COPERA SALT 2024

A CONDITIONAL SAFETY CASE AND FEASIBILITY STUDY

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Foreword

The principal objective of this report is to present an overview of the results and conclusions of the on-going work in the Netherlands on developing safety cases for a Geological Disposal Facility

(GDF) in a Permo-Triassic salt formation. A major milestone in the Dutch disposal programme was reached in 2017 with the publication of an initial Clay Safety Case based on the R&D work completed in the OPERA research programme, which focussed on a GDF in Boom Clay (one of the Paleogene clays). The present report presents a first conditional Salt Safety Case, taking into account progress in the Netherlands and elsewhere.

As with the OPERA Safety Case, the present report is termed a 'conditional' safety case, as it is recognised that, for eventual implementation of a GDF, various parameters will need to be updated, especially to match site-specific conditions, evolution of the GDF design and the exact waste inventory at the time of implementation.

This Salt Safety Case is accompanied by a parallel milestone report on a GDF in a Paleogene Clay formation (i.e., a Clay Safety Case) since both geological options are being considered. Because our intent is to ensure that the reports are consistent and can be read as stand-alone documents, significant sections of text are common. In addition, as much of the information in both reports has changed little since the 2017 OPERA Safety Case, some text has been brought forward from that report, amended with updated information if necessary.

Because both these reports mark major milestones in the Dutch overall radioactive waste management programme, they cover a wider and somewhat different scope from safety cases in other disposal programmes that are closer to implementation. The principal objectives of the work described in both new COVRA reports are:

- To propose practical conceptual designs for a GDF and to examine their engineering feasibility;
- To assess the post-closure safety of a GDF based on these designs;
- To use the design information to provide a basis for estimating future costs and therefore to allow determination of the level of financial provisions to be made today by COVRA;
- To use the experience gained in producing the report to strengthen the national competences in scientific and technical areas related to geological disposal;
- To use the findings of the report to select and prioritise the R&D activities to be carried out in the Dutch disposal programme over the coming years;
- To inform decision-makers, the public and the scientific/technical community at large about the progress of geological disposal planning in the Netherlands.

The predecessor programme OPERA was financed by the Dutch Ministry of Economic Affairs and the public limited liability company Electriciteits-Produktiemaatschappij Zuid-Nederland (EPZ) and was coordinated by COVRA. The present on-going work is part of COVRA's OnderzoeksProgramma Voor Eindberging van Radioactief Afval (COPERA) work and is financed from the COVRA budget.

COVRA acknowledges all the researchers in Dutch and foreign research organisations that are contributing to COPERA. In line with current international practice, it was decided to structure COVRA's programme on geological disposal around the development of a series of safety cases for a GDF. However, the wider than usual range of objectives and the correspondingly wider target readership means that there are significant differences between COVRA's safety cases and GDF safety cases from other countries, which have often been prepared in order to meet some specific permitting or licensing requirements. The COPERA Safety Case is less comprehensive, being an early-stage report in a series of analyses that will be regularly updated and extended by further iterations as implementation comes closer.

This report focuses on salt as a host rock. The Netherlands has benefited greatly through the close cooperation with BGE and the US DOE, which manage the German and US waste disposal programmes. These organisations have carried out comprehensive investigations and have practical experience with salt as a host rock for many years. However, no decisions on possible locations for a GDF in the Netherlands will be taken for many years into the future and the next iterations of safety cases, whether in salt or in clay, are expected to continue to be generic and conditional in nature.

The present report extends beyond the scope normally used for a safety case for geological disposal of waste, in that:

- It contains additional material on some key engineering aspects of GDF implementation. This gives a firm basis for the safety assessments and allows early estimation of future costs;
- Emphasis is placed on embedding the safety case studies into the wider Requirements Management System (RMS) being developed to cover all of COVRA's radioactive waste management work;
- Additional information is included on the overall scope and structure of the R&D projects that underpin COPERA. Proposals for future scientific and technical studies leading eventually to implementation of a GDF are included at the end of the current report;
- As in the predecessor 2017 OPERA Safety Case, the wish to make the report accessible to a wide readership has required additional explanatory material to be included, to describe the basic concepts involved in geological disposal and to summarise current international consensus on the recognised approaches.

As with all its publications, COVRA welcomes any comment readers might have.

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Summary

The objective of this report is to present an overview of results and conclusions of the on-going work in the Netherlands on developing safety cases for a Geological Disposal Facility (GDF). COPERA is COVRA's ongoing long term research programme expected to run for decades and includes research for GDFs in poorly indurated clay, rock salt and multinational solutions. The COPERA programme and future work on geological disposal is being structured around the development of a series of Safety Cases for a GDF in the Netherlands. The research programme has a structure that can be used for several programming periods; each decade will result in an iteration of two safety cases, one for GDF in rock salt and another for a GDF in poorly indurated clay. The present report documents the latest safety case for a GDF in rock salt; it has been prepared in parallel with a second iteration of the safety case for a GDF in clay.

The national context of the geological disposal programme, the wider than usual range of objectives and the wide target readership, means that there are significant differences between the report presented here and recent national Safety Cases published in other countries. The COPERA Salt 2024 Conditional Safety Case & Feasibility study is, for example, less comprehensive, given that it is an initial analysis that will be followed by further iterations. On the other hand, the report is wider in scope than many other national Safety Cases. Explanatory material has been included, for example, to describe the basic concepts involved in geological disposal and to summarise the current international consensus on the recognized approaches to achieving safety and to structuring a technical Safety Case for a GDF. This is done to make the report accessible to a wide readership. In addition, proposals for future scientific and technical studies have been developed, using the information gathered during the preparation of the Safety Case. These are presented in a roadmap, laying out all COVRA's ongoing activities leading eventually to implementing a GDF in the Netherlands.

We are, however, fully aware that a successful GDF programme must address both societal and technical issues, as well as scientific and technical matters. Globally, the greatest obstacles to the geological disposal of waste have been related to gaining sufficient public and political support for the concept itself and, more specifically, for siting activities. The Rathenau Institute has explored a society-based approach to identifying potential siting areas and locations for a GDF.

What's new in COPERA

The structure of the COPERA project focuses on development of safety cases for rock salt and clay repositories. The present report documents an Initial, Conditional Safety Case for rock salt: this also gives a framework for future planning.

An updated disposal concept has been produced for the Geological Disposal Facility (GDF) – with an engineered barrier concept including a waste package specifically designed for the disposal of the most active wastes.

Developments in other countries considering deep disposal in rock salt have been fully integrated: in particular, there has been close cooperation with both the disposal programmes in Germany and the United States. Both these countries have repositories in rock salt.

COVRA is developing a Requirements Management System (RMS) that will structure all its activities from waste conditioning, through temporary waste storage to disposal operations, including ensuring that safety is provided after closure of the GDF. Further levels are defined, considering the need to be compatible with the parallel safety case in poorly indurated clay, and also with COVRA's waste storage programme.

The cost estimate for a GDF in rock salt has been updated based on demonstrated construction and emplacement techniques from both the disposal programmes in Germany and the United States.

Based on the results, priorities and specific goals have been developed for work in the next phase of the COPERA research programme.

Introduction

Nuclear technologies are used in electricity generation, medicine, industry, agriculture, research and education. These technologies generate radioactive wastes that must be managed in a way that always ensures safety and security. For materials that remain hazardous up to hundreds of thousands of years, the recognised approach to long-term isolation and confinement is disposal in a GDF constructed in a stable geological environment far beneath Earth's surface.

The Netherlands, along with other countries with significant quantities of long-lived radioactive wastes, has chosen geological disposal as the official national policy. The reference date for implementing a national GDF is around 2130, more than 100 years from now, although this might change. The extended timescales allow flexibility in case alternatives to disposal in a national GDF become available; one such option is disposal of Dutch waste in a shared, multinational repository.

COPERA is COVRA's current, on-going long-term research programme that started in 2020. COPERA is not the first Dutch programme on geological disposal. It builds on predecessor programmes, OPLA (1985 - 1992), CORA (1995 - 2001) and OPERA (2010 – 2017).

The focus of this COPERA salt safety case report is to provide an overview of the arguments and evidence that can lead to enhancing technical and public confidence in the levels of safety achievable in an appropriately designed and located GDF. As in the previous programme OPERA, it addresses three important objectives:

- Increase technical, public and political confidence in the feasibility of establishing a safe GDF in the Netherlands.
- Enhance the knowledge base in the Netherlands related to geological disposal.
- Guide future work in the overall COPERA programme in the Netherlands.

The development of scientific and technical understanding, data and arguments that support this safety case has been structured by addressing specific research questions using a multidisciplinary approach, involving tasks covering many areas of expertise.

How much waste is destined for geological disposal?

Three scenarios for future waste arisings were developed in 2022 as part of the national programme. Waste Scenario 1 is identical to the scenario used in OPERA: the operation of the Borssele Nuclear Power Plant until 2033 and the replacement of the High Flux Reactor in Petten with Pallas. The expected eventual inventory of wastes from all sources destined for geological disposal in Waste Scenario 1 is shown in Table 1. The design of the GDF in rock salt presented here can be easily adapted to the other 2 waste scenarios, provided that their waste characteristics match those used in waste scenario 1.

Waste Scenario 1 - Current installations +

Type	Volume in storage (m^3)	Number of canisters / containers in storage	Number of canisters / containers for disposal	Volume for disposal (m^3)
200 L drums	38,141	100,000	100,000	31,461
1000 L Containers		8,400	8,400	
Decommissioning waste	3,814	-	826	3,814
(TE)NORM	49,360	-	12,600	58,070
CSD-c	90	502	84	504
CSD-v	86	478	80	530
ECN-cansiter	49	244	122	643

Table 1) The expected number of waste packages for disposal in Waste Scenario 1.

What could a Dutch geological disposal facility look like?

The GDF design developed for COPERA (2020 – 2025) is based on the universally adopted 'multibarrier system' of natural and engineered barriers that contain and isolate the wastes and prevent, reduce, or delay migration of radionuclides from them to the biosphere.

The repository described here is assumed to be constructed in a salt dome: a massive body of salt that can extend a few kilometres in both the vertical and horizontal direction. The repository consists of surface and underground facilities, connected by three vertical shafts (Fig. 1). To make optimal use of the vertical extent of a salt dome, the underground facilities are at two levels for different categories of waste. The upper and lower levels are located at depths of about 750 and 850 m below the surface: depths chosen to ensure that deep erosion (glacial channels) will not disturb the repository during a future ice age. Both levels have an infrastructure area, with the larger, main infrastructure area located at the upper level. The upper-level infrastructure includes a mechanics workshop, material depot, personnel break rooms, equipment for dose rate measurements and decontamination, storage areas for vehicles, vehicle workshop, battery loading room, electricity supply room, transformer station, surveyors' office and bunker for backfill. The smaller, lower level infrastructure area is used to store equipment that is needed on a day-to-day basis at that level. In addition to the shafts, there is an inclined spiral ramp that connects the upper with the lower level. A minimum thickness of about 200 m of rock salt around the waste is considered sufficient to provide an adequate principal natural barrier. In the generic salt dome upon which the current concept is based, there is 350 m of salt surrounding the waste, which is 150 m more than the assumed minimum.

The lower level is for the disposal of vitrified high-level waste (vHLW), spent fuel from research reactors (SRRF) and non-heat-generating high-level waste (HLW). Here, most radioactive wastes are encapsulated in HLW packages optimized for disposal in rock salt (Fig. 2). These are thick-walled, carbon steel (TStE335) containers to hold the various types of HLW canisters, and have a thickness of 220 mm, as shown in Figure 2: For ECN canisters for SRRF on the left for and CSD-v and CSD-c canisters for vitrified HLW and fuel assembly debris on the righ. The upper level is for the disposal of low and intermediate level waste (LILW) and depleted uranium.

How do we analyse the safety of the GDF?

Quantitative analysis of the safety of the GDF is the central theme of this Safety Case. Estimates of potential radiological impacts on people are described for various future scenarios of how the disposal system might evolve. The Normal Evolution Scenario (NES) is the central case considered; it assumes normally progressing, undisturbed construction, operation and closure of the GDF, with no significant external disturbance of the disposal system in the future. However, the COPERA safety assessment recognizes that there are uncertainties about the long-term behaviour of some parts of the disposal system, as well as the potential for the GDF to be affected by various natural or human-induced processes and/or events, about which there are also some uncertainties. These uncertainties might perturb the normal evolution of the GDF and need to be assessed. One of the most important natural scenarios to consider is major climate change, which could lead to periods of global cooling, lowering of sea level and the formation of permafrost and mid-latitude ice sheets, which might cover the GDF area in the distant future. Accordingly, COPERA also identified other types of scenarios, including a range of alternative evolution, some of which are addressed in the COPERA safety assessment - failure of the HLW packages directly after closure of the repository, failure of all tunnel seals, failure of a spiral ramp seal, less probable characteristics of radionuclide mobilization and transport, and reduced long-term sealing by backfill. Others will be addressed in a future assessment - failure of a shaft seal, flow path between a brine pocket and nearby mine excavations, pressure-induced permeation of fluids in salt formations. In addition, three different

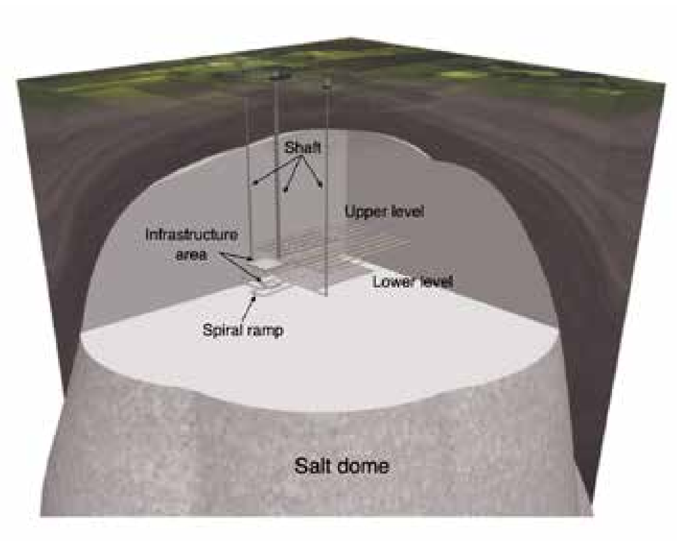


Figure 1) The general layout of a two-level repository in a generic salt dome. The upper level will be used for the disposal of LILW and (TE) NORM while the lower level will be used for the disposal of HLW.

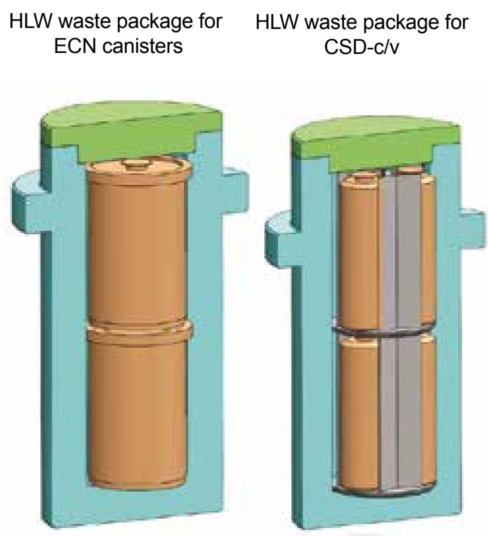


Figure 2) The two designs of HLW package. Left: the HLW package for 2 ECN canisters. Right: the HLW waste package for 6 CSD-c/v canisters.

human intrusion/influence scenarios have been identified and these will also be addressed in a future assessment.

For each of the scenarios considered, the potential future evolution of the GDF system is assessed, based on detailed studies needed to understand how each component will perform in the short and long term. Using this information, the migration of radionuclides that may be released from the wastes in the GDF is modelled and the impacts of any releases to the biosphere is calculated.

How much will the GDF cost?

The GDF design and the proposed implementation process allow estimates to be made of the future costs that will be incurred. These estimates determine the financial contributions to be paid by current waste producers to ensure that the national waste fund will be sufficient for GDF implementation. The total costs for disposal in 2130, based on the timetable, are estimated to be 3.5 billion Euro. The cost estimate assumes that a definitive decision on the disposal method will be made around 2100. An underground observation phase of 10 years is included, to facilitate retrieval of waste packages before closure, if required. If this phase is extended to 50 or even 100 years, costs will not change significantly. The development of the disposal concept is not included in the cost estimate.

The multibarrier basis of the GDF

The basis of geological disposal has been firmly established internationally for the last 45 years on the concept of the multibarrier system, whereby a series of engineered and natural barriers act in concert to isolate and contain the wastes and their hazardous content (Fig. 3). The relative contributions to the safety of the various barriers at different times after the closure of a disposal facility and the ways that they interact with each other depend upon the design of the disposal system. The design itself is dependent on the geological environment in which the facility is constructed.

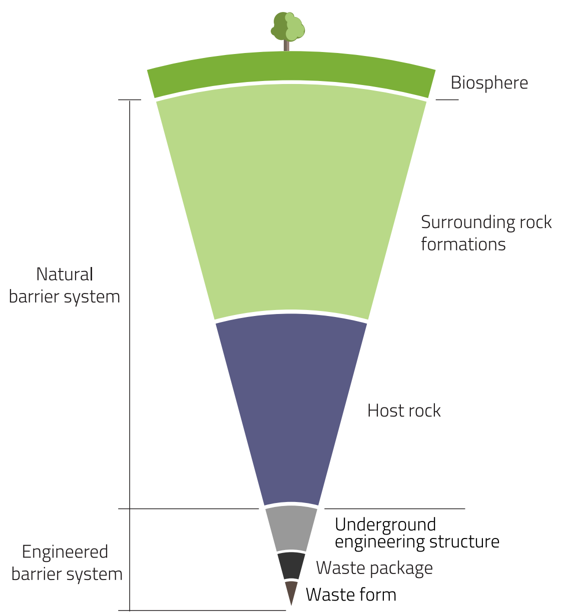


Figure 3) Components of multibarrier systems at the time of completion and closure of the geological disposal facility.

Consequently, the multibarrier system can function in different ways at different times in different disposal concepts.

What is the Natural Barrier System?

The host rock for the GDF, rock salt, and the overlying geological formations comprise the natural barriers within the multibarrier system.

Rock salt is the principal natural barrier. Undisturbed rock salt is practically impermeable and can thus provide complete containment. In the Netherlands, deposits of rock salt are very old and stable. Rock salt from the Zechstein Group, for example, was deposited over 250 million years ago during the Permian, while the rock salt in the Triassic Röt formation is over 145 million years old. Both have the capability to isolate the waste from people and the environment for at least the one million year timescale examined in safety assessments and probably for much longer, and both are present in a potentially appropriate depth range across large parts of the northeast and southeast Netherlands, in thicknesses of greater than 200 m. While COPERA considers a GDF in a generic salt dome, with some minor changes, the repository could also be constructed in other salt structures, such as bedded salt and salt sills. Bedded salt formations have a roughly horizontally layered structure, while a salt sill is an intermediate form between bedded and dome salt.

Because salt is plastic, deforming under load, and soluble in water, diapirism and subrosion are processes that must be assessed when considering the long-term stability of a formation. Diapirism is the gradual upward movement of a salt dome through overlying sedimentary formations, while subrosion is the dissolution of salt by groundwater. In principle, these processes could lead to the disruption of the geological barrier (salt) around the GDF and release of radionuclides into groundwaters over timescales of millions to tens of millions of years. However, these timescales are long after the hazard potential of the wastes has diminished, and well beyond the period of concern for safety assessment. In the Netherlands, diapirism rates of salt domes are estimated to be between 0.001 and 0.1 mm/year and possibly even lower, while the subrosion rates are estimated to be in the order of 0.01 and 0.1 mm/year and possibly even lower.

It is recognised that there are uncertainties related to the properties of the rock salt and that these need to be studied in the future. Three areas of uncertainty are currently considered, namely the thickness and depth of salt formations of potential interest for a GDF, their internal structure and homogeneity, and their short- and long-term evolution. The quality and coverage of the data on the thickness and depth of the rock salt of the Zechstein group and the Röt formation (the two most promising formations for a GDF) are not yet high enough to allow proper consideration of potential siting areas. This is particularly the case for the Röt formation. There is also a large uncertainty in the internal structure and homogeneity of salt structures in the Netherlands, in part because it is challenging to image salt structures seismically and interpret the data. With respect to the long-term evolution of salt structures, the subrosion and diapirism rates have been determined for specific salt domes in this and previous research programmes but are not precise. Better data would help to improve understanding of the evolution of salt structures through time. With respect to short-term evolution (tens to hundreds of thousands of years), the specific interest is on how major climate driven changes such as an ice age could influence the diapirism and subrosion rate.

Overlying and underlying geological formations

The bedded and dome salt formations of the (late Permian) Zechstein Group and the Röt formation lie within a thick sequence of sedimentary formations. Depending on location, this can range from salt deposits of the middle Permian Rotliegend Group,sandstones and conglomerates of the Early Triassic Germanic Triassic Group, salt of the Muschelkalk and Keuper formations and clay in the Upper North Sea Group. Some of the sediments in the overburden have high permeability and act as aquifers, through which radionuclides could potentially migrate to the surface if they were to leave the repository. These aquifers contribute to post-closure safety because any releases that might occur would be dispersed and diluted in these large bodies of groundwater, thus reducing their concentrations and their consequent hazard potential.

How might climate change impact the natural barriers?

During the Quaternary glacial cycles, the Netherlands has periodically been covered by ice sheets extending down across the Baltic and North Sea areas from a Scandinavian ice cap. Not every glaciation has been sufficiently intense to cause ice cover as far south as the Netherlands and, even in the more intense glacial periods, not all the present country has been covered by ice. Ice-sheet loading can affect subrosion and diapirism rates and glacial meltwaters at the end of an ice age can cause deep erosion. In a future GDF siting programme, it will be essential to look in more detail at the likelihood and consequences of such a scenario. The current understanding is that interglacial conditions similar to the present day are likely to persist for around 100,000 years – possibly longer. If deep erosion does not affect a GDF until sometime after 100,000 years, the radioactivity of the HLW will already have been markedly reduced.

The current COPERA safety assessment makes the conservative assumption that the next ice age will occur much sooner, in 50,000 years' time.

What is the Engineered Barrier System?

Undisturbed rock salt is practically impermeable and should thus, on its own, provide complete containment. Construction of the repository, however, perturbs the host rock by excavating shafts, tunnels and other open spaces needed to emplace the wastes. To ensure the closure and sealing of these open spaces, multiple engineered barriers are used. These are concrete backfill and seals, granular salt backfill, the HLW package and the HLW and LILW waste forms themselves, along with their containers. For the various types of HLW, engineered containment after closure of the GDF is initially provided by the steel HLW package (Fig. 2) and the concrete seals in the shafts and disposal tunnels that prevent the inflow of water from overlying formations. For the various forms of LILW, containment during the operational period is provided by the waste forms, their containers, concrete seals and the cement backfill of the disposal rooms, but after closure of the GDF, our safety case currently (and conservatively) assigns no containment function to their waste forms, containers and the cement backfill of the disposal room.

It is expected that the initial engineered containment barriers of the HLW (waste package, seals) will degrade with time and additional engineered containment must be provided, in the form of granular salt backfill. This backfill is in any case an important component of the engineered containment during the operational and immediately post-closure stages, as it stabilises the openings in the GDF. This because it is used to backfill the transport, ventilation and service tunnels in the upper and lower level, as well as the shafts, between the concrete seals. Granular salt backfill initially has a relatively high porosity and permeability but, compacts with time, so that its properties becomes comparable to the undisturbed host rock: impermeable. In the HLW disposal tunnels it is emplaced dry, to limit corrosion of the HLW packages and the production of gas but is moistened during emplacement where it is used in the shafts and other openings within the lower and upper level (e.g., transport, service and ventilation tunnels), as this increases the compaction rate to ensure it achieves the required low permeability faster. In the unlikely case that brine inflow to the waste occurs, the engineered barriers contribute to the containment of the radionuclides by restricting the movement of contaminated brine or allowing only very slow dissolution and mobilisation of the radionuclides. For backfilling the infrastructure areas in both the upper and lower level, gravel will be used. This is done to help to minimise the gas pressure within the repository: within the gravel, gas can accumulate if it is generated.

How will the backfill and seals behave in the multibarrier system?

Granular salt backfill is a key component of the salt GDF multibarrier concept and contributes to the long-term containment function of the repository system by achieving a very low permeability by compaction. Three successive stages of compaction can be recognised (Fig. 4). In the first stage, the host rock converges (creeps) to fill open spaces between it and the backfill; these result from both the settling of the backfill over time and the inability to fill an open space entirely during backfill emplacement. During this phase, the backfill begins to self-compact and microcracking may occur within the host rock closest to the tunnel, in the so-called excavation disturbed zone (EDZ). In the second stage, the backfill starts to compact more strongly, due to stress imposed by the convergence of the host rock, since both are now in direct contact. The rate at which the backfill compacts depends on many factors including, for example, intrinsic properties such as the grainsize of the backfill, the temperature and the moisture content, but also the rate of convergence of the surrounding host rock, coupled with the resistance of the backfill. Under repository conditions, it is expected that stages 1 and 2 take in total about 1,000 years for a moisturised granular salt backfill. At that point, the granular salt backfill will have a permeability of about $1 \cdot 10^{-19}$ m^2. In the third stage, compaction of the backfill has essentially ceased. At this point, static healing/sealing of both the granular salt backfill and the EDZ in the host rock is expected to become the dominant process; this will eventually result in the granular salt backfill attaining the same properties as the host rock: it will become impermeable. Dry granular salt backfill will take much longer to reach stage 3 but will only be used in tunnels where HLW will be disposed of, to minimise gas generation.

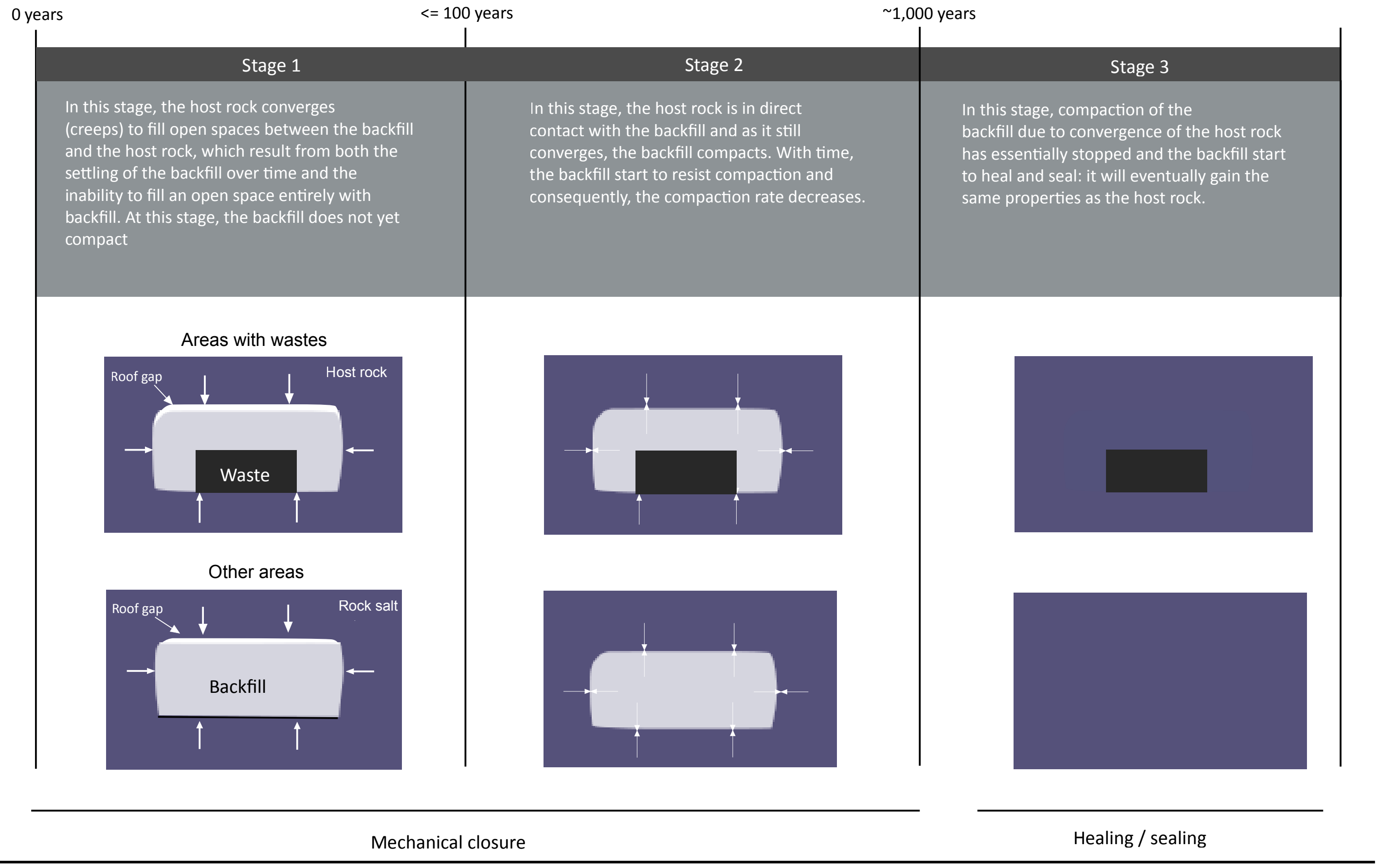


Figure 4) The three different stages of compaction and healing/sealing affecting the HLW disposal tunnels, and the dominant processes involved. During the first stage, convergence of the host rock closes the crown-space gap. In the second stage, the granular backfill compacts. Stages 1 and 2 together are referred to as mechanical closure. In Stage 3, compaction due to convergence of the host rock has ceased and the backfill starts to heal and seal.

For the COPERA safety assessment, we conservatively consider only the second of the three compaction stages mentioned. The first stage is not modelled, as it is expected to last for only a few decades so that the initial state of the disposal system assumed in the safety case is reached almost immediately. The third stage is not modelled at present, as it is still part of ongoing research. Thus, in the COPERA safety assessment, the assumption is that compaction will stop when a residual porosity of 1% is reached and the associated permeability is about $1 \cdot 10^{-19}$ m^2. This is a conservative assumption, as it is expected that the porosity will decrease further due to healing and sealing (stage 3; disconnection of pores in the salt) which in turn will decrease the permeability even further, until the granular salt backfill becomes impermeable.

It is recognised that there are uncertainties related to the compaction of the granular salt backfill – most importantly, how long it takes to attain the same properties as the host rock (stage 3). Better understanding of this will help to refine our requirements and optimise the other concrete and steel engineered barriers. A further uncertainty that needs to be quantified relates to the minimum thickness of the granular salt backfills in the shafts between the GDF tunnels and the top of the salt dome. Finally, the effect on compaction of gases generated by corrosion processes in the repository needs further investigation.

In the period when the granular salt backfill still has a high permeability, the necessary containment is provided by strategically placed seals in both tunnels and shafts. The moisturised granular salt backfill is expected to obtain a low permeability in about 1,000 years. However, the concrete seals are expected to maintain their effectiveness for much longer: 50,000 years. After this period, it is assumed that glaciation may alter hydrogeological and geochemical conditions, introducing significant uncertainty in predicting the chemical composition of incoming waters which could lead to the degradation of concrete seals—particularly those in direct contact with the overburden formations. Based on practical experience in Germany, there are two current options for tunnel seals: Sorel and salt concrete.

A simplified shaft design is assumed in COPERA as the eventual shaft design will depend on the local geology (e.g., the presence of anhydrite layers). The simplified closure system, where it passes through the rock salt, consists of, from top to bottom, a concrete seal, moisturised granular salt backfill and a further concrete seal. Detailed shaft seal designs have not been considered in any of the previous Dutch studies on rock salt, but work has been done elsewhere. Extensive research in both Germany and the USA has yielded a design that consists of different elements that, together, provide the necessary short and long-term properties to ensure containment. The proposed shaft sealing for the Gorleben repository in Germany, for example, consists of three short-term sealing elements, one long-term sealing element, abutments, and materials that can trap water or gas in their pores. Together, these would delay the inflow of groundwater into the repository sufficiently long for the shaft granular salt backfill to gain a sufficiently low permeability. The type and thickness of the materials used depend on the structure, properties and mineralogy of the evaporite formations through which a shaft passes. It should be noted that these seals are designed only for the part of the shaft that is located within the salt: in the overburden formations, the shaft is backfilled conventionally. Further work will be required on the design of shaft closure system and seals, and on the appropriate materials to use in the seals themselves.

How will the waste packages behave in the multibarrier system

Conservatively, only the HLW package is assigned a post-closure containment role: LILW containers are assumed to provide no containment after closure of the GDF. The HLW package is designed to provide complete containment for at least 1,000 years, which is the time the granular salt backfill needs to attain a low-enough permeability to ensure that there is no significant brine flow. However, it is likely that the HLW package will provide containment for a significantly longer period. In the COPERA safety assessment, the HLW packages are assumed to fail 1,000 years after closure but an additional alternative scenario was assessed in which the HLW packages are assumed to fail directly after the closure of the repository.

For all the waste packages used for LILW and depleted uranium, an effective zero 'failure time' for all LILW waste packages is used in the safety assessment and COPERA conservatively assumes that radionuclides are released into the concrete backfill of the disposal rooms immediately after the closure of the GDF.

What happens to gases produced in the GDF?

As part of the COPERA research programme, a scoping study was undertaken to estimate the potential for gas generation within a repository in rock salt. Gas pressure can delay, or even halt, the compaction of the granular salt backfill. The study considered three main gas generation mechanisms: corrosion, microbial breakdown of organic substances and radiolysis (can be important in waste with high beta/gamma activity). The model results suggest that gas generation depends primarily on the availability of brine, which is likely to be very limited, not only because a repository in rock salt is dry, but also due to the low permeability of the granular salt backfill. Limiting the availability of brine reduces gas generation significantly, but some gas is likely to be generated within the GDF, because there will be some brine available, for example, in the granular salt backfill. The next step will be to expand the model to include the compaction of the granular salt backfill in the safety assessment model.

How will the disposal system evolve over time?

The information available to COPERA to quantify GDF performance is subject to different types and levels of uncertainty. COPERA allows for this by making conservative simplifications, assuming poor performance, using pessimistic parameter values and omitting potentially beneficial processes. The results of the COPERA safety assessment are thus expected to be pessimistic forecasts of system performance. However, it is also essential for system engineering optimisation purposes to make best estimates of how we expect the system to behave in reality, acknowledging the uncertainties along the way. This allows a balanced view that will inform later decisions on GDF design optimisation and, eventually, on acceptable site characteristics. For example, this approach avoids over-engineering system components or rejecting otherwise acceptable GDF sites.

COPERA compares best estimates of the behaviour of system components in different timeframes (expected evolution) with the simplified assumptions of the safety assessment. The expected behaviour is summarised below.

From closure to 1,000 years

It is expected that the characteristics of the biosphere and the overlying sediments remain similar to the present day, with only some minor erosion, which will not affect the repository's performance. After the repository is sealed, the (moisturised) granular salt backfill in the tunnels and shafts will begin to compact, reducing porosity and permeability over the next 1,000 years, thereby effectively sealing the repository. The heat generated by high-level waste (HLW) will temporarily speed up the compaction process, while any small fractures in the surrounding rock will heal during this period.

Within the repository, brine displacement will occur as the granular salt backfill compacts, but the flow will be limited due to its low brine content and the low permeability of the backfill. In the first 1,000 years, radionuclides from LILW and (TE)NORM will primarily be transported through advection, but after this period, diffusion will become the dominant mode of transport. HLW radionuclides will remain fully contained within the HLW packages.
Gas generation from corrosion of the steel HLW packages and radiolysis will be minimal, as the granular salt backfill surrounding the HLW packages has no added moisture: in other placer moisture is added to the granular salt backfill to increase compaction. In addition, any gas generated will migrate to areas of the repository that do not compact. These are the infrastructure areas that will be backfilled with gravel and the concrete seals. After 1,000 years, the moisturised granular salt backfill will have attained a low permeability within the lower and upper levels and stage 2 of compaction will end. This is followed by the healing and sealing of the backfill (stage 3). In contrast, dry granular salt backfill used in the tunnels with the HLW packages will still be in stage 2 as its compaction is significantly slower. Also in the shaft, the moisturised granular salt backfill will still be in stage 2 at the end of this period as the temperature and pressure are lower than in the upper and lower levels of the repository and hence the compaction is slower.

In terms of subrosion and diapirism, 0.1 m of salt will have been removed by subrosion and the salt dome will have risen 0.1 m at the end of this 1,000 year period. In both cases, the current subrosion and diapirism rate of 0.1 mm/year is assumed.

A simplified behaviour is modelled in the COPERA safety assessment. It is assumed that the HLW packages will remain intact during this period, while LILW packages are assumed to provide no containment and consequently LILW radionuclides are assumed to be released immediately after the repository's closure. Temperature and lithostatic pressure, which influence backfill compaction, are considered constant, with temperatures based on a geothermal gradient and pressure calculated from the sediment and salt density at the Gorleben site. Conservatively, no solubility limits are included, and the granular salt backfill will remain permeable throughout this period. Gas generation is not considered in this assessment but will be addressed in the next safety evaluation.

1,000 years after closure – start next glacial period (assumed at around 50,000 years)

As for the first 1,000 years, the biosphere, including climate, vegetation and groundwater, are expected to remain similar to present day conditions, though sea levels may fluctuate. Subrosion, which will continue at the current rate of 0.1 mm/year, will result in about 5 m of salt being dissolved at the end of this period. Likewise, with a rate of 0.1 mm/year assumed here, the depth of the repository will decrease by 5 m due to diapirism. It is assumed that the next ice age will occur in 50,000 years which is, as a result of global warming, unlikely. Subrosion, diapirism and changes in the biosphere are not expected to affect the repository's performance. Within the repository, moisturized granular salt will start to heal and seal (stage 3) within the lower and upper levels. Moisturized granular salt used in the shaft will still be in stage 2 at the start of this period but after a few hundred more years it will also start to heal and seal (stage 3). In contrast, the dry granular salt in the disposal tunnel will take an additional several thousand years to reach this stage. As healing progresses, the pores in the backfill will disconnect, preventing diffusion and effectively immobilizing any mobilised radionuclides within the granular salt backfill. Additionally, no water will be able to enter the repository via the shaft, ensuring the full containment of radionuclides.

In the safety assessment, it is conservatively assumed that the HLW package will fail 1,000 years after repository closure. At this point, radionuclides from the CSD-v, CSD-c and ECN canisters are considered instantly available for transport. Furthermore, the granular salt backfill is assumed not to heal in the safety assessment, so advective and diffusive transport of radionuclides remains possible, though very limited and slow due to the low permeability of the granular salt backfill. Additionally, gas generation is considered zero.

Next glacial period (duration assumed to be 100,000 years)

This period covers the next ice age, during which several geological changes are expected. The uppermost 50 m of sediment may erode, rivers could incise by 20 - 120 m, and glacial basins up to 150 m deep may form. As an ice sheet advances over the salt dome, differential loading could temporarily increase diapirism rates. Permafrost may penetrate up to 270 m underground, reducing groundwater recharge, increasing salinity and slowing subrosion. However, glaciations will lower the sea level, increase groundwater flow velocities and raise subrosion rates. The movement of the ice sheet could also reactivate old faults, temporarily increasing their permeability. Melting ice sheets may force fresh water into overburden sediments, further increasing subrosion rates, while glacial channels up to 600 m deep may form and fill with sediment.

By this time, compaction of all the granular salt backfill in the repository will have reached the final stage, with properties equivalent to those of the host rock, thereby immobilizing all radionuclides within the GDF. Over a 100,000-year ice age, the salt dome is expected to rise by 10 m, with an equal amount of salt dissolving due to subrosion: in other words, the repository itself is still too deep to be affected by an ice age. The total amount of salt dissolved by subrosion at the end of this stage is 15 m. The salt dome will have risen by the same amount.

In the safety assessment, it is conservatively assumed the granular salt backfill still has sufficient permeability to allow both advective and diffusive transport of radionuclides. Furthermore, gas generation is assumed to be zero.

End of the glacial period – 1,000,000 years

The next stage covers the period from the end of the first ice age to one million years. During this time, multiple glacial periods could occur, potentially forming multiple glacial channels and increasing subrosion and diapirism rates temporarily. These glacial periods could significantly change the biosphere and reduce the overburden formations above the repository, possibly bringing it closer to the surface, especially if sedimentation does not occur. At most, 10 glacial periods might be expected to occur within one million years, with varying durations and intensities. Only some of these might be expected to extend far enough south to affect the Netherlands. Moreover, sedimentation is expected to continue, increasing the thickness of the overburden, and no major tectonic events are anticipated that would result in regional uplift of the Netherlands. While the future development of the overburden formations is uncertain, it is improbable that the salt dome will pierce the surface: the repository is expected to remain several hundred meters deep.

Within the repository, conditions are therefore expected to remain stable, with the backfill maintaining the same properties as the host rock, and all radionuclides remaining contained. Over one million years, about 100 m of salt will dissolve due to subrosion, assuming a 0.1 mm/year, leaving at least 250 m between the repository and the top of the salt dome. Even in a scenario with double the subrosion rate, there would still be 150 m of separation. Diapirism will cause the repository to rise by 100 m at the end of this period, assuming a diapirism rate of 0.1 mm/year. Hence, neither subrosion nor diapirism will affect the repository's performance.

On even longer time scales, subrosion and diapirism may eventually release small amounts of immobile, long-lived radionuclides into overburden formations. By then, the repository's hazard potential will be comparable to, or lower than, naturally occurring ore bodies.

In the COPERA safety assessment, as for the previous periods, advective and diffusive transport of radionuclides can still occur during this period. As in the previous period, no gas generation is expected.

How safe is the GDF?

The COPERA safety assessment calculates the potential impacts of the GDF on the environment over the timescales discussed above. It takes a simple and largely conservative modelling approach that adopts a similar methodology and assumptions to those of other international exercises. The approach captures the widely accepted, most critical processes of advection, diffusion and compaction that control the behaviour of a GDF in salt.

The Normal Evolution Scenario

The Normal Evolution Scenario (NES), is the reference case for this initial stage of COPERA. The safety assessment shows that even after a million years remain in place within the repository: no radionuclides have migrated out of the repository into overlying formations and biosphere. The multibarrier system has effectively performed its isolation and containment task by this time. Over much longer periods, many millions of years, releases are likely to occur eventually in the normal evolution scenario in locations where there is a significant combined effect of subrosion and diapirism, if these rates are high. However, by such times the hazard potential of the waste has reduced to levels well below those of natural uranium ore deposits.

Overall, even using pessimistic approaches, the performance assessment calculations for the NES show that potential radiation exposures to people in the first million years after closure are zero. The NES is the most likely evolution and remains the focus for calculations in the far future. Long-term interactions between degradation products of the different types of waste are limited, since the different types of waste are disposed of at different sections of the GDF. These interactions need, however, to be evaluated. Also, gas generation needed to be included in future post-closure safety assessments, as it could slow down or even hinder compaction, or impact other processes.

In conclusion, in the normal evolution scenario, and for at least one million years after closure, what is placed in salt stays in salt.

The Alternative Scenarios

Alternative evolution scenarios are less likely but it is important to calculate their consequences because these calculations show the redundancy of the multibarrier system. In total, five of eight identified alternative scenarios were modelled in the COPERA safety assessment. These are failure of all HLW packages directly after the closure of the repository, failure of all tunnel seals directly after the closure of the repository, failure of a spiral ramp seal directly after the closure of the repository, less probable characteristics of radionuclide mobilization and transport, and reduced long-term sealing by backfill. Although differences exist in the extent to which radionuclides travel within the repository for each alternative scenario, in no cases are radionuclides predicted to leave the repository within one million years after closure. The results indicate that the shaft seals play an important role in long-term safety by limiting the amount of brine entering or leaving the repository. Future work will evaluate the remaining alternative scenarios and human intrusion scenarios.

Can the disposal system be optimised?

Optimising the radiological protection provided by the GDF is an important objective for the future. In COPERA, examination of optimisation options has been limited, especially as the release in the normal evolution is zero. However, the safety assessment shows that the designs of the HLW package, tunnel seals and spiral ramp seal do not affect the outcome of the safety assessment: these are not critical factors for the safety concept. This is because the shaft seals, when they function as expected, inhibit any contaminated brine from entering or leaving the repository. Nevertheless, using a robust HLW package has advantages, in particular during the operational period, when it eases handling. Likewise, the use of tunnel seals also has operational advantages, and may also play a critical role should the shaft seals not function as expected, although this needs to be tested.

A potential cost optimisation is the reduction of the centre-centre distance between HLW packages. In the current disposal concept, this is 10 m, but could be reduced to allow for more HLW packages within a disposal tunnel. However, its effect is expected to be limited. Another potential optimisation is to use the depleted uranium waste as an aggregate in concrete, which is then used as a backfill material. A further option would be to use the conditioned depleted uranium directly as a backfill for the disposal rooms used for the disposal of LILW. Irrespective of where it is used, the time needed for disposal of all the waste would be reduced by ten years, and 21 fewer disposal rooms would be needed, thus reducing to less than half the number of upper-level disposal rooms. However, the impact on sealing effectiveness of using concrete with depleted uranium as an aggregate is not yet well studied. In addition, backfill containing depleted uranium must be treated as a radioactive material, which complicates operations

Conclusions of the initial COPERA Salt 2024: A Conditional Safety Case & Feasibility study

The feasibility of constructing a GDF in salt in the Netherlands

The COPERA GDF concept is based on the well-developed German concept for disposal of HLW in salt domes and on the operational WIPP repository in bedded salt in New Mexico, USA. It also builds on the previous Dutch concepts. There are decades of practical experience in both commercial salt mining and even in constructing an actual repository for radioactive wastes.

Geotechnical assessment within the COPERA research programme indicates that a stable and robust two-level GDF can be constructed and operated in a salt dome at depths of >700 m, with the model adopted for COPERA having levels at 750 and 850 m depth. For the construction of the GDF, existing salt mining techniques and equipment (e.g. continuous miners and scalers) can be used.

Existing international studies also show that there are practical techniques for sealing tunnels and shafts in a GDF. It is expected that further progress and operational experience will become available over the next 100 years, well before these techniques need to be deployed in the Netherlands.

Overall, there is considerable scope to adapt and optimise the engineering design of the GDF in future years and it is expected that any eventual design will be significantly further developed from the current COPERA concept.

The feasibility of siting a GDF in salt in the Netherlands

COPERA was not a siting study, but it is important to have confidence that suitable locations for a GDF might be available if rock salt is eventually selected as the host formation. Rock salt is present in appropriate thicknesses and depth ranges across large parts of the northeast and north of the Netherlands, but there are significant uncertainties in the depth-thickness distribution of some rock salt formations. Also, the internal structure of salt structures, in particular of salt domes, is not yet well known. The eventual GDF design can be adapted to be compatible with the specific properties of candidate locations, thus allowing flexibility in depth and layout aspects that are not critical to safety.

A siting programme will need to avoid certain geological structures and features, and guidelines and criteria for doing this will need to be developed. Factors that will need to be considered include other uses to which a salt dome might have been subject (e.g., the presence of caverns for storage of oil, gas et cetera), the variability of the rock salt properties, the potential for deep glacial erosion and diapirism and subrosion rates.

Other potential GDF host rocks exist in the Netherlands, some of which have been evaluated in the past and all of which will be studied in more detail in the future. These include Paleogene clays for which a safety case is presented, in parallel to this report.

It is recognized by COVRA that siting a GDF involves considerably more than evaluating technical factors. Any future siting programme will need to take account of societal requirements and will be staged, progressive and consensual in nature.

The COPERA salt GDF provides a completely safe disposal solution

The GDF concept is expected to provide complete containment for at least 1 million years and possibly for much longer. Beyond this period, a minute fraction of highly mobile radioactivity might eventually, due to disruption of the geological barrier by subrosion or diapirism, move into surrounding geological formations, but will be diluted and dispersed in deep porewaters and groundwaters, resulting in biosphere concentrations that cause no safety concerns and are expected to be well below natural levels of radioactivity in drinking water.

Confidence in safety

The safety case for geological disposal relies on understanding processes that have been active for millions of years in deep rock formations. By studying geological settings similar to those considered for a GDF, we can gain confidence in our understanding of these processes. Natural analogues provide evidence of rock salt's ability to offer long-term containment. For instance, the existence of 250-million-year-old rock salt indicates its impermeability, as any permeable salt would have been dissolved by groundwater. Examples include gas trapped beneath the Zechstein salt in the Netherlands and CO_2 trapped in the Werra/Fulda salt deposit in Germany, which demonstrate rock salt's effectiveness as a seal.

Additionally, rock salt's dryness leads to exceptional preservation of organic materials. In the Hallstatt salt mines, artefacts from the Bronze Age, including wooden tools and textiles, have been preserved. Similarly, ancient human remains found in the Chehrabad Salt Mine in Iran are remarkably well-preserved due to the dryness of the salt.

For understanding the compaction of granular salt backfill, analogues are crucial, since laboratory experiments can only simulate short time periods and may not accurately reflect real conditions. The Sigmundshall mine in Germany, where granular salt (halite) waste has compacted to low porosity over 40 years, provides insights into the compaction process. The findings from this mine indicate that pressure solution creep is a significant mechanism at low stresses and must be considered when determining the time-scales for sealing backfilled repositories in rock salt - as is done in the COPERA safety assessment.

Confidence in the reliability of the COPERA performance assessment calculations is also enhanced by the fact that they are compatible with those estimated independently by other national programmes and also in previous Dutch rock salt safety assessments.

Optimisation of the design and the Safety Case is possible

Several processes and scenarios that could affect or alter the normal evolution have not yet been treated at this stage of COPERA and thus constitute open issues that will require further R&D and safety assessment. The principal uncertainties have been identified in each COPERA work package and will be addressed by future studies. Not all the work is required in the next decades; it can be staged over several iterations of the future COPERA research programme.

Over the five years of its operation, COPERA has achieved its principal aims and has been a valuable exercise to progress and support national policy in the Netherlands. A GDF in the rock salt at around 750 m depth can clearly fulfil its task of permanently isolating Dutch wastes and protecting current and future generations.

The results obtained to date give confidence that the disposal of all the current Netherlands inventory of long-lived and highly active radioactive wastes at depth in the rock salt is feasible. The approach evaluated is sufficiently flexible to handle any likely future inventory changes or respond to changes in disposal schedule.

The COPERA GDF concept, if implemented at a site with an appropriate geological setting, can provide high levels of safety that match those estimated in other national programmes. It would clearly meet international standards for this type of facility. However, more work remains to be done and continued RD&D will enhance and optimise the GDF design, giving a clearer picture of future costs and implementation flexibility. COPERA has built upon OPERA, which built upon CORA and OPLA, and it is essential to maintain continuity of expertise and knowledge amongst the scientific and technical community in the Netherlands in this way.

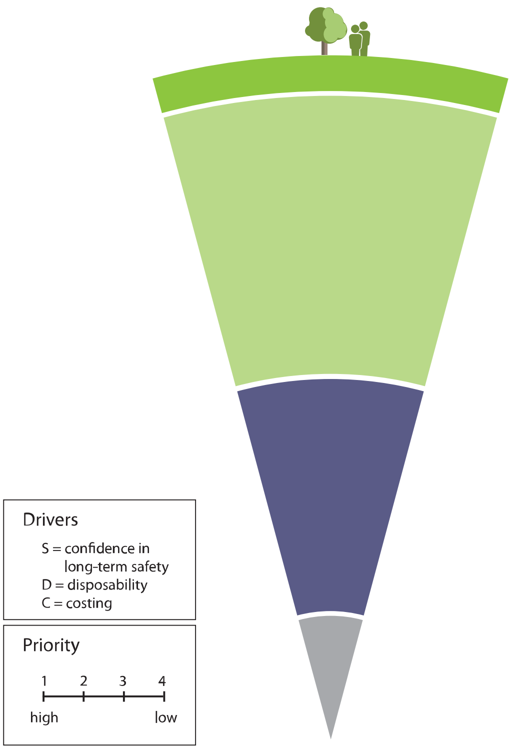
Looking forwards

The information generated in COPERA can be used to support waste management policy development in the Netherlands and to provide a more accurate basis for establishing future financial provisions for waste management. The availability of a safety assessment reference case and approach allows COVRA to make disposability assessments of any future waste arisings or of packaging proposals from waste producers.

The COPERA results are compatible with the policy decision to provide long-term storage and to carry out a staged programme of research, development, and demonstration (RD&D) into geological disposal. They illustrate that an endpoint of geological disposal can be implemented. COPERA has developed a roadmap for future RD&D for disposal in rock salt that starts with the identification of the key topics that need to be addressed in future work. The illustration below (Figure 5) shows these key topics for the main components in the disposal system, along with the drivers for carrying out further work and the priorities currently attached to each component. The highest priority is associated with obtaining further information on the host rock: rock salt.

Awareness of the GDF design concept and its requirements in terms of depth, area and geological conditions will facilitate fitting this facility into national planning policies and priorities for the use of underground space.

The existence of COPERA and its findings are important contributions to satisfying the Netherlands' obligations under both EC Directive 2011/70/EURATOM and the IAEA Joint Convention, showing that substantial progress has been made on the national programme. The project also supports the Netherlands' position of carrying out a dual-track (national and potential multinational) policy for radioactive waste management. The results can be used as the Netherlands' contributions to the development of multi-national projects.



Component	Key topics	Drivers	Priority
Society	Integrating societal aspects into technical research	S ■■■ D ■□□ C ■□□	2
Biosphere	- (Current knowledge sufficient)	S ■□□ D ■□□ C ■□□	4
Surrounding rock formations	Subglacial channels and climate change	S ■■□ D ■□□ C ■■□	3
Hostrock	Geotechnical properties Long-term evolution (subrosion/diaprism) Salt deposits (extend and structure)	S ■■■ D ■■□ C ■■■	1
Engineered barrier system	Shaft and drift seals Granular salt backfill compaction	S ■■□ D ■■■ C ■■□	2

Figure 5) Key topics for research into geological disposal in salt, organised according to the components of the multi barrier system.


DRYFAST

1. Introduction

1.1 Purpose and context of the present report

The objective of this report is to present an overview of results and conclusions from ongoing work in the Netherlands on developing safety cases for a Geological Disposal Facility (GDF). One of the options being examined is a GDF in a salt formation. It is anticipated that COVRA will produce safety cases every 10 years, with updates every 5 years. The present work is part of COVRA's ongoing COPERA programme. The report updates and expands on previous Dutch research in disposal in rock salt, considering progress in the Netherlands and elsewhere in the intervening years.

The progress made in the salt studies is mostly related to the development of a disposal concept and improving understanding of the barriers that determine the safety of a rock salt GDF. However, effort has also been put into examining more closely the practicability and efficiency of constructing and operating a GDF in salt, which explains why the present report title refers to both safety and feasibility. Because our intent is to ensure that the report can be read as a stand-alone document, information remaining unchanged since the 2017 OPERA Safety Case has been carried forward from that report and amended only as necessary with updated information.

In addition, a parallel study on the safety and feasibility of disposal in a GDF in clay has been prepared (Neeft et al., 2024b). This is also intended to be a stand-alone document, so that information common to both salt and clay studies (e.g., on the Dutch waste inventory) is included in both reports.

1.2 Why do we need geological disposal?

Radioactivity and radiation were discovered at the end of the nineteenth century. Today, about four generations later, the use of nuclear technologies in electricity generation, medicine, industry, agriculture, research and education has become indispensable. Consequently, radioactive wastes are generated. Since radioactivity decays naturally over time, safety can be achieved by ensuring that the wastes are isolated from the human environment until they have decayed sufficiently and no longer pose a hazard. The period for which the wastes must be isolated depends on the type of waste, but it can range from a few days to over 100,000 years.

Isolation in the short time (decades) is achieved by storing the radioactive materials in safe and secure surface facilities. This is a proven technology and is applied globally. However, storage of radioactive wastes in surface facilities is not a long-term or final solution for waste that remains radioactive for many thousands of years. Active monitoring, inspection, security and maintenance of a surface facility cannot be guaranteed over such a long timescale, especially as it is impossible to predict how society will develop over such period. Even on a much shorter timescale, the last 100 years for example, society has changed radically. For waste that remains hazardous for thousands to hundreds of thousands of years, the only currently accepted solution is geological disposal. This is the emplacement of waste several hundreds of metres below the surface in a stable geological environment. The special build facility is generally referred to as a GDF or repository, while emplacement of wastes in a GDF is generally referred to as disposal.

The materials within a GDF, including the wastes and their packaging, will degrade slowly over time. Even the most stable geological environments will eventually undergo changes over geological timescales of hundreds of thousands to millions of years. Complete containment of all radionuclides indefinitely is therefore not a realistic objective. Thus, the release and subsequent transport of any radionuclides that leave the GDF must be sufficiently delayed so that, before reaching the human environment, they will have decayed and diluted to levels that can never result in significant radiation doses to people or the environment. Defining doses that would be considered insignificant is both a technical and societal issue.

1.3 The Dutch Context

The Central Organisation for Radioactive Waste (COVRA) in the Netherlands is responsible for collecting, processing, storing, and disposing of all Dutch radioactive waste. The current Dutch policy is that current and future high level radioactive and long-lived waste is stored above ground for at least a hundred years, until approximately 2130 (Ministerie van Infrastructuur en Milieu, 2016). After this period of storage, this waste must be permanently disposed of in a GDF. A dual-track policy is being pursued for the GDF, i.e., the disposal can be exclusively national or multinational if several countries agree to share a common GDF. Research on disposal is an integral part of Dutch radioactive waste policy.
The coordination of this research is part of the core tasks of COVRA.

The decision-in-principle to dispose of Dutch radioactive waste in a GDF was taken by the government in 1984 (Minister van volkshuisvesting ruimtelijke ordening en milieubeheer, 1993). A definitive decision on implementing this disposal method will, however, be taken around 2100. Thus, while deep geological disposal in a national GDF is currently considered the reference solution, this could change in the future, if alternative options become available. As noted above, an alternative option is disposal of the Dutch waste in a shared multinational repository. Another potential option for some of the wastes is deep borehole disposal (Verhoef et al., 2021) in boreholes that can be kilometres deep. The current policy thus provides a certain flexibility (Ministerie van Infrastructuur en Milieu, 2016). It is possible to keep options open, because waste inventories accumulate slowly and facilities ensuring safe surface storage for decades have been implemented. The present rock salt safety case and feasibility study focusses on analysing the safety that could be achieved by implementation of a dedicated national GDF in rock salt.

Although this will be the first Dutch rock salt safety case and feasibility study, the research programme (COPERA) leading up to this safety case is not the first research programme on disposal in rock salt in The Netherlands (See also Appendix 1). The first research (ICK) started in 1972 and continued until 1979. It concluded that rock salt formations might be a suitable option for the disposal of radioactive waste. The research did not continue directly after the end of this programme but started again in 1985 with the OPLA research programme, which focused on disposal in onshore rock salt deposits. It ran in parallel with a research programme, DORA, which investigated offshore rock salt deposits. The latter was abandoned due to objections to the disposal of radioactive waste directly into the ocean, even though this concept was unconnected with the sub-seabed DORA concept. The subsequent research programme, CORA, was initiated to investigate the retrievability of wastes from a GDF in rock salt and poorly indurated clays. This was the first time that poorly indurated clay was considered as a potential host rock in the Netherlands. Retrievability was a new requirement introduced by the Dutch government (Minister van volkshuisvesting ruimtelijke ordening en milieubeheer, 1992). The CORA research programme was followed by the OPERA research programme (Verhoef et al., 2017), which focused mainly on poorly indurated clay with only two reports produced on rock salt (Hart et al., 2015a; Hart et al., 2015b). This research programme ended in 2017 with the publication of the OPERA safety case (Verhoef et al., 2017). The most recent research programme is the COPERA research programme which started in 2020 with the publication of the COVRA research programme (Verhoef et al., 2021). The COPERA research programme is a continuous research programme with a rolling agenda that will be updated every 5 years. This avoids periods without any research on geological disposal in the Netherlands. During the first phases of COPERA, referred to as COPERA (2020 -2025), research will be mostly fundamental in nature, but will address more specific solutions with time (Verhoef et al., 2021).

Although the reference date for implementation lies relatively far into the future (2130), starting the COPERA research programme in 2020 allows for learning from the development of geological disposal programmes in other countries, via international collaboration in research projects. Examples of international research projects of relevance for salt disposal that COVRA has joined during COPERA (2020 - 2025) are KOMPASS (e.g., Friedenberg et al., 2022b) and DECOVALEX (decovalex.org). These collaborations, together with an ongoing national research programme, allow continuous progress towards the resolution of all open scientific, technical and societal issues. Decisions and actions taken today influence the disposal concept; the technologies used today for the collection and treatment of radioactive waste must consider the characteristics of a future GDF. This is to ensure that the wastes will be acceptable for disposal in the facility and requires that research on the disposal of radioactive waste is started at this early stage.

The current research programme, COPERA, is part of the larger national radioactive waste disposal programme. In this national programme, different actors are involved and the roles and responsibilities of each of these actors must be clear. Responsibility starts with the waste produced. Any company in the Netherlands licensed to work with radioactive materials under the nuclear energy act, is bound by law to tender its waste to COVRA, which collects, treats, conditions and then stores the radioactive waste. The total amount of radioactive waste expected to be produced over the next 100 years can easily be stored at the COVRA site. As COVRA accepts ownership and full liability for radioactive waste, the coordination of research on geological disposal and the implementation of a GDF are also responsibilities of COVRA.

COPERA (2020 - 2025), will end in 2025, with the last year of the research programme being used to prepare for the next cycle. Many of the results of the COPERA (2020 - 2025) research programme can already be presented to the public as input for a wider discussion on future progress. Furthermore, publishing results now allows the Ministry of Infrastructure and Water management time for its reporting duty to the European Commission under the Waste Directive (EU 2011). The results of the COPERA (2020 - 2025) research programme is presented in numerous, detailed reports; the specific publications on the salt programme are described in (Appendix 2) These reports have been published on COVRA's website or as papers in scientific (open access) journals. In the present report, we provide an overview of the results in the framework of an overarching synthesis of the arguments and evidence that can lead to enhancing technical and public confidence in the achievable safety levels of a GDF in rock salt. To achieve this goal, the report is structured in the form of a Safety Case, as is recommended by international bodies and as has been done in numerous national geological disposal programmes (NEA, 2017).

1.4 Roles of a Safety Case in Geological Disposal

Safety Case is a term commonly applied in many industries where potential hazards to workers and the public must be assessed. Since the early 2000s, the safety case concept for GDFs has been developed by the NEA (Nuclear Energy Agency, NEA, 2004, 2013c) and IAEA (International Atomic Energy Agency, IAEA, 2006, 2011c, 2012b) and in many national waste disposal programmes. Here, as in the OPERA safety case (Verhoef et al., 2017), we use the definition from the IAEA Safety Standards for Geological Disposal (IAEA, 2011a):

"The safety case is an integration of arguments and evidence that describe, quantify and substantiate the safety, and the level of confidence in the safety, of the geological disposal facility".

In Chapter 3, we give details on the structure of a safety case. For the present safety case, key generic points concerning safety cases can be made and their relevance to this Dutch rock salt safety case and feasibility study pointed out:

- Safety cases are made at various stages in a repository development programme, and it is therefore an iterative process. This safety case and feasibility study is the first in a series of safety cases for a GDF in rock salt. With time, they will become more detailed.
- At earlier stages of a disposal programme, key data may be incomplete or based on certain assumptions. Incomplete data, or data with a significant degree of uncertainty, will be clearly marked.
- When no data are available, conservative (i.e., pessimistic), well-founded assumptions are made. When more, and more accurate data become available, they will lead to a higher certainty of predicted levels of safety and/or to design modifications in future work.
- In addition, an early-stage safety case may make preliminary assumptions that will have to be justified before a definitive safety case can be made. This is certainly the case for site specific data, which are currently lacking. These assumptions must be clearly stated and the approaches to confirming their validity laid out. For this reason, the final chapter of the present report outlines the roadmap for future work on radioactive waste disposal in rock salt in the Netherlands.
- A safety case made under these conditions can be characterised as a 'conditional safety case'. The present conditional rock salt safety case and feasibility study is clearly of this nature and represents the first of a series of safety cases.

1.5 Context and objectives of the Dutch Safety Case in rock salt

As noted above and described in more detail in Chapter 3, safety cases are produced throughout the lengthy process of repository development. This first rock salt safety case and feasibility study is far from what will be required for a license application. However, this initially generic safety case will gradually become more specific over time. The present safety case addresses three significant objectives:

1. The primary objective is to increase technical, public and political confidence in the feasibility of establishing a safe GDF in rock salt in the Netherlands for all the radioactive waste that is currently expected to have been produced up to 2130.

2. A further key aim is to update and enhance the knowledge base in the Netherlands related to geological disposal in rock salt. While some research on rock salt has been done during OPERA (e.g., Hart et al., 2015b), this was very limited. Hence, the last research programme that focused on rock salt ended over 20 years ago. The COPERA programme aims to renew and update the earlier technical and scientific studies and look more deeply into various topics, in the process enhancing national capabilities and providing information for a wider debate on the topic of geological disposal.

3. Finally, a specific purpose of this report is to summarise the work that has been done as part of the COPERA (2020 - 2025) and guide future work in the Netherlands on disposal in rock salt.

1.6 COPERA

The main goal of the COPERA research programme is to develop knowledge for implementing safe and efficient geological disposal of radioactive waste in poorly indurated clays and rock salt in the Dutch subsoil, considering all steps throughout the radioactive waste management chain. COPERA has the same three objectives that drove OPERA (Verhoef et al., 2021).

- Confidence in long-term safety (S): Increasing technical and societal confidence in feasible, long-term and safe disposal of radioactive waste in the Dutch subsurface, thereby supporting the requirements of the EC Waste Directive.
- Costing (C): Improving cost estimates by reducing uncertainties and optimising costs for the construction and operation of a GDF for radioactive waste in the Netherlands.
- Disposability (D): Improving the disposability of radioactive waste: optimising processes for efficient waste processing throughout the waste chain to be suitable for geological disposal.

The COPERA research programme is expected to contribute to:

- a strengthened national nuclear knowledge infrastructure and international network for geological disposal and radioactive waste;
- a societal discussion on geological disposal, informed by up-to-date knowledge and based on the recognition of the societal responsibility for implementing a safe final solution for radioactive waste;

- the consideration of the multinational repository option as a part of the dual-track strategy (Ministerie van Infrastructuur en Milieu, 2016).

To accomplish the main goal of the research programme, various work packages have been developed (Fig. 1.1). The overarching work package, WP 0, includes all tasks related to programme management and coordination. The tasks in this work package are performed by the programme office at COVRA. Work packages 1 and 2 (WPs 1-2) have a more strategic and integrative character. WP1 is related to strategic aspects of the research programme, such as costing and shared solutions and other strategic studies. WP2 covers integration of the knowledge obtained through the research programme and the production of safety cases in rock salt and poorly indurated clay (Neeft et al., 2024b).

Four work packages (WPs 3-6) are structured around key topics that must be studied to produce these safety cases; these are related to the components of the multibarrier system and are specific for the host rocks. The tasks (projects) in these work packages differ depending on the programming period and contain the main activities of the research programme. The last work package (WP 7) covers all interactions with society, including education, communication and public participation in the long-term research programme.

Within each work package and, specifically, WP 3-6, different research tasks have been identified. The prioritisation of these research tasks is based on the prioritisation of the components of the overall disposal system to which it is related (society, biosphere, surrounding rock formations, host rock, engineered barrier, Figure 1.2, Verhoef et al., 2021). Therefore, a task will generally receive the same priority as the component to which it is related. In cases where a lower priority was given, the argumentation is provided within the task description in (Verhoef et al., 2021). Additionally, there were specific tasks that are characterised below as a unique opportunity for collaboration, or support strategic policy needs, and can involve participation in a co-funded (international) research activity or international collaboration. Below is a list of external research tasks that have been addressed in COPERA (2020 - 2025). For a more comprehensive list, we refer to Verhoef et al. (2021) and Appendix 3.

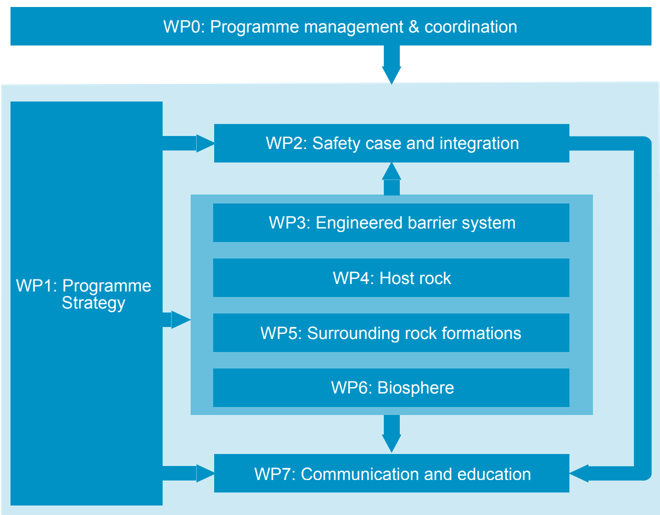


Figure 1.1) Overview and relation between the programme's work packages. Figure from Verhoef et al. (2021).

Cost estimate for a GDF in rock salt – Priority 1. In this task, the total cost of constructing a GDF in rock salt was estimated using the SSK method which is the standard approach for cost estimates used in the Netherlands for large construction works. Carrying out a new cost estimate is of particular significance, considering that the most recent cost estimate for a repository in rock salt is over 20 years old (Grupa and Jansma, 1999). Furthermore, the cost estimate is used to set COVRA's waste fees, which generate funds for all of COVRA's waste management work. The results of this project have been published in reports on our website (Herold and Leonhard, 2023b; Oudenaren and Browning, 2023).

Review of different disposal concepts in rock salt - Priority 1. As part of this task, a new disposal concept was developed, reviewed and optimised where possible. The optimised disposal concept was then used as an input for the cost estimate. The final report is published on our website (Herold and Leonhard, 2023a).

Development/improvement of numerical methods & tools for modelling coupled processes - Unique Opportunity: This task, part of the EURAD research programme, aimed to investigate how uncertainties and sensitivities in typical numerical safety assessment models can best be analysed with classical and modern methods.

Uncertainties related to human aspects - Unique Opportunity: In this task, the objective was to identify, characterise and assess the significance and evolution of uncertainties related to social, economic and other human aspects that are deemed relevant to safety and the decision-making process. This task, like the previous task, is a part of the EURAD research programme. It was finished at the start of 2023 and the results have been published on the EURAD website (Dumont et al., 2024).

Waste package for HLW - Priority 2: In previous Dutch disposal concepts for rock salt, encapsulation of the HLW in an overpack to create a robust waste package for disposal was not considered (See Appendix 1), unlike in the German disposal concept (Bollingerfehr et al., 2018a). The task addressed the advantages and disadvantages of utilisation of a self-shielded waste package for disposal. It then investigated the feasibility of designing a self-shielded waste package for a repository in rock salt, which would ensure complete containment during the period when there may be significant advective transport of brine in the GDF. The task has been finished and the results are published on our website (Wunderlich et al., 2023).

Waste package for (TE)NORM – Priority 2: Currently, (technologically enhanced) naturally occurring radioactive materials ((TE) NORM) are stored in standardised DV-70 containers (see section 6.2.4), which are suitable for above-ground storage. However, it is uncertain whether the DV-70 container can also be utilised as the disposal package within a GDF in rock salt. The objective of this task was to examine whether the standardised DV-70 container can be used in a highly saline environment. If not, alternative container options would be explored. Additionally, the possibility of utilising (TE)NORM in a different, more practical manner was also investigated. The task has been finished and the results are published on our website (Browning and Grupa, 2023).

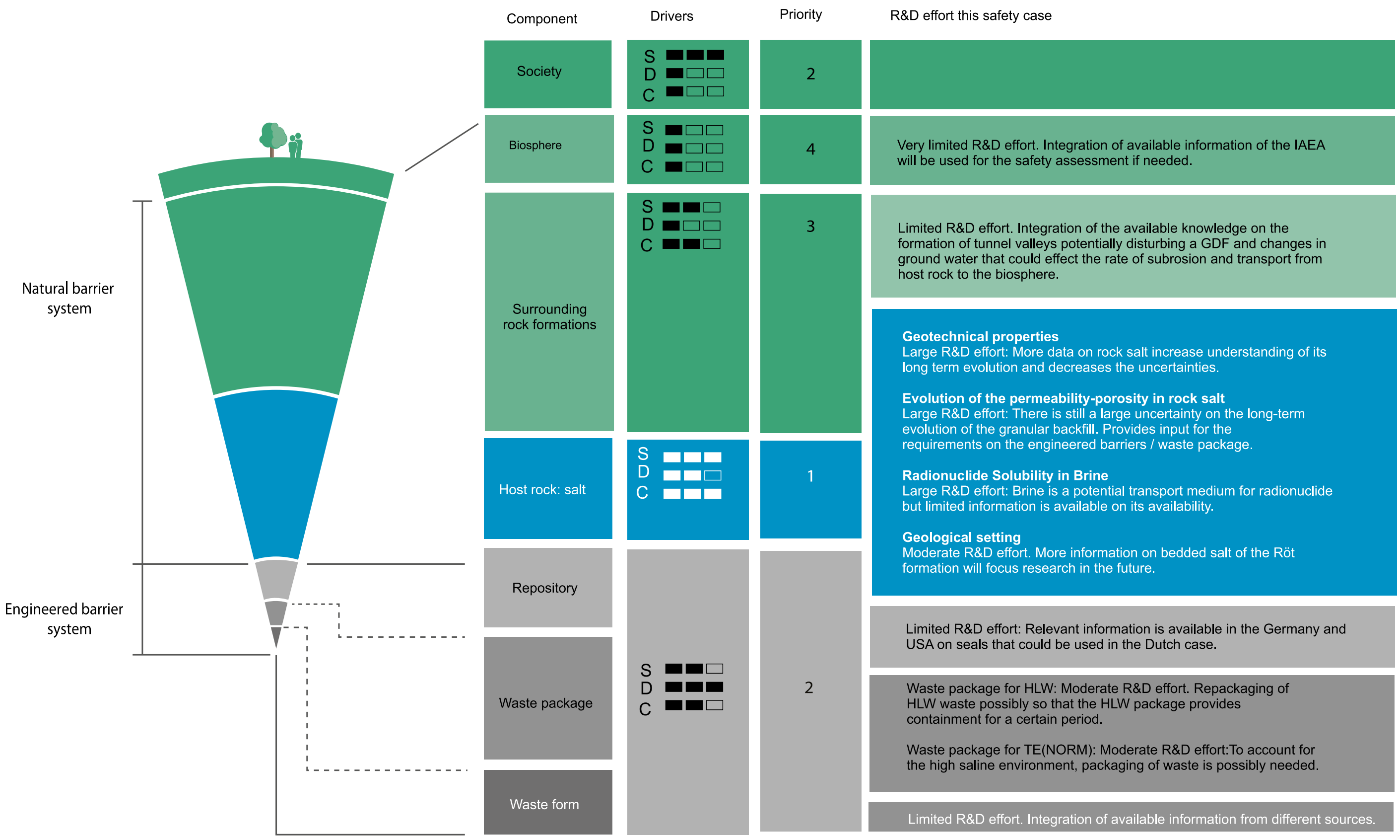


Figure 1.2) An overview of the different tasks within the COPERA (2020 - 2025) and their relationship with the different components within a repository. Figure from Verhoef et al. (2021).

Solubility of radionuclides in brine – Priority 1: The mobility of radionuclides in a saline environment (with and without concrete packages) depends mostly on their solubilities. As part of this task, a programme was developed to quantify the solubility of radio-nuclides under different high-saline environments. This task has been finished and the report can be found on our website (Oving and Meeussen, 2024).

Geotechnical properties of salt - Priority 1: In this task, the objective was to establish a database containing the thermal, hydrological and mechanical properties of rock salt. These properties were utilised in the safety assessments. While the primary focus was on rock salt from the Zechstein formation, other types of salts and formations can also be incorporated into the database. This task has been finished: the report and the database with the salt properties have been published on our website (Hunfeld et al., 2023).

Evolution of the permeability-porosity in rock salt - Priority 1: Understanding the long-term (1,000 – 1,000,000) evolution of the permeability and porosity of the granular salt backfill that would be used in the GDF tunnels is essential for assessing the long-term safety of a repository in rock salt. This task is expected to be finished in 2025, but the first report (Oosterhout et al., 2022) has been published on our website and a second will follow (Oosterhout, 2023). The results of Oosterhout et al. (2022) have been implemented in the safety assessment by Nicholas and Thatcher (2023).

Gas Production - Priority 2: Gas production can have an impact on the compaction of the backfill material. For instance, it has the potential to delay the compaction process. The objective of this task was to examine the extent of gas production that can occur within a repository located in rock salt. This task has been finished and three reports have been published on our website, describing the waste specification used as an input for the gas production calculations (Watson, 2023), the equations used to calculate gas production (Benbow et al., 2023a) and the expected gas production in specific cases (Benbow et al., 2023b).

Brine availability - Priority 1: In rock salt, brine is an important medium for the transportation of radionuclides. This task aimed to assess the availability of brine within a repository in rock salt, identify the factors that influenced this availability, including temperature gradients due to heat-generating waste, and explore the feasibility of developing a numerical model for predicting brine availability. This task was part of DECOVALEX, and the results are published on the DECOVALEX website (decovalex.org). A summary of the results has been published in the journal Geomechanics for Energy and the Environment (Kuhlman et al., 2024a).

Bedded Salt of the Röt Formation - Priority 1: In the Netherlands, there are several salt deposits of potential interest for GDF development, including one within the Röt formation. However, there was a lack of reliable information on the depth and thickness of the salt deposits within this formation. The objective of this task was to map and characterise the Röt formation, focusing specifically on the rock salt contained within it. This task has been finished and results have been published on our website (Altenburg, 2022).

Diapirism Rates in the Netherlands (Past-Present-Future) - Priority 3: Diapirism is the process by which a salt dome rises upwards through the surrounding sedimentary formations towards the surface. It is one of the processes that could potentially lead to the release of radionuclides in the far future. Therefore, it is important to understand the historical diapirism rates in the Netherlands, the current rates and the expected rates in the future. The aim of this task was to investigate and provide insights into these aspects. Two reports have been published on the past rates of diapirism in the Netherlands as part of this task (Almalki, 2023; Lauwerier, 2022).

Understanding Past, Present, and Future Subrosion Rates in the Netherlands - Priority 3: Subrosion refers to the dissolution of salt by groundwater flowing in surrounding sediments. Due to subrosion, the salt formation within which the GDF is located will gradually dissolve, which could eventually lead to contact between the wastes and the groundwater system. Thus, subrosion could potentially result in the release of radionuclides into the geosphere. It is, therefore, an important process to understand for the long-term safety of the GDF. This task aimed to gather information on past and current subrosion rates in the Netherlands and evaluate how subrosion rates can be predicted for the future using numerical models. Two reports have been published on our website on past subrosion rates in the Netherlands (Almalki, 2023; Lauwerier, 2022).

FEP-Catalogue and Scenario Development for a generic HLW Repository in a Salt Dome – Unique opportunity: To understand the possible evolution of the disposal system, the first step is to create a generic list of the Features, Events and Processes (FEPs) that can affect it over time. Using these FEPs, different types of scenarios (evolutions) can be derived in a systematic way to form the basis for the performances assessment. This work created a generic FEP list that was subsequently used to derive different types of scenarios for a GDF in a salt dome in the Netherlands. The report (Lommerzheim, 2023) including the FEP list has been published on our website.

1.7 Structure of this conditional rock salt safety and feasibility study

This document describes the work carried out on disposal in salt in parallel with the clay safety case and feasibility study (Neeft et al., 2024b) during the 2020 – 2025 period of the COPERA research programme. As a prelude to presentation of the results of this research programme and of the current disposal concept, Chapter 2 explains the general concept of geological disposal and the life cycle of a geological disposal facility. In addition, it provides a short historical overview of previous disposal concepts in rock salt and an international perspective on the current state of geological disposal development in different countries, with a focus on rock salt. This is followed in Chapter 3 which describes the approach used to assess post-closure safety, the structure of the safety case, the different requirements for geological disposal of waste and the contributions to safety of different components in a multibarrier system with rock salt host rock. Chapter 4 describes the inventory of wastes expected to be disposed of in 2130 and how this waste can potentially be emplaced in a GDF in rock salt. Chapter 4, also discusses the cost of a repository in rock salt, including potential optimisations. Thereafter, we discuss in more detail the different components of the GDF multibarrier system, namely the natural barrier (Chapter 5) and the engineered barriers (Chapter 6). For each of these barriers, the uncertainties that are relevant for long-term safety are described. Chapter 7 describes both the normal and alternative evolution scenarios for the geological disposal system over the next 1 million years. A central part of the safety case is the quantitative safety assessment, in which potential future doses or risks are evaluated. Chapter 8 shows the results of this safety assessment. Chapter 9 discusses whether the expected long-term safety of the salt GDF concept modelled in the current work justifies proceeding to further stages in the geological disposal programme of the Netherlands and Chapter 10 gives a justification for the prioritisation of future research. References are listed in Chapter 11. A series of Appendices gives more detailed information on some of the topics introduced at a broader level in the main text.

2. Geological disposal

This chapter describes the concept of geological disposal of radioactive waste, covering the objectives and showing how components in the multi barrier system contribute to post-closure safety. In addition, it describes the practical activities to be carried out throughout the long period from planning through to the closure of the GDF, which may last several decades to a century or more. It also provides an international perspective on the status of geological disposal, with a focus on rock salt.

2.1 Disposal objectives

Geological disposal aims to remove hazardous material from the immediate human and dynamic, natural surface environment to a stable geological environment deep underground where it will be protected from disturbance by natural or human processes. Deep geological repository concepts and designs aim to provide an initial period, usually many thousands of years, when all the radioactivity in the wastes is completely contained within the engineered barriers of the disposal system. However, the wastes, their packaging and other containment materials will degrade slowly, and even the most stable geological environments will eventually change with the passage of geological time. Complete containment of all radionuclides for all time is thus not feasible. However, the radioactivity of the wastes decreases with time, by natural radioactive decay, and the engineered and natural barriers in the system delay any migration through to the human environment, so allowing further decay as well as dilution and dispersion.
Over the very long term, the safe performance of a disposal system thus depends on the balance of the rates of radioactive decay and the processes involved in radionuclide mobilisation and interaction with the rocks and groundwaters of the natural barrier system.

The basis of geological disposal of waste has been firmly established internationally for the last 45 years on the concept of the so-called 'multi-barrier system', whereby a series of engineered and natural barriers act in concert to isolate the wastes and enclose the radionuclides that they contain (IAEA, 2011a):

Isolation: removes the wastes safely from direct interaction with people and the environment. To achieve this, locations and geological environments identified for a GDF must be deep, inaccessible and stable over long periods (for example, where rapid uplift, erosion and exposure of the waste will not occur) and should be unlikely to be drilled into or excavated in a search for natural resources in the future.

Containment: means retaining the radionuclides within the multi-barrier system until natural processes of radioactive decay have reduced the potential hazard considerably. For many radionuclides, a multibarrier system can provide total containment until they decay to insignificant levels of radioactivity within the waste packages. However, the engineered barriers in a multibarrier system will degrade progressively over hundreds and thousands of years and eventually lose their ability to provide complete containment. Because some radionuclides decay extremely slowly and/or are mobile in water, their complete containment is not possible. Assessing the safety of geological disposal involves evaluating the mobilisation and transport of these radionuclides and their potential impacts, if they eventually reach people and the surface environment, even in extremely low concentrations and many thousands of years into the future.

The inner components of the multibarrier system for geological disposal (see Figure 2.1) comprise the engineered barrier system (EBS). This includes the waste forms, containers, any overpack around the containers, buffer materials directly surrounding the waste packages and backfill material filling the void spaces in tunnels or galleries. These are all within the so-called 'near-field' of the disposal system – a loosely defined term that includes those

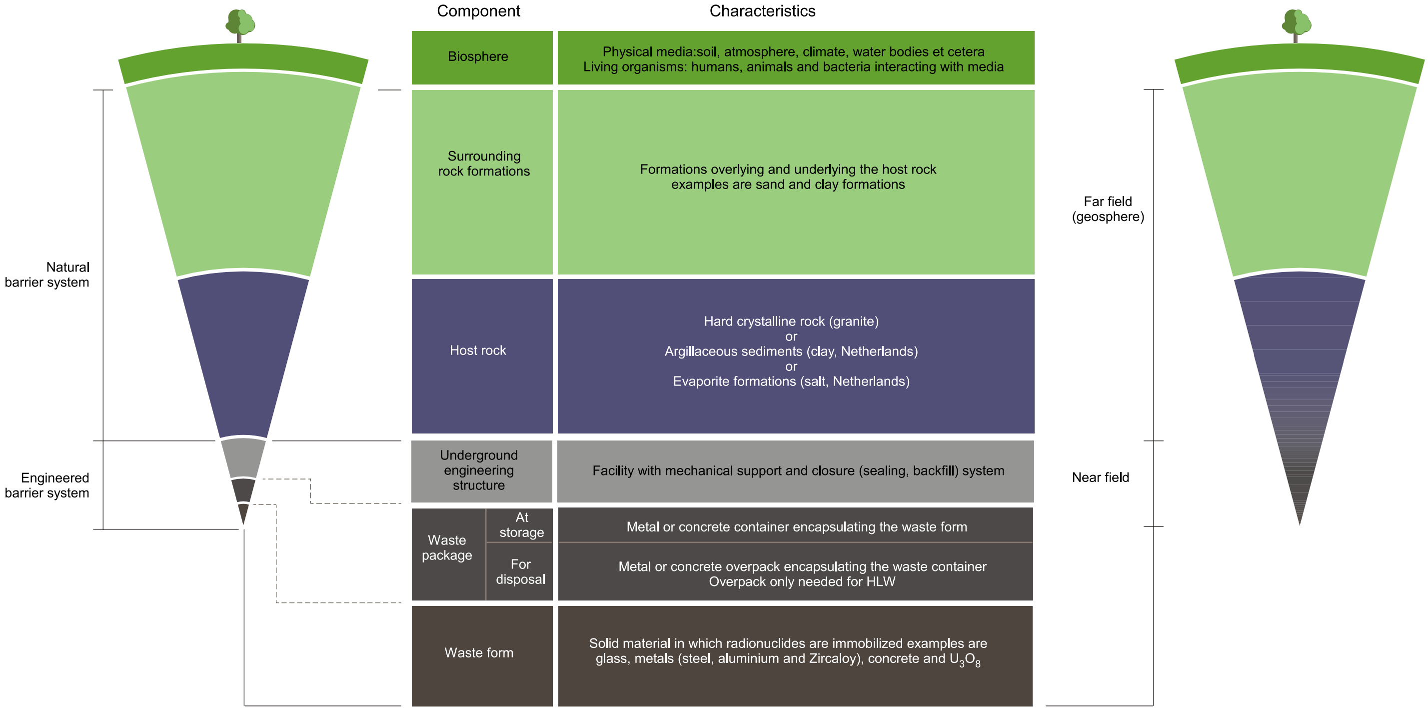


Figure 2.1) Components of multibarrier systems at the time of completion and closure of the geological disposal facility, adapted from the OPERA Safety case (Verhoef et al., 2017).

core regions of the GDF that are involved in, or physically or chemically affected by, the evolution and degradation of the waste and its containers. The 'far-field' is comprised of the natural barriers: the host rock and the surrounding rock formations. Tunnel and shaft seals are further engineered barriers, but generally lie outside the near - field. Each of the components in the multi-barrier system contributes to ensuring isolation and containment.

