Evaluation of methods and tools to develop safety concepts and to demonstrate safety for an HLW repository in salt

Final report

Peine, March 2017

Evaluation of methods and tools to develop safety concepts and to demonstrate safety for an HLW repository in salt

Final report

W. Bollingerfehr
D. Buhmann
S. Dörr
W. Filbert
A. Gehrke
U. Heemann
S. Keller
J. Krone
A. Lommerzheim
J. Mönig
S. Mrugalla
N. Müller-Hoeppe
A. Rübel
J. R. Weber
J. Wolf

Peine, March 2017

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 02E10719 and 02E10729. The authors alone, however, are responsible for the contents of this study.

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

Table of Contents

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.

After completion of the VSG project in 2013 complementary work has been performed within the framework of the ISIBEL programme. In this context e.g. potential contributions of natural and antropogenic analogs to confidence building were addressed as well as the feasibility and limits of deriving a repository concept strictly from requirements.

The report in hands provides a comprehensive summary of the results of R&D work regarding HLW disposal in domal salt formations that has been performed after launching the ISIBEL programme in 2005. 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 that had been gained up to now and demonstrates that the tools required for safety evaluations are available and allow reliable safety assessments of HLW repositories in salt formations. It completes the ISIBEL programme that was lasting for 11 years. A condensed summary of the results presented in this study is given in chapter 6 Conclusions.

1 Introduction

Rock salt as a disposal medium for High Level Radioactive Waste, including spent fuel and reprocessing waste (HLW)[1], has been investigated in Germany for more than fourty 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. In the beginning, the German nuclear waste management programme focused on the development of generic waste repository concepts in salt domes. In recent studies in-depth investigations, the Gorleben geology was used as a specific site. 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 results become 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) established a scientific working group "AKEnd" 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

[1] 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 2010][2]. 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 the evolutions' occurrence.

In the context of the national discussion on the Safety Requirements and the international discussion on the safety case the R&D project ISIBEL (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 [Buhmann et al. 2008c]. 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.

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 from the repository depth 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 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,
- demonstration and assessment of the “safe containment“ concept,
- assessment of uncertainties within long-term safety analysis, and

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

- development of a proposal for the structure and the content of a safety case report for an HLW-repository in salt formations.

In 2010, the project results were judged to be applicable to future safety analysis and safety demonstration. They provided the basis for the 'Preliminary Safety Assessment of the Gorleben Site' (VSG) as is discussed in [Fischer-Appelt et al. 2013]. The practical application and further development of the safety and demonstration concepts originally developed in the ISIBEL work were an important step forward for the safety case in salt in Germany. In VSG different site-specific repository concepts for spent fuel, reprocessing waste and negligibly 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 a rock-zone providing the retention of radionuclides for the safety demonstration period (Containment Providing Rock Zone). 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.

In parallel to VSG a follow-up project of ISIBEL was launched, that focussed on the following topics:

- repository issues: specification of additional types of waste to be considered for disposal, analysis of the methodological approach for development of repository concepts,
- long-term safety demonstration for geotechnical barriers: relationship between barrier size and failure propability, applicability of partial safety factor method on long periods,
- evaluation of tools for process analyses: thermomechanical, geomechanical, hydrological and geochemical calculations, gas processes,
- specific investigations on reference and alternative scenarios: methodological approach to reflect uncertainties in scenario development,
- supplementary work on the FEP catalogue: contribution to preparation of NEA-FEP catalogue and data base, and

- applicability of natural analogues.

In 2013 a summary report was given to reflect the state of the art in safety analysis and safety assessments for HLW repositories in domal salt formations and including the interim status of the ISIBEL-project [Bollingerfehr et al. 2013]. The present report is an upgrade of this report considering the results of the final investigations of ISIBEL and the essential outcomes of other R&D projects.

The overall work carried out in the described R&D projects covers the main elements of a safety case. Figure 1.1 assigns the chapters of this report to the elements of the safety case according to [IAEA 2012]. The approach to demonstration of safety refers to the safety objectives and safety principles that must be applied and the regulatory requirements that must be met.These issues are described in chapters 2 and 3. The description of the disposal system with information and knowledge about the disposal system is provided in chapter 4. Safety assessment is the main component of the safety case and involves assessment of a number of aspects as explained in detail in chapter 5. The importance of addressing uncertainties in safety assessment is reflected in chapter 5.2. The integration of safety arguments includes all of the different evidence, arguments and analyses that are available to support the assessment of the quality and performance of the disposal facility. In the ISIBEL project especially the applicability of natural analogues was investigated (chapter 5.12).

The main elements of a safety case are shown in Figure 1.1.

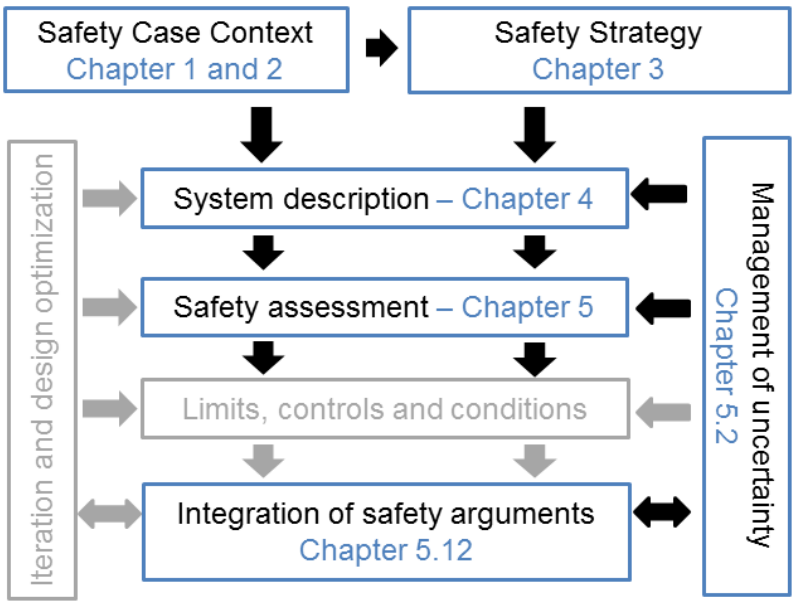

Figure 1.1: Assignment of the elements of the safety case according to [IAEA 2012] by the descriptions in this report (grey elements and interactions not covered by the ISIBEL projects).

The iteration process and the limits, controls and conditions (for example limits and conditions for the acceptable concentration levels for specific radionuclides in the waste) were not topics of the ISIBEL projects.

This report is a summary of a wide variety of reports generated in the context of the R&D project ISIBEL:

a.) Reports related to the ISIBEL project:

1. Geologisches Referenzmodell für einen HAW-Endlagerstandort im Salz [Keller 2007].
2. Konzeptionelle Endlagerplanung und Zusammenstellung des endzulagernden Inventars. [Bollingerfehr et al. 2008].
3. Bewertung der Betriebssicherheit [Filbert & Pöhler 2008].
4. FEP-Generierung und Szenarienentwicklung [Weber & Keller 2008].
5. Nachweis der Integrität der geologischen Barriere [Heusermann 2007].
6. Nachweiskonzept zur Integrität der einschlusswirksamen technischen Barrieren [Kreienmeier et al. 2008].
7. Untersuchungen zur Ermittlung und Bewertung von Freisetzungsszenarien [Buhmann et al. 2008a].
8. Nachweiskonzepte für die Einhaltung der nicht radiologischen Schutzziele in der Nachbetriebsphase [Hippler et al. 2008].
9. FEP-Katalog für einen HAW-Standort im Wirtsgestein Salz [Buhmann et al. 2008b].
10. Überprüfung und Bewertung des Instrumentariums für eine sicherheitliche Bewertung von Endlagern für HAW – ISIBEL Phase 1 [Buhmann et al. 2008c].
11. FEP-Katalog für einen HAW-Standort im Wirtsgestein Salz. Revision 01 [Buhmann et al. 2010a].
12. Erläuterungen zur Revision des FEP-Kataloges [Buhmann et al. 2010b].
13. Entwicklung und Test einer Methodik zur Ableitung eines Referenzszenarios [Buhmann et al. 2010c].
14. Nachweis und Bewertung des Isolationszustandes „Sicherer Einschluss" [Buhmann et al. 2010d]
15. Behandlung von Ungewissheiten im Langzeitsicherheitsnachweis für ein HAW-Endlager in Salzgesteinen [Buhmann et al. 2010e].
16. Überprüfung und Bewertung des Instrumentariums für eine sicherheitliche Bewertung von Endlagern für HAW – ISIBEL Phase 2 [Buhmann et al. 2010e].

17. Status of the Safety Concept and Safety Demonstration for an HLW Repository in Salt [Bollingerfehr et al. 2013].

18. Probabilistische Methoden als Hilfsmittel zur Bemessung von Verschlussbauwerken für ein Endlager im Salinar [Röhlig et al. 2014].

19. Ergänzung des endzulagernden Inventars [Dörr 2013].

20. Geochemical modeling in the near field of a HLW repository in a high-saline environment [Moog et al. 2016].

21. Aspects on the Gas Generation and Migration in Repositories for High Level Waste in Salt Formations [Rübel et al. 2013].

22. Probabilistische Bewertung von Szenarien in Langzeitsicherheitsanalysen. Ergebnisse des Projekts ISIBEL [Buhmann et al. 2016].

