Evaluation of current knowledge for building the Safety Case for salt-based repositories

OPERA-PU-NRG221B

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.

OPERA-PU-NRG221B
Title: Evaluation of current knowledge for building the Safety Case for salt-based repositories
Authors: J. Hart, J. Prij, T.J. Schröder, G-J. Vis, D.-A. Becker, J. Wolf, U. Noseck, D. Buhmann
Date of publication: 3 Augustus 2015
Keywords: Safety Case, safety assessment, salt, disposal

Contents

List of figures

List of tables

Summary

The present report describes the current status of the Safety Case of the final disposal of radioactive waste in rock salt[a] in the Netherlands. The evaluation comprises an overview of the state-of-the-art of the safety and feasibility of a final disposal facility in rock salt. In addition, recommendations and a detailed roadmap are provided for the further development of the Safety Case on rock salt in the Netherlands.

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

Each chapter deals with a particular aspect of the Safety Case; at the end of every chapter separate sections summarise and evaluate the main issues. Chapter 8 collects the main outcomes of these evaluations and recommendations, and in Chapter 9 a roadmap is presented that describes how this Safety Case can be realized in a stepwise manner.

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

The main conclusion is that, although currently still quite a lot of loose ends have to be tied off, an initial Safety Case for disposal in rock salt in the Netherlands - comparable to the OPERA Safety Case on Boom Clay - can be built with reasonable efforts (about 5 to 8 man-years), by efficient use of the OPERA results and existing knowledge. This can be achieved in a step-wise manner over a time span of 3 to 5 years. Such an undertaking requires - next to the implementation of methodological improvements of the Safety Case framework - an updated, integrated disposal concept for all currently considered waste fractions, and a subsequent update of the post-closure safety assessment.It is therefore recommended to reconsider the Safety Case related aspects treated in this report, taking into consideration the most recent information from other programmes for salt-based disposal. This will lead to an up-to-date and well-established initial Safety Case for the disposal of radioactive waste in rock salt in the Netherlands.

Samenvatting

Dit rapport beschrijft de huidige status van de Safety Case voor eindberging van radioactief afval in steenzout[b] in Nederland. 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. Daarnaast zijn aanbevelingen gegeven en is er een gedetailleerde ‘roadmap’ uitgewerkt voor de verdere ontwikkeling van de Nederlandse Safety Case voor de eindberging van radioactief afval in steenzout.

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

Alle onderdelen van een Safety Case worden hoofdstuksgewijs besproken. Elk hoofdstuk eindigt met een aparte paragraaf, waarin een samenvatting en evaluatie van de belangrijkste bevindingen van het betreffende onderdeel van de Safety Case wordt behandeld. Hoofdstuk 8 verzamelt de belangrijkste bevindingen van de voorafgaande hoofdstukken, en in Hoofdstuk 9 worden concrete aanbevelingen gegeven voor

[a] In the present report often abbreviated to ‘Safety Case on rock salt’ or ‘salt Safety Case’
[b] In dit rapport vaak afgekort tot ‘Safety Case voor steenzout’ of ‘zout Safety Case’

vervolgstappen om een Nederlandse Safety Case voor de eindberging van radioactief afval in steenzout te realiseren.

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

De belangrijkste conclusie is dat, hoewel er op dit moment nog veel open eindjes moeten worden afgehecht, een eerste Safety Case voor steenzout in Nederland - vergelijkbaar met de OPERA Safety Case voor Boomse Klei - met relatief beperkte inspanningen (ongeveer 5 tot 8 manjaar) te realiseren is, door efficiënt gebruik te maken van bestaande kennis en de resultaten van het OPERA programma. Dat kan op een stapsgewijze manier binnen een periode van 3 tot 5 jaar gerealiseerd worden. Voor de realisatie zal het nodig zijn om - naast de implementatie van de methodologische verbeteringen van de Safety Case aanpak - een vernieuwd, geïntegreerd bergingsconcept te ontwikkelen, waarin alle tegenwoordig beschouwde afvalfracties veilig opgeslagen kunnen worden. Op basis hiervan kan een revisie van de lange-termijn veiligheidsberekeningen uitgevoerd worden.

Daarom wordt aanbevolen om de verschillende in dit rapport behandelde aspecten van de Safety Case te bekijken en zo nodig te herzien, gebruikmakend van de meest recente informatie van andere op steenzout gerichte eindbergingsprogramma's. Dat resulteert in een up-to-date en goed ontwikkelde initiële Safety Case voor eindberging van radioactief afval in steenzout in Nederland.

1. Introduction

1.1. Background

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). A Safety Case comprises the findings of a safety assessment and a statement of confidence in these findings. The Safety Case also acknowledges areas of uncertainty and unresolved issues and provides guidance to resolve these issues in future development stages of the repository.

The Safety Case methodology is also a powerful tool for structuring and conducting research programmes for the disposal of radioactive waste. The structure of a generic Safety Case for a disposal facility in Boom Clay in the Netherlands has been developed in the framework of Task 2.1 of the OPERA programme. 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.

During the last 40 years much effort has been devoted in the Netherlands to the geologic disposal of radioactive waste in rock salt, for example in the framework of the ICK[c] (ICK, 1979), OPLA[d] (OPLA, 1989), and CORA[e] (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).

The present 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 aims to provide the OPERA programme a first Safety Case for the geological disposal of radioactive waste in rock salt[f] in the Dutch context. This Safety Case is based on a review of the state of the art on geological 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 for guiding future activities in accordance with the radioactive waste management strategy in the Netherlands.

1.2. Objectives

The OPERA Salt Safety Case (OSSC) project aims:

- 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,
- to process this current knowledge according to the methodology of the Safety Case for deep geological disposal,
- to identify knowledge gaps in the Dutch Salt Safety Case, and

[c] Interdepartementale Commissie Kernenergie (Interdepartmental Nuclear Energy Commission)
[d] Commissie Opberging op Land (Commission on Onshore Disposal)
[e] Commissie Opberging Radioactief Afval (Commission on Disposal of Radioactive Waste)
[f] In the present report often abbreviated to ‘Safety Case on rock salt’ or ‘salt Safety Case’

- to provide recommendations and detailed guidance for future development of the Safety Case for final disposal of radioactive waste in rock salt in the Dutch context.

1.3. Realization

The present report summarises 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 taken into account. It expends the main aspects of the Dutch salt Safety Case as compiled in the first comprehensive and detailed deliverable of the OSSC project, OPERA-PU-NRG221A (Hart, 2014a).

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. In addition, a detailed roadmap is presented that shows how an initial Safety Case on rock salt, comparable to the OPERA Safety Case on Boom Clay, can be realized in a stepwise manner. The evaluations and recommendation in this report are presented in a compact manner; for a more detailed discussion of the topics see also OPERA-PU-NRG221A.

1.4. Explanation of contents

The objective of the present report is to present and evaluate the main elements of the Safety Case for final disposal of radioactive waste in rock salt within the Dutch context.

Chapter 2 elucidates main aspects of geological disposal and the concept of the Safety Case.

Chapter 3 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 as well as legal national and international commitments and guidance.

Chapter 4 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.

Chapter 5 presents an overview of the characteristics of the waste intended for disposal, 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 6 provides an overview of the safety assessments applied in the previous Dutch studies VEOS and PROSA, compared with those performed in CORA and PAMINA. Additionally, more recent views and developments on safety assessment methodologies have been identified.

Chapter 7 reviews and compares the safety arguments resulting from OPLA and CORA and current insights in process understanding of relevant safety features. Information provided in the documentation from Germany and US has been taken into account, too.

Chapter 8 provides recommendations to proceed to the next phase of the salt Safety Case in the Netherlands.

Chapter 9 presents a roadmap that describes how a Safety Case on rock salt, comparable to the Safety Case on Boom Clay developed in OPERA, can be realized in a stepwise manner.

Appendix 1 contains a set of maps of locations of the Zechstein Group in the Netherlands.

Appendix 2 gives a detailed outline of the roadmap to a Dutch Safety Case on rock salt.

2. Geological Disposal

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 radioactive (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 additionally 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 represent a Safety Case are illustrated in the figures below (NEA, 2004; p.19) (IAEA, 2012; p.17).

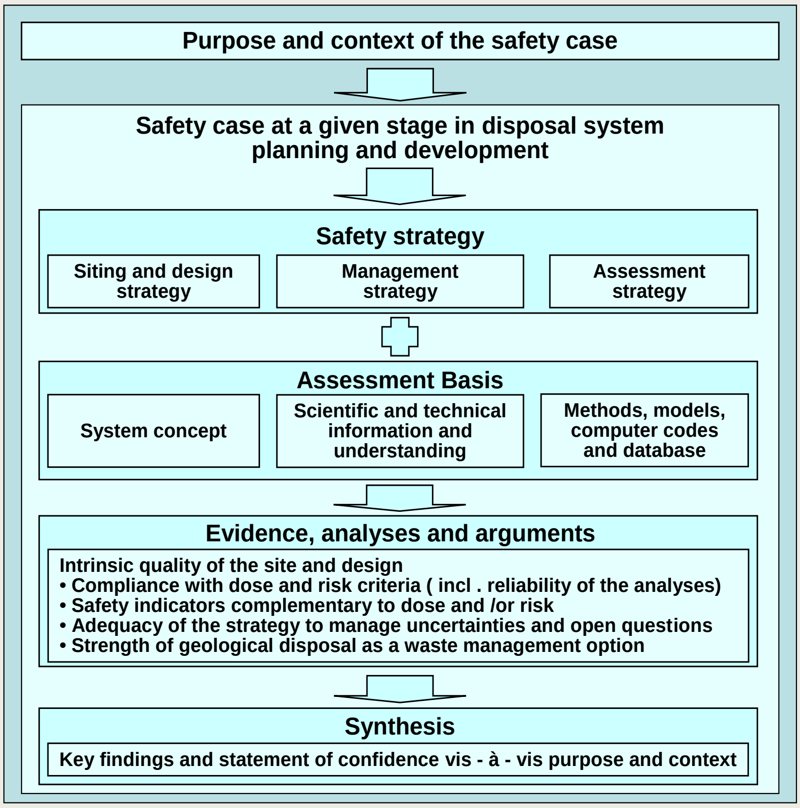


Figure 2-1 Representation of the different stages of a Safety Case (NEA, 2004).

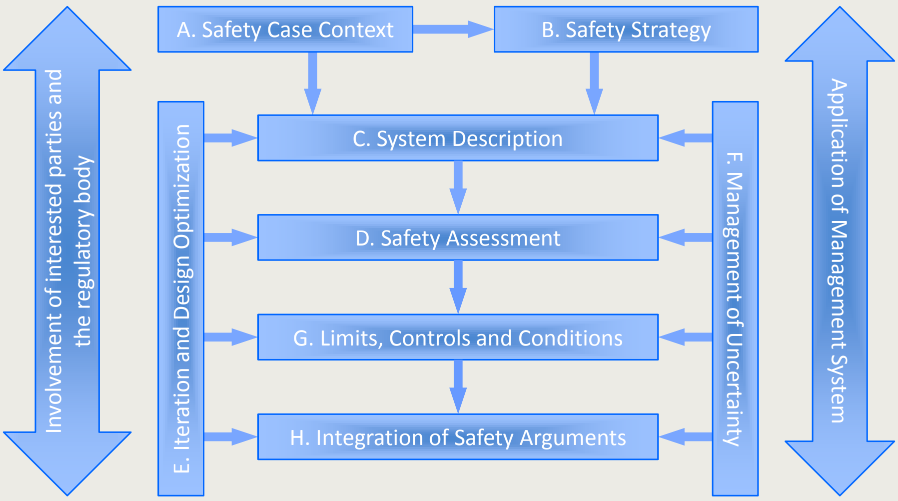


Figure 2-2 Components of the safety case; application of the management system and the process for interaction with the regulatory body and interested parties (IAEA, 2012).

As time progresses and the implementation of a disposal facility evolves (see Figure 2-3; 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 recognised that these decisions will be taken under joint responsibilities of the involved parties, which may change during the disposal programme (IAEA, 2012b).

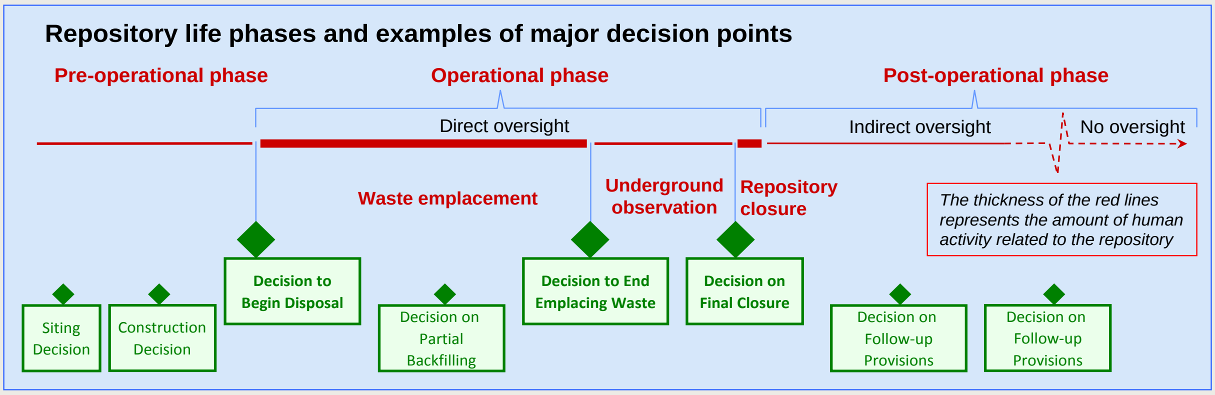


Figure 2-3 Phases of a disposal facility.

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 project 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, during the last 40 years much effort has been devoted to the geologic disposal of radioactive waste in rock salt, for example in the framework of the ICK[g] (ICK, 1979), OPLA[h] (OPLA, 1989), OPLA-1A (RGD, 1993) and CORA[i] (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 an initial Safety Case for the geological disposal in the Dutch context. This Safety Case is based on a review of the state of the art on geological disposal of radioactive waste in rock salt 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.

[g] Interdepartementale Commissie Kernenergie (Interdepartmental Nuclear Energy Commission)
[h] Commissie Opberging op Land (Commission on Onshore Disposal)
[i] Commissie Opberging Radioactief Afval (Commission on Disposal of Radioactive Waste)

3. Safety Case Context

3.1. 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.

3.2. Nuclear profile of the Netherlands

The Netherlands has a small nuclear programme, with only one 483 MW_e 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 MW_{th} 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 MW_{th} 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 3-1 (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 3-1 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, 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 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 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. Spent fuel from any new-built research reactors will also be transferred to COVRA for long-term surface storage.

3.3. Types of waste to be disposed of

The Dutch policy on the management of spent fuels from nuclear power plants 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 past the NPP operators have decided in favour of reprocessing the spent fuel for economic reasons, reuse of plutonium and reduction of the waste volume. After reprocessing of the NPP spent fuel at Areva, France, the remaining vitrified high-level wastes (HLW) are returned to the Netherlands (MinEZ, 2013; p. 20).

In addition, also spent fuel is presently being produced in the Netherlands; it 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.

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) from research reactors;
- 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 various types of radioactive waste intended for final disposal is provided in Section 5.3.

3.4. Legal framework

The treatment of radioactive waste is determined to a large extent 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 following section gives a summary overview of the Dutch legal framework.

3.4.1. 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 3-1.

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. The law sets out the basic rules on nuclear energy, to make provisions for radiation protection, designate the various competent authorities, and outline their responsibilities.

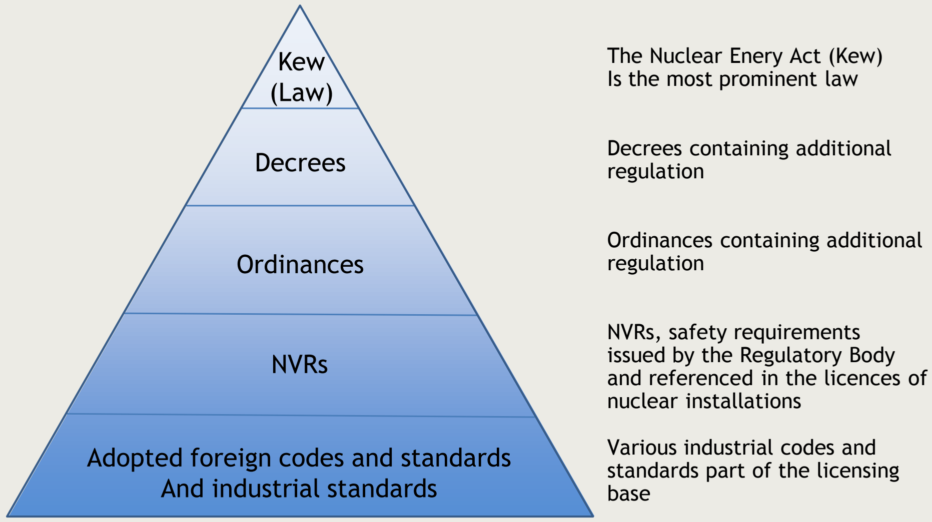


Figure 3-1 Simplified representation of the hierarchy of the Dutch legal framework.

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.

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).

3.4.2. 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).

At present the Authority for Nuclear Safety and Radiation Protection (Autoriteit Nucleaire Veiligheid en Stralingsbescherming, or ANVS), is the responsible RB for regulating the nuclear sector and radiation protection. The ANVS falls under the responsibility of the Minister of Infrastructure and the Environment (I&M). The structure of the ANVS is depicted in Figure 3-2[j].

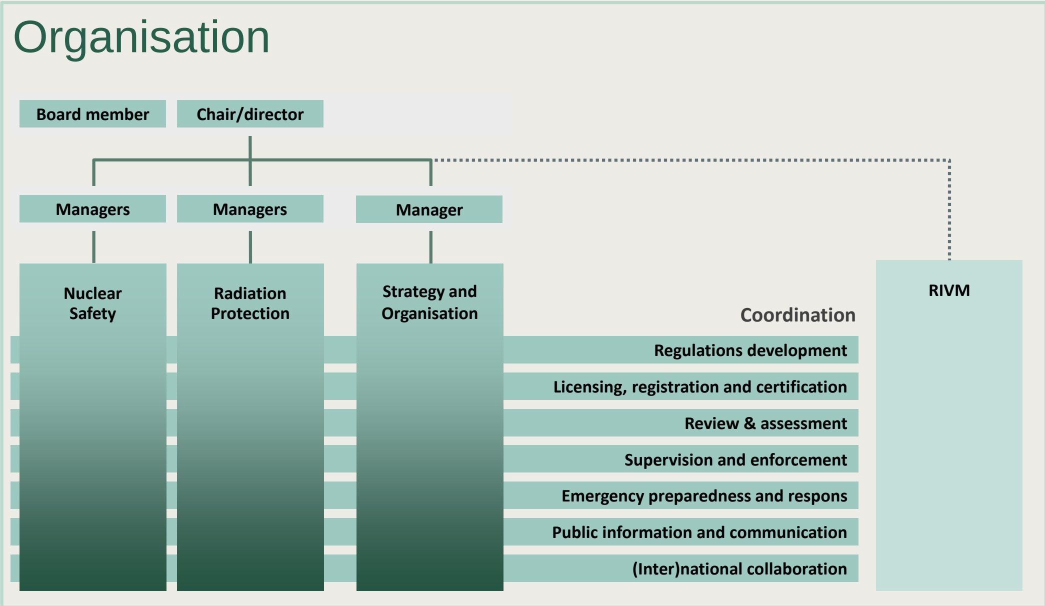


Figure 3-2 Structure of the new RB, the ANVS.

The RB meets international standards, including those published by the International Atomic Energy Agency (IAEA). The 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 radiative material.

The RB has access to external specialists of Technical Support Organisations (TSOs) or other consultancy organizations. Some major supporting organizations are the *Nuclear Research & consultancy Group* (NRG) in Petten and Arnhem, and the *Gesellschaft für Anlagen- und Reaktorsicherheit* (GRS, Germany), providing consultancy and training services to government and industry.

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 organise the implementation of the basic laws. That said, the information provided in this section may be revisited on a regular basis.

At present it cannot be foreseen how the laws and responsibilities will change in the next decades. As time progresses, and the prospects of an implementation of a deep geological disposal facility will come closer it is mandatory to further develop appropriate regulations.

3.4.3. Legislation and geological disposal

The main line of the present Dutch policy is to isolate, control, and monitor radioactive waste in above ground structures for at least a hundred years, after which geological

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

disposal is foreseen. The Dutch policy on the geological disposal of radioactive waste 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).

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 be implemented. 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 (EU, 2011).

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).

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.

3.4.4. 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, for example by drafting the required 'National Programme' according to the definition provided by this Directive.

3.5. 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).

3.6. Evaluation

At present no major changes are foreseen for the Dutch nuclear profile: except the elaboration of the PALLAS research reactor.

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 organise 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.

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). This flexibility especially relates to the consideration of surface disposal of low and intermediate level radioactive waste, as well as the possibility of a joint final disposal facility with other countries, and opens up an important deviation from the *Policy Document*.

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.

4. Safety Strategy

4.1. *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 formations. 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 have recently been described in detail in (Bollingerfehr, 2013) and (US DOE; 2011; Section A.5.1) respectively.

4.2. *Basis of the Dutch waste management strategy*

The 1984 Governmental policy plan (VROM, 1984) together with the policy statement on retrievability (VROM, 1993), form the basis for the Dutch strategy principles, which can be summarised 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 summarised 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.

4.3. *Present views on the Dutch waste management strategy*

In the Netherlands, the development of a geological disposal facility for radioactive waste will last more than a century. Based on the present Dutch nuclear profile (cf. Section 3.2) 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 concerning 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 (IAEA, 2011; p.19). Figure 4-1 shows the common elements in the decision-making processes on geological disposal and the planned timing for the Netherlands that follows from the current Dutch policy (Verhoef, 2014; Figure 1). Indicated on the right are the stakeholders involved in the decision.

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.

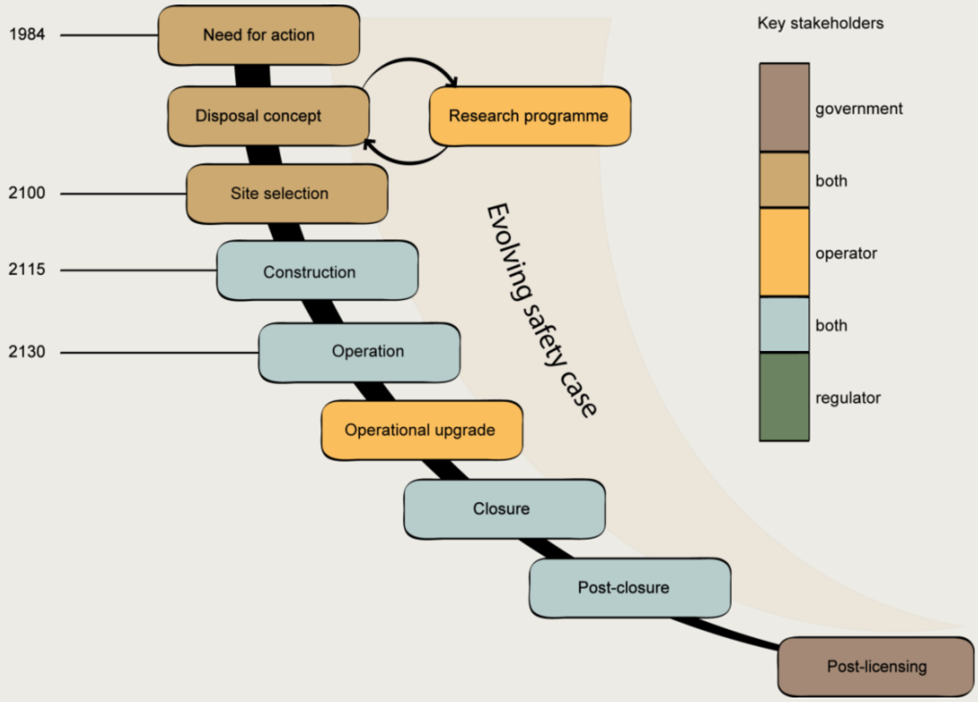


Figure 4-1 Common elements in the decision-making processes on geological disposal including the timeline for the Netherlands.

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

4.4. 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. 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 conditions is given below.

- *The ICM criteria (isolate, control and monitor) form the basis of the radioactive waste management policy.*
- *Radioactive waste is stored above ground for a period of at least 100 years.*
- *A single organisation (i.e. COVRA) has been established for management of all steps of the radioactive waste management process.*
- *In addition to a national geological disposal facility (GDF), the option of a multinational GDF is not excluded.*
- *All radioactive waste is intended to be disposed of in a single, deep GDF operating in 2130.*
- *The GDF has to be designed, operated and closed such that the process is reversible and the waste is retrievable.*
- *Both rock salt and clay formations are being considered as potential host rocks for geological disposal in the Netherlands.*
- *Specific regulatory criteria for the siting or the performance of a geological disposal facility have not yet been defined.*
- *The public has to be given the necessary opportunities to participate effectively in the decision-making process regarding radioactive waste.*

On the basis of these boundary conditions strategic measures have been engaged, which are elucidated in the following section.

4.5. 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, but they are also applicable 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 GDF will be constructed within a Tertiary Clay formation or Zechstein rock salt formation.*
- *The materials and implementation procedures should not unduly perturb the safety functions of the host formation, or of any other component.*
- *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.*
- *Waste types will be divided into groups to be emplaced in separate sections of the geological disposal facility.*
- *The various 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 procedure.*
- *Geological disposal planning will assume that surveillance and monitoring will continue for as long as deemed necessary.*
- *There are preferences for using shielded waste packages that minimise operations and consequent operational radiation doses in the underground.*
- *There are preferences for materials and implementation procedures for which broad experience and knowledge already exists.*

4.6. Siting strategy

Considering the present stage in the decision-making process in the Netherlands, no siting of a radioactive waste repository is foreseen. The efforts 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.

4.7. Safety concept

The safety concept for geological disposal is the understanding of why the disposal system is safe, irrespective of identified uncertainties and unfavorable phenomena. 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). 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 is discussed in Chapter 5 of this report.

4.7.1. The Dutch Safety Concept

At present, the Dutch safety concept for the geological disposal of radioactive waste in rock salt is less well developed than the Boom Clay concept. The Boom Clay concept 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. However, the concept of safety functions for salt-based repositories in the Dutch context have not yet been established. The primary reason is that the consideration of safety functions has emerged only after the previous Dutch national disposal programmes OPLA and CORA programme have ended.

An overview of aspects related to the Dutch safety concept in the framework of disposal in a salt-based repository is summarised below.

Multiple barriers

The principle of a multi-barrier system was already recognised in the OPLA (see Figure 4-2; OPLA, 1989; p. 34) and PROSA studies (Prij, 1993; p. 0.11). 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 (OPLA, 1989; p. 35).

The geological disposal concept in Boom Clay, adopted in OPERA, also relies on a sequence of complementary and/or redundant barriers (defence-in-depth, see also Figure 4-2, right) (Verhoef, 2011a; p. 8). These barriers can be natural (geological) and man-made (engineered). 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. Consequently, the principle of a multi-barrier system in salt formations is still applicable and does not require an update at present.

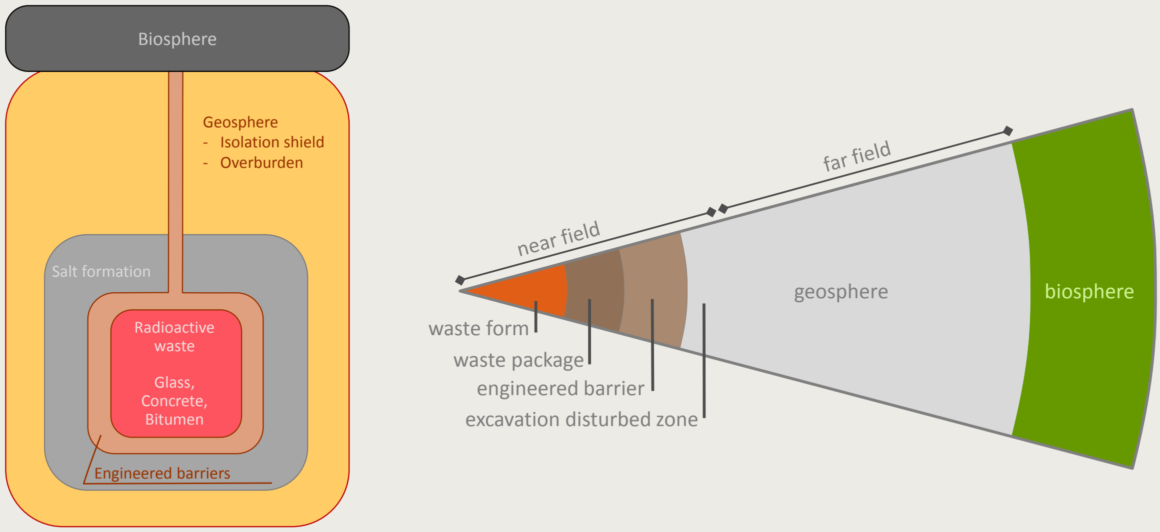


Figure 4-2 Principle of a multi-barrier system in salt formations (OPLA, left), and in OPERA (right).

Safety Functions
In the OPLA and PROSA reports, no specific safety functions were explicitly mentioned to characterise the above-mentioned barriers of the multi-barrier system[k]. 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).

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. In order to be in line with recent developments safety functions need to be established for a salt-based disposal concept.

Retrievability
The Dutch government issued a policy directive in 1993 stating that underground disposal of highly toxic waste (including radioactive waste) was permissible in the Netherlands, provided that it remains retrievable over the long term (VROM, 1993).

Whereas no explicit legislation or (practical) guidelines for waste retrieval have been developed in the Netherlands, the general concept is discussed internationally and worked out in recent years 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).

Within the CORA program, the Dutch METRO concept included some features to facilitate the retrieval of emplaced waste canisters, for example the TORAD-B design comprising a steel liner for enhanced stability. These features are discussed in Section 5.2.3.

4.7.2. The German safety concept

Within the German R&D project ISIBEL[l] 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 barriers for safe containment are the rock salt, the drift seals and the shaft 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 (see Figure 4-3 ; Bollingerfehr, 2013; p.20).was upgraded and described in more detail in the R&D project VSG[m] (Mönig, 2012).

Based on the safety principles set out in the German "*Safety Requirements governing the final disposal of heat-generating radioactive waste*" (BMU, 2010), three **guiding principles** have been defined as follows:

- the radioactive waste must be contained as widely as possible in the containment

[k] 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.
[l] ISIBEL: Überprüfung und Bewertung des Instrumentariums für eine sicherheitliche Bewertung von Endlagern für HAW
[m] VSG: Vorläufige Sicherheitsanalyse für den Standort Gorleben

providing rock zone (CRZ[n]),

- 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.

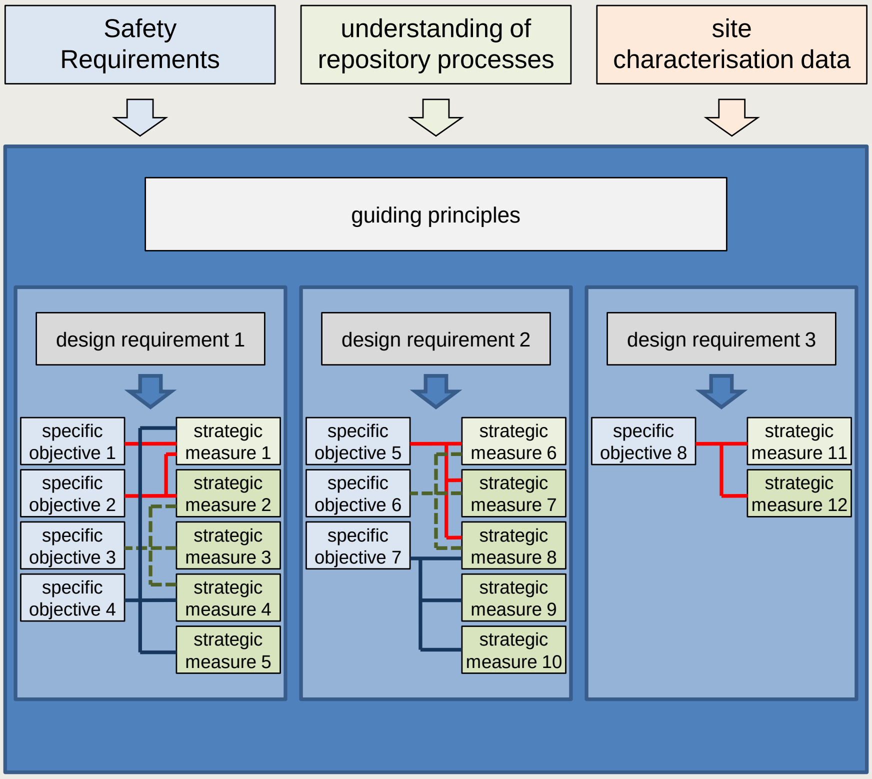


Figure 4-3 German principle approach to derive specific objectives and strategic measures.

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 **design requirements** were defined as follows (Bollingerfehr, 2013; Section 3):

- **Containment**: The emplaced waste canisters shall be enclosed quickly and as tightly as possible by the salt;
- **Performance of CRZ**: During the safety 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 for developing design specifications, for example with respect to the mine position in the salt dome. The strategic measures together meet the objectives of the safety.

[n] The English translation of the German Safety Requirements uses the expression „isolating rock zone“ and defines this zone as the part of the repository system that, in conjunction with the technical seals ensure the containment of the waste. Since this zone refers explicitly to the safety function “containment”, the term “containment providing rock zone”, CRZ, is used in this report.

4.7.3. The WIPP Safety Concept

In the United States the terms “safety concept” and “safety functions” are not commonly used. Instead, the term “performance goal” is utilised. In the US, an assessment of repository safety after closure addresses the ability of a site and repository facility to meet safety standards.

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 in such way that, working in combination with natural barriers, releases of radionuclides into the accessible environment and radiological exposures are within the limits specified by US regulation (US-NRC, 2013).

Moreover, the presently operating Waste Isolation Pilot Plant (WIPP) near Carlsbad, New Mexico, is required to have disposal system barriers to meet specific regulatory requirements of 40 CFR § 194.44(a)[o] 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 WIPP licence application, the so-called Compliance Certification Application (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). Magnesium oxide, MgO, is used in the WIPP where it acts as an engineered barrier by decreasing actinide solubilities through the consumption of carbon dioxide possibly produced by microbial activity (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
- Utilise available construction methods and materials

The functions allocated to the shaft were assessed as part of a study to investigate the feasibility and suitability 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).

The WIPP safety concept has been translated into a design of the actual repository, which is elucidated in more detail in Section 5.2.5 “*Designs considered in US*” of the present report.

4.8. Approach to managing 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 part of the safety strategy (IGSC, 2011; p. 61) (NEA, 2013; p. 8). Within a step-by-step approach to the disposal facility

[o] 40 CFR 194.44 - Engineered barriers; http://www.law.cornell.edu/cfr/text/40/194.44

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.

Sources of uncertainty can be categorised into the following 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 (e.g. NEA, 2012; p. 40). These three categories, unlike the last two mentioned, have also been considered in the Dutch OPLA and CORA programmes.

Strategies of treating uncertainties within the safety assessment are well established. Generally, the following approaches may apply (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 adopting 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.

Additional information on 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.

4.9. 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 be undertaken in mock-up facilities or on the site of the disposal facility, either at the surface or underground.

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 (summarised in Buhmann, 2008) and VSG (Fischer-Appelt, 2013) 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 the Netherlands, Germany and US.

4.9.1. 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 various mining techniques and disposal concepts (e.g. salt mines, deep boreholes). A scenario analysis has been performed and several aspects of safety have been considered in OPLA:

- Scenario uncertainties;
- Thermal and mechanical aspects of rock salt;
- Transport of radionuclides through the subsurface;
- Radiation damage in NaCl (radiolysis);
- 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 research facilities 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 investigated, 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;
- 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 for geological disposal has been summarised in (Hart, 2015).