The relative contributions to the safety of the various barriers at different times after the closure of a disposal facility and the ways that they interact with each other depend upon the design of the disposal system. The design itself is dependent on the geological environment in which the facility is constructed. Consequently, the multi-barrier system can function in different ways at different times in different disposal concepts. The multi-barrier system shown in Figure 2.2 distinguishes between the EBS and the surrounding natural barriers, comprising the host rock (purple) and surrounding rock formations (green). Box 2-1 discusses the declining radiotoxicity of wastes as a function of time, showing that this radiotoxicity is reduced by factors of many thousands over a period of some hundreds to a few thousands of years, depending upon the waste type. Providing safe isolation and containment over this 'early' period of the highest hazard potential is perhaps the most important role of a multibarrier system.

The natural barriers in the present safety and feasibility study are rock salt (evaporite) deposits and their surrounding rock formations. How each of the barriers in a multi-barrier system for a repository in rock salt contributes to the isolation and containment is shown in Table 2.1.

The operational life of the GDF will be many decades, depending on how much waste exists in storage when the facility becomes operational, and how much will be produced thereafter. An essential aspect of the geological disposal system is that it provides protection and safety in a completely passive manner. After a GDF is completed and closed, no further actions are required from people to manage the wastes. Over immensely long times, the engineered barriers and the wastes become part of the deep, natural environment, with conditions in the salt host rock returning to those of the natural, undisturbed environment before the GDF was constructed (see Chapter 7).

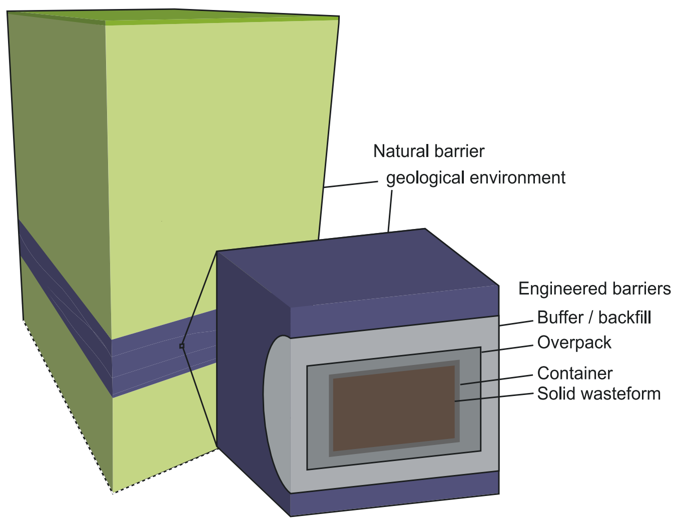


Figure 2.2) General concept of the multiple barrier system for geological disposal of radioactive waste in a salt formation adapted from Chapman and Hooper (2012) and adapted as presented in the OPERA Safety case.

Barrier component	Generic contributions to post-closure safety
Waste form: The solid waste material.	▪ Provide a stable, low-solubility matrix that limits the rate of release of the majority of radionuclides by dissolving slowly in brine that come into contact with it.
Waste container: Generally metal or concrete: for higher activity wastes the container might have an outer metal overpack.	▪ Protect the waste form from physical disruption (e.g., by movement in the bedrock). ▪ Prevent groundwaters from reaching the waste form for a period of time. ▪ Act as a partial barrier limiting the movement of water in and around the waste form after corrosion has breached the container. ▪ Control the redox conditions in the vicinity of the waste form by corrosion reactions, thus controlling the solubility of some radionuclides. ▪ Allow the passage of any evolved gases from the waste form out into the surrounding engineered barrier system.
Buffer or backfill: Around the waste container, separating the package from the rock. In many designs, a natural clay buffer (bentonite) is used.	▪ Protect the waste container from physical disruption (e.g., by movement in the bedrock). ▪ Control the rate at which groundwaters can move to and around the waste container (e.g., by preventing flow). ▪ Control the rate at which chemical corrodents in groundwaters can move to the waste container. ▪ Condition the chemical characteristics of groundwater and pore water in contact with the container and the waste form so as to reduce corrosion rate and/or solubility of radionuclides. ▪ Control the rate at which dissolved radionuclides can move from the waste form out, into the surrounding rock. ▪ Control or prevent the movement of radionuclide-containing colloids from the waste form into the rock. ▪ Suppress microbial activity in the vicinity of the waste. ▪ Permit the passage of gas from the waste and the corroding container out into the rock.
Mass backfill: For access and service openings. Various natural materials and cements in different parts of the GDF, chosen to be compatible with the geological environment.	▪ Restore mechanical continuity and stability to the rock and engineered barrier region of the facility so that the other engineered barriers are not physically disrupted (e.g., as a clay buffer takes up water and expands). ▪ Close voids that could otherwise act as groundwater flow pathways within the facility. ▪ Prevent easy access of people to the waste packages.
Sealing systems: Emplaced locally in tunnels and shafts at key points in the system.	▪ Cut off potential fast groundwater flow pathways within the backfilled facility (e.g., at the interface between mass backfill and rock). ▪ Prevent access of people into the backfilled facility.
Natural geological barrier: The host rock in which the waste emplacement tunnels or caverns are constructed and all the overlying geological formations, which might be different to the host formation.	▪ Isolate the waste from people and the natural surface environment by providing a massive radiation shield. ▪ Protect and buffer the engineered barrier system from dynamic human and natural processes and events occurring at the surface and in the upper region of the cover rocks (e.g., major changes in climate, such as glaciation). ▪ Protect the engineered barrier system by providing a stable mechanical and chemical environment at depth that does not change quickly with the passage of time and can thus be forecast with confidence. ▪ Provide hydrogeological rock properties that, together with low hydraulic gradients, limit the rate at which deep groundwaters can move to, through and from the backfilled and sealed facility, or completely prevent flow. ▪ Ensure that chemical, mechanical and hydrogeological evolution of the deep system is slow and can be forecast with confidence. ▪ Provide properties that retard the movement of any radionuclides in groundwater – these include sorption onto mineral surfaces and properties that promote hydraulic dispersion and dilution of radionuclide concentrations. ▪ Allow the conduction of heat generated by the waste away from the engineered barrier system so as to prevent unacceptable temperature rises. ▪ Disperse gases produced in the facility so as to prevent over pressures leading to mechanical disruption of the engineered barrier system.

Table 2.1) Contributions to post-closure safety of the principal barriers in multibarrier systems adapted from Chapman and Hooper (2012) as presented in the OPERA Safety case.

2.2 Different options for the geological host rock

Over the last 45 years, geological disposal of waste has developed from a concept to reality, with the world's first GDF for spent nuclear fuel currently being licensed for operation in Finland, and others in advanced stages of siting and development in France, Sweden, Switzerland and China. In that period, most countries have focussed their attention on three broad groups of rocks as host formations:

- **Hard 'crystalline' rocks:** such as granite, gneiss and other metamorphic or plutonic rocks can be extremely stable, especially with respect to future erosion (e.g., by ice sheets) and are generally easy to construct in, allowing large, stable underground openings to be used for waste emplacement. Extensive worldwide studies have been performed on hard crystalline rocks of varying compositions and ages, including ancient Pre-Cambrian shield rocks (e.g., in Canada, Sweden and Finland).
- **Argillaceous sedimentary rocks**: such as clays, mudstones and marls can provide a high level of physical containment owing to their low permeability, which can lead to their pore-waters remaining essentially immobile, with little or no groundwater flow occurring on timescales of interest for post-closure safety. This characteristic has been demonstrated in the Jurassic and Paleogene clay formations being targeted in France, Switzerland and Belgium, using environmental isotopic and chemical compositional profiles of their pore waters (Mazurek et al., 2008). A parallel study, describing COVRA's current preliminary safety and feasibility study for clay as a host formation for the Dutch waste inventory, is published together with this report (Neeft et al., 2024b).
- **Evaporite formations**: are principally dome and bedded salts, with the host rock of interest being halite. These formations, although they can be structurally and compositionally complex in the case of domal salts, are often cited as providing ideal containment properties. In homogeneous regions of either bedded or dome formations, there is essentially no fluid that is sufficiently mobile to transport radionuclides to the surrounding rock formations. These formations were the first to be identified as potential hosts for radioactive waste disposal, as long ago as 1950 (NRC, 1957) and have been studied in the Netherlands as well as several other countries, including Germany, Italy and the USA.

Each of these groups has its own strengths, advantages and challenges with respect to containment and isolation. There is also a wide range of variability of these strengths within any one group and between specific sites that have been investigated for disposal internationally. It is recognised that safety can be achieved by different balances of these characteristics and strengths of the safety functions of the natural, geological barrier, so that there is no unique solution that is the 'best rock' or the 'best environment'.

Over the last 45 years, a range of both generic and site and host rock specific GDF designs has been developed around the world and a range of materials proposed for various components of the EBS. Both the design and the materials selected depend upon the category of waste to be disposed of and the geological environment under consideration. In some countries, there is a preference for a single GDF for all wastes that require geological disposal, with separate sections that have different designs to accommodate the different wastes. Many further design considerations are involved in fitting a generic concept to a specific site, including the ability to be flexible and adapt design, depth and geometry to local conditions by exploiting the best volumes of rock or avoiding certain geological features. This provides scope for optimising operational procedures and costs, accommodating local community requirements and minimising the environmental impacts of construction and the operation of both surface facilities and the GDF.

2.3 Activities for the lifecycle of a geological disposal facility

The major activities through the lifecycle of a geological disposal facility (Fig. 2.3) are site selection, construction, operation and

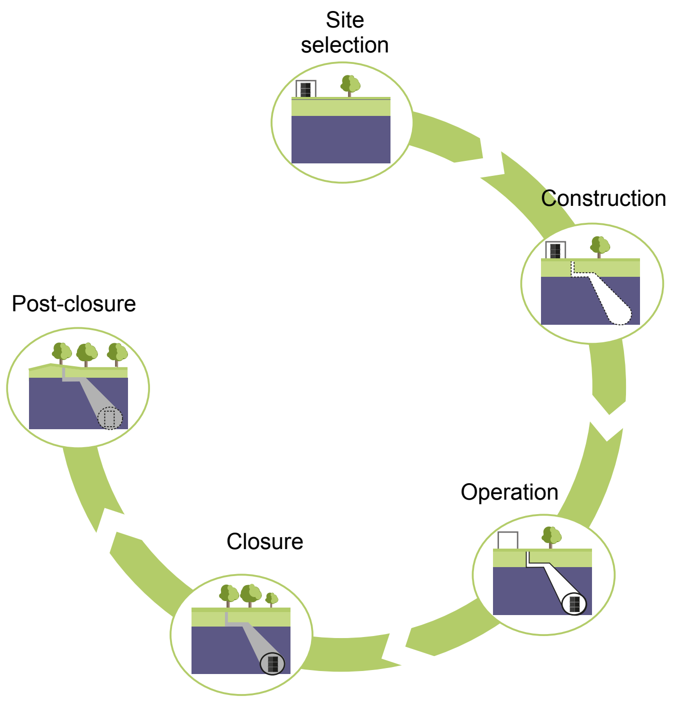


Figure 2.3) The life cycle of a GDF.

closure. There is relevant international experience in design and/or implementation for each of these phases. The role of a safety case in each stage is described in Chapter 3. In this report, we look at the potential Dutch approach to each stage and, where possible, at an international example related to rock salt.

2.3.1 Site selection

Selecting a suitable location for the Dutch GDF is an activity that lies decades in the future. At the request of the Dutch Ministry of Infrastructure and Water Management, the Rathenau Institute is developing policy advice on how the Netherlands can best organise the decision-making process for siting the GDF (Cuppen, 2022); will be based on extensive European experience on radioactive waste governance (van Est et al., 2023). The institute has already concluded that aiming for decision-making around 2100 negatively impacts people's perception of the need for actions today, making public participation a complex challenge. They advise establishing criteria allowing reservation of potential locations for a GDF and emphasise the importance of embedding the role of public participation within research and various national and decentralized political decision-making processes (Dekker et al., 2023).

COVRA assumes that the siting strategy will be based on a volunteering model incorporating stakeholder involvement at all stages. It would be technically guided at the outset only insofar that clearly unsuitable regions are excluded at the start. For example, a relevant geological criterion could be that candidate sites should have a potential host rock formation that shows no evidence of past local, deep glacial erosion, because potential similar future events could impact post-closure safety. It is considered important today that the eventual siting strategy will incorporate the flexibility to evaluate objectively any proposal that might emerge from volunteer communities or regions. A visualisation of what a site selection process might involve is described below, but no process has yet been established in Dutch policy. We use examples from various disposal programmes, including the most recent international developments and their associated time frames.

Many national geological disposal programmes have suffered setbacks and delays because their GDF siting projects have proved difficult or impossible to implement. In general, this is because it has proved hard for implementers to prepare and present the appropriate mix of technical, societal and political inputs that is required to achieve consensus amongst the stakeholders. However, the recent success of several national programmes indicates that this problem can be overcome, largely by recognizing that siting needs to be an open and inclusive process for all parties concerned.

Gathering technical information to help identify suitable regions and, eventually, specific locations, involves iterative programmes of data evaluation and site investigation to characterise the geological environment in sufficient detail. At each stage, information is generated in progressively more detail, to refine the design of the GDF and to improve the system modelling that is central to the post-closure safety assessment. Generally, GDF design and safety evaluation will go through several cycles of development, as more, and more specific, information becomes available. The basic geological and geotechnical characteristics of the host rock and surrounding formations must be adequately understood, and, for the safety case, an integrated picture must be built up of the dynamic evolution of the deep environment during tens and hundreds of thousands of years.

This requires the compilation and interpretation of observations made by many field, laboratory, and remote sensing techniques, at a wide range of spatial scales. It will also involve the use of data from other geotechnical, survey and exploration activities in the Netherlands, and from dedicated deep drilling, testing and sampling in boreholes. Identifying, scoping and managing technical uncertainties will be a key activity within the siting programme. Underground Research Facilities (URFs) can be involved in a site characterisation process. Such a facility, at the location of a GDF, could be expanded into a disposal facility if sufficient confidence is available that radioactive waste can safely be disposed of at that site. URFs in salt formations have operated over many decades since the 1960s, with the principal work being carried out at Project Salt Vault (Kansas, USA) and at Asse and Gorleben, in Germany.

2.3.2 Construction

Construction works start after a construction license has been obtained from the relevant authority. For the EU Member States, the procedure also includes consultation with, and obtaining an opinion from, the European Commission, as required by the Euratom Treaty (Carbol et al., 2022).

Experience in constructing a GDF in rock salt is available: the Waste Isolation Pilot Plant located in New Mexico, USA, was constructed in a bedded rock salt formation approximately 650 m below the surface and has been in operation since 2000. In addition to the construction of a GDF, there is also ample experience in the construction of mines and other open spaces in salt formations. Numerous conventional mines in rock salt and other evaporite formations are currently operational worldwide, for example the Boulby potash mine in the UK and the salt mine at Heilbronn Germany, which has a production capacity of just under 5 million tons annually. These can provide insights into construction techniques, best practices et cetera, that could potentially be used in a GDF.

Figure 2.4) A road header used in a salt mine. Photo by Jeroen Bartol, COVRA

In both conventional salt mines and in WIPP, road headers (see Figure 2.4) and continuous miners are commonly used for tunnel construction or expansion, demonstrating the considerable experience in civil and mining engineering that can be leveraged when constructing a GDF in salt. The Netherlands also has experience in salt mining and constructing large open spaces (caverns) within salt domes for gas and oil storage. For both salt mining and the construction of large caverns in the Netherlands, a technique known as dissolution mining is generally used. With dissolution mining, a solution (typically water or brine), is injected into the formation through a well. This water dissolves the salt, and the resulting brine is pumped back to the surface. However, dissolution mining is not considered suitable for constructing a GDF, as the precise requirements on dimensions and shape of rooms cannot be achieved with this technique.

Other examples of radioactive waste disposal in salt (the Asse and Morsleben repositories in Germany) are not considered representative of how a GDF would be constructed today.
Both were initially conventional commercial salt mines before they were used for the disposal of low and intermediate level radioactive waste. Both sites were thus not sited or designed using current criteria for a GDF. In mines, the goal is to remove as much salt as possible and this results in significant disturbance to the host rock close to the outer sections of the salt domes. In a purpose built GDF, disturbance to the host rock will be minimised and the repository will be constructed away from the outer regions of the salt deposits. In the case of the Asse repository, the significant disturbance of the host rock resulted in the inflow of small amounts of brine into the open facility (Minkley, 2009), and a decision has been made to retrieve all the waste. The Morsleben repository is in the process of being backfilled and decommissioned with the waste in place, which will eventually lead to its final and safe closure.

2.3.3 Operational phase

There is growing international experience in operating underground disposal facilities that can be applied to any type of GDF. This includes experience from the operational WIPP facility where intermediate level waste (ILW) and transuranic radioactive waste is disposed of, and repositories for short lived low and intermediate level waste (SL-LILW) in granitic host rocks in Hungary, Finland and Sweden. SL-LILW is also currently disposed of in surface facilities in many European countries, such as France, Spain, Bulgaria and the Czech Republic. All these disposal facilities, as well as surface storage facilities (such as COVRA's storage facilities at Nieuwdorp), use approaches and techniques that will be required in a GDF for the handling and emplacement of waste packages and for overall active facility operational management.

2.3.4 Closure phase

The disposal tunnels with the emplaced waste may either be continuously backfilled after emplacement of waste or backfilled later in the closure phase. Plugs in shafts and other accessways may be required to prevent ingress of water into the disposal facility and seals may be needed to maintain the isolation of the waste.
Methods for the closure of a disposal facility have also been demonstrated in trials in URLs. In salt, for example, there multiple full-scale experiments have been performed in the Morsleben repository in preparation for its closure (Preuss et al., 2002).
These full scale tests are complemented by many small-scale laboratory experiments performed to understand the evolution of the material used to fill the open spaces (Friedenberg et al., 2022a; e.g., Oosterhout et al., 2022).

Depending on the design concept, some of the installations in the underground and surface facilities (e.g., cranes used for package handling) need to be dismantled and removed before closure and some components may need to be either decontaminated or disposed of as active waste. Site remediation activities allow the site to be returned to normal use (Carbol et al., 2022). There are currently no closed GDFs, but there are closed surface disposal facilities, such as the Manche disposal facility in France, which operated for 25 years (ANDRA, 2020).

2.3.5 Post-closure phase

The post-closure phase begins with a period of active institutional control over access to and activities at the disposal site. This is primarily intended to increase the level of confidence in the isolation and containment provided by the multibarrier system. Active institutional control is not required to assure long-term safety, as the multiple barriers function as an entirely passive system after closure, but it does help to prevent or minimise the probability of inadvertent human intrusion into the multibarrier system, so long as monitoring is maintained. The active institutional control period may extend over several decades, depending on the national regulations and licence requirements in place. Eventually, at some point in time to be agreed by future generations, active institutional control will be terminated.

Passive institutional control primarily consists of record keeping and preserving knowledge on the waste, the disposal facility and the site. Propagating this knowledge into the future will require a range of provisions to be made with local, national and international organisations. The longer that knowledge of the GDF can be preserved and communicated, the greater the reduction in the hazard potential of the wastes by decay and the lower the likelihood and consequences of inadvertent human intrusion.

The scope and duration of institutional control must be defined in national regulations and requirements set out by the regulatory body, but in many countries such requirements have not yet been fully defined (Carbol et al., 2022). The closed Dutch chemical waste disposal facilities such as Volgermeerpolder, near Broek in Waterland (Berkers et al., 2023), are analogues of demonstrated long-term institutional control.

2.4 Background

This safety case is built not only on work that has been performed as part of the COPERA research programme, but also on research performed as part of previous Dutch programmes. Some of the choices made in developing the COPERA disposal concept are based on previous evaluations of potential disposal concepts.
While research on disposal in rock salt in the Netherlands has been extensive, we only briefly discuss how the Dutch salt disposal concept has evolved. For more information, we refer to Appendix 1.

In total, five different research programmes in the Netherlands have evaluated disposal in rock salt, starting in 1972 with the ICK (1972 – 1979) research programme, followed by OPLA (1985 – 1993), CORA (1995 – 2001), OPERA (2010 – 2017) and the current research programme COPERA (2020 – 2025) (Fig. 2.5). It should be noted that research on disposal in rock salt was very limited in the OPERA research programme: no new disposal concept was developed, nor was a previous one updated.

For the disposal of LILW or similar types of waste, disposal rooms, bunkers, or large caverns were the options considered up to 1993 (OPLA). The disposal room and bunker were to be constructed using explosives in combination with conventional mining. For a large cavern, on the other hand, construction by solution mining was considered. After construction, the emplacement of LILW packages was envisioned to take place in a variety of ways (e.g., truck, crane, tipping), but all in an uncontrolled fashion. Uncontrolled disposal of waste (i.e., when packages are not regularly stacked and supported) makes retrieval (if needed) difficult, and perhaps even impossible. Consequently, when retrievability became a requirement in 1993 (Minister van volkshuisvesting ruimtelijke ordening en milieubeheer, 1993), the approach to LILW disposal proposed prior to 1993 was no longer considered suitable. As CORA focused on the disposal of HLW, the COPERA (2020 - 2025) disposal concept (Chapter 4) will be the first Dutch disposal concept in rock salt to take the retrievability requirement into account.

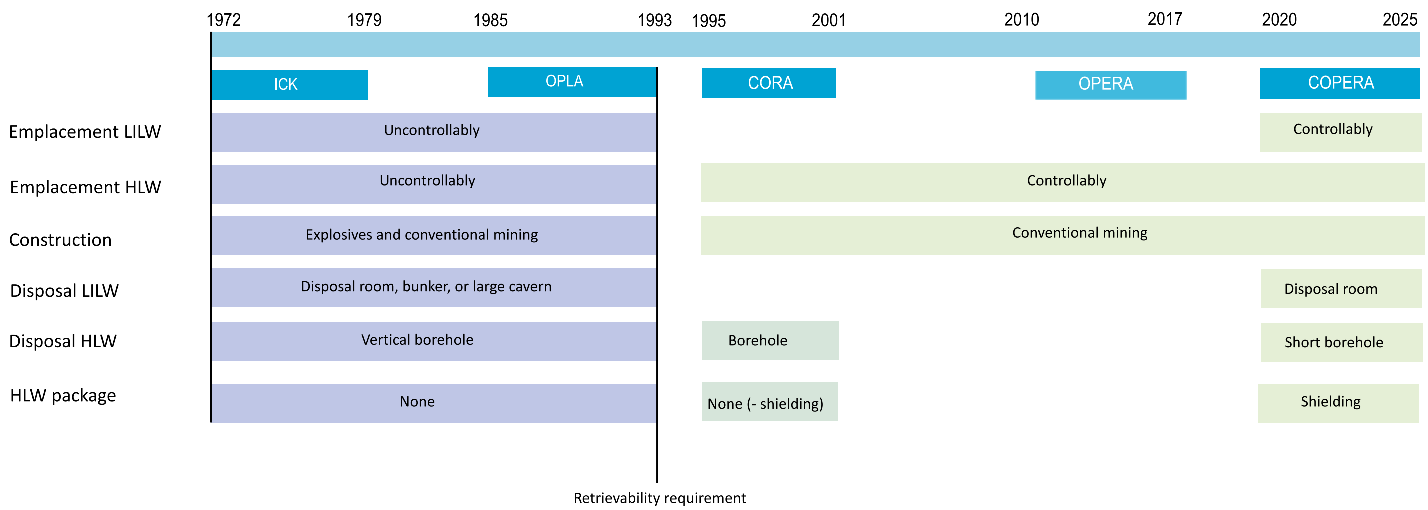


Figure 2.5) Simplified overview of how the Dutch disposal concept for disposal in rock salt has evolved over time During CORA, 2 different disposal concepts were developed for the disposal of HLW: 1 with horizontal boreholes (METRO) and HLW package and 1 with vertical boreholes (TORAD – B), in which in a non-shielding disposal overpack was used. This overpack was required to ensure that the waste was retrievable. See also appendix 1.

For the disposal of HLW, both vertical and horizontal boreholes were proposed in the OPLA (vertical) and CORA (Vertical: TORAD - B; Horizontal: METRO) disposal concepts. The procedure by which the HLW was to be emplaced changed over time, from lowering the waste into the borehole and subsequently releasing the package in an uncontrolled manner (e.g., OPLA), to controlled emplacement using cranes (e.g., TORAD-B) or even special vehicles (e.g., METRO). The latter two disposal concepts also demonstrated that the retrievability of waste in rock salt was possible.

In addition to the changes in the disposal method, the proposed construction method of the GDF also changed over time. The first studies proposed a combination of explosives, conventional mining or leaching, but later only conventional mining was considered (from CORA onwards). In the current disposal concept, the repository is intended to be constructed using conventional mining techniques, to limit damage to the host rock.

The salt disposal concepts changed significantly over the past decades, but none of them considered the use of HLW package, except for the TORAD-B concept. The functions of the HLW package in the TORAD-B concept were only to ensure the retrievability of the waste and provide minimal radiation shielding. The absence of a HLW package differs from the German disposal concepts (e.g., Bollingerfehr et al., 2013), in which the self-shielding POLLUX-10 HLW package is used. This led to consideration of the potential advantages and disadvantages of using a self-shielding HLW package in the Dutch disposal concept (Verhoef et al., 2021).

2.5 Salt repositories for disposal of radioactive waste

While a decision on the disposal of radioactive waste in the Netherlands may still be decades away (Ministerie van Infrastructuur en Milieu, 2016), three repositories in rock salt already exist elsewhere. As already mentioned in previous chapters, two of these are in Germany—the Asse II mine and the Morsleben repository. The third repository, the Waste Isolation Pilot Plant (WIPP), is in the United States, in New Mexico. WIPP has been accepting waste for over two decades and is regarded as a pioneering project for radioactive waste disposal in salt formations. In the sections below, we briefly discuss the history and status of these repositories, and the lessons learned.

2.5.1 Asse II mine

The Asse II mine (Fig. 2.6) is located south-east of Braunschweig in Germany and was constructed at a depth of 765 m between 1906 and 1908 in a salt dome. Initially, the mine was constructed to mine potash (until 1925), but this was later extended to include halite (1916 - 1964). Following the end of the mining activities in 1964, after being bought by the federal government, the Asse II mine was converted into a test facility for low and intermediate radioactive waste disposal. In total, 125,787 drums of mostly 100 and 400 litres of low radioactive waste were disposed of between 1967 and 1978. In addition, 1293 containers of intermediate level radioactive waste were emplaced between 1972 and 1977. After disposal of the waste ended, the Asse mine was turned into an Underground Research Facility (URF), where a range of experiments was carried out, including, for example, borehole heating experiments

Figure 2.6) Drums with concrete shielding emplaced within the Asse II mine. Photo: Bundesgesellschaft für Endlagerung mbH.

(e.g., Vons and Heijdra, 1993) and backfill experiments (e.g,. BAMBUS Poley, 2000). As the Asse II mine was originally built to maximise the extraction of salt, large open volumes had been excavated. These open volumes extend out to the edge of the salt dome where mineable salt is located. LILW was disposed in some of these open spaces, (at a depth of 725 - 750 m) and intermediate (at a depth of 511 m).

Eleven years after the emplacement of the waste, in 1988, brine began to enter the mine. However, where the brine originated from was not clear. To remediate this situation, three different options were considered, and, in 2010, it was decided to retrieve all the waste from the Asse II mine and relocate it above ground. This was judged by those involved in the decision to be the best solution to ensure the long - term safety of people and the environment.
The retrieval of the waste will be challenging, as it is unclear what the current storage situation is, or how to retrieve the waste safely, and how high the costs and occupational radiation doses will be.
An obvious question is how brine could have entered the Asse II mine if rock salt is nearly impermeable, and what can be learned from this? The situation in the Asse mine are such that one of the current German requirements for implementing a repository in salt (Minkley, 2009) on dilatancy stress and fluid pressure criteria would be violated, meaning that the integrity of the natural barrier (the host rock) cannot be guaranteed (Bollingerfehr et al., 2018b).
To preserve the impermeability of the rock salt, there should be a sufficiently large (salt) barrier between the waste and surrounding formations that contain groundwater. In the Asse II mine, however, the thickness of the salt that remained after mining the upper levels is insufficient and the disposal caverns had not been backfilled. As a result, brine started to intrude into the mine (Minkley, 2009). This experience has reinforced the requirement in our current programme that a purpose-built GDF should be constructed, in which the waste is surrounded by a sufficient thickness of intact salt as the principal natural barrier.

2.5.2 Morsleben

The Morsleben radioactive waste repository (Endlager für Radioaktive Abfälle Morsleben: ERAM. Fig. 2.7) is a GDF in a salt dome located near the town of Morsleben, in Germany. Like the Asse II mine, Morsleben is an old salt mine, where mining began in 1897.

After mining ended, the salt mine was designated for disposal of low and intermediate level waste by the former East German government, in 1965. Disposal of waste started in 1971 and ended in 1991 but started again in 1994 and lasted until 1998. Around 36,800 m^3 of low and intermediate level waste, with negligible heat generation, was emplaced.

It was decided that backfilling and sealing the facility was the preferred option for closing the repository, which is now under licensing for closure. About 75% of the cavities will be backfilled with salt concrete also referred to as saltcrete and 22 sealing structures will be constructed (Strahlenschutz Bundesamt für Öffentlichkeitsarbeit, 2015). Filling the empty cavities with salt concrete will enhance the mechanical stability of the mine and prevent the development of new pathways (fractures) between the mine and the surrounding geological formations. Some cavities will, however, remain open, to act as storage space to accommodate gas produced by corrosion processes and decomposition of organic substances in the wastes. Gas pressure could potentially lead to new pathways through rock salt, and the gas storage spaces are connected by a borehole filled with gravel to ensure that gas exchange between cavities can take place. In addition to backfilling, seals will be emplaced near the disposal areas.

2.5.3 Waste Isolation Pilot Plant

The Waste Isolation Pilot Plant (Fig. 2.8), or WIPP facility, in Carlsbad, New Mexico, USA is the first custom-built salt repository and is currently the only operational GDF in rock salt. It is constructed in the bedded salt of the Salado formation.

The Salado formation, like the Zechstein group in the Netherlands, is of Permian age. The site for the WIPP facility was selected in 1974, construction started in 1984, and the first waste arrived at the facility in 1999. The single-level repository is located at a depth of 655 m; it currently consists of eight large disposal panels but can be expanded if needed. Each of the large disposal rooms consists of 7 smaller rooms that have a width of about 10 m, a length of about 91 m and a height of about 4 m, separated from each other by a 30 m thick salt wall/pillar.

The WIPP is licensed only for disposal of transuranic waste that is a by-product of the nuclear defence programme in the USA.

Figure 2.7) Drums with radioactive waste in the Morsleben repository. Photo: Bundesgesellschaft für Endlagerung mbH.

Figure 2.8) A borehole is drilled into the wall of the WIPP for experimental purposes. Source: Kristopher Kuhlman, Sandia

The waste comprises two categories: waste that can be directly handled and stacked on the ground within a disposal room, and waste that must be remotely handled. For the latter, waste canisters are placed horizontally in the walls of the disposal rooms. Although the operation of WIPP will continue for many years, research has been carried out on its closure. As a backfill, bentonite/sand, salt mixtures and many other materials have been considered (Brush et al., 1999; Papenguth et al., 2000), but currently pure MgO in palletised form is used. MgO has the advantage that it will reduce the mobility of radionuclides and has a self-sealing function, since it will swell. The MgO is placed on top, next to and between the waste and the salt in polyethylene sacks and hence, they do not completely backfill entire rooms. Within the WIPP, the MgO is the only engineered barrier in the system (United States Department of Energy, 2014).

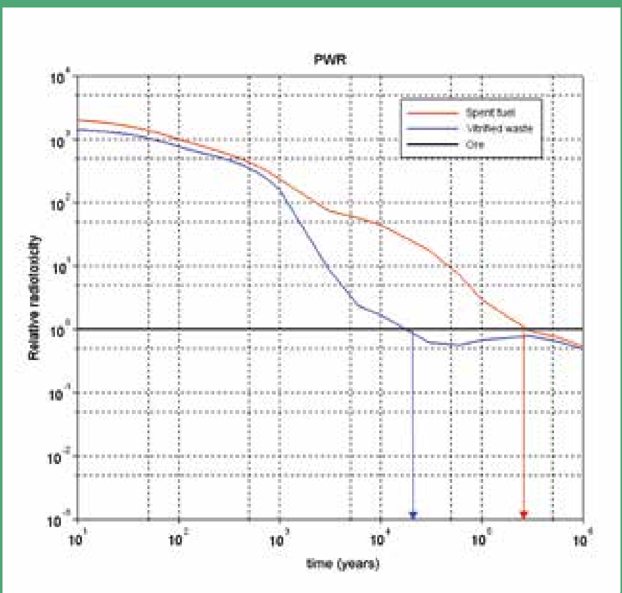
Box 2-1: Addressing the long-time scales in the Safety case

There is a commitment among those managing radioactive wastes to ensure safety at all times to levels at least as protective as those provided today. This has meant looking farther into the future than has been attempted for any other engineering project – not just a few generations (the design life of most engineered structures), but tens of thousands of generations. Typical GDF safety assessments model potential impacts on future generations out to a million years – a timescale that is hard to imagine for most people. However, even such an immense period of time is relatively short for a geologist used to considering how our natural environment has evolved over hundreds of millions of years. The long times over which we wish to provide protection are put into a different perspective when we consider our ability to characterise and understand natural geological processes occurring deep below the surface over much longer periods. This is what underpins the concept of geological disposal and provides confidence in the achievable safety.

Of course, forecasting the future behaviour of a disposal system for such long times brings with it increasing uncertainty as we look farther into the future. The level of uncertainty depends on the particular geological environment being studied, the materials used in the multibarrier system, and the physical and chemical processes being evaluated. For some materials or processes, we can only be confident in our predictions of behaviour for thousands of years. For others, particularly many geological processes, we can have confidence in our predictions for hundreds of thousands, or even millions of years.

Furthermore, radioactive wastes exhibit one key characteristic that sets them apart from many other hazardous materials and that puts the issue of long timescales in a different perspective - owing to the natural process of radioactive decay, their radioactivity reduces with time. If the multibarrier system prevents radionuclides returning to the human biosphere for sufficiently long, they will no longer pose hazards for humans. The rate and scale of reduction in radioactivity depends on the radionuclides contained in the wastes and is known exactly. Because much of the original activity in the most radioactive categories of COVRA's waste is due to radionuclides that decay relatively quickly (e.g., Sr-90 and Cs-137, whose activity halves every 30 years), most of the activity disappears within 1,000 years. This early decay in radioactivity significantly reduces concerns about the long timescales that are being considered. However, the potential impacts of longer-lived radionuclides must also be considered - and this is a central aspect of the safety assessment in Chapter 8. It is important, therefore, to consider in more detail how the total radioactivity of the wastes changes with time.

In practice, when considering the potential impacts of radionuclides on people, it is their 'radiotoxicity' rather than their radioactivity that is more relevant, since the radiation dose (in Sv) from ingesting a given amount of a radionuclide (in Bq) differs between radionuclides. The radiotoxicity of a given amount of waste is thus a measure of the radiation doses that would result if all the radionuclides in a given amount of waste were to be dissolved in water which was then drunk by a person (Hamstra, 1975; Hamstra and Van der Feer, 1981). This situation is entirely hypothetical, but it does allow comparison of how hazardous different types of radioactive materials can be. For example, it allows comparisons between the radiotoxicity of spent fuel or HLW and the radiotoxicity of the natural uranium ore from which the fuel was produced. An example of the calculation of the relative radiotoxicity of spent nuclear fuel from a pressurized water reactor (PWR, such as the Borsele nuclear power plant) is shown in the graph below, which also shows the radiotoxicity of vitrified HLW resulting from reprocessing such spent nuclear fuel (Gruppelaar et al., 1998). The graph plots the declining radiotoxicity of spent fuel and vitrified HLW as a function of time after the fuel has been taken out of the reactor or, for vitrified HLW, after it was manufactured, following the reprocessing of the equivalent quantity of spent fuel. These curves are shown normalised to the radiotoxicity of the amount of uranium ore that was originally used to make the fuel (the horizontal line). To determine the radiotoxicity of each radionuclide Gruppelaar et al. (1998) used slightly different dose conversion factors to determine the radiotoxicity of each radionuclide than those used by Hamstra (1975) and Hamstra and Van der Feer (1981) since the International Commission on Radiation Protection (ICRP) updated the radiotoxicity data.



With a burn-up of 47,500 MWd/t, spent fuel is more radiotoxic than the uranium ore[1] from which it was manufactured for a period of about 200,000 years. At present, direct disposal of spent fuel from power reactors is not considered in the Netherlands, so the more relevant curve in the figure is that for vitrified HLW (the principal part of COVRA's higher activity waste inventory). In the reprocessing process, the long-lived uranium and plutonium are removed to manufacture more nuclear fuel. The resulting HLW is more radiotoxic than the uranium ore only for a period of for around 20,000 years.
By this time, the large reduction in hazard potential that has occurred means that the primary functions of the multibarrier system have, largely, been achieved.
The waste has been isolated and contained until it presents a hazard potential equivalent to materials found in nature and, specifically, to those materials from which it was originally manufactured. It must also be acknowledged that uranium ores themselves can present hazards and that the wastes are now in a different location from the original ores. Accordingly, the safety case still needs to consider the possible impacts on people and the environment of the residual radionuclides that do not decay for very long times. These are predominantly radioisotopes of the heavy elements such as uranium, neptunium and plutonium, and of fission products such as I-129, Tc-99 and Se-79. However, the former group is strongly retarded in the clay host rock and the fission products, although mobile in groundwaters, have low radiotoxicities (Chapman and Hooper, 2012).

This illustrates that, in the design and safety assessment of a multibarrier system, it is essential to ensure that complete isolation and containment are achieved over the first hundreds of years after closure. In the early period after closure, it is appropriate to judge possible health impacts on people using normal radiological protection standards. In the longer term, the hazard potential is much less, and in the very long-term we are dealing with something like naturally radioactive materials. Consequently, as the timescale increases beyond a few tens of thousands of years and out to a million years, it becomes more appropriate to assess hazards using other measures, more related to our daily exposure to natural radioactivity.

1. Note that the radiotoxicity of U-238 is central when comparisons are made to uranium ore. The calculations shown here use 2.4×10^{-7} Sv/Bq for the dose conversion coefficient for uranium-238, from ICRP (1994). ICRP (1996) proposed 1.2×10^{-6} Sv/Bq: i.e., uranium-238 was then believed to be more radiotoxic. The impact in cross-over time for spent fuel with a burn-up of 50 MWd/ton is a reduction from 170,000 years to 130,000 years (Magill et al., 2003). An even higher dose conversion coefficient for uranium-238 was used by NAGRA (2002): 2.5×10^{-6} Sv/Bq. NAGRA estimated a cross-over time for vitrified HLW relative to uranium ore at about 2000 years. The latest ICRP-119 (2012) report proposes again 2.4×10^{-7} Sv/Bq, as proposed previously in ICRP (1994). This change makes the calculations by Gruppelaar et al. (1998) with a cross-over time of about 20,000 years for vitrified HLW, currently, the most relevant one and is different from the one presented in the OPERA Safety Case from Chapman and Hooper (2012), which is closer to the value estimated by NAGRA. Regardless, the main message, is that the hazard potential diminishes over many thousands of years and should, remain unchanged.

3. Post Closure Safety

As explained in Chapter 1, demonstration of the safety of a multi-barrier system is achieved through the preparation of a series of safety cases that are assembled sequentially, at key phases of programme development. The present Chapter explains in more detail the safety strategy for disposal, the structure of the safety case prepared by COVRA and the roles that the evolving safety case will play throughout all phases in the lifecycle of a geological disposal facility. The safety strategy is designed to satisfy national and international requirements. Since the publication of the OPERA Safety case in 2017, the COVRA requirements management system has been made compatible with the parallel safety case in clay and with COVRA's waste storage programme.

The principal safety-relevant impacts of the multibarrier system are calculated in terms of radiation doses that might be received by people in the distant future. To put this into context the following section describes the permissible dose targets or limits that have been laid down in regulations.

3.1 Required level of safety

To establish that a multibarrier system will not give rise to unacceptable impacts on people, agreed limits for such impacts must be defined. Calculating the consequences of potential releases of radionuclides from a multibarrier system is, in principle, a purely technical challenge. Judging whether the calculated releases would be acceptable to people is, however, also a societal issue. The most common metrics for quantifying radiological impacts are calculated radiation doses or risks. To assess whether adequate safety has been achieved, these doses and risks are then compared with regulatory limits or targets. Yet, no regulatory criteria have been defined explicitly for the implementation of a GDF in the Netherlands. However, European radiation protection criteria and standards have been established by Council Directive 96/26/Euratom, and Member States must comply with this Directive (European Commission, 2014).

The EU radiation protection criteria and standards are derived from the recommendations made by the International Commission on Radiological Protection (ICRP), in particular those made in 2007 in ICRP Publication 103 (which sets down a limit of 1 mSv per year for the total dose to any member of the public from any regulated source) and in 2013 in Publication 122 (which proposes a lower constraint of 0.3 mSv per year for a GDF, ICRP, 2013). In OPERA, a lower limit of 0.1 mSv per year was proposed (Hart and Schröder, 2017) since this has been a common choice in other national disposal programmes.

To give some perspective on these numbers, it can be noted that the average total natural radiation exposure to a person living in the Netherlands is much higher, averaging 1.7 mSv per year (Smetsers and Bekhuis, 2021). This total radiation exposure in the population is monitored and periodically updated by the National Institute for Public Health and Environment (RIVM), which analyses prevalent exposure pathways and uses radionuclide-specific dose conversion coefficients set, and periodically updated, by the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and by ICRP. Consequently, calculated values vary a little periodically – for example, a value of 1.6 mSv per year was estimated at the time of the OPERA Safety case. In fact, natural exposures of people living in the Netherlands are significantly below the global average value of about 2.6 mSv per year (United Nations Scientific Committee on the Effects of Atomic Radiation, 2008).

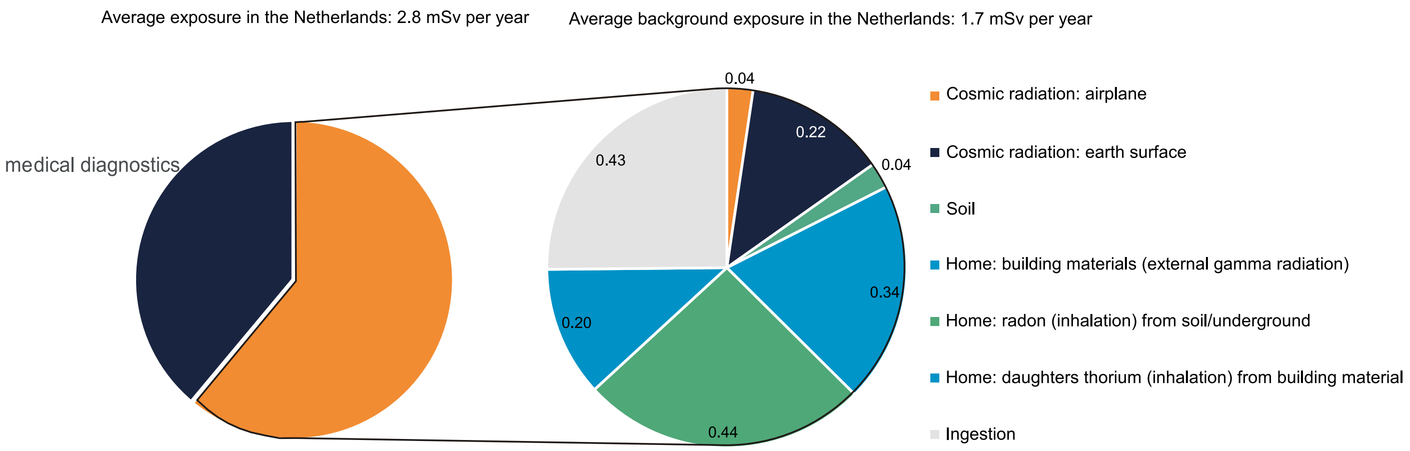


Figure 3.1) Average radiological exposure to members of the public in the Netherlands with a total exposure of 2.8 mSv per year as estimated by the National Institute for Public Health and Environment as estimated in 2021. Radiological exposure by natural radionuclides and cosmic radiation from Smetsers and Bekhuis (2021).

Figure 3.1 shows the different average contributions to radiological exposure by natural radionuclides, with a short description of exposure pathways, as well as the additional average exposure to radiation from medical diagnostics, which raises the average exposure from 1.7 mSv per year (natural background only) to a total value of 2.8 mSvper year. Radionuclides with a primordial origin have an important contribution to our natural background radiological exposure:

- Uranium-238, uranium-235 and thorium-232 occur in various concentration in all rocks and minerals and decay to radionuclides that generate radioactive radon, a noble gas. Radon is emitted from building materials containing these radionuclides and can subsequently be inhaled by people. The resulting radon dose is the largest single contributor to our average radiological exposure by natural radionuclides (see Figure 3.1).
- Potassium-40 also occurs in various concentration in all rocks and minerals and is mainly responsible for the external gamma-radiation from soil and building materials at home, and also provides the largest contribution to doses from the ingestion of food (Cinelli et al., 2019; Smetsers and Bekhuis, 2021).

The multiple barrier system for geological disposal of waste is designed to contain the artificially generated radionuclides in the wastes to such an extent that any additional radiological exposures that might arise for people in the future are negligible compared to the natural radiation exposures to which they will be subject.

3.2 Structure of a safety case

Expanding upon the concise definition of a safety case given in Chapter 1, the IAEA (2012a) and NEA (2013b) draw attention to the following key points. The safety case must:

- provide the basis for understanding the disposal system and how it will behave over time;
- address site and engineering aspects, providing the logic and rationale for the design, and be supported by safety assess ment that includes quantitative estimates of the behaviour and evolution of the disposal system.
- identify and acknowledge unresolved uncertainties that may exist at the specific stage of the repository development programme, along with their safety significance and approaches for their management.
- include additional information and evidence that supports the safety assessment, and reasoning on the robustness and reliability of the disposal system.
- present, if required, more general arguments and information to put the results of safety assessment into perspective.

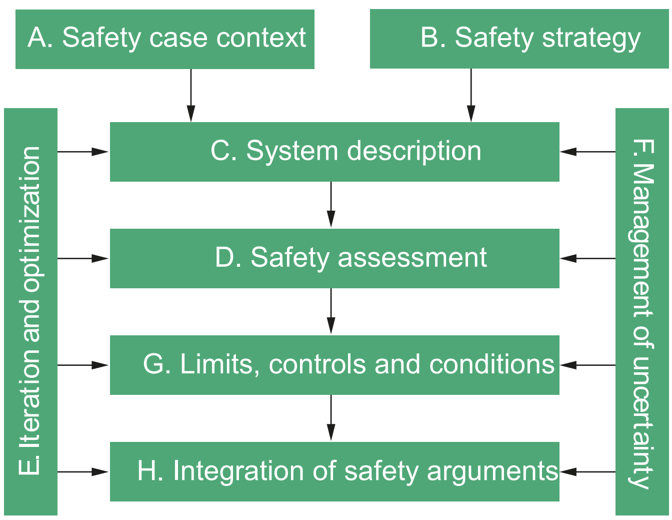


Figure 3.2) Components of a Safety Case (IAEA, 2012a).

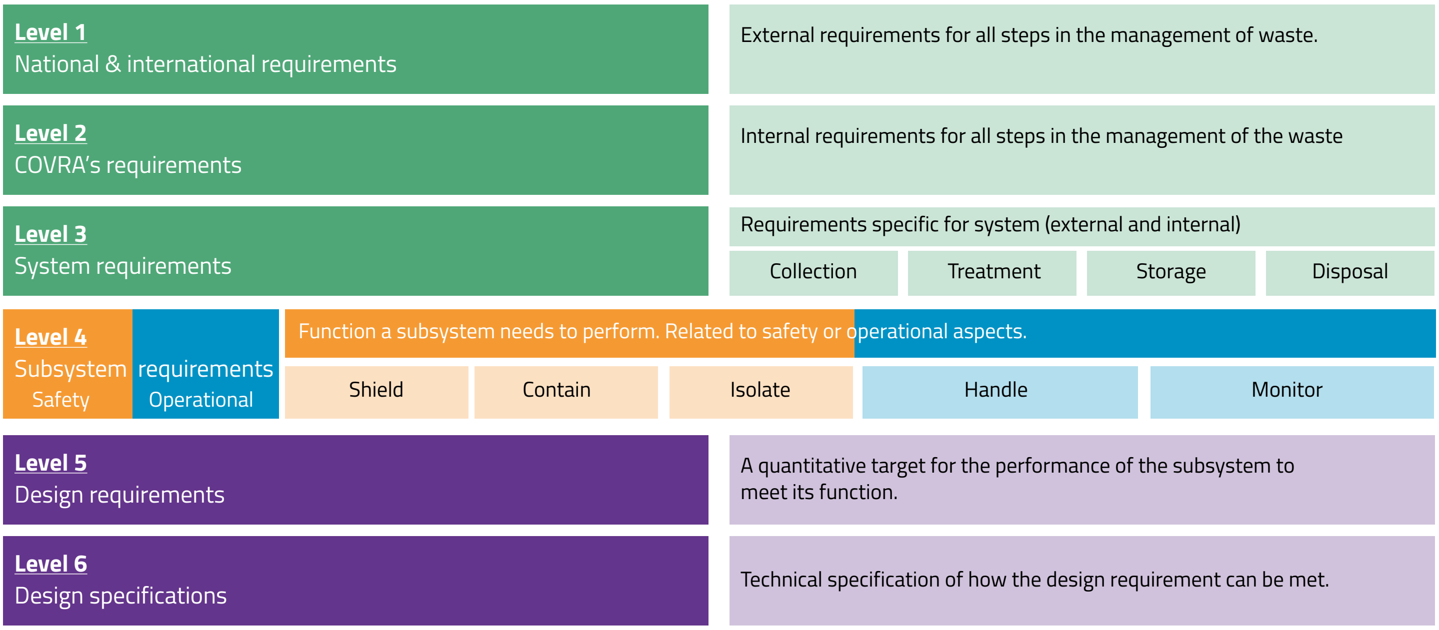


Figure 3.3) Current hierarchical arrangement of requirements. The first 3 levels of the RMS define the overall requirements on any one of our systems and the first two levels apply to all of them. At Level 3 the RMS becomes system-specific, diverging to define the requirements for the different sub-systems within one of our four main activities. Levels 5 and 6 define in detail how sub-systems for one of our main activities are designed and their engineering is specified to ensure that the completed system meets all the higher-level requirements. These requirements are described in Section 3.5

The components of the safety case as defined by the IAEA are portrayed graphically in figure 3.2. Each of these items is addressed in the present report, as follows:

- The safety case context was mentioned already in Chapter 1; to increase confidence in post-closure safety, assess the disposability of the waste and assure adequate funding for disposal. The Dutch national radioactive waste management programme is currently being evaluated in the framework of the EU Waste Directive (European Commission, 2011) and this safety case has been written to help this evaluation.
- Section 3.4 gives more details on the overall safety strategy.
- A high-level system description of the multiple barrier system for geological disposal of radioactive waste is covered in Chapter 2 and an overview description of COVRA's current concept for a multibarrier system with salt as a host rock is described in this Chapter. The system description is covered in more detail in Chapters 4, 5 and, 6. Chapters 4, 5 and 6 also discuss current design requirements and specifications, showing how the components of the multibarrier system contribute to safety.
- Uncertainties and gaps in data can arise because not all the information required for a safety assessment is available. This can be addressed by using alternative models of processes and behaviour, or by making assumptions. When assumptions need to be made, these are generally chosen to be conservative, i.e., pessimistic, so as not to overestimate the performance of the multibarrier system. However, a best estimate of the expected evolution can also be made, and this provides a perspective on how conservative the assessment assumptions are. Chapter 7 discusses the realistically expected evolution of the engineered barriers in clay host rocks.
- Chapter 8 shows the system evolution assumed for the safety assessment.
- Chapter 9 is an integration of the previous work to formulate conclusions. Discussion of uncertainties has not been allocated a specific section; instead, the uncertainties associated with each of the important processes described, or with the data employed, are addressed at the appropriate section. In addition, the final Chapter summarises uncertain ties and open questions.
- Design iterations, as indicated in the IAEA structure, have been performed and are presented in Chapters 4 and 6.

3.2.1 Safety strategy

According to both IAEA and NEA guidance documents, one of the initial components of the safety case should be a safety strategy (IAEA, 2012a; NEA, 2013b) which is defined as the high-level approach adopted for achieving safe and acceptable disposal of radioactive waste. The implementer (i.e., COVRA) should develop the safety strategy. In the current phase of work in the Netherlands, the strategy should provide for a systematic process for developing, testing and documenting the present level of understanding of the performance of a GDF and for building and maintaining the necessary knowledge and competences through successive research programmes. It is important to note that the safety strategy is presented in the form of a living document; both the strategy and the disposal concepts based on the strategy will develop iteratively over the whole implementation period, which in the Netherlands is currently planned to last almost a century (Ministerie van Infrastructuur en Milieu, 2016; van Gemert et al., 2023).

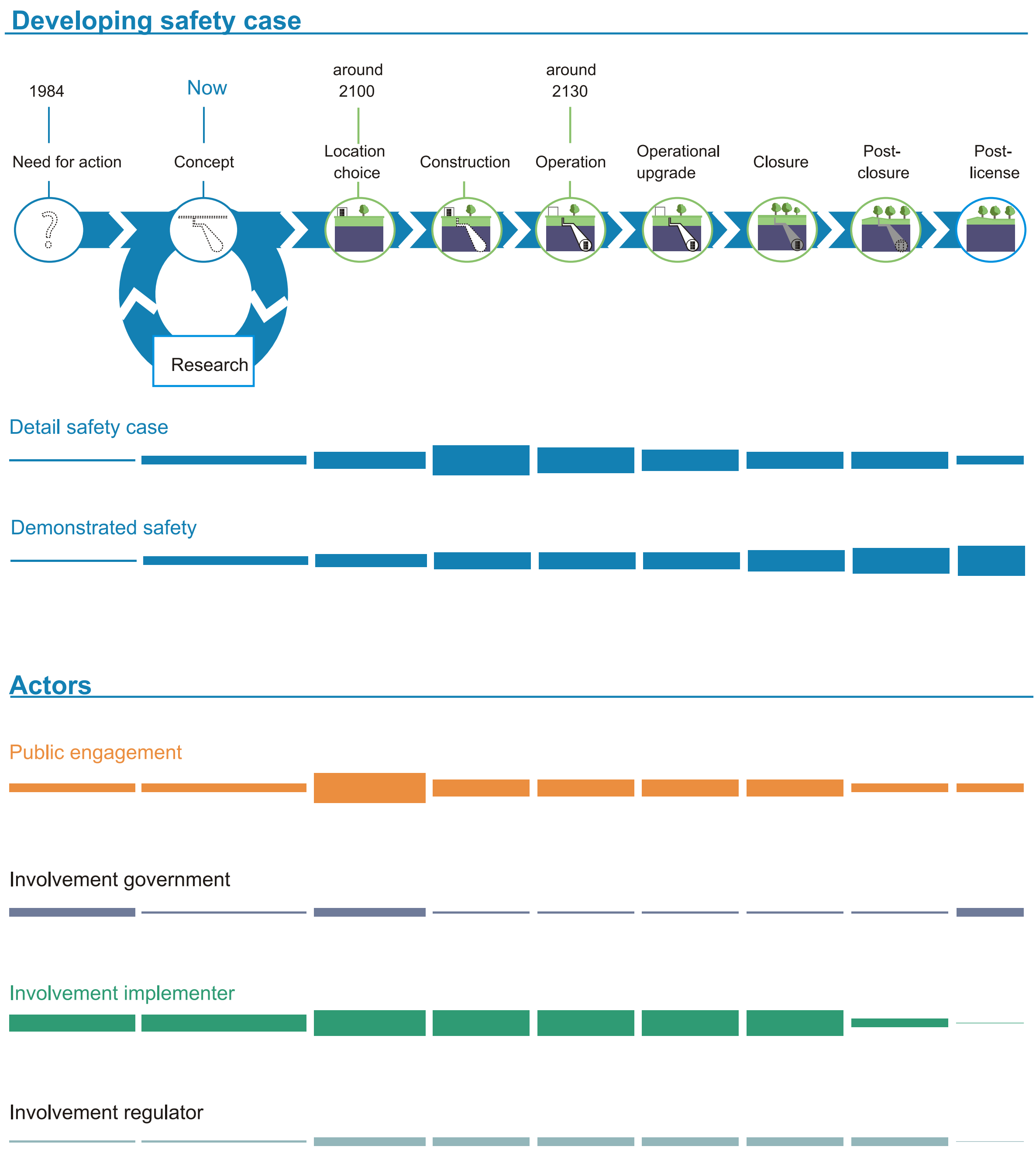


Figure 3.4) Key stakeholders and common elements in the decision making processes on geological disposal of radioactive waste with the planning as laid down in the first national programme made in the framework of the EU waste directive (Ministerie van Infrastructuur en Milieu, 2016).

The safety strategy also includes the identification of the over-arching national and international requirements to be satisfied, and the definition of the more detailed requirements made by the programme implementer to accomplish this. National and international requirements are derived from relevant national (Dutch Decree on radiation protection) and international regulatory frameworks (EURATOM)- COVRA's requirements are more specific derivations of the general international and national requirements. Both are used to define further requirements to be satisfied by the multibarrier system and its components and lead to the identification of the design requirements and design specifications for the implementation of the GDF. The safety strategy has been chosen to focus on a design-driven basis by developing a hierarchical set of different levels of requirements in a requirements management system (RMS), as shown in Figure 3.3

3.3 Roles of the safety case

The role that the safety case will take throughout the conceptual, site selection, constructional, operational, closure and post-closure phases is different. The iterative nature of the safety case is apparent when one considers. This figure 3.4 shows the common steps or stages in the decision-making processes leading to geological disposal and indicates the key stakeholders involved, as well as the planned timing in the Netherlands, as laid down in the first national programme made in the framework of the EU Waste Directive (Ministerie van Infrastructuur en Milieu, 2016). At each decision point, the safety case must provide the safety related information that allows a judgment on whether to proceed to the next stage.

The nature of the decisions to be made and the characteristics of the safety case for each of the stages in the disposal of waste are described in IAEA (2011b) and the regulatory expectations of the safety case are periodically updated in the European Pilot Study (EPS, 2016). The design basis for each phase has a different set of objectives, requirements, constraints, inputs, and outputs (IAEA, 2020b).

3.3.1 Need for action

When a country starts generating radioactive waste, there is a need for action by the government, which must define a policy to meet its responsibility for managing all the necessary steps, from collection to eventual disposal. Commonly, the government nominates or establishes an organisation responsible for developing and implementing the disposal strategy. The Netherlands already passed this stage in 1982, with COVRA being the nominated agency to manage Dutch radioactive wastes. The decision for geological disposal of waste was made in 1984.

3.3.2 Disposal concept stage

The Dutch programme is in the conceptual phase of geological disposal development, in which different disposal concepts and potential host rocks are being considered. These generic designs allow definition of associated generic, non-site-specific safety cases (how the concepts being considered would provide safe disposal for the waste inventory) and they provide a starting point for programme planning and the estimation of duration, costs and project risks (IAEA, 2020a). The government lays out the requirements for and framework within which geological disposal of waste should be implemented. The implementer (COVRA) establishes the safety strategy, incorporating these requirements into the high-level tiers of its RMS, and carries out preliminary safety assessments for post-closure. Post-closure safety should be provided by a system of natural and engineered barriers. Regulatory review of the work at this stage should guide the implementer on the likelihood of achieving the necessary demonstration of safety (EPS, 2016).

This is effectively the present stage of the Dutch programme. COVRA established a research programme for non-site specific safety cases for the next) in which disposal concepts for 2 host rocks (poorly indurated clays and rock salt) and other techniques emerging from international collaboration are investigated. COVRA makes safety cases – including the post-closure safety assessments - aligned with the review cycles of the national programmes in the Waste Directive (European Commission, 2011) which implies that a safety case is to be made each decade(Verhoef, 2020; Verhoef et al., 2017). COVRA also makes the cost estimates and research programmes for disposal of waste every 5 years and publishes updates of safety cases 5 years after publication of its safety cases.

3.3.3 Site selection stage

The government, together with the implementer, must develop a national framework for decision-making on site selection. For successful projects, this must be widely supported, and adhered to, by the relevant actors, whose roles and interrelationships must be clear. The national framework should support participatory, flexible and accountable decision-making processes. For example, the implementer identifies potentially suitable sites that are compatible with the disposal concept(s) and characterises these sites to the extent that a decision can be made on a preferred site (EPS, 2016). In the Netherlands, it is not yet decided who will identify potentially suitable sites but, in any case, a key element of the basis for this decision should be a safety case, including at least an outline of the operational safety case together with a comprehensive post-closure safety case. One of the most important inputs of this post-closure safety case is the Site Descriptive Model (SDM). The SDM can be seen as a synthesis of the descriptions of the site geology, rock mechanical properties, thermal properties, hydrogeological properties and parameters, hydrogeochemistry, transport and flow properties and the surface environment. The SDM represents an integrated suite of information and understanding of the natural systems. The SDM is not static but is continuously updated as the site knowledge base grows through further investigations (IAEA, to be published). The Environmental Impact Assessment (EIA: in Dutch, Milieu Effect Rapportage) is based on this safety case. The realisation of the SDM will require the drilling of boreholes to characterise the sub-surface environment. The implementer is expected to require a licence for drilling boreholes or for implementing site specific underground facilities. A licence application for drilling may also require an EIA. This EIA includes the impact on local and regional stakeholders of any environmental disturbances, including potential harmful emissions, noise, dust, traffic et cetera, during the drilling.

A monitoring programme and the organisations in addition to the implementer that monitor properties of the site, must be identified and agreed. The aim of the monitoring carried out by the implementer is to obtain reference values for a wide range of environmental parameters. Monitoring is therefore to be started at an early stage, with one of its aims being to quantify, against a pre-construction baseline, any additional (radiological) exposure from the construction and operation of the geological disposal facility. The National Institute for Public Health and Environment monitors radioactivity in the Dutch environment and has sensors within nuclear facilities to perform their own independent monitoring. All countries in the European Union are required to perform these measurements in their national environments, under the terms of the Euratom treaty of 1957. It is therefore expected that this National Institute will also have a role in the independent monitoring of the GDF.

Local and regional stakeholders have an important role during the lifecycle of the GDF, especially during the site selection process and onwards. Public information, consultation and/or participation in environmental or technological decision-making should represent current best practice and must take place at different geographical and political scales. Large-scale technology projects are more likely to be accepted when local and regional stakeholders have been involved in making them possible and have developed a sense of interest in, or responsibility for, their success. For the Netherlands, this stage of site selection lies far in the future, probably not beginning until the second half of the 21st Century. However, the approaches to be used and the decision processes that will be applied must be proposed, discussed by all stakeholders and agreed at an earlier phase in the disposal programme.

3.3.4 Construction licensing stage

The reference design (and application for construction) phase, is the period in which the implementer adapts the conceptual design to the site properties, substantiates and finalises the design of the disposal facility, and develops the safety case to support the implementer's application to construct, operate and close the facility. Based on the review of the safety case, the licensing body decides whether to grant a licence for the implementer to construct the facility. This is a crucial milestone in the development of a GDF (EPS, 2016). Depending on the licensing approach adopted, licensing may be the basis for going underground to enable more detailed and direct characterisation of the site than can be accomplished from the initial boreholes, or it may be the basis for extending from a URF that has already been used for underground characterisation purposes into volumes of rock that will be used for disposal. The reference design uses information from the SDM and the EIA includes assessment of the impacts on the local and regional environment as well as identifying the impacts on stakeholders of any environmental disturbances resulting from GDF construction activities (IAEA, to be published). The EIA will also address mitigation measures to reduce such disturbances, developed in agreement with the hosting community.

3.3.5 Construction and operational licensing stage

When a construction license is granted, underground access, characterisation and testing excavations can be extended into a progressive programme of GDF construction, including any surface facilities such as a waste encapsulation plant that may be required.

During construction the implementer demonstrates that the facility is being built as planned in the safety case and in accordance with the conditions of the construction licence. Towards the end of this phase the implementer will present its final approach for operation and a concept for closing the facility and apply for an operating license. In preparing for operation, the implementer will need to demonstrate safety during operation, including radiation protection of workers and members of the public (EPS, 2016). Commissioning tests are envisaged to be required to provide final assurance that the GDF will operate safely. These might include tests of the transportation and emplacement of waste packages using dummy waste containers of the same weight and shape as the final waste packages (IAEA, to be published).

3.3.6 Operation and closure stage

The operational stage is the period in which the implementer emplaces waste packages in the disposal facility. During this phase, the implementer may excavate new disposal tunnels or caverns, and possibly backfill and seal underground openings, either temporarily or permanently. Late in this phase, the implementer also develops an application to close and seal the facility, and prepares a plan for post-closure institutional controls, monitoring and surveillance. At the termination of operations, the regulator will decide whether to grant a licence for the implementer to close and seal the facility. When the licence is granted, the implementer proceeds to the closure of the facility (EPS, 2016).

3.3.7 Post-closure stage

The post-closure phase, is the period in which the implementer provides evidence to demonstrate that it has closed the disposal facility in accordance with safety and license requirements, presents a firm plan for institutional controls, and continues monitoring and surveillance as long as is required by the national legal and regulatory framework (EPS, 2016).

3.3.8 Post - licensing period

At some point after closure, the GDF will cease to be a licensed nuclear facility in the ownership of the implementer. The national government takes over responsibility for the GDF. International nuclear safeguards requirements (with respect to any fissile materials contained in the GDF) might then be satisfied by remote surveillance means (e.g., satellite monitoring, aerial photography, micro-seismic monitoring). All relevant information about the nature and location of the GDF is expected to be accessible, as obligated by implementation of the European Directive for the establishment of Infrastructure for Spatial Information in the European Community (European Commission, 2007). It is likely that the national government will introduce measures to regulate any monitoring, surveillance or safeguarding activities and to control or prohibit activities, such as exploration drilling, in the vicinity of the GDF.

3.4 How the disposal system for a GDF in salt provides isolation and containment

As mentioned in section 2.1, the two principal objectives of a GDF are isolation and containment.

In the COPERA disposal concept, isolation is provided by the 750-metre depth of the GDF, which isolates the waste from people, the environment and geomorphological processes such as river incision, sea level change, deep glacial channels and permafrost. Isolation is provided in a passive manner for at least several hundreds of thousands of years, but probably for much longer.

Containment in the COPERA salt disposal concept is provided by a combination of natural and engineered barriers (Fig. 3.5). The natural barrier is the host rock: rock salt. Undisturbed rock salt is practically impermeable and should thus provide permanent containment if in its undisturbed state. The core of the engineered system comprises the waste form and, for HLW, the outer steel waste package. During the construction of the GDF, local disturbance of the natural barrier caused by excavation of open spaces such as disposal, transport and ventilation tunnels, is inevitable. To ensure that these do not lead to breaching of the contain-

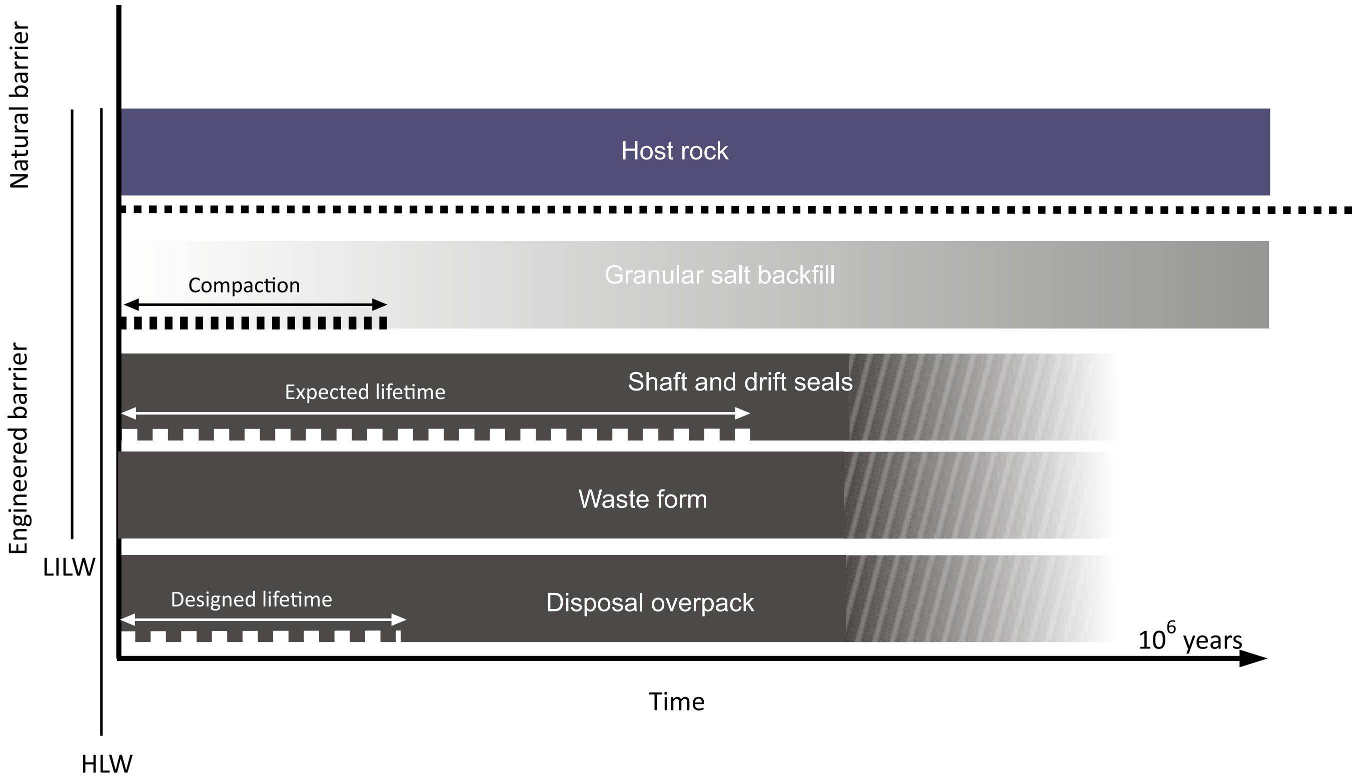


Figure 3.5) Evolution over time of the sealing effects of important barriers in the post closure phase of a repository system in salt. The colour intensity represents the degree of the respective containment effect, and the colour used corresponds with figure 3.1. Note that the figure is not to scale. Figure based on figure 2.1 of Bollingerfehr et al. (2018a).

ment, seals and backfill are used. The seals in the shaft and tunnel, together with the HLW package, provide the required containment immediately after construction. However, these engineered barriers will be affected by thermal, mechanical and chemical processes, and their performance will degrade with time. In contrast, the salt host rock in the near-field will retain its containment properties for geological periods of time, provided that it can return to its undisturbed state. For long-term sealing, granular salt will be used to backfill most of the open spaces and parts of the shafts. Initially, granular salt will have a relatively high porosity and permeability but, with time, the backfill will be naturally compacted by the convergence of the host rock. This compaction results in a decrease of the porosity and permeability in the granular salt until it has comparable properties to the host rock. Hence, the waste will eventually be encapsulated by an impermeable barrier. In the unlikely case that brine inflow to the waste occurs, the engineered barriers, in combination with the waste form, contribute to the containment of the radionuclides in different ways. This is achieved either by the seals, the HLW package or the backfill by restricting the movement of contaminated brine or, in the case of the waste form, allowing only very slow dissolution and mobilisation of the radionuclides. The evolution over time of the sealing and containment provided by the different barriers is shown schematically in figure 3.5.

To ensure that the COPERA disposal concept is properly designed and implemented to provides the necessary isolation and containment, an RMS is used, as described below.

3.5 Requirement Management system

3.5.1 What is an RMS?

There is a broad agreement in systems engineering that the identification and management of requirements that drive the design of complex systems and their components is essential if the purposes and objectives of the final system are to be achieved. In engineering, a Requirements Management System (RMS) is a hierarchical set of requirements establishing a design basis for a process or a manufactured object. This can be, for example, a piece of machinery, a building or a major piece of infrastructure.
An RMS should provide the logic and the rationale of the design. For disposal, for example, it provides a framework to assemble and manage all the requirements that are placed on the disposal system and to ensure that all these requirements are met. Moreover, it will ensure that inevitable changes in requirements and specifications that will occur over the lifetime of a disposal project are properly addressed and documented. In addition, it will help to identify knowledge gaps and potential optimisations. A key goal of an RMS is thus to ensure that what is designed and eventually built fully meets all the requirements.

3.5.2 Why do we need an RMS for disposal?

Implementing geological disposal of waste is a lengthy process that must cover waste collection, treatment, processing, storage, and disposal (Figure 3.6). Throughout this long and complex series of activities, an RMS provides a tool to identify and

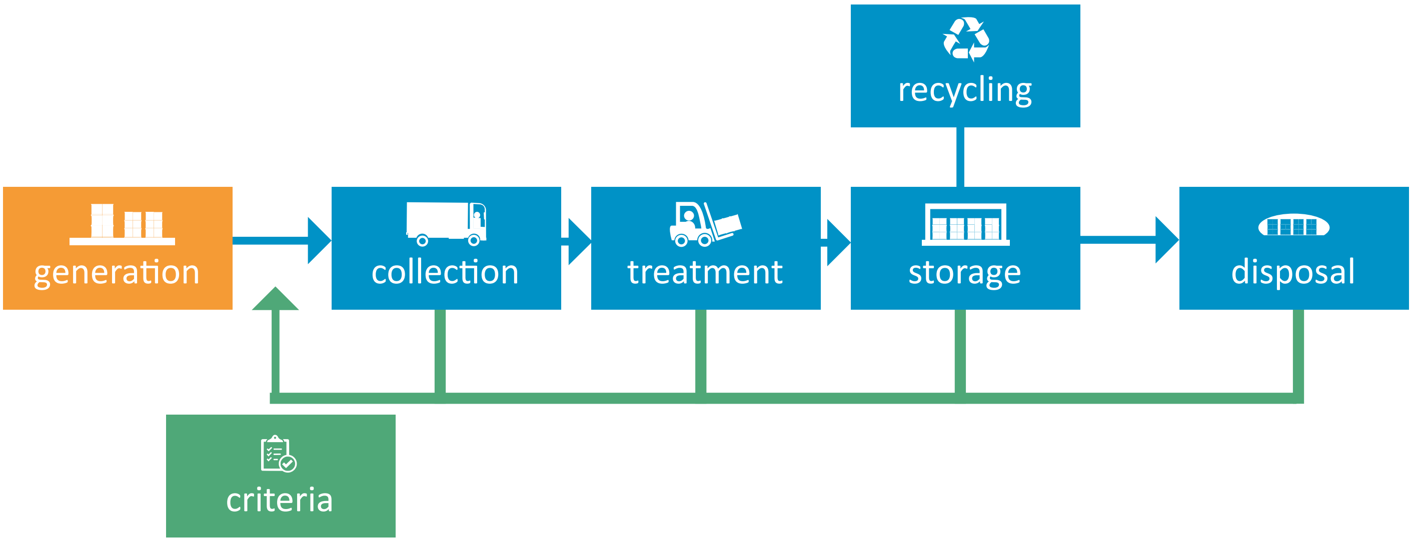


Figure 3.6) All the steps of radioactive waste management involve requirements on linked systems that must be managed by the implementer. Adapted from Verhoef et al. (2017).

manage requirements, provide traceability and transparency, and act to communicate between professionals with different expertise. Each step in the management of radioactive waste involves facilities and activities that are linked. Each procedure and each facility is a sub-system within the overall waste management system that COVRA manages. Each sub-system places requirements on one or more of the others and the overall system must be managed to meet many external requirements, arising outside COVRA. Managing any of these systems involves a broad range of disciplines including civil, electrical and chemical engineering, worker health and safety, security, geology, physics, chemistry, microbiology, and project and cost management. Requirements management takes advantage of the existing information in all these areas and the corresponding work processes available and integrates them into an overall structure to ensure the successful implementation of the management of waste. International experience has shown that the necessary integration of requirements for the disposal of waste is best addressed by the early development of a requirements-driven design basis (IAEA, 2020b).

3.5.3 Current RMS at COVRA

COVRA started with the definition of an RMS in the OPERA project (Verhoef et al., 2017). This RMS has been updated from an RMS specific to the disposal of waste into the current RMS that includes all the steps in the management of radioactive waste. This updated RMS, consisting of 6 Levels (Figure 3.3), is currently in the development stage. Thus, the RMS described here will be extended, adapted and updated in the future. The first two Levels of the current RMS contain requirements that apply to both disposal and pre-disposal (collection, treatment and storage) activities.
From Level 3 onwards, requirements become specific for disposal activities or any of the pre-disposal activities. Together, they describe how, for example, a disposal system provides a solution that meets the requirements that are encapsulated in rules, regulations and policies. Levels 4, 5 and 6 are described in detail in Chapters 4, 5 and 6.

Unlike our other, operational, pre-disposal systems, our disposal system is currently generic and conceptual, so our GDF design is at an early stage. Nevertheless, we can define the GDF safety concept and its sub-systems in sufficient detail to develop an RMS for the disposal system that already acts as a useful guide to our work. We thus use a reference GDF system design. This reference design describes a disposal system that meets the high level requirements and the system level requirements (e.g. isolation and containment) and is based on international best practices and similar GDF designs that have been shown to meet these requirements.
An example is the host rock for disposal. Both clay and rock salt have been shown in other national programmes to fulfil the requirements set in levels one to three. Subsystem requirements (level four), design requirements (level five) and design specifications (level six), on the other hand, differ between a GDF in clay and salt, largely because isolation and containment can be best provided by different designs of the multibarrier system. It also includes and assumptions that are made that do not result from the RMS but are necessary due to the early stage of the programme.

3.5.4 Structure of a requirement

Requirements must be clearly and unambiguously defined, without duplication. In COVRA's current RMS, each requirement consists of 3 parts namely (1) a UIN (Unique Identification Number), (2) requirement itself and a (3) short description of the requirement. As the RMS develops in future, other attributes are expected to be assigned to each requirement, such as measures of effectiveness in meeting the requirement, status and the responsible 'owner' within COVRA.

The UIN is a unique identifier need to keep track of the requirement, any changes in the requirement and its inter-relationships in the RMS. It is also used to identify its location in the RMS (e.g., Level and the (sub) system it belongs to) and the source of the requirement. For Levels 1 and 2 the UIN consist of three elements. From left to right, these are the number of the Level, the source on which a requirement is based and a sequential number. For example, a requirement at Level 1 that arises from the Dutch Decree on radiation protection (DCRE) will have the following UIN: L1 – DCRE – 01. A second requirement based on the same Dutch Decree on radiation protection at Level 1 will have the following UIN: L1 – DCRE – 02.

At Level 3, the UIN consists of 4 elements namely, the Level, the system of which the requirement is part of (e.g., transport, conditioning, storage and disposal), the source of the requirement and a sequential number. The first requirement at Level 3 for the disposal system (D) based on the national programme (NPRA) will have the following UIN: L3 – D – NPRA – 01. A second requirement for the system disposal (D) which is based on the national programme will end with 02, and so on. A similar logic is followed for levels 4, 5 and 6, except that the source of the requirement is replaced by the subsystem of the requirement. An example of a UIN at Level 4 is L4 – DS – WP – 01. Based on this UIN, the requirement is at Level 4, applies to a disposal system with salt (DS) and is specifically for the Waste Package (WP).

The second part of the requirement structure is its descriptive title, which should be short, clear and unambiguous. The last part of a requirement is a short description of the meaning, justification and purpose of the requirement, which could include references to reports, research or legislation on which the requirement or value is based. The latter is needed to ensure traceability of a requirement, or the justification for the values used to determine whether the requirement is met.

3.5.5 Level definitions and requirements

In the following section, the definitions of the 6 different Levels are discussed. These definitions determine at which Level a requirement needs to be placed. In addition, an overview is given of the requirements that are currently on each Level in our developing RMS. For the purposes of this safety case, we only considered the requirements for the disposal system from Level 3 onwards.

3.5.5.1 Level 1: National and international requirements

Level 1 requirements are based on rules and regulations that originate only from external national and international legislative organisations. They are requirements on all our activities and systems. Example sources are EURATOM, the EU, the Dutch government and the Dutch regulatory body. As these requirements originate from outside COVRA, they are also referred to as external

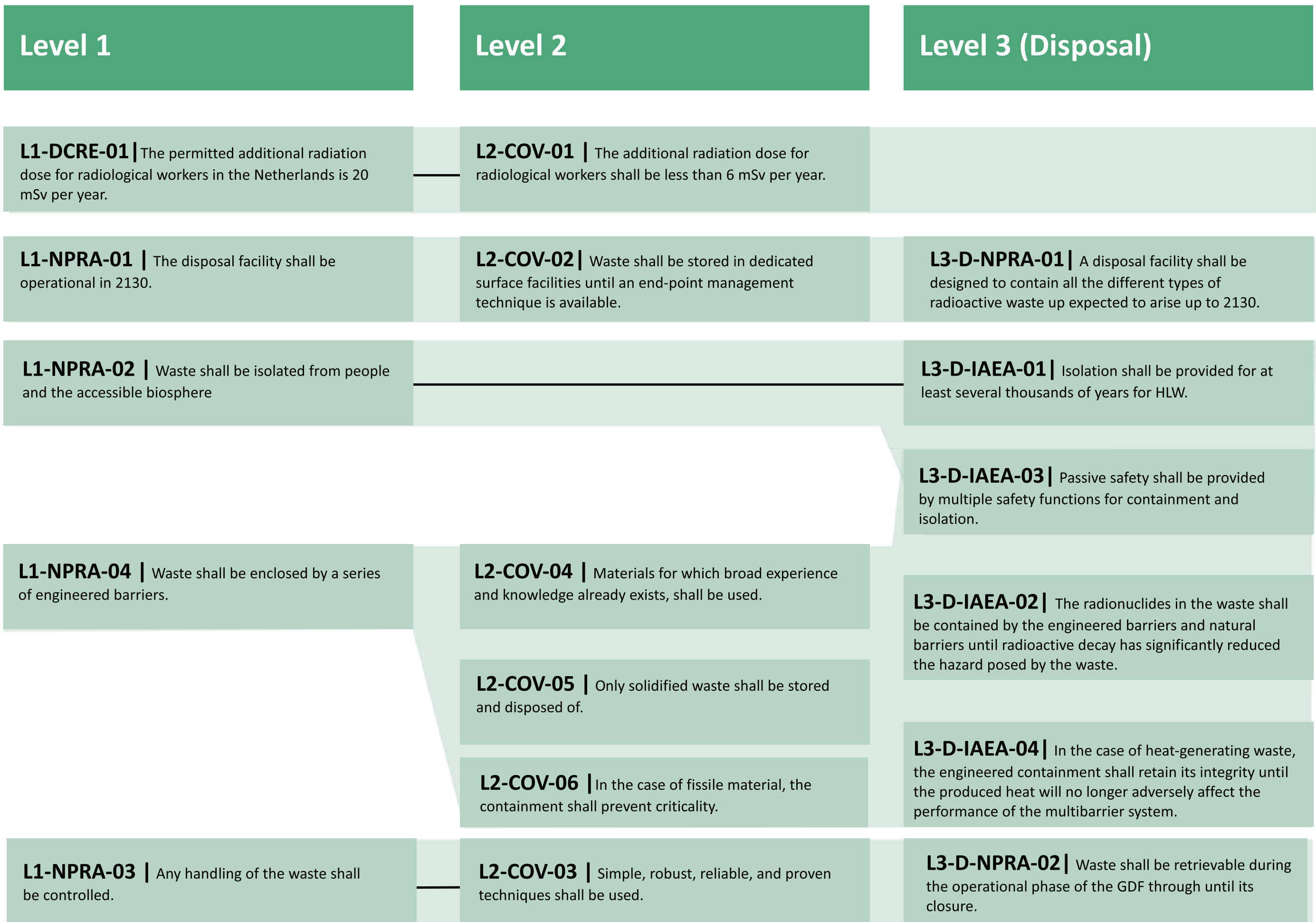


Figure 3.7) Organisation of the requirements at each level and the connection between the requirements described in this Chapter. The black line between some requirements means that the lower level requirement is a refinement of a higher level requirement.

requirements. For abbreviations, DCRE and NPRA are used. DCRE refers to the Dutch Decree on radiation protection and NPRA to the National Programme Radioactive Waste.

3.5.5.2 Level 2: COVRA's requirements

Level 2 comprises the requirements set by COVRA based on its mission and policy, which are referred to as internal requirements. This can include requirements that are adopted by COVRA from outside the organization. An example are requirements derived from the IAEA safety principles. Like Level 1, these requirements apply to the entire system of waste management (predisposal and disposal activities).

3.5.5.3 Level 3: System requirements specific for disposal of waste

At this level, the RMS is split into 4 major categories, namely collection, treatment, storage and disposal. Together, these four activities and represent the entire system of waste management that is the responsibility of COVRA. The requirements listed at this Level are specific for one of these four systems and originate from both regulations (external requirements) and requirements set by the organization based on its policy (internal requirements). Here, only requirements specific to the disposal of waste are described and the current development of the RMS has derived them from the IAEA's SSR-5 (IAEA, 2011a) and national legislation (Ministerie van Infrastructuur en Milieu, 2016). For abbreviations, D, IAEA, and NPRA are used. DCRE refers to the Dutch Decree on Radiation Protection, NPRA refers to National Programme Radioactive Waste and D refers to Disposal.