23. Natürliche Analoga im Wirtsgestein Salz. Teil 1 Generelle Studie (2011), Teil 2 Detailstudien (2012 – 2013) [Brasser et al. 2013].

b.) Reports related to the VSG project:

24. Endlagerkonzepte. Bericht zum Arbeitspaket 5 [Bollingerfehr et al. 2011]

25. Salzgeologische Bewertung des Einflusses von „kryogenen Klüften“ und halokinetischen Deformationsprozessen auf die Integrität der geologischen Barriere des Salzstocks Gorleben [Hammer et al. 2012a].

26. Untersuchungen von Kohlenwasserstoffen im Erkundungsbergwerk Gorleben. Technischer Bericht [Hammer et al 2012b].

27. Abfallspezifikation und Mengengerüst. Basis Laufzeitverlängerung der Kernkraftwerke (September 2010). Bericht zum Arbeitspaket 3 [Peiffer et al. 2011a].

28. Geowissenschaftliche Langzeitprognose. Bericht zum Arbeitspaket 2 [Mrugalla 2011].

29. Grundzüge des Sicherheits- und Nachweiskonzeptes. Bericht zum Arbeitspaket 4 [Mönig et al. 2012].

30. Abfallspezifikation und Mengengerüst. Basis Ausstieg aus der Kernenergienutzung (Juli 2011) Bericht zum Arbeitspaket 3 [Peiffer et al. 2011b].

31. Endlagerauslegung und –optimierung. Bericht zum Arbeitspaket 6 [Bollingerfehr et al. 2012].

32. FEP-Katalog für die VSG. Bericht zum Arbeitspaket 7 [Wolf et al. 2012].

33. Szenarienentwicklung: Methodik und Anwendung. Bericht zum Arbeitspaket 8 [Beuth et al 2012a].

34. Integritätsanalyse der geologischen Barriere. Bericht zum Arbeitspaket 9.1 [Kock et al. 2012].

35. Integrität geochnischer Barrieren. Teil 1 Vorbemessung. Bericht zum Arbeitspaket 9.2 [Müller-Hoeppe et al. 2012a].

36. Integrität geochnischer Barrieren. Teil 2 Vertiefte Nachweisführung. Bericht zum Arbeitspaket 9.2 [Müller-Hoeppe et al. 2012b].

37. Radiologische Konsequenzenanalyse Bericht zum Arbeitspaket 10 [Larue et al. 2013].

38. Untersuchungen zum menschlichen Eindringen in das Endlager. Bericht zum Areitspaket 11 [Beuth et al. 2012b].

39. Synthesebericht für die VSG. Bericht zum Arbeitspaket 13 [Fischer-Appelt et al. 2013].

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 [BMU 2010]. 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*[3].

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)*[4]*, 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 decision points in the development process of the final repository. An important part of the safety case is a safety analysis and safety assessment covering a period

---

[3] Text written in italics is directly taken from the English translations of the Safety Requirements.

[4] 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.

of a million years that 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[5]) 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-*

[5] 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 are met. For probable evolutions it must be demonstrated that the additional[6] 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 in-

[6] The term "additional" refers to the fact that the dose caused by release of radionuclides from the repository is additional to the naturally occurring dose. In the following, this term is omitted in the discussion of calculated doses.

tegrity 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.

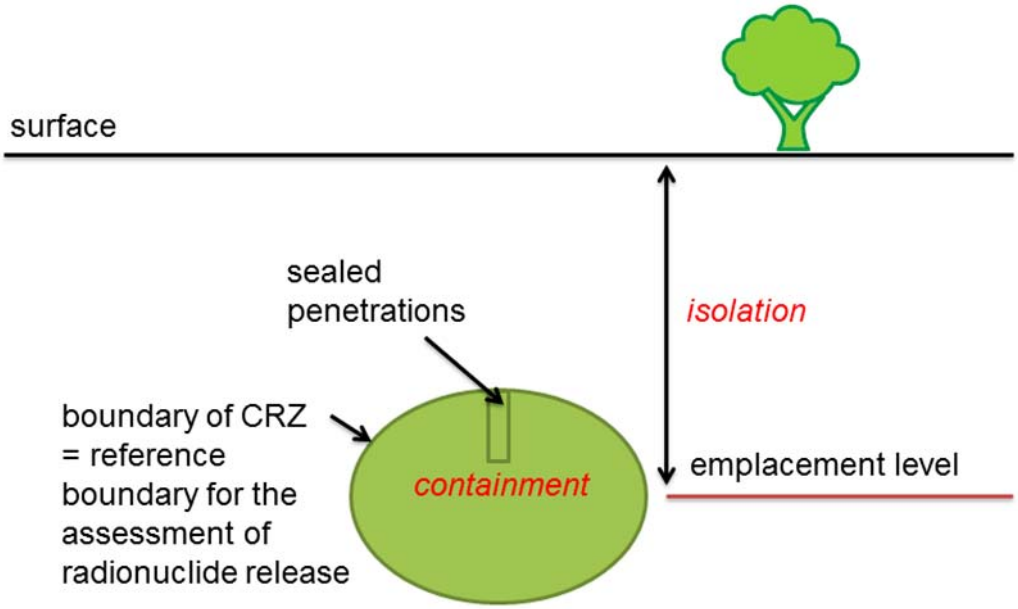

Figure A 1: Key elements for a safety concept according to the Safety Requirements

3 General approach to 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 [Buhmann et al. 2008c]. 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 over the years. During compaction, the porosity and permeability of the crushed salt decreases until, in the long run, it has comparable 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 [Mönig et al. 2012]. 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 will be built that provide the required sealing immediately upon construction. 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 sufficiently developed by the time the performance of the engineered bar-

riers 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.

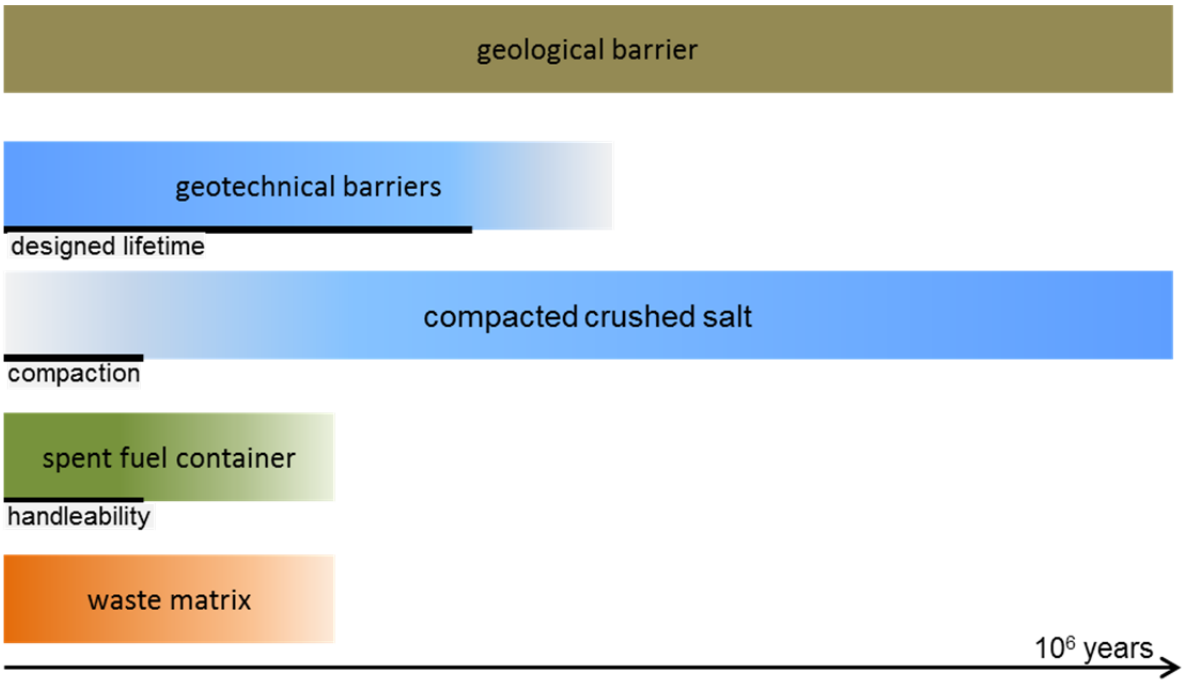

Figure 3.1: Evolution with time of the sealing effects of important barriers in the post closure phase of a repository system (the colour intensity represents the degree of the respective sealing effect, not to scale)

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.