4.9.2. Safety Demonstration concept – Germany

The fundamentals of the German safety demonstration concept are specified in the German Safety Requirements (BMU, 2010). A first implementation of this 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 validating the integrity of the geotechnical and geological barriers, and evaluating the radiological consequences for future evolutions of the disposal system.

The safety concept and the elements of the safety demonstration were further refined in the project VSG (Mönig, 2012), see Figure 4-4 (Bollingerfehr, 2013; p.58). On the basis of a comprehensive assessment of uncertainties, particularly in relation to scenario uncertainties, the containment of the waste shall be evaluated and shall include the assessment of:

- the permanence of the CRZ,
- the integrity of the geological and geotechnical barriers for probable scenarios,
- the releases of radionuclides from the CRZ employing a suitable radiological safety indicator for probable and less probable scenarios (see also Section 6.8.1).

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 presently 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 4-4. The red elements form the safety analysis, the core of the safety assessment.

The safety demonstration concept was developed for salt, but the basic elements in Figure 4-4 must be assessed for any host rock type considered for a HLW repository in Germany.

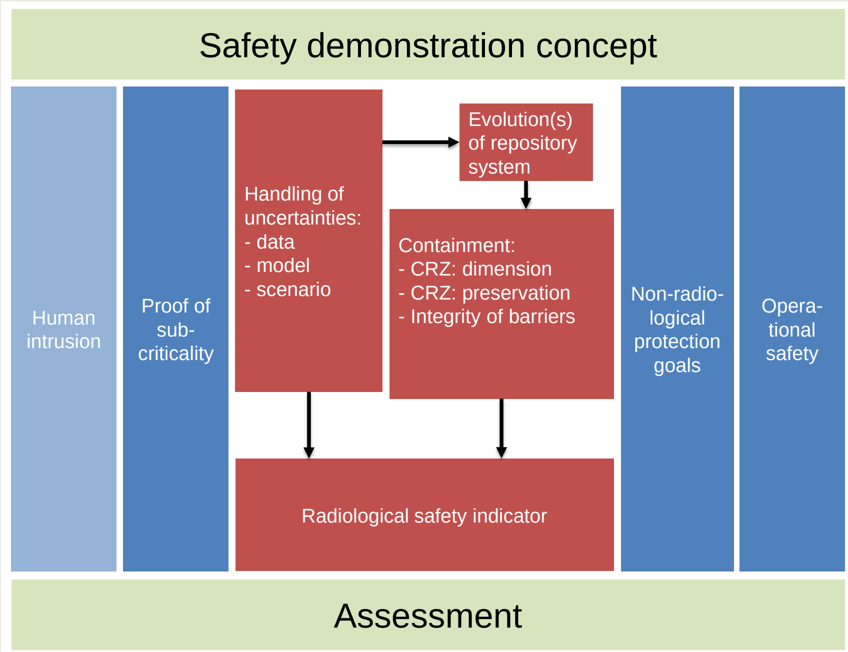


Figure 4-4 Principle Elements of the German safety demonstration concept.

4.9.3. 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 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 (MacKinnon, 2012; p. iii). 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 program, 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 concluded in (MacKinnon, 2012; p. 22) that the 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 prioritises test activities that include laboratory, field, and underground investigations.

4.10. 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 in the development of the Safety Case for 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 safety-related features of the final disposal in rock salt such as the multi-barrier system, the designation and effectuation of the appropriate safety functions, and aspects related to the Dutch retrievability requirement, needs to be established.

A critical review of the Dutch safety concept of final disposal in rock salt should take into account recent international developments, especially in Germany and US, and iterate with the most recently adopted Dutch strategy for the final disposal of radioactive waste.

5. System Description

5.1. Objective and Scope

This chapter provides an overview of the knowledge base related to the various components of the disposal system. It is captured under the Safety Case component "System Description", and includes the various facility designs considered in the past and at present, the characteristics of the waste to be disposed of, the salt formation wherein the facility is constructed, the surrounding and overlying sediments on the salt formations and the biosphere. It must be realised that in the Netherlands until now only generic designs have been considered in generic salt formations without site specific information.

5.2. Facility Designs

5.2.1. Early design studies

The radioactive waste related research started 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[p] and diapirism[q].

Based on generic safety evaluation and of conceptual designs as illustrated in Figure 5-1 (Hamstra, 1981; p.108[r]) it was concluded that the risk associated with these natural processes could be made sufficiently low by selecting a suitable site (ICK, 1979). Additionally, a rather detailed list of criteria was established for the selection of a salt formation (ICK, 1979).

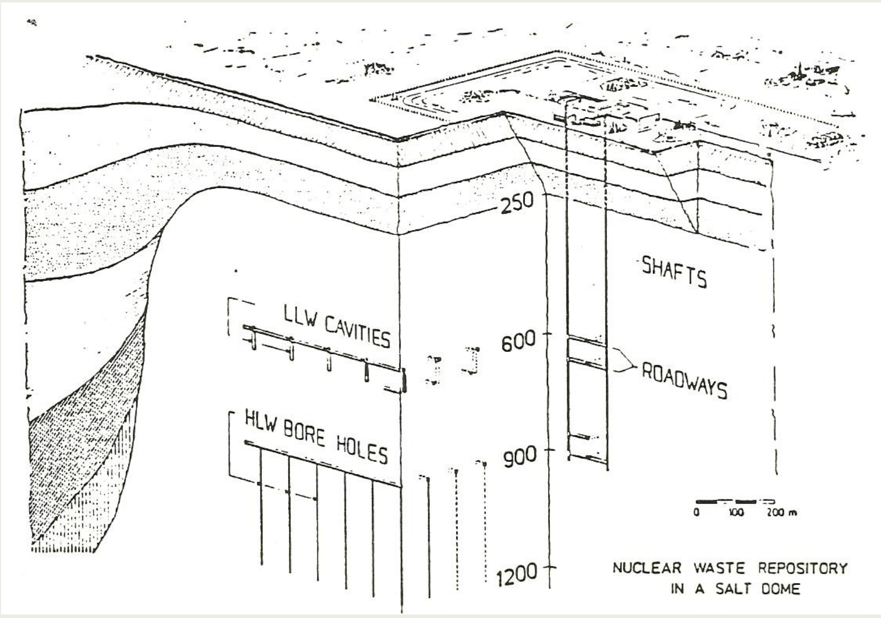


Figure 5-1 Artist's impression of a nuclear waste repository (1981).

[p] Subrosion: dissolution of the salt formation by the groundwater.

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

[r] 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).

It was also recognised that the isolation of the waste could be disturbed by interactions between the salt and the waste; specifically due to the heat generation of a portion of the HLW resulting from the spent fuel of the NPP's.

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 1984 the committee OPLA (OPberging te LAnd; Dutch for disposal onshore) launched its research proposal setting forth how research concerning a potential geological radioactive waste repository should be arranged in order to arrive at a sound decision. OPLA *Phase 1* covered research, comprising laboratory investigations, office studies and participation in foreign projects, *Phase 2* covered research in support of Phase 1, e.g. field explorations such as soil drilling for groundwater studies. *Phase 3* covered field explorations more specifically concentrated on the appropriate geological formation.

The OPLA final report (Prij, 1989) concluded that about 20 sites would be considered suitable 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 finished in 1993.

5.2.2. Designs considered in OPLA

In OPLA three generic salt formations were selected as potential host rocks. Schematic pictures are given below in Figure 5-2 (OPLA, 1989; p. 34).

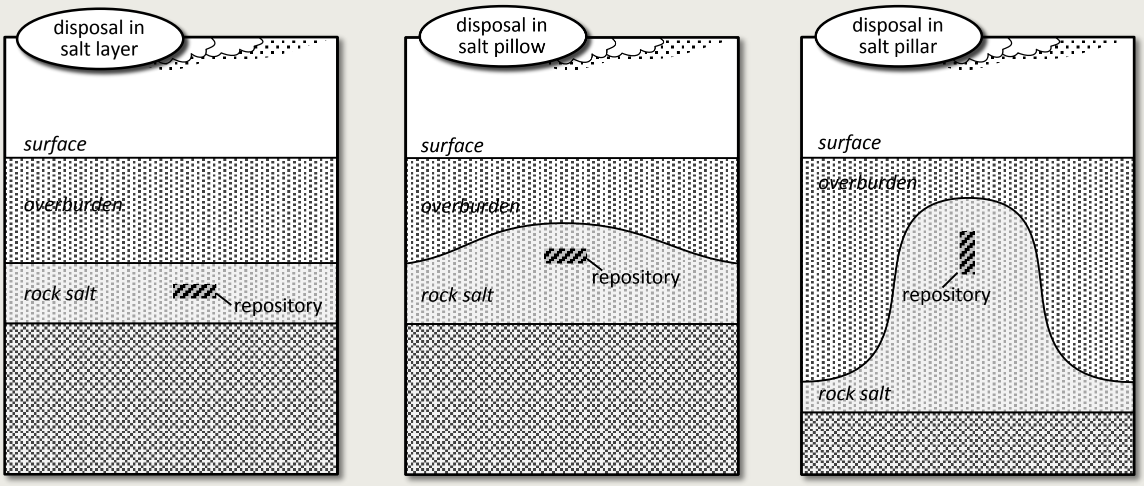


Figure 5-2 Generic salt formations considered in OPLA (1989).

The disposal techniques to accommodate the various considered types of waste (see Section 5.3.1) studied in the OPLA research programme were (Figure 5-3; OPLA, 1989; p. 27):

- A conventional mine, to be constructed in a salt dome or a salt pillow, using standard mining techniques and consisting of boreholes for all heat-generating waste, and chambers or caverns for the remaining medium and low level waste.
- Deep boreholes and caverns. The deep boreholes are mined from the surface and meant for the heat generating high level waste. The caverns, leached from the surface,

are meant for all remaining waste. This disposal technique can be applied in all type of formations.

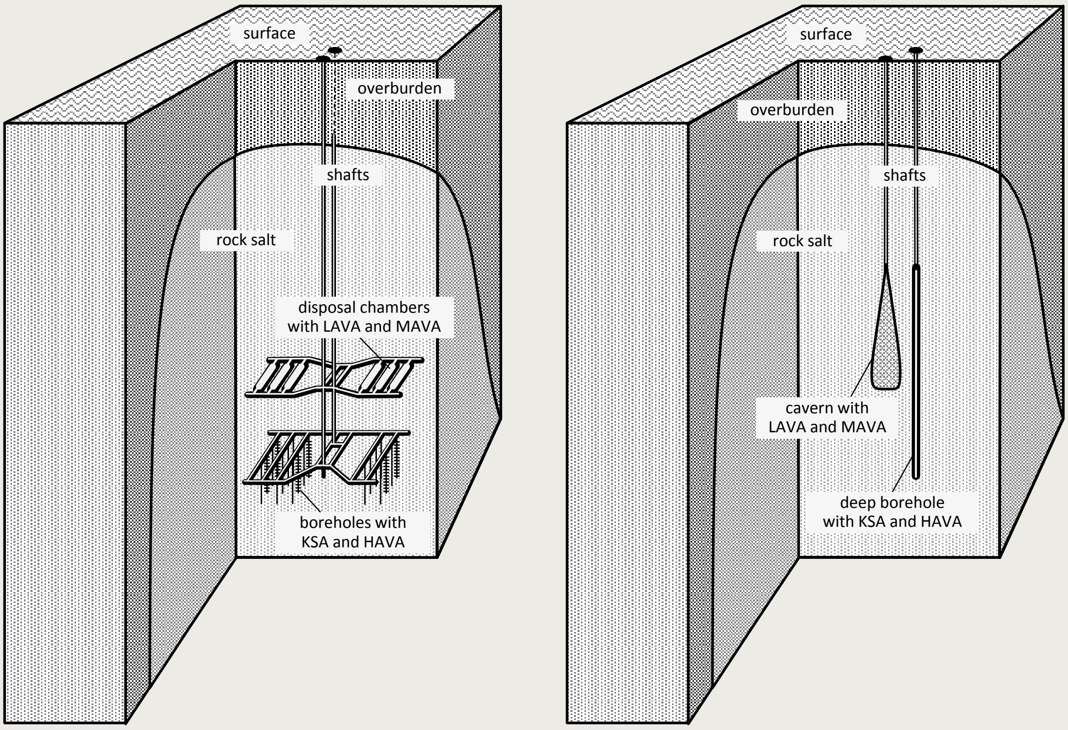


Figure 5-3 Schematic view of the disposal techniques studied in OPLA.

To support the design work an extensive research programme was carried out consisting of experimental as well as theoretical work. The experiments were performed in the research facilities in the salt production mine Asse II near Braunschweig (Germany). This work (Prij, 1991b) focused on:

- 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 effect of the radiation on the rock salt (Hartog, 1988).

The theoretical support for the (long term) safety of the various designs for the disposal of all categories of waste, viz. HLW, ILW and LLW[s], are nowadays referred to as the VEOS (Veiligheids Evaluatie van Opbergingsconcepten in Steenzout; (Prij, 1989), and PROSA (PRObabilistic Safety Assessment; Prij, 1993) studies. Chapters 6.2 and 6.3 of the present report summarise highlights of VEOS and PROSA.

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

[s] HLW, ILW, LLW: High, Intermediate, and Low Level radioactive Waste

5.2.3. Designs considered in CORA

As part of the CORA programme a disposal concept has been developed, that takes retrievability explicitly into account: the TORAD-B (Poley, 2000a; see also Figure 6-6) and METRO-I design (see Figure 5-4; Vate, 2001; p.13). The TORAD-B design comprises a steel liner for enhanced stability and to facilitate the retrieval of emplaced waste canisters.

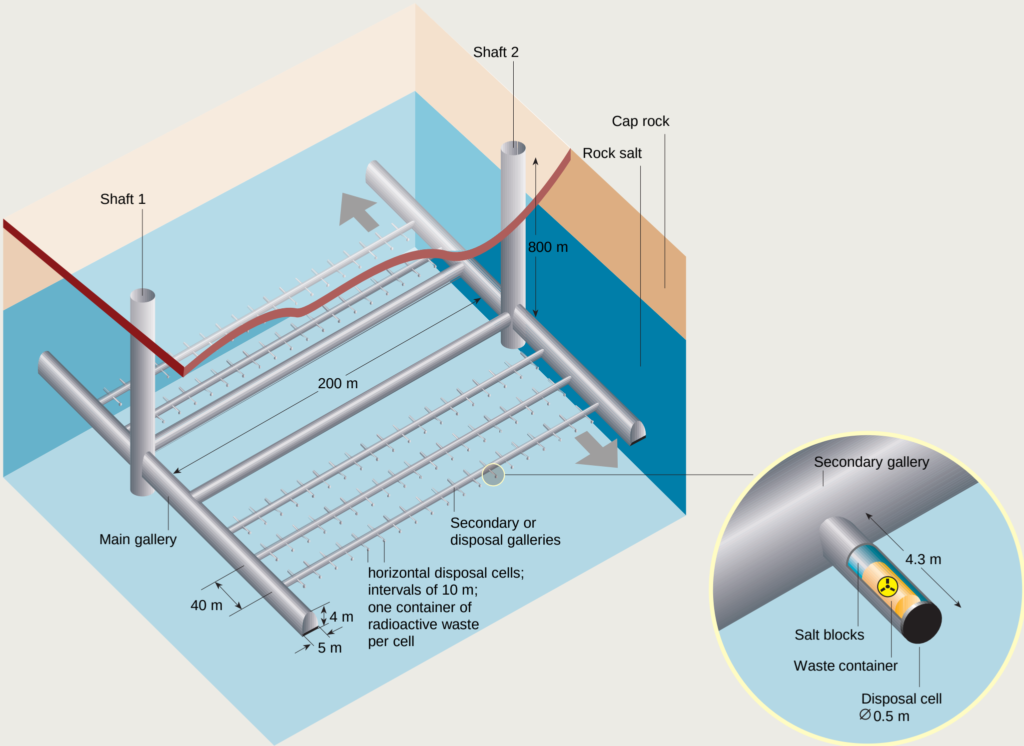


Figure 5-4 Lay-out of the METRO-I disposal concept.

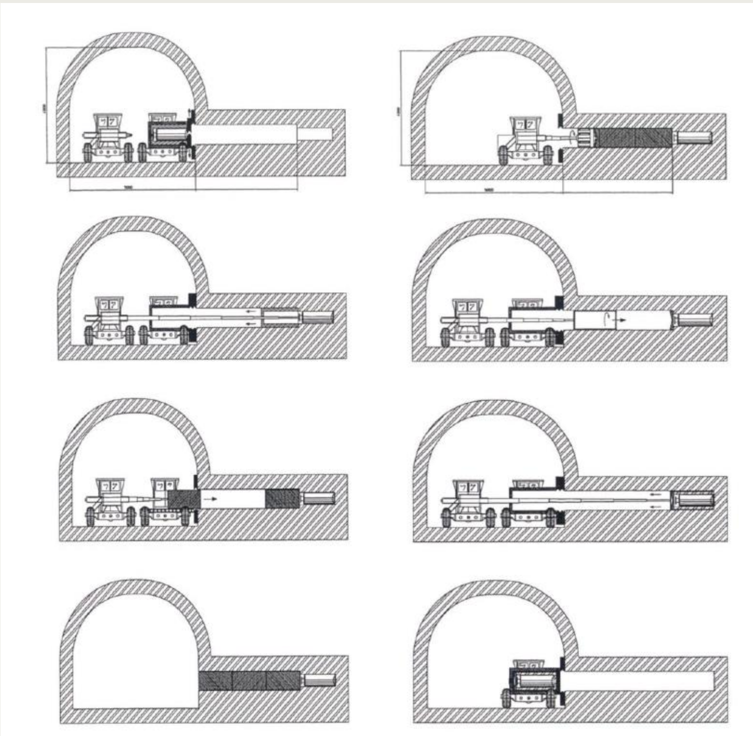


Figure 5-5 Schematic view of the CORA retrievable disposal design CORA. Left the emplacement, and right the retrieval of an HLW canister.

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 design considered in CORA consisted of short horizontal boreholes (Heijdra, 1997), drilled from a gallery as depicted in Figure 5-4 (Vate, 2001; Figure 2). A schematic of the emplacement and retrieval of canisters containing heat-generating high level waste (HLW) is depicted in Figure 5-5 (Grupa, 2000; Figure 3.1 and Figure 5.1). The consequences for the safety of the METRO-I disposal concept have been analysed in CORA-4 (Grupa, 2000), see also Section 6.4 of this report.

5.2.4. Designs considered in Germany

According to the general approach of the safety concept in Germany there are two main design requirements for the final disposal of radioactive waste. The first is to enclose the emplaced waste canisters as quickly and as 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. Taking into account these requirements, two variants of a repository in the Gorleben salt formation were developed (Bollingerfehr, 2013; p. 38).

Variant 1: Emplacement of all heat-generating radioactive waste in horizontal drifts, or horizontal boreholes.

Variant 2: Emplacement of all heat-generating radioactive waste in different types of retrievable canisters in long vertical boreholes.

Additionally, in both variants the emplacement of radioactive waste with negligible heat generation in a separate area of the salt dome in horizontal emplacement chambers was considered, see Figure 5-6.

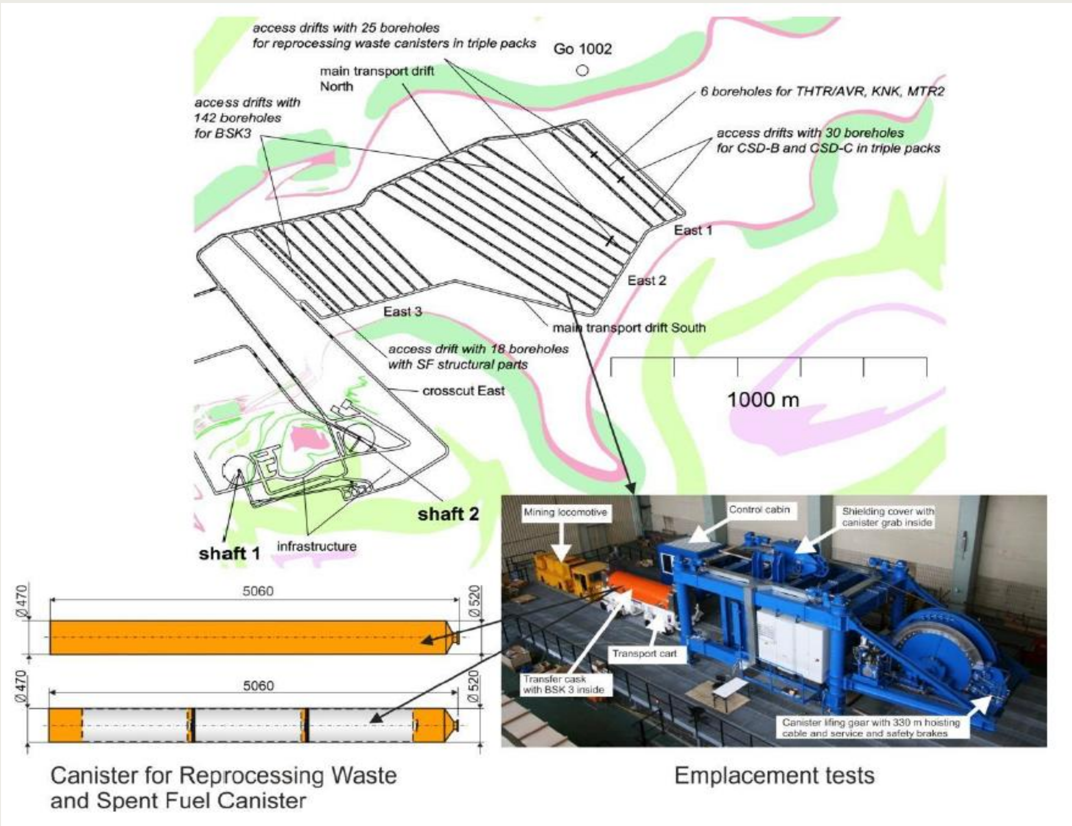


Figure 5-6 German repository design variant 2: Emplacement of all heat-generating waste in lined vertical boreholes (adjusted to the assumed geological structure).

Details of the designs were determined with thermo-mechanical analyses, providing borehole and drift distances, and demonstrating that the temperature limit of 200 ^{o}C at the contact between the borehole liner and the salt will not be exceeded. An important aspect is that the mine is adjusted to the assumed geologic structure of the Gorleben salt dome at the emplacement level.

Figure 5-6 also shows a sketch of a canister for reprocessing waste and a spent fuel canister as well as a photograph of the test set-up for the full-scale canister emplacement demonstration tests in vertical boreholes that were successfully performed in a surface facility.

5.2.5. WIPP facility in US

For accommodating transuranic (TRU) radioactive waste in the US the Waste Isolation Pilot Plant (WIPP, see Figure 5-7) was constructed in the early 1980s, following an extensive site selection study which concluded with the current selection near Carlsbad, New Mexico. The WIPP site is located at a depth of 655 m (2,150 feet) below ground and has the following favourable geological and hydrological characteristics (US DOE, 2004; p. 1-2).

- repository excavation is relatively easy;
- the host rock is impermeable and contains little interstitial brine;
- the host rock formation will eventually encapsulate the waste because of the formation's plastic behaviour;
- dissolution effects of materials in the waste are minimised and predictable;
- future development of resources (oil, minerals, potable water) is thought to be minimal and predictable;
- groundwater flow is minimal and predictable;
- permanent surface waters are absent;
- the host rock is very likely not to be affected by long-term climate changes.

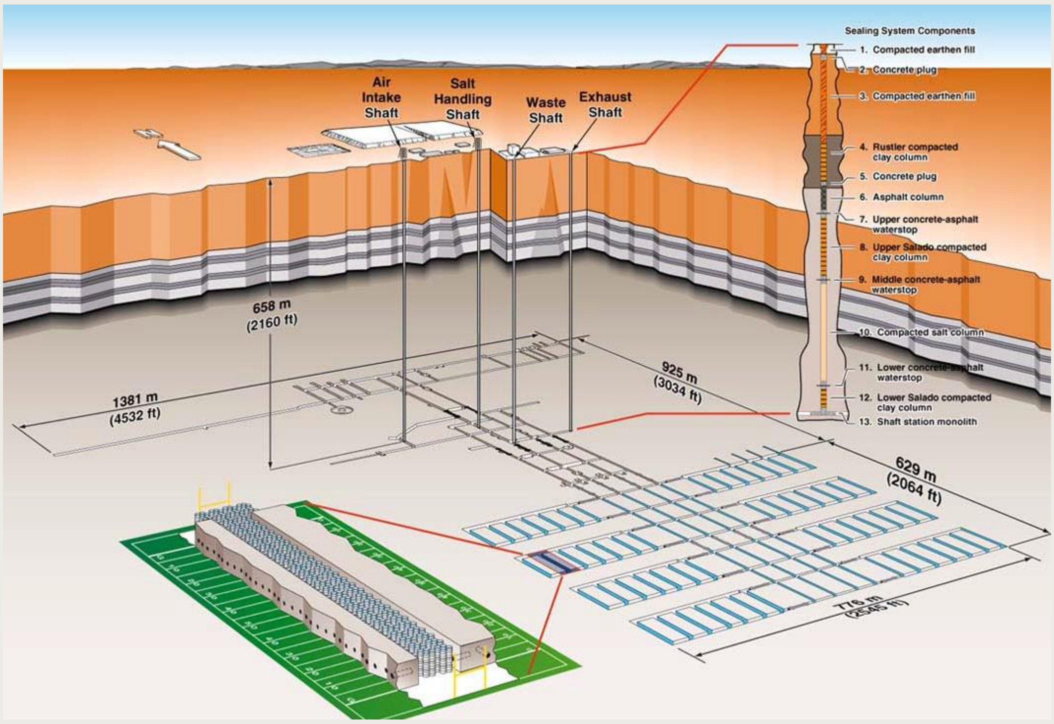


Figure 5-7 Layout of the Waste Isolation Pilot Plant.

The waste is disposed in so-called "panels", eight in total, each consisting of seven rooms, along with adjacent "access drifts" and crosscuts. An important aspect of the closure procedure is the emplacement of bags on top of the waste, containing magnesium oxide

(MgO), which decreases the solubility of the actinide elements of transuranic waste (US DOE, 2011; p. 2-20).

When a panel is filled with waste containers it is closed with a closure system 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).

5.3. Waste Characteristics

It is apparent that, in relation to the long-term safety of geological disposal, the characteristics of radioactive waste must be established sufficiently well. In general the waste characteristics depend on:

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

As will be shown in the next sections, the characteristics of the Dutch radioactive waste intended for disposal has changed considerably throughout the last decades, depending on, amongst others, the adopted policy and future nuclear prospects in the Netherlands.

5.3.1. Waste characteristics applied in previous studies

For the VEOS study as part of the OPLA programme, the waste characteristics including the radionuclide inventories of the then considered waste types have been determined in detail (Heijboer, 1988). At that time a distinction between the various waste types was made as indicated in the following table (Heijboer, 1988; pp. 138,139). In VEOS, and later also in PROSA, the total radionuclide inventory of the disposal mine (see Section 5.2.2) was determined taking into account so-called “waste strategies” (Slagter, 1988; Section 4.2), for which Table 5-1 provides a short description. The radionuclide inventories of each of these waste types/canisters have been listed in Tables 1 to 11 of (Heijboer, 1988).

The radionuclide inventories served as source term for the VEOS and PROSA safety assessments, which are described in Sections 6.2 and 6.3 of the present report.

Notable for the waste characteristics adopted in VEOS and PROSA are the following:

- The distinction of the various 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.

Prior to the OPERA program, the VEOS waste inventories have not been updated. Instead, they served (partially) as input for safety assessment demonstrations as part of the **CORA** programme for the METRO-I disposal concept (Grupa, 2000; Chapter 4; Section 5.2.3 of the present report), and for a generic concept applying long vertical boreholes as part of the EU FP6 project **PAMINA** (Schröder, 2009a).

In the 1990’s and 2000’s alternative waste classification schemes were developed and adopted by the international community, as elucidated in the next section.

Table 5-1 Overview of waste types and waste strategies considered in VEOS and PROSA[t].

Waste type	
KSA	Highly radioactive, heat generating waste, resulting from the reprocessing of spent fuel from nuclear power plants
HAVA	Highly radioactive waste mainly originating from the reprocessing of spent fuel and subsequent vitrification process
MAVA	Technological alpha waste cast in concrete consisting of spent material and parts from the reprocessing process
LAVA	Technological waste with a lower nuclide content than MAVA and negligible heat generation
Waste strategy	
A	Accommodation of the radioactive waste from the Borssele and Dodewaard NPPs (installed capacity 500 MWe, operational period of 30 years), the reprocessing of the spent fuel elements, and a surface interim storage of the waste for 50 years.
B	Accommodation of waste from the Borssele and Dodewaard NPPs supplemented by waste from additional 3000 MWe nuclear power generation, and 105 years collection of radioactive waste from hospitals and laboratories. Final disposal was foreseen between 2080 and 2090 after 50 years of interim storage.
C	Accommodation of waste generation from NPPs equal to 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.

5.3.2. Waste Classification

The most recent waste classification scheme developed by IAEA covers all types of radioactive waste and provides a generic linkage with disposal options for all types of waste. The scheme is based on considerations of long-term safety. Six classes of waste were distinguished 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 of The Netherlands is not based on a law or regulation but it is common practice to use the IAEA classification scheme. In the Dutch waste classification scheme three waste categories are considered (EL&I, 2011; Section 32.1): heat-generating High-Level Waste (HLW heat-generating), non-heat generating High-Level Waste (HLW non-heat-generating), and Low- and Intermediate-Level Waste (LILW).

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).

5.3.3. Current inventory of the OPERA reference database

As part of the OPERA programme (Verhoef, 2014; Appendix; Hart, 2014), an up-to-date ‘source term’, both in terms of the radioactive inventory and the waste matrix has been determined. 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

[t] KSA: Kern Splijtings Afval; HAVA/MAVA/LAVA: Hoog/Middel/Laag Actief Vast Afval

from the Borssele NPP, no new nuclear power plants, and a surface storage period until 2130. At present the following types of waste are considered for final disposal:

Spent fuel, mainly comprising 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). It is noted that also spent fuel from a future replacement of the HFR, currently indicated as PALLAS, has to be considered.

Vitrified high level waste (HLW) consisting 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 mainly consisting 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 LILW arising from activities with radioisotopes in industry, research institutes and hospitals. It includes lightly contaminated materials, such as tissues, plastic, metal or glass objects, or cloth. It also includes waste conditioned in cement, originating from the nuclear power plants.

Technically Enhanced Naturally Occurring Radioactive Material, (TE)NORM, foreseen for geological disposal (Verhoef, 2011a) is depleted uranium (DU) originating from the uranium enrichment facility of URENCO (EL&I, 2011; p. 22).

The currently anticipated Dutch radionuclide inventory intended for final disposal differs significantly from the inventories considered in VEOS and PROSA. As a consequence the results of the VEOS and PROSA safety assessments may not be applicable 1:1 to provide statements about the long term post-closure safety of a salt-based repository in which the presently foreseen types and amounts of waste need to be accommodated.

5.4. 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 4.7) 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.

The following sections shortly discuss borehole and gallery plugs, and shaft seals since these features are specifically related to salt-based repositories. Issues related to backfill are treated in Section 5.5.

5.4.1. Borehole plugs

After emplacement of radioactive waste in a disposal cell or cavity, the disposal cells will be closed with of a plug, usually manufactured from pre-compacted crushed salt (e.g. Heijdra, 1996; Section 3.1). Such a plug initially holds a higher porosity than the surrounding rock salt and may form a potential pathway for any brine entering the disposal cells and for contaminated brine to leave the disposal cells.

However, pressure from the overburden and surrounding rock salt induces convergence and compaction of the plugs and a decrease of their porosity, ultimately to values where they have become impermeable for brine transport. 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% (Zhang, 2006; 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, 2009a). 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 calculations revealed that a disposal cell sealing plug would become impermeable after about 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, and “close” the disposal cells.

5.4.2. 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 seals could be demonstrated.

5.4.3. 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 considered 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).

From extensive analyses performed in both Germany (e.g. Bollingerfehr, 2013; Section 4.3.4.3) and US (Hansen, 2011; Section 2.3) it appeared that a shaft seal system must be adapted to the conditions prevailing at a specific site. The components and materials of a shaft seal system should be selected in accordance with the geologic environment along the shaft length and the composition of brines that might intrude from the overburden.

An example of a shaft sealing system presently adopted in Germany is illustrated in Figure 5-8 (Bollingerfehr, 2013; p.53). Each element of the shaft system 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 uppermost element of the shaft seal below the shaft foundation is a filter layer. In the overburden formation the shaft is backfilled conventionally.

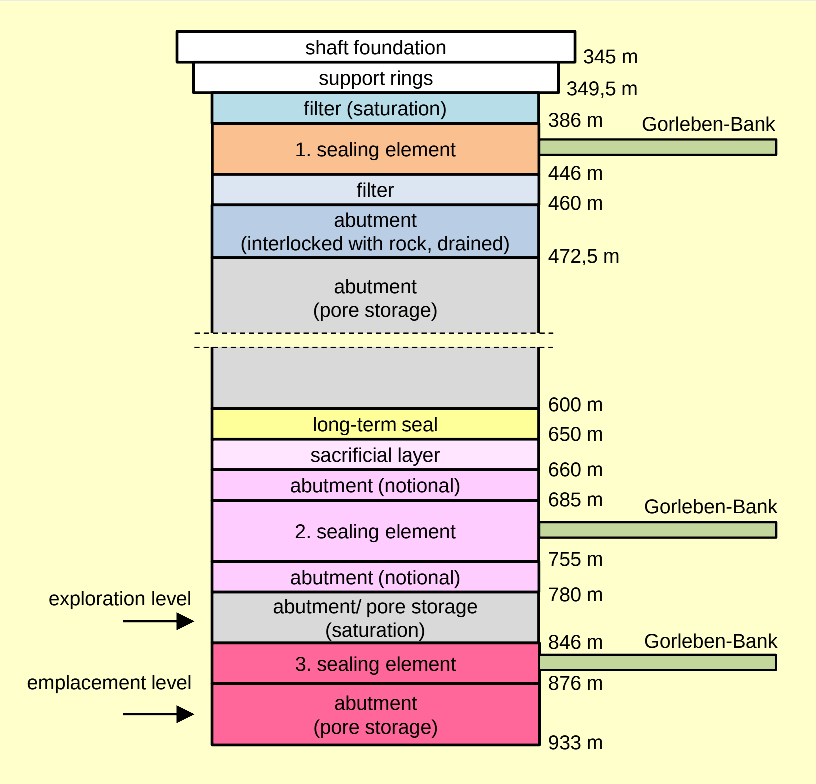


Figure 5-8 Sketch of the functional model of the shaft seal - German design.

5.4.4. EU FP7 project DOPAS

A major effort on the design and performance of seals and plugs is executed in the recently started EU-FP7 project DOPAS (Hansen, 2012), in which NRG also participates. 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. 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.

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

5.5. Salt host rock

5.5.1. Introduction

Rock salt (halite) is the main component of the evaporation cycle and is originally precipitated from a saturated surface or near-surface brine. Due to its specific features and properties rock salt has been considered a candidate host rock for radioactive waste emplacement for a long time. Over hundred years of experience in salt mining in several countries demonstrated that underground structures in rock salt can be constructed in a stable way. In ambient conditions rock salt is practically impermeable to gases and liquids, which makes it suitable as confining zone for hazardous material. In addition, rock salt shows good thermal conductivity and favourable deformation behaviour. 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.