3.5.5.4 Level 4: Subsystem requirements

Level 4 defines the requirements that a subsystem needs to fulfil to ensure the operational or post-closure safety of the disposal system. Where appropriate, the requirements can be expressed as specific safety functions or performance targets. The sub-system requirements cover the major components of the GDF and its multibarrier system, and activities associated with them, and are grouped into subcategories: shield, contain, isolate, handle and monitor. The sub-system requirements are described in their respective chapters.

3.5.5.5 Level 5: Design requirements

Level 5 comprises requirements on the design of sub-system components and associated activities, which can include quantitative values for the performance target of the subsystem to meet its safety or other function. This value can result from another requirement or be established by research into the characteristics and behavior of the sub-system component. The origin of the quantitative value must be clearly stated in the requirement description, to ensure traceability.

3.5.5.6 Level 6 Design specifications

Level 6 comprises the technical specifications for sub-system components and related activities that define how the Level 5 design requirement can be met. This could be the use of a specified material or a specified design figure, such as the thickness or density of a component. There can be multiple design specifications for a component at level 6, which in some cases must be reconciled with each other. For example, if 10 cm thick steel meets a shielding requirement, but 20 cm is necessary to meet a corrosion requirement related to a containment requirement, then the component will be constructed with the latter thickness. Management of all the design specifications will help to identify which requirements are most critical in determining the design of a subsystem, which in turn allows the sub-system to be optimised.

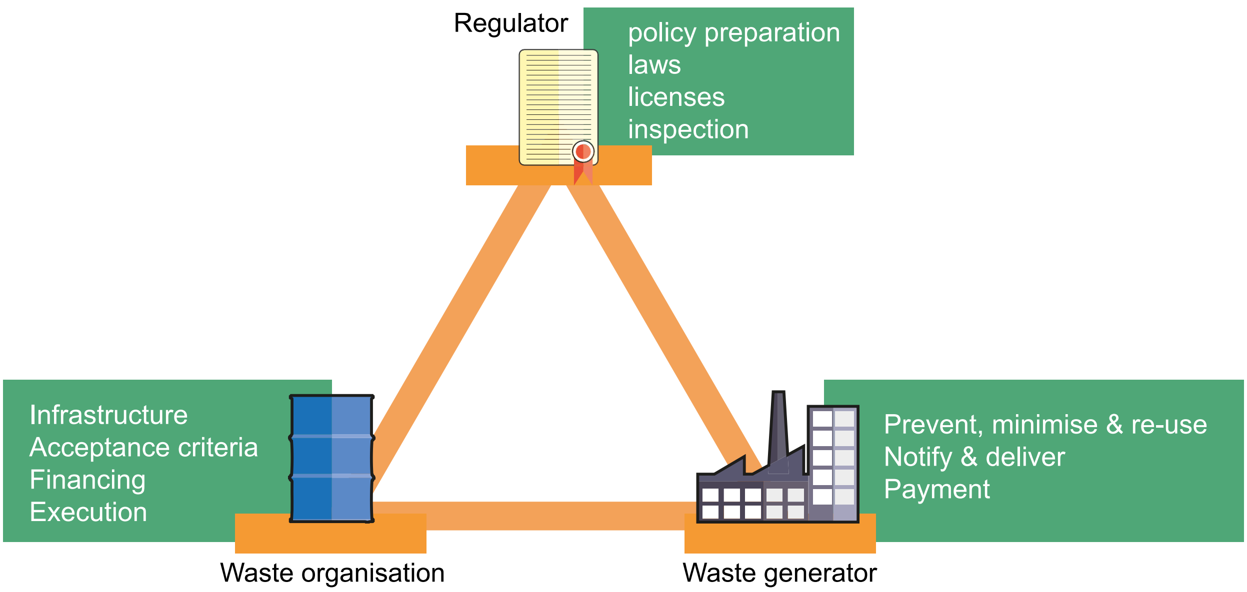
Box 3.1 Organisation of RWM in the Netherlands: roles of the parties

The institutional arrangement of actors with responsibilities in the management of radioactive waste can be viewed as a triangle in which the authorities, waste management organisation and waste generators must each fulfil clearly defined roles that are described below and must exhibit independence from the others. The full range of stakeholders in the Dutch programme also includes the public – both nationally and internationally. The Dutch public is currently kept informed about progress in geological disposal of waste through websites, governmental workshops, guided tours at COVRA's premises and lectures upon invitation by members of Dutch society

Dutch authorities: The authority that prepares policy and establishes laws governing the generation and management of radioactive waste is currently the Ministry of Infrastructure and Water Management. More specifically, the Directorate-general for Environment and International Affairs within this Ministry is responsible for policy development regarding nuclear safety, security and radiation protection (Ministerie van Infrastructuur en Waterstaat, 2020).
The Ministry prepares the national programme to comply with the European Commission Waste Directive (European Commission, 2011) and prepared the last (seventh) national report for the IAEA Joint Convention (Ministry of infrastructure and water management, 2020). The Ministry is also responsible for developing a disposal policy aimed at arriving at a publicly accepted disposal facility.

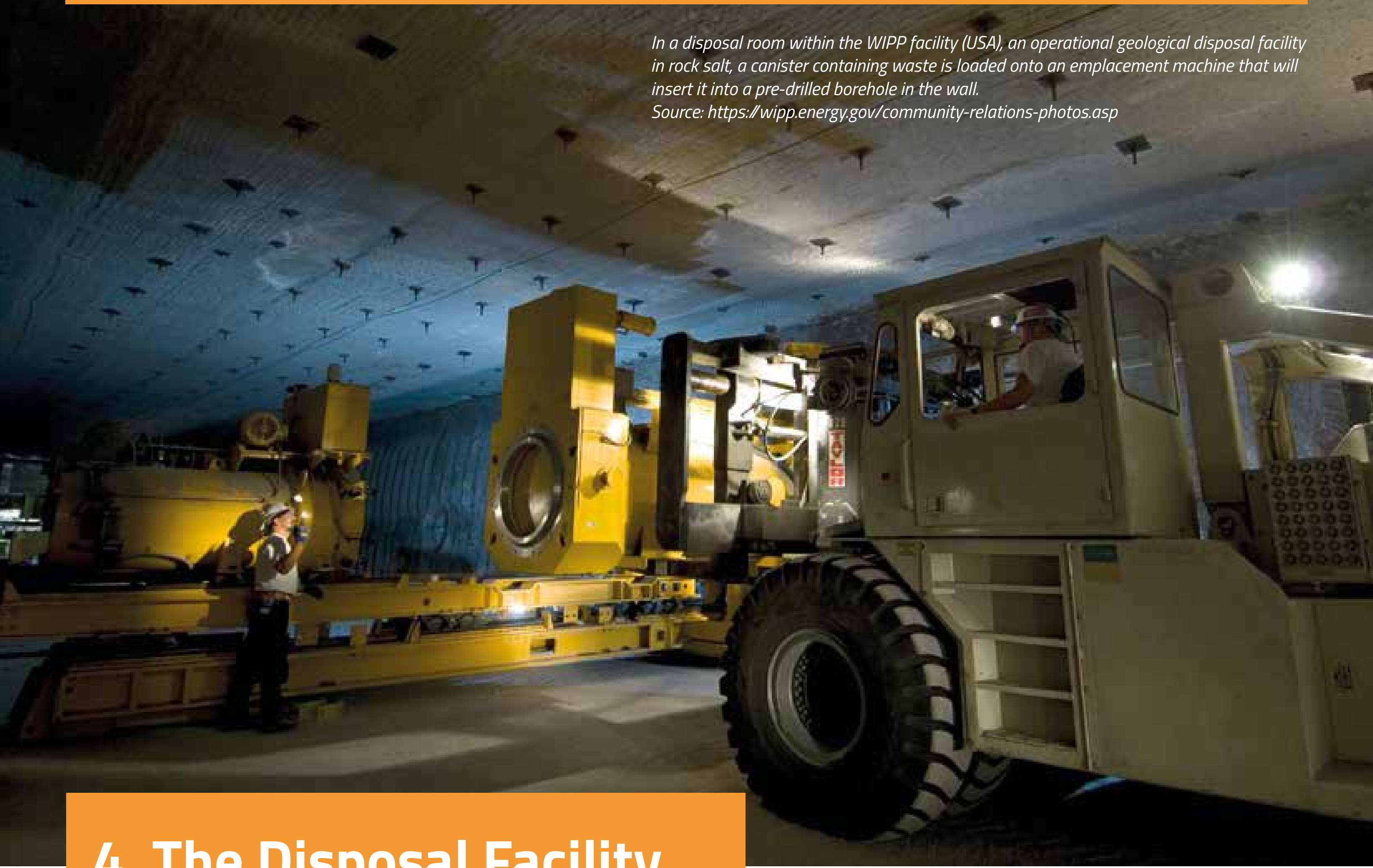
The authority that grants licences and carries out inspections is the Authority for Nuclear Safety and Radiation Protection (ANVS), which was established in 2015. Its responsibility for policy development was transferred to the Ministry of Infrastructure and Water Management in May 2020 (Ministerie van Infrastructuur en Waterstaat , 2020). ANVS focusses on the safety aspects related to the geological disposal facility. Reviewing COVRA's safety cases is therefore an important part of ANVS work. ANVS has a legal responsibility for informing the public about nuclear safety and radiation protection. The Ministry of Infrastructure and Water Management allocates the financial resources for ANVS to carry out its duties.

Waste generator: Waste generators are nuclear power plants, nuclear research reactors, hospitals and research organisations. These generators are required to minimise the generation of radioactive waste as much as is reasonably achievable. Radioactive materials for which no use, re-use or recycling is foreseen, are transferred to COVRA. The waste generators pay COVRA waste fees and notify COVRA of the types and amounts of wastes being



produced. Each generator prepares the waste according to the waste acceptance criteria set by COVRA and submits documentation on the characteristics of the waste.
This documentation has become more detailed, to ensure sufficient characterisation of the waste to be disposed of and to treat the wastes more safely. Discussions with some waste generators are on-going, to allow COVRA to confirm that all relevant details have been provided.

Waste management organisation: The government founded COVRA in 1982, to manage radioactive waste in the Netherlands from collection to an end-point management technique. This technique can be recycling of sufficiently decayed waste or geological disposal for other types of waste; no near surface disposal is planned in the Netherlands. COVRA takes ownership of the radioactive waste when they are delivered and is responsible for development and implementation of the disposal facility. COVRA is charged with implementing all necessary steps to always ensure safety into the future. These include waste collection, treatment and storage, conducting research on geological disposal and implementing final disposal.
COVRA is also responsible for ensuring that the fees paid by waste generators ensure that sufficient funding is available for all future radioactive waste management steps. Every 5 years, COVRA updates its cost estimate for the GDF, considering national developments and the international-state-of-the art, to ensure that the GDF costs will be covered by COVRA's waste fees.



In a disposal room within the WIPP facility (USA), an operational geological disposal facility in rock salt, a canister containing waste is loaded onto an emplacement machine that will insert it into a pre-drilled borehole in the wall.
Source: https://wipp.energy.gov/community-relations-photos.asp

4. The Disposal Facility

Summary:

- A new disposal design is presented for a repository in a salt dome.
- The repository will consist of two levels with the upper level at 750 m for the disposal of (TE)NORM and LILW, while the lower level is for the disposal of HLW and is located at a depth of 850 m.
- About 70 years is foreseen from the start of the construction to the final closure of the repository.
- The total cost for the repository in rock salt is estimated to be 3.5 billion euros

In this chapter, we introduce the waste materials that are destined for geological disposal in the Netherlands and the proposed design of the GDF in rock salt, including the requirements that have driven the design. The chapter then briefly describes how the GDF evolves as part of the deep geological environment. In the last section of this chapter, we discuss the total cost of the COPERA (2020 – 2025) rock salt GDF and potential optimisations.

4.1 The wastes destined for geological disposal

4.1.1 The waste scenario

In the Netherlands, radioactive waste is classified into three different groups, namely Low and Intermediate Level Waste (LILW), Naturally Occurring Radioactive Materials (NORM), which includes Technically Enhanced NORM (TE-NORM), and High Level Waste (HLW). For estimating the total waste in 2130 and how it is accumulated over time, Burggraaff et al. (2022) developed three different scenarios. Here, we discuss only Waste Scenario 1 (Table 4.1). The other two scenarios are discussed in Appendix 4, which also considers how the repository can be expanded if needed. In Waste Scenario 1, Burggraaff et al. (2022) assume that the current nuclear facilities will remain open as planned, while only one new research reactor will be constructed (Pallas). This scenario does not account for the opening of new nuclear powerplants, as currently considered by the Dutch government (Erkens, 2024).

4.1.2 LILW

Low and intermediate level radioactive waste (LILW) originates from activities with radioactive materials or radioisotopes in nuclear plants, industry, research institutes and hospitals.
It includes lightly contaminated materials, such as plastic, metal or glass objects, tissues and cloth. At COVRA, the sizes of the LILW packages are standardised and optimised to ease their handling and LILW is usually processed and conditioned using cementitious materials. In total, there are 4 types of packages currently stored at COVRA, with volumes of 200, 600, 1,000 and 1,500 litres.
The 200 and 600 litre packages consist of painted, galvanised steel drums containing the wastes embedded in cement. The 1,000 and 1,500 litre concrete containers contain a cemented waste form.
At least half of the volume of each package is cementitious material. Most of the LILW packages can be handled safely and transferred to a GDF without significant additional shielding.

LILW is currently assumed to be suitable for disposal without further packaging or conditioning. However, more research is needed. The inventory expected for disposal is estimated in a similar way to that for depleted uranium, i.e. based on linear extrapolation of the rate of wastes entering storage over the past years. This extrapolation covers the life expectancy of the nuclear reactors (HFR and NPP Borssele) in the national programme (Burggraaff et al. (2022)) and leads to an estimated volume of 31,641 m^3 which is contained in 100,000 200 litre drums and 8,400 1,000 litre packages (Table 4.1). Note that in table 4.1, the 600 and 1,500 litre are omitted as they represent only a fraction of the total waste package expected for disposal.

4.1.3 (TE)NORM

(TE)NORM (Technologically Enhanced Naturally Occurring Radioactive Materials) consists of radioactive waste from ores and other raw materials generated in processing industries. The main type of (TE)NORM is radioactive waste (depleted uranium) originating from the uranium enrichment facility of URENCO. Depleted uranium (DU) is converted to a stable oxide and stored in standardised DV-70 containers (see section 6.2.4). After OPERA (Verhoef et al., 2017), the waste generator notified COVRA of increased waste arisings, and a new storage facility for depleted uranium to store all the waste was opened on 13 September 2017. The inventory in the national programme is determined by linear extrapolation of the receipt rate of containers over the past 20 years from the currently stored uranium inventory up to 2050 (Burggraaff et al., 2022). This has resulted into a volume of 49,360 m^3 compared to the 34,000 m^3 estimated in OPERA. This volume requires 12,600 KONRAD type 2 containers which are also currently envisaged for use as the waste packages for depleted uranium.

4.1.4 HLW

The HLW consists of heat-generating waste (vitrified waste from reprocessing spent fuel from the Borssele and Dodewaard nuclear power plants, conditioned spent fuel from the research reactors and spent uranium targets from molybdenum production), together with non-heat-generating active wastes such as hulls and ends from fuel assemblies that have been disassembled during reprocessing. Heat generation is a result of the continuing radioactive decay of the radionuclides in the wastes. As time progresses, the heat output decreases due to the ongoing decay. The relatively short-lived radionuclides contribute most of the early heat output. The concentration of these radionuclides depends on the type of waste, its composition and/or the burn-up of the fuel. HLW normally requires further packaging and/or conditioning prior to disposal. In waste scenario 1, 502 CSD-c, 478 CSD-v (Colis Standard de Déchets-vitrifié) and 244 ECN canisters are expected for disposal (Table 4.1). The 304 litre stainless steel ECN canisters are for unprocessed spent nuclear fuel from research reactors. The CSD-C containers hold the metal parts from the spent fuel assemblies that have been cut up to extract the spent fuel for reprocessing. The CSD-v containers hold the vitrified HLW resulting from the reprocessing of reactor fuel. Compared to OPERA (Verhoef et al., 2017), the number of CSD-v canisters has not changed. The amount of heat generating HLW has been estimated in more detail than other types of waste since special structures (double sided wells) need to be constructed for safe interim storage. Non-heat generating waste can be held in a large, flexible storage space, so uncertainties in estimating the number of canisters have much smaller planning implications. The estimated number of CSD-c canisters has been reduced from 600 to 502, in agreement with the waste inventory determined by Burggraaff et al. (2022).

Waste Scenario 1 - Current installations +

Type	Volume in storage (m^3)	Number of canisters / containers in storage	Number of canisters / containers for disposal	Volume for disposal (m^3)
200 L drums	38,141	100,000	100,000	31,461
1000 L Containers		8,400	8,400	
Decommissioning waste	3,814	-	826	3,814
(TE)NORM	49,360	-	12,600	58,070
CSD-c	90	502	84	504
CSD-v	86	478	80	530
ECN-cansiter	49	244	122	643

Table 4.1) The expected number of waste packages for disposal in Waste Scenario 1.

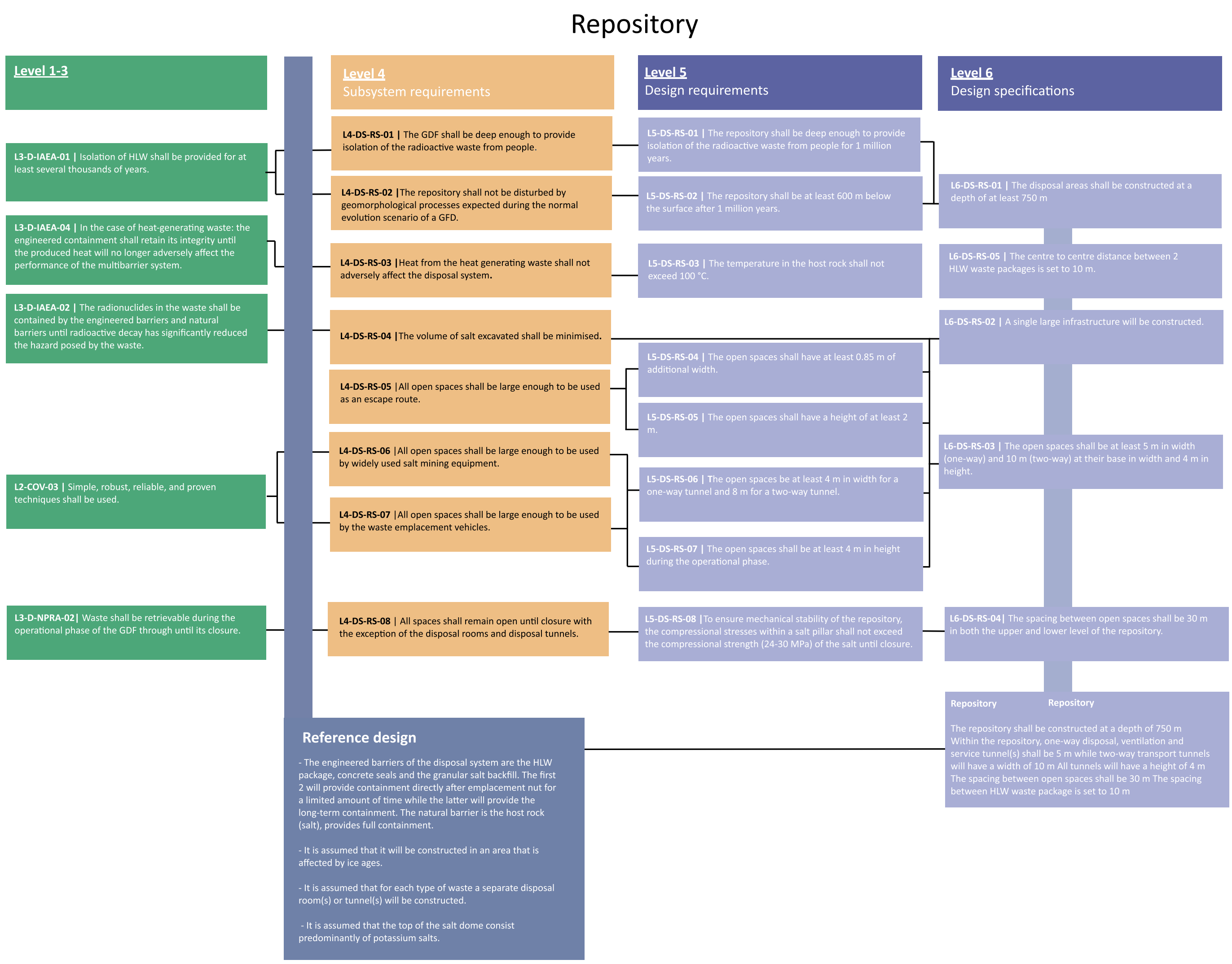


Figure 4.1) The RMS of the disposal facility. DS stands for Disposal in Salt; RS stands for RepoSitory.

4.2 The COPERA (2020 – 2025) GDF in salt

It is assumed that the COPERA (2020-2025) GDF described below will be located within a generic salt dome in the geological formations called the Zechstein Group. The Zechstein Group was selected as many of the domes that it contains are considered potentially suitable for disposal (see Chapter 5). Furthermore, there is a considerable amount of data available on, and practical experience in, the Zechstein Group. However, the disposal concept can be adapted to other salt structures, such as salt sills or bedded salt formations (Chapter 5). In a bedded salt formation, for example, it may be simpler to place the repository on a single level, making optimal use of the horizontal extent of a bedded salt formation. Multiple levels may also be possible, depending on the dimensions and geology of a salt structure.

Before discussing the layout of the COPERA (2020 – 2025) disposal facility, we shortly discuss the requirements in COVRA's current RMS that, in part, determine its layout. In total, there are 5 Level 3 requirements that directly affect the design of the repository. The first is that the - *Isolation of HLW shall be provided for at least several thousands of years (L3-D-IAEA-01).* This requirement influences the choice of the depth of the repository via L4-DS-RS-01 and L4-DS-RS-02. The second and third are - *The radionuclides in the waste shall be contained by the engineered barriers and natural barriers until radioactive decay has significantly reduced the hazard posed by the waste (L3-D-IAEA-02). In the case of heat-generating waste: the engineered containment shall retain its integrity until the produced heat will no longer adversely affect the performance of the multibarrier system (L3-D-IAEA-04).* These 2 requirements together with an additional operational requirement on Level 4 - *All open spaces shall be large enough to be used as an escape route (L4-DS-RS-05)* – determines the dimensions of the open spaces and the techniques used to make them. The dimensions are in part also determined by the Level 3 requirement - *Simple, robust, reliable, and proven techniques shall be used (L2-COV-03).* This implies that the chosen mining, transport and emplacement vehicles (e.g., emplacement vehicle developed by Posiva Oy (Finland)) need to have sufficient room to manoeuvre in the underground facility. The last requirement is that - *Waste shall be retrievable during the operational phase of the GDF through to its closure (L3-D-NPRA-02).* This requirement determines the spacing of the different open spaces within the repository, so they remain stable. In the section below, we will show how these requirements are met.

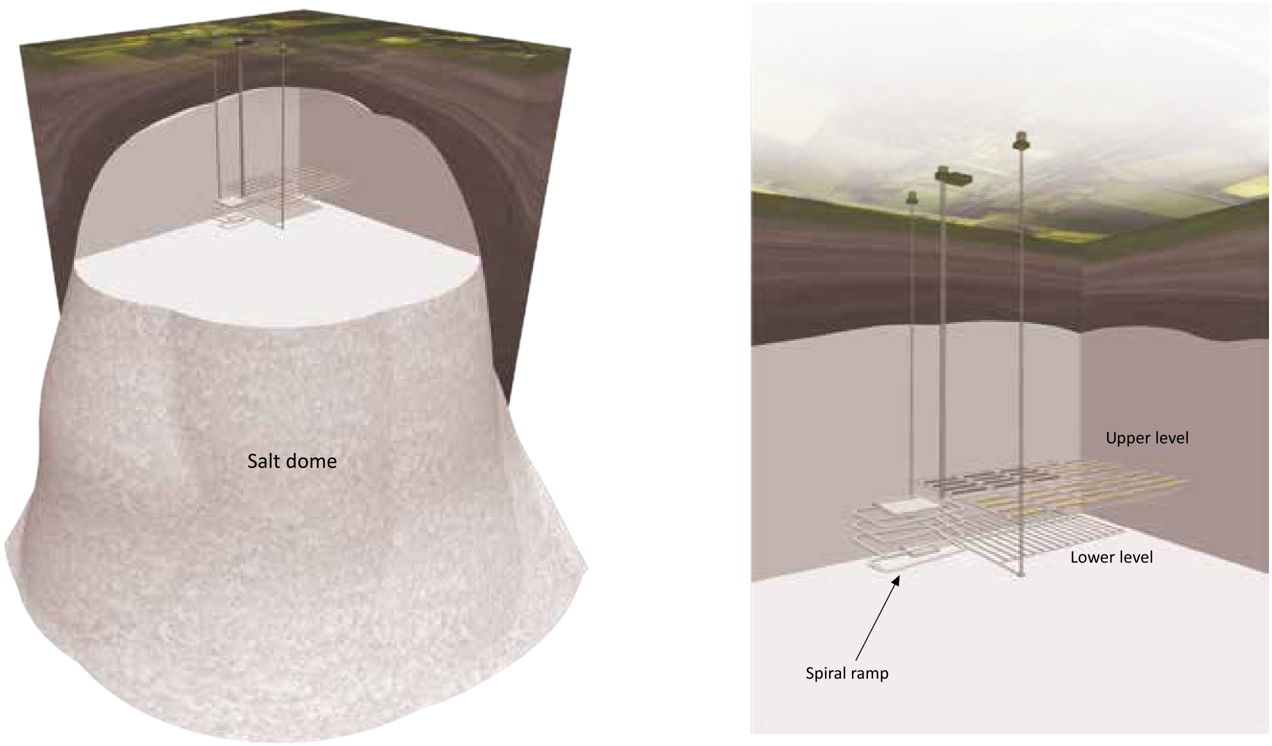


Figure 4.2) The general layout of a two-level repository in a generic salt dome. The upper level will be used for the disposal of LILW and (TE)NORM while the lower level will be used for the disposal of HLW. While this repository design is for a repository in a salt dome, it can be adopted for other salt structures, although the repository may then consist of a single layer rather than two, to make optimal use of the horizontal extent of, for example, a bedded salt formation

4.2.1 Repository Layout

The COPERA (2020 – 2025) repository in a salt dome of the Zechstein Group is, as in previous salt disposal concepts (Van Hattum en Blankevoort, 1986) in the Netherlands, envisaged to consist of 2 levels (Fig. 4.2a and b). The upper level will be located at a depth of 750 m while the lower level will be separated by 100 m at a depth of 850 m. The depth was selected to ensure that the repository provides isolation (Fig. 4.1) and that it will not be disturbed by geomorphological processes at the surface (Fig. 4.1 and Chapter 5) and potential future glacial erosion channels. The two-level design of the repository has multiple advantages. First, it makes optimal use of the large vertical extent (several kilometres) of a salt dome compared to its horizontal extent (several hundred of metres). Second, having a two-level repository reduces the footprint (dimensions in the horizontal plane) of the GDF. This in turn reduces the likelihood of inadvertent human intrusion by drilling a borehole through, or near the footprint of a GDF (IAEA, 2017a). However, it does not reduce the consequences of inadvertent human intrusion if it occurs. Lastly, having a two-level repository will maximise the horizontal salt thickness surrounding the waste. The repository is connected to the surface by three shafts and the two levels are connected by a spiral ramp.

4.2.2 Surface facilities

Surface facilities are required for receiving, inspecting, and conditioning the waste and to support the construction, operation, and closure activities of the repository (Fig. 4.3). To fulfil these requirements, the surface facilities will be separated into a radiological controlled and a non-controlled area. All the waste handling will take place in the controlled section. A buffer store will allow some waste to be held temporarily to ensure a steady supply through the conditioning facilities into the GDF. In the non-controlled section of the surface facilities, other work will be carried out, for example, the mixing of concrete for conditioning the (TE) NORM waste. Also, a processing and storage facility will be constructed for conditioning (i.e., reducing the grain size) of excavated salt and for its temporary storage before use for back-filling. In addition, a visitor centre will allow the public to gain a better understanding of the GDF.

Figure 4.3) An impression of what the surface facilities might look like in the UK. Figure from Nuclear waste services (2024).

4.2.3 Shafts

The repository is connected to the surface by three circular-section shafts down to and into the salt dome (Fig. 4.2). The transport shaft has an outer diameter of 8 m and there are 2 shafts with a diameter of 5 m that are used for both inward ventilation and personnel transport, as well as acting as emergency escape routes. The transport shaft is used to move equipment and waste to the repository, to remove excavated salt and acts as the outlet for ventilation air. Thus, fresh air from the surface will be transported via both ventilation shafts into the repository and will flow back to the surface via the transport tunnel and transport shaft. Before being released into the biosphere, the air will pass through filters and detectors to ensure that any, unlikely, operational period radionuclide releases will be detected. All shafts will have a circular section, which provides significant advantages for geomechanical stability compared to a rectangular section, since it avoids stress concentrations at the corners (Herold and Leonhard, 2023a).

All three shafts will be sunk vertically, directly to and through the GDF, as opposed to options involving lateral access to the salt dome from adjacent shafts in the surrounding formations.
A vertical shaft through the salt dome minimises the distance both materials and the waste need to travel. Additionally, roof space voids that develop in horizontal backfilled openings are avoided by using vertical shafts. These gaps will be at most tens of centimetres wide between the granular salt backfill and the tunnel roof and they take a few decades to close, depending on the convergence rate of the host rock (Morsleben 1 - 2 mm/year; Gorleben up to 7 mm/year; Bracke and Fischer-Appelt, 2013; Buchholz et al., 2020; Fischer-Appelt et al., 2013). For the closure of the repository, vertical shafts backfilled with salt (for long term containment) and other materials that provide short term containment are therefore preferred, especially as the compacted salt becomes impermeable (see section 6.1.1).

Before reaching the salt dome, the shafts will pass through a caprock, when present. This may affect any presumed protection against subrosion provided by the caprock. However, there is an ongoing debate on whether all caprocks do indeed provide additional protection (See section 5.2.4.1, Geluk et al., 1993). Furthermore, the three shafts affect only a very small part of the caprock, and the shafts will be sealed with material that aims to provide an equal level of protection to the natural caprock.

The two disposal levels will be connected by a spiral ramp with an inclination of 3% and a total length of 1920 m (Herold and Leonhard, 2023b), so that only one large infrastructure area and one loading area will be needed (Fig. 4.2). This will minimise the amount of salt that needs to be excavated and thus minimise damage to the host rock (Fig. 4.1 and 4.2, L6-DS-RS-02). Nevertheless, for efficient operations, the lower level will have a small complementary infrastructure area.

4.2.4 Lower level

4.2.4.1 Layout of the lower level

The lower level consists of a small infrastructure area, a central transport tunnel, a service tunnel and two ventilation tunnels, one on each side of the transport tunnel (Fig. 4.4a). The transport and ventilation tunnels are connected via disposal tunnels and a service tunnel. All these open spaces will be broadly rectangular in section but shaped to reduce the peak stresses around the corners (Fig. 4.4b) and their dimension will be minimised, as the rock salt has a containment safety function. The disposal, ventilation and service tunnels have a height of 4 m, a width of 5 m at the base and a maximum width of 5.6 m (Fig. 4.4b). The height of 4 m (Fig. 4.1, L5-DS-RS-07) is needed as the vertical height of the HLW disposal overpack is 3.4 m (See section 6.2.1). The remaining 0.6 m is needed to account for additional height required for lifting the disposal overpack from the ground (tens of centimetres), the expected convergence of the open spaces and an operational flexibility margin (tens of centimetres). The width of open spaces is based in part on the required width of mining equipment (Herold and Leonhard, 2023a). For example, a Sandvik MR361 road header has a width of 3.1 m while a scaler (ghhrocks LF-7.6HB or ghhrocks LF-20H) requires a width of between 3.3 and 3.6 m. Allowing an operational flexibility margin, a width of 4 m is assumed to be needed for equipment (L5-DS-RS-06, Fig. 4.1). In addition, workers should be able to walk or escape around the equipment. These routes must have a minimum height of 2 m and a minimum width of 0.85 m is required (L5-DS-RS-04 and L5-DS-RS-05, Fig. 4.1, wetten.overheid.nl, 2024). Together with the expected convergence, the width of the tunnels (at their base) is therefore set to 5 m for one-way tunnels and 10 m for two-way tunnels (L6-DS-RS-03). Similar dimensions are suggested by Van Hattum en Blankevoort (1986), Heijdra and Prij (1997) and, in the German programme, by Bertrams et al. (2020a). While their width and height are similar, the tunnels have different lengths: 230 m (disposal and service tunnels) and 530 m (ventilation tunnel). To account for the turning radius of equipment, the intersections are all curved (Fig. 4.4a inset, Herold and Leonhard, 2023a). Note that the dimensions of the open spaces are essentially a trade-off between having large enough open spaces and minimising the amount of salt removed (Fig. 4.1).

For efficiency reasons, transport through the transport tunnel should be possible in both directions concurrently. Therefore, the transport tunnel has a width of 10 m (L6-DS-RS-03) but the same height and length as the ventilation tunnels. This is about the same size as the transport tunnel in the WIPP repository (12 m) and in the German disposal concept (Bertrams et al., 2020a).

The small infrastructure area at the lower level is needed to store equipment that is needed on a day-to-day basis during the operational period. This will increase the efficiency of construction work, as there is no need to go to the upper level for equipment that is needed regularly. Note that the infrastructure area at the lower level is significantly smaller than the one at the upper level and does include a salt bunker that is used to store excavated salt temporarily.

The total number of disposal tunnels needed depends on the total amount of HLW waste packages and the distance between them in the tunnel. Assuming a centre-to-centre distance of 10 m between HLW packages, 30 disposal tunnels will be needed.
This distance is determined predominantly by operational, rather than thermal load requirements. This is based on the rest-angle slope of the loose backfill during its emplacement; about (30 - 40 degrees, which implies a salt covered floor length of about 3-4 m. The size of the emplacement device is not yet defined, and a safety margin will be required that will likely be about 2 m (Fig. 4.4c). Ensuring that the HLW tunnels are filled as completely as possible with granular salt backfill reduces the time needed for the compaction of the backfill material. The limited heat output of the waste due to the long storage period, combined with the high thermal

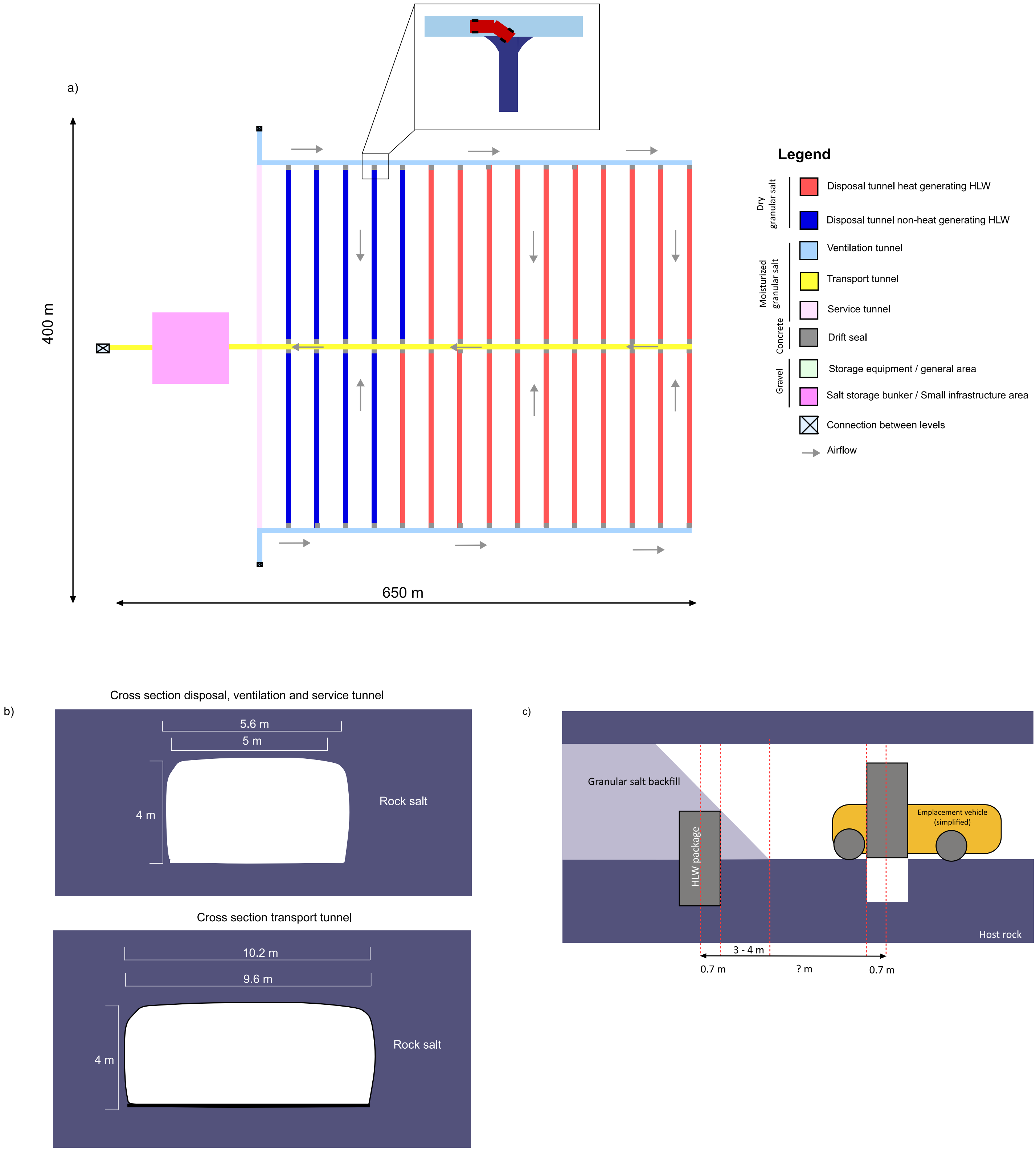
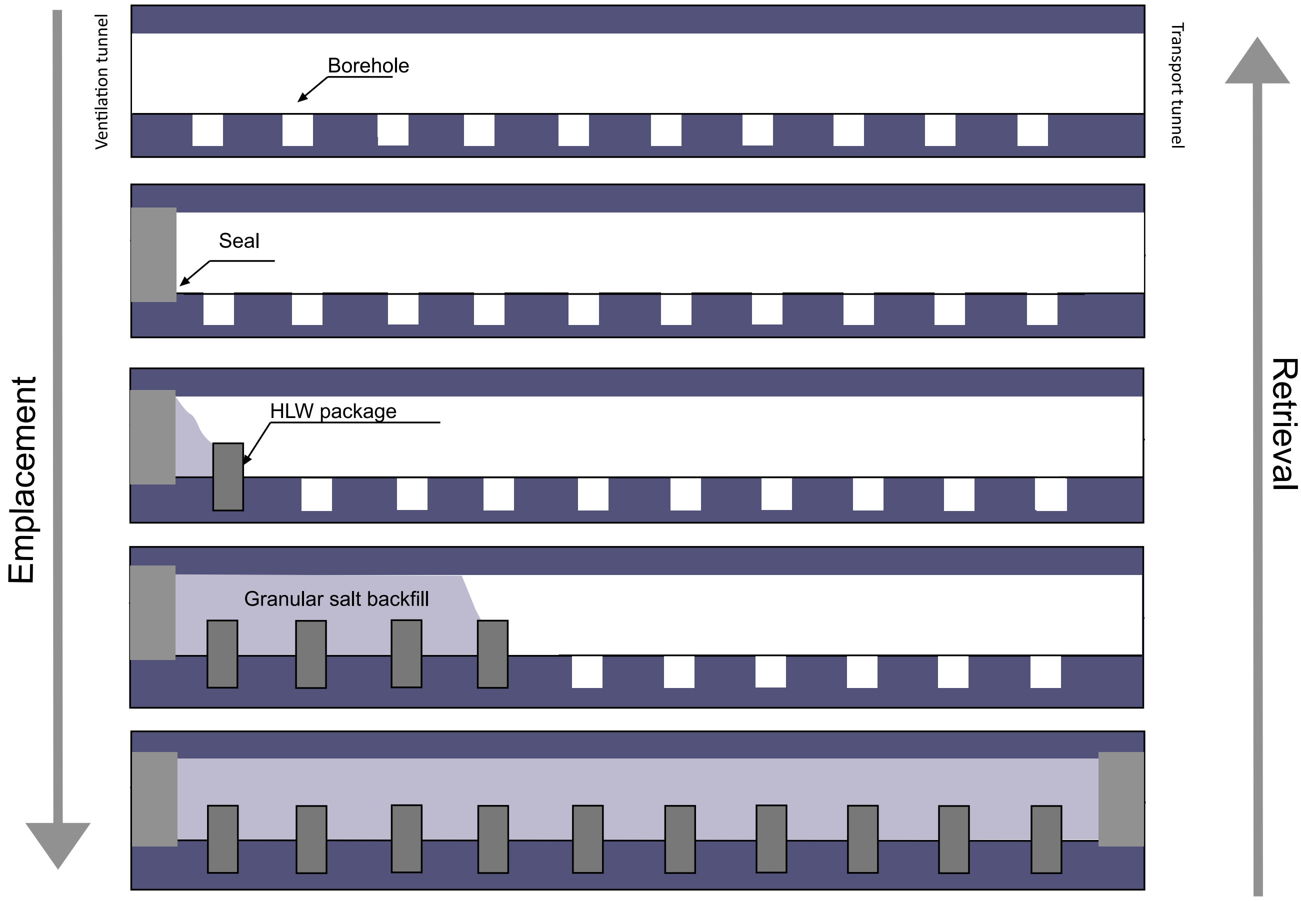


Figure 4.4a) Layout of the lower level. The lower level consists of a transport tunnel, a service tunnel, two ventilation tunnels and tunnels for the disposal of heat generating HLW (red) and non-heat generating HLW (dark blue). In the inset, the rounded corners at intersections are shown. These rounded corners are needed to ensure that mining equipment can turn round the corners. b) Shape and dimensions of the tunnels in the lower level. These tunnels are shaped to reduce the peak stress around the corners, and their dimensions are limited to minimise the impact on the host rock. c) The centre-to-centre distance needed to ensure that HLW packages can be emplaced in a disposal tunnel.

a)



b)

Figure 4.5a) The emplacement of a HLW package in a disposal tunnel. After the emplacement of the HLW package, part of the tunnel will be backfilled. For retrieval, the process is essentially reversed. The drawing is at scale, except for the centre – centre distance between the packages: the distance shown on the figure is smaller than the actual distance between two HLW packages. b) The emplacement vehicle developed by Posiva Oy (Finland). While this is not developed for a disposal concept in which the HLW package will be transported vertically, it has the same functionality required for our reference GDF design.

conductivity of salt (e.g., Hunfeld et al., 2023), limits the temperature rise within the repository: salt domes tend to have a lower temperature compared to surrounding formations at the similar depth (Bonté et al., 2012). It is therefore not likely to be a limiting factor in the centre-to-centre distance a few decades to hundreds of years after the closure of the repository when the granular salt backfill will have regained the same thermal properties as the host rock (Smit, 2022). However, more research is needed on thermal effects at shorter times, since the initial thermal conductivity of the granular salt backfill (due to its high porosity) is significantly lower (by a factor of 3 - 5) than the host rock, resulting in higher temperatures within the backfill (Bechthold et al., 2004). At present, a conservative 10 m centre-to-centre distance between HLW waste packages is assumed, but this may be reduced with more research (L6-DS-RS-05). Furthermore, the assumed maximum allowable temperature of 100 degrees Celsius, to stay below the boiling point of water, might be increased: the maximum temperature for a repository in salt is set to 160 °C in Germany (Czaikowski et al., 2024). During the operational period, active ventilation will ensure that working conditions, including the temperature, remain optimal.

To ensure the mechanical stability of the repository, the compressional stresses within a salt pillar (the unexcavated salt between tunnels) should not exceed the compressional strength of the salt. Based on the depth of the repository (850 m, see Chapter 5), 22 – 25 MPa of stress is expected inside a salt pillar (Herold and Leonhard, 2023a; Müller-Hoeppe et al., 2012). Within the Morsleben and Asse mines, the strength of the salt varies between 24 – 30 MPa (L5-DS-RS-08, Herold and Leonhard, 2023a and references therein). While the compressional stresses within the salt pillar could in some cases exceed the lower range of the compressional strength range of salt pillars, at this stage of research we assume that the 30 m distance between tunnels, as used in these facilities, is sufficient (L6-DS-RS-04). Within the ongoing planning process, as knowledge about an actual site increases, more detailed investigations and numerical analysis of the mechanical stability will be carried out, to lead to a more optimised design. As in the case of package spacing, the limited heat output of the waste will not likely be the limiting factor deciding the spacing between tunnels since the maximum temperature rise is only a few degrees (Smit 2022). However, more research will be needed, as Smit (2022) assumed that the backfill has the same thermal properties as the host rock which, as noted above, is not the case in the early period when it still has a high porosity.

In total, the lower level of the repository will be 650 m in length and 400 m in width: a total surface area of about 0.26 km^2

4.2.4.2 Lower level waste emplacement

For disposal in the lower level, a specifically designed HLW disposal package will be used. The requirements on and design of these packages are described in Chapter 6.

The HLW package can contain up to 6 CSD-v or CSD-c canisters, or 2 ECN canisters for SNF. The HLW package for the CSD-v or CSD-c canisters will have a length of 3.387 m, a diameter of 1.540 m, and a weight of approximately 29,000 kg (empty). The HLW package for the 2 ECN canisters has a height of 3.102 m, a diameter of 1.460 m, and a weight of 29,000 kg (empty). Including the wastes, an ECN waste package for disposal has a weight of around 31,000 kg and the CSD-v/CSD-c container has a weight of 35,000 kg. While this is significantly more than the OPERA supercontainer (Verhoef et al., 2017), the emplacement of the much heavier Pollux container (65,000 kg) has been successfully demonstrated (Filbert and Engelhardt, 1995) in the German programme. In addition, shaft hoisting equipment with a payload of 80,000 kg was developed and successfully tested (Filbert and Engelmann, 1994). This experience implies that the waste package for disposal developed for the Dutch repository can be transported and emplaced.

Disposal of HLW will not begin until the entire lower level has been constructed. The design concept places the HLW packages into shallow boreholes in the tunnel floor. Tunnels will be filled sequentially, and the disposal holes will be drilled within the currently operational disposal tunnel and covered with lids, to ensure the safety of the workers prior to package emplacement. These boreholes will be 1500 mm deep, about half the length of the HLW packages, and are intended to prevent tipping of the packages. Furthermore, having direct contact between part of the HLW package and the host rock will help to reduce the local near-field temperature within the repository when the granular salt backfill still has a relative high porosity and thus low thermal conductivity, so more heat is transferred through the host rock (Bechthold et al., 2004, L. Schaarschmidt, personal communication, 12 December 2023). Emplacing the HLW packages vertically also reduces the footprint of the HLW, thus reducing the likelihood (IAEA, 2017a) of inadvertent human intrusion (drilling through the waste), and no tilting of the HLW package (from vertical to horizontal) is needed and it can be carried vertically in the emplacement vehicle.

Before commencing disposal operations in a disposal tunnel, a seal will be constructed between the ventilation tunnel and the disposal tunnel (Fig. 4.5a). Emplacement of HLW begins with lowering a HLW package via the transport shaft to a reception area in the repository and placing it vertically in an emplacement vehicle (Fig. 4.5a and b). This emplacement vehicle carries the waste container through the transport tunnel to the disposal tunnel and lowers it into the vacant disposal borehole furthest from the transport tunnel. After emplacement, salt that has previously been excavated and prepared is used to cover the waste package and partially backfill the disposal tunnel (dry granular backfill). This procedure will continue until a disposal tunnel is full, at which time the concrete seal between the transport and the disposal tunnel will be constructed. The concrete seals at both ends of the disposal tunnel will provide containment immediately after construction and increase the compaction rate of the granular salt backfill within the disposal tunnel by providing a counterpressure. This ensures that the granular salt backfill cannot leave the disposal tunnel and will be compacted due to convergence of the host rock. In addition, the seals provide physical separation between the waste and workers during the operational phase.

If a HLW package needs to be retrieved (Fig. 4.5a), the seal at the transport tunnel and the backfill in a disposal tunnel need to be removed. This can be done using the same equipment that has been used for the construction of the GDF, with care being taken not to damage the HLW packages. When the backfill surrounding the waste overpack is removed, the emplacement vehicle engages with the HLW packages and lifts it from the borehole, then delivers it back to the shaft where it is hoisted to the surface. Note that excavation of a HLW package might be needed prior to its lifting from the borehole due to convergence of the salt.

4.3.1 Upper level

4.3.1.1 Upper level layout

The upper level also consists of a central transport tunnel with two parallel ventilation tunnels, connected by service tunnels. As in the lower level, the transport tunnel has a width of about 10 m to accommodate two-way transport of materials, workers and waste. The ventilation and service tunnels have a width of about 5 m, as they are only needed for one-way transport of workers. Disposal rooms will be constructed at right angles to the disposal tunnels and are modelled on the disposal rooms in the WIPP, with a width of 10 m, a length of 110 m and a height of 5 m. The room length available for disposal is somewhat smaller, as 5 m thick concrete seals will be constructed at either end of a disposal room. Each disposal room is thus effectively 100 m long. The seals will be constructed after all the waste is emplaced and after the backfilling of a disposal room. In the short term, the concrete seals will provide a physical barrier between the workers and the waste. In the long term, they provide containment.

There are multiple options for backfilling the disposal rooms. The first option is not to backfill: however a disadvantage is that the waste will eventually disintegrate, and packages could be displaced by collapse of stacking and/or by uneven convergence of the host rock. This could lead to impairment of the concrete seals at each end of a disposal room. In addition, retrievability of the waste after the observational period might become difficult (but not impossible, see section 2.5.1). Options considered for backfills are either re-use of excavated salt, providing a material that compacts (granular salt), or using a harder non-compactable material (sorel concrete, salt concrete or gravel). Compacting backfill will somewhat restrict the displacement of the waste packages. On the other hand, with non-compacting backfill (gravel or concrete), waste will remain stabilised in place and can be retrieved during the operational

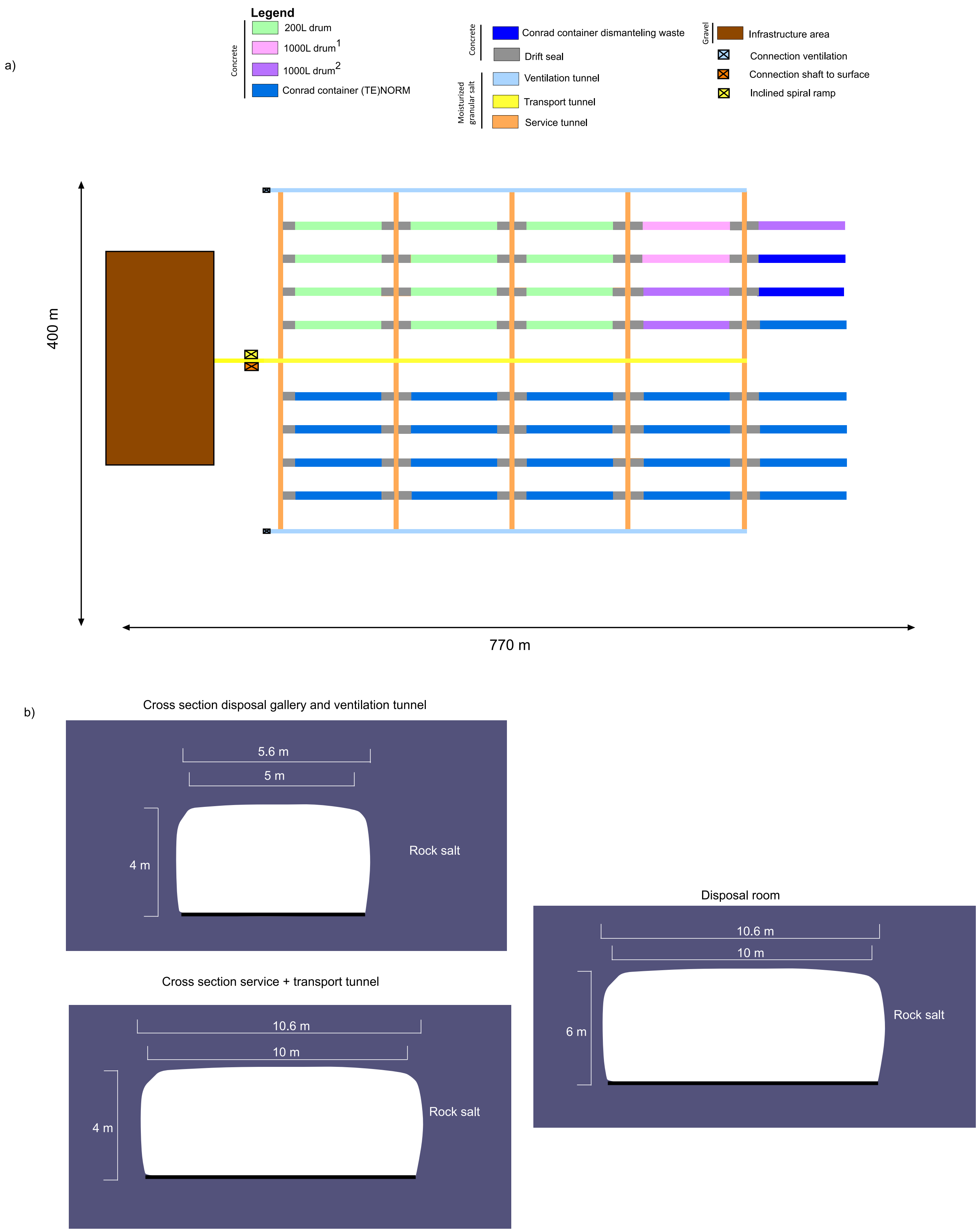


Figure 4.6a) Layout of the upper level. The upper level consists of a transport tunnel, 2 ventilation tunnels and disposal rooms for different types of wastes. The colour of a disposal room indicates which type of waste will be placed in them. b) Shape and dimensions of the tunnels in the upper level. These tunnels are shaped in such a way as to reduce the stress peaks around the corners. 1 stands for the disposal rooms for resins immobilized liquid waste and 2 for the disposal rooms for molybdenum waste liquid.

period and for some period after closure, albeit with significant effort. To ensure that the entire disposal room is filled, it is for now assumed that some form of soft excavatable concrete (salt or sorel) will be used, as it can be pumped into a disposal room. In contrast, it is much more difficult to ensure that all the open spaces between the waste can be fully backfilled with gravel.

As in the lower level, the tunnels and disposal rooms will be slightly oval shaped, to reduce stress peaks in the salt host rock (Fig. 4.6b). The disposal rooms are separated by 30 m pillars and are assumed to be stable, based on experience in German GDF projects (Fig. 4.6a, Herold and Leonhard, 2023a).

The main infrastructure area for the repository will be constructed on the opposite side of the disposal rooms (see Fig. 4.6a). This area includes a mechanics workshop, material depot, personnel break rooms, equipment for dose rate measurements and decontamination, storage areas for vehicles, vehicle workshop, battery loading room, electricity supply room, transformer station, surveyors' office and bunker for backfill. Based on a first estimate, the infrastructure area will occupy about 15,000 m^2 (120 by 125 m), of which only 5,000 m^2 (70 by 70 m) will be open. The rest of the area is taken up by the supporting salt pillars (Herold and Leonhard, 2023a). The exact layout of this large infrastructure area can be decided in the future.

The footprint of the upper level be about 770 m by 400 m: a total area of about 0.31 km^2.

4.3.2 Upper level waste emplacement

4.3.2.1 Disposal of 200L drums

Nine thousand 200L drums can be emplaced in each disposal room (see Figure 4.7 and Figure 2 in Appendix 4). The 200L drums will be placed horizontally in racks, as they are currently stored at COVRA. An electric forklift truck, as currently used at COVRA, will be employed for the emplacement of this waste. In Waste Scenario 1, 12 disposal rooms are required for the 200 L drums.

4.3.2.2 Disposal of 1,000L drums

2376 1,000L containers can be emplaced in each disposal room (see Figure 4.7 and Figure 2 in Appendix 4). They will be stacked similarly to the way in which they are currently stored at COVRA. The number of drums stacked on top of each other will, however, be less, due to the height of the disposal rooms. An electric forklift truck will be used for emplacement. Five disposal rooms are expected to be needed for the 1,000L containers for Waste Scenario 1.

4.3.2.3 Disposal of Konrad Type II Container

The Konrad type II container is used for the disposal of (TE)NORM and decommissioning waste. For the disposal of (TE)NORM, 9060 Konrad Type II containers are needed after the waste has been conditioned, requiring 14 disposal rooms, independent of the waste scenario. 660 containers will be placed in each disposal room: 55 by 6 containers, stacked in layers (see Figure 4.7 and Figure 2 in Appendix 4). A heavy forklift truck will be used for the emplacement of the containers. Currently, there are heavy forklift trucks that can lift more than 20,000 kg, the maximum weight of a Konrad Type II container. For the disposal of the decommissioning waste, 2 disposal rooms will be needed in Waste Scenario 1. As with (TE)NORM, the waste will be emplaced using a heavy forklift truck.

4.4 Final closure of the repository

The observational period begins when all the waste is emplaced. During this period, the ventilation, transport and service tunnels, at both the upper and lower level, and all shafts and disposal tunnels will remain open. This also includes the small infrastructure area (lower level) and the large infrastructure area (upper level).

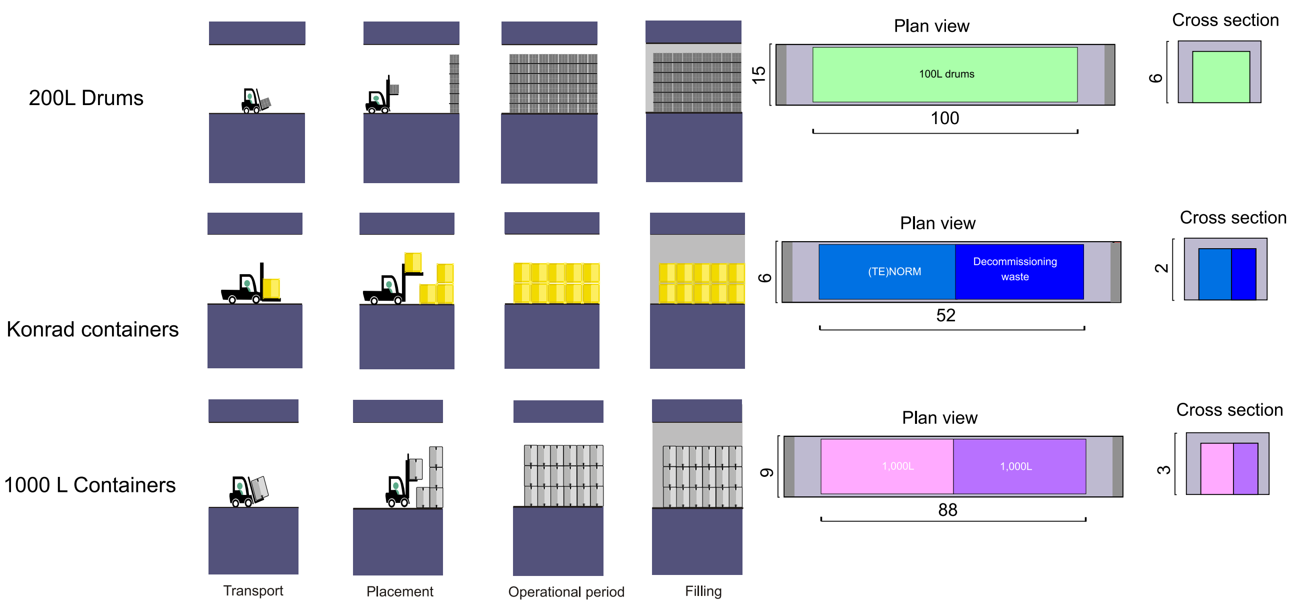


Figure 4.7) Disposal of the 200L, 1,000L and the Konrad containers in the disposal rooms. The upper figure shows how the 200L drums will be disposed in a disposal room. The crown space, space above the waste package and the roof of the disposal room, is large in the disposal rooms for the Konrad containers and the 1,000L drums. The space above between the waste packages and the ceiling is not yet optimised.

These openings will all be backfilled at the end of the demonstration period, before the final closure of the repository. For the ventilation, transport and disposal tunnels, moisturised granular salt will be used as backfill to increase the compaction rate (see section 6.1.1). Gravel will be used as a backfill material for the salt bunker and both infrastructure areas. These areas can act as a reservoir for gas that could be generated within the repository. Additionally, they will delay possible fluid migration from the shaft towards the disposal rooms and tunnels. The shaft closures will comprise backfill material and seals. The seals will be constructed from diverse types of materials that, for a limited amount of time after placement, provide complete containment (e.g., concrete and/or composite seals). The backfill will eventually, after compaction, provide long-term containment and is envisaged to be moisturised granular salt. The exact materials used, and their thickness, will depend on local geological conditions (see section 6.1.5).

4.5 Schedule

Implementation of this two-level repository with three shafts can be divided into 11 phases (Fig. 4.8). Above ground, these are (1) site characterisation and preparation and (2) construction of the surface facilities. Subsurface construction and emplacement of the waste can be divided into
(3) shaft sinking, (4) excavation of the infrastructure area and spiral ramp, (5) excavation of the lower level, (6) emplacement of the HLW waste, (7) excavation of the upper level, (8) emplacement of the LILW and (TE)NORM waste, (9) observation period and (10) final closure of the repository. While the different phases are discussed in sequence, some will overlap. For example, the construction of the shafts starts when the above ground facilities are still under construction, and this reduces the total construction time for the repository. Note that, for the schedule, it is assumed that all waste will be delivered to the disposal site on time for disposal.

To have an operational repository in 2130 (Herold and Leonhard, 2023b), activities will have to begin earlier. Herold and Leonhard (2023a) examined these activities, starting with (1) site characterisation and (2) licence application, followed by site preparation and (3) construction of the surface facilities. Once the basic site preparation is finished, subsurface construction can begin, starting with shaft sinking (4). About 4 years are needed to construct the large transport shaft and 3 years for the smaller diameter ventilation shafts. The excavation of the transport shaft is therefore started 1 year before the construction of the 2 ventilation shafts. A freezing technique is used for construction of the shafts in the overburden. After the shafts have been constructed, the infrastructure area and the upper-level service tunnel are constructed (5). This is followed by the spiral ramp, service tunnel, small infrastructure area and salt bunker of the lower level. The service tunnel is constructed to ensure steady airflow, while the infrastructure areas are needed for supporting activities such as maintenance of the underground equipment, storage rooms or utility supply.

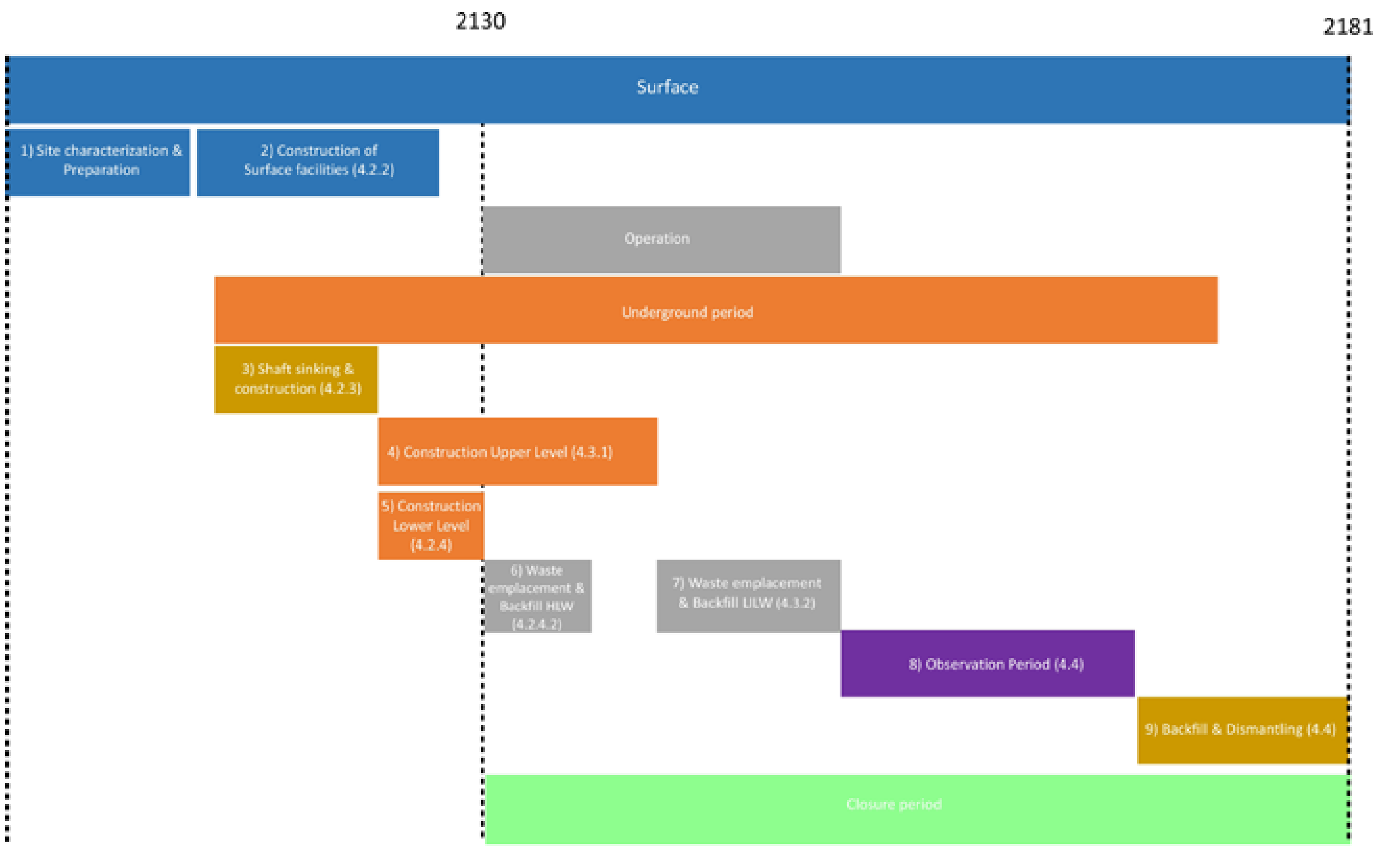


Figure 4.8) Expected construction and emplacement sequence of the GDF and the time needed for construction, emplacement, observation and closure (Herold and Leonhard, 2023b), showing the numbered activities referred to in this report. Note that the construction of the upper level can be divided into 2 phases: 1 phase is the construction of the infrastructure area and the some of the tunnels that are needed for the construction of the lower level. In the second phase, after waste emplacement and backfill HLW (7), the waste area will be constructed. Between the 2 phases, there is no construction in the upper level.

Continuous miners and scalers are used to construct all the openings in the GDF. Scaling machines are used to smooth excavated surfaces and prevent subsequent rock falls. There are advantages in using these machines compared to using explosives to excavate salt, as was proposed in previous disposal concepts (e.g., Van Hattum en Blankevoort, 1986). They minimise the damage to the rock salt, and they are more precise. A continuous miner can excavate spaces with a small cross section (Fig. 4.4b; e.g., disposal tunnels) at a rate of 2 m/hour and spaces with a large cross section (e.g., service and transport tunnel) at 1 m/hour.

When the infrastructure areas are completed, (6) the entire lower level is constructed followed by (7) the emplacement of HLW. Assuming 1 shift operation, 2 hoists per shift and 1 HLW package per hoist, about a year would be needed to emplace all the waste in the lower level. The emplacement rate of the HLW is limited by the rate at which waste containers can be hoisted down the shaft, which thus becomes the principal factor controlling the overall emplacement period. About a further 2 years is estimated to be needed for backfilling the disposal tunnels with dry granular salt using a slinger machine, which is also used to backfill other open spaces within the repository. This machine sprays the backfill into a disposal tunnel and, when needed, a vibration machine can be used to compact the backfill. Currently, large scale (practical) tests are being performed with different types of backfill and backfill techniques, as part of the SAVER project.

Since only about three years is needed for disposal of HLW and backfilling of the disposal tunnels in the lower level, it is more efficient to start emplacement after the construction period has finished, allowing separation of disruptive excavation activities from active handling of radioactive materials. Separating the construction and emplacement phase into two distinct phases is not, however, a strict requirement, as is demonstrated by the WIPP facility in which waste emplacement takes place during the construction activities expanding the facility. As the disposal tunnels will be open for a limited time, maintenance is expected to be minimal.

Following the emplacement of all the HLW, (5) the upper repository is constructed followed by the emplacement of the (8) LILW and (TE)NORM waste. Based on 220 working days per year, about 17 years will be needed to emplace all the LILW and (TE)NORM waste in the upper level. This assumes three hoisting operations per day. Allowing for delays during operation, 17 years is assumed for the disposal of the LILW waste inventory. All the estimated durations rely on mixed loading of the hoisting cage, with multiple types of waste being hoisted to optimise the overall emplacement time. An additional 2 years is needed to backfill all the disposal rooms, followed by an observational period of 10 years. This observation period is included to simplify retrieval of waste packages before closure, should this be decided (Verhoef et al., 2017). After the observation period (9), final closure of the repository includes backfilling of the remaining open spaces, followed by sealing of the repository access and dismantling the remaining surface facilities (10), which will take about 8 years.

4.6 The expected evolution of the repository

The multibarrier system of host rock salt and the EBS will provide the levels of containment and isolation required of the GDF (Chapter 2 and table 2.1). Here, we briefly describe how the containment and isolation objectives will be accomplished, if the system as described in the previous sections evolves as expected. This is referred to as the normal evolution scenario and is described in more detail in Chapter 7. The normal evolution scenario is built up by assessing all the features (components and properties) of the reference system, the events that might affect these and the processes that drive the overall evolution of the system (Lommerzheim, 2023).

The key feature of the COPERA (2020 – 2025) salt disposal concept is that the undisturbed natural barrier (rock salt) is impermeable and can, by itself, provide permanent containment form (See Fig. 3.5 in Chapter 3). The only way for radionuclides to be released from the repository is thus via previously excavated open spaces in the host rock, for example, shafts, transport and ventilation tunnels. Here, multiple engineered barriers provide the necessary containment. Directly after closure of the GDF, containment is provided by concrete seals, the HLW package and the waste form (See Fig. 3.5 in Chapter 3). These engineered barriers will inevitably degrade over time by thermal, mechanical and chemical processes. The HLW package, for example, is assumed to provide complete containment only for its design lifetime of 1,000 years (Wunderlich et al., 2023) - which is the time needed for the moisturised granular salt backfill to attain a low permeability. In practice, all HLW packages are expected to provide complete containment for hundreds of thousands of years after GDF closure. The concrete seals have a design lifetime of 50,000 years (Lommerzheim, 2023) but they will also probably retain their integrity for significantly longer.

Within the GDF, long-term containment is provided by granular salt backfills. Initially, the granular salt will have a relatively high porosity and permeability and will provide only limited containment. With time, however, the backfill will be compacted by convergence of the host rock. This compaction results in a decrease of its porosity and permeability until the backfill eventually has comparable properties to the host rock, i.e., is impermeable. Together with the other engineered barriers, it provides the necessary containment (See Fig. 3.5 in Chapter 3).

Because we have good understanding of the properties and the behaviour of the natural and engineered barriers, it is possible to describe how the COPERA (2020 – 2025) salt repository will evolve with time. Directly after the closure of the GDF, ground water slowly starts to ingress into the sides and the top of the shafts at locations that will depend on the hydrogeological properties of the locally overlying formations above the salt dome. At the same time, the granular salt used to backfill the ventilation, transport, and service tunnels within the repository (Fig. 4.4a and Fig. 4.6a) and parts of the shaft starts to compact. This compaction is a result of gravity and the natural convergence of the host rock; it could result in some fluid migration, as brine added to increase compaction rate of the backfill is pressed out of the granular salt backfill.

During this period, any radionuclides in the LILW and (TE)NORM that have been dissolved and mobilised can be mobilised by the bulk motion of the fluid. Within about 1,000 years, all the moisturised granular salt backfills (Fig. 4.4a, Fig. 4.6a), including the granular salt in the shafts, will attain a low permeability (between 10^{-17}- 10^{-18} m^2). Consequently, fluids an neither leave nor enter the repository after this time. Within tens of thousands of years after closure, connections between the pores in the backfill seal off, and diffusive transport of radionuclides also stops. Any radionuclides that have been mobilised from the wastes are, from then on, essentially fixed in the granular salt backfill. A similar evolution is expected for the dry granular salt used to backfill the HLW disposal tunnels (Fig. 4.4a and Fig. 4.6a and Fig. 6.1b). However, it will take significantly more time for the dry granular salt—a few thousand years compared to one thousand years—to achieve a low permeability (Spiers et al., 1988).

It is not expected that the HLW package will fail under the conditions of the normal evolution scenario. Contact with brine would be needed for these packages to fail by corrosion and subsequent mechanical failure. While there will be some moisture present in the granular salt backfill, it is very limited, especially as no additional moisture is added to the granular salt used to backfill the lower-level disposal tunnels. The amount of moisture available for corrosion in the first 1,000 years is expected to be too small to result in failure of the HLW package. The Hauptsalz in the Asse mine, for example, has a brine content of less than 0.02% by volume (Hansen et al., 2016) whereas considerably more brine would be required to result in the failure of the HLW package (Wunderlich et al., 2023). Furthermore, no brine will be available after compaction, healing and sealing of the backfill which is expected to have become impermeable. However, compared to the moisturised, dry granular salt backfill is likely a few thousand years to reach a low permeability. Nevertheless, as detailed later, to increase confidence in the robustness of the safety case, the COPERA (2020 -2025) safety assessment makes pessimistic assumptions about how long the HLW waste package will remain intact.

Some of the waste in the GDF comprises radionuclides with such long decay times that they will always be present in the disposal system. Even over times comparable with the expected lifetime of Earth, they will not decay significantly. Examples are long-lived fission products such as I-129 or Se-79. The main category of very long-lived material in the wastes, however, is the depleted uranium contained in (TE)NORM. As its name implies, this uranium is a naturally occurring material that is being returned to a deep natural environment; it contains no additional radioisotopes to those of the original ore. Over very long geological time scales (millions of years), these radionuclides might leave the repository in a much diluted form due to slow geological processes that disturb the natural rock salt barrier, such as subrosion and diapirism (See section 5.2.4.1 and 5.2.4.2).

The GDF thus provides total containment at all future times for most radionuclides, except perhaps the very longest-lived radionuclides, which might be mobilised after many millions of years. Chapter 7 looks in more detail at how the disposal system is expected to evolve with time and how this is modelled in the COPERA (2020 -2025) safety assessment. As the repository is located 750 m below the surface in a stable geological environment, it can provide total isolation of the wastes from the normal activities of people for as long as it remains undisturbed.

4.7 Total cost of the repository in rock salt

4.7.1 Approach

In the previous sections, we have addressed the engineering feasibility of constructing a repository in rock salt, discussed how long construction takes and summarised how the facility provides long-term safety. The repository project must also be economically feasible, so an estimate of the implementation costs is required. In addition, COVRA charges a fee to organisations from which they accept wastes for storage and disposal and the accumulated

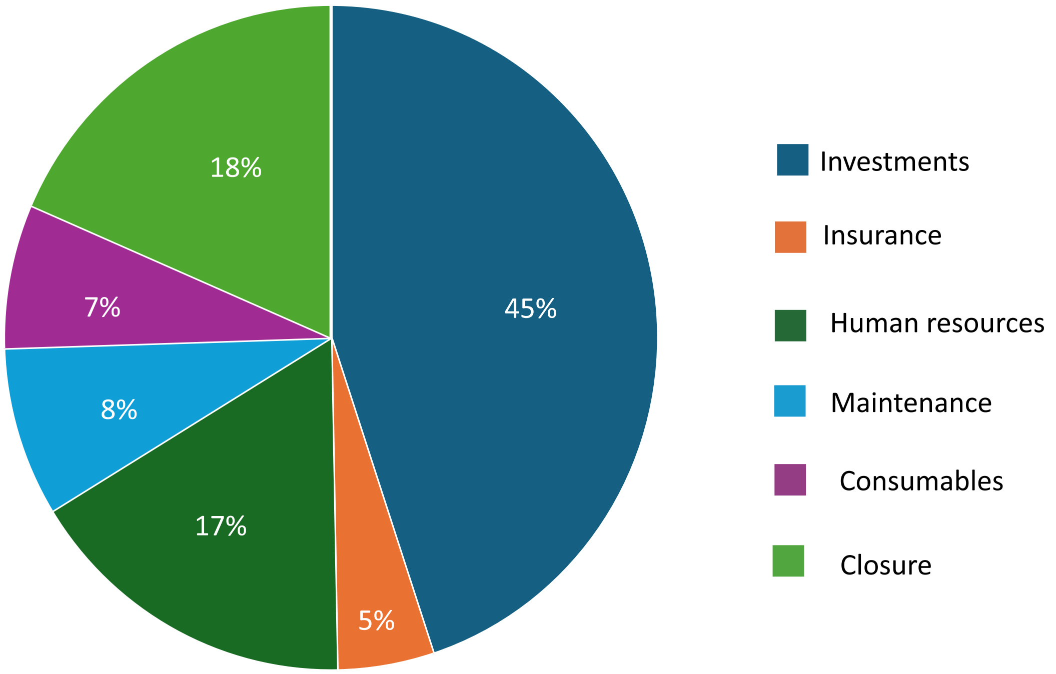


Figure 4.9) The percentage of the total cost of investment, insurance, human resources, maintenance, consumables and closure cost.

fees must cover the estimated disposal costs. Accordingly, COVRA produces regular cost estimates, allowing it to set appropriate fees. As part of the COPERA (2020 - 2025) research programme, Herold and Leonhard (2023a) estimated the total cost of disposal in rock salt, based on the COPERA (2020 - 2025) disposal concept, using the SSK (Standaard Systematiek Kostenramingen), a standardised method used in the Netherlands to estimate, record, and share cost estimates for large construction works. Using a standardised method, makes the cost estimates transparent and more directly comparable with, for example, estimated costs for disposal in poorly indurated clay (Neeft et al., 2024b), or with future updated salt disposal concepts.

To determine the costs, a stepwise method is used. Here we discuss this method only briefly. A more comprehensive description is given in (Herold and Leonhard, 2023b). In the first step, the unit costs of the different systems, structures and components that are needed were estimated. These unit prices were based on industry standards (e.g., mining equipment), external sources (e.g., reports), input from COVRA (e.g., architect and engineering fees) and practical experience (e.g., backfill materials). In the second step, the disposal schedule was established, based on practical experience and relevant external projects. In the final step, the costs were estimated for activities by assessing their duration or required quantities of materials, for capital costs. These results were subsequently used as input for the SSK method.

4.7.2 Costs

In total, the COPERA (2020 - 2025) salt repository the estimated overnight cost in 2022 Euros, excluding VAT, is around 3.5 billion euros (Table 4.2) using the same cost estimate method as in OPERA. This cost comprises 6% for preconstruction activities, 34% for repository construction, 9% for construction and operation of the conditioning facility, 7.6% for construction and operation of the (TE)NORM conditioning facility, 1.3% for the disposal of HLW, 14% for the disposal of LILW and (TE)NORM, 4% for underground observation and 22% for closure of the repository. Construction of the repository is one of the largest cost items, with above and below ground construction each comprising roughly 50% of the total construction costs. Human resources, followed by investment, consumables and closure are also major costs (Fig. 4.9). Using the SSK method, the expected cost becomes 3.5 billion euros. This is assuming an uncertainty range of -50% to +50%. The expected cost is like the estimate calculated using the deterministic method. This is likely a result of the symmetric uncertainty distribution used and that no one single item represents a significant large part of the cost. There is a probability of 5% that the repostiry will cost less than 1.7 billion euros (excluding VAT) and a 5% chance that it will cost more than 5.3 billion euros (excluding VAT).

4.7.3 Potential optimisations

4.7.3.1 HLW disposal

There are many ways that the design and the operational aspects of the repository design can be optimised and, in so doing, some costs can be reduced. In the current disposal concept, the centre-to centre distance between HLW packages is 10 m. This could be reduced to allow for more HLW packages within a disposal tunnel. However, the construction of the HLW disposal tunnels makes up only a small part of the total cost of 3.5 billion Euro (0.2% for construction: Table 4.1 and Herold and Leonhard, 2023a; Oudenaren and Browning, 2023). Reducing its cost will have a marginal effect on the total costs of a repository. Furthermore, the emplacement costs will not reduce by placing more HLW packages in a disposal tunnel; the number of waste packages and the time needed to dispose the wastes remains the same. This is because the time required to dispose all the HLW is determined by the number of HLW packages that can be hoisted down a single shift. If this number cannot be increased, then the total HLW disposal time could be reduced by, for example, increasing the number of canisters per waste package.

4.7.3.2 Multiple shifts

In the COPERA (2020 – 2025) disposal concept, the transport of the waste through the shaft is a bottleneck within the transport and emplacement process, especially for the large number of LILW and (TE)NORM waste packages that need to be disposed. Changing to a two-shift regime reduces the time needed to dispose the waste. This in turn reduces the operational period, which corresponds to a cost reduction of around 180 million Euros (about 15%) and would shorten the net disposal time by 8.5 years.

4.7.3.3 (TE)NORM

In terms of volume, one of the largest amounts of waste is (TE) NORM and, more specifically, Depleted Uranium (DU). To reduce the cost, some of the DU could be used in a different way. One example is the use of DU as an aggregate in the concrete buffer, if the HLW were to be disposed of in an waste package similar to the clay supercontainer (Verhoef et al., 2017). As explained in Chapter 6, the OPERA supercontainer contains only 1 HLW canister per supercontainer. The HLW packages designed by Wunderlich et al. (2023) for a GDF in salt can contain six HLW canisters, which significantly reduces both the construction and disposal time needed. It should be noted that in the updated COPERA (2020 - 2025) concept for a GDF in clay, seven HLW canisters are placed in a single supercontainer (Neeft et al., 2024b). The efficiency of a GDF in salt might be increased if a similar supercontainer were considered, with modifications such as including DU in the outer concrete of the container to enhance operational radiation shielding and reduce its size.

Another option is to use the DU as an aggregate in concrete backfill material (Browning and Grupa, 2023). For example, the total volumes of the lower level service (9,328 m^3), ventilation (27,328 m^3) and transport tunnel (26,288 m^3) are sufficient to dispose of all the conditioned 58,070 m^3 of DU (Burggraaff et al., 2022). A further option would be to use the conditioned DU as a backfill for the disposal rooms. Irrespective of where it is used, the time needed for disposal of all the waste would be reduced by 10 years, and 21 disposal rooms would not need to be constructed, thus reducing to less than half the number of upper-level disposal rooms.

However, the impact on sealing effectiveness of using concrete with DU as an aggregate is not yet well studied. In addition, backfill containing DU must be treated as a radioactive material which complicates operations and, if needed, retrieval of wastes (Browning and Grupa, 2023). More research is needed to better understand the possible implications of using conditioned DU as a backfill in the operational, closure and post closure phases.

	Cost (k€) [1]	Cost (k€) [2]	Cost (k€) [3]	Total (k€)	%
Preconstructional activities	229,679	612		230,292	6.5
Land purchase	62,766	72		62,839	27.3
Site infrastructure work	48,127	540		48,668	21.1
Site facility construction	1,920			1,920	0.8
Security installation	1,400			1,400	1
Utility consumption	55,462			55,463	24.1
Human resources	60,003			60,003	26.1
Repository construction	1,228,391			1,228,391	34.6
Construction and outfitting shafts	408,200				33.2
Construction and outfitting upper level main areas	6,969				0.6
Construction and outfitting HLW main area & bunker	5,947				0.5
Construction and outfitting HLW disposal area	2,175				0.2
Construction and outfitting LILW and (TE)NORM disposal galleries	5,049				0.4
Site facility construction	289,258				23.5
Maintenance surface	124,731				10.2
Insurance	34,372				2.8
Utility consumption	141,738				11.5
Human resources	209,953				17.1
CF construction and operation		360,991		360,991	10.1
Investment - Site facility construction		20,286			5.6
Investment - CF internals/construction and installation		111,800			31
Maintenance during HLW disposal campaign		41,600			11.5
Insurance during HLW disposal campaign		6,670			1.8
Material for DWP and consumables		180,635			50
(TE)NORM construction and surface operation			269,977	269,977	7.6
Investment - Site facility construction			10,223		4
Investment - CF internals/construction and installation			259,754		96
HLW	47,370			47,340	1.3
Disposal	1,850				3.9
Backfilling and sealing	18,036				38.1
Maintenance surface	5,423				11.5
Insurance	4,363				9.2
Utility consumption	6,163				13.0
Human resources	11,504				24.3

LILW and (TE)NORM	496,562			496,562	14.0
Disposal	6,020				1.2
Backfilling and sealing	71,838				14.5
Maintenance surface	104,763				21.1
Insurance	92,192				18.6
Utility consumption	26,188				5.3
Human resources	195,561				39.4
Underground observation	134,585			134,585	3.8
Maintenance surface	54,231				40.3
Insurance	16,640				12.4
Utility consumption	20,425				15.2
Human resources	43,289				32.2
Repository closure	780,899	3,109		784,008	22.1
Backfilling and sealing galleries and ramp	49,365				6.3
Backfilling and sealing shafts	450,000				57.4
Dismantling and decommissioning nuclear facilities	100,000	3,109			12.8
Site dismantling and clearance	50,000				6.4
Insurance	13,912				1.8
Utility consumption	49,300				6.3
Human resources	68,312				8.7

%	*Percentage of the total cost.*
%	*Percentage cost within phase.*
%	
%	

Table 4.2) Costs of the repository for the different phases based on the OPERA method. The first column gives the cost item, the second column the cost as estimated by Herold and Leonhard (2023b), the third column the cost of the Conditioning Facility (CF) and material needed for the disposal of HLW (Herold and Leonhard, 2023b; Wunderlich et al., 2023). The fourth column is a cost estimate for the (TE)NORM conditioning facility including the costs for concrete, human resources and the Konrad type II container (Oudenaren and Browning, 2023). The fifth column is the total cost per item and the last column gives the percentage of the total costs (dark) and the percentage of the cost per phase (light).