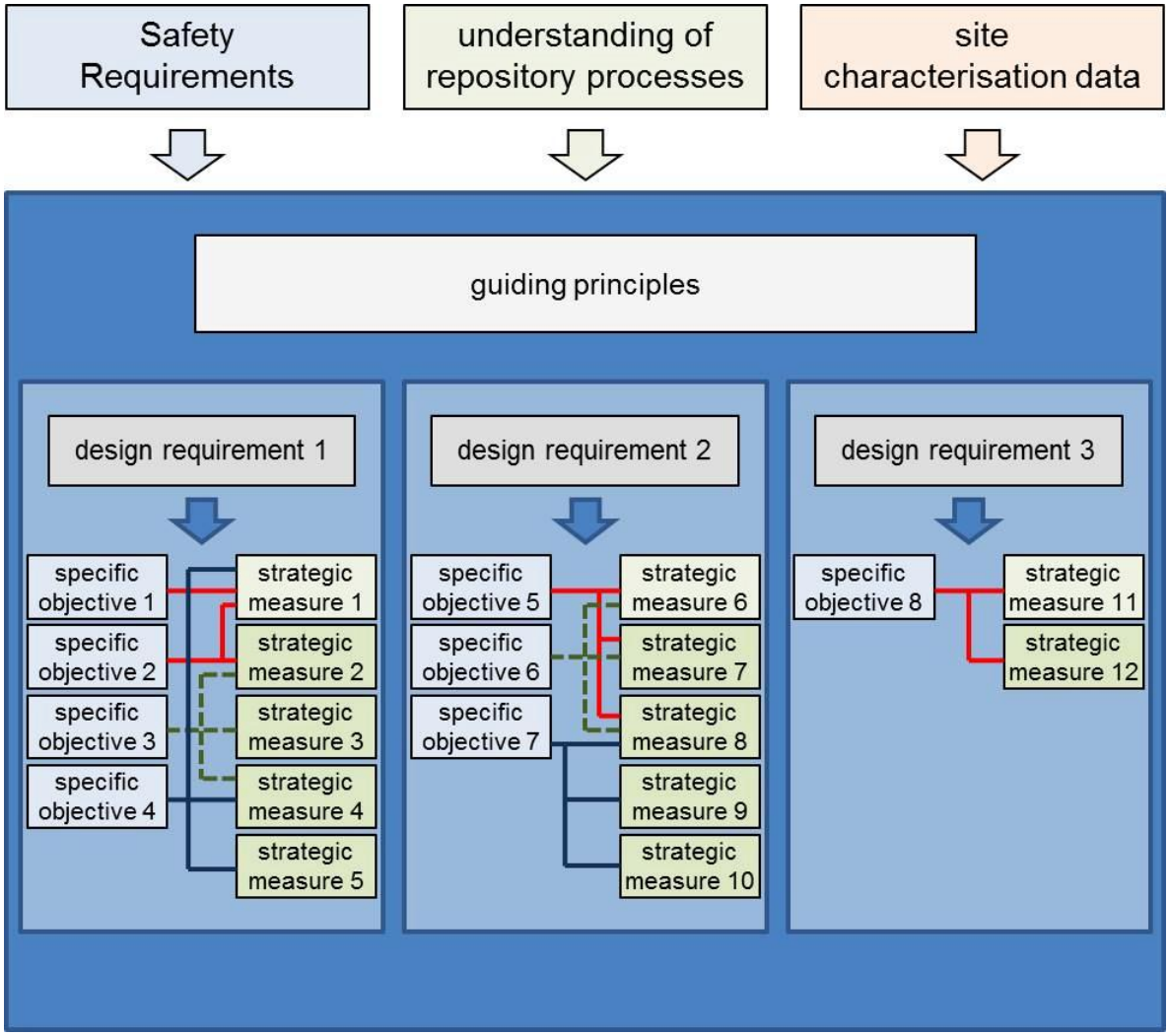

Figure 3.2: Principle approach to derive specific objectives and strategic measures for the safety concept

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.

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 domal salt as host rock. 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 after the German government decided to investigate its suitability for the construction of a repository. The exploration work was interrupted in 2000 for 10 years in the context with the first German nuclear phase-out decision and stopped in 2013 as a consequence of restarting the site selection for HLW repository in Germany. The exploration planning of the nineties provided 9 exploration areas to investigate the wide stretching central area of the Gorleben salt dome. So far the first exploration area that is located in vinicy of both shafts were investigated in detail only. Nevertheless, the Gorleben salt dome can be considered as the best investigated domal salt structure.The corresponding findings 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 established 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.

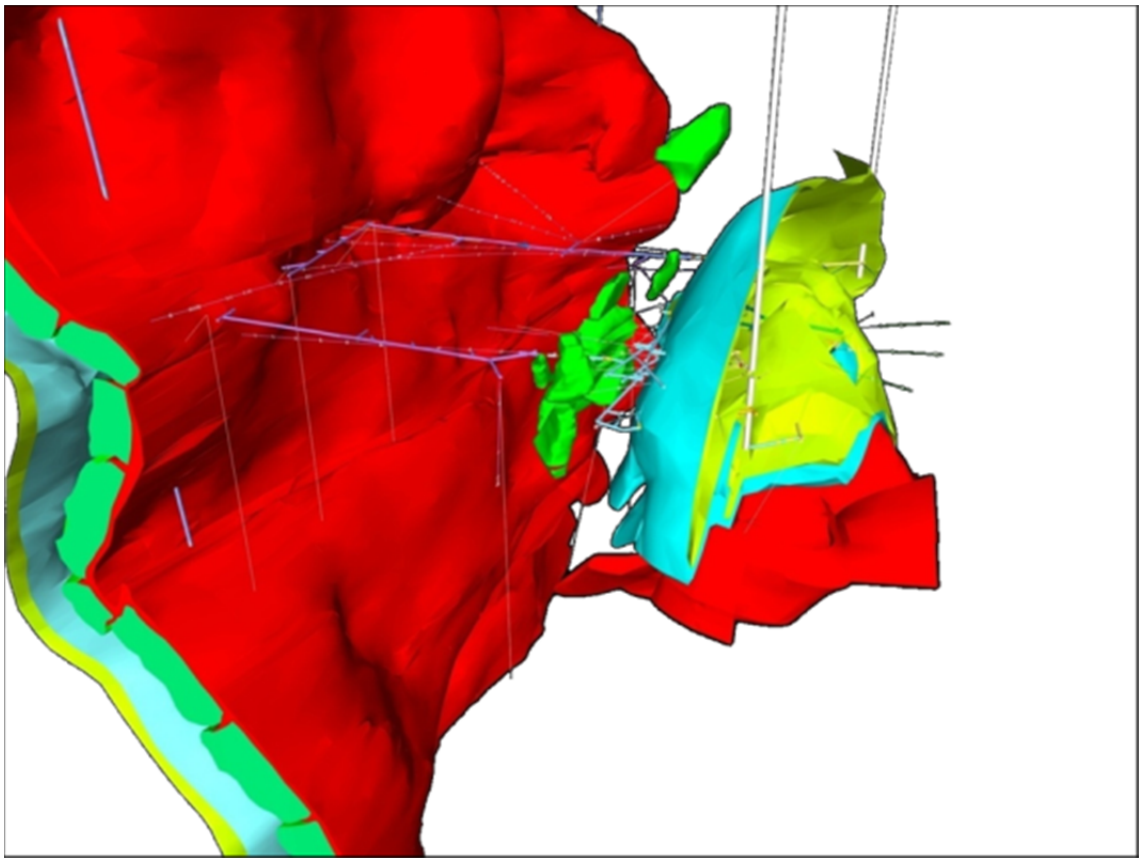

Figure 4.1: Detail of the geological 3D model of the Gorleben salt dome including the shafts and galleries of the exploration mine, potash seam (red), Liniensalz (blue) and Hauptanhydrit blocks (green)

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. Figure 4.2 and Figure 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 occur in isolated clusters of up to 1 cm in size.

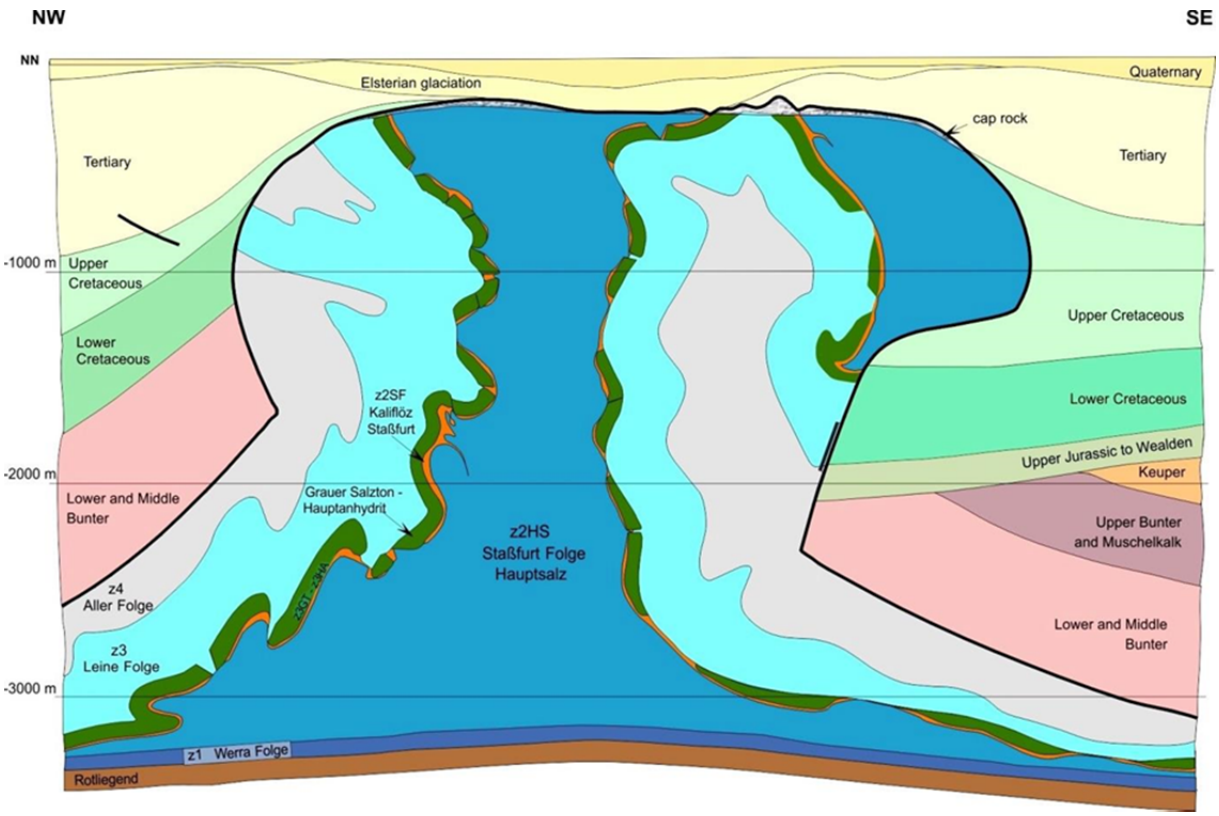

Figure 4.2: Simplified NW-SE geological cross-section of the Gorleben salt dome, [Bornemann et al. 2008], modified

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.