[u] ELSA: Schachtverschlüsse für Endlager für hochradioaktive Abfälle

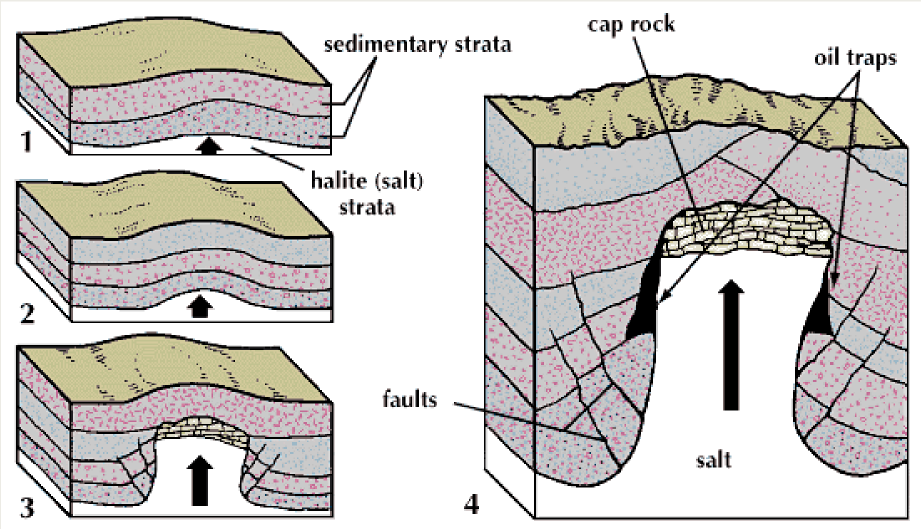


Figure 5-9 Formation of a salt dome; upper left: bedded structure; right: salt dome.

According to the geological evolution two main types of salt formations with quite different characteristics can be distinguished: bedded salt formations and domal salt formations, see Figure 5-9 (ENC, 2014). Due to its uprise the lateral extent of **domal salt** can be limited and therefore the dome margins may delimit the area useful for hosting a repository. **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.

5.5.2. Previous geological studies aimed at deep disposal

During the past decades, extensive studies have been executed concerning geological disposal of radioactive waste in salt in the Netherlands. The most elaborate study performed in the OPLA-1 phase concerned the report OPLA-12 (RGD, 1988) focussing 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, aiming at narrowing down uncertainties identified in OPLA-1. In a series of sub-studies Rijks Geologische Dienst evaluated (a) spatial data, (b) salt movement, (c) caprock formation and subrosion, (d) fluvial and subglacial erosion, and (e) the geological barrier model (RGD, 1993). Further, an overview was provided of the applicability of geophysical techniques for geohydrological exploration of shallow salt bodies and their overburden (TNO, 1993).

Within the CORA-programme (1996-2000) less attention was given to the geological aspects of salt, since these were already covered in OPLA.

5.5.3. Salt formations in the Netherlands

Salt accumulates when salt-rich lake- or sea water evaporates. 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. The Zechstein Group consists of various rock types, including clay, carbonate, anhydrite and rock salt. As part of the OSSC project a set of maps of the Zechstein Group have been established. These maps are depicted in Appendix 1.

5.5.4. Salt domes

With respect to depth and thickness of a disposal location in salt, salt domes are the most obvious target for hosting a repository. Rock salt in salt domes generally contains Zechstein Z2 salt in the core and younger salt along the margins of the dome (Van Adrichem Boogaert, 1993-1997) (Harsveldt, 1980, 1986). In some cases, however, 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 salt structure according to a well specified procedure (Geluk, 1998). It is obvious that such a procedure is highly site-specific.

5.5.5. Aquifers surrounding rock salt

There are basically two configurations in which rock salt can be in contact with aquifers (Figure 5-10). The situation probably most relevant for deep geological disposal is when an evolved salt dome intersects with present aquifers or reaches/is covered by an aquifer at the top. Fluids may then migrate from permeable aquifers through fractures in brittle inclusions into the margin of the salt dome. However, details of this mechanism are unclear and need further investigation. The other configuration involves a fault with an offset, juxtaposing salt against a permeable aquifer or fault zone.

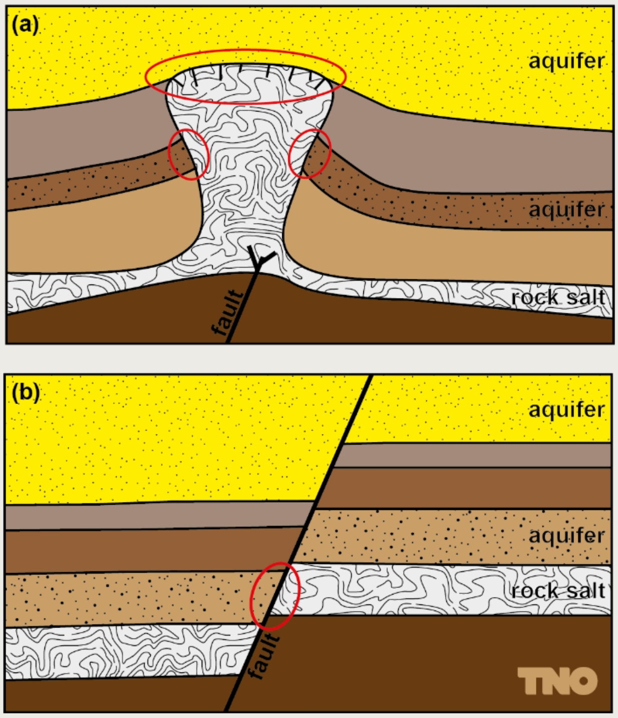


Figure 5-10 (a) A salt dome which has grown through overlying rock layers. The red circles indicate possible connections between an aquifer and the salt; (b) A salt layer offset by a fault, causing juxtaposition. Connectivity between aquifer and salt may also occur via the fault zone.

The 24 aquifers mentioned in (Van Adrichem Boogaert, 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 5-2 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
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

5.6. Safety-relevant processes

The excavation of a mine influences the stress field and the mechanical processes in the salt host rock (see also Section 5.6.1). 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 salt 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 similar isolating properties as rock salt. At that time, the salt has become impermeable for liquids and gases. Thus a discussion of the properties of salt concerns 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). Several recent summary reports discussed the various aspects and consequences of the presence and inflow of brine into a repository, e.g. (Caporuscio, 2013; Kuhlman, 2014).

5.6.1. Rock salt in the near field/excavation damaged zone (EDZ)

The originally undisturbed host formation will be affected by the excavation of a foreseen repository. As a consequence, the rock salt surrounding the excavated areas will be exposed to a geomechanical response. Figure 5-11 shows the different states of rock salt before and after excavation activities (Marschall, 2008; p. 49), from the undisturbed rock salt (see Section 5.6.2) to the backfilled state including internal pressure (see Section 5.6.3). The next sections discuss the most relevant aspects of this process.

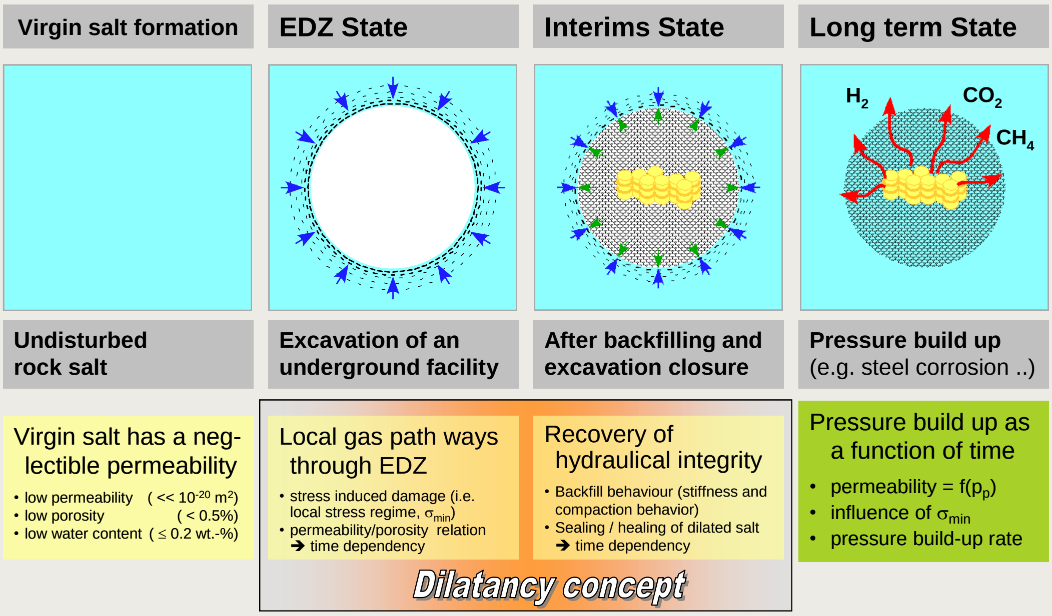


Figure 5-11 Schematic view of different states of rock salt in relation to underground mining.

5.6.1.1. Disturbed rock zone (DRZ)

A disturbed rock zone is an inevitable feature close to the excavated area in rock salt, and it may contribute to flow and transport processes in any part of the mine. The DRZ exhibits different hydraulic and mechanical properties compared to undisturbed rock salt, although the dimensions are not clearly defined. Typical properties of a DRZ are (1) dilational deformation, (2) loss of strength, floor heave, roof degradation and collapse, (3) increased fluid permeability via connected porosity (Hansen, 2011; p. 25). Many investigations have been performed to characterise these properties (e.g. Rothfuchs, 2009).

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.

5.6.1.2. Healing of DRZ

One of the advantageous properties of rock salt for hosting a repository is that fractures present in the rock salt can heal after closure of the excavations. Evidence for healing has been observed in laboratory experiments, small-scale tests, field tests (Hansen, 2011; Section 2.4.1.5), as part of the German ALOHA project in the Asse mine (Wieczorek, 1999; Wieczorek, 2004), and through observations of natural analogues. In a salt formation with elevated temperatures, as is expected for disposal of high-level waste, the healing process may be accelerated (Lux, 2007)[v].

Models for healing of the disturbed zones are based on observations and have been investigated in several projects (e.g. Wallner, 2007; Bérest, 2012). Healing is found to occur by either a recovery of the fractures by mechanical effects or a closure by chemical processes (crystallization), see e.g. (Pfeifle, 1998), (Hansen, 2013; p.143), (Hansen, 2011; p. 27), and (Pluymakers, 2014; p. 1).

In the Netherlands the healing of DRZ’s in rock salt based repositories has been studied at the Utrecht University (e.g. Houben, 2008; Hart, 2009). Although several theoretical models have been developed to assess test results concerning the development of crack geometries and the permeability evolution over time, it was concluded that more research would be needed to be conclusive about the fundamentals of the healing processes.

In the presence of a liquid phase, a DRZ can heal by recrystallization of salt minerals, additionally to healing by meachanical compaction. In general, the knowledge of thermodynamic data of these processes is good, but for elevated temperatures above 100 ^{o}C more data are necessary (Mackinnon, 2014). 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).

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

The following description of characteristics and processes of undisturbed rock salt are applicable to both bedded and domal salt structures. Differences between the rock salt types will be described, if applicable.

5.6.2.1. Thermal, hydraulic, mechanical, and chemical host rock properties

The knowledge of THMC properties of salt is the basis for investigations of all saline systems. Recently, in the project VIRTUS (Wieczorek, 2013, 2014), 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. A summary overview of the various salt properties is given below. Additional information has been compiled in (Hart, 2015; Section 4.5).

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 salt the heat conductivity, heat capacity, and thermal expansion coefficient as well as their temperature dependencies are adequately known and can be taken into account in safety assessments.

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

Hydraulic properties: Undisturbed rock salt is impermeable to fluids. This property makes rock salt well suitable to isolate radioactive waste from the environment. However, from the sedimentation process, rock salt formations can contain considerable amounts of residual liquids, i.e. 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. A movement of liquids within rock salt is only possible in fractures and fissures, e.g. in damaged zones.

Mechanical properties: Due to its mechanical properties rock salt has the tendency to creep under external mechanical load. The values of mechanical parameters describing elasticity and viscoplasticity are generally available for all salt types, although some are still under debate, see 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 5.6.4.

Chemical properties: The mineral composition of a saline system depends on its chemical composition. The various mineral phases found in salt domes are summarised in Table 5-3. A disposal concept in rock salt should preferably be located in halite (NaCl), and inclusions of anhydrite or carnallite should be generally avoided to prevent their thermal degradation as well as release of the crystal water. Additional information of the chemical system and chemical reactions of the host rock are described in Section 5.6.5.

Table 5-3 Salt minerals and their chemical composition.

Name	*Chemical composition*
Halite (rock salt)	NaCl
Anhydrite	$CaSO_4$
Gypsum	$CaSO_4 \cdot 2H_2O$
Bischofite	$MgCl_2 \cdot 6H_2O$
Carnallite	$KClMgCl_2 \cdot 6H_2O$
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	$MgSO_4 \cdot 6H_2O$

5.6.2.2. Creep / plasticity

Undisturbed salt at repository depth exhibits an isotropic, lithostatic state of stress. 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 are possible.

In case of underground mining activities the stress state in the salt around the excavations is disturbed, resulting in deviatoric stress and microfractures. 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 feature, usually referred to as viscoplasticity, is an intrinsic behaviour of salt. This will be discussed in more detail in Section 5.6.3, because it describes mainly the convergence process in backfilled areas. The effect of creep on the healing process in a DRZ has been mentioned in Section 5.6.1.2.

5.6.2.3. Thermal effects

Thermal effects play a major role in many processes in rock salt, especially in mechanical processes, since heat accelerates creep without creating additional fractures. 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 (e.g. Hansen, 2011; Section 4.3.2), see also Section 5.6.3.2.

The physics of plastic deformation of salt is dependent on the temperature and is governed by processes at the microscopic scale, for example the influence of impurities, grain boundaries, potential water contents, etc.

Thermal effects on hydraulic and chemical processes should be taken into account, but are of less importance compared to the mechanical effects. Most of the chemical reactions are temperature-dependent. Changes in solubility limits, sorption coefficients, etc. due to temperature changes are well investigated and documented. However, especially in the Dutch situation with the expected mild temperature increases, the thermal effects will be relatively limited.

5.6.2.4. 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, 1988/1999). 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.

Any effects of radiation on the salt properties would be concentrated in a small area of some metres around the HLW (Prij, 1991; pp. 178-190). Possibly formed cracks would heal in a relatively short time due to creep and recrystallization. However, if an overpack of a few cm thickness is applied radiation damage can be avoided (OPLA, 1989; p. 80).

5.6.2.5. Coupled thermal – hydraulic – mechanical – chemical effects

In general, thermal, hydraulic, mechanical, and chemical (THMC) effects as described in the previous section are coupled processes 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. Additionally, as the coupling parameters are not always known adequately, simplifications are necessary.For example, in the case of a mild or absent heat output from the disposed radioactive waste the thermal effects are insignificant. In the description of salt systems primarily the hydraulic and mechanical effects are coupled.

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. Some attempts have been made for the coupling of chemical and hydraulic processes, e.g. to increase the precision of calculated retardation of radionuclides during transport, and investigations are going on.

The coupled modelling of chemical and thermal effects 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.

The healing of a DRZ (Section 5.6.1.2) 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), and the EU-FP6 project THERESA[w] (Jing, 2010). 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 also Section 5.6.3.4.

5.6.3. 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. 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).

5.6.3.1. Thermal, hydraulic, mechanical, chemical properties of crushed salt

Crushed salt is in several aspects different from undisturbed rock salt (cf. Section 5.6.2.1). For example, the global thermal properties are different due to the high porosity: the open voids in the backfill reduce the thermal conductivity since they are filled with gas as an isolator. Additionally, due to the initially high porosity, the initial permeability for fluids is also enhanced compared to undisturbed rock salt. The relationship between porosity and permeability of crushed salt has been investigated in for instance (Prij, 1993; Section 5.2.2, and Müller-Lyda, 1999; Wallner, 2007); Schröder, 2009a; Section 3.3.2). A proper understanding of this relation is vital for assessing the long-term post-closure safety of a salt-based repository, where crushed salt is applied as backfill. This is elucidated in the following section.

The chemical properties of crushed salt are in principle similar as for undisturbed rock salt.

5.6.3.2. Compaction of crushed salt

After backfilling of open excavations, 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. As an example, Figure 5-12 shows the temporal evolution of porosity of backfilled areas in a repository with various amounts of humidity and with different temperatures (Larue, 2013; p. 125).

The physics of plastic deformation of salt is governed by processes at the microscopic scale, see also Section 5.6.1. Compared 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. Compaction of crushed salt in relation to disposal has intensively been investigated internationally, in Germany for the Asse mine and commercial salt mines in Germany (e.g. Brenner, 1999) and

[w] THERESA: “Coupled thermal-hydrological-mechanical-chemical (THMC) processes for application in repository safety assessment”

laboratories (e.g. Schulze, 2007; Wieczorek, 2010), in the context of the US WIPP disposal site (Hansen, 1995), and in the Netherlands at the Utrecht University (Zhang, 2006).

These investigations have demonstrated that the compaction process under typical repository conditions (dry or low moisture content; temperatures elevated; high convergence rate) lasts in the range of several decades to several hundreds of years. The time frame can be considerably higher if the internal (back-) pressure increases, e.g. after a complete filling of the mine with brine (flooding).

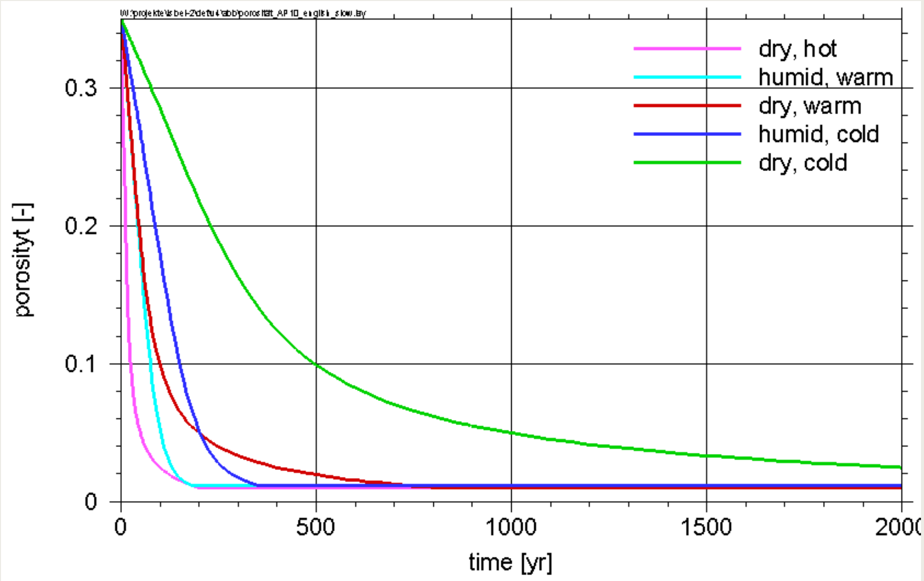


Figure 5-12 Temporal evolution of porosity at different temperatures.

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. Another open question is the final porosity that can be reached upon compaction by plasticity.

5.6.3.3. Thermal effects

As crushed salt will be used in a repository with heat-generating waste it will be exposed to elevated temperatures. Temperature plays a significant role in many processes in rock salt. For example, the thermal conductivity of crushed salt is lower than that of undisturbed salt. Consequently temperatures are higher for heat-generating containers surrounded by crushed salt compared to the undisturbed host rock. 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). In the Dutch context however, with an extended surface storage time, the maximum temperatures reached in the salt will be moderate.

Elevated temperatures may also cause dehydration of salt and thus influence creep. Usually, in mechanical models the temperature dependency is taken into account by an Arrhenius term. Other thermal effects as described in Section 5.6.2.3 are also applicable to crushed salt.

5.6.3.4. 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. The compaction of crushed salt (Section 5.6.3.2) 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. 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 chemical equilibrium in this highly saline environment is usually calculated by applying the so-called Pitzer approach for the activity correction (see Section 5.6.5). Depending on the availability of appropriate thermodynamic data for the individual processes the equilibrium composition of the fluid and the equilibrium composition of the solid phase can be estimated.

5.6.4. Constitutive modelling and computer codes

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. At present, chemical effects are rarely taken into account in these models, although there is progress in chemical modelling, see Section 5.6.5. THMC processes are strongly related under the conditions in a repository, as elucidated in the simplified representation of Figure 5-13 (Kuhlman, 2014; Figure 1.3)[x].

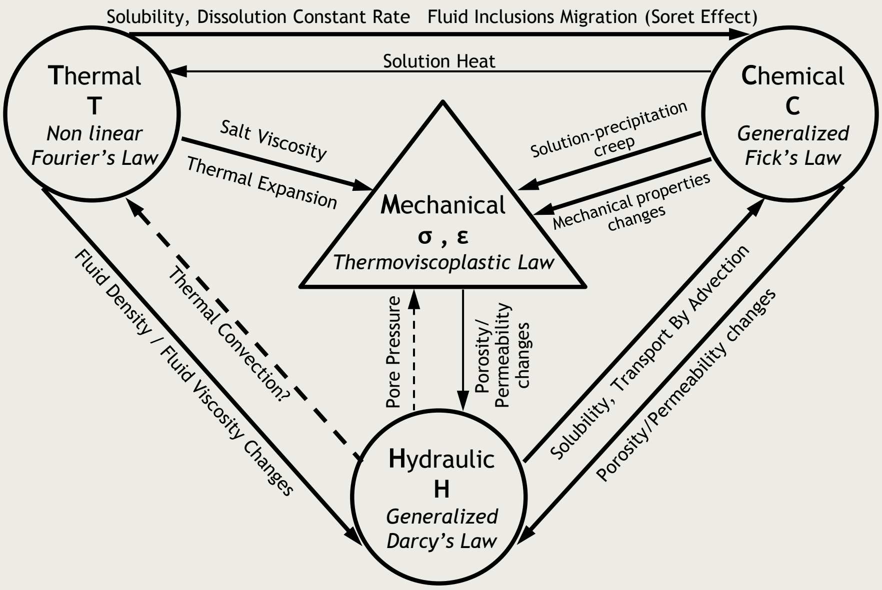


Figure 5-13 Relationship between thermal, hydraulic, mechanical, and chemical processes.

Each process can influence the other processes, and, 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).

Constitutive laws to model coupled processes have been proposed and developed by many companies and research institutions and over a long time period. In 2007, an overview of advanced models developed in Germany has been given by (Schulze, 2007). 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.

ECN/NRG has improved repeatedly the model for convergence of crushed/compacted rock salt for dry as well as brine-saturated backfill, e.g. in the framework of OPLA-1 (Prij, 1993; Section 5.2.3), the EU Framework project BAMBUS (Bechtold, 1999; Poley, 2000b), the METRO-III project of the CORA programme (Grupa, 2000; Section 4.4.3), the NF-PRO

[x] Dashed connections in this figure indicate processes that may not exist in undisturbed salt; fatter solid arrows indicate stronger relationships between processes.

project ('Coupled Creep Model', CCM; Zhang, 2006; Section 5.3), and the PAMINA project (Schröder, 2008).

The coupling of models in **computer codes** for THMC calculations is most advanced for *hydraulic-mechanical* processes. Many aspects of these models have been treated in an EC-hosted 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. 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). At present there is not "one" code that is able to solve all the existing tasks in THMC calculations, but there will be a need to develop specialised codes.

All codes 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. In modern transport codes the coupling with geochemical codes is in development to take retardation (sorption) effects into account.

The codes for hydraulic calculations that have been developed recently take better account of density effects, e.g. d^3f ("*distributed density-driven flow*", Schneider, 2012). 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.

5.6.5. Chemical conditions / geochemical modelling

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

- dissolution processes in the host rock / crushed salt 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.

Dissolution processes

The mineral composition of a saline system depends on its chemical composition. Figure 5-14 (Hartmann, 2006; p.11) shows as an example the quinary system Na-K-Mg-Cl-SO_4-H_2O to elucidate the various forms of mineral phases in a typical chemical system occurring with rock salt.

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). 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 brine and mineral volumes.

Chemical equilibrium modelling is a useful tool in assessing and understanding of geochemical reactions, not only the dissolution and precipitation reactions, but also with respect to the analysis of corrosion and migration processes. Several modelling tools exist for assessing these processes, e.g. (Allison, 1991; Parkhurst, 1999; Meeussen, 2003). For characterising interactions amongst ions and solvents of a saline system the Pitzer approach (Pitzer, 1984, 1991) is commonly used. Depending on the availability of appropriate thermodynamic data for the individual processes, i.e. the Pitzer-constants, the equilibrium composition of the fluid (brine) and the equilibrium composition of the solid phase (salt) can be estimated.

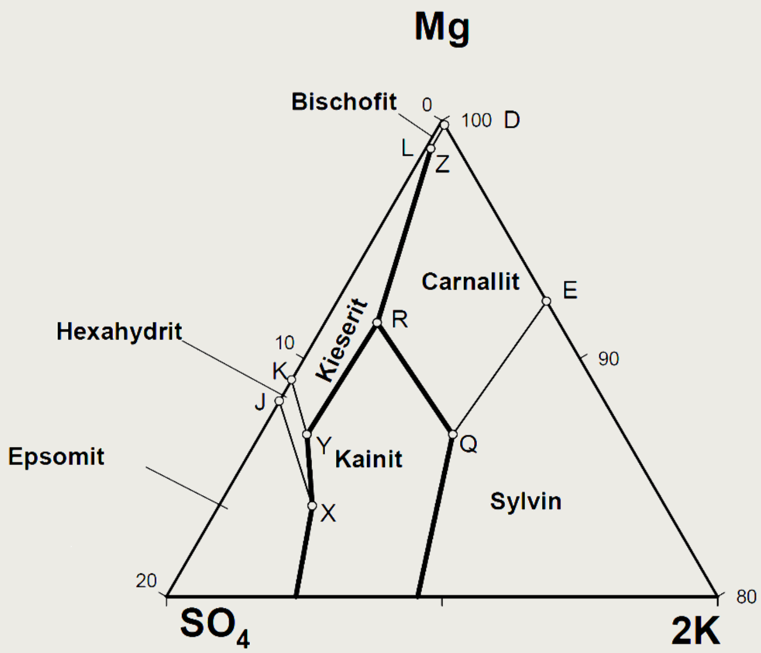


Figure 5-14 **Part of the quinary system Na-K-Mg-Cl-SO_4-H_2O in the particular case of NaCl saturation (Jänecke diagram).**

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

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. In the PROSA study 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).

The corrosion of iron in highly saline environment and at normal temperatures has been investigated in many projects (e.g. Smailos, 1992; Smailos, 1993). 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. The dissolution behaviour of cementitious material affects the pH in solution, which is a relevant parameter also for determining the solubility of radionuclides (see below).

Corrosion of cementitious barriers

Degradation and dissolution of cementitious materials 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).

Solubility of radionuclides

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[y].

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. On the other hand, 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 various mineral or soluble forms can be calculated. At present, many models exist that allow estimating 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 often occurs not in thermodynamic equilibrium, but intermediate, thermodynamically less favourable phases can appear (e.g. Vandenborre, 2008);
- in case of saline solutions, the reactivity 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, and 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[z], 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.

Models for the calculation of radionuclide dissolution and transport in a saline system 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 and the complexity of the geochemical processes are reasons why geochemical models are not fully coupled to hydraulic models.

Gas production and transport

As already mentioned in the previous sections, there may be several 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.

In **PROSA** the effect of gases on brine flow in a repository was considered in the REPOS code as exchange processes between segment models (Prij, 1993; p.8.10). Additional efforts on the gas production and gas-mediated transport were executed 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 sealing the disposal cells into the gallery, diffusive transport; gas driven transport and transport due to a temperature gradient. It appeared that under the assumed conditions the relative contribution of gas-driven flow to the overall radionuclide transport mechanisms is relatively minor (Grupa, 2000; Figuur 32). Eventually, all transport is ended when the plugs become impermeable due to their compaction Grupa, 2000; p.143).

[y] This is different from Boom Clay as host rock, where sorption plays an important role in the mobility of many radionuclides.
[z] http://www.oecd-nea.org/dbtdb/ last accessed on 1 June 2015

The contribution of gas production and gas-driven flows to the overall repository safety has also been investigated in **Germany**. (ISIBEL and VSG projects), and US (WIPP).

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, individual models and computer programmes should be reviewed for potential improvements (DBE-TEC, 2008; p.72).

The German **VSG** project concluded that, 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. 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 gas-related 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. 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).

5.6.6. Solubility and transport

If radionuclides are released from a deep geologic repository, water provides the most likely pathway for radionuclide transport. However, the low water content of rock salt limits the transport of dissolved radionuclides. In addition, the self-sealing capacity of the salt could also limit the possibility for radionuclide releases and transport away from the repository. As Hansen and Leigh (2011) noted, Generally, investigations of the potential for radionuclide transport in a salt repository assume that there will be brine available to dissolve and transport the radionuclides (Hansen, 2011; p.63). In practice, however, transport depends on at least two factors: brine sources with sufficient volume and the existence of a pressure gradient capable of moving the soluble radionuclides through the rock to the biosphere (Winterle, 2012; p.3-6).

At the very low permeabilities associated with undisturbed salt formations, there is essentially no advective movement of water, so radionuclide transport away from the repository is limited to diffusion in whatever brine may be present in interconnected pathways (e.g., along grain boundaries) in the salt.

Brine exists in bedded salt in three forms: fluid inclusions, hydrous minerals, and grain boundary water. Owing to the characteristics and environments of the brine in salt, its transport or migration occurs via three primary mechanisms: motion of the brine inclusions in a temperature gradient, vapor-phase transport along connected porosity, and liquid transport driven by the stress gradient (Hansen, 2011; p.63).

Compared to diffusion of solutes in open water, diffusion through a solid medium is further slowed by (i) physical characteristics of the solid phase, e.g. porosity and permeability; (ii) constrictivity, which depends on the size of the connections between adjacent pores and how small pores are distributed in the medium; and (iii) and tortuosity (Winterle, 2012; p.3-7).

Tortuosity and constrictivity are difficult to measure directly, so the physical characteristics of a solid medium are usually represented by an empirically measured effective diffusion coefficient. This quantity is difficult to measure, especially for the usual very low permeabilities found in rock salt, and they may vary a few order s of magnitude. Typical values measured in-situ are in the order of $2{\cdot}10^{-14}$ to $2{\cdot}10^{-13}$ m^2/s (Winterle, 2012; p.3-7).

For some solutes, diffusive transport may also be retarded by adsorption of the solute onto solid surfaces. Generally, the sorption capacity of salt minerals is low, but sorption could be more important if impurities or interbedded layers, such as clays, are present in the salt formations.

Although diffusion is expected to significantly limit the rate of radionuclide transport in salt formations, whenever a radionuclide reaches the formation boundary it will encounter potentially different transport conditions in a different rock type. Such factors are site specific and fall outside the scope of this report. In addition, layers of other rock types that are interbedded with a salt formation may locally provide pathways with different transport conditions and properties. Diffusion lengths to such pathways could be relatively short, compared to the thickness of the salt formation as a whole.

5.7. Geosphere

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

In VEOS, the rise of the overburden due to the disposal of heat generating waste was shown to be very small and moreover eliminated by surface erosion (Prij, 1989; Chapter 5). 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.

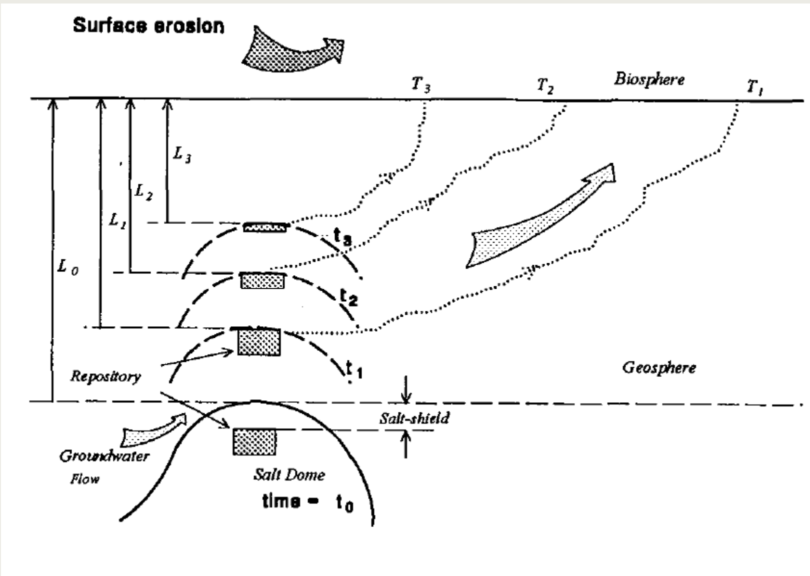


Figure 5-15 Schematization of salt rise and subrosion.

This process is outlined in Figure 5-15 (Prij, 1993; p.6.27). From an intitial state indicated by “0”, the position of the salt dome can shift upwards and the thickness of the overburden decreases. The concept of a gradually decreasing thickness of the overburden has been implemented in the METROPOL[aa] model, and nuclide residence times have been calculated by particle tracking.

In the present report, the geosphere characteristics are not explicitly treated, 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).

5.8. 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 (tens of) 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 based on the assumption that the future human beings will have the same diet as the present ones.

Assuming a release of radionuclides from contaminated groundwater several exposure pathways have been considered (Prij, 1993; Section 6.5.2):

- 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 schematically elucidated in Figure 5-16 (Prij, 1993; p. 6.56).

The biosphere characteristics themselves are independent of the "Source" as indicated in Figure 5-16, 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.

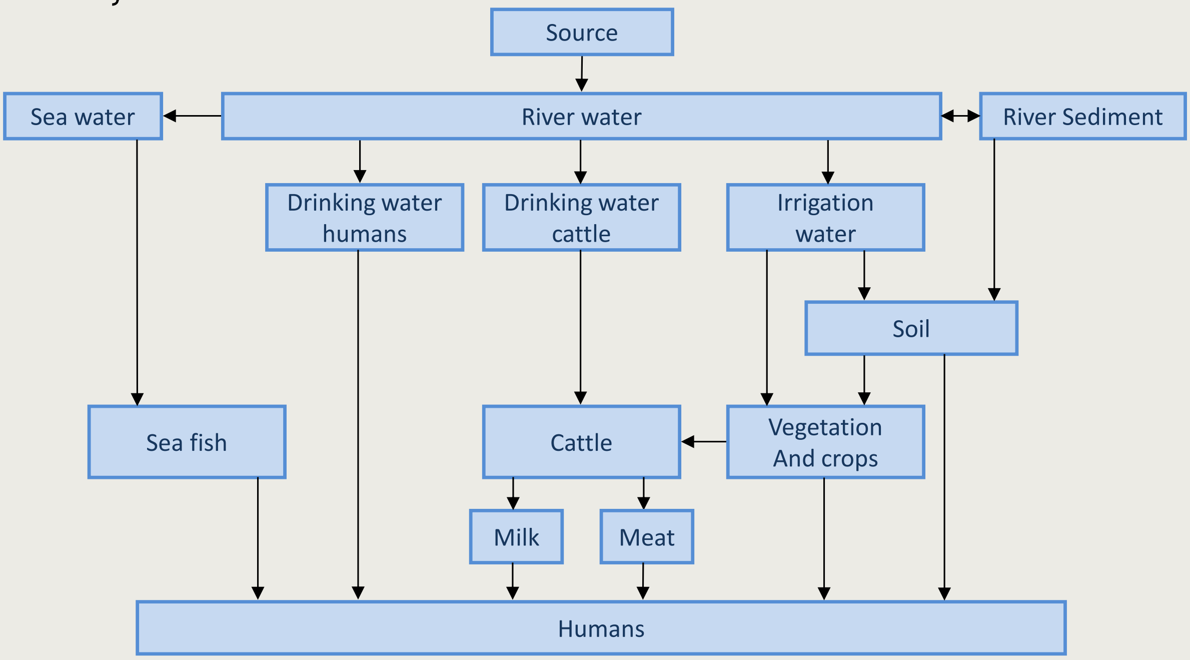


Figure 5-16 Contamination pathways for the biosphere.

[aa] METROPOL: MEthod of the TRansport Of POLlutants, computer program

5.9. Evaluation

The following sections summarise the main findings of the evaluation of the various aspects of the disposal system.

5.9.1. 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 ILW 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 ILW 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.

