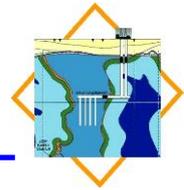


**Status of the safety concept  
and safety demonstration  
for an HLW repository in salt**

Summary report

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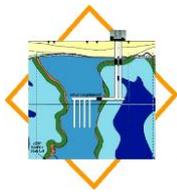
Peine, December 2013



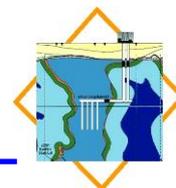
The work that forms the basis of this study was carried out by BGR (*Federal Institute for Geosciences and Natural Resources*) on behalf of BMWi (*Federal Ministry of Economics and Technology*) and by GRS and DBE TECHNOLOGY GmbH on behalf of BMWi, represented by the *Project Management Agency Karlsruhe, Water Technology and Waste Management (PTKA-WTE)*, *Karlsruhe Institute of Technology* under the project numbers 02 E 10719 and 02 E10729. The authors alone, however, are responsible for the contents of this study.

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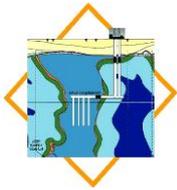
*"Überprüfung und Bewertung des Instrumentariums für eine sicherheitliche Bewertung von Endlagern für HAW"*

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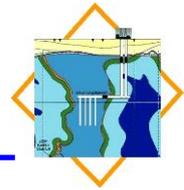
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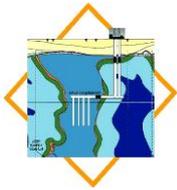


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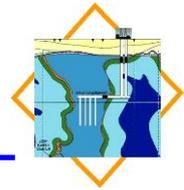
## Abstract

Salt formations have been the preferred option as host rocks for the disposal of high level radioactive waste in Germany for more than 40 years. During this period comprehensive geological investigations have been carried out together with a broad spectrum of concept and safety related R&D work. The behaviour of an HLW repository in salt formations, particularly in salt domes, has been analysed in terms of assessment of the total system performance. This was first carried out for concepts of generic waste repositories in salt and, since 1998, for a repository concept with specific boundary conditions, taking the geology of the Gorleben salt dome as an example. Suitable repository concepts and designs were developed, the technical feasibility has been proven and operational and long-term safety evaluated. Numerical modelling is an important input into the development of a comprehensive safety case for a waste repository. Significant progress in the development of numerical tools and their application for long-term safety assessment has been made in the last two decades. An integrated approach has been used in which the repository concept and relevant scientific and engineering data are combined with the results from iterative safety assessments to increase the clarity and the traceability of the evaluation. A safety concept that takes full credit of the favourable properties of salt formations was developed in the course of the R&D project ISIBEL, which started in 2005. This concept is based on the safe containment of radioactive waste in a specific part of the host rock formation, termed the containment providing rock zone, which comprises the geological barrier, the geotechnical barriers and the compacted backfill.

The future evolution of the repository system will be analysed using a catalogue of Features, Events and Processes (FEP), scenario development and numerical analysis, all of which are adapted to suit the safety concept. Key elements of the safety demonstration are the integrity proofs for the geological and geotechnical barriers and analysis of backfill compaction. In addition, any possible radionuclide release from the repository to the environment has also to be assessed.

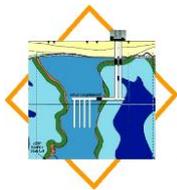
The safety and demonstration concept developed in the course of the ISIBEL project was further evolved and applied in the course of the R&D project "Vorläufige Sicherheitsanalyse Gorleben – VSG" (preliminary safety analysis Gorleben) as an example for an HLW repository in a domal salt structure. The repository concepts also consider the requirement for retrievability of stored waste during the operational phase of the repository. The results of the R&D project VSG provide evidence that a safe HLW repository within a salt dome of a suitable geologic structure is feasible. The long-term safety can be ensured using state-of-the-art science and technology.

In 2010, the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) issued new safety requirements for the disposal of heat-generating radioactive waste. These requirements have been included in the analysis.



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This study shows the depth of the geological and technical knowledge on final disposal of HLW in a salt dome with a suitable geologic structure and demonstrates that the tools required for safety evaluations are available and allow reliable safety assessments of HLW repositories in salt formations.



## 1 Introduction

Rock salt as a disposal medium for High Level Radioactive Waste, including spent fuel and reprocessing waste (HLW)<sup>1</sup>, has been investigated in Germany for more than forty years. German research institutions and companies, some directly and some indirectly involved in radioactive waste management, have performed comprehensive R&D work on conceptual and safety related topics. Several R&D projects have been completed which in combination evaluate the behaviour of an HLW repository in a salt formation in terms of the performance of the total system. These projects include the characterisation and description of the geologic host formation, the development of a repository design together with a feasibility assessment, and an evaluation of both operational and long-term safety. Prior to 1998 the German nuclear waste management programme focused on the development of generic waste repository concepts in salt domes. After this time site specific studies, using the Gorleben geology as an example, were started. Recent research has provided significant insights into the performance of a repository in rock salt. These insights have been integrated into the development of a repository reference concept and have contributed to the safety assessment.

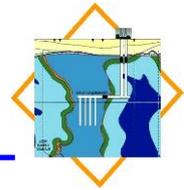
The generally accepted approach to demonstrating compliance of repository behaviour with performance goals has been to use numerical models with respect to specific legal requirements such as safety limits. More advanced programmes now combine multiple lines of reasoning in the development of a safety case. In the more recent integrated approach, numerical modelling becomes one of many inputs in the development of a comprehensive safety case.

The clarity and traceability of the integrated approach is based on a combined study of the repository concept, the availability of relevant scientific and engineering information, and the results from iterative safety assessments.

In 1999 the German Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) brought together a scientific working group tasked with defining scientific criteria as a basis for an HLW repository site selection process. The initial output of this working group was a comprehensive review of the international state of the art in HLW disposal safety concepts including performance demonstration. As a result of this review the focus of safety work moved away from assessments of worst case scenarios, including water intrusion followed by radionuclide releases, to the safe containment of radioactive waste within the host rock based on the performance of the geologic system as a complete entity. This new approach allows the inherent containment capabilities of the host rock to be fully reflected in safety assessments. This containment approach now forms the basis of BMU's Safety

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<sup>1</sup> In Germany spent nuclear fuel (SNF) for disposal is included in the waste category "High level radioactive waste". Therefore in the following text the term "HLW" means "spent fuel" and "high level reprocessing waste".



Requirements [BMU 2010a]<sup>2</sup>. A further output, related to scenario development, was a classification into probable, less probable and improbable future repository evolutions and graduated safety limits, these being based on the likelihood of occurrence.

R&D project ISIBEL, i.e. the Analysis and Evaluation of Tools for the Safety Demonstration of HLW Repositories in Salt Formations, was started in 2005. The overall objective of the study was to summarise the state of the art in HLW disposal in salt formations and to evaluate whether a demonstration of technical feasibility and repository safety was possible. For the first time, a concept that took full credit of the favourable properties of salt formations, reflecting the concept of safe containment, was developed and tested. Demonstration of the safe containment requires proof of the integrity of the geological and geotechnical barriers in the very long-term as well as an analysis of backfill compaction.

Assessment of any possible radionuclide release has to be conducted for future evolutions of the repository system that involves possible impairment of barrier system performance and the evolution of continuous transport pathways to the biosphere. An evaluation of the probability of this is conducted as part of scenario development.

As a specific example for the geology of domal salt structures, in this study exploratory data from the Gorleben salt dome were used to test the application of the new approach. Results from Gorleben give important insights into the status of the integrity of the host rock. For example it has been clearly demonstrated that the anhydrite layers were fragmented during the uplift of the salt dome and because of this there is no continuous fluid pathway through the anhydrite to the biosphere. Additionally a methodology was developed to demonstrate the integrity of the geotechnical barriers.

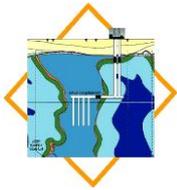
The fundamentals of a methodology for scenario development were derived and also a catalogue of Features, Events and Processes (FEP) for a repository in salt – reflecting all aspects which are relevant for the future evolution of the repository system - was developed. A broad spectrum of procedures, models and tools is available to assess scenarios which include radionuclide releases. Confirmation was obtained of the suitability of these tools for modelling of the relevant processes in the repository system and for the analysis of consequences.

During the second phase of R&D project ISIBEL (2008 - 2010), project work focused on addressing deficiencies and open issues which had been identified in the safety demonstration methodology. These included:

- Development and application of an appropriate methodology for scenario development focusing on the development of reference scenarios,
- independent peer review of the FEP catalogue developed during the first phase of R&D project ISIBEL,

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<sup>2</sup> In the following referred to as “Safety Requirements”



- demonstration and assessment of the “safe containment“ concept,
- assessment of uncertainties within long-term safety analysis, and
- development of a proposal for the structure and the content of a safety case report for an HLW-repository in salt formations.

As a result, the main issues identified for the follow-up phase of the R&D project ISIBEL included the following:

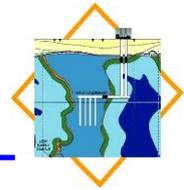
- advanced investigations on the long-term safety demonstration for geotechnical barriers;
- evaluation of specific waste forms (i.e. spent fuel from research and prototype reactors, special kinds of low and intermediate level radioactive waste);
- investigation into process analysis and modelling tools;
- assessment of gas production and migration; and
- the applicability of natural analogues in safety assessments.

A comprehensive and balanced evaluation of the relevant aspects for long-term safety was made possible by combining the diverse skill sets of the BGR, DBE TECHNOLOGY GmbH and GRS institutions, each contributing different areas of expertise related to HLW disposal. Project results were judged to be applicable to future safety analysis and safety demonstration. Together they provided the basis for the ‘Preliminary Safety Case Gorleben’ as is discussed in [Fischer Appelt et al. 2013].

In 2010 BMU notified that the suitability of the Gorleben salt dome as a site for an HLW repository would be analysed in a multi-level procedure based on a preliminary safety analysis, an upgraded repository concept and an international peer review [BMU 2010b]. GRS was contracted by BMU with the R&D project "Vorläufige Sicherheitsanalyse Gorleben – VSG" (preliminary safety analysis Gorleben). A multi-disciplinary research group was brought together including seven German research institutions and companies. Initially, the projective objectives focused on BMU's decision quoted above, and summarised below:

- based on an assessment of the results of the integrity analysis for the Gorleben salt dome and of the radiological impact analysis and taking into account the Safety Requirements, the suitability of the Gorleben salt dome for HLW disposal should be evaluated,
- site specific repository concepts considering operational safety should be developed and optimized, and
- identified research and development needs should be summarised.

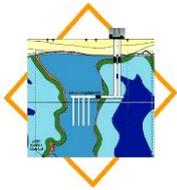
In August 2012 the project objectives were modified in such a way that no suitability statement for the Gorleben site would have to be given in the course of the project. Furthermore it was to be analysed whether the Safety Requirements could be met by the repository concepts and geological barrier such as exist at the Gorleben site.



The practical application and further development of the safety and demonstration concepts originally developed in the ISIBEL work were important issues for this project. In this context, the current status of safety demonstration was examined and exploratory data from the Gorleben salt dome formed the geologic basis for this assessment. Furthermore, different site-specific repository concepts for spent fuel, reprocessing waste and negligible heat generating waste, that is waste deemed not appropriate for disposal at the Konrad repository, were developed. These concepts also take into account the requirement for retrievability during the operational phase of the repository the technical design also included concepts for sealing and backfilling the underground facility. The basic principle implemented in the safety concept was the containment of the radioactive waste in the rock-zone providing the containment.

The containment function must become effective immediately upon repository closure and must be provided by the repository on a permanent and maintenance-free basis. Brine intrusion potentially affecting the waste must be either prevented or extremely limited. The generic FEP catalogue developed under R&D project ISIBEL was adapted to the site-specific conditions and further detailed. The ISIBEL methodology for scenario development was supplemented by a new approach for the development of alternative scenarios. For the evaluation, the scenarios are first transformed into conceptual models which can then be used for numerical calculations. Finally, proof of integrity was demonstrated for the geological and geotechnical barriers, and radionuclide releases under the different scenarios were calculated

This report provides an overview of the state of the art in safety analyses and safety assessments for HLW repositories in domal salt formations.



## 2 Specifications in the Safety Requirements

The specifications on HLW disposal in Germany are defined by the Safety Requirements governing the final disposal of heat generating radioactive waste. The relevant specifications regarding the development of a safety concept and a corresponding safety demonstration are summarised in this chapter.

The protection goals of the Safety Requirements are to protect man and environment. Unreasonable burdens and obligations for future generations should be avoided. These protection goals are to be achieved by a final disposal concept that is based on the disposal of the radioactive waste *in a deep geological formation with a high containment capacity*<sup>3</sup>.

The protection objectives and basic assumptions are expressed by eight safety principles that are defined in the Safety Requirements. In the context of the safety concept and the safety demonstration, the following three safety principles should be mentioned.

- Safety principle 4.1: *The radioactive and other pollutants in the waste must be concentrated and contained in the containment providing rock zone (CRZ)<sup>4</sup>, and thus kept away from the biosphere for as long as possible.*
- Safety principle 4.2: *Final disposal must ensure that in the long term, any release of radioactive substances from the final repository only negligibly increases the risks associated with natural radiation exposure.*
- Safety principle 4.6: *The final repository shall be constructed and operated in such a way that no intervention or maintenance work is required during the post-closure phase to ensure the reliable long-term containment of the radioactive waste in the containment providing rock zone.*

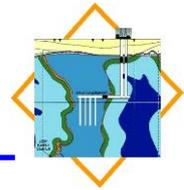
Whilst the safety concept described in chapter 3 has to consider all safety principles, safety principles 4.1, 4.2 and 4.6 above are of particular importance. The determination of the CRZ and the confirmation of its containment capacity, by proving the integrity of the geological barrier and the geotechnical barriers, are key elements for the safety demonstration (see chapter 5).

The Safety Requirements stipulate that the demonstration of the safety of the repository should be documented in a comprehensive safety case and that this safety case shall be documented for all operating states of the final repository. An important part of the safety case is a safety analysis and safety assessment covering a period of a million years that

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<sup>3</sup> Text written in italics is directly taken from the English translations of the Safety Requirements.

<sup>4</sup> The English translation of the Safety Requirements use the expression „isolating rock zone“ and defines this zone as *the part of the repository system which, in conjunction with the technical seals ensure the containment of the waste*. Since this zone refers explicitly to the safety function “containment”, the term “containment providing rock zone” is used in this report.



must be carried out to provide evidence of long-term safety. This analysis comprises the following aspects:

- description of the geological situation and the final repository concept,
- the identification, characterisation and evaluation of safety-relevant features, events and processes,
- the comprehensive identification and analysis of safety-relevant scenarios and their allocation to probability categories,
- strategy for the identification, evaluation and handling of uncertainties,
- long-term statement on the integrity of the CRZ,
- proof of the integrity of the technical components during their functional period,
- long-term radiological statement,
- proof of subcriticality, and
- monitoring and evidence preservation programme.

All of these required fundamental aspects were dealt with in the R&D projects ISIBEL and VSG, with the exception of the description of a monitoring and evidence preservation programme. The safety concept and the safety demonstration are described in chapters 3, 4 and 5.

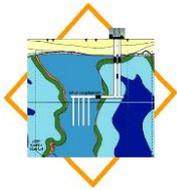
A fundamental aspect concerning all parts of the safety concept and the safety demonstration is the handling of scenario uncertainties. The Safety Requirements stipulate a scenario analysis and the consideration of three types of evolutions (developments<sup>5</sup>) of the repository system:

*Probable evolutions refer to normal evolutions forecasted for this site, and evolutions normally observed at comparable locations or similar geological situations. The forecasted normal evolution of properties should be used as a basis when considering the technical components of the final repository. If quantitative data on the probability of a certain evolution occurring is available, and the probability of it occurring in relation to the reference period is at least 10%, this shall be considered a probable evolution.*

*Less probable evolutions refer to evolutions which may occur for this site under unfavourable geological or climatic assumptions and which have rarely occurred in comparable locations or comparable geological situations. A consideration of the technical components of the final repository should be based on the normal forecasted evolution of their properties upon occurrence of the respective geological evolution. Any unfavourable evolution in the properties of the technical components that deviate from normal evolution should also be investigated. Repercussions on the geological environment should be considered. Apart from such reper-*

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<sup>5</sup> In order to avoid confusion between the commonly used term “development of scenarios” and the fact that a scenario is defined as a development in the Safety Requirements, the term “evolution” is used in this report to describe the future of the repository system. If the term “evolution” occurs in a citation and replaces the term “development”, it is written non italic.



*cussions, anticipated geological evolutions should also be taken into account. Within such an evolution, the simultaneous occurrence of several unrelated faults should not be assumed. If it is possible to make a quantitative statement on the probability of a certain evolution or an unfavourable evolution in a technical component's properties, this should be taken into account if the probability in relation to the reference period is at least 1%.*

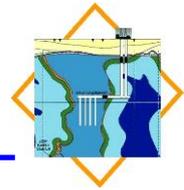
*Improbable evolutions refer to evolutions which are not expected to occur at the site even under unfavourable assumptions, and which have not been observed in comparable locations or comparable geological situations. Statuses and evolutions for technical components which can be more or less excluded by taking certain action, as well as the simultaneous, independent failure of several components, are classed as improbable evolutions.*

For probable and less probable evolutions, evidence must be provided that the radiological criteria defined in the Safety Requirements have been met. For probable evolutions it must be demonstrated that the additional effective dose for an individual due to the release of radionuclides from the repository is, at maximum,  $10^{-5}$  Sv/a. For less probable evolutions the criterion is an additional effective dose of less than  $10^{-4}$  Sv/a. *For improbable evolutions, reasonable risks or reasonable radiation exposure have not been quantified. However, where such evolutions may lead to high radiation exposure, it is necessary to investigate, within the context of optimisation, whether it is possible to reduce such effects with a reasonable input.*

*A simplified radiological long-term statement without modelling the dispersion of substances in the overburden and adjoining rock is permissible if the radioactive substances released from the CRZ lead to a maximum of 0.1 person-millisieverts per year for probable evolutions and a maximum of 1 person-millisievert for less probable evolutions. This ensures that only very low overall amounts of radioactive substances can be released. These person-millisieverts shall be calculated using a recognized generic exposure model for analyses of long-term safety, for which it should be assumed that:*

- The reference group in question contains ten persons who obtain their entire annual water requirement for nutritional purposes, including drinking water, animal watering, crop irrigation, from a well, and that*
- this well water contains all the radionuclides that have escaped from the CRZ in the year under study The dilution of the well water to a mineral content which would permit it to be used as drinking water should be taken into account.*

To conclude, the key elements for the safety concept are the isolation and the containment of the radioactive waste in a deep geological repository. Isolation is provided by the depth of the repository (see chapter 4.3 and figure 2.1). Containment is provided by the CRZ and the integrity of the CRZ in the long-term. For those areas of the CRZ which are penetrated due to the construction of the repository, an adequate technical barrier system must be provided. If it is possible to prove safety with a simplified radiological statement, near-surface processes are not safety relevant.



In defining the CRZ it should be recognised that the determination of its extension can be problematic. A large extension might contradict the inherent intention of the Safety Requirements to concentrate the waste far away from the biosphere, whilst a small extension might contradict the specification of the containment in the zone.

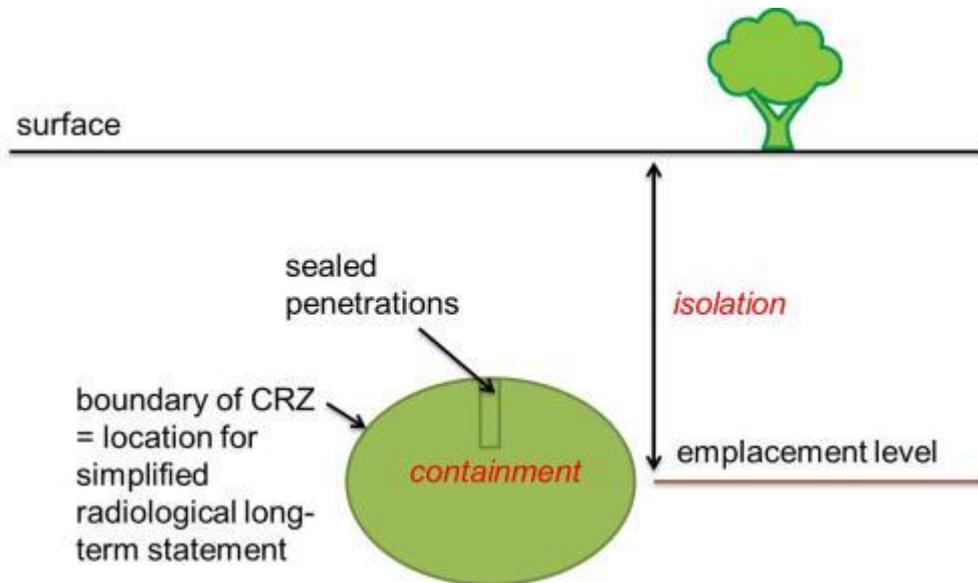
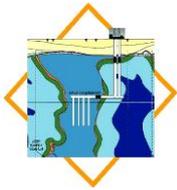


Figure 2.1: Key elements for a safety concept according to the Safety Requirements



### 3 General approach of the safety concept

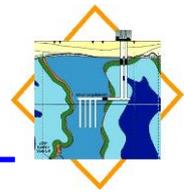
A safety concept for a repository at a given site describes which circumstances and measures contribute to accomplishing and maintaining the required level of long-term safety. In the course of the R&D project ISIBEL the fundamentals for a safety concept for salt formations were developed and a safety concept that takes full account of the advantages of the final disposal of HLW in salt was developed for the first time. The basic concept is to focus on the systematic demonstration of the safe long-term containment of the waste. The relevant barriers for safe containment are the salt rock, the shaft seals and the drift seals. Any void volume in emplacement areas is to be backfilled with crushed salt which will be naturally compacted by convergence. During compaction, the porosity and permeability of the crushed salt decreases until, in the long run, it has the same barrier properties as rock salt.

This general safety concept for a repository in a salt dome was upgraded and described in more detail in R&D project VSG. Based on the safety principles set out in the Safety Requirements, and on existing knowledge concerning the processes that could impair the safety of the repository, and the site properties, three guiding principles have been derived as follows:

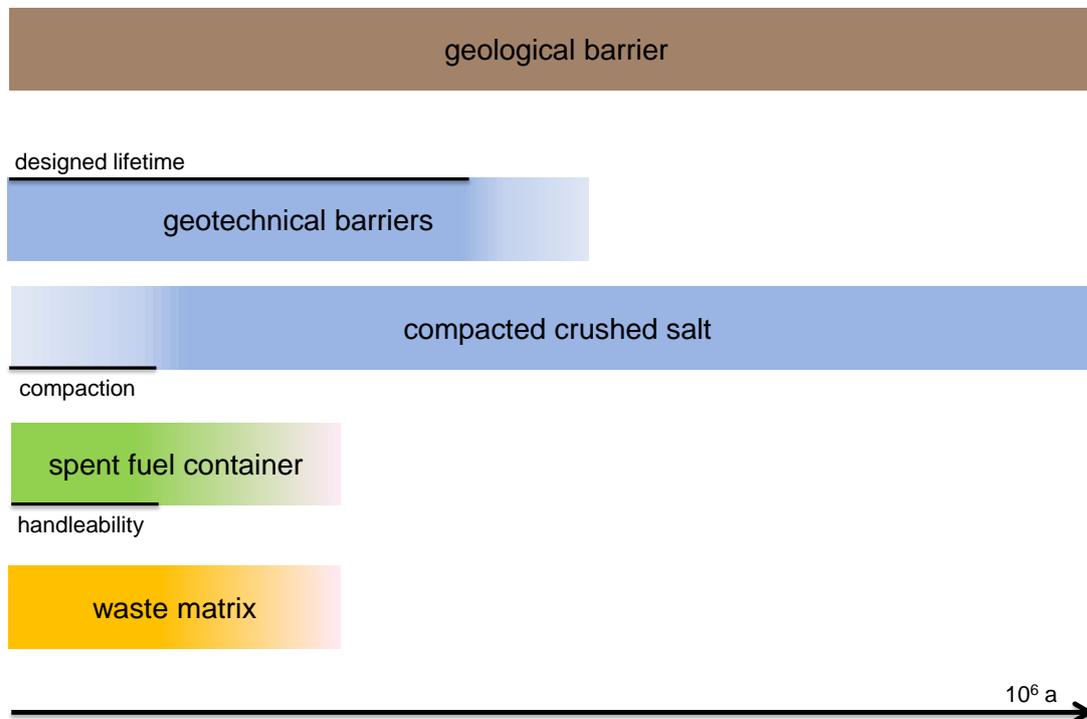
- the radioactive waste must be contained as widely as possible in the CRZ,
- the containment shall be effective immediately post-closure and it must be provided by the repository system permanently and maintenance-free, and
- the immediate and permanent containment shall be accomplished by preventing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

The geological barrier should provide the permanent containment of the radioactive waste. The salt rock within the CRZ is essential for the containment as it is practically impermeable to solutions. Accordingly the integrity of the salt rock within the CRZ must be ensured.

A penetration of the geological barrier is inevitable during mine construction and will result in its local impairment. Creep processes promoted by the visco-plastic-elastic properties of the salt rock will lead eventually to the closure of such mine openings, thus restoring the original properties of the geological barrier. Since this process requires some time, engineered high-performance shaft seals and drift seals that provide the required sealing immediately upon construction will be built. These engineered barriers may be affected by thermal and mechanical impacts and by chemical processes and, therefore, their long-term performance may not be irrefutably proven. To guarantee the long-term sealing of the penetrations, the mine workings are backfilled with crushed salt as a long-term stable material. The compaction of the crushed salt, which is driven by salt creep, results in a very low permeability of the crushed salt over a period of time. Evidence must be provided that sealing by the compacted backfill material is fully developed by the time the performance of the engineered barriers can



no longer be demonstrated. The evolution of the seal effect with time is schematically shown in Figure 3.1 for important barriers in the repository system.

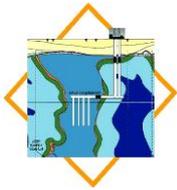


**Figure 3.1:** Evolution with time of the sealing effects of important barriers in the post closure phase of a repository system (the colour intensity represents the degree of the respective sealing effect, not to scale)

According to the Safety Requirements *handleability of the waste containers must be guaranteed for a period of 500 years in case of recovery from the decommissioned and sealed final repository. Care should be taken to avoid the release of radioactive aerosols. During the operating phase up until sealing of the shafts or ramps, retrieval of the waste containers must be possible.*

In the event that brine intrusion to the waste occurs, these barriers in combination with other barriers, contribute to the enclosure of the radionuclides in the CRZ through different processes. This is achieved either by restricting the movement of contaminated solutions along the former drifts or, as in the case of the waste immobilising matrix, by retarding the dissolution of the radionuclides.

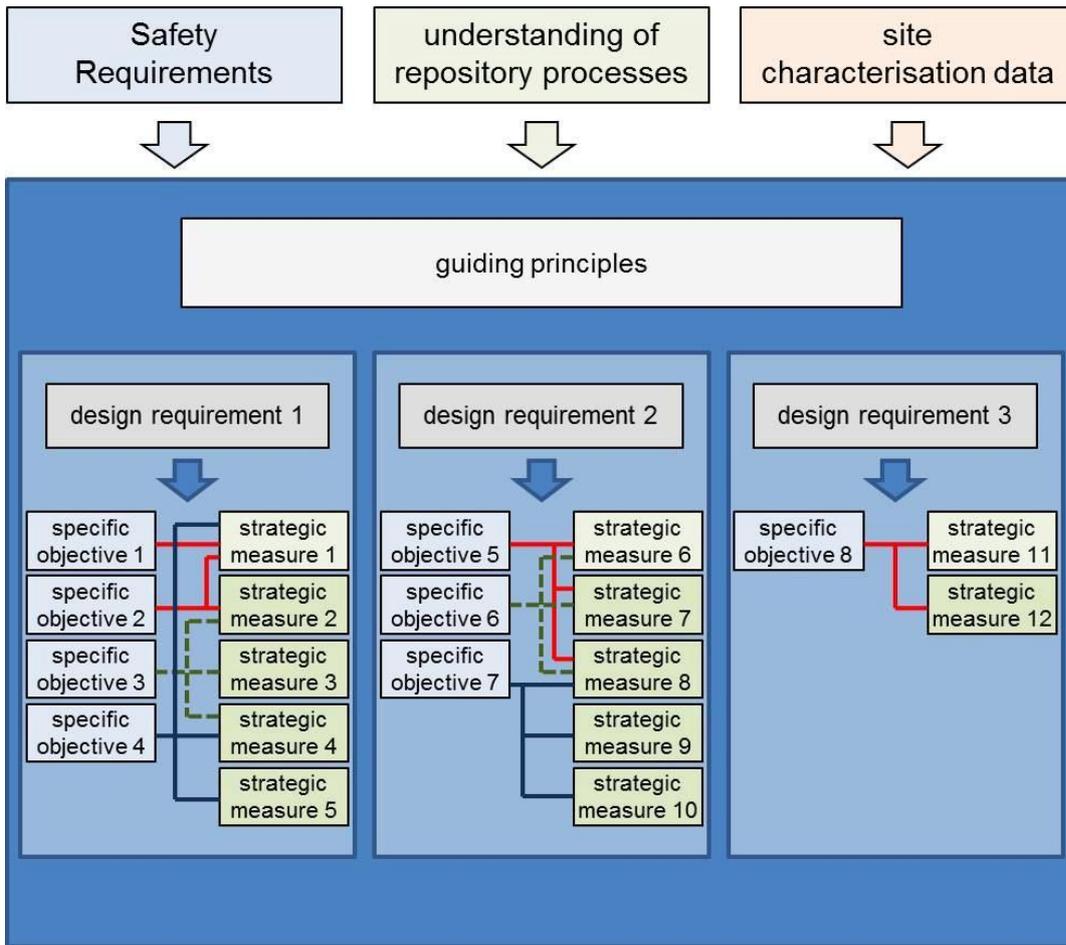
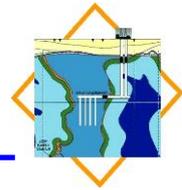
For a further development of the safety concept the guiding principles were reinforced by the definition of two further design requirements whilst a third additional design requirement stems from the regulatory requirement to avoid criticality in the repository.



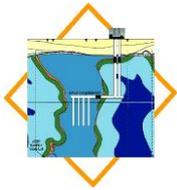
- Containment: The emplaced waste canisters shall be enclosed quickly and as tightly as possible by the salt.
- Performance of CRZ: During the demonstration period of one million years, the CRZ shall remain intact and its barrier function shall not be impaired by internal or external processes and effects,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

These design requirements were then used to derive specific objectives and to determine strategic measures which embrace design specifications, for example with respect to the mine position in the salt dome, and technical provisions. Typically, each strategic measure supports a number of specific objectives. The strategic measures in their entirety combine to meet the objectives of the safety concept. The principle types of correlation between design requirements, specific objectives and measures are schematically shown in Figure 3.2.

This approach allows mapping of the general stipulations of the Safety Requirements to objectives and measures for the safety concept of a given site to be shown in a transparent way. An implementation of this general approach is described in chapter 4.



**Figure 3.2: Principle approach to derive specific objectives and strategic measures for the safety concept**



## 4 Implementation of the safety concept

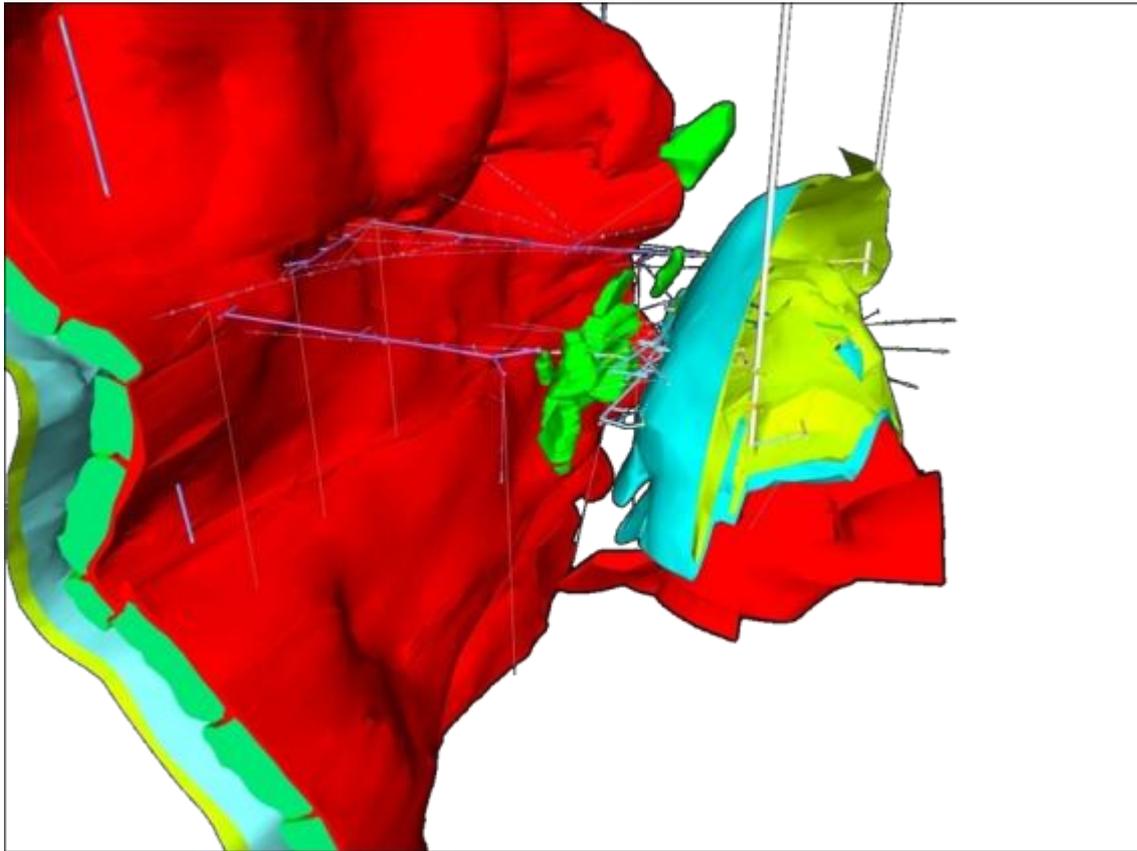
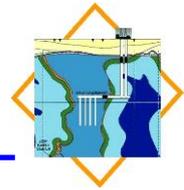
Implementation of the safety concept requires its application to a specific site. This was realised in the R&D project ISIBEL by choosing the geological situation at Gorleben as an example of a site with the host rock salt. Thus, the Gorleben site serves as a reference site for the implementation of the safety concept.

### 4.1 Geological situation at the reference site

The geological situation of the Gorleben salt dome has been intensively explored since 1979 when the German government began to investigate its suitability for the construction of a repository. The exploration has not yet been completed. The findings so far are briefly summarised below. A more detailed presentation is given in the following reports: [Bräuer et al. 2011], [Bornemann et al. 2008], [Klinge et al. 2007] and [Köthe et al. 2007].

Detailed information on the salt dome results from a combination of different exploration techniques. The exact and reliable identification of a particular stratigraphic position is possible due to the knowledge of the stratigraphy of each of the Zechstein beds at the Gorleben site and the concomitant consideration of other characteristic features such as mineralogical composition or bromide concentration. Ground Penetrating Radar (GPR) technology was further developed during the investigation of the Gorleben salt dome and was adapted to the needs of underground exploration in salt. This method has a high spatial resolution capacity and enables distances to reflecting structures to be determined to an accuracy of only a few centimetres under optimal conditions

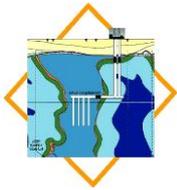
A very detailed geological model of the salt deposit was built up over the course of the underground investigation activities by correlating the underground GPR surveying results with geological information from drifts and exploration boreholes. As an example, Figure 4.1 shows part of the Gorleben 3D geological model and reveals Hauptanhydrit blocks floating within a ductile halite and potash salt matrix.



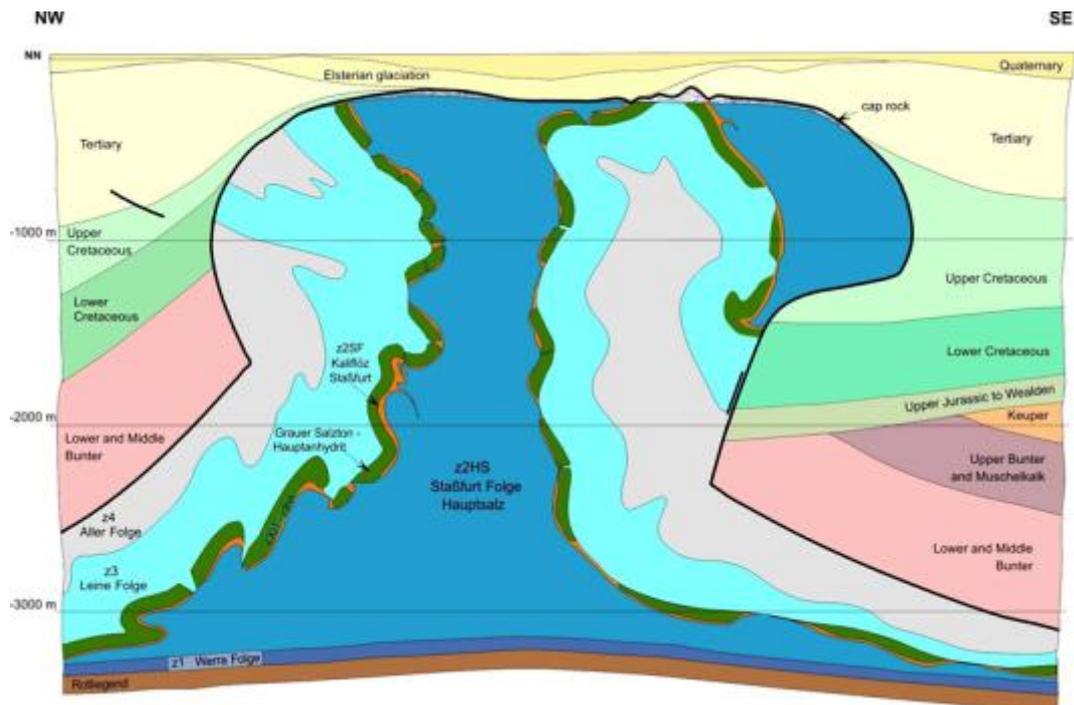
**Figure 4.1:** Detail of the geological 3D model of the Gorleben salt dome including the shafts and galleries of the exploration mine, potash seam (red), Liniensalz (blue) and Hauptanhydrit blocks (green)

#### 4.1.1 The Gorleben salt dome

The Gorleben salt dome has a horizontal outline approximately 15 km long and 4 km wide. The base of the salt dome lies at a depth of 3,200 m to 3,500 m, whilst top salt lies at a depth of approximately 250 metres. The salt dome consists of Zechstein salt sequences, and predominantly contains Staßfurt Series salt sequences in the core of the salt dome and salt sequences from the Leine and Aller Series on its margins. There is known to be a difference in the deformation behaviour of these Series, with Staßfurt rock salt creeping at a faster rate than Leine rock salt. The shafts and infrastructure facilities of the exploration mine were constructed in the Leine rock salt whilst the zones explored as potential emplacement areas were cut in the Staßfurt rock salt because canisters stored there would be more quickly enclosed by the rock mass due to higher creep rates. The salt layers in the transition zone between the Leine and the Staßfurt rock salt Series contain brittle anhydrite beds of Hauptanhydrit sequence and salt clay and carbonaceous layers at the base of the Leine Series and potash salts at the top of the Staßfurt Series. The anhydrite unit was broken up into separate blocks during the halokinetic uprise of the salt. Figures 4.2 and 4.3 show a simplified geological cross-section and a plan view of the Gorleben salt dome at 840 m depth.