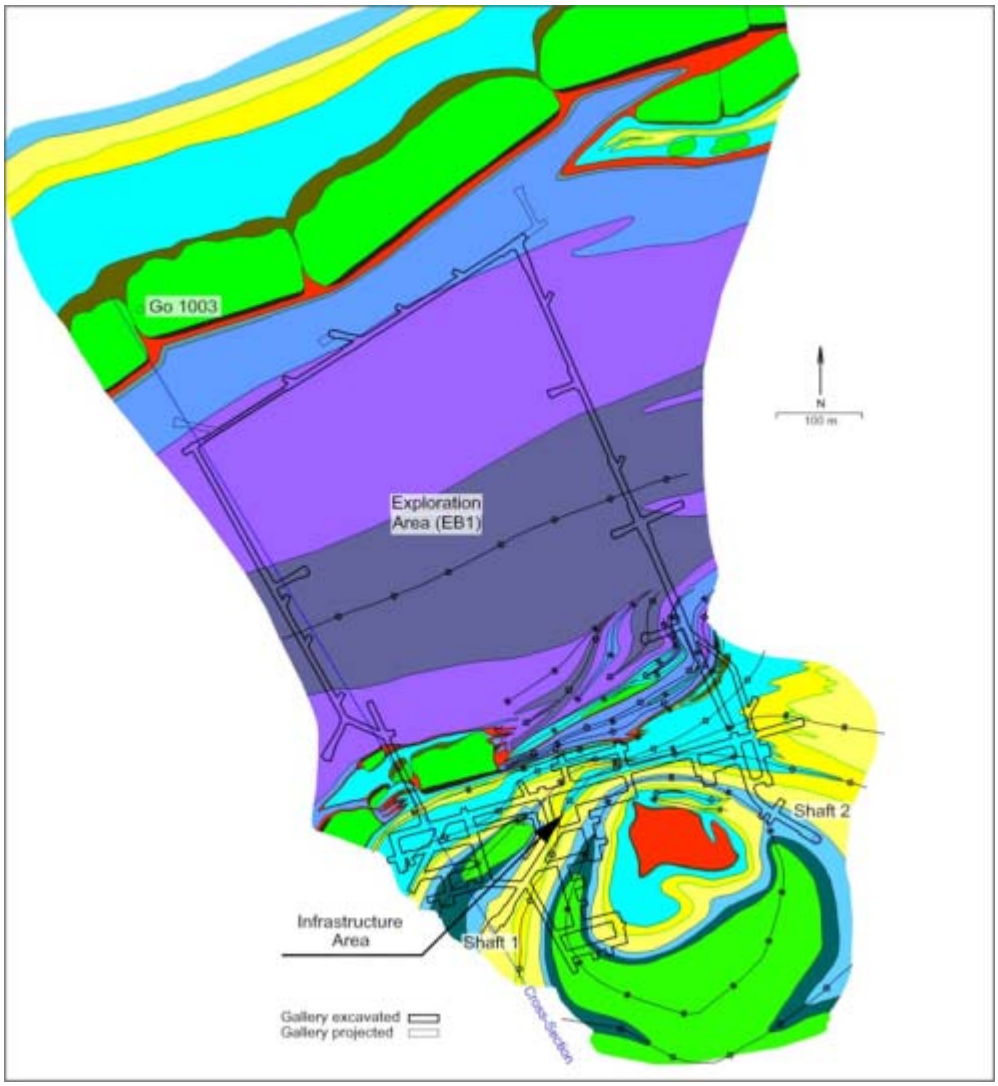

Figure 4.3: Horizontal section of the Gorleben salt dome at the depth 840 m, [Bornemann et al. 2008], modified

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 evolve thereby 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, subrosion 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 and size 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.

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.

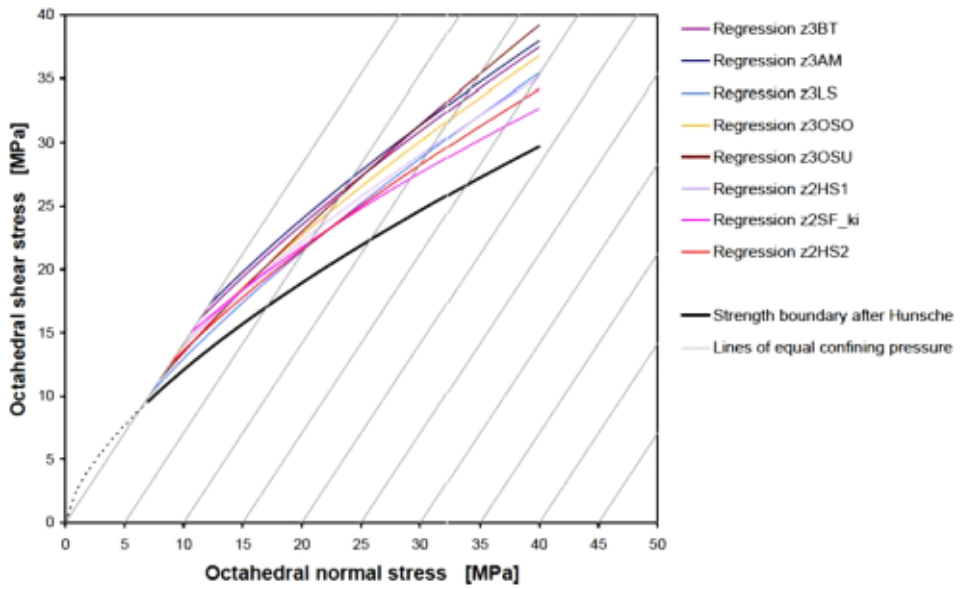

Figure 4.4: Failure strengths in Gorleben salt samples [Bräuer et al. 2011]

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/m², 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^{-5}/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 homogenously 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 intensively 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 was affected by formation pressure which was higher than the hydrostatic pressure of a water column at the same depth, the presence of an elevated pressure confirms that the reservoir was isolated.

The volume of the fluid was then estimated by carrying out a material balance. Because the solutions were accompanied by gas, but the liquid-gas ratio was unknown, considerable uncertainties arose concerning the compressibility of the fluid in the reservoir. This in turn gaves 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 were 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 metres. 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. 2012b].

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 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 glacigenic Gorleben channel that was created during the glaciation of the Elsterian ice age.

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.

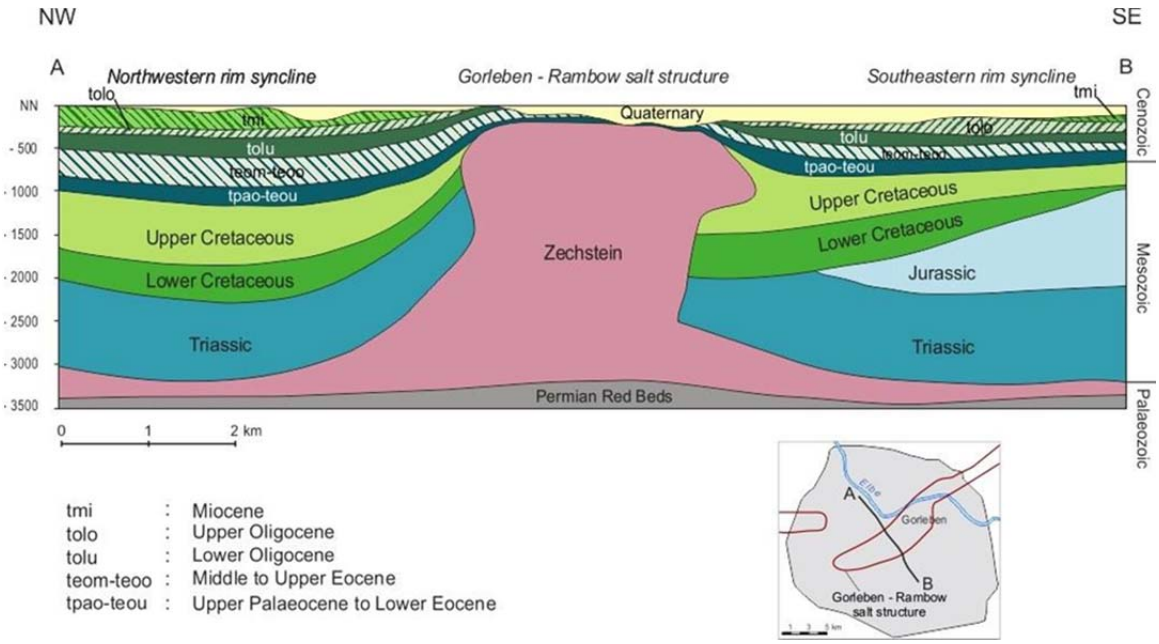

Figure 4.5: Simplified geological cross section of the study area [Köthe et al. 2007]

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 seismic evidence for any 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 layer, 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.

The hydraulic head of the water table in the Gorleben area varies between 13 and 21 m above sea level, 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.

