radionuclides;
- gas production and transport.

4.6.5.1. Dissolution processes

The mineral composition of a saline system depends on its chemical composition. Figure 4-26 (Hartmann, 2006; p.11) shows as an example the quinary system Na-K-Mg-Cl-SO_4-H_2O to elucidate the different forms of mineral phases in a typical chemical system occurring with rock salt.

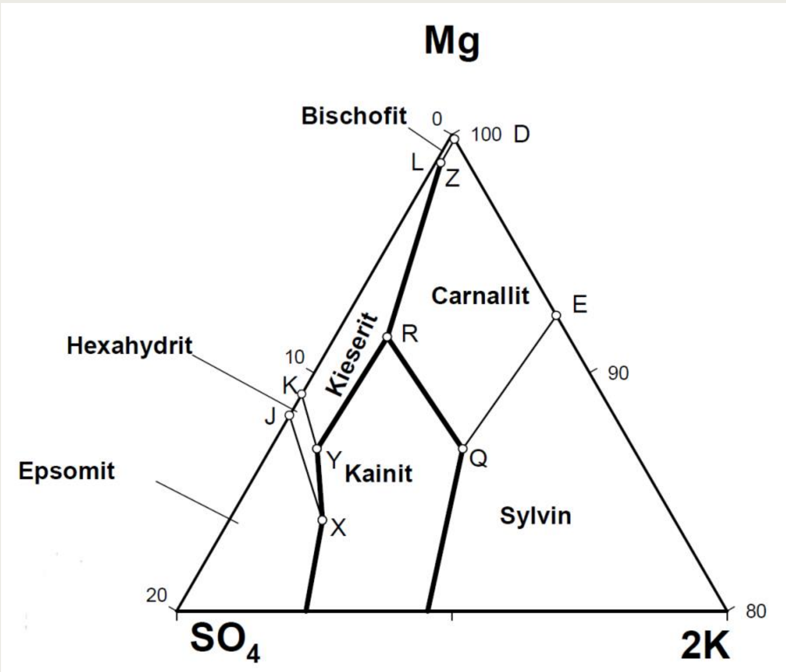

Figure 4-26 Part of the quinary system Na-K-Mg-Cl-SO_4-H_2O (Jänecke diagram).

Understanding the behaviour of a saline system is relevant related to the intrusion of a solution or to the displacement of brine already present in the host rock. In the presence of a solution or brine, chemical interactions between the components of the brine and the solid phase can occur (i.e. dissolution & precipitation) until a chemical equilibrium is reached. Of particular importance is the dissolution of carnallite, resulting in a complex sequence of dissolution/precipitation reactions that may affect the geomechanical stability of the host rock (hence the preference to construct a deep geological facility in halite). The dissolution/precipitation reactions depend on the composition of the brine. Any intrusion of solution from outside the host rock will quickly saturate during the contact with the host rock, and its composition depends on the source and minerals present along the flow path of the solution. In case of a brine intrusion from a brine pocket that has been existing in the host rock for a long time, the brine can be assumed to be in equilibrium with the surrounding host rock mineral. Potential dissolution/precipitation reactions in case of brine intrusion thus depend on the source and genesis of the brine.

Chemical equilibrium modelling is a useful tool in assessing and understanding of geochemical reactions, such as the dissolution and precipitation reactions discussed here, but also with respect to the analysis of corrosion and migration processes as discussed in the next sections. Several modelling tools exist for assessing these processes, e.g. (Allison, 1991; Parkhurst, 1999; Meeussen, 2003). For chemical equilibria of a saline system, "*salting out*" by high salt concentrations leads to a decreased activity of ions (Pitzer 1984): in geochemical modelling the so-called 'Pitzer approach' (Pitzer 1991) for the activity correction is applied in order to correct for this behaviour. Depending on the availability of appropriate thermodynamic data for the individual processes (i.e. the necessary Pitzer-constants), the equilibrium composition of the fluid and the equilibrium composition of the solid phase can be estimated.

Mineral dissolution and precipitation is a kinetic process not leading to instantaneous equilibrium between the solid and solution phase. Mineral-solution systems can undergo several conversions until equilibrium with brine is reached (Ostwald phase rule). It is evident that the equilibrium modelling approach provides useful insights in principal processes and interactions, but might only provide part of the information necessary to assess the effect of solution intrusions.

With respect to the earlier discussed THMC relations, mineral dissolution and precipitation can affect the geomechanical properties of the host rock, e.g. by changes of the mineral volumes.

The thermodynamic data in databases as THEREDA (Moog, 2015) are currently independent of pressure. However, from e.g. geothermal projects the dependence of these data on high pressures is known (see e.g. Pitzer, 1984). If appropriate, these dependencies could be included into future versions of the applied databases.

4.6.5.2. Corrosion of waste container and waste matrix

In case of the intrusion of a liquid phase into a repository in rock salt, brine that comes into contact with waste containers will interact with the containers and, once the container has failed, the waste matrix. The corrosion of iron in highly saline environment and at normal temperatures has been investigated in many projects (e.g. Smailos, 1992; Smailos, 1993). In the PROSA safety assessment chemical interactions of the waste material (steel, glass) with the rock salt are taken into account as corrosion of the canister (Prij, 1993; item 3.2.8). Presently, work is in progress to include the iron chemistry into a thermodynamic data base. Next to the corrosion of steel, vitrified waste and fuel elements, also the corrosion of cementitious material present either as shielding or as matrix element can occur. For geochemical modelling of CSH phases in cementitious material, several approaches exist (Kienzler, 1998). Dissolution behaviour of cementitious material affects the pH in solution, which is a relevant parameter also determining the solubility of radionuclides (see below).

Potential corrosion and degradation reactions depend on the composition of the brine. The composition of the brine depends on the source and genesis of the brine (see e.g. Kienzler, 2001; p.21*ff*).

4.6.5.3. Corrosion of cementitious barriers

Degradation and dissolution behaviour of cementitious materials as summarized in the previous section also occur in dams or shaft seals when these come into contact with brine. Degradation and dissolution processes can impair the proper geomechanical and hydraulic function of these barriers (see e.g. Müller-Hoeppe, 2012).

4.6.5.4. Solubility of radionuclides

The chemical conditions of rock salt can be characterized by the host rock composition and the liquid and gaseous phase eventually present in the pore space or in (larger) inclusions. The mobility of radionuclides in the host rock – once a waste container has failed and the waste matrix is dissolved – depends mainly on the solubility of the radionuclides under the specific local chemical conditions, while sorption of radionuclides to the host rock is assumed to be of lesser relevance[19].

Relevant sorption can occur on surfaces of the EBS and the waste matrix, or on their corrosion/degradation products. This is in general advantageous for the safety, because it decreases the mobility of radionuclides. However, sorption might also increase the mobility of radionuclides in case of sorption to mobile colloids (e.g. Kersting, 1999; Utsunomiya, 2009) which can be formed e.g. by degradation processes in the waste.

By the application of chemical equilibrium modelling, the distribution of the system components over different mineral or soluble forms can be calculated. At present, many models exist that allow to estimate radionuclide sorption to colloids, degradation products or components of the geosphere (e.g. Kinniburgh, 1999; Tipping, 2002; Hiemstra, 1996 & 1999; Tonkin, 2004; Bruggenwert, 1982; Bradbury, 2008). However, the application of such models faces currently three limitations:

- dissolution of a mineral system occurs often not in thermodynamic equilibrium, but intermediate, thermodynamic less favourable phases can appear (e.g. Vandenborre, 2008);
- in case of saline solutions, the reactivity (or activity) of soluble components differs from solutions in a weak electrolyte (which is the case for the major part of existing data);
- the major part of existing data is based on ambient temperatures, extrapolation to higher temperature is not always possible.

An important aspect is the availability of a consistent thermodynamic database. In parallel to the NEA thermodynamic database project (NEA-TDB[20], see also NEA, 2013c), the project THEREDA (Moog, 2015) is aiming at the compilation of a comprehensive chemical database with particular attention to reactions in saline systems. To allow to correct for the altered activity of species in highly saline environments, a correction by the "Ion Interaction Approach" (Pitzer, 1991) can be applied. However, unlike other activity correction methods (i.e. Debeye-Hückel, Davis-equation) applied successfully in less saline environments, the Pitzer approach requires a larger number of element specific correction-constants. Several organizations are involved to derive thermodynamic data for this approach, based on existing experimental data (Moog, 2015).

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[19] This is different from Boom Clay as host rock, where sorption plays an important role in the mobility of many radionuclides

[20] http://www.oecd-nea.org/dbtdb/ last accessed on 9 January 2015

Models for the calculation of radionuclide dissolution and transport in a saline system are implemented in a variety of computer codes. Most of these codes calculate thermodynamic equilibrium; the consideration of non-equilibrium reaction kinetics is in an initial state due to the lack of many reaction parameter values. Lack of data is, in addition to the complexity of the geochemical processes, also one of the reasons why geochemical models are not fully coupled to hydraulic models in appropriate codes. Another reason is the time-consuming calculation of chemical equilibria for many points in time and space.

In the PROSA project several geochemical reactions have been taken into account, e.g. chemical equilibrium reactions, processes in relation to transport of radionuclides, physio-chemical characteristics influencing chemical equilibria (Prij, 1993; p.A2.2).

4.6.5.5. Gas production and transport

As already mentioned in the previous sections, there may be different sources of gas production in the repository. Gas can be included in the rock salt and may migrate towards the repository or it can be produced by the waste in the repository, either as a result of corrosion processes, of radiolysis. Radiolysis is the chemical decomposition of water due to ionising radiation. Since radionuclide transport due to gas transport is very limited in rock salt repositories, radiolysis is considered of minor importance (Prij, 1993; p.8.17).

In **PROSA** the effect of gases on brine flow in a repository is considered in the REPOS code as exchange processes between segment models. In the PROSA scenario study the FEP “Gas mediated transport” was considered to play some role in a 'high gas pressure' scenario. However, due to a lack of models and data this scenario has not yet been analysed in PROSA (Prij, 1993; p.8.10).

Additional efforts on the gas production and gas-mediated transport were executed in **CORA**. Various FEPs have been identified that may annul the isolation effect of the plugs: bad construction, bad emplacement of the canisters, consequences of gas generation and radiation. In CORA the various radionuclide transport mechanisms were calculated using the EMOS code: squeezing the contaminated brine as a result of convergence through the plugs into the gallery, diffusive transport; gas driven transport and transport due to a temperature gradient.

The small relative contribution of gas-driven flow to the overall radionuclide transport mechanisms is shown in Figure 4-27, depicting results of EMOS calculations on the Metro-I concept (Grupa, 2000; Figuur 32). Although the transport velocity is very low, the long duration of the scenario (thousands of years) may result in a significant amount of the waste escaping from the disposal cell. Eventually, all transport is ended when the plugs become impermeable due to their compaction Grupa, 2000; p.143).

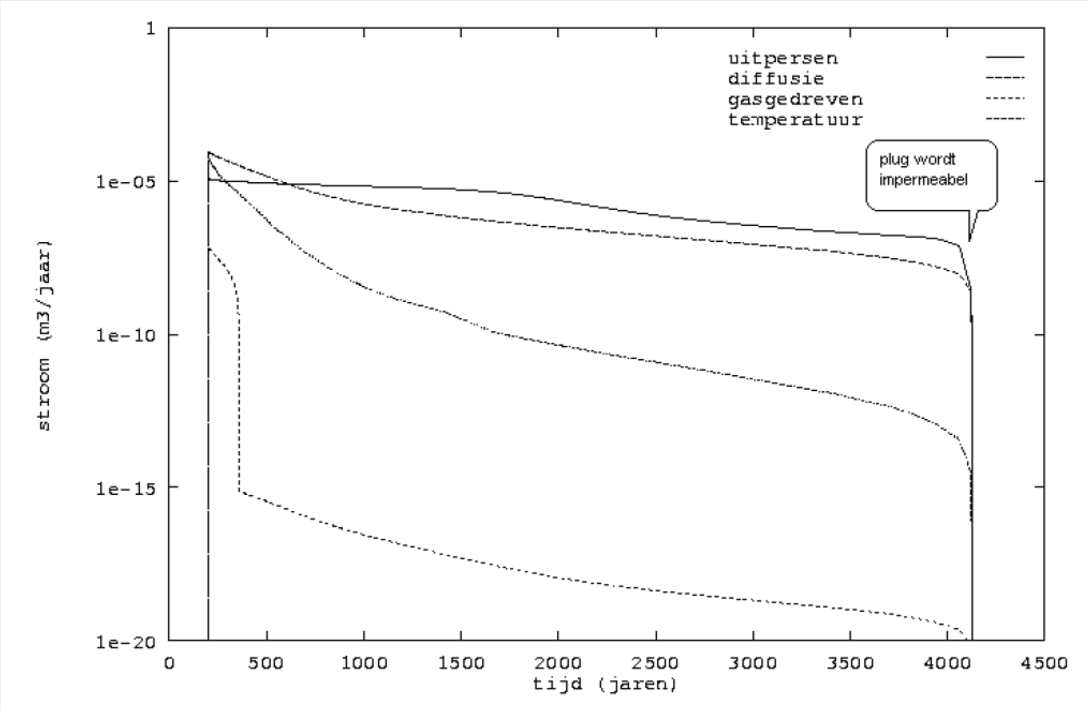


Figure 4-27 Contribution of different flow mechanisms driving dissolved radionuclides from the disposal cell to the gallery.

The contribution of gas and gas-driven flows to the overall repository safety has also been investigated in **Germany**. In the ISIBEL project , the accumulation of gas in a model of the repository waste segment and the release of gas from a segment have been considered in only in a very simplified form. The transport of gases within the underground construction could not be calculated explicitly but was modelled as a boundary condition which was determined by prior calculations using other computer codes. It was noted that, before conceptual models for the transport of gases can be developed, the volume of gas formation in a final repository for high-level radioactive waste needs to be determined. There were some doubts whether the effects of gases in the repository would be negligible so that the development of a programme would be unnecessary (DBE-TEC, 2008; p.72).

ISIBEL concluded that model predictions of gas transport and release processes within the underground constructions contain some rigourous simplifications Depending on their relevance to the assessment of the safety of the entire repository system, these individual models and programmes should be reviewed for potential improvements.

The German **VSG** project also addressed to some extent the effects of gas on repository safety (Bollingerfehr, 2013; p.110). It was recognized that there are no specifications in the German Safety Requirements regarding the consideration of the gaseous phase in a deep geological disposal facility, and thet no generally accepted calculational scheme exists at present. Due to the high degree of uncertainty of gas-related issues, it is only possible to assess radionuclide release in the gaseous phase with very conservative assumptions (if known) and parameter variations. (Larue, 2013) carried out calculations for the gaseous phase that indicated the following important influencing factors:

- the amount and location of containers with undetected failures;
- the layout of the mine;
- the gas production rate;
- the compaction process of backfill material; and
- the anticipated functional lifetime of the containers.

From the analyses performed in the VSG project It was concluded that further R&D projects would be necessary to be able to assess the complex behaviour of these processes in safety analyses (Larue, 2013; p.179).

As part of the **WIPP** safety assessment the production and potential transport of radioactive gases has been investigated too (US DOE, 2014; p.164,166). Although the transport of radionuclides from a repository in a gas phase is expected to be less significant than their transport in water (brine), researchers have identified pressure buildup of hydrogen gas due to anaerobic corrosion of steel container materials as a potential concern. Pressure changes and fluid flow due to gas production in the repository and the Salado are accounted for in the WIPP PA calculations through modeling the two-phase flow. In addition, microbial gas generation from degradation of organic material is accounted for in PA calculations.

In the near field, pressure buildup could inhibit rock convergence and consolidation of crushed rock backfill, depending on the associated volume of hydrogen and gas generation rates The associated hydrogen volumes and rates require further quantification (Hansen and Leigh, 2011; p.63).

The production and potential transport of *radioactive* gases are eliminated from the WIPP PA calculations on the basis of low consequence to the performance of the disposal system. Transportable radioactive gases are comprised mainly of isotopes of radon and C-14, Radon gases are eliminated from the WIPP PA because their inventory is small, and their half-lives are short (<4 days), resulting in insignificant potential for release from the repository. (US DOE, 2014, p.197).

4.7. Geosphere

The geosphere (overburden) characteristics themselves are affected to a minor extent by the characteristics of the host rock.

In VEOS, the rise of the overburden due to the rise of the salt formation was assumed to be eliminated by surface erosion (Prij, 1989; Chapter 5). This would lead to a gradually decreasing thickness of the overburden. Simultaneously, the salt-shield above the repository would gradually be removed by subrosion, ultimately leading to a contact between the contents of the repository and the groundwater system. Eventually, the subrosion process would cause the destruction of the repository and the release of its contents into the geosphere.

A similar approach was used in PROSA (Prij, 1993; Chapter 6). In PROSA however, the upward movement of the salt dome was treated as a function of depth. The rise of the dome was assumed to continue until it either reaches the surface or stops when the rate of subrosion is balanced by the rise of the salt inside the dome. As in the VEOS study, surface erosion may gradually decrease the thickness of the overburden. This process may continue during the release of nuclides, leading to a thickness that may vary during the release interval. Consequently, the travel times for nuclides that break through the geosphere and enter the biosphere may vary in time, not only because the velocity field changes, but also because the thickness of the overburden gradually decreases.

The process described above is outlined in Figure 4-28 (Prij, 1993; p.6.27). The solid line marks the initial position of the salt dome. The initial thickness of the overburden is denoted by the parameter L_0. Three additional stages of the rising salt diapir, at future times t_1, t_2 and t_3, are indicated by dashed lines. The thickness of the overburden is then L_1, L_2 and L_3, respectively. In the initial stage, at time t_0, a so-called salt-shield is assumed to be present between the repository and the groundwater system. As time progresses, this salt shield is gradually removed by subrosion. At some point in time t_1 the salt shield is vanished and the release of nuclides into the groundwater starts. While the repository continues to rise, it is affected further by subrosion. For subsequent stages the flow paths are indicated by dotted lines starting from the repository and ending in the biosphere after respective times T_1, T_2 and T_3. The concept of a gradually decreasing thickness of the overburden has been implemented in the METROPOL model, and nuclide residence times have been calculated by particle tracking.

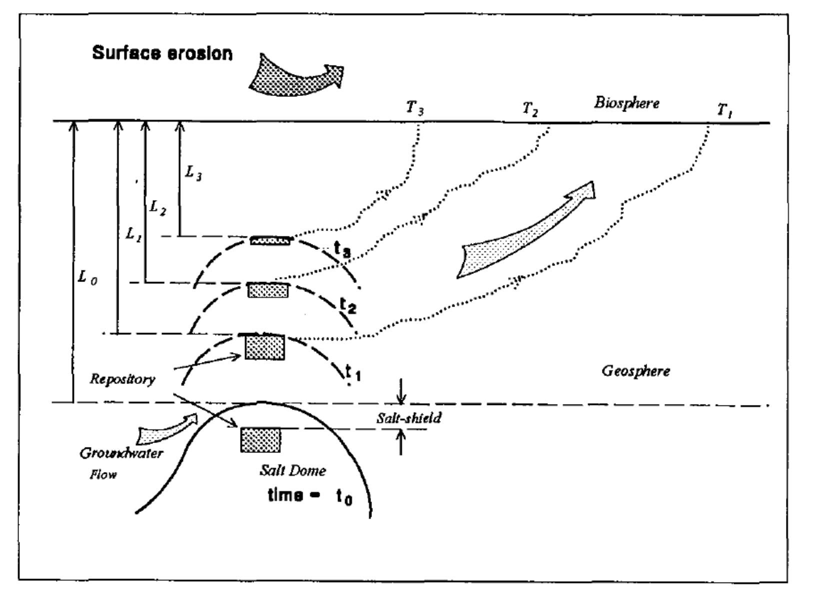


Figure 4-28 Schematization of salt rise and subrosion

In the present evaluation, the geosphere characteristics are treated summarily, mainly in Chapter 5, also taking into account that in OPERA WP4, "Geology and geohydrology", all relevant geological and geohydrological features of the geosphere at present and their expected future evolution(s) are being investigated (Verhoef, 2011b; p.17).

4.8. Biosphere

In all safety assessments of geological waste repositories it is assumed that the possible release of nuclides from the waste repository into the biosphere occurs in a far future, thousands of years from now. This implies that assumptions about the characteristics of the then relevant biosphere are rather speculative. Another aspect of the biosphere models is the fact that it is virtually impossible to determine the diet of future human beings. The standard approach followed in safety assessments is assuming that the future human beings will have the same diet as the present ones. This is consistent with the approach followed that we will protect future human beings in the same way as the present ones.

If the release comes from contaminated groundwater different exposure pathways have been considered:

- Uptake of drinking water
- Ingestion of fresh water fish from ponds
- Ingestion of plants irrigated with contaminated water
- Ingestion of milk and meat from cattle whose feed has been irrigated with contaminated water

The contamination pathways are elucidated in Figure 4-29 (Prij, 1993; p. 6.56).

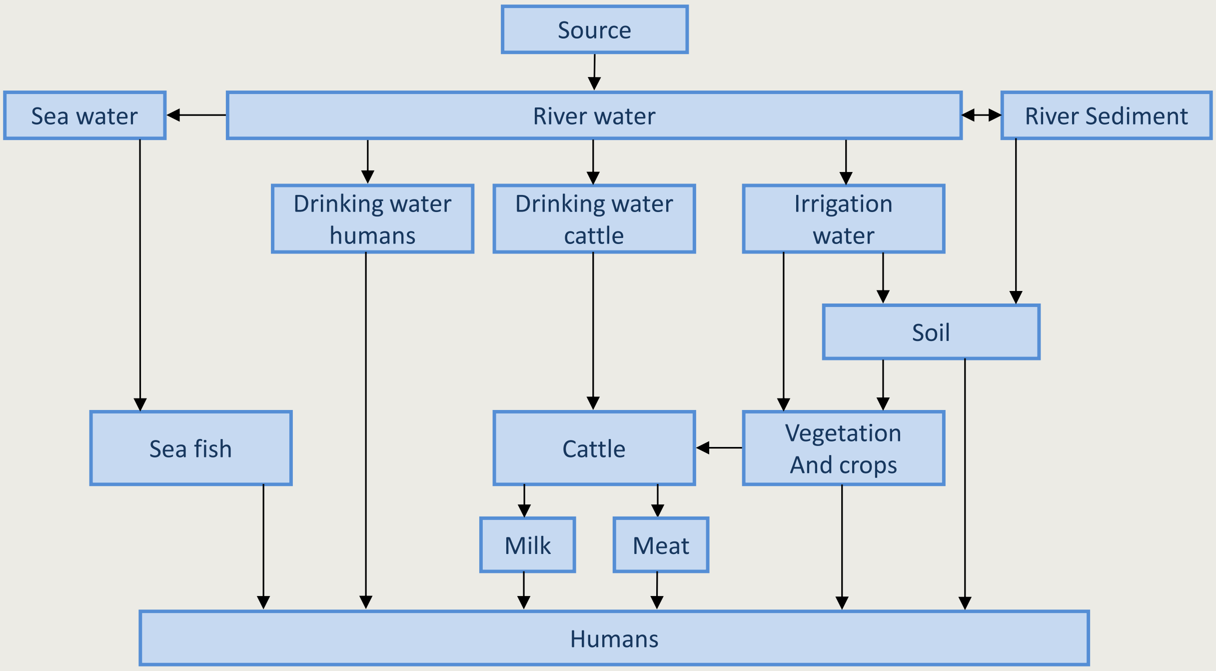


Figure 4-29 Contamination pathways for the biosphere.

If the release does not occur by means of groundwater – such as in the Diapirism scenario (Prij, 1987; pp. 162-167) - the exposure is supposed to occur in a polar desert by direct radiation and inhalation of contaminated dust.

The biosphere characteristics themselves are independent of the “Source” as indicated in the figure above, and therefore the type of host rock from which any radionuclides will be released. In the present evaluation, the biosphere characteristics are therefore treated summarily.

4.9. Evaluation

In this chapter the important components of the disposal system have been described. The following sections summarize the main findings of the evaluation of the different aspects of the disposal system.

4.9.1. Facility designs

For the disposal of all types of radioactive waste generic mine designs have been developed in the past research programmes in The Netherlands:

- Deep vertical boreholes, drilled from the surface for the disposal of heat-generating HLW.
- A conventional mine with galleries and vertical boreholes drilled in the gallery floor for the disposal of heat-generating HLW and chambers for the remaining MLW and LLW.
- To be able to retrieve the waste the mine concept has been revised. Instead of long vertical boreholes a concept has been considered in which short horizontal boreholes are drilled from the walls of the galleries. Additionally the TORAD-B design comprises a steel liner for enhanced stability and to facilitate the retrieval of emplaced waste canisters.

The conceptual facility designs considered in the Dutch research programmes until now are of a generic nature in three generic salt formations: a deep salt dome, a shallow salt dome and a salt layer.

Until now no design considerations have been made in the Netherlands for the possibility of retrieving the MLW and LLW.

In Germany and the US facility designs have been adapted to specific salt formations. They made use of the same mining technique and showed that it is relatively easy to account for the internal structure of the salt formation.

In all facilities the engineered structures together form a multi barrier system consisting of:

- waste matrix
- waste container
- backfill around the waste containers
- seals of boreholes and caverns
- backfill in the galleries
- dams in the galleries
- seals of the shafts

The characteristics of the radioactive waste intended for disposal have changed since the CORA programme, implying that the previously considered facility concepts for disposal in salt are presently not adapted to receive the newly considered waste types such as spent fuel from research reactors and depleted uranium.

The main recommendation to proceed further with the development of the Salt Safety Case in the Netherlands is to outline a final disposal facility in rock salt, taking into account the most recent waste characteristics and an up-to-date safety concept including the definition of safety functions for the disposal of radioactive waste in rock salt, and the possibility to retrieve the waste.

4.9.2. Waste characteristics

The waste categories indicated in the previous sections include a variety of radionuclides which together define a source term of the radiological safety assessment calculations, to be performed within OPERA's WP7 *Scenario development and performance assessment*.

Compared to the radionuclide inventories considered in the previous studies performed in VEOS, PROSA, CORA, and PAMINA, the following considerations apply:

- Unlike in the past, the spent fuel from research reactors, including the future replacement of the HFR (PALLAS) is part of the radioactive waste inventory. The nuclide inventory of the spent fuel elements represents a considerable part of the total nuclide inventory (Hart, 2014b; Table 6-2).
- Unlike in the past, depleted uranium (DU) currently adds to the radioactive waste inventory, foreseen to be disposed in a future geological disposal facility. Although DU represents, in radiological terms, a relatively small fraction of the total inventory (Hart, 2014b; Table 6-2), it comprises a significant volume (Meeussen, 2014; Section 3.2).
- The distinction of the different waste types KSA, HAVA, MAVA, and LAVA and their definitions , as practiced in VEOS and PROSA, currently no longer apply.
- In the previous studies VEOS and PROSA consideration was given to the so-called Waste Strategies B and C, which included radioactive waste from NPPs that in retrospect have not been built, and therefore the assumed radionuclide inventories from NPPs were considerably larger than is foreseen at present.
- In the CORA analyses and in the PAMINA probabilistic safety assessment only a single waste fraction, i.e. vitrified HLW, was considered, and only 300 HLW canisters were taken into account, resulting in too low an inventory compared to the present expected inventory in OPERA.

As a conclusion it can be stated that previously performed safety assessments for salt-based repositories, in VEOS, PROSA, CORA, and PAMINA, applied waste characteristics and radionuclide inventories which differ considerably from the presently foreseen inventories. As a consequence the quantitative results from the safety assessments performed in the past are not representative for a future geological disposal facility in rock salt under the present insights.

It is therefore recommended to update previously performed safety assessments, taking into account the presently foreseen nuclide inventories and waste fractions, and for a disposal concept that is adapted to these waste types and amounts.

4.9.3. Salt formations

First of all it is important to consider all the extensive studies which have been done with respect to disposal of nuclear waste in rock salt during the 1980’s and 1990’s. Part of the work done in those studies is still relevant and up-to-date, other parts will have to be revised using the current state-of-art.

Knowledge on salt domes is steadily increasing, due to their use for salt extraction and possible gas storage. With respect to gaps in knowledge on salt-dome properties, attention should be given to:

- Detailed 3D seismic mapping of the shape and internal structure of salt domes;
- Acquiring borehole data (logs and cores) from salt domes;
- Improving knowledge on connectivity between aquifers and salt domes;
- Assessing inhomogeneities within the salt, including encapsulated liquids and gasses;
- Understanding how inhomogeneities (stringers) will react to deformation;
- Determining the structural position (bedded salt, pillow salt, salt dome, salt wedge) and shape of the salt structure (elongated, circular, overhangs etc.);
- Unravelling the structural development during geologic history.

4.9.4. Safety-relevant thermal, hydraulic, mechanical, and chemical processes

There is an abundance of data available about thermal, hydraulic, mechanical and chemical properties of rock salt, much of which has been derived from exploration of the Gorleben and Asse sites in Germany, and of the Waste isolation Pilot Plant (WIPP) in the U.S. Extensive summaries of earlier studies, performed in the last decades of the 20th century and encompassing several thousand pages, have been provided by BGR, i.e. the “Salzmechanik” volumes (Fahland, 2013). A recent and comprehensive state of the art of discussion has been presented and discussed during the International Conferences TIMODAZ-THERESA in Luxembourg (Li, 2012), SaltMech6 in Hannover (Schulze, 2007), and SaltMech7 in Paris (Bérest, 2012). The influence of disturbed rock zones on performance assessments has been discussed in (Davies, 2005). Several recent summary reports discussed the different aspects and consequences of the presence and inflow of brine into a repository, e.g. (Caporuscio, 2013; Kuhlman, 2014).

In Germany, the VSG identified open issues ((Wolf, 2012b)), and, when compiling essential parameter values for THM calculations in the project VIRTUS (Wieczorek, 2013), uncertainties in these values and open questions have been identified. Also in international workshops and conferences dealing with THMC processes (e.g. THERESA – Jing, 2010; Li, 2012), uncertainties and open questions have been established. Additionally, in the US issues have recently been identified to put forward the US HLW disposal programme (Hansen 2011; Section 5.2).

The understanding of safety relevant processes as summarized in the previous sections is an essential prerequisite in the assessment of the post-closure safety, and thus forms the scientific basis of a Safety Case in rock salt. It is not surprising that from the above mentioned international studies a large list of scientific topics can be extracted that are marked as 'open issues' or 'knowledge gaps'. It is, however, not possible within the scope of this report to scan the current state of art of the list of topics in sufficient detail to judge which of the noted topics is of relevance for the Dutch Salt Safety Case, for the following reasons:

- Uncertainty studies that allow to make a well-supported judgment of the relevance of each of the given aspects for the long-term safety are scarce, i.e. it is difficult to judge if further understanding of these issues will add to confidence in safety;
- Not all topics identified may be of relevance for the Dutch disposal concept, in particular the expected thermal load of heat-generating HLW is restricted due to the long interim storage;
- Due to the rather small inventory of the Dutch nuclear programme, safety might be demonstrated with a more conservative, simplified performance assessment modelling approach than in case of much larger inventories, e.g. in the German programme;
- Some of the identified processes might be of no relevance in the current state of the disposal programme in the Netherlands which is presently focusing on the long-term safety. They can however become relevant when assessing scenarios related to operational safety or when investigating geotechnical aspects related to retrievability at a later stage.

On the basis of the information collected in the previous sections, aspects have been identified internationally as relevant to carry along with respect to enhancing the understanding of salt THMC topics. Since not all the identified aspects can be treated in OPERA follow-up programmes, a discrimination has been made between topics that are judged as relevant to carry forward in the post-OPERA phase, and topics that are judged as interesting to address but not directly relevant for the long-term, post-closure safety of a salt-based repository in the Netherlands.

In the following tables, topics which are considered relevant for the long-term, post-closure safety are indicated with solid bullets These topics are judged to deserve carrying forward in OPERA follow-up programmes.

Topics which are indicated by the open bullets are judged of interest to reduce remaining uncertainties, but have less relevance to the long-term post-closure safety.

Screening arguments for the distinction between these two categories are based on (1) the extended surface storage period in the Netherlands, (2) the resulting limited heat output from disposed heat-generating waste, (3) the mild temperature effects on the surrounding host rock, (4) the delayed decision for siting a repository, and (5) expert judgement (authors familiar with the Ducht Safety Case).

The issues are categorized as follows:

- ❖ General aspects
- ❖ Material parameters
- ❖ Evolution of the DRZ
- ❖ Crushed (granular) salt
- ❖ Chemical and physical aspects

General aspects		
Aspect	**Source**	**Screening argument**
● Additional research on the processes related to the interface of repository materials (buffer, backfill, canister) and the host rock	Jing, 2010; Section II.3	See Section 4.6
○ Re-evaluation of the strong temperature dependence of mechanical deformation for more precise modelling using advanced constitutive models	Hansen, 2011; Section 2.4	Limited temperature effect
○ Consideration and enhanced modelling of the heterogeneity of rock materials and anisotropy in stress fields may play a role in THM modelling	Li, 2012; Section 2.5	Delayed siting
○ Establishing the relevance of scale effects, i.e. the transformation of results in the laboratory to in-situ conditions should be verified	Li, 2012; Section 2.5	Expert Judgement (Section 5.4.4)

Material parameters		
Aspect	**Source**	**Screening argument**
○ Extended site characterization - especially if a site has already been selected - is necessary to get more detailed information about fluid inclusions, fissures and faults, morphology, etc.	Hansen, 2011; Section 4.7.1; Wolf, 2012; Section 89.11	Delayed siting
○ Additional information concerning the influence of elevated temperatures and plastic deformation on barrier (host rock) characteristics	Wolf, 2012; e.g. Section 87	Limited temperature effect
○ Knowledge enhancement on the effects of radiolysis on salt	Hansen, 2011; Table 4; Wolf, 2012; Section 80	Expert Judgement (Very limited due to shielding)

Evolution of the DRZ		
Aspect	**Source**	**Screening argument**
● Enhancement of the understanding of the DRZ influence on seal systems	Hansen, 2011; Table 4; Wolf, 2012; Sections 39, 50-52	See Section 4.6.1
● The implementation of constitutive and coupled models on the behaviour of DRZ into numerical codes enabling reliable extrapolations to long-term in-situ conditions	Rothfuchs; 2009; Section 3.3	See Section 4.6.1
○ Knowledge enhancement of the correlation and coupling of permeability with volumetric strain (i.e. porosity) to obtain more reliable porosity-permeability relations in the DRZ	Jing, 2010; Section 2.5.2; Hansen, 2011; Section 4.2	Long-term post-closure safety only, assumed healed DRZ

Evolution of the DRZ		
Aspect	**Source**	**Screening argument**
○ Extension of the experimental database of the time-dependent evolution, recompaction and self-sealing of the DRZ for sound calibration and benchmarking of theoretical models	Jing, 2010; p.IX	Long-term post-closure safety only, assumed healed DRZ
○ Validation and improvement of the constitutive models for the behavior of the EDZ	Jing, 2010; p.IX; Wolf, 2012; Sections 54, 82; Hansen, 2011; Table 4	Long-term post-closure safety only, assumed healed DRZ
○ Experimental confirmation of the long-term compaction at moderate stresses by measurement of the permeability and porosity The testing duration has to be much longer than several months, in order to evaluate the empirical equations for the extrapolation of the compaction behavior	Rothfuchs; 2009; Section 3.3; Hart, 2009; experiments performed at the Utrecht University	Long-term post-closure safety only, assumed healed DRZ
○ Development, validation (including validation of their suitability for in-situ application), and calibration of generally agreed constitutive models for the re-compaction and healing of dilated rock salt in the DRZ	Rothfuchs; 2009; Section 3.3; Hansen, 2011; Table 4	Long-term post-closure safety only, assumed healed DRZ
○ Improvement of the description and modeling of cracking processes and the reduction of conservatism	Wolf, 2012; Section 82	Long-term post-closure safety only, assumed healed DRZ
○ Understanding of the healing/sealing phase: complex interactions between convergence, seal structures, the contact zone and the DRZ	Wolf, 2012; Section 82	Long-term post-closure safety only, assumed healed DRZ
○ Development of methods for the technical sealing of the DRZ, e.g., by injection methods with long-term stable sealing materials	Wolf, 2012; Section 82	Long-term post-closure safety only, assumed healed DRZ

Crushed (granular) salt		
Aspect	**Source**	**Screening argument**
● Enhancement of the understanding of physical processes (a.o. convergence, channelling) which control the efficiency of granular salt compaction especially with respect to humidity effects	Hansen, 2011; p.68; Wolf, 2012; Sections 36, 45, 47, 56; Sneyers, 2008; p. 143	See Section 4.6.3
● Enhancement of the understanding of the porosity development at the end of the compaction process, i.e. for low porosity values, and the correlated permeability. For low porosity values the uncertainty of the porosity value is of the same magnitude as the porosity itself. Micro-structural investigations are in development to help clarify these problems	Wolf, 2012; Section 47; Zhang, 2006; Section 6); Kröhn, 2012	See Section 4.6.3

Crushed (granular) salt		
Aspect	**Source**	**Screening argument**
● Development of generally agreed constitutive models for compaction in granular salt allowing reliable extrapolations to in-situ conditions	Sneyers, 2008; p. 143	See Section 4.6.4
○ Enhancement of understanding of consolidation of backfill materials, more specifically thermal conductivity as a function of porosity, temperature-dependent constitutive modelling	Hansen, 2011; Table 4	Limited temperature effect

Chemical and physical aspects		
Aspect	**Source**	**Screening argument**
● Solubility of radionuclides	Hansen, 2011; Table4	See Section 4.6.5
● Development of surface complexation models to describe sorption and desorption under saline conditions	Wolf, 2012; Section 105	See Section 4.6.5
● Study on the contribution of colloids to the total concentration of radionuclides in solutions, especially the actinides	Wolf, 2012; Section 106	See Section 4.6.5
● Study on the transferability of laboratory experiments on colloidal systems to large-scale heterogeneous systems	Wolf, 2012; Section 106	See Section 4.6.5
● Inventory of complexing agents in waste with negligible heat generation and its influence on the mobilization of radionuclides	Wolf, 2012; Section 106	See Section 4.6.5
○ Reducing the uncertainties related to gas production and transport	Larue, 2013; p.179 Hansen, 2011; p.63	See Section 4.6.5.5
○ Better determination of the sorption behavior of some radionuclides or elements and their anionic and cationic compounds, such as iodine	Wolf, 2012; Section 105	Expert judgement (might reduce conservatism)
○ Further development of surface complex models to describe sorption in natural systems (assessment of K_d values and their bandwidths)	Wolf, 2012; Section 105	Expert judgement (might reduce conservatism)
○ Study on the influence of temperature on complex formation under saline conditions	Wolf, 2012; Section 106	Expert judgement (siting aspect)
○ Study on the influence of temperature on sorption and desorption under saline conditions	Wolf, 2012; Section 105	Expert judgement (might reduce conservatism)
○ Determination of the degradation of hydrocarbons to complexing agents under site-specific conditions (salinity, temperature, H_2 pressure)	Wolf, 2012; Section 106	Expert judgement (might reduce conservatism)
○ Enhancement of the description of diffusion processes in backfilled galleries at small porosities	Wolf, 2012; Section 111	Expert judgement (very slow process)

Chemical and physical aspects		
Aspect	**Source**	**Screening argument**
○ Assessment of coupled thermodynamic processes for salt, e.g. thermal osmosis, chemical osmosis, thermo-diffusion (Soret effect), influence of a concentration gradient on the heat transport (Dufour effect)	Wolf, 2012; Section 113	Expert judgement (might reduce conservatism)

On the basis of the collected information, aspects have been identified internationally as relevant to carry along with respect to enhancing the understanding of salt THMC topics. Since not all the identified aspects identified in this section can be treated in OPERA follow-up programmes, a discrimination has been made between topics that are judged as relevant to carry forward in the post-OPERA phase, and topics that are judged as interesting to address but not directly relevant for the long-term, post-closure safety of a salt-based repository in the Netherlands.