5.9.2. Waste characteristics

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

- The distinction of the various waste types KSA, HAVA, MAVA, and LAVA and their definitions , as practiced in VEOS and PROSA, currently no longer apply;
- In VEOS and PROSA studies consideration was given to the 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 compared to the presently foreseen inventory;
- Unlike in the past, the spent fuel from research reactors, including the future replacement of the HFR (PALLAS), is part of the present radioactive waste inventory;
- Unlike in the past, depleted uranium (DU) currently adds to the radioactive waste inventory intended for disposal. Although DU represents, in radiological terms, a relatively small fraction of the total inventory it comprises a significant volume;

As a conclusion it can be stated that the waste characteristics considered in the past differ significantly from those presently foreseen for final disposal. Since the waste characteristics provide the radionuclide source term for the safety assessment, the results of safety assessments performed in the past are not 1:1 representative for a future Dutch geological disposal facility in rock salt for hosting the presently foreseen types and amounts of radioactive waste.

5.9.3. Engineered barriers

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 creep of the surrounding host rock and finally will become impermeable. The time for the plugs to reach the threshold porosity is still a feature requiring additional confirmation.

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.

5.9.4. Salt formations

During the 1980's and 1990's extensive studies have been performed with respect to disposal of nuclear waste in rock salt. 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, topics have been identified which need additional efforts for reducing remaining uncertainties. These topics have been elucidated in Section 8.3.4 of the present report.

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

There is an abundance of data available about thermal, hydraulic, mechanical and chemical (THMC) properties of rock salt, much of which has been derived from exploration of the Gorleben and Asse sites in Germany, the Waste isolation Pilot Plant (WIPP) in the US, and a variety of European Framework projects (e.g. BAMBUS I/II, THERESA). Extensive summaries of earlier studies, performed in the last decades of the 20th century are available for further consultation.

The understanding of safety relevant processes as summarised in the present chapter is an essential prerequisite in the assessment of the post-closure safety, and thus forms the scientific basis of a Safety Case on rock salt.

On the basis of the information collected in the previous sections, THMC-related aspects have been identified internationally as relevant to carry along with respect to enhancing the understanding of salt THMC topics. Not all identified topics relate to the Duch Safety Case.

In order to identify potentially relevant aspects for enhancing the Dutch Safety Case, screening arguments for the distinction between crucial aspects and features for reducing remaining uncertainties 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 Dutch Safety Case).

The identified issues are categorised as follows (see also Hart, 2015; Section 4.9.4):

- 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 and transport of radionuclides

These topics are discussed further in Section 0 of the present report.

5.9.6. 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 and 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 realised that the details of the then existing overlying sediment and the biosphere cannot be determined exactly.
- 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.

6. Safety Assessment

6.1. 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 present chapter gives an overview of the safety assessments performed in the previous Dutch studies VEOS, PROSA, and CORA as well as in Germany and US, and summarises the recent developments on safety assessment methodologies.

6.2. 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 A, B, and C (see also Section 5.3.1).

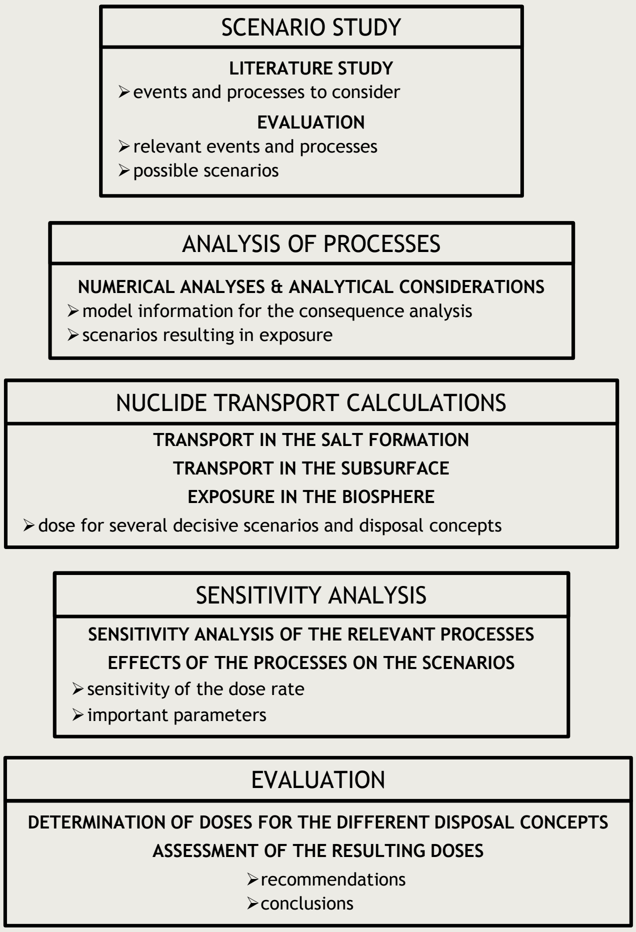


Figure 6-1 VEOS safety assessment methodology

The VEOS safety assessment was essentially a detailed and deterministic method, aiming to provide the necessary knowledge and insights concerning processes relevant to the dose rate in the biosphere. 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 VEOS safety assessment was performed taking into account a number of subsequent steps as elucidated in Figure 6-1 (Prij, 1989; p.44).

6.2.1. Definition of scenarios and process analysis

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. This procedure resulted in the identification of 11 scenarios, which were distinguished in three classes with descending probability (Prij, 1989, Chapter 4; see also Hart, 2015; Section 5.2):

- Normal evolution scenario (NES), assuming that the repository system functions according to the specifications.
- Altered evolution scenario (AES), considering events which are not likely to occur but cannot be excluded.
- Disruptive evolution scenario (human intrusion, HI), taking into account that the carefully designed system of barriers is compromised.

For the identified scenarios it was also indicated how the barriers around the waste can be destroyed and groundwater (or humans) can come in contact with radionuclides from the waste.

After having identified the scenarios, the processes playing an important role in the scenarios were analysed in more detail, in order to determine their possible impact on the release of nuclides form the GDF. Examples of the most important processes analysed are diapirism, subrosion, altered stresses due to mining activities, convergence of cavities, groundwater movement and radiation damage (OPLA, 1989; p. 85-86). The presence of brine inclusions in rock salt was considered irrelevant as a potential carrier of dissolved radionuclides to the biosphere.

6.2.2. 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 was conceptualised as three connected compartments, each analysed with a dedicated computer code:

- The disposal facility and the salt formation; in this compartment nuclide release and transport of the contaminants has been modelled;
- The overlying sediments through which the contaminants are transported with the groundwater
- The biosphere, in which the contaminated water is transported to the earth's surface leading to exposure of humans.

An example of the numerical results of the VEOS calculations is summarised in Figure 6-2 (Köster, 1989; p.5), depicting the exposure calculated for the Waste Strategy B[bb]. More results are given in the final report of OPLA (OPLA, 1989; p. 87-91).

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). However, for the Reconnaissance Drilling scenario receiving a high potential dose rate was judged extremely unlikely due to the very small probability of occurrence, and the conservative assumptions made for this scenario.

[bb] "Waste strategy B": all waste from the NPPs Borssele and Dodewaard reactors as well as new reactors of 3000 MW and assuming an interim storage period of 50 years. See also Table 5-1 of this report

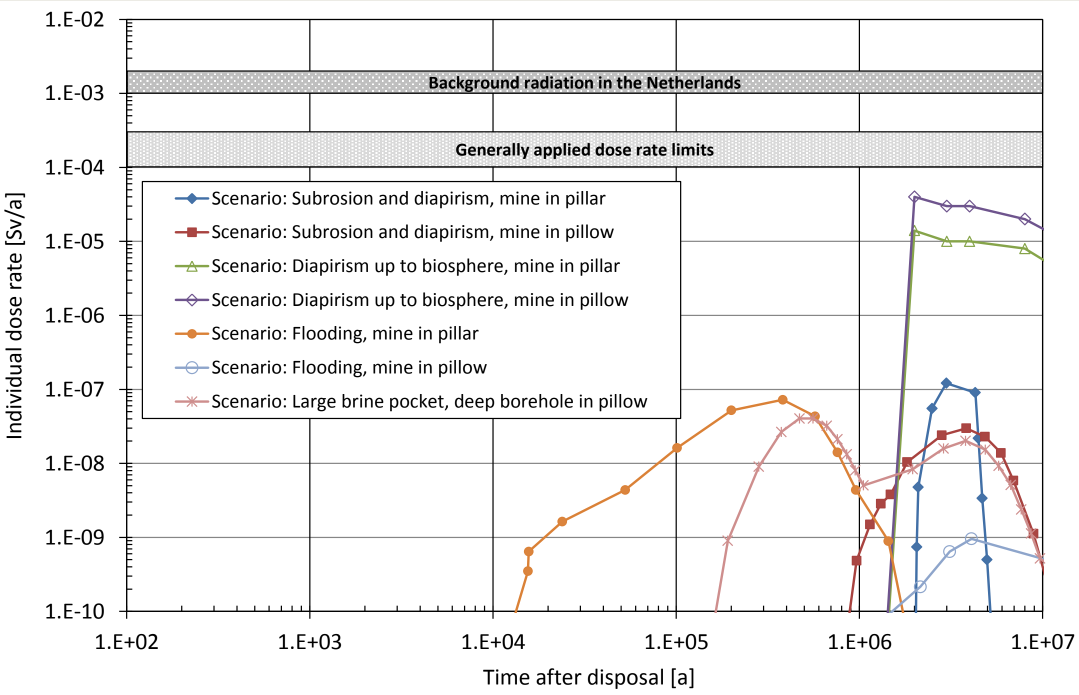


Figure 6-2 Overview of VEOS results for leading scenarios.

6.2.3. Treatment of uncertainties

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

- the sensitivity of the calculated dose rates on model parameters;
- the bandwidth in the results of the dose calculations;
- relevant design characteristics subject to potential improvement.

The results of the analyses are provided in the VEOS final report (Prij, 1989, Chapter 7). A few highlights are summarised in the following:

For the normal evolution scenario (NES) the most important parameters influencing the exposure in the biosphere are (1) the leaching time of radionuclides from the waste, and (2) the residence time of the nuclides in the salt formation. The VEOS results have also shown that in the NES the amount of brine normally present in the rock salt is much smaller than the amount of water needed to corrode the containers.

For the altered evolution scenarios (AES) a large amount of analyses have been performed to quantify the effect of variations in the values of the process parameters used in the dose calculations such as disposal depth and diapirism rate, convergence rates, nuclide migration inside the salt formation and in the groundwater, and more. An important conclusion was that most of these parameters can be influenced by the choice of the disposal technique (e.g. deep boreholes, versus caverns) and the salt formation type. For example, the selection of the **salt formation type** determines the depth of the disposal facility and directly the convergence rate of the rock salt and the radionuclide travel time.

For the human intrusion scenarios the results of the reconnaissance drilling analysis have shown that the exposure is largely dependent on the intensity of the inspection of the contaminated bore core. The difference in inspection of the bore cores (thorough versus routine) may result in differences in the exposure of one order of magnitude.

In addition to the radiological safety assessment, VEOS posed a significant effort on the determination of radiation damage to NaCl (e.g. OPLA, 1989; p. 79), see also Section 5.6.2.4 of the present report.

6.2.4. Concluding remarks

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 were therefore stated in terms of *expectations* rather than statements (OPLA, 1989; p. 19).

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

For a more systematic comparison between the various scenarios and disposal concepts it would be required to use a **probabilistic** method 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).

6.3. 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.

6.3.1. Definition of scenarios – PROSA methodology

The PROSA scenario definition was based on a comprehensive list of potentially relevant FEPs (Prij, 1993; Chapter 3). Subsequently, a screening procedure was applied to identify FEPs representative for the Dutch context. To simplify this FEP screening procedure a number of well defined "states" of the barriers in the multi-barrier system were defined (Prij, 1993; p. 2.6). For a particular state of the multi-barrier system, assumed to consist of the engineered barriers, the salt host rock, and the overburden, 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 present (or: intact) it takes a very long time before the nuclides leave the salt formation and consequently short-term FEPs in the overburden and biosphere can be neglected.

Taking three main barriers into account, a total of 8 multi barrier system states (MBSS) are possible (see Table 6-1). Within this methodology, MBSS-1, where all three barriers are present, represents the most-confining state of the disposal system. In contrast, MBSS-8, where all three barriers are assumed bypassed, represents the most unfavourable condition with regard to the confinement of radionuclides.

Table 6-1 Possible MBSSs (multi barrier system states).

State number	*Engineered barriers*	*Isolation shield (salt host rock)*	*Overburden (geosphere)*
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

After a judgement for each FEP of their impact on each multi-barrier system state, a set of scenarios could be defined and selected to be analysed further (Prij, 1993; Section 2.3). The 22 identified scenarios (Prij, 1993; Table 4.9) could be subdivided into three distinct groups or families of scenarios:

- The **subrosion** scenarios
- The **flooding** scenarios
- The **human intrusion** scenarios

6.3.2. Calculations performed

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

- Salt compartment; PROSA utilised the same radionuclide inventories as in VEOS;
- Groundwater compartment;
- Biosphere compartment (see also Figure 5-16).

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). Examples of results of the subrosion, groundwater, and human intrusion scenarios are described in the following. Extensive evaluations of all calculated results can be found in the PROSA final report (Prij, 1993; Chapter 7).

Subrosion scenarios

In the subrosion scenarios the isolation shield of the salt around the geological disposal facility is slowly dissolved by groundwater. During the time needed to dissolve the isolation shield the disposal facility slowly moves upward due to diapirism. This is schematised in Figure 5-15 of the present report (Prij, 1993; p. 6.27).

At some time into the future (more than approx. 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, where an exposure was calculated utilising dedicated computer programs (Prij, 1993; p. 7.4). An example of the PROSA dose calculations is depicted in Figure 6-3 for the case “Shallow dome, faulted overburden” for waste strategy B (Prij, 1993; p. 7.10).

A summary of all calculation results for the subrosion scenarios is given in

Table 6-2 (Prij, 1993; p. 7.12). The results indicate that in all cases both the calculated dose rates are orders of magnitude below the generally accepted limit of 0.1 mSv/a (see Section 3.4.3).

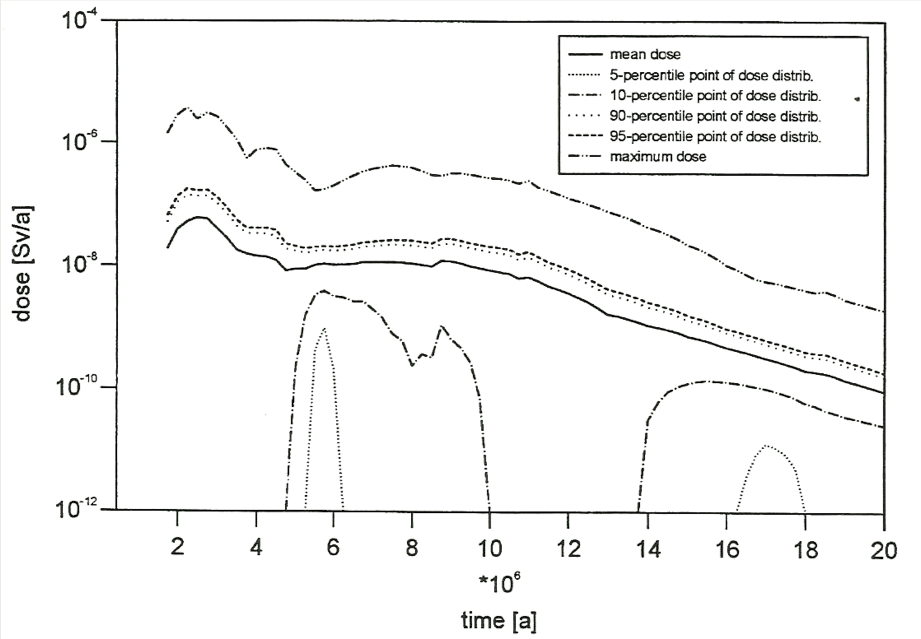


Figure 6-3 Evolution of calculated dose rate. Shallow dome with faulted overburden.

Table 6-2 Summary of calculation results for the subrosion scenarios.

Formation type	*Waste strategy*	*Fault*	*Calculation 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	$0.8\ 10^{-9}$

Groundwater intrusion scenarios

In the water intrusion scenarios the isolation shield is bypassed through an undetected water carrying layer or a large crack. As this water will rapidly be saturated with salt it turns into brine. This brine may come 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 the EMOS_ECN code for several cases, varying the rock salt formation type (shallow dome, deep dome), and the waste strategy (A and B).

An important result of the calculations was that only a small portion of the initial inventory will be squeezed out the salt formation, due to the ongoing convergence which eventually closes any open spaces.

For the groundwater intrusion scenarios large differences were found between the VEOS results and the PROSA results. The PROSA simulations resulted in about 5 orders of magnitude lower exposures than VEOS. The main explanation was found in the application of an improved model in PROSA for the convergence and compaction of backfilled openings.

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. One can presume that this sort of action only occurs after the 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 addition dose calculations have been performed assuming that the human action will occur 1000 years after discharge of the fuel from the reactor, which is considered to be a minimum duration of the efficiency of other passive, time-resisting, marking methods. The analysis of the calculated scenarios *Reconnaissance drilling*, *Solution mining*, *Leaking storage cavern*, and *Conventional salt mining*, indicated that only the scenario *Reconnaissance drilling* may lead to a significant exposure (100-200 Sv/a; Prij, 1993; p.7.22). This was also recognised in the VEOS safety assessment (see Section 6.2.2).

6.3.3. Treatment of uncertainties

In PROSA a probabilistic analysis was performed for assessing the impact of a selected number of model parameters for the salt, groundwater, and biosphere compartments on the calculated exposure (Prij, 1993; Section 9.2). This sensitivity analysis aimed to find the input parameter(s) having the largest influences on the exposure. Additionally, an uncertainty analysis aimed to quantify the variability of the output parameters on variations of the input data.

A selection of the results obtained for the subrosion scenario is presented in Table 6-3 (Prij, 1993; p. 10.7). The numbers in the table are a global measure for the sensitivity of the maximum dose rate on a specific input parameter. The higher that number is, the larger is the influence of that parameter on the exposure. It appears that for all four cases the internal rise rate of the salt dome is the most influential parameter on the maximum dose rate. It was concluded 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.

Table 6-3 Summed sensitivity coefficients of the maximum calculated dose rates.

	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
Glass dissolution rate	7	8	9	9

For the ***groundwater compartment*** it appeared that the most sensitive parameters are the characteristics of faults present in the geosphere, and the permeability of the geosphere layers. Additional efforts to characterise the geosphere may reduce the uncertainties of these parameters, and ultimately result in better predictions of the radionuclide transport rates in the geosphere.

In the PROSA study no attempts have been performed to determine the sensitivity of the human intrusion scenarios.

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 (Cadelli, 1996,; p. 182):

- 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 calculated exposure was 10^{-7} Sv/a.
- The parameters of the crushed salt applied as backfill are a considerable source of uncertainty, whereas the properties of the rock salt are a weak uncertainty source in the maximum exposure.
- The groundwater velocity in the geosphere is a considerable source of uncertainty, whereas the biosphere conversion factor and the time of groundwater intrusion are a weak source for the uncertainty in the maximum exposure.

6.3.4. Concluding remarks

A major aspect of the PROSA study was that a probabilistic calculation model was developed and applied to statistically analyse the radiological consequences of a deep geological facility for the disposal of radioactive waste. Based on the analyses performed the following conclusions were drawn (Prij, 1993; Chapter 11):

- 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 very low (< 10^{-6}/a).
- 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 consideration of a multi-barrier system.
- Due to the improved model of convergence and compaction of salt in PROSA the calculated exposure in the groundwater intrusion scenarios is reduced compared to the VEOS analysis to a negligible level.
- The hydrological properties of the overburden play a role in the transport of nuclides. The future properties can deviate significantly and unpredictably from the present values. The overburden properties are considered to be of importance for the mine's stability of the salt formation.
- 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.
- A probabilistic safety analysis can also help to identify parts of the disposal system that dominate the uncertainty, i.e. the less robust parts of the system.

The basis of the methodology as applied in the PROSA study seems the way ahead in the Netherlands, i.e. a systematic approach utilising FEPs as basis of knowledge building and scenario development, and performing sensitivity and uncertainty analyses to determine the parameters and processes mainly influencing the long-term safety.

6.4. CORA safety assessment

Following the OPLA programme, the CORA programme was executed taking into account the following conditions differing from those adopted in OPLA:

- 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.

In CORA retrievability of already emplaced radioactive waste was a main topic, unlike in the OPLA programme. 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 5.2.3 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 summarise a few highlights of the salt safety assessment carried out in CORA.

6.4.1. The abandonment scenario

For the CORA safety assessment, also referred to as the "METRO study" (Grupa, 2000; Section ES.2), 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 (Grupa, 2000; p. 140). In relation to operational safety, which is relevant in relation to the retrievability requirement, the FEP *'deserted unsealed repository'* was judged as representative and formed the basis for the so-called '*negelection scenario*', also referred to as '*abandonment scenario*'.

The abandonment scenario describes the evolution of the facility in the case of abandonment after the waste has been emplaced in the disposal cells, but before the facility is closed and sealed. The abandonment 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.
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.

6.4.2. Calculations performed

For the analyses of the 'abandonment scenario' the EMOS_ECN code package has been used. An important improvement accomplished in CORA compared to earlier EMOS versions concerned the modelling of the transition from permeable to impermeable salt plugs. This transition is caused by the convergence driven compaction of the salt plug 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 canisters to be placed in a *horizontal* borehole with a length of about 5 m, see Figure 6-4.

In each borehole one canister would be stored and a salt plug would seal the borehole. This consideration significantly deviated from the designs used in OPLA with *vertical* boreholes of 300 m with about 80 canisters (cf Section 5.2.2). In the analyses performed in CORA only HLW has been considered whereas in OPLA all categories of waste have been taken into account.

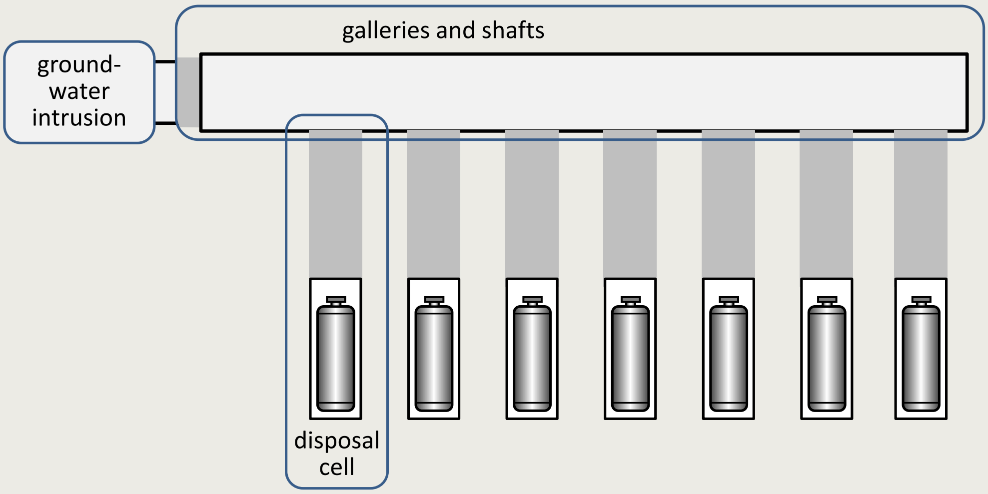


Figure 6-4 Horizontal cross section of a gallery and several disposal cells in the METRO design.

An example of the calculated results for the *abandonment scenario* is shown in Figure 6-5 (left) 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 occur after about 4000 years. From that time on, no release of radionuclides from the disposal cell would occur.

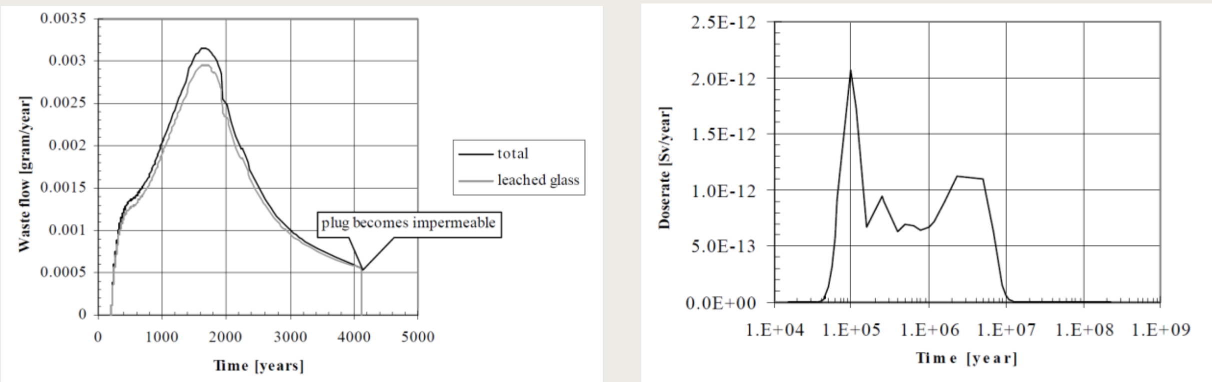


Figure 6-5 Escape of waste from one disposal cell to the gallery (left) and individual effective dose (right) - abandonment scenario.

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 6-5 (right). Compared to the internationally accepted dose limit of 0.1 mSv/year, and under the adopted assumptions, the calculated dose rate for the *abandonment scenario* is negligible.

6.4.3. 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 observation that the selection of phenomena, or FEPs, 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.

In CORA also 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 recognised several types of uncertainties that should be addressed further:

- The reliability of the data (CORA, 2001; Section 7.2.1). It was recognised 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 CORA 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).
- Calculation 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.

6.4.4. 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 abandonment and flooding of the disposal facility. This statement, relevant for all disposal concepts, needs further to be investigated.

The calculations performed in CORA were mainly focused on the abandonment scenario, and contain a number of assumptions that need further study.

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 might happen 2 or 3 orders of magnitude faster. But even then it will more than thousands years. One should seriously consider improvements of the design to shorten this period.
- When the disposal facility would flood the compaction of open spaces in the facility accelerates significantly. A convergence model for ***wet salt***, the FADT[cc] model of Spiers (1988), indicates that it would take about 4000 years to reach the 'final' porosity, i.e. a porosity similar to that of the undisturbed rock salt.

[cc] FADT: Fluid Assisted Diffusional Transfer, the transfer of material from salt grain contact interfaces to voids by diffusion through the liquid phase, driven by surface free energy gradients

- Rough estimates indicated that the calculated exposure for the abandonment scenario analysed in CORA may be significantly increased 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 it 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 was 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. This aspect has been investigated in some more detail in the PAMINA project.

6.5. PAMINA performance 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, 2009a). 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 comprehensive statistical analysis of the calculated results was performed.

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, 2009a; Sections 2.2, 2.3). In the PAMINA study only vitrified HLW was modelled as a source term. To calculate the compaction behaviour of the crushed salt backfill and the salt borehole plugs, the computer program *EMOS_ccm2* has been improved (Schröder, 2009a; Sections 3.2, 3.3).

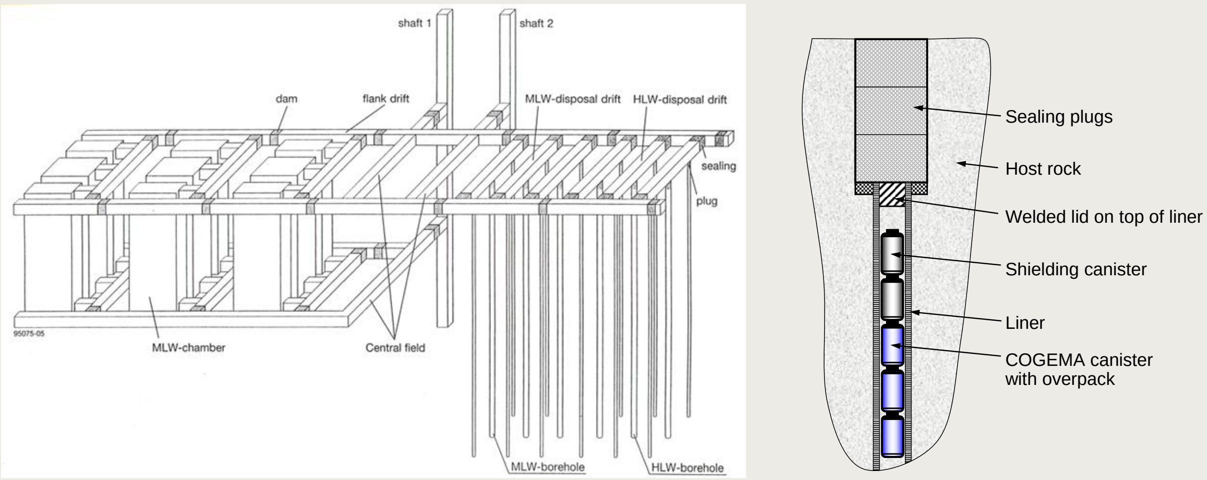


Figure 6-6 EVEREST repository and Torad-B borehole designs adopted in the PAMINA safety assessment.

An example of the calculated output is given in Figure 6-7 (Schröder, 2009a; Figure 7.2), showing the evolution of the *permeability of the sealing plug* (variable “B2”). The dry and the wet (flooding) scenarios show significant differences in compaction behaviour of the sealing plug. Ultimately compaction of the sealing plug results in complete isolation of the boreholes, but the time required is significantly longer for the flooding scenario than for the dry scenario, due to the hydrostatic pressure counteracting convergence due to rock stress. It was recommended in PAMINA to support this observation with additional studies.

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.

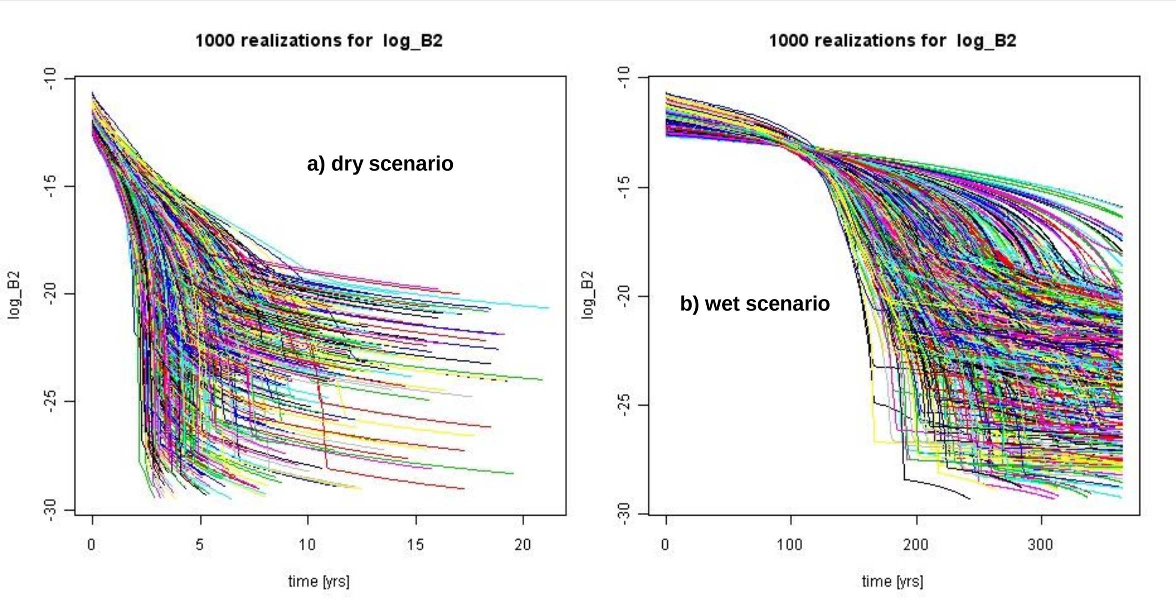


Figure 6-7 Evolution of the permeability of the plug (Variable "B2") over time for a) the dry scenario and b) the wet scenario.

The calculated potential dose rates in PAMINA 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 were far below any legal regulatory limit.

6.6. Safety assessment in Germany

The safety assessment methodology in Germany, more specifically the development of scenarios, was evolved 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 6-8 (Beuth, 2012a; Abb.3.1).

A number of basic assumptions were made in the project VSG 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 concerned 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.

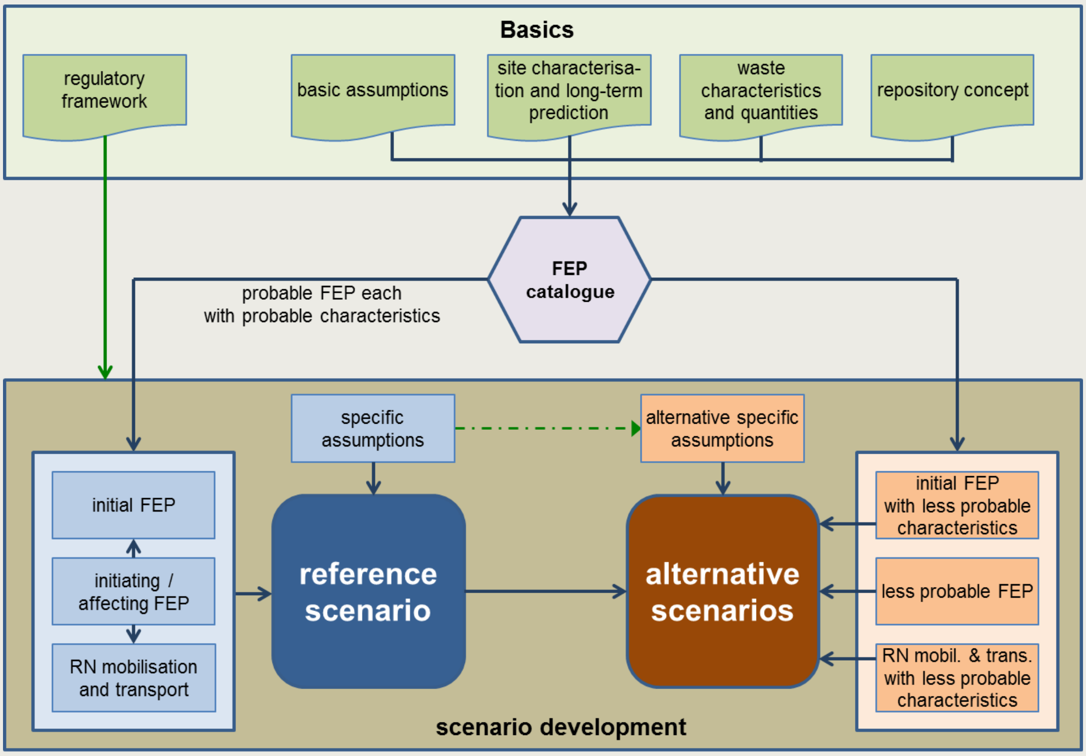


Figure 6-8 Methodology for the development of scenarios applied in R&D project VSG.

Taking specific assumptions into account, the *reference scenario* was developed by 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 and are developed from e.g. considerations regarding alternative assumptions, or less probable characteristics of FEPs adversely affecting a barrier of the system (Figure 6-8).

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 likely differ from the Dutch situation, since in the Netherlands no site selection process has been started yet. A comprehensive report on the preliminary German safety assessment was recently published as part of the VSG project (Larue, 2013).

6.7. Safety assessment in US

The WIPP programme utilised the performance assessment methodology as schematically represented in Figure 6-9 (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).