Shining a flashlight on a wall of rock salt inside the Waste Isolation Plant in the USA. The WIPP is currently the only operational deep geological disposal facility in the world. Source: Kristopher Kuhlman, Sandia.

5. The Natural Barrier

Summary:

- Rock salt is impermeable, heals, is dry, has a high thermal conductivity, is plastic and self-healing, and there are decades of practical experience with construction in rock salt and with the handling of radioactive wastes underground in GDFs in salt formations.
- Both the Permian Zechstein and Triassic Röt formation are potentially suitable for disposal in the Netherlands.
- Diapirism rates in the Netherlands vary between 0.001 and 0.1 mm/year.
- Subrosion rates in the Netherlands vary between 0.01 and 0.1 mm/year.
- Ice ages and glacial channels are not expected to disturb the repository.

The natural barrier, which is a part of the multibarrier system, comprises the host rock and the surrounding geological formations. In the safety concept, it has the function to isolate and contain the waste. This chapter focuses on understanding in more depth the natural barrier for a GDF in a salt formation. We begin by examining the properties of rock salt and why it is considered a suitable option for geological disposal. Then, we discuss the two main salt formations of interest, the wide variety of salt structures in the Dutch underground (Fig. 5.1) and the geological formations above and below them. Next, we address the uncertainties associated with the properties and evolution of these formations. Finally, we explore how changes in climate can impact the host rock, as well as the overlying and underlying formations.

5.1 Properties of salt

Evaporite is a general term used in geology to specify sediments that result from the evaporation of water with a high content of dissolved solids, principally salt, typically in lake basins and shallow marine environments. While evaporation is the dominant process by which evaporites form, freezing of saline water can also result in the deposition of evaporites (For references see Babel and Schreiber, 2014). Evaporites encompass a wide range of minerals including chlorides such as rock salt (halite: NaCl) and sylvite (KCl), sulphates, such as anhydrite ($CaSO_4$), gypsum ($CaSO_4 \cdot 2H_2O$) and polyhalite ($K_2Ca_2Mg(SO_4)4 \cdot 2H_2O$), and carbonates, such as calcite ($CaCO_3$) and magnesite ($MgCO_3$). In this report, the terms 'salt' or 'rock salt' refer to halite, the principal rock type of interest for geological disposal.

Evaporites, and rock salt specifically, are currently being considered as potential GDF host rocks in the United Kingdom and the Netherlands and are already in use for geological disposal in Germany and the USA. In the following sections, we discuss properties of rock salt that makes it suitable for a GDF (Partly based on Hansen et al., 2016) and, when possible, quantify these.

5.1.1 Hydrological properties of salt

The main safety functions of the host rock are to contain the radionuclides in the waste and, together with the other components of the natural barrier, ensure that any releases to the biosphere that do occur are in insignificant concentrations. The transport of radio-

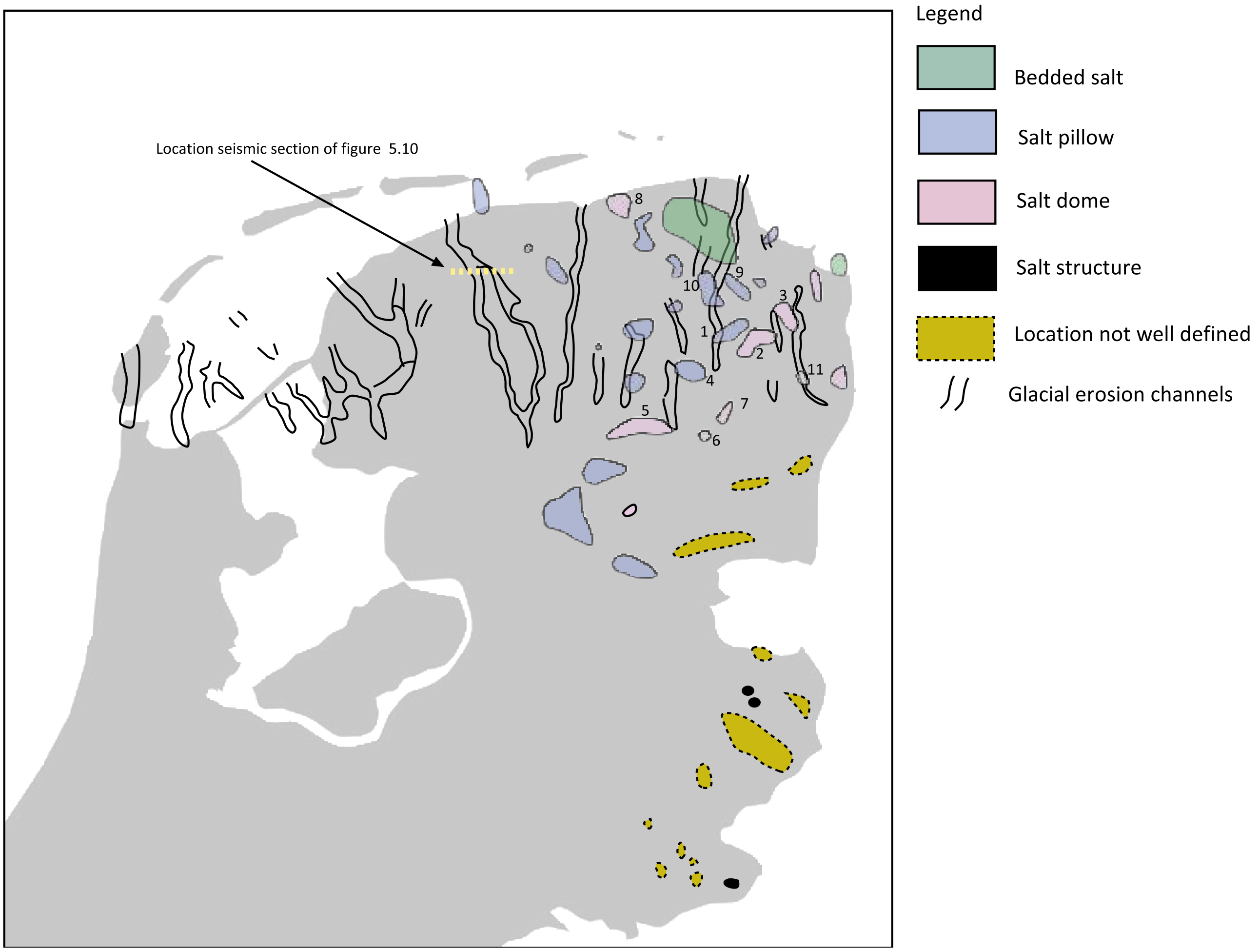


Figure 5.1) Onshore salt structures in the Netherlands with potential to host a GDF, based on the criteria set in the OPLA research programme. These criteria included the thickness of the salt above the repository, the dimension of the repository and the thickness of salt below and next to the repository (Wildenborg et al., 1993). While there are over 40 salt structures, not all will be suitable for disposal. Conversely, there might be other salt structures not indicated here that could be suitable for disposal. This figure only intends to demonstrate the variety of salt structures in the Netherlands. Figure based on Wildenborg et al. (1993). 1) Veendam, 2) Zuidwending, 3) Winschoten, 4) Anloo, 5) Hooghalen, 6) Schoonloo,7) Gasselte-Drouwen, 8) Pieterburen, 9) Slochteren/Noordenbroek, 10) Hoogezand, 11) Onstwedde. The location of the seismic section of figure 5.10 is shown as a dotted yellow line. Note that 'location not well defined' means that there is a salt structure, but its extent cannot be defined well due to the lack of good seismic data. Furthermore, no offshore salt structures are shown.

nuclides in solution in groundwaters and porewaters in rock can occur predominantly via two processes: advection and diffusion. In-situ measurement of the Salado formation (bedded salt, the host rock for the WIPP GDF in the USA) shows that its average permeability is $6.65 \cdot 10^{-22}$ m^2 (Gorham et al., 1992). This is based on a compilation of multiple experiments (e.g., Permeability Testing Program). A similar value, was found in Germany for salt from multiple locations that include the formations that host the GDF at Asse and the previously proposed GDF at Gorleben (Minkley et al., 2020) and from in-situ measurements using brine in France (Cosenza et al., 1999). These measured permeabilities are so low that brine would find it impossible to flow through it and therefore rock salt is generally considered to be impermeable. This is demonstrated by the fact that salt formations exist millions of years after their deposition (e.g., van Balen, 2010). If salt had even a small permeability, water would have flowed through, and, owing to its high solubility, would have dissolved it, such that 250 million years old salt formations would not exist today. Furthermore, in the Netherlands, rock salt of the bedded Zechstein Group forms a seal for many hydrocarbon fields, onshore and offshore (Breunese et al., 2010.). It has been a seal for gas accumulations from underlying or adjacent sediment formations for over 150 million years, with the last gas generation taking place in the Late Cretaceous (de Jager and Visser, 2017; Lee et al., 1985). If the rock salt had even a small permeability or were fractured or degraded, these hydrocarbons would not have been preserved for over 65 million years.

When a repository is constructed, salt around the openings will be disturbed (up to six metres away from the open space; Hansen, 2003), increasing the permeability in this region to between $1 \cdot 10^{-15}$ m^2 and $1 \cdot 10^{-18}$ m^2 (Peach and Spiers, 1996). A similar range in permeabilities, using various test setups for measurements, was found by Bechthold et al. (2004). This disturbed salt region around the openings is generally referred to as the excavation disturbed zone (EDZ). Since it has a higher permeability, the EDZ might provide a pathway for fluid movement. However, the EDZ is still expected to have a permeability several orders of magnitude lower than granular salt backfill in the openings at any point during the evolution of the repository (Oosterhout et al., 2022): the backfill permeability directly after closure is about $1 \cdot 10^{-12}$ m^2, assuming a porosity of 40% and the permeability – porosity relation as described by Oosterhout et al. (2022).

The other potential transport process is diffusion. Measured in-situ, diffusion in intact salt varies significantly between $6.40 \cdot 10^{-7}$ m^2/s and $4.70 \cdot 10^{-11}$ m^2/s (Gorham et al., 1992). Therefore, as with previous GDF safety assessments in Germany (which used an effective diffusivity of $1 \cdot 10^{-11}$ m^2/s, Bertrams et al., 2020b), we assume that diffusive transport through any significant distance in the host rock on timescales of relevance in the safety assessment is not possible: assuming the effective diffusivity of $1 \cdot 10^{-11}$ m^2/s of Bertrams et al. (2020b), any radionuclides mobilised from the waste would diffuse less than 20 m into the host rock in 1 million years.

5.1.2 Salt is dry

While traces of brine exist between and within salt crystals, and minerals such as carnallite ($KMgCl_3 \cdot 6H_2O$) and bischofite ($MgCl_2 \cdot 6H_2O$) contain high proportions of mineralogically bound water, rock salt is generally characterised as being very dry. This is especially the case in salt domes, since the intrinsic brine from original deposition has been squeezed out during the deformation involved in the development of the dome structure (Hansen et al., 2016). For example, the Hauptsalz in the Asse mine has a brine content of less than 0.02% by volume (Hansen et al., 2016). Evidence of the dryness is provided by some unique archaeological finds in old salt mines, for example, a 2,000-year-old child's shoe (Reschreiter and Kowarik, 2019) and finger-loop braiding and intricately woven cords and bands (Grömer et al., 2015). Both would not have been preserved if salt contained a significant amount of brine. This property of salt is used today to provide a dry storage environment. For example, in the UK, mediaeval manuscripts, hand drawn artwork, fabrics, dresses, wallpaper and other materials are stored in the worked-out parts of Winsford Rock Salt Mine in Cheshire (www.deepstore.com).

5.1.3 Salt creeps

Rock salt is a plastic medium that can flow slowly under stress and encapsulate buried material. This natural "flow" in response to load pressures, density differences between geological formations and stress variations, for example, around openings, is also referred to as creep. With time, all open spaces within the repository will slowly close naturally, due to salt creep. The salt will slowly surround other materials, including the waste, and form a tight geological barrier around them (Hansen et al., 2016). This behaviour is observed in salt mines, laboratories (e.g., Spiers et al., 1986; Urai et al., 2008), field experiments (e.g., Bérest et al., 2014; Heijdra and Prij, 1992) and natural examples (Waltham, 2008). The rate at which salt creeps varies significantly, as it depends on a wide variety of variables, including temperature, grainsize and stress differences (Oosterhout et al., 2022). In situ measurements from the Morsleben repository in Germany show that the convergence rate of mined openings is about 1 to 2 mm / year (Buchholz et al., 2020). A similar creep rate was found in experiments in the Asse mine (Heijdra and Prij, 1992). A creep rate one to two orders of magnitude higher was observed in the WIPP (Munson, 1997) which is located in bedded salt that tends to creep faster than domal salt (Hansen et al., 2016). The microscale mechanisms that are responsible for salt creep are discussed in more detail in section 6.1.1.

5.1.4 Salt heals

Salt creep can close fractures and openings and heal previously damaged areas to restore the low permeability. Three stages can be distinguished in the host rock, namely two mechanical closure stages due to convergence via creep, followed by static healing when convergence has essentially ended. During mechanical closure, a number of processes can be active but, at the expected depth and temperature for a repository in rock salt, the two dominant processes are likely to be dislocation creep and pressure solution creep (See section 6.1.1 for more details. Oosterhout et al., 2022). The timescale for mechanical closure is a few hundreds to a thousand years (Houben et al., 2013; Koelemeijer et al., 2012). Static healing (disconnection of pores), on the other hand, occurs by diffusive crack healing or recrystallization (See section 6.1.1 for more details; Houben et al., 2013; Koelemeijer et al., 2012). The timescale for disconnecting pores is unclear, but has been estimated to range from several years (Houben et al., 2013) to a few thousand years (Koelemeijer et al., 2012). A further, unpublished, study by Grupa and Houkema (2000), who implemented the percolation model of Peach and Spiers (1996), suggests that this process takes about 400 – 26,000 years, when starting with a porosity of 3%. When starting at a lower porosity (0.3%), it would take about 5 – 230 years (Grupa and Houkema, 2000). Both estimates are for fully brine-saturated conditions. Assuming unsaturated but humid conditions, it was estimated to take at least 9,000 years. These calculations were made for salt with a grain size of 10 mm.

5.1.5 Practical experience

There are currently many operational salt mines around the world. The Wieliczka salt mine in Poland was active from the 18th century until production stopped in 1996 (Wiewiórka et al., 2009). An older and still active mine is at Khewra, in Pakistan. Here, salt was being extracted from around 320 BC, although salt trading started much later, in the Mughal era (1526–1761). More recently, the Boulby potash mine in the UK opened in 1968 (extracting sylvite, polyhalite and rock salt) and is currently still active. Furthermore, the only purpose-built, operational GDF to date is in bedded salt: the WIPP facility in New Mexico, USA. There is also a long history of salt mining in Germany, with disused mines currently being used for both commercial hazardous waste disposal (e.g., Herfa Neurode mine) and radioactive waste disposal (the Morsleben mine). The German GDF programme carried out advanced R&D into radioactive waste disposal in salt domes from the 1970s to the 1990s. This included engineering studies on emplacement techniques for heat emitting wastes, on GDF sealing at the Asse mine, and on construction of a deep underground research facility at Gorleben. In the Netherlands, salt caverns are used for the storage of gas (Zuidwending), gasoil (De Marssteden) and nitrogen (Heiligerlee). There are also plans to store hydrogen produced by energy generated from wind turbines in caverns in salt domes (van Gessel et al., 2021). In the UK, hazardous waste is currently being stored in the Winsford Rock Salt Mine by Veolia (Veolia, 2015) while in Australia, TELLUS is currently considering building a multinational GDF in bedded salt at a depth of 850 meters that would accept both chemical and radioactive waste (Tellus, 2024).

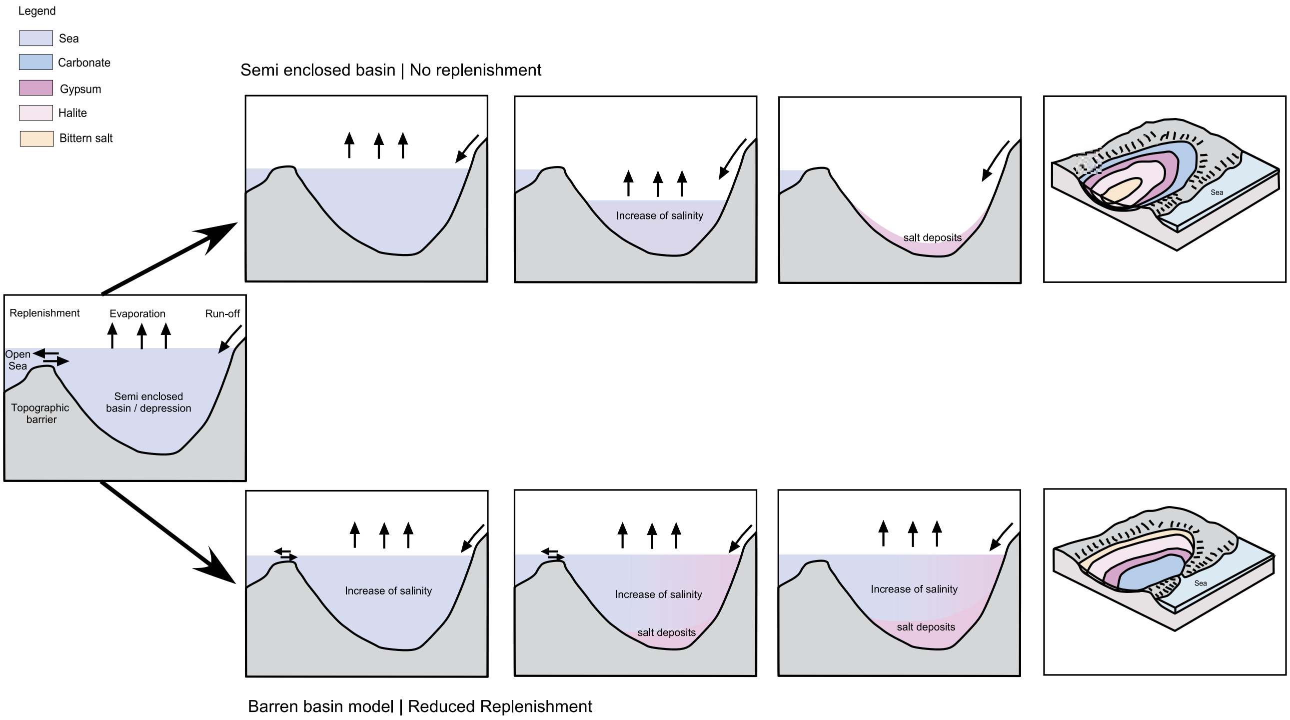


Figure 5.2) The two main mechanisms by which evaporites can be deposited. The evolution of the closed basin model is shown in the upper figures, while the evolution of the barren basin model is shown in the lower figures. The last figure on each evolution line shows the expected types of salt deposits and their distribution. While salt is mainly deposited within the basin, some salt will also be deposited along the basin margins in, for example, sabkhas.

Thus, there is a wealth of underground experience and data in the Netherlands and other countries on which to base a GDF programme in rock salt formations (Jackson, 1995; Minkley, 2009). Some of this information has been collected as part of the current COPERA (2020 – 2025) research programme (Hunfeld et al., 2023).

5.1.6 Thermal conductivity

Rock salt has a relatively high, temperature-dependent thermal conductivity (~3.5 W/mK) (e.g., Hunfeld et al., 2023), compared to other GDF host rocks, for example, granite (3.1 W/mK; Miranda et al., 2019) and clay (e.g., 1.45 W/mk Weetjens, 2009). Due to its high thermal conductivity, heat from the waste will be transported more easily away from the GDF out into the rock salt and the surrounding formations. Consequently, heat-generating wastes can be placed closely together, reducing the footprint of a GDF, which, in turn, reduces the potential for inadvertent human intrusion (IAEA, 2017a). In addition, the increase in temperature with depth in a salt dome is relatively low compared to the normal geothermal gradient. At 850 m depth, the expected geothermal temperature in a salt dome is about 35 °C (Bonté et al., 2012; Smit, 2022).

5.2 Origin of salt

As the host salt formation for a GDF provides the principal barrier in the multibarrier system, it is worth looking in detail how salt deposits have formed and how their mode of formation affects their properties, and the way that these might vary within a formation.

Typically, marine evaporites are deposited in semi-enclosed depressions (basins or seas) having a negative water budget and an intermittent recharge connection to the open sea, possibly over a topographic barrier (Fig. 5.2; Top). If inflow is interrupted due to a global sea level drop or tectonic uplift of a topographic barrier, the level of water in the basin will drop, increasing its salinity and eventually leading to the deposition of evaporites. On average, the evaporation of 1,000 m depth of seawater would result in a 13 m thick layer of evaporites, with halite being the dominant salt type (Geluk, 2010). Moving from the margin towards the centre of the basin, highly insoluble minerals are deposited first, followed by more soluble minerals. Consequently, gypsum, anhydrite, and halite are deposited first, around the margins of the basin, followed by bittern salt (magnesium and potassium salts). This deposition pattern is commonly referred to as the bullseye pattern. Typically, one evaporation event results in an evaporite layer of only a few metres although it will depend on the initial depth of the basins (Geluk, 2010). Multiple episodes of recharge and evaporation can, however, progressively build up into deposits that can be many tens of metres thick.

This mechanism cannot explain the thick Permian evaporite deposits observed in the Netherlands (up to 800 m; Zechstein Group: section 5.2.2,Geluk, 2007) or the thick Messinian (Late Miocene) evaporites in the Mediterranean Sea (up to 2 km, Rouchy and Caruso, 2006). To account for these kilometre-thick evaporites, it is suggested that inflow to a semi-enclosed basin is severely restricted but not completely interrupted (the so-called barren basin model, Fig. 5.2; lower figure), allowing a continuous process of evaporation and deposition. With restricted inflow and continuous evaporation, the salinity of the water increases towards the centre of the basin, eventually resulting in the deposition of evaporites. The most insoluble minerals (carbonates) are deposited near the inlet, while, further away, more soluble minerals (gypsum, anhydrite, halite and bittern salt) are deposited. This deposition pattern is known as the teardrop pattern and can result in significantly thicker salt deposits compared to the semi enclosed basin model previously discussed. This is because, in the barren basin model, salts are continuously deposited over an extended period, resulting in thick salt deposits.

Besides deposition within a marine basin, salt can also be deposited at the margins of a basin in coastal environments with an arid or semi-arid climate. In these areas, so-called sabkhas can form. Sabkhas are areas that are occasionally flooded with saline water during events such as high tides or storm surges. Sabkhas are thus generally located just a few centimetres above sea level and are therefore influenced by small changes in sea level. They are characterised by salt deposits intercalated with mud or other types of sediment. Additionally, the sequence may contain local erosional surfaces and possible indications of salt dissolution and precipitation. Sabkhas can also develop along the margins of the open ocean. Examples of regions where sabkhas can be found include the Arabian Peninsula, the Persian Gulf, the coastlines of North Africa, parts of Australia and other arid coastal areas around the world.

5.2.1 Salt deposits in the Netherlands

In the Netherlands, there are several onshore formations of different geological ages known to contain rock salt (Geluk et al., 2000). The ages of these formations are Permian (298.9 – 251.9 million years ago), Triassic (251.9 – 201.3 million years ago) and Jurassic (201.3 - 145 million years ago, Geluk, 2010). From oldest to youngest these are the Permian Rotliegend Group, the Permian Zechstein Group, the Triassic Röt formation, the Triassic Muschelkalk formation, the Triassic Keuper formation and the Jurassic Weiteveen formation. Among these formations, only two currently appear suitable for the disposal of radioactive waste: the Zechstein Group and the Röt formation. Salt deposits in the other formations and Groups (Muschelkalk, Keuper or Weiteveen formation and the Permian Rotliegend Group) appear either too thin, typically being tens of metres in thickness, are only present offshore, or are laterally discontinuous. This last point means that, with the information currently available, they do not form a sufficiently large and continuous deposit for waste disposal purposes (Geluk, 2010; Geluk, 2005). We therefore focus in the current project only on the Permian Zechstein Group and the Triassic Röt formation. As the Permian Rotliegend Group is generally below the Permian Zechstein Group and construction of a repository in the former would be more costly owing to the greater depth, we focus further here on the Permian Zechstein Group as being most likely to be suitable for disposal. Note that offshore salt deposits can still potentially be used since they could be accessed by tunnel from onshore, as is currently being considered for a GDF in the UK (Nuclear waste services, 2024), or via an artificial island if permissible.

5.2.2 Zechstein Group

Permian Zechstein salt is characterised by five depositional cycles of evaporite rocks that represent repeated phases of marine transgression followed by evaporation (semi enclosed basin model) with periods in-between during which the inflow into the basin was severely restricted but not completely interrupted to cause a marine transgression (Fig. 5.2, barren basin model). Together, these resulted in the five depositional cycles that are currently observed and the relatively thick salt deposits. During the first cycle (Z1, Werra formation, Fig. 5.3a top-left), ≈ 250 m (Geluk, 2010) of salt was deposited in a small band that now stretches from Hengelo in the east to Alkmaar in the west. During the second cycle (Z2, Stassfurt formation, Fig. 5.3a top-right), a > 600 m thick layer of relatively pure (> 95% halite) salt was deposited in the eastern and north-eastern part of the Netherlands (Geluk, 2005). During the third (Z3, Leine formation, Fig. 5.3a bottom-left) and fourth (Z4, Aller formation, Fig. 5.3a bottom-right) cycles, only 200 – 300 m (Z3 cycle) and 150 m (Z4 cycle) thick layers of evaporites were deposited. During the Z3 cycle, salt was deposited in the north-eastern and western part of the Netherlands, while during the Z4 cycle, deposition was restricted to the most northern part of the Netherlands. In the last cycle, Z5 (Ohre formation), salt deposition was limited to the offshore and north-eastern part of the Netherlands. The Ohre formation is, however, relatively thin (< 20 m, McCann, 2008). To the south, the Zechstein Group thins and becomes more continental in nature (Fig. 5.3a – Sabkha and fluvial). Based on the thickness of the Zechstein evaporites, they were probably deposited in a barren basin setting.

The depth of the Zechstein Group of evaporites and other sediments, varies across the Netherlands (Fig. 5.3b). In the north, the depth to the base of the Zechstein Group varies between 3,000 and 3,500 m. An exception is an area in the east of the Netherlands where the base of the Zechstein Group is at a depth of around 4,000 m. This area is also known as the Saxony basin and has been subject to local subsidence after deposition of the Zechstein Group (Pharaoh et al., 2010). In the centre of the Netherlands, the depth to the base of the Zechstein Group decreases to about 1,000 m and even less in the east. Going further south, the depth to the base of the Zechstein Group increases again to over 2,000 m. An exception is again the south-eastern part of the Netherlands where the base of the Zechstein Group is at a depth of less than 1,000 m. While the Zechstein Group covers most of the Netherlands, Zechstein salt was only deposited in the north and central part of the Netherlands (compare Fig. 5.3a and b). In the south, the Zechstein Group has a more continental / coastal character (sabkha, fluvial).

Over much of the Netherlands, the thickness of the Zechstein Group is generally less than 100 m, except for the Northern and Eastern part of the Netherlands (Fig. 5.3c). In the east, the Zechstein Group reaches a thickness of a few hundreds of metres. In the north, the thickness of the Zechstein Group varies significantly laterally. At some places, it is less than 100 m thick, while only a few kilometres away, it reaches a thickness of over 1,300 m. These large variation in thickness are a result of post-depositional salt movement and the formation of salt structures. The processes that have resulted in these salt structures are discussed in more detail in section 5.2.4.

5.2.3 Röt formation

The younger Triassic Röt formation can be subdivided into three members: the Upper Röt Claystone, intermediate Röt Claystone and Main Röt Evaporite. The thickest rock salt layer within this formation is part of the Main Röt Evaporite. From bottom to top, this member is characterised by a thin anhydrite layer and massive halite, with up to 5 sub–cycles (TNO-GDN, 2023i).

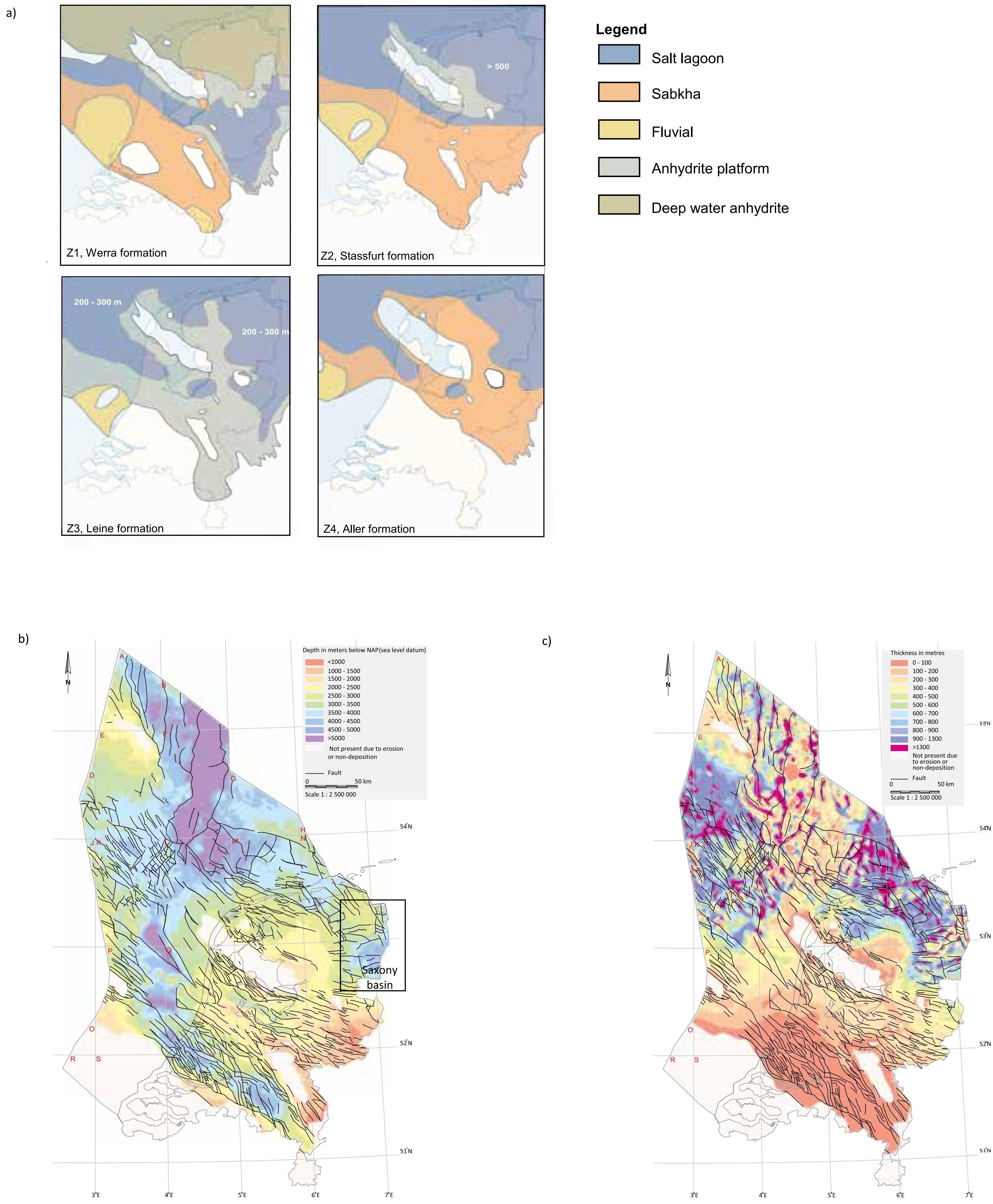


Figure 5.3a) The depositional environments of the Permian Zechstein Group. Colours indicate different depositional environments. Salts are deposited in the salt lagoons, sabkha, anhydrite platform and deep-water anhydrite. Figure is based on Geluk (2005). b) Depth in metres below Amsterdam Ordnance Datum of the base of the Zechstein Group (Figure from Duin et al., 2006). c) The thickness of the Permian Zechstein Group. This thickness includes not only the evaporites, but also clay and sandstone in the south (Figure from Duin et al., 2006).

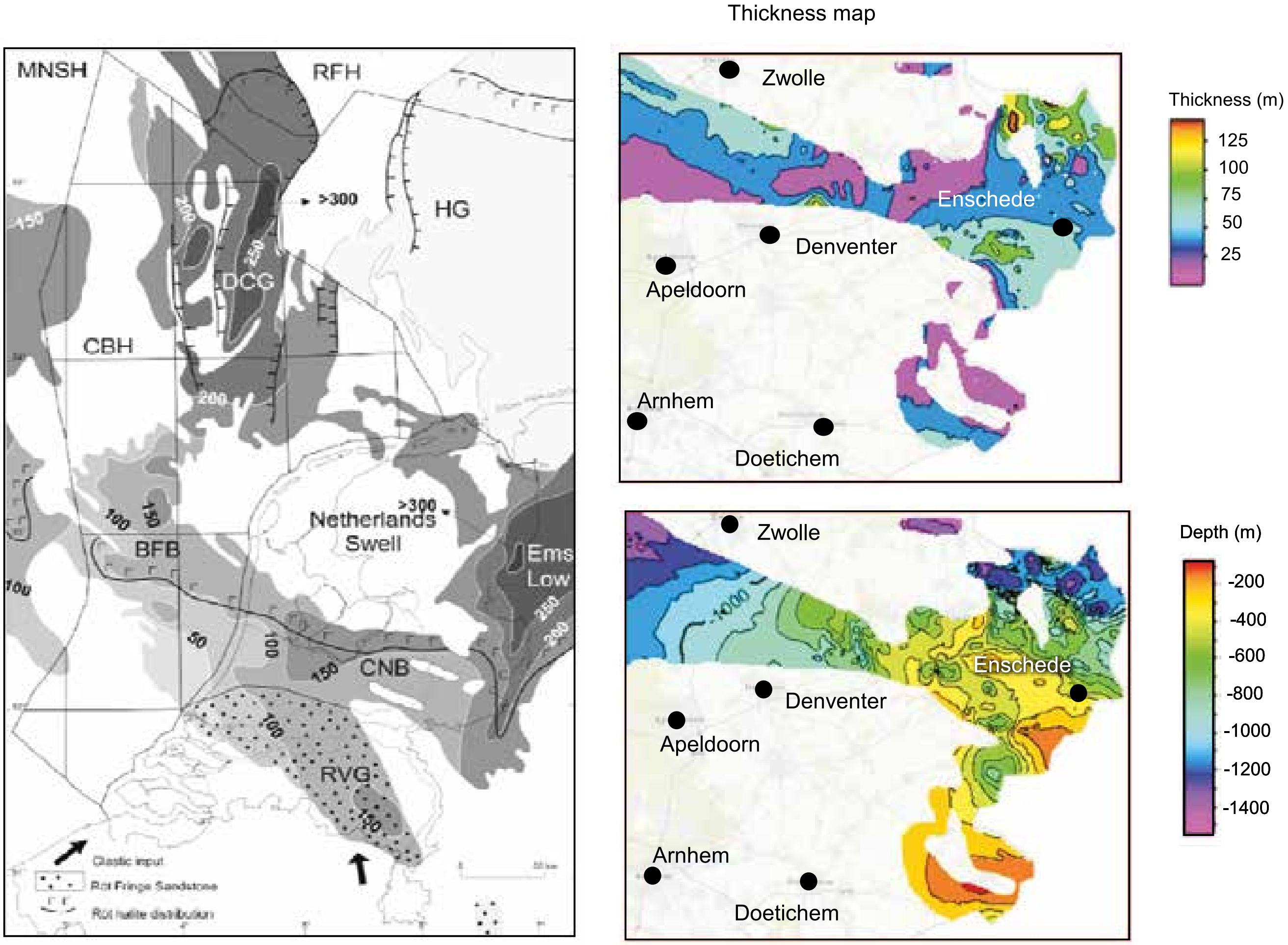


Figure 5.4) Thickness of the Röt formation (left) based on Geluk (2005) and the thickness (right-top) and depth (right-bottom) of the Main Röt Evaporite member in the eastern part of the Netherlands. Figure from Altenburg (2022).

Like the Zechstein salt, this member was deposited mainly in the northern Netherlands, where a salt lagoon or sabkha was located during the Triassic. In more southern, fluvial environments, no salt deposits are expected (Geluk, 2005).

The Röt Formation, including all three members, has a localised maximum thickness in the eastern part of the Netherlands, which decreases gradually southwards (Geluk, 2005). In the north, it locally has a thickness of more than 300 m (the Netherlands Swell), while the average is between 200 and 150 m. The depth of this formation varies widely. In the northern part of the Netherlands, the top of the formation is at a depth of more than 1500 m.
In contrast, in the central-eastern part of the Netherlands, the top of the formation is at or below 100 m depth (Altenburg, 2022).
As part of the COPERA (2020 – 2025) programme, Altenburg (2022) performed a detailed seismic study of the Röt formation, finding that the Main Röt Evaporite member is thickest (100 m) near Enschede and Tubbergen, just north of Enschede (Fig. 5.4). This is also close to the area where salt is extracted by the Nobian company. In other areas, the Main Röt Evaporite member is relatively thin (< 75 m) and has a limited lateral extent, which results from an inversion and subsequent erosion of the member in the Late Cretaceous (Altenburg, 2022). Note that, in contrast to the Zechstein Group, the Röt formation has no salt structures formed by post-depositional salt movement (e.g., salt domes) and is present mostly in semi-horizontal layers.

Altenburg (2022) also studied the lithology of the Main Röt Evaporite member in borehole TWR-480, in which ten distinct depositional facies were identified. Only four of these intervals are estimated to contain at least 95% microcrystalline halite. Some of the other intervals do contain halite, but they are generally mixed with other lithologies such as claystone, anhydrite or clay.
The salt in these intervals is thus impure. Based on the borehole data, Altenburg (2022) suggested that the Main Röt Evaporite member in this region was deposited in a salt pan, surrounded by sabkhas that had restricted marine access. This implies that the area was at the margins of the basins in which the salt was deposited, which agrees with work of Geluk (2005).

5.2.4 Shaping the salt

Initially, all evaporites are deposited in semi-horizontal geological layers, generally referred to as bedded salt. Being horizontal, their internal structure is relatively easy to understand as they have large lateral extent, and salt creep rates are generally high. However, the horizontal structures can deform into salt pillows (also known as salt swells, or salt mounds), which usually have a thickness of

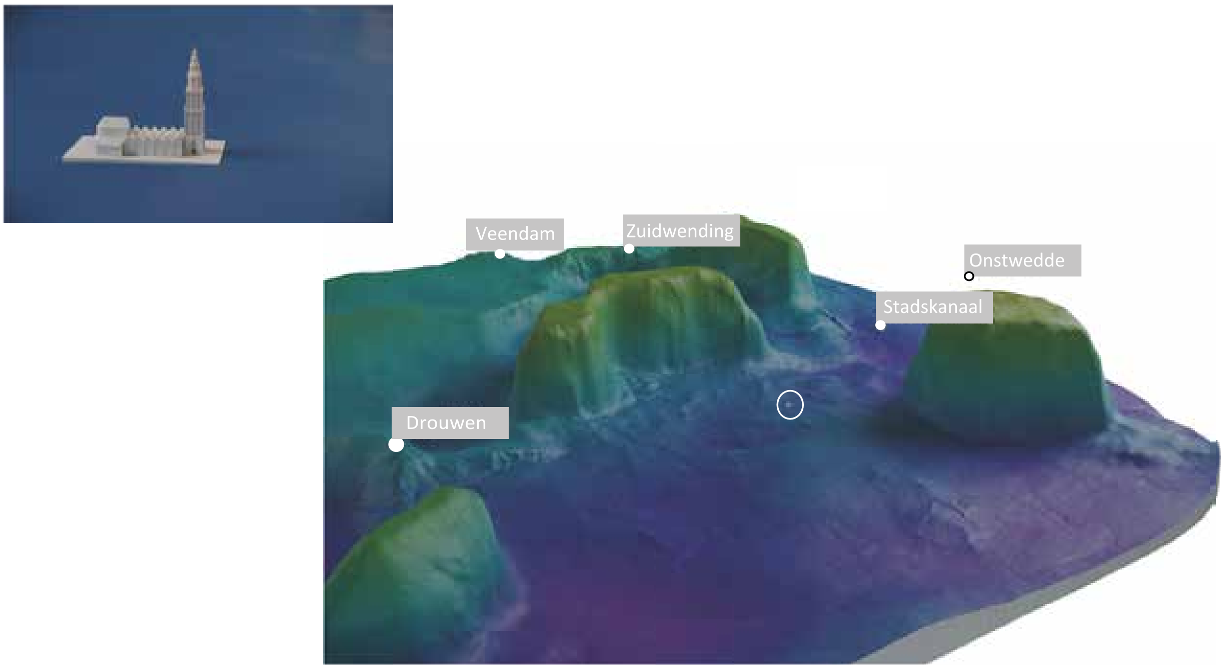


Figure 5.5) Using the of Top of the Zechstein Group, the figure shows the salt structures in the subsurface in the North of the Netherlands, with all the surrounding sedimentary formations removed, for illustrative purposes. These salt structures are 1 to 2 km in height and can be several kilometres in both length and width although this varies between them. To illustrates further the immense scale of these salt structures, the Martinikerk in Groningen (100 m in height, shown on the left figure) is placed in the right figure: inside the white circle. The names indicate the location of cities and villages on the surface. Image is from Andreas Kruisselbrink |TNO – Geologische Dienst Nederland.

several hundreds of metres more than the bedded salt from which they originated. Their lateral extent is a few hundreds of kilometres, and the inner structure of a salt pillow is complex, due to salt flow. Salt pillows are essentially an intermediate form between bedded salt and salt domes, although, unlike the latter, they have not yet pierced the overburden. The last type of salt structure is the salt dome, which typically has a large vertical and somewhat limited horizontal extent (kilometres). Overall, the centre of salt domes appear to consist primarily of halite with other minor evaporite minerals, although isolated anhydrite banks are occasionally observed in seismic surveys (e.g., Lauwerier, 2022; Van Gent et al., 2011). The diapiric rise of a salt dome through the surrounding sediments (see section 5.2.4.2) means that faults can be formed in the overburden and the caprock. They do not, however, affect the inner structure of the dome (Rijks Geologische Dienst, 1993; ten Veen et al., 2015; Wildenborg et al., 1993). The three different types of salt structures described above are present in a wide variety of geometric shapes (Jackson and Talbot, 1991).

The two dominant processes responsible for the salt structures (which can appear like 'mountains' if the surrounding sediments are stripped away graphically for illustrative purposes: see Fig. 5.5) that are currently observed in the Netherlands are subrosion and diapirism.

5.2.4.1 Subrosion

Evaporites are among the most soluble rock formations. Halite has a solubility in water of between about 35 and 360 g/L (Babel and Schreiber, 2014 and references therein). For comparison, calcium carbonate (the main constituent of limestone) has a solubility in water of 0.013 g/L (Rohleder and Kroker, 2012). When chemically unsaturated groundwater encounters soluble rock formations, it will lead to their dissolution. At depth, this process is also referred to as subrosion, and its rate and impacts are primarily determined by the flux of groundwater available, the dissolution rate and the solubility of minerals present in a salt dome (Geluk et al., 1993). Solubility and dissolution rates of salt deposits are controlled by pressure, temperature and the types of soluble minerals present (Babel and Schreiber, 2014). The groundwater flux depends on the porosity, permeability and thickness of the sediment layer through which it flows, the hydraulic gradient and the climate, which affects recharge rates and sea level. During periods of glaciations, for example, the lowering of the sea level will lead to higher groundwater velocities due to a larger hydraulic gradient. Subrosion thus depends on many variables and varies significantly over time, between salt formations and even around a single salt formation.

Subrosion of a salt dome can lead to the formation of a caprock, although this is not always present above a salt dome (Posey and Kyle, 1988). The caprock is a hard and less soluble rock type overlying the more soluble salt dome. From the bottom upwards, the caprock typically consists of anhydrite, gypsum and calcite/carbonate (Posey and Kyle, 1988). The anhydrite is the residue from the dissolution and removal of halite from the evaporite formation, as it has a significantly lower solubility in groundwater than the predominant halite content of the dome. Since dissolution occurs at the interface of the salt dome and the caprock, the anhydrite is progressively younger towards the base of the caprock. The gypsum between the anhydrite and calcite results from the hydration of the anhydrite. The calcite on top of the gypsum results from the interaction of carbon-bearing fluids with anhydrite at the interface

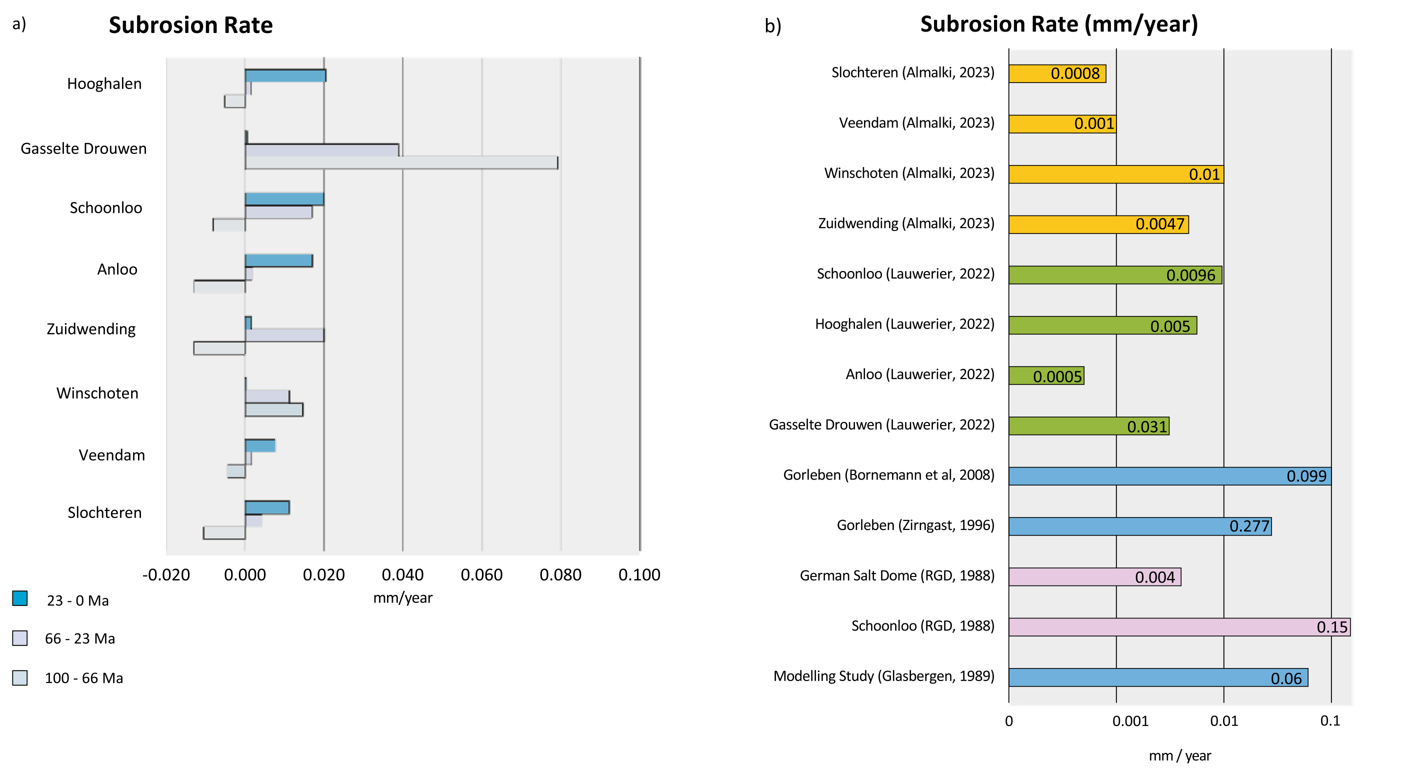


Figure 5.6a) Subrosion rates of eight salt structures in the Netherlands and three different time intervals that were studied in detail during the COPERA (2020 - 2025) research programme (Verhoef et al., 2021). For location of the salt structures, see figure 5.1. b) When comparing the average subrosion rates in Figures 5.6 a and 5.5b, we note that there is a difference between the calculated subrosion rate of the Schoonloo diapir by Rijks Geologische Dienst (1988) and that of Lauwerier (2022). In the study of Rijks Geologische Dienst (1988), it was assumed that the caprock has formed within 10 million years, although it could have taken much longer, resulting in a lower subrosion rate. The calculated negative erosion rates result from the method used and the uncertainty in the data. The negative rates suggest that the subrosion rate in these cases was actually very close to zero, or even zero: the subrosion rate are too low to be accurately estimated with the method used (Zirngast, 1996). In Figure 5.1b the scale is logarithmic and in this safety case, we assume a conservative value of 0.1 mm/year for the subrosion.

between the anhydrite and overlying sediments, possibly involving microbial activity (sulphate reducers, Caesar et al., 2019). While anhydrite, gypsum and calcite/carbonate are a typical cap rock sequence, there is a large variety in the presence, composition, and thickness of cap rocks between salt domes.

The caprock can either have a low permeability and protect against further subrosion of the dome top or have a high permeability (Geluk et al., 1993), such as the caprock of the Zuidwending salt dome, which has open cavities at the bottom of the caprock, between 20 and 100 cm in size, where fluid transport has occurred (Geluk et al., 1993). Subrosion can also affect the flanks of a salt dome (flank subrosion; Glasbergen, 1989). Over long-time scales, subrosion can lead to the disappearance of entire soluble rock formations from the geological record.

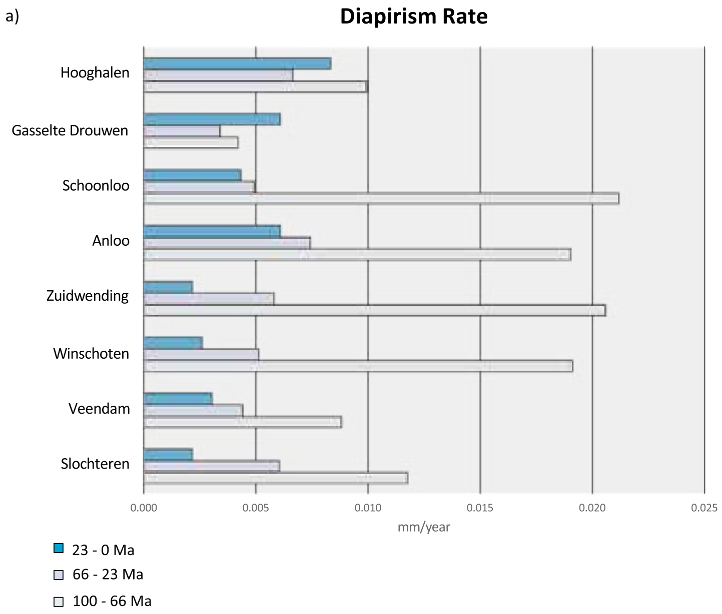
Based on the thickness of the caprock of the Schoonloo and Pieterburen diapir, the Rijks Geologische Dienst (1993) estimated that the subrosion rate for these two diapirs is about 0.15 mm/year. In Germany, estimates of subrosion rates range between 0.004 – 2.2 mm/year (See overview in Rijks Geologische Dienst, 1988). In Prij et al. (1993), the subrosion rate was modelled with a power function as a function of depth, based on a limited number of observations, giving a subrosion rate of less than 0.1 mm/year at depths of 200 m and more. A range from 0.1 – 0.2 mm/year was obtained by Bornemann et al., (2008) for the subrosion rate at the top of the Gorleben salt dome (~250 m depth) since the Elsterian glacial stage.

As part of the COPERA (2020 – 2025), Lauwerier (2022) and Almalki (2023) found that the average subrosion rate is 0.08 mm/year for the salt domes studied. In general, they found that the subrosion rate increases with time. This might be related to the depth of a salt structure: the closer the salt dome is to the surface, the higher the subrosion rate due to groundwater flow (Prij et al., 1993). Lower values for subrosion of 0.01 mm/year for the top and 0.001 mm/year for the side of a salt dome were found by Li et al. (2024) although the rate depended strongly on the assumed groundwater flow.

In summary, the generic subrosion rate that we assume in the present work for our generic repository is in the order of 0.01 to 0.1 mm/year, but the uncertainty in the data is large (Fig. 5.6a and b).

5.2.4.2 Diapirism

Diapirism, also known as halokinesis, is the process by which a salt dome rises upwards. The prevailing view is that differential loading is the driving force behind the development of salt domes (Hudec and Jackson, 2007; Vendeville, 2002). In this model, for example, a local thick sequence of overlying sediments could displace bedded salt laterally and locally push evaporites upwards through the over-



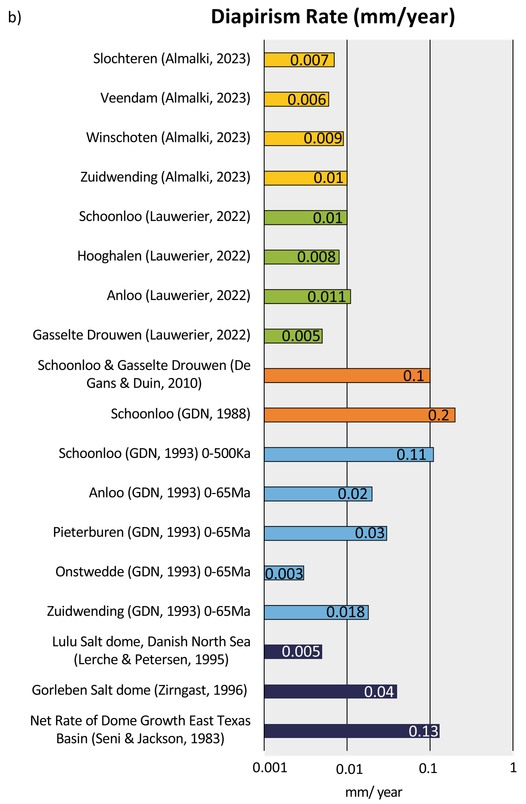


Figure 5.7a) The net diapirism rates of eight salt structures in the Netherlands and three different time intervals that were studied in detail during (COPERA (2020 – 2025), Verhoef et al., 2021). For the location of the salt structures, see figure 5.1. b) comparison of the average diapirism rate in the Netherlands with salt domes in Denmark, USA and Germany. In this safety case, we assume a value of 0.1 mm/year for the subrosion. Note that the scale is logarithmic.

burden. Differential loading can also result from the a passage of thick ice sheets (Lang et al., 2014). Extensional tectonic activity can also promote diapirism, by reducing the thickness and strength of the overburden. Indeed, most salt structures have developed during periods of tectonic activity (Geluk, 2005; Harding and Huuse, 2015; Hudec and Jackson, 2007; Jackson and Vendeville, 1994; Lauwerier, 2022; Vendeville, 2002; Zirngast, 1996). Although differential loading is thought to be the main driving force for the development of diapiric salt structures, other forces such as upward buoyancy of salt, due to density differences with surrounding formations, also play a role. Evaporite formations can have a lower density than surrounding sedimentary rocks, which become progressively denser with thickening and compaction (Fuchs et al., 2011) while the density of evaporites does not change significantly with depth, resulting in a density inversion.

For salt domes in the Netherlands, the rate of diapirism was initially estimated to vary between -0.04 – 0.06 mm / year (Rijks Geologische Dienst, 1993) with some outliers, such as the Schoonloo salt dome (0.11 mm / year). However, the latter estimate might be too high, according to Baker et al. (2001). A more recent study by Lauwerier (2022) and Almalki (2023), using the salt balance method (Zirngast, 1996) suggests that the average diapirism rate varies between less than 0.005 mm/year and 0.02 mm/year, with the highest average diapirism rates (Fig. 5.7a) having occurred during the Late Cretaceous (100 – 66 Ma). This high rate coincides with a major tectonic phase (see section 5.5) in the Netherlands. Overall, diapirism rates tend to have decreased throughout the last 100 million years, or to have remained roughly similar. These values are in line with what is observed in other countries, although salt dome growth in the east Texas basin appears to be significantly faster (Fig. 5.7b), which is presumably a result of the specific regional tectonic evolution. Hence, diapirism rates from the relatively small area of north–western Europe that contains salt domes and has a common tectonic history, are considered the most representative for the Netherlands. It should be noted that the diapirism rates of both Lauwerier (2022) and Almalki (2023) are average rates over a long-time span. Their estimate includes periods of no uplift and high uplift. Li et al. (2024) found that the highest diapirism rate in the Netherlands (onshore) during Neogene period (last 23 million years) was 0.006 mm/year.

In summary, the diapirism rate assumed for the present study is between 0.001 and 0.1 mm/year (Almalki, 2023; Lauwerier, 2022; Wildenborg et al., 1993), but the uncertainty in the data is large (Fig. 5.7a and b).

5.2.4.3 Long term safety: diapirism and subrosion

In the context of geological disposal, diapirism and subrosion processes are important to consider as they could, over time scales of millions to tens of millions of years, lead to disruption of the geological barrier around the GDF. If diapirism alone is active (Fig. 5.8), the depth of the GDF relative to a static surface gradually decreases, while the thickness of the salt surrounding the GDF does not change until the salt reaches the surface, where it would begin to erode rapidly (Waltham, 2008). Eventually, the GDF itself would reach the surface, and be eroded into the biosphere (Prij et

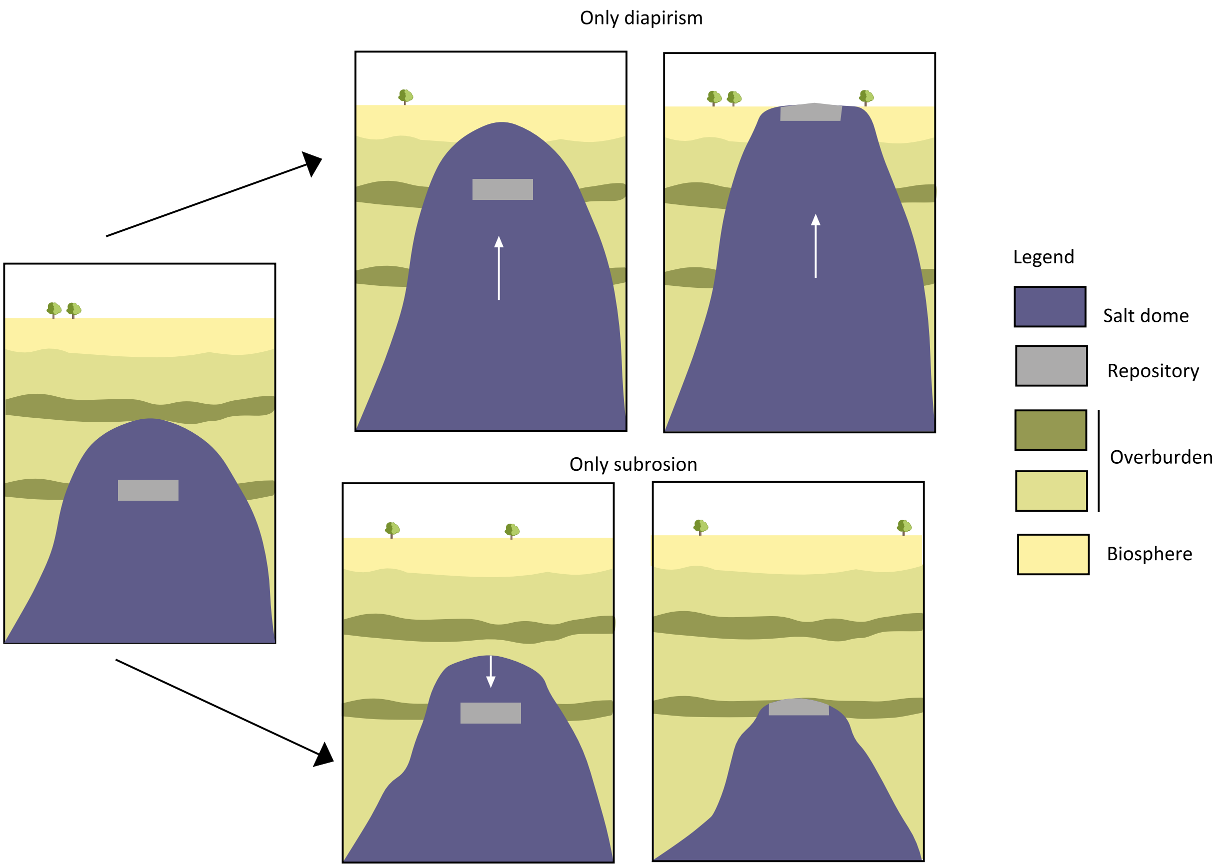


Figure 5.8) Highly schematic overview of how diapirism and/or subrosion could eventually result in a release. In the upper figures, only diapirism is active. Over long times and when the diapirism rate is large enough, diapirism could bring the repository to the surface, leading to a release into the biosphere (Prij et al., 1993). When only subrosion is active, the salt surrounding the repository could eventually be fully eroded, resulting in the contact between groundwater and the waste, and hence a release. Note that both processes are likely to be active concurrently and possibly influence each other.

al., 1993). If only subrosion is active (Fig. 5.8), the natural rock salt barrier around the GDF is gradually dissolved while the depth of the repository does not change. At some point in time, the GDF is exposed to groundwaters from the surrounding sediments and radionuclides are released into the groundwater. However, the geological environments of most salt domes in the Netherlands indicate that both processes are likely to be active concurrently, so it is important to consider both diapirism and subrosion of salt domes and their evolution through time. In all cases, however, any exposure of the GDF and any release of radioactivity are expected to be well beyond the period in which the hazard potential of the waste has decayed to such an extent that it is of no significant concern for safety assessment.

5.3 Identification of uncertainties

Three areas of uncertainty are currently considered of most significance when considering using a salt formation for geological disposal, namely its thickness and depth, its internal structure and homogeneity, and its short- and long-term evolution. The quality and coverage of the data on the thickness and depth of the rock salt of the Zechstein group and the Röt formation (the two most promising formations for a GDF), are not yet high enough to justify consideration of potential siting areas. This is particularly the case for the Röt formation. Although the thickness of the Zechstein Group in the northern-eastern part of the Netherlands has been well documented, in other places the available data are limited. These uncertainties will need to be studied in the future, but this is not urgent as the GDF siting programme is still decades away.

While data are available on the internal structure and homogeneity of salt structures, the uncertainty is large, in part because it is challenging to image salt structures seismically (particularly, within salt domes) and to interpret the data. Salt has a high P-wave velocity, which makes it difficult to post-process the data to produce a correct image, and some salt domes have complex internal structures (Jackson and Hudec, 2017), which makes interpretation of seismic sections challenging. These uncertainties are recognised and need to be studied in the future. An initial scoping study could be performed using publicly available data (Matenco and Beekman, 2023) on the homogeneity of salt structures (e.g., Hunfeld et al., 2023; TNO, 2014; Van Gent et al., 2011). Some of these data have been collected in COPERA (2020 -2025) (Hunfeld et al., 2023) from publicly available data sources (cores). However, the dataset is limited. To obtain a better understanding of the variations among potential host rocks, this database needs to be extended, specifically with respect to the intrinsic properties of the salt, such as diffusion rate and permeability. Data on the mechanical properties (creep) of salt are readily available.

With respect to the long-term evolution of salt structures, while the subrosion and diapirism rates are relevant (Lang et al., 2014), they are only expected to impact the GDF system long after its hazard potential has diminished and well beyond the period of concern for safety assessment. As described above, some of these rates have been determined for specific salt domes, but the methods used are not precise and they derive rates over a relatively long time interval (millions of years, Almalki, 2023; Lauwerier, 2022). They are essentially an average for a whole salt dome that can include multiple subrosion and diapirism stages. While these values are sufficient for this stage of the programme, more precise diapirism and subrosion rates would help to understand better the evolution of salt structures through time. With respect to short-term evolution (tens to hundreds of thousands of years), which is of relevance to the safety case, the specific interest is on if and how major climate-driven changes such as an ice age could influence the diapirism and subrosion rate. Based on these observations, numerical models may need to be developed to predict the future evolution of salt structures.

For the much shorter term evolution (hundreds of years), inflow of brine is one of the processes that needs to be studied as it could affect, for example, the corrosion of the waste packages which could result in gas generation. While salt is essentially dry, some inflow of brine has been observed inside experimentally heated, freshly drilled boreholes (e.g., Finley et al., 1992; McTigue, 1986). More recently, the Brine Availability Test in Salt (BATS) was conducted at WIPP. This experiment included a borehole with a heater and a liquid brine sampler, surrounded by measurement equipment, such as temperature and acoustic emissions sensors. The results of the BATS experiments show that when the heater is turned off, there is a temporary, significant increase in brine inflow (from 0.0004 grams per minute to 0.03 grams per minute in a borehole with a surface area of approximately 0.75 m^2). While the processes that led to this peak inflow are not yet fully understood, observations together with computer simulations suggest that abruptly turning off the heater caused shock thermal shrinkage of the bedded salt near the borehole, leading to micro-fracturing that enhanced the local permeability of the Excavated Damaged Zone (EDZ) which caused the peak inflow (Kuhlman, 2023; Kuhlman et al., 2024a). The BATS experiment was performed over a relatively short period of time (weeks and months) and the heater was abruptly turned off while the EDZ was not healed. In a GDF, however, the heat output of the waste will gradually decrease over hundreds of years and the EDZ is expected to heal within a few decades, due to salt creep. Both processes will limit and potentially stop any brine inflow. Although domal salt tends to be drier than the bedded salt in the BATS test (see section 5.1.1), it is still important to understand this processes and how it would potentially affect the behaviour of the COPERA (2020 - 2025) salt GDF concept.

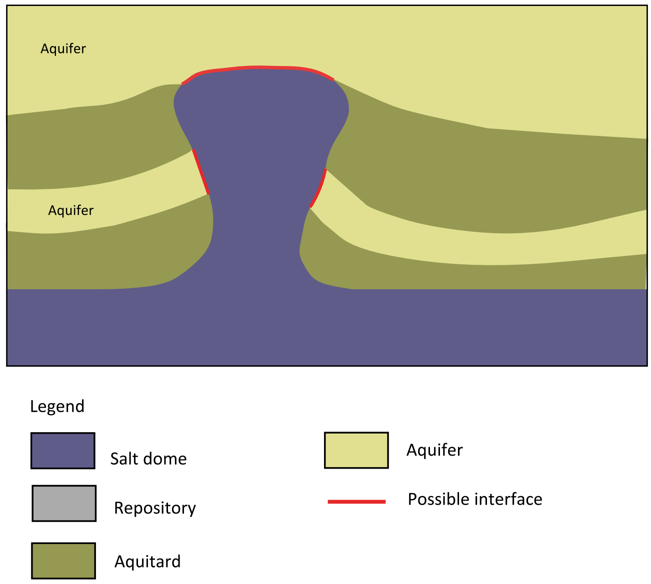


Figure 5.9) salt dome which has grown through overlying rock layers. The red lines marked 'interface' indicate possible connections between an aquifer and the salt. At these locations, subrosion takes place.

5.4 Surrounding geological formations

5.4.1 Groups and formations

The bedded and dome salt formations of the Zechstein Group and the Röt formation lie within a thick sequence of sedimentary formations. Textbox 5.2 provides a brief description of the characteristics of the major lithostratigraphic units, based on the stratigraphic nomenclature used by the Dutch Geological Survey (Van Adrichem Boogaert and Kouwe, 1993-1997). In summary, there can be a wide variety of sediments below and above a salt structure. Depending on location, this can range from sandstones and conglomerates of the Early Triassic Germanic Triassic Group, salt of the Muschelkalk and Keuper formations and clay in the Upper North Sea Group. These deposits vary laterally and can be absent in some regions. For example, the Zechstein Group is characterised by sandstone in the south and evaporites in the north, while it is not present in south-west of the Netherlands, due to non-deposition or erosion. The wide variation in overburden lithologies across the Netherlands means that, for the purposes of this safety case, it is not possible to select a single stratigraphy for the overburden that would be representative for all salt domes.

Some of the sediments in the overburden have a high permeability and act as aquifers, through which radionuclides can potentially migrate to the surface if they leave the repository. Diapiric evolution of a salt dome means that it may have encountered an aquifer as it has risen (Fig. 5.9, Hart et al., 2015a). Currently, there are 24 aquifers (Hart et al., 2015b; Van Adrichem Boogaert and Kouwe, 1993-1997) that occur on top of, next to or below the Zechstein Group deposits in the north and northeast of the Netherlands; depending on location, some of these could be in direct contact with salt. In contrast, other sediments in overlying formations have a low permeability and are aquitards, including the Boom and Rupel clays that are currently being considered as the alternative host rocks for the Dutch GDF (Neeft et al., 2024b).

5.4.2 Deep glacial erosion features

The Upper, Lower and Middle North Sea Group sediments form the uppermost part of the stratigraphic column over most of the Netherlands and are thus the formations that have been most affected by surface processes such as ice ages. The Quaternary Period, spanning approximately the last 2.6 million years, is characterised by cyclic glaciations that resulted in extensive ice cover in the northern hemisphere. In the last million years these have occurred approximately every 100,000 years, alternating with

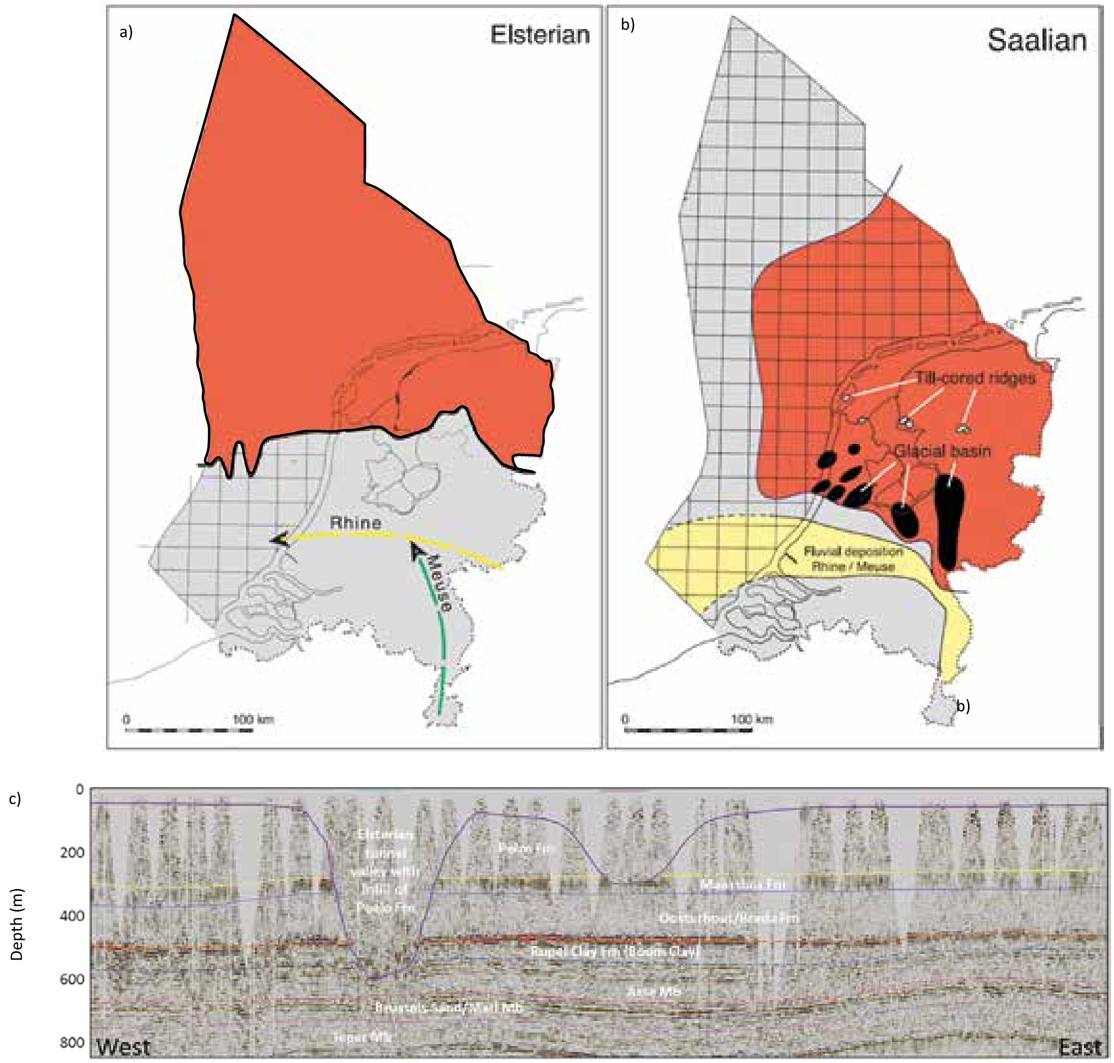


Figure 5.10) The maximum extents of ice cover during the Elsterian (left) and Saalian (right) glaciations (shown in red) and the location of glacial basins and till-cored ridges in the Netherlands. The glacial basins are rarely deeper than 150 m. The lower figure shows deep 'buried valleys' discussed in the main text. For the location of these deep glacial erosion channels and the line of the seismic section, see Figure 5.1. Figure from Verhoef et al. (2017).

warm, interglacial periods, such as the present day. The current interglacial, the Holocene, began around 11,000 years ago as global warming terminated the Weichselian glaciation, resulting in melting of the remaining ice sheets in Europe and North America. During these Quaternary glacial cycles, the Netherlands experienced periodic coverage by ice sheets originating from a Scandinavian ice cap, extending across the Baltic and North Sea regions (Fig. 5.10). The growth, movement and eventual decay of these ice sheets had significant impact on the subsurface, for example, influencing the deep and shallow groundwater flow regimes and causing occasional deep erosion.

The intensity of these glaciations has varied over time, and not every glaciation has been sufficiently intense to cause ice coverage as far south as the Netherlands. The extent of ice coverage in the Elsterian (475 – 410 ka ago) and the Saalian (370 – 130 ka ago) glaciations, both of which caused ice sheet development over parts of the Netherlands, is shown in figure 5.10. The ice sheet of the most recent, Weichselian (115 – 10 ka ago) glaciation, did not extend across the continental Netherlands and is therefore not shown.

Some glacial periods have resulted in erosion that, were it to occur again with similar intensity, could affect a salt dome in which a GDF is located (Prij et al., 1993), depending on its depth and location. During the Elsterian glaciation, subglacial channels, also known as "tunnel valleys" or "buried valleys", were formed. These subglacial channels cut down into pre-Quaternary strata in Germany by up to 500 m (Keller, 2009) and up to 600 m in the Netherlands (See figure 3-4 in ten Veen et al., 2015 and Figure 5.9, ten Veen et al. (2015)) and have a width of between 1 to 2 km. After their formation, the subglacial channels were rapidly backfilled with glacial sediments. As shown in Figure 5.1 and 5.10c, some of these channels are located above salt formations, but both bedded salt and salt pillows are not affected, as they are situated well below the maximum 600 m erosion depth. In contrast, the tops of salt domes can be located at a shallower depth. In total, only one salt dome appears to coincide with the glacial channels, namely (Fig, 5.1, number 11) Onstwedde, whose top is located at a depth of 250 m (Bosch et al., 2009) although the relationship of this dome to glacial channel erosion is still under discussion (Rijks Geologische Dienst, 1993). By locating the repository at a depth of 750 to 850 m, it is highly unlikely that future subglacial channels could cause a problem.

However, research is needed to understand better the formation of these channels and whether a greater depth could potentially be reached in future glaciation scenarios.

In addition to tunnel valleys, subglacial depressions (glacial basins: see Figure 5.10) developed in the Netherlands during the Elsterian and Saalian glaciations. The Saalian glacial basins are rarely deeper than 150 m and are not expected to affect a GDF (de Gans et al., 2000; Van Dijke and Veldkamp, 1996). Till-cored ridges, on the other hand, are geological landforms primarily composed of glacial till—a mixture of unsorted sediment (including clay, sand, gravel, and boulders) deposited by glaciers. They are associated with the movement and retreat of glaciers and are often found in formerly glaciated regions. The ridges can reach up to 110 m in height in the Netherlands (ten Veen et al., 2015) and only affect groundwater flow locally, near the surface. Therefore, like glacial basins, they are not expected to affect a GDF.

Protracted cold episodes during glacial periods can result in permafrost conditions developing over large areas ahead of advancing icesheets. During the most recent, Weichselian glaciation, the Netherlands experienced permafrost conditions in the soils and sediments above the Zechstein formations, which penetrated to varying depths. The previous OPERA research programme studied the potential for future permafrost development. It was concluded that, for any location in the Netherlands, the depth of permafrost could range from 120 to 200 m but will not exceed 270 m, using the best estimate of air temperature evolution during the Weichselian glaciation (Govaerts et al., 2015). This depth is not sufficient to directly impact the GDF. However, permafrost will also affect groundwater movement and could lead to lower subrosion rates during this glacial period.

Future post-glacial seismicity is a further possibility that needs to be addressed. During periods of ice cover, seismic activity tends to be suppressed, leading to the accumulation of continental-scale tectonic stresses. When the ice melts, the reduction in load on the lithosphere can trigger earthquakes. This post-glacial seismic activity is expected to concentrate mainly along existing major fault zones (ten Veen et al., 2015), which would be avoided when siting a GDF. There is no evidence to show that Quaternary glaciations have reactivated major faults in the Netherlands area to the extent that the containment properties of salt formations would be impaired. This is evident by the fact that the Zechstein Group is currently an important seal for many hydrocarbons in the Netherlands, which have been trapped by the salt since the Late Jurassic. During and after trapping of these hydrocarbons, the Netherlands was affected by both extensional and compressional tectonics that are generally associated with earthquakes (see section 5.5). The conclusion that rock salt retains its integrity after an earthquake was also reached by Minkley et al. (2010) for a sufficiently thick salt barrier, based on the analysis of rock burst in salt mines.

As explained in section 5.2.4.1 and 5.2.4.2, glaciation can also influence the diapirism and subrosion rate. If an ice sheet encroaches progressively over a salt dome, it will result in differential loading, which could increase the diapirism rates (Lang et al., 2014). This increase is only temporary, occurring during periods of ice sheet advance and retreat. Subrosion rates could also vary during a glacial period. During the melting of an ice sheet, large volumes of fresh, sub-glacial water can be forced into permeable formations, possibly resulting in an increased subrosion rates. In contrast, permafrost will limit the amount of fresh water entering the ground, resulting in a higher salinity of groundwater and consequently lower subrosion rate. More research is needed to fully understand the impact of ice ages on a range of processes, including permafrost depth, groundwater flow patterns and rates in sediment overlying salt formations, deep erosion and rates of diapirism and subrosion.

5.5 Generalised geological history

This section briefly discusses how the horizontal bedded Zechstein group and Röt formation evolved over geological time (Fig. 5.11), which can be used as an indicator of future evolution over the next million years.

The Upper Rotliegend group underlying the Zechstein Group was deposited during the Middle to Late Permian (300 – 259 Ma). The Late Permian was characterised by the final assemblage of the tectonic plates that formed the supercontinent Pangea (Stampfli and Borel, 2002). The southern part of Pangea consisted of South America, Africa, Antarctica, Australia and the subcontinent of India. Together, these are generally referred to as Gondwana (Fig. 5.11). To the north, the supercontinent consisted of North America and Eurasia, generally referred to as Laurasia. Laurasia and Gondwana began to collide in the Carboniferous, which eventually resulted in the formation of the Central Pangean Mountains. North-western Europe and, more specifically, the Netherlands, was located just north of these mountains (also known as the Variscan mountain belt, Gast et al., 2010). Following the continental collision and widespread volcanism, the northern part of the Netherlands was affected by large scale subsidence (Geluk, 2005; van Wees et al., 2000), which resulted in the formation of the so-called Northern and Southern Permian basin. This Permian basin covered most of the present-day North Sea, the northern part of the Netherlands, Germany and Poland. While marine intrusions into the Permian basin occasionally occurred, these did not result in the development of a thick evaporite layer (Legler and Schneider, 2008).

At about 257 Ma (Brauns et al., 2003; Szurlies, 2013), the Permian basin, which was below the global sea level, became flooded multiple times in less than 4 million years (Szurlies, 2013), as a result of a connection with the Arctic Ocean in the north. During high sea levels in the Permian basin, carbonates were deposited around its margin, while no sediments were deposited in its centre. During periods of low sea level, thick sequences of Zechstein salt were deposited in the centre of the basin, while at the margin of the basin, sediments were eroded. This eventually resulted in the five evaporite cycles that are currently observed in the Dutch subsurface (TNO-GDN, 2023i). Hence, the Zechstein Group was deposited in a relatively short period due to subsidence of the northern and southern Permian basin.

In the Triassic (251 – 201 Ma), the North Atlantic rift started to develop between Greenland and Scandinavia (Fig. 5.11; Late Createcous, Lundin, 2002). A branch of this North Atlantic rift system was located in the North Sea: the North Sea Rift system (Pharaoh et al., 2010). This rifting marked the start of the break-up of the supercontinent Pangea and is characterised by localised large scale crustal extension, which, together with the arid climate, resulted in the development of a large salt basin that covered most of the Netherlands, except the southern part of the country. Within this salt basin, the Triassic Röt formation was deposited (Bachmann et al., 2010). The widespread extension in the Triassic also resulted

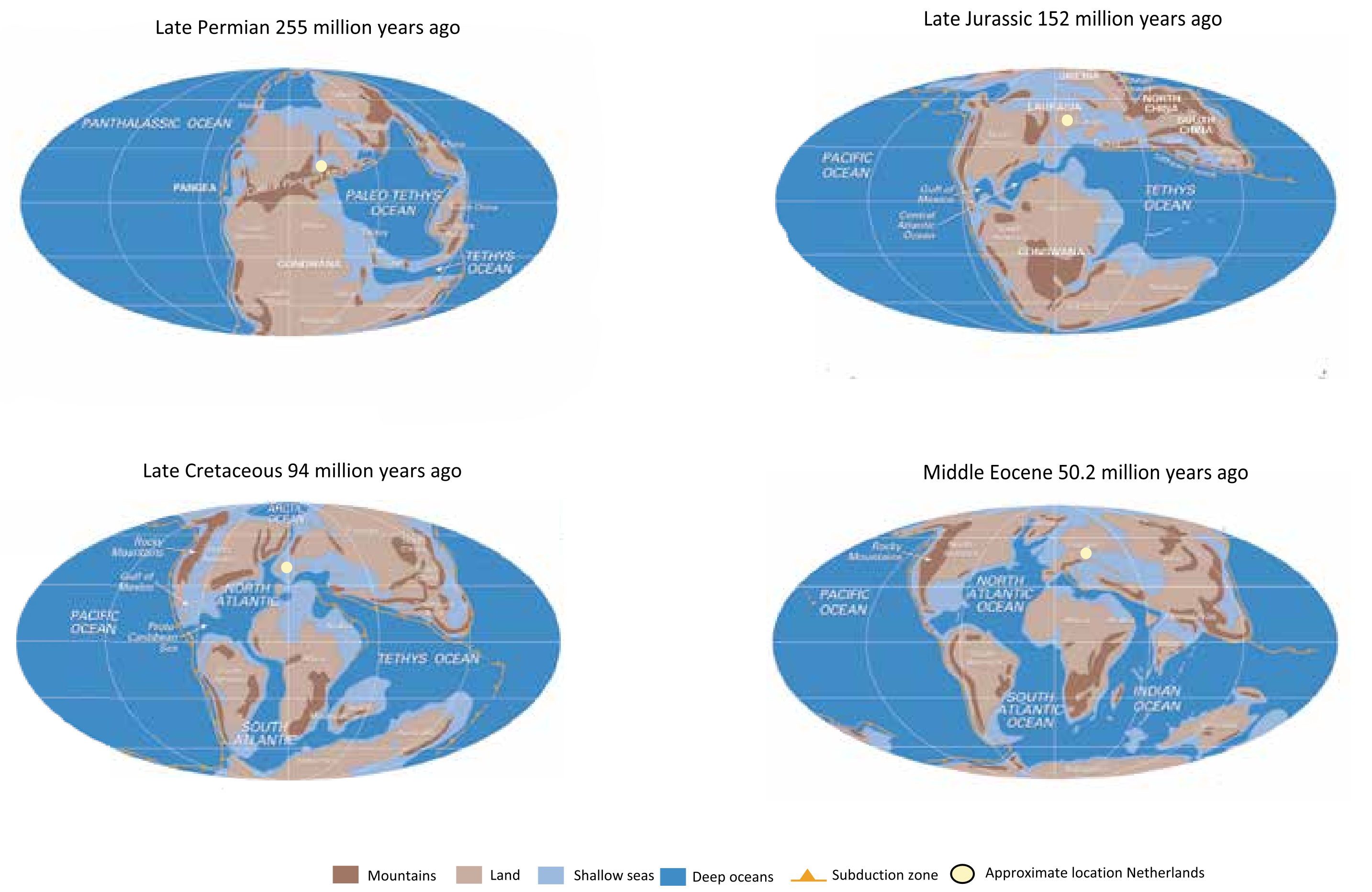


Figure 5.11) Location of the Netherlands through time (yellow dot) during the Late Permian (255 million years ago), Early Triassic (237 million years ago), Late Jurassic (152 million years ago) and the middle Eocene (50.2 million years ago). Figures from www.britannica.com.

in the first period of salt movement (Pharaoh et al., 2010; Remmelts, 1995; Remmelts et al., 1993).

Rifting and associated extension in the North Sea Rift System continued up to the Early Cretaceous. While the Atlantic Ocean continued to widen, the ocean separating Africa (and Arabia) from Eurasia started to close in the south, due to plate subduction along the southern margin of Europe. This led to north to north-east directed compressional stresses in the Netherlands, resulting in an inversion and uplift which, in turn, caused widespread erosion in most areas, although the Lower Saxony basin was less affected by this inversion. This was also the period in which widespread salt diapirism occurred and some salt domes might even have reached the surface (Almalki, 2023; Harding and Huuse, 2015; Lauwerier, 2022). Their movement has continued up to recent times, although its rate diminished significantly and was, at least in the North Sea (Harding and Huuse, 2015), driven by sediment loading and possibly differential loading by ice sheets.

The south of the Netherlands was also affected by the Northwest European rift system during the Cenozoic (66 – 0 Ma), particularly during the Late Oligocene (33.9 - 23.03 Ma). This rift reactivated the Permian aged Roer Valley Graben, which resulted in the deposition of a thick package of sediments (Geluk, 2005). This area is still tectonically active: most natural earthquakes in the Netherlands occur in this region (Muntendam-Bos et al., 2022).