The Staßfurt rock salt Series mainly consists of Hauptsalz which was sedimented with an original thickness of 700 m and more. The remaining beds of the Staßfurt Series, including the Staßfurt potash seam, only add up to a few metres in thickness. Hauptsalz consists of around 95 % to 98 % halite and 2 % to 5 % anhydrite. The only other minerals in the upper parts of the Hauptsalz sequence are the minerals polyhalite and carnallite which occurs in isolated clusters of up to 1 cm in size.



**Figure 4.2: Simplified NW-SE geological cross-section of the Gorleben salt dome, [Bornemann et al. 2008], modified**

The top of the salt dome, known as the salt table, is covered by a cap rock with a thickness varying predominantly between 10 m and 50 m. The cap rock consists of altered, residual, constituents of low solubility which are relicts of subrosion of the upper part of the salt dome. The cap rock is overlain by a few Cretaceous relicts and by Tertiary and Quaternary sediments. In some places the Quaternary sediments lie directly on top of the cap rock. In a few restricted places the sediments lie on the salt in the zone of the Gorleben channel which penetrated up to approximately 100 m into the salt dome as a result of glacial erosion processes taking place around 400,000 years ago.

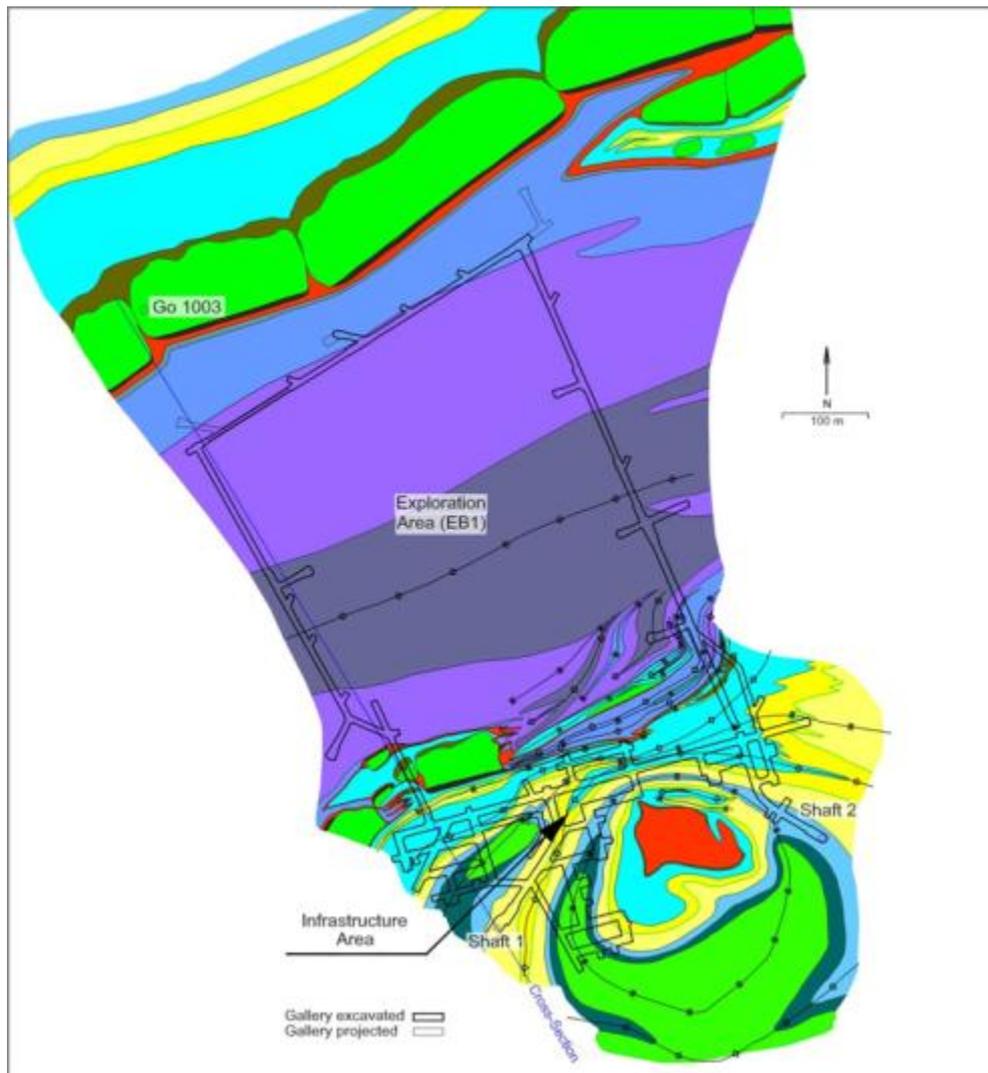
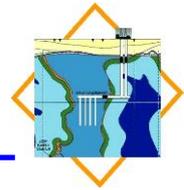
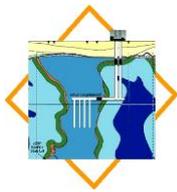


Figure 4.3: Horizontal section of the Gorleben salt dome at the depth 840 m, [Bornemann et al. 2008], modified

#### 4.1.1.1 Formation of the Gorleben Salt Dome

Evaporation of sea water within the North German Basin during the Zechstein period, approximately 260 – 250 million years ago, led to the deposition of salt. There were several cycles involving the inflow of fresh sea water into the basin, the subsequent isolation of the basin and evaporation of the sea water to form salt deposits of considerable thickness. The initial thickness of the evaporitic Zechstein beds in the area around the Gorleben salt dome was up to 1,400 metres.

In the period that followed, the basin subsided and the salt was overlain by various other sediments. The salt was then pressed upwards to form the salt dome because of the ability of salt to creep and its lower density compared to the overlying sediments.



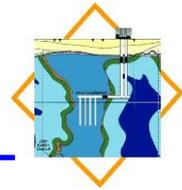
It is generally assumed that in response to the overburden pressure of the overlying sediments, tectonic impulses in the pre-evaporitic rocks initiated the migration of salt into other areas such as into the site of the subsequently formed salt dome. Because of its elongated shape, an initial tectonic impulse of this kind along a basement fault was assumed for the Gorleben salt dome. Seismic surveys with a depth resolution of 50 m carried out specifically to answer this question have, however, failed to find any faults beneath the salt dome.

Interpretation of the stratigraphy and structure of the overlying rocks and the surrounding rocks revealed that changes in salt thickness preceding the formation of the salt dome (development of a salt pillow) had started in the Upper Bunter Sandstone and Muschelkalk periods – in other words, approximately 250 million years ago – and continued into the Dogger up to approximately 170 – 160 million years ago [Zirngast 1991]. Approximately 140 million years ago during the Malm, or in the following Lower Cretaceous period, the salt pillow stage ended and the diapir stage began, i.e. the penetrative break-through of the salt through the overlying sediments. The salt dome continued to develop further during the Cretaceous (approximately 140 – 65 million years), leading to the development of salt overhangs during the Upper Cretaceous and the Tertiary. During these developments, when at times the salt structure reached up as far as the ground surface, around half of the original salt volume was subroded, i.e. dissolved. During the diapiric uplift of the salt dome, the salt travelled distances of up to 5 km laterally and 3 km vertically at flow rates of up to 0.14 mm/year. The average diapiric speed of the top of the salt dome was calculated as a maximum of 0.086 mm/year from the speed of the salt in the catchment area of the salt dome and the largest cross-sectional area of the salt dome at that time. This maximum upward movement took place during the Upper Cretaceous. The upward movement during the last approximately 20 million years took place at an average speed of 0.018 mm/year. During the alternating glacial and interglacial periods in the past, subsidence took place at rates between 0 and 0.4 mm/year or at an average rate over the last 100,000 years of less than 0.1 mm/year [Köthe et al. 2007].

#### 4.1.1.2 Hydraulic Properties of the Rock Salt

The original sedimentary structure changed considerably during the intensive movement of the salt rocks as diapirism proceeded. This is especially true for the particularly mobile Hauptsalz of the Staßfurt Series which forms the inner core of the salt dome. During the diapiric movement of the salt, the effective stresses repeatedly fractured the rock salt and then healed it again. This caused the Hauptsalz to become homogenised into a mixture in which blocks of primary rock salt crystals, and shredded and crushed fragments of anhydrite lines, float in a matrix of recrystallised rock salt.

Original brine trapped in the salt was either squeezed up to the salt table due to diapiric uprise of the salt or squeezed into specific zones, such as the boundary between the Staßfurt Series and Leine Series or the fissured Hauptanhydrit blocks. Dissolution films on grain



boundaries were altered, by the recrystallisation processes, to form isolated fluid inclusions. This means that the rock salt in the Gorleben salt dome outside of fluid accumulations contains no interlinked pore spaces.

Rock salt which has not been affected by mining and is not in the areas where there are accumulations of fluids is therefore practically impervious to diffusion processes and impervious to hydraulic flow processes.

Micro fractures will form around the drifts as a result of the physical excavation process. Drifting therefore gives rise to excavation damaged zones (EDZs) where the barrier effect of the rock salt is diminished. The extent of the mining induced EDZ is typically a function of the depth, the shape of the opening and the time it is unsupported. The growth rate of the EDZ declines in the course of time. Given the depth of the exploration mine of 840 m and the width of the drifts of less than 10 m, the extension of the EDZ around an individual drift will not exceed a few decimetres. Borehole tests in individual drifts at the exploration level within the Gorleben salt dome only encountered raised permeabilities in the proximal zone around the drifts extending distances of less than 2.5 m into the rock salt from the walls of the drifts.

#### 4.1.1.3 Mechanical Properties of Rock Salt

The static moduli of elasticity measured in the rock salt from the Gorleben salt dome are 33 to 36 GPa. The static Poisson's ratios in all the different types of rock salt lay between 0.25 and 0.32. Brazilian tests carried out to determine the tensile strength of the different rock salt types produced values between 1.5 and 2 MPa. At deformation rates of  $10^{-5}/s$ , the uniaxial compressive strengths measured for the various rock salt types lay between approximately 20 and 35 MPa. The measured triaxial strength limits are shown in Figure 4.4.

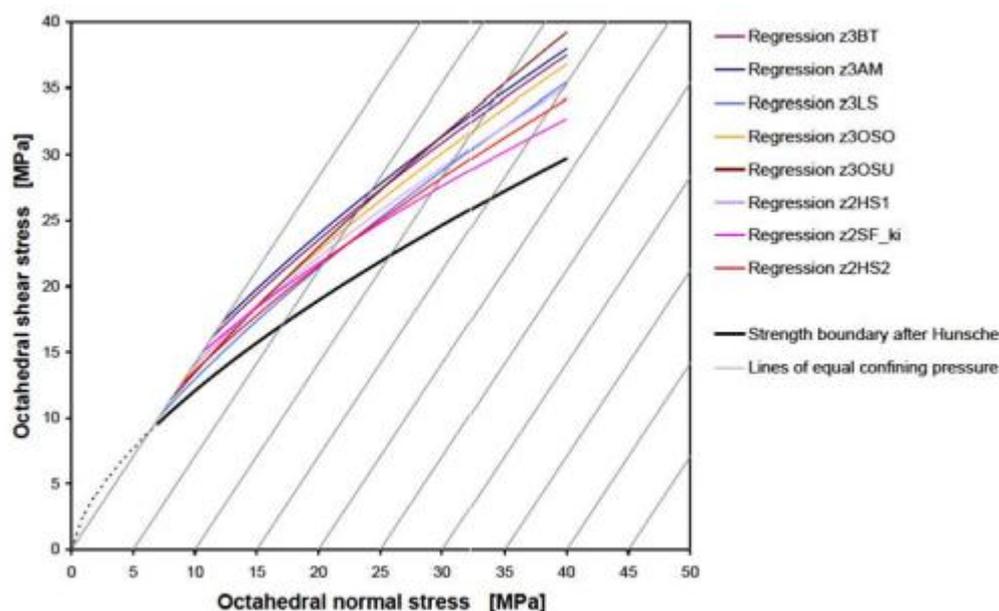
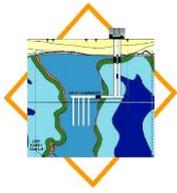


Figure 4.4: Failure strengths in Gorleben salt samples [Bräuer et al. 2011]



The measured strengths were lower at higher temperatures. The strengths at around 180 °C were around half of the strengths at room temperature.

Although the deformation of the rock salt associated with stresses exceeding the strength limits can damage the salt, rock salt can creep when the stress conditions remain below certain stress limits. This means that salt can dissipate the mechanical stresses by constant-volume deformation. The salt remains completely undamaged during this process [Cristescu & Hunsche 1998]. The creep deformation over time depends on the stress state, the temperature and the salt type. There are various creep laws which describe this dependency, see, for example [Bräuer et al. 2011].

The creepability of the Hauptsalz of the Staßfurt Series is generally higher than that of the rock salt types in the Leine Series.

#### 4.1.1.4 Thermal Properties of the Rock Salt

The natural in-situ temperature at the depth of the exploration level is approximately 311 K which is around 5 K above the temperature expected at a depth of 840 m below ground level when using the average continental geothermal gradient.

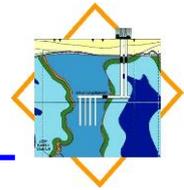
This temperature elevation is due to the heat flow density which, at approximately 115 mW/m<sup>2</sup>, is about twice as high inside the salt dome as in the surrounding rocks. This is caused by the high thermal conductivity of the salt, measured on samples from the Gorleben salt dome at around 5.5 W/(m K) at 293 K and around 3.5 W/(m K) at 473 K.

The linear thermal expansion coefficient of rock salt from the Gorleben salt dome varies between 3.5 and 3.9 x 10<sup>-5</sup>/K at temperatures between 293 K and 373 K.

The specific thermal capacity was determined to be between 0.85 and 0.90 kJ/(kg K) at temperatures between 293 K and 523 K.

#### 4.1.1.5 Brine Reservoirs

Brines occur as natural constituents in salt structures, however they are not randomly distributed but occur in specific stratigraphic horizons. When salt sediments accumulate, large volumes of water are initially present between the still uncompacted grains of salt. As the overburden pressure increases, the water is progressively squeezed out and the degree of interconnectedness of pores becomes lower by recrystallisation of grain contacts. This reduces the possibility for water to be extruded into the overlying beds, and fluids therefore collect preferentially in those areas where the load is dissipated to a lesser degree via the fluids, in other words, in more competent rocks. The mobilisation of the formerly more ho-



mogenously distributed fluids and their concentration within localised reservoirs is proportional to the amount of rock deformation which takes place – the more intensive the rock deformation, the more thorough the mobilisation and concentration.

In the Gorleben salt dome, the preferential locations for brines are the Hauptanhydrit, the Gorleben-Bank, the Anhydritmittel (anhydrite horizons) of the Anhydritmittelsalz and its accompanying layers, and the transition zone between the Staßfurt Series and Leine Series.

No accumulations of solutions were encountered in the intensely folded Hauptsalz of the Staßfurt Series.

When solution accumulations were encountered in the Gorleben salt dome, a pressure measurement was first made to verify the isolated nature of the reservoir. Since an isolated solution reservoir is affected by formation pressure which is higher than the hydrostatic pressure of a water column at the same depth, the presence of an elevated pressure confirms that the reservoir is isolated.

The volume of the fluid is then estimated by carrying out a material balance. Because the solutions are accompanied by gas, but the liquid-gas ratio is unknown, considerable uncertainties arise concerning the compressibility of the fluid in the reservoir. This in turn gives rise to uncertainties in the calculated minimum and maximum reservoir volumes. The maximum calculated fluid content of a reservoir was several thousand cubic metres. The maximum volume of solution which flowed out of a reservoir was almost 200 cubic metres.

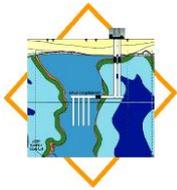
The brines encountered were highly concentrated magnesium chloride solutions. These are relicts of the Zechstein Sea, where the bromine, lithium and rubidium content, and other constituents, are attributable to metamorphosis within the salt dome, and did not involve any solutions originating from outside [Schramm et al. 2009].

In addition to macroscopic solution reservoirs in anhydrite rocks and in accompanying layers, water can also be present at grain boundaries in the salt or in fluid inclusions. Water contents between 0.012 and 0.017 weight per cent were measured in the Hauptsalz of the Gorleben salt dome.

#### 4.1.1.6 Hydrocarbon Occurrences

In addition to the natural occurrence of solutions, a salt dome may also contain hydrocarbons and some of the solution reservoirs within the Gorleben salt dome did contain hydrocarbons. Hydrocarbons were also encountered in the Hauptsalz without significant accompanying aqueous solutions.

Hydrocarbons occur in the Hauptsalz of the Gorleben salt dome in the form of visible staining of the rock salt in clearly localised zones measuring some square decimetres to square me-



tres. The hydrocarbon concentrations in the samples of Hauptsalz from the Gorleben salt dome varied between 0.02 and 443 ppm (parts per million) with respect to weight (mg hydrocarbons/kg rock salt), with a median value of 0.3 – 0.4 ppm [Hammer et al. 2012].

The molecular composition of the hydrocarbons encountered in the Hauptsalz of the Gorleben salt dome was determined by analysis early on in the investigation [Gerling et al. 2002]. Interpretation of this composition, and the variations of the carbon isotopes within the hydrocarbons, indicated that the hydrocarbons did not originate in the Hauptsalz but rather in the Staßfurt Carbonate that is in rocks of the Staßfurt Series deposited before the formation of the Hauptsalz. This interpretation assumes that during the intensive movement of the Hauptsalz during the diapiric growth of the salt dome, the hydrocarbons migrated upwards into temporary fractures which developed in the Hauptsalz and then became trapped and moved to their current positions along with the transported salt. The measured isotope ratios in gaseous hydrocarbons found in the Hauptanhydrit of the Leine Series indicate a partial origin from the Rotliegendes beds underlying the Zechstein sequence.

#### **4.1.2 Overburden and adjoining rock zone**

For the safety case it is necessary to assess the whole geologic system. The aspects of the overburden and adjoining rock and the climatic effects that influence some geologic processes were accordingly added.

The overburden and the adjoining rock in the Gorleben investigation area consist of Triassic to Quaternary sequences whose geologic structure has been strongly influenced by salt diapirism and glacial processes, see Figure 4.5. The main structures are the northwestern and southeastern rim synclines on the flanks of the Gorleben-Rambow salt structure, which were formed because of the diapirism, and the glacialenic Gorleben channel that was created during the glaciation of the Elsterian ice age.

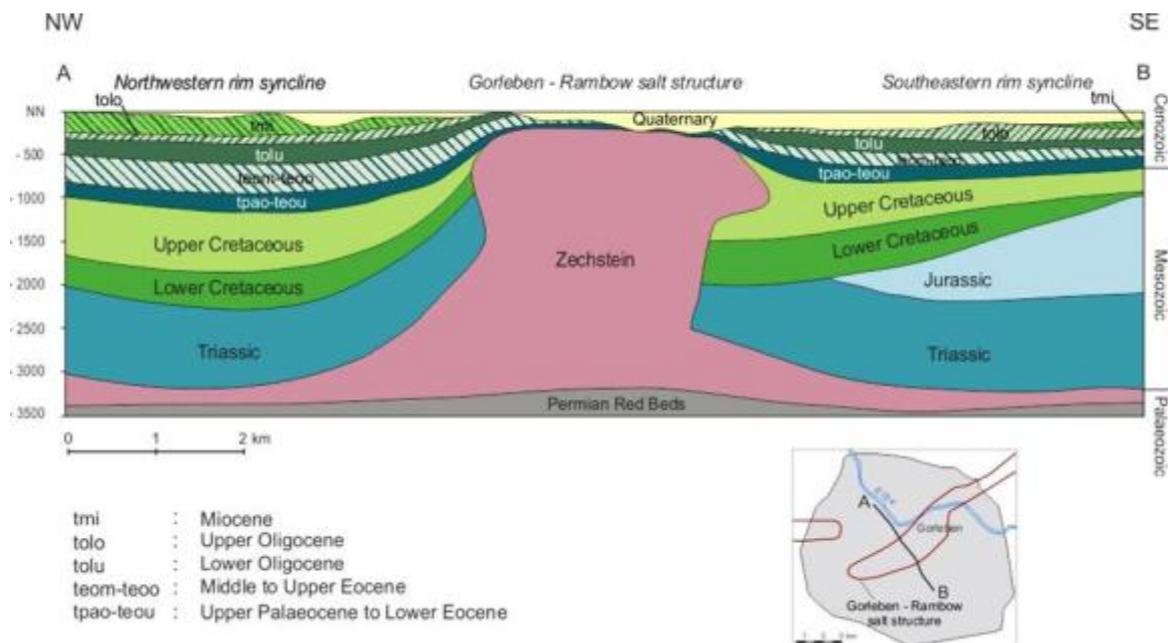
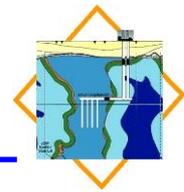


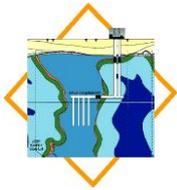
Figure 4.5: Simplified geological cross section of the study area [Köthe et al. 2007]

Due to the diapirism, the Upper Cretaceous and Tertiary sediments in the rim synclines show a greater thickness than is normal. However the sediments above the salt dome are much thinner for the same reason. In addition, the rim syncline strata have been dragged up in some parts on the flanks of the salt dome. The absence of some stratigraphic units on the top of the salt dome is attributable to either the later erosion of some of the horizons lifted up by the salt structure, or because they were never deposited on the elevation which developed above the salt structure. A large hiatus is the reason why the Middle Miocene sediments are overlain by the Quaternary with a thickness of approximately 50 – 100 metres. Above the salt dome and within the Gorleben channel, the sediments reach a much greater thicknesses of 250 – 300 metres.

#### 4.1.3 Tectonics

In common with the rest of northern central Europe, the absence of plate tectonic activity means that the Gorleben site is only affected by minor regional stresses. No graben formation or orogenic activity has taken place in North Germany in the last 10 million years. The Alpidian orogeny affected North Germany only indirectly. There is nothing to indicate that this stable tectonic situation will change during the next million years.

Earthquakes are rare events in North Germany. The tectonic situation can be estimated on the basis of the historic earthquake catalogue which compiles all of the earthquakes documented since 800 AD. The historic earthquake catalogue reveals that the Gorleben site has not been affected by earthquakes over the last 1,000 years [Leydecker et al. 2008].



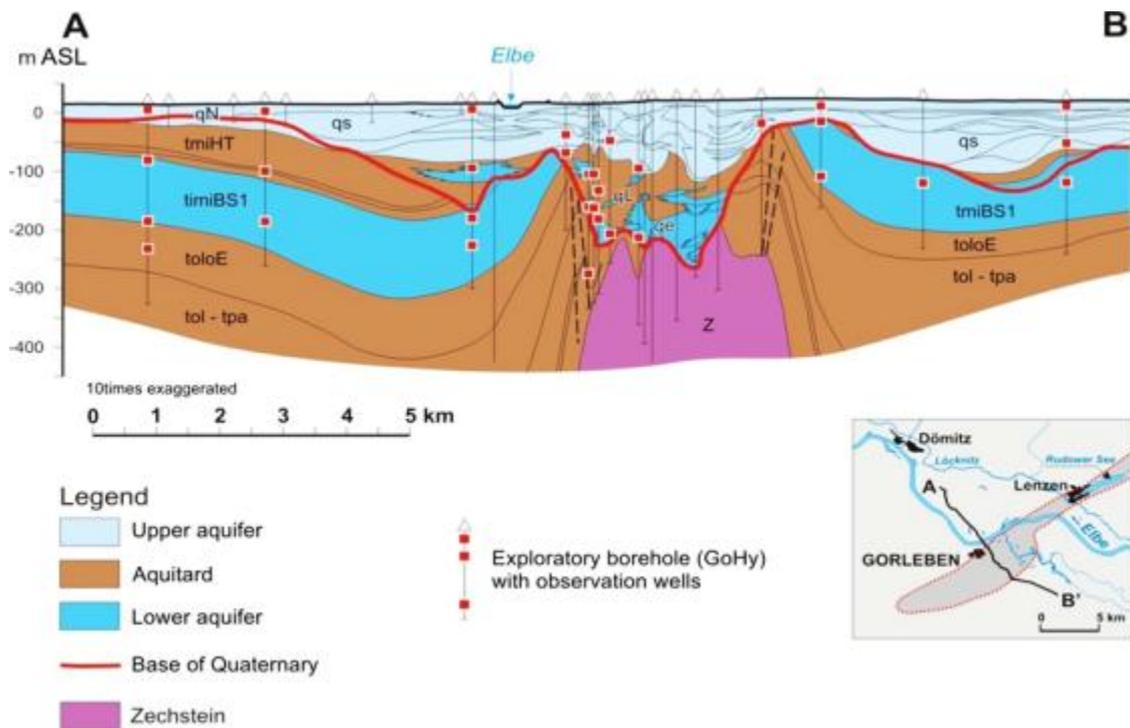
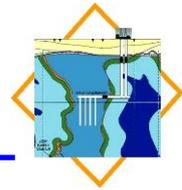
Only six tectonic earthquakes with magnitudes greater than 2.5 on the MSK scale have been recorded by instruments in the whole of North Germany since 1995. The upgrading of the seismograph network which took place in 2007 means that all earthquakes with a magnitude of between 2.0 and 2.5 can now also be recorded. Three earthquakes have been registered since 2007. The strongest earthquake occurring since 1995 had a magnitude of 3.4. The earthquake locations are spread throughout North Germany and are not concentrated in a specific zone. The focus depths of the earthquakes all lay beneath the Zechstein basement.

An analysis of the faults in the pre-Zechstein basement revealed that the faults formed in the Permian approximately 300 – 250 million years ago, and are largely independent of the previously formed Variscan fault pattern. The faults in the pre-Zechstein basement have no corresponding faults above the Zechstein evaporite sequence [Brückner-Röhling et al. 2002].

In conclusion, there is no evidence for any seismic activity giving rise to major movement along active fault zones near the Gorleben salt dome.

#### **4.1.4 Groundwater flow and hydrochemical regime in the overburden and adjoining rocks**

The unconsolidated sediments of the overburden and adjoining rocks at the Gorleben site form an aquifer system up to 430 m thick where the aquifers and aquitards are subdivided into an upper and a lower groundwater stockwork, Figure 4.6. The base of the system in the rim synclines is formed by clays of Lower Oligocene age and the underlying clays of Palaeocene to Eocene age. In the Gorleben channel, the base of the aquifer is formed, in part, by either the salt formation or the cap rock.



**Figure 4.6:** Aquifer subdivision in the overburden of the Gorleben salt dome [Klinge et al. 2007], cross-section 10 times vertically exaggerated

The hydraulic head of the water table in the Gorleben area is only 6 m, reflecting the flat topography, with a groundwater recharge of 160 mm/year. The direction of the groundwater flow close to the surface is from the high in the south of the salt dome and then radially through the surrounding synclines. In the beds in the Gorleben channel above the salt dome, the water flows from southwest to northeast, and outside of the salt dome, towards the Elbe River. The water management which is practiced in the local agricultural areas influences the direction of groundwater flow. The groundwater flow is also influenced by differences in groundwater density according to differences in the salt content. The groundwater salinity increases mainly with depth and ranges from fresh water with less than 1 g/l TDS (total dissolved solids) particularly in shallow zones, limited deposits with high salt concentrations to saturated brine with approximately 320 g/l TDS [Klinge et al. 2007]. Figure 4.7 shows the main constituents of the water in the different vertical zones in the Gorleben study area.

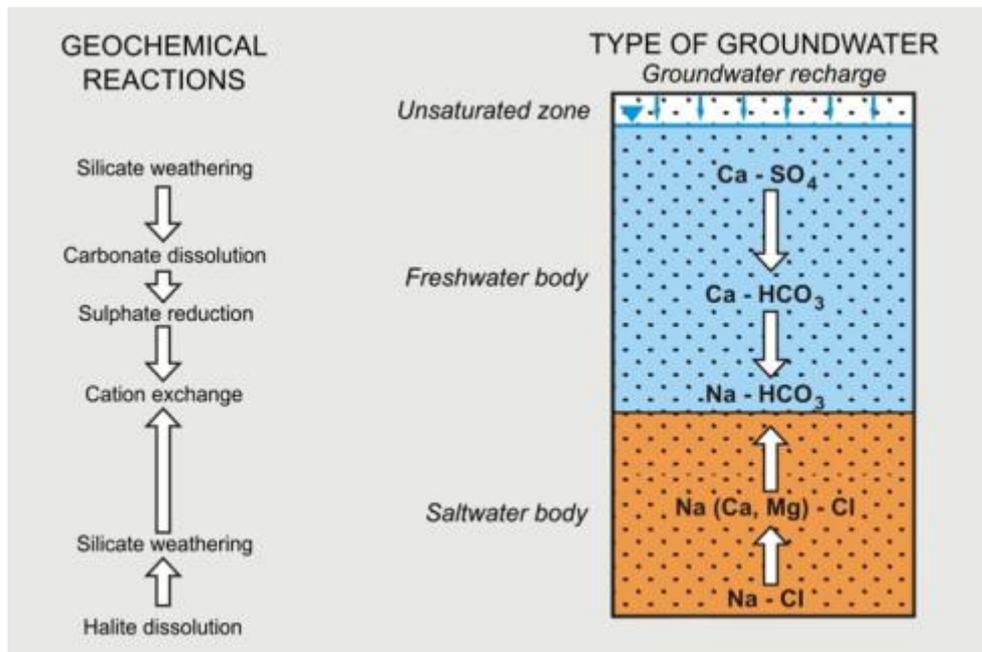
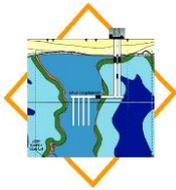


Figure 4.7: Schematic diagram of the vertical groundwater type zoning within the Gorleben study area (from [Klinge et al. 2007])

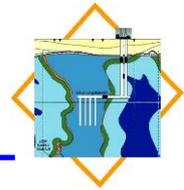
#### 4.1.5 Climate

The supra-regional climatic effects have a controlling effect on many processes in the geologic system or are considered to be an important cause of other significant effects. In the past glacial and interglacial periods were followed by a 100,000 year cycle. The Gorleben area was covered with ice during the Elsterian and Saalian glacial. In Holsteinian interglacial the sea probably flooded the morphologically deeper lying zones of the Gorleben channel formed during the Elsterian glacial [Mrugalla 2011].

#### 4.2 Site-specific safety concept

According to the general approach explained in chapter 3 the following design requirements were established from the guiding principles and the safety requirements in order to derive specific objectives and strategic measures [Mönig et al. 2012]:

- Containment,
- permanence of CRZ, and
- subcriticality.



In total, 14 objectives and 17 strategic measures were set up in R&D project VSG, and these are summarised in the following subsections. More details and explanations can be found in [Mönig et al. 2012]. The strategic measures as a whole provide the basis for the site-specific design and layout of the repository (chapter 4.3).

#### 4.2.1 Design Requirement: Containment

For the requirement to enclose the emplaced waste canisters quickly and as tightly as possible by the salt, eight specific objectives have been derived:

**O1:** No solution, or at best very minute amounts of solution, shall come into contact with the waste canisters emplaced in all probable evolutions of the repository system.

**O2:** Only limited amounts of solution shall come into contact with the waste canisters in less probable evolutions of the repository system.

**O3:** In the event of radionuclides being mobilized from the waste, transport of these pollutants shall be retarded by chemical and physical processes.

**O4:** The properties of both, the salt rock and the engineered barriers that are responsible for the containment of the radionuclides, shall be readily predictable.

**O5:** The repository shall be designed such that no intervention is necessary from the outside during the post-closure period.

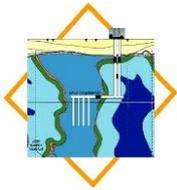
**O6:** The engineered barriers shall be designed robustly, in order to ensure their functional efficiency, taking into account different load cases and possible degradation processes.

**O7:** The immediate and long-term containment of the radioactive waste in the CRZ shall be ensured by a staggered barrier system. The individual elements shall act redundantly or on diverse processes and they shall complement each other in their temporal effectiveness.

**O8:** The disposal concept shall provide a segmentation of disposal areas to allow a fast emplacement of the waste and the prompt backfilling and sealing of the corresponding drifts and boreholes. Disposal areas with different waste types shall be separated in a way that no physical and/or chemical interactions with a negative effect on long-term safety occur.

The strategic measures to meet these objectives are as follows:

**M1:** The excavation volume of the repository will be as small as possible. The excavation will be performed using gentle methods in order to limit the impact on the geological barrier.  
(→ O1, O2, O3)



**M2:** The mine openings of the emplacement areas will be excavated in salt regions with homogeneous structure and properties, e.g. in the Staßfurt rock salt Series (Hauptsalz). (→ O4)

**M3:** The mine openings of the emplacement areas will be excavated in salt regions that are free of brine pockets of significant volume and that provide favourable creep properties. These emplacement areas will be excavated in the Staßfurt rock salt Series (Hauptsalz). (→ O1, O2)

**M4:** The mine openings of the emplacement areas will be excavated with sufficient safety pillars to the shafts, to rock strata with potentially larger brine pockets, and to potential transport paths for solutions. Based on the existing experience in salt mining, a safety pillar of 50 m was determined. It is necessary to demonstrate that this safety pillar is sufficient to maintain the integrity of the geological barrier in the CRZ. (→ O1, O2)

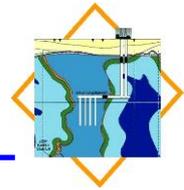
**M5:** Engineered barriers with defined hydraulic properties will be erected in the shafts and in the access drifts between infrastructure area and the emplacement areas. Their design is based on load cases which should cover the potential range of future impacts during the required duration of effectiveness. The engineered barriers must be adequately tight until the hydraulic resistance of the compacted crushed salt effectively hinders the brine intrusion to the waste. (→ O1, O2, O3, O5, O6, O7)

**M6:** The mine openings of the emplacement areas will be backfilled with crushed salt. The convergence process will result in a compaction of the crushed salt with a reduction in its porosity and permeability. Owing to the higher temperatures, this process is faster in the vicinity of the heat-generating high-level radioactive waste than elsewhere in the mine. The crushed salt limits the extent of the convergence process. In addition, this measure reduces the void volume in the emplacement areas that can be filled with solution. (→ O1, O2, O5, O6, O7)

**M7:** Small amounts of moisture will be added to the crushed salt that is used to backfill the access drifts, at least in the vicinity of the emplacement areas, in order to increase the plasticity of the crushed salt and thus to accelerate its compaction. (→ O1, O2, O5, O6, O7)

**M8:** The amount of humidity in the vicinity of the emplaced waste will be minimised in order to constrain the corrosion of waste canisters and thus to limit gas production. Crushed salt having only the small natural aqueous content of the Staßfurt rock salt Series, Hauptsalz, will be used as backfill material in the emplacement drifts. To this end operational provisions will be taken for handling the crushed salt upon excavation. If necessary, the crushed salt will be dried before backfilling. (→ O1, O2, O5, O6, O7)

**M9:** The shaft seals will be designed such that their seal efficiency relies on several different seal elements that are independent from each other and that have diverse functionality owing to their configuration. (→ O6, O7)



**M10:** Disposal areas will be segmented in order to minimise the simultaneously open volume to be backfilled with crushed salt and to guarantee a prompt backfilling of the disposal areas already filled with waste. (→ O8)

**M11:** The disposal areas with waste that differ in expected gas production will be separated in two different areas in the repository in order to prevent physical and/or chemical interactions between both waste types. The separation will be guaranteed by seals. (→ O8)

#### 4.2.2 Design Requirement: Permanence of CRZ

The second design requirement states that the CRZ must remain intact and that its barrier function is not impaired by internal or external processes and effects. This requirement results in five additional specific objectives and five additional strategic measures.

**O9:** The quality of the containment shall not be impaired by surficial processes during the assessment period.

**O10:** The quality of the containment shall not be impaired by thermal processes during the assessment period.

**O11:** Salt minerals containing crystal water, e.g. carnallite, shall not be thermally degraded.

**O12:** The quality of the containment shall not be impaired by gas production and gas production rate during the assessment period.

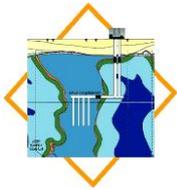
**O13:** The possibility of unintended human intrusion in the CRZ shall be reduced by administrative measures so long as these measures do not have negative effects on long-term safety.

The following measures are established to meet these objectives:

**M12:** The repository's drifts and boreholes will keep a sufficient distance from crystal water containing salt minerals such as carnallite in order to avoid their thermal degradation and a release of the crystal water. (→ O11)

**M13:** The disposal level at depths greater 800 m below surface will guarantee a sufficient thickness of the salt formation above the disposal level. Additionally a sufficient distance to the salt dome's flank will be observed. The depth of disposal level will also reduce the possibility of human intrusion. (→ O9, O13)

**M14:** The repository will be built in a salt dome whose uplift is almost completed, i.e. the salt dome has a negligible uplift rate. In combination with a calm tectonic regime with low subsid-



ence rates this will guarantee that the uplift of the disposal level to the surface is not safety relevant. (→ O9)

**M15:** The maximum temperature in the salt formation will be limited to 200°C<sup>6</sup> by applying an appropriate thermal loading and disposal geometry of the containers (→ O10, O11).

**M16:** The gas production and gas production rate is limited by minimizing the moisture in the backfill and, if necessary, by using appropriate container materials. (→ O12)

#### 4.2.3 Design Requirement: Subcriticality

The third design requirement is to maintain subcriticality. This relatively explicit requirement leads to one further objective and one further strategic measure.

**O14:** Subcriticality has to be maintained for the operational and post-closure phase by the loading of the containers and the design of the repository.

**M17:** The loading and the design of the waste container must guarantee subcriticality. Corresponding analyses have to be carried out on the basis of the radioactive inventories of the waste containers including the consideration of uncertainties. (→O14)

### 4.3 Technical description of the repository design

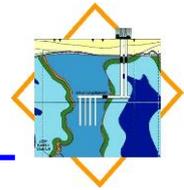
#### 4.3.1 Fundamentals and boundary conditions

The repository design is governed by the safety concept. It has to meet the requirements of the national mining law ABV [ABergV 2009], the nuclear energy act [AtG 2010] and the Safety Requirements [BMU 2010a]. The Safety Requirements include an obligation to provide for the retrieval of the waste packages during repository operation and to remove waste packages in case of intervention up to a time period of 500 years after repository closure.

There are three important fundamentals for the repository layout and design, these being the geological situation and boundary conditions at the repository site and the type and amount of waste for disposal:

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<sup>6</sup> This temperature criterion is only used for the repository layout. It aims at two things: firstly, it shall ensure that the maximum temperature within the waste containers is low enough to avoid negative alterations of the waste (reduced stability of fuel rods or degradation of the glass matrix in vitrified waste). Secondly, taking the foreseen safety distance of at least 50 m from the emplacement fields to the bounding salt layers (including Carnallite) around Staßfurt Hauptsalz into account, it can be assumed that by limiting the temperature in the emplacement fields Carnallite is not thermally degraded and no crystallization water can be released.



- The description of the geological situation of the salt dome Gorleben is summarised and described in chapter 4.1.
- The amounts and types of expected heat-generating waste were assessed on the basis of the political decision in 2011 to phase out nuclear power by 2022 and compiled in [Dörr 2013]. In total the expected waste inventory amounts to 10,550 tonnes of heavy metal resulting from 35,564 spent fuel elements and 7,973 canisters, types CSD-V, CSD-B and CSD-C, of waste from the reprocessing of 6,700 tonnes of heavy metal. In addition spent fuel from prototype and research reactors was taken into account.

An important design parameter regarding the footprint of the repository mine is the maximum permissible temperature for the host rock. For rock salt the temperature should not exceed 200 °C at any time and in any place in the repository [Mönig et al. 2012].