These geological data are the base to assess the existing methods and tools for hydrogeological model calculations. These calculations generally aim at simulating the density-driven flow and the transport of contaminants in the overburden of a potential repository for radioactive waste. For details of the following information see Appendix A.

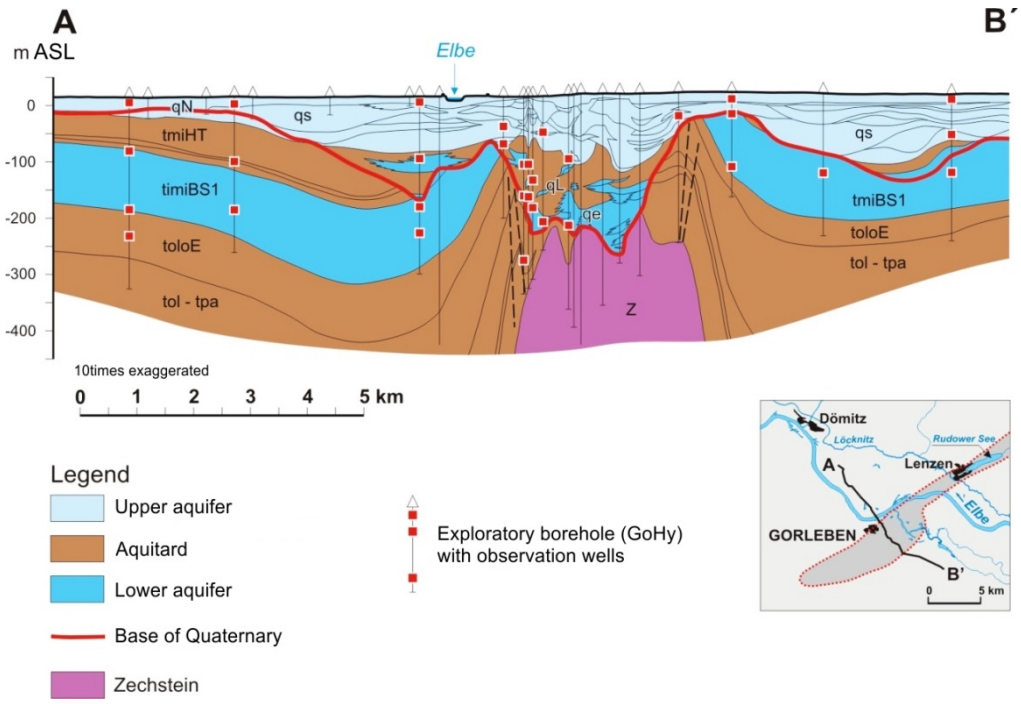

Figure 4.6: **Aquifer subdivision in the overburden of the Gorleben salt dome [Klinge et al. 2007], cross-section 10 times vertically exaggerated**

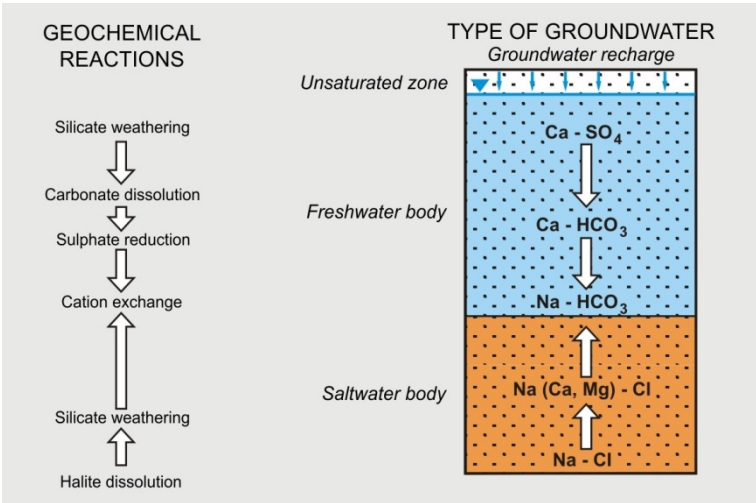

Figure 4.7: **Schematic diagram of the vertical groundwater type zoning within the Gorleben study area (from** [Klinge et al. 2007]**)**

The flow model of d^3f++ solves the equation system of density-driven flow and transport, which includes processes that are relevant for long-term safety assessments such as radioactive decay, sorption, precipitation, complexation, and colloidal transport [Fein 2004]. Appropriate pre- and postprocessors are applied to construct the geometry and grid files and to visualize simulation results.

The starting point for the development of the hydrogeological model was data from hydrogeological structural model described in [Ludwig 2001]. As the hydrogeological system forms a 3-dimensional structure of high geological complexity, a 2-dimensional model was built to characterise and analyse the flow field. The 2-dimensional model enables to analyse the general behaviour of the flow system and provides at the same time the great advantage of reduced complexity. The Gorleben data set was prepared for the construction of a 2-dimensional geometry and grid by some corrections, see Appendix A.

A comparison between the geometry before and after the manual corrections is shown in Figure 4.8.The most prominent changes concern the areas where the generation algorithm yielded lines that were implausible or expendable. Furthermore, the lines are to some extend less smooth due to the removal of nodes. However, the main characteristics of the geological structure were maintained.

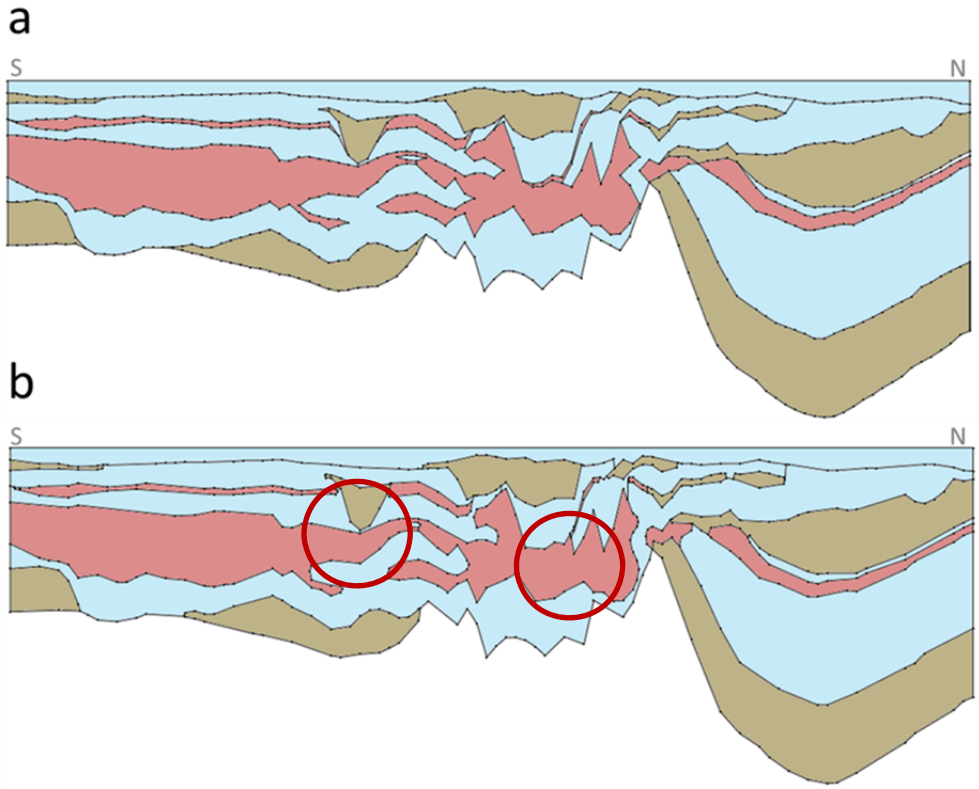

Figure 4.8: **Automatically generated geometry (a) and the same geometry after the manual corrections (b); aquifers are coloured blue, aquitards brown and aquicludes red, significant changes marked with red circles (15x exaggeration)**

From the model geometry, a grid was generated applying the software ProMesh 4 [Reiter 2014]. Grid generation is an important step for the construction of a numerical model. The resulting grid contained 4,316 nodes and 7,798 elements. It had a total length of 19,800 m and a depth of up to 440 m.

The grid was used as basis for flow simulations with variable density and viscosity. As already shown in Figure 4.8 three hydrogeological units were distinguished according to [Ludwig 2001]: aquifers with a moderate to high permeability, aquitards with a very low to low permeability and aquicludes with an extremely low to very low permeability. The hydraulic parameters of the three units are listed in Appendix A.

Figure 4.9 shows the initial and boundary conditions of the model. Initially, the bottom part of the model except for the northern rim syncline is filled up to a depth of 190 m with pure brine (relative salt mass fraction ω_{rel} = 1). Then a transition zone of 20 m is defined where the salt mass fraction declines linearly to ω_{rel} = 0. Above the depth of 170 m, the entire model is filled with fresh water (ω_{rel} = 0). Pressure according to the pressure profile is assigned to the upper boundary to represent the groundwater level. No salt enters the model area through this boundary (i. e. ω_{rel} = 0). The other boundaries were defined to be impermeable for flow. Again, ω_{rel} = 0 is assigned on the boundaries except for the contact zone with the salt dome where ω_{rel} is set to 1.