Based on our understanding of the geological history of the Netherlands, what can be said about the next 1 million years? While the Netherlands has experienced both compressional and extensional tectonic periods during the last roughly 250 million years, it is expected that, from a tectonic point of view, the large-scale stress and tectonic regime over the next million years will be like the last few million years. Hence, subduction along the southern margin of Europe is expected to continue as it did for roughly the last 65 million years, and tectonic stresses will not change significantly in the next million years. Consequently, the average diapirism rate is expected to remain like that of the last 23 million years (0.001 and 0.1 mm/year, see section 5.2.4.2). Likewise, the average subrosion rate will also remain like that of the last 23 million years, which does include Quaternary Ice Ages (0.01 and 0.1 mm/year, see section 5.2.4.1). After 1 million years, a GDF located at depth in a salt dome repository is expected to have risen about 100 m relative to the present-day surface, while the same amount of salt is removed from the top of the salt dome by subrosion. Furthermore, it is expected that large earthquakes remain restricted to the Lower Rhine Graben in the south of the Netherlands, where there are few or no salt deposits (Fig. 5.12). The number of anthro-

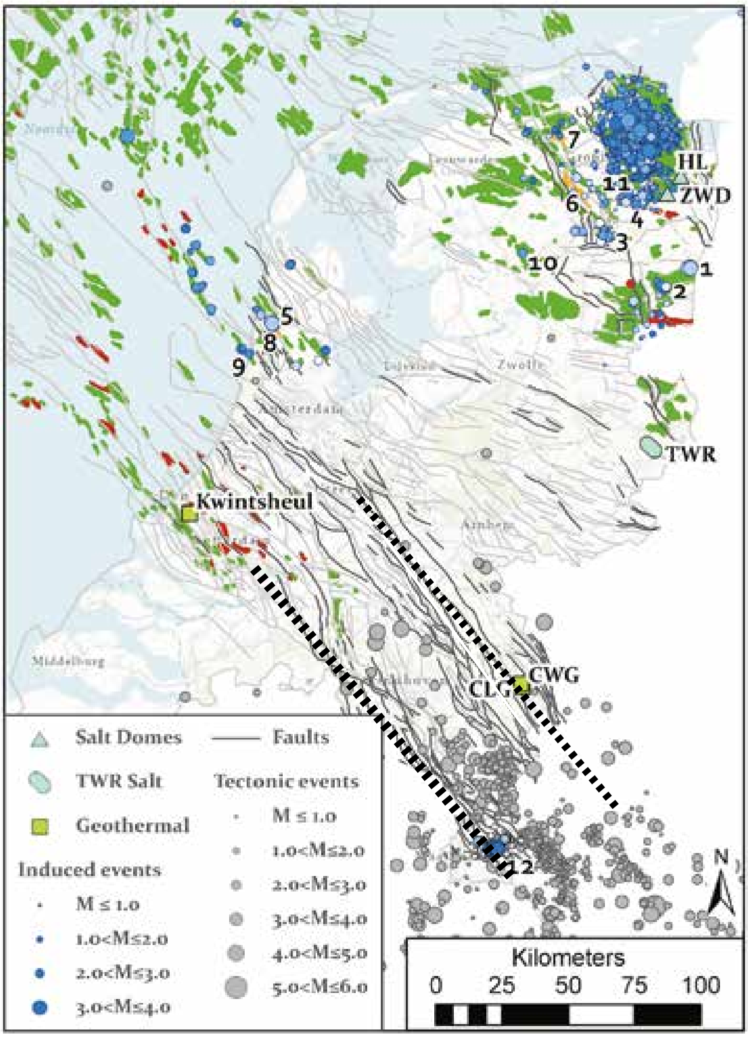


Figure 5.12) Overview of seismicity in the Netherlands. The dark lines indicate potentially tectonically active faults, the light grey lines indicate faults in Permian formations. Oil and gas fields are in red and dark green, respectively. The gas fields using underground gas storage (UGS) are shown in orange. The approximate location of the Roer Valley Graben is indicated by the grey area. Figure from Muntendam-Bos et al. (2022). TWR: Twente-Rijn salt dome; HL: Heiligerlee salt dome; ZWD: Zuidwending salt dome.

pogenic earthquakes in the north resulting from gas extraction are expected to decrease with time, although it is unclear how long the re-establishment of the natural stress field, which results in the earthquakes, will take (Zöller and Hainzl, 2023).

On shorter time scales, diapirism and subrosion are both influenced by climate and glaciations. The times of glaciations have been predominantly a result of natural variations in Earth's orbital behaviour (Milankovitch cycles), but other factors are involved and it is unclear when the next glaciation will occur. Fischer et al. (2021) provide an overview of different model studies on the onset of the next glaciation. These models agree that, with an increased level of CO_2, a new ice age will not occur in the next 50,000 years and possibly not in the next 100,000 years or even much longer (Ganopolski et al., 2016; Lord and Thorne, 2020).

5.6 Assumptions for the post-closure safety assessment

5.6.1 Host rock

For the safety assessment, we assume that the GDF is in a salt dome and the unperturbed host rock is essentially impermeable. Hence, we assume that no advective and diffusive transport will occur through the rock salt. In the COPERA (2020 - 2025) safety assessment, it is also assumed that any fracturing in the EDZ will have closed by creep of the salt before the final closure of the repository. As the EDZ has a relatively low permeability compared to the backfill (Oosterhout et al., 2022), its importance as a pathway for radionuclide migration is limited.

5.6.2 Surrounding formations

The wide variation in overburden lithologies across the Netherlands means that it is not possible to select a single stratigraphy for the overburden that would be representative for all salt domes. Therefore, a stylised generic overburden is assumed for the performance assessment. It is assumed that the overburden consists of a combination of both unconsolidated and consolidated sediments with a relatively high permeability and porosity (an aquifer) and is intersected by faults. The assumption that an aquifer is present directly above the top of the salt dome means that any radionuclides that leave the GDF via one of the shafts could eventually be transported to the surface. Faults in the overburden could reduce the travel time of the radionuclides. When a location is selected, the stylised generic overburden can be replaced by a site-specific stratigraphy and lithologies for the overburden. While the stylised

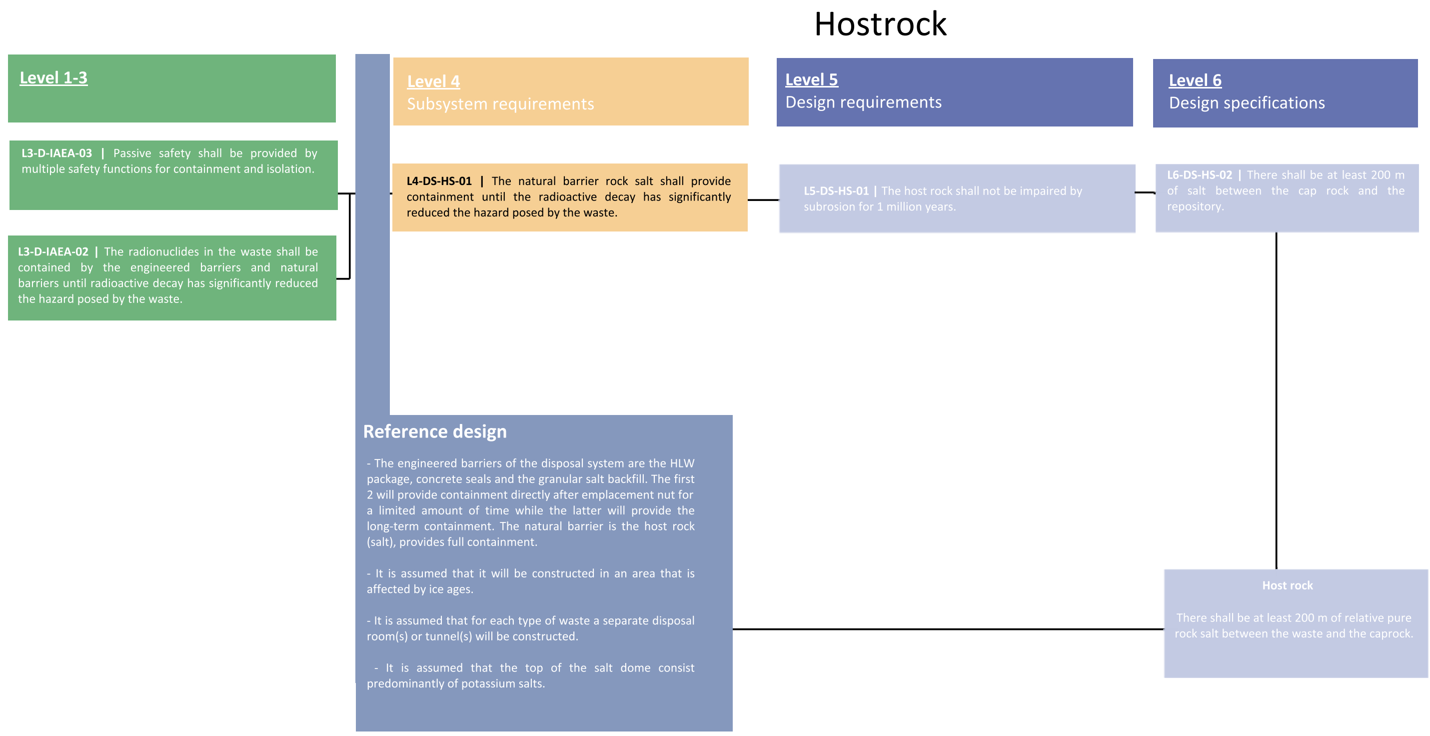


Figure 5.13) RMS for the host rock. DS stands for Disposal in Salt; HS stands for Host Rock. Currently, the RMS for the host rock is still under development and will be expanded in future.

generic overburden is only an approximation, its influence is found to be limited in the normal evolution scenario (See Chapter 8).

5.7 Host rock and the RMS

The containment function of the host rock is shown as a requirement at level four of the RMS: L4-DS-HS-01. The natural barrier rock salt shall provide containment until the radioactive decay has significantly reduced the hazard posed by the waste. This stems from requirements L3-D-IAEA-02 and L3-D-IAEA-03, as shown in Figure 5.13. For the rock salt to provide the required containment, it needs to have a minimum thickness and must be undisturbed. The minimum thickness is determined by requirement L5-DS-HS-01: The host rock shall not be impaired by subrosion for 1 million years. Our assumption that the subrosion rate is between 0.01 and 0.1 mm/year (see section 5.2.4.1), means that 100 m of rock salt is expected to erode within one million years. Taking uncertainties into account (in an alternative scenario, with double the expected subrosion rate, Chapter 7), there should be at least 200 m of undisturbed salt between the waste and the caprock. However, other processes could adversely affect the host rock salt, possibly increasing the minimum thickness required. For example, the evolution of temperature and the stress state within the host rock, which might cause extensional fractures to develop in the host rock, potentially leading to the ingress of ground water. More research is thus needed on these processes to develop models to determine whether, and to what extent, they could affect the integrity of the natural barrier and, if so, the implications for the minimum thickness of the host rock.

Box 5.1: Analogues

Since the start of research into geological disposal, it has been recognised that rock formations naturally demonstrate a significant capacity for containment and isolation—qualities that underpin the widely accepted finding that geological disposal is the most suitable strategy for managing long-lived radioactive wastes. The safety case revolves around understanding processes that have been active for millions of years in deep rock formations. Examining geological settings akin to those foreseen for a GDF helps increase confidence in our understanding of the nature, extent, and pace of such processes. Here, we discuss a few examples of natural analogues relevant to a rock salt GDF (Miller and Noseck, 2014).

Starting with the host rock, there are multiple natural analogues that demonstrate that rock salt can provide full containment for at least one million years, and probably for much longer. As already mentioned in section 5.1.1, the fact that 250 million years old rock salt exists indicates that it is impermeable. If salt deposits had even a very small permeability, they would not have existed for such a long time, since they would easily have been dissolved by groundwater (van Balen, 2010). Also, the fact that the rock salt acts as a seal formation for many oil and gas hydrocarbon deposits demonstrates that it is impermeable. In the Netherlands, gas has been trapped beneath the bedded Zechstein salt for millions of years (Pletsch et al., 2010). Another example is the Werra/Fulda salt deposit in Germany, which trapped large quantities of CO_2 produced by magmatic intrusions approximately 20 million years ago. This CO_2 still resides in the salt deposit (NEA, 2013a).

Another characteristic of rock salt is that it is very dry, which allows for exceptional preservation of organic materials. In the Hallstatt salt mines, for example, organic material from the Bronze Age had been exceptionally preserved (e.g., Reschreiter and Kowarik, 2019). This includes wooden tools, wood used in the construction of the salt mine (see Figure 1), leather, textiles and even human remains. Numerous tools used for mining and related activities have also been found, including pickaxes and shovels made from wood and bronze that were probably used in the process of extracting and processing salt. Organic material from the past has also been preserved in salt mines in Iran. A series of ancient, salt-preserved human heads have been discovered in the Chehrabad Salt Mine, dating back 2,400 years, based on DNA analysis and radiocarbon dating. The unique, dry preservation conditions in the salt mine have allowed the retention of hair, skin and even facial features on the heads (e.g., Aali et al., 2012). Both examples demonstrate that salt is dry and can preserve a wide range of materials over long periods. For this safety case and feasibility study, natural analogues suggest that it is unlikely that a thick, metallic HLW package will fail 1,000 years after emplacement, which

Figure 1) Fractured mine timber in the Hallstatt salt mine. This wood is over 2,000 years old and is exceptionally well preserved. Photo taken by D. Brandner, NHM Vienna and is from Reschreiter and Kowarik (2019).

is the behaviour conservatively assumed in the PA (See Chapter 8).

Analogues can also be useful for observing the compaction of granular salt backfill. This is because laboratory experiments used to understand compaction are limited in timescale (months to years) while compaction is likely to take a few hundreds of years. To reduce the length of experiments, they are performed at stresses that are higher than expected in situ (at a given porosity). This results in the activation of different compaction mechanisms than those expected within the repository, and this results in uncertainties in extrapolation to lower stresses. This in turn results in uncertainties in the timescales required for the sealing of open spaces in rock salt and in quantifying the active processes.

An analogue study that addresses this issue has looked at backfill that has been compacted in-situ. Forty years ago, the Sigmundshall mine in Germany was filled with granular halite. This backfill has compacted down to 1% porosity within 40 years, under low stress conditions. Microphysical modelling of the Sigmundshall backfill body indicates that such rapid compaction can only occur if fluid-assisted grain boundary diffusion (pressure solution creep) is the dominant compaction mechanism. The microstructure of the Sigmundshall backfill shows indentation and truncation at the grain-scale, indicating diffusional processes taking place at stressed interfaces between adjacent grains. Based on the study of this analogue, it can be concluded that pressure solution is the dominant compaction mechanism at low stress and must be considered when determining the timescales for the sealing of a backfilled repository in rock salt, especially at high porosity when the resistance of the backfill is low and stresses on the backfill are small.

Box 5.2: Overlying and underlying geological formations

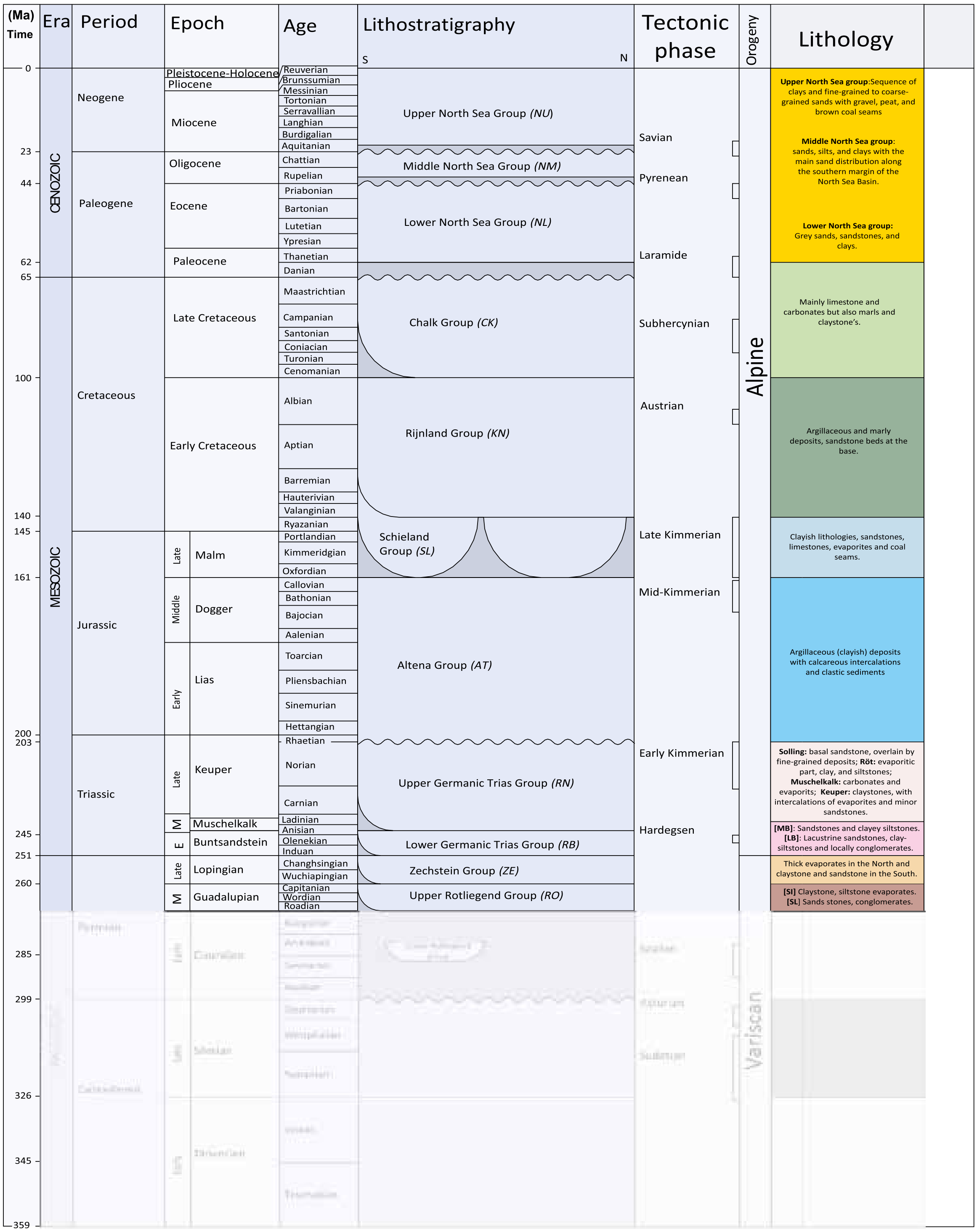


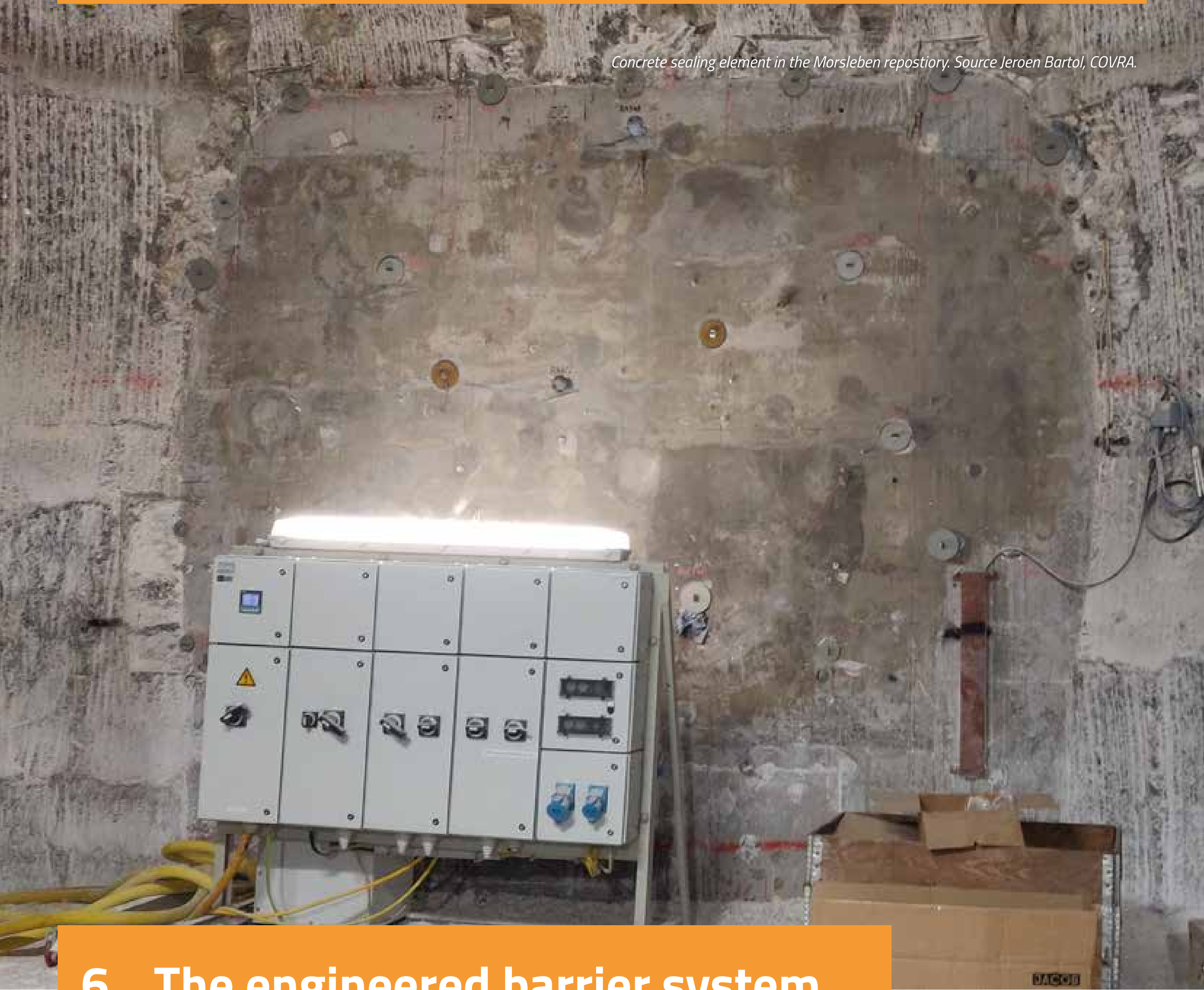
Figure 1) Tectonic and stratigraphic chart of the Netherlands. The figure has been modified after Van Dalfsen et al. (2006) and in turn is based on Van Adrichem Boogaert and Kouwe (1993-1997).

This description begins with the formation immediately underlying the bedded Zechstein Group (the Upper Rotliegend group) and moves upwards through the overlying formations. The Upper Rotliegend Group can be subdivided into two formations: the Slochteren (SL in Figure 1) and the Silverpit formation (SI in Figure 1, Geluk, 2005; TNO-GDN, 2023h). These two formations are lateral equivalents, with the transition between the two occurring in a narrow zone in the north of the Netherlands. The two formations consist mainly of sandstones and conglomerates (Slochteren formation), and claystone, siltstone and evaporites (Silverpit formation, Geluk, 2005). The Slochteren formation is the reservoir for the Groningen gas field. Overlying the Upper Rotliegend is the Zechstein Group (for description see section 5.2.2) which in turn is overlain by the Early Triassic Lower Germanic Triassic Group. It consists of lacustrine sandstones, clay-siltstones and local conglomerates (Lower Buntsandstein, LB in Figure 1), sandstones and clayey siltstones (Main Buntsandstein Subgroup, MB in Figure 1, Geluk, 2005). On top of this formation is the Upper Germanic Trias Group consisting of the Solling (basal sandstone, overlain by fine-grained deposits), Röt (evaporites, clay and siltstones; see section 5.2.3), Muschelkalk (carbonates and evaporites) and Keuper formations (clay stones, with intercalations of evaporites and minor sandstones, Geluk (2005)). The group was deposited in the Middle and Late Triassic (Van Adrichem Boogaert and Kouwe, 1993-1997).

The Upper Germanic group is overlain by the Altena Group, which is characterised by mainly argillaceous deposits with calcareous intercalations and clastic sediments (TNO-GDN, 2023a; Van Adrichem Boogaert and Kouwe, 1993-1997). The Altena group is overlain by the Niedersachsen Group, which is characterised by mainly argillaceous lithologies, but also some sandstones (restricted to the basin fringe), limestones and evaporites, and is Late Jurassic – Early Cretaceous in age (TNO-GDN, 2023e). These are overlain by the Rijnland Group (argillaceous and marly deposits, with sandstone beds at the base, which are Early Cretaceous in age (TNO-GDN, 2023f)), the Chalk Group (mainly limestone and carbonates but also marls and claystones that are Late Cretaceous in age (TNO-GDN, 2023b; Van Dalfsen et al., 2006)) and the Lower, Middle, and Upper North Sea Group. The Lower North Sea group is characterised by grey sands, sandstones and clays (TNO-GDN, 2023c). The middle North Sea group, which is Oligocene in age, is characterised by sands, silts and clays, with the main sand formations distributed along the southern margin of the North Sea basin (TNO-GDN, 2023d; Van Adrichem Boogaert and Kouwe, 1993-1997). The uppermost layer is the Upper North Sea Group, which is Neogene – Recent in age and is characterised by a sequence of clays and fine-grained to coarse-grained sands with gravel, peat and brown coal seams (TNO-GDN, 2023g). Note that the latter three groups are considered to be one in VELMOD-1 (Van Dalfsen et al., 2006) and is characterised by clays, silts, fine- to coarse-grained sands and sandstones. In both Lauwerier (2022) and Almalki (2023), the North Sea group is subdivided into only two groups, namely the Lower and Upper North Sea Group.

Box 5.3: Availability of salt domes

Salt domes in the Netherlands are already used in a variety of ways. The Zuidwending dome, for example, is used for gas storage, while Marssteden is used for storing gasoil. Until 2030, only 1 to 4 salt caverns are expected be needed for the storage of hydrogen as part of the energy transition from fossil fuels to renewables, according to the report of van Gessel et al. (2021), commissioned by the Ministry of Economic Affairs. This can be accommodated by the salt domes that are currently already being used for dissolution salt mining or storage. Between 2030 and 2050, the number of salt caverns and domes that might need to be developed for storage purposes will depend on whether hydrogen storage becomes a preferred option for providing flexibility in both energy and heating. If it does, more than 200 salt caverns might eventually be needed. To accommodate all these salt caverns, exploitation of multiple onshore salt domes would be required (van Gessel et al., 2021). As more salt domes are allocated for other uses, fewer will remain available for radioactive waste disposal, making it possible that no suitable salt domes will eventually be available for a GDF, especially since siting is still decades away (Ministerie van Infrastructuur en Milieu, 2016). To avert this, one could adopt the German approach, i.e., reserving at least some salt domes for radioactive waste disposal. Before repurposing one of these reserved salt domes, extensive research should be conducted to assess its suitability for disposal. If a salt dome is deemed unsuitable for a GDF, it could then be utilized for hydrogen storage or other applications. This approach would ensure that the Netherlands can continue using salt domes for storage while preserving sufficient options for a GDF.



Concrete sealing element in the Morsleben repostiory. Source Jeroen Bartol, COVRA.

6 The engineered barrier system

Summary:

- There are four engineered barriers: the vitrified waste form (which dissolves only slowly), the HLW package (which provides total containment for 1,000 years), concrete seals (providing containment for 50,000 years) and moisturised granular salt backfill (for which full containment is achieved 1,000 years after closure and continues thereafter).
- For all types of HLW, instant mobilisation of radionuclides is assumed in the COPERA (2020 - 2025) safety assessment 1,000 years after the closure of the repository, i.e. slow corrosion/dissolution of the waste form is not assumed
- For all LILW and depleted uranium, instant mobilisation is assumed directly after the closure of the repository so that the solidified waste form itself is conservatively not treated as a safety barrier.

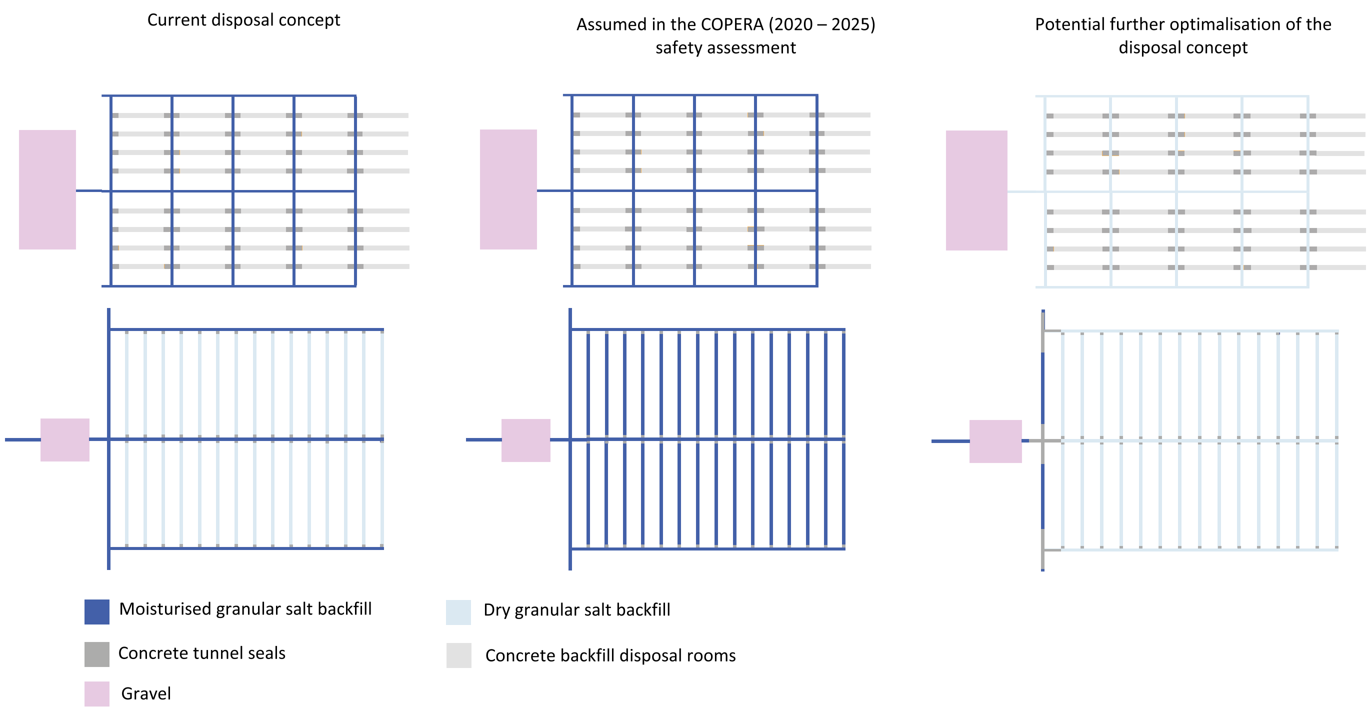
Containment in the COPERA (2020 - 2025) disposal concept is provided by a combination of the natural and engineered barriers (Fig. 3.5) in the multibarrier system. The natural barrier is the host rock: rock salt (see Chapter 5) and the engineered barrier system described in this Chapter comprises the following components:

- the LILW waste containers;
- the HLW canisters;
- the steel containers holding the HLW canisters (referred to here as the HLW packages);
- concrete backfill in the LILW disposal rooms of the upper level;
- dry granular salt backfill emplaced in the HLW disposal tunnels in the lower level;
- moisturised granular salt backfill emplaced in the transport, ventilation and service tunnels in both levels, and in parts of the shafts;
- concrete seals at the ends of disposal rooms and tunnels;
- concrete and/or composite seals in the three shafts of the GDF;
- gravel backfill used in the infrastructure area.

a)


Upper level -- LILW - (TE)NORM disposal
Lower level - HLW Disposal
Konrad containers
200L drums
1,000L drums
4
2
4
2
4
2
3
1
2

b)


Current disposal concept
Assumed in the COPERA (2020 – 2025) safety assessment
Potential further optimalisation of the disposal concept
Moisturised granular salt backfill
Dry granular salt backfill
Concrete tunnel seals
Concrete backfill disposal rooms
Gravel

c)

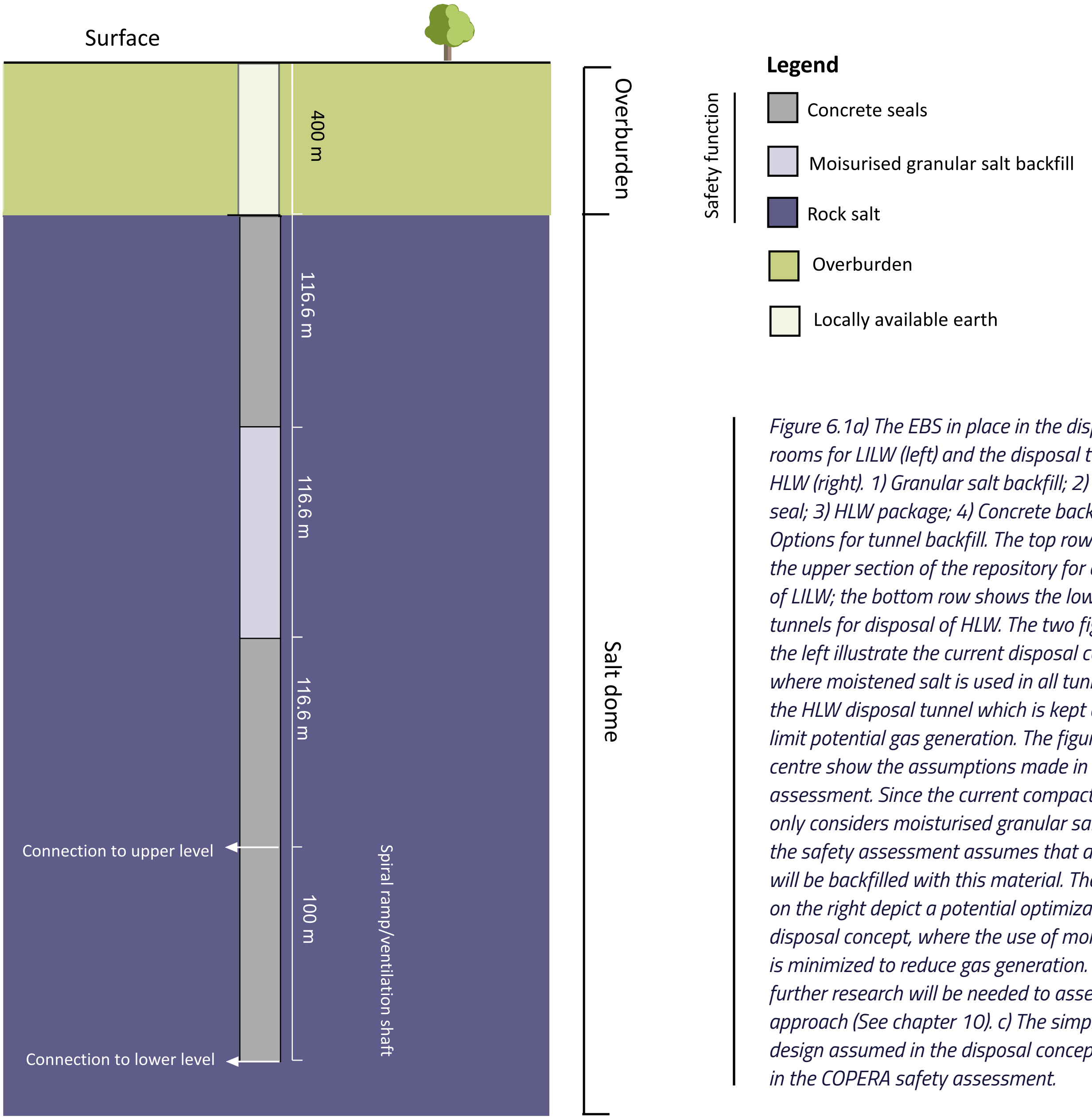


Figure 6.1a) The EBS in place in the disposal rooms for LILW (left) and the disposal tunnels for HLW (right). 1) Granular salt backfill; 2) Concrete seal; 3) HLW package; 4) Concrete backfill. b) Options for tunnel backfill. The top row shows the upper section of the repository for disposal of LILW; the bottom row shows the lower level tunnels for disposal of HLW. The two figures on the left illustrate the current disposal concept, where moistened salt is used in all tunnels except the HLW disposal tunnel which is kept dryer to limit potential gas generation. The figures in the centre show the assumptions made in the safety assessment. Since the current compaction model only considers moisturised granular salt backfill, the safety assessment assumes that all tunnels will be backfilled with this material. The figures on the right depict a potential optimization of the disposal concept, where the use of moistened salt is minimized to reduce gas generation. However, further research will be needed to assess this approach (See chapter 10). c) The simplified shaft design assumed in the disposal concept and used in the COPERA safety assessment.

Undisturbed rock salt is impermeable and should thus, on its own, provide complete containment. Construction of the repository, however, perturbs the salt by excavating shafts and other open spaces. To ensure the stability of the GDF and sealing of these open spaces, a combination of backfills and seals (listed above) is used (Fig. 6.1a and b). Together, these provide both the required short and long term containment. Short term containment for hundreds to thousands of years is provided by the HLW package and the concrete seals in the shafts, lower and upper level. For long-term containment, extending out to a million and more years, granular salt is used; this initially has a relatively high porosity and permeability but, compacts with time, and its properties become comparable to the undisturbed host rock. It will be emplaced dry in the HLW tunnels and, to speed up the compaction rate, with additional moisturisation elsewhere in the GDF, including the shafts (Fig. 6.1b; left figure and Fig. 6.1c).

In the unlikely case that brine inflow to the GDF occurs, the engineered barriers contribute to the containment of the radio-nuclides by restricting the movement of contaminated brine and, in the case of some waste forms, allowing only very slow dissolution and release of radionuclides from the wastes. In this Chapter, we describe the various materials, their behaviour, safety functions and the evolution of these different components of the engineered barrier system.

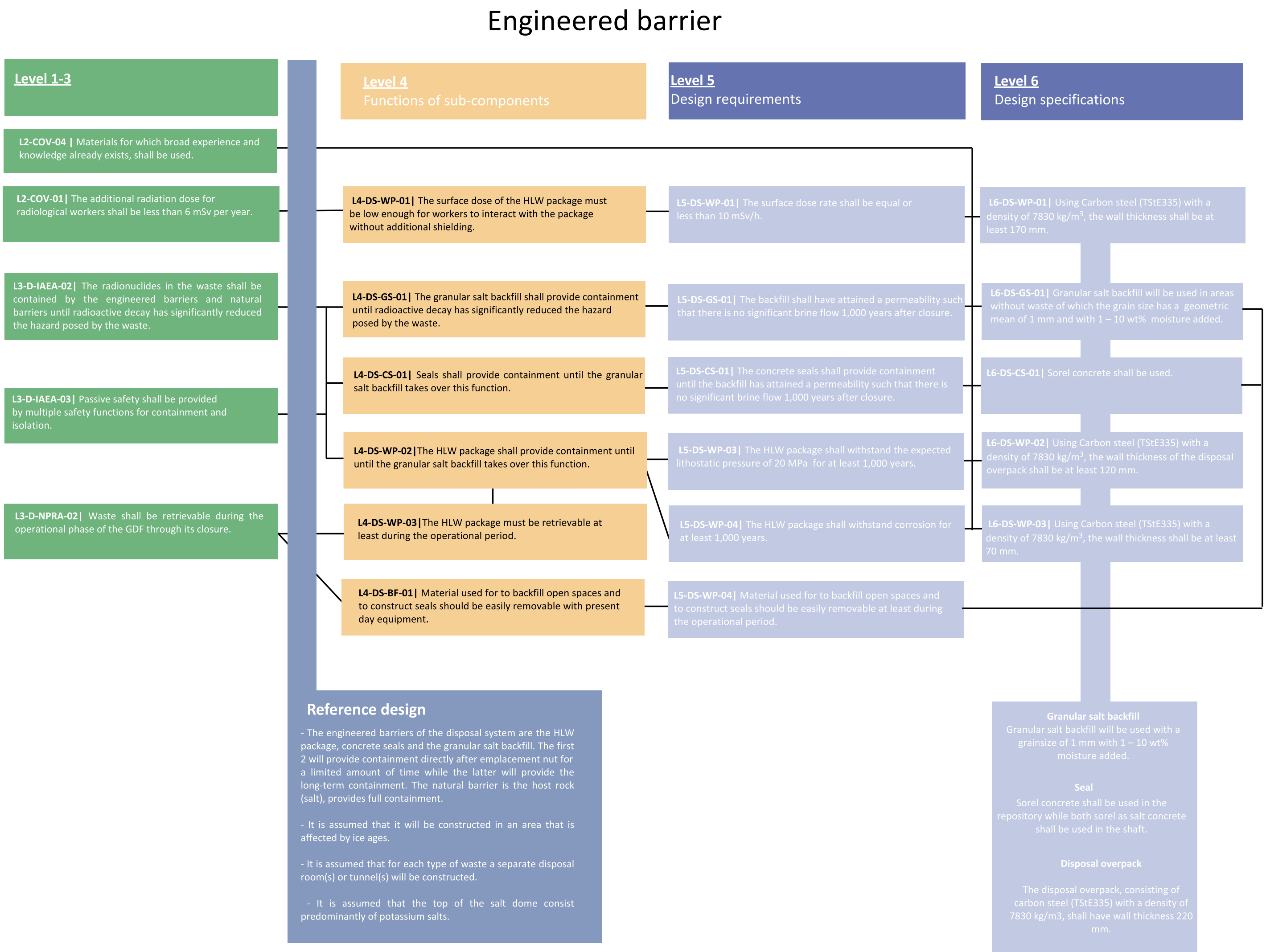


Figure 6.2) RMS for the engineered barrier. DS stands for Disposal in Salt; WP stands for Waste Package; GS stands for the Granular Salt barrier, CS for the Concrete Seal; BF for BackFill.

6.1 Backfill and seals: design, behaviour and safety functions

6.1.1 Granular salt backfill for shafts, tunnels and other openings

The granular salt backfill is a key component of the safety concept for the salt GDF. In the shafts and connecting tunnels, it ensures the necessary long-term containment of all the wastes in the GDF by preventing water flow into the disposal rooms and tunnels (Fig. 6.1b and c). As it compacts, it develops a very low permeability, as specified in requirements L3-D-IAEA-02 and L3-D-IAEA-03 in the COVRA's RMS (Fig. 6.2). Based on L3-D-IAEA-02, the direct design requirement for the granular salt backfill is that: The granular salt backfill shall have attained a permeability such that there is no significant brine flow 1,000 years after closure (L5-DS-GS-01). The requirement for 1,000 years results from the necessity for achieving sufficiently low permeability within the minimum containment period that the HLW package should provide (section 6.2.1). As explained in the sections below, the design specification for granular salt backfill that can meet this requirement is that the geometric mean of the grainsize is 1 mm and the moisture content is 1-10 wt% (L6-DS-GS-01). A further high-level requirement that affects the use of a granular salt backfill is L3-NPRA-02 which specifies that wastes must be retrievable during the operational phase of the repository; this is relatively easily achieved with a granular salt backfill. Other high-level requirements are L2-COV-04 and L2-COV-03, which specify that materials for which broad experience and knowledge already exists shall be used and that simple, robust and proven techniques shall be used. Granular salt backfill has been extensively investigated (e.g., Friedenberg et al., 2022b; Oosterhout et al., 2022; Spiers et al., 1988) and is currently considered for use in the German geological disposal programme (Bollingerfehr et al., 2018a).

As explained in section 5.1.3, salt flows or creeps in response to stress gradients in and around the repository openings, eventually leading to their closure. Three successive stages of granular salt backfill compaction can be recognised (Fig. 6.3). In stage 1, the host rock converges (creeps) to fill the open spaces between the backfill and the host rock that result from both the settling of the backfill over time and the inability to fill an open space entirely with backfill. During this phase, when the backfill has not yet started to compact, microcracking may occur within the host rock (EDZ, section 5.1.3).

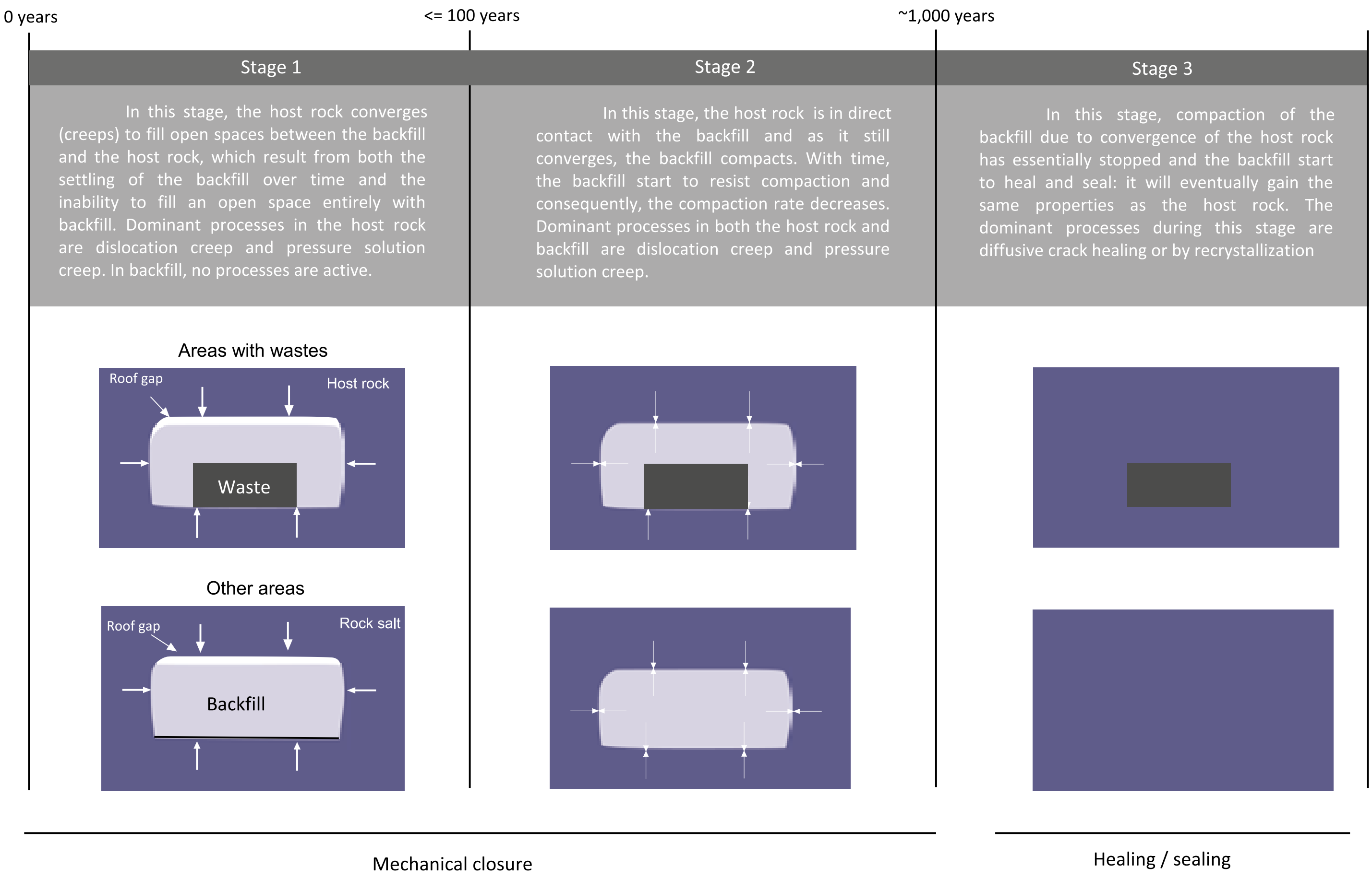


Figure 6.3) The three different Stages of compaction and healing/sealing affecting the HLW disposal tunnels, and the dominant processes involved. During the first stage, convergence of the host rock closes the crown-space gap. In the second stage, the granular backfill compacts. Stages 1 and 2 together are referred to as mechanical closure. In Stage 3, compaction due to convergence of the host rock has ceased and the backfill starts to heal and seal.

This first stage can best be described by two processes that are concurrently active in the host rock, namely dislocation creep and pressure solution creep (van Oosterhout et al., 2022). The backfill has not yet begun to compact as a result of convergence of the tunnels.

Dislocation creep is relatively well understood since the conditions (temperature and deviatoric stresses) under which this mechanism is dominant can be reproduced relatively easily in the laboratory. In dislocation creep, the mechanism is the movement of defects in the crystal lattice, which are always present in natural rock salt. As a deviatoric stress is applied to the salt (shear stress), bonds between atoms will stretch and eventually break, but subsequently recombine, allowing the defects to effectively move on. This process is generally described by a power law which depends on the temperature among other parameters (Oosterhout et al., 2022). During pressure solution, on the other hand (Spiers et al., 1990; Urai et al., 1986), salt dissolves at grain boundaries where there is a high stress and subsequently migrates in a liquid phase to places with low stresses and precipitates there. This mechanism depends on the availability of brine, the temperature and the grainsize. Based on the average creep rate of the host rock in Morsleben (1- 2 mm/year) and Gorleben (up to 7 mm/year) and assuming a crown-space gap of 10 cm above the backfill, this stage is expected to take less than 100 years (Bracke and Fischer-Appelt, 2013; Buchholz et al., 2020; Fischer-Appelt et al., 2013).

After some decades, in Stage 2 (Fig. 6.3), the granular salt backfill starts to compact due to the convergence of the host rock which is now in direct contact. The rate at which the backfill compacts depends on many factors including its intrinsic properties such as its grainsizes, the temperature and its moisture content, but also the rate of convergence of the surrounding host rock coupled with the resistance of the backfill. As the backfill compacts, it will slow down the convergence of the host rock, as the increase in stress in the backfill will cause a decrease in stress difference between the surrounding host rock and backfill. As in the first stage, the dominant processes in the host rock are dislocation creep and pressure solution creep. Also in the backfill, both processes can be active. Experimental work in combination with an analytical solution shows that for a backfill with a grain size equal to or less than 1.0 mm, the porosity of the backfill will decrease in less than 1,000 years from 40% to 1%. For a grainsize of 3 mm, this decrease would take over 8,000 years. Therefore, to ensure fast compaction, a grainsize of 1 mm or less is recommended (Oosterhout, 2023). This size also has the advantage that only one of the two mechanisms is active: pressure solution, which will make the compaction of the backfill more predictable. As pressure solution depends on the availability of brine, it is proposed to add 0.5 – 10 wt% moisture to increase the compaction rate of the granular salt backfill significantly (Oosterhout et al., 2022; Spiers et al., 1988) and this is incorporated into our present conceptual design. At the end of Stage 2, the expected permeability will be between 10^{-17} – 10^{-18} m^2 (Oosterhout, 2023; Oosterhout et al., 2022). Both Stages 1 and 2 are referred to as mechanical (creep) closure.

It is, however, unlikely that the backfill will consist of a single grainsize. Experiments as part of the COPERA (2020 - 2025) research programme were performed to understand how the distribution of grainsize would influence the compaction and how to predict the strain rate (Oosterhout, 2023). These experiments showed that, for a relatively small grain size distribution, the geometric mean of the grain size within a granular salt backfill can be used to predict its compaction behaviour (Oosterhout, 2023). Hence, it is not necessary to use a backfill with a single grain size equal to or less than 1 mm; it suffices if the geometric mean of the grain size within the backfill is equal to or less than 1 mm.

In Stage 3, compaction of the backfill has essentially ceased (no further mechanical closure, Fig. 6.3). At this point, static healing/sealing of both the granular salt backfill and the EDZ (host rock) is expected to become the dominant process. Healing and sealing of pores, cracks and grain boundaries can occur by two different mechanisms: diffusive crack healing or recrystallization (Houben et al., 2013; Koelemeijer et al., 2012). In crack healing, healing occurs by reduction in surface energy, which eventually results in isolated fluid inclusions. Salt dissolves at one area and precipitates at another area, with the transport of salt occurring by diffusion through a thin layer of brine (Houben et al., 2013). In recrystallization, driven by stored energy remaining from earlier plastic deformation, grain boundaries migrate and overgrow brine filled cracks.
This will result in isolated, spherical fluid inclusions within the newly formed grain. Study of Stage 3 processes is still part of an ongoing investigation (Oosterhout et al., 2022) but it is expected that with time, these processes will decrease the permeability of the granular salt backfill further. However, it is still unclear (Houben et al., 2013; Koelemeijer et al., 2012) how long healing would take; this is currently being studied and the process is expected to be complete within a few thousand years. At the end of this stage, both the backfill and the EDZ will have porosity and permeability equivalent to that of undisturbed salt. It should be noted that it is not clear when Stage 2 ends and Stage 3 starts, but until a porosity of at least 1% is reached, Stage 2 is expected to be the dominant phase (Oosterhout et al., 2022).

In conclusion, to fulfil the requirements for long term containment, a granular salt will be used, with a grainsize that has a geometric mean of 1 mm. To speed up the compaction, 0.5 – 10 wt% moisture will be added to this granular salt backfill when it is used to fill transport tunnels and shafts. Together, experiments and models show that moisturised granular salt will compact sufficiently within 1,000 years to provide the necessary containment. To limit the amount of brine available for corrosion, dry granular salt (without added moisture) is used to backfill the HLW disposal tunnels in the lower level (Fig. 6.1b; Light blue in the current disposal concept).

While the dry salt backfill will still compact, it will be at a significant slower rate than the moisturised backfill (Spiers et al., 1988) so that its containment function comes into play later - but this is likely to have little impact on safety, since the wastes in the lower level are enclosed in engineered barriers that last for long times.

6.1.2 Safety Case Assumptions on salt compaction behaviour

For the COPERA (2020 - 2025) safety assessment, we consider only Stage 2 of compaction. The first stage is not modelled, as it is expected to last for only a few decades up to 100 years, so that the initial state of the disposal system assumed in the safety case is reached almost immediately. For Stage 2, both dislocation creep and diffusion/humidity creep are considered in the safety assessment (Oosterhout et al., 2022). The third stage is not modelled at present as it is still part of ongoing research (Verhoef et al., 2021). Thus, in the COPERA (2020 – 2025) safety assessment, the assumption is that compaction will stop when a residual porosity of 1% is reached, in line with the safety assessment of the Gorleben repository (Fischer-Appelt et al., 2013). This is a conservative assumption, as it is expected that the porosity will decrease further due to healing and sealing (Stage 3; disconnection of pores in the salt), although how long it takes before the backfill is healed is still unclear (Houben et al., 2013; Koelemeijer et al., 2012).
For the quantitative safety assessment calculation, we assume that 1 wt% moisture will be added to the granular salt backfill, resulting in an initial saturation of about 4.5% and that all granular salt backfill, including the backfill in the HLW tunnel, will be moisturised (Fig. 6.1b; safety assessment). This is because the compaction model developed during COPERA (2020 – 2025) is specifically for moisturised salt only (Oosterhout et al., 2022).

Since it is still unclear how compaction impacts the effective diffusivity in the backfill (Flügge et al., 2016), we assume that the effective diffusivity equals the diffusion in free water multiplied by the porosity of the backfill. To calculate the permeability, we use the relationship between porosity and permeability as described by Oosterhout et al. (2022).

6.1.3 Uncertainties and further work

While the simple analytical solution proposed by Oosterhout et al. (2022) is used as a good first approximation to describe the compaction of the granular salt backfill over time, it would be beneficial to have models that simulate the interaction between the host rock and the backfill in a direct manner. This will give better insight into how the granular salt backfill will compact over time. This has a high priority, as it can influence the total time needed for the backfill to gain a low permeability. Furthermore, a better understanding of the final stage of sealing (Stage 3) of the backfill is also needed, as there are still large uncertainties (e.g., Houben et al., 2013; Koelemeijer et al., 2012). A better understanding will help to refine our requirements and optimise the other engineered barriers. Also, the minimum length of the section with granular salt backfill in the shafts needs to be quantified (Fig. 6.1c). Finally, the effect of gas generation is not yet considered in the compaction models. Gas resulting from corrosion of waste packages or from hydrolysis could delay compaction of the backfill by exerting a counterpressure. This will also be one of the focuses of the next phase of the COPERA (2020 - 2025) research programme.

6.1.4 Concrete backfill of the disposal rooms

As discussed in Section 4.3.1.1, there are several options for backfilling the disposal rooms. For now, it is assumed that they will be backfilled with a form of soft, excavatable concrete, such as salt or Sorel concrete to ensure the stability of the disposal rooms. Since both materials are also being considered for the shaft and tunnel seals, we assume, for the purposes of the safety assessment, that the concrete backfill in the disposal rooms will have the same properties and characteristics as the concrete used for the seals. This includes the change in permeability by four orders of magnitude after the expected lifetime of the concrete (Beuth et al., 2012).

In the next phase of the COPERA research programme, a review should be undertaken to recommend a material (or materials) for backfilling the disposal rooms. This is particularly important because using concrete as a backfill for LILW and (TE)NORM would require significant effort, even during the operational period. Moreover, concrete contains water, which could lead to gas generation via corrosion or radiolysis. Therefore, the use of concrete as a backfill may not be optimal, and a systematic review should be conducted to identify more suitable materials for backfilling the disposal rooms if available.

6.1.5 Concrete seals

Here we discuss the concrete seals located in the shafts above and below the salt seal, in the inclined ramp and the ends of the tunnels or rooms in which the wastes are emplaced. These components of the engineered barrier system are designed to provide containment during the period when the granular salt backfill in the shafts and the repository still has a relative high permeability (Fig. 6.2 and Fig. 3.5 in Chapter 3). As formulated in the RMS, the function of the concrete seals during the post closure phase is to provide containment until the backfill takes over this function (L4-DS-CS-01, Fig. 6.2). This stems directly from the level 3 requirement L3-D-IAEA-02. The design requirement for the concrete seals is that: The concrete seals shall provide containment until the backfill has attained a permeability such that there is no significant brine flow for 1,000 years after closure (Fig. 6.2, L5-DS-CS-01). In reality – and in line with the German safety concept (Bollingerfehr et al., 2013; Fischer-Appelt et al., 2013) - , the concrete seals are expected to last much longer, as long as the chemical environment does not change significantly. As explained below, it is judged that we can assess seal performance with confidence for up to 50,000 years, beyond which time the seals may be affected in less predictable ways by thermal, hydrological, mechanical and chemical alteration. L3-D-NPRA-02, requiring retrievability, is not highly restrictive, since concrete can be removed relatively easily during the operational phase and even for some time after closure of the repository. Likewise, L2-COV-04, requiring well known materials, is not restrictive, as there are several concrete materials that could be used for which there is broad experience and knowledge (Engelhardt et al., 2021).

6.1.5.1 Design and emplacement of tunnel seals

Within the two levels of the repository, concrete tunnel seals will be constructed at each end of a disposal room (upper level) or disposal tunnel (lower level) as soon as these are filled (Fig. 6.1b). The seals will limit potential interactions between the different waste types and provide physical separation between the waste and the open spaces (e.g., transport tunnel, ventilation tunnel, service tunnel) during the operational and the observation period. Based on practical experience and experiments in Germany (Engelhardt et al., 2021; Jantschik et al., 2018; Jantschik et al., 2016), two options are currently preferred for these tunnel seals: Mg-oxychloride compositions (sorel concrete) or salt concrete (saltcrete).
Sorel concrete is stable in Mg/K rich brines, but not in NaCl rich brine, and the opposite is true for salt concrete (Jantschik et al., 2018; Jantschik et al., 2016). This highlights the importance of understanding the chemical environment (hydrochemistry) in which concrete seals are emplaced. When properly installed, the seals will remain stable until there are changes in the chemical environment (from Mg/K rich brines to NaCl rich brine), which can lead to degradation of the concrete and an increase in permeability.
This could occur, for example, during an ice age which in turn could potentially lead to the degradation of concrete seals—particularly those in direct contact with the overburden formations.

Before a tunnel seal is constructed, the EDZ will be removed by careful recutting of the host rock to remove damage that may have developed within the host rock, up to a distance of a few metres, during the operational phase. Any remaining fractures will be sealed by concrete injections. This will reduce the permeability of the EDZ around the tunnel seal. During the operational phase, the tunnel seals have an additional function, to provide physical separation between the waste and the workers. In the lower level, they have a second function, to constrain lateral movement of the backfill and thus increase the compaction rate of the granular salt in the disposal tunnels.

6.1.5.2 Shaft and ramp seals

Detailed shaft closure and seal designs have not been considered in any of the previous Dutch studies on rock salt (Hart et al., 2015a). In previous research programmes, a shaft sealing system was suggested consisting entirely of granular salt that would, with time, become impermeable. However, using granular salt alone as a sealing material is not sufficient. It would take time for the granular salt to compact and fulfil its long-term sealing requirements, and more immediate sealing is also required to prevent access of waters from overlying formations directly after closure of the GDF. Extensive research in Germany and the USA has therefore resulted in shaft closure designs that consist of different elements that, together, provide the necessary short and long-term containment. The proposed shaft closure design for the Gorleben repository in Germany consists of three short term sealing elements, one long term sealing element, abutments, and materials that can trap water or gas in their pores (Fig. 6.4a, Herold and Leonhard, 2023b and references in there). The three sealing elements, from top to bottom, consist of bentonite (Fig. 6.4a, 1. sealing element), which is a low permeability swelling clay. Furthermore, the swelling pressure of the bentonite will help close the EDZ at relatively shallow depths and low (lithostatic) pressure. The second sealing element is salt concrete, which is stable against the expected brines in this area (Fig. 6.4a, 2. sealing element). The third sealing element consists of sorel concrete, which is chemically stable against Mg-rich brines (Fig. 6.4a, 3. sealing element), as the potash salt in the surrounding rock could alter the brine composition.
The long-term sealing element consists of granular salt. Together,

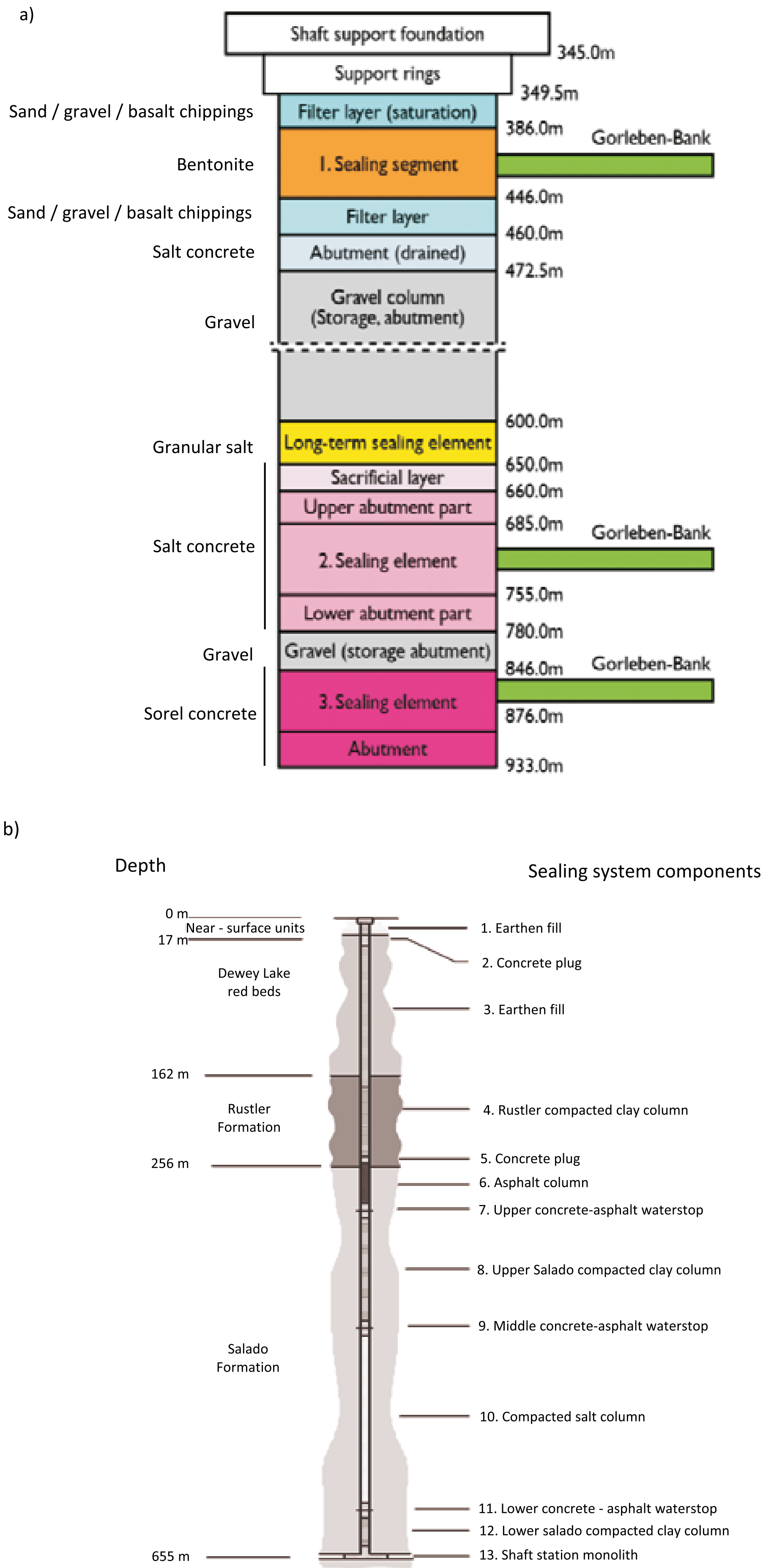


Figure 6.4a) Shaft sealing for the Gorleben (Herold and Leonhard, 2023b and references in there). b) Shaft sealing for the WIPP (Hansen, 2011).

these would delay the inflow of groundwater into the repository sufficiently long for the backfill in the tunnels (Rübel et al., 2016) to gain a sufficiently low permeability. The shaft seal design developed for Gorleben is expected to maintain its effectiveness for the next 50,000 years, extending up to the start of a next assumed ice age. After this period, it is assumed that glaciation may alter hydrogeological and topographic conditions, introducing significant uncertainty in predicting the chemical composition of incoming waters. In the long term, after the next ice age, the primary sealing role will be fulfilled by the surrounding host rock and the compacted backfill (Mauke and Herbert, 2015; Müller-Hoeppe et al., 2012). It should be noted that this seal is designed only for the part of the shaft that is located within the salt: in the overburden formations the shaft is backfilled conventionally.

Like the shaft seal system designed for the Gorleben site, the shaft seal system for WIPP (Fig. 6.4b) is designed to limit the entry of water into, and the release of contaminants out of, the repository. It also consists of multiple elements. From the bottom to the top (Fig. 6.4b), these are a monolith of a salt saturated Portland cement-based concrete, a clay column, a granular salt column and an asphalt column. This is followed by the upper Salado clay column, concrete and asphalt. The clay columns will limit any water movement. In addition, clay promotes the healing of fractures in the surrounding rock by swelling and removing the excavation damaged zone as a potential pathway. The moisturised granular salt column will attain its low permeability due to compaction within 1,000 years (see section 6.1). The asphalt column is highly impermeable, durable, resistant against most acids, salt and alkalis. Because of its viscoelastic properties, cracks are not likely to form within the asphalt. The uppermost part of the shaft, through overburden sediments, is filled with compacted clay (Rustler formation) and with locally available earth.

For the COPERA (2020 - 2025) safety case, we propose a simplified design for the shafts and for the spiral ramp (Fig. 4.2 in Chapter 4) connecting the upper with the lower level (Fig. 6.1c) in order to provide the required short and long-term containment. The eventual shaft closure and seal design will depend on the local geology (e.g., presence of anhydrite layers). The simplified shaft seal in the rock salt consists of, from top to bottom, a concrete (either sorel or salt concrete) shaft seal, moisturised granular salt backfill, and a further concrete seal. As for the tunnel seals, likewise, two options are considered for the type of concrete for construction of the seals: sorel or salt concrete. The uppermost part of the shaft above the salt dome is filled with locally available earth. Although this simplified shaft closure design can capture the most important aspects (short and long-term containment) of the shaft evolution, it needs to be further developed. For example, it does not account for any gaps that can form between the concrete seals and the salt during the first few years, as the latter settles. This could, in turn, affect the performance and stability of the concrete seal. In the German concept, the salt and the concrete seal are separated by abutments of salt or sorel concrete (See figure 6.4a) which ensure the stability of the concrete seals.

6.1.5.3 Assumptions of seals characteristics in the Safety Case

The key characteristic of the seals determining their safety performance is their permeability to liquids. In practice, gas permeability is usually measured rather than liquid permeability as measurement of the latter is more practicable. In the COPERA (2020–2025) safety assessment, it is assumed that all tunnel, spiral ramp, and shaft seals are constructed from sorel concrete (Fig. 6.1b; dark grey). This material has an initial gas permeability of approximately $4.5 \cdot 10^{-18}$ m^2, based on laboratory and in situ-experiments (Jantschik et al., 2018 and references therein). Salt concrete, as demonstrated by an in situ sealing element of a former German salt mine at a depth of 945 m, has a significantly lower gas permeability of $5 \cdot 10^{-20}$ m^2 and is thus effectively impermeable to gas. Therefore, the sorel concrete assumed for the seal in our safety assessment results in conservative estimates of performance. If improved performance is required, this could be achieved by using salt concrete, although it will also depend on the hydro chemical environment in which the seals will be constructed. The permeability is assumed to remain unchanged during the expected lifetime of the seals, through the next ice age, after which the seals are assumed to have failed. This failure will be simulated in the safety assessment by increasing the assumed permeability by 4 orders of magnitude, following the German approach (Beuth et al., 2012).

For effective diffusivity, the diffusion in free water is multiplied with the porosity of 0.13 determined from MgO cement with silica sand aggregate (Záleská et al., 2019), which gives an effective diffusion coefficient of $2.3 \cdot 10^{-10}$ m^2/s. This is likely to be a conservative assumption as laboratory experiments indicate that the actual effective diffusivity of Portland Type I and Type II cement, which is expected to be in the same range as sorel concrete (Jantschik et al., 2018), is around $1 \cdot 10^{-14}$ m^2/s.

6.1.5.4 Uncertainties and future work

During this first phase of the COPERA (2020 - 2025) research programme, the focus was on the design of a HLW package (see section 6.2.1) and the HLW tunnel backfill (see section 6.1.1). Therefore, the designs for the seals and the shaft closure system used in this disposal concept are simplified. In future, the suitability of concrete, and more specifically the type of concrete, for use as seals in the salt formations in the Netherlands will need to be evaluated and a review should be done in the next phase of the COPERA (2025 - 2030) research programme, focusing on the optimum tunnel/spiral ramp and shaft sealing material for a Dutch salt GDF. This review should take into account the variety of salt structures observed in the Netherlands, using data from the examination of borehole cores (Hunfeld et al., 2023) and should be based on the current disposal concept. Selection of seal materials should be accompanied in the next phase of the COPERA research programme by the design of a generic shaft sealing system applicable to a range of Dutch salt structures. Furthermore, in the current disposal concept, each HLW disposal tunnel is planned to be closed with a tunnel seal at each end. To reduce the use of moisturized salt and, consequently, minimise potential gas generation and associated costs, a study is needed to evaluate whether isolating the entire waste disposal area from the shaft area by three large concrete tunnel seals with moisturized granular salt backfill in between—while the rest of the repository uses dry granular salt (see Fig. 6.1b; right figure)—can provide the same level of safety while minimising gas generation and advective flow within the repository (less brine). This study should not only look at the lower level but also at the upper level for which a similar design with three large tunnels seals could also be used, although tunnels seals might not be needed as only LILW and (TE)NORM waste is disposed here.

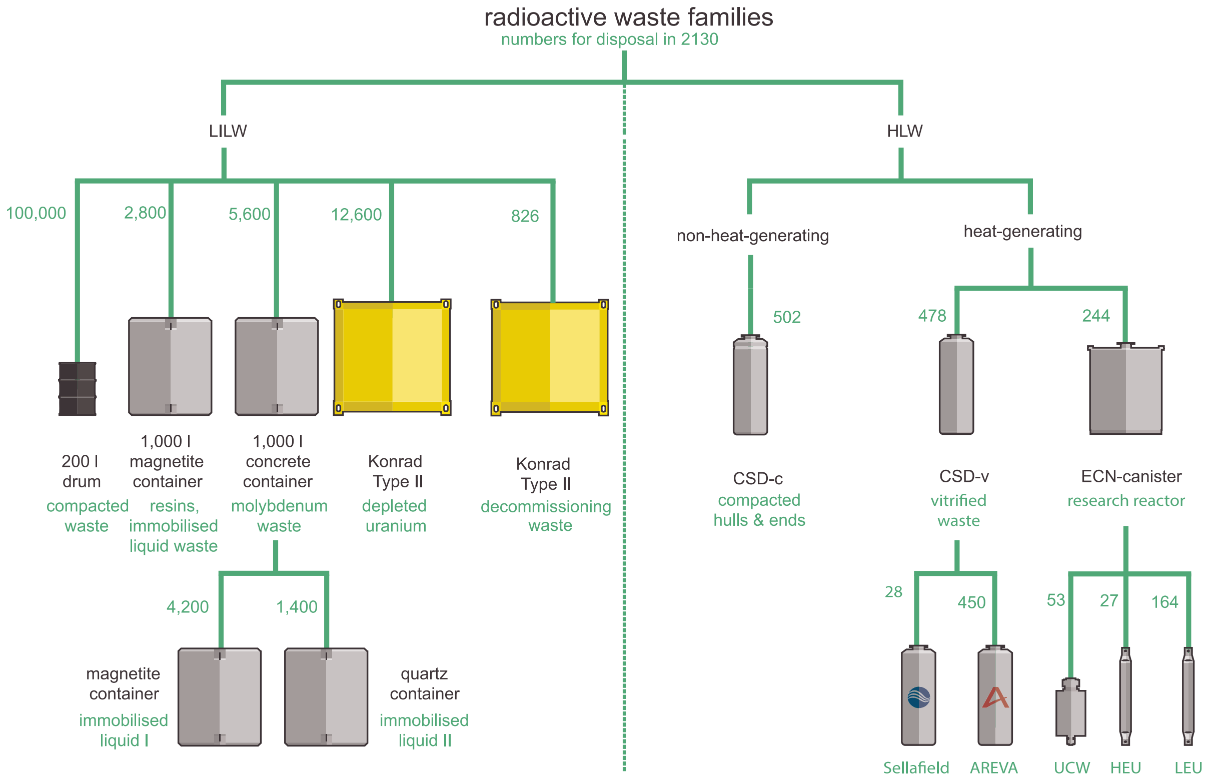


Figure 6.5) Waste families in the Dutch inventory considered in here and their relevant containers and canisters in which the waste is stored.

6.2 The waste packages

One of the focusses of the previous OPERA research programme (Verhoef et al., 2017), was on documentation of the compete inventory of the wastes that will be emplaced in the GDF.
This resulted in a comprehensive description of the different types and volumes of wastes expected for disposal. This section is based on the work of Verhoef et al. (2017) and updated using data from Burggraaff et al. (2022). The different types and volumes of wastes assumed in the present report are identical to those used in the parallel clay study (Neeft et al., 2024b). The updated inventory data for COPERA (2020 – 2025) programme is described in Appendix 4. Some examples of important changes are as follows: decommissioning waste is a newly defined waste category of LILW; other non-heat generating HLW has been separated out as a type; the expected number of waste packages for disposal has been up-dated. Figure 6.5 shows the different types of containers/canisters of waste that will be stored prior to packaging for disposal. When repackaged for disposal, only 4 types remain, these are:

- Waste package for HLW which holds both CSD-C, CSD-V and ECN (see section 6.2.1);
- Konrad Type II containers for depleted uranium (see section 6.2.4);
- 1,000L concrete or magnetite containers for LILW (see section 6.2.5);
- 200L drums for LLW (see section 6.2.5);

6.2.1 The HLW package

In the safety concept for the salt GDF, the carbon steel of the HLW package provides relatively short-term containment, over the period of highest hazard potential and heat output of the wastes (Fig. 6.1), until the surrounding granulated salt backfill has sufficiently compacted, taking over the principal containment function in the EBS (Fig. 6.2, L4-DS- WP-02). To fulfil its containment requirement, the HLW package should withstand the expected lithostatic pressure of 20 MPa for at least 1,000 years (L5-DS-WP-03); this is the expected lithostatic pressure (including a safety margin) at 850 m depth. Furthermore, to ensure its containment function, the HLW package should also withstand penetration by corrosion for at least 1,000 years (L5-DS-WP-04). The 1,000 years follows from the expected time the backfill needs to attain a permeability such that there is no significant brine flow. Ensuring radiation safety during the handling of the HLW packages (L4-DS-WP-01) places further requirements and design limits on the package: the surface dose of the HLW waste package shall be equal or less than 10 mSv/h (L5-DS-WP-01). This value is based on the current regulations for the transport of radioactive waste since it is still too early to determine the maximum permissible surface dose rate when taking the expected exposure time of workers during disposal into account. The retrievability requirements on the system and the waste package (L3-NPRA-02, L4-DS-WP-03) during the operational period are satisfied since the HLW waste package must provide full containment until the granular salt backfill has become impermeable (L4-DS- WP-02) which is a much longer time than the operational period. Note that, L3-D-IAEA-04 (Fig. 4.1) - In the case of heat-generating waste: the engineered

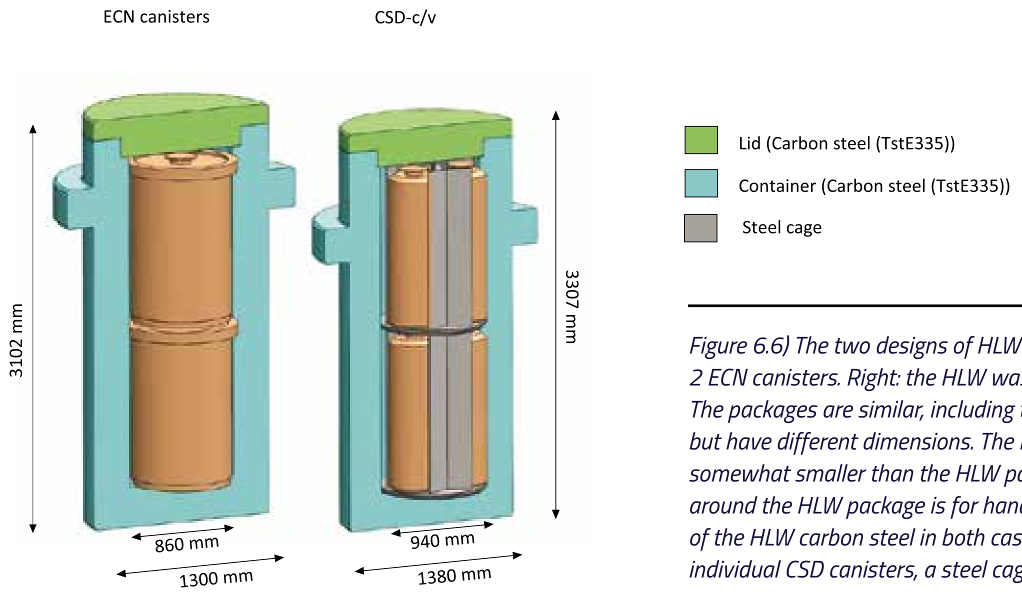


Figure 6.6) The two designs of HLW package. Left: the HLW package for 2 ECN canisters. Right: the HLW waste package for 6 CSD-c/v canisters. The packages are similar, including the material used, carbon steel (TstE335), but have different dimensions. The HLW package for the ENC canister is somewhat smaller than the HLW package for the CSD canisters. The flange around the HLW package is for handling and emplacement. The thickness of the HLW carbon steel in both cases is 220 mm. To secure and separate individual CSD canisters, a steel cage will be used.

containment shall retain its integrity until the produced heat will no longer adversely affect the performance of the disposal system - is not a requirement for a HLW waste package in rock salt. As explained in section 5.1.6, rock salt has a high thermal conductivity. Together with the limited amount of heat that is generated by the Dutch heat generating HLW, this means that the expected temperature increase in the host rock will be limited and will not adversely affect the performance of the disposal system.

A potential waste package which could satisfy the above requirements is the OPERA supercontainer (Verhoef et al., 2017). From the inside to the outside this supercontainer consists of a carbon steel overpack, a concrete buffer and a stainless-steel envelope around the waste containers. However, the specific type of concrete used in the buffer of the supercontainer was designed for a GDF in clay, with specific functions designed to meet the requirements of the safety concept in clay. The design and materials are thus not optimised for the rock salt GDF concept and a supercontainer design is therefore not assumed for this safety case.

As part of the COPERA (2020 – 2025) research programme, Wunderlich et al. (2023) developed a new HLW package that is optimised for disposal in salt. This HLW package is a metal container holding the various types of HLW canisters. It is manufactured from carbon steel (TstE335), which provides a good combination of strength and ductility, and has predictable corrosion properties (no pitting or granular corrosion). In addition, there is extensive experience with fabricating and sealing these types of containers and carbon steel is relatively abundant and low in cost (L2-COV-04). Furthermore, repositories in salt formations are relatively dry and gas generation due to corrosion of iron is expected to be very limited (Wunderlich et al., 2023). The use of an alternative, low corrosion, more expensive stainless steel is not necessary to meet the containment requirement time of the waste package.

Two waste packages have been developed: one for the disposal of CSD-c and CSD-v canisters, and one for the ECN-canisters that contain SRRF (Fig. 6.6). The difference between the two package designs is only in their geometries, which are designed to minimise the cost and potential gas generation (less metal implies less potential for gas generation); all other properties of the waste packages will be the same. For the CSD canisters, each waste package contains two layers of three CSD canisters, separated and secured by a steel cage. For the ECN-canisters, only two are placed in each waste package (Fig. 6.6).

The HLW containers will be manufactured by forging and will be temporarily stored until needed. After waste is placed in them, an airtight lid (Fig. 6.6) is welded on to ensure that brine does not enter. Welding can be done by electron beam or by submerged arc. It is uncertain whether attachment points such as bolted or screwed trunnions would be reliable in the long-term for retrieval operations. Therefore, a lifting flange is proposed to ensure that the disposal package is retrievable over long time periods.

For the carbon steel (TStE335, L2-COV-04), a thickness of 170 mm would be needed to stay within the contact dose limit of 10 mSv/h (L6-DS-WP-01). The HLW package only needs a thickness of 120 mm to withstand the expected lithostatic pressures (L6-DS-WP-02) and a thickness of 70 mm (L6-DS-WP-03) is needed to account for corrosion during the first 1,000 years, assuming the maximum possible corrosion rate (Wunderlich et al., 2023). Hence, to ensure containment during the first 1,000 years, the thickness of the stainless steel should be at least 190 mm to retain sufficient thickness to sustain the load, even after an assumed maximum amount of corrosion. Together with an additional safety margin, the wall thickness has been set at 220 mm.

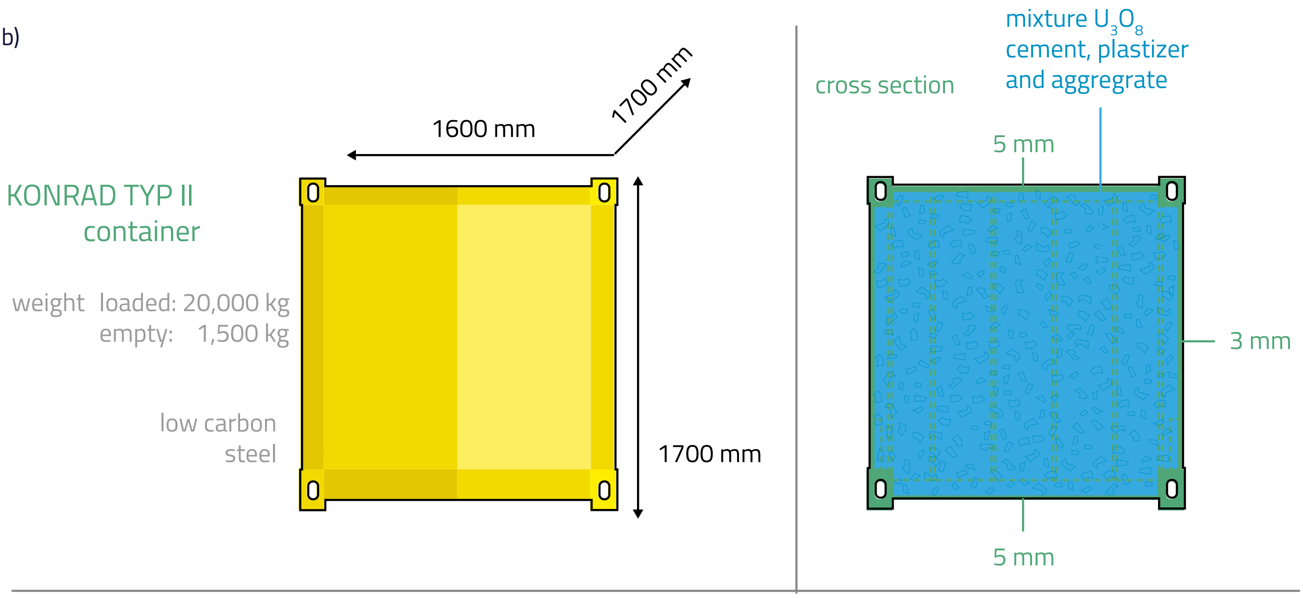
In the previous disposal concepts of the METRO project (Grupa and Houkema, 2000) and of TORAD-B (Poley, 1999), the HLW canister was either placed directly in the salt host rock with no additional overpack (METRO) or with only a relative thin overpack (TORAD-B). In both cases, temporary shielding during emplacement had to be provided by the emplacement equipment. However, a self-shielded HLW package will reduce the time needed to emplace the HLW. Furthermore, it provides passive safety during the operational period.

In summary, a HLW package made from TStE335 with a thickness of 220 mm adheres to all the requirements set in the RMS.

a)

Figure 6.7a) The DV-70 container. b)The Konrad type 2 container for the disposal of (TE)NORM. Figure from Verhoef et al. (2016).

b)



6.2.2 Safety Case Assumptions for HLW package behaviour

For the COPERA (2020 - 2025) safety assessment, it is assumed that the HLW package will fail 1,000 years after closure of the repository. This is a very conservative assumption, based on the maximum possible corrosion rate and the presence of an unlimited amount of brine to achieve this degree of corrosion. Most likely, the HLW package will remain intact for a significantly longer period, because the amount of brine available for corrosion is expected to be very small (See section 5.1.1).