The operational lifetime of the repository was assumed to be 40 years, and the year of commissioning to be 2035 on the basis of the discharged core of the last operational reactor.

#### 4.3.2 Design approach

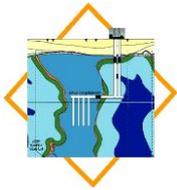
According to the general approach of the safety concept there are two main repository design requirements. The first is to enclose the emplaced waste canisters quickly and as tightly as possible within the containment providing rock zone (CRZ) and the second states that the CRZ must remain intact and that its barrier function is not impaired by internal or external processes and events. A third design requirement “Exclusion of criticality” (see chapter 4.2) is of no relevance for the repository layout according to the safety concept (chapter 3). In line with the appropriate strategic measures described in chapter 4.2, a site-specific design and layout of the repository was developed in the course of the R&D project VSG [Bollingerfehr et al. 2012].

The objective of the repository design is to detail those parts of the safety and safety demonstration concept [Mönig et al. 2012] that are relevant to the demonstration of operational and long-term safety.

This requires the development of a repository concept which has to be optimised in further design steps. For this purpose, two main repository concepts were considered:

- Variant 1: Emplacement of all heat-generating radioactive waste in POLLUX<sup>®</sup> and CASTOR<sup>®</sup> casks in horizontal drifts.

In addition – for comparison only – the emplacement of all heat-generating radioactive waste in transport and storage casks (CASTOR<sup>®</sup>) in horizontal boreholes was considered.



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

As an option, the emplacement of a certain amount of radioactive waste with negligible heat generation in a separate area of the salt dome in horizontal emplacement chambers was considered.

For both variants (emplacement in drifts / emplacement in vertical boreholes), the respective technical design of the repository is described in the following chapters. Components and systems such as surface facilities, shaft hoisting and ventilation systems which are essential to the operation of a repository, but which do not vary much in the variants mentioned above and do not have a direct impact on the long-term safety of the repository will not be described in detail. These components and systems were elaborated previously in the context of mostly generic repository designs [Bollingerfehr et al. 2008] in compliance with conventional and nuclear regulations.

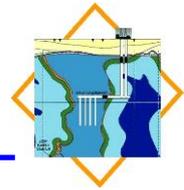
### 4.3.3 Design of the repository

The repository design was adapted to the specific geologic environment at the Gorleben site according to the fundamentals and boundary conditions and to the type and amount of waste previously mentioned.

#### 4.3.3.1 Access to the underground and excavation of the underground facilities

Access to the repository will be provided by two shafts. The shafts are erected in the Leine rock salt Series, which is much stiffer than the Staßfurt rock salt Series and thus provides stability for the shaft construction during the operational phase. The shafts will be surrounded by safety pillars to protect the shafts as far as possible against thermo mechanical loading. Protective distances must be left between the shafts and the emplacement fields for heat-generating waste, between the emplacement fields and the carnallite (Potash seam Staßfurt) and the anhydrite (Hauptanhydrit) on one hand and to the salt dome flanks on the other. According to [Filbert & Engelmann 1998] a distance of approximately 300 m to the first emplacement field was determined to be safe. [IFG 2010] provisionally specified a 50 m safety distance for galleries, chambers and boreholes to the anhydrite/carnallite. This safety distance was verified by thermomechanical (TM) calculations.

The heat-generating waste will be disposed of in the Staßfurt rock salt which is characterised by particularly high convergence rates [Blase et al. 1989] thus resulting in a potentially rapid enclosure of the waste canisters.



To avoid any impact from the surface, the repository mine openings are considered to be excavated at a level of at least 870 m below the surface. The exploration mine already developed to a certain extent since the late 1980's was excavated some 30 m above the emplacement level (840 m below surface) so as not to impair the later repository excavations. In a previous conceptual repository design [Filbert & Engelmann 1998] the main mine openings necessary to operate the exploration and the repository mine are described (e.g. infrastructure rooms at both levels including functional areas such as electrical and mechanical workshops and stores, connection of both levels via a spiral, emplacement drifts etc.).

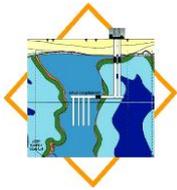
#### 4.3.3.2 Disposal variants

The first step for developing an emplacement concept is the selection and description of waste-specific (SF/reprocessing waste) packages for the emplacement variants on the basis of the waste inventory [Dörr 2013]. 3D thermal calculations were carried out (see chapter 4.3.3.3) to demonstrate compliance with the 200°C criterion (see M15 in chapter 4.2.2) and to determine the distances of waste packages, drifts, and boreholes. The results provided suitable design parameters for the layout of the emplacement drifts and fields and the entire repository including infrastructure areas and drifts for the transportation of waste packages and the excavated rock salt material. The respective repository design approach was performed in two steps; first a conceptual design was developed, followed by a technical design which includes suggestions for optimisation. The two variants of the considered reference design are characterised as follows:

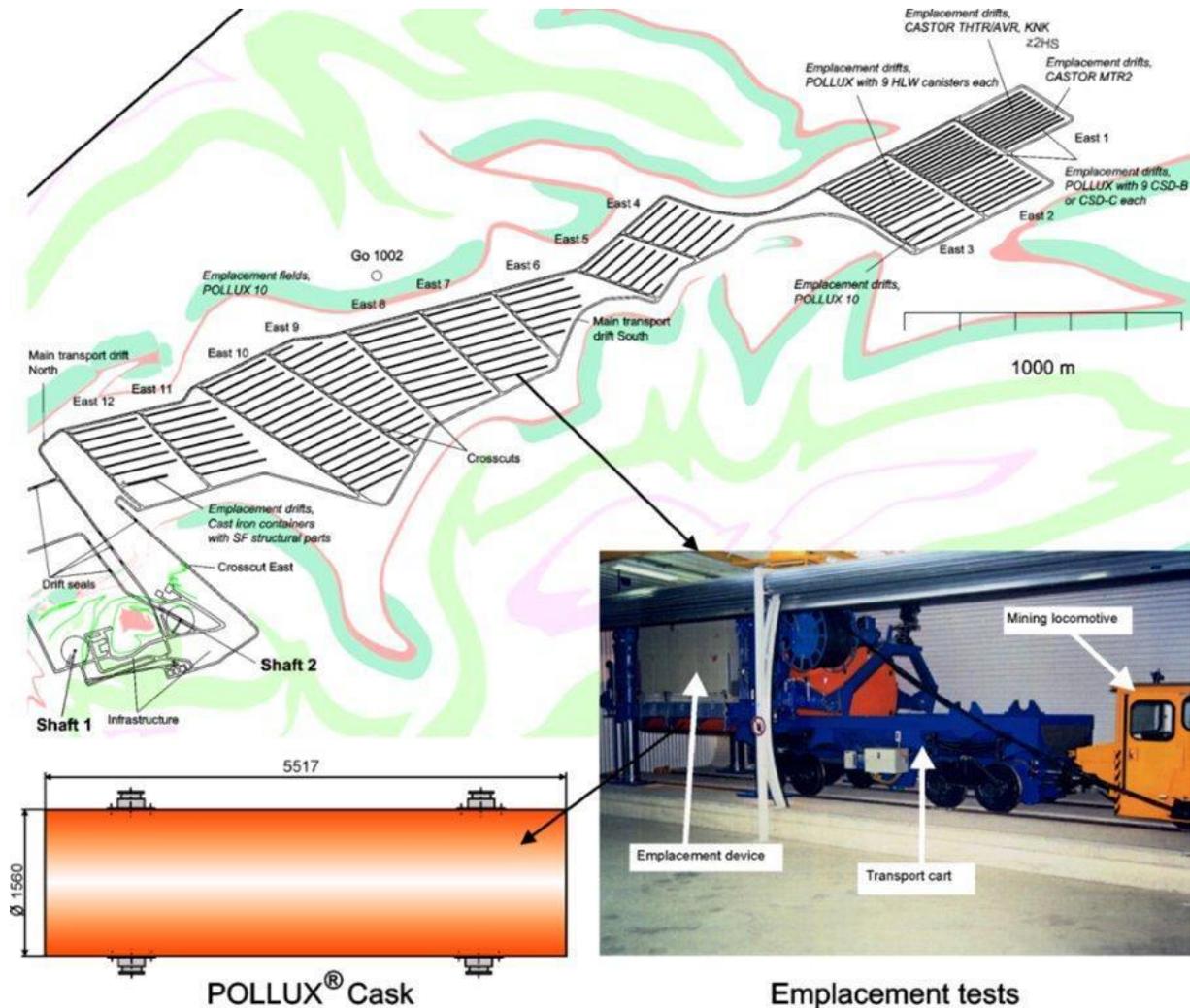
##### 4.3.3.2.1 Variant 1: Disposal of POLLUX<sup>®</sup> and CASTOR<sup>®</sup> casks in horizontal drifts

The total inventory of heat-generating waste will be disposed of in heavy (weight max. 65 metric tons), self-shielded POLLUX<sup>®</sup> casks [Bollingerfehr et al. 2012] containing the fuel rods of disassembled spent fuel elements or waste from reprocessing in horizontal drifts of the salt mine. A small quantity of structural parts from the conditioning of spent fuel elements will be disposed of in cast iron containers. The expected numbers of waste packages are: 2,105 POLLUX<sup>®</sup> casks with spent fuel from NPPs, 887 POLLUX<sup>®</sup> casks with reprocessing waste, and 527 containers (specific CASTOR<sup>®</sup> casks) with spent fuel from prototype and research reactors. After emplacement of each waste package, crushed salt will be used to backfill the remaining drift openings.

The thermal calculations (see chapter 4.3.3.3) for cask and drift distances, demonstrate that the temperature limit of 200°C at the contact between cask and salt will not be exceeded (see strategic measure M15) and the results have been transferred into a repository mine layout which is shown in Figure 4.8. In total, the north-eastern part of the mine – adjusted to the assumed geologic structure of the Gorleben salt dome at the emplacement level – will have a length of approximately 4 km and a width varying between 300 m and 700 m. As mentioned above, a certain amount of non-heat-generating waste was also considered.



As a result, three separate emplacement fields were designed in the south-eastern part of the repository mine (left of shaft 1, but not displayed in Figure 4.8) [Bollingerfehr et al. 2012].



**Figure 4.8:** Repository design for variant 1: Emplacement of all heat-generating spent fuel and reprocessing waste in POLLUX<sup>®</sup>-casks in horizontal drifts (adjusted to the assumed geologic structure)

Figure 4.8 also shows a sketch of the POLLUX<sup>®</sup> cask and a photograph of the test set-up for full-scale container emplacement demonstration tests in a drift. These tests were successfully performed in the 1990s [DBE 1995].

For the purpose of comparison, the emplacement of all heat-generating waste in self shielding transport and storage casks (CASTOR<sup>®</sup>) was considered. The conceptual approach is to locate the very heavy casks (up to 160 t, loaded) into 10-m-long horizontal boreholes which are accessible from a central transport drift.

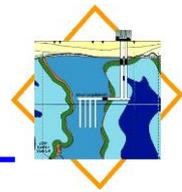


Figure 4.9 shows the emplacement situation underground (plan view) and the main components.

The footprint of this repository mine will be similar to that of variant 1 (POLLUX<sup>®</sup> concept) due to the thermal criterion (200°C). Since the concept of emplacing transport and storage casks has not been investigated in detail, all technical aspects such as shaft transport, emplacement technique, and the suitability of the casks as a waste package will have to be analysed in future R&D projects.

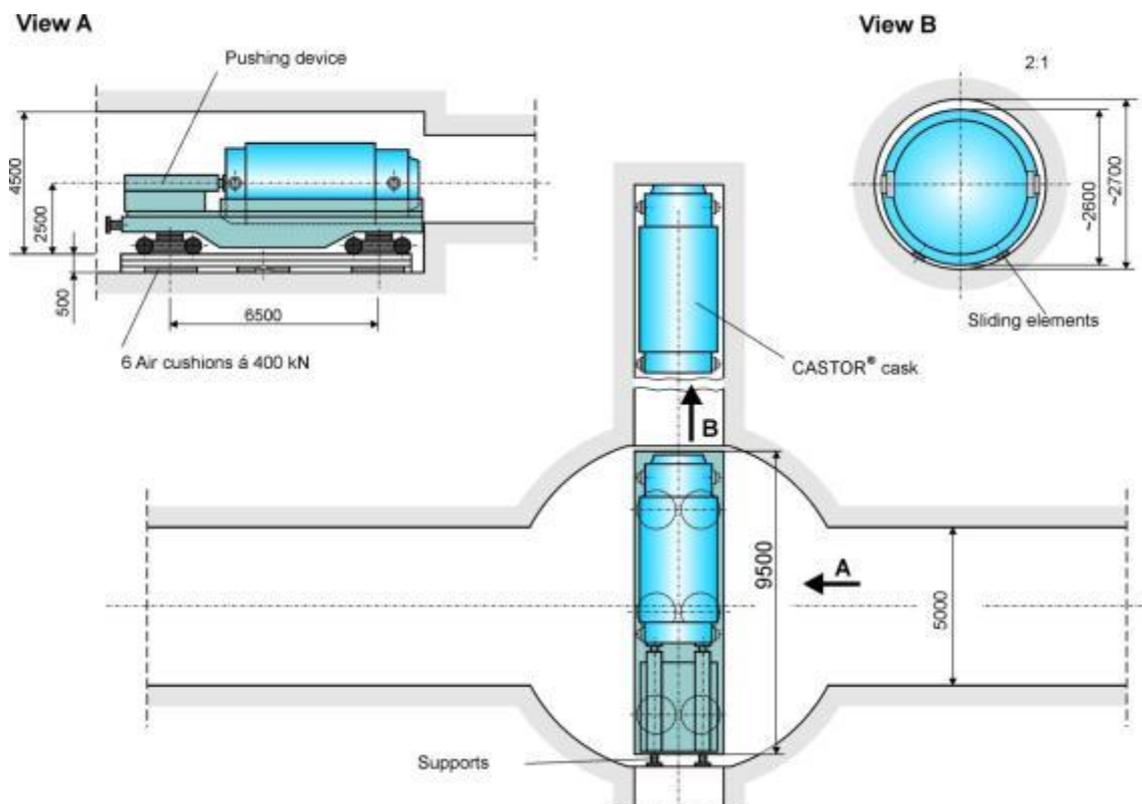
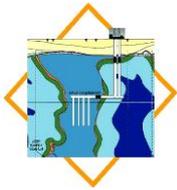


Figure 4.9: For comparison with variant 1: Disposal of transport and storage casks (CASTOR<sup>®</sup>) in horizontal boreholes

#### 4.3.3.2.2 Variant 2: Disposal of canisters for fuel rods and reprocessing waste in vertical boreholes.

A completely different approach consists of emplacing all heat-generating waste in deep vertical boreholes fulfilling the retrievability requirements according to the Safety Requirements. In this concept the borehole will be equipped with a casing prior to canister emplacement. The space between the canisters and the annular gap surrounding them will be filled with backfill material such as sand. When full, the top of the borehole will be closed tightly by a lid. The casing and lid provide additional technical barriers. At the same time, they facilitate ac-



cess to the canisters in case they need to be retrieved (see chapter 4.3.5). After filling of all the boreholes in an emplacement gallery, crushed salt will be used to backfill the remaining drift openings.

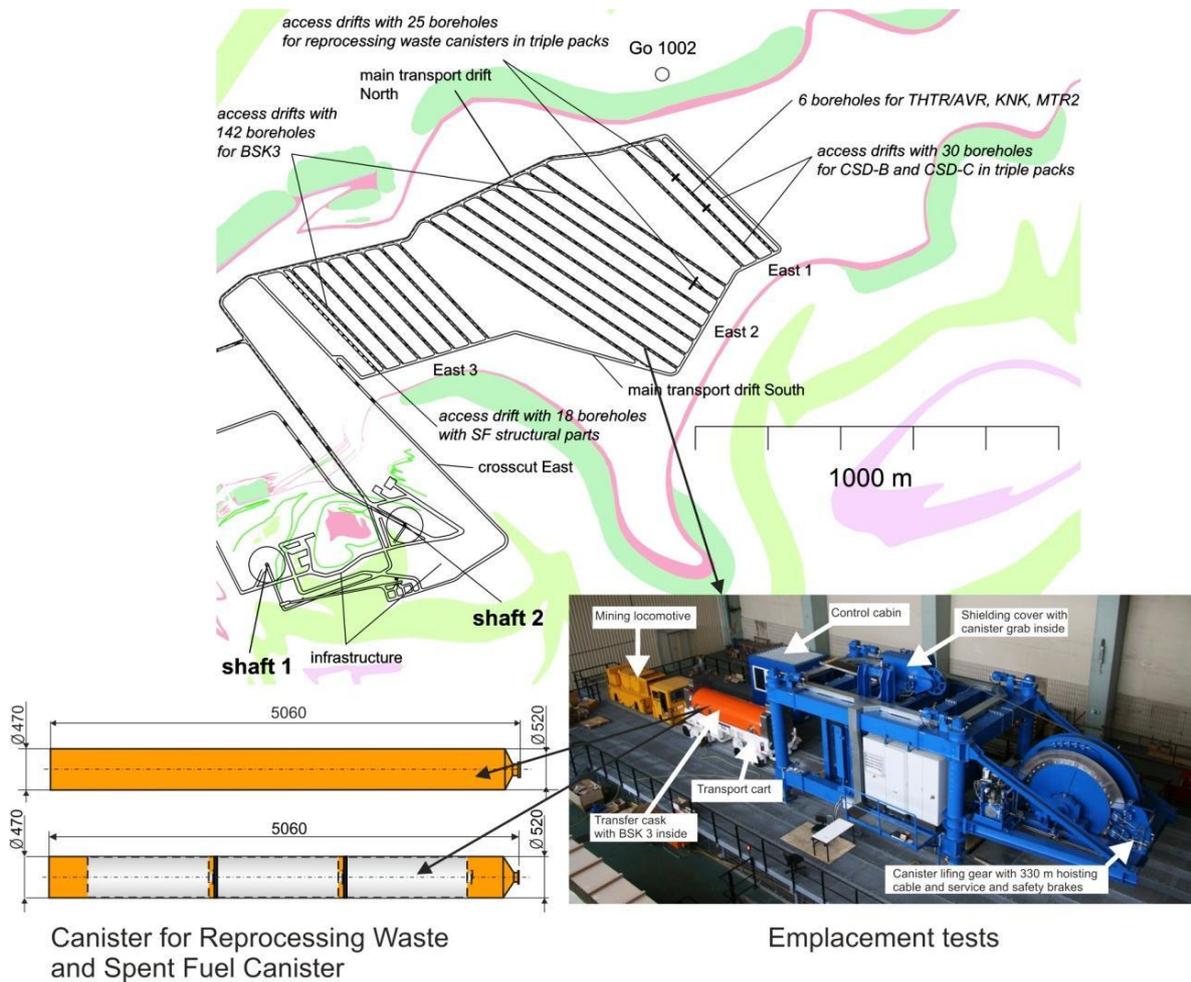
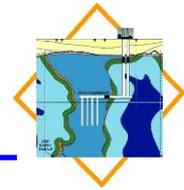
The expected amounts of waste containers have been estimated for the borehole concept as 7,015 containers with spent fuel from NPPs, 2,659 containers with reprocessing waste, and 295 containers with spent fuel from prototype and research reactors. The canisters contain fuel rods, reprocessing waste, or structural parts from the conditioning of spent fuel assemblies.

The thermal calculations for the borehole emplacement concept took into account heat transfer into the surrounding rock salt over the length of the boreholes (> 300 m). The results provided borehole and drift distances, and again demonstrated that the temperature limit of 200 °C at the contact between the liner and the salt will not be exceeded. The corresponding repository mine layout is shown in Figure 4.10. In total, the north-eastern part of the mine – adjusted to the assumed geologic structure of the Gorleben salt dome at the emplacement level – will have a length of approximately 1 km and a width varying between 400 m and 800 m. Again, the emplacement of non-heat-generating waste in the south-eastern part of the mine was considered as an option.

Figure 4.10 also shows a sketch of a canister for reprocessing waste and a spent fuel canister and a photograph of the test set-up for the full-scale canister emplacement demonstration tests in vertical boreholes that were successfully performed in a surface facility in 2008 / 2009 [Bollingerfehr et al. 2008].

#### 4.3.3.3 Thermo-Mechanical Design Calculations

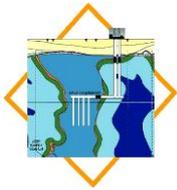
From a thermomechanical (TM) point of view, the design of a repository for heat-generating waste strongly depends on a demonstration that thermal limits are met and that the mine openings withstand the mechanical loadings (see strategic measure M15, chapter 4.2.2). The temperature behaviour in specific locations of the repository such as the potash layer, dam locations, and on top of the salt dome were investigated by means of numerical calculations.



**Figure 4.10: Repository design for variant 2: Emplacement of all heat-generating waste in lined vertical boreholes (adjusted to the assumed geologic structure)**

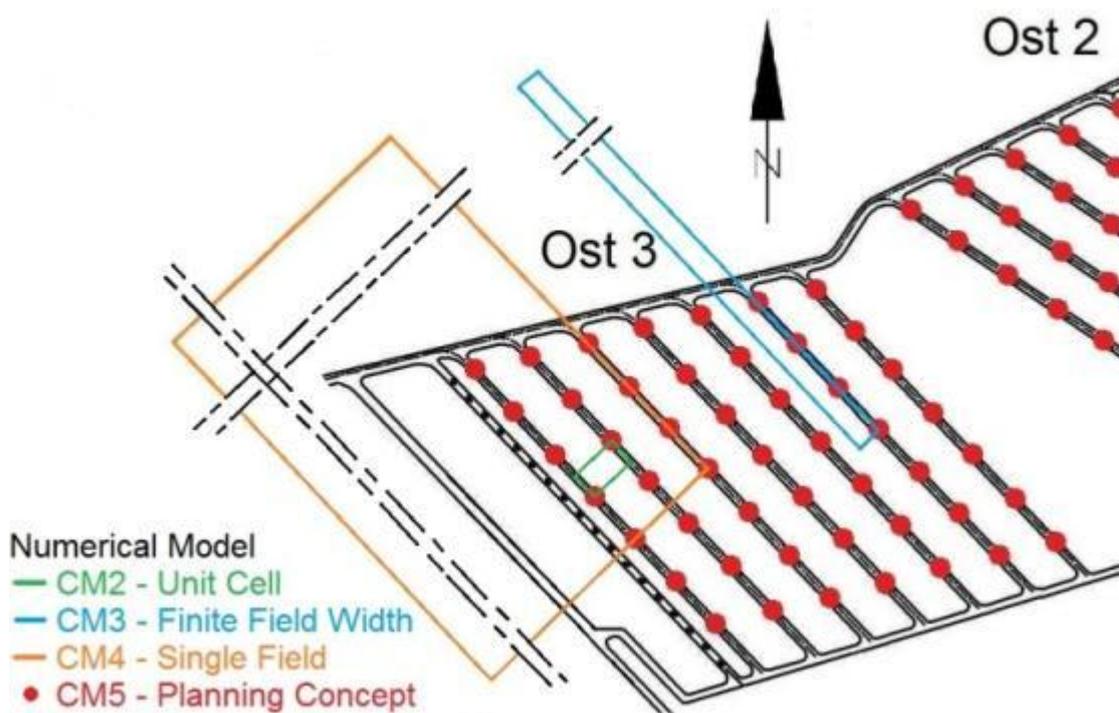
Pure thermal (T) or combined TM calculations were performed depending on the disposal concept. TM calculations were necessary due to the mechanical influence on thermal state variables, e.g. for crushed salt used as backfill material within the drift emplacement concept. Crushed salt has an initial porosity of approximately 35 % and will be compacted over time. The material behaviour changes depending on this compaction process, specifically the heat conductivity increases from approximately 1 W/m K to approximately 5.5 W/m K. Generally and independent of the mechanical behaviour, the temperature dependence of the material properties has to be considered and is particularly important for the geo-materials. The thermal layout of the repository is based on three-dimensional (3D) TM calculations of transient heat conductivity with compaction-state-dependent parameters [Lerch et al. 2012].

Calculations for both emplacement variants (drift disposal and borehole disposal) were performed as a three stage process:

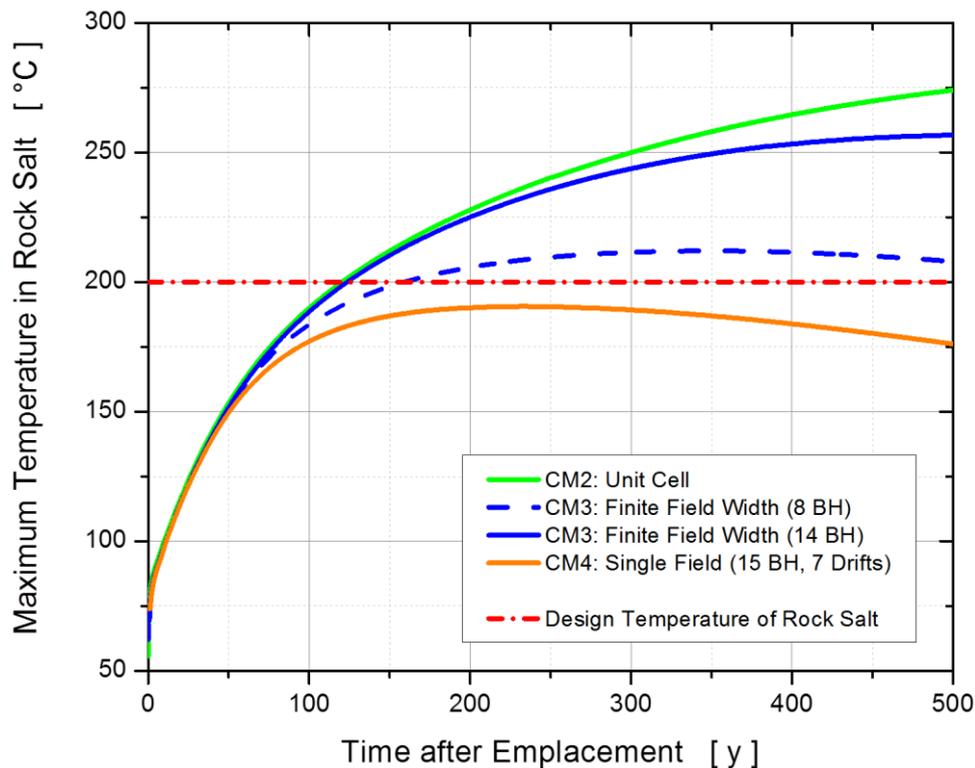
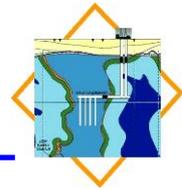


- investigation of design variations,
- layout of the repository based on suitable subsystems, and
- estimation of the thermal behaviour in the entire repository.

The calculation of the first two steps was carried out with the programme FLAC3D, a code of finite-difference method for 3D TM calculations. LinSour was used in the third step. LinSour is an analytic code for 3D T calculations of heat conductivity based on line sources. Figure 4.11 shows a sketch of the different models used in the last two steps. The models were generated to investigate the influence of the heat dissipation not only in one direction (model CM2: unit cell), but also in all three dimensions, (models CM4 = single field and CM5 = planning concept). The boundary conditions significantly influence the results of the different numerical models as shown in Figure 4.12. The intensity of the heat transport also depends on the field size examined in the different models (comparison of CM2 to the two CM3s with 8 and 14 boreholes respectively).



**Figure 4.11:** Borehole disposal – Section of the emplacement concept and scheme of appropriate numerical models [Bollingerfehr et al. 2012]



**Figure 4.12: Borehole disposal – Maximum temperature influenced by the boundary condition within the numerical models [Bollingerfehr et al. 2012]**

Figures 4.13 and 4.14 show the temperature evolutions for drift disposal and borehole emplacement at the emplacement level, which is the level of the backfilled and sealed drifts. Exceptions are the top of the salt dome and, in the case of the borehole concept, points on the mid plane of the active borehole. In both disposal concepts, the temperature limit of 200 °C is met. For the fields where waste with negligible heat generation is emplaced, at drift seal locations and at the shafts, slightly higher temperatures were calculated for the borehole disposal concept than for the drift disposal concept. Also, the thermal impact on the rock layers surrounding the host rock is increased in the borehole concept because of the more compact emplacement of the waste canisters, see the thermal behaviour at a distance of 50 m from the eastern main access drift (Figures 4.13 and 4.14). No differences exist at the top of the salt dome where the maximum temperature is about 35 °C approximately 2,500 to 3,000 years after emplacement.

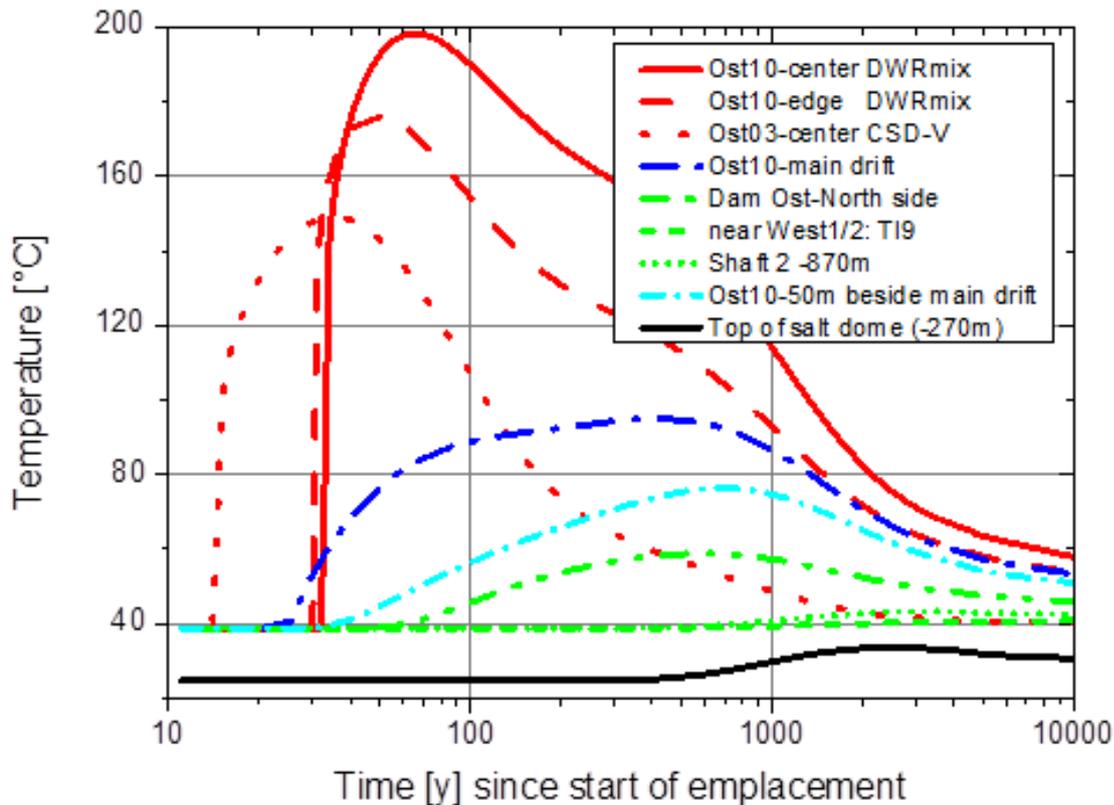
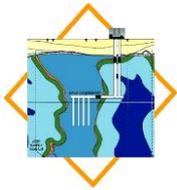
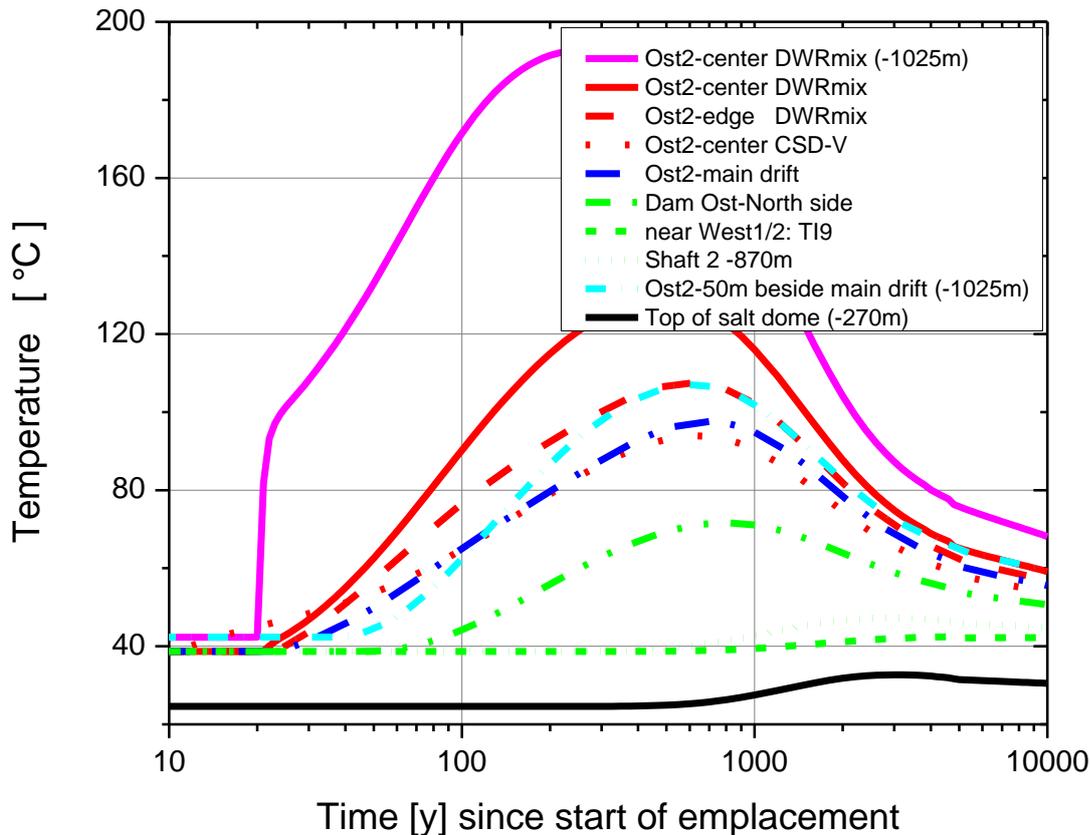
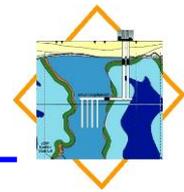


Figure 4.13: Variant 1 (drift disposal) – Temperature over time at different points in the repository [Bollingerfehr et al. 2012]

The thermomechanical calculations showed that both disposal variants (drift disposal and borehole emplacement) for the heat-generating waste meet the thermomechanical design criteria, the only difference being the smaller horizontal foot print in the case of borehole disposal. In comparison with previous calculations (e.g. BAMBUS Project [Bechthold et al. 2004]), a more comprehensive and refined set of parameters (e.g. higher stress levels and faster creep classes of the host rock) was used in the thermomechanical calculations. These led to a substantially faster enclosure of the waste canisters.

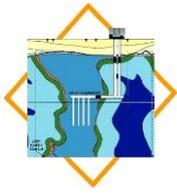


**Figure 4.14:** Variant 2 (borehole disposal) – Temperature over time at different points in the repository [Bollingerfehr et al. 2012]

#### 4.3.3.4 Optimization of repository layout

Optimisation of the repository design is an iterative task until the detailed design documents are submitted to the licensing authority. In this context two aspects were investigated in the R&D project VSG-project; the position of drifts in the mine and the ventilation system for repository operation. Usually, drifts and boreholes that are not necessary for repository operation, for example at the exploration level, could be used for conveying the exhausted air. This would require connecting boreholes between both levels and this might be a contradiction to the safety requirement to minimise excavations and to define the scope of the CRZ. Preliminary calculation results showed that it is possible to provide sufficient amounts of fresh air in all mine areas, and to transfer all exhaust air to the surface, using only the drifts at the emplacement level. In addition to this the structures of drifts and emplacement fields were reconsidered in order to identify schemes which would keep excavation to a minimum.

However, there is room for further optimisation. This future R&D task should include a systematic comparison of emplacement concepts with regard to all technical components such as transport and emplacement technique, technique and process of retrieving waste packag-



es and arrangement of drifts and fields in the geologic environment. Based on the results of thermomechanical calculations, the repository layout and optimisation should not only focus on minimizing the footprint, but should also consider aspects such as the overall drift lengths and the duration of the operational phase.

#### **4.3.4 Design of the backfilling and sealing system**

The objective of the repository closure concept, and in detail, of the sealing system, is to create the conditions for safe containment of the radioactive waste inside the host rock. As a first approach to safe containment, a continuous advective transport path for liquids through man-made openings from the overburden to the radioactive waste canisters and vice versa has to be prevented.

In accordance with the safety and safety demonstration concept (see chapter 3 and [Mönig et al. 2012], backfilling measures and engineered barriers are provided in all excavations of the repository mine. The drifts will mainly be backfilled with crushed salt, and other engineered barriers will be implemented in the drifts at selected locations and in the shafts.

##### **4.3.4.1 Backfill**

Crushed salt is compacting under rock pressure and is thus intended to close the excavations tightly on the long-term. While this backfill material around the waste containers will be as dry as the naturally available material during excavation, typically less than 0.02 % of moisture content, the backfill material in the main transport drift will have a moisture content of 0.6 %. The intention is to accelerate the compaction of the backfill material in order to heal the rock salt barrier as quickly as possible.

The compaction of crushed salt mainly depends on pressure, temperature and moisture content. As an example, Figure 4.4.15 shows the results of calculations with the computer code CodeBright [Müller-Hoeppe et al. 2012a]. While in a cold drift the compaction process needs about 800 years until a porosity of 0.05 is reached, this porosity is reached after a few years in a hot emplacement drift and after about 30 years in a main drift with lower temperature but higher moisture content.