On the basis of the topics discussed in Section 4.6, and the evaluation summarized in the tables above, a number of topics have been identified as relevant for the long-term, post-closure safety, and for carrying forward in the post-OPERA phase:

- Influence of Disturbed Rock Zone (DRZ)
- Compaction behaviour of crushed (granular) salt
- (T)HMC effects related to the dissolution of rock salt
- Corrosion of waste container and waste matrix
- Corrosion of cementitious barriers
- Solubility of radionuclides

Influence of Disturbed Rock Zone (DRZ)

Enhanced understanding of the evolution of the DRZ adds to the reduction of the uncertainties related to post-closure safety. However, the evolution of the DRZ is considered not to affect the post-closure safety to a major extent.

On the other hand, the DRZ may affect the isolation properties of seal systems. In investigating the evolution of the DRZ and its impact on the post-closure safety it is therefore recommended to focus on the enhancement of the understanding of the influence of the DRZ on seal systems, including the complex interactions between convergence, seal structures, the contact zone and the DRZ.

In addition, the implementation of constitutive and coupled models describing the behaviour of the DRZ into numerical codes enabling reliable extrapolations to long-term in-situ conditions is another important aspect to consider in subsequent research programmes.

Compaction behaviour of crushed (granular) salt

In a salt-based repository any void volume in emplacement areas has to be backfilled with crushed salt which will be naturally compacted by creep induced convergence. During the compaction process , the porosity and permeability of the crushed salt decreases until, in the long run, it has similar barrier properties as pristine rock salt. In the post-closure safety assessment the compaction behaviour of crushed salt plays a dominant role as it affects the timing of complete isolation of the emplaced waste from the environment. In the case of an inundated repository the compaction behaviour of crushed salt also determines the displacement of contaminated brine, if any. In the post-closure safety assessment evidence must be provided that sealing by the compacted backfill material is fully developed by the time the performance of the engineered barriers can no longer be demonstrated.

Taking these considerations into account it is recommended to proceed further with the following topics related to crushed salt:

- Enhancement of the understanding of physical processes, amongst others convergence and any effects of the presence of brine, which control the efficiency of granular salt compaction especially with respect to humidity effects;
- Enhancement of the understanding of the porosity development during the compaction process, and the correlated permeability. In this repect the timing of reaching sufficiently low porosities to accomplish attain a complete isolation of the disposed waste is relevant for the post-closure safety.

At a later stage, further development and verification may be considered of coupled constitutive models for compaction in granular salt allowing reliable extrapolations to in-situ conditions.

HMC effects related to the dissolution of rock salt

As discussed in Section 4.6.5understanding of the dissolution behaviour of a saline system is relevant when assessing the effects of intrusion of brine or to the displacement of brine already present in the host rock. Mineral dissolution and precipitation can affect the geo-mechanical properties of the host rock, e.g. by changes of the mineral volumes and are therefore of relevance for the assessment of the geo-mechanical integrity of the barrier function of the host rock. Mineral dissolution and precipitation are kinetic processes, potentially undergoing several conversions until equilibrium with brine is reached, and experimental support for these processes is therefore essential. When assessing a generic system in rock salt (NaCl), as it was performed for previous Dutch disposal concepts summarized in Section 4.2, it is judged that currently sufficient knowledge is present to assess chemical dissolution and precipitation processes.

In a later stage when an updated Safety Case will focus more on the technical feasibility of the disposal concept, the layout and extend of the concept has to be matched in some way to the extend and composition of existing rock salt formations available in the Netherlands. In this case, the assessment of solid-solution interactions getting more complex due to the expected presence of mixed mineral compositions in the vicinity of the facility. The general guideline is to dispose waste in NaCl, and to avoid the presence of anhydrite and carnalite as much as possible, maintaining a minimum barrier distance to the disposed waste. However, from the current generic study performed on the Dutch disposal concepts it cannot be judged whether the experimental support provided mainly by the German programme will be sufficient, or whether more complex dissolution reactions need to be assessed that may be insufficiently covered by experimental data.

Assuming that the next phase of the OPERA Safety Case for the geological disposal in rock salt will focus on the long-term, post-closure safety - in accordance with the main objective of the OPERA programme - an assessment of a *generic* disposal concept in a *generic* NaCl host rock might appear appropriate. In that case, dissolution processes can be considered to be sufficiently well understood.

However, focussing on the longer time scale, and the need to provide evidence on the technical feasibility in subsequent steps of the Dutch Salt Safety Case, recommendations are to perform the following extensive actions:

1. To inventory which mineral dissolution processes can be of principal interest considering the size of a Dutch disposal concept in comparison with size and (chemical) composition of available rock salt host formations in the Netherlands;
2. To judge, based on the previous inventory, whether the range of expected mineral interactions is sufficiently well covered by current experiments;
3. To evaluate what site-specific features of a disposal concept in rock salt can reasonably well be addressed in the next stage of the Dutch programme, and what geological features need to be addressed in order to perform a Safety Case on the feasibility in the stage thereafter. Such an evaluation might also be used to define exclusion criteria for siting locations.

It is expected that in the near future better computational options exist to model the complex hydrological, (geo)mechanical and (geo)chemical processes[21] in salt formations in their natural hydrogeological setting in 3D. Such modelling is expected to provide more realistic evidence for the barrier function of the host rock, eventually leading to the development of new safety and performance indicators (see e.g. Becker, 2009; p.65*ff*).

It is therefore recommended to keep track on these computational developments by e.g. participation in international working groups (NEA Salt Club), and visiting technical meetings on this matter (e.g. ABC-meeting, US-German Workshops).

With respect to the constitutional modelling of chemical processes, the activities of the German THEREDA project (Moog, 2015) are of interest.

Corrosion of waste container and waste matrix

Much is known on the corrosion of iron in highly saline environment. For geochemical modelling of Calcium Silicate Hydrate (CSH) phases in cementitious material, several approaches exist, but there is some need to understand the effect of the dissolution behaviour of cementitious material on relevant features determining the solubility of radionuclides, e.g. the pH. It is recommended to address this aspect further in a future programme, in a two-step approach:

1. Analyse CSH corrosion processes in saturated brine, its influence on radionuclide solubility and mobility, and existing uncertainties by geochemical process models, based on existing literature on CSH in saline environments;
2. Consider experimental evaluation of critical parameters and processes based on the outcome of the modelling study.

Since potential corrosion and degradation reactions depend on the composition of the brine, the studies should carefully consider source and genesis of the brine for the Dutch situation (see also previous paragraph).

Corrosion of cementitious barriers

As discussed in Section 4.6.5, degradation and dissolution processes can impair the proper geo-mechanical and hydraulic function of technical barrier of a disposal design (see e.g. Müller-Hoppe, 2012). However, in the Dutch disposal concepts summarized in Section 4.2 no cementitious barriers like shaft seals or dams has been defined (yet). Consequently, no information is available on the relevance of these barriers, and no well underpinned statements apply regarding the impact of corrosion processes of cementitious barriers in a salt-based repository. Instead, in line with the evaluation provided in Section 4.9.1, it is

[21] Thermal effects may be considered, too, but seems of less relevance taking into account the presently adopted extended surface storage period of heat generating waste.

recommended first to clarify constructional aspects of these barriers and their role in the safety concept.

From the point of view of process understanding of corrosion of cementitious barriers in a salt environment, it is recommended to consider the ease of assessment as an important design criterion, thus focusing as much as practically eligible on the current knowledge of the involved materials as well as the relevant processes and their remaining uncertainties.

Solubility of radionuclides

The mobility of radionuclides in the salt host rock – once a waste container has failed and the waste matrix dissolved – depends mainly on the solubility of the radionuclides in brine under the specific local chemical conditions, while sorption of radionuclides to the host rock is assumed to be of lesser relevance.

Relevant sorption of nuclides can occur on the surfaces of the EBS, the waste matrix, and on their corrosion/degradation products, but from current knowledge it is difficult to judge whether these processes relevantly contribute to safety in a salt environment. Furthermore, colloids formed by degradation processes in the waste might also increase the mobility of radionuclides.

In first instance, it can be assumed that the transport of radionuclides in the geosphere and biosphere is sufficiently covered by the related work packages of the current OPERA programme. However, two aspects should kept in mind:

- due to minor sorption on the host rock, other radionuclides can become relevant for the long-term safety than in the case of a disposal concept in Boom Clay, where sorption of nuclides is often significant;
- the contribution of the geosphere to the overall safety might be greater in case of a disposal concept in rock salt than for Boom Clay, eventually raising the need to develop more detailed geosphere transport modelling approaches than currently developed and applied in OPERA.

It is recommended to focus on these aspects in subsequent geological disposal research programmes.

With regards to the modelling of radionuclide dissolution and transport in a saline system the elaboration of non-equilibrium reaction kinetics, still being in an initial state due to the lack of many reaction parameter values, may be taken forward.

4.9.5. Overlying sediments and biosphere

Considering the sediments overlying the salt formations, and the biosphere, the following observations apply:

- The analyses have shown that, apart from human intrusion, a release of nuclides from the salt formation might conceivably occur not earlier than 10,000 up to a million years after disposal. The final hurdle to the biosphere is formed by the overlying sediment. Given these time scales it must be realized that the details of the then existing overlying sediment and the biosphere cannot be determined exactly[22].
- It is meaningful to give extra attention to the overlying sediments with respect to the construction of the GDF. This aspect should be worked out further.

[22] This aspect is currently treated in OPERA WP4 *Geology and Geohydrology*

- One even can imagine that the properties of the overlying sediments play a role in the scenarios for human intrusion, both for the consequences of future intrusion as well as in preventing human intrusion. This should be elaborated in more detail.
- The approach followed in the studies performed sofar applying a generic model for the overlying sediments and the biosphere. The uncertainties in the models have been taken into account with pdf’s of the relevant parameters.
- With respect to the biosphere it has been assumed in the several previously performed safety assessments that the biosphere and the diet of future human beings will be the same as present. This is consistent with the approach followed that we will protect human beings in the same way as the present ones.
- The biosphere characteristics themselves are independent of the radionuclide “source” originating from the subsurface, and therefore the type of host rock from which any radionuclides will be released.

5. Safety Assessment

5.1. Objective and Scope

Safety assessment encompasses evaluating the performance of a disposal system and quantifying its potential radiological impact on human health and the environment (IAEA, 2012; p. 5). Safety assessment is a major component of the Safety Case for a disposal facility and should take account of the potential radiological impacts of the facility, both in operation and after closure. Radiological impacts may arise from gradual processes after closure that may cause the facility and its components (e.g. natural and engineered barriers) to degrade, and from discrete disturbing events that could affect the isolation of the waste (e.g. earthquakes, faulting and inadvertent human intrusion). Safety assessment should demonstrate whether the disposal facility complies with applicable regulatory requirements.

The Safety Case component “Safety Assessment” obtains the input, information and knowledge in the first place from the “System Description”. In subsequent stages of a disposal programme input and experiences from iterations and design optimizations will also be relevant.

The present methodologies for the execution of performance assessments in geological disposal are for a significant part based on the ideas that have been developed in the EU project PAGIS[23] (Storck, 1988), a collaborative research programme of the European Union.

In the present chapter the safety assessment methodology used in the previous Dutch studies VEOS, PROSA, and CORA will be summarized and compared with more recent views and developments on safety assessment methodologies.

5.2. VEOS safety assessment

The VEOS safety assessment was conducted as part of the Dutch programme OPLA, and comprised dose calculations for 21 different disposal concepts (Prij, 1989). These concepts were based on three generic formation types (salt dome with 230 m thick overburden; salt pillow with 800 m thick overburden; bedded salt at a depth 1200 m), three disposal techniques (mine with boreholes for the high activity waste (HAW) and mined chambers for the remaining waste; deep boreholes drilled from the surface for the HAW and dry caverns for the remaining waste; deep boreholes drilled from the surface for the HAW and wet caverns for the remaining waste), and three waste strategies (see also Section 4.3.1):

- Strategy A: waste from 15 GWa nuclear energy with 50 year of interim storage;
- Strategy B: waste from 105 GWa nuclear energy with 50 year of interim storage;
- Strategy C: waste from 105 GWa nuclear energy with 10 year of interim storage.

The VEOS safety assessment was essentially a detailed and deterministic method, aiming to provide the necessary knowledge and insights what processes would be relevant to the dose rate in the biosphere. As part of the VEOS programme, the computer program TASTE (Three dimensional Analysis of Salt TEmperatures) was developed, to estimate temperature distributions in the salt host rock due to the heat input from heat generating radioactive waste.

An important aspect of the VEOS safety assessment was the identification of scenarios and the most important processes affecting the long term safety of the various disposal concepts. The following sections provide a summary of these aspects.

[23] Performance Assessment of Geological Isolation Systems

The VEOS safety assessment was performed taking into account a number of subsequent steps as elucidated in Figure 5-1 (Prij, 1989; p.44):

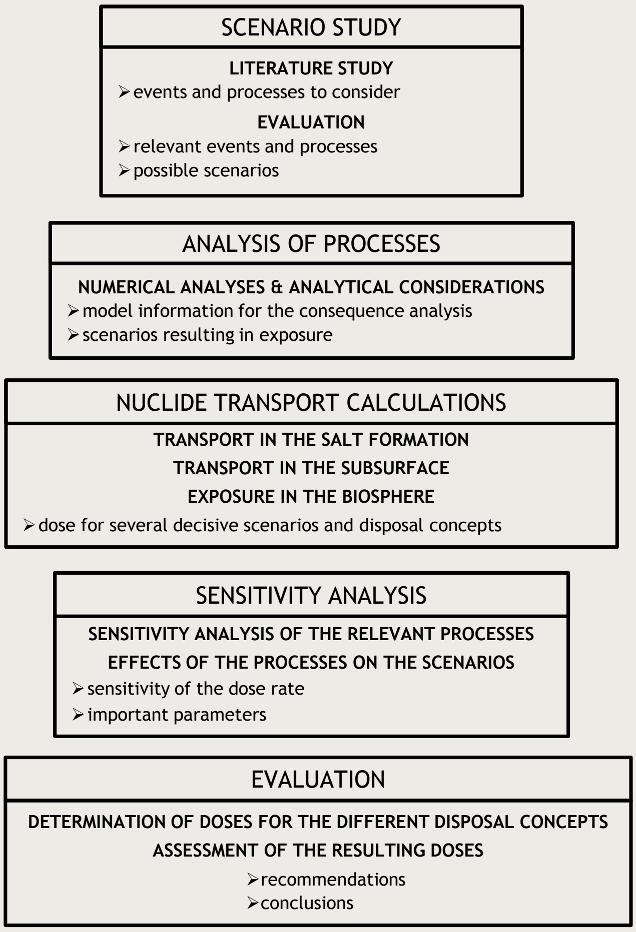


Figure 5-1 Overview of VEOS safety assessment methodology.

5.2.1. Definition of scenarios

The aim of the VEOS scenario study was to identify possible combinations of features, events and processes (FEPs) leading to (1) a **release** of nuclides from the waste matrix, (2) the **transport** of these nuclides to the biosphere and ultimately (3) the **exposure** of human beings.

For the identification of scenarios in the VEOS study the following method has been applied:

a) Literature study. The VEOS study produced a list of scenarios used in other safety assessments and an overview of events and processes which were thought to be relevant. This overview was taken as a starting point for further study as it was more comprehensive.

b) Analysis to determine which features, events and processes could be considered to be relevant for the disposal concepts in the salt formations considered.

c) Further analysis leading to possible scenarios. According to the system used in the European study PAGIS the scenarios are distinguished in three classes with descending probability (Cadelli, 1984):

   - Normal evolution; here it is assumed that the repository system functions according to the specifications.

- Altered evolution; here events are considered which are not likely but cannot be excluded.
- Disruptive evolution (human intrusion); here it is taken into account that the carefully designed system of barriers is destroyed.

The study resulted in 11 scenarios summarized in Table 5-1.

Table 5-1 Scenarios considered in VEOS for the 3 formation types (FT) and disposal concepts (DT) considered.

Scenario	*Evolution*	*FT**	*DT*
Subrosion and diapirism	normal	P, D	all
Brine migration	normal	all	all
Thermo-mechanical crack	normal	all	all
Flooding of the GDF	altered	P, D	mine
Undetected large brine pocket	altered	all	all
Casing leakage	altered	P, B	deep bore hole
Diapirism into the biosphere	altered	P, D	all
Reconnaissance drilling	human intrusion	all	all
Salt production (wet)	human intrusion	P, D	all
Salt production (dry)	human intrusion	P, D	all
Leakage of a production cavern	human intrusion	P, D	all

* P: salt pillow; D: salt diapir; B: bedded salt

In the following short description of these scenarios it is indicated how the barriers around the waste can be destroyed and groundwater (or humans) can come in contact with the waste.

Table 5-2 Description of scenarios considered in VEOS

Subrosion and diapirism	Slow rise of the geological disposal facility rises slowly due to ongoing diapirism of the formation. Simultaneously salt dissolution around the GDF by subrosion. Ultimately this might lead to contact of the waste with groundwater and after transport in the groundwater eventually to exposure in the biosphere.
Brine migration	Driven by the temperature gradient in the salt surrounding heat generating waste brine droplets confined in the rock salt might migrate to the waste, corrode the waste container, and dissolve nuclides form the waste matrix.
Thermo-mechanical crack	Due to the thermo-mechanical stresses cracks might occur in the salt formation. Through these cracks groundwater could flow into the mine and eventually come in contact with the waste.
Flooding of the GDF	After closure of the GDF groundwater flows into the GDF and comes in contact with the waste.
Undetected large brine pocket	The GDF is constructed in the near vicinity of a large brine pocket. Due to the mining activities and thermo-mechanical stresses this brine can come in contact with the waste.
Casing leakage (deep borehole only)	A leak occurs from the casing of the borehole and groundwater comes in contact with the waste.
Diapirism into the biosphere	Subrosion occurs so slowly that the salt surrounding the GDF is not dissolved as the salt formation comes at the surface where the remaining salt is eroded.
Reconnaissance drilling	The future knowledge of the presence of the GDF is lost and a reconnaissance drilling will come in contact with the HLW.
Salt production (wet)	The future knowledge of the presence of the GDF is lost and the production of salt by means of a leaching technique will take place exactly in the GDF.

Salt production (dry)	The future knowledge of the presence of the GDF is lost and the production of salt with a production mine will take place exactly in the GDF.
Leakage of a storage cavern	The future knowledge of the presence of the GDF is lost and a storage cavern is made in the near vicinity of the GDF. This cavern is assumed not correctly closed leading to contact with the waste.

5.2.2. Analyses of relevant processes

The processes playing an important role in the scenarios have been analysed in more detail. Based on these analyses it has been determined whether the processes could be intense enough to lead to the release of nuclides form the GDF. The results have been used for the subsequent transport and dose calculations. Based on these analyses the following is concluded (OPLA, 1989; p. 85-86):

a) Diapirism. In the next 1 to 4 million years the maximum internal rise of the salt in a salt pillow or salt dome is about 0,4 mm/a. The presence of heat generating waste is judged not have a significant influence on the rising rate (Prij, 1991; pp. 173-177).

b) Subrosion. It is concluded that the in the coming 1 to 4 million years the maximum rate of subrosion is 0,15 mm/a. At larger depth the subrosion rate is significantly lower.

c) Brine migration. In a salt formation small inclusions of brine cannot be excluded. These brine inclusions can migrate towards a heat source. It was judged that under natural conditions the amount of brine is usually too small to corrode the containers. The scenario Brine migration therefore does not lead to an exposure in the biosphere.

d) Temperatures. Due to heat generation in part of the radiactive waste local temperatures may rise up to about 165°C. The calculated temperatures rises have to taken into account in the further analyses of the temperature dependent process such as: brine migration, stress changes, convergence of cavities and leaching of nuclides.

e) Stresses. Due to the mining activities and any heat generation of the waste the stress state in the salt formation will change. For the analysis of these changes numerical tools have been developed (Prij, 1991; p. 17-18) and tested with experiments (Prij, 1991; pp. 103-117). Some important conclusions from the stress-state analyses were that (1) the thermal stresses are not large enough to cause cracks in the salt formation, (2) due to the thermally accelerated deformation of the rock salt in the vicinity of the waste the waste containers will be effectively confined within several years (experimentally verified with in situ experiments in the Asse mine; Prij, 1991; pp. 93-98), and (3) due to the heat generation the surface above the salt formation will rise maximal 5 cm.

f) Convergence of a cavity. Due to the creep behaviour of rock salt openings will gradually decrease (converge). Convergence is highly temperature dependent and is thoroughly studied analytically and experimentally (Prij, 1991; pp. 103-122). Convergence also is a driving mechanism for the transport of contaminated brine out of the salt formation. Based on the numerical analyses the initial convergence rate of an cavity without any internal pressure is 2 %/a.

g) Groundwater movement. The path and travel time of a particle with groundwater from the salt formation towards the earth surface have been analysed in detail and depends on several factors such as possible changes in sea level, diapirism, subrosion, glaciation, fracture in the overlying sediment and variations in permeability.

h) Radiation damage. The effects of the possible consequences of radiation damage in rock salt has been investigated both in Germany and the Netherlands (e.g. Hartog, 1988). It was established that the effects of radiation are concentrated in a small area of some metres around the HLW (Prij, 1991; pp. 178-190), see also Section 4.6.2.6 of the present report.

5.2.3. Nuclide transport and dose calculations

Nuclide transport and dose calculations have been performed for the scenarios identified in VEOS for which an exposure could be expected. For these analyses the disposal system is supposed to consist of three connected compartments:

a) The salt formation and the GDF. In this compartment there is nuclide release and transport of the contaminants.
b) The overlying sediments through which the contaminants are transported with the groundwater.
c) The biosphere in which the contaminated water is transported and ultimately exposure of humans will occur.

An overview of the models used for the different scenarios is given in Table 5-3. As can be seen for some of the scenarios the analysis of transport and or exposure can be performed with an analytical model. For the more complex ones the following computer models have been used.

a) EMOS (Endlagerbezogene Modellierung von Szenarien) developed by the GSF (Gesellschaft für Strahlenschutz und Umweltforschung) in Braunschweig (Stork, 1990) for PSE (Project Sicherheit Endlager: PSE, 1985);
b) METROPOL (Method for the transport of pollutants) developed by RIVM (Glasbergen, 1987);
c) BIOS developed by NRPB (National Radiation Protection Board) in Harwell UK.

Table 5-3 Models used for the calculation of transport and exposure.

Scenario	*Salt compartment*	*Groundwater*	*Biosphere*
Diapirism and subrosion	Analytical	METROPOL	BIOS
Flooding	EMOS	METROPOL	BIOS
Brine pocket	EMOS	METROPOL	BIOS
Casing leakage	EMOS	METROPOL	BIOS
Diapirism into the biosphere	Analytical	Short cut	Analytical
Reconnaissance drilling	Analytical	Short cut	Analytical
Salt production (wet)	Analytical	Short cut	Analytical
Salt production (dry)	Analytical	Short cut	Analytical
Leakage of cavern	Analytical	METROPOL	BIOS

An example of the numerical results of the VEOS calculations is summarized in Figure 5-2 (OPLA, 1989; p.21) and Table 5-4. The exposure given is calculated for waste strategy B, i.e. all the waste from the two existing NPPs Borssele and Dodewaard as well as new NPPs of 3000 MW and assuming an interim storage period of 50 years. More results are given in the final report of OPLA (OPLA, 1989; p. 87-91).

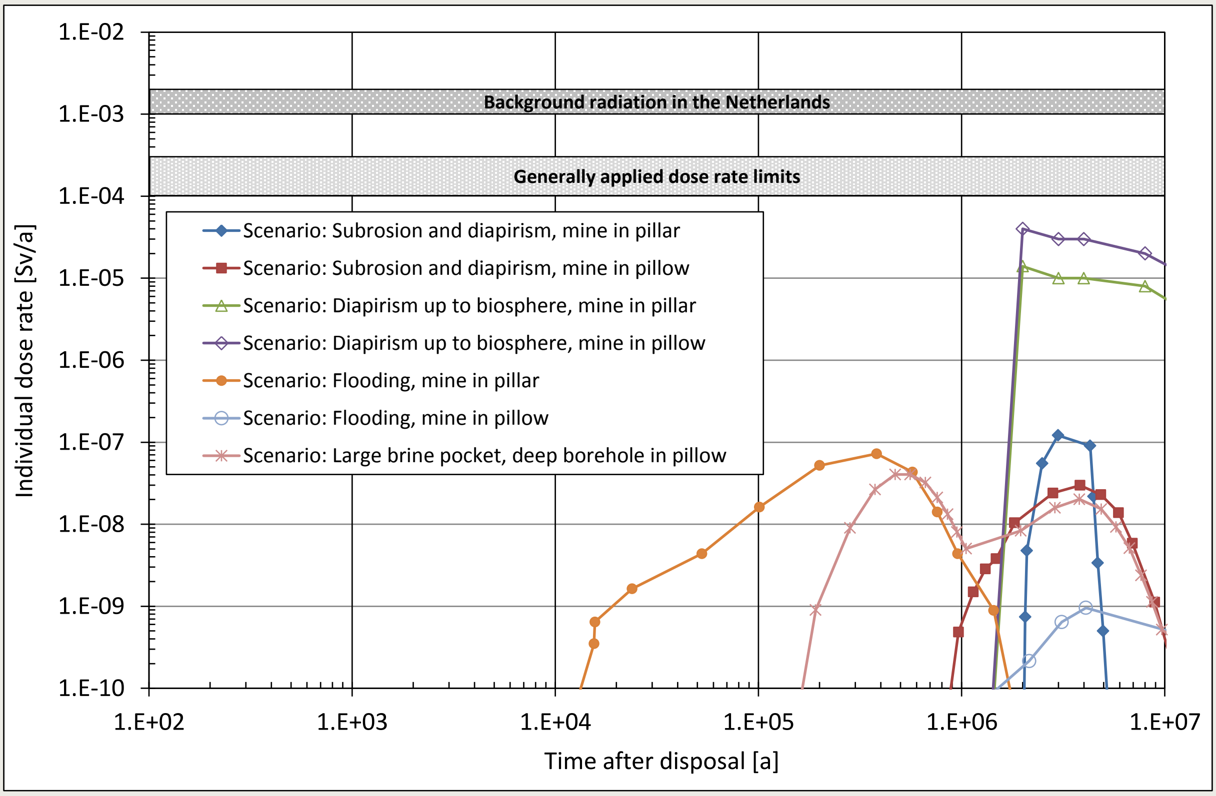


Figure 5-2 Overview of VEOS results for leading scenarios.

Table 5-4 A summary of results of the VEOS dose calculations - waste strategy B.

Scenario	*Max exposure [Sv/a]*	*Estimate of probability*	*Time of maximum exposure [a]*
Diapirism and subrosion	$1\ 10^{-7}$	~ 1	$3\ 10^{6}$
Flooding	$7\ 10^{-8}$	< 1	$5\ 10^{5}$
Brine pocket	$3\ 10^{-8}$	< 1	$4\ 10^{5}$
Casing leakage	$1\ 10^{-5}$	< 1	$1\ 10^{4}$
Diapirism into the biosphere	$4\ 10^{-5}$	$< 10^{-5}$	$2\ 10^{6}$
Reconnaissance drilling	90	$2\ 10^{-8}$	250
Salt production (wet)	$8\ 10^{-5}$	<< 1	250
Salt production (dry)	$2\ 10^{-5}$	<< 1	250
Leakage of cavern	$1\ 10^{-5}$	<< 1	$1\ 10^{6}$

From the results of the VEOS safety assessment calculations it appeared that for all waste strategies and considered scenarios, except the human intrusion scenario reconnaissance drilling, the dose rates are significantly below the natural background radiation in the Netherlands of maximal $3\ 10^{-3}$ Sv/a (OPLA, 1989; p.92). For the reconnaissance drilling scenario receiving a high potential dose rate is however extremely unlikely due to the very small probability of occurrence, and the conservative assumptions made for this scenario.

5.2.1. Treatment of uncertainties

In VEOS uncertainties have been treated by performing a large amount of **deterministic** analyses aiming at:

- The determination of the sensitivity of the results of the safety study with respect to parameters used.

- The determination of a bandwidth in the results of the dose calculations
- The determination of the relevant characteristics with respect to improvements in the design.

First of all the sensitivity of several relevant parameters and processes have been determined such as temperature, brine migration, subrosion, diapirism, rock stresses, convergence rates, nuclide migration in the salt formation and groundwater, and exposure in the biosphere (Prij, 1989; pp. 177-185). It was shown that the effects of these parameters is different for the considered exposure pathways. In none of the cases the effect is more complicated than simply linear.

Based on the results of these deterministic calculations the influence of the above-mentioned parameters on the different scenarios of the safety assessment has been determined.

5.2.1.1. Normal evolution scenario

Subrosion and diapirism. The GDF induced processes in the salt formation have no significant effect on the rate of subrosion and diapirism. Parameters influencing the exposure are the leaching time, the residence time of the nuclides in the salt formation and the travel time in the geosphere. Analyses have shown that the travel time and residence time in the geosphere have only a moderate effect. Within the ranges to be considered a factor 2 in the dose could be expected (Prij, 1987; p. 185).

A factor which could have more influence is the leaching rate of radionuclides from the waste. In the dose calculation the leaching rate is considered to be relatively fast implying that the source term is determined by the subrosion rate. Natural analogue studies in the JSS project (Jercinovic, 1987) have indicated that the leaching rate can be orders of magnitude lower. In the extreme case this could lead to the situation where the salt is dissolved by subrosion whereas the glass matrix is still intact. In that case the possible exposure can be compared with the exposure in the scenario 'diapirism into the biosphere' 1 à 4 10^{-5} Sv/a.

Brine migration. The VEOS results have shown that in the normal evolution scenario the amount of brine is much smaller than the amount of water needed to corrode the containers. In these analyses the expected variations have been taken into account.

Thermo-mechanical cracks. The analyses of this scenario have shown that the stresses induced by the mining activities and/or the storage of the waste are too small to cause cracks being large enough to act as a pathway for nuclides.

5.2.1.2. Altered evolution scenarios

Diapirism into the biosphere. A large amount of analyses have been performed to quantify the effect of variations in the values of the following parameters used in the dose calculations:

- Disposal depth and diapirism rate
- Amount of dust in the polar dessert
- Flow behaviour of the contaminated dust compared to that of salt
- The amount of dust covering the salt and waste
- The dispersion of the waste caused by natural phenomena
- Protection of human beings in a polar dessert.

It wa concluded that due to variations in these parameters the dose rate will vary between 0 and 3 10^{-3} Sv/a (Prij, 1987; p. 185).

Flooding of the GDF. It has been shown that the dominant effects for this scenario are:

- Temperatures in the rock salt
- Stresses in the rock salt
- Convergence
- Nuclide migration inside the salt formation
- Nuclide migration with the groundwater
- Transport in the biosphere.

Most of these parameters can be influenced by the choice of the **disposal technique** and the formation type. The results indicate that (Prij, 1987; pp. 189-190):

- The release from the deep bore holes is a factor 2 larger than from a disposal mine.
- The release from a leached cavern is almost the same as the release from a dry cavern.
- The dam permeability is very important. A ten times lower value of the dam permability reduces the radionuclide release by a factor of 100. There is no release if the dam permeability is lower than 1.3 10^{-18} m^2.

The choice of the salt **formation type** determines the depth of the disposal and directly the convergence rate of the rock salt and the radionuclide travel time:

- If the volumetric convergence rate is larger than 5.5%/a the release from the salt formation is zero.
- For depth larger than 1100 m the volumetric convergence rate is higher than 5.5%/a.
- The releases into the biosphere are proportional with the inverse of the travel time.

With respect to the **waste strategy** the numerical results have shown that:

- For the deep boreholes and caverns the release is proportional to the amount of HLW.
- For a GDF in the form of a mine the release is proportional with the release from the amount of ILW and LLW waste emplaced in the chambers.
- The amount and activity of the waste from industry and hospitals is very small compared to the amount of waste from the NPP's. For dose rate calculations this type of waste is therefore negligible.

5.2.1.3. Human intrusion scenarios

Reconnaissance drilling. The results have shown that (Prij, 1987; p. 190):

- The exposure is largely dependent by the intensity of the inspection of the contaminated bore core. The difference in inspection of the bore cores (thourough versus routine) may result in differences in the the exposure by one order of magnitude.
- The probability of hitting the waste containers will be lower at greater disposal depths.

Wet salt production. The results have shown that (Prij, 1987; pp. 190-191):

- The exposure is linearly proportional to the leaching rate of the glass matrix.
- The probability of occurrence of the scenario will be lower if the GDF is constructed at greater depth.

Dry salt production. Results have shown that the exposure strongly depends on the thickness of the salt between the mined gallery and the HLW container. If this thickness is more than 1 m the exposure will be more than two orders of magnitude lower. If the thickness is smaller it is unrealistic to assume that the waste will not be discovered.

5.2.2. Concluding remarks

It was recognized that, due to the lack of site-specific data and knowledge, the OPLA-I programme was of a generic nature and therefore incorporated relatively large uncertainties. The conclusions formulated were therefore stated in terms of *expectations* rather than statements (OPLA, 1989; p. 19).

Although not treated in a systematic way, the Dutch OPLA-I programme, executed in the 1980's, addressed several aspects in relation to safety and the treatment of uncertainties. The most important ones were:

<u>Mining techniques</u>. OPLA-I assessed disposal in a conventional (dry) salt mine versus deep boreholes and concluded that significant uncertainties exist in deep borehole disposal. An extensive additional experimental programme was recommended (OPLA, 1989; p. 108).

<u>Safety assessment</u>. The OPLA-I programme considered several scenarios with regard to natural phenomena and the types and amounts of waste to be disposed (OPLA, 1989; p. 14). In addition, several scenarios with regard to human intrusion were analysed (OPLA, 1989; p. 14). With regard to the safety assessment, a significant effort was posed on the determination, modelling, and assessment of radiation damage to NaCl (e.g. OPLA, 1989; p. 79).

The **deterministic** VEOS analyses performed provided a thorough understanding of the processes and phenomena important to the post-closure safety of radioactive waste disposal in rock salt.