6.2.3 Uncertainties and future work

Failure of the HLW package at 1,000 years is a very pessimistic assumption that can be refined by having more realistic corrosion models which, for example, consider the availability of brine. Having more accurate corrosion models can also help to optimise the HLW package, although the thickness is mostly determined by the required strength (120 mm) and the potential to optimise is thus limited. Furthermore, a more detailed mechanical analysis should be done of the lid and the base of the package. This has no high priority and it is recommended to be done later in the design processes (Wunderlich et al., 2023).

6.2.4 The Konrad Type II Container for depleted uranium

Depleted Uranium (DU) is different from other waste types because the activity of the waste at the time of disposal remains virtually unchanged thereafter, due to the extremely long half-lives of U-235 and U-238. Presently, the DU is stored at COVRA in the form of granules in DV-70 containers, without any conditioning. The DV-70 is rectangular shaped with outer dimensions, including the lifting points, of about 1.7 by 1.4 by 1.8 m (Fig. 6.7a). The walls of the container are 5 mm thick steel and the container has an empty weight of 750 kg and an allowable load of 12 metric tonnes (COGEMA, 2002). The container was designed for long term surface storage. In OPERA, it was assumed that for disposal, the depleted uranium will be conditioned and contained in Konrad type II containers (Fig. 6.7b, Verhoef et al., 2016), although it was not studied whether

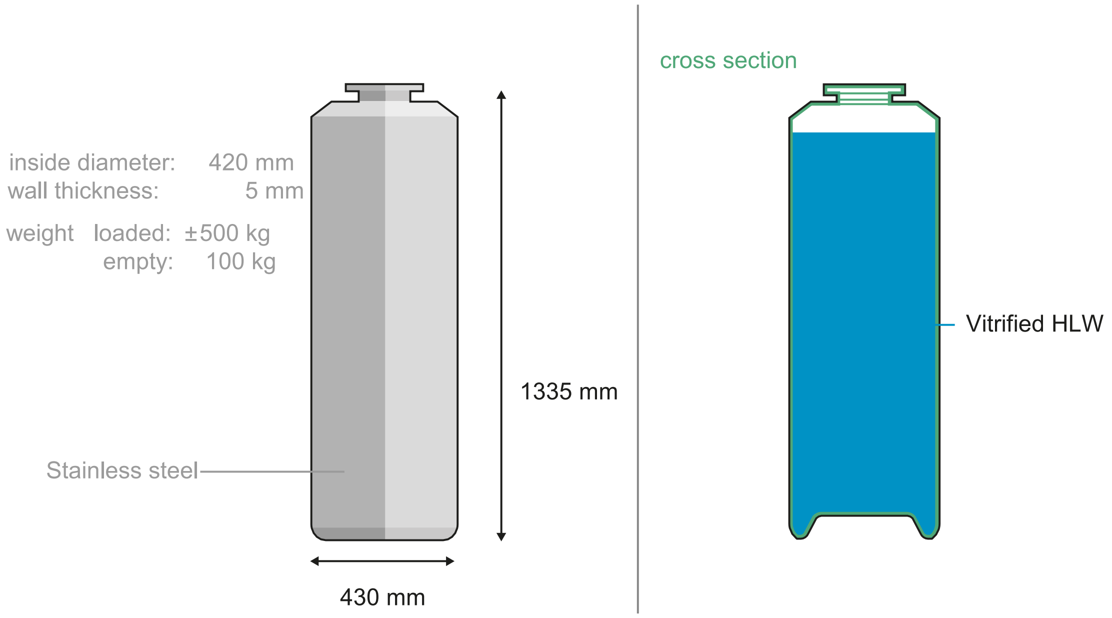


Figure 6.8) COGEMA: CSD-v (Colis Standard de Déchetsvitrifié) canister for vitrified HLW. Figure from Verhoef et al. (2016).

the use of different types of container would be more efficient. Compared to the DV-70, this container is somewhat bigger (1.6 by 1.7 and 1.7m) and therefore has a somewhat larger available volume (4.2 m^3).

Based on the available volume of a container, the total cost and the number of disposal rooms needed, the Konrad type II container has indeed been found to be the optimal container for the disposal of DU assuming granules will be conditioned. For conditioning, the use of CEM I/42.5 N HS LA (LH), with a low water/cement ratio, was judged to be most suitable. This type of concrete will have a cylindrical compressive strength of more than 20 MPa. This ensures that the Konrad type II containers will not fail under expected lithostatic pressures, provided that there is no unacceptable degradation of the concrete. The type of concrete proposed is also expected to decrease the solubility of some radionuclides in the DU, should they come into contact with brine (Oving and Meeussen, 2024).
The dose at the surface of the Konrad container is calculated to be 29 μSv/hr, which is below the permissible surface dose rate for contact handling.

6.2.4.1 Safety Case Assumptions on DU behaviour

Even though the Konrad type 2 container is expected to provide complete containment for at least 100 years after closure, and more likely for about 2,000 years, depending on the environment (Browning and Grupa, 2023), we assume in the safety assessment that it provides no containment. Therefore, all radionuclides of the DU are conservatively assumed to be instantaneously available for dissolution when the repository is closed.

6.2.4.2 Uncertainties and further work

For the current stage of the programme, the available data are sufficient to model the DU in the safety assessment. Conditioning of the waste can be done today, but this would result in additional radiation exposures of workers and more research will be needed to justify this additional dose. Furthermore, as already indicated in Chapter 4, instead of disposing of the DU as waste, it could potentially be used as a component of the backfill for some disposal rooms and tunnels. This would decrease the cost of the repository. However, more research is needed on this issue.

6.2.5 Containers for LILW

The 200 and 1,000 litre LILW containers depicted in Figure 6.5 will be placed directly into the disposal rooms/tunnels, as shown in Figure 6.1, and will be surrounded by the previously described concrete backfill. While the cementitious materials and steel of the containers might provide some containment, the COPERA conservative assumption is that radionuclides are released instantaneously into the concrete backfill of a disposal room directly after the GDF closure.

It is unlikely that further study of the evolution of the LILW packages would provide useful information to add to the realism of a future safety case, so this conservative approach is likely to continue to be the most appropriate.

6.3 The waste form

In this section, we discuss whether and how each waste form can contribute to delaying or limiting the release of radionuclides. In addition, for each waste form, we also discuss current uncertainties and potential future work.

6.3.1 Vitrified waste

Vitrified HLW (vHLW) contains the largest proportion of the radioactivity in the GDF at the time of closure. It results from the reprocessing of used (spent) nuclear fuel from nuclear power plants. Reprocessing of the COVRA spent nuclear fuel is carried out in France. The vHLW is manufactured in 170 litre stainless-steel canisters - COGEMA: CSD-v (Fig. 6.8). The radionuclides are incorporated in a glass matrix, which must dissolve for them to be

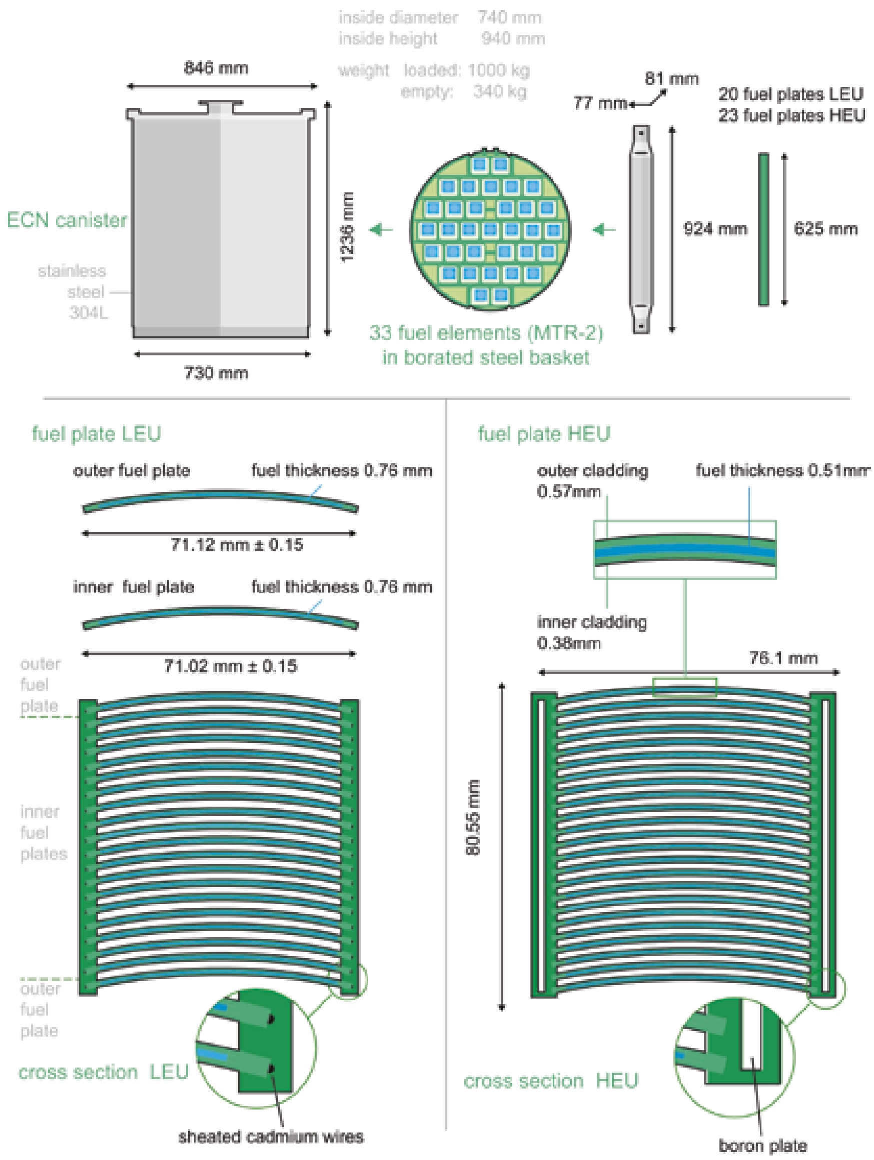


Figure 6.9) Schematic overview of spent research reactor fuel and the ECN canister that will be emplaced in the HLW package (Verhoef et al., 2016).

released. The very low dissolution rate of the glass matrix limits the release of the radionuclides. The long term dissolution rate of glass can be modelled using the empirical relation of Kienzler et al. (2012), which is based on observed behaviour in long-term laboratory studies in brine. The total glass dissolution rate depends on the temperature and the available reactive surface of the vitrified waste. At a temperature of 45 degrees Celsius (Smit, 2022), the expected dissolution rate is around $1.6 \cdot 10^{-5}$ kg/(m^{2}a). This is similar to the most likely value ($1.81 \cdot 10^{-5}$ kg/(m^{2}a)) for the dissolution rate in seawater according to Prij et al. (1993). However, the dissolution rate, as extensively discussed by Neeft et al. (2024b) for clay, will also depend on the availability of brine which is likely to limit the dissolution rate even further.

The rate that glass is dissolved depends on the specific rate of dissolution and the surface area available for dissolution (reactive surface). The surface area of the glass might increase during cooling of the vitrified waste when it is manufactured since cracks can form. To account for this, a cracking factor is used to estimate the increase in available surface area relative to the integral glass block. In the OPERA safety case, this cracking factor was assumed to be between 0 and 40 (Verhoef et al., 2017). A value between 4 and 17 was proposed by Strachan (2004) for US defence HLW glass while a value of 10 was estimated by (Kienzler et al., 2012).

6.3.1.1 Safety Case assumptions on HLW behaviour

In the COPERA (2020 – 2025) safety assessment, we do not use any of the above information on glass behaviour and instead make the highly conservative assumption that all vitrified HLW is instantaneously dissolved directly after failure of the HLW waste package. This conservative assumption also reduces the computational time needed for the safety assessment calculations. It is also assumed that the stainless-steel container surrounding the waste (COGEMA - container) will fail instantly when the HLW package loses containment. This might appear to be a conservative assumption, as it takes about 10^4 years to fully corrode the stain-less steel COGEMA container (Benbow et al., 2023a).
However, after the failure of the carbon steel of the waste package, it is unlikely that stainless-steel canisters would be able to with-

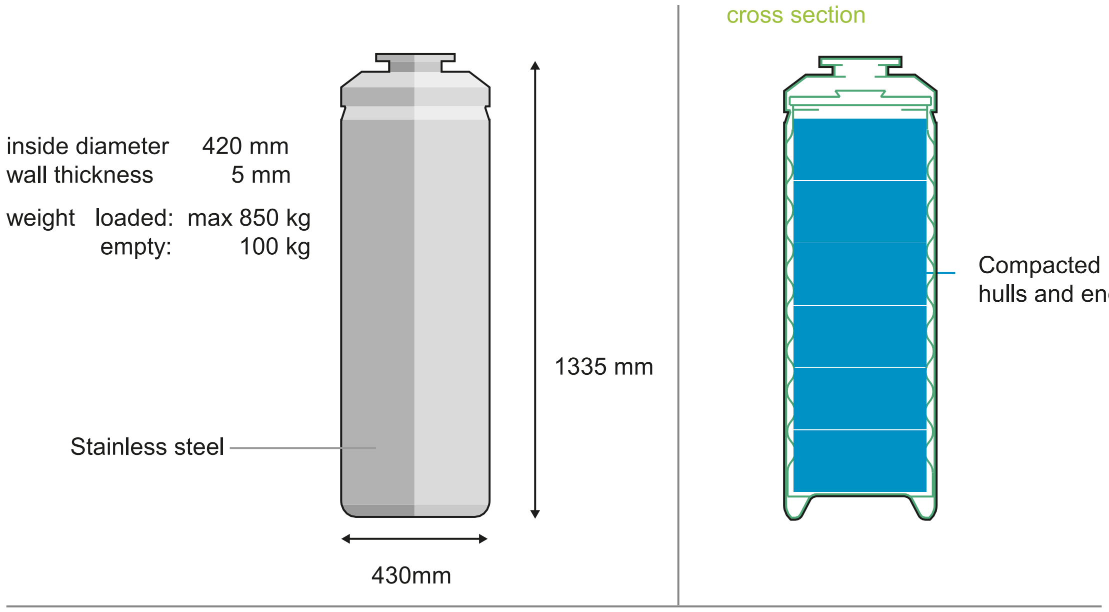


Figure 6.10) Schematic of CSD-c canister with 6 pucks (compacted drums). Figure from Verhoef et al. (2016).

stand the expected lithostatic pressure and they would probably fail mechanically shortly afterwards. The assumption of instantaneous dissolution also implies that there is a sufficient volume of brine available, which is highly improbable.

6.3.1.2 Uncertainties and further work

The fundamental processes of glass degradation in the small volumes of brine that may be present in the progressively compacting backfill material, and in the presence of iron corrosion products, have not yet been addressed. It is expected that the backfill will have attained a low permeability limiting advective transport before the HLW package fails. The evolution of the backfill-waste package-glass interfaces needs to be considered further, but the current COPERA assumptions are already highly conservative with respect to feasible release mechanisms and rates from the glass.

6.3.2 Spent nuclear fuel

In the Netherlands, spent nuclear fuel (SNF) originates from two research reactors: the Petten High Flux Reactor (Petten) and the TU Delft Reactor (Delft). Until 1996, SNF was repatriated to the USA. This changed in 1996, when the Netherlands began to store the spent fuel at COVRA. The fuel comprises both Highly Enriched Uranium (HEU: 93% 235U) and Low Enriched Uranium (LEU: 19.75% 235U). The fuel is uranium-aluminide (UAlx for HEU) or uranium-silicide (U3Si2 for LEU) in the form of particles, ranging from 40 to 150 µm in size, which are dispersed within an aluminium matrix. These particles are bonded to aluminium cladding (Deissmann et al., 2016). The fuel is in plates, with a thickness of 0.51 mm for HEU and 0.76 mm for LEU, as illustrated in Figure 6.9 (Verhoef et al., 2016). For storage, the SNF is placed in an ECN canister. This ECN canister is a 305 litre stainless steel container with a thickness of 10 mm. During storage the storage building, with its thick walls, provides the necessary shielding.

6.3.2.1 Safety Case assumptions on SNF behaviour

Owing to the rapid corrosion rate of the aluminium metal and fuel matrix, an assumption is made of instant release of all radionuclides upon failure of the HLW package (Benbow et al., 2023a). This is conservative, but less so than in the case of the vitrified HLW.

6.3.2.2 Uncertainties and further work

For the current stage of the programme, assuming instant releases from spent nuclear fuel is sufficient, especially as it is expected that the backfill will have attained a low permeability and advective transport has essentially ceased before the HLW package fails. Having more detailed models on the corrosion of spent nuclear fuel will contribute little to the overall safety assessment. Future work will, however, need to consider the potential for criticality to occur in the SRRF packages. This topic is discussed more fully in the report on the parallel COPERA safety case for a GDF in clay.

6.3.3 Non-heat generating HLW: technological waste, compacted hulls and ends

Compacted waste Standard Residues (Collis Standard de Déchets Compactés: CSD-c) arise from reprocessing spent fuel from nuclear power plants. They comprise metal parts from the spent fuel assemblies that have been cut up to extract the spent fuel.
A canister of about 170 litres internal volume is filled with either hulls or end pieces. The hulls are made of Zircaloy, while other metal parts are usually made of Inconel. End pieces are solid stainless-steel sections. Drums with other wastes arising from reprocessing fuels, such as pumps, stirrers and filters, are primarily made of stainless-steel. These drums are compacted to produce pucks, essentially compacted discs of radioactive material waste, that are loaded into CSD-c canisters with the same outer dimensions as those used for vitrified waste and then welded closed (see Figure 6.10). There is about 20% void space in the canisters.

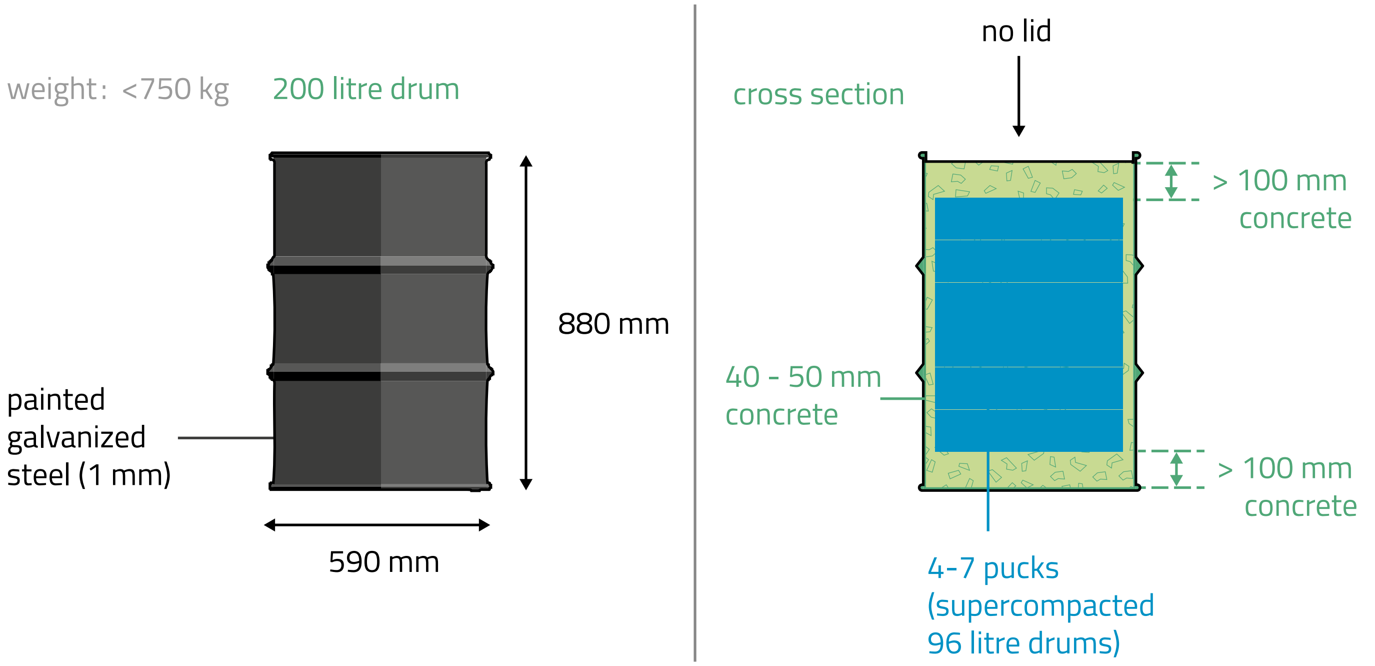


Figure 6.11) Schematic of a 200L drum with super-compacted pucks of LILW. Figure from Verhoef et al. (2016).

6.3.3.1 Safety Case assumptions on CSD-c behaviour

For the current stage of the programme, as for the other categories of HLW, assuming instant release of CSD-c is sufficient, especially as it is expected that the backfill has obtained a low permeability before the HLW package fails. Having more detailed models on the corrosion of the CSD-c will contribute little to the overall safety assessment at this stage.

6.3.4 Low and Intermediate Level (LILW) waste forms

In total, there are 4 categories of LILW: depleted uranium (DU), compacted waste, processed liquid molybdenum waste, processed liquid waste containing spent ion exchangers and decom-missioning waste (Fig. 6.5). In terms of volume, DU is the largest category of LILW, and its disposal has been discussed separately in section 6.2.4.

The second largest waste category in terms of volume is compacted waste. This waste originates from around 200 organisations, including nuclear power plants, research institutions, industry and hospitals. It consists predominantly of contaminated substances such as organic cellulose-based materials (cloth, paper and tissue), sludge, metals (steel and aluminium), halogenated and non-halogenated plastics, glass, concrete, inorganic adsorption materials, salts et cetera. To handle this waste, the drums containing the waste materials undergo a compaction process, to produce pucks, which are subsequently encased in concrete within 200L drums, as shown in Figure 6.11.

The production of medical isotopes gives rise to the third largest volume of waste within this category, comprising processed liquid molybdenum waste which results from the production of ZrO^2 through the irradiation of uranium targets. To manage this waste, the highly alkaline waste mixture is combined with a cementitious mortar within a 200L drum. These drums are then placed into a larger, 1,000L concrete container. This container is selected because it provides shielding against the high levels of radiation during both storage and disposal stages. The concrete container can be made with magnetite aggregate, instead of silica (Fig. 6.12).

The fourth largest category is processed liquid waste containing spent ion exchangers. These ion exchangers are resins used to clean water during the operational phase of a nuclear plant. The conditioning of this waste involves a process in which the liquid waste (referred to as sludge) is combined with a cementitious mortar within a 200 L drum. Subsequently, these drums are placed in a larger 1,000 L concrete container (Fig. 6.12). This container is used to provide shielding throughout the stages of storage and disposal. For all the various types of Low- and Intermediate-Level Waste (LILW) described above, the conditioning matrix utilised is a cementitious substance, made from blast furnace slag cement.

6.3.4.1 Safety Case assumptions on LILW behaviour

Since the waste packages and waste forms for LILW are not allocated a safety function, it is assumed that all radionuclides are released instantaneously from the packages at the time of closure of the repository.

Assuming instant release of the radionuclides in LILW is sufficient for the current stage of the programme, especially as the expected containment period is only a few decades at most. Hence, more detailed models on the corrosion are likely to contribute little to the overall safety assessment.

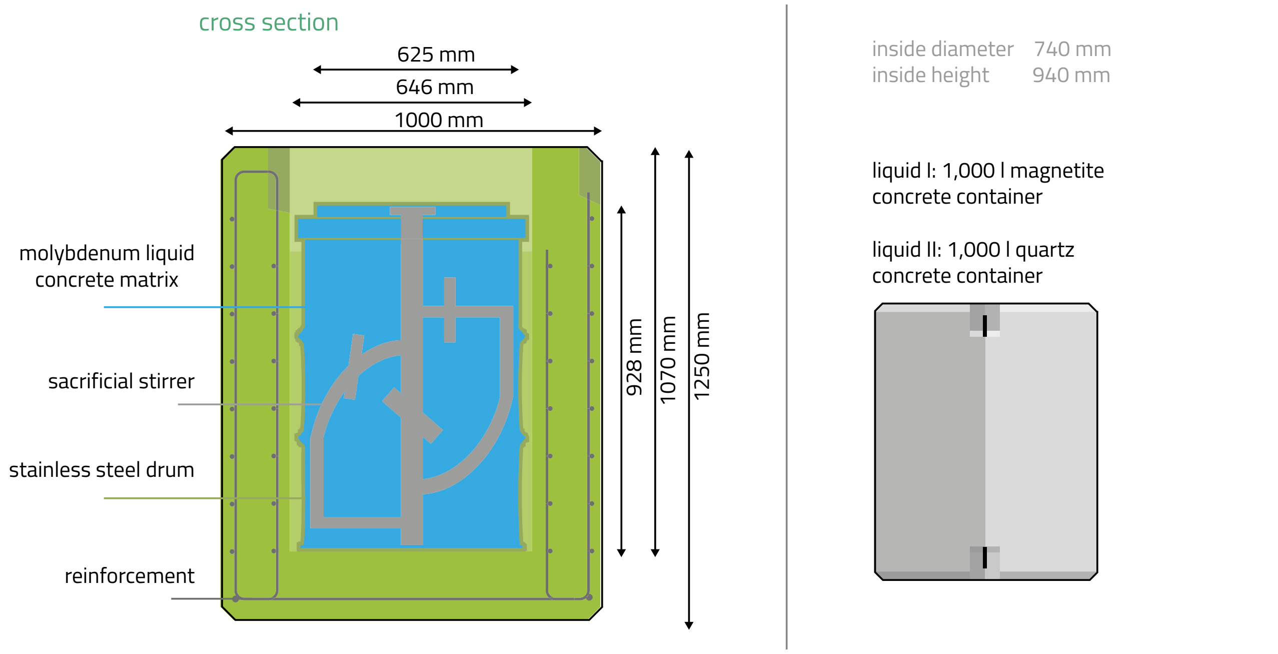
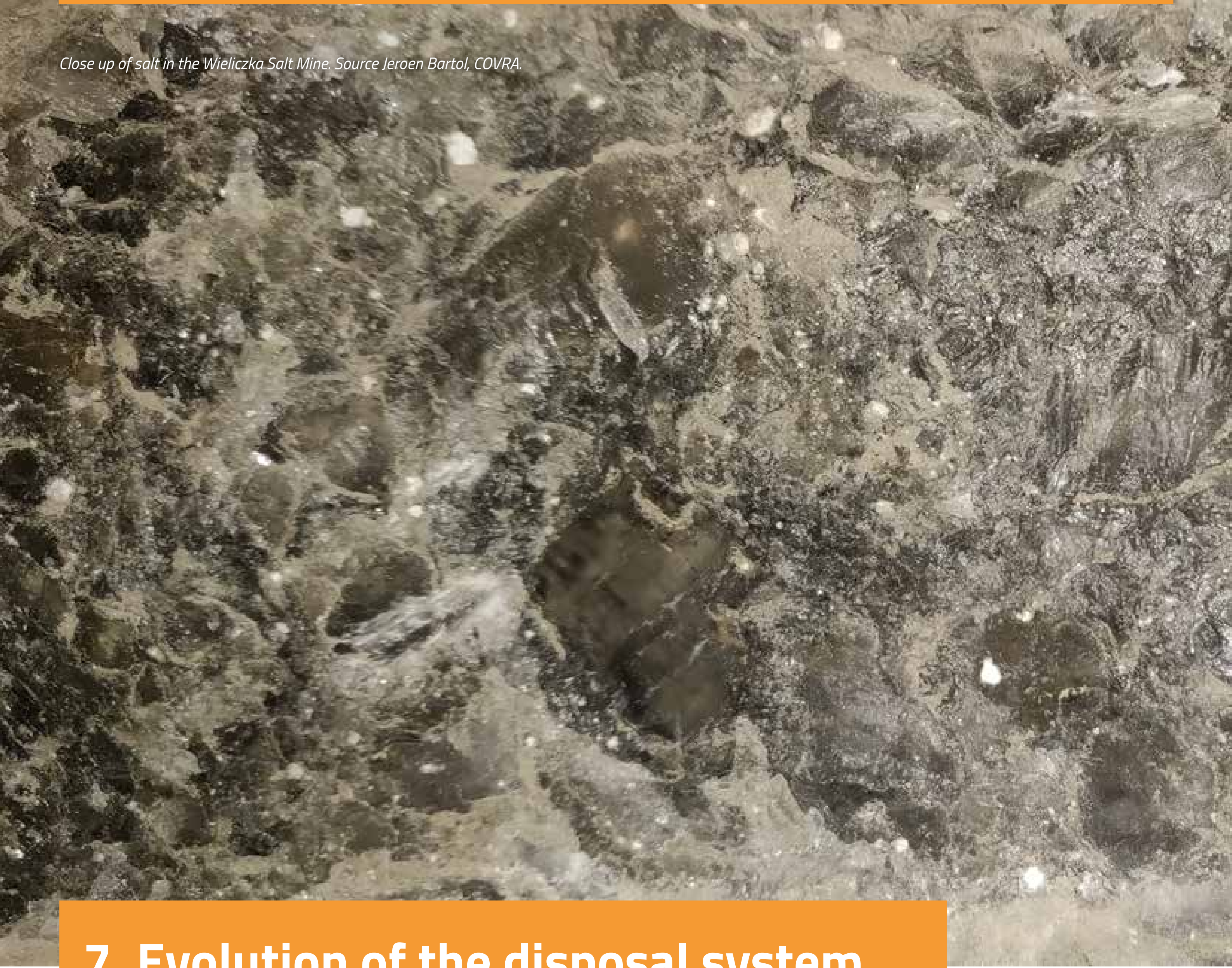


Figure 6.12) Schematic of 1,000L container holding a 200L stainless steel drum of molybdenum wastes. The same configuration is used for disposal of ion-exchanger wastes. Figure from Verhoef et al. (2016).

Box 6-1: Gas Generation

As part of the COPERA (2020 – 2025) research programme, a scoping study was undertaken to estimate the potential gas generation within a repository in rock salt (Benbow et al., 2023a; Benbow et al., 2023b; Watson, 2023). Knowing how much gas is generated within a repository is important, as gas pressure can delay, or even halt, the compaction of the granular salt backfill. The study considered three main gas generation mechanisms: corrosion, microbial breakdown of organic substances and radiolysis. The last of these can be important in waste with high beta/gamma activity. To model gas-generating reactions, two sources of brine were assumed. These are the initial saturation of the moisturised granular salt backfill and a steady continuous geosphere inflow based on work of Kuhlman et al. (2024a). Compaction of the granular salt backfill was not included in the model and thus the backfill remains permeable throughout the modelled period. Furthermore, the model does not account for the reduction in alpha radiolysis that might result from the waste form restricting direct access of brine to the radionuclide inventory: the model will likely overestimate the gas production via radiolysis.

Model results suggest that gas generation depends primarily on the availability of brine, which is likely to be very limited in a repository in rock salt (See section 5.1.2). Furthermore, HLW will be emplaced in a HLW package that is likely to last significantly longer (tens of thousands of years) than the 1,000 years conservatively assumed in this safety assessment, in which case the amount of gas produced by radiolysis will be less than that calculated (Benbow et al., 2023a; Benbow et al., 2023b; Watson, 2023). Furthermore, if dry granular salt backfill is used in the disposal tunnels (as currently assumed in the repository design – but not in the safety assessment), this will furth limit any gas production (Fig. 6.1b current disposal tunnel and Fig. 6.2). Although limiting the availability of brine reduces gas generation significantly, some gas is likely to be generated within the repository, because there will be some brine available in, for example, the granular salt backfill. The use of gas-permeable seals and the inclusion of engineered void spaces in which gas can accumulate (e.g., the gravel backfilled infrastructure areas), will help to minimise the gas pressure within the repository. The study of Benbow et al. (2023b) was scoping in nature and the results are preliminary, so that the next step will be to expand the model and include the compaction of the granular salt backfill.



Close up of salt in the Wieliczka Salt Mine. Source Jeroen Bartol, COVRA.

7. Evolution of the disposal system

Summary:

- During the first 1,000 year after closure of the repository, slow advective transport of fluids resulting from compaction of the backfill, is the dominant mode by which transport of radionuclides could occur within the repository.
- After about 1,000 years, diffusion becomes the dominant mode by which transport of radionuclides could occur within the repository, as compaction has essentially ended and the backfill has gained low permeability.
- Over longer times, the backfill starts to heal and pores within the backfill become disconnected so that the permeability becomes even lower: radionuclides become essentially trapped within the repository.
- Neither subrosion nor diapirism are expected to affect the repository; both its depth and the thickness of the salt surrounding the waste continue to provide sufficient isolation and containment.
- Although limited movement of radionuclides can occur, it is expected that no release occurs within a million years after closure of the repository.

Our understanding of the properties and behaviour of the natural and engineered barriers underlies the concept of isolation and containment provided by geological disposal. The safety assessment in Chapter 8 quantifies this behaviour to predict the performance of each component of the system and of the whole multibarrier system over a long period of time.

The data available on the performance of the different barriers contain varying degrees of uncertainties. The COPERA (2020 – 2025) safety assessment takes this uncertainty into account by making conservative simplifications, assuming limited barrier performance, using pessimistic parameter values and excluding potentially beneficial processes that lack sufficient quantification. As a result, this COPERA safety assessment yields conservative results. However, for the purposes of systems engineering and cost optimisation, it is important to make the best possible estimations regarding the system's behaviour, taking existing uncertainties into account. A balanced perspective is required, between realistic, expected behaviour, and demonstrating the system's safety by using a significant level of built-in conservatism. Striking this balance is important for subsequently making well-informed decisions on how

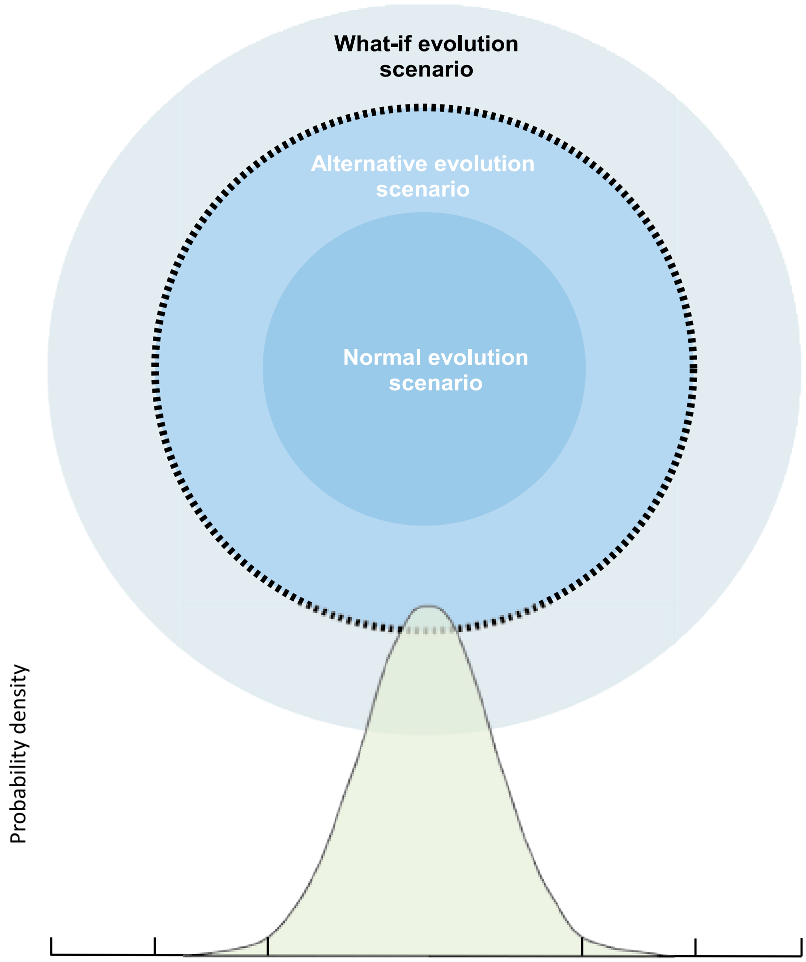


Figure 7.1) Schematic of the different types of scenarios and their relative probability.

to optimise the design of the GDF and to determine acceptable site characteristics. By following this approach, over-engineering of system components can be avoided, and more potentially suitable GDF sites can be considered. In this chapter, we thus assemble information from the previous chapters to develop a narrative on the expected 'best estimate' evolution of the GDF system.

7.1 Different scenarios

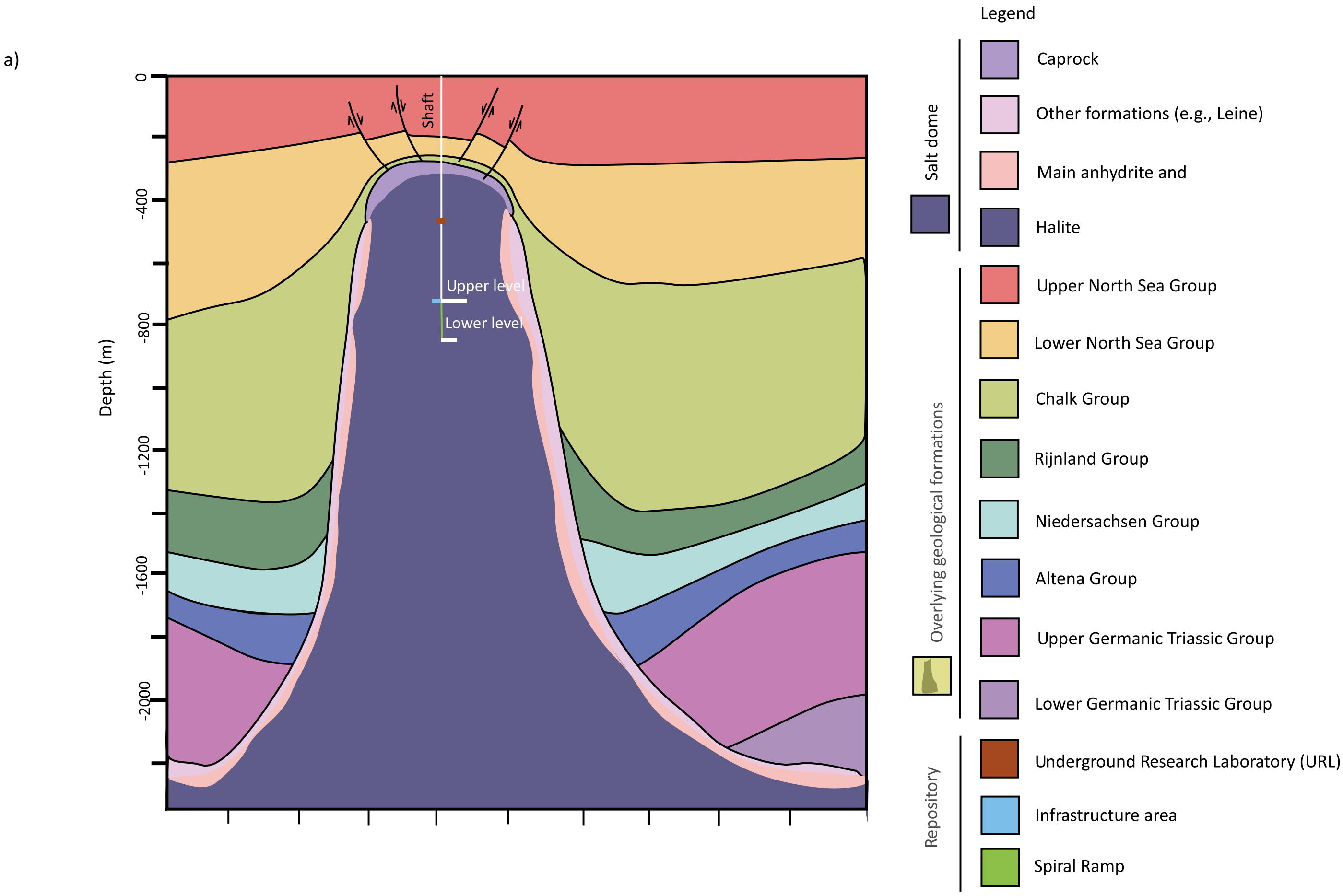
As part of the COPRA (2020 - 2025) research programme, Lommerzheim (2023) developed 4 different types of scenarios for a GDF in rock salt, namely the normal evolution scenario, alternative evolution scenarios, what if scenarios, and inadvertent human intrusion scenarios.

The normal evolution scenario refers to the expected evolution of the GDF (Fig. 7.1). In this scenario, it is assumed that there are no undetected geological features (e.g., faults), all the barriers are assumed to work as expected and the climate evolves according to current understanding of the likely timing of future climate states.

The normal evolution scenario contains a range of alternative cases (or realisations) to encompass the expected range of variability and uncertainty in key parameters that affect system behaviour (e.g., diffusion and compaction rates). One of these cases is based on best estimate values for all the key parameters and is termed the Reference Case.

Alternative evolution scenarios (Fig. 7.1) comprise a set of cases in each of which the normal evolution scenario is changed in a specific way. This can be, for example, by postulating an undetected fluid reservoir within the salt (a brine pocket), by assuming the earlier than expected failure of the shaft or tunnel seals, or by imposing a different climate evolution. An alternative evolution scenario can also include less likely properties and characteristics of the disposal system or alternative credible models of critical processes to the models assumed in the Reference Case scenario.

The third type of scenarios are the what-if scenarios (Fig. 7.1). These are generally highly unlikely or entirely hypothetical cases for analysis, to test the robustness of the system or highlight key sensitivities and points of focus for optimising the system design.



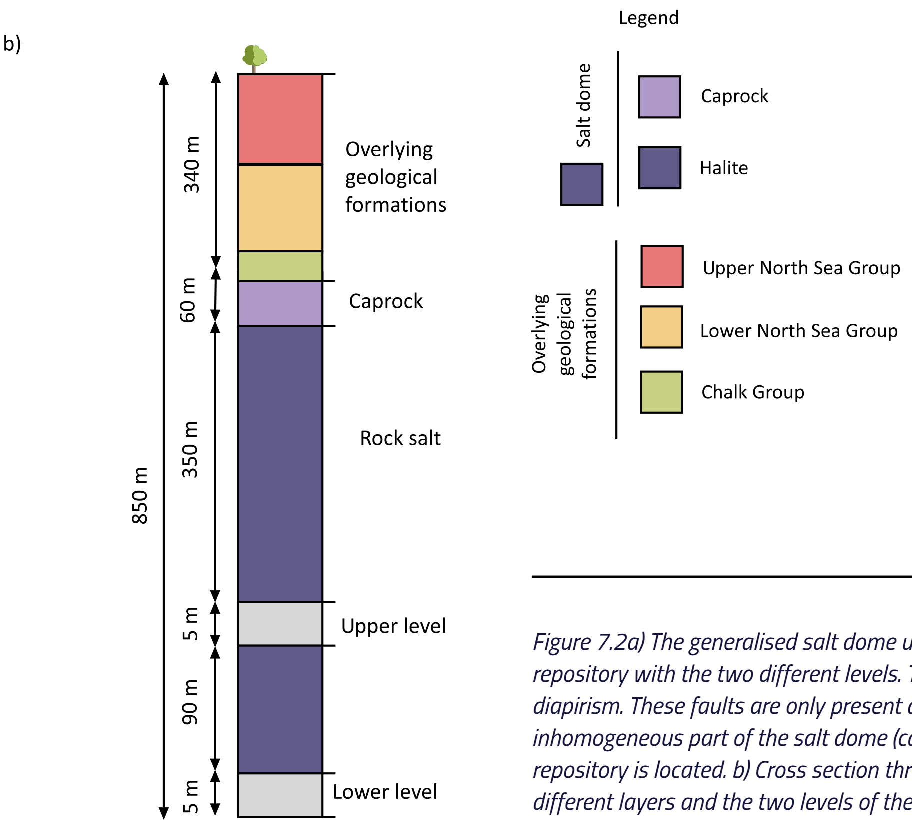


Figure 7.2a) The generalised salt dome used for the performance assessment. In white is the repository with the two different levels. The faults on the top of the salt dome are a result of diapirism. These faults are only present associated with some salt domes and only affect the inhomogeneous part of the salt dome (caprock) and not the homogenous part in which the repository is located. b) Cross section through the generic salt dome with the thickness of the different layers and the two levels of the GDF.

The fourth type of scenarios are inadvertent human intrusion scenarios. These describe consequences resulting from future human actions intruding into the repository. As the future evolution of the biosphere and future human actions are not predictable, stylised scenarios are commonly used that are derived from common recent human activities.

7.2 Normal evolution scenario for the GDF

Since no site or host rock formation has been selected for disposal, we assume here a generic salt dome, whose properties are based on current data on salt domes in the Netherlands. When more data become available, they will be used to update the generic salt dome model for future safety case exercises. The generic salt dome has a length of approximately 10 km and a width of about 5 km and is assumed to extend to considerable depth; there is thus sufficient space to host the repository. It is assumed to have a 60 m thick caprock consisting of a combination of anhydrite, gypsum, and calcite/carbonate (see section 5.2.4.1, Fig. 7.2a and b), based on the cap rock of the Gorleben salt dome in Germany, which varies between 111 and 0 m, with an average between 20 and 40 m (Köthe et al., 2007). It is overlain by the Chalk Group, the Lower and Upper North Sea Group (Paleogene, Neogene and Quaternary). These have a total thickness of 340 m (Fig. 7.2 a and b – overlying geological formations). Adjacent to the salt dome are the (Upper and Lower) Germanic Triassic group, Alterna group, Niedersachen group, Rijnland group, Chalk group and the Lower and Upper North Sea group. With the upper level of the GDF at a depth of 750 m, there is thus about 340 m between the top of the salt dome (excluding caprock) and the top of the upper level of the repository. Furthermore, there is at least 300 m between the anhydrite at the flanks of the salt dome and the repository.

The salt dome can be divided into a relatively homogenous part and an inhomogeneous part. The homogenous salt is located within the centre (Fig, 7.2, dark blue) and the inhomogeneous salt is located at the flanks of the salt dome (Fig, 7.2, pink). While homogenous, the salt in the centre of the salt domes contains secondary mineral components such as anhydrite and polyhalite (Biehl et al., 2014), along with the main halite component. The inhomogeneous flanks contain, for example, halite, potash seams, anhydrite and other types of salts. Being heterogenous, the permeability of the flank salt is higher ($<10^{-21}$ m²) than the homogenous salt ($<10^{-22}$ m²) in the centre of the salt dome. The temperature inside the salt dome at repository depth is expected to be around 34.5 °C (upper level) and 36.5 °C (lower level, Bonté et al., 2012; Smit, 2022).

As the Netherlands is located relatively far from any plate boundary, few changes to the tectonic stress field are expected over the next million years (See section 5.5). Consequently, tectonically induced seismicity is expected to be low. Because it is imperme-able, gas and oil can be trapped below a salt deposit. The extraction of these resources can result in human induced seismicity (e.g., Zöller and Hainzl, 2023). Depending on the site-specific evolution of the salt dome structures, the diapirism rates may vary between 10-3 and 10-1 mm/y. This bandwidth is based on the observed geological diapirism rates in the Netherlands and other countries although the uncertainty is high (See section 5.4.2.1). The subrosion rates is expected to vary between 10-2 and 10-1 mm/y, based on geological observations in the Netherlands and other countries (See section 5.4.2.2). Because the intensity of the next glacial period is unpredictable, the characteristics (e.g., depth of permafrost, thickness of ice sheet and depth of glacial channels) of the Elsterian glaciation are assumed here: the glacial period with the highest impacts in this respect.

The normal evolution scenario begins when all the open spaces in the GDF are backfilled, and the repository is closed. To facilitate this evolutionary narrative, we look at 4 different periods after closure of the GDF, namely:

- closure until 1,000 years after closure;
- 1,000 years after closure to the start of an assumed next glacial period
- the duration of the glacial period (assumed to last 100,000 years);
- from the end of that glacial period until 1,000,000 years.

These periods are selected as they represent major changes in the external environment or within repository. While currently considered unlikely (see section 5.4), we assume here that the next glacial period will happen in about 50,000 years and, even more improbably, that ice cover will extend over the Netherlands.
The evolution of the repository would, however, not change were the next ice age to occur later: the second period would only last longer.

7.2.1 Closure – 1,000 years after closure

Starting with the biosphere and the overlying units, it is expected that characteristics of the biosphere (climate, morphology, vegetation, animals, land use) will remain like today: a temperate warm climate, subdued topography with a sea level that might rise episodically. Some erosion might occur, by rivers for example, but this will be limited to the uppermost 50 m of the biosphere and overlying formations (Lommerzheim, 2023) and will not affect the performance of the repository (Fig. 7.3). Deeper, the top and sides of the salt dome will undergo some dissolution due to subrosion. Since no major changes are expected in the groundwater regime, the subrosion rate will be as currently observed (maximum of 0.1 mm/year) and about 0.1 m of salt will be dissolved in the first 1,000 years after closure. Similarly, since no major tectonic stress changes are expected, the depth of the repository will have decreased by about 0.1 m, due to diapirism (maximum of 0.1 mm/year). As the geometrical changes to the system are insignificant, neither process will affect the performance of the repository.
Also, a sea level rise and potential submerging of the site will not affect the evolution of the repository. Submergence under the sea could increase the salinity of the groundwater in the upper parts of the overburden sediments, to an extent that would depend on the duration of marine cover. These conditions reduce the likelihood of human intrusion.

When the repository is closed, the salt backfill in the shafts will have an initial porosity of around 40%. Likewise, the granular salt backfill within the transport, ventilation and service tunnels (Fig. 6.1b; Current disposal concept) will also have an initial porosity of around 40% (Oosterhout et al., 2022). In addition, there is a crown space of a few tens of centimetres between the ceiling and the granular salt backfill in both upper and lower level, due to the settling of the latter. The granular salt backfill in the HLW disposal tunnel of the lower level will have a somewhat lower porosity at the time of closure of the repository, as it will already have had a few decades to compact, although more slowly due to the dry granular salt used (Fig. 6.1b; Current disposal concept). In the dry

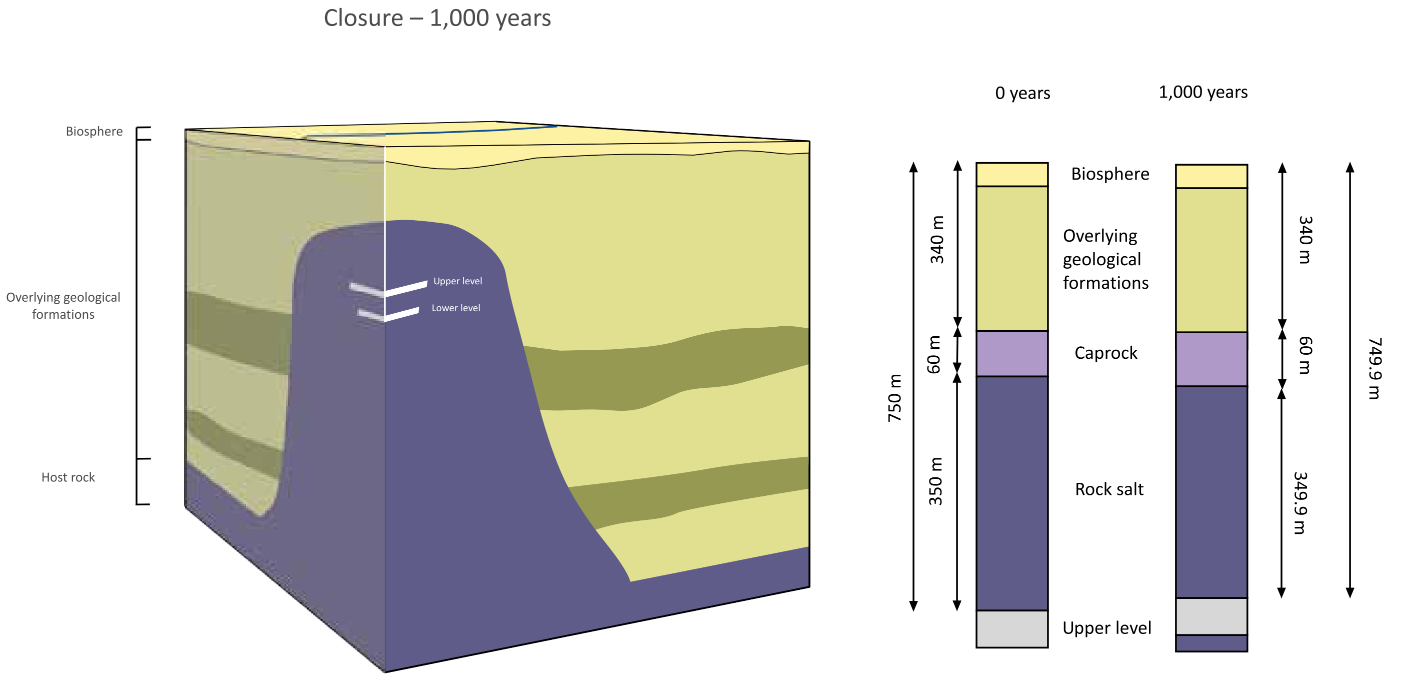


Figure 7.3) Closure of the repository to 1,000 years after closure. On the left, is the generic salt dome with the two-level repository. Initially, there is 750 m between the upper level and the surface and 350 m between the upper level and the caprock. There has been no effective change due to subrosion and diapirism. Note that in this figure we assume that the caprock does not change in thickness.

granular salt backfill of these HLW disposal tunnels the voids will predominantly contain air, in addition to some naturally occurring brine present in natural salt. Together, they result in an average saturation of around 0.1% in the dry granular backfill.
In the transport, ventilation and service tunnels and the salt backfill in the shafts, these void spaces will have a higher water-air ratio, as 1 wt% moisture will be added to the backfill to aid the compaction rate (Fig. 6.1b; see section 6.1.1). With this added moisture, the saturation is expected to be around 4.5%.

Directly after the closure of the repository, the host rock will start to converge, and the salt backfill in the shafts will start to compact. Initially, compaction is relatively fast, as the backfill does not provide any resistance (small gap due to settling, stage 1, section 6.1.1). But as convergence continues, the backfill will start to resist compaction (start of stage 2, section 6.1.1) which in turn decreases the compaction rate until it eventually stops (end of stage 2, section 6.1.1) and the healing of the backfill to form a salt seal starts (beginning of stage 3, section 6.1.1). Within about 1,000 years after closure, the granular salt backfill in the shaft will have attained a permeability 10^{-17}- 10^{-18} m^2. During the first 1,000 years after closure, groundwater will not reach the salt backfill in the shaft from above, due to the low permeability of the concrete sealing elements that are on top of it.

As in the shafts, the host rock around the disposal tunnels in the upper and lower levels will start to converge directly after the closure of the repository. Here, convergence will be aided by the heat released by the heat generating HLW and thus elevated temperatures within the repository. This effect is, however, temporary and within 1,000 years or less the temperature within the host rock returns to normal ambient temperatures (Smit, 2022). During the first 100 years after closure, the tunnel crown space is closed by convergence of the host rock (Stage 1, section 6.1.1). Only then, all the host rock will be in direct contact with the granular salt backfill so that compaction starts (start of stage 2, section 6.1.1). Initially, compaction is relatively fast, but as the convergence continues, the resistance of the backfill increases, which in turn decreases the compaction rate until it eventually stops, (end of stage 2, section 6.1.1) and the healing of the backfill starts (beginning of stage 3, section 6.1.1). Compaction of the moisturised granular salt backfill is expected to end within 1,000 years after closure (end of stage 2, section 6.1.1 and Fig. 6.1b). Likewise, the small extensional fractures that have formed in the host rock (EDZ) will heal during this period. The HLW disposal tunnels are already closed by seals during the operational period (Chapter 4) and they will follow a very similar evolution, although the backfill porosity and permeability will likely be higher at the end of this period, as dry granular salt is used here (Fig. 6.1b). The salt is still in stage 2.

Due to compaction, saturation levels also start to increase within the backfill and the in-situ brine is squeezed out of the backfilled parts of the repository and moves to other regions of the repository, including, the seals, disposal rooms and the infrastructure areas. This effect is most pronounced in those areas in which moisture was added to the backfill (Fig. 6.1b) to increase the compaction rate. The displacement of the brine is expected to be aided by differential suction pressures within the repository, which develop due to differential compaction. While some flow is thus likely to occur, it will be limited, as both the host rock, and the (both dry and moisturised) granular salt backfill contain very little brine. In addition, the permeability of the compacted granular salt will be small, inhibiting the flow of brine.

While the brine is being displaced out of the granular salt backfill, the dominant mode of transport for any radionuclides mobilised

1,000 years after closure – Start next glacial period (assumed at 50,000 years)

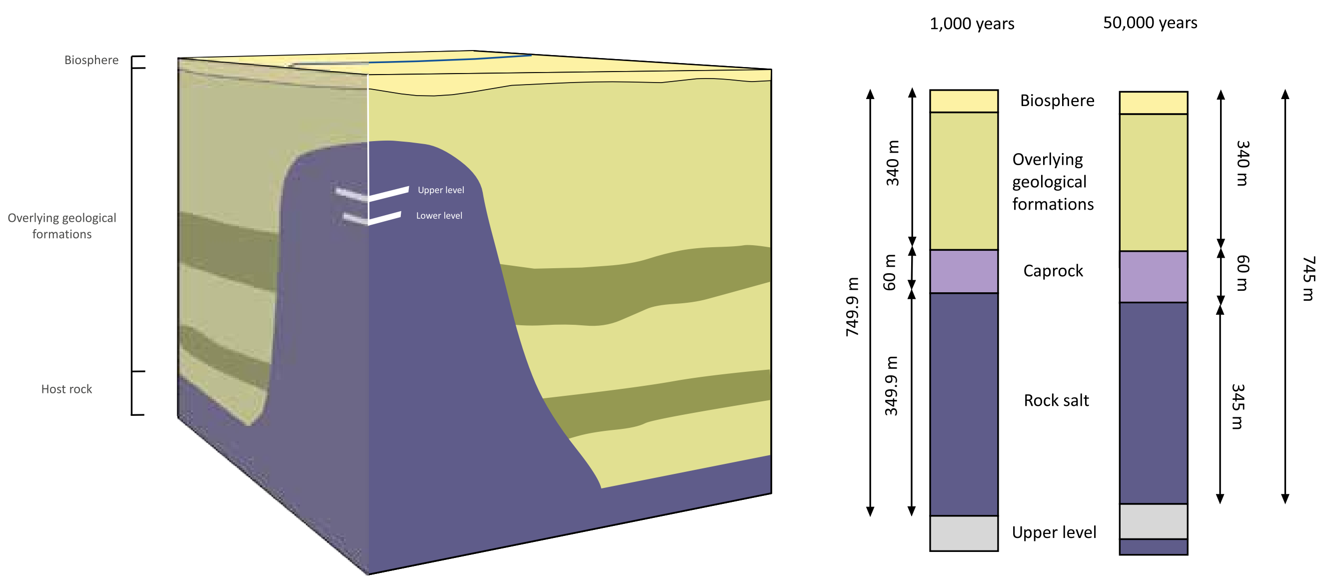


Figure 7.4) Closure of the repository to the start of the next glacial period. On the left, is the generic salt dome with the two-level repository. At the surface, some erosion (e.g., river erosion) occurs, but at most to a depth of 50 m (Lommerzheim, 2023): it does not affect the performance of the repository. The right-hand figure shows a cross section through the salt dome and the repository. At the start of this period (left), there is 749.9 m between the upper level and the surface and 349.9 m between the upper level and the caprock. After 50,000 years, this has reduced, due to subrosion and diapirism to 745 m and 345 m respectively. Note that in this figure we assume that the caprock does not change in thickness and we assume that the next ice age occurs in 50,000 years.

from waste will be by advection; this phase lasts for the first few hundreds of years (up to 1,000 years) after closure of the repository (LaForce, 2024a). This can occur within the LILW disposal rooms, where the inflow of brine could result in mobilisation of LILW radionuclides and subsequent advective transport. However, the concrete used for backfilling these rooms for stability, along with concrete seals at the openings, will severely restrict the amount of brine able to enter a disposal room, thereby minimizing the corrosion of the waste packages and dissolution of the LILW and (TE)NORM. Furthermore, any dissolved radionuclides are likely to remain within the disposal rooms into which the overall brine flow is directed during this period, due to (on average) lower saturation of these rooms. In contrast, the HLW package will still be intact and thus no HLW radionuclides are transported during this period. When backfill compaction has reduced the permeability sufficiently (at the end of stage 2), the dominant mode of transport changes from advection to diffusion although some, very slow, advective transport might locally still take place due to residual pressure differences within the repository (Fig. 7.3, LaForce, 2024a).

In the presence of brine, corrosion of the HLW packages and radiolysis can result in gas generation. The amount of gas generated depends on the availability of brine, which depends on, for example, the degree of compaction. However, the backfill used in the disposal tunnels where the HLW is located will not be moisturised and will contain a very limited amount of brine (0.1% saturation). In the disposal rooms containing LILW, as pointed out above, the concrete used for backfilling, along with concrete seals at their openings, will severely limit brine inflow and therefore the amount of brine available for corrosion and radiolysis. Therefore, the amount of gas expected to be produced is limited. As the compaction of the granular salt backfill progresses and the permeability decreases during this phase, the amount of brine that can potentially reach the LILW and HLW packages and waste will continuously decrease. In the initial stage when the backfill is still permeable, any gas produced will be able to migrate to the available void spaces in the two areas of the GDF that are filled with gravel and do not compact.

7.2.2 Conditions assumed in the safety assessment

In the safety assessment for the normal evolution scenario and its reference case, it is assumed that the HLW packages will remain unbreached and provide containment for all HLW radionuclides during this period. In contrast, the LILW packages and their contents are assumed to dissolve instantly after the closure of the repository, with the radionuclides becoming immediately and homogeneously distributed within the concrete backfill of the disposal rooms. The temperature and lithostatic pressure, both of which influence the compaction of the granular salt backfill, are assumed to remain constant throughout the safety assessment. For estimating the temperature, a representative geothermal gradient is assumed (Bonté et al., 2012; Smit, 2022). The lithostatic pressure is calculated using the average density of sediments and salt at the Gorleben site (2,240 kg/m^3, Müller-Hoeppe et al., 2012). Furthermore, no solubility limits are assumed in the safety assessment.

For the granular salt backfill, only stage 2 is considered in the COPERA (2020 - 2025) safety assessment. This implies that advective and diffusive transport of radionuclides within this safety assessment is assumed to continue even in the final stage in which, more realistically, the granular salt backfill is expected to have healed. In the safety assessment, we assume that the permeability is a function of porosity and that backfill does not reach a lower porosity then 1%: the residual porosity. Transport by advection will, however, be limited owing to the low permeability of the granular salt backfill and the small difference in pressure. For diffusive transport of radionuclides, molecular diffusion in free water is used. To account for the porosity within the granular salt backfill, the molecular diffusion in free water is multiplied by the porosity. It is assumed that all the radionuclides have the same diffusivity. Currently, due to restrictions in calculational capabilities, the backfill in the HLW tunnels is modelled as moisturised salt, although dry salt is proposed in the repository design (See figure 6.1b).

To calculate brine flow due to advection, Richard's equation can be used, as we assume in the safety assessment that no gas will be generated within the repository: gas generation will be incorporated in the next safety assessment.

7.2.2 1,000 years after closure – start next glacial period (assumed at 50,000 years)

As for the first 1,000 years, it is expected that characteristics of the biosphere (climate, morphology, vegetation, animals, land use) will remain as they are today although the sea level might change periodically. There are also no major changes expected in the groundwater regime and therefore the subrosion rate will be like that currently observed (maximum of 0.1 mm/year). Consequently, about 5 m of salt will be dissolved in the 50,000 years since the closure of the repository. Similarly, since no major stress changes are expected, the depth of the repository will have decreased by about 5 m, due to diapirism (maximum of 0.1 mm/year). At the end of this period, 345 m of salt will remain between the upper level of the GDF and the top of the salt dome. The upper level will be at a depth of around 745 m, having moved up by 5 m (Fig. 7.4). As in the first 1,000 years, the geometrical changes to the system are insignificant and neither process will affect the performance of the repository.

Within the repository, compaction of the moisturised granular salt has ceased (stage 2), and its healing has started (stage 3) with the possible exception of the backfill in the shafts. Compaction of the moisturised salt there is slower, due to the lower temperature and pressure, and might continue for an additional few hundreds of years before stage 3 starts. In contrast, the dry granular salt in the HLW disposal tunnels, will take an additional few thousand years to eventually reach stage 3 (Oosterhout et al., 2022; Spiers et al., 1988). In this stage, with time, the pores within the backfill will become disconnected due to healing of the salt, further limiting transport by diffusion, which eventually becomes zero within a few thousand years: any radionuclides released from HLW and LILW in brine are from then onwards essentially immobilised in the salt and concrete backfills of the disposal rooms and tunnels. In addition, no water from outside the salt dome will be able to enter the rest of the repository below the granular salt backfill in the shafts. The sections of the shafts above the salt backfill will eventually become fully saturated but groundwater will not be able to pass the shaft salt backfill, which will have compacted sufficiently to form an impermeable seal. Although it is very unlikely, due to the very limited amount of brine available, the HLW package might fail at some point during this period because of corrosion and lithostatic load.

7.2.3 Conditions assumed in the safety assessment

In the COPERA (2020 - 2025) safety assessment for the normal evolution scenario and its reference case, it is assumed that the HLW package will fail 1,000 years after closure. The HLW radionuclides in the CSD-v, CSD-c and ECN canisters are assumed to be instantaneously available for transport directly after the failure of the HLW package. No credit is taken for the slow dissolution of vitrified waste that would be expected in reality. As it is assumed that the LILW radionuclides have already dissolved instantly after the closure of the repository, they are available for transport during this period.

Since no complete healing of the granular salt backfill is assumed to occur in the safety assessment, advective and diffusive transport of radionuclides will still be possible during this period. However, it will be very limited due to the low permeability of the granular salt backfill. As in the previous period, we also assume that gas generation is zero, because salt is relatively dry, and brine flow will be limited due to the low permeability of the granular salt backfill. Consequently, no brine can flow towards the waste and waste packages.

7.2.4 Next glacial period (50,000 to 150,000 years after closure)

This period encompasses the whole of the ice age (Fig. 7.5) that we conservatively assume to begin 50,000 years after GDF closure. We assume that this period last 100,000 years. During the glacial period, we assume that the uppermost 50 m of sediments is eroded (Lommerzheim, 2023) while rivers might incise by up to 20 - 120 m (Lommerzheim, 2023; ten Veen et al., 2015) and glacial channels may form up to 600 m deep (Van Dijke and Veldkamp, 1996).
As an ice sheet encroaches over a salt dome, as explained in section 5.2.4.2, it will result in differential loading, which can increase the diapirism rates. This increase is only temporary, as the ice sheet eventually grows to extend completely over the salt dome and will eventually retreat as the climate warms. For some period prior to the ice age, it is assumed that permafrost will develop up to 270 m into the underground (Govaerts et al., 2015), minimising groundwater recharge and influencing the hydrochemistry (increasing salinity) in the underlying formations, which will decrease the subrosion rate during this period. However, glaciation will also cause a lowering of the sea level, resulting in larger groundwater flow velocities than those occurring presently. This in turn leads to a lowering of the fresh water - salt water interface and therefore higher subrosion rates (Lommerzheim, 2023). The movement of an ice sheet over the repository can also reactivate old faults which can locally result in a higher permeability. At the end of the glacial period, melting of the ice sheet could force fresh water into the overburden sediments, resulting in an increase in the subrosion rate. Furthermore, glacial channels up to a depth of 600 m might form during this period (ten Veen et al., 2015). These glacial channels are almost immediately filled with sediments.

In the shafts, the uppermost concrete seals could potentially begin to degrade and lose their low permeability, as the geochemical environment may become unfavourable during and after an ice age. Below the granular salt backfill in the shafts, however, the concrete seals will remain intact (Fig. 6.1c). This is because the granular salt backfill in the shaft will have the same properties as the host rock by that time: impermeable. Consequently, these seals will not be affected by any changes in the geochemical environment within the biosphere.

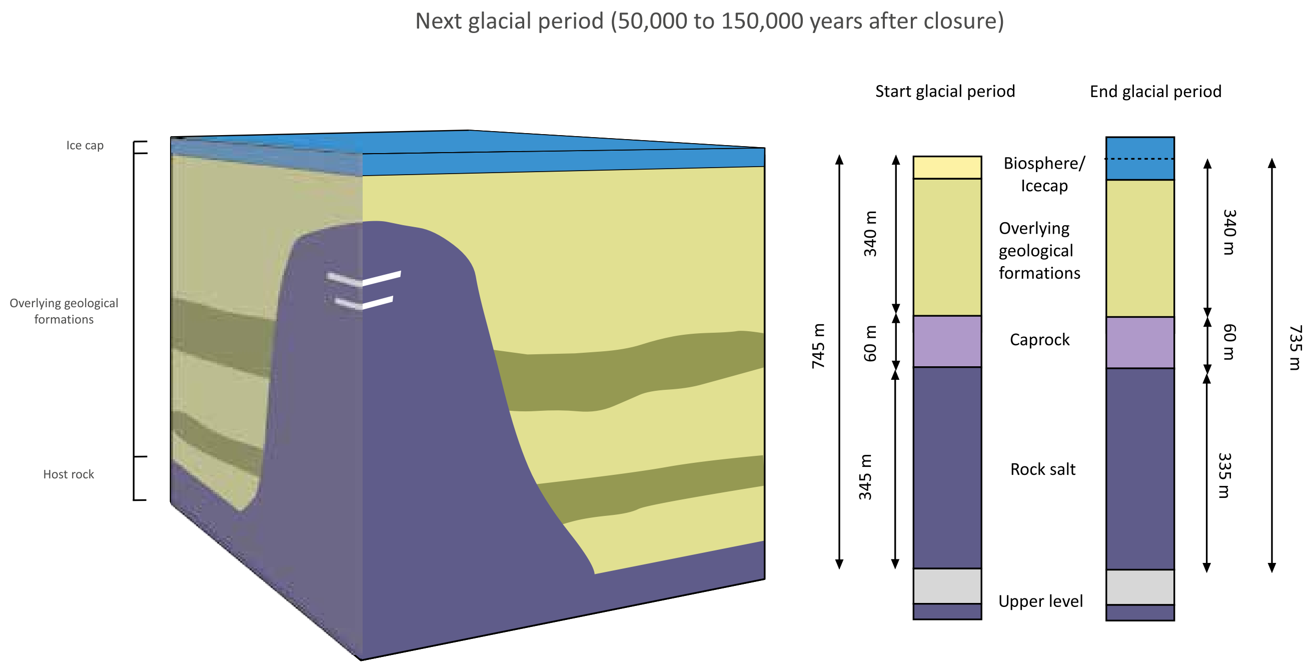


Figure 7.5) Next glacial period. As shown in the left figure, a large ice sheet covers the salt dome and the repository and has removed the biosphere and part of the upper shaft. Within the repository, there are essentially no impacts on the presence or behaviour of fluids as the backfill is expected to have healed (stage 3). At the end of this period, there is 335 m between the caprock and the lower level which will be located at a depth of 735 m. We assume that the next glacial period will last for 100,000 years and occurs 50,000 years after closure.

Within the repository, there are no significant impacts of the passage of ice and permafrost conditions, and there are essentially no impacts on the presence or behaviour of fluids. By this time, compaction of the backfill has reached the end of stage 3 and will have the same properties as the host rock so that all radionuclides from LILW and HLW remain immobilised within the granular salt backfill if released. Furthermore, if the HLW package has not failed by the end of stage 3 as is expected, it is likely to remain intact as there is no brine available for corrosion. Likewise, corrosion of the LILW packages and Konrad Type II containers will have halted due to the lack of brine and any dissolved LILW radionuclides will either remain trapped in the disposal or within the concrete backfill of the disposal room.

If the next ice age has a duration of 100,000 years, it is expected that the salt dome will have risen by about 10 m during this period. Concurrently, about 10 m of salt will be dissolved during this period, due to subrosion (Fig. 7.5).

7.2.5 Conditions assumed in the safety assessment

In the COPERA (2020 - 2025) safety assessment, it is assumed that the concrete tunnel and shaft seals will fail at the start of this period. Their containment function cannot be guaranteed after significant changes in their environment and it is difficult to predict the hydrochemistry of the geosphere in 50,000 years, after the assumed start of the first ice age, to thus determine whether the concrete would still be stable. Their failure is modelled by increasing the permeability by four orders of magnitude (Beuth et al., 2012). Since complete healing of the granular salt backfill is not assumed to occur in the safety assessment, advective and diffusive transport of radionuclides will still be possible during this period. However, advective flow will be very limited due to the low permeability of the granular salt backfill. As in the previous period, we also assume that gas generation is zero, because salt is relatively dry and relatively soon after the closure of the repository (1,000 years) advective flow will be minimal due to the ultra-low permeability of the backfill. Consequently, no brine can flow towards the waste and waste packages. We assume, for the normal evolution, that there are no large brine pockets that could migrate through the salt host rock.

As in the previous period, the safety assessment assumes that all the radionuclides in the LILW and HLW are available for transport during this period.

7.2.6 End of next ice age – 1,000,000 years

The next stage encompasses the period between the end of the first ice age after the closure of repository and 1,000,000 years. During this period, multiple glacial periods could occur, which might again result in the formation of tunnel valleys and temporarily increased subrosion and diapirism rates. These will, again, change the biosphere completely and could begin to affect the overburden formations. Depending on the number of glacial periods that will occur in the next one million years, this could result in a significant reduction of the overburden overlaying the repository, bringing the repository close to the surface, especially if no sedimentation occurs. As explain earlier, it is unlikely that more than 10 glacial periods will occur within 1 million years and some of those that do

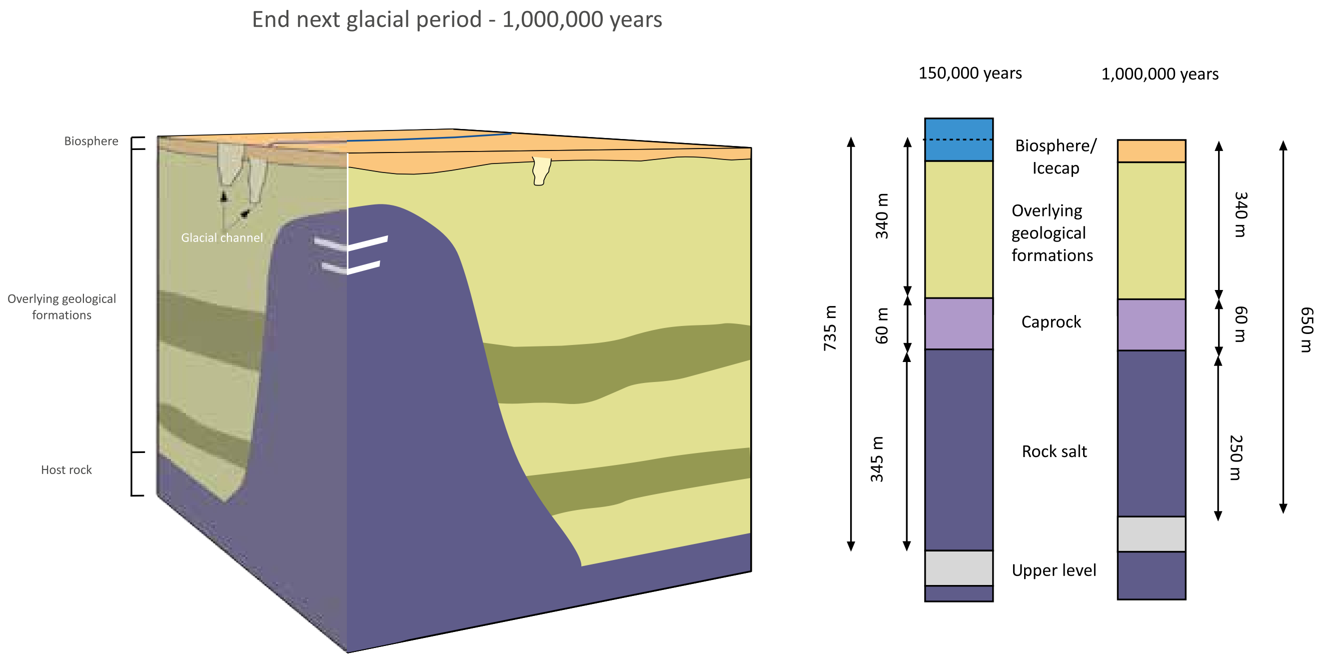


Figure 7.6) The next period spans from the end of the next ice age until 1 Ma after closure of the repository. The glacial period has completely changed the biosphere and during the melting of the ice cap, deep glacial channels developed as shown in the left figure. Within the repository, nothing happens as the backfill is expected to have healed (both dry and moisturised stage 3). At the end of this period, there is 250 m between the caprock and the lower level, which will be located at a depth of 650 m below the surface. Note that this figure only shows a single ice age, although it is expected that multiple ice ages may occur within 1 Ma after closure of the repository.

occur will not be intense enough to cause ice cover as far south as The Netherlands. Nevertheless, there may be several periods of intense glaciation in this period. It is likely that sedimentation will occur during interglacial periods over the next million years, increasing the thickness of the overlying formations, as the Netherlands is in the delta region of some major European rivers. In the last millions of years, the sediment thickness on top salt domes has tended to increase and hence the depth of the salt domes has effectively increased. Also, no major tectonic events are expected that could result in significant uplift of the Netherlands. How the overburden formations will develop in the next one million years is, however, largely uncertain, although it is considered unlikely that the salt dome will pierce through to the surface.
As discussed later, during the first million years the repository is still located hundreds of metres below the surface

Within the repository, essentially nothing has changed since the first ice age: the backfill still has the same properties as the host rock and all radionuclides are still within the repository.

In terms of subrosion, about 100 m of salt will be dissolved due to subrosion during the first million years. There is thus still 250 (350) m between the upper (lower) level repository and the top of the salt dome. Even in the unlikely situation that the subrosion is twice as high (alternative scenario), there is still 150 (250) m between the upper (lower) level repository and the top of the salt dome.
Due to diapirism, the upper (lower) level will be located at a depth of around 650 (750) m after one million years: both having moved up by 100 m. Both processes will thus not affect the performance of the repository (Fig. 7.6).

At even longer time scales (millions of years), subrosion and diapirism could eventually result in the release of residual, immobile and long-lived radionuclides into overburden formations. By this time, however, the GDF will have a hazard potential like or lower than naturally occurring ore bodies.

7.2.7 Conditions assumed in the safety assessment.

Within the repository, essentially nothing has changed since the previous period: the granular salt backfill has the same properties as the host rock and all radionuclides are still within the repository. However, in the COPERA (2020 - 2025) safety assessment, it is assumed that advective and diffusive transport of radionuclides (although unlikely) is possible in this period. This is because the granular salt backfill is assumed not to heal, and hence pores within the granular salt remain connected. However, transport will be limited by the very low permeability of the granular salt backfill.
As in the previous period, we also assume that gas generation is zero.

7.3 Alternative scenarios

Each alternative evolution scenario differs in a single aspect from the normal evolution scenario. As part of the COPERA (2020 - 2025) research programme, Lommerzheim (2023) derived a set of alternative scenarios. While there are multiple possible alternative evolution paths, many have equivalent impacts on parts of the disposal system and have similar consequences on repository system evolution. For example, destruction of the shaft seal due to a glacial channel is covered by the alternative scenario: failure of the shaft seal. Therefore, all alternative evolutions that are currently considered feasible are covered by one of eight representative alternative scenarios:

- Failure of the HLW packages.
- Failure of a shaft seal.
- Failure of the tunnel seals.
- Failure of a spiral ramp seal.
- Flow path between brine pocket and mine excavations.
- Less probable characteristics of radionuclide mobilisation and transport.
- Reduced long-term sealing by backfill.
- Pressure-induced permeation of fluids in salt formations.

7.4 Inadvertent Human Intrusion

Salt has been a resource for thousands of years. Halite is part of the human diet, while potash salts are important for fertiliser production and other chemical processes. Salt has not only been used as a resource, but salt structures are also used for the storage of gas and nitrogen, and for the disposal of hazardous and radioactive wastes. Moreover, hydrocarbon reservoirs (e.g., natural gas) are frequently discovered close to salt deposits. In Groningen, for example, the bedded Zechstein Group is the seal for the large Groningen gas field (Breunese et al., 2010.).

Since salt is a resource, this raises the possibility of inadvertent human intrusion: an individual or group of individuals being exposed to the radioactive waste while, at least initially, unaware of the potential hazard (Cooper, 2002). Since future human evolution and human society are unpredictable over long (thousands of years) and even short (hundreds of years) time frames, stylised scenarios are used to assess such scenarios. These simplified or idealised scenarios are based on present day knowledge: they do not attempt to predict the future. Inadvertent human intrusion may occur if all knowledge about the existence of the repository has been lost. According to IAEA (2017b) ,this is perhaps most probable to occur several centuries after closure of the repository. The consequence of inadvertent human intrusion depends on the location of intrusion (e.g., directly through the waste) and the status of the repository system (if boreholes cause water ingress). In total, four potential human actions have been found to be important for the long-term safety of the repository (Lommerzheim, 2023): drilling activities, mining, human influence on climate and human influence on water management.

7.5.1 Inadvertent Human Intrusion: Drilling activities

This encompasses a range of scenarios depending upon various factors such as the location of drilling with respect to the GDF structure and the state of evolution of the repository system (e.g., the state of backfill compaction; presence of liquids/gases in the mine). Any direct intrusion into the repository excavations could lead to liquid ingress.

7.5.2 Inadvertent Human Intrusion: Mining

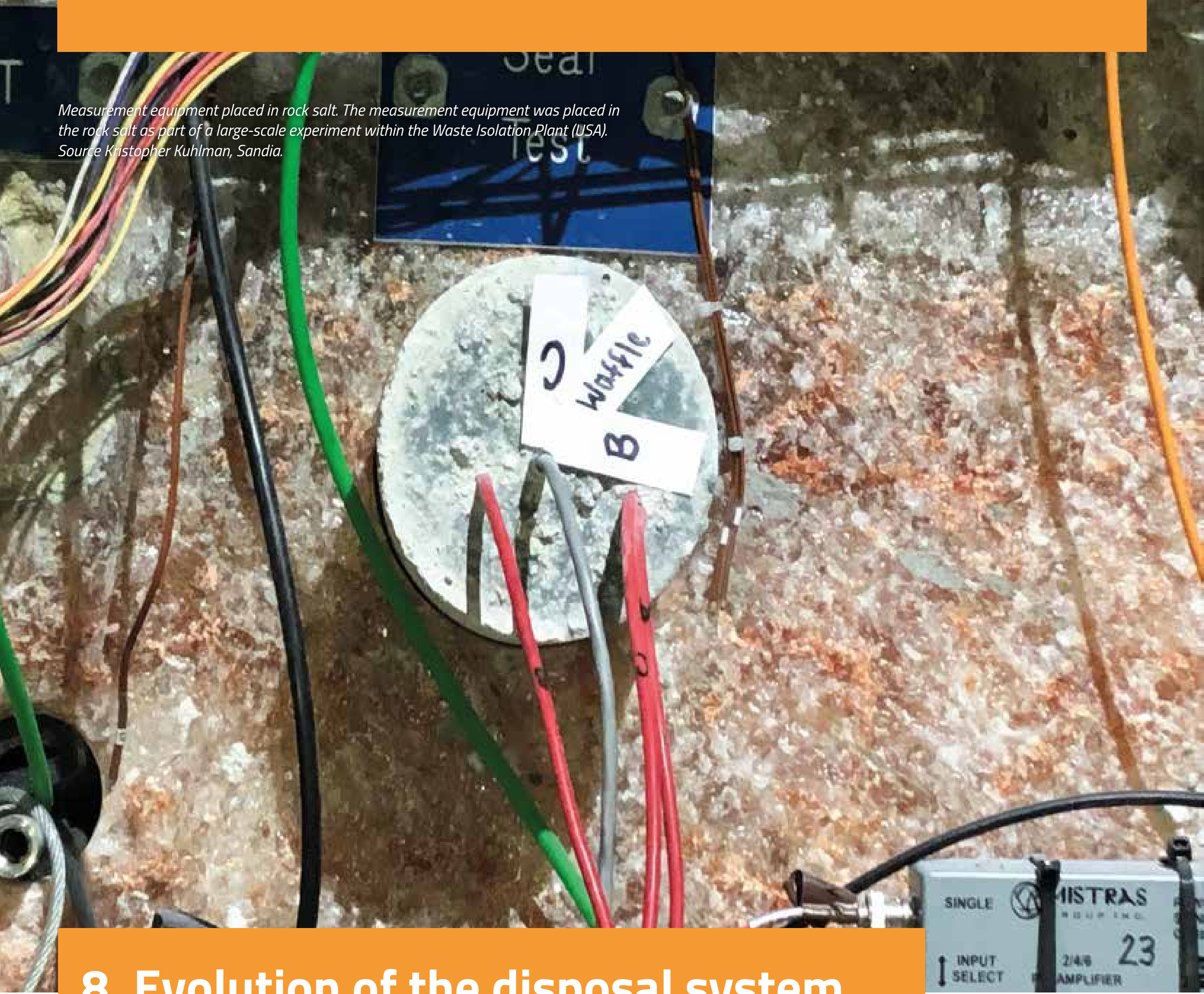
This encompasses the construction and operation of a conventional salt mine or leaching caverns located near or within the repository. Both a conventional salt mine and the leaching of caverns would disrupt the geosphere surrounding the repository and could impact the performance of the engineered barrier system. The consequences depend on how closely the conventional salt mine or leaching caverns are located to the repository. Since monitoring is an integral part of the construction process, any contamination during construction is expected to be brief and would likely be detected promptly. Thus, contamination is likely to remain confined to the mine entrances or caverns, with minimal release into the surrounding biosphere. Note that in the Netherlands, no conventional salt mines exist.

7.5.3 Human influence: Climate

This encompasses the influence of humans on climate, driven (locally or in a region) by the human-induced increase in atmospheric CO_2 levels. While a higher temperature has no direct impact on the barriers, it can delay the start of the next glacial period. A qualitative assessment of the impact of this climate scenario, including the associated delay in the onset of the next glacial period, on the evolution of the repository system indicates that the potential consequences are already accounted for in the reference climate evolution model.

7.5.3 Human Influence: Water management

Water management encompasses various activities within the overburden and surrounding formations. These activities involve the establishment of water wells for sourcing, pumping, infiltration, underground storage, groundwater reservoirs et cetera. These interventions are relevant only in the later stages of the repository system's development: if contaminated fluids from the repository find their way into the aquifers of the overburden formations through shafts. Hence, water management can potentially accelerate and intensify the spread of contaminated fluids within aquifers and surface water bodies.