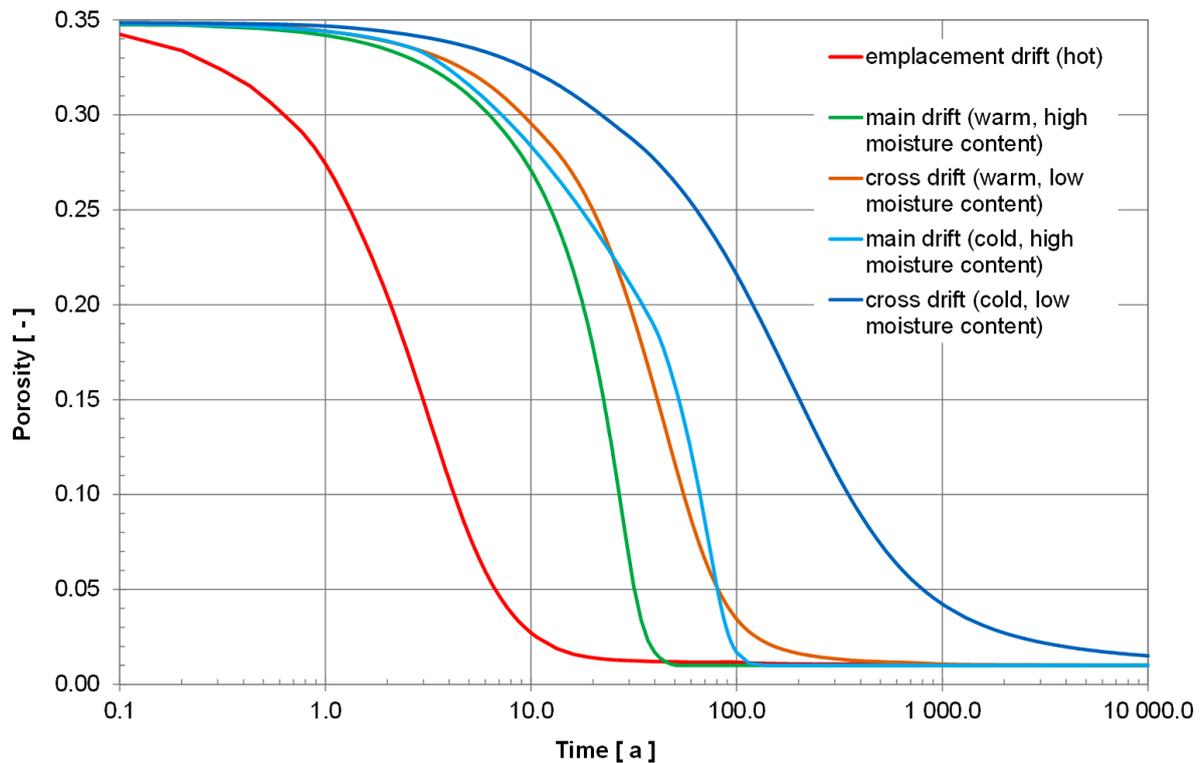
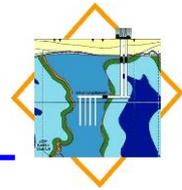
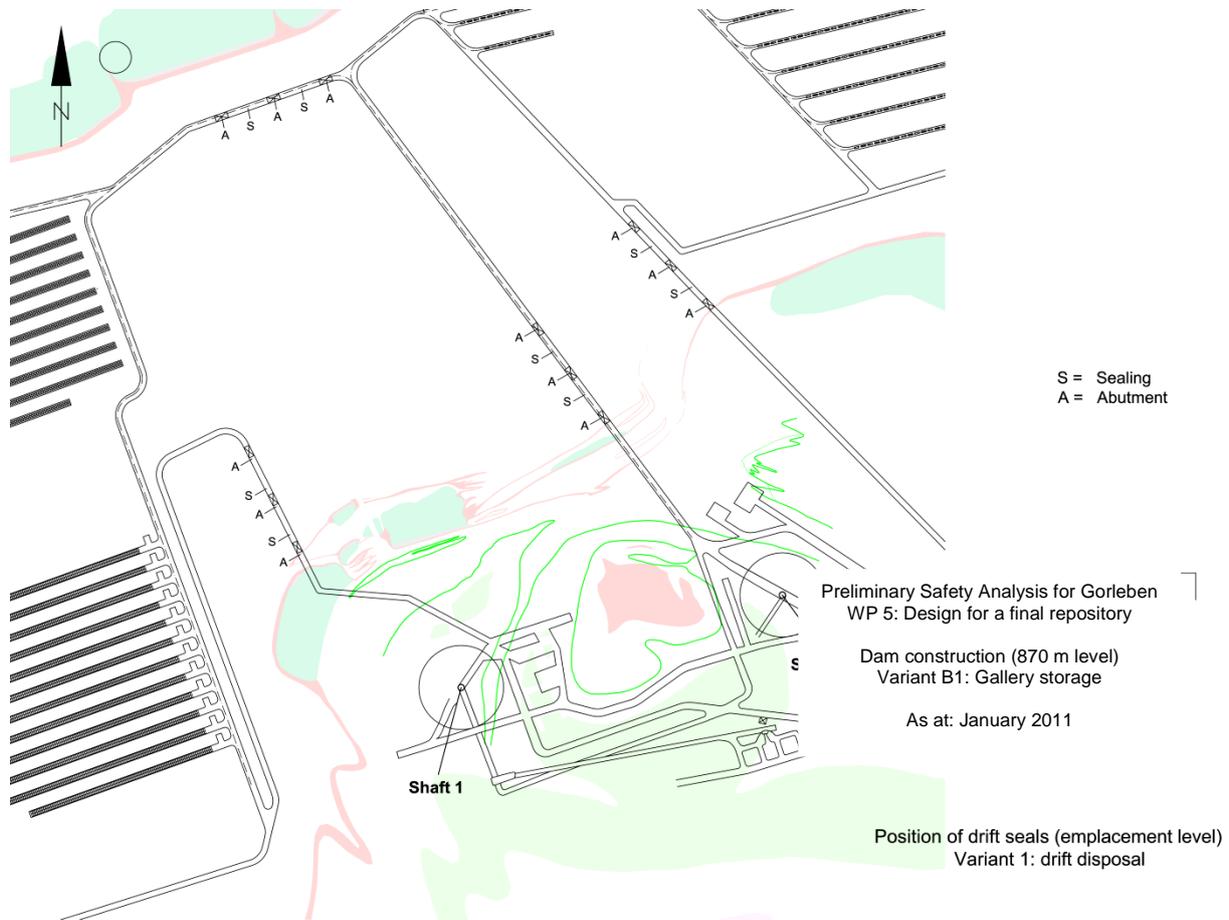
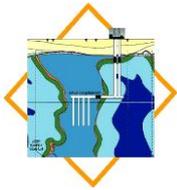


Figure 4.4.15: Comparison of temporal evolution of porosities in drifts backfilled with crushed salt; adopted from [Müller-Hoeppe 2012]

#### 4.3.4.2 Drift Seals

In addition to backfilling, drift seals (engineered barriers) will be located close to the shaft filling station and infrastructure areas at selected positions in all drifts connected to the shaft at the exploration level and at the emplacement level (Figure 4.16). This will ensure that potential fluid pathways to the shaft will be sealed and the heat-generating radioactive waste will be separated from the waste with negligible heat generation.



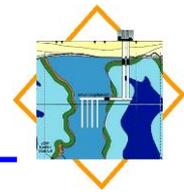
**Figure 4.16: Position of the four drift seals on the 870 m level [Bollingerfehr et al. 2012]**

Calculations related to the safety demonstration were carried out for the drift seals. In all relevant scenarios, the stability and tightness of the drift seal could be demonstrated ([Müller-Hoeppe et al. 2012a], [Müller-Hoeppe et al. 2012b]).

The shaft filling station and the infrastructure areas are backfilled with gravel which has negligible compaction capabilities and which forms permanent pore storage to significantly delay an increase in brine pressure at the drift seals.

#### 4.3.4.3 Shaft Seals

Both shafts will be sealed by shaft seals which are engineered barriers consisting of several components comprising sealing elements, abutments and pore storage. The components and materials (see Table 4.1) were selected in accordance with the geologic environment along the shaft length and the composition of brines that might intrude from the overburden. Based on the geological mapping of the existing shaft 1 in Gorleben, a concept for the shaft seal and a functional model were compiled (Figure 4.17). This concept took into account the



detailed stratigraphic situation along the shaft length, the existing shaft accesses at the exploration level and the planned emplacement level and the composition of potentially intruding brine and its potential timing. Figure 4.17 shows the functional elements of the shaft seal. Starting from the bottom of the existing shaft, first a static abutment will be erected, followed by a sealing element (item 3 in Figure 4.17) which has a twofold task. Firstly the sealing element separates the emplacement level from the shaft and secondly it seals an anhydrite layer with higher hydraulic conductivity called the “Gorleben Bank”. This is followed by another abutment which is connected to the pore storage of the infrastructure rooms at the exploration level, and a second sealing element (item 2 in Figure 4.17), again with the additional task of sealing the “Gorleben Bank”. A long-term sealing element then forms the middle portion of the shaft seal. On top of this long-term seal, a combination of static abutment and porous material will be placed, followed by filter material. The sealing element (item 1 in Figure 4.17) on top of the filter material again has the additional function of sealing the “Gorleben Bank”. The uppermost element of the shaft seal below the shaft foundation is a filter layer. In the overburden formation the shaft is backfilled conventionally.

The design of the shaft seal was investigated by preliminary design calculations. Geochemical calculations were carried out to analyse the influence of water/brine ingress from the top of the shaft through the sealing elements in order to confirm that the selected materials are suitable. The geochemical calculations were followed by preliminary mechanical calculations in order to determine the length of each abutment and to assess potential settling effects in order to avoid damage to the sealing elements. It was estimated whether the shaft seal would be capable of retaining any intruding brine for such a period of time that the compacted crushed salt backfill would be capable of preventing the transport pathway further on [Müller-Hoeppe et al. 2012a].

Based on a scenario analysis for repository development, five design situations for the shaft seal were derived. These comprised the reference scenario with or without seismic impact, failure of a shaft seal, failure of a drift seal, and the impact of low/high convergence rates below or above the anticipated values. In the second step, the overall functional model and the functional elements were investigated in detail by a set of calculations related to the safety demonstration (thermomechanical and hydromechanical). The results showed that the time period of approximately 1,000 years is sufficiently long that in the meantime the backfill compaction has attained values which prevent advective brine flow to the waste canisters, even if a seal fails. The description of the methodological approach and the calculations performed are compiled in a technical report [Müller-Hoeppe et al. 2012b]. Table 4.1 shows the materials selected for the backfill and seals.

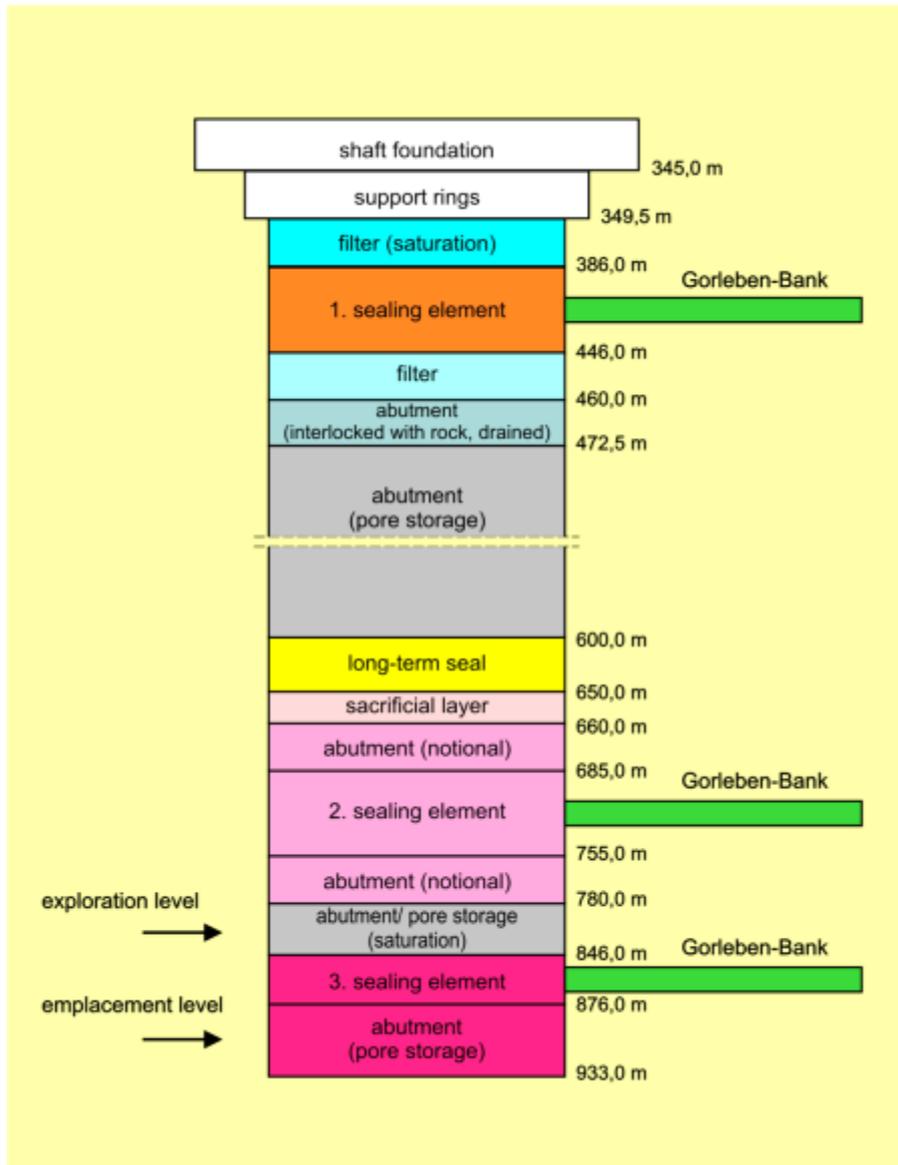
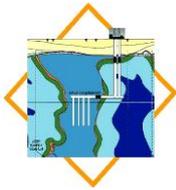
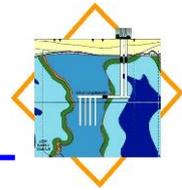
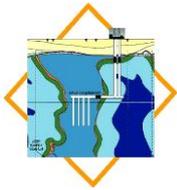


Figure 4.17: Sketch of the functional model of the shaft seal



location in repository	backfill material	sealing material	material specification
emplacement fields East 1 - East 12	dry crushed salt	/.	<ul style="list-style-type: none"> <li>grain size: up to 64 mm</li> <li>initial porosity &lt; 35%</li> </ul> see material specification [Müller-Hoeppe et al. 2012]
emplacement fields West 1 - West 3	dry crushed salt	/.	
cross cuts	dry crushed salt	approx. 10 m plug made of magnesium oxychloride concrete	
shaft 1 and 2	crushed salt (0.5-1.0 mass % content of moisture)	list of sealing materials (top down) <ul style="list-style-type: none"> <li>filter layer/saturation layer: sand/gravel/basalt grit</li> <li>1. sealing element: bentonite</li> <li>filter layer/saturation layer: sand/gravel/basalt grit</li> <li>abutment: salt concrete</li> <li>pore storage: basalt gravel</li> <li>long-term sealing: wet crushed salt</li> <li>sacrificed layer: salt concrete</li> <li>abutment (notional): salt concrete</li> <li>2. sealing element: salt concrete</li> <li>abutment (notional): salt concrete</li> <li>abutment/pore storage: basalt gravel, serpentinit gravel or concrete</li> <li>3. sealing element: salt concrete</li> <li>abutment: magnesium oxychloride concrete</li> </ul>	see material specification [Müller-Hoeppe et al. 2012]
main transport drift North	crushed salt (0.5-1.0% mass content of moisture)	/.	see material specification [Müller-Hoeppe et al. 2012]
main transport drift South	Crushed salt (0.5-1.0% mass content of moisture)	/.	see material specification [Müller-Hoeppe et al. 2012]
cross cut East (for mining transport)	Crushed salt (0.5-1.0% mass content of moisture)	drift seal with core seal and abutment made of magnesium oxychloride concrete	see material specification [Müller-Hoeppe et al. 2012]
cross cut center (for container transport)	Crushed salt (0.5-1.0% mass content of moisture)	drift seal with core seal and abutment made of magnesium oxychloride concrete	see material specification [Müller-Hoeppe et al. 2012]
cross cut parallel to cross cut West (for mining transport)	Crushed salt (0.5-1.0% mass content of moisture)	drift seal with core seal and abutment made of magnesium oxychloride concrete	see material specification [Müller-Hoeppe et al. 2012]
infrastructure area	serpentinit or basalt gravel from shaft filling station up to drift seals	/.	see material specification [Müller-Hoeppe et al. 2012]
ventilation boreholes	/.	magnesium oxychloride concrete	see material specification [Müller-Hoeppe et al. 2012]

Table 4.1: For the R&D project VSG selected materials for backfilling and sealing [Bollingerfehr et al. 2012]



#### 4.3.5 Retrievability

Retrievability of emplaced waste packages has not played an important role in the discussion of the German waste management programme in the past. In 2002 the advisory group to BMU, AkEnd [AKEnd 2002], did not see any reason to consider retrievability in the context of a siting process for an HLW repository. However, in 2010 BMU issued Safety Requirements for the final disposal of heat generating waste [BMU 2010a]. These requirements were taken into account in the repository design within the R&D project VSG [Bollingerfehr et al. 2012] for both emplacement concepts (variant 1 and 2). In the context of retrievability, the most relevant statements of [BMU 2010a] are as follows:

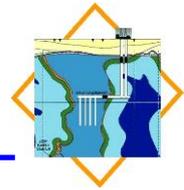
- *retrievability is the planned technical option for removing emplaced radioactive waste containers from the repository mine,*
- *during the operating phase (of the repository) until sealing of the shafts or ramps, retrieval of the waste containers must be possible.*

As boundary conditions, it was considered that a decision to retrieve waste canisters relates to all waste canisters being disposed of at a point in time and that the integrity of such waste canisters had not been impaired by external influences.

In the case of drift disposal the retrieval of waste canisters does, in principle, imply a reversion of the emplacement process. As a first step, a new drift would be excavated parallel to the embedded POLLUX<sup>®</sup> casks while increased cooling is effected using extensive cooling and ventilation systems over a period of approximately 1 year. In the second step, the material above, to the side, and at the end of the canister is removed. Subsequently, the integrity of the canister is verified. After the canisters have been recovered, they are moved into the new parallel drift. The POLLUX<sup>®</sup> casks can be picked up by a modified emplacement device. The transport process underground and to the surface is similar to the emplacement process. The total time necessary to retrieve all POLLUX<sup>®</sup> casks and to transport them to the surface is estimated to be approximately 40 years.

In the case of borehole disposal the original borehole disposal concept (unlined 300-m-deep vertical boreholes) was adapted, that is the boreholes would be equipped with a casing closed/sealed tightly at the top. In addition to this, the design of the waste canister was modified. The casing has been designed to absorb the expected geomechanical pressure of the surrounding rock masses. For this purpose, calculations were carried out to determine a casing cross-section and an appropriate wall thickness. For the Gorleben site the casing would have an outer diameter of 800 mm and a wall thickness of 50 mm.

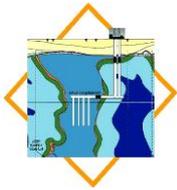
The canisters were designed with a slightly conical shape in order to facilitate retrieval. The canister heads were also sloped to allow evacuation from the backfill material. The waste canisters would be placed in the center of the casing, and the annular space around the waste containers would be filled with a backfill material such as sand. The backfill provides



heat transfer through the casing into the surrounding rock mass and retains its physical properties even under the expected high temperatures. Thus, in case of waste canister retrieval, a reverse extraction of backfill material out of the borehole would be possible. All design steps of emplacing waste canisters in, and retrieving them from, vertical lined boreholes were analysed. A more detailed design of the casing, the waste containers, the backfill material, and the overall processes would need to be carried out in a future R&D project.

One further aspect has to be taken into account when considering retrievability. Before the canisters are retrieved, a concept for their subsequent storage above-ground must be implemented. Either a fully constructed and licensed interim storage facility or a newly constructed and approved final repository might be used.

The handleability of the waste canisters must be shown to be viable for a period of up to 500 years after repository closure [BMU 2010a] and this imposes additional durability requirements on the canisters. The release of radioactive aerosols must also be prevented. In the course of the R&D project VSG this requirement was not specifically addressed in the repository design for Gorleben.



## 5 Safety demonstration

This chapter describes the means, such as analyses and arguments, which are used in the safety case to demonstrate the safety of a repository system on the basis of the safety concept and its guiding principles and design requirements. In the first step a concept is developed and is applied in the safety demonstration (chapter 5). In the second step, the elements of this concept are explained in detail and results from their implementation in the R&D projects ISIBEL and VSG are summarised (chapters 5.1 to 5.11).

### 5.1 Safety demonstration concept

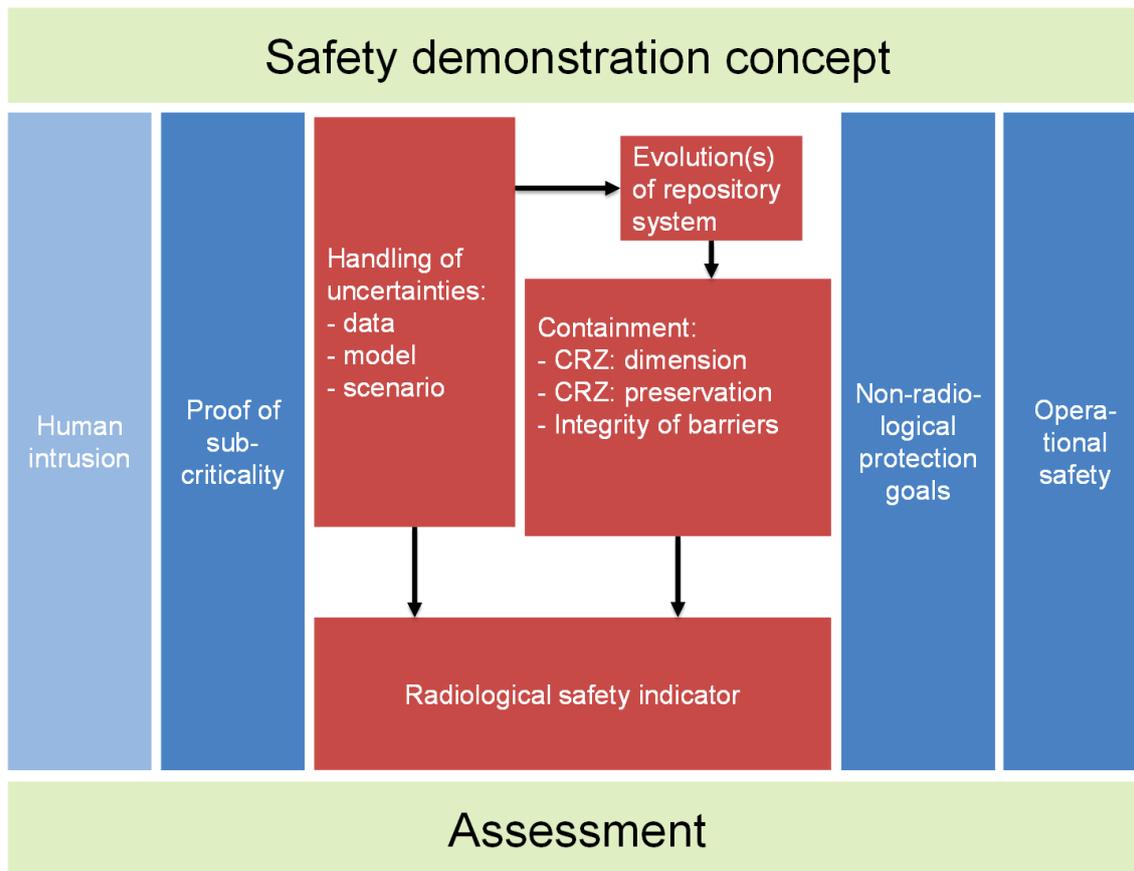
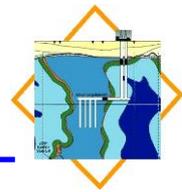
The fundamentals of the safety demonstration concept were developed in the R&D project ISIBEL [Buhmann et al. 2008] and were then refined in R&D project VSG. In the ISIBEL project, the concept focused on the demonstration of the long-term safe containment of the waste by demonstrating the integrity of the geotechnical barriers and the geological barrier. An evaluation of radionuclide release was carried out for evolutions of the repository system for which an impairment of the barrier integrity, and therefore the development of a continuous pathway for radionuclides, could not be excluded. Whether these evolutions are probable or less probable, or whether they can be excluded from further consideration, is covered by the scenario analysis [Buhmann et al. 2008].

According to this concept the decisive elements are:

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

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

The safety concept and the elements of the safety demonstration were further refined in the R&D project VSG [Mönig et al. 2012]. The key elements of the safety demonstration concept are schematically shown in Figure 5.1.

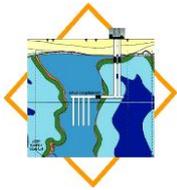


**Figure 5.1: Elements of the safety demonstration**

On the basis of a comprehensive handling of uncertainties, particularly the handling of scenario uncertainties by a scenario analysis, the containment of the waste is evaluated for all scenarios that need to be considered. This evaluation includes the assessment of

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

If the safe containment of the radionuclides in the CRZ can be demonstrated, this assessment is extended by evaluating subcriticality, non-radiological protection goals and operational safety. The safety demonstration concept now also includes an assessment of human intrusion as required by the Safety Requirements, the results being used for optimisation of the repository concept. These additional elements can be regarded as standalone analyses and are represented as single (blue) columns in Figure 5.1.



## 5.2 Handling of uncertainties

Based on the evaluation of international experience, a status report on the handling of uncertainties was published in the R&D project ISIBEL [Buhmann et al. 2010c]. According to this report, a generally accepted strategy for the handling of uncertainties can be summarised as follows:

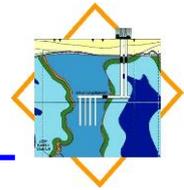
- Identify,
- assess and quantify, and
- reduce and avoid.

This strategy is an iterative process which must be repeated when a new version of the safety case is presented.

In the early stage the safety case primarily aims at site exploration and the layout of the repository. This stage offers substantial possibilities to reduce, and eventually eliminate, identified uncertainties, but new uncertainties can evolve. The knowledge data base of a site can be improved by site characterisation and so uncertainties from incomplete knowledge of the site can be reduced at every step of the safety case by further site characterisation. Remaining uncertainties in this context can be treated in a safety case by general and specific assumptions which have to be plausible and justified by technical expertise. If possible they must be proven by future site investigation and results from R&D programmes. For example, in R&D project VSG [Mönig et al. 2012] the following general assumptions concerning the site were made:

- The lateral extension of the site is consistent with a geological section of the salt dome presented in the literature [Bornemann et al. 2011] and this section can be extended to the depth of the emplacement floor.
- The properties of the salt rock in the exploration field EB1, and in the infrastructure field, can be extrapolated to the salt rocks in the CRZ and eventually to those parts of the emplacement floor outside the CRZ.
- The Hauptsalz of the Staßfurt formation is of sufficient thickness to contain the emplacement areas for all the concepts developed in the project. Accordingly the safe distance to potentially water-bearing rock formations or to rock formations which potentially can develop water pathways must be taken into account.

Another important source of uncertainty at an early stage of a safety case is the emplacement and sealing concept. This is based on the assumptions above, but the realisation of the concept must address many uncertainties. Practical knowledge or confirmation of most of the components of a repository system, their constructability, and compliance with requirements is limited and so additional assumptions regarding operability and quality must also be made. In project VSG assumptions were made, for example, regarding requirements for the emplacement container (impermeability, structural safety and retrievability) and for details of the



drift and shaft seals [Bollingerfehr et al. 2012]. At each step of the safety case more detailed knowledge concerning these technical aspects can be obtained, and the related uncertainties reduced.

The waste data base is also uncertain. This uncertainty, particularly in the early stage of a safety case, is related to the future operation of nuclear power plants including the duration of their operation, the enrichment and burn-up of fuel elements. In Germany the political decision has been taken that the use of nuclear energy will be terminated in 2022. Thus the amount of waste from spent fuel from nuclear power production can be estimated on a firm basis [Peiffer et al. 2012], [Dörr et al. 2012], [Dörr 2013]. Nevertheless, there is still incomplete knowledge on the total amount of waste, because not only spent nuclear fuel, but also other kind of waste, including waste from reprocessing, must be considered. Waste uncertainties must not only be taken into account in numerical calculations, but are also relevant to scenario and model uncertainty.

It will not be possible to reduce or totally avoid all uncertainties. Remaining uncertainties have to be specified and their consequences to the safety statement have to be assessed. The treatment of uncertainties is fundamental to the safety concept and for the demonstration concept (Figure 5.1).

In a safety analysis the inherent uncertainties with respect to the actual site data, the knowledge of the repository system, the future evolution of the repository system, and the description of individual processes and the interaction of processes have to be identified and their consequences assessed. It is international practice [Vigfusson et al. 2007], [Galson & Khursheed 2007] to classify uncertainties as:

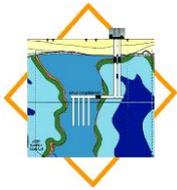
- scenario uncertainties (i.e. uncertainties about the future evolution of the system),
- model uncertainties, and
- data and parameter uncertainties.

In the following sections will be discussed how these uncertainties can be examined in detail.

Many of the uncertainties are listed in the FEP catalogue (chapter 5.3.1). It is thus a relevant basis for numerical calculations in which all kind of uncertainties have to be taken into account.

### 5.2.1 Scenario uncertainties

The future evolution of the repository system is influenced by geological and climatic processes at the site, the influence of these processes being dependent on the layout and concept of the repository and the disposed waste. In reality, the site and the repository will only undergo one evolution. Despite a detailed understanding of the various influencing factors this real evolution cannot be predicted, the reason being that points in time and the



characteristics of future events at the reference site cannot be unequivocally determined. The resulting uncertainty with regard to the future evolution of the repository system can be marginally reduced by additional research and site investigations. For example, it can be assumed that several glacial periods with permafrost formation will occur at the reference site within the next million years and may be associated with glaciations of the site. An exact prediction of when these extreme cold periods will occur and which specific areas will be affected by glaciers advancing from the north is however not possible.

To tackle this problem, scenarios are developed which describe a comprehensive range of plausible evolutions of the repository system. In R&D project VSG, for instance, a reference scenario and alternative scenarios were developed for all variants of the repository concept [Beuth et al. 2012a]. In total, the scenarios should comprehensively represent the reasonable range of repository system evolutions and should thus cover the uncertainties with regard to the real future evolution.

### 5.2.2 Model uncertainties

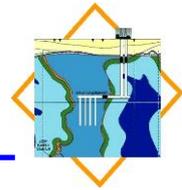
Uncertainties arising from an incomplete knowledge or lack of understanding of the behaviour of engineered systems, physical processes, site characteristics and their representation using simplified models and computer codes are termed model uncertainties [Galson et al. 2009].

In order to understand the repository system, and to analyse its future potential evolutions, different types of models are applied in a safety case:

- models to characterise the situation at the site,
- models of technical barriers,
- mathematical models of individual processes in the system, and
- Performance assessment models.

The model of the situation at a site and the models of repository layout concepts depend mainly on the state of exploration. Uncertainties in these models are based on insufficient knowledge for example due to an early stage of exploration. If numerical calculations must be performed, then specific assumptions regarding the situation at the site have to be made. In the final stage of a safety case, these specific assumptions should no longer be necessary and should, as far as possible, be replaced by models that are results of R&D.

In the case of technical barriers and transport processes, a number of uncertainties in the models exist. Examples are the long-term evolution of the chemical environment at elevated temperatures, the model for the release of radionuclides from the waste forms, the corrosion of metal containers, sorption processes, or the long-term evolution of the excavation disturbed zone (EDZ).



All assumptions and simplifications in the performance assessment model must be supported by benchmarking the results of such performance assessments with the results of mathematical models for individual processes. It has to be shown that the assumptions and simplifications are sufficiently conservative to cover uncertainties in these models and processes.

Some model uncertainties can be dealt with in a numerical calculation by parameter or data uncertainties, see chapter 5.2.3. For instance, the model for failure of a technical barrier (sealing) can be numerically represented for the time period at which the barrier fails. This number can then be treated as an unknown using a suitable bandwidth for its value. Another example is the corrosion rate which applies to the corrosion of the metal containers.

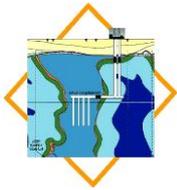
### 5.2.3 Data and parameter uncertainties

All parameter and data used in a safety case are subject to uncertainties. These uncertainties arise from the natural variability in the repository system, statistical inexactness, data relevance or insufficient knowledge [Vigfusson et al. 2007]. Some data such as the gravitational acceleration constant, the Avogadro constant, and others, are well known and must not be treated in uncertainty analyses in greater detail. Other data are known only within large bandwidths, for example the solubility or sorption of elements in a specific environment. But much of the data is characterised by a small bandwidth and can be well represented by best estimate values for deterministic calculations and by values in the bandwidth that can be described by a well defined functional relationship.

Some of the parameter uncertainties arise from model uncertainties. There are two alternative ways to take this into account. A special number for the model may be used as a representative parameter for example the time of a barrier failure. Alternatively a pre-existing parameter can be used as a representative and covering value; for instance, the uplift rate of a salt dome can be used as a representative value that covers some effects of cold periods.

## 5.3 Scenario analysis

The development of scenarios is one of the key elements of a safety case, see for example the discussion in [NEA 2012]. In 1999, a workshop was held by OECD-NEA concerning scenario development methods and practice [NEA 2001]. In that workshop, a variety of methods was presented and discussed, for example the PROSA method [Grupa 2001] that was applied in Germany for safety analyses related to the Asse mine. All of these methods are based on characterising future evolutions of the repository system by features, events, and processes (in short FEP), that are compiled into an FEP catalogue. Recently, in a research project [Beuth & Bracke 2010] a method has been applied to develop scenarios which aim at a comparison of repositories in salt and clay formations. In this project, safety functions have been used as a basis for the development of the scenarios.



In the R&D project ISIBEL [Buhmann et al. 2010b], a methodology has been developed that is closely related to an FEP catalogue. This methodology has been refined in R&D project VSG [Beuth et al. 2012a] and will be described in the following sections.

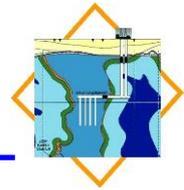
A limited number of reasonably possible evolutions have been derived based on a systematic assessment of relevant influencing factors with the objective of identifying and describing in detail relevant scenarios which allow assessing post-closure repository safety. Due to the safety concept, in the course of the R&D projects ISIBEL [Buhmann et al. 2010b] and VSG [Beuth et al. 2012a] special emphasis was placed on the inclusion of evolutions which would lead to an intrusion of solutions to the waste canisters or which result in a mobilisation of radionuclides from the waste, in both liquid and gaseous phases.

The development of scenarios is based on an FEP catalogue, chapter 0. For every FEP in the catalogue, information regarding its probability of occurrence and characteristic, and information about missing information and future R&D needs are provided. The FEP catalogue of the R&D project VSG [Wolf et al. 2012a], [Wolf et al. 2012b] is considered to be state-of-the-art for such a catalogue, taking the German regulatory requirements into account. In this catalogue, the following information related to uncertainties is included:

- probability of occurrence of every FEP (three classes: probable, less probable and improbable),
- description of the characteristics of every FEP, and
- R&D needs due to incomplete knowledge. This includes open questions that are not typical R&D items, but which must be answered prior to continuation of work.

To reduce uncertainties related to subjective decisions, a number of measures have been taken in the R&D project VSG [Wolf et al. 2012a]. These comprise, for instance, the participation of many scientists from different disciplines or the use of earlier research results. As an example, the FEP catalogue of R&D project ISIBEL [Buhmann et al. 2010a] was taken as a basis. The catalogue was reviewed by external experts and iteratively revised. Despite strong efforts to increase the knowledge base for the FEP catalogue, uncertainties will always remain regarding the completeness of identifying the relevant FEP and their characteristics, and these can only be partially reduced by further R&D.

The scenario development methodology employed in R&D project VSG aims at deriving one reference scenario and a number of alternative scenarios. The methodology enables the assignment of probability classes to the scenarios in line with the regulatory framework [BMU 2010a].

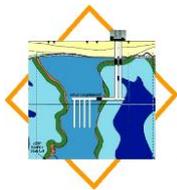


### 5.3.1 FEP Catalogue

An FEP catalogue gives a summary of features characterising the initial properties of the repository system at the end of the operational period and relevant information on events and processes which might influence the future development of that repository system. In the context of a safety assessment, the FEP catalogue is highly relevant as it is the connecting link between the fundamentals (site description, geoscientific long-term prognosis and radioactive waste inventory), the repository concept, and the system analysis. Apart from the compilation of the most relevant basics, the FEP catalogue reflects the interrelation between the site specific conditions and the modifications resulting from the disposal of radioactive waste. Therefore it is important for scenario development and also for demonstrating the integrity of geological and geotechnical barriers and for the analysis of the radiological consequences.

In the course of the R&D project ISIBEL, a host-rock specific, generic FEP catalogue for salt formations was developed based on reference data from the Gorleben site [Buhmann et al. 2008]. This FEP catalogue described all features, events and processes that might influence the future evolution of the repository system. To reflect all relevant future evolutions, the completeness of the FEP database is very important, but it cannot be demonstrated. The FEP catalogue was systematically compiled. The starting point was a comparison with the NEA-FEP database taking into account that the only salt project considered in this catalogue was the WIPP site, which is located in a bedded salt formation [Weber & Keller 2008]. Complementary approaches to identifying potentially relevant FEP were pursued and finally consolidated in order to contribute towards the completeness of the FEP catalogue. Another approach was the bottom-up approach meaning the identification of all FEP that might be relevant for the future evolution of the repository system. For the top-down-approach conceivable scenarios possibly resulting in a radionuclide release (e.g. failure of a geotechnical barrier) were assumed and all FEP that could play a role in these evolutions were identified. To check that the consolidated list was comprehensive, a plausibility check of the sequences and interdependences of the FEP was carried out and this identified some missing FEP. Further indications came from the geoscientific long-term prognosis and, as an iterative process, from the scenario development and the process analyses.

The structure and content of the FEP catalogue reflect the safety concept and the scenario development methodology [Buhmann et al. 2010b]. For each FEP, detailed information was provided that enables the direct selection of all FEP which are relevant for scenario development. In this context, the direct impairment of “initial barriers” was an important criterion and the starting point for scenario development [Beuth et al. 2012a]. An initial barrier is that part of the barriers of the repository system which has, at least temporarily, several safety functions with regard to the safe containment of the radionuclides in the CRZ. These functions are completely developed at the beginning of the post-closure period. For final disposal in salt formations the host rock, the shaft seals, the drift seals and the spent fuel canisters were defined as initial barriers.

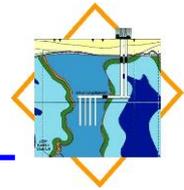


FEP number	FEP title	Date
General description of FEP		
Probabilities; etc	probable <input checked="" type="checkbox"/>	less probable <input type="checkbox"/> not to be considered <input type="checkbox"/> etc. <input type="checkbox"/>
Justification of probabilities, etc.		
General information	situation	consequences
Text entries for additional input (file cards)		

Figure 5.2: Overview of FEP database entry

In Figure 5.2 a sketch of the input mask for FEP descriptions in the FEP database is shown. Each FEP entry in the catalogue comprised a definition, general information, a description of the circumstances at the site and site-specific impacts, a classification of the conditional probability of occurrence, details on the impairment of the initial barriers, effects in the different parts of the repository system, and information regarding the time frame of action. The “conditional probability of occurrence” of the FEP is an important criterion with regard to the methodology of scenario development. These probabilities were derived either from the boundary conditions such as geology and repository concept, from natural principles and causalities such as brine intrusion where there will be metal corrosion and resulting gas production, or by expert judgments including the prognosis of further evolutions. The direct relationship with other FEP were specified and explained, thereby distinguishing between initiating FEP, resulting FEP, affecting FEP, and affected FEP, respectively. Where it was found possible, probable and less probable characteristics of the FEP were indicated. Sometimes it was possible to describe a characteristic but not to attribute a probability. This could be due to restricted data or information or to a situation where only bounding values are of interest with respect to the scenario analysis. In those cases, representative characteristics were described. The FEP catalogue also documented the results of expert discussions in explanatory statements and compiled all literature relevant to the descriptions of the FEP. In doing so, it contributed to the reliability and transparency of the information used. To facilitate its effective use, all relevant information was recorded in a data base.

The FEP catalogue had been analysed and evaluated in the course of a national peer review by five expert organisations. As a result it was confirmed that the FEP catalogue complies



with the international state of safety assessments for repositories [Buhmann et al 2010a], [Krone et al. 2008].

For project VSG, the FEP catalogue was site-specifically adapted and substantiated [Wolf et al. 2012a], [Wolf et al. 2012b] so that the most relevant data were compiled characterising the present site status, including the status of site exploration and the provided waste inventory and the developed repository concepts. This FEP catalogue contained 115 FEP entries. In four FEP the most relevant data of the inventories of radionuclides, metals, organics and other materials were compiled. These data were based on two inventory reports [Peiffer et al. 2011], [Dörr et al. 2012]. Experience from application of the methodology for scenario development in project VSG has shown that it was not necessary to include the inventory FEP in the FEP catalogue as they describe only boundary conditions for repository system evolution and a description in a comprehensive inventory report would be sufficient. With regard to the “conditional probability of occurrence” of the FEP, 98 FEP were categorised as “probable” and 4 (dealing with malfunctions of geotechnical barriers) as “less probable”. For 19 FEP, including 6 probable FEP, the FEP catalogue demonstrated that they need “not be considered” for scenario development because of geological or technical boundary conditions or because of their weak site-specific characteristics (FEP screening). By considering “conditional probability of occurrence” and “effects in the different part of the repository system” the number of FEP relevant for scenario development can be further decreased.