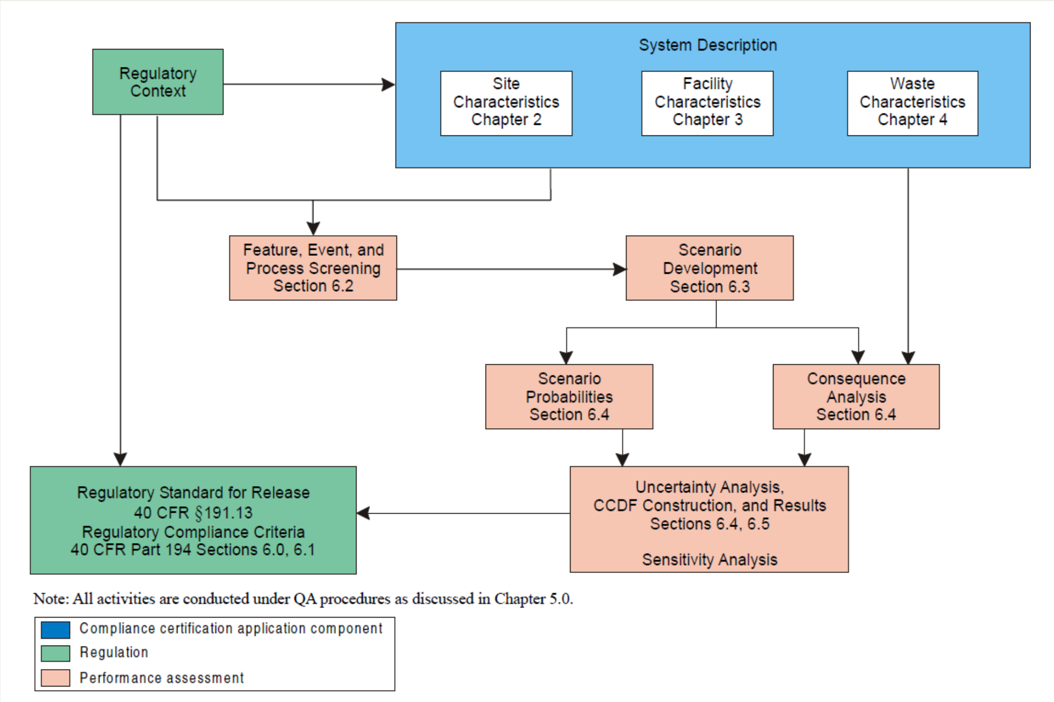


Figure 6-9 **Methodology for Performance Assessment of the WIPP (US DOE, 2004; p. 1-14).**

For developing the relevant scenarios, the WIPP FEP catalogue was screened, identifying 236 features, events and processes (FEPs), which were subdivided into 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 6-10 (US DOE, 2004; p. 6-34, p. 6-55).

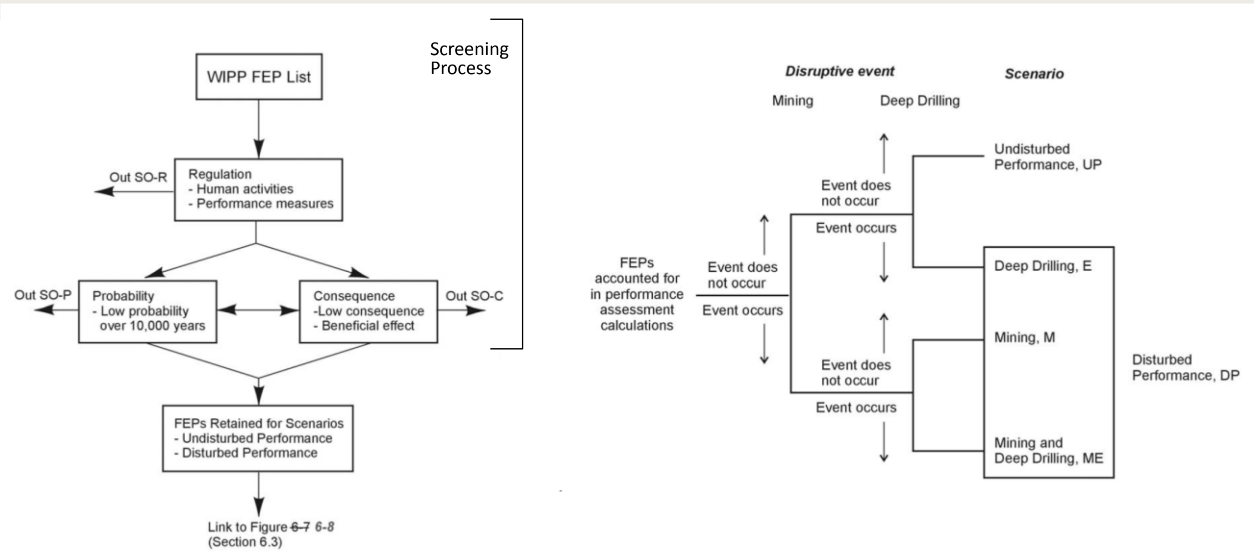


Figure 6-10 **Screening process based on screening classifications and logic diagram for scenario analysis.**

After screening, the remaining combinations of FEPs are included in logic diagrams to illustrate the formation of scenarios for consequence analysis. As in Figure 6-10 (right), a combination of occurrence/non-occurrence of potentially disruptive FEPs defines the derived scenarios. 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.

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.

The most recent results of the WIPP performance assessments have been reported as part of the 2014 “Compliance Recertification Application” CRA-2014 (US DOE, 2014b/c).

6.8. Compliance with safety criteria

Key to the safety assessment is the post-closure radiological impact. One important use of quantitative assessment results is for comparison with safety criteria; in particular with dose and risk limits or constraints. The calculated doses and/or risks are thus compared to regulatory defined limits in order to demonstrate the ‘safety’ of the system (e.g. IAEA, 2012; p. 46).

For example performance indicators can be used for the evaluation of the performance of parts of the disposal system. Because performance indicators provide a measure of the behaviour of an individual repository component or sub-system, they are more design- and site-specific than safety indicators. A comprehensive summary of the various types of indicators is given in (Rosca-Bocancea, 2013), treating the development of safety and performance indicators in geological disposal.

6.8.1. Safety indicators and reference values

Usually the following types of safety indicators are identified (Becker, 2009; p. 9):

- *‘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 other than dose and risk (Becker, 2009; p. 70), (NEA, 2012; p. 5). In most cases site-specific reference data are used, because these provide the most relevant situational context.

An example of a salt-specific safety indicator is the *Radiologischer Geringfügigkeits-Index* (index of marginal radiological impact), RGI. This indicator was developed within the ISIBEL project and acknowledges the specifications in the German Safety Requirements for a radiological indicator (Mönig, 2012; Section 4.6.1). The calculation of the RGI is based on a stylised calculational scheme, assuming that the total annual radionuclide flux released from the CRZ is diluted in the annual water consumption of one adult individual.

Complete containment of radionuclides 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 6-11). 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 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.

Consequently, the parameter RGI can be regarded as a quantitative measure of the safety function "containment" in the CRZ. This concept is straightforward to implement in a safety assessment and is worthwhile to consider in future safety assessments within the Dutch context.

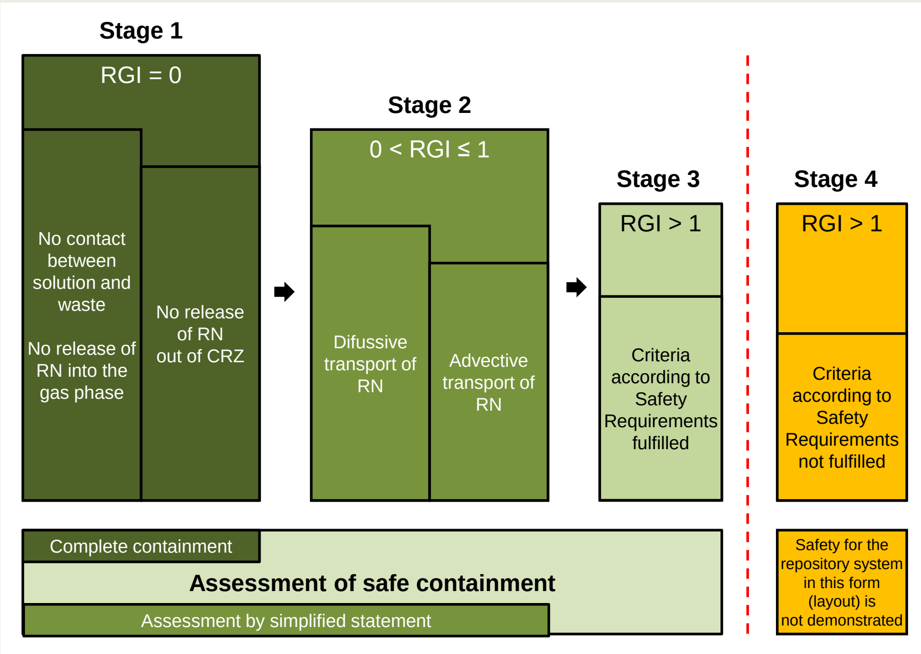


Figure 6-11 Staged approach for the long-term safety assessment - German approach.

6.8.2. Compliance with safety criteria

From the results of the safety assessment calculations performed in the Netherlands, Germany, and US, it consistently appears that calculated dose rates are generally significantly below regulatory limits for all considered scenarios, except for some human intrusion scenarios. For example, from the VEOS and PROSA 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). Note that the generally accepted regulatory limits concerning exposure are even lower (see e.g. Section 3.4.3, and Figure 6-2). However, for the Reconnaissance Drilling scenario receiving a high potential dose rate was judged extremely unlikely due to the very small probability of occurrence, and the conservative assumptions made for this scenario.

Also the results of the WIPP performance assessment in the US 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.

The results of the various safety assessments consequently and consistently demonstrate the isolating and confining properties of rock salt as a host rock for the final, deep geological disposal of radioactive waste, for a large variety of normal evolution and alternative evolution scenarios. However, for human intrusion scenarios in a disposal facility substantial exposures have been calculated in some cases. These cases may be re-evaluated to see whether the sometimes conservative modelling assumptions are still valid, or whether the repository design may be adapted to reduce the already small risk even further.

6.9. 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 subjects will be summarised in the following sections.

6.9.1. 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 6-12 (IAEA, 2004; p.17). Each of these components was extensively analysed and discussed during the project.

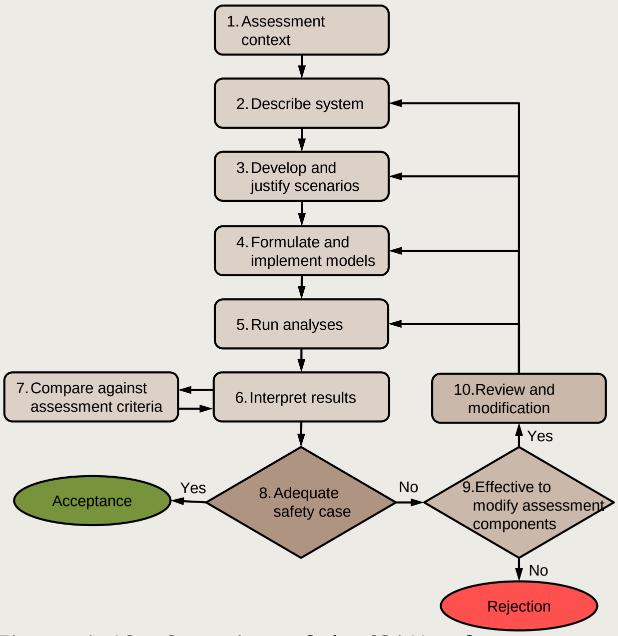


Figure 6-12 Overview of the ISAM safety assessment methodology.

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 itself. However, the idea that assessment results can serve as a basis for system optimisation (i.e. improving system performance and/or robustness by adaptations 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.

Despite the limitations of the ISAM flowchart many common elements and linkages may be identified with other recent flowcharts, such as is presently being applied in the OPERA safety assessment of a disposal facility in Boom Clay (Grupa, 2014).

6.9.2. Features, events, and processes

As already stated in the previous sections, a structured inventory of features, events, and processes (FEPs) which may affect the disposal system is essential for the evaluation of the safety of a disposal system. In addition, FEPs also may assist in the development of scenarios, describing the possible evolutions of a disposal system.

Since the CORA programme has ended several new initiatives have been engaged, primarily in Germany and US, on the development and extension of FEP catalogues.

As part of the German 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). The FEP catalogue was systematically compiled, starting from a comparison with the NEA-FEP database.

The ISIBEL FEP list evolved into a more site-specific FEP list, dedicated to conditions prevailing at Gorleben, viz. the VSG FEP catalogue (Wolf, 2012a), (Wolf, 2012b). That FEP catalogue is adapted to the German context as it takes the German regulatory requirements into account. The VSG FEP catalogue contains 115 FEP entries.

The WIPP FEP catalogue adopted in the US 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 FEP catalogue for disposal of heat-generating waste in salt (Hansen, 2013; p.32). 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 on rock salt.

Appendix 1 of OSSC Deliverable OPERA-PU-NRG221A (Hart, 2015) 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 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.

In conclusion it can be stated that FEP catalogues for the disposal of radioactive waste in salt-based repositories are still under development, mainly in Germany and the US. 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.

Another issue that may be relevant for the Dutch salt Safety Case is that structure and content of the German and US FEP catalogues reflect the respective safety concepts established for the Gorleben and WIPP disposal sites, respectively, as well as their specific scenario development methodologies. Therefore these FEP catalogues are not entirely applicable to the situation in the Netherlands.

6.9.3. Scenario development – extended PROSA methodology

As elucidated in Section 6.3.1, 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. However, that 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.

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 those ("transport") FEPs describing the transport of nuclides through the subsystem (or barriers) for the given scenario as illustrated in Figure 6-13 for the subrosion scenario (Grupa, 1999, Figure B).

Radioactive Material (RM)	Transport FEPs
Waste Matrix (WM)	*Leaching*
Host Rock (HR)	*Subrosion*
Overburden (O)	*Advection, dispersion*
Biosphere (B)	*Uptake of nuclides*
Man (M)	

Figure 6-13 Multi Barrier System for the subrosion scenario

Subsequently, FEPs are identified which are judged to adversely impact the subsystem (barriers). This methodology of FEP screening provides more details and information about the potential impact of features and events on the state of the barriers of the disposal system.

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 disposal parts and experimental facilities (29th January 2007[dd]), as well as 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).

6.10. Evaluation

As described in the previous sections, in the past several safety assessments have been carried out in the Netherlands as part of several national programmes and projects performed for the EU Framework Programme. The various studies were executed taking several disposal concepts in rock salt into account, and assuming different waste inventories that varied substantially between the various studies.

The following statements apply to the topic of safety assessment of a geological disposal facility in rock salt:

- The results of the VEOS and PROSA safety assessments indicated that the scenarios induced by natural events and processes result in a very low future exposure. High exposures can result from human intrusion scenarios, but the corresponding risk is very low (< 10^{-6}/a, assuming that the probability of these scenarios is 1).
- The Dutch 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 waste characteristics utilised in VEOS and PROSA were partly based on nuclear fuel cycle strategies assuming significantly more nuclear power generation than eventually applied, and therefore the results of these safety assessments can be judged as conservative, i.e. representing an upper limit in terms of long-term exposure.
- The methodology for the development of scenarios to assess the future, long-term safety of salt-based repositories by means of the screening of relevant features, events and processes (FEPs) influencing the repository's safety is quite well established.

[dd] The closure plan and safety report were delivered to the State Ministry of Mining, Energy and Geology (LBEG) in Clausthal-Zellerfeld on 29 January 2007.

- A careful evaluation of the results of the uncertainty and sensitivity analysis is a favourable method to underpin the robustness of the conclusions that can be drawn from the safety assessment. The work performed as part of the PAMINA project (see Section 6.5 of the present report) would be a good starting point to proceed further.
- FEP catalogues for the disposal of radioactive waste in salt-based repositories are still under development, mainly in Germany and the US. 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.

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 started yet.

In a next iteration 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 an established repository design that would be able to accommodate the presently foreseen waste categories in the Netherlands, and at a later stage to a specific site.

The results of the various safety assessments consequently and consistently demonstrate the isolating and confining properties of rock salt as a host rock for the final, deep geological disposal of radioactive waste, for a large variety of normal evolution and alternative evolution scenarios. However for human intrusions in a disposal facility substantial exposures have been calculated in some cases. These cases may be re-evaluated to see whether the sometimes conservative modelling assumptions are still valid, or whether the repository design may be adapted to reduce the already small risk even further.

7. Integration of safety arguments

7.1. Objective and scope

The present chapter provides a synthesis of the available evidence, arguments and analyses supporting the safety of a deep geological disposal facility in rock salt in the Netherlands. As no new safety assessment calculations have been performed in the OPERA OSSC project the safety arguments as obtained for OPLA and CORA have been reviewed and compared with current insights in process understanding of relevant safety features of the disposal design.

7.2. Safety assessment methodology and results

This section evaluates the similarities and differences between the main studies performed in the Netherlands, OPLA and CORA, and integrates common aspects of the various steps in the methodology:

1. scenario selection
2. determination of the probabilities of the scenarios
3. determination of the calculational model
4. determination of the parameters and their probabilities
5. dose calculation
6. sensitivity and uncertainty analysis.

7.2.1. Scenario selection

The main question to be answered with respect to the identification of future developments of the disposal system, i.e. the scenarios, is whether all 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), considering a list of phenomena which were assessed relevant for the release of radionuclides from the waste, their migration through host rock and overburden, and human exposure. The documentation of the selection method was rather poor as is common with subjective expert-judgement based methodologies. The main attention was focused on the identification of the scenarios and the most important processes, whereas less attention was paid to describing why the other features, events and processes (FEPs) were not relevant. The scenario selection process in VEOS identified 11 scenarios, which have been discussed in Section 6.2.1 of the present report.

In the scenario selection in **PROSA** (Prij, 1993; Ch. 2) a more systematic method has been developed. The main elements in that methodology were the consideration of FEPs, and their possible impact on the subsequent barriers of the disposal system. In the screening process and classification of the FEPs some subjectivity remained, which however has been documented properly. In the identification of the impact of FEPs on each multi barrier system state also some subjectivity could not be avoided. 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 abandonment scenario' (see also Section 6.4.1). 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 various salt-based disposal Safety Cases it is therefore recommended to define safety functions for the Dutch concept in rock salt, and to check for completeness of already identified scenarios by re-visiting the impact of FEPs on safety functions instead of multi barrier system states.

7.2.2. Probabilities of the scenarios

Nor in VEOS, PROSA, or CORA the probabilities of the scenarios have been determined. In these studies it was 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 were considered for the two scenarios leading to a relevant exposure: 'diapirism into the biosphere being a polar desert' and 'reconnaissance drilling'. For both cases the effects of a limited number of model parameters on the calculated exposure were calculated. The assumptions for these model parameters were selected to vouch for an upper limit of the calculated exposure.

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.

7.2.3. 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 by three different institutes, using three different computer programmes. In PROSA the three compartments were analysed by ECN only, applying however three different codes for modelling the three compartments.

For the **salt compartment** it appeared from the various modelling efforts executed in the Netherlands that, depending on the scenario, the variation in modelling of the salt compartment may in some cases significantly affect the calculated results (i.e. exposure). As a consequence a further study on recently developed salt compartment models and their implementation into safety assessment codes is recommended to assess their impact on calculated exposures, and to further reduce identified uncertainties for the salt compartment.

The **groundwater compartment** has been analysed extensively in VEOS and in PROSA by means of deterministic as well as probabilistic simulations. Also in Germany the effects of salt content in groundwater and the transport processes in the geosphere have been examined comprehensively (e.g. Buhmann, 2005). From the various modelling efforts it appeared that the variation in modelling of the groundwater compartment did hardly affect the calculated results. On the basis of the VEOS and PROSA results it can be stated that the existing knowledge of modelling the groundwater compartment sufficiently acknowledges the long-term safety-related issues. At present, the groundwater compartment and modelling aspects are being addressed in OPERA WP6.2: *Radionuclide migration in an aquifer*. The output of these efforts may also be revisited for a salt-based repository.

The **biosphere compartment** was modelled in **VEOS** with the dedicated BIOS code, whereas in PROSA the biosphere was simply modelled with stochastically distributed dose conversion factors which allowed for performing the probabilistic analyses. From the various modelling efforts in VEOS and PROSA it appeared that the variation in modelling of the biosphere compartment does not significantly affect the calculated results. This was confirmed in the German project VSG, where a variety of exposure pathways were analysed (Bollingerfehr, 2013; p. 111).

In OPERA the biosphere characteristics and modelling 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.

7.2.4. Parameters and their probabilities

In VEOS single values for all parameters were selected for performing the deterministic calculations. In most cases the values were selected with the intention to overestimate the exposure, implying a conservative approach. For some parameters the best estimate values were chosen.

In PROSA best estimate values have been used for most parameters. For parameters relevant to 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 the BIOS code with a set of randomly selected input parameters.

From the various modelling efforts performed in the past (VEOS, PROSA, CORA) it can be concluded that presently sufficient knowledge exists about which parameters need to be addressed in a safety assessment. However, obtaining numerical values of the parameters, including their uncertainty ranges, is not always straightforward since relevant parameters can depend on the temperature or be coupled through THMC processes and are often site-specific. Consequently, 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.

7.2.5. Safety criterion: dose rates

In **VEOS** extensive dose calculations have been performed for 21 different disposal concepts, three generic salt formation types (salt dome with 230 m thick overburden; salt pillow with 800 m thick overburden; bedded salt at a depth 1200 m), three waste strategies (see Table 5-1), and three disposal techniques (see Section 5.2.2).

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 that 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 mankind*" (Köster, 1989; p.58).

In **PROSA** dose calculations have been performed for 4 disposal concepts. 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, that the dose rates were again significantly below internationally accepted limits.

Additional efforts in the **EVEREST** and **CORA** confirmed that calculated dose rates for salt-based repositories are extremely low for the considered scenarios. In CORA however it was assessed that it will be necessary to study some variants of short-cut transport pathways, e.g. by assuming that a well produces water from the groundwater close to the facility, implying that the well water may contain traces of the waste.

On the basis of the calculated dose rates it was also concluded in PROSA that the possibility of human intrusion does not depend on a particular disposal technique, and can be reduced by disposal at greater depths. 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.

7.2.6. Sensitivity and uncertainty analysis

In **VEOS** the sensitivity analyses were performed in a deterministic and qualitative way or based on some global and simplified analyses. In VEOS no systematic uncertainty analyses have been performed.

In **PROSA** and its extension **EVEREST** the probabilistic sensitivity analyses have been performed in a more systematic way. Due to the approach followed in PROSA it was concluded that the internal rise rate of a salt dome (diapirism) is a considerable factor of uncertainty in the calculated exposure. In addition, the parameters of the crushed salt applied as backfill and the groundwater velocity are considerable sources of uncertainty. On the other hand, the properties of the rock salt and the biosphere dose conversion factors are a weak uncertainty source in the maximum exposure.

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 assessment has been performed in a deterministic way (scoping sensitivity).

Additional probabilistic efforts on modelling of the permeability of salt with low porosities were undertaken in the **PAMINA** project (Schröder, 2009a). The aim of the calculations was primarily to test probabilistic tools for the analysis of the results, rather than undertaking a radiological safety assessment. Since the efforts concentrated on the dynamic behaviour of sealing plugs the PAMINA exercise was essentially a performance assessment of such plugs.

A main outcome of the performed sensitivity and uncertainty analyses was an increased understanding of the behaviour of the geological disposal system, and deepened insight regarding the influence of parameters and processes on the long-term safety.

7.3. Implications for the design choices/site selection

From the performance assessments (cf. Chapter 6), 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 the safety-related aspects of these scenarios within the various salt disposal research programmes in the Netherlands.

7.3.1. Geological system

The diapirism/subrosion scenarios represent the response of the salt formation and its environment on natural processes. The results of the performance assessment give an indication of the isolation capacity of this geological system, and the sensitivity results can help to identify possibilities for design improvements and to support criteria for site selection. The following can be observed:

- The construction of the GDF and the disposal of the waste do not significantly influence the initial isolating capacity of the geological formation;
- The limited effects of thermal aspects on the isolating capacity of the host rock was demonstrated by the elaboration of a thermo-mechanical model for assessing the effects of changes in temperature and stresses caused by the disposal of heat generating waste(Prij, 1991; pp. 172-177).
- The results of the sensitivity analyses have shown that the uncertainties in the salt dome's internal rise rate and the groundwater velocity in the geosphere give the largest contributions to the uncertainty in the exposure and therefore in the isolation capacity of the geological system. The repository's design and site selection should help in reducing the uncertainty in these parameters. The site selection is, however, complicated because a prediction must be made of the magnitude of the relevant parameters in the near and far future. This must be done on the basis of data from the past.
- The sensitivity analyses have shown that uncertainty in the (hydrological) properties of the overburden significantly contribute to the uncertainty in the isolating capacity of the geological system, especially for the future properties. It is not straightforward to select a site where these uncertainties are bounded.
- Rather detailed site selection criteria were already formulated in 1979 (ICK, 1979). These criteria may be reformulated with the goal to reduce the uncertainty in the rising rate and the properties of the overburden.

7.3.2. Engineered barriers

A test of the performance of the engineered barriers concerns the consideration of the groundwater intrusion scenario, and its effects on the long-term safety. In this respect, the selection of the backfill and the designs of the seals and dams have been checked with performance assessments of this scenario. The following observations can be made:

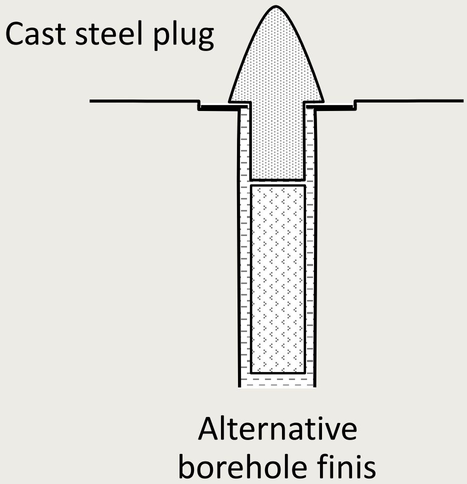
- From theoretical analyses and experiments it was demonstrated that backfilling of the excavated openings with crushed salt bounds the disturbances of the virgin rock salt in time. Due to the creep driven convergence of the excavation the crushed salt will compact and the porosity and permeability decrease from initially relatively high values to values comparable to that of the impermeable salt. So ultimately the disturbance will be 'repaired' (see also Section 5.6).
- 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.
- In PAMINA it has been shown that, assuming a dry, non-flooding scenario, a small amount of moisture present in compacted salt is extremely helpful in accelerating the compaction and ultimately reaching the undisturbed state in a relatively short period.

Although much information is already available about the behaviour of especially crushed salt as engineered barrier, additional research may contribute to the confidence of modelling the dynamic response of crushed salt to external forces, and the impacts of moisture.

7.3.3. 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 so far in the Netherlands have shown that a significant dose may be encountered when in the future human beings are trying to explore and use the underground.

Unintentional human intrusion can only occur after the knowledge of the presence of radioactive waste has been vanished. The following approaches are identified to enhance safety in relation to human intrusion:



- In the ICK period it was suggested to place a cast iron cone at the top of a borehole to make it impossible that a reconnaissance drilling hits the HLW directly (Hamstra, 1981; p. 40). This obviously only applies to disposal in vertical boreholes;
- In Germany the most promising methods to optimise the design within this respect would be (Bollingerfehr, 2013; p. 113) dyeing of backfill or adding coloured material to the backfill and placement of gravel in the openings on exploration level. These measures may indicate to the acting persons in the future that there is a special situation in the deep underground;
- In the US extensive studies have been carried out to design landmarks above the repository. At the landmarks symbols should be placed warning that dangerous material is stored at this location (US DOE, 2004; p. 1-10). In addition, storage is foreseen of the information of the location of the GDF at many different archives in the world.

With respect to the type of formation to be used for radioactive waste disposal, PROSA concluded that disposal at greater depths is the most important factor for reduction of the probability and radiological consequences of human intrusion. Because the human-intrusion scenarios were associated with the highest exposure rates and possibly the highest risk, these scenarios should be among the subjects of further research.

7.4.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).

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 for 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.

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.

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 Section 4.4), 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[ee] projects, both partially funded by the European Commission and the Dutch Ministry of Economic Affairs, in which NRG was respectively is an active participant.

7.5. Indicators and comparison with safety criteria

In IAEA document SSG-23 (IAEA, 2013; p. 30) it is stated that one of the aims of safety assessment is to compare the end points for the safety assessment with safety criteria. Most national regulations relating to repositories for nuclear waste give safety criteria in terms of dose and/or risk (cf. Section 3.4.3), and these indicators are evaluated for a range of evolution scenarios for the disposal system using quantitative analyses.

In recent years it has become evident that an overall system safety assessment can be augmented with additional analyses and complementary indicators, e.g. within the MeSA[ff] project (NEA, 2012). The review confirmed that indicators complementary to dose and risk are now accepted by the majority of regulators and implementers as an important component of a Safety Case.

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 are 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). This has not been done yet in the Netherlands.

Additionally, performance indicators show more clearly the repository´s intrinsic performance as well as the behaviour of an individual repository component or sub-system. Because these insights have only recently been developed, i.e. after the previous Dutch national programmes on geological disposal, indicators complementary to dose and risk have only limitedly been assessed in the Netherlands. The PAMINA project applied complementary indicators, but the PAMINA exercise cannot be regarded as a full safety assessment of a Dutch, salt-based disposal facility (Schröder, 2009a).

[ee] http://cordis.europa.eu/project/rcn/196921_en.html, last accessed 30 June 2015
[ff] MeSA: Methods for Safety Assessment

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.

7.6. Robustness

In (NEA, 2004) the term robustness is described as a feature that is favoured by the multi-barrier concept, i.e. a system of features operating 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.

Robustness refers to various aspects of the Safety Case and 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:

- 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;
- Application of good engineering practices (demonstrability, feasibility).

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 7-1.

Table 7-1 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[gg]	Through a system of Regulations[hh]
Repository design	Generic design	Specific for Gorleben	Through a system of Regulations
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 4.9.1)	Safety demonstration concept (see Section 4.9.2)	By the construction and operation of WIPP

7.7. Natural analogues

It is not possible to simulate in laboratory studies the very long-term processes that might affect the safe performance of a repository. An important group of arguments to overcome this and to contribute to building confidence in the safety of salt-based repositories concerns natural analogues (IAEA, 2013; p. 41). The main value of such studies is to

[gg] Bollingerfehr, 2013; Section 4.2
[hh] E.g. 10 CFR 60 (US DOE, 2014)

provide information of the full complexity of the repository system and of the characteristics of salt-specific processes over long time scales.

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).

Within that 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 and assessed for their applicability in the Safety Case. The first two aspects are most relevant for the safety demonstration concept, since they are related to the containment providing barriers.

The major conclusion from the ISIBEL study is that for assessing the integrity of the *geological barrier* a lot of well-described natural analogues are available. Additional potential natural analogues were 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 of *geotechnical barriers*. This is especially true for the compaction of crushed salt.

A recent workshop “Natural Analogues for Safety Cases of Repositories in Rock Salt” (NEA, 2013b) provided a comprehensive overview of studies regarding features which might be considered as analogues to support the salt safety case.

7.8. Evaluation

During the last four decades much effort has been devoted in the Netherlands to the geologic disposal of radioactive waste in rock salt, more specifically in the framework of the ICK, OPLA, OPLA-1A, and CORA programmes. Additional work has been done in several EU Framework projects like EVEREST, BAMBUS, PAMINA, and THERESA. In addition, dedicated programmes in the US and Germany resulted in an abundance of information on the development and implementation of a Safety Case for geologic disposal in rock salt.

All the efforts, together addressing safety-related aspects of salt-based geological disposal, resulted in sufficient confidence, to construct and operate a dedicated repository for the disposal of radioactive waste, e.g. in the US (the WIPP). In Germany, there is also significant practical Safety-Case-experience with the ERAM and Asse facilities. Additionally, the Dutch and international efforts demonstrate that the option of geological disposal in rock salt can be pursued for finally isolating radioactive waste from the biosphere.

On the other hand, in the Netherlands there has been limited activity with regard to the development of a disposal facility in rock salt since the CORA programme. 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 laid down in this chapter is necessarily partly based on generic considerations.

For the next phases of the development of the Dutch salt Safety Case it is recommended to revisit the aspects treated in this report relevant to safety, also taking into consideration the most recent information from other disposal programmes for salt-based disposal. That would lead to an increased understanding of safety-related aspects and a comprehensive Safety Case for the salt-based disposal of radioactive waste in the Netherlands.

8. Recommendations

In the last 40 year many research activities have been performed, leading to a deep understanding on the suitability of rock salt domes as host rock for the disposal of radioactive waste. After 15 years of limited research activities on rock salt in the Netherlands, the OPERA project Salt Safety Case (OSSC) assessed the current knowledge base concerning the safety and feasibility of the geologic disposal of radioactive waste in a rock salt formation, processed this knowledge according to the methodology of the Safety Case for deep geological disposal and identified existing knowledge gaps.

The present report 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. Following the systematic, pointwise evaluations of the various components necessary to build a Safety Case in the previous chapters, in this chapter recommendations to iterate and further develop the Dutch salt Safety Case are formulated. For a more detailed discussion of the topics see also OPERA-PU-NRG221A (Hart, 2014a).

Following the systematic structure of the previous chapters, the sections in this chapter discuss the main topics that need additional elaboration to proceed to the next phase of the Dutch Safety Case for the geological disposal in rock salt in the Netherlands. A distinction has been made between short-term themes which may be initiated and executed in an OPERA follow-up program, and long-term themes which require continuous efforts or are of interest in later stages.

Below, recommendations for the short-term are marked in orange, and recommendations for the long-term in blue. All recommendations are numbered for reference (in Chapter 9 and Appendix 2), where the recommendations derived in this chapter will be applied to determine the necessary steps to arrive at a national Safety Case for a disposal facility in rock salt, comparable to the Safety Case on Boom Clay as developed in the OPERA programme. This results in a roadmap showing how the elements discussed in this chapter can be tied together and addressed in a stepwise manner in order to build and update the Safety Case on rock salt.

8.1. Safety Case context

8.1.A Although the 1984 *Policy Document* (VROM 1984) indicates that geological disposal of **all** Dutch radioactive wastes is foreseen, a recent discussion document related to the potential consideration of surface disposal of low and intermediate-level radioactive waste opens up the possibility to flexibly treat the selected option of long-term management (Kamp, 2013). It is recommended to closely monitor these developments and to re-evaluate their consequences for the Dutch waste management strategy.

8.1.B It cannot be excluded that in 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 organise the implementation of the basic laws. It is therefore recommended that the information provided (MinEZ, 2013) should be updated on a regular basis.

8.1.C It cannot be foreseen how the laws and responsibilities will change in the coming 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 recommended to develop specific regulatory aspects with regard to (1) safety criteria, (2) siting issues, (3) licensing of such a facility.

8.2. Safety Strategy

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[ii]. The Netherlands is drafting the required 'Nationaal Programma' (Kamp, 2013) according to the definition provided by this Directive. As these documents are of a generic nature no explicit recommendations related to salt-specific issues apply here, except from regular assessing and updating their contents if necessary.

In the previously executed programmes and projects OPLA, OPLA-1A and CORA, the Dutch safety concept for the geological disposal of radioactive waste in rock salt has been worked on. Since 2001 no systematic activities have been performed on this topic in the Netherlands, whereas in Germany and US substantial progress has been made.

It is recommended to critically review the Dutch (multi-barrier) safety concept of final disposal in rock salt, both in the light of recent international developments and the most recently adopted Dutch strategy with regard to the following topics:

8.2.A
- Safety concept: the elaboration of specific safety functions for a disposal facility in rock salt, more explicitly for the various components of the facility, e.g. engineered barriers, the waste matrix, waste container, backfill, seals, dams, and shafts, as well as the host rock.

8.2.B
- Safety demonstration concept: the development of a more systematic approach for demonstration of the long-term safe containment of the waste by validating the integrity of the geotechnical and geological barrier, as well as an evaluation of radiological consequences for future evolutions of the disposal system. Here, the principles recently developed in Germany can serve as a guideline (cf. Section 4.9.2). These activities can be elaborated on short term, to assess the need and benefits of participation in international cooperations.

8.3. Disposal system

The following sections summarise the main recommendations in relation to the various parts of the disposal system.