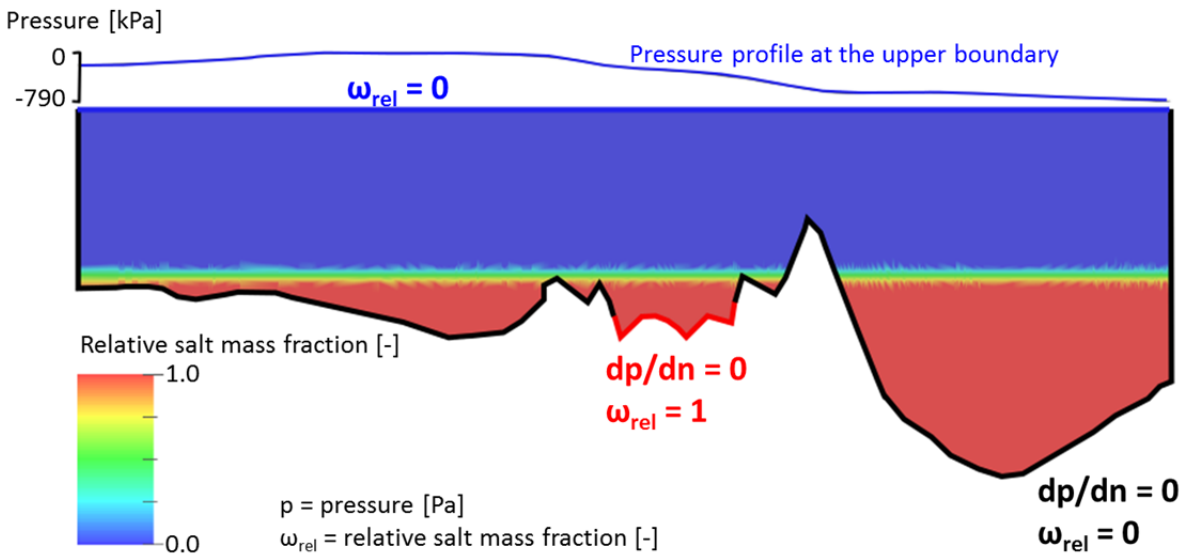

Figure 4.9: Initial and boundary conditions for the 2-dimensional model

Simulations were performed for 10,000 years model time (see Figure 4.10). Within this time, the initial salt distribution dissolves and zones of high salt concentration remain only above the salt dome and in the north-western rim syncline. The salt spreads not only within the aquifer but also through aquitard and aquiclude layers such that a characteristic salt distribution evolves.

The flow velocity corresponds mainly to the permeability. Highest flow velocities are found near the top boundary and at narrow passages within the aquifer. Water enters the model area through the upper boundary where the hydraulic pressure imposed by the boundary condition is highest. Effects of the density on the flow field can be observed above the salt dome where the flow appears to be non-directional and in the north-western rim syncline where large flow vortices evolve.

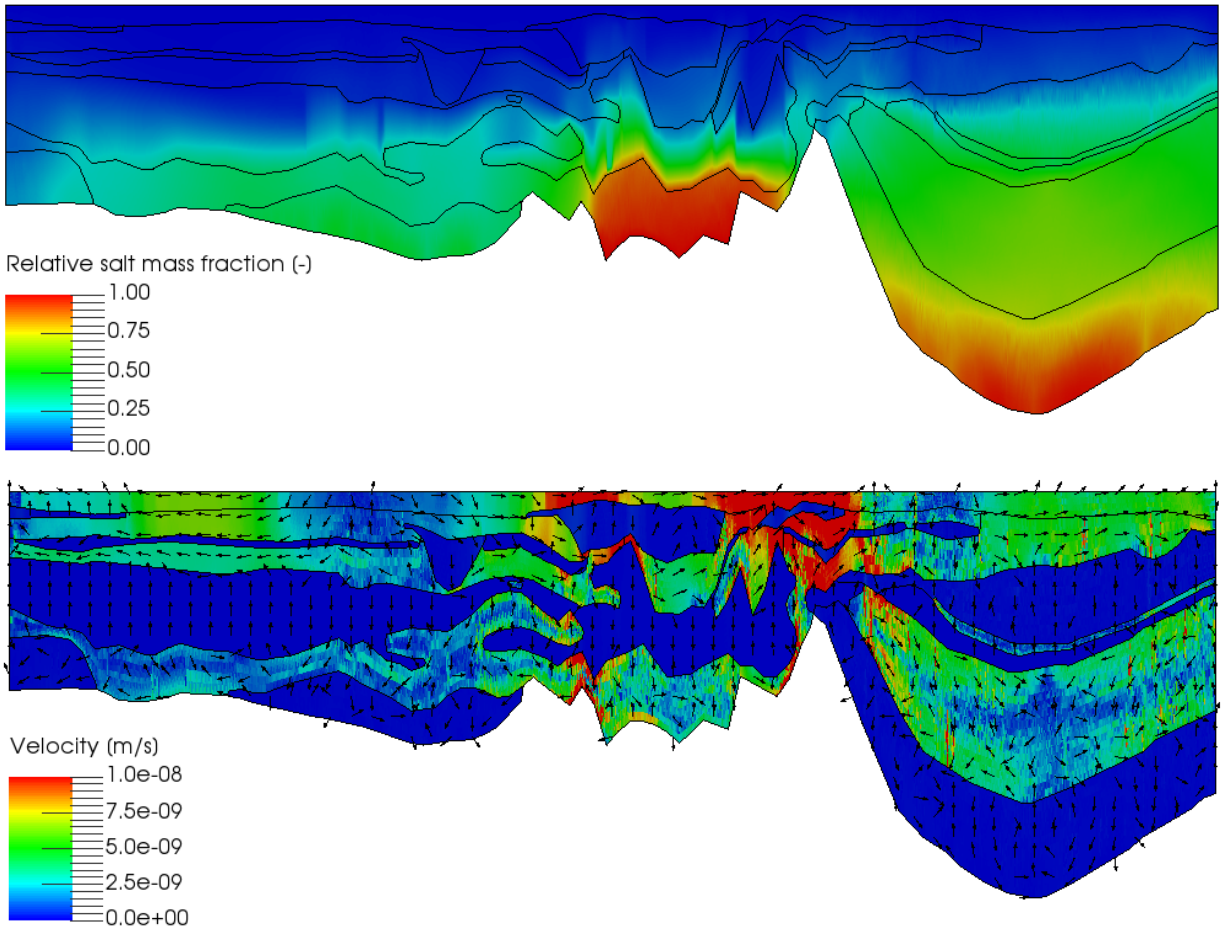

Figure 4.10: Simulation results after 10,000 years model time for the hydrogeological model: relative salt mass fraction (top) and flow velocity (bottom)

4.1.5 Climatic influence

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 are characterized 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].

In addition to the geological description of a site the knowledge of the climatic evolution is important for a long-term forecast of geological processes.

Glacigenic channels of various extents are known in North Germany associated with the Elsterian, Saalian and Weichselian glacials. Their presence is limited to areas which were covered by an ice sheet at the time (Figure 4.11).

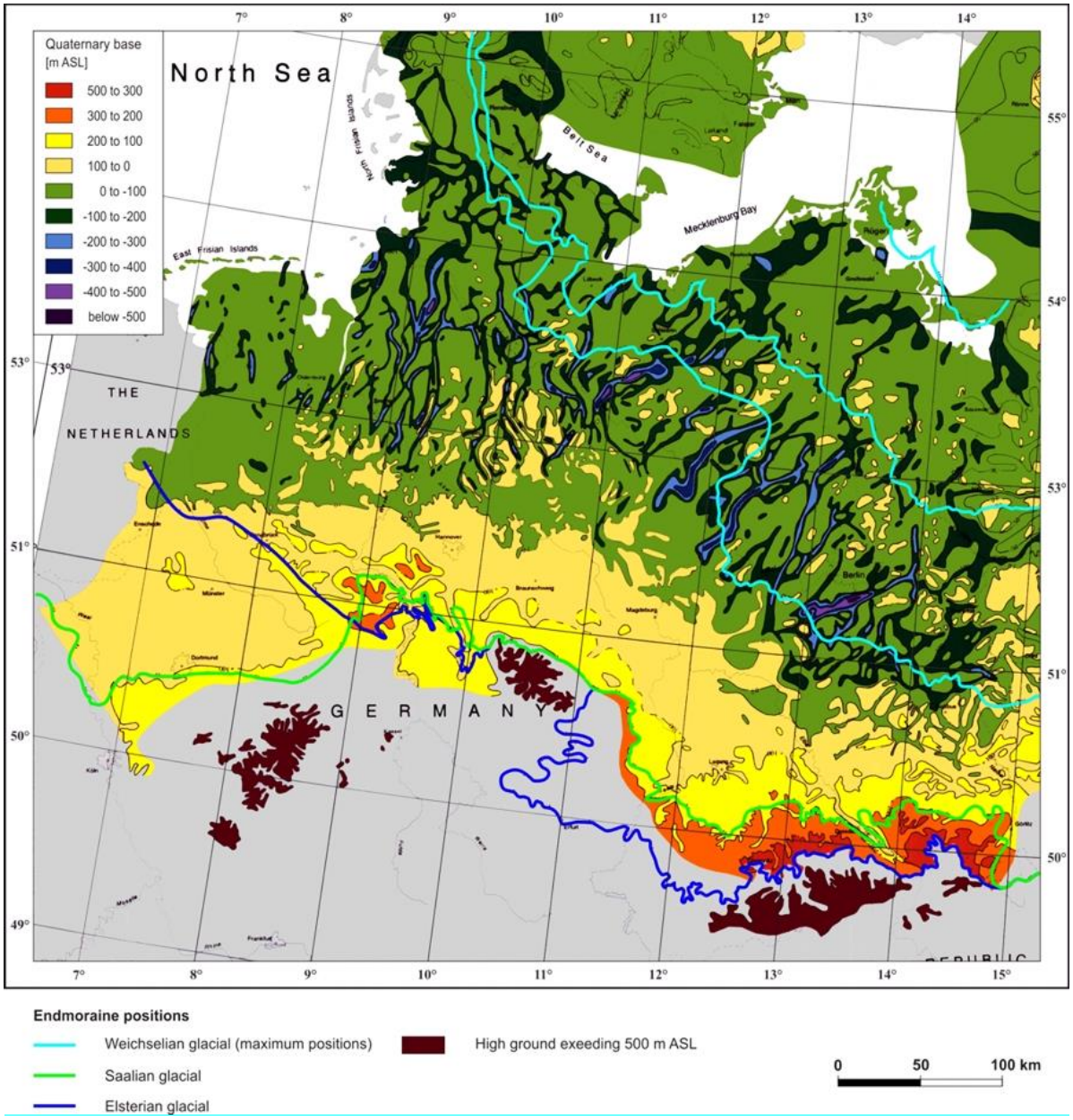

Figure 4.11: Tunnel-valley distribution and edges of the ice sheets in North Germany [modified after Stackebrandt 2001]

Figure 4.12 shows the glacial channel at the Gorleben site. Its width above the salt dome at a depth of 100 m below sea level is around 2 – 4 km. And it extends down to approx. 300 m below sea level.