Eventually, 92 probable and four less probable FEP built the basis for scenario development [Beuth et al. 2012a]. In this context, initial-FEP, that means probable FEP with direct impairment of the initial barriers, and the FEP Radionuclide Mobilisation and Radionuclide Transport were of particular relevance.

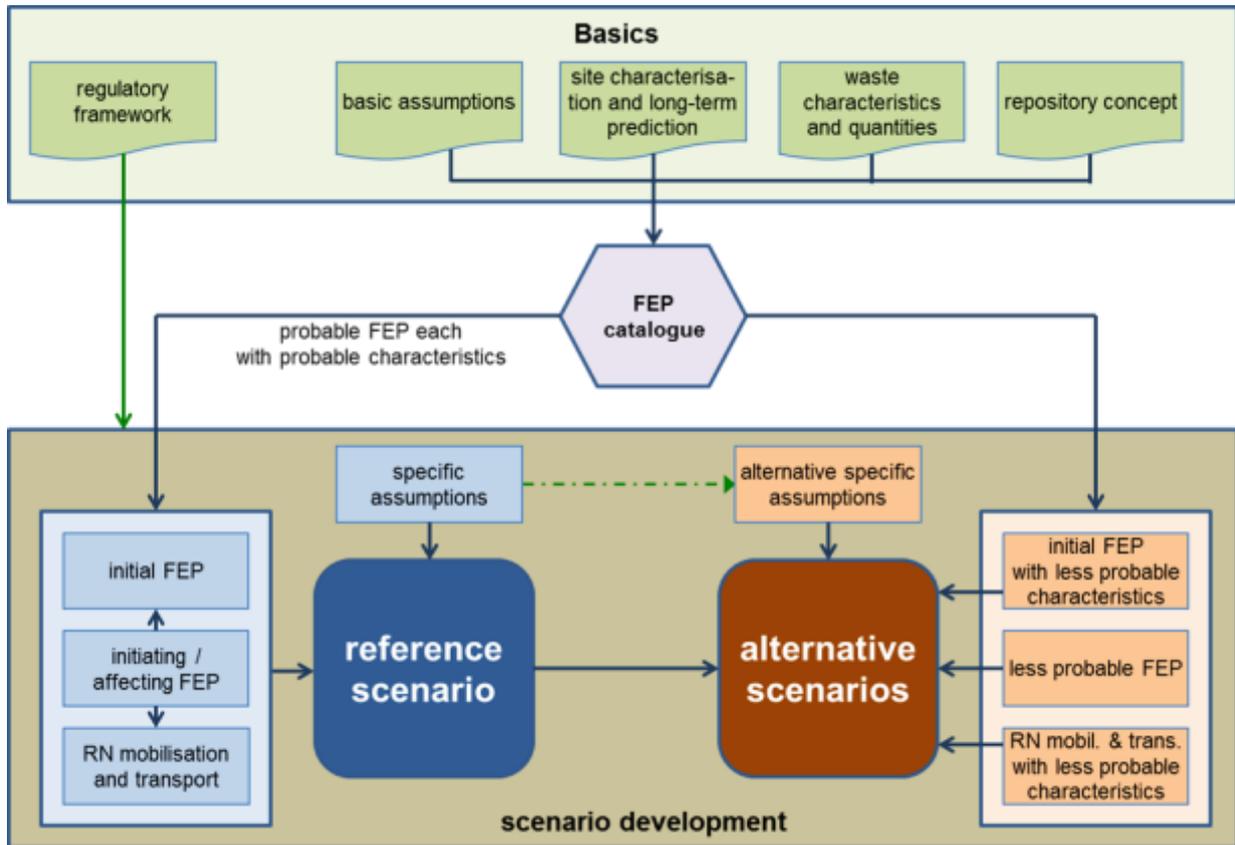
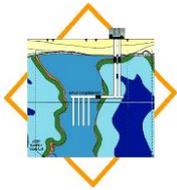
### 5.3.2 Scenario-Development

#### Methodology

##### Fundamentals

The methodology described in this chapter was developed in two projects. Initially, in an early phase of the R&D project ISIBEL [Buhmann et al. 2010b], the methodology was deduced and tested for reference scenarios. This methodology was then expanded in the course of R&D project VSG for the development of alternative scenarios [Beuth et al. 2012a].

The methodology aims at deriving, in a systematic manner, a limited number of plausible scenarios specifically one reference scenario and a number of alternative scenarios. Overall, the scenarios should comprehensively represent the reasonable range of repository system evolutions. The methodology allows direct assignment of probability classes to the scenarios in accordance with the safety requirements of [BMU 2010a] and is depicted schematically in Figure 5.3.



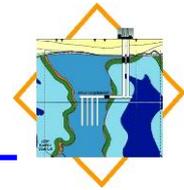
**Figure 5.3:** Scenario development methodology applied in R&D project VSG [Beuth et al. 2012a]

The fundamentals of scenario development include basic assumptions, specific assumptions, site description and geoscientific long-term prognosis.

A number of basic assumptions are essential in order to deal with uncertainties resulting from incomplete site investigations below ground (see chapter 4.1), since so far only a relatively small region of the salt dome has been investigated *in situ*. The basic assumptions relate to the lateral size and geological structure of the Gorleben salt dome in the emplacement depths, the properties of the salt rocks in the CRZ, and the available dimension of the rock salt of the Staßfurt Series.

Specific assumptions provide means to deal in a transparent and traceable way with particular uncertainties, some of which may be minimised in future, while others such as prognosis of future climatic evolution may never be reduced. The specific assumptions for the reference scenario are defined as follows [Beuth et al. 2012a]:

- Climate evolution: For site evolution, the climate evolution is characterised by a 100,000 year cycle with periodical changes of ice ages and warm intervals. A certain sequence of climate changes is defined as a standard climate evolution for the reference scenario.



During an ice age similar to the former Elster-type, the onset of a new glacial channel following an existing channel is assumed.

- Geotechnical Barriers: The shaft and drift seals will be constructed in such a manner that they comply with all design requirements.
- Safety pillar: There are neither misinterpreted results from exploration nor undetected geological characteristics which could result in a reduction of the scheduled safety pillar of 50 m between the excavations of the emplacement area and liquid bearing formations or pathways.
- Exploration level: The exploration level will be isolated from the repository level by appropriate backfilling and sealing, thus avoiding any pathways to the emplacement level or any disturbance of the evolution of the repository system.

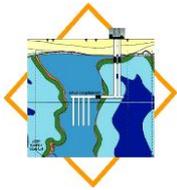
The site description includes the current status of exploration (geological data of host rock and overburden formations, hydrogeology), the provided waste inventory and the designed repository concepts. The repository concept options analysed in the course of the R&D project VSG ([Bollingerfehr et al. 2011], [Bollingerfehr et al. 2012]) include the combined disposal of low and intermediate level waste with negligible heat generation in the western area of the repository and of heat-generating high level waste in the eastern area of the repository (see chapter 4.3). For heat-generating waste, two main repository variants have been analysed: drift emplacement of POLLUX<sup>®</sup> and CASTOR<sup>®</sup> casks (variant 1 + for comparison: horizontal borehole emplacement of CASTOR<sup>®</sup> casks) and vertical borehole emplacement of retrievable canisters (variant 2).

The long-term geoscientific prognosis describes the future evolution of the site based on an analysis of site evolution in the past (termed the 'actuality principle').

Scenario development commences at two starting points that follow directly from the guiding principles for deriving the safety concept:

- A number of initial barriers (host rock, shaft seals, drift seals, and spent fuel canisters) are identified that constitute a subset of all barriers acting in the repository system via diverse mode of operations and, partly, in different time frames. Their collective characteristic is that these barriers prevent the contact of solutions with the emplaced waste immediately on closure of the repository system. Initial FEP that could impair the functionality of the initial barriers provides the one starting point for scenario development.
- In addition, all potential system evolutions which involve a release of radionuclides from the waste form, including those without any contact between external solutions and the waste form. Those FEP which relate to the mobilisation and transport of radionuclides constitute the second starting point for scenario development.

All information required by the methodology of scenario development can be directly extracted from the FEP catalogue due to its comprehensive structure and content.



Since the repository design may influence the system development in a significant manner, a reference scenario and several alternative scenarios have been developed for each disposal variant.

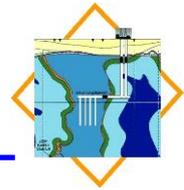
The Safety Requirements [BMU 2010a] distinguish between probable, less probable and improbable evolutions and define assessment criteria for these classes. Improbable evolutions need not be considered. Therefore the scenarios developed in the R&D projects ISIBEL and VSG have been assigned to the corresponding probability classes. The classification of the scenarios is based on the conditional probability of occurrence of the scenario-defining FEP and the probability of their characteristics. For example, if all initial FEP and their characteristics are probable, the resulting scenario (reference scenario) is probable. For the same reasons most alternative scenarios are less probable. Only alternative scenarios resulting from alternative specific assumptions may also be probable, the reason being that such specific assumptions cannot be characterised with regard to their probability and characteristics. Therefore the classification of their probability is based on expert judgment.

### Reference Scenario

A reference scenario does not only include one specific evolution but describes as broadly as possible the covering spectrum of potential evolutions of the repository system. Following the methodology characterised above, the reference scenario has been derived from a combination of probable initial FEP and the FEP 'Radionuclide Mobilisation and Radionuclide Transport' (Figure 5.3). For these FEP, probable characteristics are assumed which are derived from direct impact by other probable FEP. These FEP have either an initiating or affecting effect on the initial FEP respectively on the FEP 'Radionuclide Mobilisation and Radionuclide Transport'.

For the description of the future evolution of the repository system, it must be taken into account that the relevant FEP may have different characteristics at different times and in different parts of the repository system. Therefore to optimise the clarity and the traceability of the description, it is useful to subdivide the description of the reference scenario into four subsystems – near field, host rock, drifts and shafts, and overburden formations – and to consider the interrelation between the subsystems and possible chronological limits of the initial FEP. When a repository system for different kinds of radioactive waste has to be analysed, the two subsystems 'near field' and 'drifts and shafts' should be also subdivided for the different emplacement areas.

Apart from evolution options of the initial barriers, evolutions resulting in a radionuclide mobilisation and a subsequent radionuclide transport via liquid or gaseous pathways have to be considered because of their relevance for the demonstration of compliance with radiological safety criteria.



Several relevant characteristics of the reference scenario for combined emplacement of waste with negligible heat generation and heat-generating waste (drift emplacement, variant 1) are given in [Beuth et al. 2012a] as follows.

Near field, eastern area (heat-generating waste): The waste containers have a durability of 500 years. There are only very small amounts of liquids resulting from fluid inclusions in the host rock and in the crushed salt and the residual moisture in the waste containers. Therefore only very limited corrosion of the waste containers and a low gas generation rate are expected. The heat production of the waste increases the convergence of the drifts and thus forces the compaction of the backfill ("dry" crushed salt).

Near field, western area (waste with negligible heat generation): There is no requirement for container durability, therefore failure of all containers just after closure has been assumed. The waste contains some liquids, therefore intensive corrosion and gas generation will start just after closure and therefore, early mobilisation of radionuclides will occur. The compaction of the crushed salt will be retarded because of low temperatures.

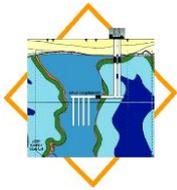
Drifts and shafts: Drift seals and shaft seals are designed to have a durability of 50,000 years, therefore glacial impacts on the shafts are not relevant. The EDZ at the barrier locations will be removed during construction and remaining fissures will be sealed by convergence after closure. The infrastructure area that is located between the shafts and the sealed access drifts to the emplacement areas will be backfilled with gravel and will act as a reservoir for fluids intruding from the host rock or via the shafts. A brine intrusion from reservoirs in anhydrite layers of the Leine-Series is assumed for the reference scenario.

The access drifts are separated from the infrastructure area by drift seals. They are to be backfilled with crushed salt with a moisture content of 0.6 % to enhance the backfill compaction. The fluid pressure will be mainly controlled by the enclosed mine air.

Host rock: The Hauptsalz of the Staßfurt-Series only contains minor fluid inclusions (0.02 % moisture) and does not show any open fractures. The heat input from the waste will result in stress redistribution in the rock. There are very low uplift and subsidence rates of the salt dome. Only the top of the salt dome will be affected by glacial impacts, such as glacial channels.

Overburden formations: These include several aquifer levels. A fluid pressure resulting from the groundwater will influence the top of the salt dome and hydrochemistry will affect the subsidence of the cap rock. During ice ages, the overburden formations may be completely altered.

Interactions between the different waste types in the western and eastern areas are not assumed for the reference scenario because correctly working drift seals effectively separate the emplacement areas.



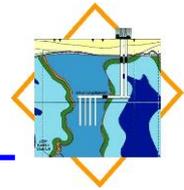
As an example for the application of the ISIBEL / VSG methodology, the reference scenario for repository variant 1 has been derived in a systematic manner and described in detail in the course of the R&D project VSG [Beuth et al. 2012a]. For other alternative repository variants, only an analysis of differences from the reference scenario for variant 1 has been performed. Specific assumptions, initial FEP and the FEP 'Radionuclide Mobilisation and Radionuclide Transport' are the same for the different emplacement variants, but the characteristics of these FEP may be different due to varying boundary conditions. For example, the geometry of the emplacement areas, the number and properties of the different waste containers, the heat and stress field in the host rock and appropriate impacts on the excavations and barriers are specific for the emplacement variants. Many far field processes of geosphere and boundary conditions like climatic evolutions are only marginally influenced by the different emplacement variants, if at all. Therefore the main focus is on the near field and the adjacent areas.

### Alternative Scenarios

The reference scenario covers a range of probable evolutions of the repository system, where the range is as large as possible. Alternative scenarios are evolutions which differ in only one aspect from the reference scenario. The methodology for derivation of alternative scenarios is modified related to the reference scenarios and based on four different starting points (Figure 5.3):

- Evolutions arising from alternatives concerning the specific assumptions for the reference scenario. It has to be checked whether resulting scenarios are probable or less probable.
- Evolutions resulting from less probable characteristics of the initial FEP that may impair the functionality of the initial barriers. If plausible consequences on resulting and affected FEP seem to be relevant, the impairment of the reference system has to be described in an alternative scenario. Because of the less probable characteristics of the initial FEP, the resulting alternative scenario is also less probable.
- Evolutions resulting from less probable characteristics of the FEP describing mobilisation and transport of radionuclides. By means of resulting and affected FEP the requirement to define a new less probable alternative scenario has to be checked.
- Evolutions resulting from less probable FEP. The differences of the system evolution from the reference scenario can be specified by the characteristics of the resulting and affected FEP. The alternative scenarios are similarly less probable.

If possible, information for scenario development is directly taken from the FEP catalogue concerning less probable characteristics of FEP that could impair the functionality of the initial barriers or that describe mobilisation and transport of radionuclides respectively, and less probable FEP. Otherwise, the characteristics are directly controlled by the initiating and affecting FEP.



It is possible that similar alternative evolutions result from different starting points. In this case, various evolutions may be consolidated into one representative alternative scenario that covers the characteristics of various evolutions.

In R&D project VSG, 17 alternative scenarios were developed and described in detail for the drift emplacement disposal concept. Regarding the alternatives to specific assumptions, alternative scenarios have been defined because of divergent glacial impacts, misinterpreted and undetected geological properties and a new pathway between exploration and emplacement level. From less probable characteristics of initial FEP and the FEP Radionuclide Mobilisation and Radionuclide Transport the definition of 9 alternative scenarios was necessary. Evolutions resulting from less probable characteristics of the other initial FEP may be consolidated into other representative alternative scenarios that cover the characteristics of various evolutions. All less probable FEP need the definition of additional alternative scenarios except for 'Piping of seal elements' which can be assigned to the alternative scenarios 'early loss of integrity of a drift seal' and 'early loss of integrity of shaft seal'.

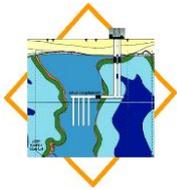
Alternative scenarios for the different emplacement variants were derived in the same way as the procedure for the reference scenarios, that is by difference analyses. As a result, no additional alternative scenario in comparison to the alternative scenarios for the repository variant A / B1 had to be defined. This was due to identical specific assumptions, initial FEP, relevant FEP (Radionuclide Mobilisation and Radionuclide Transport) and less probable FEP. However, in detail, the development of the alternative scenarios will vary because of different characteristics and the strength of interrelation between the FEP.

#### 5.4 Determination and dimensioning of the CRZ

The determination of the containment-providing rock zone, in particular its dimensions, is essential for the assessment of the containment of radionuclides in this zone. According to the Safety Requirements *the applicant should provide a clear spatial and temporal definition of the containment providing rock zone.*

The CRZ is part of the geological barrier and the demonstration of its integrity is essential for the safety assessment. Nevertheless it is important to analyse the surrounding parts of the CRZ since they play an important role (protective function) for the preservation of the CRZ during the assessment period (chapter 5.5). The part of the barriers which is involved in the demonstration of integrity is dependent on the repository system. For domal salt, the demonstration of integrity is carried out for the whole salt dome (= integrity of the geological barrier, chapter 5.6) and also for the shaft and drift seals (= integrity of the geotechnical barriers, chapter 5.6).

A release of radionuclides from the CRZ may happen locally along the man-made access system, such as the shafts, or across the boundary surface of the CRZ to the surrounding host rock. Assuming that the integrity of the geological barrier can be demonstrated, a re-

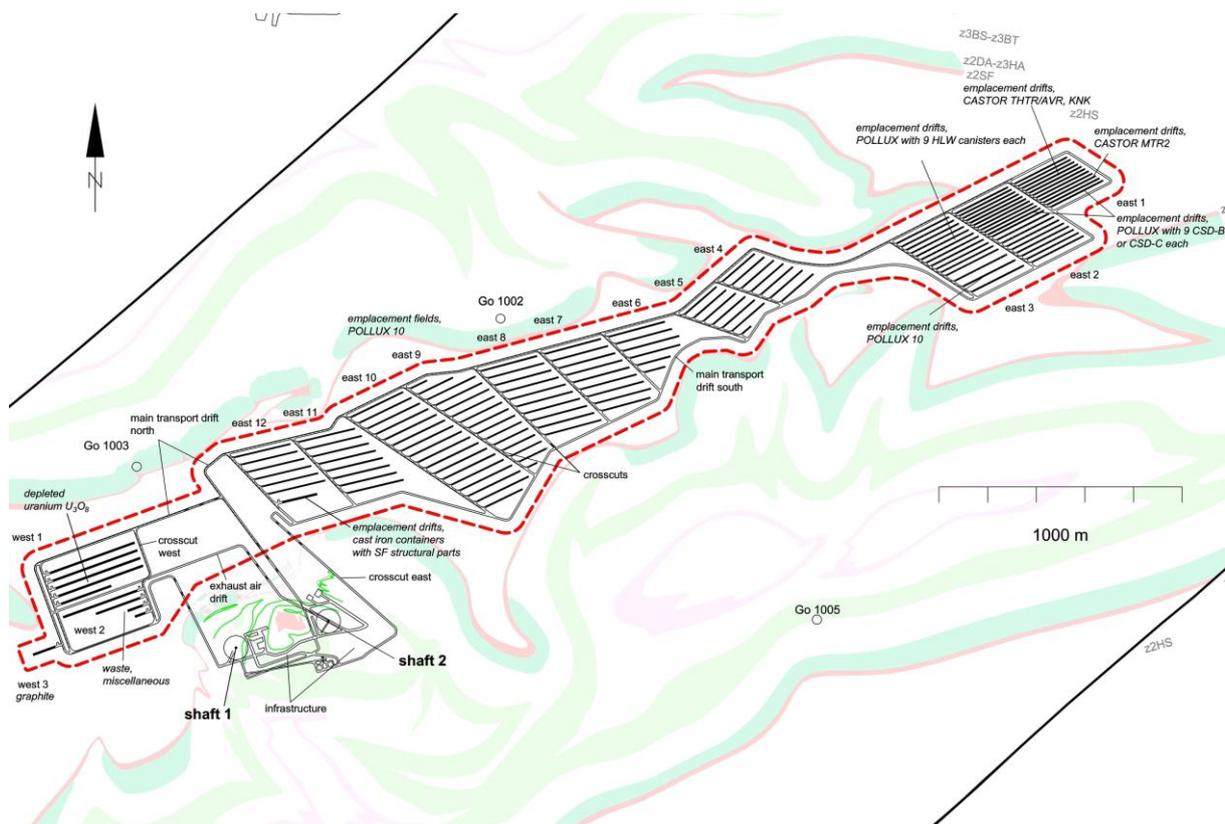
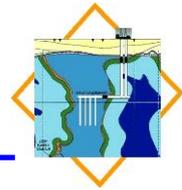


lease of radionuclides must be assessed solely for pathways along the mine system of the repository. Here, a potential reference point for the evaluation of the radionuclide migration from the CRZ is at either the shaft seals or the drift seals, which separate the flank drifts from the infrastructure system.

If the numerical results lead to a range of dimensions for the CRZ, then the CRZ is determined by a process which takes into account different criteria:

- the ability to characterise the properties and to predict the future evolution of the corresponding rock zone, the geotechnical barriers and the general data situation,
- the quality of containment (cf. RGI in chapter 5.9) taking into account the uncertainties of release calculations including assumptions about the evolution of the repository system, and
- the certainty of the statement of the proof of integrity, taking into account the uncertainties such as homogeneity of geochemical material features in the proof of integrity

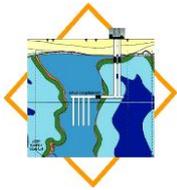
As an example, in R&D project VSG the CRZ was finally determined [Fischer-Appelt et al. 2013] to surround the mining structures at a distance of 50 m (this distance was proposed in [Mönig et al. 2012] and ending at the sealings in the flank drifts close to the infrastructure area (Figure 5.4).



**Figure 5.4:** Example of a CRZ (dashed red line) in a repository in a salt formation [Fischer-Appelt et al. 2013], modified

## 5.5 Preservation of the CRZ in the assessment period

The salt barrier above the CRZ serves as a protective layer for the CRZ. However the containment capability of a repository system is not directly affected by the removal of parts of the geological salt barrier outside the CRZ. The removal of parts of the geological salt barrier outside the CRZ can therefore be an acceptable process during the future evolution of the repository system. However, it is a prerequisite for the functioning of the geological salt barrier within the CRZ that the salt rock within the CRZ is preserved. Several geological processes introduce the potential to eliminate parts of the bedrock directly or indirectly to assist erosion or subsrosion processes. Removal of the bedrock can occur mechanically (erosion) and also chemically as dissolution of rock salt by groundwater (subsrosion). Relevant for the preservation of the CRZ are therefore those features and processes which can influence groundwater flow conditions or abrasive activity at the ground surface. Future geological and climatic evolutions in this regard could include geological features and processes such as diapirism, erosion, crustal deformation and the future climate with the expected sequence of glacial and interglacial epochs.



Another geological process that might be considered in the context of the preservation of the CRZ is magmatism. In this perception a volcanic intrusion into a repository is interpreted as an extrusion of barrier rock material by the intruded magma rather than as damage to the barrier. Classifying magmatism as damage to the barrier would result in assessing it in the next chapter dealing with the integrity of the barrier.

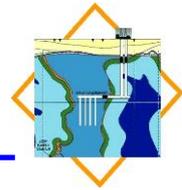
Finally, processes have to be examined during which the CRZ remains in existence but where waste canisters leave the CRZ. This can be imagined if waste canisters tend to sink downwards in the viscous rock salt due to density differences.

Meteor impact with an energy level sufficient to remove parts of the CRZ need not to be considered as it is of such low probability. Furthermore, the devastating non-radiological consequences of such an event significantly exceed any radiological consequences.

### 5.5.1 Uplift and subsrosion

When rock salt comes into contact with water it can be dissolved as long as the water is sub-saturated. If dissolution takes place at the top of a salt formation, the top will be lowered at the location of dissolution. By this process a depression is formed wherein the groundwater flow intensity is reduced, and thereby the dissolution rate decreases. The dissolution process is further diminished by a layering of the groundwater due to an increase in density resulting from increasing salinity. Finally, the dissolution rate will reduce to a midget extend, controlled by the diffusional transport rate of dissolved salt through the density-layered groundwater. Therefore, the depth of the top of a salt formation can remain constant at a certain depth for a geologic period. This steady state will be disturbed if the top of the salt formation is shifted by salt uplift into a shallower depth with lower groundwater salinity. For this reason, the future subsrosion rate is linked to the future salt uplift.

Salt uplift is a component of salt tectonics or diapirism. The diapiric stage at the Gorleben site started when the salt broke through the overburden during the Cretaceous. Diapirism slowed down from the start of the Tertiary when the salt dome entered the so called replenishment phase [Köthe et al. 2007], Table 15 and 16]. A reconstruction of the evolution history of the Gorleben salt dome based on an interpretation of the stratigraphy and structure of the overburden and adjoining rocks can be found in [Zirngast et al. 2004]. According to this interpretation, the original thickness of the bedded salt in the area of investigation was approximately 1,400 m, whereby the minimum thickness must have been 1,150 m in order for the rim synclines to form. According to [Köthe et al. 2007], p. 178 ff] 64 % of the primary Zechstein rock in the source area migrated into the salt dome. This left behind only residual thicknesses of Zechstein rocks of 100 – 500 m in the immediate vicinity of the salt dome. It can be assumed that these residual beds consist of a range of different evaporitic horizons which have different creep properties. Because of the diapirism of the Gorleben salt dome which has taken place up to the present time, it is likely that a large proportion of the remain-

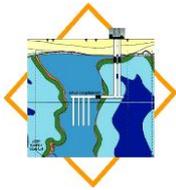


ing 36 % of the Zechstein consists of rocks with low creep properties. The conclusion that can be drawn from this is that if diapirism were to continue, there is only a small amount of the residual volume which would be capable of migrating into the salt dome. More details on the volumes of migrated salt and the analysis of the rim synclines are contained in [Zirngast 1991].

The speed at which diapirism took place was estimated for the Hauptsalz at a depth of around 900 m to be a maximum flow rate of 0.34 mm/year during the Upper Cretaceous, and up to 0.07 mm/year from the Miocene to the Quaternary. The associated uplift rates at top salt were estimated at 0.08 mm/year in the Upper Cretaceous and approximately 0.02 mm/year from the Miocene to the Quaternary. This indicates that diapirism reached its maximum in the Upper Cretaceous, and has since diminished through into the Quaternary. During this diapirism, the different creep properties of the various lithological units, and their differences in competence, led to the formation of complex flow folds in parts of the salt dome. The creation and healing of joints in the salt formation can also be attributed to this process.

It can be assumed that diapirism at the Gorleben site will continue to diminish over the next million years because a build-up of high compressive stresses in the rock sequence is not expected in this tectonically quiescent zone. It is also considered unlikely that additional layers of sediment exceeding several hundred metres in thickness will be deposited in the area over the next million years which, by raising the overburden pressure, would give rise to a considerable increase in the rate at which the salt is moving upwards. The low proportion of sequences with good creep properties in the residual thicknesses of the rim synclines also supports the conclusion that diapirism will continue to diminish. If the rate of uplift of top salt during the Miocene to the Quaternary is used as a benchmark, then top salt can be expected to rise by another 20 m in the next million years. This is not an absolute figure, because the salt dome is also affected by the forecast slowing down in the rate of diapirism, and counter-acting processes such as subsidence and erosion which take place at the same time. Moreover, the flow rates of the Hauptsalz group at significant depths (approximately 800 – 900 m) in the central part of the anticline could have much higher values as in the past. This could cause a deformation of the geometry of a repository constructed in the Hauptsalz. However, these movements do not equate exactly with those which developed in the top salt. Extrapolating the estimated rate of 0.07 mm/year for movement within the Hauptsalz group at this depth from the Miocene to the Quaternary gives a salt movement of 70 m in a million years at this level. This could give rise to the formation of new folds or a continuation in the evolution of old folds within the salt formation.

The rates of uplift and the flow rates described above are averages for the given time period of approximately 23 million years. Short-term fluctuations in these values over a single million year period are therefore possible. An increase in these rates could be initiated for instance by a lengthy glacial period which led to the formation of a thick ice sheet above the site – a situation which cannot be completely excluded. The additional load of the ice, particularly at



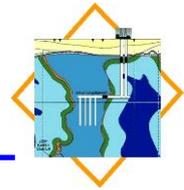
the edge of the ice sheet, would have an effect and would modify the stress conditions in the underlying geology.

Because of the large distance to the potential Scandinavian centres of ice accumulation with very significant ice thicknesses, and the climate changes interpreted in the past, the evolution of this process in the vicinity of the Gorleben site would be limited to a period of only a few tens of thousands years. The short duration of this effect, and the ice thicknesses predicted to exist at the site, which are not expected significantly to exceed 1,000 m, means that if such a situation ever arose, it would not give rise to an increase in diapirism which lasted for a significant length of time. At a marginal ice sheet position in particular, the thicknesses of the ice at the site would be such that when combined with the previously discussed geological conditions, this would not be expected to have an influence on diapirism.

Predicting the future evolution of subsidence at the Gorleben site is based on a climate forecast, which anticipates glacial periods with intensities ranging from the Elsterian to the Weichselian glacials and intervening interglacial periods. This forecast most closely resembles the period from the Elsterian to the recent for which [Bornemann et al. 2008] reported in [Köthe et al. 2007], p. 175], estimated a subsidence rate of 0.1 to 0.2 mm per year.

If this rate is extrapolated to a period of a million years and ignores superimposed processes such as the formation of a new channel, subsidence of around 100 to 200 m of salt would be possible. [Köthe et al. 2007], p. 178], forecasts a future subsidence rate of 0.01 to 0.05 mm per year taking into consideration the depth of the recent top salt and using as a basis reports [Keller 2001] and [Keller 1990] which compile and statistically analyse data from several North German salt domes. These subsidence rates would then give rise to subsidence of 10 to 50 m of salt over the investigated period of a million years. Because it is thought unlikely that diapirism will increase over this period, it is not expected that the subsidence rates will be significantly enhanced by this process.

If it is also assumed that top salt would only reach depths at which subsidence can no longer take place during the later stages of the million year investigation period, it can then also be assumed that subsidence rates during the glacial periods taking place before this time would again be reached and would range from 0.1 to 0.2 mm per year. In this case, it is possible that the reported value of 50 m for the overall forecast could be exceeded. Subsidence between 50 and 100 m is therefore expected as an approximation for the future evolution corresponding to a rate of 0.05 to 0.1 mm per year which incorporates the glacial effect and the influence of the depth of top salt.



### 5.5.2 Erosion

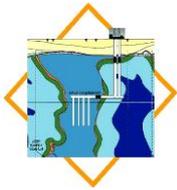
Erosion includes all those processes which modify the geomorphology by linear or areal erosion. Linear erosion is often associated with fluvial processes but can also be caused by glaciers. The main erosional components in this case are sub-glacial erosion and the outflow of melt water generated by the pressure conditions. Significant erosion can take place in parts of the area covered by the glacier and can also give rise to the formation of channels.

Glacigenic channels could form at the Gorleben site during future glacial periods. According to present understanding, the channels can only form beneath an ice sheet, however, this situation is only thought likely to develop during the next but one ice age. If history repeats itself at the Gorleben site and another channel of a similar size to that formed during the Elsterian glacial were to form, then this could again erode the overburden in a similar way. However, neither the existing channel nor the salt dome itself would be preferential locations for channel formation. The material filling the Gorleben channel does not, for instance, differ significantly from the sediments into which it was eroded. Moreover, the salt rocks forming the salt dome behave in a similar way to hard rock and are eroded more slowly than unconsolidated sediments. As previously described, the depths of channels known to have been cut into hard rocks are of the order of 50 m deep, and similar values can be assumed for the salt formation. If it is assumed that an extremely deep channel similar to the Hagenower channel with a depth of over 500 m were to develop in the Gorleben area in future, this would first mean eroding the overburden sediments which currently have a thickness of approximately 250 to 340 m. When the base of the channel reaches top salt, channeling slows down just as it does in hard rock. This slowing down of channel formation means that the time available for channeling until the start of the following interglacial will probably not be sufficient to cut a channel with a total depth of approximately 500 m. Because channels do not remain open but are filled with sediments during glacial retreat, there is no additive effect of superimposed channel formation even at one and the same site.

### 5.5.3 Magmatism

Magmatism is a collective term for all processes and formations associated with magma (molten rock) rising out of the mantle or deeper parts of the earth's crust. Magmatism can involve intrusive and extrusive processes depending on whether the uprising magma cools down and solidifies at depth and forms plutonites or whether it reaches up as far as the surface (volcanism). Magmatism is observed along plate boundaries, for example, during orogenesis or graben formation.

Because of the conditions existing at the Gorleben site, no magmatic processes and associated geothermal activities are expected in the next million years. This also applies to the formation of plutonites and vulcanites, and the release of gas or the formation of thermal springs which are often associated with the late phases of magmatic events. The only places in Germany where volcanic activity may be reactivated are considered to be in the Eifel and



in the Vogtland. These areas are distant from the Gorleben site so that any direct influence can be excluded. The tectonic conditions existing at the site support the conclusion that active Alpidic orogenesis would not give rise to the formation of a fault which could enable magma to rise up and affect the Gorleben site within the next million years.

#### **5.5.4 Container sinking**

An estimate of the sinking range of the heavy waste containers in the viscous salt yielded a maximum distance of not more than a few decimetres [Wolf et al. 2012b]. No additional effect on the preservation of the CRZ for the Gorleben site due to container sinking therefore has to be considered.

### **5.6 Integrity of the geological barrier**

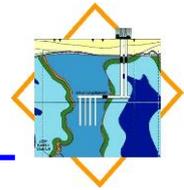
The term “integrity” is used to describe the containment capability of the rock salt. Integrity means, in the case of domal rock salt, the absence of interconnected pore spaces such that practically no hydraulic or diffusive flow processes can occur. Thus the integrity indicates the ability of the salt barrier to permanently prevent the inflow of fluids into the repository from the overlying and surrounding rock and to prevent leakage of contaminated fluids and gases from the repository to the outside environment.

Virgin rock salt in salt diapirs, apart from isolated fluid reservoirs, is practically dry and impermeable and thus its integrity is existent.

#### **5.6.1 Concept for the evaluation of the integrity of the geological barrier**

To assess the long-term behaviour of the integrity of a natural rock salt barrier, all processes which might lead to the evolution of microcracks and subsequent permeation pathways must be analysed.

Comprehensive laboratory analysis of the petrophysical properties of rock salt showed that stress states below the dilatancy boundary do not cause any damage even in the long-term, which means that the preservation of the impermeability of the salt rock is proven for stress states below the dilatancy boundary. But if the applied stress exceeds the dilatancy boundary, microcracks are formed within the salt. The initially impermeable rock salt becomes permeable if the microcracks interconnect at a persisting stress state above the dilatancy boundary [Hunsche & Schulze 2002]. For the assessment of the long-term barrier behaviour, therefore, all processes that might influence the state of stress must be analysed.



The natural state of stress in the surroundings of a repository is predominantly altered by mining activities and by heat input from the waste.

In addition to the impact of mining activities and heat input from the waste, a few further processes and features have to be considered due to their influence on the stress conditions. The stress level directly depends on the geotectonic stress field, this stress field being governed by large-scale tectonics. A rapid modification of the stress field could occur during an earthquake if the source is located close to the salt formation. In this case the formation of individual faults has to be examined if the induced stress exceeds the short term shear strength of the salt rock. The state of stress in the rock salt can also change if chemical reactions within the rock salt, such as thermochemical sulphate reduction, produce reaction products with volumes different to the starting material. Exothermic reactions of hydrocarbons, which occur as natural constituents of rock salt deposits, can be neglected, because the heat from such reactions is very small compared to the heat supplied by the waste [Bracke et al. 2012], even on highest estimates. The state of stress can also be considerably modified if a glacier approaches or overlies the site. Another phenomenon having an effect during a glaciation period is ground cooling. Due to the different thermal expansion coefficients of different rock types, the cooling can cause unfavourable stress conditions especially at rock type boundaries. In this regard the potential for the evolution of cryogenic faults at the top of salt diapirs has been discussed in the literature.

To determine the resulting state of stress in the rock salt barrier, numerical model calculations are performed using finite-element methods. Modelling has to consider the specific repository characteristics and the geological structure of the repository site.

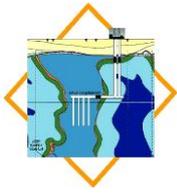
It is known from laboratory and in-situ testing that the integrity of the barrier can also be impaired if a fluid pressure which exceeds the minimum principal rock stress exists. The integrity indicator “state of stress” is thus not only analysed in relation to the dilatancy boundary value, but also by a depth related theoretical fluid pressure.

## **5.6.2 Practical evaluation of the integrity of the geological barrier**

### **5.6.2.1 Evaluation of the mechanical rock properties**

For the evaluation of the integrity indicator “state of stress” by means of numerical model calculations, it is necessary to know the critical indicator values, which are the depth related theoretical fluid pressure and the dilatancy boundary, and the mechanical rock properties as input parameters for the numerical model calculations.

The theoretical hydrostatic fluid pressure according to depth can easily be determined from the height and density of a hypothetical fluid column which extends to the ground surface.



The dilatancy boundary was measured in laboratory investigations. The results from these laboratory investigations confirm that the dilatancy boundary is a function only of stress and does not depend on the type of salt, the load geometry or the loading rate. According to the results of the laboratory measurements, the position of the dilatancy boundary with respect to octahedral stress can be expressed as

$$\tau_{Dil} = 0.8996 \sigma_o - 0.01697 \text{ MPa}^{-1} (\sigma_o)^2$$

with  $\sigma_o = (\sigma_1 + \sigma_2 + \sigma_3)/3$  octahedral normal stress

$$\tau = [(\sigma_1 - \sigma_2)^2 + (\sigma_2 - \sigma_3)^2 + (\sigma_3 - \sigma_1)^2]^{1/2} / 3$$
 octahedral shear stress

Dilatancy occurs if the octahedral shear stress exceeds the dilatancy boundary [Cristescu & Hunsche 1998]:  $\tau > \tau_{Dil}$ .

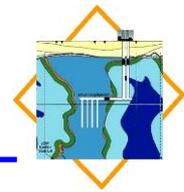
The mechanical rock properties which are required as input parameters for the numerical model calculations depend on the applied constitutive law. One of the constitutive laws which describe the dependency of creep deformation over time on the stress state is the BGRa creep law [Cristescu & Hunsche 1998]

$$\dot{\epsilon} = V \cdot A \cdot \exp\left(\frac{-Q}{R \cdot T}\right) \cdot \left(\frac{\sigma}{\sigma^*}\right)^n$$

with  $A = 0.18 \text{ d}^{-1}$ ,  $Q = 54 \text{ kJ/mol}$ ,  $R = 8.314 \cdot 10^{-3} \text{ kJ/(mol} \cdot \text{°K)}$ ,  $n = 5$ ,  $\sigma^* = 1 \text{ MPa}$  (reference stress) and  $T = \text{temperature in K}$ .

The only rock-dependent parameter in this equation is the prefactor  $V$ . It is a measure of the ability of the rock to creep. Nearly 1,000 samples from the different stratigraphic units of the Gorleben salt diapir were used in laboratory measurements of the mechanical rock properties [Bracke et al. 2012], [Bräuer et al. 2011]. The results show distinct and large differences in the steady state creep rates of different rock salt types, which are due to differences in the impurity distribution and are mostly related to the stratigraphic position. Average values of the pre-factor were measured as between 1/32 and 2 for various rock salt types in the Leine Series. The average values for pre-factor  $V$  in the Hauptsalz of the Staßfurt Series lay between 1/2 and 2 [Heusermann et al. 2012a].