The study, however, could not discriminate between the relative importance of the processes and phenomena because it was impossible to determine the amount of conservatism in the best estimates of the health effects. Should a more systematic comparison between the different scenarios and disposal concepts be required a **probabilistic** method would have been used in addition to the deterministic one (Prij, 1991; p. 201). This research, OPLA phase 1A, started in July 1990 and was finished in 1993. The safety assessment in OPLA phase 1A was called PROSA (PRObabilistic Safety Assessment).

5.3. PROSA safety assessment

In the early 1990's a generic probabilistic safety analysis (PROSA) of the Dutch generic reference disposal concept has been performed (Prij, 1993). The PROSA study had two equally important aims, viz. the determination of the radiological effects on humans and the derivation of safety relevant characteristics of a disposal concept for radioactive waste. These characteristics have been derived from sensitivity analyses of the radiological consequences of some disposal concepts in rock salt formations. The PROSA study was restricted to the safety in the post-closure period.

The PROSA study comprised an extensive analysis of features, events and processes (FEPs) that could affect the long-term safety, and a newly developed methodology to identify scenarios, viz possible future evolutions of the disposal system.

5.3.1. Definition of scenarios – PROSA methodology

The starting point of the PROSA scenario definition was a comprehensive list of potentially relevant FEPs (Prij, 1993; Chapter 3). Subsequently, a screening procedure was applied to the FEP list in order to identify and compile a manageable number of representative scenarios, i.e. future evolutions of the disposal system. To simplify this screening procedure it was proposed to perform this screening for a number of well defined "states" of the barriers in the multi barrier system (Prij, 1993; p. 2.6). For a particular state of the multi barrier system it is easier to screen the FEPs because:

- In bypassed barriers transport related FEPs can be neglected.
- Each multibarrier system state implies a relevant time scale for the nuclides to arrive in the biosphere. If for instance the isolation shield in the salt formation is not bypassed it takes very long times before the nuclides leave the salt formation and consequently short time FEPS in the overburden and biosphere can be neglected.

Taking three main barriers into account, a total of 8 MBSSs (multi barrier system states) are possible:

Table 5-5 Possible MBSSs (multi barrier system states).

State number	*Engineered barriers*	*Isolation shield*	*Overburden*
1	Present	present	present
2	Present	present	bypassed
3	Present	bypassed	present
4	Present	bypassed	bypassed
5	Bypassed	present	present
6	Bypassed	present	bypassed
7	Bypassed	bypassed	present
8	Bypassed	bypassed	bypassed

Having defined the possible states the methodology for the definition of the scenarios contains the following steps:

i) *Identification of FEPs* which might influence the state of the barriers, the release, the transport and the state of the radionuclides. The list should be comprehensive and not be restricted to FEPs induced by nature or the waste but also contain human induced FEPs.

ii) *First screening of the list of FEPs* with respect to the host rock (salt formation) and the probability of occurrence.

iii) *Classification into primary and secondary FEPs*. A primary FEP directly attacks or bypasses one or more barriers of the multi barrier system. The primary FEPs define the state or the evolution of the GDF. In particularly they lead to a change in the size or the short circuiting of the barriers. The remaining FEPs are defined as the secondary FEPs. These FEPs influence the transport and the state of the nuclides and should be included in the transport model or code.

iv) *Assignment of the primary FEPs* to each of the MBSSs taking into account that some processes attack more than one barrier.

v) *Screening of the FEPs for each of the MBSSs*. In this screening a classification of the FEPs with respect to time showed to be very helpful.

vi) *Definition and selection of the scenarios to be analysed* further. This also includes the selection of the processes to be taken into account in the consequence analysis.

vii) *Determination of the secondary FEPs* for each of the MBSSs.

As in this methodology it is assumed that the evolution of the repository can be defined in terms of barrier states the methodology can be considered to be a top down approach in which the different multi-barrier system states are used as scenario elements. As the consequence analysis will be performed on the repository as a whole the methodology is not in conflict with the total systems approach.

Based upon available literature the FEPs are subdivided in three different categories, see Table 5-6.

Table 5-6 Categorisation of FEPs.

NATURAL PHENOMENA	*HUMAN-INDUCED PHENOMENA*	*WASTE & REPOSITORY INDUCED PHENOMENA*
1.1 Extra-terrestrial 1.2 Geological 1.3 Climatological 1.4 Geomorphological 1.5 Hydrological 1.6 Transport 1.7 Geochemical 1.8 Ecological	2.1 Design and construction 2.2 Operation and closure 2.3 Post-closure subsurface activities 2.4 Post-closure surface activities	3.1 Thermal 3.2 Chemical 3.3 Mechanical 3.5 Radiological

The PROSA FEP list consists of 63 natural phenomena, 48 human-induced phenomena and 36 waste & repository induced phenomena. Following the PROSA methodology 22 scenarios were identified, see Table 5-7.

Table 5-7 Summary of scenarios in PROSA.

Id.	*SCENARIO* Dominant primary FEP	*MBS Nr.*
1	Subrosion, diapirism, denudation	1
2	Subrosion, diapirism, fault in overburden	2
3	Subrosion, diapirism, glaciation[1)]	2
4	Flooding,	3
5	Flooding, large brine pocket	3
6	Flooding, fault in overburden	4
7	Flooding, large brine pocket, fault in overburden	4
8	Subrosion, high gas pressure[1)]	5
9	Subrosion, release of irradiation induced stored energy	5
10	High gas pressure, fault in overburden[1)]	6
11	Release of stored energy, fault in overburden[1)]	6
12	High gas pressure, glaciation[1)]	6
13	Release of stored energy, glaciation[1)]	6
14	Flooding, very high gas pressure[1)]	7
15	Flooding, release of stored energy[1)]	7
16	Leaky storage cavern	7
17	Flooding, very high gas pressure, fault[1)]	8
18	Flooding, release of stored energy, fault[1)]	8
19	Reconnaissance drilling	8
20	Solution mining	8
21	Conventional mining	8
22	Archaeological investigation[1)]	8

[1)] Scenarios not considered in VEOS.

The scenarios identified can be divided into three distinct groups or families of scenarios:
1) The **subrosion** scenarios (nr. 1, 2, 3, 8, 9,10, 11, 12, 13)
2) The **flooding** scenarios (nr. 4, 5, 6, 7, 14, 15, 17, 18)
3) The **human intrusion** scenarios (nr. 16, 19, 20, 21, 22)

5.3.2. Conceptual models

The PROSA analyses have been performed taking into account the the following compartments:

- Salt compartment
- Groundwater compartment
- Biosphere compartment

Salt compartment. For the salt compartment the REPOS (REPOSitory) module from EMOS (Storck, 1990) has been used. In REPOS a complete disposal mine can be modelled. REPOS has models for the release of nuclides from different types of waste containers and waste forms. The most relevant physical and chemical effects influencing the transport of radionuclides can be modelled with the code. Taking into account the then adopted facility design (see Figure 4-4), the GDF was conceptualized as:

- Waste canister: with a leaching model for vitrified HLW and for cemented MLW and LLW.
- Borehole: a cylindrical excavation for heat-generating waste (HLW and MLW) with a sealing.
- Chamber: excavation for non-heat-generating waste (MLW and LLW), backfilled and sealed.
- Disposal drift: rectangular excavation with specified dimensions, backfilled and sealed.
- Flank drift: rectangular excavation with specified dimensions and additional volume representing connecting drifts, all parts backfilled and sealed by dams.
- Central field: rectangular excavation with specified dimensions and additional volume representing the infrastructure of the mine, all parts backfilled and sealed by dam.

Additional details of these components and the method of modelling of the relevant processes are extensively described in the final report of PROSA (Prij, 1993; pp. 5.11-5.18).

Groundwater compartment. The nuclide transport with groundwater was analysed with MASCOT, developed by Battelle Memorial Institute USA for US DOE. In addition to the transport processes in the aquifer, various relevant processes have been accounted for, such as advection, dispersion, diffusion, sorption, and decay of the radionuclides released in the groundwater compartment (Prij, 1993; pp. 5.38-5.45).

In PROSA a two step approach was used to establish the groundwater flow (Prij, 1993; pp. 6.24-6.53). First, a detailed 2-D groundwater flow model was applied using METROPOL, the finite element code also used in VEOS. Subsequently, a uniform ‘effective’ velocity was obtained from particle tracking calculations.

For the groundwater compartment a probabilistic safety assessment was performed taking into account sampled variations of the following parameters:

- Thickness of the geohydrological units; 4 for a shallow and 8 for a deep dome.
- Porosity of these geohydrological units
- Permeability of the geohydrological units
- Ratio of permeability between the fault and the adjacent layer
- Hydraulic gradient at the surface.

The results are a statistically distributed effective velocity being the input parameter for the radionuclide transport calculations with MASCOT.

Biosphere compartment. The radiation exposure in terms of maximum dose rates is calculated with EXPOS (EXPOSure), the biosphere model from EMOS. The calculation is straightforward by multiplying the output of MASCOT (flow of groundwater) with nuclide specific dose conversion factors. The stochastic distribution of these dose conversion factors have been calculated separately with the computer code miniBIOS (Prij, 1993; pp. 5.46-5.51 and pp. 6.54-6.68) a simplified version of BIOS used in VEOS. In miniBIOS several release points are possible resulting in different exposure pathways as indicated in Figure 5-3.

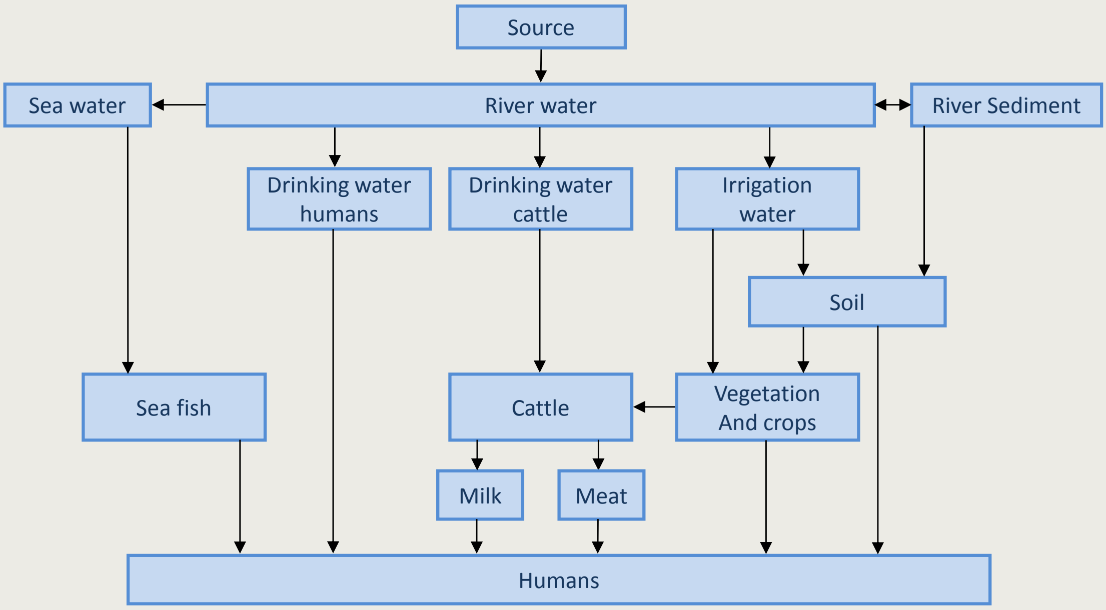


Figure 5-3 Contamination pathways considered in the PROSA biosphere model.

Integral model for the Subrosion scenarios
The subrosion scenarios have been probabilistically analysed with the code PANTER (Probabilistic Analysis of Nuclide Transport in the Environment of a Repository).
PANTER integrates the following models:

- SUBROS_ECN, the compartment model that calculates the release from the GDF
- MASCOT, calculating the transport through the overburden
- EXPOS, calculating the exposure in the biosphere, and
- A probabilistic shell.

The module for the analysis for subrosion SUBROS as incorporated in EMOS was improved at several points[24]:

- The subrosion rate and the diapirism rate were both modelled as independent stochastic variables.
- The modelling of the dissolution of glass in groundwater wa based on the chemical reaction of spherical glass in brines. The radius of the glass spheres was taken as a stochastic variable.
- The subrosion rate was shown to be dependent on the depth from the top of the diapir.

Integral model for the Flooding scenarios
The flooding scenarios in PROSA have been analysed deterministically with the code package EMOS_ECN consisting of:

- REPOS_ECN,
- MASCOT, and
- EXPOS

[24] To distinguish this adapted module from the original EMOS model it was called SUBROS_ECN.

Human intrusion scenarios
These scenarios have not been reanalysed in PROSA as there was no reason to adapt the models used in VEOS.

5.3.3. Calculations performed

Examples of results of the subrosion, water intrusion, and human intrusion scenarios are described in the following. Extensive evaluations of the calculated results can be found in the PROSA final report (Prij, 1993; Chapter 7).

Subrosion scenarios
In these scenarios the isolation shield of the salt around the GDF is slowly dissolved by groundwater. In the time needed to dissolve the isolation shield the GDF slowly moves upward due to diapirism. This is schematised in Figure 5-4 (Prij, 1993; p. 6.27).

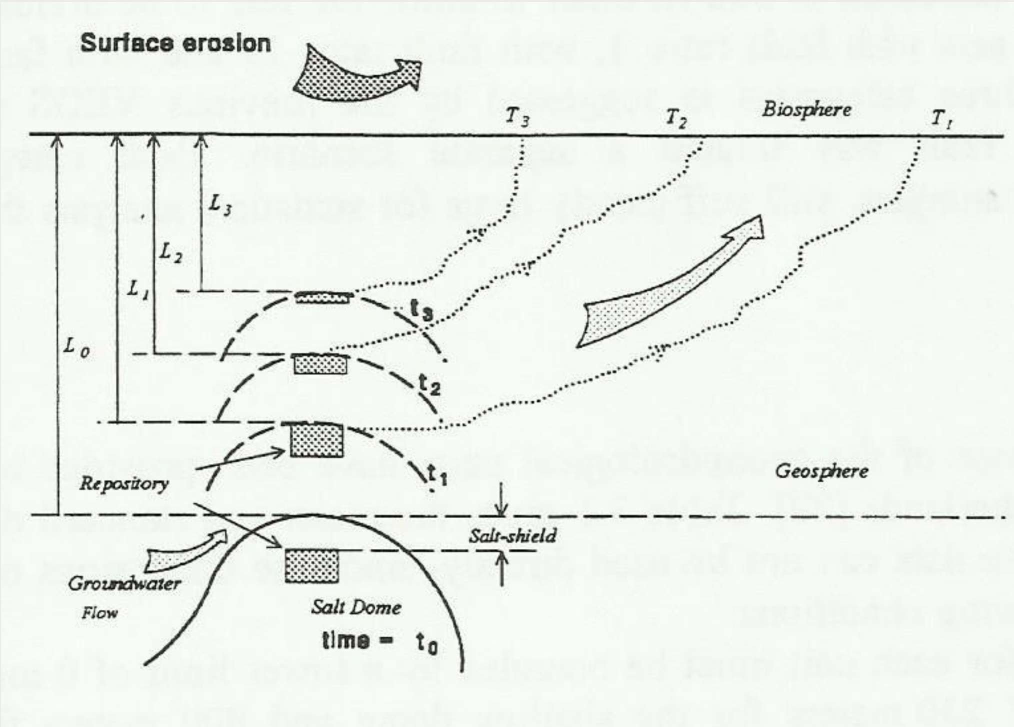


Figure 5-4 Schematisation of the subrosion and diapirism scenario.

After more than 0.1 Ma the waste comes in contact with the groundwater. After leaching the radionuclides out of the waste matrix the groundwater transports the radionuclides towards the biosphere. The outcome of SUBROS_ECN consisted of a time dependent activity flux with a statistical distribution at every discrete time step. The release of the salt formation was used as input for MASCOT, the module calculating the groundwater transport.

Due to the combined effect of subrosion of the top of the salt formation and erosion at the earth surface the thickness of the overburden gradually reduces with time. Therefore the travel times have been calculated as also accounting for the variability in the properties of the geohydrological layers. A typical result of these analyses is given in Table 5-8 (Prij, 1993; p. 6.46).

Table 5-8 **Statistics of particle velocity (ln v) for different depth intervals.**

Overburden thickness [m]		*Mean velocity [m/a]*		*Standard deviation [m/a]*	
min	max	fault	no fault	fault	no fault
0	28.75	-5.86	-6.64	2.58	2.24
28.75	86.25	-5.86	-6.64	2.58	2.24
86.25	143.75	-7.70	-8.39	3.48	3.13
143.75	201.25	-9.66	-10.4	3.11	2.95
201.25	230	-9.98	-10.9	2.93	2.99

Finally, the exposure in the biosphere was calculated with EXPOS using statistical distributions of the dose conversion factors. These factors were calculated with the code miniBIOS. The following miniBIOS input parameters were treated stochastically:

- The nuclide specific concentration factors (soil —> crop, pasture —> animal, water —> fish), sedimentation factor and distribution coefficient in soil and sediments.
- Biosphere data: depth and volume of the various compartments, flow of river and sediments, sediment transfer to land, erosion rate, rainfall, diffusion coefficients, suspended sediment load and porosity of soil and sediment.

A typical example of the resulting dose conversion factors is shown in Figure 5-5 (Prij, 1993; p. 6.61).

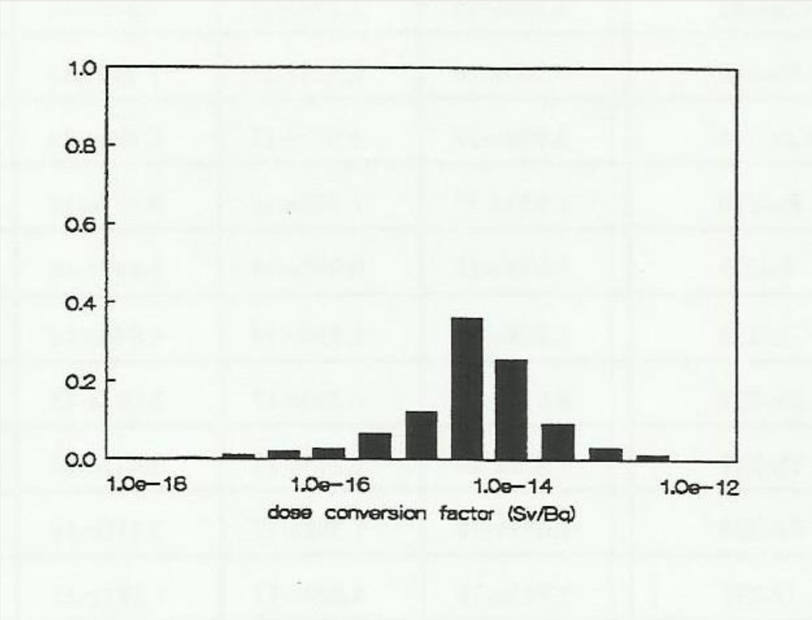


Figure 5-5 Calculated distribution of the dose conversion factor of Pu-239 (Sv/Bq).

With PANTER the probabilistic analyses have been performed for the following four cases (Prij, 1993; p. 7.4):

Table 5-9 Calculations preformed with PANTER.

Formation type	*Waste strategy*[*)]	*Fault*
Shallow dome	B	No
Shallow dome	B	Yes
Deep dome	B	No
Deep dome	B	Yes

*) Definition of Strategy B: see Section 4.3.1

The evolution of the results of the PANTER dose calculations is depicted in Figure 5-6 for the case “Shallow dome, faulted overburden”(Prij, 1993; p. 7.10).

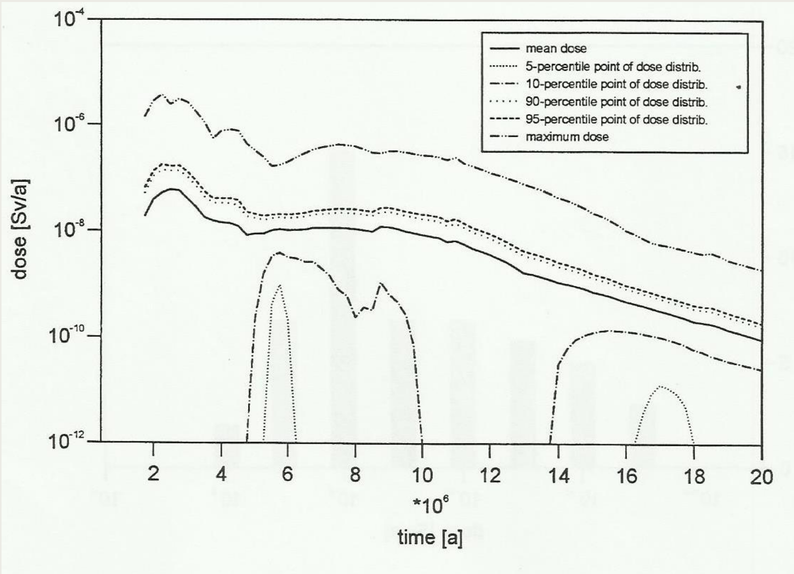


Figure 5-6 Evolution of calculated dose rate. Shallow dome with faulted overburden.

A summary of all calculation results for the subrosion scenarios is given in Table 5-10 (Prij, 1993; p. 7.12). The results indicate that in all cases both the calculated dose rates are orders of magnitude below the natural background radiation in the Netherlands (1-2 10^{-3} Sv/a)

Table 5-10 Summary of calculation results for the subrosion scenarios.

Formation type	*Waste strategy*	*Fault*	*Calculational result*			
			Max. Dose [Sv/a]	Max. mean [Sv/a]	Time [Ma]	Max.Risk [1/a]
Shallow dome	B	No	2.7 10^{-6}	4.0 10^{-8}	3.0	1.0 10^{-9}
	B	Yes	3.8 10^{-6}	5.9 10^{-8}	2.8	1.5 10^{-9}
Deep dome	B	No	2.1 10^{-6}	2.5 10^{-8}	2.5	0.6 10^{-9}
	B	Yes	2.0 10^{-6}	3.2 10^{-8}	3.0	0.8 10^{-9}

Water intrusion scenarios

In the water intrusion scenarios the isolation shield is bypassed through an undetected water carrying layer or a large crack. This water will rapidly be saturated with salt and converted to brine. This brine comes into contact with the waste as long as there is some void space left in the boreholes and chambers in which the waste is disposed. This is only the case in the first hundreds of years after disposal, hereafter the void space is closed due to the convergence behaviour of the rock salt. This same convergence behaviour extrudes the contaminated brine out of the GDF into the groundwater in the overburden where it is transported and eventually reaches the biosphere.

Dose calculations have been analysed deterministically with EMOS_ECN for several different cases (shallow/deep dome, Waste Strategy A/B, fault/no fault in overburden). Also for the water intrusion scenarios calculated dose rates ((Prij, 1993; p. 7.20) are negligible compared to the natural background radiation in the Netherlands.

It was argued that the presence of a large undetected brine pocket in combination with the groundwater intrusion will be very limited due to the early groundwater intrusion. For the results it appears that in the groundwater intrusion scenario it takes 50 years for the voids in the bore holes to be completely filled with groundwater. It is expected that filling these voids with brine from a undetected large brine pocket will take at least the same time (Prij, 1993; p. 7.20).

Human intrusion scenarios

The human intrusion scenarios are to be distinguished from the others as exposure is a direct or indirect result of a deliberate human action in or close to the geological formation where the waste is disposed. This sort of action can only occur supposing knowledge of the presence of radioactive waste has vanished. Therefore in the dose calculations it has been assumed that these actions will not take place earlier than 250 years after discharge of spent fuel from the reactor fuel. In that way consideration has been given to a period of interim storage, a period of operating the GDF and a period of at least hundred years in which the knowledge of the GDF is retained through administrative measures.

In addition dose calculations have been performed assuming that the human action will occur 1000 years after discharge of the fuel from the reactor. This period of 1000 years is considered to be a minimum duration of the efficiency of other passive, time resistance, marking methods. The results are summarized in Table 5-11 (Prij, 1993; p. 7.22), indicating that the scenario “Reconnaissance drilling” may lead to a significant exposure. This was also recognized in the VEOS safety assessment (see Section 5.2.1.3).

Table 5-11 Summary of the estimated dose due to human intrusions scenarios at different times after discharge of the fuel from the reactor.

Scenario	*After 250 years*	*After 1000 years*
Reconnaissance drilling	1 10^{2} Sv	2 10^{1} Sv
Solution mining	8 10^{-5} Sv/a	3 10^{-5} Sv/a
Leaking storage cavern	<< 1 10^{-7} Sv/a	<<1 10^{-7} S/a
Conventional salt mine	2 10^{-5} Sv/a	2 10^{-7} Sv/a

5.3.4. Treatment of uncertainties

In the PROSA study, the dose calculations were performed taking into account best estimate values for most model parameters. For parameters of importance in the dose calculations and with large uncertainties, probability density functions were selected. Probability density functions (PDFs) were used for a selected number of model parameters for the salt compartment, the groundwater “compartment”, and the biosphere “compartment”.

The definition of the PDFs was mainly done on the basis of careful engineering judgement. For several parameters, mean values and their standard deviation were obtained from literature or, in case of a few parameters, from measurements. For the majority of the considered parameters the statistical distributions were lognormal. For other parameters, e.g. the dose conversion factors, the 50 percentile value was used from a stochastic distribution that was calculated separately.

In addition to the dose calculations with statistically-spread input parameters, a sensitivity analysis has been performed aiming to find the input parameter(s) having the strongest influences on the exposure, as well as an uncertainty analysis aiming to quantify the output variability.

With statistical techniques described in the PROSA study (Prij, 1993; Section 9.2) sensitivity/uncertainty coefficients have been calculated for the set of **collected dose maxima** of each run.

These results are presented in Table 5-12 (Prij, 1993; p. 10.7). The table gives the sum of the ranks with respect to the four sensitivity measures, SPEA, PRCC, SRRC, and SMIR. That number is considered to be a measure for the sensitivity. The higher that number is, the larger is the influence of that parameter on the exposure. It can be observed from the table that for all four cases the internal rise rate is the most important parameter followed by the groundwater velocity, the dose conversion factors and the subrosion rate.

Table 5-12 Summed sensitivity coefficients of the maximum dose rates from the PANTER calculation.

	Shallow dome		*Deep dome*	
	fault	no-fault	fault	no fault
Internal rise rate	27	27	28	28
Groundwater velocity	23	23	24	22
Dose conversion factor	19	19	18	14
Exponent in subrosion relation	16	16	16	18
Distribution coefficient k_d	13	13	13	15
Dispersivity	8	8	5	8
Glas dissolution rate	7	8	9	9

It was concluded that the internal rise rate is clearly the most influential parameter with respect to the dose rate. This implies that for the implementation of a salt-based repository this parameter, which is unmistakably a site-specific issue, is the most relevant to address in further examinations. The sensitivity coefficients for the groundwater velocity, the dose conversion factors, and the exponent in the subrosion relation are often close to each other.

Perhaps a more interesting result of the analyses is the relatively low sensitivity of the exponent in the subrosion rate. In VEOS it was recognized that both the subrosion and diapirism rate were very important parameters. Due to the modelling assumptions a distinction between these two parameters could not be made. In PROSA the depth dependency of the subrosion rate could be taken into account which implies that the influence of the subrosion is reduced to a secondary one. With the data set of PROSA there will always be an equilibrium between the internal rise rate and the subrosion rate.

For the ***salt formation*** it appears that for the subrosion scenario the internal rise rate was the most sensitive parameter while for the groundwater intrusion scenario the model for convergence and compaction of backfilled openings is most sensitive. The glass dissolution, the subrosion rate as well as the time of groundwater intrusion were found to be less sensitive.

For the ***groundwater compartment*** statistical techniques have been used to derive the sensitivity and uncertainty of the groundwater velocity. It appears that the most sensitive parameters are the characteristics of the fault, the permeability of the layers and the hydraulic gradient at the surface. The porosity and the location of the nuclide release point do not significantly influence the groundwater velocity.

The most important sources of uncertainty in the calculated ***dose conversion factors*** were determined using statistical techniques. Sensitive parameters were especially the flow of the river and the various distribution coefficients and concentration factors. The level of technology and the consumption pattern are assumed to resemble the present situation. Consequently, the uncertainty caused by these assumptions is not included in the dose conversion factors.

The release position of the radionuclides in the biosphere will also be of importance. In the biosphere model used the radionuclides are released in a river. Numerical analyses showed that another release point e.g. a well can result in three orders of magnitude higher dose conversion factors while release in a sea can lead to three orders of magnitude lower dose conversion factors. It is clear that the release position has an important influence on the resulting dose. As the dilution in a well is low the resulting dose will he high. The dose due to the release in a sea will be low as a result of the strong dilution.

For the groundwater intrusion scenario a deterministic technique has been applied. For this scenario special attention has been given to the sensitivity of the model for convergence and compaction of backfilled openings.

As an extension of the work in PROSA a probabilistic sensitivity and uncertainty analysis of the groundwater intrusion scenario has been carried out in the framework of the EU programme EVEREST. The most important conclusions were (Prij, 1995; pp. 7.3.7.34):

- Of the 300 simulations 25 resulted in a zero release of nuclides from the salt formation, since the final porosity was reached before any groundwater could reach the waste containers.
- Only 153 out of the 300 simulations resulted in a maximum exposure > 10^{-15} Sv/a.
- The maximum and mean dose rates are 4 10^{-8} Sv/a respectively 1 10^{-9} Sv/a. These results are in good agreement with the deterministic analysis where the maximum dose rate was found to be 2 10^{-9} Sv/a (shallow dome, waste strategy B).
- Sensitivity and uncertainty analyses have been performed of the amount of extruded ^{135}Cs[25] at different times after disposal and the maximum exposure.
- The parameters of the crushed salt are a considerable source of uncertainty, whereas the properties of the rock salt are a weak uncertainty source in the mass of extruded ^{135}Cs and the maximum exposure.
- The groundwater velocity is a considerable uncertainty source, whereas the biosphere dose conversion factor and the time of groundwater intrusion are a weak source for the uncertainty in the maximum exposure.

In the PROSA study no attempt has been taken to determine the sensitivity of the human intrusion scenarios.

5.3.5. Concluding remarks

Chapter 11 of the PROSA study (Prij, 1993) provides an evaluation of the PROSA results compared to the VEOS results. From the evaluation it appeared that for the subrosion scenario the deterministically calculated so-called 'conservative' dose rates in VEOS are lower than the maximum values from the probabilistic analyses in PROSA. The VEOS values are almost equal to the mean value of the probability density function of the calculated

[25] This nuclide is the dominant nuclide. In 95% of the simulations with a dose rate >10^{-12} Sv/a this nuclide determines more than 99% of the total exposure

dose rates in PROSA. Based on this finding it was concluded that one must be very careful with the meaning of the notion 'conservative' result.

For the groundwater intrusion scenarios large differences were found between the VEOS results and the PROSA results. The PROSA results which can be considered as best estimate results are about 5 orders of magnitude lower than the VEOS. The main explanation was found to be the use of an improved model for the convergence and compaction of backfilled openings.

Based on the analyses performed the following conclusions were drawn:

i) The results of PROSA confirmed the main conclusion of VEOS: The scenarios induced by natural FEPs result in a very low radiation exposure of future generations. Relatively high exposures can result from human intrusion scenarios. For all scenarios considered the health risk is < 10^{-6}/a.

ii) The methodology developed for the identification of scenarios is transparent and flexible. It is based on a long list of potentially important FEPs and the multi-barrier system.

iii) Based on theoretical and experimental evidence some models were improved. These improvements concerned:

- the introduction of transient creep in the convergence of an opening,
- the explicit use of the compaction properties of the backfill,
- the introduction of stochastic dimensions of the glass particles in the leaching model,
- the depth dependence of the subrosion rate,
- the salt rise induced reduction of the thickness of the overburden in the nuclide transport.

iv) A probabilistic calculation model has been developed and applied for the subrosion scenarios.

v) Due to the improved model of convergence and compaction the exposure in the groundwater intrusion scenarios is reduced significantly to a negligible level.

vi) A low internal rise rate and the possibility of deep disposal are the safety relevant characteristics from a salt formation.

vii) The hydrological properties of the overburden play a role in the transport of nuclides in the future. These properties then, at that time can deviate significantly and unpredictably from the present value. The overburden properties are considered to be of importance for the mine's stability of the salt formation.

viii) Although only limited attention was paid to the human intrusion scenarios in PROSA, these scenarios are considered to be the scenarios with the highest individual risk.

The basis of the methodology as applied in the PROSA study seems the way ahead in the Netherlands, i.e. a systematic approach as bulleted above, which includes sensitivity and uncertainty analysis.

Uncertainty analysis is a sound scientific ingredient of a safety assessment. A probabilistic analysis gives additional endpoints such as total risk or percentile values of dose rates. It can also help to identify parts of the disposal system that are robust as one of the characteristics of robustness is that it will not contribute significantly to uncertainty in the endpoints. This potential has not been explored in PROSA where the main attention was drawn to the parts of the system that dominate the uncertainty, i.e. the less robust parts of the system. Additional performance indicators as endpoint of an uncertainty analysis and a different look at the results of the analysis may help in showing robustness of the system.

5.4. CORA safety assessment

In the Netherlands, safety reports concerning the disposal of radioactive waste in salt formations have been published during the OPLA research programme (OPLA-I: 1984 - 1989; OPLA-IA: 1989 - 1993). During that time the intention was to build such a disposal facility, maybe even on a short term, but only if sufficient political and societal support was obtained.

The CORA programme was executed taking into account the following conditions:

- The waste will be stored for at least 50 to 100 years in a surface storage facility managed by COVRA NV.
- Alternatives for the disposal in salt formations should be studied.
- The waste has to be disposed of in such a way that the waste is retrievable for a significant period of time.

As an alternative for the disposal in salt formations, the disposal in clay formations has also been studied in CORA. That research was performed in close cooperation with colleagues from SCK in Belgium and reported elsewhere (CORA, 2001).

During the OPLA programme only very limited research into retrievability of already emplaced radioactive waste has been carried out. In CORA retrievability was the main topic. This also implied that the research should focus on the operational phase of the disposal facility. The results for the post-operational facility, obtained during the OPLA research programme, were considered to remain valid.

In CORA the design of the repository, METRO-I has been adopted, see also Chapter 4.2.3 of the present report. The developers of that disposal concept assumed that a phased disposal strategy would provide 'passive safety', i.e. even in the case of unexpected events during the operational phase, the facility as a whole will evolve into a safe condition without a need for human actions.

With respect to safety, the OPLA-programme focussed on the 'post-operational phase', the phase after sealing and closure of the facility. The models developed and applied to study the post-operational phase do not directly apply to the operational phase.

As a consequence, the research in CORA aimed to achieve a further development of the models required for safety studies, with the intention of applying them to the operational phases in the METRO disposal concepts.

The following sections summarize the work performed on the salt safety assessment carried out in CORA.

5.4.1. Definition of scenarios

For the development of the method for scenario development two conditions have been applied in the CORA safety assessment, also referred to as the “METRO study” (Grupa, 2000; Section ES.2):

- The method of scenario development was aimed to show what has been taken into account in the analysis, and what has not (yet) been taken into account.
- Definitions (e.g. for 'scenario') have been developed that are consistent with their application in a probabilistic assessment.

The FEP-catalogues developed for waste disposal in clay or rock salt have been inspected for FEPs that threaten the isolation of the waste whilst the facility has not been closed and sealed. These FEPs are given in Table 5-13 below (Grupa, 2000; p. 140). A problem has been that many of the FEPs were described rather loosely, making it hard to classify them.

Table 5-13 FEPs that threaten the isolation of the waste during the operational phase.

FEP	***Safety aspect***
Seismicity	Integrity of the disposal cell. Flooding of the disposal mine due to activation of a fracture
Undetected geological features	Flooding of the disposal mine
Canister defects	Integrity of the disposal cell
Poor quality construction	Integrity of the disposal cell and/or the mine
Improper waste placement	Integrity of the disposal cell
Inadequate backfill compaction, voidage	Integrity of the disposal cell
Co-disposal of reactive wastes	Integrity of the disposal cell
Inadvertent inclusion of undesirable materials	Integrity of the disposal cell
Deserted unsealed repository	Flooding of the disposal mine
Injection of fluids in (abandoned) repository	Integrity of the disposal cell
Malicious intrusion	Malicious use of the disposed materials
Recovery of repository materials	Integrity of the disposal cell
Underground nuclear testing in (abandoned) repository	Integrity of the disposal mine
Gas generation effects	Integrity of the disposal cell
Subsidence, collapse	Integrity of the disposal mine
Nuclear criticality	Integrity of the disposal cell
Release of stored (gamma) energy	Integrity of the disposal cell

The models that have been developed in the METRO study were meant to analyse the consequences of the so-called 'neglection scenario'. The FEP *'deserted unsealed repository'* forms the basis for this scenario. It will be obvious that the other FEPs will lead to new variants of this scenario, or even completely new scenarios.

The neglection scenario describes the evolution of the facility in the case of neglection after the waste has been emplaced in the disposal cells, but before the facility is closed and sealed. The neglection scenario consists of the following steps (Grupa, 2000; p.141):

1. At some time after maintenance of the facility has ceased, the facility will flood with groundwater, as a result of leakages, leading to brine formation within the facility. For the analysis it is assumed that the flooding occurs 200 years after the cease of the maintenance.
2. The brine is pressed through plugs that seal the disposal cells.
3. The brine reaches the waste (vitrified high level waste) in the disposal cell. A part of the waste will dissolve in the brine (leaching).
4. Due to the creep of the rock salt the mined volumes converge and contaminated brine is pressed out of the facility, and enters the ground water flow system.
5. Through various routes contaminants originating from the waste may reach the surface, where they may eventually lead to exposure.