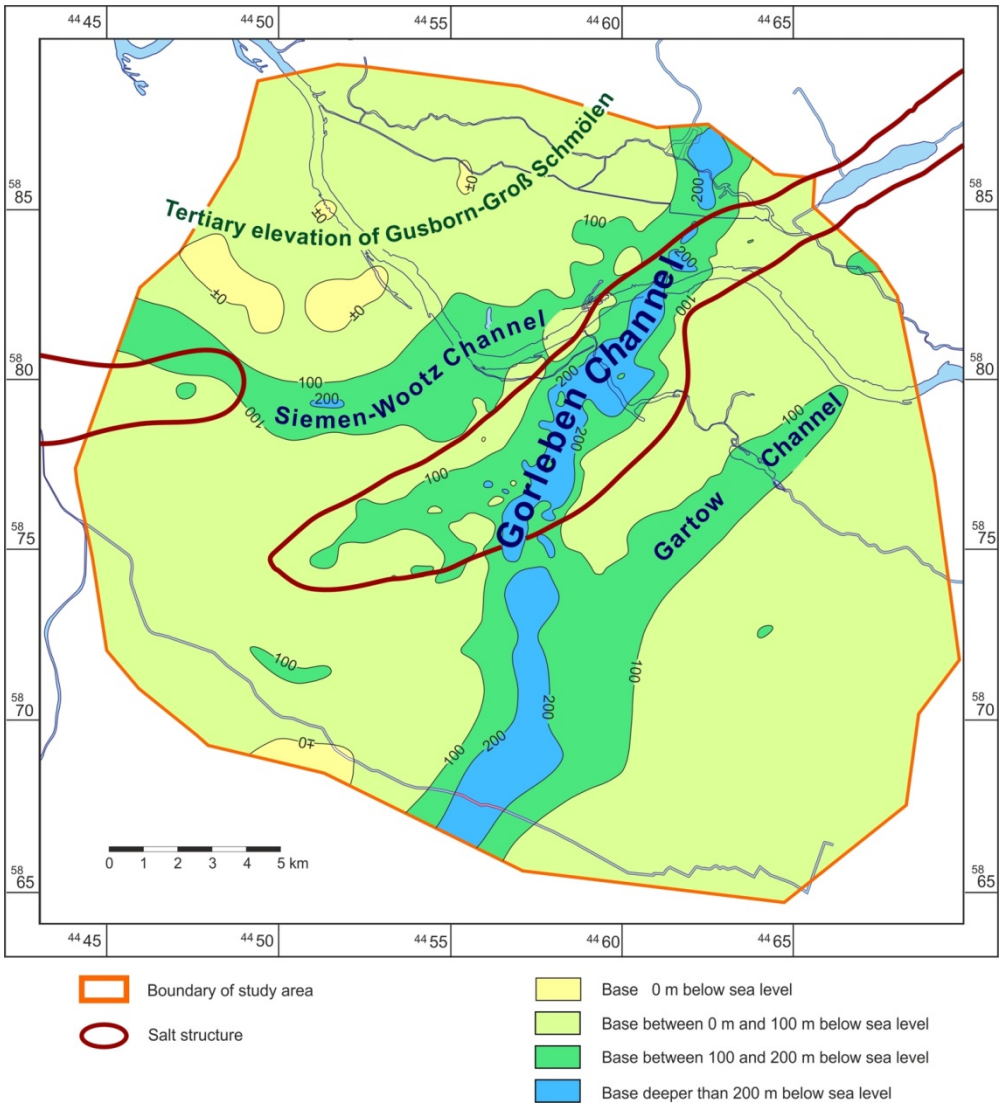

Figure 4.12: Depth contour of the Quaternary base and Quaternary channels [modified after Köthe 2007]

The examples show that the erosion is a process which is directly influenced by the climate and that it affects the geosphere in a typical way depending on it.

Developments which are directly caused by the climate are permafrost and the glaciation. The formation of permafrost in the soil or rocks has a major impact on their internal processes. And as a consequence it has strong effect on hydrogeological conditions like the groundwater flow and chemistry. A glaciation changes the hydrogeological conditions too and it also causes mechanical effects depending on the ice thickness.

For a scenario development it is necessary to generate a likely or a band of likely climate evolutions for the future. It is needed to assess the effects of the climate influenced geologic prosesses.

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 and its consequences shall be reduced by administrative measures and by the repository design 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 subsidence 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[7] 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)

---

[7] 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 into account the foreseen safety distance of at least 50 m from the emplacement fields to the bounding salt layers (including Carnallite) around Staßfurt Hauptsalz, 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.

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 [ABBergV 2009], the nuclear energy act [AtG 2010] and the Safety Requirements [BMU 2010]. 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.

Apart from the regulatory requirements mentioned above, the following fundamentals are very important for the development of the repository layout and design at the Gorleben site:

- The description of the geological situation (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 chapter 4.3.3 [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 4.2.3) 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® and CASTOR® casks in horizontal drifts.
  In addition – for comparison only – the emplacement of all heat-generating radioactive waste in transport and storage casks 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 will not be described in detail. They 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. 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 Radioactive Waste Inventory

Design and operation of a repository for high level waste (HLW) are significantly determined by the inventory of the radioactive waste and the disposal casks. For that reason, detailed information about the waste inventory and about the casks is necessary. Furthermore, this information is necessary for the evaluation of the repository safety.

Within the framework of this project, the waste quantities and the corresponding number of casks for the disposal in a HLW repository were estimated. For this purpose, the radioactive waste with negligible heat generation that will not comply with the waste acceptance criteria for the Konrad repository from the present point of view were identified from information of the waste producers as answer to the request of BfS for the Preliminary Safety Assessment for the Gorleben Site (VSG) and from own research. The casks which seem to be suitable for this type of radioactive waste were selected. Furthermore, the quantities of spent nuclear fuel from Nuclear Power Plants (NPP), spent nuclear fuel from prototype and research reactors,

and radioactive waste from reprocessing were estimated based on information from the Joint Convention report [IAEA 1997], information from the waste producer, information from the BfS and information from own research. In addition to this, the material inventory of the radioactive waste, spent fuel, and the casks was complied.

The streams of radioactive waste with negligible heat generation that will not comply with the waste acceptance criteria for the Konrad repository are:

- 200-l-drums conditioned at the former Karlsruhe nuclear research center before 1995,
- graphite/carbon bricks from high temperature and research reactors,
- depleted Uranium from the Uranium enrichment plant at Gronau,
- 560-l-drums with cemented fission product solution from the reprocessing of research reactor fuel elements in Dounreay (UK), and
- structural components from spent fuel from NPPs.

At the Karlsruhe site, 590 200-l-drums were processed before the waste acceptance criteria for the Konrad repository were developed in 1995. They will not comply with the waste acceptance criteria for different reasons, e.g. fissible material content > 50 g per waste package or absence of a database for reliable comparison with the waste acceptance criteria. For the 200-l-drums overpacks of Konrad Container Type IV of concrete or heavy concrete or Concrete Packaging Type I were selected.

Graphite and carbon brick were used as reflector and/or moderator material in research reactors and in both high-temperature reactors (AVR and THTR-300). The graphite/carbon brick material contains C-13 and impurities. Due to the activation of C-13 and the impurities, the graphite/carbon brick contain significant amounts of C-14, H-3 and Co-60. The activity of the graphite/carbon brick of the AVR is known. For the THTR-300 and the research reactors it was estimated that the activities are lower than the activities for the AVR due to the shorter operational phase and the lower neutron flux. The graphite/carbon brick of the AVR should be packed in 200-l-drums. It was assumed that these drums will be packed in Konrad Container Type IV made of concrete. For the graphite from THTR-300 and research reactors, Konrad Container Type VI made of concrete were selected.

Depleted uranium, so-called uranium tails, arises when uranium is enriched at the uranium enrichment plant at Gronau. Uranium tails are declared as resource, but there is no possibility for its use. For that reason it is not excluded that the uranium tails have to be declared as radioactive waste and have to be disposed of. In total 320,000 Mg of uranium tails in the chemical compound U_3O_8 have to be considered containing 3.4×10^{15} Bq of U-238. For the Konrad repository a maximum activity of 1.9^{12} Bq of U-238 was approved. The 3.4×10^{15} Bq overshoot the maximum activity and the uranium tails cannot be disposed of in the Konrad repository. For the uranium tails the Konrad Container Type VI of steel seems to be suitable.