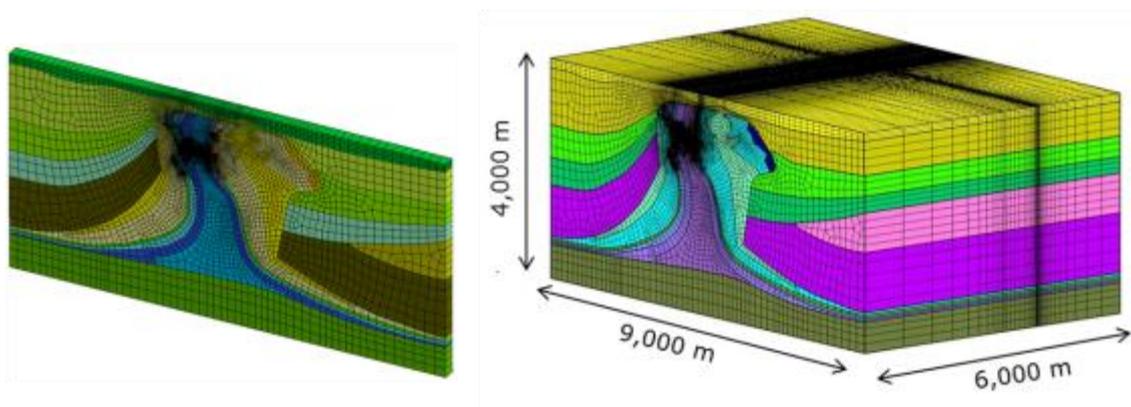
The other mechanical rock properties needed for the numerical model calculations are the static moduli of elasticity and the static Poisson's ratio. These were also determined in laboratory measurements for the different rock salt types and are documented in numerous technical reports (see [Bracke et al. 2012] and [Bräuer et al. 2011]).



### 5.6.2.2 Numerical model calculations

A finite-element model for geomechanical calculations can be based on the geological model of the salt formation by assigning the different salt types and the surrounding rock according to creeping capacity and other mechanical properties to homogeneous zones of the model. For geomechanical modelling, the geological structure of the host rock and the overburden and the geometry of the rooms are idealized and simplified. A geological model of the Gorleben salt dome is shown in Figure 4.2.

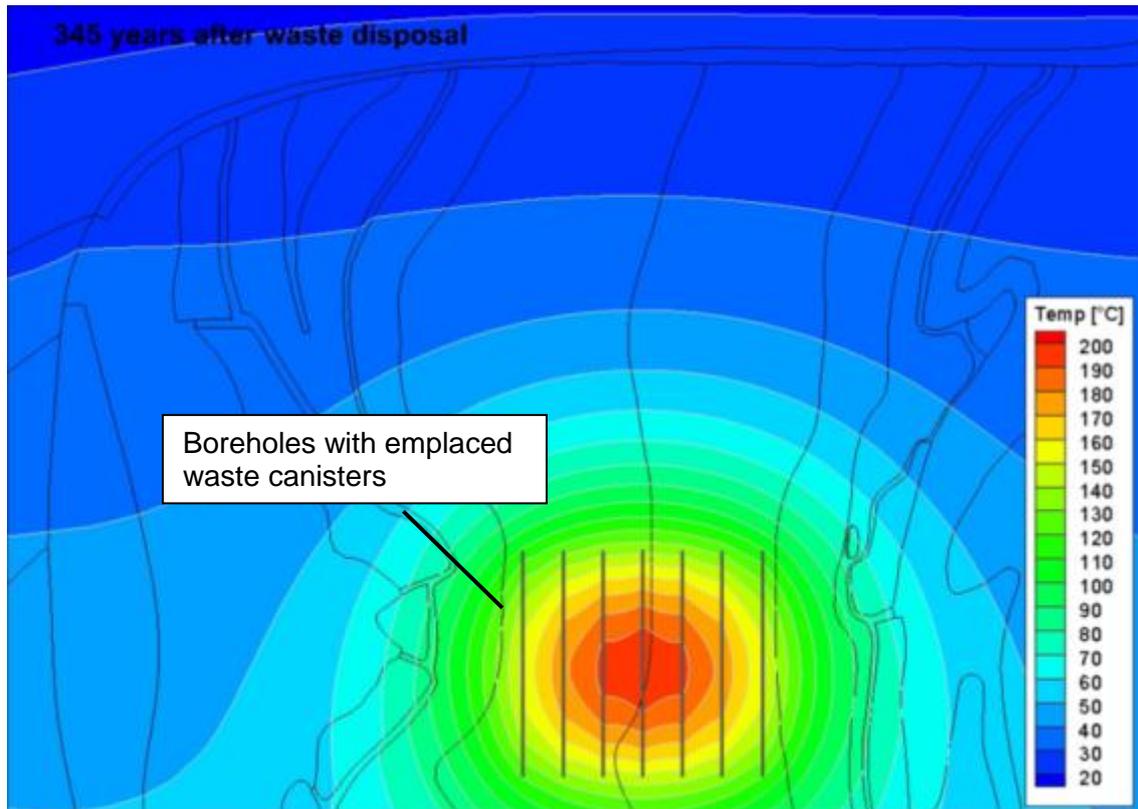
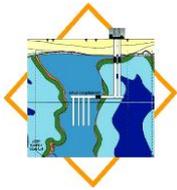
For some problems it is necessary to consider three-dimensional spatial effects, an example being in connection with time variable temperature fields. Appropriate models can be created by extruding a two-dimensional cross section into the third spatial direction, if the geologic variability in the extrusion direction is low. Figure 5.5 shows a finite-element model of the Gorleben salt dome which was used to analyse the temperature effects of different repository designs.



**Figure 5.5:** Set-up of a finite-element model of the Gorleben salt dome (left: 2-D model, right: extruded 3-D model) [Heusermann et al. 2012a], [Heusermann et al. 2012b]

The model was based on a geological cross section of the Gorleben salt dome which was extruded into the direction of the longitudinal extension of the salt dome. It consists of nearly 20 different homogeneous zones, according to different salt and overburden layers. The model observes thermomechanical interactions and is therefore also parameterised with respect to thermal conductivity, thermal expansion and thermal storage capacity. The emplacement of the total volume of HLW from electricity production in Germany until the end of the nuclear phase out in 2022 was assumed. The results show the temperature field at any time step and the corresponding state of stress compared to the dilatancy boundary and to the depth related hypothetical fluid pressure as indicators of the integrity.

**Fehler! Verweisquelle konnte nicht gefunden werden.** shows the temperature field for the borehole disposal concept at the date at which the maximum temperature in the centre of the emplacement area is reached, which is the case after about 350 years.



**Figure 5.6:** Predicted temperature difference in the rock salt 345 years after waste emplacement (borehole disposal concept) [Heusermann et al. 2013]

The maximum vertical lifting of the surface above the repository area due to the thermal expansion of the heated underground amounts to 4 m after approximately 1,000 years.

Figure 5.7 shows the difference between the minimum principal stress and the hypothetical depth related fluid pressure. Areas where the minimum principal stress remains higher than the hypothetical depth related fluid pressure are plotted blue whilst green represents rocks where the integrity is not defined and where the integrity indicators cannot be analysed. The minimum principal stress goes below the hypothetical depth related fluid pressure in only a small region at the top of the salt dome. This zone reaches its maximum extent after 30 years and disappears after approximately 3,000 years.

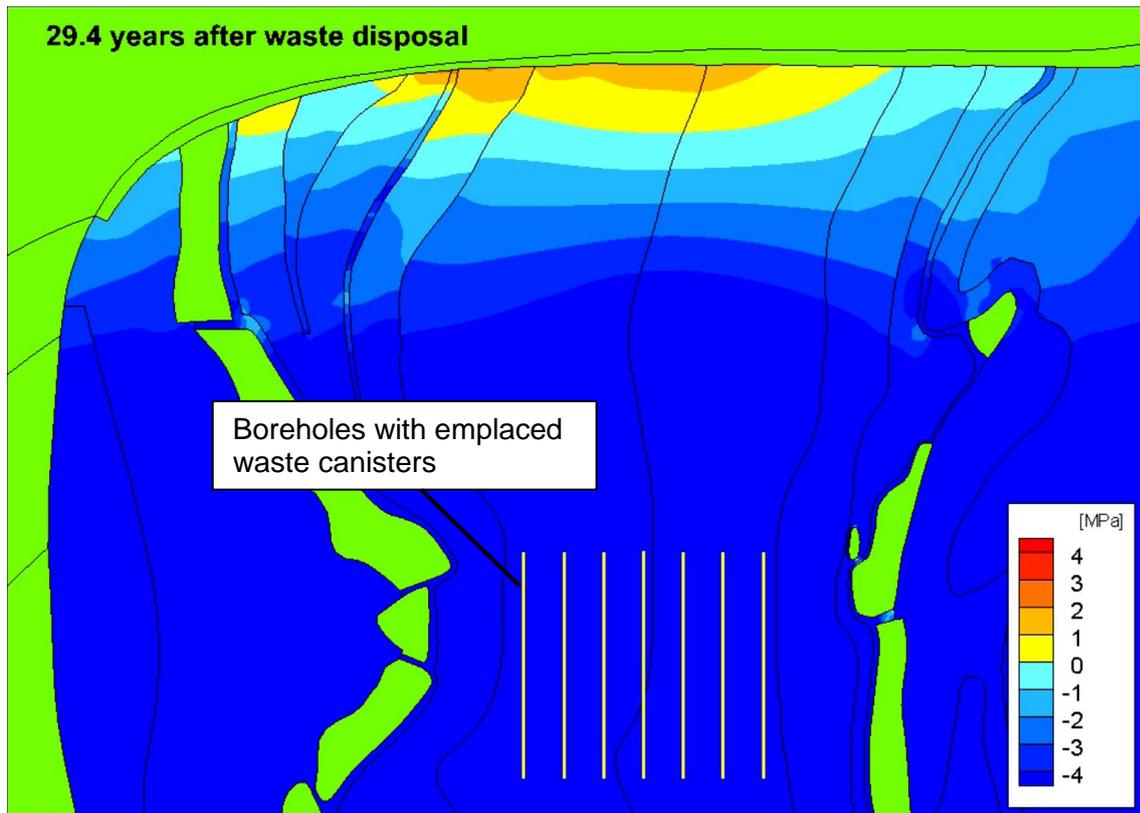
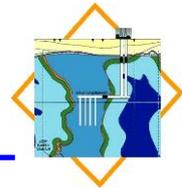
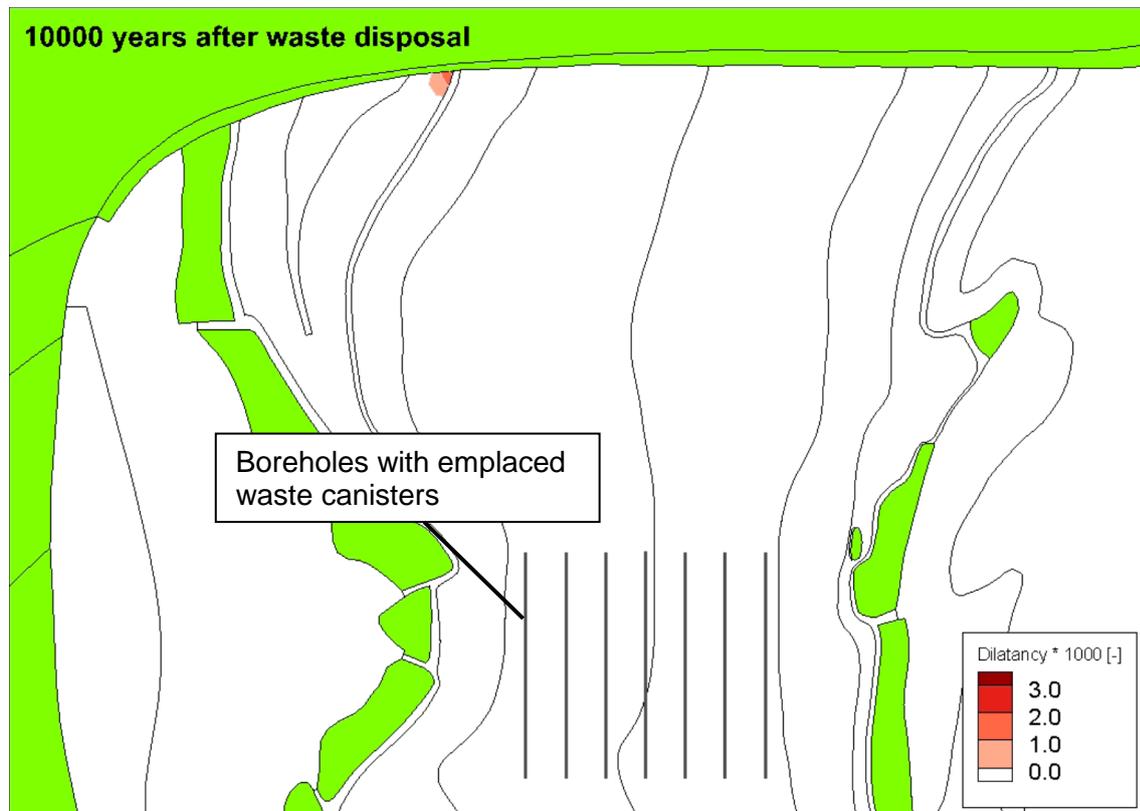
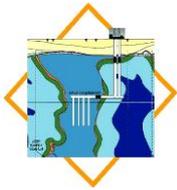


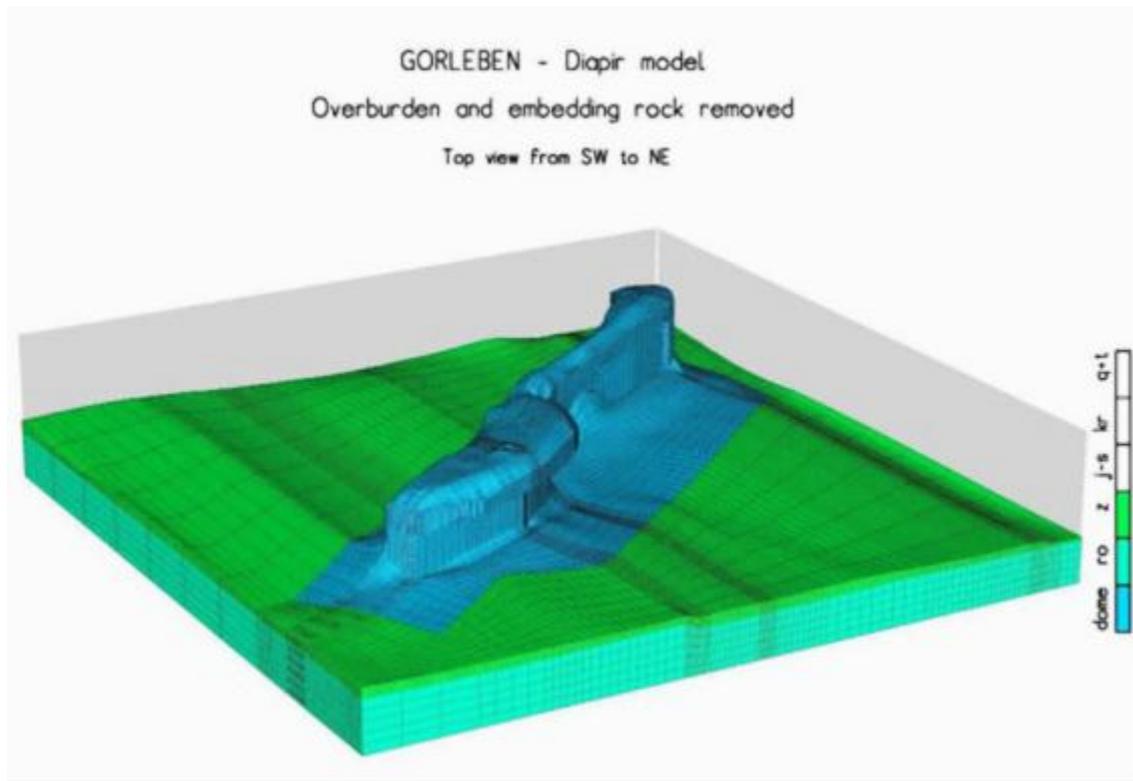
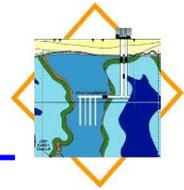
Figure 5.7: Difference between minimum principal stress and hypothetical depth related fluid pressure [Kock et al. 2012]

The zone in which the octahedral shear stress exceeds the dilatancy boundary reaches its maximum extent after 10,000 years. Even at this maximum stage its extent is fairly small compared to the thickness of the salt barrier (Figure 5.8).



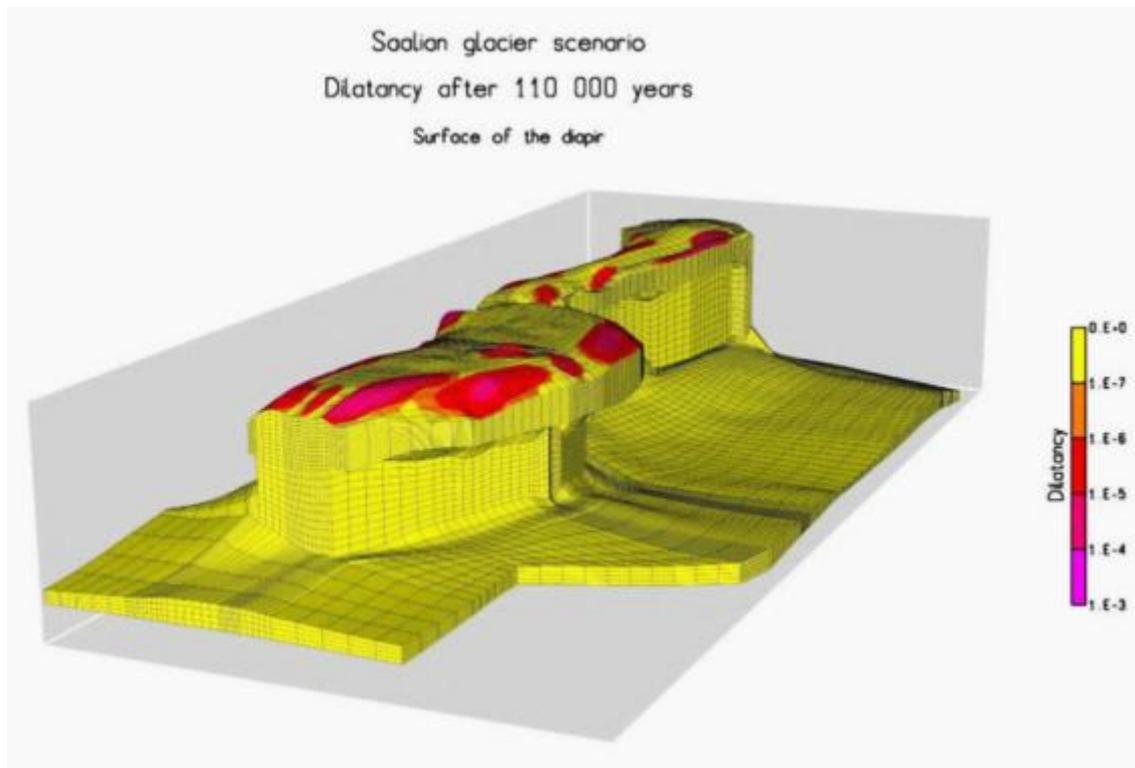
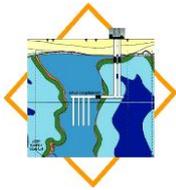
**Figure 5.8: Areas in the rock salt with stresses above the dilatancy boundary [Heusermann et al. 2013]**

A finite-element model would be even closer to reality if it is based on a real three-dimensional geological model instead of extruding a two-dimensional geological cross section. Such a real three-dimensional finite-element model was set up for the Gorleben salt dome to determine the effect of glaciation of the site on the integrity of the salt rock. Figure 5.9 shows the elements representing the Zechstein and older units. Overlying and surrounding rocks are removed in this illustration.



**Figure 5.9:** Three-dimensional finite-element model of the Gorleben salt dome [Heusermann et al. 2012a], [Heusermann et al. 2012b]

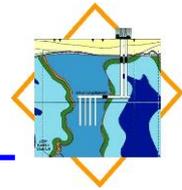
The effect of glaciation was investigated with respect to thermal and mechanical impacts. Thermal impacts arise from the cooling of the overlying rock and the salt dome due to a decrease in the surface temperature during the glacial cycle. Mechanical effects are caused by a glacier moving close to the site or overlaying it. The results of the finite-element calculations show that the integrity of the salt barrier is preserved during a glacial cycle. Only in the case of the complete glaciation of the site and during periods of melting of the glacier are there some planar zones at the top of the salt dome in which the state of stress can exceed the dilatancy boundary (see Figure 5.10). These zones are restricted to the top of the salt dome and do not extend into it.



**Figure 5.10:** Dilatant zones at the top of the Gorleben salt dome during glacier melting [Kock et al. 2012], see also [Heusermann et al. 2012b]

### 5.6.2.3 Additional considerations

The preceding chapter described results from finite-element calculations which consider the impact of mining activities and heat input from the waste on the state of stress in the rock salt barrier. Additionally, the effects of a glacial period on the state of stress with regard to the cooling of the surface and a potential traverse of the site by a glacier were considered. Other processes and features which might influence the state of stress, such as the geotectonic stress field or thermochemical sulphate reduction, were disregarded in the calculations. This is considered credible due to the deformation behaviour of rock salt. Because of the ability of rock salt to creep, the state of stress within a rock salt formation always tends to isotropy. Therefore the state of stress inside a salt dome is, in the long-term, only a function of the depth and the density of the overlying rock. Every perturbation of this stress-isotropy is diminished over time by creep deformation of the rock salt. It is assumed that any change of the geotectonic stress field at the Gorleben site will happen sufficiently slowly, that an impact on the state of stress within the salt dome will immediately be countered by creep deformation. It is therefore appropriate to ignore the geotectonic stress field in the finite-element calculations of the state of stress. Any impact of thermochemical sulphate reduction on the state of stress is ignored in the finite-element calculations for the same reason.



The effect of earthquakes on the integrity of the salt rock barrier is assessed completely differently. With regard to the stress change rate of earthquakes, the ability of rock salt to creep is futile and it can be deemed to behave elastically. Some finite-element calculations were made with respect to the impact of earthquakes on the rock salt barrier of the Gorleben salt dome. According to the rules for the determination of site specific design earthquakes, an MSK-intensity of 7.3 was derived for the Gorleben site [Leydecker et al. 2008]. Assuming an earthquake of this intensity resulted in a slight enlargement of the zones with impaired integrity [Minkley et al. 2010].

For all scenarios with different combinations of impacts on the state of stress which were considered in the Preliminary Safety Analysis for the Gorleben site (VSG), the integrity of the rock salt was maintained for a barrier thickness of at least several hundred metres [Kock et al. 2012].

## 5.7 Integrity of the geotechnical barriers

### 5.7.1 Objectives of the draft of the sealing system

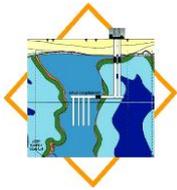
An adequate closure system for the Gorleben site was developed, taking into consideration the Safety Requirements of and based on the salt-specific safety and demonstration concept (chapters 3 and 5) that was derived in the course of the first phases of the R&D project ISIBEL [Buhmann et al. 2008] and further developed in the R&D project VSG [Minkley et al. 2010]. The prerequisite for the long-term safe containment of radioactive waste is the integrity of the geological and, during their anticipated functional lifetime<sup>7</sup>, the geotechnical barriers. The barriers should prevent or minimise solution inflow<sup>8</sup> to the waste or a release of contaminated solutions. The closure concept is comprised of shaft seals, the infrastructure areas backfilled with gravel and the drift seals (cf. chapter 4.3). The remaining excavations are backfilled with crushed salt. The final disposal containers also have a containment function and must be retrievable during the operational phase. Handleability must be guaranteed together with prevention of a release of radioactive aerosols during the first 500 years of the post closure phase according to the Safety Requirements [BMU 2010a]. Compliance with these requirements will be assured by the container design and the repository concept (chapter 4.3) and is not further considered here.

Underground facilities that must be excavated for the repository operation may become potential fluid pathways through the geological barrier during the post closure period. Therefore they should be tightened by the sealing structures until the backfill material crushed salt put

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<sup>7</sup> In construction engineering a special wording of terms is common which is sometimes different from the wording in long-term safety assessment. So "functional lifetime" is identical with the term "design working life" (in the sense of construction engineering, see glossary of [DIN 2010])

<sup>8</sup> In this chapter "solution inflow" is understood the "Inflow from above ground, overburden and formation solutions"



in the mine excavations can take over the containment function. The crushed salt will become compacted over a few thousand years such that it then has a lower hydraulic conductivity than the sealing constructions and hydraulic properties comparable to the surrounding rock mass. To cover uncertainties in the prognosis of the compaction of the crushed salt, a proportionately longer functional lifetime of the sealing structures is required. Since predictions on the influence of the overburden water, which is relevant for the design of the shaft seals, can only be made up to the next ice age, a functional lifetime of 50,000 years is assumed for the sealing structures.

Significant requirements for the sealing system design are derived from study of the most likely evolution of the repository system, which will be described in the reference scenario [Beuth et al. 2012a]. In developing the barrier design, it must be ensured that any release of radionuclides into the biosphere does not exceed permissible levels.

Additionally, any possible impairment of the function of the sealing system and resulting radiological releases based on less probable evolutions (alternative scenarios) must be investigated [Beuth et al. 2012a]. If the results indicate that during these evolutions the predicted level of release is unacceptable, the closure concept must be optimised so that the radiological protection goals of the Safety Requirements will be met. The previously considered scenario of a solution inflow through the main anhydrite for safety demonstration in salt formations can be ruled out for the Gorleben site (chapter 4.1). Limited solution inflow from the Leine Series salt formations cannot be excluded, whereas only very small solution inclusions occur in the main salt of the Staßfurt Series.

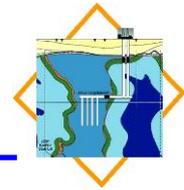
### 5.7.2 Constructional design of the sealing components

The design of the sealing components was carried out in the detail of a technical functional proof following technical regulations [DIN 2009], DIN-EN-1997-1 and [DIN 2010], DIN-EN-1990 (=implementation of the EUROCODE), [DGGT 1997] and [DAfStb 2004], the goal of which is to verify the required level of reliability of barrier constructions. In constructional design practices for the design, the concept of ultimate limit states is used in combination with the partial safety factor method. Thus the limit state conditions of the sealing system must be determined and the associated impacts<sup>9</sup> and impact combinations and resistances must be identified. In addition the ultimate limit state analysis must be assigned to design situations.

Scenario development for the repository concept is based on specific assumptions and on the FEP catalogue [Beuth et al. 2012a]. The FEP catalogue (chapter 5.3.1) compiles all features, events and processes that are relevant for the future evolution of the repository system. Impacts, resistances and design situations are assigned to the FEP as much as

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<sup>9</sup> The corresponding term in the sense of construction engineering is "actions" (cf. [DIN 2009])

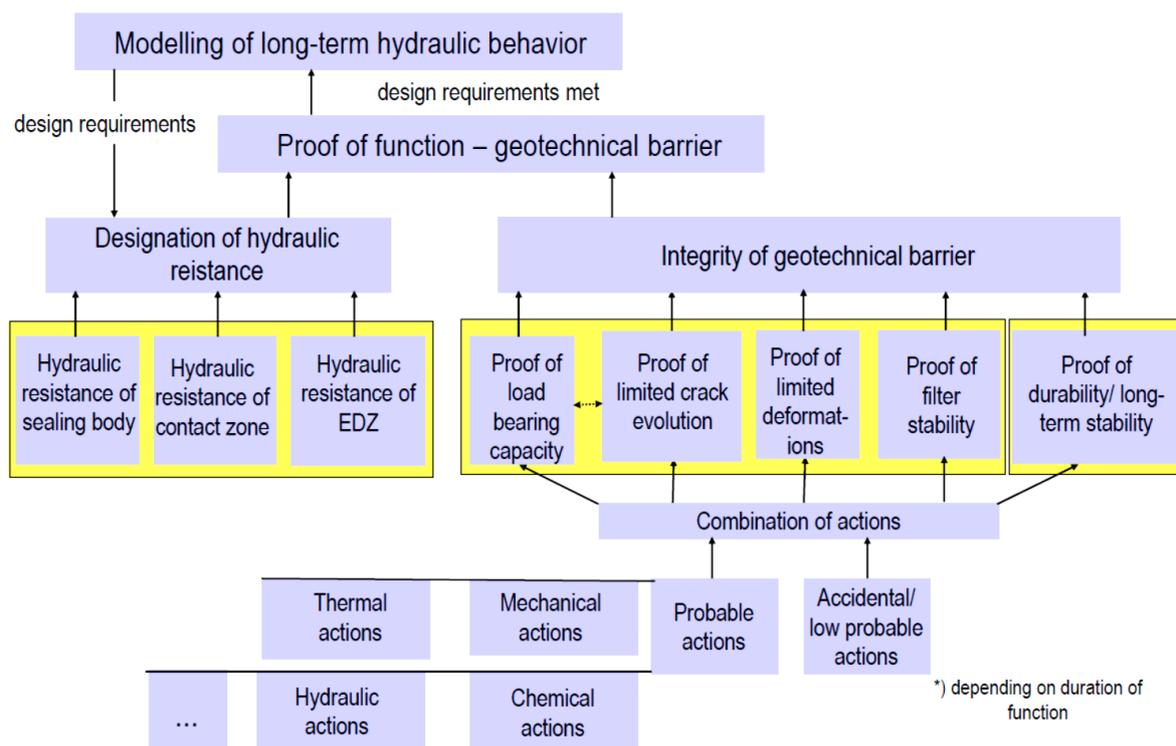


possible, for consideration during the technical functional proof. By this step the FEP and scenarios can be considered in the technical function proof and a sizing of the components of the sealing structures can be carried out with the methods of partial safety factors.

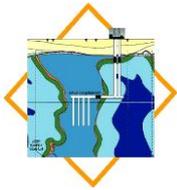
Hence the technical functional proof is carried out, taking into consideration impacts, reflected by the FEP, which can impair the function of the sealing structures (Figure 5.11). Differences are made between chemical, thermal, mechanical and hydraulic impacts. Design situations for the technical functional proof are derived from the following FEP:

- "Earthquakes"
- "Early loss of integrity of a shaft seal", and
- "Early loss of integrity of a drift seal".

Resistances are specific to the material and design and therefore depend on the outline design of the sealing structures. The resistance and the ultimate limit state functions must be determined for the selected design.



**Figure 5.11:** The structure of the technical functional proof. Here "actions" are synonymous with "impacts". Hydraulic impacts = hydromechanical impacts, thermal impacts = thermochemical and thermomechanical impacts



A functional lifetime of the repository sealing system that goes far beyond the 50 – 100 years that is normally required for conventional structures must be set. Nevertheless, the procedure can be used for the dimensioning of the sealing structures, since the relevant impacts and resistances are derived based on the functional lifetime frame [Müller-Hoeppe et al. 2012b].

### 5.7.3 Design of the sealing structures

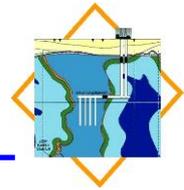
For the sealing structures of the closure concept described in chapter 4.3 for an HLW repository concept for the Gorleben site, a conceptual design is made which provides only preliminary dimensioning, since final dimensioning of the sealing structures can only be carried out when all the planning work is completed. The preliminary dimensioning is necessary, since an idea of the dimensions of the structures is required during the planning process and also the basic technical feasibility of the sealing system must be verified. The preliminary dimensioning is divided into the following steps:

- Chemical preliminary dimensioning for clarification of whether a relevant permeability increase or a mechanical detriment, such as loss of strength through chemical impacts, can be prevented by the stipulated concept
- Mechanical preliminary dimensioning for clarification of whether relevant crack formation or disaggregation with the result of significant increase of permeability through mechanical impacts can be prevented by the stipulated concept
- Hydraulic preliminary dimensioning for clarification of whether a solution inflow to the radioactive waste can be prevented by the stipulated concept.

After completion of the planning, the final dimensioning of the sealing system is carried out by numerical calculations in the scope of a more detailed integrity proof.

To evaluate the functionality of the sealing system, different scales with regard to geometric resolution are required for numerical modelling. Thus crack evolution can impair the functionality on a small scale, whilst the thermomechanical impacts of the repository are described on the large scale of a far-field model due to the distance, particularly of the shaft, from the emplacement areas. Various models are applied for different scales for handling the impacts, whereby the initial and boundary conditions of the submodels assigned to the substructures are based on results of other models or are derived from them. This method which is termed 'submodelling technique' is consistently applied.

For integrity proof, shaft and drift seals can be handled separately based on their spatial distance, as can the three sealing elements of the shaft, which are decoupled by design in the results of the preliminary dimensioning [Müller-Hoeppe et al. 2012a]. Individual proofs are



therefore made on the submodels, which include the separate sealing elements and the assigned abutments. The initial and loading conditions of the submodels, however, result from the interaction of the individual sealing elements within the total system. Accordingly, the models are closely linked to each other and in some cases a reverse coupling is also made.

A prognosis can be derived for sealing system functionality from the totality of the results generated for the substructures including results from preliminary dimensioning, whereby plausibility evaluations and trend analyses must also be taken into account.

#### **5.7.4 Design of the sealing system for the Gorleben site**

As an example from using the procedure to design the sealing structures, the results from the R&D project VSG [Müller-Hoeppe et al. 2012a], [Müller-Hoeppe et al. 2012b] will be summarized in the following.

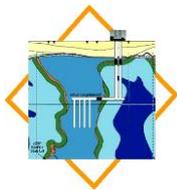
##### **5.7.4.1 Site-specific boundary conditions**

Shafts have a water-tight lining with a foundation located significantly below the salt level for the control of the water bearing overburden formations. This lining remains in the shaft after repository closure. Because of the stability of the salt formations, the shafts in the salt dome are not lined below the shaft lining foundation. In the shaft area, salt geology is characterised by complex folded layers of the Leine formation. The small Gorleben anhydrite layer can have an increased hydraulic conductivity; however, it does not provide a discrete pathway to the salt table. In Gorleben shaft 1 the potash seam lying near the salt table is leached, at Gorleben shaft 2 the potash seam is not exposed.

The drift seals separate the western and eastern emplacement areas in the Staßfurt salt formations from each other and from the infrastructure area which is excavated in the rock of the Leine Series (chapter 4.3, Figure 4.15). The drift seals there are also located in the Staßfurt salt formations. At the locations of the seals and at drifts, where credit of the sealing properties of the crushed salt will be taken in the long-term, all installations such as roadway, tracks and cables will be dismantled.

##### **5.7.4.2 Proof of constructability**

In order to ensure that the sealing structures can be constructed with adequate properties, materials are exclusively selected (e.g., Ca-bentonite type Salzdetfurth, salt concrete typeASSE, magnesium oxychloride concrete A1, compacted crushed salt, cf. chapter 4.3) that are already used on a large scale for comparable purposes and therefore construction experience already exists for them [Breidung 2002], [BfS 2010], [Mauke et al. 2012], [Kamlot et al. 2012], [Knoll 2005], [Hurtado et al. 1997]. Organic substances (e.g., asphalt and bitu-



men) will not be used, since otherwise biological impacts would have to be considered [DGGT 1997] and there are insufficient data available regarding this type of impacts under the boundary conditions of the repository system. Therefore, the long-term behaviour of these substances over the set functional lifetime of 50,000 years is unclear.

#### 5.7.4.3 Preliminary dimensioning

The procedure for carrying out a preliminary dimensioning will be shown in the following example for the sealing system of the Gorleben 1 shaft (chapter 4.3, Figure 4.16).

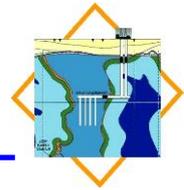
##### Geochemical preliminary dimensioning of the sealing components

The potential impact of the practically unlimited volume of overburden water is decisive for the preliminary geochemical dimensioning of the shaft seal. Two different compositions of overburden water result from the hydrogeology of the surrounding rock formations as well as from the chemical properties of the construction materials of the shaft lining and their degradation. These two water types have been termed the rainwater sequence (originating from surface water and overburden water with negligible salinity) and the cap rock sequence (originating from overburden solutions of a high salinity with the non-negligible content of sulphate and magnesium ions).

Leaching capacities are calculated for the impact of these solution sequences on the respective construction materials (normal concrete, salt concrete, magnesium oxychloride concrete). The composition of the overburden solutions (initial solutions) has a very low or low corrosion potential for normal concrete of the shaft lining, and will be modified by the degradation products.

The two solution sequences mentioned above will then affect the shaft sealing elements and have an average corrosion potential for salt concrete (middle shaft seal) and no relevant corrosion potential to magnesium oxychloride concrete (lowermost shaft seal). The bentonite of the upper shaft seal is involved in the chemical interactions by ion exchange.

Chemical model calculations showed that the sealing system has sufficient resistivity against a corrosion induced solution path to the radioactive waste. For the salt concrete seal the long-term function is ensured by an appropriately dimensioned sacrificial layer and for the magnesium oxychloride concrete seal the corrosion is prevented by a Bischofit reservoir that is located in the gravel backfill between both sealing elements (Figure 4.16) Alteration processes in bentonite will proceed very slowly under thermal and geochemical boundary conditions in the Gorleben 1 shaft and are therefore negligible [Xie & Herbert 2012]



### Mechanical preliminary dimensioning of the shaft seal

The most important impacts that have to be considered for the mechanical preliminary dimensioning are the load of the surrounding rock mass and the hydraulic pressure of the overburden water. Furthermore, mechanical loads as a result of an earthquake are considered, although Gorleben is located in a low earthquake intensity region. The reason is that earthquakes could possibly trigger design-defining deformations (settlements), which could impair the sealing function of the sealing structures.

The mechanical preliminary dimensioning is done in four sequential steps:

- Estimation of the depth of the excavation damaged zone to determine the recutting depth (basis for definition of the dimensions of the sealing elements)
- Determination of the required abutment length
- Estimation of the settling of the filter layer as a result of the superimposed load
- Estimation of the additional settling under earthquake loading.

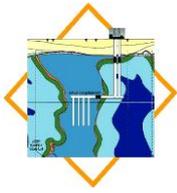
To determine the required abutment lengths for the sealing elements, the pressure of the solution column in the shaft reaching to the surface plus an additional 50 m water column due to a potential rise of the sea level through an anthropogenically induced climate change is assumed as a load. The determination of the abutment lengths are based on the rated values of the shear strength of the contact zone (between abutment and the shaft contour) and the tensile strength of the construction material (salt concrete or magnesium oxychloride concrete).

The layers of sand and gravel introduced for guaranteeing the filter stability of the bentonite sealing element are subject to settling. During the determination of this settling, the superimposed load of the overburden, the weight of the bentonite sealing element and the climate induced rise in sea level are taken as a basis.

For the evaluation of the consequences of an earthquake an early incidence is assumed at which the fixing of the bentonite sealing element and the abutments by convergence is still low. Therefore, it is conservatively assumed that contact between the filter layer and the shaft contour will be completely neutralized by the earthquake acceleration for the duration of the strong earthquake phase.

To assess the settling, the calculated total settling and the potentially resultant loosening of the bentonite can be compared with the acceptable settlements [Wagner 2005].

As a result of the mechanical preliminary dimensioning it can be concluded that no relevant impairment of the sealing function of the sealing system will result from mechanical impacts.



### Hydraulic preliminary dimensioning of the sealing system

The hydraulic pressure of the overburden water represents the primary impact to be controlled by the design of shaft seals. In the scope of the preliminary dimensioning it will first be checked whether the sealing system meets the design objectives under the boundary conditions of the reference scenario with function as designed [Beuth et al. 2012a]. In addition, the flow rate and the flow volume of brines through the bentonite sealing element as well as the time duration for filling of the downstream storage volume are estimated. The objective of the shaft seal design is to guarantee a sufficient time delay of the fluid pressure increase in front of the drift seals in the connecting drifts to the emplacement areas. Crushed salt compaction has to be sufficiently advanced so that the sealing system in combination with the compacted crushed salt is able to prevent a solution inflow to the radioactive waste.