5.4.2. Conceptual models

For the analyses of the ‘neglection scenario’ the EMOS_ECN code package has been used. An important modification is the improvement of modelling of the transition from permeable to impermeable salt plugs. This transition is caused by the convergence driven compaction of the saltplug from an initial porosity to a final porosity, similar to that of undisturbed rock salt. The percolation model developed at the Utrecht University by Peach (Peach, 1991) was implemented to model that transition.

To be able to simplify the retrieval of the waste the METRO design considered the HLW canister to be placed in a horizontal borehole with a depth of about 5 m, see Figure 5-7. In each borehole one canister would be stored and a saltplug of 3 m would seal the borehole. This consideration signififantly deviated from the designs used in OPLA with vertical boreholes of 300 m with about 80 canisters and a salt sealing plug of 20 m (cf Section 4.2.2). In the analyses performed in CORA only HLW has been considered whereas in OPLA all categories of waste have been taken into account. It must be noted that more than 95% of the radioactivity and radiotoxicity is contained in the HLW (Hart, 2014b; Table 6.2).

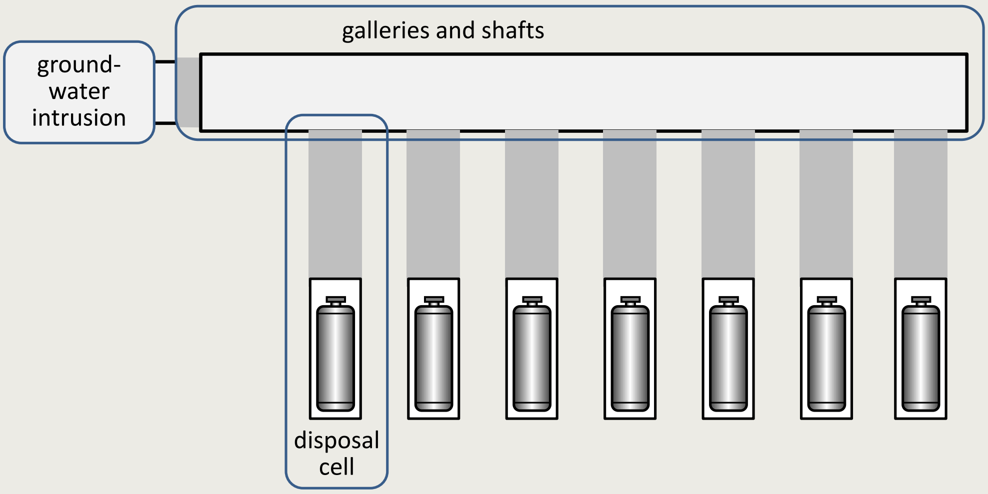


Figure 5-7 Horizontal cross-section of a gallery and several disposal cells in the METRO design.

In CORA it has been assumed that the behaviour of the salt plug after its emplacement would be similar as that of compacting crushed salt.

In the CORA study only attention has been paid to radiological doses as a result of future exposures to the waste that may escape from the facility. Additional research is needed to determine whether other (quantitative) measures are needed to evaluate risks.

5.4.3. Calculations performed

As stated in the previous section, only the *neglection scenario* has been analysed in detail in CORA. An example of the calculated results is shown in Figure 5-8 depicting the amount of waste that escapes each year from one disposal cell to the gallery. A noticeable process is that, due to convergence of the surrounding rock salt, the salt plug will be compacted resulting in a porosity where it becomes impermeable. For the considered analysis this would occut after about 4000 years.

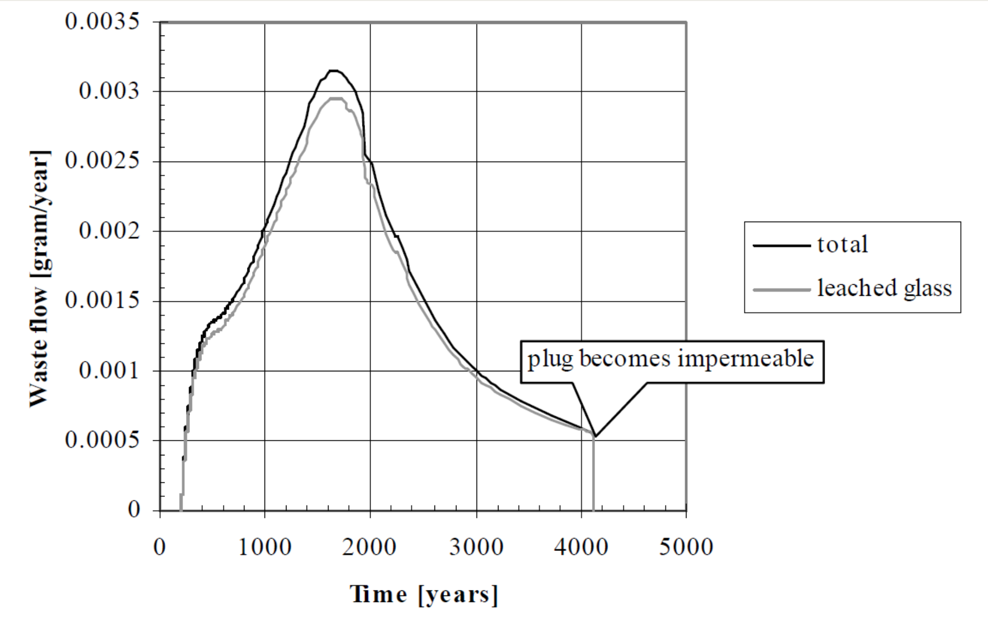


Figure 5-8 Escape of waste from one disposal cell to the gallery (neglection scenario).

As long as radionuclides will be released from the waste packages and the disposal cell (i.e. up to about 4000 years), they will migrate through the flooded galleries and reach the groundwater system. Ultimately radionuclides may reach the biosphere resulting in an individual dose rate, see Figure 5-9. Compared to the internationally accepted dose limit of $1 \cdot 10^{-3}$ Sv/year, the calculated dose rate for the *neglection scenario* and under the adopted assumptions is negligible.

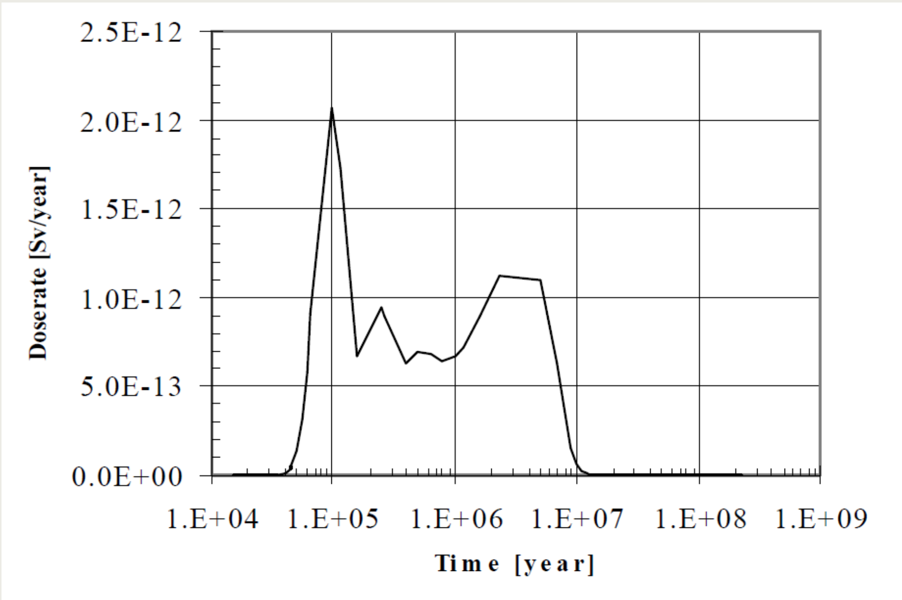


Figure 5-9 Individual effective dose (neglection scenario).

This result is determined assuming that the waste reaches the surface through ground water flow. However, it will be necessary to study some variants of this transport pathway, e.g. by assuming that a well produces water from the ground water close to the facility, so the well water contains traces of the waste. Such scenarios have not been consider in CORA.

5.4.4. Treatment of uncertainties

In the framework of the CORA study some attention has been given to procedures that would enable the systematic development of new scenarios. These considerations led to the following insights:

- A rational system for scenario development may be envisaged to ensure fulfilment of Safety Case requirements such that no omissions will occur.
- The use of a structured approach for the description of scenarios, e.g. the use of FEPs, provides a better description of each scenario, and improves the traceability of phenomena that are accounted for in the scenario.
- The use of scenarios in safety studies is very common. However, the selection of phenomena that are included in a given scenario is subjective. This becomes evident when the work of different research groups is compared: one group may incorporate many more phenomena than another group in a scenario with the same name. This discretion in the coverage of a given scenario may cause some confusion.
- Assuming a consistent description of the scenarios, it will be possible to compare the results of different safety studies. This will also provide means of publishing results more widely.

Also some attention has been given to an analysis of the uncertainty in the modelling of the permeability of salt with low porosities. This had been performed in a deterministic way.

The CORA final report recognized several types of uncertainties that should be addressed further:

- The reliability of the data (CORA, 2001; Section 7.2.1). It was recognized that for several data still uncertainties do exist, e.g. concerning the validity of data that have been obtained under laboratory conditions in in-situ conditions (CORA, 2001; p. 27). It was mentioned that these uncertainties could be reduced by R&D under in-situ conditions.
- The applicability of data in the long term (CORA, 2001; p. 27). On time scales of several hundreds of thousands of years uncertainties in the extrapolation of features and processes are inevitable.
- In previous Dutch programmes the R&D efforts were focused on the retrievability of high-level waste. Retrievability aspects of low- and intermediate level waste have not yet been investigated (CORA, 2001; p. 27).
- Calculational uncertainties. The CORA Commission considered the uncertainties in the calculations of the radiation doses and the underlying assumptions such that further analysis would be required. The uncertainties were mainly attributed to a situation of abandonment of the facility, for which the risk was considered maximal. (CORA, 2001; p. 55).
- Uncertainty about the development of processes when the structure of the facility has degraded. For that situation further research would be required.

5.4.5. Concluding remarks of the METRO (CORA) study

An important question in the study on retrievable concepts is whether the METRO design is fail safe. In the design studies it is stated that some passive safety can be given in the operational phase in which the mine is not yet closed. The plugs of the disposal cells are considered to assure sufficient isolation and protection in unexpected situations such as neglection and flooding of the GDF.

This statement, relevant for all disposal concepts, need to further be investigated. To that aim the models used in OPLA have been developed further and for some processes it was necessary that new models had to be developed.

With the improved models some analyses have been performed giving a first impression of the consequences of neglection and some points at which the design can be improved.

It was concluded that the models developed in OPLA for the analysis of the flooding scenario could be used for the analysis of the neglection scenario. This specially holds for the models of convergence and compaction (Grupa, 2000; pp. 135-136) built in the salt compartment model REPOS_ECN build in the code package EMOS_ECN.

The calculations performed in CORA were mainly focused on the neglection scenario, and contain a number of assumptions that need further study. Research is needed into the transition from permeable to impermeable material for the options available for sealing the disposal cells. Further, the behaviour of the 'technical barriers' (glass that fixates the waste, canister and the plugs) regarding gas-generation and radiation should be studied and the levels of uncertainty quantified.

Trends observed from the CORA safety assessment are (Grupa, 2000; p. 135):

- The models for dry salt formulated by Spiers (1988) indicate that it takes about a million years before the plugs become impermeable. Based on the results of BAMBUS this would be 2 or 3 orders of magnitude shorter. But even then it will more than thousands years. One should seriously consider improvements of the design to shorten this period.
- When the GDF would flood the compaction of open spaces in the facility accelerates significantly. A convergence model for wet salt, the FADT model of Spiers (1988), indicate that it would take about 4000 years to reach the ‘final’ porosity, i.e. a porosity similar to that of the undisturbed rock salt.
- Rough estimates indicated that the exposure can be much higher than calculated for the neglection scenario analysed in CORA if other exposure paths are considered such as a shortcut of the groundwater path through drilling.

As the key question is whether a salt-based disposal facility is fail safe it is important to know whether the physical processes will result in complete isolation of the waste and what the consequences would be if that were not the case.

In OPLA is was assumed that a section filled with crushed salt and closed with a salt plug would become impermeable if the porosity reaches the same porosity as natural rock salt. In CORA a first attempt has been made to explain this behaviour with the percolation model. This also gives an explanation of the behaviour of the processes of permeability and dispersion when the compaction of the salt reaches the percolation threshold.

5.5. PAMINA safety assessment

In the EU FP6 project PAMINA, Task 2.1.D “Techniques for Sensitivity and Uncertainty Analysis”, JRC-IE and NRG have undertaken a probabilistic uncertainty analysis for repository facilities in rock salt and in clay by combining their tools for PA calculations and statistical analysis and demonstrated the applicability of the combined approach (Schröder, 2009). For a limited number of input parameters and their associated statistical distributions, samples were generated and used as input for PA calculations. Several output parameters were extracted from the results and analysed statistically to describe the uncertainty of the output parameter and their dependence on the variability of the input parameter. NRG has used the basic scenario description and the input parameters from the Dutch national programme (CORA) as a starting point for the analyses, and performed the PA calculations. JRC-IE has provided the statistical framework and performed the statistical analyses.

The work performed in the PAMINA project built on the experiences gained in the EU-FP4 Framework project BAMBUS-II (Grupa, 2003). Both in the BAMBUS-II and PAMINA studies the focus was on the compaction behaviour of compacted salt sealing plugs of boreholes, under dry and wet (flooding) conditions. Both studies applied a similar disposal concept and waste inventory. The main difference was that in PAMINA a larger set of safety an performance indicators was considered and that a comprehensive statistical analysis of the calculated results was performed. In the following therefore only the PAMINA study is discussed in more detail.

The PAMINA study described the results for an analysis of a repository design in rock salt, based on the EVEREST concept (Cadelli, 1996; Section 3.2.1.2.2), with the geometry of the boreholes adapted from the Torad-B design (Schröder, 2009; Sections 2.2, 2.3). It is noted that the EVEREST concept in essence is quite similar to the VEOS (Prij, 1989; Section 2.3) and PROSA (Prij, 1993; Figure 6.1) disposal concepts.

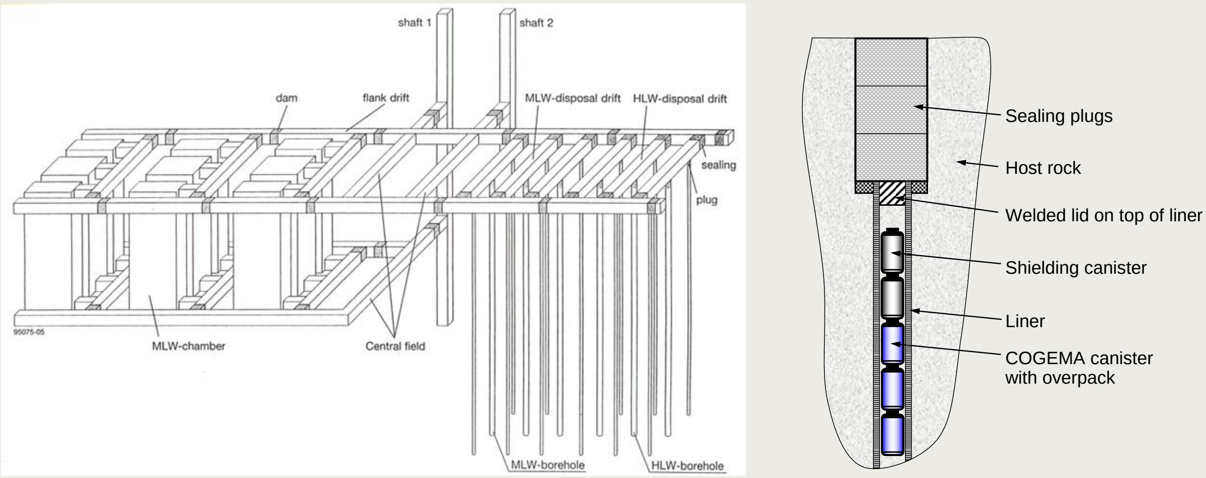


Figure 5-10 EVEREST repository and Torad-B borehole designs adopted in the PAMINA safety assessment.

The reference repository was assumed to be located at a depth of 840 m in a salt dome and consists of shafts, underground working places, warehouses, connecting galleries and vertical disposal cells that contain 300 canisters of vitrified HLW each. For the study, a simplified repository design (Figure 5-11) based on the *Torad-B* concept was adopted, consisting of three segments:

- the waste canisters
- the disposal cells
- the central field.

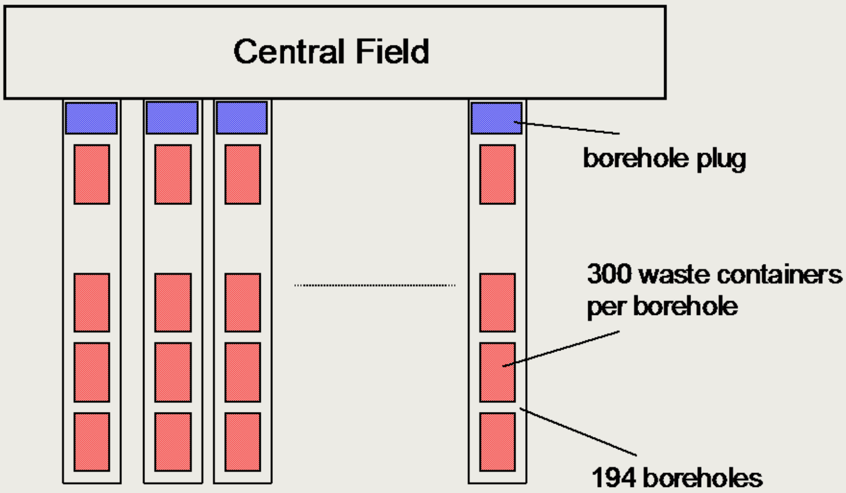


Figure 5-11 Compartment representation of the Torad-B facility design

From the 194 boreholes of the *Torad-B* concept, for the simplicity only one borehole is modelled in the PAMINA study, assuming the case of a brine intrusion in one borehole only and a consecutive instant failure of the lining and of all 300 containers.

In order to calculate the compaction behaviour of the crushed salt backfill and the salt borehole plugs, as well as the mobilisation and release of nuclides from the glass-matrix of the containers in this large-scale repository design, the computer program *EMOS_ccm2* has been applied. NRG improved several modules of this programme, mostly with regard to the modelling of the compaction behaviour of crushed salt and the porosity-permeability relation (Schröder, 2009; Sections 3.2, 3.3).

The probabilistic uncertainty analysis of that study focused on the closure behaviour of the rock salt plug that seals the disposal cell. For six input parameters, relevant for the creep behaviour of the compacted salt plug and the porosity-permeability relation, average values and their distribution were derived from experimental data (Zhang, 2006). A sample with 1000 realizations was generated for all six input parameters and used for performance assessment calculations of the normal evolution scenario and the brine intrusion scenario. As output, five point values and nine time-dependent output parameters were selected, all related to the safety and performance of the repository. Statistical analysis covered uncertainty analysis (mean, standard deviation, skewness, kurtosis and percentiles) as well as sensitivity analysis (R^2, Pearson, Spearman, SRC, PCC, SRRC and PRCC[26]).

An example of the calculated output is given in Figure 5-12 (Schröder, 2009; Figure 7.2), showing the evolution of the *permeability of the sealing plug* (variable “B2”). The dry and the wet (flooding) scenarios showsignificant differences in compaction behaviour of the sealing plug. Wheras in the dry scenario no run needs more than 25 years to get the porosity of the plug return - as a result of compaction - to the value of rock salt (i.e. 0.003), in the wet scenario the plug needs always more than 175 years. Reason for the large differences in the associated so-called “closure times” between the dry and the wet scenario is the hydrostatic-pressure that is built up in the plug once the plug gets saturated after the central field and shafts have been flooded, and that counteracts the pressure applied to the plug by the surrounding rock salt.

[26] Respectively: standardized regression coefficient, partial correlation coefficient, standardized rank regression coefficient, Partial Rank Correlation Coefficient

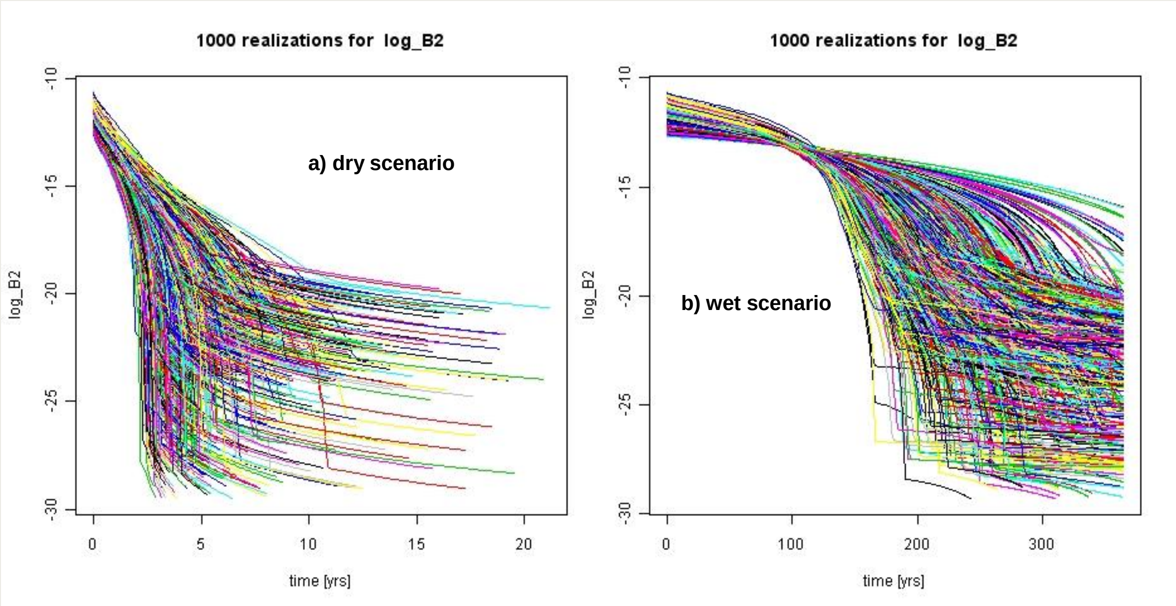


Figure 5-12 Evolution of the permeability of the plug (Variable "B2") over time for a) the dry scenario and b) the wet scenario.

The use of safety assessment tools like EMOS for sensitivity analyses was found to have certain limitations. The calculations carried out for this study could be performed in an acceptable time period because of two reasons: the repository set-up chosen was relatively simple and the number of variable input parameters was small, resulting in a relatively small sample size.

Another limitation concerned the variety of time spans for the different EMOS simulations. The simulations were not necessarily terminated at fixed end-points. Because the calculations by EMOS where terminated when the porosity of the sealing plug reached its threshold value of 0.0003, different simulation times occurred for the set of calculations. From the time of termination of a particular simulation, no values of output parameters were available for that simulation to perform the statistical analysis. As a consequence the number of time steps and output parameters available for the sensitivity and uncertainty analysis decreased progressively with simulation time for the sets of 1000 simulations.

The calculated potential dose rates varied only to very small extent, despite the large variation of input parameters used. The main reason for the small variations found was the rather small time interval necessary to seal the plug (95 percentile at 785 years) compared to the long travel times (hundred thousands of years) of the radionuclides to the biosphere. The potential dose rates calculated for the wet scenario are far below any legal regulation limits with the used parameterization of *MASCOT*.

From the point of view of repository safety, the variability of the parameterization of the plug model has only a limited influence on the repository safety in terms of potential dose rate for the wet scenario. For the chosen scenario, the process of recrystallization creep was only relevant duting the first *days*. The porosity-dependency of both creep processes - the lower the porosity, the lower the compaction rate - levels out the variability of the creep parameter and stabilizes therefore the system behaviour.

The lack of experimental data on the permeability behaviour of crushed salt at very low porosities does not influence the safety of the repository: for the given set-up, any monotonic assumption on the permeability evolution at porosities below 0.005 is suitable to show that the expected dose rates will be far below any regulatory limit.

5.6. Safety assessment in Germany

The safety assessment methodology, more specifically the development of scenarios, in Germany was developed in two projects. Initially, in an early phase of the R&D project ISIBEL (Buhmann et al. 2010b), the methodology was deduced and tested for reference scenarios. The methodology was then expanded in the course of R&D project VSG for the development of alternative scenarios (Beuth, 2012a).

The methodology aims at deriving, in a systematic manner, a limited number of plausible scenarios specifically one reference scenario and a number of alternative scenarios. Overall, the scenarios should comprehensively represent the reasonable range of repository system evolutions. The methodology allows direct assignment of probability classes to the scenarios in accordance with the safety requirements of (BMU 2010) and is depicted schematically in Figure 5-13 (Beuth, 2012a; Abb.3.1).

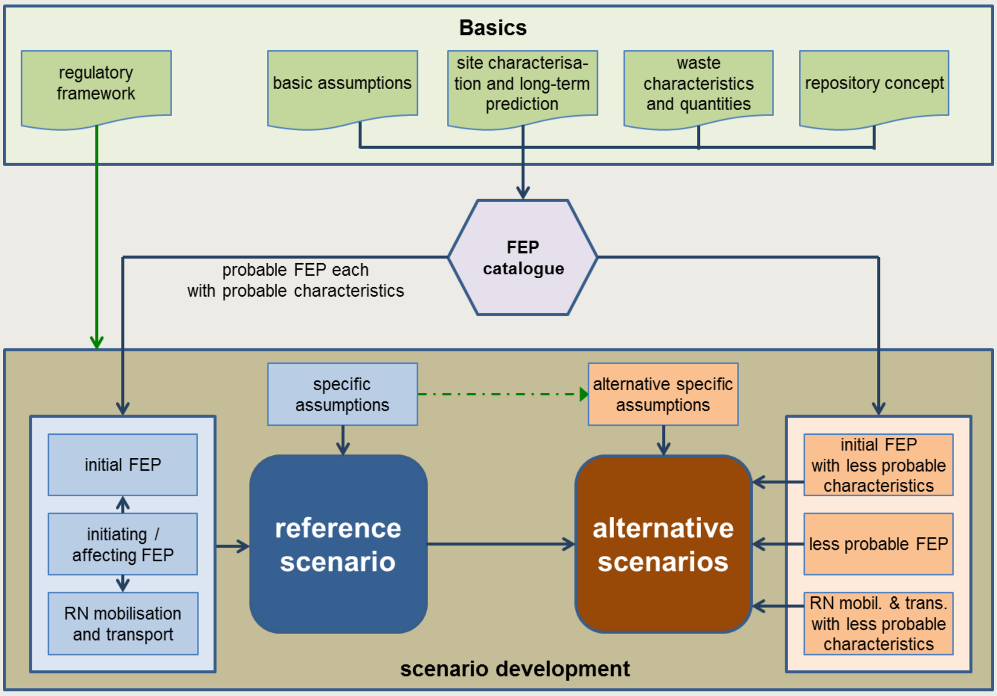


Figure 5-13 Methodology for the development of scenarios applied in R&D project VSG.

A number of basic assumptions were made in the project VSG in order to deal with uncertainties resulting from incomplete site investigations below ground. So far, only a fairly small region of the salt diapir has been investigated in situ. The basic assumptions pertain to the lateral size and geological structure of the Gorleben salt diapir in the emplacement depths, the properties of the rock salt in the CRZ, and the available dimension of the halite body of the Staßfurt Series (z2HS).

Taking specific assumptions into account, the *reference scenario* results from considering all probable FEPs

- that may impair the functionality of the initial barriers (Initial FEPs), and
- that determine the mobilization of radionuclides from the waste and their subsequent transport, both in the gas phase and in the liquid phase.

Alternative scenarios are evolutions which differ in exactly one aspect from the reference scenario. Alternative scenarios are developed from the following starting points (see also Figure 5-13):

- evolutions resulting from alternatives concerning the specific assumptions for the reference scenario,

- evolutions resulting from less probable characteristics of the FEP that may impair the functionality of the initial barriers,
- evolutions resulting from less probable characteristics of the FEP describing mobilisation and transport of radionuclides, and
- evolutions resulting from less probable FEP.

It is feasible that similar alternative evolutions result from the different starting points. In this case, various evolutions may be abstracted into one representative alternative scenario that covers the characteristics of the various evolutions.

Within the framework of the VSG project the relevance of future human activities has been discussed in detail (Beuth, 2012b). That study deals with human activities after closure of the repository, which directly damage the CRZ or the technical barriers. Only those actions have been considered which are unintended, i.e. without knowledge of the presence of a repository and its hazards.

The FEP screening process and subsequent scenario development in Germany is tailored to the specific circumstances at Gorleben, and therefore significantly dependent on the location of the site. These site-specific circumstances will probably differ from the Dutch situation, since in the Netherlands a site selection process has not been started yet.

A comprehensive report on the preliminary German safety assessment was recently published as part of the VSG project (Larue, 2013).

5.7. Safety assessment in US

The WIPP programme utilized the performance assessment methodology as schematically represented in Figure 5-14 (US DOE, 2004; p. 1-14). The process includes developing scenarios, scenario probabilities, and consequence models to estimate performance (US DOE, 2004; p. 1-13).

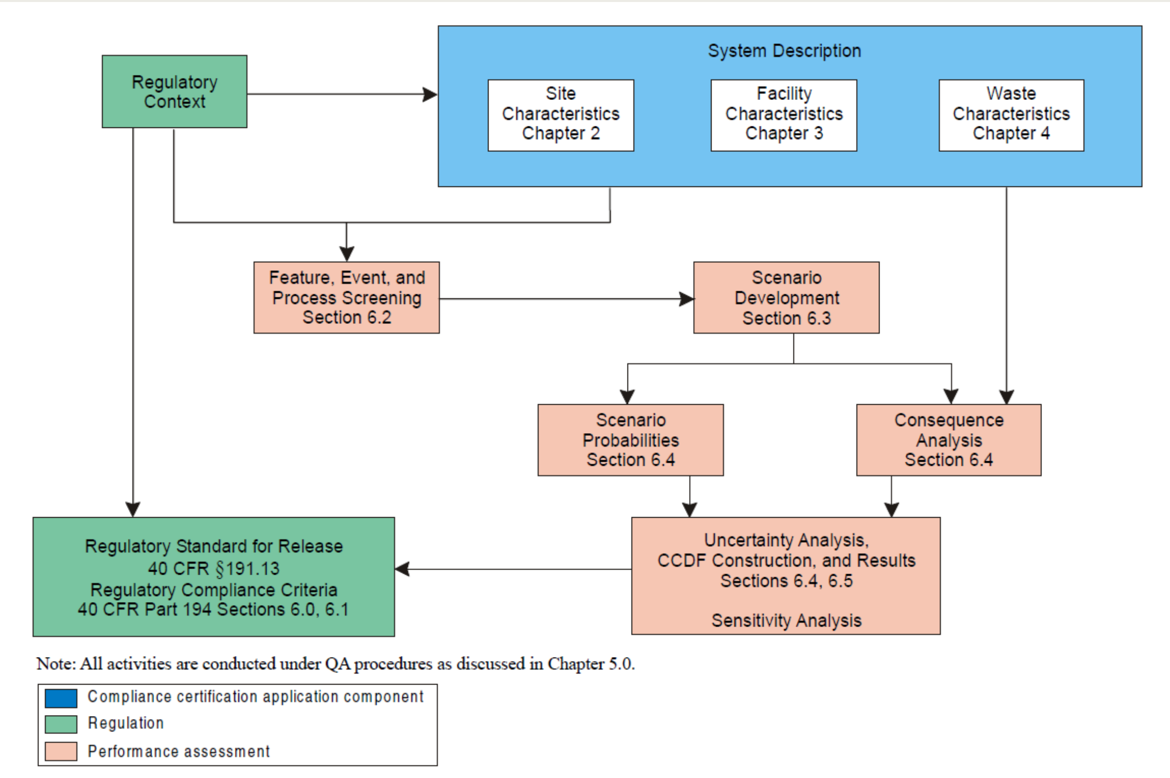


Figure 5-14 Methodology for Performance Assessment of the WIPP (US DOE, 2004; p. 1-14).

A thorough understanding of the disposal system itself and the possible interactions of the repository, emplaced waste and the surrounding geology are required for the PA (US DOE, 2004; p. 6-3). In its recertification process the DOE claimed passive institutional controls, including notification devices which reduce the likelihood of inadvertent human intrusion, but these controls were not included in the PA (US DOE, 2004; p. 1-10).

For developing the relevant scenarios, the WIPP FEP catalogue was screened, identifing 236 features, events and processes (FEPs) in the CCA; these included natural FEPs, waste- and repository-induced FEPs, and human-initiated FEPs (US EPA, 1997; p. 32-3) The DOE then screened out FEPs based on regulatory considerations, the identified FEP consequence, and on the probability of the occurrence of the FEP (US EPA 1997; p. 32-4). This screening process is shown in Figure 5-15 (US DOE, 2004; p. 6-34, p. 6-55).

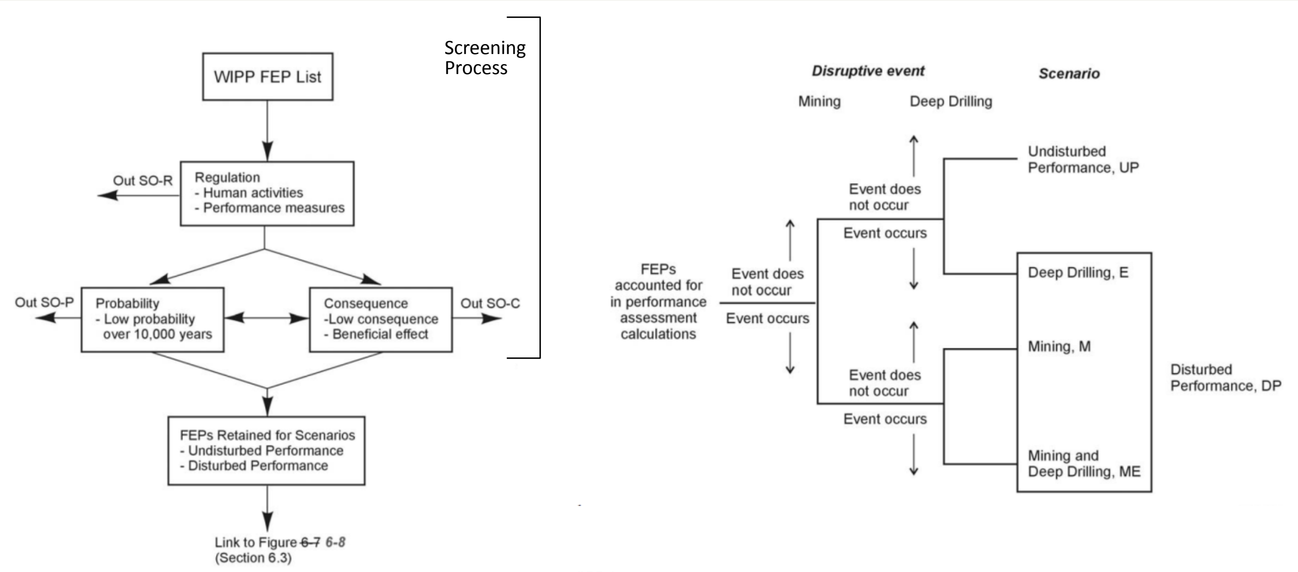


Figure 5-15 Screening process based on screening classifications and logic diagram for scenario analysis

After screening, 16 remaining combinations of FEPs are included in logic diagrams to illustrate the formation of scenarios for consequence analysis. As in Figure 5-15 (right), a combination of occurrence/non-occurrence of potentially disruptive FEPs defines the derived scenarios.

Regulation (Sections 191.15 and 191.24 of 40 CFR) require the evaluation of undisturbed performance, which does not include human intrusion nor unlikely natural events (DOE, 2004; p. 6-4).

Because human intrusion / drilling scenarios may have the most significant impact on the disposal system's containment abilities of all the FEPs, EPA regulation require that the PA consider scenarios of intrusions by drilling for resources (US DOE, 2004; p. 6-6).

The results of the PA show that the only possible mechanism for significant radionuclide release from the system comes from human intrusion (US DOE, 2004; p. 6-7). These human intrusion scenarios include single as well as multiple borehole intrusions and are of two different types: those that penetrate the Castile Formation's pressurized brine reservoir and those that do not (US DOE, 2004; p. 6-12). Should a borehole drill through the repository nuclides may be released and reach pressurized groundwater. Groundwater may be carried up to the surface through the borehole, transporting radionuclides along with the mobilized groundwater (US DOE, 2004; p. 6-70).