8.3.1. Facility designs

For the disposal of all types of radioactive waste all conceptual facility designs considered in the Dutch research programmes until now are of a generic nature and have been considered in three generic salt formations: a deep salt dome, a shallow salt dome and a salt layer. However, since the CORA programme the characteristics of the radioactive waste intended for disposal have changed, implying that the previously considered facility concepts for disposal in salt are presently not adapted to receive the presently considered waste types such as spent fuel from research reactors and depleted uranium.

8.3.1

An important recommendation to proceed 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 recently identified waste characteristics;
- an up-to-date safety concept including the definition of safety functions for the various components of the multi-barrier system; and
- the possibility to retrieve the waste, including intermediate- and low-level radioactive waste.

[ii] 'Establishing a Community framework for the responsible and safe management of spent fuel and radioactive waste', 19 July 2011 (EU, 2011)

Such a disposal facility outline may be based on the existing METRO-I concept elaborated in the CORA programme (see Section 5.2.3), or the German disposal concept (see Section 5.2.4). In the last case it must be noted that the current German disposal concept is adapted to the conditions prevailing at the Gorleben site, which likely differ from the Dutch situation.

8.3.2. Waste characteristics

The characteristics of radioactive waste intended for final disposal serve as a source term for the long-term, post-closure safety assessment.

Previous safety assessments in the Netherlands, performed for salt-based repositories, in VEOS, PROSA, CORA, and PAMINA, assumed waste characteristics and radionuclide inventories which differ considerably from the presently foreseen inventories (Hart, 2014b). More specifically, the spent fuel from research reactors (HFR, and Pallas), and the significant quantities of depleted uranium have not been taken into account in the previous studies. As a consequence, the quantitative results from the past safety assessments would require quite some interpretation for extrapolation to a future geological disposal facility in rock salt for the presently foreseen waste inventory.

8.3.2 It is therefore recommended to update previously performed safety assessments, taking into account the presently foreseen nuclide inventories and waste fractions, thus including the spent fuel from research reactors, and the significant quantities of depleted uranium.

8.3.3. Engineered barriers

The main applied engineered barriers in salt-based repositories are pre-compacted salt plugs for sealing the disposal cells, crushed salt backfill, dams to isolate repository modules, and shaft seals to finally isolate the subsurface structures from the biosphere.

Considering the present state of the Dutch strategy for the final disposal of radioactive waste there is no urgent need to design plugs, dams, and shaft seals. Additionally, in Germany significant and relevant information is available concerning the design and performance of gallery seals (drift seals, dams) in salt-based repositories.

8.3.3 It is however recommended to actively participate in international research initiatives, especially collaboration with the German and US programmes, as well as in the EU framework projects, like the FP7 project DOPAS, to keep up with the knowledge and safety aspects of seals, dams and plugs.

Recommended aspects considering crushed salt as backfill are mentioned in Section 0.

8.3.4. Salt formations

Part of the extensive studies performed during the 1980's and 1990's in relation to disposal of nuclear waste in rock salt formations in the Netherlands is still relevant and up-to-date.

8.3.4 However, knowledge on salt domes is steadily increasing, mainly due to their use for salt extraction and possible gas storage. With respect to gaining additional knowledge on salt-dome properties in the Netherlands it is recommended to focus on the following aspects:

- Acquiring detailed 3D seismic maps 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 gases;
- 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.

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

On the basis of the evaluation of the abundant information on THMC processes in salt-based repositories, and the discussion on chemical aspects in Section 5.6.5, the following 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 and transport of radionuclides

These topics are elucidated in the following paragraphs.

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.

8.3.5.A 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.

8.3.5.B 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 the same 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.

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

- 8.3.5.C 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;
- 8.3.5.D Enhancement of the understanding of the porosity development during the compaction process as well as the correlated permeability, which adds to the understanding of reaching a complete isolation of the disposed waste.

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

HMC effects related to the dissolution of rock salt

Understanding 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. They are therefore of relevance for the assessment of the geo-mechanical integrity of the barrier function of the host rock.

At present, it is judged that sufficient knowledge is available 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 properties of the concept will become more site-specific and they have to be matched to the composition and properties of existing rock salt formations available in the Netherlands.

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, it is recommended to perform the following extensive actions:

8.3.5.F
- To inventarise which mineral dissolution processes can be of principal interest considering the anticipated size of a Dutch disposal concept compared to the size and (chemical) composition of available rock salt host formations in the Netherlands;

8.3.5.G
- To judge, based on the previous inventory, whether the range of expected mineral interactions is sufficiently well covered by existing experiments;

8.3.5.H
- To evaluate which geological features need to be addressed in order to elaborate a Safety Case on the feasibility in later stages. Examples are dome-specific heterogeneities and related creep mechanisms, geomechanical stability studies for determining the distances from the roof of the cavern to the top of the salt formation and between adjacent salt caverns, leading to estimates of cavern convergence and subsidence rates. Such an evaluation might also be used to define exclusion criteria for siting locations.

8.3.5.I With respect to the constitutive modelling of chemical processes, the activities of the German THEREDA project (Moog, 2015) are of great importance, and it is therefore recommended to actively pursue collaboration in that project.

8.3.5.J In addition, it is recommended to keep track on computational developments for modelling the complex hydrological, (geo)mechanical and (geo)chemical processes[jj] in salt formations in their natural hydrogeological setting in 3D. This can be accomplished by e.g. active participation in international working groups (NEA Salt Club, US-German Collaboration), and technical meetings on this matter (e.g. ABC-Salt[kk]-meetings).

Corrosion of waste container and waste matrix

Much is already known on the corrosion of iron in highly saline environments. For geochemical modelling of Calcium Silicate Hydrate (CSH) phases in cementitious material,

[jj] Thermal effects may be considered, too, but it seems of less relevance taking into account the presently adopted extended surface storage period of heat generating waste.
[kk] ABC-Salt: Actinide Brine Chemistry in a Salt-Based Repository

several approaches exist, but there is additional need to understand the effects of the dissolution behaviour of cementitious material on relevant features determining the solubility of radionuclides, e.g. the pH.

It is therefore recommended to address this aspect further in a future programme by:

8.3.5.K
- Analysing CSH corrosion processes in saturated brine, their influence on radionuclide solubility and mobility, and assessing existing uncertainties, through an evaluation of existing literature on CSH in saline environments. Since potential corrosion and degradation reactions depend on the composition of the brine, the studies should carefully consider the source of the brine in the Dutch situation;

8.3.5.L
- If necessary, establishing an experimental programme of critical parameters and processes based on the outcome of the modelling study.

Corrosion of cementitious barriers

Degradation and dissolution processes can impair the proper geo-mechanical and hydraulic function of technical barriers in the disposal facility. However, in the Dutch disposal concepts no cementitious barriers like shaft seals or dams have 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.

8.3.5.M It is therefore recommended first to clarify constructional aspects of cementitious barriers and their role in the safety concept.

8.3.5.N Additionally, 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 and transport 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 relation to the solubility of radionuclides it is recommended to focus on the following aspects:

8.3.5.O
- due to minor sorption on the salt 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;

8.3.5.P
- The elaboration of non-equilibrium reaction kinetics in a saline system, still being in an initial state due to the lack of many reaction parameter values, may be taken forward.

Gas production and transport

Gas can be included in the rock salt and may migrate towards the repository. Gas can be produced by the waste in the repository as a result of corrosion processes, too. Past investigations have indicated that the uncertainties related to gas production and transport are still significant.

8.3.5.Q It is therefore recommended to address this aspect further in a future programme by assessing the complex behaviour of gas-related processes in safety analyses, such as the pressure buildup of hydrogen gas due to anaerobic corrosion of steel container materials, microbial gas generation from degradation of organic material, and the modelling of gas production and transport in computer codes.

8.3.6. Overlying sediments and biosphere

Considering the sediments overlying the salt formations, and the biosphere, the following recommendations are proposed:

8.3.6.A
- The approach followed in the studies performed so far applied a generic model for the overlying sediments and the biosphere. As the Dutch Safety Case for geological disposal progresses, the need will arise to adapt the modelling features to a specific site.

8.3.6.B
- As the contribution of the geosphere to the overall long-term safety for a disposal concept in rock salt might be different from that for Boom Clay, e.g. due to greater depth and subrosion effects, a more detailed geosphere transport modelling approach may be developed than currently applied in OPERA. Such an approach should also account for any effects of the higher density of brine that might be forced out of the salt host rock.

8.4. Safety Assessment

With respect to the safety assessment methodology recommendations are put forward for the following topics:

- Features, events, and processes – FEPs;
- Scenario development;
- Handling of uncertainties.

8.4.1. FEPs

8.4.1.A The previous (limited) assessment of FEPs within the Dutch context dates back to 2001 (the CORA programme). Taking into account recent elaborations of FEP catalogues, and the use of FEPs in the development of relevant scenarios for the safety assessment, it is recommended to re-visit FEPs within the Dutch context. This can be achieved similarly to the elaboration of the OPERA FEP catalogue for Boom Clay (Schelland, 2014). Recent developments in Germany (VSG FEP catalogue) and US (the WIPP FEP catalogue) may be helpful in such an exercise, although it must be noted that these FEP catalogues are adapted to the specific sites.

8.4.1.B Additionally it is recommended to seek for active participation in the presently ongoing compilation of a comprehensive FEP catalogue for disposal of heat-generating waste in salt by German and US institutes in the framework of the OECD/NEA Salt Club.

8.4.2. 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 screening of FEPs with regard to the probability of their occurrence and the potential consequences on the (long-term) safety, and the subsequent formulation of scenarios which need to be analysed in a safety assessment.

In PROSA the scenario selection was done by identifying combinations of FEPs that could affect each *multi-barrier system state* of the disposal system. A more recent development is to identify (combinations of) FEPs that could affect the *safety functions*. This methodology only applies if safety functions have been defined for the Dutch concept in rock salt (cf. Section 8.2).

8.4.2.A It is recommended to re-visit the sets of scenarios which have been developed in the past in the Netherlands, utilising an updated FEP (salt) catalogue, and identify (combinations of) FEPs that could affect the *safety functions* of the salt-based disposal system. Consulting the German approach for the development of scenarios (cf. Section 4.9.2) is considered to be helpful.

8.4.2.B In previous safety assessments, the human intrusion scenarios often resulted in the largest calculated exposure, although they impose a very low-probability risk. However, these scenarios were assessed in a simplified and over-conservative manner. It is therefore recommended, when re-assessing human intrusion scenarios, to take into account recently developed information, e.g. within the IAEA project HIDRA[ll].

8.4.3. Determination of parameters

It is judged that presently sufficient knowledge exists about which parameters need to be addressed in a safety assessment.

8.4.3.A However, the effects of elevated temperatures on parameter values need to be addressed with more accuracy, especially related to assessing chemical aspects on the basis of thermochemical databases (see also Section 0).

8.4.3.B Ultimately, numerical values of parameters need to be adapted to the in-situ conditions of a specific site and the prevailing boundary conditions.

8.4.4. Uncertainties

In the PROSA study uncertainties already have been addressed to a large extent, especially in relation to identifying the main contributors to the long-term exposure of the future population. The PROSA safety assessment however assumed waste characteristics which differ significantly from presently expected ones (cf. Section 8.3.2).

8.4.4 For the next iteration of the safety assessment within the Dutch context it is therefore recommended to carefully evaluate the procedures applied in PROSA, and to perform uncertainty and sensitivity analyses to show the robustness of the various components of the disposal system. The work performed as part of the PAMINA project (see also Section 6.5 of the present report) would be a good starting point to proceed.

8.4.5. Constitutive modelling and computer codes

Under the conditions prevailing in a repository, THMC processes are strongly related. Constitutive laws to model coupled processes have been proposed and developed by many companies and research institutions and over a long period of time.

In the Netherlands, the main computer code applied for radiological safety assessments in rock salt is EMOS, developed in the last 20 years, an offspring of the German EMOS4 code. However, while EMOS allows to assess rock salt compaction behaviour and its effects on radionuclide migration, many THMC processes cannot be modelled. Other computer codes are under development and include recently developed information of salt-related THMC processes, primarily in the US (e.g. PFLOTRAN), and Germany (e.g. d^3f and r^3t).

8.4.5.A It is recommended to inventory what recently developed computer codes are available for performing safety assessments of salt-based repositories, and to compare their capabilities

[ll] HIDRA: Human Intrusion in the context of Disposal of Radioactive Waste
http://www-ns.iaea.org/projects/hidra/ - last accessed on 20 July 2015

for process modelling as basis for PA modelling.

8.4.5.B Based on a decision what computer code is best suited for performing safety assessments within the Dutch context it is recommended to join the international community for further code development and benchmarking.

8.4.6. Safety assessment - final considerations

As described in the previous sections, in the past several safety assessments have been carried out in the Netherlands as part of national programmes and projects performed for the EU Framework Programme. The various studies were executed assuming several disposal concepts in rock salt, and assuming waste inventories that varied substantially between the various studies.

8.4.6 However, the previous safety assessments are presently judged as outdated and need to be upgraded. Consequently, it is recommended to repeat the safety assessments for a dedicated repository design using recently acquired information and newly developed methodologies concerning e.g. waste characteristics, FEPs, scenario development, and uncertainty/sensitivity analysis.

8.5. Compliance with safety criteria

A crucial step in the safety assessment, being an important aspect of the Safety Case, is showing compliance with safety criteria. In the past the usually applied safety criteria were dose rate and risk. However, a modern safety assessment also encompasses the demonstration of the robustness of the individual repository components of the disposal system, such as engineered barriers and the host rock itself. The definition and estimation of safety and performance indicators is crucial in that respect.

In relation to the topic of compliance with safety criteria the following topics are considered to be taken forward in the next iteration of the Dutch Safety Case for the disposal of radioactive waste in rock salt:

8.5.A
- Safety and performance indicators are valuable tools to quantitatively assess the long-term safety of a geological disposal facility, and should be analysed in more detail than previously done. The indicators evaluated in the PAMINA project are a thorough basis for such an assessment.

8.5.B
- Calculated values of safety and performance indicators make sense if they can be compared to reference values or other '*yardsticks*'. While OPERA already provides a large set of safety and performance indicators, it is recommend to give focus to performance indicators that help to analyse the very specific, and more complex system behaviour of a disposal concept in rock salt. In some cases it might be useful to develop '*yardsticks*' for these indicators as well.

8.5.C
- 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. That 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.

8.5.D
- Upon the earlier stated recommendation to formulate safety functions for a salt-based disposal concept in the Dutch context, it is recommended to define properly outlined *safety function indicators*.

8.6. 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).

Given the present status of the Dutch geological disposal programme, there is no urgent need to develop monitoring equipment for a geological disposal facility. However, since monitoring is considered relevant in the Dutch safety strategy (see Section 4.4), it is important to develop monitoring *strategies* in support of decision-making based on the recorded monitoring data, and to establish its role for retrievability. Part of these topics are subject of the MoDeRn (White, 2013) and recently commenced Modern2020[mm] projects.

With respect to retrievability - of lesser relevance in Modern2020 - it can be noted that general topics as why, and how long to monitor go beyond the specific Safety Case on rock salt, and should be addressed within the general developments of RWM. For recommendation with respect to this topic see (Schröder, 2015).

8.6 However, when developing safety functions and the accompanying system design for a disposal in rock salt, it is recommend to closely incorporate the lessons learned from MoDeRn (NDA, 2013), Modern2020 and (Schröder, 2015) with respect to options and potentials to allow surveillance as part of the Dutch policy on "isolation, control and monitoring" (ICM). Although ideas on this are currently under development (Modern2020), it is recommend to consider how performance indicators can be linked to (future) monitoring activities, in order to integrate the ICM policy more closely into the Safety Case.

8.7. International collaboration

8.7 Since the Dutch community involved in research regarding geological disposal is limited in capacity and resources, international collaboration is a very cost-effective manner to advance salt repository sciences and implementation in the Netherlands. It is therefore recommended to actively participate in international programmes and projects, e.g.:

- The OECD/NEA Integration Group for the Safety Case (IGSC), aiming to develop Safety Cases supported by scientific technical basis;
- The OECD/NEA Salt Club, aiming to develop and exchange scientific information on rock salt for hosting deep geological repositories for radioactive waste;
- IAEA-hosted coordinated research projects related to Safety Case development and radioactive waste disposal (currently[nn] e.g. PRISMA, GEOSAF-II, HIDRA);
- The US-German workshop on salt repository research, design, and operation, a collaboration between US and German institutes and companies aiming to combine the technical basis for salt disposal;

[mm] http://cordis.europa.eu/project/rcn/196921_es.htm - Last accessed on 17 June 2015

[nn] **PRISMA**: Application of the Practical Illustration and Use of the Safety Case Concept in the Management of Near-Surface Disposal Project
GEOSAF: International Project on Demonstration of the Operational and Long-Term Safety of Geological Disposal Facilities for Radioactive Waste
HIDRA: International Project on Human Intrusion in the Context of Disposal of Radioactive Waste

8.7

- Collaborative projects partially funded by the European Commission, currently[oo] e.g. Modern2020 and DOPAS, and future relevant projects;
- IG-DTP - Implementing Geological Disposal of Radioactive Waste Technology Platform, a scientific and technical forum hosted by the European Commission to provide the necessary focus in the lead up to the operation of geological repositories for high-level nuclear waste in Europe.

[oo] **Modern2020**: Development and demonstration of monitoring strategies and technologies for geological disposal)
DOPAS: Full-Scale Demonstration Of Plugs And Seals

9. Roadmap towards a Safety Case for rock salt

The previous chapters summarized the substantial knowledge on rock salt presently available in a systematic, pointwise manner, and highlighted relevant aspects that need further attention in future. The main conclusion is twofold:

- On the one hand it can be concluded that currently still quite a lot of loose ends concerning the final disposal of radioactive waste in salt-based deep geological disposal facilities in the Netherlands that have to be tied off before a quantitative safety assessments can be performed that forms the basis of an initial Dutch Safety Case for disposal in rock salt;
- On the other hand, although specific shortcomings are identified, in general sufficient knowledge and experience exist to realize such a Safety Case with reasonable efforts, by efficient use of the OPERA results. This includes methodological and contextual aspects that can adopted 1:1 from OPERA, approaches to set-up and perform safety assessments, parts of the PA model, and others.

In this chapter, a roadmap is presented that describes how such a generic, location independent initial Safety Case on rock salt - comparable to the Safety Case on Boom Clay developed in OPERA - can be realized in the near future. Following the methodological approach proposed for the Boom Clay Safety Case (e.g. Becker, 2013; Hart, 2014a; Grupa, 2015; Rosca-Bocancea, 2013; Schröder, 2013), the Safety Case on rock salt should allow a principal statement on the safety provided by such a concept, clarify which processes are covered sufficiently well (and which not), and provide guidance on the relevance of various processes for the long-term safety.

Before analysing the necessary efforts, it can be noted that compared to the Safety Case on Boom Clay currently being developed within OPERA, both the starting point as well as specific features of a Safety Case on rock salt clearly differ:

- Many components of the salt Safety Case do not need further development but can be adopted - incidentally with minor adaptions - from the OPERA Safety Case on Boom Clay (e.g. methodological aspects, inventory, biosphere model).
- Relevant elements of the Safety Case on rock salt have already been elaborated in the past, their necessary update will only involve minor to medium efforts (e.g. FEP-list, scenario development).
- While the modelling of radionuclide migration in Boom Clay is built on approaches used in many other fields of science, the processes that determine the migration of radionuclides in rock salt are unique in their combination of physical, geomechanical and chemical interactions. As a consequence, tool development is much more important for rock salt, because computational modelling tools or approaches cannot just simply be adapted from other fields of science. It is also to be noted that only two other countries, Germany and US, are working on the development of such tools.
- While the integrated modelling framework for a repository in Boom Clay developed in OPERA allows to be used <u>both</u> for detailed process studies and PA analyses (and all steps in between)[pp], it is judged that, from current state of code development, such an elegant and time-efficient approach will not be achievable for rock-salt in the near future. As a consequence, different codes have to be used for process modelling and PA.

[pp] Note that this is a unique feature of the chosen approach and underlying modelling framework!

- Some relevant processes, although sufficiently well elaborated and understood, are not yet available as computational models allowing investigation of the relevance of these processes for the overall system behaviour of rock salt (e.g. EDZ behaviour, THMC processes). Such computational models were out of reach for the research performed in the previous Dutch research programmes. Additionally, the limited Dutch research efforts performed on rock salt in the last 20 years did not allow for building or utilizing modelling tools that allow studying combinations of processes discussed in Section 8.4 and evaluating their relevance for the safety.
- Due to the much more complex system behaviour of a disposal system in rock salt (see e.g. Schröder, 2009a), and the lack of tools modelling coupled processes as discussed above, a judgement on the contribution of features or processes on the safety is more difficult than in case of Boom Clay. Although general guidance can be given based on past experience, it is not possible to give further suggestions on the relevance of the various processes discussed in Section 8.4. This leads to the recommendation to perform further steps in an iterative manner, with regular sensitivity analyses performed to provide guidance for process-related research activities.

Based on the consideration above, the general recommendation for further development of the salt Safety Case is to follow an extensive, stepwise approach that takes maximum benefit from the outcomes of OPERA, existing knowledge and future developments, either as part of national projects or bilateral cooperations. The approach recommended and described below is divided into three phases, with each phase allowing to critically evaluate the relevance of the topics discussed in the previous chapter for the overall safety, which is necessary to focus resources and give guidance to further research activities:

- ***Phase 1***: Base model compilation & First assessment
- ***Phase 2***: Completion of process representation & Refinement of disposal concept
- ***Phase 3***: Delivery of Rock Salt Reference Model & Development of initial Safety Case

In the ***first*** phase quantitative assessments are performed, based on already existing computer models and tools, and by using actual, updated data or (partial) process models obtained from OPERA and elsewhere (e.g. on the inventory, solubility of radionuclides, transport in geosphere, transport and uptake in biosphere). This includes also an updated disposal concept, based on the currently considered waste inventory.

In the ***second*** and ***third*** phase, two main groups of presently identified shortcomings are addressed sequentially:

- ***tool development***: a number of processes are rather well understood, but not sufficiently covered by existing models;
- ***increasing process understanding***: some processes are currently not well understood and/or need further refinement with respect to their embedding in disposal modelling.

The overall effort necessary to prepare a Safety Case is estimated at about 5 to 8 man-years, spread over a period of 3 to 5 years. Table 9-1 summarizes the three phases and their scope.

In the following sections, the activities of each phase are further elaborated, linked to the various recommendations presented in the previous chapter.

Table 9-1 Overview of project phases and estimated efforts and time period

Phase	Scope	Efforts [man-years]	Time [years]
1	Base model compilation & First assessment	1-2	1
2	Completion of process representation & Refinement of disposal concept	2-3	1-2
3	Delivery of Rock Salt Reference Model & Development of initial Safety Case	2-3	1-2

9.1. Base model compilation & First assessment

The objective of the first phase is to provide a PA modelling platform that allows recalculating existing PA representations with new data input. This approach follows the Safety Case methodology, equivalent to the OPERA Safety Case for disposal in Boom Clay. The outcome can be used to identify the relevance of various processes. It also provides first evidence for safety. However, based on the discussion in the previous chapter (particularly Chapter 8.4), it must be emphasized that in this phase, the model will not cover a number of processes that may affect the long-term safety. That means that, although the outcome will provide experts valuable information, it has limited value in providing evidence for safety to a broader public.

This first phase could be completed in a period of about one year, with efforts estimated at about 1 to 2 man-years[qq].

Table 9-2 gives an overview of the most important elements necessary to build a Safety Case for a Dutch disposal concept in rock salt, following the structure of the previous chapters. The table also gives an estimate of the efforts necessary to realize each element. A more complete overview, addressing all aspects of the Safety Case and covering links to all recommendations of the previous chapter, is provided in Appendix 2.

Table 9-2 Elements of the Safety Case and estimated efforts

Component	Elements	Efforts
Safety Case Context	Contextual aspects, Types of waste (inventory)	none to minor
Safety Strategy	Management strategy, boundary conditions, Strategic choices, siting strategy, Approach to manage uncertainties, Safety concept, Demonstrability	minor - medium
System Description	Waste characteristics, Biosphere	none to minor
	Facility design, Engineered barriers, Salt host rock, Geosphere	minor to medium
	Safety relevant processes	medium to major
Safety Assessment	FEP database, Safety function 'Containment', 'Geosphere', and 'Biosphere',	none to minor
	Uncertainty, Safety & Performance Indicators (SPIN)	minor - medium
	Dissolution of waste, Safety function of barriers & host rock, Scenario definition	medium

[qq] This rather small effort is related to the generic, site-independent nature and post-closure safety focus of the initial Safety Case.

Relevant efforts in this phase are related to:

- the definition of safety functions for the host rock and engineered barrier system;
- the development of a first, generic disposal concept for all waste fractions covered in OPERA (Verhoef, 2011a);
- an analysis and update of safety-relevant processes and their current model representation, including model parameterisation with respect to radionuclide solubility, waste dissolution behaviour, gas formation;
- the development of rock-salt specific scenarios;
- the definition of rock-salt specific performance indicators; and
- an update of the FEP-list.

9.2. Completion of process representation & Refinement of disposal concept

The objective of the second phase is to evaluate processes that, although currently sufficiently well described, are not yet implemented in existing PA models used to analyse the behaviour of a repository system in rock salt. Process modelling studies have to be performed, and if necessary, the existing baseline model PA representations will be updated and adapted. The reference disposal concept will be refined based on the lessons learned, and evidence for the feasibility of the disposal concept will be elaborated based on geomechanical analyses and eventually THMC modelling. The resulting PA model should represent all processes that are assumed to be sufficiently well understood, and should allow to assess their relevance for safety.

This second phase could be completed in about 1 to 2 years, with efforts estimated at about 2-3 man-years[rr].

9.3. Delivery of Rock Salt Reference Model & Development of initial Safety Case

The objective of the third phase is to develop further understanding of less well understood and possible safety relevant processes - if necessary - and to deliver a fully integrated, up-to-date safety assessment model, on the basis of presently best available knowledge. Safety assessment calculations will be performed and an initial Safety Case will be prepared. Like for the OPERA Safety Case on Boom Clay, it is expected that the salt Safety Case will still lack experimental evidence for some processes, and will leave a limited number of well-defined, open questions including concrete advice on how to address these.

This third phase could be completed in about 1 to 2 years, with efforts estimated at about 2-3 man-years[rr].

9.4. Other activities

More general activities performed throughout all phases in support of the above include participation in international projects and working groups, and general development of Safety Case-related aspects.

[rr] Again, this rather small effort is related to the generic, site-independent nature and post-closure safety focus of the initial Safety Case, effectively limiting the ambitions.

Glossary

actinides	elements with an atomic number between 89 and 103
anhydrite	a very brittle material ($CaS0_4$) 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 organisms 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
ICM	‘Isolate, control, and monitor’ (in Dutch: IBC ‘isoleren, behersen, controleren’)
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

References

(Allison, 1991) Allison, JD, DS Brown, KJ Novo-Gradac, MINTEQA2/PRODEFA2, A Geochemical Assessment Model for Environmental Systems: Version 3.0 User’s Manual. US EPA, Athens, Georgia, 1991.

(Bechtold, 1999) Bechthold W, Rothfuchs T, Poley A, Ghoreychi M, Heusermann S, Gens A, Olivella S, Backfilling and Sealing of Underground Repositories for Radioactive Waste in Salt (BAMBUS Project). Final report, EUR 19124 EN, 1999.

(Becker, 2009) Becker D-A, Cormenzana J L, Delos A, Duro L, Grupa J, Hart J, Landa J, Marivoet J, Orzechowski J, Schröder T-J, Vokal A, Weber J, Weetjens E, Wolf J, *Safety Indicators and Performance Indicators, Performance Assessment Methodologies in Application to Guide the Development of the Safety Case (PAMINA),*EC , Deliverable D-N:3.4.2, 2009, 1-75.

(Becker, 2013) Becker D-A, 2013, Grupa JB, Wolf J, Methods for uncertainty analysis, OPERA-PU-GRS7321, 13 December 2013.

(Bérest, 2012) Bérest P, Ghoreychi M, Hadj-Hassen F, Tijani M (eds.): Mechanical Behavior of Salt VII. Proc. 7th Intern. Conf. , Paris, France, 16-19 April 2012. CRC Press/Balkema, London, 2012.

(Beuth, 2012a) Beuth, T.; Bracke, G.; Buhmann, D.; Dresbach, C.; Keller, S.; Krone, J.; Lommerzheim, A.; Mönig, J.; Mrugalla, S.; Rübel, A. & Wolf, J. (2012a): Szenarienentwicklung: Methodik und Anwendung. Bericht zum Arbeitspaket 8. Vorläufige Sicherheitsanalyse für den Standort Gorleben, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-284, ISBN 978-3-939355-60-1, Köln, 2012.

(Beuth, 2012b) Beuth, T.; Baltes, B.; Bollingerfehr, W.; Buhmann, D.; Charlier, F.; Filbert, W.; Fischer-Appelt, K.; Mönig, J.; Rübel, A. & Wolf, J. (2012b): Untersuchungen zum menschlichen Eindringen in ein Endlager. Bericht zum Arbeitspaket 11. Vorläufige Sicherheits-analyse für den Standort Gorleben, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-280, Köln, 2012.

(Bkse, 1969), Besluit van 4 september 1969, tot uitvoering van de artikelen 16, 17, 19, eerste lid, en 21 van de Kernenergiewet, Tekst valid on: 23-05-2014.

(BMU, 2010), Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit (BMU): Sicherheitsanforderungen an die Endlagerung wärmeentwickelnder radioaktiver Abfälle. Bonn, Stand: 30. September 2010.

(Bollingerfehr, 2013) Bollingerfehr W, Buhmann D, Filbert W, Keller S, Krone J, Lommerzheim A, Mönig J, Mrugalla S, Müller-Hoeppe N, Weber JR, Wolf J, Status of the safety concept and safety demonstration for an HLW repository in salt - Summary report, TEC-15-2013-AB, FKZ 02 E 10719 and 02 E10729, Peine, December 2013.

(Bradbury 2008) Bradbury, M, B Baeyens, Far Field Sorption Data Bases for Performance Assessment of a High-Level Radioactive Waste Repository in a Undisturbed Opalinus Clay Host Rock, PSI Bericht 03-08, Villingen, Switzerland, 2008.

(Brenner, 1999) Brenner J, Feddersen H-K, Gies H, Miehe R, Rothfuchs T, Storck R; Untersuchung von Altversatz als Analogon zur Konvergenz und Kompaktierung versetzter untertägiger Hohlräume im Salz über lange Zeiträume – Phase I. Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH. GRS-148. BMBF-FKZ 02E 8996-3. Köln, 1999.

(Bruggenwert 1982) Bruggenwert, MGM, A Kamphorst, Survey of experimental information on cation exchange in soil system, In: Bolt, GH (ed.): Soil Chemistry B. Physico-chemical models. Elsevier, Amsterdam, p.141-203, 1982.

(BS, 2001) Besluit Stralingsbescherming, Besluit van 16 juli 2001, houdende vaststelling van het Besluit stralingsbescherming, Ministerie van Sociale Zaken en Werkgelegenheid, Identificatienummer: BWBR0012702.

(Buhmann, 2005) Buhmann, Grupa JB, Hart, Poppei J, Resele G, Projekt Langzeitsicherheit Asse, Transportmodellierung - Fluid- und Radionuklidtransport am Standort Asse, ALSA-C-1.5B-NR145, Petten, 27 July 2005.

(Buhmann, 2008) Buhmann, D, Mönig J, Wolf J, Keller S, Mrugalla S, Weber JR, Krone J, Lommerzheim A, Nachweis und Bewertung des Isolationszustandes "Sicherer Einschluss". ISIBEL Projekt. Gemeinsamer Bericht von DBE TECHNOLOGY GmbH, BGR und GRS, Peine.

(Buhmann, 2010b), Buhmann D, Mönig J, Wolf J, Keller S, Mrugalla S, Weber JR, Krone, Lommerzheim L, Behandlung von Ungewissheiten im Langzeitsicherheitsnachweis für ein HAW-Endlager im Salzgestein. ISIBEL Projekt. Gemeinsamer Bericht von DBE TECHNOLOGY GmbH, BGR und GRS, Peine.

(Cadelli, 1996) Cadelli N, Escalier des Orres P, Marivoet J, Martens K-H, Prij J, Evaluation of elements responsible for the effective engaged dose rates associated with the final storage of radioactive waste: Everest project, Contract No FI2W/CT90/0017, European Commission, Nuclear Science and Technology, EUR 17122 EN, ISBN 92-827-8621-8, Brussels/Luxembourg, 1996.

(Caporuscio, 2013) Caporuscio FA, Boukhalfa H, Cheshire MC, Jordan AB, Ding M, Brine Migration Experimental Studies for Salt Repositories, FCRD-UFD-2013-000204, Los Alamos National Laboratory LA-UR-13-27240, 25 September 2013.

(CORA, 2001), Commissie Opberging Radioactief Afval, "Terugneembare berging, een begaanbaar pad? Onderzoek naar de mogelijkheden van terugneembare berging van radioactief afval in Nederland" Ministry of Economic Affairs, The Hague, February 2001.

(Cosenza, 1993) Cosenza Ph, Ghoreychi M: Coupling between mechanical behavior and transfer phenomena in salt. In Ghoreychi, M. (ed), 3rd Conference on Mechanical Behavior of Salt, pages 271–293. Trans Tech Publications, 1993.

(Davies, 2005) Davies C, Bernier F (eds): Impact of the excavation disturbed or damaged zone (EDZ) on the performance of radioactive waste geological repositories. Proc. EC Cluster conf. and workshop, Luxembourg, Nov. 3-5, 2003. EUR 21028 EN, Brussels, 2005.

(DBE-TEC, 2008) Review and Appraisal of the Tools available for a Safety Assessment of Final Repositories for HLW ISIBEL (Status of the Total Performance Assessment for a Repository for HLW in Rock Salt), Joint report by DBE TECHNOLOGY GmbH, BGR, and GRS, Peine, April 2008.

(DGV-TNO, 1987) Inventarisatie en analyse van geofysische methoden voor onderzoek van zoutvoorkomens. OPLA Fase 1, OPLA-13. DGV-TNO rapport OS-87-01, 110 p.

(Dodd, 2000) Dodd DH, Grupa JB, Houkema M, de Haas JBM, Van der Kaa Th, Veltkamp AC, Direct Disposal of Spent Fuel from Test and Research Reactors in the Netherlands - A Preliminary Investigation, 21406/00.30934/P, Petten, 31 January 2000, 1-90.

(EL&I, 2011) Ministry of Economic Affairs, Agriculture and Innovation, Ministry of Foreign Affairs, 2011, National Report of the Kingdom of the Netherlands Fourth review conference (May 2012), Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, Ministry of Economic Affairs, Agriculture and Innovation Ministry of Foreign Affairs The Hague, September 2011, 1-148.

(ENC, 2014) Britannica Online for Kids. Web. <http://kids.britannica.com/comptons/art-53934>; 27 Nov. 2014.

(EU, 2011) 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, Official Journal of the European Union L199, 2.8.2011, 48-56.

(Fahland, 2013) Fahland S, Reports „Salzmechanik", Index of contents "Salzmechanik I" to "Salzmechanik XVIII", BGR, January 2013.

(Fischer-Appelt, 2013) Fischer-Appelt K, Baltes B, Buhmann D, Larue J, Mönig J, Synthesebericht für die VSG. Bericht zum Arbeitspaket 13, Vorläufige

Sicherheitsanalyse für den Standort Gorleben, GRS-290, ISBN 978-3-939355-66-3, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln, 2013.

(Freeze, 2014) Freeze G, Sevougian SD, Leigh C, Gross M, Wolf J, Mönig J, Buhmann D, Development of a Salt Repository FEP Catalogue – Preamble, March 2014.

(Geluk, 1998) Geluk MC, Internal tectonics of salt structures. Journal of Seismic Exploration 7, 237-250, 1998.

(Grupa, 1999) Grupa, 1999, The PROSA methodology for scenario development, Paper 3 of Scenario Development Methods and Practice, An evaluation based on the NEA Workshop on Scenario Development, Madrid, May 1999.