According to the calculated flow volumes passing through the bentonite seal element, the reservoir volume available in the filter layer and the gravel column will be filled up after approximately 1,100 years. Thus, the required timeframe of approx. 1,000 years for crushed salt compaction [Czaikowski & Wieczorek 2012] is available, without taking credit from other sealing elements and storage volume.

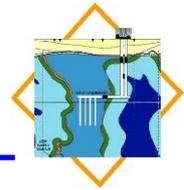
In a further step the consequences of the less probable evolution "early loss of integrity of shaft seal" (with intact drift seals) must be examined, and it must be proven that the sealing system can also prevent solution inflow to the radioactive waste even under these boundary conditions. For this scenario, the most unfavourable case is the loss of integrity directly after construction, long before crushed salt compaction has been completed.

The large void volume of the infrastructure area will result in a significant retardation of fluid rise and fluid pressure build-up so that, even if there is a loss of integrity of a shaft seal (in combination with properly functioning drift seals), compaction of the crushed salt in the emplacement fields will be so advanced [Popp et al. 2012] that it can prevent solution inflow to the radioactive waste.

#### **5.7.4.4 Detailed integrity proof**

Numerical calculations were carried out in the scope of the detailed integrity proof for the geotechnical barriers, to analyse non-linear and time dependent, as well as thermomechanical and hydromechanical coupled, performance of the shaft seal and drift seal [Müller-Hoeppe et al. 2012a]. The calculations are used for various objectives. The following objectives are mentioned here:

- Verification the assumptions of the preliminary dimensioning,



- verification of models: Admissibility of model simplifications, check of model parameters, correct transfer of time dependent initial and boundary conditions of each assigned model and submodel,
- determination of time-dependent initial and boundary conditions for submodels,
- reduction of the number of computational cases or calculation variants,
- comparison of the results of various modelling tools, and
- finally, performance of the respective individual verifications to prove structural integrity

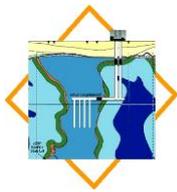
The required individual steps to prove the structural integrity of the geotechnical barriers are listed in the following:

- compilation of possible, relevant impacts and list of the design situations
- derivation of scenarios which may result in an impairment of the seals and determination of the associated ultimate limit states,
- determination of design-relevant combinations of impacts and derivation of computational cases,
- carrying out the calculations to the identified computational cases, and
- evaluation of the results of the computational cases for the calculation model based on design criteria, which are described by the ultimate limit states

The ultimate limit states are described by the criteria for integrity proof of the geotechnical barriers. They include material characteristic values, which are couched in terms of stress and deformations. For the functional proof of the sealing elements, the contact zone (between sealing body and shaft or drift contours) and the excavation damaged zone (EDZ) have to be considered supplementary to the sealing body. For these three elements the following integrity proof criteria must be used:

- For the evaluation of the EDZ in salt formations, the dilatancy criterion and the fluid pressure criterion are used as proof criteria. Both criteria are evaluated, depending on the selected modelling approach, either for the hypothetical pore pressure used for the value comparison or for the effective pore pressure [Müller-Hoeppe et al. 2012b].
- For the evaluation of the contact-zone, the rated values of the adhesive tensile strength and the shear strength are used as proof criteria for the limitation of cracks in the case of cohesive materials. For bentonite, the fluid pressure criterion is used, and the restriction of the local bulking is examined.
- For the barrier structure, in the case of cohesive materials (concrete), the material-specific fracture strength limits are used. In doing so, the design values are selected in such a manner that the limitation of cracks is also kept. For the bentonite seal, the fluid pressure criterion in combination with the criterion for restriction of the local bulking is used.

In the R&D project VSG, the total structure of one of the two shaft seals, the individual sealing elements of one shaft seal and a fictitious drift seal – covering the properties of all drift



seals – are analysed in the course of the detailed integrity proof and by application of the submodelling technique [Müller-Hoeppe et al. 2012b]. For the submodels, selected computational cases are examined. The compilation of the results of all these computational cases for detailed integrity proof taking into account the results of the preliminary dimensioning is done first for the shaft and the fictitious drift seal and finally for the entire sealing system.

### 5.7.5 Evaluation of the sealing components for the Gorleben-specific repository concept

The functionality of the sealing system has been evaluated for the repository concept, using the procedure described in the previous chapters, [Müller-Hoeppe et al. 2012a], [Müller-Hoeppe et al. 2012b]. The results for the detailed integrity proof are summarised taking into account the results of the preliminary dimensioning. A prerequisite for the reliability of the integrity proof is that the assumptions for crushed salt compaction compiled in [Czaikowski & Wieczorek 2012] and [Popp et al. 2012] can be confirmed by the results of further investigations.

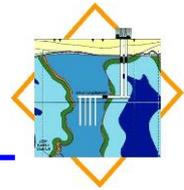
The constructability of the sealing components has been ensured by utilizing existing prototypes and established construction methods.

For the reference scenario, it is assumed that all barriers will function as designed [Beuth et al. 2012a]. For the reference scenario no significant inflow of solutions to the radioactive waste has been stated. This situation exists even if a design basis earthquake were to occur immediately after repository closure.

Also for the less probable evolutions (alternative scenarios) "Early loss of integrity of a drift seal" and "Early loss of integrity of a shaft seal", no significant inflow of solutions to the radioactive waste from above ground, overburden or salt formations is anticipated. However, there are still uncertainties with regard to the modelling and the data base. Thus, the results are not well established and only give a first indication.

In summary, the sealing system meets the requirement to prevent or limit a solution inflow to the radioactive waste. For the reference scenario and the alternative scenario "Early loss of integrity of a drift seal" the standard of reliability of a structural design has already been reached. In contrast, the conclusions for "Early loss of integrity of a shaft seal" are not well established and only give a first indication.

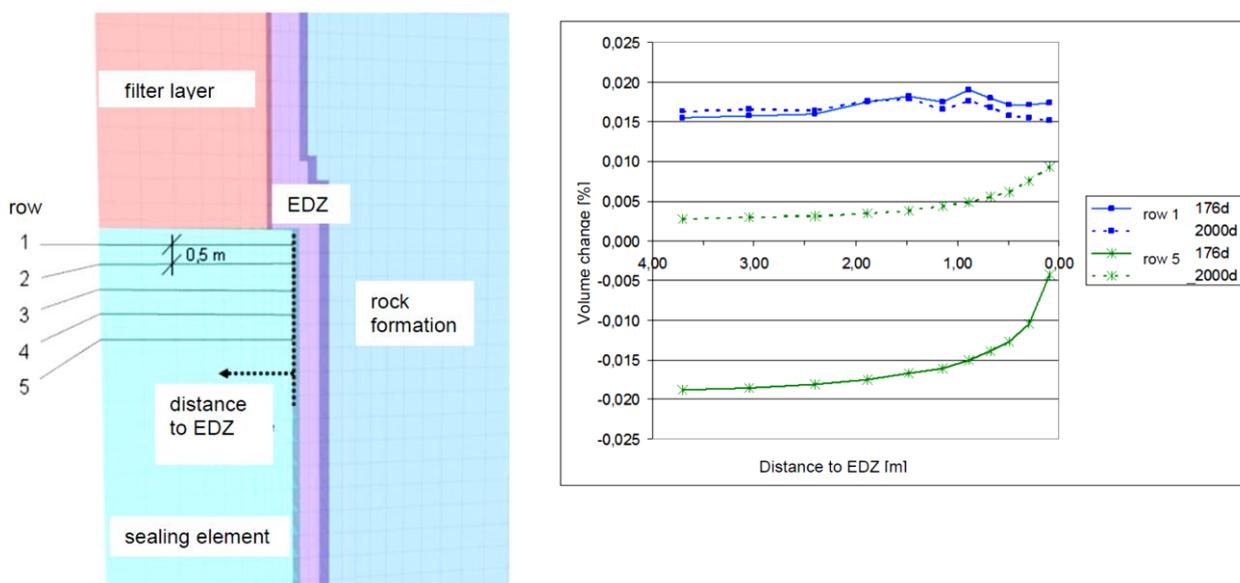
The following results were achieved in detail for the shaft seal and the hypothetical drift seal.



### 5.7.5.1 Shaft seal

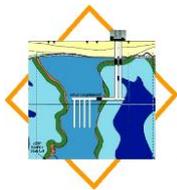
Individual integrity proofs for the various sealing elements were performed for the integrity proof for the shaft seal. The design situations "Reference scenario without earthquake" and "Reference scenario with earthquake" were treated. Based on the thermomechanical model calculations, it can be shown that for the shaft seal consideration of the thermomechanical boundary conditions also covers the influences of diapirism. With regard to the thermomechanical impacts, a bandwidth from the undisturbed geothermal conditions (taking into consideration cooling through the shaft ventilation) up to impacts from the disposal of heat-generating waste was considered. These cover the various repository configurations.

The model calculations show that the bentonite sealing element in combination with filter layer and abutment will fulfill its sealing function with regard to mechanical, hydraulic and thermal impacts in an appropriate manner if the barrier has been properly constructed and fluid pressure has slowly increased during the "Reference scenario without earthquake". The local bulking criterion is complied with but the fluid pressure criterion will be temporarily and locally infringed, (Figure 5.12). Even though only a trend can be derived from the calculation results based on the long computational times for the required hydromechanical coupled calculations, this trend is additionally supported by the test results of the R&D project "shaft seal Salzdetfurth" [Breidung 2002].



**Figure 5.12:** Thermomechanical calculations for the bentonite shaft seal. Volume change on selected sampling points for the dates 176 days after construction (maximum fluid pressure load) and 2,000 days after construction (increasing saturation) [Müller-Hoeppe et al. 2012a]

The chemical impact on the bentonite sealing element results from the potential compositions of the overburden waters, which may also contain cement phases of the shaft liner



(corrosion products). No computational analyses to analyse this impact were carried out but the effect is classed as negligibly low [NEA-IGSC 2012].

In summary, there is a good indication as to the functionality of the bentonite sealing element.

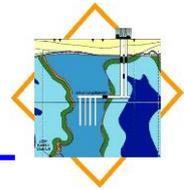
The salt concrete sealing element will fulfill its function with regard to the mechanical, hydraulic and thermal impacts if it has been properly constructed and the fluid pressure increases in the expected manner. For calculations that take into account the thermomechanical impacts of the repository, this statement is unrestricted. For calculations without thermomechanical impacts of the repository, only a trend statement is possible, since substantiated results from hydromechanical coupled calculations are not available (Figure 5.13). However, exceeding the fluid pressure criterion in calculations for a cold shaft is insignificant so that with the reduction of conservatism by hydromechanical coupled calculations the fluid pressure criterion will be met. For the evaluation of the calculation results it must be taken into account that uncertainties still exist with regard to the stress evaluation when approaching the contact zone which may be subjected to possible restrained stresses. Despite this uncertainty, the initial investigation results on a pilot seal structure [BfS 2010] confirm that the restrained stresses in the contact area do not lead to permeabilities that effect the functional requirements.

The chemical impacts on the salt concrete sealing element result from shaft internal solutions and from the overburden waters that pass through the bentonite sealing element and are accordingly modified. Geochemical calculations have shown that the provided sacrificial layer is, in principle, sufficient to reduce the corrosion potential of the inflowing salt solutions. The impact of localized corrosion is classed as low because the corrosion products build a protective layer at the top of the sealing element.

For the mentioned impacts or impact combinations, a good indication as to the functionality of the sealing element made of salt concrete can be made.

For the magnesium oxychloride concrete sealing element, the same conclusions as drawn for the salt concrete sealing element are true with regard to the mechanical, hydraulic and thermal impacts. The solutions passing through the salt concrete sealing element will be saturated with  $MgCl_2$  up to magnesium oxychloride concrete stability when passing a Bischoffit-layer in the gravel column between the two shaft seal elements. Thus, these solutions have no corrosion potential for magnesium oxychloride concrete. The geochemical long-term stability is thus guaranteed.

Within the scope of the preliminary dimensioning, the earthquake-induced settling of the gravel column was conservatively covered, estimated on an analytical basis. Based on the drained concrete abutment below the bentonite sealing element, loosening in the bentonite remains within tolerable limits. Plausibility evaluations were performed for the impacts on the



concrete sealing elements. From low additional accelerations, as they are set in the case of the design basis earthquake, the design is not determined, since the dynamic material strength and stiffness as a rule grow stronger than the stresses. A different evolution will occur if a shear failure occurs which is not blocked geometrically. Shear strengths can drop due to the normal stress reduction caused by an earthquake. This evolution possibly affects the contact zone, since failure exceeding the admissible shear stresses and strains in the contact zone is a potential failure mode. Since the calculated shear stresses are well below the acceptable shear strength of the material, a failure in the contact zone due to an earthquake is improbable. However, this evaluation must still be verified. In the course of the detailed integrity proof, no additional investigations were performed for earthquakes.

In summary, the results from the preliminary dimensioning and the integrity proof for the Gorleben 1 shaft seal show, that this barrier system will fulfil its stipulated functions during the examined future evolutions of the repository system. However, the integrity proofs are not yet definitive. Of special relevance is that the shaft seal has three independent sealing elements that are decoupled from each other, each with its own abutment, so that it is improbable that the entire shaft seal would fail. The less probable alternative scenario "early loss of integrity of a shaft seal" (meaning a synchronous failure of all shaft seal components) is thus very conservative.

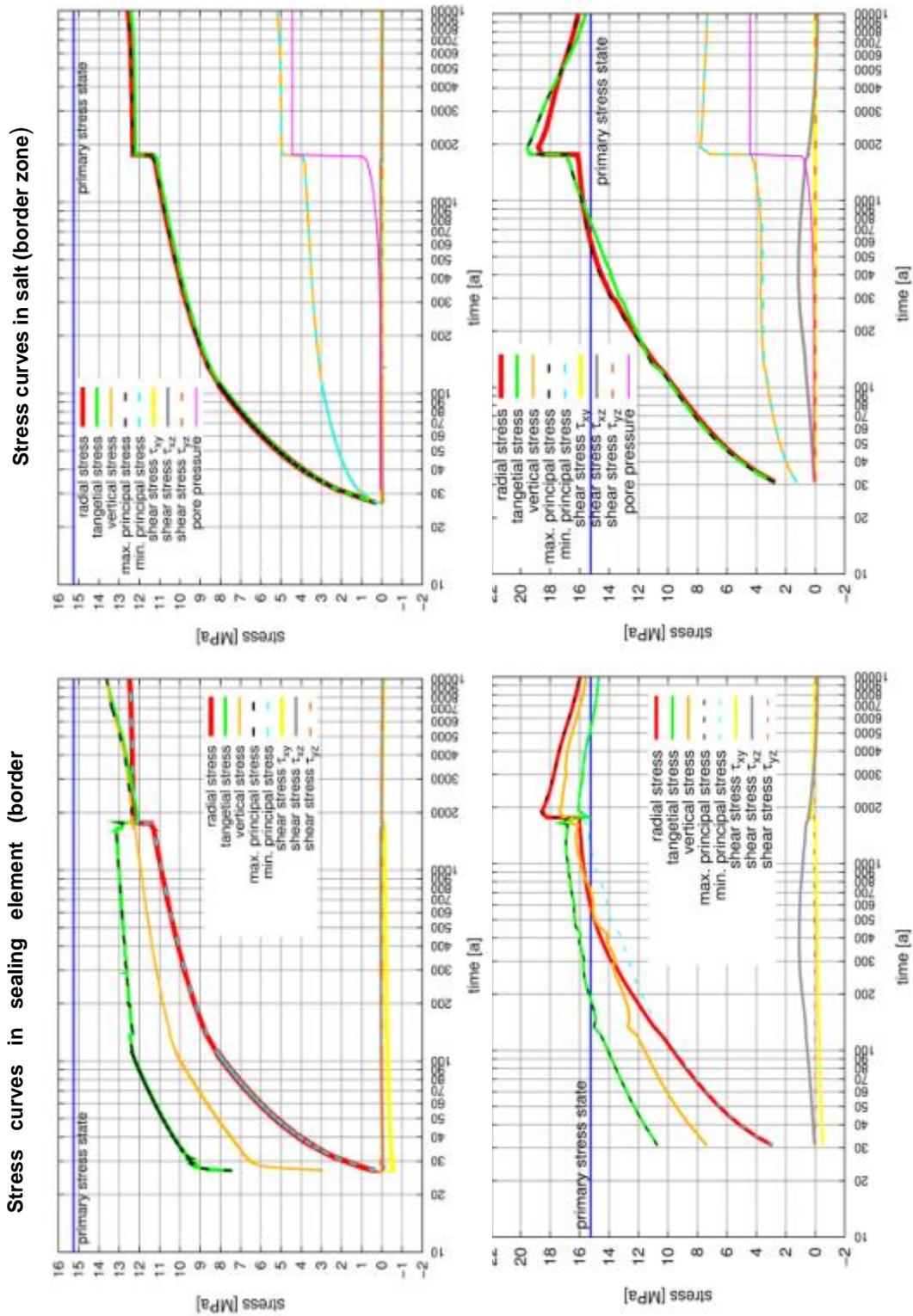
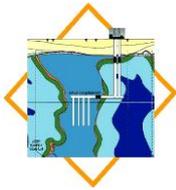
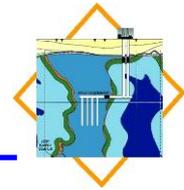


Figure 5.13: Stress curves on two selected sampling points in the contour area of the salt concrete sealing element (top: without thermomechanical impacts of the repository, bottom: with thermomechanical impacts of the repository) [Müller-Hoeppe et al. 2012a]



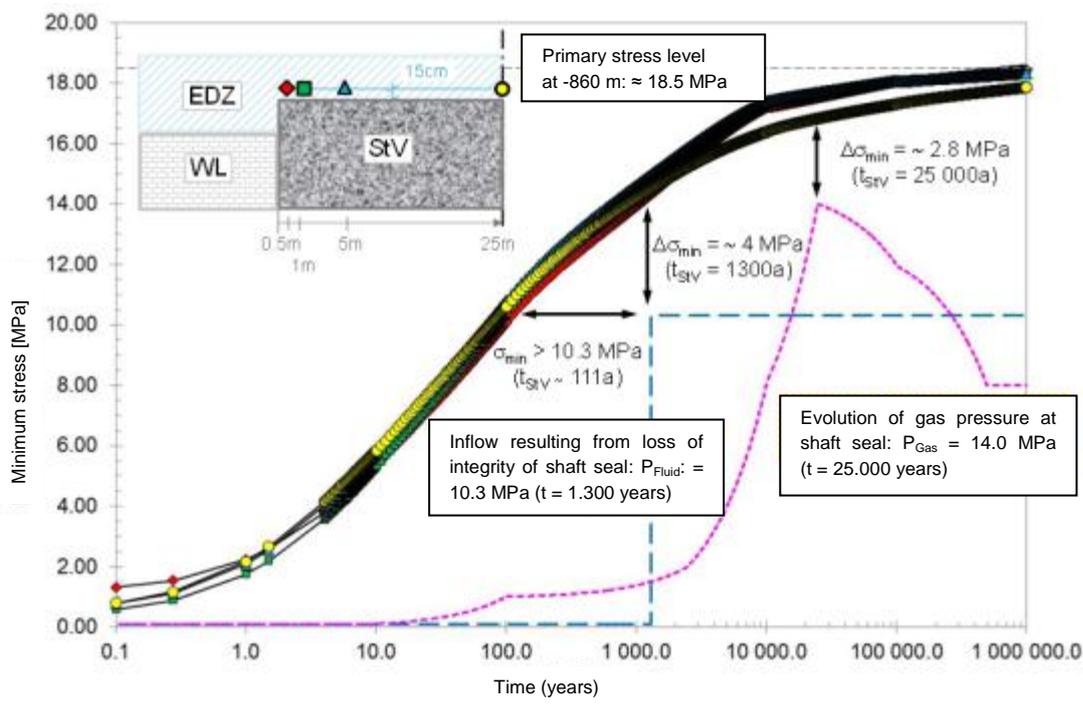
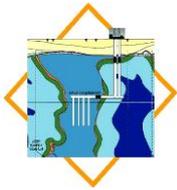
### 5.7.5.2 Drift seals

Corresponding to the closure concept (chapter 4.3), there were four drift seals for the repository concept to be investigated and evaluated. In order to reduce the number of required integrity proof calculations, a fictitious drift seal is considered, in which the valid boundary conditions (geology, geometry) and impacts (temperature, fluid pressure) for the four drift seals are unfavourably combined in a comprehensive manner (cf. chapter 4.3, Figure 4.15). Since calibrations in the shaft confirm that variations of the salt creep did not have to be considered here, corresponding in-situ measurements in the area of the drift seals were not done. Therefore here also the consequences of a slower salt creep were to be considered on the function of the barriers.

Important design situations are the "Reference scenario without earthquake" and "loss of integrity of the shaft seal". The "Reference scenario with earthquake" was not evaluated for the drift seal, since the plausibility evaluations for this evolution done for the shaft seal were also valid for the drift seal.

The model calculations show a slight increase of porosity in the EDZ for an evolution with gas pressure on both sides ("Reference scenario without earthquake") and slower salt creep, which does not however increase the integral hydraulic permeability of the barriers in an impermissible manner. In the calculation case for solution pressure from one side only, to which the design situation "early loss of the integrity of a shaft seal" is assigned, at the time of the solution pressure increase in the EDZ there was already a rock pressure build up so high that the fluid pressure criterion for the assumed pore pressure is met (Figure 5.14). This conclusion can be transferred to the case of the impact combination of solution from one side only and gas pressure from one side only, which is assigned to the same design situation (chapter 5.5.4). The designs for the calculation case of gas pressure on both sides are covering this, so that – in accordance with the requirements - a sufficient hydraulic resistance is always present.

In summary, the model calculations show that the drift seals in the scheduled arrangement for the anticipated repository system evolution meet the required sealing function. Especially important for the functionality of the drift seals are the retardation function of the sealing elements of the shaft seal system and the fluid pressure reducing large storage volume of the infrastructure areas, Therefore, even in the case of a malfunction of a shaft seal, the slow increase of fluid pressure will enable a fast convergence-induced pressure build up in the EDZ and the contact zones of the drift seals. With regard to the geochemical long-term stability of the drift seals, the saturation of the solutions with  $MgCl_2$  is of highest relevance. Therefore, a Bischofit depot is located in the infrastructure area.



**Figure 5.14:** Comparison of the stress conditions in the EDZ and the solution and gas pressure from one side only for the fictitious drift seal [Müller-Hoeppe et al. 2012a]

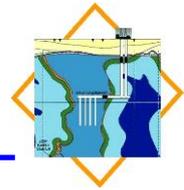
### 5.7.5.3 Complete closure system

For the sealing system described in chapter 4.3, which consists of shaft seals, drift seals and further functional elements, the following separate integrity proofs for mechanical, thermal, hydraulic and chemical impacts have been performed resp. considered in the scope of the preliminary dimensioning and detailed integrity proof:

- Load bearing capacity
- Limitation of crack formation
- Limitation of deformations
- Durability (long-term stability)

Based on the current planning status the functional proof of the filter stability could not yet be performed, but only described. The principle feasibility is given, however, since on the one hand, such a functional proof is the state of the art [Schneider 2004] and, on the other hand, there is still flexibility for the design of the filter layers.

The individual integrity proofs were handled at different levels of detail. Calculations and plausibility evaluations were performed and trends were derived taking credit from test results. For the evaluation of the evolution "Reference scenario with earthquakes" empirical experience was used ". Since the individual integrity proofs for different types of the sealing



structures had positive results, a positive prognosis in regards to the functionality of the planned closure system can also be given.

In the integrity proofs all the impacts listed in [Müller-Hoeppe et al. 2012a] for the preliminary design or the detailed integrity proof – with the exception of the chemical impacts triggered by temperature changes – were directly or indirectly considered.

With regard to the increased leaching potential of warm brines there is still need for R&D. With regard to the  $MgCl_2$  saturation this issue empirically reflected, by dimensioning the Bischoffit depot in the infrastructure area with surplus [Xie & Herbert 2012]. The increased leaching potential of NaCl was not quantified. However, it is plausible, that it is negligible, since  $MgCl_2$  saturation and temperature increase result and under isothermal conditions with  $MgCl_2$  saturation even NaCl is precipitated. Detailed examinations for quantifications of this issue have not been performed.

Displacements of the sealing elements generated by salt uplift (diapirisms) are covered by the barrier design.

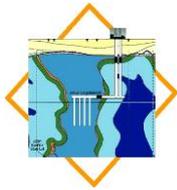
Since exploration data from the barrier locations can be used, and the technical data basis used for the components is validated to the most possible extent, the separate integrity proofs have a high prognosis reliability.

The boundary conditions defined for the integrity proofs of the barriers cover a broad spectrum, e.g. different thermal impacts, so that different arrangements of emplacement fields for heat-generating, radioactive waste were covered – irrespective of later exploration results. In all possible constellations the positive prognosis with regards to the functionality of the sealing system remains valid.

## 5.8 Proof of Subcriticality

The radioactive waste in a repository contains fissile material and it must be confirmed that this material cannot assemble into a critical arrangement. Criticality is only possible if a sufficient amount of fissile material in a sufficient amount of water – or another neutron moderator – is available. Critical arrangements must be excluded for all potential evolutions of the repository system that are classified as either probable or less probable [BMU 2010a].

If subcriticality has to be demonstrated, the boundary conditions and process evolutions of the considered scenarios must be taken into account. This comprises the condition of the spent fuel itself, i.e. its constitution, enrichment, burn-up, and geometric arrangement, the amount of moderator which can come into contact with the fissile material, the potential time of occurrence of criticality, and consideration of any uncertainties in parameter values. As an alternative, the assessment might also be based on overall calculational cases and unfa-



unfavourable model assumptions, such as neglecting the reduced content of fissile Uranium by burn-up processes in the reactor.

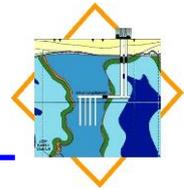
As an indicator for the demonstration of subcriticality, the multiplication factor  $k_{eff}$  is used. It is defined by the ratio of neutrons produced due to the fission process to the neutrons initially present. Criticality can be excluded, if this parameter has a value

$$k_{eff} < 0.95$$

In R&D project VSG, criticality was analysed for several scenarios and for waste containers of the type POLLUX<sup>®</sup>, BSK canister, and transport and storage casks [Kilger et al. 2012]. Fuel assemblies and fissile material contents representative of German NPPs were taken as the basis. The fuel was conservatively assumed to be unirradiated so that no reactivity decrease by burn-up was considered. The calculations performed showed that the occurrence of a self-sustaining nuclear chain reaction in a repository in a chlorine-based salt host rock based on the inventory of a single BSK or POLLUX<sup>®</sup> cask with spent fuel from light water reactors, and in case of the direct disposal of CASTOR<sup>®</sup> transport and storage casks, can all be excluded.

In the calculation results a significant decrease in reactivity occurred as compared to pure water and this is essentially caused by Cl-35, which is present at a high concentration in saturated brine in the salt dome. This also holds for cases such as flooding of the cask interior with water or brine, loss of implemented neutron absorbers (boron) or the dissolution of the basket structure within the cask. Considering the saturation concentration of Cl-35 in brine, all the systems and cases studied remained subcritical. Thus, the demonstration concept for subcriticality can be based on the neutron absorbing properties of Cl-35. In a dry system, without ingress of water into the mine, criticality may generally be excluded.

For special nuclear fuel such as highly enriched fuel from research reactors, subcriticality has not yet been demonstrated for the planned waste containers. In such cases if subcriticality cannot be proved, the spent fuel would need to be re-packed in different containers.



## 5.9 Calculation of radiological safety indicators

The specifications for radiological safety indicators are given in the Safety Requirements (chapter 2). Two indicators are possible:

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

The calculation of effective dose has been applied for many years and the application scheme is straightforward (chapter 5.9.2). Criteria for the additional effective dose are specified in the Safety Requirements (chapter 2).

To implement the specifications in the Safety Requirements for a radiological indicator the RGI (Radiologischer Geringfügigkeits-Index (index of marginal radiological impact)) was developed in the R&D project ISIBEL [Buhmann et al. 2010b]. The calculation of the RGI is based on a stylised calculational scheme. It is assumed that the total radionuclide flux released from the CRZ is diluted in the annual water consumption of one adult individual. In [Buhmann et al. 2010b] this value ( $W$ ) is set to 500 m<sup>3</sup>/year. The calculation does not consider how the radionuclides are transported from the boundary of the CRZ to the water body used by the individual considered. To determine the radiological consequences of the radionuclide concentration in the water body, a biosphere model equivalent to the calculation of the effective dose in the biosphere is applied. The calculated exposures are normalized in order to highlight the fact that this calculation is an indicator for safety and not a prognosis of future exposures. In [Buhmann et al. 2010b] the criterion  $K_{RGI}$  is set to 0.1 mSv/year.

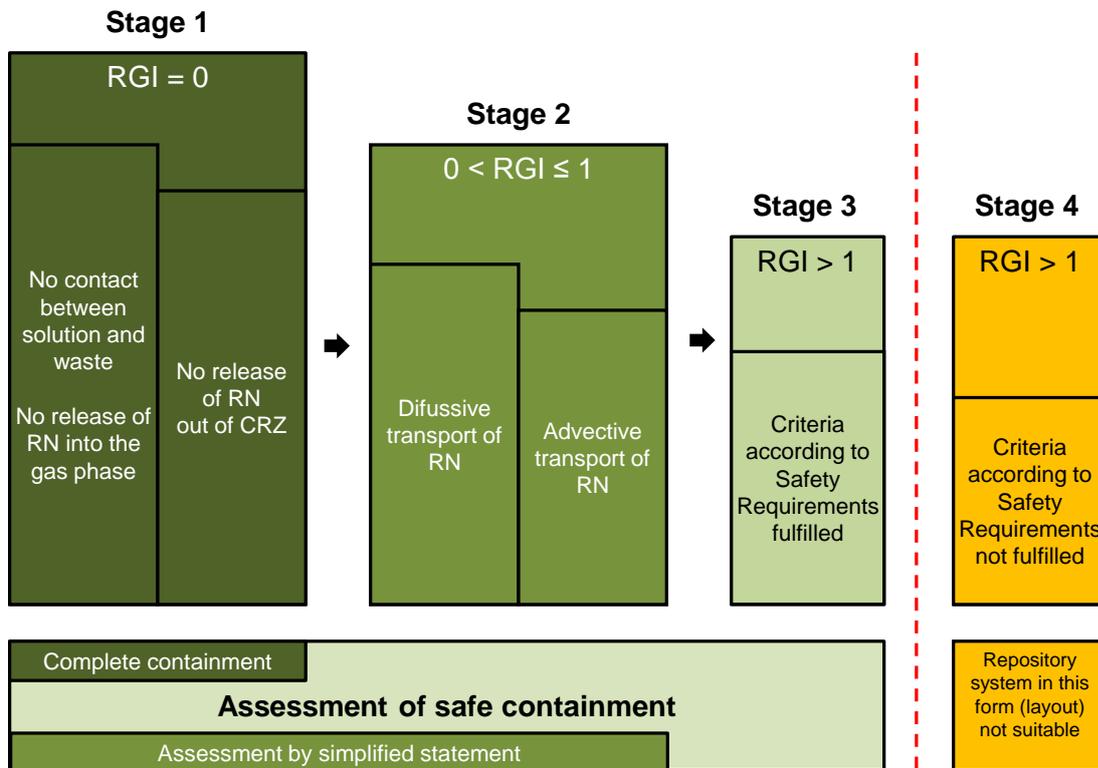
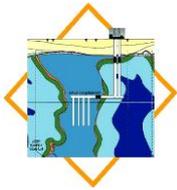
The calculation scheme is

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

In the R&D project ISIBEL, the application was set up for the assessment of a radionuclide release in the liquid phase. The concept of the RGI can easily be expanded to assess both the gaseous and liquid pathways of radionuclide migration. At the moment there are no specifications in the Safety Requirements regarding consideration of the gaseous phase and no generally accepted calculational scheme exists.

### 5.9.1 Containment of radionuclides in the CRZ

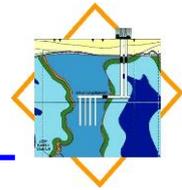
The assessment of the containment is based on the simplified radiological statement defined in the Safety Requirements (chapter 2). The qualitative and quantitative evaluation of fluid and radionuclide transport processes yields a staged assessment (Figure 5.15).



**Figure 5.15: Staged approach for the long-term safety assessment (after [Mönig et al. 2012])**

Complete containment is regarded as the most stringent form of containment and this is achieved if there is no contact between intruding solution and the waste, when no radionuclides are released into the gaseous phase, or when no radionuclides are released from the CRZ (stage 1 in Figure 5.15). If radionuclides are released from the CRZ, safe containment has to be demonstrated. For this purpose the RGI is applied.

Based on the calculational scheme for the RGI developed in the R&D project ISIBEL, it is assumed that the total radionuclide flux released from the CRZ is diluted in the water consumed annually by a reference group of 10 individuals. Analogous to the calculational scheme developed in ISIBEL the calculation does not consider how the radionuclides are transported from the boundary of the CRZ to the water body used by the considered individuals. To determine the radiological consequences of the radionuclide concentration in the water body for the reference group, a biosphere model is applied. In the biosphere model the same dose conversion factors have to be used as are applied for the calculation of the effective dose rate in the biosphere. The requirement for a safe containment is considered to be fulfilled if the exposure caused by the radionuclides released from the CRZ does not result in any significant increase in the consequences that ensue from natural conditions. According to the Safety Requirements, the criterion for an additional effective dose is  $10^{-4}$  person-Sievert per year for probable and  $10^{-3}$  person-Sievert per year for less probable evolutions. The calculation of the RGI for one individual of the reference group is carried out by multiply-



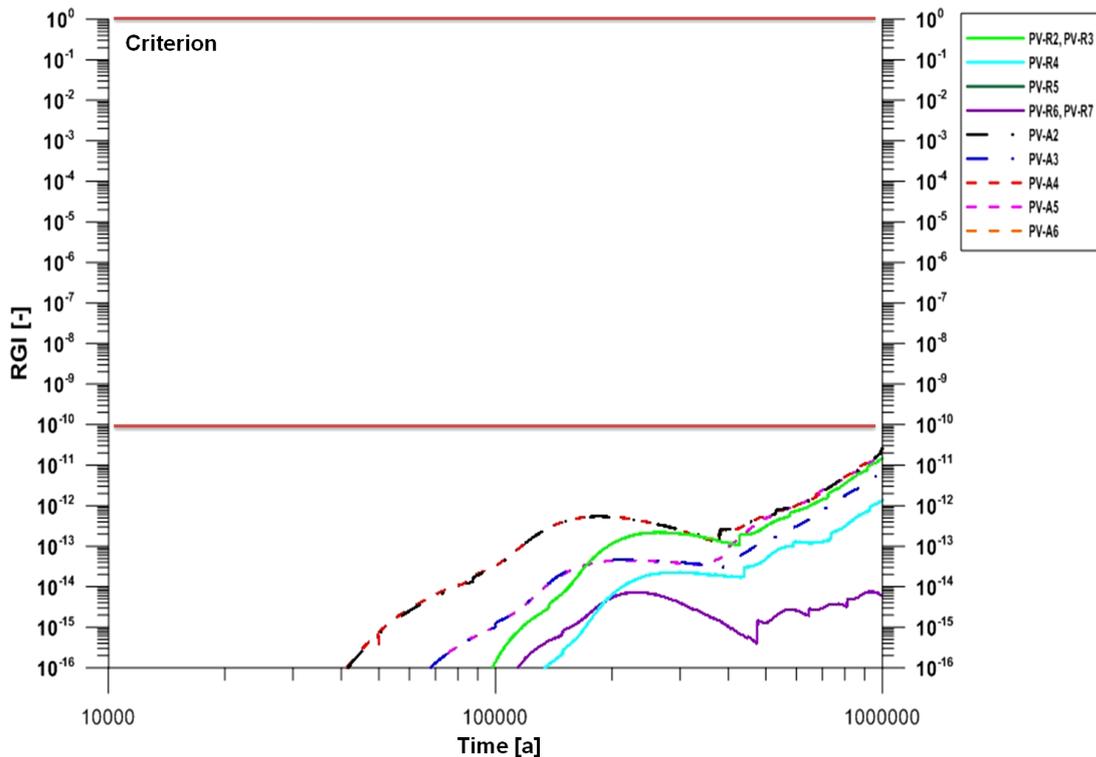
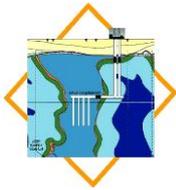
ing the radionuclide flux out of the CRZ of every radionuclide  $S_i$  [Bq/a] with its dose conversion factor  $D_{CFi}$  [(Sv/a)/(Bq/m<sup>3</sup>)] and dividing it by the annual consumption of water  $W$  [500 m<sup>3</sup>/a] and the criterion  $K_{RGI}$  for one individual [0.01 mSv/a for probable and 0.1 mSv/a for less probable evolutions].

The calculation of the RGI results in an index which indicates whether the released radionuclides cause any significant harm for human health. If the RGI is below 1, a safe containment of the radionuclides within the CRZ is demonstrated (stage 2). If the RGI is above 1, the radionuclide release from CRZ is not insignificant (stage 3). This does not mean that the repository system is not safe, but further investigations, especially the calculation of the effective dose in the biosphere (chapter 5.9.2), are required in order to identify whether the consequences of the analysed scenario can be considered to meet the criteria of the Safety Requirements. If not (stage 4) the defined repository system is not suitable.

In the R&D project VSG the reference scenario (chapter 5.3.2) was implemented in a performance assessment model [Larue et al. 2013]. Important parameter uncertainties (particularly regarding compaction of crushed salt and container corrosion rates) were handled by parameter variations, probabilistic calculations were not carried out. To obtain information on the robustness of the system “what-if” cases were also included in the performance assessment.

The calculations show that no solution from outside the CRZ reaches the disposal areas. Only the infrastructure area is filled with solution after the designed functional time of the shaft seal (50,000 years). When hydraulic pressure at the drift seals is increasing, the permeability of the compacted crushed salt is already lower than the permeability of the drift seals. The calculations yield an RGI of 0 (complete containment for a boundary of the CRZ at the drift seals, see Figure 5.16) for radionuclide release in the liquid phase.

Parameter variations for the compaction process illustrate that with a slower compaction rate and a higher final porosity of the compacted crushed salt, solutions reach the waste containers and a mobilisation of radionuclides occurs. Transport of radionuclides towards the drift seals is dominated by diffusion and is thus very slow. Calculated activity flows through the drift seals during the demonstration period are smaller than 10<sup>-5</sup> Bq/year (1 radioactive decay per 100,000 years), the corresponding RGI is smaller than 10<sup>-10</sup> (Figure 5.16).



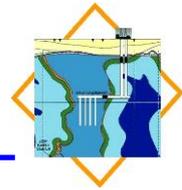
**Figure 5.16:** Calculated RGI at the eastern drift seal for different scenarios (PV = parameter variation, R = reference scenario, A = alternative scenario, after [Larue et al. 2013])

Two alternative scenarios are analysed by calculations in [Larue et al. 2013]: early loss of integrity of the shaft seals and early loss of integrity of the drift seals. Both calculations are supported by parameter variations. As for the reference scenario none of these calculations result in an RGI higher than  $10^{-10}$  (Figure 5.16).

The results from the R&D project VSG project are supported by calculations carried out for a generic repository in the R&D project ISIBEL [Buhmann et al. 2010a]. For a radionuclide release in the liquid phase, both projects illustrate that transport of radionuclides by advection within the CRZ does not play a role for probable and less probable evolutions of the repository system, for some alternative scenarios considering a slower compaction process a radionuclide release by diffusion occurs.