Other possible pathways for radionuclide releases from drilling include cuttings (solids cut by the drill and brought to the surface), cavings (solids that fall from the borehole wall), and spallings (solids forced into the borehole during drilling by pressurized gas) (US EPA, 2006b; p. 2).

The WIPP FEP screening process and subsequent scenario development is significantly dependent on the location of the site. In the vicinity of the WIPP site oil drilling and the mining of potassium salt is a common practice. These circumstances may not apply to the Dutch situation. On the other hand, the methodology of FEP screening and scenario development adopted for the WIPP site is comparable to the PROSA methodology.

5.8. Recent views and safety assessment methodology

The recent views on safety assessment methodology mainly concern the identification of the relevant FEPs, the scenario development and the method for dealing with uncertainties. These three subjects will be summarized in the following.

5.8.1. The ISAM methodology

Taking into consideration the more recent approaches to safety assessment for near surface disposal facilities, the ISAM project identified the need to address the following key components (IAEA, 2004; p.1):

- Specification of the assessment context;
- Description of the waste disposal system;
- Development and justification of scenarios;
- Formulation and implementation of models;
- Analysis of results and building of confidence.

The sequence of these steps is elucidated in Figure 5-16 (IAEA, 2004; p.17). Each of these components was extensively analysed and discussed during the project.

The ISAM flowchart as presented in Figure 5-16 contains each of the main steps shown in the earlier 1991 NEA Review of Safety Assessment Methods (NEA, 1991; p.43), although the components of the assessment basis are not explicitly represented, but are rather lumped together in the box “describe system” (NEA, 2012; p.89). In addition, the ISAM flowchart highlights the importance of the assessment context in determining the scope of the safety assessment and also shows by “feedback arrows” the iterative nature of the assessment process.

The ISAM methodology focuses on safety analyses and their results, rather than on the broader range of evidence, analyses and arguments that are synthesised in a Safety Case. Consequently, the ISAM iteration loops shown are limited to the assessment, while the idea that assessment results can serve as a basis for system optimisation (i.e. improving system performance and/or robustness by changes in siting and design) is missing.

In addition, the scope of the ISAM flowchart is limited to an “acceptance versus rejection” situation, i.e. to the typical circumstances of a licensing application. Another aspect which was not accounted for in the ISAM flowchart is that iterations take place between assessment activities such as scenario development, model formulation, numerical analyses, and result interpretation. Experience shows that such iterations occur during the whole assessment process and not only in an acceptance/rejection situation.

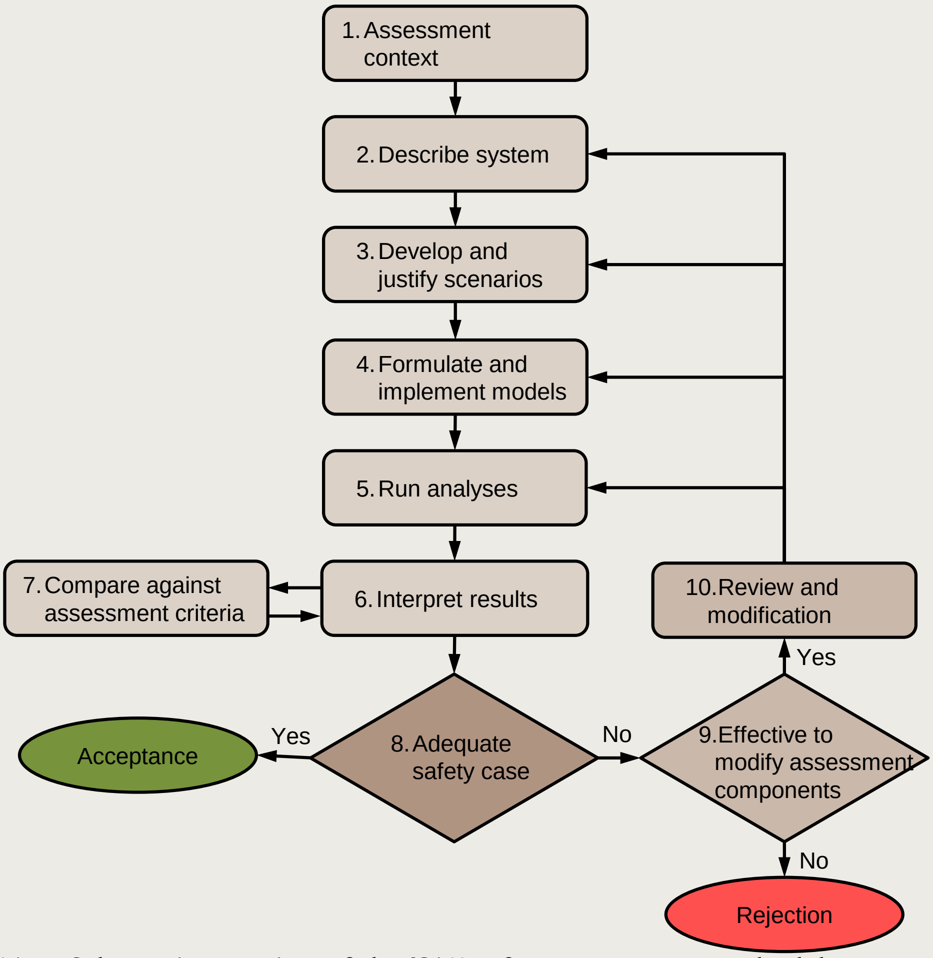


Figure 5-16 Schematic overview of the ISAM safety assessment methodology.

Summarising, the ISAM flowchart has limitations as a generic depiction of assessment strategies adopted by the different waste management programmes and internationally. Nonetheless, although they differ in scope, in their degree of detail and in the terminology adopted, many common elements and linkages may be identified between the ISAM flowchart and other recent flowcharts, such as is presently being applied in the OPERA safety assessment of a disposal facility in Boom Clay (Grupa, 2014).

5.8.2. Features, events, and processes

One of the requirements for the safety case is that "The site for a disposal facility shall be characterized at a level of detail sufficient to support a general understanding of both the characteristics of the site and how the site will evolve over time." *Such a characterisation must also include a specific understanding of the impact on safety of features, events and processes (FEPs) associated with the site and the facility*" (IAEA, 2011; Requirement 15).

A structured list of FEPs and their detailed descriptions are valuable for the evaluation of the impact of features, events, and processes on the performance of safety functions attributed to the disposal system. In addition, FEPs also may assist in the development of scenarios, describing the possible evolutions of a disposal system. Scenario generation methodologies usually include a systematic identification and consideration of FEPs that can directly or indirectly affect the release and transport of radionuclides from a disposal facility.

The development of scenarios is one of the key elements of a safety case, see for example the discussion in (NEA 2012; p. 10). In modern safety assessments the development of scenarios is based on characterising future evolutions of the repository system by FEPs that are compiled into a FEP catalogue. Based on the geological situation at the investigated site, the repository concept and the different types of waste, a site-specific FEP catalogue must be compiled. For each FEP detailed information is provided that allows selecting directly all FEP which are relevant for formulating a reference or baseline scenario and alternative scenarios.

FEP catalogue in PROSA and CORA

The basis of the approaches of the selection of scenarios and the safety assessment in PROSA was a list of FEPs that are potentially important for the repository safety. The initial list was based on existing literature and was neither site nor host-rock specific. A screening process resulted in a list with all FEPs considered relevant for the safety assessment of a salt-based repository contained 63 natural phenomena, 48 human-induced phenomena and 36 waste and repository-induced phenomena (Prij, 1993; Ch.3). The FEP list was sub-divided into three main categories and additional sub-categories:

Table 5-14 FEP categories applied in PROSA.

Main category	*Sub-category*
Natural phenomena	Extra-terrestial
	Geological
	Climatological
	Geomorphological
	Hydrological
	Transport
	Geochemical
	Ecological
Human induced phenomena	Design and construction
	Operation and closure
	Post-closure and sub-surface activities
	Post-closure surface activities
Waste and repository induced phenomena	Thermal
	Chemical
	Mechanical
	Radiological

For each FEP a short description of the feature, event, or process was provided, as well as a classification of the relative importance in relation to the potential impact on each of the engineered barriers of the disposal system.

The PROSA FEP list was used as a tool for the selection and definition of scenarios that were assessed further in the probabilistic safety assessment.

As part of the CORA programme the PROSA list was re-evaluated and compared to the WIPP FEP database, resulting in a set of FEPs that were added to the PROSA FEP-catalogue (Grupa, 2000; Appendix A). In addition a FEP classification scheme alternative to PROSA was proposed (Grupa, 2000; Appendix B).

FEP catalogue developed in Germany

In the course of the R&D project ISIBEL, a host-rock specific, generic FEP catalogue for salt formations was developed based on reference data from the Gorleben site (Buhmann et al.

2008). This FEP catalogue describes all features, events and processes that might influence the future evolution of the repository system. The FEP catalogue was systematically compiled, starting from a comparison with the NEA-FEP database. To check that the consolidated list was comprehensive, a plausibility check of the sequences and interdependences of the FEP was carried out and this identified some missing FEPs. Further indications came from the geoscientific long-term prognosis and, as an iterative process, from the scenario development and the process analyses.

The ISIBEL FEP list evolved into a more site-specific FEP list, dedicated to conditions prevailing at Gorleben, the VSG FEP catalogue (Wolf, 2012a), (Wolf, 2012b). That FEP catalogue is dedicated to the German context as it takes the German regulatory requirements into account. In addition, the most relevant data were compiled characterising the present site status, including the status of site exploration and the provided waste inventory and the developed repository concepts. This FEP catalogue contained 115 FEP entries.

The German VSG FEP catalogue had been analysed and evaluated in the course of a national peer review by five expert organisations. The peer review confirmed that the FEP catalogue complies with the international state of safety assessments for salt-based repositories (Bollingerfehr, 2013; p. 65).

US: WIPP FEP Catalogue

The development of the FEP catalogue applicable to the salt-based WIPP repository is an ongoing process that started in 1994, and was re-assessed in 2004, 2009, and recently in 2014.

As part of the 1994 WIPP Compliance Certification Application (CCA) and Compliance Recertification Application (CRA) processes, the US DOE compiled a FEP catalogue identifying a comprehensive set of features, events and processes. The original FEPs generation and screening were documented in the CCA and the resulting FEP list became the FEPs compliance baseline. The baseline contained 237 FEPs and was documented in Appendix SCR of the CCA (US DOE, 1994).

An updated version if the FEP catalogue was produced in 2004 (US DOE, 2004). The main difference with the 1994 version was that the FEPs were subdivided into three classes and several subclasses, and re-numbered in consistency with these classes.

For the CRA-2009, a reassessment of FEPs concluded that of the 235 FEPs considered for the CRA-2004, 188 have not been changed, 35 have been updated with new information, have been split into 20 similar, but more descriptive FEPs, 1 screening argument has been changed to correct errors discovered during review, and 1 has had its screening decision changed. Therefore, there are 245 WIPP FEPs for the CRA-2009. None of these new or updated FEPs required changes to PA models or codes; existing models represented these FEPs in their current configurations. All changes were tabulated in *Table SCR-1. FEPs Change Summary Since CRA-2004* (US DOE, 2009).

The purpose of the 2014 update FEP screening (US DOE, 2014) was to identify those FEPs that should be accounted for in PA calculations, and those FEPs that need not be considered further. The DOE’s process of removing FEPs from consideration in PA calculations involved the structured application of explicit screening criteria, which were derived from regulatory requirements. The CRA-2014 considers 245 WIPP FEPs. A complete list of the CRA-2014 FEPs is provided in *Table SCR-1. FEPs Summary for CRA-2014* of (US DOE, 2014).

US: FEP Catalogue for Spent Fuel and HLW

At present there is a programme underway in the US to assess the feasibility to also accept, besides defence waste, commercial spent fuel (Used Nuclear Fuel, UNF) and heat generating HLW at a salt-based repository. This idea has emerged as one of the waste management options after ceasing the activities performed in the Yucca Mountain project and the DOE forming the Blue Ribbon Commission on America's Nuclear Future in 2010. In (Hansen, 2011) an overview is provided about all aspects that are needed to finally emplace heat generating UNF and HLW in a salt based repository. One of the first activities to be undertaken if a salt repository option were to be investigated as a disposal option for UNF and HLW would be a re-consideration of the FEPs screening table provided in (US DOE, 2014). The introduction of new waste forms with attendant heat and radioactivity will likely add some key FEPs to the performance assessment analyses (Hansen, 2011; p. 86). The FEP list differs somewhat from other FEP lists as it includes several aspects related more to the Safety Case than to the safety assessment.

Evaluation of FEPs

Appendix 1 of this report evaluates FEPs relevant for salt. The German FEP catalogue for salt served as the basis of the evaluation (Wolf, 2012b), and has been compared with the PROSA FEP list (Prij, 1993), and the WIPP FEP catalogue (US DOE, 2014). The designations and numbering of the FEPs have been indicated as in the respective documents.

Due to budget and time constraints a thorough and comprehensive comparison was outside the scope of the OSSC project. However, the information provided may serve as a basis to consult the respective documentation and look in more detail to the FEPs considered relevant for the disposal in rock salt.

FEPs: Conclusions

In conclusion it can be stated that FEP catalogues for the disposal of radioactive waste in salt-based repositories are still under development in the US. The site-specific WIPP FEP catalogue is regularly updated according to the latest developments and insights gained from e.g. PA. In addition, there is the FEP catalogue for a generic salt-based site for the disposal of heat-generating spent fuel and HLW, that also is being iterated on a regular basis.

An evaluation of FEPs relevant for the disposal of radioactive waste in salt reveals that a number of FEPs still need more substantiation and elaboration. In particular the effects of temperature on specific FEPs and THMC-coupled processes, and the relation to long-term safety should be assessed in more detail.

5.8.3. Scenario development

The development of scenarios is one of the key elements of a safety case, see for example the discussion in (NEA, 2012; Chapter 5). In that document the term "Scenario" represents (and is understood as) a simplified description of a potential evolution of the repository system from a given initial state. Scenarios describe the compilation and arrangement of safety relevant features, events and processes as a fundamental basis for the assessment of post closure safety which includes assessing the potential consequences on human and its environment.

Also the IAEA SSG-23 (IAEA, 2012; p. 52) states that w*hen assessing the safety of a waste disposal facility, it is important to consider the performance of the disposal system under both present and future conditions. This means that many different factors (e.g. future human actions, climate and other environmental changes as well as events or processes that could affect the performance of the disposal facility) need to be taken into account. This may be achieved through the formulation and analysis of a set of scenarios. In this* *respect, development of scenarios constitutes the fundamental basis for the quantitative assessment.*

The development of scenarios is in essence not constrained to a host rock envisaged to facilitate a repository. There are however scenarios that are specific to a type of host rock, as will be elucidated in the remainder of this section.

In general the methods to develop scenarios are based on characterising future evolutions of the repository system by features, events, and processes (FEPs), that are compiled into an FEP catalogue (NEA, 2012; Section 5.1), see Section 5.8.2.

Several categories of scenarios are usually defined, usually one normal evolution or reference scenario and one or several altered scenarios (NEA, 2012; p. 126). The latter correspond to the main categories of FEPs potentially initiating or causing significant deviations or disturbances from the reference scenario. This might be the case e.g. due to uncertainties concerning the assumptions on safety functions, or due to effects such as climate change, repository issues (such as canister or seal defects), or future human actions.

5.8.3.1. Extended PROSA methodology

As elucidated in Section 5.3.1, the PROSA methodology for the identification of scenarios was fit for its purpose. In the original approach those FEPs would be identified that short circuit one or more of the barriers of the multi-barrier system (see Figure 5-17; Grupa, 1999; Figure A). However, this approach ignores the difference between a short-circuit and a process that describes the transport through a barrier (or subsystem). The evaluation options in PROSA of the FEPs in terms of “short circuit” or “no short circuit” lacked the level of detail that is needed for a proper evaluation.

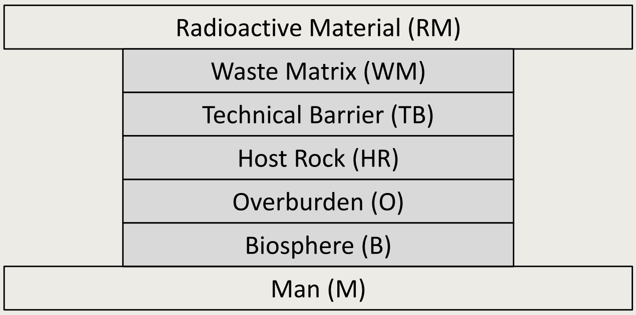


Figure 5-17 Initial barriers between radioactive material and man.

The experience gained in CORA was that those FEPs that were associated with short circuits actually would allow new transport processes, which would only sometimes really behave as a short circuit. In the improved method elucidated in (Grupa, 2000) this was recognised.

The improved method first identifies “old” transport processes, and subsequently identifies those FEPs that allow “new” transport processes.

The improved method consists of the following steps:

1) Definition of a Multi Barrier Systems (MBSs) for each known scenario. As part of defining the MBS, a FEP catalogue is screened for those FEPs describing the transport of nuclides through the subsystem (or barriers) for the given scenario. Those “transport FEPs” are then defined as “secondary FEPs”, as illustrated in Figure 5-18 for the subrosion scenario (Grupa, 1999, Figure B). At the end of step 1 each known scenario should be described as an MBS including the secondary FEPs.

Radioactive Material (RM)	Secondary FEPs
Waste Matrix (WM)	*Leaching*
Host Rock (HR)	*Subrosion*
Overburden (O)	*Advection, dispersion*
Biosphere (B)	*Uptake of nuclides*
Man (M)	

Figure 5-18 Multi Barrier System for the subrosion scenario.

2) Classification of all FEPs for each subsystem (or barrier) in each MBS. In this step **primary** FEPs are identified, which are judged to change a multi-barrier subsystem (or more subsystems) to such an extent that the secondary FEPs identified in Step 1 are not adequate to describe the transport through the altered subsystem.
3) One or more new scenarios are added as a result of the FEP classification. Having identified the primary FEPs identified in step 2, new scenarios may be developed. For example, in the PROSA FEP catalogue 34 primary FEPs have been identified that affect “subrosion-transport” through the rock salt. From these 34 primary FEPs six cause a (water-) permeable connection between the subsystem “technical barriers” and the subsystem “overburden”. Due to a water-permeable connection the facility can be flooded. This newly identified situation should be analysed in a separate scenario.
4) Goto step 1 (iterative process). For the newly identified scenarios the FEP-catalogue has to be re-evaluated to identify possible (new) primary FEPs, starting with the development of the MBS for the new scenarios as in step 1.

Given the MBS and the primary FEPs, it is often not straightforward to develop a new MBS. The experience was that the optimal description of the MBS has to be developed in close interaction with the actual analyses (e.g. dose calculations) of the scenario related to the new MBS. By doing that it can be judged which barriers are most relevant for a particular scenario. Knowledge of the relevant barriers is essential for defining the new Multi Barrier System for the scenario.

An issue encountered in the extended PROSA methodology was that some transport processes are not related to a single FEP (Grupa, 1999; p. 108). For example the transport of radionuclides through the subsystem in case of the brine intrusion scenario in a repository in rock salt would be determined by three FEPs: (1) creep of rock, (2) water being squeezed out from the cavities, and (3) advective transport through the permeable connection). That problem has been solved only provisionally by the afore mentioned interaction between the performance assessment and the development of the MBS. This solution, however, is not sufficiently transparent and decreases the persuasiveness of the method. It was therefore proposed to further develop this method.

The extended PROSA method has been applied for the safety study underlying to the license application for the closure of the Asse (D) salt mine including the experimental disposal facilities (29. January 2007[27]) and for a review on behalf of the Ministry of Agriculture and Environment of Sachsen-Anhalt (MLU) of two supporting reports issued in 2002 in preparation of the licensing process for the Morsleben Repository for radioactive waste (Endlager für radioaktive Abfälle Morsleben - ERAM) (Grupa, 2003b).

[27] The closure plan and safety report was delivered to the German State Ministry of Mining, Energy and Geology (LBEG) in Clausthal-Zellerfeld on 29 January 2007.

5.9. Evaluation

With respect to the safety assessment methodology some evaluating and concluding remarks are provided in the following sections for the following topics:

- FEPs
- Scenario development
- Handling of uncertainties

5.9.1. FEPs

A comparison between the PROSA FEP list and the FEP catalogue of WIPP (US DOE, 1996) has been performed as part of the CORA programme. Also FEPs have been identified which were common or missing in both catalogues. The results of the comparisons are tabulated in Appendix A of (Grupa, 2000). It was concluded that the WIPP catalogue was the most complete and comprehensive by that time.

The structure and content of the German FEP catalogue reflect the German safety concept established for the Gorleben disposal site, and the German scenario development methodology (Buhmann et al., 2010b). Therefore it is not completely applicable to the situation in the Netherlands.

According to the current German law, the conditions that have to be assumed in the safety assessment are determined by regulation. Therefore, FEPs related to Human Behaviour and the biosphere (exposure factors) are not considered in the VSG FEP catalogue. For the FEP category Surface Environment, only FEPs that have influence on the geosphere are considered in the VSG catalogue.

The WIPP FEP catalogue has been developed specifically for the considered site and waste characteristics, i.e. transuranic, non-heat generating waste. The WIPP FEP catalogue is regularly revised and updated. A recent development is the development of a WIPP-specific FEP catalogue for heat-generating HLW, that may be disposed of at the WIPP site in the future (Freeze, 2014).

In the framework of the US-German collaboration on the topic of radioactive waste disposal in salt-based geologic disposal facilities, experts from the US and Germany are in the process of compiling a comprehensive FEPs catalogue for disposal of heat-generating waste in salt. The ultimate goal of the joint effort is to populate an international FEP database for salt repositories that can promote easy searching for FEPs and pertinent information. The populated FEP matrix can be a useful tool for developing a PA model and a robust Safety Case in salt repositories.

Both the German FEP catalogue and the WIPP FEP catalogue may serve as a basis for the development of a salt-specific FEP catalogue within the Dutch context. This would require a re-evaluation of all FEPs in the catalogues in order to assess the applicability of the FEPs to the Dutch context.

5.9.2. Scenario development

The methodology to develop scenarios for the safety assessment of salt-based repositories is principally independent of the specific host rock. The methodology usually applied broadly consists of the following features:

- The substantiation of a FEP catalogue. The FEP catalogue must be host-rock specific and take into account country-specific regulatory aspects, waste characteristics (e.g. non-heat generating versus heat generating waste), as well as the repository concept at hand.
- The screening of FEPs with regard to the probability of their occurrence and the potential consequences on the safety, both during the operational phase and thereafter.

- The formulation of a reference / normal evolution / most probable scenario, taking into account the FEPs that were judged to be relevant for the safety.
- The formulation of alternative scenarios, by considering disruptive events or taking into account less-probable FEPs.

In general such an analysis results, broadly speaking, in three classes of scenarios:

- The normal evolution scenario, describing the most likely evolution of the repository and the radiological consequences to the population and the environment.
- Alternative scenarios with a natural cause, such as flooding, subsurface effects (hydrological, geochemical, etc.) earthquakes, undetected features, etc. e.g. variations of the waste characteristics.
- Human intrusion scenarios, or alternative scenarios with a human cause, such as drilling activities, fluid extraction or injection, mining activities etc.

Typical scenarios that have to be assessed for salt-based repositories are:

- subrosion scenarios, i.e. the dissolution of rock salt by groundwater in combination with diapirism,
- scenarios where flooding or undetected brine pockets in the rock salt are considered,
- human intrusion, e.g. in relation to the mining of salt.

A scenario specific for the Dutch policy of potential waste retrieval is the so-called "neglection scenario". This scenario may come into play when social and political developments may impair the control activities during the extended operational phase, which is closely related to the requirement of retrievable disposal. This may lead to the abandonment of the facility before it is properly closed and sealed.

The FEP screening process and subsequent scenario development activities in Germany and US are tailored to the specific circumstances at Gorleben and WIPP respectively, and therefore significantly dependent on the location of the site. These circumstances may differ from the Dutch situation, since in the Netherlands no site selection process has been followed. On the other hand, the methodology of FEP screening and scenario development adopted in Germany and US are comparable to the PROSA methodology. The PROSA methodology is however more specific with respect to assessing any adverse effects of events and processes on the functioning of the engineered barrier system.

5.9.3. Treatment of uncertainties

Various techniques for uncertainty analysis as part of the safety assessment have been described and applied in the past. However, it is not obvious how to choose between these techniques. For making a choice (NEA, 2013; p. 41) describes a number of strategies to handle uncertainties.

1. Addressing the uncertainty explicitly.
2. Demonstrating that the uncertainty is irrelevant to the safety assessment.
3. Bounding the uncertainty
4. Ruling out the event or process being uncertain
5. Using an agreed stylised approach.

In the safety assessment process used in OPERA, a scenario analysis is a key step as a preparation for the quantitative analyses[28]. From the viewpoint of strategy, it is recognised that it is uncertain how the disposal system will evolve. This uncertainty is addressed

[28] In OPERA, scenario analysis as part of the safety assessment for Boom Clay is treated in Task 7.1

explicitly (strategy 1) by developing a set of scenarios that each may be a description of the system evolution.

To keep the number of scenarios manageable, strategies 2 and 4 are applied in the process of identifying scenarios: irrelevant events and processes are screened out, and events and processes that are extremely unlikely are ruled out. In addition, for each scenario one or more assessment cases are formulated in such way that they are bounding cases for scenario uncertainties (strategy 3).

For some at present uncertain parameters it makes no sense to use a probabilistic approach. This is the case, for example, for the waste inventory. At present there is a relatively well known uncertainty about the waste inventory that will accumulate in the Netherlands for final disposal as of 2130. If this uncertainty were included in the probabilistic analysis, the results of this probabilistic analysis would lose their relevance in 2130 when the waste inventory is definitely known. In such cases it is better to establish some 'waste scenarios', for which separate analyses are performed. The same goes for all "technical parameters" relating to the facility design, the host rock and the geosphere.

The identification of the parameters that should be considered in a probabilistic investigation is not trivial and requires some expertise. Since a high number of probabilistically varied parameters require a lot of work for pdf quantification and a high numerical effort to perform sensitivity analysis, the number should be kept as low as possible. Parameters that allow a clearly conservative, but not over-conservative quantification can be left out of the probabilistic investigation, as long as there are no specific reasons to include them. Often, however, it is not fully clear whether a parameter will under all circumstances influence the model results in the same direction, so that a conservative value can be selected. In such cases it has to be assessed in an adequate manner whether it makes sense to include the parameter in the probabilistic analysis. A graphical screening can be very helpful in this task and is probably the better choice compared with other screening methods like Morris screening (Saltelli, 2000). For orientation a limited number of runs can be performed, nevertheless taking into account all questionable parameters. If no proper pdfs are known, uniform or log-uniform distributions between reasonable limits should be used. A CSM or mean rank plot will then normally show, which parameters are probably important and which do not have significant effects to the model output. The proper sensitivity analysis should then be restricted to those parameters that appear most important, if possible no more than about 10.

A well understood result of a sensitivity analysis is firstly a good basis for deepening the understanding of the system behaviour. A robust list of parameter sensitivities can improve the confidence in the model results. Secondly, it is important to know about the parameters that have a high influence to the model output. Further research can then be concentrated on reducing the most important uncertainties.

Based on the presently available information and knowledge, roughly the following steps are foreseen for the uncertainty analysis process:

1. Establish a set of scenarios
2. Establish for each scenario a number of assessment cases (i.e. various inventories, focus on specific waste types)
3. Establish models and (conservative) parameter values for the assessment cases
4. For each assessment case, identify the parameters that are of interest for the probabilistic process
5. Establish pdf's for these parameters
6. Perform the analyses

7. Evaluate, using one or more of the techniques described (e.g. regression- and correlation-based methods, or rank transformation, etc.)

A careful evaluation of the results of the uncertainty and sensitivity analysis is required to show the robustness of the conclusions that can be drawn from the safety assessment. The work performed as part of the PAMINA project (see Section 5.5 of the present report) would be a good starting point to proceed further.

5.9.4. Final considerations

As described in the present section, in the past several safety assessments have been carried out in the Netherlands as part of several national programmes as well as in projects performed for the EU Framework Programme. The various studies were executed taking different disposal concepts in rock salt into account, and assuming different waste inventories that varied substantially between the different studies.

Roughly said the following statements would apply to the topic of safety assessment of a geological disposal facility in rock salt:

- The safety assessments performed in the past applied waste characteristics that differ significantly from the presently foreseen inventories and compositions to be finally disposed of in a geological facility.
- The evaluation of FEPs is judged to be in a mature state and has been applied successfully for the various generic disposal concepts.
- The methodology of scenario development, both for the normal evolution scenario and for alternative evolution scenarios, on the basis of the screening of FEP databases is quite well established.
- In a next step of the topic of safety assessment of a salt-based geological disposal facility the evaluation of FEPs and the further development of scenarios should be focused more to a fixed repository design that would be able to accommodate the presently foreseen waste characteristics, and at a later stage to a specific site.

6. Integration of safety arguments

6.1. Objective and scope

The Safety Case component “Integration of safety arguments” is elucidated in IAEA SSG-23 as follows (IAEA, 2012; p. 38): *“The safety case should provide a synthesis of the available evidence, arguments and analyses. These should explain how relevant data and information have been considered, how models have been tested, and how a rational and systematic assessment procedure has been followed. The safety case should also acknowledge any limitations of currently available evidence, arguments and analyses, and should highlight the principal grounds on which a judgement has been made that the planning and development of the disposal system should nevertheless be continued. The safety case should include the approach by which any open questions and uncertainties with the potential to undermine safety will be addressed and managed. If the evidence, arguments and analyses do not provide sufficient confidence to support a positive decision, then the safety case, the facility design or even the disposal concept may need to be revised.”*

As no new safety assessment calculations have been performed in the OSSC project the safety arguments as stated in OPLA and CORA have been reviewed and compared with current insights in process understanding of relevant safety functions of the disposal design.

6.2. Safety assessment methodology and results

To be able to evaluate the calculational results of the main studies performed in the Netherlands, OPLA and CORA, and integrate the common aspects it is necessary to start with the main similarities and differences between the two safety studies. An important similarity between the studies is the general methodology. So the similarities and differences can be discussed based on the different steps in the methodology:

i) scenario selection
ii) determination of the probabilities of the scenarios
iii) determination of the calculational model
iv) determination of the parameters and their probabilities
v) dose rate calculations
vi) complementary indicators
vii) sensitivity and uncertainty analysis.

6.2.1. Scenario selection

The main question to be answered with respect to the identification of future developments of the disposal system, i.e. the scenarios, it to verify whether the most relevant scenarios and those leading to the highest dose rates to the population are covered, and no relevant scenarios are ignored, neglected or forgotten.

The scenario selection in **VEOS** was mainly based on expert judgement (Prij, 1987). In this judgement the waste was the starting point and possible release mechanisms and transport ways were imagined. This judgement was based on a list of phenomena which were considered to be somehow involved in the release of radionuclides from the waste during its migration or diffusion, and into human exposure. The documentation of the method of selection was rather poor as is usual with expert judgement based methodologies in which subjective elements are dominant. Most attention was focused on the identification of the scenarios and the most important processes. Less attention was focused in describing why the other features, events and processes were not relevant. The scenario selection process in VEOS identified 11 scenarios, which have been discussed in Section 5.2.1 of this report.

In the scenario selection in **PROSA** (Prij, 1993; Ch. 2) a more systematic method has been developed. The main elements in this methodology are (see also Section 5.3.1 of this report):

- a comprehensive list of FEPs, although it was recognized that probably not all possible FEPs have been taken into account,
- the subdivision into primary (barrier state related) FEPs and secondary (transport and nuclide related) FEPs,
- an evaluation of each FEP, to identify its relevance for the Dutch situation, and at what part of the multi-barrier-system it could have a potential effect,
- the introduction of a limited number of multi barriersystem states, taking into account (1) the engineered barriers, (2) the isolation shield, and (3) the overburden.

In the screening process and classification of the FEPs some subjectivity remains, which however has been documented properly. In the identification of the combinations of primary FEPs for each multi barrier system state also some subjectivity could not be avoided. The PROSA screening process resulted in identifying 22 different scenarios. The methodology in PROSA did lead to the scenarios previously identified in VEOS and some new ones related to glaciation and gas generation.

The research performed in the framework of **CORA** was devoted to retrievable disposal concepts. The performance assessment was primarily focussed on the so-called 'neglection scenario' (see also Section 5.4.1). The FEP '*deserted unsealed repository*' formed the basis for that scenario.

A more recent development in scenario selection is to identify combinations of FEPs that could affect the safety functions instead of each multi-barrier-state. This methodology can be regarded as an extension of the PROSA methodology, as has been adapted recently in Germany (Bollingerfehr, 2013; p. 62), and also in OPERA for defining scenarios in a clay based repository (Grupa, 2014; p. 13). A requisite of that methodology is that a set of safety functions is defined for the final disposal of radioactive waste in a salt-based repository. This has been done in Germany (Bollingerfehr, 2013; Section 2), but not yet in the Netherlands (neither in the US).

Based on the evaluation of the different salt-based disposal Safety Cases it is therefore recommended to define safety functions for the Dutch concept in rock salt, and to re-visit the PROSA methodology of scenario development by evaluating the impact of FEPs on safety functions instead of multi barrier system states.

6.2.2. Probabilities of the scenarios

Nor in VEOS, PROSA, or CORA the probabilities of the scenarios have been determined. In these studies it is argued that, due to the low maximum exposure resulting from each of them, it is not needed to spend much effort on this task.

In VEOS some probabilistic aspects of the scenarios were considered for the two scenarios leading to an exposure which was higher than the adopted limit (0.1 mSv/a). These scenarios were the 'diapirism into the biosphere being a polar desert' and the 'reconnaissance drilling'. For both cases not the probability of the scenario was calculated but the probability of exposure given that the scenario occurs. With the methodology in VEOS it was difficult to do more in that respect.

In PROSA the introduction of the multi barrier system states concept gave an opportunity to determine the probability of the groundwater intrusion scenario. That opportunity, however, was not used due to the extreme large reduction of the exposure resulting from the more realistic modelling of the convergence and crushed salt compaction.

6.2.3. Calculational models

In the three Dutch studies the calculational model contains three compartments:

- the salt compartment,
- the groundwater compartment, and
- the biosphere compartment.

In VEOS the three compartments were analysed separately and by three different institutes, and using three different computer programmes. The release from the salt compartment was calculated at ECN, the transport with the groundwater was calculated at RIVM/LBG, and the transport in the biosphere compartment and the ultimate exposure was calculated at RIVM/LSO. In PROSA and CORA the analyses were performed at ECN with one integrated code covering all three compartments.

The modelling of the three compartments in PROSA and CORA differs substantially from the ones used in VEOS.

Salt compartment

The model for salt compartment was based on EMOS which was also used in VEOS but has been modified for two types of scenarios.

For the *groundwater intrusion scenarios* the main difference with VEOS is the model for convergence and compaction of backfilled openings. The models used in PROSA and CORA for these processes differ from the one used in VEOS regarding the following points:

- Transient free borehole convergence was accounted for while the VEOS model was based on stationary free convergence;
- The compaction behaviour of the backfill was explicitly taken into account while the VEOS model used an implicit and fixed backfill resistance;
- The change in compaction due to flooding was accounted for with relations derived from experiments while the model in VEOS uses a simple reduction factor;
- The modelling of the transition from permeable to impermeable salt plugs was significantly improved, by introducing a model for the convergence driven compaction of the salt plug from an initial porosity to a final porosity;
- The theoretical basis was better understood;
- Well- defined experiments supported the modelling improvements (i.e. the BAMBUS tests).

For the *subrosion scenarios* the determination of the release from the salt formation is modelled differently. In PROSA the process of salt dissolution and glass leaching is modelled simultaneously. For the glass dissolution it has further been assumed that the glass consists out of a finite number of spheres with a statistically distributed size. This is considered to be better than the approach followed in VEOS. Here it has been assumed that the leaching rate is much higher than the subrosion rate which implies that the nuclide release from the glass is only determined by the subrosion rate. In PROSA the subrosion rate depends on the depth of the top of the salt formation whereas in VEOS it is a constant value.

In CORA the subrosion scenarios were not analysed.

From the different modelling efforts executed in the Netherlands it appeared that the variation in modelling of the salt compartment for the subrosion scenario did hardly affect the calculated results. As a consequence it can be stated that the present knowledge of modelling the salt compartment sufficiently acknowledges the long-term safety-related issues as far as the subrosion scenario is concerned.

From the different modelling efforts executed in the Netherlands it appeared that the variation in modelling of the salt compartment for the groundwater intrusion scenario significantly affects the calculated results. As a consequence it can be stated that a further study into these models is worthwhile to confirm the validity of the recent models.

Groundwater compartment

In **VEOS** the nuclide transport from the salt formation into the biosphere has been analysed with the solute transport code METROPOL-4. The analyses were performed for a 1-D flow tube and the travel times were 'conservatively' selected from large sets of particle velocities. The thickness of the overburden was kept constant in the analyses except for one case where the thickness was reduced to account for some effect of diapirism and erosion.

In **PROSA** the nuclide transport from the salt formation has been calculated with the code MASCOT, a 1-D code with a numerical integration of a closed form analytical solution. This approach has been selected to be able to perform the probabilistic analyses with PANTER. It has been verified that the results of MASCOT were comparable with the METROPOL results.