Measurement equipment placed in rock salt. The measurement equipment was placed in the rock salt as part of a large-scale experiment within the Waste Isolation Plant (USA). Source Kristopher Kuhlman, Sandia.

8. Evolution of the disposal system

Summary:

- The safety assessment for normal evolution of the system indicates that no release of radioactivity will occur during the first one million years – even when conservative/ pessimistic assumptions are made in the analyses.
- Five of the alternative scenarios were studied- failure of the HLW packages, failure of the tunnel seals, failure of the spiral ramp seal, less probable characteristics of radionuclide mobilization and transport, and reduced long-term sealing by backfill. These also do not result in any releases during the first one million years.
- The result of this safety assessment for a repository in rock salt is in line both with previous Dutch safety assessments and with international safety assessments.

A central part of this conditional safety and feasibility study is the modelling and calculation of the potential impacts of the GDF on people and the environment in the future. This is done using quantitative safety assessment, based on a representation of the long-term evolution of the repository. An assessment based on conservative or pessimistic data and assumptions aims at enhancing confidence in post-closure safety. Assessments using data that are more realistic are used to optimise the design, identify knowledge gaps, steer the development of knowledge and guide future research.

The safety assessment involves developing computer models of all the significant processes to simulate the evolution of a GDF and quantifying the necessary parameter values for input to these models. Here, we first summarise briefly how the COPERA (2020 – 2025) disposal concept is expected to provide long-term safety. This is followed by a short description of the computer model that is used to calculate the evolution of the GDF over time and its potential impacts in terms of radiation doses. Then, the results of the safety assessment calculations are described and compared to performance yardsticks, where appropriate. Lastly, the outcome of this safety assessment is compared with previous Dutch and international safety assessments for a repository in rock salt.
For a more detailed description of the modelling setup and results, we refer to Bartol (2025).

8.1 Modelling approach

To demonstrate in a quantitative way how the salt GDF multibarrier system provides the required safety, a numerical computer model is used. This is a mathematical representation of the evolution of the disposal system through time, considering all the processes that could result in significant movement of radionuclides. In our model, both advection and diffusion are responsible for the movement of radionuclides at some point in the evolution of the repository. Advection is the transport of the radionuclides by the bulk motion (flow) of brine, which can result from compaction of the granular salt backfill or following ingress of brine into the repository through the shafts. Diffusion, on the other hand, is the movement of radionuclides in a static volume of brine from a region of high concentration to a region of low concentration.

Only the processes contributing to radionuclide transport in the repository opening needs to be modelled, as the undisturbed host rock is so impermeable that no transport will take place in it (see section 5.1.1). A similar approach was taken in previous Dutch safety assessments (Prij et al., 1993), in German studies (Bollingerfehr et al., 2018a) and in international generic safety assessments (LaForce et al., 2023). In common with international safety assessment practice, the present safety assessment is restricted to one million years after closure. On this time scale, processes such as subrosion and diapirism will not affect the barrier function of the host rock (see Chapter 7), and the hazard potential of the waste will reduce considerably – to a level much below that of natural uranium ore bodies (see box 2.1).

To model both the hydrogeological evolution (brine flow) and the transport of radionuclides within the repository, a two-dimensional, computer model is used, based on the configuration of the GDF shown in Fig. 8.1a, b, c. A two-dimensional model was selected since transport of radionuclides through the repository levels and the shafts will predominantly be in, respectively, the horizontal or vertical plane. This effectively minimises the travel distance to the overburden formations as the radionuclides can only travel in two dimensions in the granular salt backfill rather than three and is thus a conservative assumption. In addition, this reduces the computational time needed significantly.

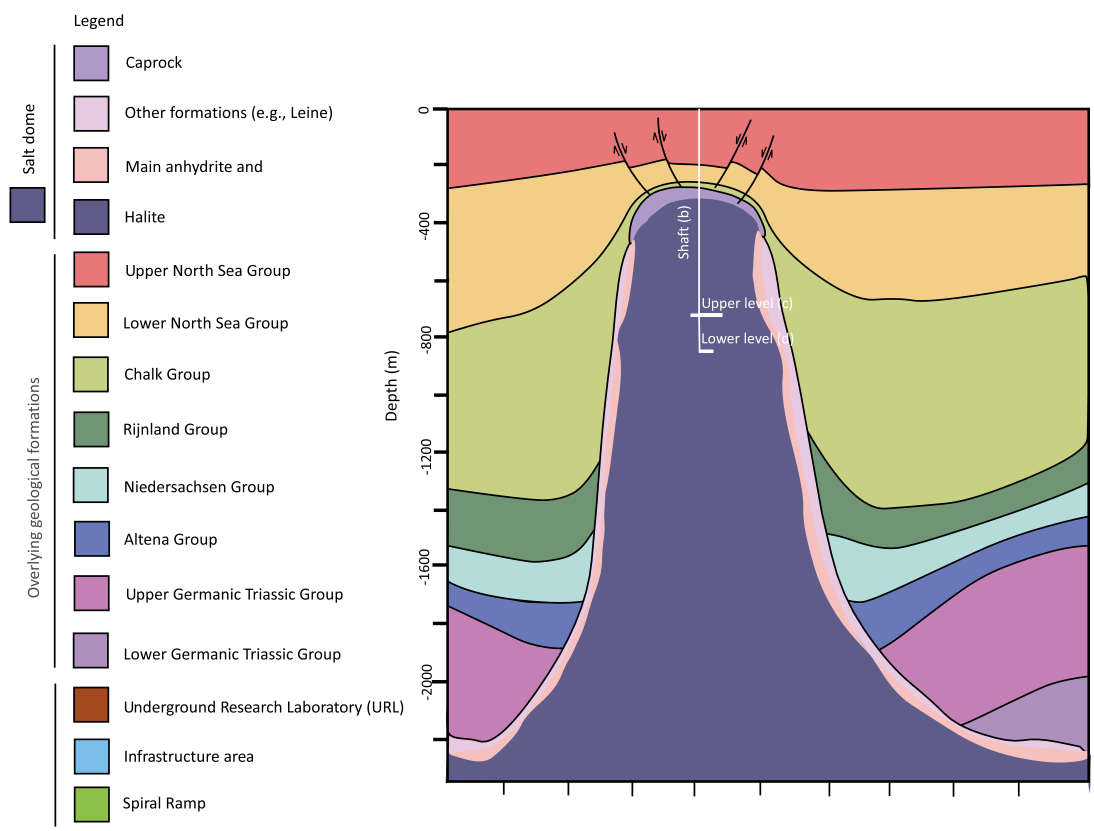
The two-dimensional model consists of an overburden, the transport shaft, the two ventilation shafts, the two repository levels and the inclined spiral (transport) ramp connecting the upper with the lower level. Above the salt dome an overburden 100 m thick is modelled (table 8.1), this layer ensures that both brine and radionuclides can migrate into the overburden. In this safety assessment, each shaft is modelled with its own overburden to prevent the development of convective flow that enters one shaft, flows through the upper or lower level and exits a second shaft. While this is conceivable, it is outside the scope of the current safety assessment, as it would require a three-dimensional model to position the different shafts realistically. Of the shafts connecting the upper and lower level with the surface (850 m), only the 450 m section that is in salt is modelled (Fig. 8.1b). The remaining upper part (400 m) in the overlying sediments does not have a safety function and is excluded in the model to reduce the computational time needed. The shafts connecting the repository to the surface are 5 m (ventilation shaft) and 8 m (transport shaft) in diameter. The spiral ramp connecting the upper with the lower level, is modelled as a vertical transport shaft of 100 m. Modelling the inclined spiral ramp connecting the upper with the lower level as a purely vertical shaft is conservative, as the distance that radionuclides would travel between the two levels is shortened (100 m vs 1920 m).

The safety assessment models the behaviour of four types of material: the granular salt backfill, the concrete used for the tunnel seals and shaft seals and to backfill the LILW disposal rooms in the upper level, gravel used to fill the infrastructure area, and unconsolidated sediments (overburden). For the granular salt backfill, only the second phase of compaction is considered in this safety assessment (Nicholas and Thatcher, 2023; Oosterhout et al., 2022). Healing of the granular salt backfill and thus the disconnection of the pores in stage 3, is conservatively not modelled (See section 6.1.2). The granular salt backfill consequently is treated as remaining permeable throughout this safety assessment, with its permeability depending on the porosity, as described by Oosterhout et al. (2022) and references therein. Furthermore, in the safety assessment, as mentioned above, we assume that all granular salt backfill, including the backfill in the HLW tunnels, is moisturised (Fig. 6.1b; centre figure). This is because the compaction model used here (Oosterhout et al., 2022) is developed specifically for moisturised granular salt backfill. The compaction of the granular salt backfill depends on the temperature and pressure (Oosterhout et al., 2022); constant temperatures of 36.5 and 34.5 °C and lithostatic pressures of 18.7 and 16.5 MPa are used for the lower and upper levels respectively. The temperatures are based on the stable geotherm (Smit, 2022) while the lithostatic pressure at both levels is calculated based on depth and the average density of the overlying formations, using analogue data from the Gorleben site (Müller-Hoeppe et al., 2012). The assumed temperature is conservative, as the heat-generating waste will temporarily increase the temperature within the repository, thereby increasing the compaction rate.

For the concrete seals in both the tunnels and shafts, including the spiral ramp that is modelled as a shaft, and for the concrete used as backfill in the disposal rooms, a permeability of $4.5 \cdot 10^{-18}$ m^2 is used (See section 6.1.5). As the expected lifetime of both tunnel and shaft concrete seals cannot be guaranteed after 50,000 years (See section 6.1.5), their permeability is conservatively modelled as gradually increasing over a period of 1,000 years from 50,000 years to $4.5 \cdot 10^{-14}$ m^2: i.e., by four orders of magnitude (Beuth et al., 2012). The gradual increase in permeability is necessary to ensure model stability, as the model has difficulty handling sudden jumps or sharp transitions, such as abrupt changes in permeability. Gravel is assumed have a permeability of $1.0 \cdot 10^{-14}$ m^2 and a porosity of 0.3. Following LaForce et al. (2023), the unconsolidated sediments in the overburden are allocated a permeability of $1 \cdot 10^{-15}$ m^2 and a porosity of 0.2 (See table 8.1).

To ensure that the overburden cannot become desaturated due to slow leakage into the shafts, we assume that is replenished by ground water flow in the overburden. A head gradient of 0.005 m per m, together with the assumed permeability of $1 \cdot 10^{-15}$ m^2 in the overburden, results in an average Darcy velocity of the ground water of about $5 \cdot 10^{-11}$ m/s. The repository, on the other hand, is initially only partially saturated. Therefore Richards' equation, in combination with the Van Genuchten retention curves (Van Genuchten, 1980), is used to calculated brine flow. For the Genuchten retention curves, values are set for each material (See table 8.1). To obtain the relative permeability function, the permeability is multiplied by the saturation.

a)



b)

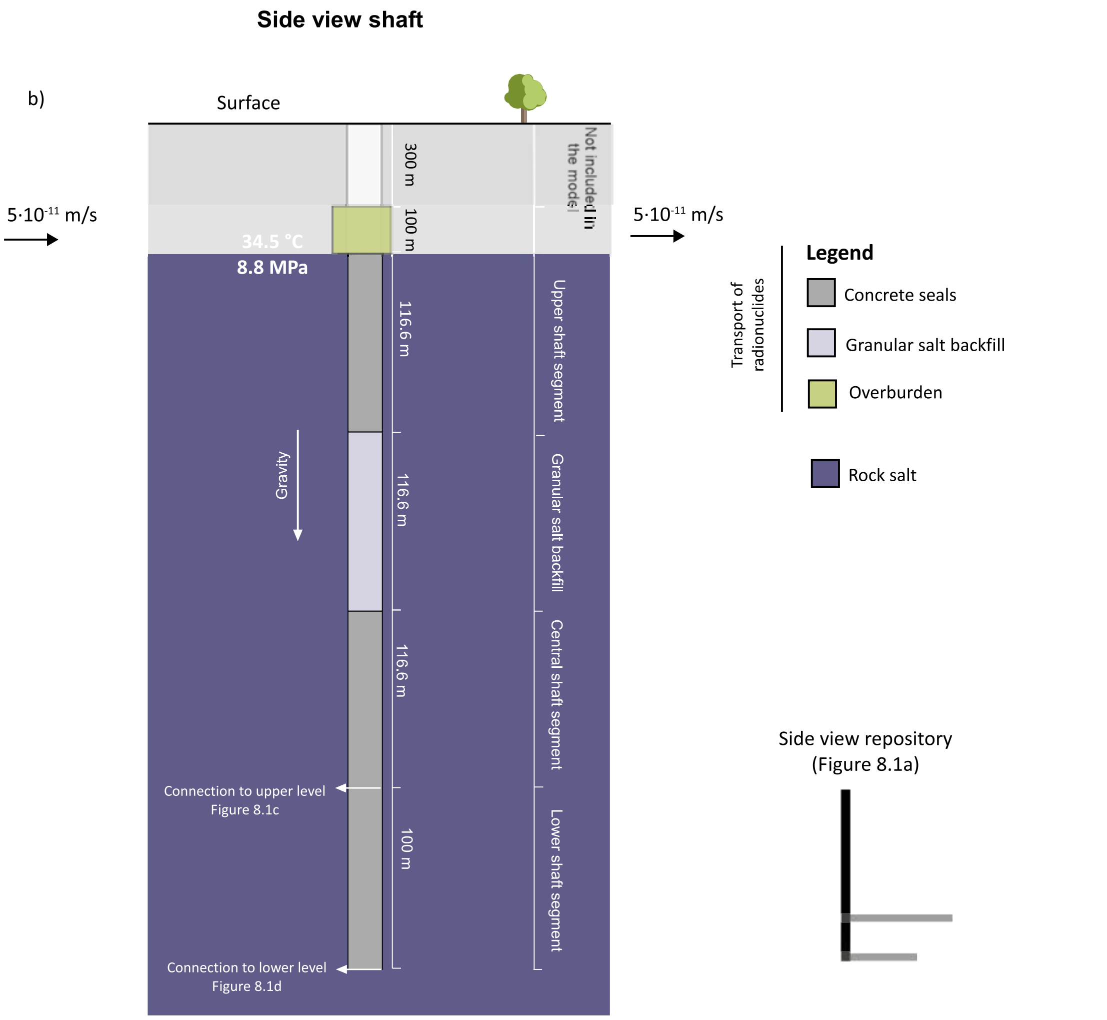
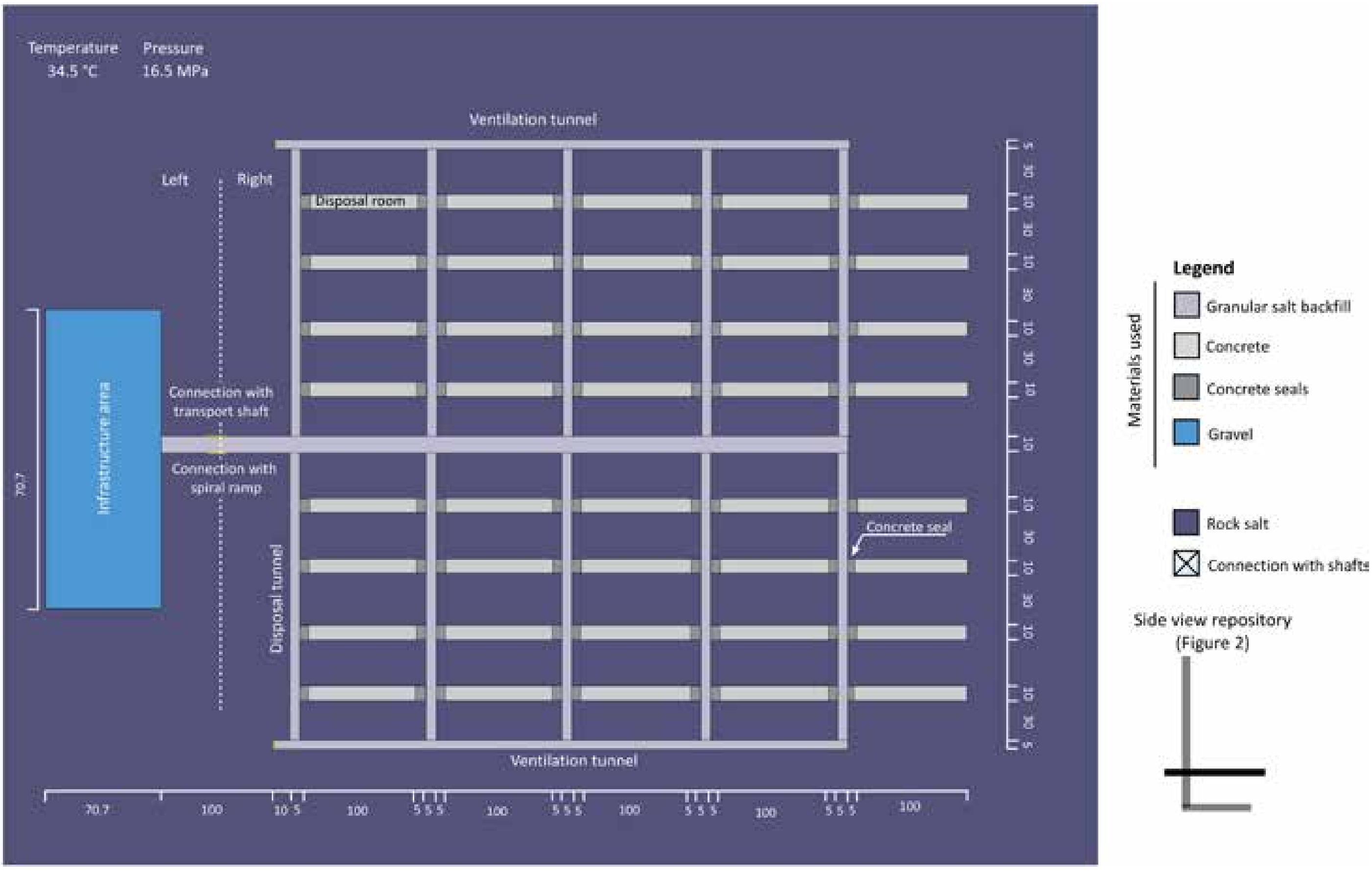
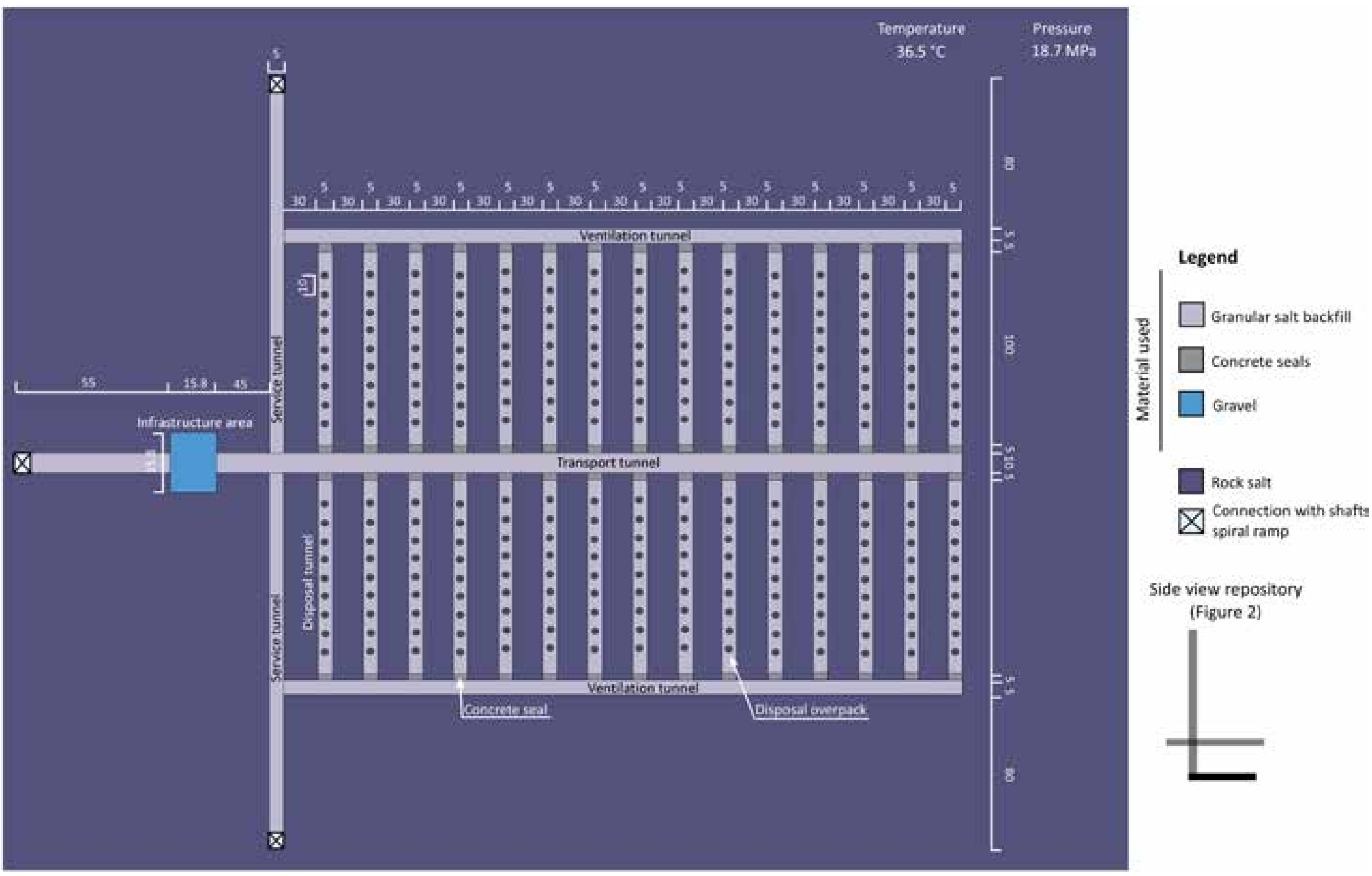


Figure 8.1a) Configuration of the two-level repository in a generic salt dome. As the host rock is impermeable, only the two levels of the repository and the shafts connecting the lower and the upper level with the surface are modelled. b) Model setup of shafts, including the materials modelled. Model setup of the lower (c) and upper (d) levels, including the materials modelled. The values of the parameters for the different materials used within the repository are given in Table 8.1. Note that the concrete backfill of the disposal rooms in the upper level has the same material properties as the concrete seals. Figure from Bartol and Vuorio (2025)

c)


Temperature
34.5 °C
Pressure
16.5 MPa
Ventilation tunnel
Left
Right
Disposal room
Infrastructure area
70.7
Connection with transport shaft
Connection with spiral ramp
Disposal tunnel
Concrete seal
Ventilation tunnel
Legend
Materials used
Granular salt backfill
Concrete
Concrete seals
Gravel
Rock salt
Connection with shafts
Side view repository (Figure 2)

d)


Temperature
36.5 °C
Pressure
18.7 MPa
Ventilation tunnel
Service tunnel
Infrastructure area
Transport tunnel
Disposal tunnel
Ventilation tunnel
Concrete seal
Disposal overpack
Legend
Material used
Granular salt backfill
Concrete seals
Gravel
Rock salt
Connection with shafts spiral ramp
Side view repository (Figure 2)

Parameter	Overburden	Granular salt backfill	Concrete	Gravel
Initial saturation (-)		4%	46%	0.01%
Permeability (m^2)	$1 \cdot 10^{-15}$ *(a)*	Depends on porosity *(c)*	$4.5 \cdot 10^{-18}$ - $4.5 \cdot 10^{-14}$ *(g)*	$1.00 \cdot 10^{-14}$ *(j)*
Porosity (-)	0.20 *(a)*	0.40 *(d)*	0.13 *(h)*	0.30 *(k)*
Tortuosity (-)	-	-	-	-
Relative Permeability Function	-	Se	Se	Se
Van Genuchten's Labda(-)	-	0.6 *(e)*	0.56 *(i)*	0.675 *(l)*
Van Genuchten's P0 (MPa)	-	1.6 *(e)*	7.7 *(i)*	1.6 *(l)*
Van Genuchten's S_{lr}	-	0.02 *(e)*	0.0 *(i)*	0.0 *(l)*
Diffusivity (m^2/s)	$1 \cdot 10^{-9}$*(b)*	$9.2 \cdot 10^{-10}$*(f)*	$2.99 \cdot 10^{-10}$*(f)*	$6.9 \cdot 10^{-10}$*(f)*
Conservatism (-)	-	Healing of the salt is not modeled	Of the two options, the higher permeability concrete is selected.	-

Table 8.1) Parameters used for the different materials in this safety assessment. (a) Values taken from LaForce et al. (2023). (b) Default COMSOL diffusivity. (c) Permeability – porosity relation from Oosterhout et al. (2022). (d) Average porosity of the backfill when a slinger machine is used from (Mischo et al., 2021). (e) Water retention properties taken from Camphouse et al. (2012) and Jové-Colón et al. (2012) similar to the ones used in DECOALEX task F (LaForce et al., 2023). (f) Following the German safety assessment, the self-diffusion coefficient of free water is multiplied with the porosity to obtain the diffusion coefficient (Bollingerfehr et al., 2018b). (g) Of the two types of concrete being considered by COVRA, sorel concrete has been conservatively assumed, as it has the highest permeability. Permeability is based on Jantschik et al. (2018) and references therein. After 50,000 years, when the performance of the concrete seals can no longer be guaranteed, the permeability of the seal is assumed to increases by 4 orders of magnitude (Beuth et al., 2012). (h) Porosity from Záleská et al. (2019) consistent with MgO cement with silica sand aggregate. (i) Based on Ecay et al. (2020). (j) Permeability from Rübel et al. (2016). Note that these values are for Vosges sandstone and not for gravel, following LaForce et al. (2023). (k) Typical values of porosity for gravel are between 0.24 - 0.38 (Domenico and Schwartz, 1997). Following LaForce et al. (2023), 0.30 is used here. (l) Based on Osselin et al. (2015).

For the transport of radionuclides, two transport mechanisms are modelled throughout the repository: advection and effective diffusion. Advective transport results from brine flow and hence depends on the compaction rate and, via pressure, on the van Genuchten retention curves (Van Genuchten, 1980). For transport by effective diffusion, molecular diffusion in free water ($2.3 \cdot 10^{-9}$ m^2/s) is multiplied by the porosity, in line with the approach taken in the German safety assessment (Bollingerfehr et al., 2018a). Note that the effective diffusivity assumes that a material is fully saturated. This is a conservative assumption, as the effective diffusivity in a partially saturated material will be less, since there is less fluid to travel through.

In the overburden, on the other hand, only diffusion is assumed, rather than both diffusion and advection. This is done to ensure numerical stability: when both are used in combination with the current boundary conditions, radionuclides may accumulate along one of the model boundaries, resulting in a large concentration gradient and potential numerical instabilities. No flux flow boundary conditions are then applied along the boundaries of the modelled overburden compartment. Together, these conditions ensure that all radionuclides leaving the GDF are retained within this compartment, making it possible to calculate the total number of moles of radionuclides leaving the repository through the shafts. This, in turn, can be used to calculate the radiation exposure of people using water from the overburden if radionuclides reach it. Furthermore, these boundary conditions ensure mass is conserved within the model. If needed, when a release is predicted, the model can be adjusted to calculate the radionuclide flux out of the repository into the biosphere.

The degradation of the different wastes and their packages will eventually lead to the release of radionuclides into the granular salt or the concrete backfill. Gradual releases are the most realistic assumption for most waste families. However, this safety assessment makes the highly conservative assumption that all the HLW forms will dissolve instantaneously and are distributed homogenously, in the backfill of the disposal tunnels, directly after the failure of the HLW package at 1,000 years after closure of the repository (see section 6.3). LILW is also assumed to be dissolved instantaneously and to be distributed homogenously within a disposal room in both the backfill and the waste packages, directly after the closure of the repository. These assumptions are made to ensure that the related impacts are estimated very conservatively (See chapter 6).

A form of particle tracking is used to mimic the behaviour of mobilised radionuclides. By doing so, the computational time reduces significantly, as only a limited number of particles need to be modelled compared to the number of different radionuclides present in the waste. The actual concentration of a specific radionuclide can be calculated when needed, by taking the concentration and the relevant half-life. Particles can be used since no interaction is assumed between the radionuclides and the host rock, granular salt backfill, concrete seals and gravel. Furthermore, no sorption of radionuclides is assumed to take place within the repository and no solubility limits used in the model: the latter will be included in the next safety assessment. In total, two types of particles are used to mimic the behaviour of the radionuclides from the instantaneously dissolved HLW and LILW - (TE)NORM.

In total, six scenarios are modelled (Lommerzheim, 2023): the normal evolution scenario in which the GDF evolves as expected, and five alternative scenarios: failure of the HLW packages, failure of the tunnel seals, failure of a spiral ramp seal, less probable characteristics of radionuclide mobilization and transport, and reduced long-term sealing by backfill. The remaining alternative scenarios cannot be modelled with this specific model setup, due to computational instabilities and the absence of the host rock in the model.

8.1.2 Uncertainties in the modelling

The safety case needs to consider 4 different types of uncertainties whose effects propagate through the overall performance assessment. The first is system uncertainty, which arises from incomplete understanding or characterisation of the disposal system. The uncertainties related to the performance of each individual component of the safety system are discussed in Chapters 5 and 6. The second is scenario uncertainty, which depends on how complete and well understood the features, events and processes of a scenario are. The scenarios and the normal evolution of the system are described in the previous chapter. The third is model uncertainty, which relates to whether the conceptual models sufficiently describe the behaviour of the disposal system and to calculational modelling uncertainties that may be introduced in the translation of the conceptual models into mathematical models and their integration into a safety assessment model. This involves model simplifications that need to be well-argued and, preferably, tested, to find out whether the calculational models correctly represent conceptual understanding. The fourth category is parameter uncertainty. The calculational and safety assessment models require values for all parameters, and here numerical uncertainty can occur, for example related to the measurement technology and sampling methodology. Parameter values must also be selected considering the range of variability and heterogeneity of natural material properties.

In the COPERA (2020 – 2025) safety assessment, realistic or best estimate data and assumptions are used, when possible, together with evaluation of the uncertainties in the results that this produces. In practice, however, a combination of best estimates and conservative assumptions is often employed to avoid overprediction of achievable safety levels. The numerical uncertainties are commonly dealt with by performing sensitivity analyses in which the relevant parameters are varied throughout their potential ranges. This can be done through deterministic modelling of multiple cases or by probabilistic models, in which parameter distributions, rather than specific values, are employed. In the COPERA (2020 - 2025) safety assessment, deterministic modelling has been employed.

8.2 The Normal Evolution Scenario

In the next sections, we briefly discuss the model results for the normal evolution scenario, starting with the shafts. Since the shafts do not have a lining within the salt dome, convergence of the host rock salt due to differential pressure results in the compaction of the granular salt backfill (Fig. 8.1b and 8.2a). Initially, this compaction is relatively fast, as the backfill is only partially saturated. Brine within the granular salt is squeezed out of the backfill into the non-compacting concrete shaft seals above and below, which act essentially as sinks for the brine. Within 1,200 years, the backfill reaches its residual porosity of 1% and its average permeability has dropped to $5 \cdot 10^{-19}\,m^2$. However, the granular salt backfill is still sufficiently permeable after reaching residual porosity that, if the repository is only partially saturated, brine from the fully saturated overburden will, albeit very slowly, continue to migrate through the concrete and the salt in the shafts and into the repository. Consequently, the shafts connecting the overburden with the upper and lower levels will, like the rest of the repository, become fully saturated about 380,000 years after closure.
Note that, compared to the granular salt backfill in both the upper and lower level (Fig. 8.2a, green and purple line), compaction of the salt backfill in the shaft is somewhat slower due to the lower lithostatic pressure and temperature. The maximum average brine flows in all 4 shaft segments occur around 1,000 years after closure after which it decreases to zero around 380,000 years after closure. The flow is dominantly into the repository (Fig. 8.2b).

In the tunnels of the upper level of the repository, as in the shafts, the granular salt backfill starts to compact immediately after closure of the repository. Initially, when the granular salt backfill is still only partially saturated, the rate of compaction is high, and brine is slowly squeezed into adjacent areas that do not compact. These are the concrete tunnel seals, the concrete used to backfill the LILW disposal rooms, and the gravel in the infrastructure area. In addition, brine near the shafts is squeezed into the lower concrete-filled part of either a ventilation or a transport shaft.
All these non-compacting volumes act as a sink for brine. The compaction rate changes dramatically around 300 years after closure (Fig. 8.2a, green line). Around this time, the granular salt backfill locally becomes fully saturated and starts to resist compaction and, around 500 years after closure, the residual porosity in the upper level of 1% is reached. This results in a temporary peak in the average saturation of the upper level. Although the permeability of the granular salt backfill is low at this point (about $1 \cdot 10^{-19}\,m^2$), it is still sufficient for brine to continue flowing into non-compacting areas, resulting in a decrease in the overall saturation of the upper level. Around 2,000 years after closure, the average saturation in the upper level starts to increase again. This increase is related to brine flowing from the lower level, where it is being squeezed out due to compaction until about 4,500 years after closure (Fig. 8.2b; Lower shaft segment).

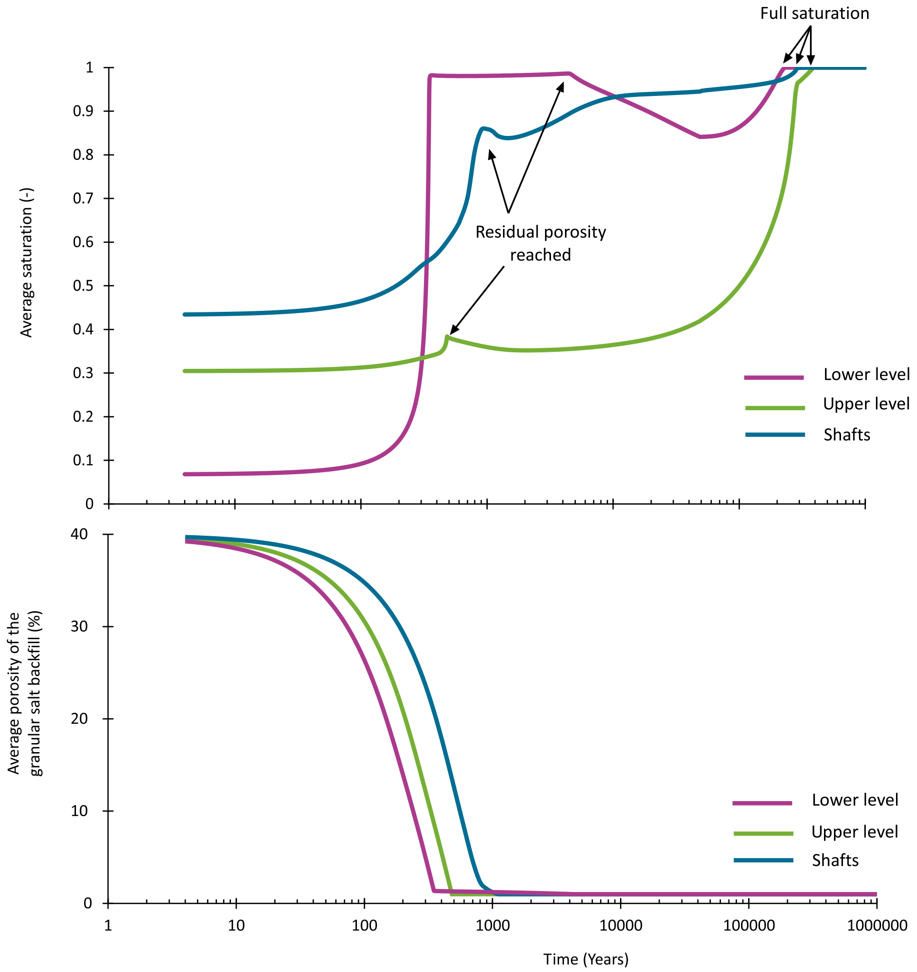
Compartment	Component	Normal evolution	Failure of HLW package
Waste - Engineered barrier system	LILW	Instantaneously release	
	HLW	Instantaneously release after 1,000 years	Instantaneously release after closure
	Effective diffusivity	All materials: molecular diffusion in free water times porosity	
Host rock	Concrete seal	Permeability decreases to $1.5 \cdot 10^{-14}$ m^2 during ice age	
	Gravel	Constant permeability	
Overburden	Minimum porosity granular salt backfill	1%	
	Permeability	Constant permeability	
Biosphere	Effective diffusivity	Constant diffusivity	
	Not modeled		

Table 8.2) The six scenarios that are modelled in the COPERA (2020 - 2025) safety assessment. These scenarios are the normal evolution scenario and five of the alternative scenarios proposed by Lommerzheim (2023).

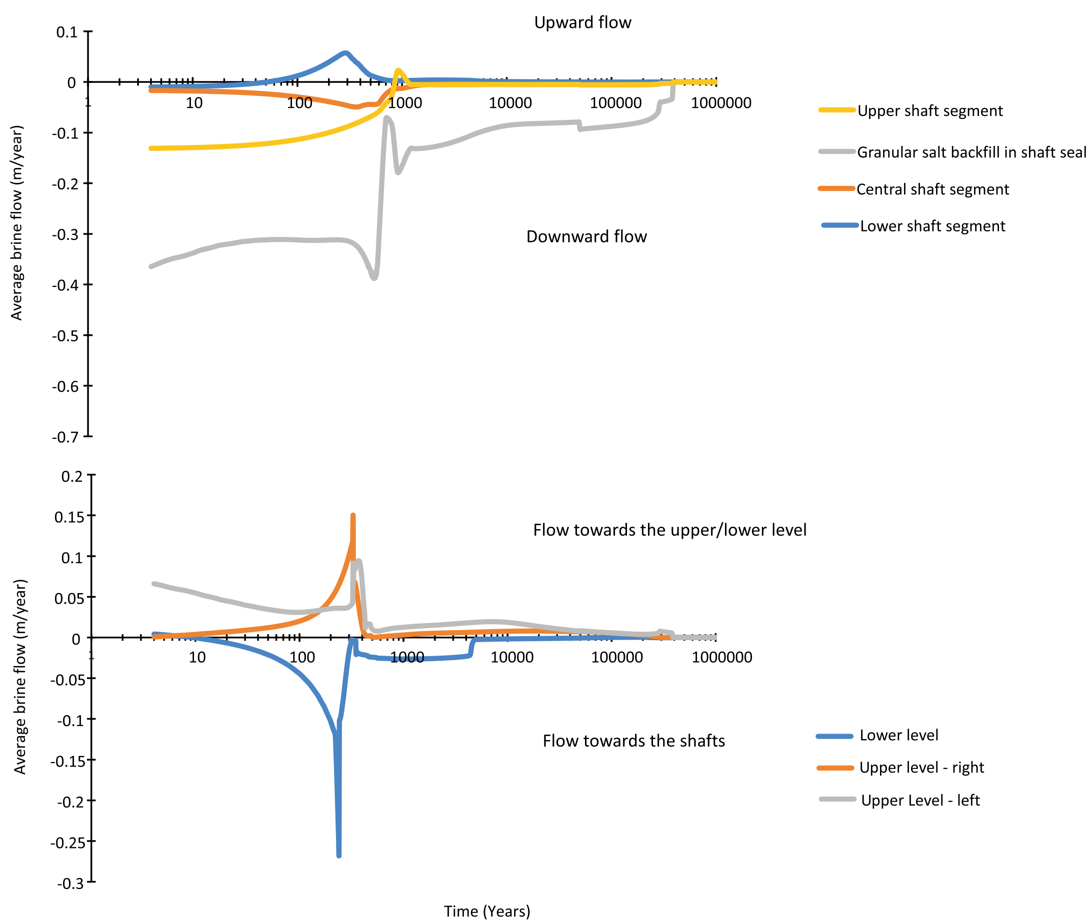
Alternative scenarios

Failure of a tunnel seal	Failure of a spiral ramp seal	Less probable characteristics of radionuclide mobilisation and transport	Reduced long-term sealing by backfill
		All materials: molecular diffusion in free water times porosity times 2	
Tunnel seals have permeability of $4.5 \cdot 10^{-15}$ m^2 directly after closure	Spiral ramp has has permeability of $4.5 \cdot 10^{-15}$ m^2 directly after closure		
			Residual porosity of 2% and 5%

a)


Full saturation
Residual porosity reached
Average saturation (-)
1
0.9
0.8
0.7
0.6
0.5
0.4
0.3
0.2
0.1
0
Lower level
Upper level
Shafts
Average porosity of the granular salt backfill (%)
40
30
20
10
0
1
10
100
1000
10000
100000
1000000
Time (Years)

b)


Upward flow
Downward flow
Average brine flow (m/year)
0.1
0
-0.1
-0.2
-0.3
-0.4
-0.5
-0.6
-0.7
10
100
1000
10000
100000
1000000
Upper shaft segment
Granular salt backfill in shaft seal
Central shaft segment
Lower shaft segment
Flow towards the upper/lower level
Flow towards the shafts
Average brine flow (m/year)
0.2
0.15
0.1
0.05
0
-0.05
-0.1
-0.15
-0.2
-0.25
-0.3
Lower level
Upper level - right
Upper Level - left
Time (Years)

c)

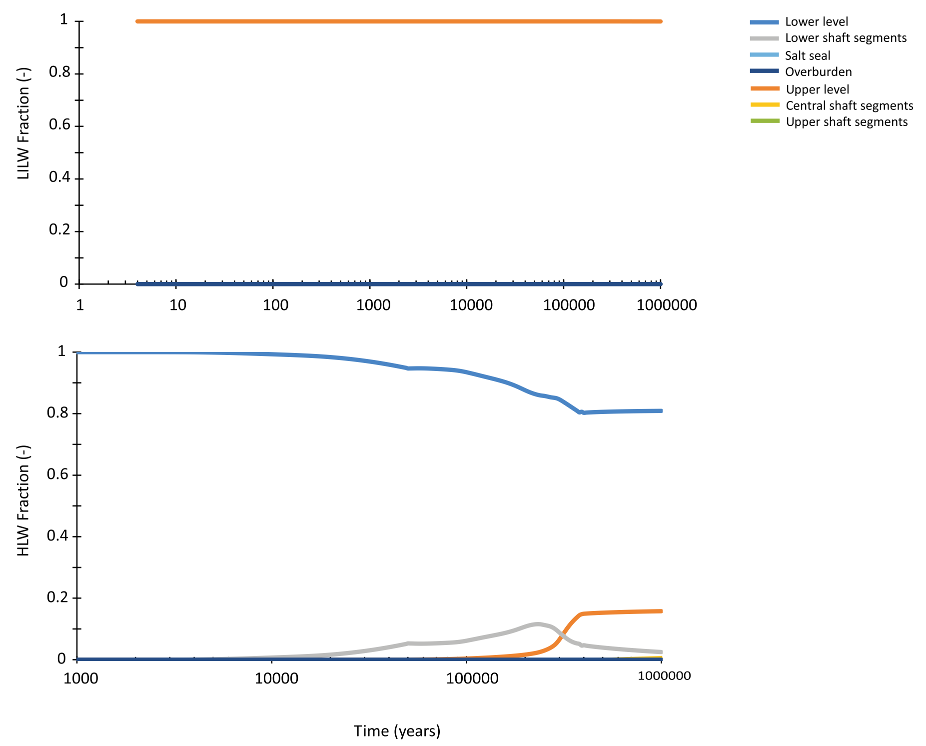


Figure 8.2a) Evolution of the average saturation (upper figure) and average porosity of the granular salt backfill (lower figure) in the shaft (blue), and in the upper (green), and lower levels (purple) of the repository. The arrows indicates either when the repository becomes fully saturated or when residual porosity is reached. b) The upper graph shows the average direction of brine flow in the four different shaft segments (see Figure 8.1b). The lower graph shows the average brine flow toward the transport tunnel in the upper and lower levels. For the upper level, there are two lines: one represents the average brine flow from the infrastructure towards the transport shaft while the other represent the average brine flow from the disposal rooms, ventilation and transport tunnel towards the transport shaft (See figure 8.1c). This distinction is made because the direction of flow, while toward the transport shaft, is opposite. For comparison, we assume that that flow is positive into the upper and lower level and negative towards the shafts. The peaks observed in the average brine flow are a result of the van Genuchten curves. c) Fraction of the total LILW (upper graph) and HLW (lower graph) radionuclides in the different compartments of the repository over time.

After compaction has ceased here, there is still some brine flow from the lower to the upper level via the shafts due to differences in pressure (saturation) within the repository (Van Genuchten, 1980). In addition, since neither the salt nor the concrete in the shafts is completely impermeable, a small brine flow from the overburden via the shafts also contributes to further saturation of the upper level (Fig. 8.2b). The upper level becomes fully saturated around 300,000 years after closure (Fig. 8.2a).
Significant brine flow ends well before 1,000 years (Fig. 8.2b)

Particle tracking from the LILW - (TE)NORM in the upper level indicates that radionuclides mobilised from the LILW and (TE)NORM are not able to leave this level (Fig. 8.2c). This is because brine flow from the upper level is dominantly into the lower level: into waste areas (disposal rooms) and into the large infrastructure area. Hence, the upper level acts as a sink for brine and consequently for radionuclides via advective transport (Fig. 8.2b). Consequently, after one million years, all the LILW - (TE)NORM radionuclides are still in the upper level.

The evolution of saturation and permeability of the granular salt backfill in the lower level differs from that in the upper level. In the lower level, only the concrete seals between the waste and the ventilation/transport tunnels, along with the small infrastructure area, can act as a sink for brine and these have only a small volume compared to the volume of granular salt backfill in the disposal tunnels. During the first 300 years, compaction in the lower level occurs rapidly, because the granular salt backfill is not yet fully saturated, and both temperature and pressure are higher compared to other parts of the GDF (Fig. 8.2a, purple lines).
This rapid compaction leads to brine flow towards the shafts and the small infrastructure area (Fig. 8.2b), which are filled with non-compacting backfill material. After 300 years, when the average granular salt backfill porosity reaches 1.3%, the compaction rate decreases dramatically as it becomes (locally) fully saturated, resulting in slower compaction. It then takes approximately an additional 4,000 years for the granular salt backfill in the lower level to reach residual porosity of 1% and for compaction to cease (Fig. 8.2a). When compaction ceases, the average saturation in the lower level starts to drop as brine continues to flow, although more slowly, to the upper level, while compaction no longer closes the vacant pore spaces (Fig. 8.2b). Around 30,000 years after closure, average saturation starts to increase again as brine from the overburden reaches the lower level. It will take another 200,000 years

for the lower level to become fully saturated. The flow of brine is dominantly out of the lower level and the maximum brine flow is reached before 1,000 years (Fig. 8.2b).

Brine flows out of the lower level into the shafts, although very slowly (Fig. 8.2b). Consequently, after one million years, the HLW radionuclides are found to be still mostly in the lower level (81 %), with less in the upper level (16%), and still less in the central shaft segment, below the shaft salt backfill (0.8%, Fig. 8.2c). No radionuclides from the HLW are present in the shaft backfill or the overburden. As in the upper level, any significant brine flow ends well before 1,000 years.

8.3 Sensitivity analysis and opportunities to optimise the system

As the GDF project progresses, the conceptual model of the disposal system that is discussed in this report will evolve, and the design, engineering and operational aspects of the repository will undergo a process of optimisation as they become more focussed and as more information becomes available. Design and function optimisation will be central to the developing project. Sensitivity analysis can also provide further insight into the behaviour of the GDF system. In this safety assessment, the impacts of varying parameter values are tested. This is done by looking at some alternative scenarios, as defined by Lommerzheim (2023). These analyses are discussed below.

Early Failure of the HLW packages: In the normal evolution scenario, it is assumed that all the HLW packages will fail 1,000 years after the closure of the GDF. Although this is already a conservative assumption, in this alternative scenario, it is assumed that all the HLW packages fail immediately after the closure of the repository. As expected, since radionuclides have more time to travel, a somewhat larger fraction of the HLW radionuclides are present after 1 Ma within the central shaft segment below the salt backfill compared to the normal evolution scenario (1% vs 0.8%) of the total amount of HLW radionuclides). However, the HLW radionuclides do not reach the backfill in the shaft or the overburden. Based on these results, the post-closure longevity of the HLW package is not a critical factor in the safety concept. Nevertheless, using a robust package has advantages, especially during the operational period.

Failure of all the tunnel seals: In this alternative scenario, the permeability of all tunnel seals is set at $4.5 \cdot 10^{-14}$ m^2 directly after the closure of the repository, i.e., their permeability is increased by four orders of magnitude (Beuth et al., 2012). Failure of the tunnel seals in the GDF immediately after its closure does not result in any releases: there is also very little difference in the distribution of

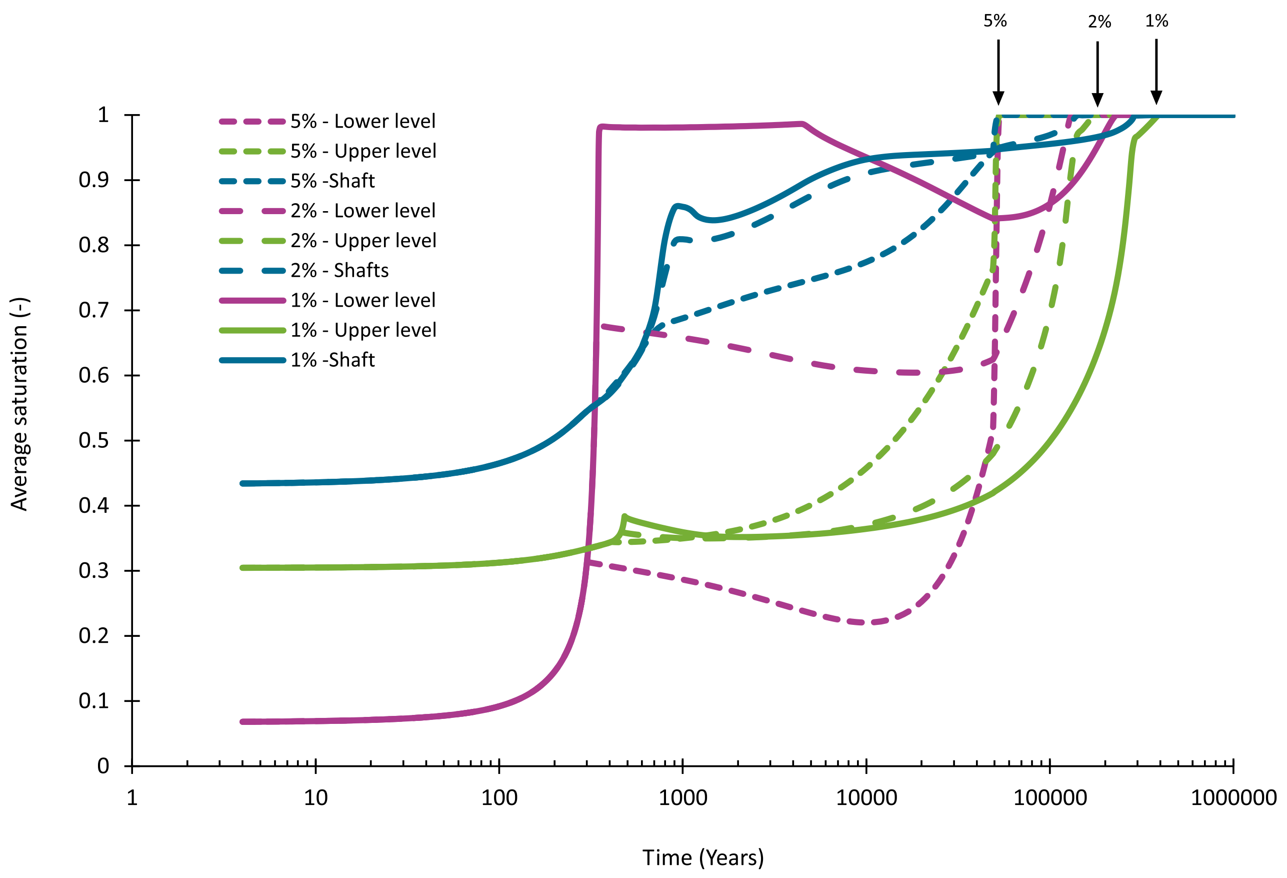


Figure 8.3) Evolution of the average saturation of the shaft (blue), upper level (green), and lower level (purple) for the different residual porosities. The figure shows that a higher residual porosity of the granular salt backfill will result in the repository reaching full saturation at an earlier stage indicated by the arrows. When the repository is fully saturated, diffusion becomes the dominant mode of transport.

radionuclides within the GDF compared to the normal evolution scenario. This can be explained by the limited dimensions of the tunnel seals, which are only 5 m thick, so their influence on the evolution of the GDF is minimal. Like the HLW package, this suggests that the longevity of the tunnel seals is not critical for long-term safety.

Failure of the spiral ramp seal: Like the alternative scenario failure of the tunnel seals, the permeability of the spiral ramp seal is set to $4.5 \cdot 10^{-14}$ m^2 in this alternative scenario, directly after the closure of the GDF. This alternative scenario also does not result in a release. This can be explained by the fact that the failure of the ramp seal does not significantly change the hydrological evolution within the repository except for the evolution of the lower level. If the ramp seal has a higher permeability, brine can flow more easily from the lower to the upper level than in the normal evolution scenario. Consequently, the lower level has on average, a significantly lower saturation (50 % vs 90%), while the upper level has a somewhat higher saturation, but the difference is only a few percent, since large parts of the upper level consist of non-compacting materials. As brine can more easily leave the lower level, compaction here is also faster: residual porosity is reached about 350 years after closure. Consequently, when the HLW packages fail, compaction has ceased and the HLW radionuclides cannot be squeezed out of the lower level. Therefore, more HLW radionuclides are expected to remain at the lower level: 93% vs 80% in the normal evolution scenario.

Less probable characteristics of radionuclide mobilisation and transport: For this alternative scenario, the diffusivity is increased by a factor of two. In terms of hydrological evolution, this scenario is like the normal evolution scenario. However, the higher diffusion rate alters the distribution of the radionuclides from the HLW. While the advective transport of the HLW radionuclides is the same as in the normal evolution, a somewhat large fraction (18% vs 16%) travels through the shaft segments to the upper level of the repository due to the higher diffusion rates. However, no radionuclides are present in the overburden, or the shaft salt backfill.

Reduced long-term sealing by backfill: For this alternative scenario, we assume residual porosities of 2% or 5%, instead of 1% as in the normal evolution scenario. As shown in Figure 8.3, the hydrological evolution of the repository changes in these two cases, due to the higher residual porosity and the associated permeability of the granular salt backfill. As in the normal evolution scenario, there is an initial increase in saturation due to the compaction of the granular salt backfill (a reduction in pore space while the amount of brine remains similar). This initial increase is followed by a short period of decreasing average saturation, and then a subsequent increase in saturation related to brine entering the GDF via the shafts. Compared to the normal evolution scenario, this brine inflow is higher due to the larger residual porosity and associated permeability of the shaft backfill. The relatively higher brine inflow results in the full saturation of the entire repository at approximately 100,000 years (for 2% residual porosity) or 52,000 years (for 5% residual porosity) after closure, compared to about 300,000 years in the normal evolution scenario (1% residual porosity, Figure 8.3). Once the repository is fully saturated and compaction has ended, diffusion becomes the dominant mode of transport, as advective transport ceases. Therefore, in the models with higher residual porosity, diffusion is the dominant mode of transport for a longer period compared to the normal evolution scenario. Since diffusion is a very slow process, essentially all radionuclides (>>99.99%) of HLW and LILW remain within their respective levels. This does not mean that a higher residual porosity is beneficial. With a higher residual porosity, the permeability of the granular salt remains high and hence it does not become as effective as a barrier. For example, the permeability of the backfill in the shaft becomes $1.6 \cdot 10^{-18}$ m² and $5.3 \cdot 10^{-17}$ m^2 for 2% and 5% residual porosity compared to $1.0 \cdot 10^{-19}$ m^2 in the normal evolution. Note that in the case of 5% residual porosity, the shaft backfill will have a higher permeability than the concrete shaft seal until the latter fails.

Based on the results of the safety assessment, granular salt backfill compaction, most especially of the shaft backfill, is important. When the shaft backfill has compacted and attained a low permeability, brine cannot enter the repository, while radionuclides cannot leave it. In addition, both advective and diffusive transport are important with advective transport being the initial dominant processes followed by diffusion.

8.4 Simplification in the safety assessment

The following section highlights assumptions that have been made to simplify the safety assessment. Most of these simplifications consist of not taking credit for potentially positive processes that could enhance safety levels but are not yet sufficiently understood or quantified to allow their use in a robust assessment.
Some simplifications, however, relate to effects that could potentially be negative. The simplifications include the following factors:

Gas generation: It is currently assumed that no gas generation takes place. However, as shown in Chapter 6, gas generation may be unavoidable. While COPERA (2020 - 2025) has carried out some scoping studies on gas generation (Benbow et al., 2023b), it has not yet been studied in sufficient detail to include in the current safety assessment. Its effect could be deleterious to the modelled strong containment of the disposal system because gas generation can delay compaction, although it is expected that the total amount of gas generated in the repository will be limited (See Chapter 6).
This is a non-conservative assumption, and more research is needed to understand its effect. In the Gorleben safety assessment, gaseous radionuclides were found to travel further than those dissolved in brine (Fischer-Appelt et al., 2013).

Two phase flow: In this safety assessment, Richards' equation is used, which implicitly assumes that flow is single-phase and that any gas present in the GDF (including air) can readily escape to the surface. Over time, however, it will become more difficult for gases generated within the GDF to escape, and consequently, gas pockets may form. These pockets could potentially hinder the compaction of the granular salt backfill, leaving parts of the granular salt backfill with high porosity and permeability. On the other hand, a gas pocket (or gas cushion) could also surround waste, which in turn would stop any brine from coming in direct contact with the waste and thus have a positive effect on containment.
As with gas generation, this needs to be addressed in future safety assessments, especially as this assumption could have both negative and positive implications for containment.

Effective diffusion: Molecular diffusion in free water is multiplied by the porosity to obtain effective diffusivity. This is a conservative assumption because effective diffusivity also depends on the tortuosity of a migration path or trajectory, compared to a straight-

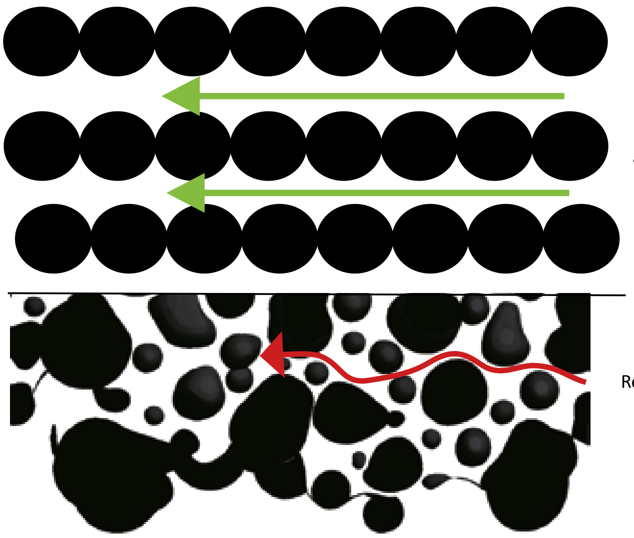


Figure 8.4) A schematic figure of a porous medium with travel paths of radionuclides (red). In the COPERA (2020 - 2025) safety assessment, it is assumed that radionuclides travel in a straight line (green), while a radionuclide must diffuse around the solids (black), increasing the distance it must travel (tortuosity) and hence the time needed to travel. The assumption that the effective diffusivity only depends on the porosity is a conservative assumption.

line distance. The interconnected pores in a porous medium create a convoluted path for diffusion. In the safety assessment, we assume that radionuclides travel in a straight line between pores, which is therefore a conservative assumption (Fig. 8.4).
For example, the diffusivity of the concrete seals is $2.3 \cdot 10^{-9}$ m^2/s in the safety assessment, while experiments show that the effective diffusivity is actually in the order of $1 \cdot 10^{-14}$ m^2/s (Jantschik et al., 2018; Lüthi et al., 2008). Nevertheless, even with this conservative assumption, radionuclides will not leave the repository within one million years. More realistic lower values will further reduce the likelihood of releases. Furthermore, it is assumed that effective diffusion does not depend on saturation, whereas, when material is only partially saturated, effective diffusion will be less.

Healing of granular salt backfill: In this safety assessment, only stage two of the compaction of the granular salt backfill is modelled. Consequently, it is implicitly assumed that the pores in the granular salt backfill will remain connected throughout all the periods considered in the safety assessment, with both advection and diffusion assumed to be possible at all times. However, it is expected that, eventually, all pores in the granular salt will become disconnected due to the healing of the salt (stage 3), thus trapping any mobilised radionuclides in the processes. While there is still a large uncertainty on the time required for granular salt to compact enough that the pores become disconnected (phase 3), based on experiments it is currently estimated to be within a few thousands of years (Grupa and Houkema, 2000; Houben et al., 2013; Koelemeijer et al., 2012). Assuming pores in the granular salt remain connected throughout the safety assessment is conservative.

Release rate from the waste: Instant post-closure mobilisation of radionuclides is assumed for both LILW and HLW. This is modelled by assuming that radionuclides are evenly distributed throughout the disposal room or tunnel directly after closure of the repository (LILW) or after failure of the HLW waste package. For the LILW, this is a conservative assumption, as it will take time for the concrete to disintegrate and for the radionuclides to dissolve (See section 6.3.4.1). In the previous long-term safety assessment for a repository in rock salt, for example, it was assumed that the concrete waste matrix would provide containment for up to 30 years (Prij et al., 1993). For HLW, this is also a conservative assumption, since vitrified HLW has a very low dissolution rate.

No solubility limits and no sorption: Elemental solubility limits are not applied to the radionuclides from the wastes and consequently the safety assessment assumes all radionuclides are immediately in solution and able to migrate. This is a conservative assumption. While less important in a repository in rock salt, assuming no sorption is also a conservative assumption. For some radionuclides, solubility limits can be important, especially in alternative scenarios when radionuclides could potentially leave the repository. Therefore, as part of COPERA (2020 – 2025) a model has been developed to calculate the solubility of radionuclides in brine (Oving and Grupa, 2024; Oving and Meeussen, 2024), and this will be used in the next performance assessment.

Compaction model: A simple salt backfill compaction model is used, based on experimental work. This does not explicitly consider the interaction between the host rock and the backfill. A similar approach is taken in LOPUS, a computer model used for safety assessments in German and previous Dutch safety assessments. Compaction rates from the German study are of the same order of magnitude as in this study, and both are in agreement with experimental work (Oosterhout et al., 2022). This is therefore probably a neutral assumption. However, compaction models that explicitly consider the interaction between the gas and backfill need to be developed, although these types of models will be computationally demanding.

Van Genuchten parameters for granular salt backfill: The Van Genuchten parameters are used to model the relationship between brine content and crystalline salt behaviour in partially saturated backfill, allowing estimation of the behaviour of granular salt back-fill that has not been treated in any way. Whether these parameters can be used for compacting granular salt is currently unknown. While the Van Genuchten parameters do affect compaction and advective transport, the effect on the model results is expected to be limited. This is because the period in which advection transport is significant is limited (< 1,000 years). This is therefore a neutral assumption.

Moisturised salt: In this safety assessment, only moisturised granular salt backfill is assumed to be used in this safety assessment. Consequently, advective transport is the dominant mode of transport in the lower level resulting in the HLW tracer being transported to the upper level and shaft. However, for dry granular salt the compaction rate differs significantly (Spiers et al., 1988), since more pore space will be available for brine from the moisturised granular salt backfill to flow into. This will result in faster compaction of the latter. Thus, using mostly dry granular salt with moisturised granular salt backfill only in specific places, would limit the transport of radionuclides even more. Indeed, scoping models suggest that radionuclides from the HLW will remain entirely in the lower level. This is supported by Bartol et al. (Submitted) who showed that by using dry granular salt in the disposal tunnels, radionuclides will travel less far, which is in agreement with the safety assessment of Gorleben (Fischer-Appelt et al., 2013). Therefore, assuming that all salt will be moisturised is considered neutral and possibly conservative.

8.5 Comparison with other safety assessments

8.5.1 Comparison with previous Dutch assessments

In the Netherlands, 3 salt safety assessments have been performed in the past, namely VEOS (Commissie Opberging te Land, 1989), PROSA PRObabilistic Safety Assessment (Prij et al., 1993) and CORA (Grupa and Houkema, 2000). Here, we only highlight the two most recent salt safety assessments, as PROSA was based on the results of VEOS (Prij et al., 1993). These safety assessments were performed for different disposal concepts with different types and volumes of waste, compared to the current disposal concept, but they all share the same safety concept.

In PROSA, an extensive FEP analysis was undertaken, which resulted in three different families of scenarios (Prij et al., 1993): subrosion, flooding and human intrusion scenarios. In the subrosion scenarios, the host rock is gradually dissolved by groundwater, while diapirism reduces the distance from the repository to the surface and results in higher subrosion rates. Eventually, the impermeable salt barrier surrounding the waste is dissolved entirely and groundwater encounters the waste. Results of the PROSA safety assessment show that the first exposure is expected in this group of scenarios ("early release") only after 1.5 million years. This supports the results of the present safety assessment, which shows no release of radionuclides within the assessment period of one million years. It should be noted that in the PROSA subrosion scenario group, "early" release only occurs in salt structures that have a diapirism rate of more than 0.2 mm/a: a factor of two higher than observed in the Netherlands (see section 5.2.4.1). For more realistic values of diapirism rates (< 0.2 mm/a), Prij et al. (1993) showed that at least 4.5 million years is needed for radionuclides to reach the near-surface. Note that the subrosion rate is assumed to depend on depth.

In the second, water intrusion group of scenarios, the host rock is intersected by an undetected water-bearing formation (an anhydrite layer) or a large fracture. Consequently, groundwater starts to enter the repository directly after closure and inundates it in 138 years, encounters the waste and is subsequently squeezed out of the repository due to the convergence of host rock and compaction of the backfill. Eventually, the backfill will become impermeable and the release of radionuclides from the repository stops. Prij et al. (1993) showed that the maximum dose rate would occur, at the earliest, 0.5 million years after the closure of the repository. The calculated maximum dose rates are small (at least six orders of magnitude lower than the current natural background radiation exposure in the Netherlands), since only a small portion of the initial inventory will be released. Thus, even if the natural barrier is impaired in this way, the consequences will be limited. Furthermore, this group of scenarios is unlikely, as a large fracture or water-bearing formations would not go unnoticed during the very detailed characterisation of a salt structure. Moreover, the salt creep together with self-healing properties of rock salt mean that it is unlikely that a fracture would remain open for a long period in a salt structure. These results suggest that the dose rate for the shaft failure scenario, not modelled here, will likely be limited, although both the amount of and type of waste is different in the PROSA safety assessment compared to the COPERA analysis.

While the focus in PROSA was on the long-term containment behaviour of a repository in salt, in CORA a scenario was modelled based on an abandoned, unsealed repository (Abandonment scenario, Grupa and Houkema, 2000). In this scenario, maintenance of the repository stops during the operational phase, for no specific reason, at a time when the seals within the repository are not water-tight and the remaining open spaces are not filled with granular salt backfill. Thus, the only functional engineered barriers are the waste form, the disposal overpack and a 1.5 m salt plug in a horizontal borehole separating the waste from the rest of the repository (Heijdra and Prij, 1997). After flooding, contaminated brine is squeezed out of the repository due to the convergence of the host rock and compaction of the salt plug. About 4,000 years after flooding of the repository, the salt plug become impermeable, halting the further release of radionuclides. The calculated maximum individual annual dose was still estimated to be about six orders of magnitude lower than the present-day background radiation in the Netherlands. As with the PROSA water intrusion scenario, this scenario essentially demonstrates the robustness of a repository in salt. Again, this suggest that the consequences of the shaft failure scenario, which is not modelled here, are likely to be limited.

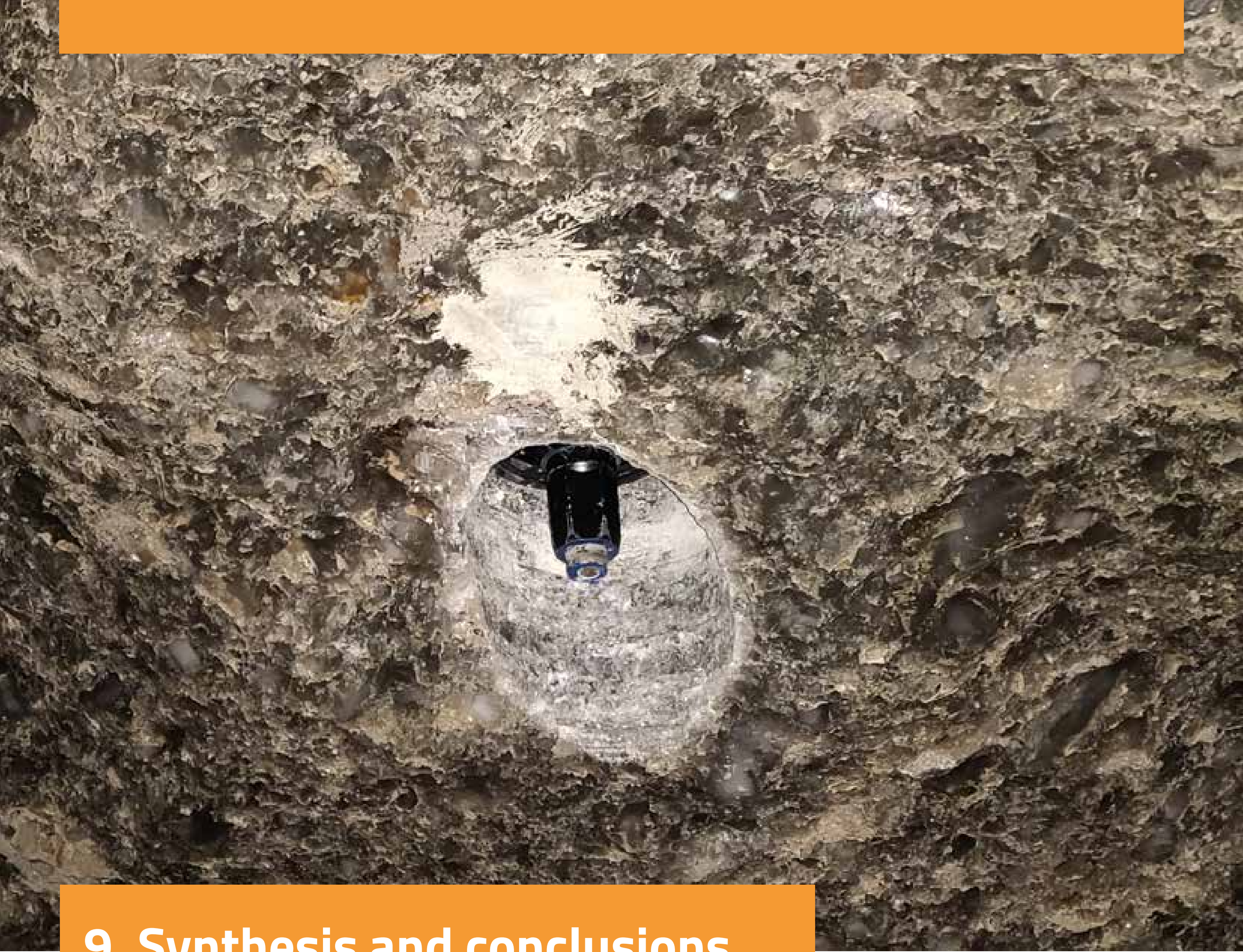
8.5.2 Comparison with international safety assessments

A safety assessment of a generic salt dome repository was performed within DECOVALEX 2023 (LaForce et al., 2023). The primary objectives were to build confidence in the models, methods and software used for performance assessment. Five organisations participated, from four countries. Each team developed a different modelling strategy. For example, compaction was modelled in four different ways, resulting in different compaction rates for the granular salt backfill. Despite differences in the modelling strategies, all models indicate that salt compaction and radionuclide diffusion are the key processes. Furthermore, none of the models indicated that radionuclides would leave the repository or enter the shaft connecting the repository with the surface during a period of 100,000 years (LaForce, 2024a). Similar results were also obtained in a separate independent study of a generic repository in bedded salt by Bollingerfehr et al. (2018b). Both safety assessment exercises have a similar safety concept to that modelled here for the COPERA (2020 - 2025) disposal concept.

Likewise, the safety assessment for the proposed Gorleben repository, assuming single phase flow and transport, as done in this safety assessment, showed that no release is expected in both the normal evolution and alternative scenarios, including the shaft failure scenario (Bracke and Fischer-Appelt, 2013; Fischer-Appelt et al., 2013). The Gorleben safety assessment also showed that there is too little brine available to corrode the containers or dissolve the containers, and that radionuclides do not travel far, or at all. Likewise, no gaseous radionuclides were released through a shaft seal in the two phase flow model, although they were able to travel further within the repository compared to radionuclides transported in the fluid phase (Bracke and Fischer-Appelt, 2013; Fischer-Appelt et al., 2013). One of the key differences between COPERA and the Gorleben safety assessment is that, in the latter, radionuclides do not travel far or at all in brine, while in this safety assessment, radionuclides from the HLW migrate to the upper level and enter the lower parts of the shaft. This difference is a result of the amount of brine modelled as being present in the repository from the start. In our model, we assume that all granular salt backfill is moisturised. This results in significant advective transport of radionuclides. In contrast, in the Gorleben safety assessment, most granular salt will be as dry as possible. While some brine will still be present in the backfill (about 0.02 wt.%), this is significantly less compared to the moisturised backfill assumed in COPERA; this results in slower compaction but also in radionuclides traveling less far, especially when dry backfill is used in disposal areas (Bracke and Fischer-Appelt, 2013; Fischer-Appelt et al., 2013). Note that, in the granular salt backfill between the two concrete seals, the Gorleben study also assumed that moisturised granular salt is used to increase the compaction rate.

8.6 Conclusion

The COPERA (2020 - 2025) safety assessment has taken a simple and largely conservative modelling approach that adopts a similar methodology and assumptions to those of other international exercises. The approach captures the widely accepted, most critical processes of advection, diffusion and compaction that control the behaviour of a GDF in salt. Based on the results of this safety assessment, it is concluded that the engineered and natural barriers will provide complete containment during the assessment period of one million years, despite some significant conservative assumptions being included in the modelling. Even with these conservative assumptions, there is no release expected in the normal evolution scenario during the first million years after closure. The same applies to the normal evolution and five alternative scenarios modelled. However, more research is needed on the alternative scenarios that have not been modelled here, as these could potentially result in releases during the first million years after closure. Over much longer periods (several millions of years), releases are likely to occur eventually in the normal evolution scenario in locations where there is a significant combined effect of subrosion and diapirism, if these rates are high. However, by such times the hazard potential of the waste has reduced to levels well below those of natural uranium ore deposits. In conclusion, in the normal evolution scenario, and for at least one million years after closure, what is placed in salt stays in salt.



9. Synthesis and conclusions

The COPERA (2020 - 2025) research programme has been carried out to progress and support national policy, which calls for eventual geological disposal of all Dutch long-lived radioactive wastes in either rock salt (discussed in this report) or in poorly indurated clay (Neeft et al., 2024b). Building upon previous research programmes such as OPLA, CORA and OPERA, COPERA (2020 - 2025) marks the start of a new and continuous long-term research programme (Verhoef et al., 2021). Here, we provide a synthesis of the scientific and technical aspects of disposal in salt, our current conclusions, and our plans for future developments. It should be noted that at the start of COPERA (Verhoef et al., 2021), implementation of an operational disposal facility was foreseen in 2130 (Ministry of infrastructure and the Environment, 2016). Recent documents preparing the next update of the national programme have not amended this date. Site specific safety cases are therefore assumed to be needed only after 2050 (van Gemert et al., 2023).

This conditional safety and feasibility study for a geological disposal facility (GDF) in salt has gathered, collected and integrated a substantial amount of existing information from previous research programmes, and from our international partners. In addition, new studies have been carried out into the disposal of the Dutch inventory in rock salt as part of the first phase of the COPERA (2020 - 2025) research programme.

9.1 Aims of COPERA

The overall aims of the COPERA (2020 – 2025) research programme and the safety case are:

- To show that appropriate engineering designs for a GDF can be developed which would be feasible for construction at depths of over 700 m in rock salt, and specifically in a Zechstein salt dome, in the Netherlands.
- To implement a state-of-the-art methodology for producing a conditional safety and feasibility study for a GDF in salt, using the capabilities of national expert organisations in the Netherlands and abroad.
- To use current best practice safety case methodologies to show that long-term safety of the disposal of all the wastes in the Dutch inventory is achievable
- To use the results of engineering and safety evaluations to identify further R&D needs that will progressively enhance design and underpin future iterations of the safety case.
- To communicate the approach to implementing geological disposal and to assess its safety in a comprehensive set of documentation, including this high-level overview, which is intended to be transparent to all parties.

To proceed with the phased development of a geological disposal programme in the Netherlands, COPERA (2020 - 2025) has gathered and integrated a considerable amount of existing information and carried out a wide range of new studies on one possible salt GDF concept, which is designed to contain the expected inventory of radioactive wastes that will arise over the next decades and is aimed to be constructed in a Zechstein salt dome. However, many aspects of the concept are expected to be transferable to any salt structure or salt deposit.

The COPERA (2020 - 2025) programme is conditional because it is not site specific and it is based on a simple outline design for a feasible GDF concept. The evaluation contains some significant areas of uncertainty. The experience gained in the current study will be used to guide future work that will progressively address these conditional factors and refine the practicalities. This will allow the conditional safety and feasibility studies to become more refined over time. Nevertheless, it is possible to say that a combination of Dutch work and broad international experience in construction and operation of GDFs and other structures in rock salt indicate that a solution for the Netherlands is entirely feasible, as discussed below.

9.2 Feasibility of constructing a GDF in a salt dome

The COPERA (2020 - 2025) GDF concept is based on the well-developed German concept for disposal of HLW in salt domes and the operational WIPP repository in bedded salt in New Mexico, USA. It also builds on the previous Dutch concepts by considering all the types of waste in the Dutch inventory and the need for retrievability of these wastes.

Geotechnical assessment within the COPERA (2020 - 2025) research programme indicates that a stable and robust two-level GDF can be constructed and operated in a salt dome at depths of greater than 700 m, with the model adopted for COPERA (2020 - 2025) having disposal levels at 750 and 850 m depth. For the construction of the GDF, existing salt mining techniques and equipment (e.g., continuous miners and scalers) can be used. There are decades of practical experience in both commercial salt mining and in constructing an actual repository for radioactive wastes. There is thus abundant experience that can be used.

Existing international studies already show that there are practical techniques for sealing tunnels and shafts in a GDF. It is expected that considerably more progress and operational experience will become available over the next 100 years, well before these techniques need to be deployed in the Netherlands.

Overall, there is considerable scope to adapt and optimise the engineering design of the GDF in future years and it is expected that the eventual design will be significantly further developed from the current COPERA (2020 - 2025) concept.

9.3 Feasibility of siting a GDF in rock salt

COPERA (2020 - 2025) is not a siting study. However, it is important to have confidence that suitable locations for a GDF are available in the Netherlands, were rock salt to be selected as the preferred host geological formation. Rock salt is present in an appropriate depth range in parts of the north-east and east of the Netherlands. However, the depth and thickness of the Röt Formation and other salt deposits in the Netherlands need to be studied further. Uncertainties also exist on the internal structure and the purity of the salt in these salt deposits. More research is needed to reduce these uncertainties. Nevertheless, it is expected that the eventual GDF design can be adapted to match the specific properties of many potential locations, thus allowing flexibility in depth and layout aspects that are not critical to safety. Owing to the limited amount of data, no attempt has yet been made to optimise appropriate depths and thicknesses. This will be done when more data become available.

A siting programme will need to avoid certain geological structures and features. When a salt structure is selected, it needs to have good containment properties. Factors that need to be considered are, for example, the size of the dome, the overburden, the diapirism rate, the subrosion rate, the homogeneity of the salt et cetera. Guidelines and criteria for doing this will need to be further developed.

It is recognised by COVRA that siting a GDF involves considerably more than evaluating technical factors. Any future siting programme must take societal requirements into account and needs to be phased, progressive and consensual in nature.

9.4 The objective and design of a GDF

The COPERA (2020 - 2025) concept is for a GDF that will contain all high-activity and long-lived radioactive wastes that are currently in storage in the Netherlands or are likely to be generated over the next 100 years.

The safety concept for this GDF aims to isolate and contain the radioactivity in these wastes so that they do not present a hazard for people or the environment in the future. The hazard potential of the wastes (their capacity to cause harm if people encounter them) is initially high. However, it diminishes rapidly over the first hundreds of years after they are placed in the GDF, followed by a slower decrease in the next thousands of years. The safety concept thus places emphasis on assuring complete isolation of the wastes over the early period, knowing that small amounts of radioactivity might migrate into the surrounding geological and surface environment in the far distant future, due to degradation of the GDF through natural processes. The multiple safety barriers in the concept ensure that any releases will be so small as to be harmless to future generations.

The GDF system comprises the 'engineered barriers' of the GDF itself, situated in the 'natural barrier' provided by the rock salt and the surrounding geological formations. The engineered barriers comprise solidified waste matrices in a variety of containers, surrounded by backfills of granular salt or salt / sorel concrete seals and, in the case of high-level waste, a carbon steel waste package. Understanding of their properties and behaviour, including that of the host rock, is central to the safety case.

The most highly active wastes (vitrified HLW, fuel assembly debris and spent research reactor fuel: SRRF) are contained in a HLW package, designed to facilitate emplacement of the wastes and to provide radiation shielding during operations and complete containment for at least 1,000 years after closure, when the granular salt backfill still has a relative high permeability, but it will likely function for much longer.

Until such time as the GDF is close to construction, options for design and operation will remain open and their feasibility will continue to be evaluated and compared, so that an optimised solution develops progressively.

9.5 How the COPERA (2020 - 2025) GDF is expected to perform

As noted above, the most critical time over which the performance of the GDF system must be assured is the first few hundreds to a few thousand years, owing to the initially high hazard potential of the wastes.

However, assessments look much further into the future and consider how the GDF will continue to perform for tens and hundreds of thousands of years. Eventually, anticipated changes in the natural environment, but also uncertainties in the properties of materials, makes quantitative estimates of future performance less reliable. Nevertheless, in common with other international safety cases for geological disposal, we consider potential environmental impacts over the next million years. At such long times, it is appropriate to use other indicators of performance rather than calculated radiation doses to far-future humans.

COPERA (2020 - 2025) has assessed how we expect the GDF system to evolve over these long periods and has taken a conservative approach to modelling the behaviour of radioactivity in the system. This approach involves making pessimistic assumptions about the system behaviour of the engineered barriers and omitting some potentially beneficial processes from the assessment.

The expected behaviour is that the host rock, in combination with the engineered barriers, will provide total containment of the radioactivity inside the GDF for at least one million years and probably for even longer. This is because the host rock is impermeable, and the granular salt backfill used to fill most of the open spaces is also expected to become impermeable within 1,000 years. Before this time, concrete seals will ensure containment. Within one million years, the hazard potential of almost all wastes will be less than that of a natural uranium ore body.

9.6 What the COPERA (2020 - 2025) salt safety assessment shows

The normal evolution scenario on which the safety assessment calculations for COPERA (2020 - 2025) are based uses a mix of 'best estimate' parameter values, for which variability and uncertainty are reasonably quantified. Conservative (i.e., pessimistic) parameter values are used when there is not a solid basis for setting a best estimate. A conservative value allocates low or zero effect to a beneficial containment property. Conservative assumptions are made not only about parameter values, but also with respect to some processes that are not well understood.

Even using pessimistic approaches, the performance assessment calculations show that people in the future are highly unlikely to be exposed to radiation by materials originating from the repository-All radionuclides are expected to remain within the repository for at least one million years after closure. This is supported by other, independent safety assessments for repositories in rock salt.

The calculated impacts for the normal evolution scenario are thus well below typical, internationally accepted, radiation protection constraints for members of the public.

A key observation is that, within a few hundred thousand to a million years, almost all the radioactivity initially in the GDF has decayed within the GDF itself. The GDF has effectively performed its isolation and containment task by this time and will likely do that for millions of years to come.

9.7 Conservatisms and open issues in the COPERA (2020 - 2025) conditional safety and feasibility study

As noted previously, the normal evolution scenario on which the present safety assessments calculations are based contains several conservatisms. The main conservatisms are:

- All waste packages for HLW fail at the same time.
- The lifetime of the HLW packages is limited to 1,000 years.
- There is instant mobilisation of the radionuclides in all the LILW (after repository closure) and in the HLW (after HLW package failure).
- The granular salt used as backfill in the most critical parts of the GDF remains permeable throughout the assessment period.
- Molecular diffusion in free water is assumed, instead of considering pore-space tortuosity, resulting in a diffusivity for mobilised radionuclides that is significantly higher than would be expected.
- No waste or radionuclide solubility limits are applied, and no sorption is assumed to take place on any materials in the disposal system.

At the same time, it is acknowledged that several processes and events that might lead to greater predicted impacts have not yet been addressed in this stage of COPERA (2020 - 2025) and thus constitute open issues that will require further R&D and safety assessments. These include but are not limited to:

- A full assessment of alternative evolution scenarios that might lead to behaviour different to that of the normal evolution.
- Assessment of how radioactivity might move in a gas phase (e.g., as radioactive carbon in methane) or how the presence and behaviour of a gas phase generated in the GDF (such as hydrogen, from the corrosion of metals) could affect the salt backfill compaction.

9.8 Other evidence underpinning confidence in safety

Natural and archaeological analogues of materials preservation in salt show that degradation processes can be much slower than typically modelled in safety cases. The preservation of artefacts in salt mines is a good example of how rock salt can preserve materials. The stability and impermeability of salt formations is demonstrated by their ability to provide a seal for many hydrocarbon deposits, which have been trapped and contained by the salt since the Late Jurassic, i.e., for more than 150 million years.

At a broader scale, natural radioactivity, present in all rocks, soils and waters around us, provides a useful yardstick against which to compare potential impacts of wastes in the GDF. Natural radioactivity levels in the Netherlands are typical of those across Europe and the unavoidable natural radiation exposures to which we are all subject are much higher than those estimated to result from any releases from the GDF, even for the most pessimistically calculated cases. We live in, and humankind has evolved in, a naturally radioactive environment.