(Grupa, 2000) Grupa JB, Houkema M, Terughaalbare opberging van radioactief afval in diepe zouten kleiformaties Modellen voor een veiligheidsstudie, 21082/00.33017/P, Petten, March 2000.

(Grupa, 2003) Grupa JB, Hart J, BAMBUS-II - WP 7.3. The Safety Implications of Retrievability, NRG 20553/02.48069/P, Petten, February 2003.

(Grupa, 2003b) Grupa JB, Task 4.1 - Comparison with International Recommendations Abschätzungen und Bewertungen zum Themenkomplex "Langzeitsicherheit" 20992/03.53495/C Petten, 18 June 2003.

(Grupa, 2014) Grupa JB, Report on the safety assessment methodology, OPERA-PU-NRG2121, 2 April 2014.

(Grupa, 2015a) Grupa JB, JCL Meeussen, E Rosca-Bocancea, AFB Wildenborg, D Buhmann, E Laggiard, Report on migration model in rock formations surrounding the host rock, OPERA report OPERA-PU-GRS7222, 30 June 2015.

(Grupa, 2015b) Grupa JB, JCL Meeussen, E Rosca-Bocancea, L Sweeck, T Wildenborg, Report on migration and uptake of radionuclides in the biosphere, OPERA report OPERA-PU-NRG7232, to be submitted.

(Grupa 2015c) Grupa JB, J Hart, AFB Wildenborg, Final report on the description of relevant scenarios for the OPERA reference concept in Boom Clay, OPERA report OPERA-PU-NRG7111 - revision 1, to be submitted.

(Guillaumont, 2003) Guillaumont, R., Fanghänel, T., Fuger, J., Grenthe, I., Neck, V., Palmer, D. A., Rand, M. H., Update on the Chemical Thermodynamics of Uranium, Neptunium, Plutonium, Americium und Technetium, OECD/NEA Data Bank, Ed., vol. 5 of Chemical Thermodynamics, North Holland Elsevier Science Publishers B. V., Amsterdam, The Netherlands, (2003).

(Hamstra, 1981) Hamstra J. et al., The Dutch geological waste disposal project. EUR 7151 EN, 1981.

(Hansen, 1995) Hansen FD, Ahrens EH, Tidwell VC, Tillerson JR, Brodsky NS: Dynamic compaction of salt: Initial demonstration and performance testing. Rock Mechanics, Daemen & Schultz (eds); ISBN: 90 5410 552 6; Balkema, Rotterdam, 1995.

(Hansen, 2011) Hansen FD, Leigh CD,Salt Disposal of Heat-Generating Nuclear Waste, Sandia Report, SAND2011-0161, January 2011.

(Hansen, 2011) Hansen FD, Repository Performance Confirmation. - Sandia Report SAND2011-6277, Sandia National Laboratories, October 2011.

(Hansen, 2012) Hansen J, FP7-Fission-2012-1.1.1: Technical feasibility and long-term performance of repository components – full-scale demonstration of plugs and seals (DOPAS), Euratom Programme for Nuclear Research and Training, Reference number: 214477, 27 March 2012.

(Hansen, 2013) Hansen FD, 2013, Steininger W, Biurrun E, Proceedings of the 4th US-German workshop on salt repository research, design, and operation, FCRD-UFD-2014-000335, SAND 2013-10592P, December 2013.

(Harsveldt, 1980) Harsveldt HM, Salt resources in the Netherlands as surveyed mainly by AKZO. Fifth International Symposium on Salt (Hamburg), 11-16, 1980.

(Harsveldt, 1986) Harsveldt HM, The Netherlands. In: Dunning, F.W. & Evans, A.M. (eds.): Mineral Deposits of Europe, Vol. 3: Central Europe. The Institute of Mining and Metallurgy and The Mineralogical Society (London), 113-116, 1986.

(Hart, 2009) Hart J, Grupa J, Spiers CJ, Microphysical Modelling of EDZ Healing in Rock Salt, Proc. EC TIMODAZ-THERESA Intern. Conf., Luxembourg, Sept. 29 - Oct. 01, 2009. EUR 25527 EN, pp 197-207. Luxembourg, 2012.

(Hart, 2014a) Hart J, Davis P, D. Becker DA, Noseck U, Hendriksen L, Wildenborg AFB, Winegram M, Evaluation of current state-of-the art on Safety Case methodologies, OPERA-PU-NRG211A, January 2014.

(Hart, 2014b) Report on the determination of the inventory, OPERA-PU-NRG1112A, Petten, 21 February 2014.

(Hart, 2015) Hart J, Prij J, Vis G-J, Becker D-A, Wolf J, Noseck U, Buhmann D, Evaluation of current knowledge for building the Safety Case for salt based repositories, OPERA-PU-NRG221A, 15 July 2015.

(Hartmann, 2006) Hartmann RFP, Visualisierung der Dichte im quinären System mariner Evaporite, Studienarbeit, Institut für Informatik, Technische Universität Clausthal 12 April 2006.

(Hartog, 1988) den Hartog HW, Stralingsschade in NaCl, Eindrapportage RE03 over fase I van het OPLA onderzoek, OPLA 21, University of Groningen (1988).

(Hartog, 1999) den Hartog H W, Vainshtein D I, Dubinko V I, Turkin A A, Gann V V and Jacobs J 1999 Radiation damage in NaCl: retrievability, smart backfill materials, monitoring, CORA Research Project for the Dutch Ministry of Economic Affairs Final Report, December 1999.

(Haverkate, 2002) Haverkate BRW, Waste Management Strategy in The Netherlands, Part 2: Strategy Principles and Influencing Issues, 20881/02.50512/P, Petten, 22 November 2002.

(Heijboer, 1988) Heijboer RJ, de Haas JBM, van Dalen A, Benneker PBJM, Nuclide-inventaris, warmteproductie en gammastraling van kernsplijtingsafval, Eindrapportage Deelrapport 11, Petten, maart 1988.

(Heijdra, 1995) Heijdra JJ, Hamilton LFM, Prij J, Slagter W, *Transient convergence and compaction of crushed salt as incorporated in the computer code EMOS*. ECN-I--95-029, 1995.

(Heijdra, 1997) CORA 8, J.J. Heijdra and J. Prij. Concept ontwerp terughaalbare berging in steenzout. Eindrapport METRO 1, maart 1997.

(Hiemstra 1996) Hiemstra, T, WH van Riemsdijk, A surface structural approach to ion adsorption: The charge distribution (CD) model, J. Colloid Interface Sci. 1996, 179, p.488-508.

(Hiemstra 1999) Hiemstra, T, WH van Riemsdijk, Surface structural Ion adsorption Modeling of Competitive Binding of Oxyanions by Metal (Hydr)oxides, J. Colloid Interface Sci. 1999, 210, p.182-193.

(Houben, 2008) Houben ME, Spiers CJ, Peach CJ, Microcrack healing and sealing effects in excavation damaged rock salt: Theory, experiment and applications in relation to radioactive waste management. Report to NRG on work performed under EU 6th Framework Project THERESA (WP3), 1 November 2008.

(IAEA, 2004) Safety Assessment Methodologies for Near Surface Disposal Facilities (ISAM), Vol. I – Review and Enhancement of Safety Assessment Approaches and Tools; International Atomic Energy Agency, ISBN 92–0–104004–0, Vienna, Austria.

(IAEA, 2009) IAEA, Classification of Radioactive Waste, IAEA Safety Standards for protecting people and the environment, General Safety Guide No.GSG 1, Vienna 2009; p. 1-48.

(IAEA, 2012) The Safety Case and Safety Assessment for the Disposal of Radioactive Waste, Specific Safety Guide No. SSG-23, STI/PUB/1553, Vienna, September 2012.

(IAEA, 2012b), International Atomic Energy Agency *PRISM: The international project on practical illustration and use of the Safety Case concept in the management of near-surface disposal, Overview Report*, Draft Working Material, November 2012 (to be finalised in 2014).

(ICK, 1979) Interdepartementale Commissie Kernenergie (Interdepartmental Nuclear Energy Commission), Report on the feasibilities of radioactive waste disposal in salt formations in the Netherlands. Ministry of Economic Affairs. April 1979.

(IFCE, 1980) International Fuel Cycle Evaluation, Waste Management and Disposal Report by Working Group 7, IAEA Vienna 1980.

(Jing, 2010), Jing L, Gens A, Rothfuchs T, THERESA - Coupled Thermal-Hydrological-Mechanical-Chemical Processes for Application in Repository Safety Assessment, Final report, Contract Number: FP6-036458, December 2010.

(Kersting, 1999) Kersting, AB, DW Efurd, DL Finnegan, DJ Rokop, DK Smith, JL Thompson, 1999, Migration of plutonium in groundwater at the Nevada Test Site, Nature, 397, p.56-59.

(Kinniburgh, 1999) Kinniburgh, DG, WH van Riemsdijk, LK Koopal, M Borkovec, MF Benedetti, MJ Avena, Ion binding to natural organic matter: competition, heterogeneity, stoichiometry and thermodynamic consistency. Colloids Surf. A, 1999, 151, p.147-166.

(Köster, 1989) Köster HW, de Jong EJ, Lembrachts JF, de Vries WJ, Dosisberekeningen – Eindrapportage, Deelrapport 10, Petten, April 1989.

(Kröhn, 2009) Kröhn K-P, Stührenberg D, Herklotz M, Heemann U, Lerch C, Xie M: Restporosität und -permeabilität von kompaktierendem Salzgrus-Versatz. REPOPERM – Phase 1. Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH. GRS-254. ISBN 978-3-939355-29-8. FKZ 02E 10477 (BMWi), Köln, 2009.

(Kröhn, 2012) Kröhn K-P, Zhang CL, Wolf J, Stührenberg D, Jobmann M, von Borstel L, Lerch C: The compaction behaviour of salt backfill at low porosities. In: Mechanical Behavior of Salt VII – Berest, Ghoreychi, Hadj-Hassen & Tijani (eds); Taylor & Francis Group, ISBN 978-0-415-06122-9, pp. 161-168; London, 2012.

(Kuhlman, 2014) Kuhlman KL: Summary Results for Brine Migration Modeling Performed by LANL, LBNL, and SNL for the UFD Program. Sandia National Laborat¬ories, SAND2014-18217 R, Albuquerque, US. Sept. 2014.

(Larue, 2013) Larue, J, Baltes B, Fischer H, Frieling G, Kock I, Navarro, M, Seher H: Radiologische Konsequenzenanalyse. Bericht zum Arbeitspaket 10, Vorläufige Sicherheitsanalyse für den Standort Gorleben, GRS-289, ISBN 978-3-939355-65-6, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH: Köln, 2013.

(Li, 2012) Li X, Jing L, Blaser P (Eds.): Impact of thermo-hydro-mechanical-chemical (THMC) processes on the safety of underground radioactive waste repositories. Proc. EC TIMODAZ-THERESA Intern. Conf., Luxembourg, Sept. 29 - Oct. 01, 2009. EUR 25527 EN. Luxembourg 2012.

(Lux, 2007) Lux K-H, Eberth S: Fundamentals and first application of a new healing model for rock salt. Proc. Sixth Conf. (SaltMech6), Hannover, Germany, 22-25 May 2007; pp. 129-138. ISBN 978-0-415-44398-2. Taylor & Francis Group, London, 2007.

(MacKinnon, 2014) MacKinnon RJ, Salt Disposal Research, Development, and Demonstration (RD&D), 5th US/German Workshop on Salt Repository Research, Design and Operation Santa Fe, New Mexico, US September 7 – 11, 2014.

(Marivoet, 2008), Marivoet J, Beuth T, Alonso J, Becker D-A, Task reports for the first group of topics: Safety Functions Definition and Assessment of Scenarios Uncertainty Management and Uncertainty Analysis Safety Indicators and Performance/Function Indicators, Deliverable D1.1.1) of PAMINA - Performance Assessment Methodologies in Application to Guide the Development of the Safety Case (Contract Number: FP6-036404), March 2008.

(Marschall, 2008) Marschall, P, Cuss R, Wieczorek K, Popp T: State of the Art on Gas Transport in the Tunnel Nearfield / EDZ. EU project NF-PRO. Deliverable 4.4.1, 2008.

(Meeussen, 2003) Meeussen, JCL, 2003, ORCHESTRA: an object-oriented framework for implementing chemical equilibrium models. Environ. Sci. Technol. 37, p.1175–1182.

(Meeussen, 2014) Meeussen JCL, E Rosca-Bocancea, Determination of the inventory: part B matrix composition, OPERA-PU-NRG1112B, Petten, 15 December 2014.

(MinEZ, 1980) Memorandum on Energy Policy, part III, Lower House, 1979-1980 session, 15802, nrs 11, 12, 13. Ministry of Economic Affairs, 1980.

(MinEZ, 1984) Proposal for a Research Program into Geological Disposal of Radioactive Waste in the Netherland. OPLA Committee, Ministry of Economic Affairs, June 1984.

(MinEZ, 2008) Ministry of Economic Affairs. Energierapport 2008. Publication number 08 ET 14, The Hague, Juni 2008.

(MinEZ, 2013) Ministry of Economic Affairs, Convention on Nuclear Safety (CNS), National Report of the Kingdom of the Netherlands for the Sixth Review Meeting in April 2014, The Hague, July 2013.

(MinEZ, 2014) Ministry of Economic Affairs, Netherlands' NAcP - edition 2014, 2nd National Action Plan (NAcP) for the follow-up of post-Fukushima Dai ichi related activities, Report for the ENSREG-led Peer Review Meeting to be held in April 2015, December 2014.

(Mönig , 2012) Mönig J, Buhmann D, Rübel A, Wolf J, Baltes B, Fischer-Appelt K, Sicherheits- und Nachweiskonzept - Bericht zum Arbeitspaket 4 - Vorläufige Sicherheitsanalyse für den Standort Gorleben, GRS-277, June 2012.

(Moog, 2015) Moog HC, Bok F, Marquardt CM, Brendler V: Disposal of Nuclear Waste in Host Rock formations featuring high-saline solutions – Implementation of a Thermodynamic Reference Database (THEREDA), Appl. Geochem. (submitted), 2015.

(Müller-Lyda, 1999) Müller-Lyda I, Birthler H, Fein E: Ableitung von Permeabilitäts-Porositätsrelationen für Salzgrus. GRS-148, ISBN 3-931995-10-0, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH. BMWi-FKZ 02E 8855-2; Braunschweig, 1999.

(Müller-Hoeppe, 2012) Müller-Hoeppe N, Buhmann D, Czaikowski O, Engelhardt H-J, Herbert H-J, Lerch C, Linkamp M, Wieczorek K, Xie M: Integrität geotechnischer Barrieren. Teil 1: Vorbemessung. Vorläufige Sicherheitsanalyse für den Standort Gorleben. Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH. GRS-287. ISBN 978-3-939355-63-2. Köln, 2012.

(NDA, 2013) NDA, Nagra & NRG (eds.), MoDeRn Monitoring Reference Framework report, MoDeRn Deliverable D-1.2, November 2013.

(NEA, 2006) Organization for Economic Co-operation and Development - Nuclear Energy Agency (OECD-NEA), NEA International FEP Database : Version 2.1, November 2006.

(NEA, 2008) Organization for Economic Co-operation and Development - Nuclear Energy Agency (OECD-NEA), Safety Cases for Deep Geological Disposal of Radioactive Waste: Where Do We Stand? Symposium Proceedings, Paris, France, 23-25 January 2007, NEA Report No. 6319, ISBN 978-92-64-99050-0, OECD, 2008.

(NEA, 2008a) Organization for Economic Co-operation and Development - Nuclear Energy Agency (OECD-NEA),Moving forward with geological disposal of radioactive waste – A collective statement by the NEA Radioactive Waste Management Committee (RWMC), NEA Report No. 6433, ISBN 978-92-64-99057-9, OECD, 2008.

(NEA, 2012), Organization for Economic Co-operation and Development - Nuclear Energy Agency, Methods for Safety Assessment of Geological Disposal Facilities - for Radioactive Waste - Outcomes of the NEA MeSA Initiative, NEA No. 6923, ISBN 978-92-64-99190-3, OECD, Paris, 2012.

(NEA, 2013), Organization for Economic Co-operation and Development - Nuclear Energy Agency (OECD-NEA), The Nature and Purpose of the Post-closure Safety Cases for Geological Repositories, NEA/RWM/R(2013)1, NEA No. 78121, OECD, Paris, 2013.

(NEA, 2013b) OECD/NEA, Natural Analogues for Safety Cases of Repositories in Rock Salt, Salt Club Workhop Proceedings, 5-7 September 2013, Braunschweig, Germany, NEA/RWM/R(2013)10, 17-Mar-2014.

(NEA 2013c) Nuclear Energy Agency (NEA), Chemical Thermodynamics Series Volume 1-13a, OECD/Elsevier, 1992-2013.

(OPLA, 1989) Commissie Opberging te Land (OPLA): “Onderzoek naar geologische opberging van radioactief afval in Nederland. Eindrapportage Fase 1”. Ministerie van Economische Zaken, Den Haag, May 1989.

(Parkhurst 1999) Parkhurst, DL, CAJ Appelo, User’s Guide to PHREEQC (Version 2). US Geol. Surv. Water-Resour. Invest. Rep. 99-4259, 1999.

(Peach, 1991) Peach CJ, *Influence of deformation on the fluid transport properties of salt rocks*. Proefschrift Utrecht, 1991.

(Pfeifle, 1998) Pfeifle TW, Hurtado LD: “Permeability of Natural Rock Salt from the Waste Isolation Pilot Plant (WIPP) During Damage Evolution and Healing.” International Journal of Rock Mechanics and Mining Sciences and Geomechanics Abstracts 35(4): 637–38, 1998.

(Pitzer 1984) Pitzer KS, JC Peiper, RH Busey, Thermodynamic Properties of Aqueous Sodium Solutions, J. Phys. Chem. Ref. Data, 13 (1), 1984, 1-102.

(Pitzer, 1991) Pitzer KS, Ion interaction approach: Theory and data correlation, in: Activity coefficients in electrolyte solutions, 2nd Edition, (Pitzer, K.S., ed.) CRC Press, 75-153, 1991.

(Pluymakers, 2014) Pluymakers AMH, Spiers CJ Compaction creep of simulated anhydrite fault gouge by pressure solution: Theory vs. experiments and implications for fault sealing. In: Faulkner, D.R., Mariani, E., and Mecklenburgh, J. (eds.), Rock Deformation from Field, Experiments and Theory: A Volume in Honour of Ernie Rutter, Geological Society, London, Special Publications 409, doi:10.1144/SP409.6.

(Poley, 2000a) Poley AD, Concept ontwerp terughaalbare opslag radioactief afval in diepe boorgaten in steenzout. (Torad-B), CORA 11, 19 mei 2000.

(Poley, 2000b). Poley AD, BAMBUS project - Nederlandse bijdrage, 20038/00.30991/C, Petten, 17 May 2000.

(Prij, 1987) Prij J, Glasbergen P, and Römer JC, “Scenario’s en Analysemethode; VEOS Eindrapportage, deelrapport 2”. Petten, 1987.

(Prij, 1989) Prij J, Veiligheidsevaluatie van opbergconcepten in steenzout (VEOS), Eindrapportage Deelrapport 1, Samenvatting en evaluatie, ECN, Petten, January 1989.

(Prij, 1990) Prij J, “Safety evaluation of disposal concepts in rock salt”. In: Proceedings of the Symposium on Safety Assessment of Radioactive Waste Repositories, pp. 247-256, Paris 9-13 October 1989, OECD/NEA, 1990.

(Prij, 1991), Prij, J, van Dalen A, Roodbergen HA, Slagter W, van Weers AW, Zanstra DA, Glasbergen P, Köster HW, Lembrechts JF, Nijhof-Pan I, Slot AFM, “Safety evaluation of geological disposal concepts for low and medium-level wastes in rock–salt” (PACOMA project), EUR 13178 EN, 1991.

(Prij, 1991b) J. Prij. On the design of a radioactive waste repository. Proefschrift UT Twente, June 1991.

(Prij, 1993), Prij J, Blok JBM, Laheij GHM, van Rheenen W, Slagter W, Uffink GJM, Uijt de Haag P, Wildenborg AFB, Zanstra DA, PRObabilistic Safety Assessment, Final report, of ECN, RIVM and RGD in Phase 1A of the OPLA Programme, 1993.

(Prij, 1995) Prij J, Blok BM, Laheij GHM, Oostrom M, Polle AN, van Rheenen W, Slagter W, Uffrink GJM, Uijt de hag P, Wildenborg AFB, Zanstra DA. ECN contribution to EVEREST: Sensitivity of the release from a repository in a saltformation in the Netherlands, Petten May 1995.

(RCN, 1972) Werkgroep RCN. De verwerking en opslag van radioactief vast afval van kernenergiecentrales in Nederland. Atoomenergie, 14 (1972), 235-248.

(RGD, 1988) Rijks Geologische Dienst, Geologische inventarisatie en ontstaansgeschiedenis van zoutvoorkomens in noord- en oost-Nederland. OPLA Fase 1, OPLA-12. Rijks Geologische Dienst (RGD) rapport 10568, Haarlem, 260 p.

(RGD, 1993) Rijks Geologische Dienst, 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 OPLA, Fase 1A. RGD rapport 30.012/ER, Ministerie van Economische Zaken, 116 p.

(Rosca-Bocancea, 2013) Rosca-Bocancea E, Schröder TJ, Development of Safety and Performance Indicators, OPERA-CF-NRG010, 29 October 2013.

(Rosca-Bocancea, 2014) Rosca-Bocancea E, JCL Meeussen, Modelling of radionuclide migration in the rock formations surrounding the host rock, OPERA Interim Report OPERA-IR-NRG623, 12 December 2014.

(Rothfuchs, 2003) Rothfuchs T, Dittrich J, Droste J, Müller J, Zhang C-L: Final Evaluation of the Project "Thermal Simulation of Drift Emplacement" (TSDE-Project). Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH. GRS-194. ISBN 3-931995-62-3; BMWi-FKZ 02E 9269. Köln, 2003.

(Rothfuchs, 2009) Rothfuchs T, Schulze O, Wieczorek K, Buhmann D, Müller-Hoeppe N: Summary of Salt Group Discussions. Conference on the Impact of Thermo-Hydro-Mechanical-Chemical (THMC) Processes on the Safety of Underground Radioactive Waste Repositories. Luxembourg, September 29 – October 1, 2009.

(Saltelli, 2000) Saltelli, A. (ed.): Sensitivity Analysis. Wiley, 2000.

(Schelland, 2014) Schelland M, Hart J, Wildenborg AFB, Grupa JB, OPERA FEP-database, OPERA-PU-TNO2123A, June 2014.

(Schneider, 2012) Schneider A (ed.): Enhancement of the Codes d3f and r3t. Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH. ISBN 978-3-939355-68-7. GRS-292, BMWi-FKZ 02E 10336, Köln, 2012.

(Schröder, 2008) Schröder TJ, EMOS versie ccm2 voor PAMINA WP2.1D: aanpassingen van de porositeit-permeabiliteit relatie en de convergentiesnelheid in EMOS ccm1 2.1952.21/08.89397, Petten, 4 June 2008.

(Schröder, 2009a) Schröder TJ, Hart J, Costescu-Badea A, Bolado Lavin R, PAMINA Task 2.1.D, Techniques for Sensitivity and Uncertainty Analysis - Analysis of a repository design in a rock salt formation, PAMINA Milestone Report M2.1D.7, NRG-21952/09.95945, December 2009.

(Schröder, 2009b) Schröder TJ, Rosca-Bocancea E, Hart J, *PAMINA. Calculation results of a repository design in argillaceous rock,* PAMINA Milestone Report M3.4.21, 2009.

(Schröder, 2013) Schröder TJ, E Rosca-Bocancea, Safety and performance indicator calculation methodology, OPERA report OPERA-PU-NRG7312, December 2013.

(Schröder, 2015) Schröder TJ, BRW Haverkate, AFB Wildenborg, Topic report on retrievability, staged closure and monitoring, OPERA report OPERA-IR-NRG123, to be submitted.

(Schulze, 2007) Schulze O, Heemann U, Zetsche F, Hampel A, Pudewills A, Günther R-M, Minkley W, Salzer K, Hou Z, Wolters R, Rokahr R, Zapf D: Comparison of advanced constitutive models for the mechanical behavior of rock salt - results from a joint research project. I. Modeling of deformation processes and benchmark calculations. In: M. Wallner, K.-H. Lux, W. Minkley & H.R. Hardy, Jr. (eds.). The Mechanical Behavior of Salt. Understanding of THMC Processes in Salt (SaltMech6), pp 77-88, ISBN 978-0-415-44398-2, Taylor and Francis Group, London, 2007.

(Sevougian, 2012) Sevougian DS, Geoff A. Freeze, Michael B. Gross, Joon Lee, Christi D. Leigh, Paul Mariner, Robert J. MacKinnon, Palmer Vaughn, TSPA Model Development and Sensitivity Analysis of Processes Affecting Performance of a Salt Repository for Disposal of Heat-Generating Nuclear Waste, FCRD-UFD-2012-000320, Rev. 0, September 28, 2012.

(Slagter, 1988) Slagter, W, Roodbergen HA, Analyses van het radionuclidentransport in steenzoutformaties bij opberging van radioactief afval, Petten, September 1988.

(Smailos, 1992) Smailos, E., W. Schwarzkopf, R. Köster, K. H. Gruenthaler (1992): Advanced corrosion studies on selected packaging materials for disposal of HLW canisters in rock salt, in Corrosion Problems Related to Nuclear Waste Disposal: A Working Party Report, Eu-ropean Federation of Corrosion, Ed.: The Institute of Materials: 23-31, 1992.

(Smailos, 1993) Smailos E, Corrosion of high-level waste packaging materials in disposal relevant brines, Nuclear Technology, vol. 104: 343-350, 1993.

(Spiers, 1988) Spiers CJ, *Long-term rheological and transport properties of dry and wet salt rocks*. EUR-11848, 1988.

(Storck, 1990) Storck R, D Buhmann, RP Hirsekorn, A Nies, H Rausch, EMOS: Programmpaket zur Langzeitsicherheitsanalyse eines Endlagers für radioaktive Abfälle - Version 4. GSF-Bericht 90, September, 1990.

(Sweek, 2014) Sweeck L, Preliminary list of reference dose conversion factors, OPERA report OPERA-PU-SCK632, 14 August 2014.

(SZW, 2001) Besluit van 16 juli 2001, houdende vaststelling van het Besluit stralingsbescherming, Staatsblad 2001, 397, 6 September 2001, 1-260.

(Tipping, 2002) Tipping E, Cation binding by humic substances, Cambridge University Press, Cambridge, United Kingdom, 2002, 1-x.

(TNO, 1993) Geophysical methods for the hydrogeological reconnaissance of salt domes and the adjacent sediments. OPLA Phase 1A, TNO report OS-93-15-B, 299 p.

(Tonkin, 2004) Tonkin, JW, LS Balistieri, JW Murray, Modeling sorption of divalent metal cations on hydrous manganese oxide using the diffuse double layer model. Appl. Geochem. 2004, 19, p.29-53.

(US DOE, 1996) United States Department of Energy, Title 40 CFR 191 Subparts B & C Compliance Certification Application, 1996.

(US DOE, 2003) US DOE, Step-By-Step Guide for Waste Handling at WIPP. Fact Sheet. U.S. Department of Energy. Carlsbad Field Office. Office of Public Affairs Carlsbad, NM (http://www.wipp.energy.gov/fctshts/wastehandling.pdf). Revised January, 2003.

(US DOE, 2004) United States Department of Energy, Title 40 CFR 191 Subparts B & C Compliance Recertification Application. DOE/WIPP 2004-3231. U.S. Department of Energy, March 2004.

(US DOE, 2011) US DOE, Waste Isolation Pilot Plant. Documented Safety Analysis. DOE/WIPP 07-3372, Rev. 3. Prepared by Washington TRU Sollutions, LLC Carlsbad, NM. U.S, department of Energy Contract DE-AC29-01AL66444 for U.S. Department of Energy Carlsbad Field Office Carlsbad, NM. May 2011.

(US DOE, 2014) Title 40 CFR Part 191 Subparts B and C, Compliance Recertification Application 2014 for the Waste Isolation Pilot Plant, Appendix SCR-2014, Feature, Event, and Process Screening for PA, DOE/WIPP-14-3503. Carlsbad, NM: Carlsbad Field Office.

(US DOE, 2014b) Title 40 CFR Part 191 Subparts B and C, Compliance Recertification Application 2014 for the Waste Isolation Pilot Plant, Results of Performance Assessments (40 CFR § 194.34), Carlsbad, NM: Carlsbad Field Office.

(US DOE, 2014c) Title 40 CFR Part 191 Subparts B and C, Compliance Recertification Application 2014 for the Waste Isolation Pilot Plant, Appendix PA-2014, Performance Assessment, Carlsbad, NM: Carlsbad Field Office.

(US EPA, 1997) Compliance Application Review Documents for the Criteria for the Certification and Re-certification of the Waste Isolation Pilot Plant's Compliance with the 40 CFR PART 191 Disposal Regulations: Proposed Certification Decision. EPA 402-R-97-013. U.S. Environmental Protection Agency.Office of Radiation and Indoor Air Washington, D.C. 20460. October 1997.

(Utsunomiya, 2009) Utsunomiya, S, AB Kersting, RC Ewings, 2009, Groundwater Nanoparticles in the Far-Field at the Nevada Test Site: Mechanism for Radionuclide Transport, Environ. Sci. Technol. 43, p.1293-1298.

(Van Adrichem Boogaert, 1997) Van Adrichem Boogaert HA, Kouwe WFP, Zechstein Group. In: Stratigraphic Nomenclature of the Netherlands. (1993-1997), Retrieved 01-11-2013 from www.dinoloket.nl/Zechstein-group-ze.

(Van Gent, 2011) Van Gent H, Urai JL, De Keijzer M,. The internal geometry of salt structures e A first look using 3D seismic data from the Zechstein of the Netherlands. Journal of Structural Geology 33, 292-311, 2011.

(Vandenborre, 2008) Vandenborre, J, A Abdelouas, B Grambow, 2008, Discrepanties in Thorium oxide solubility values: a new experimental approach to improve understanding of oxide surface at solid/solution interface, Radiochim. Acta 96, p.515-520.

(Vate, 2001) Van de Vate L, Retrievable disposal of radioactive waste in the Netherlands, TNO-NITG – Information, December 2001.

(Verhoef, 2011a) E. Verhoef, E. Neeft, J. Grupa, and A. Poley. OPERA. Outline of a disposal concept in clay. OPERA report OPERA-PG-COV008, COVRA N.V., July 2011, 1-17.

(Verhoef, 2011b), Verhoef E, Schröder TJ, OPERA Research Plan, OPERA-PG-COV004, COVRA N.V., 21 June 2011.

(Verhoef, 2014) Verhoef E, Neeft E, Towards a safety strategy - Developing a long-term Dutch research programme into geological disposal of radioactive waste, OPERA-PG-COV014, 26 March 2014.

(VROM, 1984) Ministry of Housing, Spatial Planning and the Environment (VROM), Radioactive waste policy in The Netherlands; An outline of the Government's position, September 1984.

(VROM, 1993) Opbergen van afval in de diepe ondergrond, Kamerstukken II, 1992-1993, 23163, nr. 1, 1-9.

(Wallner, 2007) Wallner M, Lux K-H, Minkley W, Hardy HR jr (eds.): The Mechanical Behavior of Salt. Understanding of THMC Processes in Salt. Proc. Sixth Conf. (SaltMech6), Hannover, Germany, 22-25 May 2007. ISBN 978-0-415-44398-2. Taylor & Francis Group, London, 2007.

(White, 2013) White MJ (Ed.), Monitoring During the Staged Implementation of Geological Disposal: The MoDeRn Project Synthesis MoDeRn Deliverable D-6.1, 24 October 2014.

(Wieczorek, 2013) Wieczorek K, Masik S, Raab M, Behlau J, Jobmann M: The virtual underground research laboratory. Radwaste management – Repository. Nuclear Engineering International. www.neimagazine.de. July 2013.

(WM, 1979), Dutch Government, Environmental Protection Act, “Wet Milieubeheer”, Wm, Law identification BWBR0003245, Stb. 1979, No. 442.

(WOB, 1991), Wet van 31 oktober 1991, houdende regelen betreffende de openbaarheid van bestuur, Id. BWBR0005252, text in force on 20 January 2014.

(Wolf, 2012a) Wolf J, Behlau J, Beuth T, Bracke G, Bube C, Buhmann D, Dresbach C, Hammer J, Keller S, Kienzler B, Klinge H, Krone J, Lommerzheim A, Metz V, Mönig J, Mrugalla S, Popp T, Rübel A, Weber JR, FEP-Katalog für die VSG. Konzept und Aufbau. Bericht zum Arbeits-paket 7, Vorläufige Sicherheitsanalyse für den Standort Gorleben, GRS-282, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH: Cologne, 2012.

(Wolf, 2012b) Wolf J, Behlau J, Beuth T, Bracke G, Bube C, Buhmann D, Dresbach C, Hammer J, Keller S, Kienzler B, Klinge H, Krone J, Lommerzheim A, Metz V, Mönig J, Mrugalla S, Popp T, Rübel A, Weber JR, FEP-Katalog für die VSG. Dokumentation. Bericht zum Arbeitspaket 7, Vorläufige Sicherheitsanalyse für den Standort Gorleben, GRS-283, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH: Cologne, 2012.

(Zhang, 2006) Zhang X, and Grupa JB, NF-Pro deliverable 3.5.7 – Compaction behaviour and permeability of low porosity compacted salt grit (dry and wet). NRG Report 21146/06.77412/P, December 2006.

Appendix 1 Salt formations in the Netherlands

The Zechstein Group consists of various rock types, including clay, carbonate, anhydrite and rock salt (Figure A-1). 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 A-2;
- Depth base Zechstein Group in the Netherlands, Figure A-3;
- Thickness Zechstein Group in the Netherlands, Figure A-4;
- Salt domes Zechstein Group, Figure A-5;
- Salt extraction licences in the Netherlands, Figure A-6;
- Zechstein top between 500-1500 m, Figure A-7.

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 A-5) 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 A-6 shows the three areas with salt extraction licences in the Netherlands, with superimposed on that the salt domes from Figure A-5. From these locations more information on the composition and structure of the salt may be obtained, resulting from the mining activities.