From the different modelling efforts it appeared that the variation in modelling of the ground compartment did hardly affect the calculated results. As a consequence it can be stated that the present knowledge of modelling the groundwater compartment sufficiently acknowledges the long-term safety-related issues.

In OPERA the groundwater compartment and modeling aspects are being addressed in WP6.2: *Radionuclide migration in an aquifer*. The output of these efforts may also be revisited for a salt-based repository.

Biosphere compartment

The biosphere compartment receives its “input” in the form of radionuclides released from the underlying salt formation and overburden including any aquifers. The biosphere compartment therefore cannot necessarily be seen as a separate, stand-alone feature which is decoupled from the other compartments of the disposal system, including the salt host rock: salt-specific features, events and processes may influence the biosphere characteristics and consequently radionuclide transfer in the biosphere (cf. Chapter 5).

In **VEOS** the biosphere was modelled with the BIOS code, whereas in PROSA the biosphere is simply modelled with dose conversion factors. This approach is followed to be able to perform the probabilistic analyses. The dose conversion factors are randomly selected from a statistical distribution which was calculated with a large number of runs with the MiniBIOS code. The codes MiniBIOS and BIOS are strongly related to each other. They use the same kind of biosphere, exposure pathways etc.

In both **VEOS** and **PROSA** the nuclides enter the biosphere via a river. A reason for this entry into the biosphere is the compatibility between VEOS and PROSA. It further is considered to be consistent with the assumption that the input flow of the radionuclides in the biosphere is the sum of all output flows from the different flow tubes in the overburden.

In **CORA** the methodology applied in PROSA has been followed (Grupa, 2000; Appendix C). It has been assumed that all radionuclides which are transported with the groundwater are released to the surface in a single river, and that that river water will be used for a small farming community.

In OPERA the biosphere characteristics and modeling aspects are being addressed in WP6.3: *Radionuclide migration and uptake in the biosphere*, and Task 7.2.3: *PA model for* *radionuclide migration and uptake in the biosphere*. The output of these efforts may also be revisited for a salt-based repository.

As part of the German project VSG, additional exposure pathways are added (Bollingerfehr, 2013; p. 111):

- Unintended ingestion of soils,
- Inhalation of resuspended contaminated soil particles,
- Uptake of contaminated soil by cattle and
- external radiation by dwelling on areas irrigated with contaminated water and in buildings erected with contaminated materials.

The effects of these additional exposure pathways proved to be minor.

From the different modelling efforts it appeared that the variation in modelling of the biosphere compartment does not significantly affect the calculated results. As a consequence it can be stated that the present knowledge of modelling the biosphere compartment sufficiently acknowledges the long-term safety-related issues.

6.2.4. Parameters and their probabilities

In VEOS single values for all parameters were selected. In most cases the values were selected with the intention to overestimate the exposure in the scenario. For some parameters the best estimate value were chosen.

In PROSA for most parameters the best estimates have been used as well. For parameters, of importance in the dose calculation and with a large uncertainty, probability density functions have been selected, both for the salt compartment and the groundwater compartment (Prij, 1993; Section 6.4). For the biosphere compartment the dose conversion factors were calculated with MiniBIOS in which a set of input parameters were randomly selected from a statistical distribution.

In **CORA** a similar approach as in PROSA was adopted (Grupa, 2000; Appendix C). Several parameters and processes were however re-assessed and implemented in the EMOS code, e.g. (Grupa, 2000; Sections 4.4, 4.5):

- convergence by the creep of rock salt
- compaction of salt plugs (dry compaction and wet compaction)
- advective flow of brine
- permeability of salt plugs (percolation model of Peach)

In addition, in order to enhance the retrievability of the waste containers, CORA adopted the METRO-I design, consisting of short horizontal boreholes with the capability to hold a single container per borehole. (Grupa, 2000; p. 133).

From the different modelling efforts performed in the past it can be concluded that presently sufficient knowledge exists about what parameters need to be addressed in a safety assessment. Obtaining numerical values of the parameters however is not always straightforward since quite a number of relevant parameters depend on the temperature, are coupled through THMC processes, or are site-specific. As a consequence, the determination of numerical values of parameters needs ultimately to be adapted to the in-situ conditions of a specific site and the prevailing boundary conditions.

6.2.5. Safety criterion: dose rates

In **VEOS** dose calculations have been performed for 21 different disposal concepts. These concepts are based on three generic formation types (salt dome with 230 m thick overburden; salt pillow with 800 m thick overburden; bedded salt at a depth 1200 m), three disposal techniques (mine with boreholes for the high activity waste HAW and mined chambers for the remaining waste; deep boreholes drilled from the surface for the HAW and dry caverns for the remaining waste; deep boreholes drilled from the surface for the HAW and wet caverns for the remaining waste), and three waste strategies:

- Strategy A: waste from 15 GWa nuclear energy with 50 year of interim storage;
- Strategy B: waste from 105 GWa nuclear energy with 50 year of interim storage;
- Strategy C: waste from 105 GWa nuclear energy with 10 year of interim storage.

The doses calculated in **VEOS** revealed that for all waste strategies and considered scenarios, except the human intrusion scenario reconnaissance drilling, the dose rates are significantly below the natural background radiation in the Netherlands of maximal 3 10^{-3} Sv/a (OPLA, 1989; p.92). For the reconnaissance drilling scenario receiving a high potential dose rate is however extremely unlikely due to the very small probability of occurrence, and the conservative assumptions made for this scenario. The main conclusion of the VEOS safety assessment was that the final disposal of radioactive waste in rock salt could very unlikely lead to a future dose to the population (Köster, 1989; p.58).

In **PROSA** dose calculations have been performed for 4 disposal concepts. The disposal techniques with deep boreholes drilled from the surface and the interim storage of 10 years have not been considered in PROSA.

Similar as for the VEOS safety assessment calculations, the PROSA calculations indicated for all waste strategies and considered scenarios, except the human intrusion scenario reconnaissance drilling, the dose rates were again significantly below the natural background radiation in the Netherlands. For the reconnaissance drilling scenario receiving a high potential dose rate is however extremely unlikely due to the very small probability of occurrence, and the conservative assumptions made for this scenario.

The project **EVEREST**, carried out in the framework of the EU programme, was an extension of the work in PROSA and comprised a probabilistic sensitivity and uncertainty analysis of the groundwater intrusion scenario. The analyses yielded the maximum and mean dose rates of respectively 4 10^{-8} Sv/a and 1 10^{-9} Sv/a (Prij, 1995; p. 7.3.7.34). These results were in good agreement with results of the deterministic analysis performed in PROSA where the maximum dose rate was found to be 2 10^{-9} Sv/a (shallow dome, waste strategy B; Prij, 1993; Tabel 7.10).

In **CORA** only doses for the neglection scenario have been determined. It appeared that the maximum dose rate was 2 10^{-9} Sv/a. This result is determined assuming that the waste reaches the surface through ground water flow. However, it will be necessary to study some variants of this transport pathway, e.g. by assuming that a well produces water from the ground water close to the facility, so the well water contains traces of the waste.

In **Germany** the calculation of the radiological consequences in the biosphere taking into account the complete migration pathway (incl. the geosphere outside the CRZ) is required if the simplified approach results in a RGI > 1 (i.e. stage 3 of Figure 3-6). However, in the project VSG all calculated RGI values for the liquid phase were lower than 1 and thus no calculation for the consequences in the biosphere after transport through the geological barrier outside the CRZ were carried out. The calculations carried out for a generic repository in the ISIBEL project (Buhmann, 2010) show that dilution effects in the geosphere yield exposures several orders of magnitude lower than the exposures calculated at the boundary of the CRZ.

An **evaluation** of the results showed long interim storage to be favourable in terms of calculated dose rates because any finally disposed waste encountered unintentionally by man would be less active (Prij, 1993; p. 7.24). On the other hand there are also risks associated with long surface storage which have to be considered.

On the basis of the calculated dose rates it was also concluded in PROSA that the possibility of human intrusion does not lead to a preference for a particular disposal technique.

With respect to the type of formation to be used for radioactive waste disposal, it was felt that storage at greater depths is the most important factor for reduction of the effects of human intrusion. Disposal at great depths might also reduce the probability of human intrusion.

Because the human-intrusion scenarios were associated with the highest exposure rates and possibly the highest risk, these scenarios should be the subject of further research. In all other cases/scenarios calculated dose rates for salt-based repositories are far below any regulatory limits.

6.2.6. Complementary indicators

Complementary indicators can avoid to some extent the difficulties faced in evaluating and interpreting doses and risks that are expected to occur in the future. In particular, individual human behaviour and near-surface processes, which are important factors in the calculation of dose and risk, are difficult or impossible to predict over long time scales. In contrast the possible evolutions of a well-chosen host rock and geological site can be bounded with reasonable confidence over much longer time scales of up to about one million years into the future (depending on the site). Hence, there is a trend in some recent safety cases towards evaluating indicators in addition to dose and risk, which show more clearly the repository´s intrinsic performance without requiring any assumptions concerning the surface environment and the biosphere. The use of such indicators may support the statement that radionuclide release to the surface environment will be minor and of low consequence and, thereby, increase the robustness of the safety case.

An important requisite is that complementary indicators are to be considered that specifically address issues related to disposal in salt. Section 3.8.1 of the present report provides a summarily overview of these indicators, viz. the safety indicators, performance indicators, as well as their reference values, if applicable.

In comparison to a safety indicator, a performance indicator provides a measure of the behaviour of an individual repository component or sub-system. For this reason, performance indicators are usually more concept or site-specific than safety indicators. Multiple performance indicators may be applied in a safety case and they could be used to evaluate performance of the disposal system barrier-by-barrier, and to determine what redundancy and performance ‘head room’ is available in the repository design. Various performance indicators have been used or proposed, and they typically relate to such things as the containment times provided by individual barriers or the migration rate (flux) of radionuclides across them. Such performance indicators are not specifically related to disposal in rock salt (Morris, 2009).

As already stated in Section 3.8.2 a difficulty with the use of safety indicators (and also performance indicators) lies in the derivation of appropriate reference values. There is a trend, however, towards using site-specific reference values, such as local or regional groundwater concentrations, because these are often considered to provide the most relevant situational context.

CORA In addition to the main indicators considered in the Netherlands, i.e. dose rate and risk, there has been a focus on the self-sealing behaviour and “closure times” of plugs and sealing materials of boreholes of salt-based repositories (Grupa, 2000; Poley, 2000a). The "closure times" of plugs and seals in a salt-based repository are defined as the times for which compacted salt reaches the percolation limit (1% porosity), for which the possible fluid flow paths in the compacted salt are cut off through the ongoing compaction process (Grupa, 2000; p. 95).

The CORA safety assessment (see also Section 5.4) concentrated on the neglection scenario, assuming an open flow path through the flooded galleries to the groundwater system. Also in that case the calculated dose rate was negligible under the adopted assumptions is negligible.

PAMINA

The behaviour of plugs and seals and the assessment of safety and performance indicators has been elaborated systematically for a model of a salt-based repository as part of the EU-FP6 project PAMINA (Schröder, 2009). That model is essentialy similar to that adopted in CORA (Grupa, 2000). The probabilistic uncertainty analysis in PAMINA focused on the closure behaviour of the compacted salt plug that seals the disposal cell of a generic repository design in rock salt. A selection of safety and performance indicators has been assessed in a probabilistic assessment, using statistical techniques.

The indicators have been sub-divided in "*point values*" and "*time-dependent values*", as summarized in Table 6-1 (Schröder, 2009; p. 87). The point values represent a certain value of the indicator at one point in time, e.g. a maximum value or a total amount. The time-dependent values may vary as a function of time. The "dry" scenario, in which no presence of brine in the repository has been assumed, refers to the normal evolution in a salt-based repository. In the "wet" scenario it has been assumed that the repository has been flooded with brine that may serve as a transport medium for radionuclides.

Table 6-1 Indicators of a salt-based repository analysed in PAMINA.

Point values	unit	"dry" scenario	"wet" scenario
A.1 Closure time of the plug	[year]	x	x
A.2 Saturation time of the plug	[year]		x
A.3 Time when pressure drop equals stress in plug	[year]	x	
A.4 Maximum dose rate in the biosphere	[Sv/year]		x
A.5 Total amount of brine pressed out of disposal cell	[m^3]		x
Time dependent values			
B.1 Porosity of the plug	[-]	x	x
B.2 Permeability of the plug	[m^2]	x	x
B.3 Strain rate divided by stress as function of porosity	[1/MPa·year]	x	x
B.4 Dose rate in the biosphere	[Sv/year]		x
B.5 Amount of brine pressed out the disposal cell	[m^3/year]		x
B.24 Radiotoxicity in the disposal cell	[Sv]		x
B.25 Radiotoxicity in the center field	[Sv]		x
B.32 Radiotoxicity flux from disposal cell to center field	[Sv/year]		x
B.33 Radiotoxicity flux from center field to geosphere	[Sv/year]		x

Since in the dry scenario no contaminated brine enters the biosphere, the dose rate was only calculated for the wet scenario. From the 1000 realizations of this output variable in the wet scenario, it appeared that the distribution of the maximum dose rate in the biosphere has a small dispersion (the ratio between the 99% percentile and the 1% percentile was less than 2) and that it was spread around $6.8 \cdot 10^{-13}$ Sv/year (Schröder, 2009; p.92), which is next to negligible.

Germany - VSG

An example of a complementary *safety indicator*, analysed in the VSG project, is the contribution to radiotoxicity in groundwater, which was calculated as the radiotoxicity flux for a repository in rock salt at the CRZ, and is illustrated in Figure 6-1 (Noseck, 2012; p. 74). The study performed in (Noseck, 2012) showed that, if an appropriate calculation scheme and an accepted yardstick are applied for the calculation of this indicator (for example by a given scheme from the regulator) this indicator gives a strong argument for the safety of the repository system independent of the barrier function of geological formations outside the CRZ.

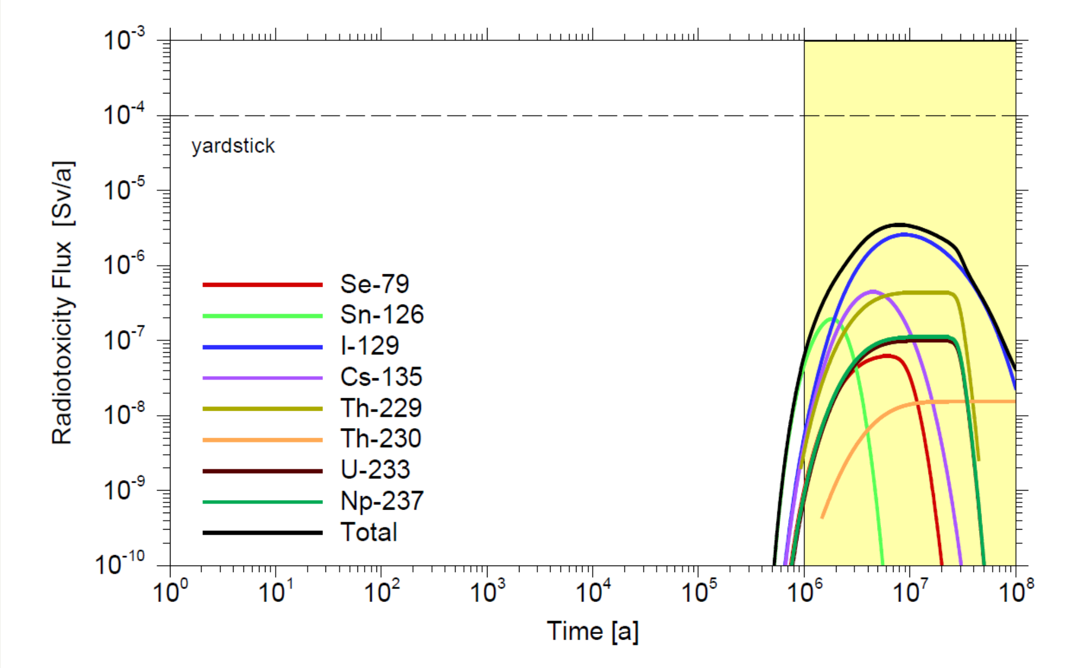


Figure 6-1 The contribution to radiotoxicity at the CRZ for a HLW repository in rock salt.

As an example of a *performance indicator* the evolution of the radiotoxicity inventory in the different compartments of a repository in salt for a brine intrusion scenario, as calculated by GRS is shown in Figure 6-2 (Becker, 2009; Section 8.1). In principle this figures shows the “containment” of radionuclides in the subsequent compartments of the repository in a single plot and gives a good indication of the confinement provided by the integrated repository system. The results calculated for this particular case indicate that, from the original inventory a part of only about 10^{-9} may reach the biosphere within one million years.

A comprehensive overview of performance indicators that may be applied in a safety assessment is provided in PAMINA Deliverable 3.4.2 (Becker, 2009).

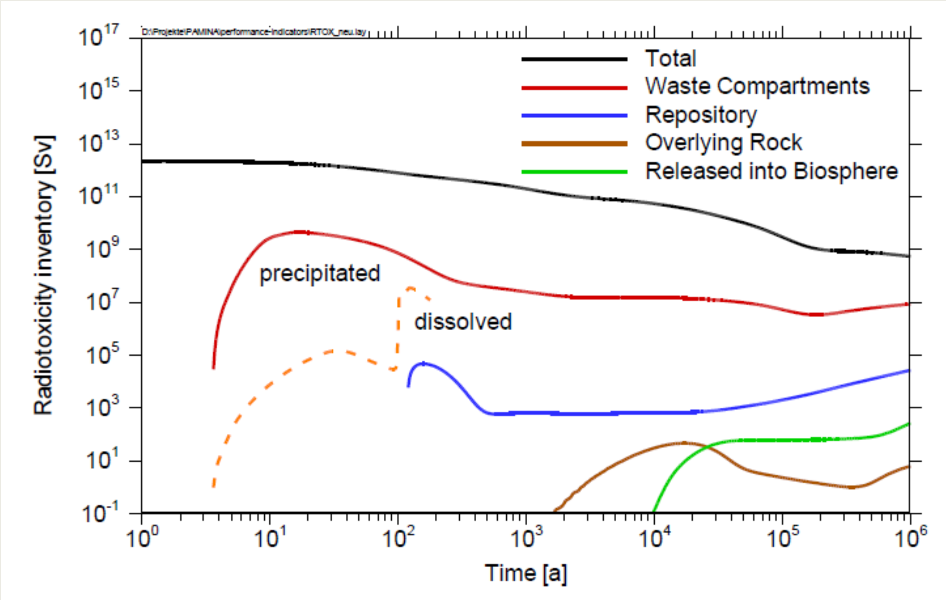


Figure 6-2 Radiotoxicity inventory in different compartments of a repository in salt.

In (Wolf, 2008) appropriate reference values were identified by determining natural background values and considering a safety margin by constraining the reference value to about one third of the natural background value. For the effective dose rate and the radiotoxicity concentration in the biosphere water two global (regional) reference values were determined. The radiotoxicity flux from the geosphere requires a local site-specific reference value. The general philosophy of this procedure was to keep the reference values comparatively low in order to enhance the confidence in the safety statement given by the corresponding safety indicator. For both case studies the following reference values were identified (Wolf, 2008; p. 83):

- for the effective dose rate: $1 \cdot 10^{-4}$ Sv/a,
- for the radiotoxicity concentration in the biosphere water: $2 \cdot 10^{-6}$ Sv/m^3, and
- for the radiotoxicity flux from the geosphere: 0.1 Sv/a.

In (Wolf, 2008; p. 89) it was recognized that a key problem associated with safety indicators is the determination of appropriate reference values, based on reliable data, in order to give solid safety statements. In particular, the derivation of the reference value for the radiotoxicity flux from the geosphere is subject to high uncertainties. This indicator is to a large extent site-specific. More work is necessary to derive a conclusive reference value for a particular considered site.

6.2.7. Sensitivity and uncertainty analysis

In **VEOS** the sensitivity analyses were performed in a deterministic way and mainly restricted to the release from the salt formation during the groundwater intrusion scenarios. The sensitivity of the other compartments and scenarios has been performed in a qualitative way or was based on some global and simplified analyses. In VEOS no systematic uncertainty analyses have been performed.

In **PROSA** the sensitivity analyses have been performed in a more systematic way. The release from the salt formation during the groundwater intrusion scenario was performed in a deterministic way and concentrated on the sensitivity of the model for convergence and compaction of a backfilled opening. The sensitivity of the results of the subrosion scenarios has been analysed in a probabilistic way.

For that scenario the uncertainty caused by the uncertainty in the data was also analysed. Due to the approach followed in PROSA it could be made clear that the internal diapirism rate is a much more sensitive parameter than the subrosion rate.

In the framework of **EVEREST** the sensitivity and uncertainty in the results of the groundwater intrusion scenario has been performed in a probabilistic manner. It could be concluded that the uncertainty in the properties of crushed salt, whereas the uncertainty in the properties of the rock salt are a weak source for the uncertainty in the exposure. Furthermore, it was shown that the uncertainty in the groundwater velocity is a strong source, whereas the uncertainty in the biosphere dose conversion factor and the time of groundwater intrusion are a weak source for the uncertainty in the maximum exposure.

In the framework of **CORA** some attention has been given to rational systems that would enable the development of new scenarios systematically. This has led to some improvements in the existing system. However, the system for scenario development is still incomplete.

In CORA also some attention has been given to an analysis of the uncertainty in the modelling of the permeability of salt with low porosities. This had been performed in a deterministic way.

Additional efforts on this topic were undertaken in the **PAMINA** project (Schröder, 2009). In relation to the quality and reliability of the science, it was concluded (Schröder, 2009; p. 103) that the use of safety assessment tools for sensitivity analysis has certain limitations. The calculations carried out for the PAMINA study could be performed in an acceptable time period because (1) the repository set-up chosen was relatively simple and (2) the number of variable input parameter was low. However, for more elaborated repository systems and more extensive lists of variable input parameters, calculation times may be a limiting factor. The same is true for the required data-storage capacity. In case of a more complex system analysis other data post-processing strategies may solve at least part of the problem of resources.

The handling of the massive amount of output files needed special (external) program routines to extract, interpolate and summarize all data. The development of tailor-made data post-processing tools may therefore be an essential part to handle the outcomes of the calculations.

6.3. Implications for the design choices/site selection

In the performance assessments, described in more detail in Chapter 5 of the present report, we have seen that three main classes of scenarios have been identified in PROSA:

- The diapirism/subrosion scenarios: a first test of the performance of the geological isolation;
- The groundwater intrusion scenario: a check of the performance of the engineered barriers;
- The human intrusion scenarios: a check of the geological isolation system.

The next sections evaluate, when applicable, the iterations of these scenarios during the various salt disposal research programmes in the Netherlands.

6.3.1. Geological system

The diapirism/subrosion scenarios represent the response of the salt formation and its environment on the natural processes. The results of the performance assessment give an indication of the isolation capacity of this geological system. The sensitivity results can help to identify possibilities for design improvements and to support criteria for site selection.

The following can be observed:

i) One first should show that the construction of the GDF and the disposal of the waste does not significantly influence the initial isolation capacity of the geological formation. The engineered barrier should be such that this influence is limited and bounded in time; see below.

ii) Then, one should show that the disposal of the waste does not significantly influence the initial isolation capacity. A first attempt on this matter has been given by Prij (Prij, 1991; pp. 172-177), presenting a thermo-mechanical model for the semi stationary state of halokinesis of a salt formation. That model has been used to determine the sensitivity of the rising rate to changes in temperature and stresses caused by the disposal of heat generating waste. The conclusions were:

   a. The rising rate of the salt formation is influenced by the heat effects during a period of about 1000 years, which is a geologically short period;

   b. The temperature rise itself may cause a maximum increase of the initial rising rate of 5% whereas the thermal stresses reduce the rising rate by less than 6%.

iii) The results of the sensitivity analyses have shown that the uncertainty in the internal rise rate and the groundwater velocity give the largest contribution to the uncertainty in the exposure and thereforein the isolation capacity of the geological system. The design and site selection should help in reducing the uncertainty in these parameters. One hereby must keep in mind that the release of nuclides from the salt formation, the transport through the overburden and the migration and exposure in the biosphere occurs in a very far future, more than 0.5 Ma after disposal.

iv) The parameter having the highest influence on the dose rate is the internal rise rate of the salt formation. This uncertainty in this parameter can be influenced to a certain extent by the selection of the site. The selection, however, is not very easy because a prediction must be made of the magnitude of the internal rise of the salt formation in the near and far future. This must be done on the basis of data from the past. A beneficial, which is a low internal rise rate in the past 1 Ma does not automatically imply a low internal rise in the next 1 Ma. The same holds for the high rise rate in the past, does not guarantee a high rise rate in the near and far future. Detailed numerical analyses of the halokinesis of the salt formation is needed to accurately predict the values of the internal rise rate for the near and far future. In these analyses the tectonic stresses must be taken into account (Cloetingh, 1993).

v) The sensitivity analyses have shown that uncertainty in the (hydrological) properties of the overburden significantly contribute to the uncertainty in the isolation capacity of the geological system. This however hold primarily for the properties in the very far future. It is not straight forward to select a site in such a way that these uncertainties are significantly reduced.

vi) As the properties of the overburden also determine how easy the GDF can be constructed and operated this gives a easier criterion for site selection.

vii) The depth of the GDF is a parameter that is not varied in the analyses. The fact that the internal rising rate seems to be a very sensitive parameter also means that the disposal depth is an important parameter. The deeper the repository, the longer it takes for the nuclides can be released and the lower the exposure.

viii) It is interesting to compare these findings with the rather detailed selection criteria formulated in 1979 (ICK, 1979). It seems for sure that a formation selected according to these criteria has almost no uncertainty but it is questionable that such criteria can be fulfilled. Some more work has to be done to reformulate the criteria with the goal to reduce the uncertainty in the rising rate and the properties of the overburden.

6.3.2. Engineered barriers

Although the groundwater intrusion scenario is a hypothetical one and the design of the GDF should made so that the probability of groundwater intrusion is low the results of the groundwater intrusion scenario can be considered as a test of the performance of the engineered barriers. The choice for the backfill and the designs of the seals and dams can be checked with performance assessments of a groundwater intrusion scenario.

The following observations can be made:

i) The shaft and galleries are the main disturbances of the virgin salt formation due to the construction of the GDF. It has to be shown that the backfilling of these openings is such that this disturbance is bounded in time. It has been shown in theoretical analyses and experiments that this can be performed by backfilling with crushed salt. Due to the creep driven convergence of the excavation the crushed salt will compact and the porosity and permeability decrease from initial values to final ones which are equal to that of the impermeable salt. So ultimately the unavoidable disturbance of the salt formation due to the mining activity will be 'repaired'.
ii) The convergence of openings in salt formations has been analytically and experimentally studied in OPLA (Prij, 1991b).
iii) The behaviour of dry and wet crushed salt was studied at the RUU (Utrecht State University) and in PROSA the models for convergence and compaction have been incorporated in EMOS_ECN (Schröder, 2009; Ch. 5).
iv) The sensitivity analyses performed in EVEREST confirm the finding that the properties of the crushed salt are dominating the flow behaviour in the engineered barriers.
v) In CORA an attempt has been undertaken to better understand the transition from permeable to impermeability of crushed salt.
vi) In PAMINA it has been shown that, in an unflooded repository, a small amount of moisture is extremely helpful in accelerating the compaction and ultimately reaching the undisturbed state in a relatively short period, viz. < 800 years (Schröder, 2009; p. 105).

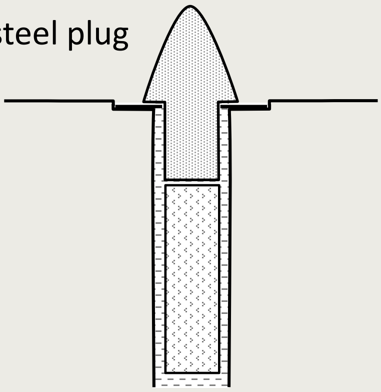
An crucial aspect in the design and construction of shaft seals is that the successive layers of the seal system must be adapted to the neighbouring geological environment. This implies that a shaft seal is a site-specific feature of the disposal system, and can only be detailed if a site has been established.

6.3.3. Geological disposal system and human intrusion

The human intrusion scenarios can be considered as a final check of the total geological disposal system. The analyses performed sofar have shown that a large dose can be encountered when in the future human beings are trying to explore and use the underground. It is a challenging task to make design choices which reduce the probability that such an exploration takes place in the formation and location where the GDF is situated.

Approaches proposed to enhance safety in relation to human intrusion followed:

- In the ICK period it is suggested to place a cast iron cone (Hamstra, 1981; p. 40) at the top of a borehole to make it impossible that a reconnaissance drilling hits the HLW directly:



- In Germany the most promising methods to optimize the design within this respect are found to be (Bollingerfehr, 2013; p. 113):
  a. Dyeing of backfill or adding coloured material to the backfill and
  b. Placement of gravel in the openings on exploration level (requiring an increase in the distance between exploration and emplacement level)
- In the US extensive studies have been carried out to design landmarks above the repository. At the landmarks symbols should be place which warn the future being that dangerous material is stored at this location (US DOE, 2004; p. 1-10).
- In the US extensive measures have been identified to assure that the location of the GDF will be recognized en remembered:
  a. Placing of well-designed landmarks above the repository. At the landmarks symbols should be place which warn the future being that dangerous material is stored at this location.
  b. Storage of the information of the location of the GDF at many different archives in the world.

The human intrusion scenarios are to be distinguished from the others as exposure is a direct or indirect result of a deliberate human action in or close to the geological formation where the waste is stored. This sort of action can only occur supposing knowledge of the presence of radioactive waste has been vanished. Therefore in the dose calculations performed in VEOS (Prij, 1989; Section 6.5.3) and further assessed in PROSA (Prij, 1993; 7.25) it was assumed that these actions will not take place earlier than 250 years after discharge of the reactor fuel for reprocessing. In that way consideration has been given to a period of interim storage, a period of operating the GDF and a period of at least hundred years in which the knowledge of the GDF is retained through administrative measures.

Within the framework of the German VSG project the relevance of future human activities has been discussed in detail (Beuth, 2012b). That study deals with human activities after closure of the repository, which direct damage the CRZ or the technical barriers. Only those actions have been considered which are unintended, i.e. without knowledge of the presence of a repository and its hazards. In the VSG study a few stylized scenarios have been analysed using arguments instead of quantitative analyses. From the assessment of these scenarios, optimization measures to impede or prevent human intrusion could be derived. These measures may in general result e.g. in signals for the acting persons in the future, that there is a special situation in the deep underground, or in a reduction of the consequences, if the intrusion in the CRZ will not be recognized by the acting persons.

6.4. Robustness

In (NEA, 2004) the term robustness is described as a feature that is favoured by the multi-barrier concept, i.e. the concept of multiple barriers that operate in concert to isolate the waste, and prevent, delay and attenuate the potential radionuclide release to the biosphere. The barriers should be complementary, with diverse physical and chemical components and processes contributing to safety, so that uncertainties in the performance of one or more components or processes can be compensated for by the performance of others to a significant extent. A system based on the multi-barrier concept typically comprises the natural barrier provided by the repository host rock and its geological environment, and the engineered barrier system. Initially, a number of engineered components may, to some extent, be "over-designed" to avoid or mitigate the effects of early uncertainties.

IAEA SSG-23 states that the safety strategy should set out how robustness of the safety functions will be provided for and how the adequacy of such robustness will be demonstrated (IAEA, 2012; p. 26). A component of the disposal system may be considered robust if it will continue to fulfil its expected safety function(s) no matter what kind of perturbations may reasonably be expected to occur. The disposal system may be considered robust if it continues to provide adequate protection and safety under a wide range of conditions and scenarios that may reasonably be expected to occur.

Robustness can be implemented through the selection of engineered materials and their properties as well as by the selection of a suitable host rock at a sufficient depth.

Taking into account the above-mentioned descriptions, the term "robustness" refers to various aspects of the Safety Case, e.g.

- The safety concept, more specifically safety functions and multiple barriers;
- The location of the repository;
- The design of the repository, including the engineered barriers;
- The safety assessment including its methodology and comprehensiveness, e.g. the consideration of normal and alternative evolution scenarios;
- Verification that good engineering practices (demonstrability and feasibility) have been applied.

Robustness is a qualitative denotation, that cannot be judged in terms of indicator values, as it comprises several aspects of the safety case that are very different in scope and performance. A qualitative overview of the consideration of aspects of robustness in salt-based repository concepts in the Netherlands, Germany and US is provided in Table 6-2.

Table 6-2 Overview robustness in repository concepts.

Robustness Aspect	***Netherlands***	***Germany***	***US***
Safety concept	• Multiple barriers • No explicitly defined safety functions ("Isolation" mentioned in Grupa, 2000)	• Multiple barriers • Safety functions	• Multiple barriers • Safety functions
Repository location	N/A	Through site-specific Design Requirements[29]	Through a system of Regulations[30]
Repository design	Generic design; engineered barriers considered	Specific for Gorleben; engineered barriers considered[31]	Through a system of Regulations

[29] Bollingerfehr, 2013; Section 4.2
[30] E.g. 10 CFR 60 (US DOE, 2014)

Robustness Aspect	***Netherlands***	***Germany***	***US***
Safety assessment	Normal Evolution Scenario and Alternative Evolution Scenarios considered	Normal Evolution Scenario and Alternative Evolution Scenarios considered	Normal Evolution Scenario and Alternative Evolution Scenarios considered
Demonstrability	Several aspects treated in OPLA and CORA (cf. Section 3.10.1)	Safety demonstration concept (see Section 3.10.2)	By the construction and operation of WIPP[32]

6.5. Natural analogues

The use of multiple lines of reasoning may add value to the safety case by providing a range of different arguments that together build confidence in certain data, assumptions and results. Furthermore, certain arguments may be more meaningful to specific audiences (IAEA, 2013; p. 40).

One, if not the most important group of arguments are natural analogue aspects of the safety case. The main value of such studies is to provide information of the full complexity of the repository system and of the characteristic of processes over long time scales. It is not possible to simulate in laboratory studies the very long-term processes that might affect the safe performance of a repository. Therefore, natural, archaeological and industrial analogue studies are often used as one of several multiple lines of reasoning that, when combined, help to build understanding and confidence.

A thorough review of how existing studies can be used as analogues supporting important elements of the safety assessment and safety case for a repository in rock salt was performed in the German project ISIBEL (Summary provided in NEA, 2013b; pp. 87-97). The objective of the review and the assessment of Natural Analogues studies was to answer the following questions:

- For which aspects can Natural Analogues contribute to the assessment of safety?
- What is the status of the identified Natural Analogues?
- How does the Natural Analogues contribute to the confidence building of the Safety Case (communicability)?

Within the study (i) analogues for the integrity of the geological barrier, (ii) Analogues for the integrity of the geotechnical barriers, and (iii) analogues for release scenarios have been considered. The first two aspects are most relevant for the safety demonstration concept, since they are related to the two kinds of containment providing barriers. The first question was answered by starting a systematic review of the required key information needs of the Safety Case and an assessment if these requirements can be fulfilled by Natural Analogues information.

In order to answer the second and third question, the ISIBEL assessment considers two different schemes. The first scheme assesses the status of Natural Analogues for a Safety Case in salt. Five classes are distinghuished for this assessment:

++ Natural Analogue is identified and documented

\+ Natural Analogue is identified and need to be better documented

● Natural Analogue is identified (no documentation)

\- Natural Analogue is not identified

-- Natural Analogue is (probably) not identifiable

[31] Bollingerfehr, 2013; Sections 4.2, 5.1
[32] MacKinnon, 2012; p.13

The second scheme assesses the communicability of a Natural Analogue for the Safety Case in salt. Two classes are distinghuished for the assessment:

- oo Public confidence building: Natural Analogue is tangible and understandable to the lay stakeholder
- o Technical confidence building: Natural Analogue is an argument to support the understanding of complex behavior

The results for potential Natural Analogues studies are compiled in Table 6-3 and Table 6-4. In order to assess the situation at a potential site and to compare it to Natural Analogues studies, the exploration results of the Gorleben salt dome were used as self-analogue.

Table 6-3 Compilation and assessment of Natural Analogues (NA) for the geological barrier.

Nr.	Aspect / Study	Applicability in Safety Case (Key information)	Status Site	Status NA	Confidence Building
1	Occurrence of salt domes	Long-term stability of salt domes	++	+	oo
2	Neotectonic conditions	Occurrence of earthquakes and magmatic events	++	--	oo
3	Analysis of the salt flow	Uplift rates (halokinesis)	++	+	oo
4	Thickness and composition of the cap rock	Subrosion rates	++	+	o
5	Behaviour of competent salt formations	Possible water pathways	+	●	o
6	Br- (and Rb)-distribution in salt formations	Interaction between formation and external solutions	++	++	o
7	Chemical and isotope composition of fluid inclusions	Interaction between formation and external solutions and gases	++	++	o
8	Openings from salt mining	Behaviour of salt at disposal level	●	+	o
9	Basalt intrusions	Behaviour of salt at high temperatures	--	++	oo
10	Basalt intrusions	Sealing of fissures	--	++	o
11	Kryogenic fractures	Occurrences of fractures formed by salt contraction during cooling	-	+	o

Table 6-4 Compilation and assessment of Natural Analogues (NA) for geotechnical barriers.