In the far future (many millions of years), the degraded GDF, with its considerably reduced radiotoxic hazard, will have similar properties to a uranium ore body, containing mainly the residues of the depleted uranium wastes. It will either become more deeply buried and isolated in Earth's crust by further deposition of sediments, or it will be eroded away by natural processes, with its contents being distributed among and becoming part of the natural radioactive background.

Confidence in the reliability of our performance assessment calculations is enhanced by the fact that they broadly agree with those estimated independently for a wide range of wastes and host rocks, in other national programmes.

9.9 Improving the design and the safety case

The principal uncertainties that have been identified as work progressed in each of the COPERA work packages will be addressed by future COPERA studies. The main areas identified for further work are:

- Improving knowledge of the lithological, geotechnical, mineralogical and geochemical properties of rock salt.
- Collecting data on subrosion and diapirism rates in the Netherlands with a higher time resolution.
- Increasing knowledge of the internal structure, lithology and extent (depth and thickness) of salt deposits.
- Developing models that simulate the evolution of salt structures through time.
- Understanding better the long-term evolution of the porosity and permeability in the granular salt back fill.
- Improving understanding of the interaction between gas and the compaction of the granular salt backfill.
- Designing a conceptual shaft and tunnel seal.
- Developing models that simulate granular salt compaction at repository scale and include the interaction between the backfill and the host rock.
- Definition and evaluation of alternative GDF design concepts that might be suitable for rock salt.
- Performing analyses of additional alternative scenarios, especially of reduced long-term sealing by backfill and of shaft seal failure.
- Developing a scientifically and societally base approach to identifying possible siting areas and locations for a GDF.
- Establishing mechanisms for knowledge maintenance and transfer over the decades and generations leading up to eventual disposal.

Clearly, not all this work is required in the next decades. It is expected to be staged over several iterations of the future COPERA (2020 - 2025) programme. An early task is to prioritise and schedule this work. This is discussed further in Chapter 10.

9.10 Looking forward

The information produced by the COPERA (2020 - 2025) programme can be used to support national waste management policy development in the Netherlands and to provide a more reliable basis for establishing future financial provisions for waste management. In particular, the availability of a conditional safety assessment approach and a reference case for both a salt and a clay GDF allows COVRA to make disposability assessments of any future waste arising, or packaging proposals from waste producers.

Major programmes such as COPERA have been completed in the past in the Netherlands (OPLA, CORA, OPERA), but there has been no continuity to maintain expertise. This situation needs to be avoided in future, and it was therefore decided to start the long-term COPERA (2020 - 2025) research programme, of which this report presents one of the results of the first phase. This first phase provides a strong launching point for a planned programme of technology development, and national knowledge management for the future, together with continued cooperation with national and international waste management initiatives. The final Chapter 10 below, outlines a roadmap for future COPERA work.

Finally, we note that the COPERA project team is fully aware that a successful GDF programme must address both technical and societal issues. Globally, the greatest obstacles to geological disposal have been those related to achieving sufficient public and political engagement and support. COPERA (2020 - 2025) has initiated work on communication with the Dutch public, to which this report is a contribution, and this important activity will be continued in future projects.

10. Roadmap for the future Dutch GDF programme

COVRA's 5-year rolling research agenda is aligned with the review cycles of the national programme on radioactive waste (NPRA) by the Ministry of Infrastructure and Water Management (Ministerie van Infrastructuur en Milieu, 2016). Accordingly, the publication of this safety case and feasibility study coincides with the ten-year review cycles of the NPRA.

While the nuclear landscape has been stable in the last decades, it will change drastically in 2035 if all proposed nuclear plants, are realized. Consequently, the volume and the characteristics of waste would change, especially if the spent fuel from the nuclear power plants were not to be reprocessed. Currently, the target date of 2130 for emplacement of all radioactive waste in a GDF has been left unaltered in the public consultation about the scope of the NPRA (van Gemert et al., 2023). This roadmap for a future GDF in rock salt, like that for a GDF in poorly indurated clay (Neeft et al., 2024b), therefore assumes a long-term storage period of slightly more than 100 years. If the target date of 2130 for the emplacement of all radioactive waste in a GDF changes to a much earlier date, research into disposal should significantly accelerate and should include more site-specific research.

10.1 Drivers for the COVRA GDF programme

The roadmap is aligned with the decision-making process on geological disposal of radioactive wastes. Choices and decisions are made in the development of a disposal concept over a very long period. The argumentation for these choices and decisions must be traceable to validated documentation and research, even after many years during which scientific and societal insights may have changed. Also, the right of autonomy and self-determination implies that crucial information must not be withheld from future generations. Furthermore, knowledge about the waste generated and the future GDF must be kept alive and accessible. Therefore, to support the decision-making process, robust and consistent knowledge management is necessary. An essential part of the knowledge management is an active, continuous research programme on geological disposal.

The long period of above-ground storage will provide time to learn from experiences in other countries, to carry out research and to accumulate the knowledge to make a well-founded decision.

A choice for GDF location or host formation can only be made after a decision on the disposal method, and the research up to then will remain at a conceptual level. COVRA will make conditional generic (i.e., non-site-specific) safety cases during the next decades. In this period, the principal driving forces for research are to:

- Strengthen confidence in the safety of disposal: investigating the different host rock options and potential GDF design options, the post-closure performance and level of public confidence and acceptability.
- Ensure that adequate funding for disposal is made available by waste producers, based on regularly updated cost estimates for the GDF: identifying and, where possible, optimising cost-determining features of a GDF.
- Assess the disposability (see Box 10-1) of different waste and waste packaging families: investigating waste packaging options and requirements on collection, treatment and conditioning of waste families to facilitate their eventual disposal.

Steering research using the safety case: COVRA is responsible for the development of the safety cases as an instrument to steer research and manage knowledge over decades. Conditional safety cases will be developed and periodically updated for a GDF in rock salt. As components of the safety cases, COVRA will carry out performance assessments to assess the relevance of knowledge and research for the post-closure safety of the GDF in the different host rocks considered in the Netherlands.

Knowledge infrastructure and community: Many research entities and researchers in the Netherlands and abroad have contributed to this first phase of COPERA. Meetings (online and in-person) were organised to share experience and new knowledge, and to foster different insights and approaches to problems. In the future, COPERA will continue to rely on national research entities as well as disposal programmes from other countries to improve understanding of processes taking place in the post-closure phase and to carry out experimental investigations and provide input data for these assessments. COVRA will also participate in international groups such as the OECD-NEA Salt Club, and in European and international projects (e.g., DECOVALEX). COVRA will encourage organisations involved in future research to share their work and experience in international fora and scientific journals. In this way, the necessary knowledge, management, infrastructure and community can be maintained over the long-term.

10.2 Key topics for further research

Previous chapters of this report have discussed current knowledge and the properties and performance of individual components of the multibarrier system (host rock and engineered barriers, Chapters 5 and 6), along with their contributions to overall safety (Chapters 7 and 8). Based on this evaluation, we have identified the primary points of focus for future research. The results, including an updated cost estimate, will be published in a summary report in 2030. This follows the cycle of the long-term research programme in which an update of the cost estimate and a summary report will be published every five years, followed five years later by a generic (non-site specific) rock salt conditional safety and feasibility study. The next generic (non-site specific) conditional safety and feasibility study will thus be published in 2035 to ensure that the safety cases are ready before the Ministry reports its updated national programme on radioactive waste (NPRA) to the European Commission. The publication of the conditional safety and feasibility study is thus aligned with the review cycles of the NPRA. The next generic (non-site specific) conditional safety and feasibility study will include a safety assessment and a cost update (Fig. 10.1).

The programme's budget, the timeframe of the work programme and the capacity of the programme office requires the setting of priorities for the research activities. COVRA has therefore

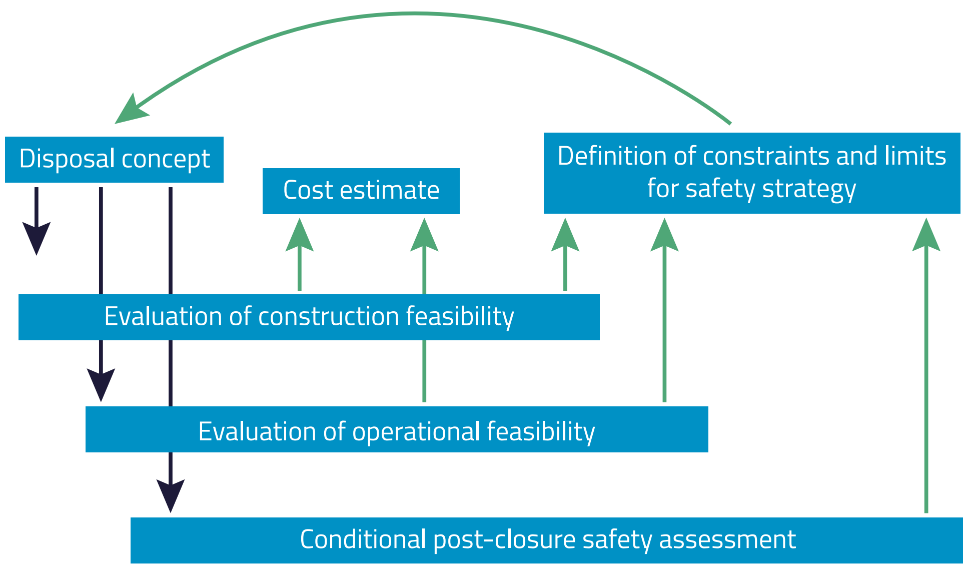


Figure 10.1) Cycle for updating the disposal concept and costs. Figure from Verhoef et al. (2021).

developed a framework to assess and assign priorities to each task in the COPERA research programme. For tasks related to the geological disposal system, this prioritisation is based on the three drivers for research presented in the previous section: confidence in safety, assuring adequate funding and disposability of the Dutch inventory. In addition, there are unique opportunities to participate in co-funded (international) research activities that allow for efficient knowledge gathering, even though their priority may be lower. In those cases, participation is considered to support our long-term strategic aims and is therefore important to fund. Figure 10.2 illustrates these key research topics for each component of the disposal system, and a more detailed description of them is provided below.

10.2.1 Biosphere

Current knowledge on the biosphere is sufficient, since the Netherlands disposal planning is at a conceptual stage. However, a study will be required on the origin and transport of naturally occurring radionuclides in drinking water in the Netherlands, to verify the models employed in the safety assessment calculations.

All Dutch research programmes have looked only at the impacts of radioactive elements that might move to the biosphere. There are also chemically toxic elements in the waste materials that could have health effects if they migrate to the biosphere, and this requires evaluation. The evaluation of the potential impacts of chemically toxic materials in the engineered and natural barriers is expected to be performed when the radiological exposure scenarios have been completed.

10.2.2 Surrounding rock formation

The GDF host rock is surrounded by other rock formations. The main safety function of the surrounding rock formations in any disposal system is isolation. Changes in the surrounding rock formation, however, can affect the host rock.

Tunnel valley formation (Priority 3): Research is needed to understand better the formation of these channels and whether a larger (or smaller) maximum depth could potentially be reached in future glaciation scenarios. An improved understanding of their formation will help to optimise the design of the repository and potentially reduce the cost. Research should not only focus on how these tunnel valleys are formed, but also when, and under what circumstances, as recent studies suggest that there will be a prolonged warm interglacial period.

10.2.3 Host rocks

Priority should be given to confirming the main assumptions underpinning the salt GDF safety concept and to improve and increase our knowledge of this host rock and its distribution in the Dutch underground.

Geotechnical properties (Priority 1): Numerical models will play a key role in demonstrating the long-term safety of a GDF in rock salt. In the first phase of the COPERA (2020 - 2025) research programme, readily available rock salt data from the Netherlands, Germany and the USA have been collected and made available through the COVRA website (Hunfeld et al., 2023). There is, however,

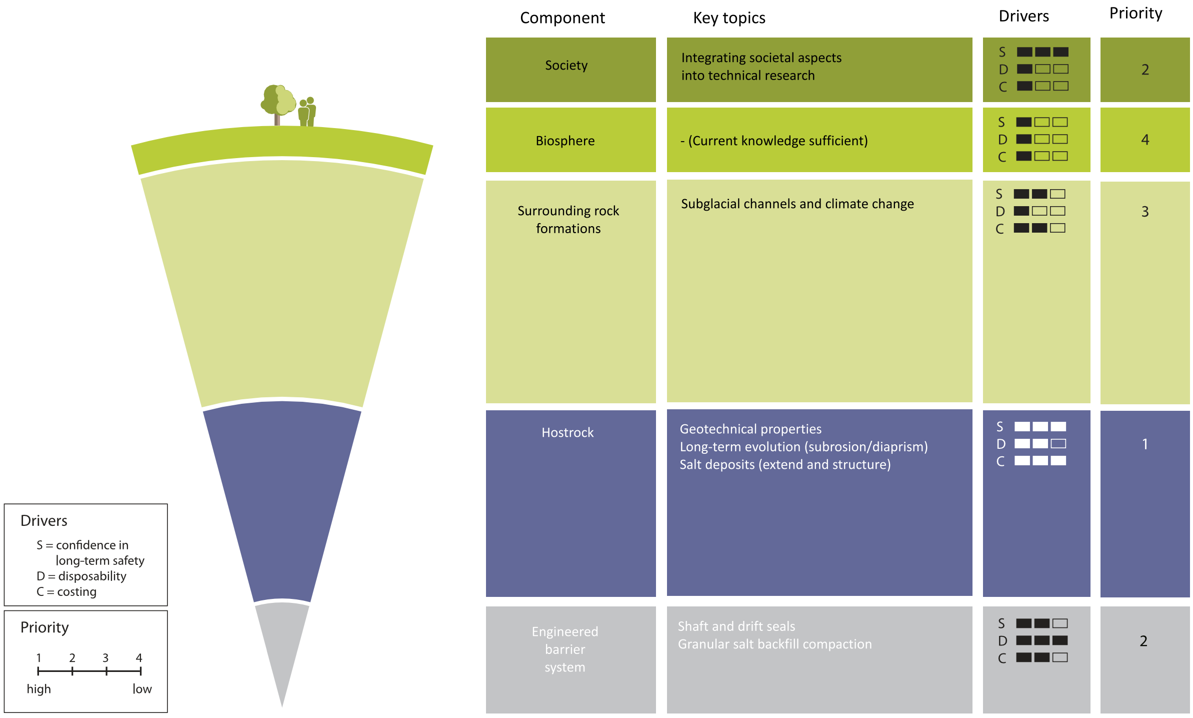


Figure 10.2) Key topics for research into geological disposal in salt, organised according to the components of the multi barrier system.

still a wealth of data that can be collected from peer-reviewed journals and reports, and from experiments on salt cores currently stored at the TNO. Expanding the salt database helps to reduce uncertainties and optimise the design of the GDF, potentially leading to an overall reduction in cost. As this work would be useful for other countries that are considering disposal in salt, it would greatly benefit from becoming an activity of the NEA salt club.

Internal structure (Priority 2): The internal structure of salt sills and domes (see section 5.2) is poorly understood, in part because it is challenging to image salt structures (especially salt domes) seismically, but also because it is difficult to interpret these data, and the number of available salt cores is limited. These uncertainties are recognised and need to be addressed by using newly acquired, detailed 3D seismic mapping and borehole data (logs and cores), when available. Having a better understanding of the internal structure of the host rock will help to optimise the disposal concept and reduce uncertainty and, possibly, cost. Understanding the internal structure can also help to increase our understanding of the geological history of a salt structure and possibly make better predictions about its future.

Diapirism rates (Priority 1): While COPERA has made a conservative assumption on diapirism rates, these are averages over tens of millions of years and thus fail to capture potential short term variations in rates that might occur due to differential surface loading during ice ages (Lang et al., 2014). For the long-term safety of the GDF, diapirism rates in the Netherlands should be better quantified, with a higher time resolution.

Subrosion rates (Priority 1): The spread in subrosion data is large, and the estimated subrosion rates used in COPERA are an average over a relative long period (millions of years). For the long-term safety of the GDF, subrosion rates in the Netherlands should be better quantified, with a higher time resolution, which is important to understand the influence, for example, of an ice age but also other geological or climate-driven processes such as sea level change.

Evolution of a salt structure (Priority 1). Having collected data on past subrosion and diapirism rates, and the internal structure and THM properties of salt, the next step is to develop a numerical model to simulate the evolution of salt structures through time. This numerical model should be able to reproduce the past subrosion and diapirism rates and should include all the major processes and factors (e.g., ice age, ground water flow, salinity of ground water) that have, or could have, an influence on the evolution of a salt structure. Having a better understanding of the evolution of salt structures through time will help build confidence in the long-term safety of the repository.

Salt deposits (Priority 2). The thickness of the most promising Zechstein Group (northern-eastern) and Röt formation (eastern) part of the Netherlands has been well documented, but in other places available data are limited. In addition, data on the salt deposits in the Muschelkalk formation, Keuper formation or Weiteveen formation are limited. Although considered unlikely, it is not currently known whether salt deposits in one of those 3 formations might be suitable for disposal. A desk study should be undertaken using a combination of seismic data and publicly available cores to map the thickness and depth of the different salt deposits in more detail and to assess the quality of the salt.

Brine availability (Priority 1): For a better understanding of the availability of brine, in combination with the evolution of the EDZ, experiments have been performed in the WIPP facility, in New Mexico (USA) by the US Department of Energy. As part of DECOVALEX 2024, work package BATS2 aims to reproduce these experiments using numerical models to understand the underlying processes that result in brine migration formation. An improved understanding of the short-term evolution will help to improve understanding of longer, safety-assessment relevant timescales (i.e., thousands of years). These improved representations are expected to be a tool to assess the total amount, timing, and distribution of brine inflow to a repository, and therefore inform the design of the GDF and assessments of the potential for gas generation.

10.2.4 Engineered barriers

Understanding the evolution of the different engineered barriers and their evolution through time is important for the long-term safety of a GDF in rock salt.

Shaft backfill and seals (Priority 1): COPERA (2020 - 2025) has not so far assessed possible shaft closure designs, so a systematic review should be undertaken of existing shaft backfill and seal concepts, designs and the materials used, leading to a recommendation for a closure system that could be used in a Dutch GDF. This should be linked to an evaluation of shaft construction and support systems. The review should also include an overview of the processes that could potentially affect a shaft closure system. Improving knowledge will increase confidence in the long-term safety of the repository and reduce uncertainty in the cost.

Tunnels seals (Priority 2): The current disposal concept includes tunnel seals, assuming that some form of concrete will be used. A systematic review should be undertaken of existing tunnel seal designs and the materials that have been used, leading to a recommendation that can be used in a Dutch repository in rock salt. The review should also include an overview of the processes that could potentially affect a tunnel seal, to increase confidence in the long-term safety of the repository and reduce uncertainty in the costs.

Granular salt backfill (Priority 1): Work in the first phase of the COPERA (2020 - 2025) programme increased our understanding of granular salt backfill compaction and, specifically, of the second phase of compaction (Oosterhout, 2023). However, the last phase of compaction, the complete healing of the backfill, still has large uncertainties and the estimated time that the backfill needs to heal varies between a few years to over 26,000 years. A better understanding of the timescales needed to heal granular salt backfill will help to set requirements for, and optimise, other engineered barriers. Furthermore, compaction of dry salt needs to be investigated.

Gas - Rock salt / backfill interaction (Priority 2): In the first phase of the COPERA (2020 - 2025), scoping calculations were performed on the potential for gas generation in a salt repository. These calculations indicated that gas generation and build-up within the repository potentially has an influence on the compaction rate of the granular salt backfill. It could, for example, delay or even halt compaction. It is therefore important to expand the model of backfill compaction and porosity/permeability development versus time to include the gas-rock salt interaction.

Repository-scale compaction model (Priority 1). Having experimental data on the compaction of salt on a small scale, the next step is to develop a computer model to simulate the compaction of salt with potential gas generation on a repository scale. This model should take into account the interaction between the host rock and the granular salt backfill, while keeping computational time manageable. The computer model results should ideally be compared to the results of existing models and natural analogues.

10.2.5 GDF design options

Thickness of the host rock (Priority 1): In the current disposal concept, it is assumed that the rock salt surrounding the waste should be at least 200 m thick, based on the subrosion rate. However, other processes could also affect the rock salt which could lead to a requirement for a greater minimum thickness. In this task, the processes that could affect the rock salt barrier should be identified, and numerical models need to be developed to simulate them. The result will help to optimise the design of the repository and can help in siting.

Backfill disposal rooms (Priority 1): In the upper level of the repository, disposal rooms are used for the disposal of LILW and (TE)NORM. To ensure that the waste package stacks remain stable, and the waste packages can be retrieved, concrete was proposed to be used a backfill. However, retrieving LILW and (TE)NORM would then require significant effort, even during the operational period. Furthermore, concrete contains water that might result in gas generation via corrosion or radiolysis. The use of concrete as a backfill might, therefore, not be optimal, and a systematic review should therefore be undertaken to identify materials that can be used to backfill disposal rooms. This material should be chosen to ensure that the waste remains in place, has a low permeability to limit the potential inflow of brine and gas generation, can easily be removed and can be used in salt. The review should result in a recommendation for a material (or materials) that can be used to backfill the disposal rooms.

(TE)NORM (Priority 2): In terms of volume, one of the largest amounts of waste to be disposed of is (TE)NORM and, more specifically, depleted Uranium (DU). In the current disposal concept, DU is conditioned and disposed in Konrad Type II containers. Because of the large volume, it is currently estimated that it would take about 10 years to dispose of all the DU (Herold and Leonhard, 2023b). To reduce the cost incurred by such a long operational period, some of the DU could be used as a component of construction materials. For example, DU can be used as an aggregate for concrete, which in turn can be used as a backfill material (Browning and Grupa, 2023). Using the DU in this way could improve the overall waste and backfill emplacement schedule to reduce the cost of a repository significantly. However, the impact on sealing effectiveness of using concrete with DU as an aggregate is not yet well studied. In addition, backfill containing DU must be treated as a radioactive material, which could complicate operations and, if needed, remedial actions (Browning and Grupa, 2023).

Scalability and optimisation (Priority 1): The current disposal concept assumes that the existing nuclear facilities will remain open as planned, with only one new research reactor (Pallas) being constructed. The current waste scenario does not account for the opening of four new nuclear power plants currently under consideration by the Dutch government (Erkens, 2024), which would result in significantly more waste. On the other hand, COVRA is currently exploring the use of an electric plasma furnace to reduce the volume of LILW to be disposed of. While the total change in waste volume requiring disposal is not yet known, it is important to study how the repository can be scaled up or down and optimised to account for potential changes in waste volume. The study should not only consider the repository size but also the waste emplacement and closure, as both represent a relatively large fraction of the total cost. Furthermore, the study needs to consider ventilation and how it can be optimized, as this is one of the largest uncertainties in the current upscaling of the disposal concept, and it can represent a significant fraction of the cost.

New waste forms (Priority 1): In this rock salt safety case, research has been limited to the currently expected wastes for disposal. However, COVRA is considering the use of an electric plasma furnace to reduce the volume of LILW to be disposed of. If implemented. this will produce a new type of waste, in the form of plasma slags. The disposability of this new waste form in a repository in rock salt, as well as the potential waste package that may be required, needs to be studied.

Optimising the engineered barrier system (Priority 1). In the current disposal concept, most of the granular salt backfill will be moisturised (Fig. 6.1b in Chapter 6). While this will help increase the compaction rate, the additional brine added to the granular salt backfill could potentially lead to more gas generation, which in turn could affect compaction. Gas generation can be reduced by limiting the amount of moisturised backfill used. The German concept involves using moisturised salt between two concrete seals in both the shaft and in the tunnels between the waste and the shafts, but uses dry granular salt in the rest of the repository, as this could significantly reduce the brine available for gas generation. The placement of this so-called salt seal, how it influences the design of the Dutch disposal, and whether it is needed for the upper level at all, need to be studied.

Generic safety assessment: A central part of a safety case is the quantitative assessment of the long-term evolution of a repository. In the previous phase of DECOVALEX, COVRA participated in a work package that developed a safety assessment for a generic salt dome to increase the confidence in the models, methods and software used. This generic safety assessment assumed single phase flow and did not include any gas generation. While this approach was simple, the knowledge and experience gained during this work package has been instrumental in the Dutch safety assessment (Chapter 8). In the next phase of DECOVALEX, the safety assessment will be expanded to include gas generation and two-phase flow, this work will provide improved input for the next Dutch salt safety assessment. Developing a generic safety assessment for a salt dome and comparing the outcomes within an international framework will increase knowledge, experience and confidence in the Dutch safety assessment. This generic safety assessment should also consider the very long-term evolution of the repository and, specifically, the fate of the depleted uranium.

10.2.6 Other key topics

Depending on available resources and priority, COVRA will also support national or international initiatives on other key topics.

APPENDIX 1. Historical background

This safety case and feasibility study is not only built upon research that has been performed as part of the COPERA (2020 - 2025) research programme but also on research performed as part of previous research programmes. Some of the choices made in the COPERA (2020 – 2025) disposal concept is a result of previous Dutch disposal
concepts. While research on rock salt in the Netherlands has been extensive, for brevity, we discuss here only the highlight of the disposal concept and how it has evolved through time. For more detailed information about the different research programmes, we refer to the references given in the text.

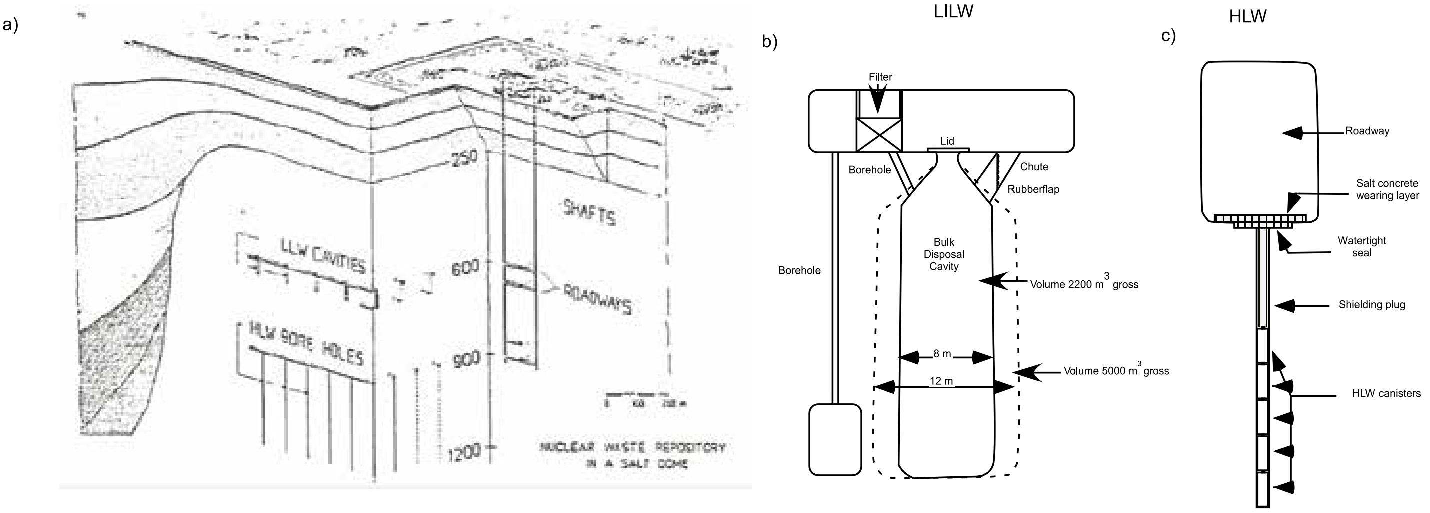
12.1.1 Prior to 1982

One of the earliest disposal concepts in the Netherlands was developed in the early 1980s for the disposal of Low Level Waste (LLW), Medium Level Waste (MLW), and High Level Waste (HLW, Hamstra, 1984). This concept considered only disposal in a salt dome, and no other types of salt formations were explored. The GDF was designed to consist of two levels. The first (upper) level was planned for the disposal of LLW and MLW in bunker-type cavities, and the second (lower) level was intended for the disposal of reprocessed HLW (Appendix 1 - Figure 1a). Two shafts were to connect the repository with the surface; one shaft was designated for the transport of materials, waste, and workers by a winding machine and for the intake of fresh air, while the second shaft was intended for the outflow of air.

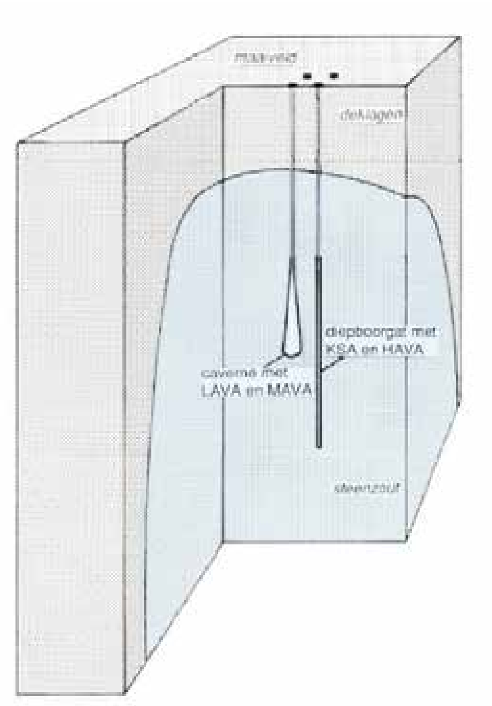
At the first level, planned to be located at a depth of 500 m, non-heat producing LLW and MLW were to be disposed of in a vertical, cylindrical bunker (Appendix 1 - Figure 1b). For the construction of this bunker, it was proposed to install a raise borer or ring cutter, and a central borehole of the future bunker was to be drilled. Subsequently, the rest of the bunker was to be constructed using explosives. After completing the bunker, waste packages could be dropped into the bunker in an uncontrolled manner. Two observations can be made regarding this concept: the LLW and MLW waste would not be neatly stacked, making it very difficult to retrieve the waste, if needed. Also, the use of explosives to construct the bunker would damage the host rock more than other excavation methods.

Heat-producing HLW was to be placed in vertical boreholes 300 or 600 m in length without an overpack. The boreholes were to be constructed in the middle of the disposal tunnel and finished with a concreted steel liner, bound into a steel and concrete collar construction at floor level (Appendix 1 - Figure 1c).

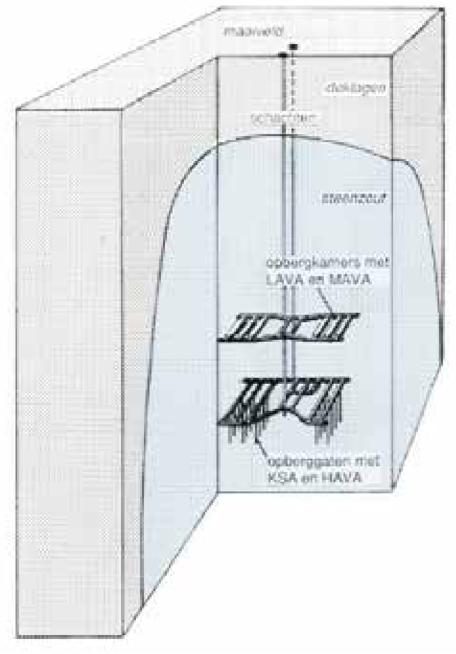
The plan for the emplacement of a HLW canister involved using either a wire or free fall. In the latter scenario, a relatively small annulus between the canister and the wall of the borehole was intended to compress the air below the canister and slow its descent during the fall. Precautionary measures were to be taken to minimize the impact. Lowering by wire or rope, although more time-consuming, was favoured because it offered much greater control over the descent of a canister. After filling a borehole with waste, the remaining space was planned to be filled with a mixture of salt, clay dust, and fly ash, topped by a plug of granulated bentonite. The borehole was then to be sealed with salt concrete and a steel cover plate.



Appendix 1 - Figure 1a) The two level GDF developed before 1983. In the concept, HLW will be disposed in vertical borehole while the LLW, MLW will be disposed in cavities. Note that the depth of the repository differs in the figure from the text but the outline of the GDF is the same (Figure from Hamstra and Verkerk (1981)). b) Disposal of LLW and MLW in cylindrical shaft (bunker). The chute on the right will be used to dump waste into the cavity.
c) Disposal of HLW in a borehole. The length of the borehole varies between 300 - 600 m depending on the number of canisters placed in the borehole.



a)



b)

Appendix 1 - Figure 2a) The OPLA boreholes and caverns disposal concept. In this concept, a cavern will be used to dispose LAVA and MAVA waste while KSA and HAVA waste will be disposed in a deep borehole. b) The conventional mine disposal concept. At the upper level of the repository, LAVA and MAVA waste will be disposed in disposal rooms. At a lower level, KSA and HAVA waste will be disposed in boreholes / cavern. For both disposal concepts, the number of caverns, (deep) boreholes and disposal rooms will depend on the nuclear power scenario and the selected salt formation (domal, pillow or bedded salt).

12.1.2 OPLA (1985 - 1993)

In 1984, a new research programme started called OPLA (OPberging op Land; disposal on land), in which two generic disposal concepts (boreholes - caverns and conventional mine) and three salt formations (bedded, pillow and domal salt) were considered. No provisions were included to allow retrieval of the waste.

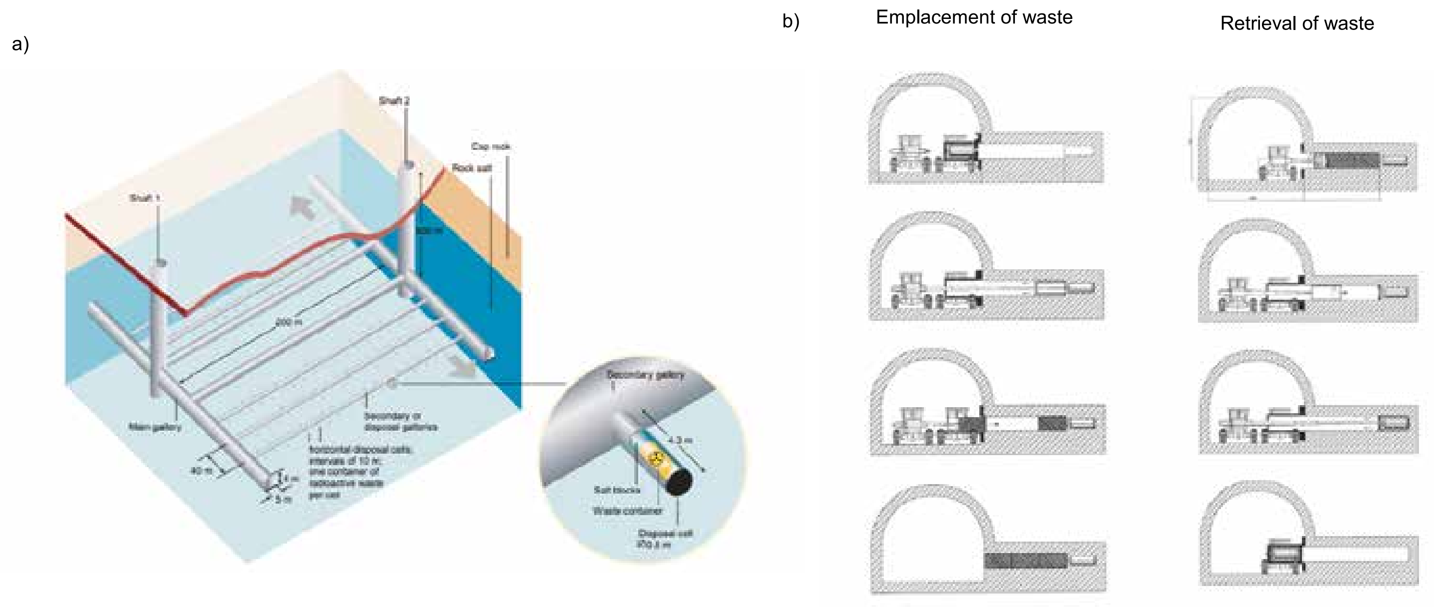
In the OPLA disposal concept (Appendix 1 - Figure 2a), dry or wet (brine filled) caverns would be used for the disposal of LAVA (Laag Actief Vast Afval, low level waste) and MAVA (Middel Actief Vast Afval, intermediate level waste) waste: this differs from what is considered currently. A wet cavern can be constructed by leaching and keeping the brine within the cavern: hence the designation 'wet'. A dry cavern is constructed in the same way, but the brine is pumped to the surface. Following the construction of the caverns, LAVA and MAVA waste would be disposed in the caverns. In a dry cavern, the waste is lifted down in a basket, the basket opens, and the waste is dumped. In a wet cavern, the basket will be opened just above the brine level. A different method is proposed for waste resulting from the decommissioning of old nuclear power plants. For that type of waste, a truck dumps the waste in a hopper that, via a conveyor chain, dumps the waste directly into the cavern.

For the disposal of KSA (Kernsplijtingsafval, high level waste) and HAVA (Hoog Actief Vast Afval, also high level waste), deep boreholes were proposed. These 1300 m deep boreholes are drilled from the surface into a salt body (either bedded, pillow or domal salt). As the deep boreholes will be filled with brine, the waste will be lowered in the borehole and subsequently released: the waste will sink at a relatively slow speed, but without direct control.

In the conventional mine disposal concept (Appendix 1 - Figure 2b), two types of salt formations (pillow and salt dome) were considered. Bedded salt was not considered as it is situated too deep for the conventional mining techniques available in 1984. In the disposal concept, the GDF consists of two levels. The upper level is for the disposal of LAVA and MAVA (low and intermediate level waste) while the lower level will be used for the disposal of KSA and HAVA (high level waste).

LAVA, MAVA and waste originating from decommissioning of nuclear power plants would be disposed at the upper level of the GDF, at a depth between 550 - 700 m (salt pillow) and 600 - 1500 m (salt dome), in a so-called rectangular shaped LAVA-rooms, in which the waste would be dumped by a truck. Alternatively, the waste could be disposed in a more controlled manner, using a gantry crane.

At the second lower level, at a depth between 590 (salt pillow) and 900 m (salt dome), both KSA and HAVA would be disposed in either a borehole or cavern, depending on the waste type. The boreholes have a length between 110 and 290 m. In a salt dome, longer boreholes would be possible, but the maximum practical depth is about 1300 m due to the convergence of a borehole. The waste would be lowered into the boreholes. To make sure the waste package will not buckle or break under the weight, a salt layer would be placed between two waste packages. For specific HLW wastes, caverns instead of boreholes would be used. This would be a relatively small cavern with a volume between 750 - 3500 m^3, constructed using a combination of conventional mining techniques (road header) and smooth blasting. The waste would be lowered slowly into the cavern and when this cavern is filled with brine, an overpack can be used to make sure the waste sinks to the bottom of the cavern.



Appendix 1 - Figure 3a) The METRO disposal concept proposed in CORA. This disposal concept only considers HLW and uses short horizontal boreholes for the disposal of waste without overpack. b) Retrieval of the waste. Figures from Heijdra and Prij (1997).

No provisions were made for the waste to be retrieved, if necessary, as it was deemed to conflict with the long-term isolation of the waste.

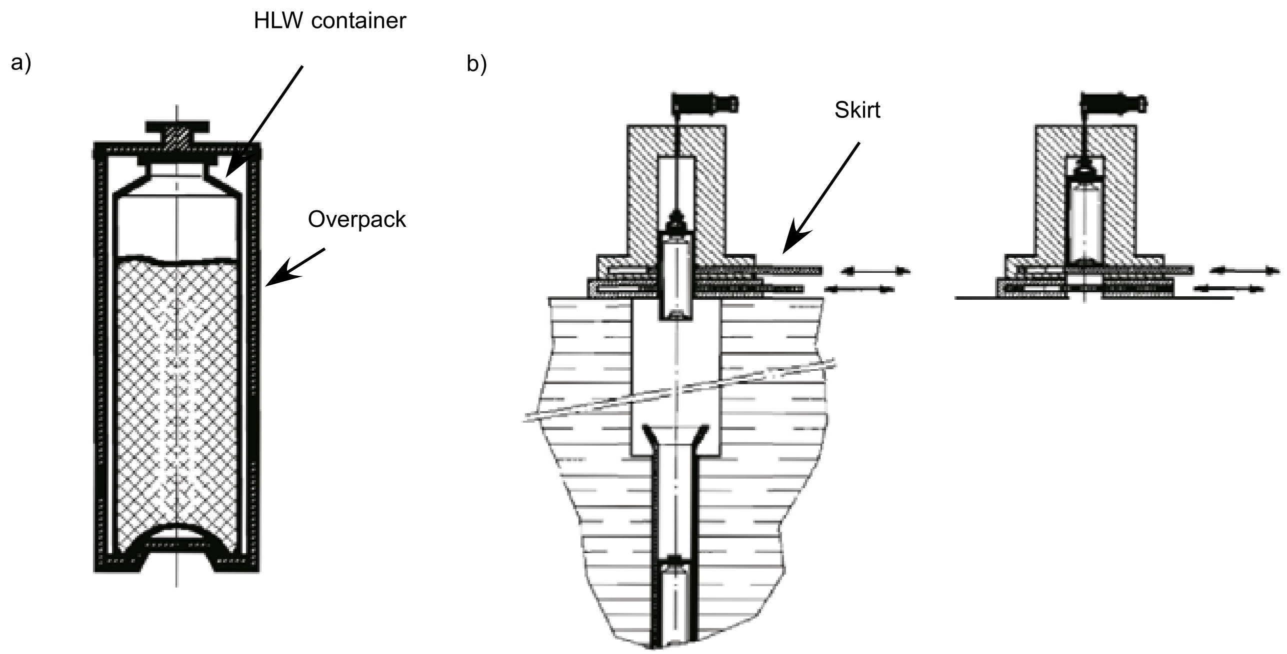
12.1.3 CORA (1995 -2001)

In the previous two disposal concepts, no provisions were made for a possible retrieval of the waste as it was not a requirement. For some of the waste types, it would be impossible to do so in both previous disposal concepts. In 1992, the retrievability of the waste was set as a new requirement by the Dutch government (Minister van volkshuisvesting ruimtelijke ordening en milieubeheer, 1992). In the research programme CORA, the successor of the OPLA research programme, two new generic disposal concepts were developed that take retrievability into account. One disposal concept uses short horizontal boreholes (METRO concept) while the other uses vertical boreholes (TORAD-B).

12.1.4.1 Disposal of HLW: METRO concept

In the METRO disposal concept (Heijdra and Prij, 1997), which only considers HLW (both vitrified waste from reprocessing and spent fuel assemblies), the depth of the GDF is set at 800 m (Appendix 1 - Figure 3). The GDF is connected to the surface via two vertical shafts. Each vertical shaft is also connected to a main gallery: there are thus two main galleries. In turn, the two main galleries are also connected by horizontal cross galleries used for disposal, with two of them not being used for disposal. As in previous programmes, a 200 m thickness of salt is assumed between the waste and the surrounding formations.

In the METRO disposal concept, reprocessed HLW waste is emplaced in short horizonal boreholes of 4.3 m in length and 0.5 m in diameter without overpack. This is different from the current disposal concept. Emplacement of the waste would be done with a specially designed emplacement vehicle. This avoids the use of lifting eyes or cams, which can be affected by corrosion. After the emplacement of waste, the horizontal borehole is closed with a pre-compacted, 150 cm cylindrical salt plug. For the retrieval of the waste, the salt plug would have to be drilled out up to a few decimetres from the waste. Subsequently, using an over-core, the waste is pulled out together with some surrounding salt and placed in a specially designed vehicle that will bring it back to the surface. An advantage of this disposal concept is that the short boreholes can be constructed on both sides of the disposal gallery and therefore the total length of all the disposal galleries would be less, compared to the use of vertical boreholes. For the spent fuel assemblies, it is proposed to dispose of them in a disposal overpack with dimensions matching those of the COGEMA container used for vitrified waste. This approach would allow the spent fuel assemblies to be disposed using the same container as vitrified waste, although it may require reducing their length, potentially resulting in additional waste. Alternatively, the spent fuel assemblies could be disposed of in a larger disposal overpack than the COGEMA container. This option has the advantage of eliminating the need to cut the spent fuel assemblies, but requires the development of a second container.



Appendix 1 - Figure 4a) Emplacement and b) retrieval of waste as proposed in the METRO concept. Figure from Heijdra and Prij (1997).

12.1.4.2 Disposal of HLW: Torad-B concept

A second disposal concept was developed during the CORA research programme. This disposal concept is referred to as Torad-B (Appendix 1 - Figure 4 Poley, 1999). In this concept, which only considers the disposal of waste and not the overall GDF design, vertical cased boreholes are used in combination with an overpack. These vertical boreholes would have a length of 500 m and be drilled from a disposal gallery at a depth of 800 m. The boreholes would be cased with steel, with a thickness of 60 mm, to prevent bending of the borehole casing. The steel corrosion rate was assumed to be between 0.02 - 0.2 mm per year, and the 60 mm thickness would ensure that the casing remains intact long enough to ensure retrievability.

For the emplacement of the waste, special lifting equipment is used. This lifting equipment protects workers from radiation during the emplacement of the waste. It includes a skirt to protect the workers when the waste is hoisted down the borehole. For final closure, the borehole would first be filled with granular salt or bentonite to cover the uneven surface of the top of the overpacks. Then, salt plugs are placed on top, and the open spaces between the salt plug and the borehole are filled with granular salt. It was suggested that heaters could be used to speed up the salt creep.

12.1.5 Summary

In summary, in all disposal concepts, LILW is dumped in a type of disposal room, bunker or large cavern. These spaces are generally constructed using explosives in combination with conventional mining.

The emplacement of waste into a bunker or cavern would make it very difficult to retrieve, which became a requirement in 1992. Furthermore, construction of parts of the repository using explosives will disturb the host rock more than other methods and could have a negative impact on the performance of the repository. Since METRO did not consider the disposal of LILW, the COPERA (2020 - 2025) disposal concept (Chapter 4) will be the first Dutch disposal concept in salt that takes retrievability of the LILW into account. It will also be the first Dutch disposal concept for salt that will utilise only those techniques for construction that limit the disturbance to the host rock.

The earlier concepts for HLW disposal generally use vertical and horizontal boreholes. The way in which the waste is lowered in the borehole changed from dropping it in an uncontrolled manner (e.g., OPLA) into a deep borehole, to a more controlled emplacement of the waste, using cranes (e.g., TORAD-B) or special vehicles (e.g., METRO). In the COPERA (2020 - 2025) disposal concept (Chapter 4), the emplacement of the HLW will be done in a controlled way. As in TORAD-B and METRO, the COPERA (2020 - 2025) disposal concept will also take retrievability into account.

APPENDIX 2. Publications in COPERA (2020 - 2025) related to rock salt

The COVRA's long-term research programme for the period 2020-2025 has resulted in multiple reports, peer-reviewed publications, and abstracts. In general, reports are published on COVRA's website, while peer-reviewed publications and abstracts are generally published on the website of a journal or the organizer of a conference. The publications, limited to the reports and publications only, related to a GDF in rock salt are described in this appendix.

Work package 0: Programme management and monitoring

This work package covers all management and coordination tasks of the programme office, including international collaboration. COVRA participates in the NEA Salt Club, which promotes knowledge exchange between Member States that consider rock salt as a host rock formation for deep geological repositories. This collaboration has resulted in one report on a Generic FEPs Catalogue and Salt Knowledge Archive (Geoff Freeze et al., 2020) and a paper on scenario development (Kuhlman et al., 2024b). COVRA also participated in DECOVALEX, which is an international research and model comparison collaboration. More specifically, COVRA participated in Task F, which had the primary objective of building confidence in the models, methods, and software used for performance assessment. This task has been instrumental for the Dutch performance assessment and resulted in one report (LaForce, 2024a) and three journal papers in Geomechanics for Energy and the Environment (Bartol and Vuorio, 2025; Bartol et al., 2025, LaForce et al., 2024).

Geoff Freeze, S., Sevougian, S.D., Kuhlman, K., Gross, M., Wolf, J., Buhmann, D., Bartol, J., Leigh, C., Mönig, J., 2020. *Generic FEPs Catalogue and Salt Knowledge Archive (SAND2020-13186), Sandia report.*

Kuhlman, K.L., Bartol, J., Carter, A., Lommerzheim, A., Wolf, J., 2024b. *Scenario development for safety assessment in deep geologic disposal of high-level radioactive waste and spent nuclear fuel: A review. Risk Analysis doi: 10.1111/risa.14276.*

LaForce, T., 2024. *DECOVALEX-2023 Task F2-Salt Final Report, p. 198.*

LaForce, T., Bartol, J., Becker, D.-A., Benbow, S., Bond, A., Dietl, C.R., Frank, T., Kock, I., Magri, F., Nicholas, J., Jayne, R., Pekala, M., Stauffer, P.H., Stein, E., Stone, J., Wolf, J., 2024. *Comparing Modelling Approaches for a Generic Nuclear Waste Repository in Salt Geomechanics for Energy and the Environment doi.org/10.1016/j.gete.2024.100621.*

Bartol, J., Vuorio, M., 2025. *Safety assessment for a geological disposal facility in domal salt: the Dutch case. Geomechanics for Energy and the Environment.*

Bartol, J., Becker, D.-A., Benbow, S., Bond, A., Frank, T., LaForce, T., Nicholas, J., Jayne, R., Stauffer, P.H., Stein, E.R., Stone, J., Wolf, J., 2025. *Designing a repository in domal salt: what is important? Geomechanics for Energy and the Environment.*

Work package 1: Programme strategy

As part of this work package, the disposal concept for a repository in rock salt was reviewed and optimized where possible (Herold and Leonhard, 2023a). This optimization included, for example, the diameter of the different shafts, the spacing between the tunnels, and the width of the tunnels. The optimized disposal concept was then used to estimate the overnight cost of a repository in rock salt (Herold and Leonhard, 2023b). Both reports have been published on our website.

Herold, P., Leonhard, J., 2023. *Cost Estimate for a Repository in Rock Salt in the Netherlands – Phase I - Review Repository Concept (BGE TEC 2023-07).*

Herold, P., Leonhard, J., 2023. *Cost Estimate for a Repository in Rock Salt in the Netherlands (BGE TEC 2023-13).*

Work package 2: Safety case and integration

The integration of the knowledge obtained through COPERA (2020 - 2025) for disposal of waste in rock salt is presented in this rock salt safety case and feasibility study. In addition, a report has been published on our website on a FEP (Features, Events and Processes) - Catalogue and Scenario Development for a Dutch generic HLW Repository in a Salt Dome (Lommerzheim, 2023). This report used FEPs to derive the potential evolution of the GDF in rock salt systematically. These scenarios have subsequently been used as input for the safety assessment.

Lommerzheim, A., 2023. *FEP-Catalogue and Scenario Development for a generic HLW Repository in a Salt Dome. www.covra.nl*

Work package 3: Engineered Barrier System

The COPERA (2020–2025) research programme in the EBS focused on HLW waste packages, (TE)NORM waste packages and granular salt backfill.

Task 3.3. 1 Waste package for HLW

In the COPERA (2020 – 2025) disposal concept, a HLW package was envisioned to be used. Using the requirements set by COVRA, Wunderlich et al. (2023) developed an HLW package that is optimised for a Dutch repository in rock salt. This optimised HLW package is constructed of carbon steel (TStE335, L2-COV-04) and contains up to 6 CDS-V or CDS-C or 2 ECN canisters. Using the newly designed HLW waste package, Smit (2022) used numerical models to calculate the temperature within the host rock for different spacings. Smit (2022) found that the temperature within the host rock is expected to remain below 100 degrees Celsius even if the HLW waste packages are in contact: no spacing between them. However, the backfill was not considered by Smit (2022) which is likely to have an initial lower thermal conductivity resulting in higher temperatures. The report of Wunderlich et al. (2023) is published on our website

Wunderlich, A., Engelhardt, H.-J., Herold, P., Marggraf, U., Schmidt, W.A., Seidel, D., 2023. *Waste Package for disposal of High-Level waste (HLW) in rock salt (BGE TEC 2023-02). www.covra.nl*

Smit, H., 2022. *Long term disposal of nuclear waste in the Netherlands: possibilities and questions to be asked.*

Task 3.3.2 Waste package for (TE)NORM

Currently, (TE)NORM held at COVRA is stored in standardised DV-70 containers. In OPERA, it was assumed that for disposal in poorly indurated clay, (TE)NORM would be conditioned with Portland cement. In turn, the uranium oxide - Portland cement mixture would be contained in a Konrad type 2 container. Browning and Grupa (2023) looked at whether the same disposal method could be used to dispose (TE)NORM in rock salt. They found that the use of the Konrad type 2 container for disposal would be the most favourable solution when (TE)NORM is conditioned. The DV-70 container, on the other hand, could be used for final disposal when the waste is not conditioned. However, the DV-70 container will not be able to withstand the pressure loading caused by the creep of the host rock after the closure of the GDF, which could hamper its retrievability. In addition, Browning and Grupa (2023) found that conditioning the waste will reduce the solubility of most radionuclides in (TE)NORM. The report Oudenaren and Browning (2023) estimated the cost for disposal of (TE)NORM and a third report looked at whether it would be useful to design a disposal pack that would provide containment over very long time (>> 1000 years, Oving and Grupa, 2024). The latter is important to set the requirement for a container used to dispose (TE)NORM.

Browning, K., Grupa, J., 2023. *Disposal concept for DU in Rock Salt | Phase II: Final report. www.covra.nl*

Oving, J., Grupa, J.B., 2024. *The calculated leaching rate of DU within a salt repository after container failure. www.covra.nl*

Oudenaren, G.I.L., Browning, K., 2023. *Cost estimation for processing and disposal of (TE)NORM. www.covra.nl*

Task 4B.2: Evolution of the permeability-porosity in rock salt

It is important to understand how long it will take for the granular salt backfill to compact, how the porosity and permeability develops through time develops within and how the compaction rate could be increased. As part of COPERA (2020 – 2025), this was addressed by Oosterhout et al. (2022) and Oosterhout (2023) in cooperation with other organisations from the US and Germany via the KOMPASS project (Friedenberg et al., 2024). Results showed that granular salt backfill with an average diameter of 1 mm would require a few hundred years to reach a residual porosity of 1%. Furthermore, the addition of brine would speed up compaction significantly. The work by Oosterhout et al. (2022) was subsequently implemented in COMSOL, the software used in the COPERA (2020 – 2025) safety assessment, by Nicholas and Thatcher (2023). Reports are available on our website.

van Oosterhout, B.G.A, Hangx, S.J.T., Spiers, C.J., de Bresser, J.H.P., 2022. *Preliminary recommendations for modelling compaction creep and porosity permeability evolution in crushed salt backfill Phase 1 Deliverable. www.covra.nl*

Oosterhout, B.G.A.v., 2023. *Experimental investigation on the effect of stress, grain size (distribution) and porosity on compaction rates in granular rock salt backfill and implications for sealing timescales of a radioactive waste repository.*

Friedenberg, L., Bartol, J., Bean, J., Beese, S., de Bresser, H., Coulibaly, J.B., Czaikowski, O., Düsterloh, U., Eickemeier, R., Gartzke, A.-K., Hangx, S., Jantschik, K., Laurich, B., Lerch, C., Lerche, S., Liu, W., Lüdeling, C., Mills, M., Müller-Hoeppe, N., Oosterhout, B.G.A.v., Popp, T., Rabbel, O., Reedlunn, B., Rogalski, A., Rölke, C., Saruulbayar, N., Spiers, C.J., Svensson, K., Thiedau, J., Zemke, K., 2024. *KOMPASS-II Compaction of Crushed Salt for Safe Containment – Phase 2 (GRS - 751).*

Friedenberg, L.B., Bartol, Jeroen; Bean, James; Czaikowski, Oliver; Düsterloh, Uwe; Gartzke, Ann-Kathrin; Hangx, Suzanne; Laurich, Ben; Lerch, Christian; Lerche, Svetlana; Lippmann-Pipke, Johanna; Liu, Wenting; Lüdeling, Christoph; Mills, Melissa; Müller-Hoeppe, Nina; Popp, Till; Rabbel, Ole; Rahmig, Michael; Reedlunn, Benjamin; Rölke, Christopher; Spiers, Christopher; Svensson, Kristoff; Thiedau, Jan; van Oosterhout, Bart; Zemke, Kornelia; Zhao, Juan; Bresser, J.H.P. de; Drury, M.R.; Fokker, P. A.; Gazzani, M.; Hangx, S.J.T.; Niemeijer, A.R.; Spiers, C.J., 2022. *Compaction of crushed salt for safe containment: Overview of phase 2 of the KOMPASS project, The Mechanical Behavior of Salt X.*

Nicholas, J., Thatcher, K., 2023. *COVRA Salt Creep Modelling – PA model. www.covra.nl*

Work package 4: Host rock

Task 4B.1: Geotechnical properties

Numerical models will play a key role in demonstrating the safety of a GDF in rock salt and specifically its long-term safety: the safety assessment. They do, however, require extensive knowledge of the Thermal, Hydrological and Mechanical (THM) properties of salt in both the short and long term. In this task, Hunfeld et al. (2023) collected THM of Dutch salt which is now available on our website. To expand the database, a task within the salt club will aim to collect salt data from other member states.

Hunfeld, L.B., Pluymaekers, M., Larède, V., 2023. *Database with Thermal, Hydrological, Mechanical, Chemical (THMC) properties of rock salt. www.covra.nl*

Task 4B.2.1: Gas Production

In this task, a model was developed to provide an initial estimate of the potential for gas generation in a salt repository. The model considered three main gas generation mechanisms: corrosion, microbial breakdown of organic substances and radiolysis. Additionally, two sources of brine were modelled, while the compaction of the granular salt backfill was not included in the results. The results showed that gas generation strongly depends on the availability of brine, with more brine leading to more gas production. Furthermore, it showed that in most cases it is necessary to reduce the amount of brine that is initially available in the waste and in the void space (granular salt backfill) surrounding it to limit the amount of gas generated. This is because in many cases the brine/water that is initially present (in waste or in the granular salt backfill) is sufficient to support all the potential gas generation from the package. The reports can be found on our website (Benbow et al., 2023a; Benbow et al., 2023b; Watson, 2023).

Benbow, S., Newson, R., Bruffell, L., Bond, A., Watson, S., 2023. *Potential Gas Generation in a Salt Repository | Gas Generation Model – Functional Specification and Definition of Central Case for Analysis (QDS-10075A-T2-FS). www.covra.nl*

Benbow, S., Watson, S., Newson, R., Bruffell, L., Bond, A., 2023b. *Potential Gas Generation in a Salt Repository | Gas Generation Model – Results (9QDS- 10075A-T3-RESULTS). www.covra.nl*

Watson, S., 2023. *Potential Gas Generation in a Salt Repository | Waste Groups and High-level Conceptualisation (QDS-10075A-T1-WASTES). www.covra.nl*

Task 4B.2.3: Brine availability

Brine can occur in three locations within undisturbed rock salt: within a salt crystal (intragranular brine), between salt crystals (intergranular brine), and as water or hydration bound to hydrous minerals. The availability of brine is crucial to the safety case because (1) it is the primary off-site radionuclide transport medium, (2) it could lead to the corrosion of metallic and glass waste forms and waste packages, (3) chloride in brine can reduce criticality concerns, and (4) brine can provide back-pressure to resist long-term creep or potentially increase the compaction rate.

In this work package, COVRA developed numerical models using COMSOL to study the availability of brine and compared the results of these models with experiments conducted at the WIPP facility. Interestingly, both experimental and numerical model results indicated that the highest influx of brine from the host rock is expected during the rapid cooling of the host rock. This is possibly related to the enhanced permeability of the EDZ due to extensional cracks forming because of the rapid cooling. This task was carried out via DECOVALEX, in cooperation with Sandia National Laboratories (USA), BGR/GRS (Germany), and Quintessa (UK). The results of this task are published in one report and one paper published in Geo-mechanics for Energy and the Environment.

Kuhlman, K.L., 2024. *DECOVALEX-2023 Task E Final Report. decovalex.org*

Kuhlman, K.L., Bartol, J., Benbow, S., Bourret, M., Czaikowski, O., Guiltinan, E., Jantschik, K., Jayne, R., Norris, S., Rutqvist, J., Shao, H., Stauffer, P.H., Tounsi, H., Watson, S., 2024. *Synthesis of Results for Brine Availability Test in Salt (BATS) -DECOVALEX-2023 Task E. Geomechanics for Energy and the Environment.*

Task 4B.3: Radionuclide Solubility in Brine

The mobility of radionuclides in the host rock – once a waste container has failed and the waste matrix is in direct contact with brine – depends on the solubility and sorption of the waste. With rock salt being the host rock, solubility of the waste in very high salinities (brine) is the most relevant process while sorption of radionuclides is assumed to be of lesser relevance. As part of the COPERA (2020 – 2025), Oving and Meeussen (2024) developed a computer model that can stochastically determine the solubility of radionuclides in various possible chemical environments expected in a repository in rock salt. This will allow for the evaluation of the solubility of radionuclides, using the ThermoChimie and NEA thermodynamic database, under varying chemical conditions while showing the sensitivity of an element with respect to the given chemical parameters. The report of the model is available on our website.

Oving, J., Meeussen, J.C.L., 2024. *Radionuclide solubility in varying chemical environments: The solubility of relevant radionuclides in chemical environments occurring in a rock salt repository. www.covra.nl*

Task 4B.4.1: Bedded salt of the Röt formation

Research in the past (e.g., Hamstra, 1984; Van Hattum en Blankevoort, 1986) mainly focused on Dutch salt domes of the Zechstein Group, due to their large size and available data. Data on other salt deposits are more limited. To map and characterise the Triassic Röt formation seismically, a student project was funded, which has resulted in a more detailed map of the thickness and depth of the main Röt evaporite layer in the east Netherlands (Altenburg, 2022). In addition, Altenburg (2022) also studied the cores of this deposit. While the main evaporite layer does indeed consist typically of evaporites (halite), there are also layers present of, for example, red claystone and anhydritic dolomite: the halite is not pure. The results of this student project are published on our website.

Altenburg, R., 2022. *Storage and disposal potential of the Triassic Röt Formation in the east of the Netherlands: Subsurface mapping and facies interpretation, Department of Earth Sciences. Utrecht University. www.covra.nl*

Task 4B.4.2: Understanding past, present and future subrosion rates in the Netherlands / Task 4B.4.3: Diapirism rates in the Netherlands (Past-Present-Future)

To understand past subrosion and diapirism rates, COVRA funded two student projects (Almalki, 2023; Lauwerier, 2022). Using seismic data, both students obtained subrosion and diapirism rates for eight different salt structures in the Dutch underground. Both found that the subrosion rate is in the order of 0.01 and 0.1 mm/year and the diapirism rate is between of 0.001 and 0.1 mm/year. However, the uncertainty in both is still large. The reports of both students are published on our website

Lauwerier, W., 2022. *The evolution of the Zechstein salt diapirs in the north-eastern Netherlands., Internship research for Earth, Life and Climate. Utrecht University, Utrecht. www.covra.nl*

Almalki, B.A.S., 2023. *Zechstein salt structures evolution in the north-eastern Netherlands, Department of Earth Sciences. Utrecht University, p. 36. www.covra.nl*

Task 7.2: Knowledge transfer to students

The transfer of knowledge to students is an integral part of the COPERA research programme. Therefore, multiple internships and master theses were funded by COVRA. More specifically, two students worked on Task 4B.4.2 and 4B.4.3 (Almalki, 2023; Lauwerier, 2022) while another student worked on Task 4B.4.1 (Altenburg, 2022). COVRA is also funding a PhD student at Utrecht University (Oosterhout, 2023; Oosterhout et al., 2022) and two lecture course were given in the past two years as part of the Earth Mineral Resources course at Utrecht University. In addition, COVRA, together with the Dutch Geological Survey, wrote a chapter about geological disposal of radioactive waste (Neeft et al., 2024a) for the book Geology of the Netherlands:

Neeft, E.A.C., Bartol, J., Vuorio, M.R., Vis, G.-J., 2024. *Chapter 21: Geological disposal of radioactive waste in Geology of the Netherlands*

APPENDIX 3. RMS

Level 1 requirements

Level 1 requirements are applicable to all the steps in the management of waste. These requirements have been selected from documents prepared by international and national organisations that must be taken account of in COVRA's activities. Preferably, reference is made to documents from national organisations.

L1-DCRE-01: The permitted additional radiation dose for radiological workers in the Netherlands is 20 mSv per year.

Any doses shall be as low as reasonably achievable (ALARA). Some quantification is necessary for the definitions of design requirements. The Netherlands adopts the recommendations provided by the International Commission of Radiation Protection (ICRP) in the Dutch Decree on radiation protection. ICRP has recommended the dose limit of 20 mSv per year. Article 7.34 in this Decree requires that the dose rate shall not exceed 20 mSv per year.

L1-NPRA-01: The disposal facility shall be operational in 2130.

The first national programme states that: The definitive decision on the disposal method will be taken around 2100. At that time, society may opt for a different (end-point) management option, depending on the state of understanding at that time, and assuming that other alternatives are available by that time. The relatively long period of above ground storage will provide time to learn from experiences in other countries, to carry out research and to accumulate knowledge. In this way, sufficient money can also be set aside to make eventual disposal possible. As a consequence, in the future, a well-argued decision on the management of radioactive waste can be taken, without unreasonable burdens being placed on future generations (I&E, 2016).

In Article 10.10 in the Dutch radiation protection decree of 2023, it is required that COVRA's waste fees need to be set in a transparent, objective and non-discriminatory manner. Any contribution to the waste fees includes the provision by COVRA of collection, transport, processing, storage and disposal services. The financial scheme determines the contributions in the waste fees for the long-term storage of the waste in the surface facilities and the emplacement of waste packages in an underground facility in 2130.

L1-NPRA-02: Waste shall be isolated from people and the accessible biosphere.

The Dutch policy for waste was established in 1984 (Berkers et al., 2023; I&E, 2016; Winsemius and Kappeyne van de Coppello, 1984) with the following three objectives for the management of chemically toxic waste and radiologically toxic waste: Isolation, Control and Surveillance/Monitor (In Dutch Isoleren, Beheersen en Controleren so called IBC-principle). The implementation of the IBC-principle reduces the possibility of having unacceptable amounts of toxic materials in our living environment. This principle is, for example, also applied to soil remediation (I&E, 2016).

L1-NPRA-03: Any handling of the waste shall be controlled.

Control is also an objective for the management of waste in the Dutch IBC-principle (I&E, 2016; Winsemius and Kappeyne van de Coppello, 1984).

L1-NPRA-04: Waste shall be enclosed by a series of engineered barriers.

People will be protected by placing a series of barriers between the radioactive waste and the human environment. The packaging of the waste is an engineered barrier that ensures that the waste is enclosed (I&E, 2016). The description for enclosure of waste has only been written for disposal of waste in the national programme. For other steps in the management of radioactive waste, we have looked at the meaning of the IBC principle for chemically toxic waste and radiologically toxic waste as published upon parliamentary questions submitted by Boois (VROM, 1985). The term Isolation is used for two meanings: 1) to prevent direct contact with the waste as well as 2) to prevent contaminants from the waste spreading into the soil by the use of an impermeable layer. The second use meant by COVRA as containment.

Level 2 requirements

Level 2 requirements are extracted from the national and international requirements, but have been specifically developed into a form that expresses by COVRA's policy. These level 2 requirements are also applicable to all the steps in the management of waste.

L2-COV-01: The incremental radiation dose for radiological workers shall be less than 6 mSv per year.

A lower dose constraint than the Dutch radiation protection decree is used by the organisation (COVRA, 2022).

L2-COV-02: Waste shall be stored in dedicated surface facilities until an end-point management technique is available.

The surface facilities should provide the optimal conditions to keep the waste form and packing in a suitable condition for an end-point management technique. Three examples of end-point management techniques are:

1. Disposal of waste;
2. Recycling of materials used as a waste form or for packaging of the waste;
3. Treating as a conventional waste stream for sufficiently decayed (exempt) waste whose materials cannot be recycled.

L2-COV-03: Simple, robust, reliable and proven techniques shall be used.

From packaging to disposal, wastes might require handling over a period of more than 100 years. The procedures for handling the waste need to be demonstrated and effective. The use of technologies that have been proven will therefore be used for all the steps in the management of waste. These demonstrated techniques also minimize the uncertainty in cost estimates for the operation and closure of the disposal facility.

L2-COV-04: Materials for which broad experience and knowledge already exists, shall be used.

From storage to disposal, engineered barriers need to be stable over a period of more than 100 years There are many processes that may affect the behaviour of barriers used for waste packaging during storage and disposal. Uncertainty in long-term prediction of the behaviour of the engineered barrier system is smaller if demonstrated materials are used. During storage, the behaviour of the waste package can be monitored, and if needed, any deleterious behaviour that is not predicted can be mitigated by active measures. After disposal of waste, safety cannot rely on monitoring, owing to the long-time scales foreseen. The identification of the relevant processes for the prediction of the long-term behaviour of engineered barriers can be performed by studying natural analogues and archaeological analogues.

L2-COV-05: Only solidified waste shall be stored and disposed of.

All waste is solidified in advance of storage. The properties of the solid waste form minimize any release of radionuclides under all circumstances considered for storage (e.g., flooding) and disposal of waste.

L2-COV-06: In the case of fissile material, the containment shall exclude criticality.

Fissile material contained in a package lacks water to moderate a possible chain reaction. Following Posiva (2021), COVRA assumes that the possibility of criticality can be excluded by containment.

Level 3 requirements

The level 3 requirements described here are only applicable to disposal of waste.

L3-NPRA-01 A disposal facility shall be designed to contain all the different types of radioactive waste expected to arise up to 2130.

The policy in the Netherlands is that most of the radioactive waste produced in the Netherlands is managed by a single organisation: COVRA. The exceptions are radioactive waste with a half-life less than 100 days that is allowed to decay at the sites where it is generated and large amounts of NORM waste that are disposed of (or re-used) at two designated landfills. COVRA therefore collects radioactive waste from different waste generators, for example, nuclear power plants, nuclear medicine production plants, research organisations, universities and hospitals. The waste generators transfer different types of waste to COVRA according to their classification based on activity, radionuclide content and chemical and physical characteristics. Low and Intermediate Level Waste is generated by all organisations that consign wastes to COVRA. High Level Waste is only generated by nuclear plants.

L3-IAEA-01: Isolation shall be provided for at least several thousands of years for HLW.

This requirement is a short abbreviation of requirement 9 in SSR-5 (IAEA, 2011a): *This requirement is a short abbreviation of requirement 9 in SSR-5 (IAEA, 2011a): The disposal facility shall be sited, designed and operated to provide features that are aimed at isolation of the radioactive waste from people and from the accessible biosphere. T he features shall aim to provide isolation for several hundreds of years for short lived waste and at least several thousand years for intermediate and high level waste. In so doing, consideration shall be given to both the natural evolution of the multibarrier system and events causing disturbance of the facility.*

COVRA makes a distinction between heat-generating HLW and non-heat generating HLW. Non-heat generating HLW is classed as intermediate level waste by the IAEA. Dutch HLW has a contact dose rate larger than 10 mSv per year. LILW as currently stored by COVRA has a contact dose rate smaller than 10 mSv per year. This type of waste is called low level waste in many countries. COVRA does not have a category of short-lived waste for disposal, as this type of waste may be recycled.

L3-NPRA-02: Waste shall be retrievable during the operational phase of the GDF through until its closure.

The requirement for the retrievability of waste has been introduced in Dutch policy in order to have active control over the emplacement of waste packages and closure of the disposal facility (Alders, 1993). This requirement is to prevent the investigation of disposal concepts that do not comply with the IBC-principle.

L3-IAEA-02: The radionuclides in the waste shall be contained by the engineered barriers and natural barriers until radioactive decay has significantly reduced the hazard posed by the waste.

This requirement is split in two requirements (L3-IAEA-02 and L3-IAEA-04). These requirements are short descriptions for requirement 8 in SSR-5 (IAEA, 2011a): The engineered barriers, including the waste form and packaging, shall be designed, and the host environment shall be selected, so as to provide containment of the radionuclides associated with the waste. Containment shall be provided until radioactive decay has significantly reduced the hazard posed by the waste. In addition, in the case of heat generating waste, containment shall be provided while the waste is still producing heat energy in amounts that could adversely affect the performance of the disposal system.

L3-IAEA-03: Passive safety shall be provided by multiple safety functions for containment and isolation.

We combined the following three requirements in SSR-5 related to passive safety and multiple safety functions. L3-IAIA-03 is a short abbreviation of:

- requirement 5: *Passive means for the safety of the disposal facility: The implementer shall evaluate the site and shall design, construct, operate and close the disposal facility in such a way that safety is ensured by passive means to the fullest extent possible and the need for actions to be taken after closure of the facility is minimized.*

- requirement 6: *Understanding of a disposal facility and confidence in safety: The implementer of a disposal facility shall develop an adequate understanding of the features of the facility and its host environment and of the factors that influence its safety after closure over suitably long time periods, so that a sufficient level of confidence in safety can be achieved.*
- requirement 7: *Multiple safety functions: The host environment shall be selected, the engineered barriers of the disposal facility shall be designed and the facility shall be operated to ensure that safety is provided by means of multiple safety functions. Containment and isolation of the waste shall be provided by means of a number of physical barriers of the disposal system. The performance of these physical barriers shall be achieved by means of diverse physical and chemical processes together with various operational controls. The capability of the individual barriers and controls together with that of the overall disposal system to perform as assumed in the safety case shall be demonstrated. The overall performance of the disposal system shall not be unduly dependent on a single safety function.*

Requirement 6 in SSR-5 has overlaps to a great extent with L2-COV-04.

As explained in Chapter 2, the multi-barrier system addresses two principal objectives providing safety: isolation of the waste and containment of the radionuclides associated with them.
These objectives should be provided by nature i.e. passive safety. Each of the barriers (components) in the multi-barrier system has one or multiple safety functions. The safety concept in the conceptualisation stage is the description of how the barriers in the disposal concept are integrated to provide safety after closure. Safety functions with assigned time frames are used for this description. A safety function is the action or role that a natural and/or engineered barrier performs after closure of the GDF to prevent radionuclides in the waste from ever posing an unacceptable hazard to people or the environment. The necessary engineered barrier system (EBS) can be host rock specific.

L3-D-IAEA-04: In the case of heat-generating waste: the engineered containment shall retain its integrity until the produced heat will no longer adversely affect the performance of the multibarrier system.

See description for L3-IAEA-02.

APPENDIX 4. Waste scenarios

In the COPERA safety case & feasibility study, the design of the repository and thus its dimensions are based on Waste Scenario 1 (See section 4.1) in which the Nuclear Power Plant (NNP) in Borssele remains open until 2033 while the High Flux Reactor in Petten is replaced by Pallas. This is in line with the waste scenario considered by Verhoef et al. (2017). However, the government is currently considering extending the lifetime of the Borssele NPP by 10 years and the construction of 2 new NPP's and possibly even 4 (Erkens, 2024). Therefore, two additional scenarios were developed. Waste Scenario 2 is similar the Waste Scenario 1 with the expectation that in this scenario, the Borssele nuclear powerplant will remain open until 2044: an additional 10 years. Waste Scenario 3 is like Waste Scenario 2 but includes the construction of 2 additional new nuclear powerplants. Both are in line with what the Dutch government is currently considering. In the last 2 scenarios, not only does the amount of HLW increase, but also the volume of LILW. The additional volume of LILW will only be produced in the form of 200L drums and 1,000L: resins immobilized liquid waste (containers). Thus, the molybdenum waste in both these scenarios does not increase compared to Waste Scenario 1 as NPP do not produce this type of waste. The three different Waste Scenarios shown here are in line with the waste scenarios considered Burggraaff et al. (2022) and in the clay safety and feasibility study (Neeft et al., 2024b).

12.4.1 Appendix: Waste Scenario 1

While Waste Scenario 1 is similar to the waste scenario considered in the OPERA safety case, there are some differences in both types and volumes of the waste (Burggraaff et al., 2022). Starting with LILW, in the Opera safety case, a total volume of 45,563m^3 was expected for disposal. This has been reduced to 31,461 m^3 (Burggraaff et al., 2022) based on the linear extrapolation of the LILW received by COVRA over the past 20 years until 2130. To calculate the number of LILW waste packages expected for disposal, the volume expected for disposal is divided by the volume of each waste package and subsequently rounded to the highest 10,000 number for the 200L drums and to the highest 100 number for the 1,000L containers. Note that it is implicitly assumed that the volume and the types of waste received by COVRA do not significantly change in time.

For (TE)NORM, the OPERA safety case (Verhoef et al., 2017) used the storage capacity of the storage facility for depleted uranium that was opened in 2017. This resulted in a volume of 34,000 m^3 expected for disposal after conditioning. Burggraaff et al. (2022), on the other hand, linearly extrapolate the volume of (TE)NORM received over the past 20 years by COVRA up to 2050. This results in a volume of 58,070 m^3 expected for disposal after conditioning. Subsequently, this volume is divided by the volume of the Konrad type 2. The latter is expected to be used for the disposal of this type of waste. The dimensions of the Konrad type 2 used here are different from OPERA. In the latter, the container was erroneously assumed to be 1.7 by 1.6 by 1.6 m while it is 1.7 by 1.7 by 1.6 m. The latter values are used here. In addition, a new category LILW was included by Burggraaff et al. (2022) namely dismantling and decommissioning. Its volume has been estimated to be 3,814 m^3 and it is expected that the Konrad type 2 container will be used for disposal without any conditioning. Like (TE)NORM, the expected number of containers for disposal is calculated by dividing the expected volume for disposal by the volume of the Konrad type 2 container.

For HLW, legacy waste (Verhoef et al., 2017) was omitted. This is because better characterization of the waste allowed the major volume of the waste to be characterized as LILW, rather than HLW. For the other HLW types, there are some differences with Verhoef et al. (2017). The estimation of the number of CSD-c canisters has been reduced from 600 (Verhoef et al., 2017) to 502 while the number of ECN canisters has increased from 150 to 244. This agrees with the waste inventory determined by Burggraaff et al. (2022). The number of CSD-v canisters has not changed.

In terms of disposal, 40 disposal rooms are needed for the disposal of LILW with most of the disposal rooms needed for the disposal of (TE)NORM. As shown in Chapter 4, it is expected that the disposal of LILW will require a surface area of 0.31 km^2. For the HLW, 30 disposal tunnels are expected to be needed and would require a surface area of about 0.26 km^2. Both are small enough to be constructed in different types of salt structures.

12.4.2 Appendix: Waste Scenario 2

Waste Scenario 2 is thus like Waste Scenario 1, with the expectation that in this scenario, the Borssele nuclear powerplant will remain open until 2044: an additional ten years. To determine the increase in the volume of waste, it is assumed that the volume of HLW (5.6 m^3) and LILW (70 m^3) currently produced per year by the Borssele NNP does not change. To calculate the number of waste packages expected for disposal, the increase in volume is divided by the volume of each type of waste. This results only in a slight increase in LILW which does not require the construction of additional disposal rooms. The area needed for the disposal of LILW is thus like Waste Scenario 1: 0.31 km^2. For HLW, the extension of Borselle NNP will result in an increase of 116 CSD-v canisters and 140 canisters of CSD-c. As each HLW package can contain 6 canisters, only 43 additional HLW packages will be required. With each disposal tunnel having 10 HLW packages, only five additional disposal tunnels are needed. With two disposal tunnels (each on one side of the transport tunnel, see Chapter 4) increasing the dimension of the repository by 35 m (30 m of spacing between tunnels plus 5 m for the width of the tunnels). Consequently, its length will increase with 105 m and area needed for disposal of HLW will be about 0.30 km^2 assuming a similar two-level repository as presented in Chapter 4. In terms of cost for disposal, it will increase but probably be very limited as only additional 22 days (see section 4.5) would be needed to place the waste in the repository while only five additional disposal tunnels would be needed which construction will take less than a year (see section 4.5).

12.4.2 Appendix: Waste Scenario 3

In Waste Scenario 3 is like Waste Scenario 2 but includes the construction of two additional new NPP in line with what the Dutch government is currently considering. To determine the volume of waste expected for disposal, the volume of HLW and LILW produced by the two new NPP are expected to have a combined capacity of 3200 MW. This is about 6.6 times more capacity than the Borssele NNP (485 MW) which produces 5.6 m^3 of HLW and 70 m^3 of LILW per year. Thus, the two new NPP combined are expected to produce 6.6 times more waste per year: 37 m^3 of HLW and 500 m^3 of LILW in the form of 200L and 1,000L: resins immobilized liquid waste (containers) per year. To calculate the number of waste packages expected for disposal, like in the previous scenarios, the increase in volume is divided by the volume of each type of waste. Like Waste Scenario 2, this implicitly assumes that the characteristics of the waste arising from the two new NPP's are the same as the waste that is currently produced by Borssele NPP: i.e., recycling of spent nuclear power fuel in France. In addition, the dismantling waste is also assumed to increase with the power of the plant by which its volume is more than four times larger than in Waste Scenarios 1 and 2.

In terms of disposal, the significant increase in LILW requires the construction of an additional 28 disposal rooms compared to the two other Waste Scenarios. With six additional disposal rooms requiring an additional 105 m (5 m for the service tunnel and 100 m for the disposal rooms), the upper level would increase in length by 490 m. Thus, the area needed for the disposal of LILW is expected would be around 0.5 km^2 (1260 by 400 m) assuming a similar two-level repository as presented in Chapter 4. The increase is thus limited and might be even further reduced by 21 if depleted uranium could be used as aggregates in cementitious materials to backfill open spaces (see section 4.7.3.2): only seven additional disposal rooms need to be constructed. The volume of HLW expected for disposal in this scenario requires the construction of 267 disposal tunnels which would require an area of around 2.2 km^2 (5480 by 400 m) assuming a similar two-level repository as presented in Chapter 4. This increase is significant and could mean that some salt structures might be too small in terms of dimensions to be used. To reduce the area of the repository, two levels for the disposal of HLW could be constructed rather than a single level. This would decrease the area needed for the disposal of HLW by a factor of two. Furthermore, the spacing between the HLW packages could potentially be reduced, resulting in more HLW packages in a single disposal tunnel. However, this requires more research, specifically on the thermal evolution of the repository as the heat generating HLW of the two new NPPs will probably be warmer when disposed of. This does not necessarily pose a problem, as salt has a relatively high thermal conductivity. Alternatively, two or more HLW packages could be lowered in a single borehole capped by a salt seal. This would also reduce the surface area required for disposal and potentially increase the number of salt structures that could be used.

Type	Waste scenario 1 Current installations + Expected for disposal		Waste scenario 2 + +10 Expected for disposal		Waste scenario 3 + Expected for disposal	
200 L drums	100,000	31,461 m³	100,000	32,161 m³	300,000	8,4646 m³
1000L : resins immobilized liquid waste (Containers)	2,800		2,900		8,500	
1000L : molybdenum waste liquid 1 (Containers)	4,200		4,200		4,200	
1000L : molybdenum waste liquid 2 (Container)	1,400		1,400		1,400	
Decommissioning waste	826 (3,814 m³)		826 (3,814 m³)		3600 (16,299 m³)	
(TE)NORM	12600 (58,070 m³)		12600 (58,070 m³)		12600(58,070 m³)	
CSD-V	478 (504 m³)		618 (649 m³)		7,918 (8,316 m³)	
CSD-C	502 (530 m³)		670 (706 m³)		9,382 (9,853 m³)	
ECN-canister	244 (634 m³)		244 (634 m³)		244 (634 m³)	

Figure Appendix 4.1) The expected number of waste packages for disposal for the 3 different scenarios. The first column gives the number of waste packages expected for disposal in 2130. In parentheses is the volume of waste expected for disposal for the LILW (all types together) and the number of canisters for HLW followed by the volume of waste expected for disposal. For the latter, it is assumed that the HLW package for CSD-C and CSD-V have a volume of 6.3 m³ and contain up to 6 canisters. The HLW package for the ECN canisters is assumed to have a volume of 5.2 m³ and can contain up to 2 ECN canisters.

	Dimensions waste packages			*Number of waste packages*			
Waste package	Height (m)	Length (m)	Width / Diameter (m)	Height (Number of waste packages)	Length (Number of waste packages)	Width (Number of waste packages)	Number per room
200 L drums	0.88	-	0.59	6 (3.54 m)	100 (88 m)	15 (8.85 m)	9000
1000 L containers	1.25	-	1	3 (4.75 m)	88 (88 m)	9 (9 m)	2376
Conrad Type 2	1.7	1.7	1.6	2 (3.4 m)	52 (88.4 m)	6 (9.6 m)	624

Figure Appendix 3.2) For LILW, 3 different types of waste packages are expected for disposal namely the 200L drum, the 1,000L containers (Molybdenum, resins, and immobilized liquid waste) and the Konrad type 2 container ((TE)NORM and dismantling waste)). The first three columns show the dimensions of each waste package. Columns 3 to 6 show the number of waste package that can be placed in either the length, width, or height of a disposal room (100 by 10 by 6 m) with between parentheses the exact number of metres needed. Hence, a total of three 100L containers can be stacked on top of each other which would require a height of 4.57 m: less than the height of the disposal room. Note that the 200L drums will be disposed in the same way in which they are currently stored at COVRA: they are placed horizontally. In total, 9000 200L drums, 2375 1,000L containers and 624 can be disposed of in 1 single disposal room.

	Waste scenario 1 Current installations +	Waste scenario 2 +	Waste scenario 3 +
Type	Disposal rooms/tunnels needed	Disposal rooms/tunnels needed	Disposal rooms/tunnels needed
Number of disposal rooms for 200 L drums	12	12	34
Number of disposal rooms for 1000L : resins immobilized liquid waste (Containers)	2	2	4
Number of disposal rooms for 1000L : molybdenum waste liquid 1+2 (Containers)	3	3	3
Decommissioning waste	2	2	6
Number of disposal rooms for (TE)NORM	21	21	21
Total number of disposal rooms	40	40	68
Number of disposal tunnels for CSD-V/C	17 (80/84)	22 (103/112)	289 (1320/1564)
Number of disposal tunnels for ECN canisters	13 (122)	13 (122)	13 (122)
Total number of disposal tunnels	30	35	302

Figure Appendix 3.3) The expected number of disposal rooms (LILW) and disposal tunnels (HLW) needed for the 3 different scenarios. For HLW, the number of waste packages is shown in the parentheses for each type of waste (CSD-C, CSD-V and ECN canisters). As each disposal tunnel can contain up to 10 waste packages, the number of disposal tunnels needed is calculated by taking the total number HLW package for each type of waste (CSD-C/V and ECN canisters) and divide it by 10.

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This report presents an overview of the results and conclusions of the Safety Case for a geological disposal facility in Permo-Triassic salt of the Netherlands. The report is a scientific/technical document that describes engineering and geological requirements needed to assure that a safe GDF can be implemented in the Netherlands.

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