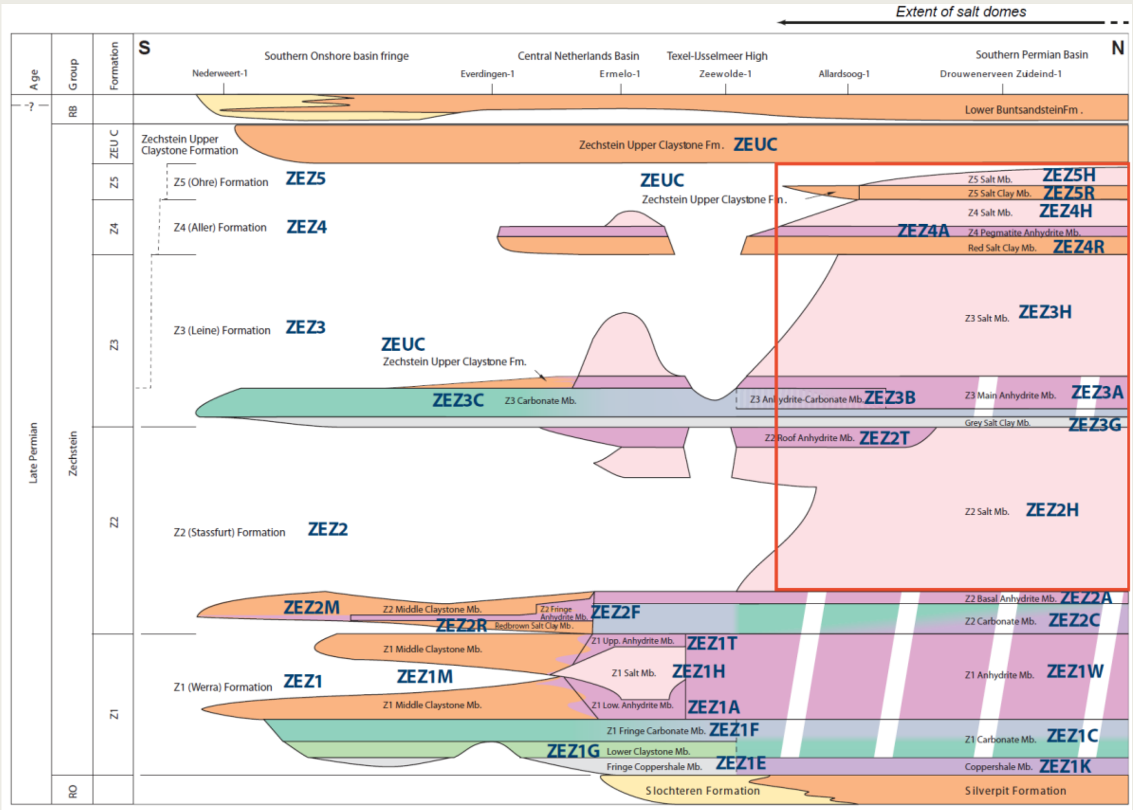


Figure A-1 **Lithostratigraphy of the Zechstein Group in the Netherlands (Van Adrichem Boogaert and Kouwe, 1993-1997).**

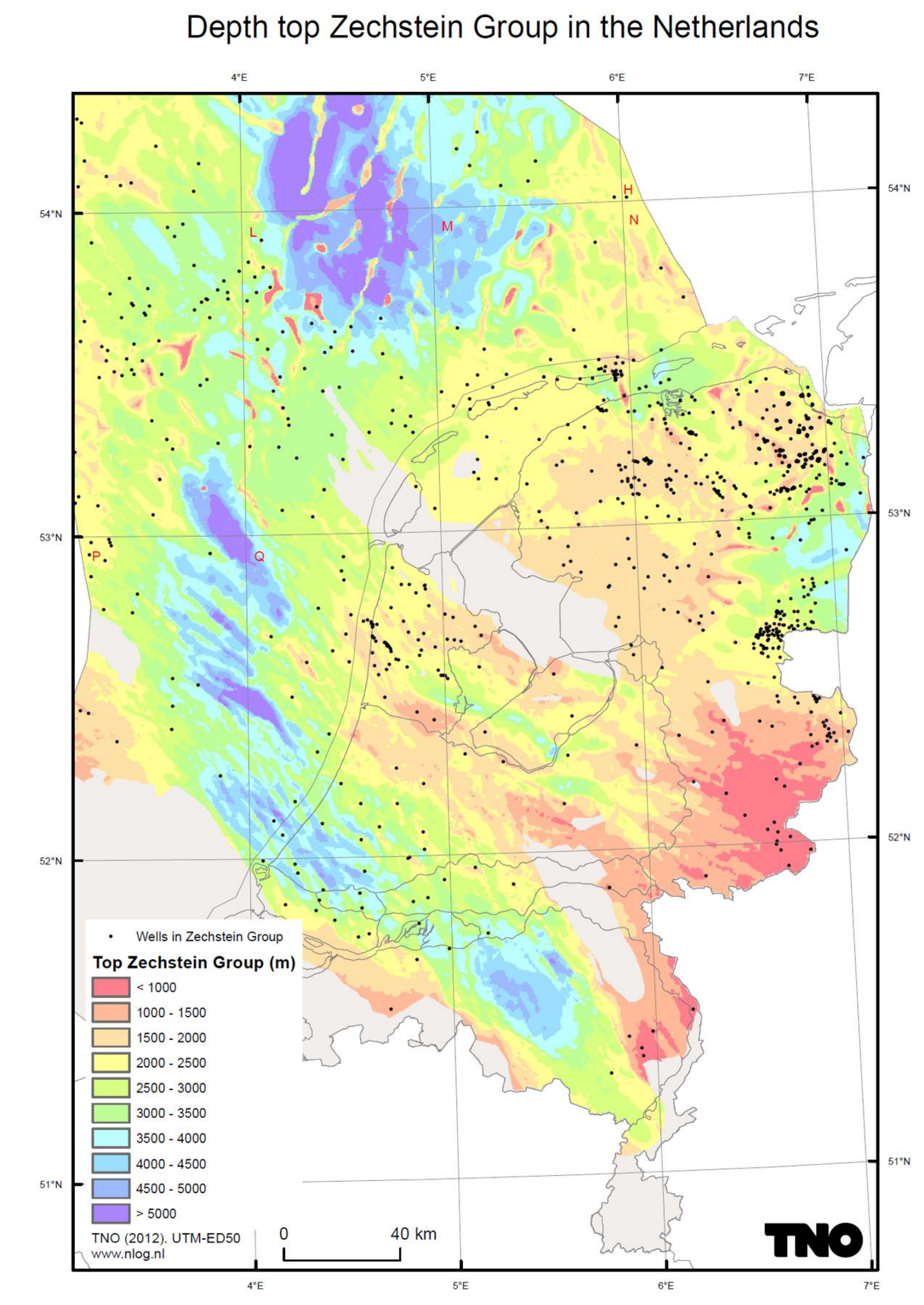


Figure A-2 Depth top Zechstein Group in the Netherlands (2013).

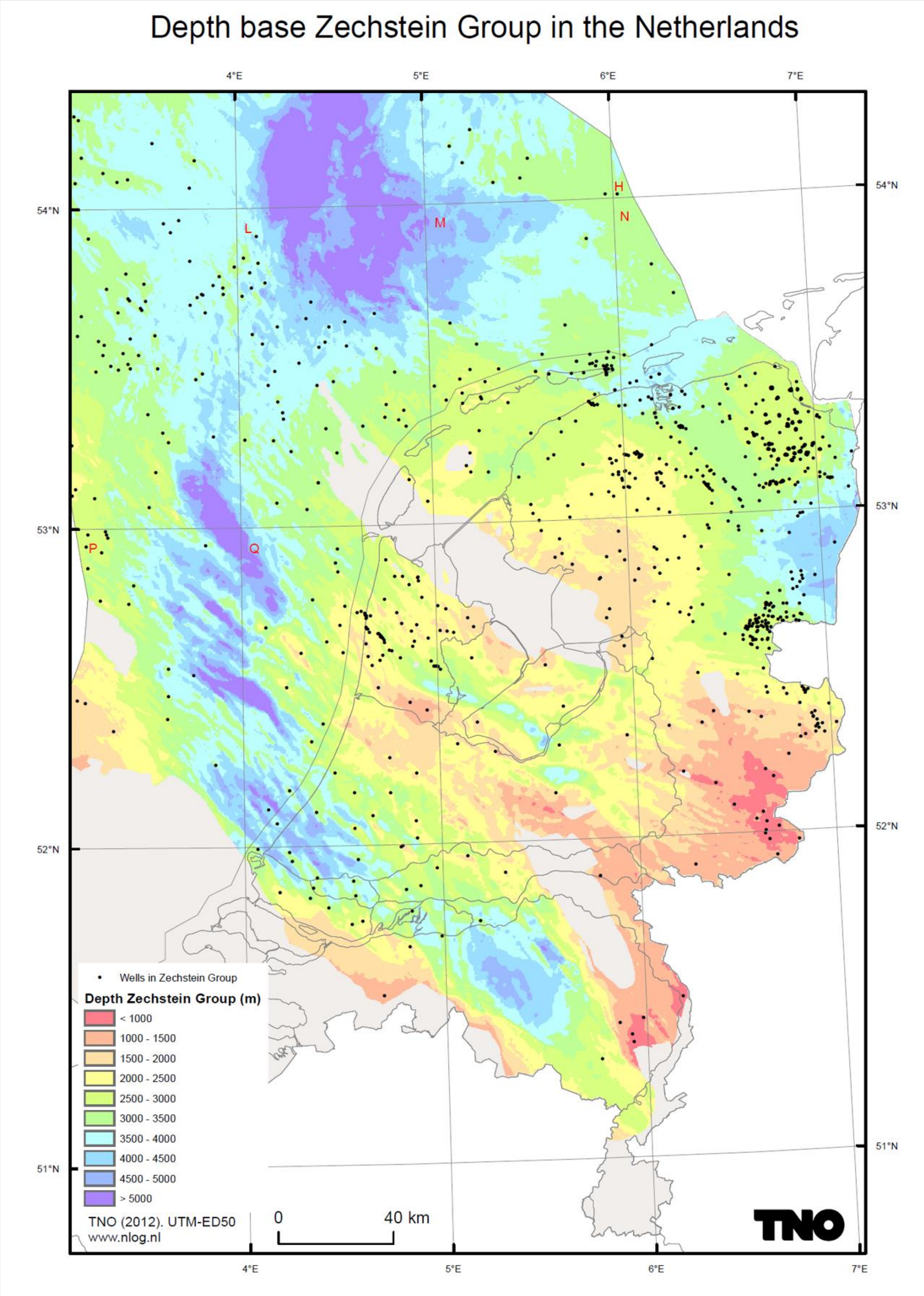


Figure A-3 Depth base Zechstein Group in the Netherlands (2013).

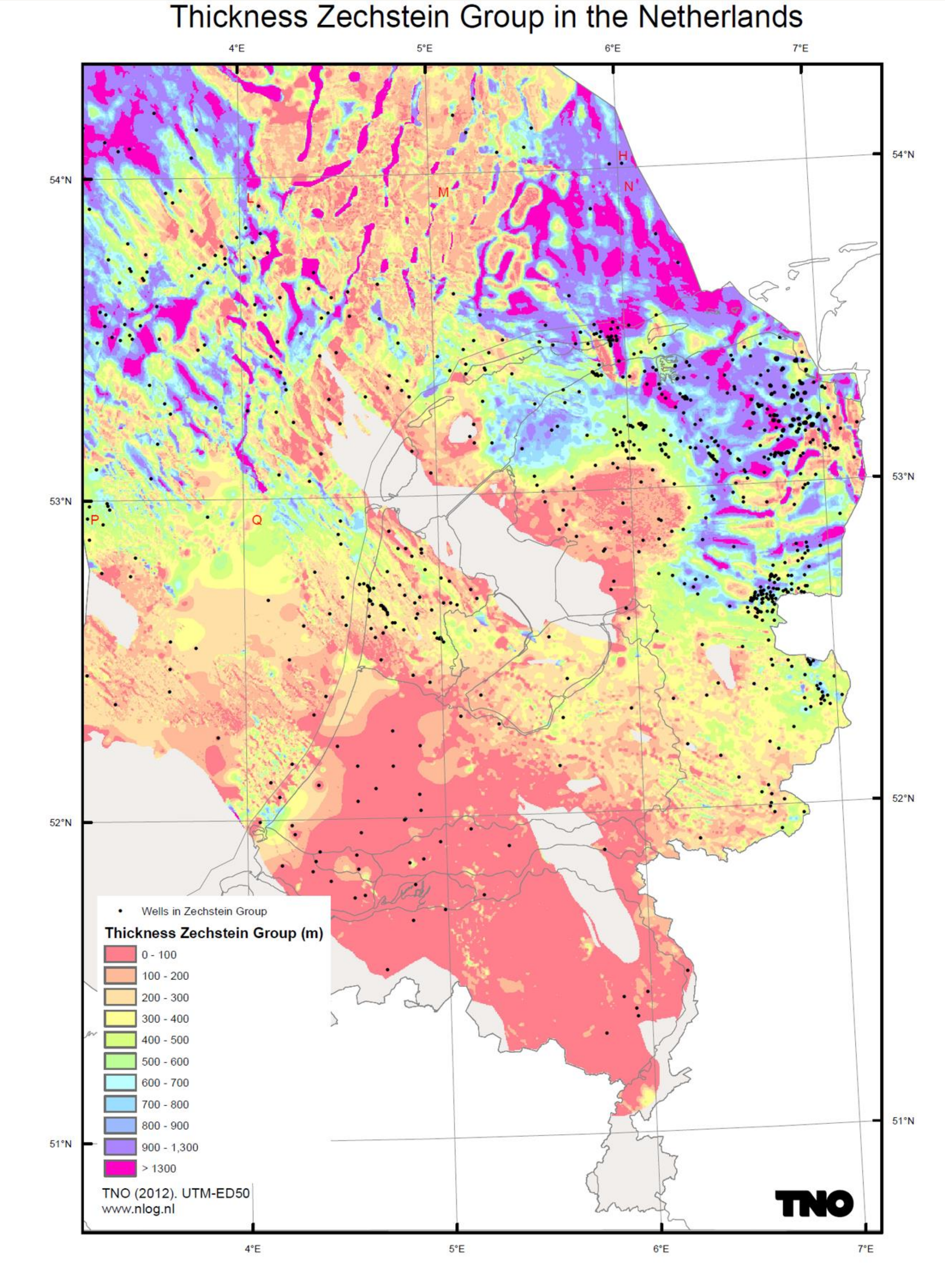


Figure A-4 **Thickness Zechstein Group in the Netherlands (2013).**

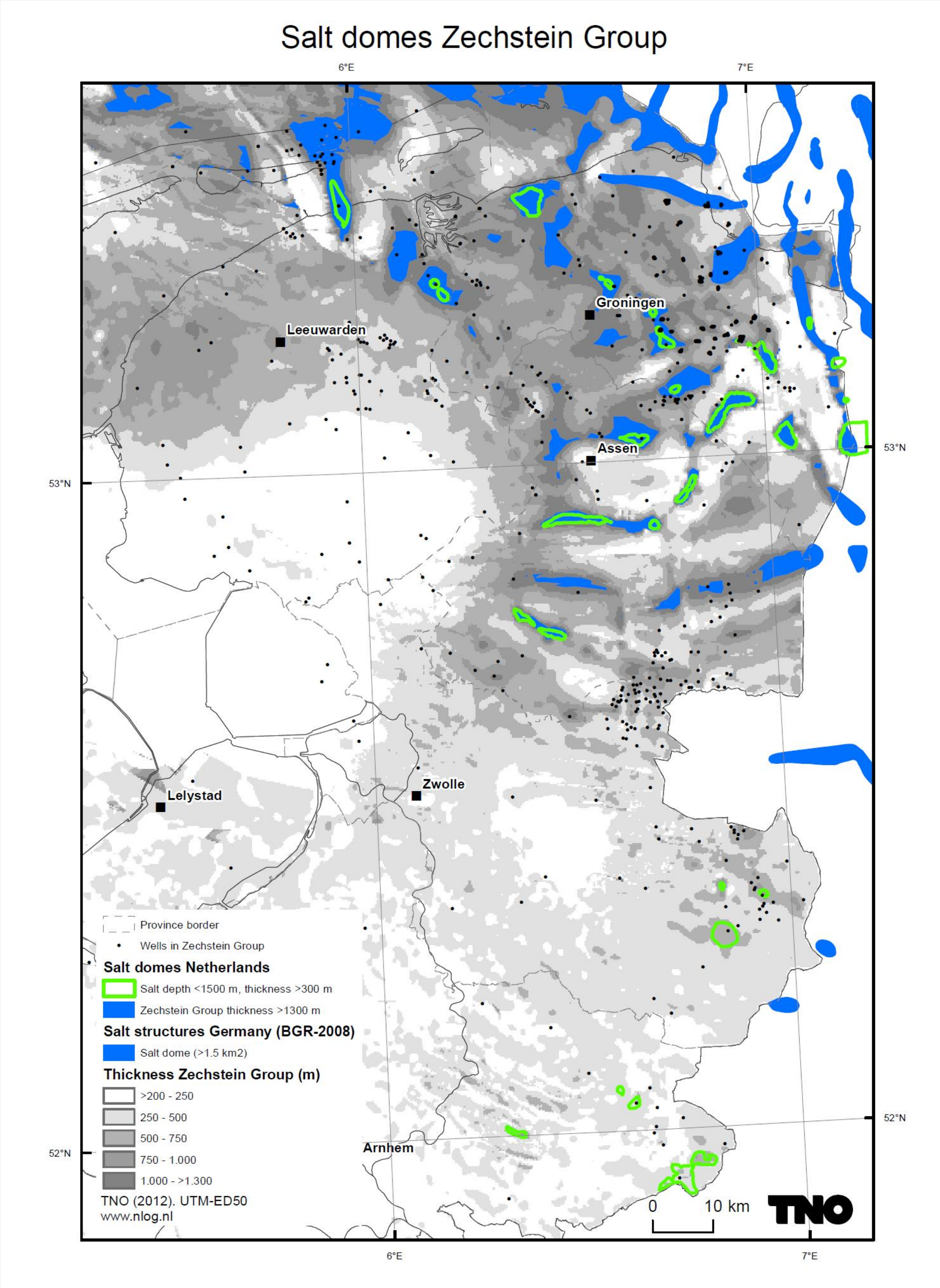


Figure A-5 **Salt domes Zechstein Group.**

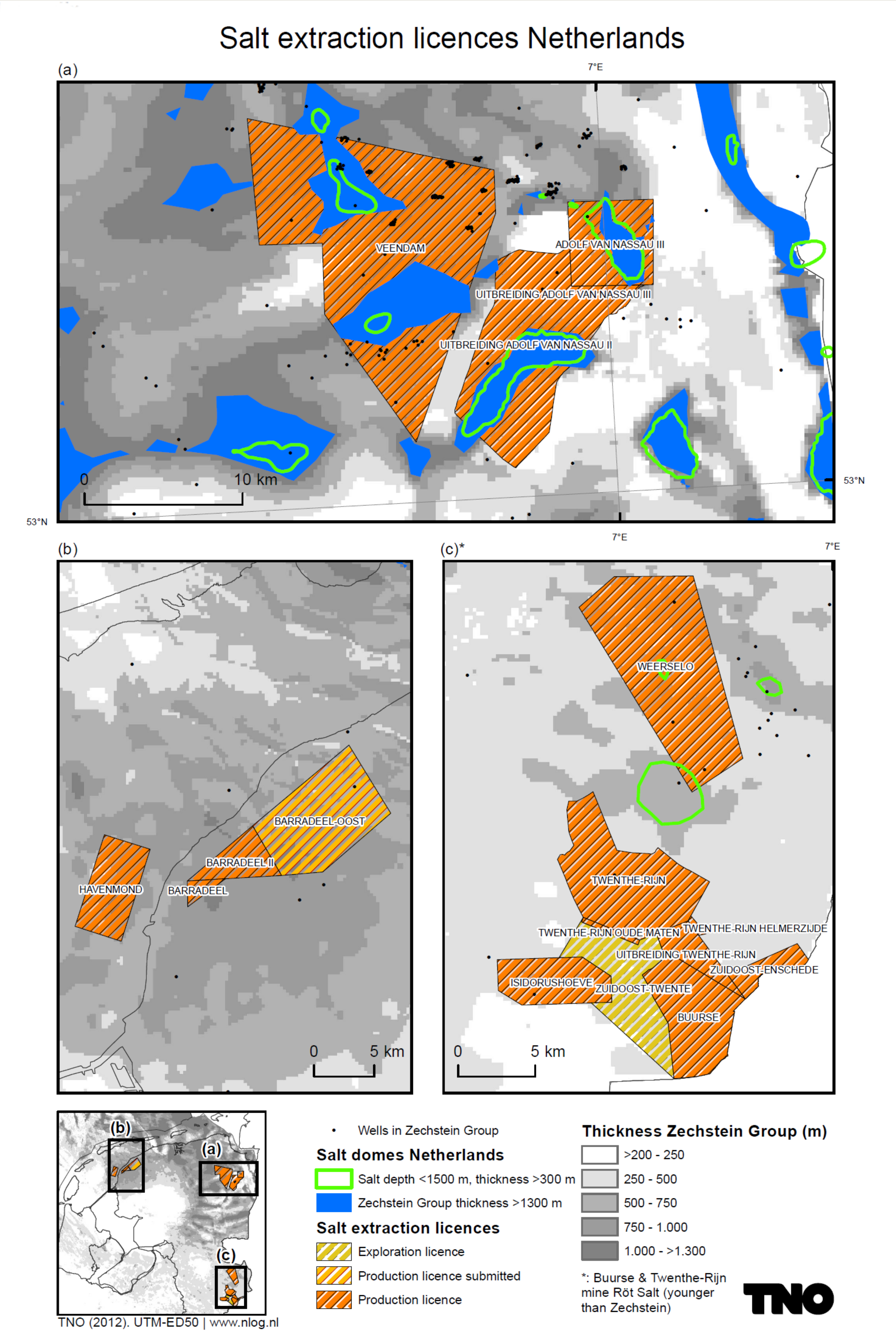


Figure A-6 Salt extraction licences in the Netherlands.

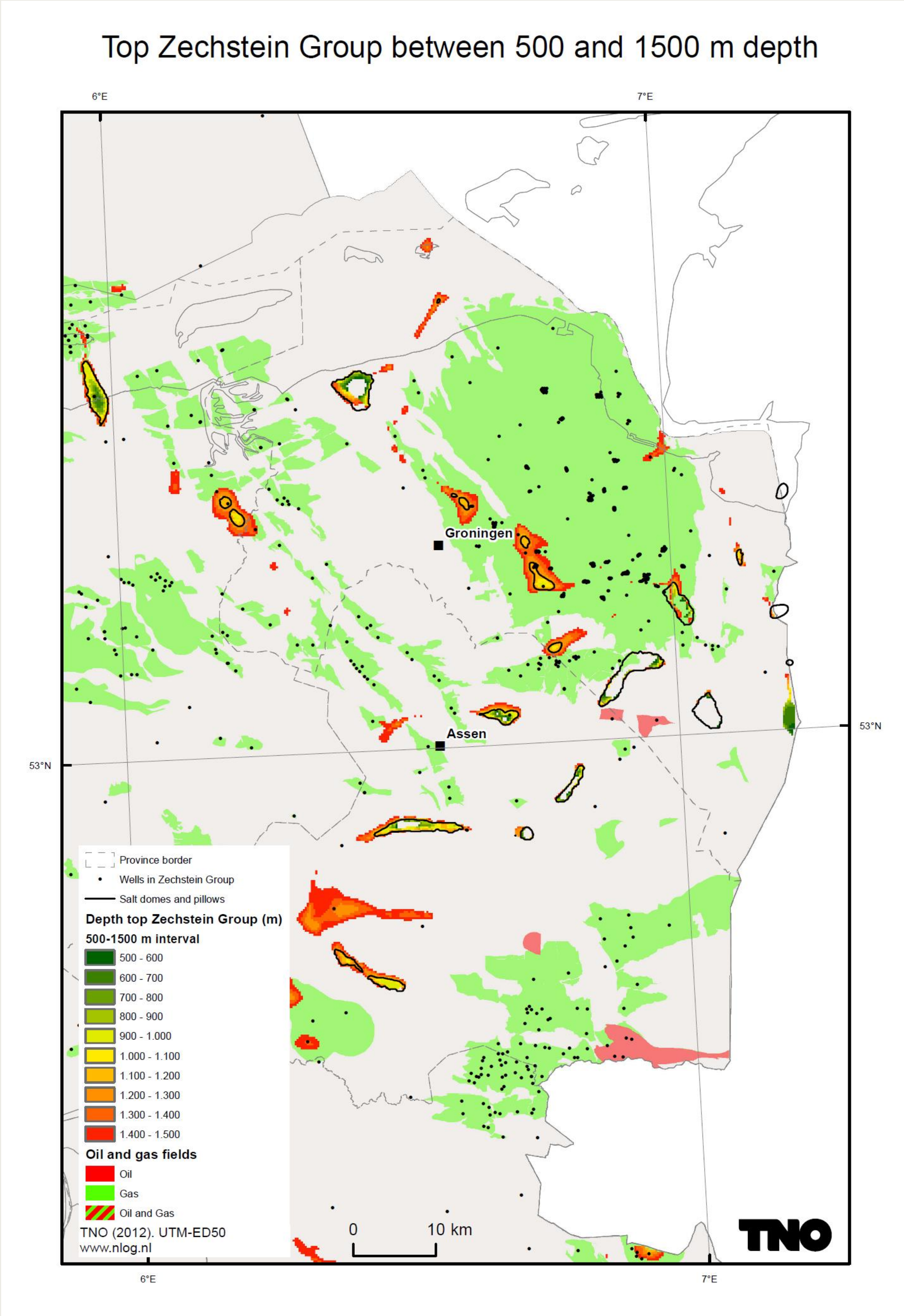


Figure A-7 **Zechstein top between 500 - 1500 m.**

Appendix 2 Elements of the Safety Case on rock salt

In this appendix, a more detailed elaboration of the roadmap in Chapter 9 is given, following the components of the Safety Case as outlined in Chapter 3 to 7, and integrating the recommendations of Chapter 8.

Phase 1: Base model compilation & First assessment

The objective of the first phase is to provide a PA modelling platform that allows to recalculate existing PA representations with new data input. The approach follows the Safety Case methodology, equivalent to the OPERA Safety Case for disposal in Boom Clay. The outcome can be used to identify the relevance of various processes. It also provides first evidence for safety. However, based on the discussion in Chapter 8 (particularly Chapter 8.4), it must be emphasized that in this phase the model will not cover a number of processes that may affect the long-term safety. That means that, although the outcome will provide experts valuable information, its value as published safety statement will still be limited.

This first phase could be completed in a period of about one year, with efforts estimated at about 1 to 2 man-years[ss].

Table B-1 to Table B-4 give for each component of the Safety Case an overview of the elements, the sources of information to be used, an indication of the efforts necessary to elaborate these, and - where applicable - a link to the specific recommendation of Chapter 8.

Table B-1 Elements of the *Safety Case Context*, the source of information and estimated efforts.

Element	Source	Efforts
Contextual aspects	based on several policy documents (e.g. VROM 1984 & 1993) and OPERA documents (e.g. Verhoef 2011a, 2011b & 2014)	minor
Types of waste to be disposed (Inventory)	OPERA disposal concept (Verhoef, 2011a), OPERA project OPCHAR Milestone reports (Hart, 2014b; Meeussen, 2014)	none

Table B-2 Elements of the *Safety Strategy*, the source of information and estimated efforts.

Element	Source	Efforts	Related recommendations
Management strategy, Boundary conditions, Strategic choices, Siting strategy, Approach to manage uncertainties	based on several policy documents (e.g. VROM 1984 & 1993) and OPERA documents (e.g. Becker, 2013; Verhoef 2011a, 2011b & 2014)	minor	8.4.6
Safety concept	Safety functions • waste matrix, waste container, geosphere, biosphere: equivalent to related OPERA safety functions • host rock: to be developed on basis of past experience	minor	8.2.A 8.2.B 8.3.5.M 8.4.6
Demonstrability	based on experiments performed in the past	medium	8.4.6

ss This rather small effort is related to the generic, site-independent nature and post-closure safety focus of the initial Safety Case.

Table B-3 Elements of the *System Description*, the source of information, estimated efforts and related recommendations.

Element	Source	Efforts	Related recommendations	Comment
Facility design	to be developed on basis of past experience and (Verhoef, 2011a)	minor - medium	8.3.1 8.3.5.M 8.3.5.N	only as far as necessary to perform SA
Waste characteristics	OPCHAR Milestone reports (Hart, 2014b; Meeussen, 2014)	none	8.3.2	
Engineered barriers	to be update on basis of prior concepts	minor	8.3.5.M 8.3.5.N	
Salt host rock	to be developed on basis of past experience & updated maps provided in this report (Appendix 1)	minor - medium	-	
Safety relevant processes	description on basis of past experience	medium - major	8.3.5.O 8.4.6	limited to processes covered by existing models
Geosphere	on basis of OPERA project RAMROCK and OPAP-II (e.g. Roca-Bocancea, 2014; Grupa 2015a)	minor - medium	-	too conservative due to higher brine density
Biosphere	on basis of OPERA project OPAP-II (e.g. Sweek, 2014; Grupa, 2015b)	minor	-	

Table B-4 Elements of the *Safety Assessment*, the source of information and estimated efforts.

Element	Source	Efforts	Related recommendations	Comment
Dissolution of waste	OPERA PA model (ORCHESTRA, Meeussen 2003), estimation on basis of OPERA & existing knowledge	minor - medium	-	depends on the aim depth of argumentation
Containment	OPERA PA model (ORCHESTRA, Meeussen 2003), estimation on basis of OPERA & existing knowledge	minor	-	assuming minor relevance
Barriers & host rock	EMOS (Storck 1990; Heijdra, 1995; Schröder, 2008)	medium	8.4.5.A	or other code, if more suitable
Geosphere	OPERA project OPAP-II (Grupa, 2015a), ORCHESTRA (Meeussen, 2003)	minor	-	
Biosphere	OPERA project OPAP-II (Grupa 2015b) ORCHESTRA (Meeussen 2003)	minor	-	
Scenario definition	on basis of OPERA (e.g. Grupa 2015c) & PROSA (Prij, 1993)	medium	8.4.2.B 8.4.6	
FEP database	PROSA (Prij, 1993, Ch. 2 - 4), German FEP-list (Wolf, 2012a & 2012b), OPERA FEP-list (Schelland, 2014)	minor	8.4.1.A 8.4.2.A 8.4.6	not going too far in this stage
Uncertainty	OPERA PA model (ORCHESTRA, Meeussen 2003), PAMINA results (e.g. Schröder, 2009a)	minor - medium	8.4.4 8.4.6	partial integration at this stage
Safety, Performance Indicators (SPIN)	to be developed, on basis of OPERA SPINs (Rosca-Bocancea, 2013; Schröder, 2013) and PAMINA results (Becker 2009, Schröder, 2009b)	minor - medium	8.4.6 8.5.A 8.5.B 8.5.D	

Phase 2: Completion of process representation & Refinement of disposal concept

The objective of the second phase is to evaluate processes that, although currently sufficiently well described, are not yet implemented in the PA models used to analyse the behaviour of a repository system in rock salt. Process modelling studies have to be performed, and if necessary, the existing baseline model PA representations will be updated and adapted. The reference disposal concept will be refined based on the lessons learned, and evidence for the feasibility of the disposal concept will be elaborated based on geomechanical analyses and eventually THMC modelling. The resulting PA model should represent all processes that are assumed to be sufficiently well understood, and should allow to assess their relevance for safety.

This second phase could be completed in about 1 to 2 years, with efforts estimated at about 2-3 man-years[tt].

Specific actions based on the recommendations in Chapter 8 are:

- ***8.2.B:*** Re-evaluation of the initial disposal concept from a high-level, Safety Strategy point of view.
- ***8.3.1:*** Refining of initial disposal concept based on the lessons learned of the previous phase; refinement with respect to retrievability and feasibility aspects.
- ***8.3.5.A:*** Evaluate possible options for model representation of the DRZ. Perform first analyses of the relevance of DRZ to provide at the end of Phase 2 a recommendation on how to implement this feature in the final Reference Model.
- ***8.3.5.F:*** Inventory relevant mineral dissolution processes for a disposal concept situated in the Netherlands.
- ***8.3.5.K:*** Inventory relevant CSH corrosion processes in brine to be covered in Phase 3.
- ***8.3.5.P:*** Inventory modelling options to address non-equilibrium processes.
- ***8.3.5.Q:*** Integrate modelling options to address gas production and transport.
- ***8.4.4:*** Implement modelling options to fully address uncertainties, on basis of the outcome of Phase 1.
- ***8.6:*** Refining of initial disposal concept with respect to options to monitor the disposal facility in later stages, as part of the Dutch ICM-criteria and requirements on retrievability, based on the lessons learned from EU framework project Modern2020.

Phase 3: Delivery of Rock Salt Reference Model & Development of initial Safety Case

The objective of the third phase is to develop further understanding of less well understood and possible safety relevant processes - if necessary - and to deliver a fully integrated, up-to-date safety assessment model, on the basis of presently best available knowledge. Safety assessment calculations will be performed and an initial Safety Case will be prepared. Like for the OPERA Safety Case on Boom Clay, it is expected that the salt Safety Case will still lack experimental evidence for some processes, and will left over a limited number of well-defined, open questions including concrete advice on how to address these.

This third phase could be completed in about 1 to 2 years, with efforts estimated at about 2-3 man-years[tt].

[tt] Again, this rather small effort is related to the generic, site-independent nature and post-closure safety focus of the initial Safety Case, effectively limiting the ambitions.

Specific actions based on the recommendations in Chapter 8 are:

- ***8.3.5.A:*** Assessment of the impact of the DRZ on safety.
- ***8.3.5.B:*** Implementation of modelling tools that allow the analysis of the impact of DRZ on safety.
- ***8.3.5.C:*** Definition and implementation of enhanced models for the compaction of crushed, granular salt in the presence of brine.
- ***8.3.5.D:*** Definition and implementation of enhanced models for the representation of porosity development in (crushed) salt barriers.
- ***8.3.5.G:*** Critical evaluation of the necessity to perform additional experiments in support of the modelling of mineral dissolution behaviour for a disposal concept situated in the Netherlands.
- ***8.3.5.G:*** Critical evaluation of the necessity to perform additional experiments in support of the modelling of mineral dissolution behaviour for a disposal concept situated in the Netherlands.
- ***8.3.5.K:*** Implementing state-of-the art geochemical models for CSH corrosion in brine compositions expected for the Netherlands.
- ***8.3.5.P:*** If necessary, implement state-of-the art geochemical models for non-equilibrium reactions in brine compositions expected for the Netherlands.
- ***8.3.5.Q:*** Implement and parameterize models for gas generation and transport.
- ***8.4.3.A:*** Assess effects of elevated temperatures on parameter values and - if necessary - implement these in the safety assessment models.
- ***8.5.C:*** Assess the feasibility to apply for the current Safety Case the '*Radiologischer Geringfügigkeits-Index*' (RGI), and - if proved useful - apply the RGI in the safety assessments.

Other activities

More general activities performed throughout all phases in support of the above include participation in international projects and working groups, and general development of Safety Case-related aspects. Many of these are activities recommended on the long-term. Specific actions mentioned in the recommendations in Chapter 8 are:

- ***8.3.3:*** To keep up with the knowledge and safety aspects related to seals, dams and plugs, it is recommended to actively participate in international research initiatives on barrier systems, especially collaboration with the German and US programmes, as well as in the EU framework projects, like the FP7 project *DOPAS*.
- ***8.3.5.I:*** Actively pursue collaboration in the German *THEREDA* project.
- ***8.3.5.J:*** Keep track on computational developments for modelling the complex hydrological, (geo)mechanical and (geo)chemical processes in salt formations in their natural hydrogeological setting in 3D, e.g. by active participation in international working groups (NEA *Salt Club*, US-German Collaboration), and technical meetings on this matter (e.g. *ABC-Salt* meetings).
- ***8.4.1.B:*** Seek for active participation in the presently ongoing compilation of a comprehensive FEP catalogue for disposal of heat-generating waste in salt by German and US institutes in the framework of the OECD/NEA *Salt Club*.
- ***8.4.5.B:*** Join the international community for further code development and benchmarking with respect of constitutive modelling of THMC processes.

8.7: With respect to general synergies, it is recommended to actively participate in international programmes and projects, for example:

- The OECD/NEA *Integration Group for the Safety Case (IGSC)*, aiming to develop Safety Cases supported by scientific technical basis;
- The OECD/NEA *Salt Club*, aiming to develop and exchange scientific information on rock salt for hosting deep geological repositories for radioactive waste;
- IAEA-hosted coordinated research projects related to Safety Case development and radioactive waste disposal (currently e.g. *PRISMA*, *GEOSAF-II*, *HIDRA*);
- The US-German workshop on salt repository research, design, and operation, collaboration between US and German institutes and companies aiming to combine the technical basis for salt disposal;
- Collaborative projects partially funded by the European Commission, currently e.g. Modern2020 and DOPAS, and future relevant projects;
- *IGD-TP - Implementing Geological Disposal of Radioactive Waste Technology Platform*, a scientific and technical forum hosted by the European Commission to provide the necessary focus in the lead up to the operation of geological repositories for high-level nuclear waste in Europe.

Disclaimer

OPERA

Meer informatie:

Postadres
Postbus 202
4380 AE Vlissingen

T 0113-616 666
F 0113-616 650
E info@covra.nl

www.covra.nl