Nr.	Aspect / Study	Applicability in Safety Case (Key information)	Status NA	Confidence Building
1	Investigations of bulkhead drift	Reduction of the permeability of an EDZ around drift sealing	+	o
2	Basalt intrusions	Long-term behaviour of basaltic gravel	++	oo
3	Chemical and mineralogical composition of natural clays	Impact of high temperatures on clay minerals	++	o
4	Properties of natural salt clays in salt	Long-term behaviour of clays as sealing material	+	o
5	Corrosion of historical concrete buildings	Long-term behaviour of cementitious materials	++	o

Nr.	Aspect / Study	Applicability in Safety Case (Key information)	Status NA	Confidence Building
6	Bentonites in saline environment	Long-term behaviour of bentonite as sealing material	+	o
7	Compacted backfill material from old drifts in salt mines	Compaction of crushed salt over long time scales	●	o

From the ISIBEL study on natural analogues it was concluded that for the integrity of the geological barrier a lot of well described natural analogues are available. Further potential natural analogues are identified but need to be better documented. Despite the vast experience in salt mining and gas storage in Germany, it is difficult to identify natural analogues for assessing the integrity geotechnical barriers. There are only a few publications on how to use this knowledge for natural analogues for the geotechnical barrier. This is especially true for the compaction of crushed salt.

To further compile and discuss about the status and potential usefulness of natural as well as anthropogenic analogues from different countries to be potentially used within Safety Cases for radioactive waste repositories in salt formations a workshop “Natural Analogues for Safety Cases of Repositories in Rock Salt” was performed recently (NEA, 2013b). The workshop proceedings provide a good overview on studies regarding integrity of salt host rock and engineered barriers as well as on microbial, chemical and transport processes, which might be used as analogues to support the safety case. The workshop also identified significant issues and processes in rock salt where analogue information might reasonably be expected to provide further insight, understanding and quantification. It was recommended to initiate further studies on compaction of crushed salt backfill, the viability of microbes in the near-field, stability of plugs and seals, deformation of anhydrite and isotope signatures in fluid inclusions.

6.6. Evaluation

A variety of aspects in relation to the topic of Integration of Safety Arguments have been treated in this chapter. The main observations and conclusions about these topics concerning their treatment in the Safety Case of geological disposal in rock salt in the Netherlands, Germany and US are summarized below.

Scenario selection and their probabilities

- In PROSA scenario selection was done by identifying combinations of FEPs that could affect each multi-barrier-state
- A more recent development is to identify (combinations of) FEPs that could affect the safety functions
- Safety functions for the Dutch concept in rock salt have not yet been defined. Consequently appropriate safety functions have to be defined first before the assaying of the effects of FEPs on safety functions can be done.

Evaluation of calculational models

- The variation in modelling of the salt compartment did not significantly affect the calculated results
- Consequently it can be stated that the present knowledge of modelling the salt compartment sufficiently acknowledges the long-term safety-related issues

Determination of parameters

- Sufficient knowledge presently exists about what parameters need to be addressed in a safety assessment

- The determination of numerical values of parameters needs ultimately to be adapted to the in-situ conditions of a specific site and the prevailing boundary conditions
- Especially the effects of elevated temperatures on parameter values needs to be addressed

Results of dose calculations

- On the basis of the calculated dose rates PROSA concluded that the possibility of human intrusion does not lead to a preference for a particular disposal technique
- With respect to the type of formation to be used for radioactive waste disposal, it was felt that storage at greater depths is the most important factor for reduction of the effects of human intrusion. Disposal at great depths might also reduce the probability of human intrusion.
- Because the human-intrusion scenarios were associated with the highest exposure rates and possibly the highest risk, these scenarios should be the subject of further research

Evaluation of sensitivity and uncertainty analyses

- The execution of uncertainty and sensitivity analyses significantly contributes to the understanding of processes affecting the long-term safety of a disposal system
- A full probabilistic safety assessment, taking into account uncertainties of all safety related parameters, has not yet been performed in the Netherlands

Comparison with safety criteria, Complementary indicators

- Safety and performance indicators are valuable tools to qualitatively assess the long-term safety of a geological disposal facility
- In the Netherlands only the calculated dose rate has been compared with established safety criteria
- The Radiologischer Geringfügigkeits-Index (RGI), applied in Germany, can be considered as a quantitative measure of the safety function “containment” in the containment providing rock zone, CRZ, of a salt-based repository
- The “RGI-concept” would also be applicable to the Dutch situation, but it should be adapted to an agreed disposal concept, an up-to-date Dutch waste inventory, and an updated safety assessment
- A difficulty with the use of safety indicators lies in the derivation of appropriate reference values
- There is a trend towards using site-specific reference values of indicators, such as local or regional groundwater concentrations. In the Netherlands siting of a disposal facility is not yet an issue, and therefore the use of groundwater concentrations as indicator reference values should be done with care.

Robustness

- The safety concepts adopted in Netherlands, Germany, and US identify multiple barriers
- Safety functions are identified in Germany and US, but not in the Netherlands
- The safety assessments performed in Netherlands, Germany, and US consider Normal Evolution Scenarios and Alternative Evolution Scenarios
- Several operational aspects of deep geological disposal have been demonstrated in the Netherlands and Germany. The construction and operation of the WIPP facility is a demonstration in itself.
- On the basis of experiences in the Netherlands and abroad it can be stated that a robust system for the final disposal of radioactive waste can be implemented

Multiple lines of reasoning - Natural analogues

- The use of multiple lines of reasoning may add value to the safety case by providing a range of different arguments that together build confidence in data, assumptions and results
- The most important group of arguments related to the concept of multiple lines of reasoning are natural analogue aspects
- The status report from the NEA Salt Club (NEA, 2013b) provides a state of the art overview of natural analogue aspects, and gives recommendations to further expand these aspects into the Safety Case

The important message from the evaluation is that, on the basis of existing information, a number of well-characterized arguments can be formulated to qualitatively add to building confidence of implementing a salt-based repository for the final disposal of radioactive waste.

However, in the Netherlands there has been limited activity since the CORA programme with regard to the development of a disposal facility in rock salt. Moreover, at present there is no well-established Dutch concept in rock salt yet available for which a dedicated safety assessment, including all aspects and features (e.g. most recently identified waste characteristics, safety functions, complementary safety and performance indicators including reference values, full probabilistic analysis, …) can be performed.

As a consequence, the integration of safety arguments, summarized above, related to the Dutch salt Safety Case is for a part based on generic considerations that may at present not all be applicable to the Dutch context. A next iteration of the Dutch salt Safety Case, based on recently acquired knowledge in US and Germany and the Netherlands would provide updated safety arguments, better tailored to the Dutch context.

7. Concluding remarks

The present report describes the results of an evaluation of the aspects of the Safety Case of the final disposal of radioactive waste in rock salt, focusing on the Dutch context. The evaluation is part of the OPERA project Salt Safety Case (OSSC), for OPERA Task *2.1.1*: *Evaluation of the current knowledge for building the Safety Case*, and has been compiled on the basis of an analysis of the existing national and international information concerning the disposal of radioactive waste in rock salt.

It is recognized that Germany and US have the most developed programmes on the geological disposal of radioactive waste in rock salt. The necessary information from those countries is brought into the OSSC consortium by the German partner in the project, through the documentation made available by the US-DOE, and through participation in the OECD/NEA hosted Salt Club.

During the last 40 years many efforts have been devoted in the Netherlands on the geologic disposal of radioactive waste in rock salt, for example in the framework of the ICK, OPLA, and CORA programmes. In these programmes performance assessments have been accomplished for generic repository designs in rock salt. The results of all these efforts have not yet been integrated in the past according to the recently developed and generally accepted methodology of the Safety Case by NEA (NEA, 2008) and IAEA (IAEA, 2012).

In the Netherlands there is at present little activity with regard to the development of a disposal facility in rock salt. Moreover, there is no fixed Dutch concept in rock salt yet available for which a dedicated safety assessment, including all modern aspects and features (e.g. safety functions, complementary indicators including reference values, full probabilistic analysis, …), can be performed. As a consequence, the integration of safety arguments related to the salt Safety Case is for a part based on generic considerations that may not all be applicable to the Dutch context, and that does not yet take into account all relevant aspects.

The evaluations of the different aspects of the Salt Safety Case, as brought together in the present report, have been summarized at the end of the respective sections:

- Safety Case context: Section 2.6
- Safety Strategy: Section 3.12
- System Description: Section 4.9
- Safety Assessment: Section 5.9
- Integration of Safety Arguments: Section 6.6

The main recommendation to proceed further with the development of the Salt Safety Case in the Netherlands is to outline a final disposal facility in rock salt, taking into account the most recent waste characteristics and an up-to-date safety concept including the definition of safety functions for the disposal of radioactive waste in rock salt.

A more detailed overview of recommendations to proceed with the Salt Safety Case in the Netherlands is provided in the second Delivarable of the OSSC project, OPERA-PU-NRG221B.

8. Glossary

actinides	elements with an atomic number between 89 and 103
anhydrite	a very brittle material (CaS04) which occurs as an impurity in a salt formation, mainly as strata or layers, and mixed with other materials
aquifer	a porous layer beneath the Earth's surface where the groundwater moves under the influence of the hydrological cycle
biosphere	part of the earth where living organismss can be found
Bq	becquerel; the SI unit of radioactivity, equal to one disintegration per second
caprock	a harder or more resistant rock type overlying a weaker or less resistant rock type, for a salt dome consisting of gypsum and anhydrite
cavern	a hollow space in the salt formation
compaction	stress induced densification of porous material
constitutive relation	relation between stresses and strains
convergence	gradual decrease in size of an excavation or cavern
decay	process whereby radioactive elements disintegrate and thereby emit radiation
dehydration	process of releasing water from hydrate minerals
diapirism	the slow rise of the salt formation caused by density differences between salt and surrounding sediments
disposal	placement of waste in a suitable facility without intent to retrieve it at a later date. Retrieval may be possible but, if intended, the appropriate term is storage.
dose conversion factor	nuclide specific conversion factor for converting becquerel (Bq) to sievert (Sv), exposure pathway specific
EBS	Engineered Barrier System
EMOS	Endlagerbezogene Modellierung von Szenarien; computer program developed in Germany
FEP	Features, Events and Processes that might affect the current state or the future evolution of the repository system
glacial erosion	wearing away of the earth’s surface by the action of an ice layer
glaciation	period within an ice age that is marked by colder temperatures and glacier advances
HAVA	high activity solid waste
HAW	high-active waste
Kd value	relationship between the concentrations: mass nuclides per mass of rocks and liquid mass per liquid volume (m^3/kg)
KSA	vitrified waste from reprocessing of nuclear fuel
LAVA	low activity solid waste
lithostatic pressure	pressure caused by the weight of the overlying rocks
MAVA	intermediate activity solid waste

METROPOL	MEthod of the TRansport Of POLlutants, computer program
MWe	megawatt (electrical)
NEA	Nuclear Energy Agency
OPLA	Commission for Onshore Disposal
overburden	earth layers above the salt formation
percolation	movement and filtering of fluids through porous materials
threshold porosity	porosity below which open pores are no longer ‘connected’ and flow becomes impossible
radioactive waste	any material contaminated by or incorporating radioactivity about certain thresholds defined in legislation, and for which no further use is envisaged
radiolysis	changes in a chemical system brought about by ionising radiation
reprocessing	a physical or chemical separation operation, the purpose of which is to extract uranium or plutonium from spent fuel for re-use
safety case	a collection of arguments and evidence in support of the safety of a facility or activity. This will normally include the findings of a safety assessment and a statement of confidence in these findings. For a GDF, there will be a number of safety cases required covering nuclear safety, environmental safety, and transport. A safety case may also relate to a given stage of development (e.g. site investigations, construction, operation, closure, post-closure).
safety function	a function that a disposal system should fulfil to achieve its fundamental objective of providing long-term safety through the concentration and confinement strategy, while limiting the burden for future generations
safety function indicator	measurable or calculable property indicating the extent to which components of a disposal system achieve their safety function(s)
safety indicator	a quantity, calculable by means of suitable models, that provides a measure for the total system performance with respect to a specific safety aspect, in comparison with a reference value quantifying a global or local level that can be proven, or is at least commonly considered, to be safe
salt diapir	large, underground salt formation shaped like a pillar or mushroom formed by an upward pushing movement of a salt layer whereby one or more tertiary layers are ruptured
salt pillow	a large, underground salt formation formed by a salt layer pushing upwards
scenario study	investigation of which combination of physical processes and occurrences can lead to a possible by-pass of the multi-barrier system and thereby to a release of nuclides
spent fuel	nuclear fuel removed from a reactor following irradiation that is no longer usable in its present form because of depletion of fissile material, poison build-up or radiation damage
storage	the placement of waste in a suitable facility with the intent to retrieve it at a later date
subrosion	dissolution of a salt formation through contact with groundwater

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Appendix 1 Evaluation of FEPs

The following tables evaluate FEPs relevant for salt. The German FEP catalogue for salt served as the basis of the evaluation (Wolf, 2012b). That FEP catalogue has been compared with the PROSA FEP list (Prij, 1993), and the WIPP FEP catalogue (US DOE, 2014). The designation and numbering of the FEPs has been indicated as in the respective documents.

The aim of the evaluation was:

- To provide an overview of FEPs relevant for the long-term safety of the salt-based repository;
- To provide a summarily judgement of the status of knowledge of each FEP;
- To provide an overview of open questions.

Due to budget and time constraints a thorough and comprehensive comparison was outside the scope of the present report. However, the information provided in this appendix may serve as a basis to consult the respective documentation and look in more detail to the FEPs considered relevant for the disposal in rock salt.

Tectonics (1.2.01.01)		
Short description	Movements of the earth's crust that induce deformation in the currently prevailing regional stress fields	
Index	VSG	3. Neotektonische Vorgänge
	WIPP	N4 Regional Tectonics
	PROSA	1.2.11 Plate Tectonics
Judgement	Site-specific FEP Sufficient process knowledge	
Open questions	No	

Orogenesis (1.2.01.02)		
Short description	All processes contributing to the formation of a mountain range	
Index	VSG	4. Orogenese
	WIPP	N5 Regional Uplift and Subsidence
	PROSA	1.2.15 Uplift and subsidence
Judgement	Site-specific FEP Sufficient process knowledge Very likely not applicable in the Netherlands for reasons of timing – several tens of millions of years (PROSA)	
Open questions	No	

Uplift and Subsidence (1.2.01.03/04)		
Short description	Large-scale upward and downward movements of the earth's crust - epirogenese	
Index	VSG	5. Senkung der Erdkruste 6. Hebung der Erdkruste
	WIPP	N5 Regional Uplift and Subsidence
	PROSA	1.2.15 Uplift and subsidence
Judgement	• Site-specific FEP • No direct effect on the performance of the barrier system (PROSA) • Sufficient process knowledge	
Open questions	No	

Metamorphosis (1.2.05.01)		
Short description	Structural changes due to natural heating and/or pressure	
Index	VSG	11. Gesteinsmetamorphose
	WIPP	N15 Metamorphic Activity
	PROSA	1.2.9 Metamorphic Activity
Judgement	• Site specific FEP • Although rock salt is subject to metamorphic activity too, there is no indication at the moment that metamorphism lowers the isolation potential of rock salt • Metamorphosis should be excluded by the site selection process	
Open questions	No	

Diagenesis (1.2.08.01)		
Short description	Long-or short-term transformation of loose sediment into solid sedimentary rocks by e.g. pressure and temperature changes, chemical solution and precipitation	
Index	VSG	15. Diagenesis
	WIPP	N/A
	PROSA	1.2.3 Diagenesis
Judgement	• Site specific FEP • May impact porosity	
Open questions	No	

Diapirism (1.2.09.01)		
Short description	Upward salt movement as a consequence of a density contrast between the rock salt and the overburden	
Index	VSG	16. Diapirismus
	WIPP	N7 Diapirism
	PROSA	1.2.4 Diapirism and Halokinesis
Judgement	• Site specific FEP • Related to Subrosion • Considered an important process, which, in the long term, reduces the rock-salt isolation (PROSA) • Mechanical models available (WIPP)	
Open questions	Effect of glaciation periods on the movement behavior of the salt dome (VSG)	

Subrosion (1.2.09.02)		
Short description	Subsurface dissolution of rock salt by groundwater	
Index	VSG	17. Subrosion
	WIPP	N16 Shallow Dissolution N18 Deep Dissolution
	PROSA	1.4.10 Subrosion
Judgement	• Site specific FEP • Considered an important process in reducing the thickness of the salt barrier; should be be incorporated in the safety study (PROSA) • Sufficient knowledge available (VSG, WIPP)	
Open questions	No	

Cryogenic Contractions **(1.3.04.02)**		
Short description	Cryogenic contractions are emanating from the salt joints in the host rock, which are related to glaciation periods and are probably due to cooling and contraction	
Index	VSG	21. Bildung kryogener Klüfte
	WIPP	N/A
	PROSA	N/A
Judgement	• Site specific FEP • Still insufficient knowledge (VSG)	
Open questions	• Structural and regional development of the cracks and contractions • Thermal-mechanical modelling to investigate the formation of cryogenic fractures under site-specific conditions and the expected future development at the site (VSG)	

Backfill (2.1.04.01)		
Short description	Mining materials, esp. crushed salt, used to fill subsurface cavities and openings	
Index	VSG	36. Versatz
	WIPP	W9 Backfill Physical Properties W10 Backfill Chemical Composition
	PROSA	3.2.2 Degradation of buffer/backfill
Judgement	• Important in relation to compaction/convergence of backfilled spaces (PROSA) • Modeling efforts still needed (see Open questions) • Backfill physical properties eliminated from PA calculations due to low consequence to the performance of the disposal system (WIPP)	
Open questions	VSG: • Compaction behavior of crushed salt • Possible optimization by application of alternative backfill materials • Moisture content of dry and wet backfill • Thermal conductivity of high-porosity crushed salt	

Seal Material (2.1.05.01)		
Short description	Composition and properties of seal materials, used for the construction of the seals of the shafts and galleries	
Index	VSG	37. Verschlussmaterial
	WIPP	W8 Shaft Seal Chemical Composition
	PROSA	N/A
Judgement	• Seal material requirements formulated (VSG, WIPP)	
Open questions	• Material optimization by specific manipulation of material properties (VSG) • Long-term stability of the support materials or sealing materials • Transferability of the results obtained for borehole tests to the boundary conditions of a repository's shaft seal • Long-term stability of bitumen	

Shaft Seal (2.1.05.02)		
Short description	FEP describing the construction as well as the physical, chemical and hydraulic properties of the shaft seal	
Index	VSG	38. Schachtverschlüsse
	WIPP	W6 Shaft Seal Geometry W7 Shaft Seal Physical Properties
	PROSA	N/A; addressed under "Seal failure"
Judgement	• VSG formulates design requirements, property requirements, and design specifications • WIPP shaft seal extensively described in the WIPP CCA	
Open questions	No	

Gallery Seal (2.1.05.03)		
Short description	FEP describing the construction as well as the physical, chemical and hydraulic properties of the gallery seals	
Index	VSG	39 Streckenverschlüsse
	WIPP	W109 Panel Closure Geometry W110 Panel Closure Physical Properties
	PROSA	N/A; addressed under "Seal failure"
Judgement	• Interactions between seal materials and brine taken into account (WIPP) • Material-specific properties of salt and Sorelbeton and magnesia studied and modeled (VSG)	
Open questions	• Characteristics and development of the contact zone • Design optimization of the gallery seals • Quality assurance of the monitoring equipment	

Alteration of Shaft Seals and Gallery Seals (2.1.05.04)		
Short description	FEP describing the influence on mineral building materials of the engineered barriers by (geo) chemical environmental conditions over time	
Index	VSG	40 Alteration von Strecken- und Schachtverschlüssen
	WIPP	W74 Chemical Degradation of Shaft Seals
	PROSA	2.1.7 Seal failure
Judgement	• Impact by a variety of chemical and physical aspects (VSG) • Modeled by application of CDFs, cumulative distribution functions (WIPP)	
Open questions	• Long-term stability of various seal materials in salty environment (bentonite, bitumen, etc)	

Other Seals (2.1.05.05)		
Short description	FEP describing the construction as well as the physical, chemical and hydraulic properties of other sealing structures, having complementary functions during the operational phase	
Index	VSG	41 Sonstige Verschlussbauwerke
	WIPP	N/A
	PROSA	N/A; addressed under "Seal failure"
Judgement	• Variety of materials and applications (e.g. support) • Effects on various sub-systems still to be addressed (VSG)	
Open questions	• Effects of thermochemical sulfate reduction • Process understanding: dissolution in crushed salt • Sorel concrete: sealing properties and fluid transport • Adaptation of the design of the sealing plug	

Borehole Liner (2.1.06.02)		
Short description	FEP related to the liner giving support to a borehole, in order to facilitate the retrieval of waste canisters	
Index	VSG	43 Bohrlochverrohrung
	WIPP	N/A
	PROSA	N/A
Judgement	• Well established in the oil and gas exploration, and in geothermal applications • Possible effects of waste heat output and radiation • Liner foreseen to function approx. 500 years	
Open questions	• Demonstration of technical feasibility	

Failure of Borehole Liner (2.1.06.03)		
Short description	FEP describing the loss of mechanical stability of a the liner giving support to a borehole, as a result of beyond design influences or undetected production errors	
Index	VSG	44 Ausfall einer Bohrlochverrohrung
	WIPP	N/A
	PROSA	N/A
Judgement	• Regarded as unlikely (<0,01%) (VSG) • May induce moisture redistribution	
Open questions	• Probability of liner failure • Importance of liner segment failure on e.g. waste container degradation	

Convergence (2.1.07.01)		
Short description	FEP relates to the cross-sectional reduction of underground cavities and openings, starting after the excavation due to stress redistribution	
Index	VSG	45 Konvergenz
	WIPP	W20 Salt Creep W21 Change in the Stress Field
	PROSA	3.3.3 Convergence of Openings
Judgement	• Convergence leads to re-sealing of excavation-induced openings, and thereby to isolation of the waste • Convergence and compaction are important processes because convergence is the driving force for any (contaminated) brine extrusion from a flooded repository • Convergence is well understood	
Open questions	• The process of healing and sealing is yet not well understood, especially the effects of moisture-induced processes (moisture creep, fluid pressure)	

Fluid Pressure (2.1.07.02)		
Short description	Fluid pressure relates to the pressure of liquids or gases in the subsurface facility	
Index	VSG	46 Fluiddruck
	WIPP	W25 Disruption Due to Gas Effects W26 Pressurization
	PROSA	N/A
Judgement	• Fluid pressure impacts the evolution of the disposal system in various ways; an overview of all relevant aspects is provided by VSG	
Open questions	• The fluid pressure indirectly affects many issues (e.g. backfill compaction). The remaining open questions are described in the corresponding FEP (VSG).	

Compaction of Crushed Salt (2.1.07.03)		
Short description	The volume and the porosity (initially ~25-45%) of the backfill introduced into the disposal facility is reduced by the continuing convergence and resulting compaction.	
Index	VSG	47 Salzgruskompaktion
	WIPP	Not mentioned
	PROSA	3.3.3 Convergence of Openings
Judgement	• After introduction of the crushed salt, the initial compaction is relatively fast (VSG) • Process is quite well understood; some open questions remain (VSG)	
Open questions	• Compaction of crushed salt at small porosities (VSG, NF-PRO) • Thermo-mechanical interactions (VSG) • Effects of moisture content in compaction rate (VSG)	

Non Thermally Induced Volume Changes (2.1.07.04)		
Short description	Non thermally induced volume changes result from mineralogical conversions in the host rock (e.g. Anhydrite <> Gipsum), chemical processes (e.g. metal/concrete corrosion), or from reduction and compaction of backfill materials (eg gravel and sand). Thermochemical sulfate reduction is considered a separate FEP. Not included in the FEP are the crushed salt compaktion and the swelling of bentonite.	
Index	VSG	48 Nicht thermisch induzierte Volumenänderung von Materialien
	WIPP	Mineralogic Dehydration - Dehydration reactions release water and may lead to volume changes
	PROSA	Not addressed
Judgement	• The FEP includes a variety of processes, some of which can be specified, whereas other cannot • Any brine intrusion complicates this FEP	
Open questions	• Determination of the volume change of tcontainer materials by corrosion • Determination of the volume change of the backfill and barrier materials as well as the host rock by drying and moisture	

Early Shaft Seal Failure (2.1.07.05)		
Short description	Failure of the shaft seals due to beyond-design effects or a combination of different effects, leading to a significant increase in the hydraulic conductivity	
Index	VSG	49 Vorzeitiges Versagen eines Schachtverschlusses
	WIPP	W37 Mechanical Degradation of Shaft Seals W74 Chemical Degradation of Shaft Seals
	PROSA	2.1.7 Seal Failure
Judgement	• Can be accounted for in PA calculations through the permeability ranges assumed for the seal and closure systems (WIPP) • Can be minimized by quality assurance (PROSA)	
Open questions	• Mechanisms of failure (thermo-mechanical, chemical, hydraulic) (VSG) • Main uncertainties with regard to cement degradation rates at the WIPP are the effects of groundwater chemistry, the exact nature of the cementitious phases present, and the rates of brine infiltration (WIPP)	

Early Gallery Seal Failure (2.1.07.06)		
Short description	Failure of the gallery seals due to beyond-design effects or a combination of different effects, leading to a significant increase in the hydraulic conductivity.	
Index	VSG	50 Vorzeitiges Versagen eines Streckenverschlusses 51 Lageverschiebung des Schachtverschlusses (2.1.07.07) 52 Ausfall eines Dichtpfropfens (2.1.07.08)
	WIPP	W114 Mechanical Degradation of Panel Closures W115 Chemical Degradation of Panel Seals
	PROSA	2.1.7 Seal Failure
Judgement	• The excavation Damaged Zone (EDZ) is an important asset • Can be accounted for in PA calculations through the permeability ranges assumed for the seal and closure systems (WIPP) • Due to the salt construction, chemical degradation is not expected to occur to the panel closures • Can be minimized by quality assurance (PROSA)	
Open questions	• Failure mechanism (VSG) • Mathematical description and parameter quantification of the excavation damaged zone • Main uncertainties with regard to cement degradation rates at the WIPP are the effects of groundwater chemistry, the exact nature of the cementitious phases present, and the rates of brine infiltration	

Porosity (2.1.08.01)		
Short description	The porosity is defined as the ratio of void volume to total geometric (outer) volume.	
Index	VSG	53 Porosität
	WIPP	Not a separate FEP
	PROSA	Not a separate FEP
Judgement	• For flow and transport processes the effective porosity is relevant; for compaction the total porosity should be considered • A proper estimation of the porosity is necessary for a variety of affected FEPs (e.g. convergence, compaction, advection, fluid flow, …)	
Open questions	• Time evolution of porosity with very small values, e.g. occurring at crushed salt compaction • Effect of any possible increase in the porosity by over-pressure resulting from gas formation processes	

Permeability (2.1.08.02)		
Short description	Permeability is a characteristic for the transmissivity of a porous (or fractured) medium. Low permeability, as is present in rock salt, is equivalent to a high flow resistance and thus causes a slow flow	
Index	VSG	54 Permeabilität
	WIPP	Not a separate FEP
	PROSA	Not a separate FEP
Judgement	• Regardless of the many constraints and influences permeability changes result primarily from changes in the (effective) porosity; these interactions are described by Permeability/Porosity relationships	
Open questions	• Effect of of lithostatic fluid pressure in the pore space, which has consequences for the description of fluid dispersion into the surrounding rock • Time-dependent permeability-porosity relations for the process of healing after excavation (e.g. in the EDZ) • Anisotropic nature of permeability	

Presence of Liquid in the Facility (2.1.08.03)		
Short description	The presence of liquid, i.e. aqueous solutions, may result from moisture contained in disposed materials (e.g. backfill, seal materials, waste), moisture contained in the air, or due to an inflow from the surrounding rock. This FEP is different from the FEP “Brine Intrusion into the Facility”.	
Index	VSG	55 Lösungen im Grubenbau
	WIPP	Not a specific FEP
	PROSA	Not a specific FEP
Judgement	• The composition and properties of the liquids depend on the interactions of the liquids with all other substances in the facility (e.g. host rock, waste, backfill)	
Open questions	• No	

Channeling in Crushed Salt (2.1.08.04)		
Short description	The formation of flow paths (channels) in backfilled spaces in which a flowing medium may preferably spread out. Channeling may result from e.g. dissolution of (part of) the medium, or an uneven compaction of the medium by its own weight.	
Index	VSG	56 Kanalisierung im Salzgrus
	WIPP	N27 Effects of Preferential Pathways
	PROSA	1.6.1 Advection, Convection, Dispersion
Judgement	• Channeling reduces the flow resistance, and locally enhances the permeability • Chanelling is possible as long as the compaction of the crushed salt has not been finalized • Effects of preferential pathways are accounted for in PA calculations in the estimates of transmissivity (WIPP)	
Open questions	• The relevance of the channeling of crushed salt, especially in relation to its compaction	

Liquid Intrusion into the Facility (2.1.08.06)		
Short description	During the development and operation of the repository system liquid contained in the rock salt or overburden may enter the excavated spaces. This process is usually referred to as “brine intrusion”. Liquid intrusion affects the backfill compaction, and a significantly enganced corrosion of the waste containers and materials used for construction. This may result in early and enhanced dispersion of radionuclides out of the waste containers and the disposal cells/galleries.	
Index	VSG	58 Lösungszutritt ins Grubengebäude
	WIPP	W40 Brine Inflow
	PROSA	1.5.7 Intrusion of saline/fresh water (Note that in PROSA FEP 1.5.7 has not been described)
Judgement	• Liquid intrusion is an important event and must be considered in a safety assessment	
Open questions	• No	

Flow Processes in the Facility (2.1.08.07)		
Short description	Any liquid present in the facility may flow due to driving forces that result from e.g. hydraulic gradients, density differences between different fluids (buoyancy) or within a fluid (convection), as well as convergence processes in the facility. Flow processes in the gas phase can results from by gas formation, convergence and lift. In solvent-filled cavities formed gases can lead to displacement of existing fluids.	
Index	VSG	59 Strömungsvorgänge im Grubengebäude
	WIPP	W42 Fluid Flow Due to Gas Production W43 Convection (driven by heat generation in waste)
	PROSA	1.6.4 Gas Mediated Transport 1.6.6 Multiphase Flow
Judgement	• Flow processes of liquids and gases are to be considered for the entire facility • In the vicinity of heat-generating waste density-driven flows of salt solutions may the triggered due to temperature/concentration gradients, potentially resulting in an enhanced dispersion of radionuclides • Flow processes including the simultaneous displacement of liquids and gases are difficult to establish	
Open questions	VSG: • Liquid displacements at small porosities and small amounts of liquid • Two-phase flow parameters in crushed salt • Model description of two-phase flow under repository conditions, in particular the resistance to incoming liquids into a gas-filled spaces	

Geochemical Conditions in the Repository (2.1.09.01)		
Short description	The geochemical condition in a repository is primarily determined by geochemical characteristics of the salt host rock, the emplaced solid wastes, and the engineered barriers. It is affected by any present or incoming liquid and the gas phase	
Index	VSG	61 Geochemisches Milieu im Grubenbau
	WIPP	Not a specific FEP, allocated through other FEPs
	PROSA	Not a specific FEP, allocated through other FEPs
Judgement	• The geochemical conditions affect many processes, that are also	

	treated as separate FEPs	
Open questions	• Evolution of the geochemical conditions at long time scales • Coupling with transport processes • Temperature effects (above 25 °C)	

Dissolution and Precipitation (2.1.09.02)		
Short description	FEP relating to the dissolution and precipitation of solid materials in liquid, i.e. brine in the case of a salt-based repository	
Index	VSG	62 Auflösung und Ausfällung
	WIPP	W58 Dissolution of Waste W59 Precipitation of Secondary Minerals W60 Kinetics of Precipitation and Dissolution
	PROSA	1.7.2 Chemical Equilibrium Reaction (a.o. Dissolution and Precipitation)
Judgement	• In principle a large variety of dissolution / precipitation reactions should be taken into account • Many other processes influence this FEP, and are influenced by this FEP • The WIPP PA takes dissolution of the waste into account. It does not take into account precipitation of radionuclides from the solution for reasons of conservatism	
Open questions	VSG: • Impact of changes in the stress on dissolution processes • Long-time behavior of a liquid-filled cavities in the rock salt in the temperature field of heat-generating waste • Influence of temperature on the dissolution and precipitation of radionuclides	

Metal Corrosion (2.1.09.03)		
Short description	FEP refers to the (electro-)chemical reaction of metal with materials of the surroundings. The processes considered here refer to reactions in the presence of aqueous solutions or water vapor. Metal corrosion results in the formation of gas.	
Index	VSG	63 Metallkorrosion
	WIPP	W64 Effects of Metal Corrosion W66 Reduction-Oxidation Kinetics
	PROSA	3.2.9 Metallic Corrosion
Judgement	• Metal corrosion can be modeled by reduction-oxidation reaction kinetics • Metal corrosion directly affects the source term of the radionuclide release from the waste	
Open questions	• Consideration of local corrosion effects • Consideration of microbially influenced corrosion processes • The influence of temperature changes on metal corrosion • Formation of stable metal corrosion products in salt solutions • Influencing the geochemical environment (especially pH and Redox properties) by metal corrosion and formation of corrosion products • Kinetics of metal corrosion under irradiation (relevant for vitrified HLW)	

Corrosion of Spent Fuel Matrix (2.1.09.04)		
Short description	Corrosion of the spent fuel matrix describes the chemical reactions of the matrices of spent fuel (e.g. UOX, MOX) by liquids, e.g. brine.	
Index	VSG	64 Korrosion der Brennstoffmatrix
	WIPP	W64 Effects of Metal Corrosion W66 Reduction-Oxidation Kinetics
	PROSA	3.2.9 Metallic Corrosion
Judgement	• Corrosion of the spent fuel matrix includes many complex processes • Specific information generated in the different German programs	
Open questions	• effect of radiolysis on the redox potential and thus the corrosion of the fuel matrix for long time spans • Influence of temperature on the corrosion of the fuel matrix • Influence of hydrogen, radiation-chemically active liquid components and the redox potential on the corrosion of the fuel matrix • Corrosion of MOX fuel under saline conditions • Corrosion of fuels from research and prototype reactors	

Corrosion of Glass (2.1.09.05)		
Short description	The FEP relates to the chemical conversion of the borosilicate matrix of vitrified waste due to interactions with liquids, especially brine. Mobilisation of radionuclides can only commence when liquids have come into contact with the glass matrix.	
Index	VSG	65 Korrosion von Glas
	WIPP	N/A for the WIPP (no vitrified HLW)
	PROSA	No specific FEP
Judgement	• The process is quite well understood, and roughly consists of the diffusion-controlled release of alkaline ions from the glass matrix and the dissolution of the glass matrix itself (VSG)	
Open questions	• Temperature dependency of the glass corrosion, in particular temperature dependency of the long-term corrosion rate • Long-term rate of glass dissolution (after reaching the Si-saturation), and its dependency of the coupling between corrosion and precipitation reactions	

Corrosion of Cementitious Materials (2.1.09.06)		
Short description	The chemical decomposition of materials with cement or Sorel phases by liquids, more specifically brine	
Index	VSG	66 Korrosion von Materialien mit Zement- oder Sorelphasen
	WIPP	N/A for the WIPP (no concrete)
	PROSA	N/A not addressed
Judgement	• The processes include a series of complex chemical interactions and lead to changes in the mechanical and chemical properties of materials with cement or Sorel phases (VSG)	
Open questions	• Influence of temperature on the corrosion of materials with cement or Sorel phases, in particular the influence of a temperature increase in the disposal facility with a cementitious waste with negligible heat generation, which is caused by the heat input from the disposal zone for high-level waste	

Material Embrittlemen by Hydrogen Uptake (2.1.09.07)		
Short description	The penetration of hydrogen in the structure of certain materials (metals) can cause a change in the mechanical properties	
Index	VSG	67 Materialversprödung durch Wasserstoffaufnahme
	WIPP	W64 Effects of Metal Corrosion
	PROSA	3.3.4 Embrittlement, Cracking
Judgement	• Process is well investigated (VSG)	
Open questions	• Change in the mechanical stability (strength) by material embrittlement	

Behaviour of Graphite Materials and Uranium Tails (2.1.09.08)		
Short description	The FEP describes the chemical interactions of these substances with aqueous solutions, especially brine. Graphite as a waste is not considered in the Netherlands. At present uranium tails are considered to be disposed.	
Index	VSG	68 Verhalten von graphithaltigen Materialien und Urantails
	WIPP	N/A
	PROSA	Not considered
Judgement	• The interaction of uranium tails with brine should be considered • The process has been investigated under laboratory conditions(VSG)	
Open questions	• The corrosion of uranium tails under in-situ conditions has insufficiently been studied, and therefore involves considerable uncertainties	

Degradation of Organic Material (2.1.10.01)		
Short description	In this FEP all processes are combined leading to a decomposition of organics. These include the thermal decomposition of organic material and the degradation by microbial activities.	
Index	VSG	69 Zersetzung von Organika
	WIPP	W44 Degradation of Organic Material
	PROSA	Not a specific FEP
Judgement	• Detailed descriptions of processes available (VSG) • Substantial information available (WIPP)	
Open questions	• Decomposition of and gas formation from organic compounds in saline environment with particular emphasis on hard-degradable plastics (PVC).	