For the calculations with diffusional transport, the diffusion process determines the calculated consequences. It is expected that the applied parameters overestimate the diffusion transport in the calculations. To quantify diffusional transport, more information on the diffusion process in compacted crushed salt is required

In summary it can be stated that the consequence analysis for the reference scenario and the investigated alternative scenarios regarding a radionuclide release in the liquid phase are based on very detailed and profound process knowledge. A regulation exists on how to as-



assess the consequences of a radionuclide release in the liquid phase. Calculations show RGI values several orders of magnitude below the criteria in the Safety Requirements.

The picture is quite different when analysing the radionuclide release in the gaseous phase. As previously stated there are no specifications in the Safety Requirements regarding consideration of the gaseous phase and no generally accepted calculational scheme. Due to the high degree of uncertainty, at the moment it is only possible to assess radionuclide release in the gaseous phase with very conservative assumptions (if known) and parameter variations. [Larue et al. 2013] carried out calculations for the gaseous phase that reveal the following important influencing factors,

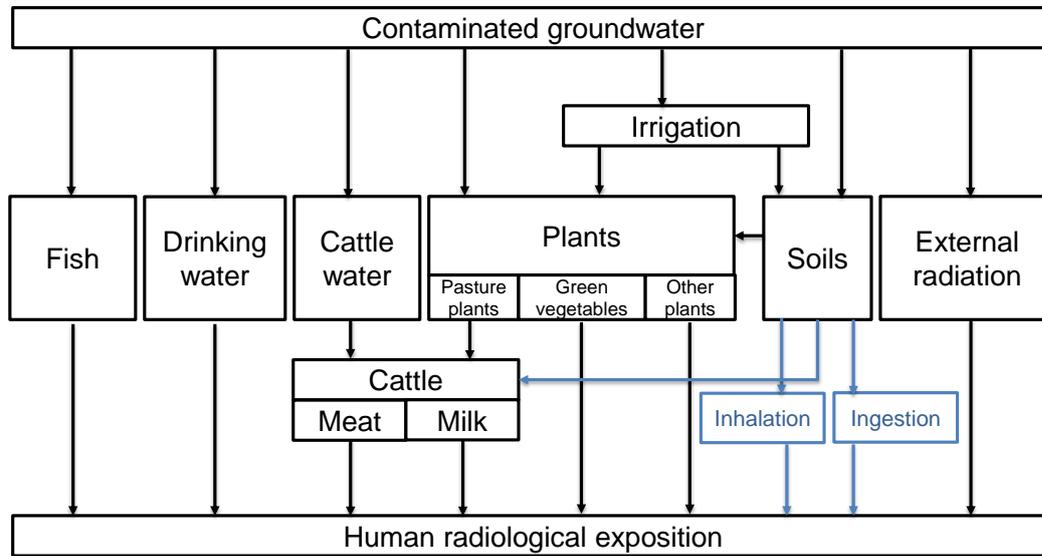
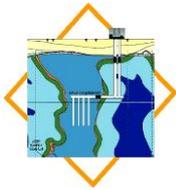
- the amount and location of containers with undetected failures,
- the layout of the mine,
- the gas production rate,
- the compaction process of backfill material and
- the anticipated functional lifetime of the containers.

Further R&D projects are necessary to be able to assess the complex behaviour of these processes in safety analyses.

### 5.9.2 Radiological consequences in the biosphere

Calculation of the radiological consequences in the biosphere comes into action if the simplified approach results in an  $RGI > 1$  (stage 3). The application of a biosphere model is mandatory under the regulations published in [AVV 2012]. This regulation defines the approach to calculate effective doses from radionuclide concentrations in aquifers by the use of dose conversions factors (DCF). According to [AVV 2012] the following exposure pathways have to be considered for contaminated groundwater:

- uptake of drinking water,
- ingestion of fresh water fish from ponds,
- ingestion of plants irrigated with contaminated water,
- ingestion of milk and meat from cattle whose feed has been irrigated with contaminated water, and
- external radiation by dwelling on contaminated riparian sediments.



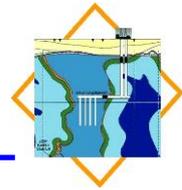
**Figure 5.17:** Exposure pathways to be modeled according to [AVV 2012]. Blue arrows and boxes represent exposure pathways added according to [Pröhl & Gering 2002]

In the implementation of the AVV by [Pröhl & Gering 2002] further exposure pathways were added:

- Unintended ingestion of soils,
- Inhalation of re-suspended contaminated soil particles,
- Uptake of contaminated soil by cattle, and
- External radiation by dwelling on irrigated areas and in buildings erected with contaminated materials.

Figure 5.17 shows the exposure pathways applied for the calculation of corresponding DCF according to [Pröhl & Gering 2002].

In R&D project VSG all calculated RGI values for the liquid phase are lower than 1 (Figure 5.16) and thus no calculations for the consequences in the biosphere after transport through the geological barrier outside the CRZ were carried out. The calculations carried out for a generic repository in the R&D project ISIBEL [Buhmann et al. 2010b] show that dilution effects in the geosphere yield exposures several orders of magnitude lower than the exposures calculated at the boundary of the CRZ. For the gaseous phase no calculations for such a transport from the CRZ to the biosphere were carried out in the VSG project. In the R&D project ISIBEL the transport in the gaseous phase has not been considered at all.



## 5.10 Human Intrusion

According to the Safety Requirements future human activities must be considered in relation to the optimisation of the final repository. Optimisation regarding human activities is of second priority compared to other optimisation targets:

- Radiation protection for the operating phase
- Long-term safety
- Operational safety of the final repository
- Reliability and quality of long-term waste containment
- Safety management
- Technical and financial feasibility

*As future human activities cannot be forecasted, a variety of reference scenarios for unintentional human penetration of the final repository, based on common human activities at the present time, shall be analyzed. Within the context of such optimization, the aim shall also be to reduce the probability of occurrence and its radiological effects on the general public.*

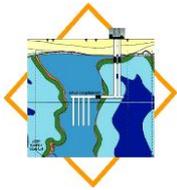
Within the framework of the R&D project VSG the relevance of future human activities has been discussed in detail [Beuth et al. 2012b]. This study deals with those human activities after closure of the repository which directly damage the CRZ or the technical barriers. Only unintended actions have been considered, that is those carried out without knowledge of the presence of a repository and its hazards.

To deal with future human activities the use of stylised scenarios is an appropriate approach (in the Safety Requirements these are mentioned as “reference scenarios”). From the investigation of these scenarios, optimisation measures can be derived. These measures may in general result, for example, in signals to the acting persons in future that there is a special situation in the deep underground or in a reduction of the consequences if the intrusion in the CRZ will not be recognized by the acting persons.

A mathematical probability for future unintended human intrusion into the CRZ cannot be given and so the probability for such an intrusion should be discussed by structured argument. A starting point for appropriate optimisation measures is then to impede the unintended intrusion. It should also be mentioned that the depth of the mine itself constrains unintended intrusion.

In [Beuth et al. 2012b] three stylised scenarios have been considered:

- drilling of an exploration borehole that crosses the CRZ of a repository,
- construction of a storage cavern that completely involves the CRZ of a repository, and
- construction of a mine for the production of salt which touches the CRZ; a preceding exploration borehole not having detected the repository.



As mentioned above, a variety of optimisation measures can be taken to minimise the consequences of future human intrusion. In [Beuth et al. 2012b] a procedure is proposed to develop such optimisation measures:

- **Compilation** of potential optimisation measures against human intrusion.
- **Identification** of potential optimisation measures against human intrusion for the scenarios to be considered.
- **Assessment** of potential optimisation measures against human intrusion for the scenarios to be considered, taking into account the primary optimisation targets of the Safety Requirements.

In [Beuth et al. 2012b] a listing of potential optimisation measures against future human intrusion is given.

After application of the method to the three stylised scenarios given above, the most promising optimisation measures can be identified and assessed. According to [Beuth et al. 2012b] these are:

- Dyeing of backfill or adding of coloured materials to the backfill, and
- Placement of gravel in the openings on the exploration level (requiring an increase in the distance between exploration and emplacement level).

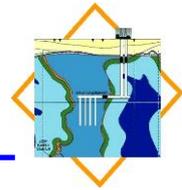
All other optimisation measures turned out to be in conflict with the primary optimisation targets or the ratio of effort to benefit is so poor that these measures need not be taken into further account.

The two measures mentioned above are designed to indicate to future acting people, that there has been previous human activity at the site. It cannot be foreseen what conclusions future populations would draw from such signs.

## 5.11 Operational Safety

### 5.11.1 Description of methodology

It is generally assumed that, by applying suitable technical and organisational measures, compliance with the safety requirements of the mining [ABBergV 2005], [ABVO 1995] and radiation protection regulations [StrlSchV 2012], including criticality safety, can be guaranteed in the case of undisturbed operation and during operational failures and incidents. The evaluation of operational safety thus concentrates on the identification and analysis of potential weak points and significant incidents. The assessment of radiological operational safety was carried out for the delivery and relocation of containers above ground and for the shaft transport and for the radiologically controlled area of the repository. In the course of the R&D



project ISIBEL [Filbert & Pöhler 2008] an assessment of conventional operational safety taking into account the relevant mining regulations was only carried out for the shaft transport process and the radiologically controlled area of the repository.

### 5.11.2 Radiological operational safety

The investigation and assessment within the scope of "Radiological Operational Safety" is limited to the requirements contained in the relevant regulations and provisions for nuclear power plants as outlined in chapter 2.

The BMU (Federal Ministry for the Environment, Nature Conservation and Nuclear Safety) states in the Safety Requirements Governing the Final Disposal of Heat-Generating Radioactive Waste [BMU 2010a]:

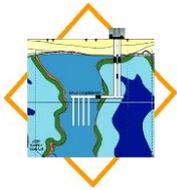
*A comprehensive safety case shall be documented for all operating states of the final repository, including the surface facilities. In particular, facility-specific safety analyses shall be conducted for emplacement operation and decommissioning, with due regard for defined design basis accidents, which should verify the protection of operating personnel, the general population and the environment as required by the Radiological Protection Ordinance. This shall include an analyses and representation of the robustness of the final repository system. Furthermore, the respective probabilities of impacts, failures or deviations from the anticipated case (reference case) of safety-related systems, sub-systems or individual components should be calculated or assessed as far as possible, and their impacts on the corresponding safety function analysed. The relevance of such analysed failures to operational safety must be investigated using probabilistic methods.*

Additionally the [BMU 2010a] states:

*For the safety of the final repository in the operating phase including decommissioning, the reliability and robustness of safety functions within the final repository must be proven in accordance with the specifications of the nuclear legislation for comparable functions in other nuclear facilities. For the operating phase, moreover, a four-level safety concept should be planned analogous to that for nuclear power plants. A "defence in depth" concept should be implemented by allocating these four levels to plant statuses and by specifying the protection measures to be taken or provided for such plant statuses.*

*The following four safety levels should be taken into account:*

<i>Normal operation</i>	<i>Measures prevent the occurrence of operational failures</i>
<i>Anomalous operation</i>	<i>Measures prevent the occurrence of design basis accidents</i>
<i>Design basis accidents</i>	<i>Measure control design basis accidents</i>
<i>Beyond design basis accidents/incidents</i>	<i>Measures reduce probability or limit environmental impacts</i>



The fourth safety level is not detailed as in [BMU 2012].

*"The safety concept should outline and justify the potential operational failures and accidents that could occur in the final repository. Decisions regarding which events are to be classified as design basis accidents as defined in §49 of the Radiological Protection Ordinance [StrlSchV 2012] should be based in particular on the results of the safety analyses and the effects in the vicinity of the repository. This should include an account of which accidents the repository is designed to withstand. Allowance should be made for human error when analysing potential accidents.*

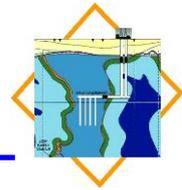
*Events which cannot be classified as design basis accidents due to their low frequency of occurrence should be evaluated and, where necessary, measures proposed to reduce their likelihood of occurrence and impacts."*

In this context, it should be mentioned that the existing nuclear regulations pertaining to the safety of nuclear power plants are updated in accordance with the state of the art in science and technology by the BMU announcement "Sicherheitsanforderungen an Kernkraftwerke" (Safety requirements for nuclear power plants) [BMU 1977], [BMU 2012].

The BMU announcement stipulates a "defence-in-depth" concept. This concept is a combination of different technical systems and administrative measures to prevent and/or control abnormal operation and incidents (levels of defence: 1 – normal operation, 2 – abnormal operation, and 3 – incidents), measures to mitigate the consequences of accidents (level of defence 4a – very rare events, 4b – events with multiple failures of safety installations, 4c – accidents with severe fuel element damages). In this report, only level 3 was considered.

Within the scope of updating the nuclear regulatory guidelines, GRS compiled a set of safety requirements for nuclear power plants and defined the radiological safety goals for level 3 [GRS 2005a]. According to this, the pertinent limit values are to be taken as maximum values for the radiation exposure of personnel when it comes to measures to control events, activities to mitigate the effects of such events, or activities to eliminate potential consequences of such events. For the design of the repository in terms of protecting the public against release-related radiation exposure, the incident dose limits imposed by the *German Radiation Protection Ordinance [StrlSchV 2012]* are to be taken as a maximum.

In addition to this, [GRS 2002a] also identifies 'events to be considered'. According to this 'a comprehensive range of events, the occurrence of which is unlikely during the operational phase of the repository, but which, nevertheless, have to be assumed as possible (level 3)', must, among others, be taken as a basis for the measures and the facilities to be implemented. The completeness and the comprehensive character of the events to be assessed must be ensured by taking into account the conditions specific to the site. When determining design-basis incidents (level 3), not all of the theoretically imaginable incidents are taken into



consideration. In fact, only those causes of incidents and resulting evolutions which, due to considerations on the probability of their occurrence, appear to be potential causes of damage are taken into account,

Due to their low risk of occurrence, the following events are not regarded as design-basis incidents (level 4). Measures against such events only serve to minimise such risks:

- events as a consequence of an aircraft crash
- events as a consequence of external impacts of hazardous substances, and
- events as a consequence of external shock waves resulting from chemical reactions.

The adequacy of the measures to control such events is verified by means of a safety analysis. In this context, a distinction is made between deterministic and probabilistic safety analyses.

It seems reasonable to take these basic principles, which were developed for nuclear power plants, and – taking into account the characteristics of a repository – to adopt them to the safety documentation verifying the adequacy of the preventive measures which are required for the construction and operation of a repository. With regard to accident prevention in the operational phase of a repository, the radiological protection goal "safe confinement of radioactive substances despite internally or externally initiated events" can be deduced irrespective of the disposal concept or the host rock.

In the following subchapters the methodology of deterministic safety analysis and probabilistic safety analysis are described.

#### **5.11.2.1 Deterministic safety analysis**

The methodology of a deterministic safety analysis consists of the steps described below:

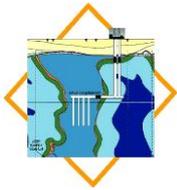
##### Demonstration of completeness

The demonstration of completeness serves to compile a complete list of all possible incidents and accidents. It essentially consists of a (brief) description of the emplacement process specifying the goods to be emplaced, a compilation of activity inventories, a list of possible scenarios concerning the release of radioactive substances taking into account existing barriers, a determination of the radiological principles and a preliminary analysis.

##### Compilation of significant events

The release of radioactive substances can be initiated by internal and external events. Internally initiated events are:

- criticality
- drop/crash of waste package



- mechanical damage due to crashing of heavy loads or due to collision of vehicles
- crash of hoisting cage
- cable slippage
- overrunning of hoisting cage
- impairment of heat dissipation
- failure of systems relevant to safety
- fire
- explosion
- rock-mechanical incidents
- intrusion of mine waters
- gas formation
- gas leakage
- operating error (human action)
- third-party interference

In addition to this, the following externally initiated events have to be considered:

- aircraft crash
- impacts of hazardous substances
- shock waves from chemical reactions
- earthquakes
- lightning, floods, storms, ice and snow
- external fires
- other site-specific impacts

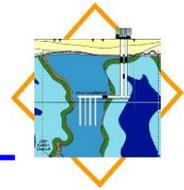
#### Identification of design-basis incidents

It is assessed whether the events, or combinations of events, which are regarded as possible could indeed lead to considerable radioactivity releases under the specific conditions in the final repository. Relevant sub-systems of the facility will be selected for detailed analysis, for which such events/combinations of events are considered to have the most serious effects (design-basis incidents). Using this approach ensures that the possibility of such an event in other sub-systems of the facility with regard to radiological impacts is also covered.

#### Demonstration of adequate damage prevention measures

The adequacy of the damage prevention measures taken for the selected design-basis incidents will be demonstrated by means of a detailed analysis of such incidents. The individual aspects to be described, together with the preventative protection measures, are the initiating event, the sequence of the incident, the barriers taking effect and the mitigation measures. Subsequently, the activity released will be calculated taking into account the inventory discharge and its retention within the barriers, and the radiological impacts will be determined.

For events with a low probability of occurrence, the adequacy of the prevention measures taken is provided through a statement of the measures taken to minimise the risks.



### 5.11.2.2 Probabilistic safety analyses

The 'Probabilistic Safety Analysis (PSA)' which is to be carried out within the framework of the 'periodical safety analysis of nuclear power plants' relates to existing nuclear power plants. At the moment, there are no regulatory provisions concerning the necessity to perform a probabilistic safety analysis as a prerequisite for obtaining a license to construct and operate a final repository.

Notwithstanding any current licensing requirements, the following section describes a number of recommendations for a reasonable application of probabilistic safety analysis for the construction and operation of a final repository for the emplacement of HLW and spent fuel elements.

According to the "Guide Probabilistic Safety Analysis" [Leitfaden 2005], the results of the PSA are to:

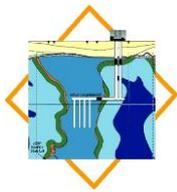
- supplement the deterministic safety assessment
- be taken as a basis to determine the necessity and urgency of safety improvements
- identify vulnerabilities of a plant through a comparison of the frequency of individual occurrences of plant conditions which cannot be controlled
- enable a balanced plant concept with regard to safety issues.

The methods suggested in [Leitfaden 2005] are a suitable set of guidelines for performing the PSA.

### 5.11.3 Conventional operational safety in the repository mine

The safety-related issues in a conventional mine are governed by numerous sectional implementation regulations to the German Federal Mining Act [BbergG 2005]. These implementation regulations also apply to an underground repository. With regard to the nuclear related issues of a repository, the German Radiation Protection Ordinance [StrlSchV 2012] additionally applies, among others.

In contrast to the deterministic and/or probabilistic safety analyses described above, neither the identification of sequences of events that are to be considered to be incidents according to §3 StrlSchV, nor the determination of comprehensive design-basis incidents are subjects for conventional operational safety. With regard to the planned design of the underground construction, and based on the planned underground operational activities, the purpose of this analysis is to investigate whether influences or impacts from the areas ventilation, power supply, rock mass, fire or damage to the means of transportation for the waste packages (transport cart) can cause situations that in addition to impeding or disturbing operations can



endanger operational safety and may require additional research. These situations do not necessarily have radiological consequences.

Events with a possible impact on the various operating activities that could endanger operational safety include:

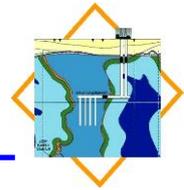
- failure of the ventilation system
- failure of the power supply system
- rock-mechanical impacts
  - cross-section reduction
  - inclination of floor
  - roof securing / formation of loose rock material
- inflow of brines and natural gases
- fire within the facility
- de-railing of a loaded transport cart.

The analysis carried out in this context revealed that failure of the ventilation system, or of the power supply, and a fire within the facility or the derailing of a loaded transport cart are controllable events and thus do not result in safety risks [Bollingerfehr et al. 2008].

Generally it is possible to control rock mechanical impacts from a safety-related point of view. For an undisturbed operation in the case of borehole emplacement it is necessary to investigate the convergences to be expected in emplacement drifts and boreholes under the expected conditions and taking into account the heat input of waste in adjacent boreholes. Based on these results, the drift diameters necessary for the emplacement operation can be calculated [Bollingerfehr et al. 2008].

Although the inflow of brines into a borehole is fairly unlikely, it cannot be completely ruled out. Unlike in a drift, it is virtually impossible to pump inflowing brines out of a borehole or at best only feasible above the canister last emplaced. Taking into account the geophysical measuring and surveying methods intended to be applied for the detection of brine reservoirs, the probability of undetected brine reservoirs – despite prior exploration – and the potential brine volumes must be investigated and assessed.

Despite prior exploration, the probability of natural gas inflow, particularly of hydrogen and methane, into a borehole and the potential quantities of such gases must also be investigated. It must be ascertained whether such an occurrence can be ruled out. If this is not the case, the potential effects of a deflagration must be determined [Filbert & Pöhler 2008].



#### 5.11.4 Demonstration tests for direct disposal of spent fuel

As previously mentioned the safety of a repository system and its components has to be demonstrated prior to implementation. In this context full-scale demonstration tests for the safe and reliable transport and emplacement of waste packages (POLLUX<sup>®</sup> casks for drift emplacement and BSK 3 canister for borehole emplacement) were successfully performed in the 1990ies and 2008 / 2009 respectively. A summary of the demonstration tests is given in the following sections.

Proof of the operational safety, in particular with regard to radiation protection, was a very important objective of the demonstration programme for direct disposal of spent fuel in the early 1990s. However the main objective was to prove the technical feasibility of the transport and emplacement systems for drift disposal of POLLUX<sup>®</sup> casks and borehole disposal of reprocessing waste and spent fuel canister. This did include the design, fabrication and functionality and safety demonstration tests of all the essential mechanical and mining engineering components and system parts. The results of the operational safety relevant demonstration tests are summarised in the following three sections.

##### Demonstration Test: Simulation of shaft transport

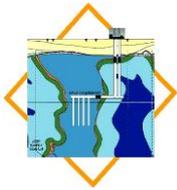
Prior to the demonstration test a probabilistic safety analysis [Filbert 1994a] was used to evaluate the proper safety criteria for implementation in the design of the hoisting system. The probability of occurrence of significant events associated with the hoisting system, including loading and unloading of waste shipments was evaluated.

Two safety related scenarios were evaluated using the probabilistic safety analysis method:

- The potential for exposure of operational personnel to increased radiation doses as a function of worker proximity to waste packages, and
- Potential for release of radioactive materials to the environment.

The combined probability of occurrence to the potential release of radioactive waste to the environment was determined to be  $1.3 \times 10^{-6}$ .

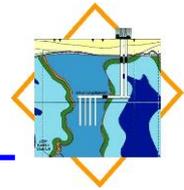
For the demonstration test performed in 1993 / 1994 [Filbert 1994b], a detailed analysis of all relevant parts and components of the shaft hoisting system regarding transferability of the state of the art to payloads of up to 85 t was carried out. If essential parts and components such as safety devices were new, or had to be built larger than in general use, it had to be shown that they worked properly. The safety devices were constructed at full scale to carry out the test programme. For this purpose a test stand as shown in Figure 5.18 was built in a former power plant, using the foundation of a decommissioned turbine. All the safety devices were designed, fabricated and tested according to the applicable specific requirements under consideration of the high weight of a loaded transport car of 85 t. 2,000 loading and unloading operations were successfully performed at the test stand, thus showing that all components were fabricated as designed and functioned reliably and safely.



**Figure 5.18: POLLUX® casks on transport cart in the hoisting cage**

Demonstration test: Drift disposal of POLLUX® casks.

Another series of demonstration tests performed in 1994 / 1995 [Filbert 1995] was aimed at demonstrating the feasibility of rail-bound handling, horizontal transportation, and drift emplacement of self-shielded spent fuel disposal casks with a weight of 65 t (POLLUX®) loaded with spent fuel. Here emphasis was put on the development and construction of components, such as an emplacement device, a transport cart and a mining locomotive. Their capabilities of working under normal operation conditions and under conditions of operational disturbance were demonstrated at a full-scale, above ground test facility in order to guarantee the safe handling of waste packages. Figure 5.19 shows the emplacement device and the dummy cask. 2,000 disposal operations were successfully performed at the test stand in Peine and showed the safety and reliability of the transport and emplacement system.

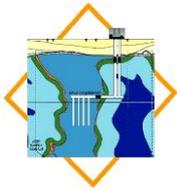


**Figure 5.19: Emplacement device for the drift disposal of POLLUX® casks**

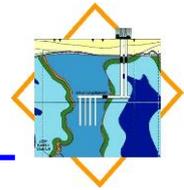
Demonstration test: Borehole disposal of canisters.

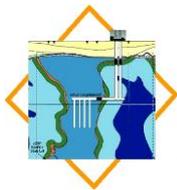
Within the third demonstration project, a transfer and emplacement system that could be used to dispose of both categories of heat-generating disposal packages (waste from reprocessing and spent fuel) in up to 300-m-deep boreholes in a repository in salt was designed, fabricated and tested, again at full scale.

In 2008 / 2009 full-scale demonstration tests [Filbert et al. 2010] were carried out in a surface facility of a former power station using inert canister dummies with the same dimensions and masses as real BSK 3 canisters (Figure 5.20). The test and demonstration campaigns comprising demonstration tests, simulation tests, and tests to identify potential operating failures and to develop preventive and corrective measures were successfully carried out. The demonstration tests comprised all the process steps, starting with the acceptance of the BSK 3 canister and finishing with the emplacement of the canister into the vertical borehole. In total, 1,004 complete emplacement operations were carried out with retrieval of the canister dummy included for continuous demonstration purposes. The entire system and each component were demonstrated to be safe, reliable, and robust.



**Figure 5.20:** Test stand for borehole disposal of canisters [Filbert et al. 2010]





## 6 Conclusions

The possibility of using rock salt as a disposal medium for HLW has been investigated in Germany for many years. Recently, in the R&D project VSG, an advanced safety analysis has been performed for this type of host rock based on international experience in safety cases and on the results of earlier research work. The geological setting of the Gorleben site was used in this study as an example of a site where rock salt would be the host.

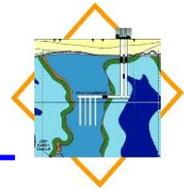
### 6.1 Assessment of the status

German Safety Requirements have been used in the development of a long-term safety concept and safety demonstration concept for a repository in a salt dome. The safety concept is based on the safe containment of radioactive waste in a containment-providing rock zone (CRZ) that will be established by the repository system immediately post closure. This shall be accomplished by preventing or limiting the intrusion of brine to the waste forms.

Key elements of the safety demonstration are the integrity proofs for the geological and geo-technical barriers and the analysis of the backfill compaction. The potential release of radionuclides must also be evaluated since it is clear that the technical barriers will always be permeable to a certain extent, even if they operate exactly as designed. In the study, a number of potential evolutions of the repository system during which an impairment of the barrier integrity is considered to occur, are evaluated on the basis of release scenarios which were developed using a comprehensive site-specific FEP catalogue. The potential releases of radionuclides from the CRZ in the selected scenarios were evaluated against suitable radiological safety indicators.

Using current state of the art in science and technology in the studies, the safety concept and the safety demonstration concept have been successfully applied to the geological situation represented by the Gorleben salt dome. The results of the R&D project VSG provide evidence that a safe HLW-repository, in a salt dome with a suitable geological structure, is feasible and its long-term safety can be demonstrated according to the state of the art in science and technology. This statement depends, however, on several assumptions which will have to be confirmed by comprehensive investigation of a specific site.

It has been established that those parts of a salt dome which are composed of highly ductile rock salt types associated with only very low fluid content, offer beneficial characteristics in respect of hosting emplacement areas of an HLW repository. The creeping capacity of the highly ductile rock salt will lead to the rapid closure of the backfilled mine openings of the emplacement areas, leading to the containment of the deposited waste. The very low fluid content of the salt will limit chemical processes such as corrosion of the waste packages.



The salt rock surrounding the CRZ will provide sufficient protection of the CRZ from any undue external impacts. In contrast, the properties of the overburden rock above the salt dome are of lesser relevance for long-term safety.

Whilst the extensive recorded results from earlier exploration of the Gorleben salt dome provide an appropriate data base for the safety assessment, these results will need to be supplemented by additional in-situ investigations. In general, a definitive statement as to the suitability of a specific site will require a comprehensive exploration of that particular site.

In the studies, repository layouts for both borehole and drift emplacement variants have been developed and these variants include some technical aspects of advanced backfilling and sealing concepts. The repository layouts explored meet the regulatory requirement to allow for the retrieval of the emplaced waste during the operational phase and a further requirement for the handling of the waste packages for a predetermined period of time after closure of the repository. The practicality of fabrication of the technical barriers as detailed has largely been demonstrated.

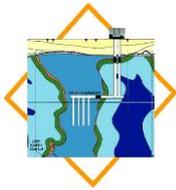
The three key aspects dominating the long-term safety of an HLW repository in salt are the geologic properties of the site, the characteristics of the repository layout and the technical plugging and sealing measures. A safety statement for a geological repository cannot therefore be based solely on the geology of the site. A site selection process which is aimed at selecting the best option, in a technical sense, from a range of options must rely not only on consideration of differences in geology, but must also consider the most appropriate site-specific layouts. This latter requirement may offer significant challenges for establishing the site selection procedure.

It is worth noting that the approach developed for domal salt to derive practical concepts from the Safety Requirements may be applicable to other rock types, e.g. clay, as well as to bedded salt.

## 6.2 Future R&D

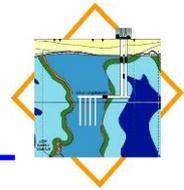
Although geological and technical knowledge on final disposal in salt formations has reached a mature status, some knowledge gaps still exist. The following R&D areas are considered to be of particular importance for the validity and robustness of the safety statement and for compliance with the Safety Requirements:

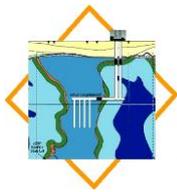
- Geological processes that are potentially highly relevant for the long-term safety of a repository in domal rock salt are
  - i. the uplift of the salt dome due to diapirism,
  - ii. the subsidence of the salt table, and
  - iii. the formation of glacial channels.



The safety statement would therefore benefit from improvements in the derivation of the uplift and subsion rates and in the evaluation of the potential depth of glacial channels.

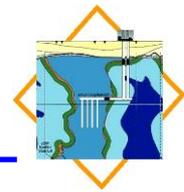
- The compaction of the crushed salt is recognized as essential for the long-term containment of the waste. Whilst the process is well understood, at the end of the process when the porosity becomes very small there are significant uncertainties concerning the compaction rate and the effective porosity values.
- Knowledge of mass transport processes offers significant scope for improvement. A database of measured values for two phase flow parameters is not available for either compacted crushed salt or natural rock salt. Uncertainties exist around the effective liquid and gaseous diffusional mass transport rates in partially saturated pores. Furthermore, the process of mobilisation of gaseous radionuclides from the waste forms is poorly understood and difficult to quantify.
- The full technical feasibility of the retrievability concept must be demonstrated as must some details of the practicality of fabrication of the technical barriers. Further studies are required in order to detail these requirements and to evaluate the impact they might have on the technical barriers. One such example is the verification of a sufficiently impermeable excavation damaged zone (EDZ) in the salt surrounding the geotechnical barriers. Another example is the demonstration of the suitability and adequacy of the procedure termed the 'method of partial safety coefficients' which is common in civil engineering, but which will need to apply for significantly longer demonstration periods in the HLW disposal situation.
- The assessment of the integrity of the geological barrier will benefit from developments in the field of numerical methods such as. the coupling of hydraulic and mechanical processes in finite element calculations.
- Finally, some aspects of the safety demonstration might benefit from conceptual improvements such as the methodological approach to deal with combinations of low-probable scenarios, or better estimation of probabilities by using alternative tools for their calculation.



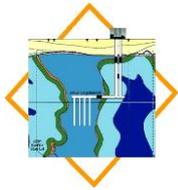


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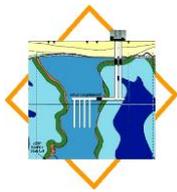
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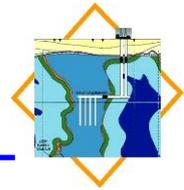
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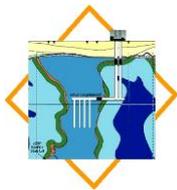


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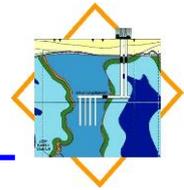
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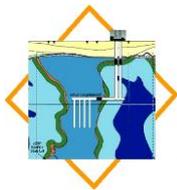
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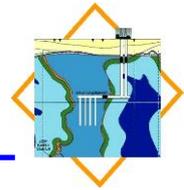
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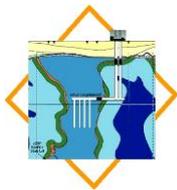
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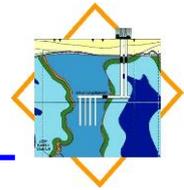
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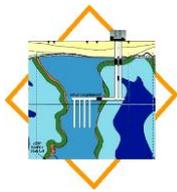
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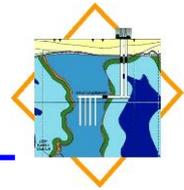
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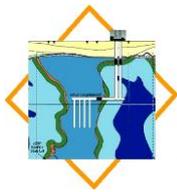
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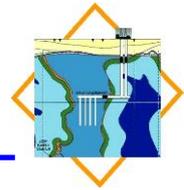
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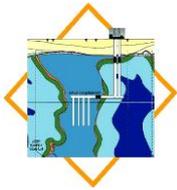
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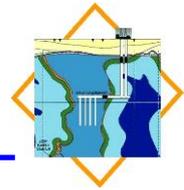
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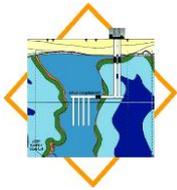
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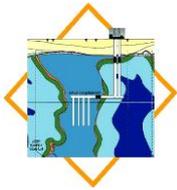


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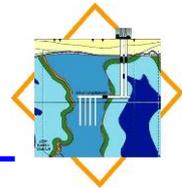
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## 9 Glossary

Waste, heat-generating radioactive	Heat-generating radioactive waste is characterised by high levels of activity concentration, and associated with this, a high thermal output during decay. Such waste places particular requirements on a repository site (final disposal in deep geological formations, use of shielded in-facility transport containers, use of special emplacement techniques, thermal design of the repository mine). In particular, heat-generating radioactive waste includes waste in the form of spent fuel rods and vitrified fission product concentrates (possibly vitrified together with feed sludges), compacted fuel cladding and structural parts from the reprocessing of spent fuel rods. In accordance with § 3, para. 2, no. 1a of the Radiological Protection Ordinance (StrlSchV), these are radioactive substances as defined in § 2, paragraph (1) of the Atomic Energy Act (AtG), which, according to § 9a of the Atomic Energy Act, are to be disposed of in a regulated manner as radioactive waste.
Containment	<p>Containment refers to a safety function of the repository system which is characterised by the fact that the radioactive waste is contained inside a defined rock zone in such a way that it essentially remains at the site of emplacement, and at best, minimal defined quantities of material are able to leave that rock zone.</p> <p>Safe containment describes the condition of the repository system in which there is at the most an insignificant release of radionuclides from the containment-providing rock zone during the demonstration period.</p> <p>Complete containment describes the status of the repository system, in which there is no release of radionuclides from the containment-providing rock zone during the demonstration period.</p>
Containment-providing rock zone	The containment-providing rock zone (CRZ) is part of the repository system which, in conjunction with the technical seals (shaft seals, cavern sealing structures, dam structures, backfill, ...), ensures containment of the waste.
Repository system	The repository system comprises the repository mine, the containment-providing rock zone, and the geological strata surrounding or overlying this rock zone up to surface level, insofar as these are relevant for safety purposes and must therefore be taken into account for the safety case.
Evolution	Probable evolutions refer to normal evolutions forecasted for this site, and evolutions normally observed at comparable locations or similar geological situations. The forecasted normal evolution of properties should be used as a basis when considering the tech-

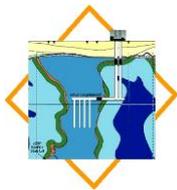


nical components of the final repository. If quantitative data on the probability of a certain evolution occurring is available, and the probability of it occurring in relation to the reference period is at least 10%, this shall be considered a probable evolution.

Less probable evolutions refer to evolutions which may occur for this site under less favourable geological or climatic assumptions and which have rarely occurred in comparable locations or comparable geological situations. A consideration of the technical components of the final repository should be based on the normal forecasted evolution of their properties upon occurrence of the respective geological evolution. Any unfavourable evolutions in the properties of the technical components that deviate from normal evolution should also be investigated. Repercussions on the geological environment should be considered. Apart from such repercussions, anticipated geological evolutions should also be taken into account. Within such an evolution, the simultaneous occurrence of several unrelated faults should not be assumed. If it is possible to make a quantitative statement on the probability of a certain evolution or an unfavourable evolution in a technical component's property, this should be taken into account if the probability in relation to the reference period is at least 1 %.

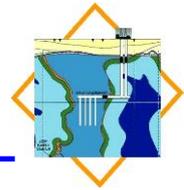
Improbable evolutions refer to evolutions which are not expected to occur at the site even under unfavourable assumptions, and which have not been observed in comparable locations or comparable geological situations. Statuses and evolutions for technical components which can be more or less excluded by taking certain action, as well as the simultaneous, independent failure of several components, are classed as improbable evolutions.

FEP	FEPs are Features, Events and Processes that might affect the current state or the future evolution of the repository system.
Geological barrier	Geological rock units between the emplacement area and the biosphere, which hinder or prevent intrusion of brine into the repository and hinder or prevent the propagation of harmful substances.
Integrity	Integrity describes the availability of containment-providing properties of a barrier.
Safety analysis	The safety analysis is used to analyse the behaviour of the final repository system under a range of different load situations and with due regard for data and model uncertainties, malfunctions and possible future evolutions in relation to safety functions. It ends with an assessment of the reliability of the final repository's compliance with the safety functions and hence also of its robustness (safety assessment).



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Safety function	<p>A safety function is a property or process occurring in the final repository system which guarantees compliance with safety-related requirements in a safety-related system or subsystem or individual component. The combined action of all such functions ensures compliance with all safety requirements, both during the operating phase and the postclosure phase of the final repository.</p>
Safety concept	<p>The safety concept describes in a verbal, argument-based way how the natural conditions, processes and technical measures in summary yield a status of safety.</p>
Safety case	<p>The safety case is based on a comprehensive safety analysis. It comprises the inspection and evaluation of data, measures, analyses and arguments indicating compliance with these Safety Requirements and hence the safety of the final repository. A comprehensive safety case includes a combination of all the evidence listed in these Safety Requirements, and may be updated in suitable depth for the various phases of final disposal according to the latest state of the art. In particular, a distinction is made between safety cases for the operating phase and for the post-closure phase of the final repository.</p>
Scenario	<p>A scenario refers to a post-decommissioning evolution of the final repository system and its safety-related properties, with a greater or lesser degree of probability, based on the current site conditions and on the basis of geoscientific and other considerations. This evolution is determined by the starting situation as well as by future events and processes. Several evolutions may also be combined into one scenario.</p> <p>The reference scenario describes an as big as possible entity of potential evolutions of the repository system, that are regarded as probable. It arises under pre-defined assumptions from the combination of probable FEP with direct influence on the function of the initial barriers and of the FEP that determine the mobilization of radionuclides from the waste and their subsequent transport. The probable characteristics of these FEP are taken as a basis, where the characteristics are generally derived from the probable characteristics of the probable FEP that are causally related to them.</p> <p>Alternative scenarios describe less probable evolutions of the repository system or those probable evolutions of the repository system, which are not covered by the reference scenario. Such evolutions can result from the appearance of a less probable FEP, the less probable characteristic of a probable FEP, or from the alternative to a specific assumption for the reference scenario.</p>



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Scenario development	The scenario development is equivalent to the systematic derivation and description of the evolution potentials of the repository system, which are relevant for a reliable assessment of the repository safety. This procedure is based on an FEP catalogue.
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