Microbiotical processes (2.1.10.02)		
Short description	FEP comprises all processes caused by micro-organisms in the facility and in the salt dome	
Index	VSG	70 Mikrobielle Prozesse im Grubengebäude und im Salzstock)
	WIPP	• W45 Effects of Temperature on Microbial Gas Generation • W48 Effects of Biofilms on Microbial Gas Generation • W46 Effects of Pressure on Microbial Gas Generation • W47 Effects of Radiation on Microbial Gas Generation • W76 Microbial Growth on Concrete • W87 Microbial Transport

	PROSA	1.8.5 Microbial interactions
Judgement	• Detailed descriptions of processes available (VSG) • Substantial information available (WIPP)	
Open questions	• Relevance of microbial processes in the facility • Importance of nitrate levels in vitrified waste	

Heat Generation (2.1.11.01)		
Short description	Part of the waste resulting from the nuclear fuel cycle generates heat. This has to be taken into account in the analyses by the appropriate model formulations of temperature-dependent phenomena.	
Index	VSG	71 Wärmeproduktion
	WIPP	W13 Heat From Radioactive Decay
	PROSA	3.4.1 Heat Generation
Judgement	• In the Dutch context of elongated surface storage the heat generation from vitrified HLW is moderate • Heat generation is well established; the effects of temperature-dependent, in-situ phenomena are often difficult to assess. • Heat generation at WIPP presently not relevant (no heat-generating waste); efforts are underway to investigate the feasibility of disposing heat-generating waste at WIPP (see e.g. Hansen, 2011).	
Open questions	• No open questions about heat generation • Many open questions about temperature-dependency of processes (addressed at individual FEPs)	

Thermal Expansion or Contraction (2.1.11.02)		
Short description	FEP describes the volume increase or decrease in the considered (rock salt) substance is caused by a change in its temperature.	
Index	VSG	72 Thermische Expansion oder Kontraktion
	WIPP	W31 Differing Thermal Expansion of Repository Components
	PROSA	3.1.4 Differential elastic response
Judgement	• Thermal expansion or contraction must be considered in coupled thermo-mechanical model calculations in conjunction with the heat conduction to correctly determine the effects of stress (re-) distributions for the integrity of the barriers, in particular salt host rock (VSG) • Addressed in German program (VSG) • FEP updated to reflect the inventory used for the CRA-2014 (heat-generating HLW) and planned thermal experiments at WIPP	
Open questions	• No	

<table>
<tr><th colspan="3">Gas Formation (2.1.12.01)</th></tr>
<tr><td>Short description</td><td colspan="2">The FEP gas formation includes all processes in which gas is formed in the repository. These are the corrosion of metals, evaporation of water, decomposition of organic components, thermochemical sulfate reduction, radiolysis and corrosion of the fuel matrix.</td></tr>
<tr><td rowspan="3">Index</td><td>VSG</td><td>74 Gasbildung</td></tr>
<tr><td>WIPP</td><td>W49 Gases from Metal Corrosion
W54 Helium Gas Production
W55 Radioactive Gases</td></tr>
<tr><td>PROSA</td><td>3.2.4 Gas generation, Explosions</td></tr>
<tr><td>Judgement</td><td colspan="2">• Requisite for most gasification processes is the presence of water (VSG)
• In a repository for high-level radioactive waste, the largest quantities of gas originate from the corrosion of metals and degradation of organic compounds (VSG)
• The FEP is strongly related to the FEP “Corrosion”, which is enhanced by the presence of brine</td></tr>
<tr><td>Open questions</td><td colspan="2">• No</td></tr>
</table>

<table>
<tr><th colspan="3">Gas Entry Pressure (2.1.12.03)</th></tr>
<tr><td>Short description</td><td colspan="2">The gas entry pressure is the pressure that a gas phase must exceed, in addition to the prevailing hydraulic pressure, to displace liquid phase in a two-phase system (liquid, gas) in a porous solid medium</td></tr>
<tr><td rowspan="3">Index</td><td>VSG</td><td>76 Gaseindringdruck</td></tr>
<tr><td>WIPP</td><td>W42 Fluid Flow Due to Gas Production</td></tr>
<tr><td>PROSA</td><td>Not addressed</td></tr>
<tr><td>Judgement</td><td colspan="2">• During the build-up of a high pressure pathways may be formed in the surrounding rock
• Pressurization of the repository through gas generation could limit the amount of brine that flows into the chambers and drifts (WIPP)
• Accounted for in WIPP PA</td></tr>
<tr><td>Open questions</td><td colspan="2">• The data base for the gas entry pressure for crushed salt with different porosities is not sufficient.</td></tr>
</table>

<table>
<tr><th colspan="3">Material Embrittlement due to Radiation (2.1.13.02)</th></tr>
<tr><td>Short description</td><td colspan="2">FEP relates to the embrittlement of materials as a result of radiation</td></tr>
<tr><td rowspan="3">Index</td><td>VSG</td><td>79 Materialversprödung durch Strahlung</td></tr>
<tr><td>WIPP</td><td>No specific FEP</td></tr>
<tr><td>PROSA</td><td>3.3.4 Embrittlement and Cracking</td></tr>
<tr><td>Judgement</td><td colspan="2">• Moderately relevant for metal components (VSG)
• Very small significance compared to other effects
• Not significant for salt</td></tr>
<tr><td>Open questions</td><td colspan="2">• No</td></tr>
</table>

Radiolysis (2.1.13.03)		
Short description	Radiolysis refers to the change of chemical compounds by the impact of ionising radiation. Possible consequences include, for example, dissociation of molecules or the formation of free radicals.	
Index	VSG	80 Radiolyse
	WIPP	• W52 Radiolysis of Brine • W53 Radiolysis of Cellulose
	PROSA	3.4.5 Radiolysis
Judgement	• Radiolysis of brine addressed in WIPP CCA/CRA; consequences are minor within the present context of WIPP • Radiolysis may induce gamma radiation energy absorption in salt, and ultimately lead to the formation of radiolytic Na-colloids and chlorine gas bubbles that remain dispersed in the NaCl crystal (VSG) • The required dose to induce gamma radiation damage in rock salt, if any, only occurs in a very narrow range of a few centimeters. Therefore the effects of radiolysis on the integrity of rock salt are negligible (VSG)	
Open questions	• Effects of radiolysis on salt (WIPP) • Efficiency of energy accumulation due to radiolysis (VSG) • Threshold for possible explosive reactions at the conditions in the repository (VSG)	

Excavation Disturbed Zone (2.2.01.01)		
Short description	An excavation disturbed zone (EDZ) is a contour-limited, damaged zone surrounding excavated cavities, where the disturbance is caused by changes of the primary stress state and the associated overflow of the dilatancy boundary. Excavation-induced changes in stress can lead to failure of intact rock around the opening and the creation of fractures.	
Index	VSG	82 Auflockerungszone
	WIPP	W18 Disturbed Rock Zone (DRZ)
	PROSA	(Related to 1.2.2 Creep of Rock)
Judgement	• Extensively studied in WIPP • Extensively studied in Germany (VSG)	
Open questions	• Understanding of the healing/sealing phase: complex interactions between convergence, seal structures, the contact zone and the EDZ • Improvement of the description and modeling of cracking processes and the reduction of conservatism • Development of methods for the technical sealing of the EDZ, e.g., by injection methods with long-term stable sealing materials	

Fractures in the Host Rock (2.2.02.02)		
Short description	Fractures are macroscopically visible disturbances in the rock, which have no or only very slight dislocations at the interfaces. They are to be distinguished from faults, which are characterized by a significant dislocation of a fault zone adjoining the rocks. Fractures originate from deformation or heat input from emplaced waste.	
Index	VSG	84 Störungen und Klüfte im Wirtsgestein
	WIPP	N8 Formation of Fractures N9 Changes in Fracture Properties
	PROSA	1.2.7 Fracturing 3.1.5 Fracture aperture changes
Judgement	• The Formation of fractures near the repository is accounted for in PA through treatment of the EDZ (WIPP) • Fractures can be eliminated by creep as a result of pressure from the overburden (VSG)	
Open questions	• Healing and sealing of fractures in rock salt (VSG)	

Faults in the Overburden (2.2.04.01)		
Short description	A fault is generally understood a dividing line in the overburden on which a displacement of two adjacent rock packets occurs. Faults can extend along the range of centimeters to kilometers. In case a more three-dimensional-stretched zone is present where several faults occur that zone is indicated as a fault zone.	
Index	VSG	86 Störungen und Störungszonen im Deck- und Nebengebirge
	WIPP	N10 Formation of New Faults N11 Fault Movement
	PROSA	(1.2.7 “Fracturing” relates to the “Isolation Shield”)
Judgement	• Site-specific FEP, effect can be minisimed by site selection (WIPP) • The effects of newly formed faults may extend over the very long term, i.e. only beyond the time span addressed in a safety assessment (VSG)	
Open questions	• No	

Stress change and stress redistribution (2.2.06.01)		
Short description	Changes in the local stresses and stress redistributions can occur as a result of the excavation of a disposal facility or the heat input from heat generating waste.	
Index	VSG	87 Spannungsänderung und Spannungsumlagerung
	WIPP	N3 Changes in Regional Stress W19 Excavation-Induced Changes in Stress W21 Changes in the Stress Field W30 Thermally-Induced Stress Changes
	PROSA	2.1.8 Stress Field Changes 3.3.2 Changes in in-situ stress field
Judgement	• FEP relates to EDZ formation • Thermomechanical stresses affect the integrity of the seals or containers. This is taken into account, inter alia, in the design of containers and technical barriers (VSG)	
Open questions	• No	

Cave-in (2.2.06.02)		
Short description	Cave-in refers to the filling of excavated, open cavities with material breaching from the surrounding rock as a result of stress redistributions.	
Index	VSG	88 Selbstversatz
	WIPP	Not a specific FEP
	PROSA	Not addressed
Judgement	• Cave-in can be minimized by early backfilling the open spaces	
Open questions	• No	

Fluid Pockets in the Host Rock (2.2.07.01)		
Short description	During the formation of evaporate materials fluids, present in saline host rock, can be collected in pockets, i.e. grain boundaries or in the pore spaces or fissures, or in larger reservoirs	
Index	VSG	89 Fluidvorkommen im Wirtsgestein
	WIPP	N2 Brine Reservoirs
	PROSA	2.1.10 Undetected Geological Features (a.o. Brine Pockets)
Judgement	• In principle, (parts of) the repository can be flooded with brine from the fluid pockets, leading to early and enhanced corrosion (VSG) • This process must be considered in a safety assessment (VSG)	
Open questions	• Nature and extent of the presence of lye films in intergranular spaces	

Presence of Hydrocarbons in Host Rock (2.2.07.02)		
Short description	Hydrocarbons are part of the natural material inventory of the host rock. In Saline rock hydrocarbons are usually found at grain boundaries or in Anhydrite fractures	
Index	VSG	90 Kohlenwasserstoffvorkommen im Wirtsgestein
	WIPP	Not considered
	PROSA	Not considered
Judgement	• The presence and consequences of hydrocarbons has been assessed in the German program (VSG) • The effects and consequences are judged as minor • The presence of hydrocarbons is one of the requisites of the possibility of thermochemical sulphate reduction	
Open questions	• In relation to the long-term safety assessment: further specification of the types and amounts of hydrocarbons • The effects of hydrocarbons on particular processes such as oxidation, pyrolysis, radiolysis and thermochemical sulfate reduction, as well as the effects of increased HC levels on the geomechanical properties of rock salt	

Hydrochemical Conditions of Host Rock and Surrounding Strata (2.2.08.01)		
Short description	The hydrochemical conditions of the host rock and the surrounding strata are determined by the type and quantity of dissolved substances, the composition of the rock and the resulting physical and chemical properties (pH, Eh) of water.	
Index	VSG	93 Hydrochemische Verhältnisse im Deck- und Nebengebirge
	WIPP	N36 Changes in Groundwater Eh N37 Changes in Groundwater pH
	PROSA	Not a specific FEP
Judgement	• Changes in Groundwater Eh and Changes in Groundwater pH have been eliminated from PA calculations on the basis of low consequence to the performance of the disposal system (WIPP) • FEP may play a role for long-term stability of the upper sealing element of the shaft seal	
Open questions	• No	

Heat-Related Uplift or Subsidence of the Overburden (2.2.10.01)		
Short description	Expansion of the host rock due to heat input from heat-generating waste. Subsequently, after the decay of the heat generation, the host rock contracts.	
Index	VSG	95 Wärmebedingte Hebung oder Senkung des Deckgebirges
	WIPP	W23 Subsidence; presently not applicable (no heat-generating waste)
	PROSA	1.2.15 Uplift and Subsidence
Judgement	• Effect can be estimated (VSG)	
Open questions	• Clarification of open questions arisen from the comparison of the results obtained with two-and three-dimensional models • Recent model calculations based on new exploration results	

Thermomigration (2.2.10.02)		
Short description	Migration of liquids as a result of thermal gradients in the repository. FEP relates to inter-crystalline water inclusions, which may slowly move through the host rock as a result of thermomigration and due to the resulting re-crystallization of NaCl. It may lead to enhanced corrosion when the migrating inter-crystalline water inclusions have come into contact with the waste containers.	
Index	VSG	96 Thermomigration
	WIPP	Not applicable; no heat generating waste
	PROSA	Not addressed
Judgement	• Only applicable under conditions of significant heat generation • Likely less relevant in the Dutch context of long-term surface storage, and the resulting decreased decay heat	
Open questions	• It is unclear how far the evaporation front, and thus the process of thermal migration, can penetrate into the intact rock salt	

Thermal Modification of Carnallite (2.2.10.03).		
Short description	The mineral carnallite contains six molecules of water (40 wt -%), which are expelled during various stages of dehydration as the temperature increases. Expelled water may move towards the emplaced waste and induce corrosion upon contact.	
Index	VSG	97 Thermische Carnallitzersetzung
	WIPP	Not applicable; no heat-generating waste
	PROSA	Not addressed
Judgement	• The presence of carnallite is site specific • Process studied in laboratory tests (VSG) • Effect can be minimized by siting (VSG)	
Open questions	• No	

Melting of Rock Salt (2.2.10.04)		
Short description	The melting of rock salt due to enhanced temperature and pressure.	
Index	VSG	98 Schmelzen des Salzgesteins
	WIPP	Not applicable; no heat-generating waste
	PROSA	Not addressed
Judgement	• Not applicable within the Dutch context due to elongated surface storage and resulting decrease of decay heat • Depending on the salt composition, the melting point at in-site pressure is at least 660 ^{o}C. The highest temperature permitted is restricted by design is at 200 ^{o}C.	
Open questions	• No	

Thermochemical Sulphate Reduction (2.2.10.05)		
Short description	Redox reaction of organic compounds with molecular hydrogen or at elevated temperatures to form sulfates, carbonates, sulfides, water, hydrogen sulfide and CO_2	
Index	VSG	99 Thermochemische Sulfatreduktion
	WIPP	Not addressed
	PROSA	Not addressed
Judgement	• Process only relevant at high temperatures (> 80 ^{o}C): At 90 ^{o}C, the extrapolated half-life of the sulfate is around 210,000 years if no other limiting factors are present (VSG) • Process not relevant in Dutch context of elongated surface storage, and the resulting decrease decay heat	
Open questions	• No	

<table>
<tr><td colspan="3">Thermochemical Sulphate Reduction (2.2.10.05)</td></tr>
<tr><td>Short description</td><td colspan="2">Redox reaction of organic compounds with molecular hydrogen or at elevated temperatures to form sulfates, carbonates, sulfides, water, hydrogen sulfide and CO_2</td></tr>
<tr><td rowspan="3">Index</td><td>VSG</td><td>99 Thermochemische Sulfatreduktion</td></tr>
<tr><td>WIPP</td><td>Not addressed</td></tr>
<tr><td>PROSA</td><td>Not addressed</td></tr>
<tr><td>Judgement</td><td colspan="2">• Process only relevant at high temperatures (> 80 ^oC): At 90 ^oC, the extrapolated half-life of the sulfate is around 210,000 years if no other limiting factors are present (VSG)
• Process not relevant in Dutch context of elongated surface storage, and the resulting decrease decay heat</td></tr>
<tr><td>Open questions</td><td colspan="2">• The consequences of a possible increase in volume during the thermochemical sulfate reduction on the pressure conditions in the salt dome are not yet established
• The possible formation of organic acids as reaction products and, where appropriate, the determination of their influence on waste containers</td></tr>
</table>

<table>
<tr><td colspan="3">Pressure Induced Liquid Infiltration in Rock Salt (2.2.11.01)</td></tr>
<tr><td>Short description</td><td colspan="2">Fluid pressures exceeding the minor principal stress in the host rock can cause an increase in the local permeability of the rock salt (secondary permeability) due to the local expansion of the grain boundaries, and fluids (gases / liquid) can infiltrate in the affected areas.</td></tr>
<tr><td rowspan="3">Index</td><td>VSG</td><td>100 Druckgetriebene Infiltration von Fluiden in das Salzgestein</td></tr>
<tr><td>WIPP</td><td>No explicit FEP</td></tr>
<tr><td>PROSA</td><td>Nor addressed</td></tr>
<tr><td>Judgement</td><td colspan="2">• Effect comes into play during simultaneous convergence and gas formation or fluid entry from the overburden
• Extensively addressed in German programs</td></tr>
<tr><td>Open questions</td><td colspan="2">• The range of the gas or fluid diffusion processes as a result of pressure-driven infiltration in the rock salt to determine a possible contaminant release
• The consequences of throttles along the infiltration path
• The effects of inhomogeneities on the fluid infiltration
• Laboratory and in-situ tests, as well as assessment with numerical codes</td></tr>
</table>

Sorption and Desorption (3.2.03.01)		
Short description	The sorption and desorption of foreign molecules in liquids or on solids. It includes various chemical / physical processes such as adsorption, absorption, ion exchange, and surface precipitation.	
Index	VSG	105 Sorption und Desorption
	WIPP	W61 Actinide Sorption W62 Kinetics of Sorption W63 Changes in Sorptive Surfaces
	PROSA	1.7.2 Chemical Equilibrium Reaction
Judgement	• Sorption, which would serve to reduce radionuclide concentrations, has been eliminated from PA calculations for conservative reasons (WIPP) • Sorption of radionuclides on rock salt is usually insignificant (VSG)	
Open questions	• Better determination of the sorption behavior of some radionuclides or elements and their anionic and cationic compounds, such as iodine • Further development of surface complex models to describe sorption in natural systems (assessment of Kd values and their bandwidths) • Influence of temperature on sorption and desorption under saline conditions • Quantification of the sorption of radionuclides on materials with cement or Sorel phases • Additional studies on sorption of anionic radionuclides • Development of surface complexation models to describe sorption and desorption under saline conditions	

Colloids (3.2.04.01)		
Short description	Colloids are particles or droplets in another dispersion medium (solid, gas or liquid), present in finely divided form. Colloidal systems represent an intermediate state of the two limiting cases of homogeneous (single-phase) mixture and a heterogeneous (multi-phase) mixture.	
Index	VSG	106 Kolloide
	WIPP	W78 Colloid Transport W79 Colloid Formation and Stability W80 Colloid Filtration W81 Colloid Sorption
	PROSA	1.7.2 Chemical Equilibrium Reaction - Colloid formation, dissalution and transport
Judgement	• Extensively studied in WIPP program • Studied in German program (VSG)	
Open questions	VSG: • Contribution of colloids to the total concentration of radionuclides in solutions, especially the actinides • Transferability of laboratory experiments to large-scale heterogeneous systems • Temperature effects on colloid formation (low relevance)	

<table>
<tr><th colspan="3">Complexation (3.2.05.01)</th></tr>
<tr><td>Short description</td><td colspan="2">A complex compound consists of a central atom and a one or more of ligands. Ligands may be molecules or ions. Complexation results in loss of the specific properties of the building blocks.</td></tr>
<tr><td rowspan="3">Index</td><td>VSG</td><td>107 Komplexbildung</td></tr>
<tr><td>WIPP</td><td>W68 Organic Complexation
W69 Organic Ligands
W71 Kinetics of Organic Complexation</td></tr>
<tr><td>PROSA</td><td>1.7.2 Chemical Equilibrium Reaction
3.2.6 Introduced Complexing Agents, Cellulosic</td></tr>
<tr><td>Judgement</td><td colspan="2">• Extensively studied in WIPP program, especially actinide complexes
• Complexation depends strongly on types of waste and matrix materials</td></tr>
<tr><td>Open questions</td><td colspan="2">• Influence of temperature on complex formation under saline conditions
• Inventory of complexing agents in waste with negligible heat generation and its influence on the mobilization of radionuclides
• Degradation of hydrocarbons to complexing agents under site-specific conditions (salinity, temperature, H2 pressure)</td></tr>
</table>

<table>
<tr><th colspan="3">Radionuclide transport in the Liquid Phase (3.2.07.01)</th></tr>
<tr><td>Short description</td><td colspan="2">The radionuclide transport in the liquid phase comprises all types of radionuclide propagation in a liquid transport medium.</td></tr>
<tr><td rowspan="3">Index</td><td>VSG</td><td>108 Radionuklidtransport in der flüssigen Phase</td></tr>
<tr><td>WIPP</td><td>• W77 Solute Transport
• W78 Colloidal Transport
• W90 Advection
• W91 Diffusion
• W92 Matrix Diffusion</td></tr>
<tr><td>PROSA</td><td>• 1.6.1 Advection, convection, dispersion
• 1.6.2 Diffusion
• 1.6.3 Dilution of mass
• 1.6.5 Matrix diffusion</td></tr>
<tr><td>Judgement</td><td colspan="2">• Transport of radionuclides in the liquid phase can only occur if the nuclides have been mobilized. The mobilization of radionuclides depends on a number of factors (see FEP Radionunuclide Mobilization) (VSG)
• If mobilization has occurred advection / dispersion and diffusion are the main transport mechanisms for radionuclides in the liquid phase in a repository (VSG)
• Several aspects affect the RN transport such as absorption, complexation, radioactive decay etc</td></tr>
<tr><td>Open questions</td><td colspan="2">• Several open questions have been addressed at the appropriate FEPs</td></tr>
</table>

Advection (3.2.07.02)		
Short description	Advection concerns the transport of dissolved compounds with the flow of a transport medium (liquid or gas)	
Index	VSG	109 Advektion
	WIPP	W90 Advection
	PROSA	1.6.1 Advection, convection, dispersion
Judgement	• Whether an advective-dispersive transport of dissolved radionuclides takes place from the disposal zone depends on the presence of flow paths before the driving mechanisms (convergence in the mine and gas storage) have terminated. If these processes are no longer effective in the repository, only diffusive transport is considered (see FEP Diffusion).	
Open questions	No	

Mechanical Dispersion (3.2.07.03)		
Short description	Dispersion is a mixing process associated with advection and it is due to differences in the groundwater velocities on a micro scale, which occur in the groundwater flow due to the variability of the permeability of individual pores and the existence of various possible flow paths around the grains.	
Index	VSG	110 Mechanische Dispersion
	WIPP	Not considered
	PROSA	1.6.1 Advection, convection, dispersion
Judgement	• Since a precise mathematical description of the flow paths is difficult to establish, mechanical dispersion is described macroscopically in a similar way as diffusion.	
Open questions	No	

Diffusion (3.2.07.04)		
Short description	Diffusion is the mixing of different substances by the thermally and/or concentration induced motion (Brownian motion) of the involved particles, e.g. ions, atoms, molecules. The combined diffusion and mechanical dispersion (see FEP Mechanical Dispersion) is also referred to as a hydrodynamic dispersion.	
Index	VSG	111 Diffusion
	WIPP	W91 Diffusion
	PROSA	1.6.2 Diffusion
Judgement	• In case the relevant driving mechanisms for advective transport in the repository have been terminated, diffusive transport is the only driving mechanism of RN transport (VSG) • Diffusion transport is usually orders of magnitude slower than the advective transport (VSG) • Whether a relevant diffusive transport in backfilled spaces at small porosities is possible, remains to be studied (VSG)	
Open questions	• Description of diffusion processes in backfilled galleries at small porosities	

<table>
<tr><th colspan="3">Matrix diffusion (3.2.07.05)</th></tr>
<tr><td>Short description</td><td colspan="2">Matrix diffusion refers to the diffusive transfer of solutes from areas where advective transport dominates into a matrix of immobile pore water</td></tr>
<tr><td rowspan="3">Index</td><td>VSG</td><td>112 Matrixdiffusion</td></tr>
<tr><td>WIPP</td><td>W92 Matrix Diffusion</td></tr>
<tr><td>PROSA</td><td>1.6.5 Matrix diffusion</td></tr>
<tr><td>Judgement</td><td colspan="2">• Matrix diffusion has to be considered especially in rocks in which, beside water-bearing fractures with advective-dispersive dominated transport of solutes, an adjacent low permeability hard rock matrix is present
• Matrix diffusion in fractured hard rock is a significant retention process
• A similar interaction is also possible between backfilled galleries and a low permeable EDZ, but so far this has not been investigated.</td></tr>
<tr><td>Open questions</td><td colspan="2">• The impact of the EDZ on matrix diffusion</td></tr>
</table>

<table>
<tr><th colspan="3">Other Transport Processes (3.2.07.06)</th></tr>
<tr><td>Short description</td><td colspan="2">In addition to the processes advection, diffusion, matrix diffusion and mechanical dispersion, other transport processes exist that can enable or influence a transport of radionuclides. These processes are covered in this FEP. Examples are thermal osmosis, chemical osmosis, thermo-diffusion (Soret effect), Dufour effect.</td></tr>
<tr><td rowspan="3">Index</td><td>VSG</td><td>113 Sonstige Transportprozesse</td></tr>
<tr><td>WIPP</td><td>W93 Soret Effect
W94 Electrochemical Effects
W95 Galvanic Coupling (Outside the Repository)
W96 Electrophoresis
W97 Chemical Gradients
W98 Osmotic Processes</td></tr>
<tr><td>PROSA</td><td>Not a specific FEP</td></tr>
<tr><td>Judgement</td><td colspan="2">• In the WIPP CCA/CRA these processes have been screened out on the basis of low consequence (W93, W94, W96, W97, W98) or low probability (W95) to the performance of the disposal system (WIPP)
• A consistent verification of the relevance of coupled thermodynamic transport processes for salt has not yet been carried out yet (VSG)</td></tr>
<tr><td>Open questions</td><td colspan="2">• Assessment of coupled thermodynamic transport processes for salt.</td></tr>
</table>

<table>
<tr><th colspan="3">Radionuclide Transport in the Gas Phase (3.2.09.01)</th></tr>
<tr><td>Short description</td><td colspan="2">This FEP covers all types of radionuclides propagation in a gaseous transport medium.</td></tr>
<tr><td rowspan="3">Index</td><td>VSG</td><td>115 Radionuklidtransport in der Gasphase</td></tr>
<tr><td>WIPP</td><td>W55 Radioactive Gases
W89 Transport of Radioactive Gases</td></tr>
<tr><td>PROSA</td><td>1.6.4 Gas mediated transport</td></tr>
<tr><td>Judgement</td><td colspan="2">• The WIPP FEPs W55 and W89 have been eliminated from PA calculations on the basis of low consequence to the performance of the disposal system (WIPP)
• Since there are in any case radionuclides in the gas phase, a transport of radionuclides in the gas phase and is likely to be considered in all sub-systems (VSG)</td></tr>
<tr><td>Open questions</td><td colspan="2">• Determination of the chemical form of C-14
• Relevance of I-129 in the gas phase
• Relevance of the sorption of radionuclides in the gas phase</td></tr>
</table>

Appendix 2 ICK Site selection criteria

Criteria for the selection of a salt formation in which a GDF was to be constructed were first formulated in 1975 by the sub commission RAS (Radioactieve Afvalstoffen) of ICK, the Interdepartmental Nuclear Energy Commission.[34] In subsequent research coordinated by the ICK these criteria were worked out further and made more specific. In this research 5 working groups were active:

- WG A: financial, organizational and juridical questions
- WG B: investigations of salt deposits
- WG C: questions of mining technology
- WG D: risk analyses
- WG E: interim storage

The final report was presented in April 1979[35]. A summary of the conclusions with respect to the criteria will be given:

a) ICK 1975: The top of the rock salt must lie below the level where at this point groundwater still participates in the hydrological cycle.
ICK 1979: Based on the risk consideration in WG D the criterion was reformulated:

The salt dome must be surrounded by deposits in which the flow rate of the groundwater is smaller than 3 m/a.

b) ICK 1975: The thickness of the salt strata must be at least 500 m.
ICK 1979: For reasons of mining technique the State Mines Inspectorate has pointed out that for any underground disposal facility must be separate from the flanking and covering rock by at least 200 m. This was also the basis for the safety analyses in WG D and therefore the condition was reformulated:

The thickness of the salt formation must be at least 500 m. and the disposal facility must be surrounded in all directions by at least 200 m of salt.

c) ICK 1975: The salt dome must preferably be covered by a layer of so-called caprock consisting of anhydrite, gypsum or other impermeable material.
ICK 1979: The aim of this criterion is to express a preference for caprock consisting of anhydrite or gypsum. Other rocks such as dolomite, sulphur or clay may also give adequate coverage. In order to clarify this significance the criterion was reformulated:

The salt dome must be covered by a layer of so-called caprock consisting of impermeable material, preferably anhydrite or gypsum.

d) ICK 1975: Above the caprock there must be a layer of clay or sandy clay.
ICK 1979: This criterion must be seen in combination with criterion e). There was no reformulation necessary:

Above the caprock there must be a layer of clay or sandy clay

e) ICK 1975: A sealing layer of clay or sandy clay must preferably be present beneath the aquifer that takes part in the hydrological cycle.
ICK 1979: The word preferably means that in nature it is never possible to speak in an absolute sense of a ‘sealing’ layer of clay or sandy clay. In the safety calculations of

[34] ICK 1975. Radioactive wastes in the Netherlands at a nuclear power plant with a capacity of 3500 MWe. Ministry of Economic affairs.
[35] ICK 1979. Report on the feasibilities of radioactive waste disposal in salt formations in The Netherlands. Ministry of Economic affairs, April 1979.

WG D a certain permeability has been taken into account. In reports of WG D it is made clear that in fact between the caprock and the water bearing coarse sandy bottom there should be layers of clay or sandy clay, so that the uppermost aquifer is separated by layers of low permeability from the deeper aquifer. Clarification therefore has been achieved by reformulation:

Between the deposits around the salt dome and the higher aquifers with moderate to good permeability there must be a layer of clay or sandy clay with a low permeability.

f) ICK 1975: In view of salt creep a tectonically quiet salt dome is preferred.
ICK 1979: As tectonically quiet salt domes are salt domes that have shown no uplift since the Upper Tertiary (about 20 Ma ago) the criterion is formulated in more detail:

In view of salt creep, the salt dome must be tectonically quiet, i.e. it must have shown no uplift since the Upper Tertiary (about 20 Ma ago)

g) ICK 1975: The rock salt must be as pure as possible in composition and possess good mechanical properties.
ICK 1979: This criterion is particularly important for the construction of the disposal mine. Based on the views of WG C the criterion is clarified:

The rock salt must be as pure as possible in composition and possess good mechanical properties. In any case, this criterion is satisfied by an average contamination with solid materials that does not exceed 15% and an octahedral shear strength of at least 4,5 MPa

h) ICK 1975: In the rock salt at the site of the future dumping cavity no potassium/magnesium salt strata or clay, limestone, or anhydrite strata must be present.
ICK 1979: This criterion was formulated in relation with the dumping cavities leached out of a salt dome. The criterion can be adapted for disposal in a mine by replacing the words ‘dumping cavity’ by ‘disposal space’. In WG C it has been shown that choosing a suitable disposal pattern of the HLW boreholes the temperature consequences and the field of their action can be limited. Therefore, potassium/magnesium deposits at a certain distance of the HLW no longer represent an obstacle. The criterion thereforeshould be reformulated:

In the rock salt at the site of the disposal space no potassium/magnesium salt strata or clay, limestone, or anhydrite strata must be present[36].

i) ICK 1975: The presence of the inclusion of gas or salt water in the salt demands great caution.
ICK 1979: Inclusions (bubbles) of gas and salt water normally occur only sporadically and are limited in volume. During the operation period of the disposal mine, however, they must be taken into account by counter measures standard in salt mining practise. The criterion is clarified:

The presence of the inclusion of gas or salt water in the salt demands great caution in the construction of the mine.

[36] In ICK 1979 (p. 39) the words ‘dumping cavity’ were not replaced by ‘disposal space’.

j) ICK 1975: There must be no prospecting for bituminous products in or around the salt dome.
ICK 1979: It must be clarified that by prospecting for bituminous products is meant test drilling for oil and natural gas deposits. The reformulation thereforeshould be:

There must be no prospecting for bituminous products in or around the salt dome (test drillings for oil and natural gas deposits).

k) ICK 1975: There must be no tectonic action in and around the salt dome.
ICK 1979: In this criterion ‘tectonic action’ means any serious continuous dislocations (fractures). Since the region to the north of a line running roughly from Den Helder to Arnhem is tectonically quiet, such dislocations are unlikely in north-east Netherlands.

There must be no serious continuous dislocations (fractures) due to tectonic action in or around the salt dome

l) ICK 1975: There must be no concession for drilling purposes, such as the storage of oil, gas or chemical waste.
ICK 1979: No need for modification

There must be no concession for drilling purposes, such as the storage of oil, gas or chemical waste

m) ICK 1975: The salt dome must not be used for other purpose, such as the storage of oil, gas or chemical waste.
ICK 1979: No need for modification

The salt dome must not be used for other purpose, such as the storage of oil, gas or chemical waste

n) ICK 1975: Operations related to the supply of drinking water or plans for them must be taken into account.
ICK 1979: No need for modification

Operations related to the supply of drinking water or plans for them must be taken into account

o) ICK 1975: Great attention must be paid to the prevention of adverse effects on areas of natural interest in consequence of access roads and surface and undergrounds works.
ICK 1979: No need for modification

Great attention must be paid to the prevention of adverse effects on areas of natural interest in consequence of access roads and surface and undergrounds works

p) ICK 1975: Other factors of planological nature must also be taken into account.
ICK 1979: No need for modification

Other factors of planological nature must also be taken into account.

Appendix 3 Number of Runs

For the detrimental effects of a GDF there are in most countries formal criteria requiring that some specific limit must not be exceeded. This can, of course, only be proved by applying the model, taking into account all relevant uncertainties. A strong criterion could therefore cause two kinds of problems:

- Since the distribution of model output values can theoretically extend to infinity, with a finite number of model runs one can never be sure to have actually calculated the highest possible output. A proof of compliance with a strong criterion would therefore be theoretically impossible.
- The requirement that even in very improbable cases the criterion must be met could lead to the necessity of unreasonably expensive measures to avoid situations that most likely would never occur anyway.

Therefore, criteria are normally given in a statistical form that allows a low probability of exceeding the limit value. But even the requirement that, for example, 95 % of all possible model output values have to remain below the limit would still require an infinite number of model runs. For this reason, such criteria are formulated with two probability values, p and q. The probability that the model yields an output value not exceeding the limit value must be at least p with a confidence of at least q. That means that there is a probability of q or higher that the model will yield values below the limit in at least a fraction of p of all possible cases.

For a number n of independent model runs, the confidence can be calculated as

$$q = 1 - F(k; n, 1-p),$$

where k is the number of limit exceedances (k = 0 is required if no exceedance is allowed) and F denotes the cumulative binomial distribution

$$F(k; n, 1-p) = \sum_{i=0}^{k} \binom{n}{i} (1-p)^i p^{n-i}.$$

Such a p/q-criterion can easily be checked by evaluating a relatively low number of model runs, as long as the following points are made sure:

- The parameter sets for the model runs to evaluate have to be purely random (or pseudo-random) and independent of each other. Model runs based on a non-random sample must not be used.
- The model runs for evaluation have to be finally selected prior to knowing their results. It is inadmissible to replace a limit-exceeding run by another one afterwards.
- The results of all evaluated model runs remain below the limit value.

If actually one exceedance of the limit is found, it is not sufficient to run the model once more in order to reach the required number of non-exceeding runs. In such a case a considerably higher number of runs become necessary. The minimum numbers of model runs that guarantee compliance with the criterion can be calculated using the formula given above. For no exceedances and for one exceedance they are listed in Table A3-1 for different values of p and q.

Table A3-1 Minimum numbers of model runs to fulfil the *p*/*q* criterion with no exceedances (black) or one exceedance (red).

	q = 90 %	q = 95 %	q = 99 %
p = 90 %	22 (38)	29 (46)	44 (64)
p = 95 %	45 (77)	59 (93)	90 (130)
p = 99 %	230 (388)	299 (473)	459 (662)

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