From the reprocessing plant in Dounreay (UK) Germany has to revoke approximately 200 pieces of 560-l-drums with cemented fission product solution. The 560-l-drums do not comply with the waste acceptance criteria for the Konrad repository. If the 560-l-drums were packed in suitable accident-proof casks with waste cask class II a disposal in the Konrad repository is possible. A suitable cask seems to be the modified cast-iron packaging type II. The modified cast-iron packaging type II has to be licensed for the Konrad repository. It is unkown whether this is possible. Therefore it is assumed to declare this waste stream to be not suitable for the Konrad repository.

A fuel element consists of the fuel rods and the structural components. The disposal of spent fuels of NPPs is provided by the direct disposal in POLLUX® casks or alternatively by fuel rod canisters (so called BSK). Only the fuel rods were packed in the POLLUX® casks and BSK. The structural components have to be disposed of separately. The structural components consist of steel with different cobalt contents. Due to the neutron flux in the reactor the cobalt was activated to Co-60. Co-60 is one of the reference nuclides for the accident analyses in the waste acceptance criteria for the Konrad repository. It is assumed that the value of Co-60 will exceed the values for a possible disposal in the Konrad repository. Therefore the structural components from 10,550 t of heavy metal (thm) of spent fuel have to be considered here. For the structural components the cast-iron packaging Type II with 120 mm lead shielding seems to be suitable.

Table 4.1 shows the waste quantities for the radioactive waste with negligible heat generation and the corresponding number of casks and their total volume.

Table 4.1: Waste and corresponding waste casks (thm = tons of heavy metal)

Waste category	Quantity of drums and waste	Cask Type	Number of waste packages	Volume [m³]
200-l-drums Karlsruhe Site	590 drums	Konrad Container Type IV or Concrete Packaging Type I	74 or 590	578 or 708
Graphite/carbon brick from high temperature and research reators	225[1] Mg and 700 Mg[2]	Konrad Container Type IV and Konrad Container Type VI	127 and 200	940 and 1,080
Depleted uranium	320,000 Mg	Konrad Container Type VI	20,834	112,504
560-l-drums	200 drums	modified cast-iron packaging type II	20	320
Structural components	From 10,550 thm	cast-iron packaging type II	2,638	3,430

[1]: AVR; [2]: THTR-300 and research reactors

After closure of the Konrad repository radioactive waste with negligible heat generation will accrue and has to be considered in a HLW repository. That waste consists of large components at the Jülich and Greifswald sites and annually approximately 200 to 300 m³ of waste from research and industry.

For the disposal of spent nuclear fuels from NPPs and prototype and research reactors and radioactive waste from reprocessing of spent nuclear fuels from NPPs three different disposal concepts were considered: Drift disposal of POLLUX® casks and transport and storage casks (variant B1), drift disposal of transport and storage casks (variant B2), and vertical borehole disposal of BSK, Triple-Packs, and modified BSK (variant C).

The waste quantities of spent fuel from NPPs until end of 2009 were determined based on the data of the joint convention report [IAEA 1997]. The waste quantities from 2010 until the shutdown of the last NPPs in 2022 were estimated based on the run-times corresponding to the nuclear energy act, the specific unload quantities of each NPP corresponding to data from BfS and [BMUB 2015a], [BMUB 2015b]. Table 4.2 shows the estimated quantities of spent fuel and tHM from NPPs and their corresponding amount of disposal casks and volumes for each disposal variant.

Table 4.2: Spent fuel from power reactors and waste containers (thm = tons of heavy metal)

NPP Type	Quantity of fuel elements	thm	Disposal variants	Cask Type	Number of waste packages	Volumina [m³]
PWR	13,285	6,960	B1	POLLUX®-10	1,329	14,021
			B2	CASTOR® V/19	699	19,432
			C	BSK	4,428	3,188
BWR	17,279	3,007	B1	POLLUX®-10	576	6,077
			B2	CASTOR® V/52	332	8,599
			C	BSK	1,920	1,382
WWER	5,000	583	B1	POLLUX®-10	200	2,110
			B2	CASTOR® 440/84	65	1,476
			C	BSK	667	481
Total	35,564	10,550	B1	POLLUX®-10	2,105	22,208
			B2	CASTOR®	1,096	29,507
			C	BSK	7,015	5,051

Most of the spent fuel from prototype and research reactors was reprocessed. The amounts of spent fuel which have to be considered for final disposal were determined based on information from BfS. Table 4.3 contains information about spent fuel elements and rods from prototype and research reactors for final disposal, the corresponding number of disposal casks and volumes for each disposal variant.

Table 4.3: Spent fuel from prototype and research reactors and corresponding casks

Reactor type	Quantity of fuel elements	Disposal variants	Cask Type	Number of waste packages	Volume [m³]
AVR	288,033	B1,B2	CASTOR® THTR/AVR	152	632
		C	BSK THTR / AVR (FE-sphere) or BSK THTR / AVR (FE-canister)	152 or 76	110 or 147
THTR-300	617,606	B1,B2	CASTOR® THTR / AVR	305	1,296
		C	BSK THTR / AVR (FE-sphere) or BSK THTR / AVR (FE-canister)	305 or 153	220 or 297
KNK II and Otto-Hahn	2,484 (KNK) and 52 (Otto Hahn)	B1,B2	CASTOR® KNK	4	17
		C	BSK KNK	5	3
BER II	287	B1,B2	CASTOR® MTR 3	9	24
		C	BSK MTR	12	9
FRM II	190	B1,B2	CASTOR® MTR 3	38	100
		C	BSK MTR	38	27
FRMZ	89	B1,B2	CASTOR® MTR 3	1	3
		C	BSK MTR	1	1
RFR	950 (FE) and 16 (FR)	B1,B2	CASTOR® MTR 2	18	47
		C	BSK MTR	10	7
Total		B1,B2	CASTOR®	527	2,092
		C	BSK	523[1] or 295[1]	337[1] or 491[2]

[1]: with BSK THTR/AVR$_{FE\text{-}sphere}$
[2]: with BSK THTR/AVR$_{FE\text{-}canister}$

Three different types of radioactive waste from reprocessing of spent fuel from NPPs have to be considered:

- vitrified high radioactive fission products and feed clearing sludges (CSD-V),
- intermediate level radioactive vitrified waste (CSD-B), and
- compacted intermediate level radioactive fuel element casings, structural components and technology waste (CSD-C).

From the Karlsruhe vitrification facility (VEK) and from Sellafield LTD (former British Nuclear Fuels BNFL) only CSD-V have to be considered. From AREVA-NC (former COGEMA) CSD-V, CSD-B, and CSD-C have to be considered. Table 4.4 lists the amounts of waste from reprocessing, the corresponding number of disposal casks and volumes for each disposal variant.

Table 4.4: Reprocessing waste and corresponding casks

Waste stream	Conditioning plant	Quantities of canisters (CSD)	Disposal variants	Cask Type	Number of waste packages	Volume [m³]
CSD-V	AREVA-NC	3,024	B1	POLLUX®-9	336	3,545
			B2	TLB[1]	108	3,189
			C	Triple-Pack	1,008	726
	Sellafield LTd.	565	B1	POLLUX®-9	63	665
			B2	TLB[1]	21	620
			C	Triple-Pack	189	136
	VEK	140	B1	POLLUX®-9	16	169
			B2	TLB[1]	5	148
			C	Triple-Pack	47	34
		Total: 3,729	B1	POLLUX®-9	415	4,379
			B2	TLB[1]	134	3,954
			C	Triple-Pack	1,244	896
CSD-B	AREVA-NC	140	B1	POLLUX®-9	16	169
			B2	TLB[1]	5	142
			C	Triple-Pack	47	34
CSD-C	AREVA-NC	4,104	B1	POLLUX®-9	456	4,811
			B2	TGC 36	114	3,409
			C	Triple-Pack	1,368	985
Total		7,973	B1	POLLUX®-9	887	9,359
			B2	TLB[2]	254	7,505
			C	Triple-Pack	2,659	1,915

[1]: CASTOR® HAW 20/28 CG, TS 28 V, TN 85, CASTOR® HAW 28 M
[2]: like 1 with TGC 36

From the different waste streams and their corresponding waste casks material inventories were derived as metals, organics, and others like graphite, concrete, and water. The knowledge of this inventory is necessary for the assessment of repository safety. For the spent nuclear fuel from NPPs and prototype and research reactors and for the radioactive waste from reprocessing of spent fuel from NPPs, the material inventories were compiled for the three different types of disposal concepts. Additionally, the overall inventory for all waste streams and disposal concepts was determined.

4.3.4 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.

4.3.4.1 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.).

4.3.4.2 Disposal variants

The first step for developing an emplacement concept is the selection and description of waste-specific (Spent nuclear fuel (SNF) / reprocessing waste) packages for the emplacement variants on the basis of the waste inventory [Dörr 2013]. 3D thermal calculations were carried out (see chapter 4.3.5) 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:

4.3.4.2.1 Variant A: Disposal of ILW in disposal chambers

The total inventory of intermediate level waste (ILW) will be disposed of in different types of casks (see table 4.1) in disposal chambers in a separate disposal area of the repository. Due to the negligible heat generation of the ILW casks, the chamber distances can solely be based on technical aspects. The chamber distances are based on the park position (alcove) for the fork lift that is needed when stacking the waste packages.

Figure 4.13 shows the contour and plan view of the disposal chambers for the disposal of Konrad Containers Type IV [Bollingerfehr et al. 2011].

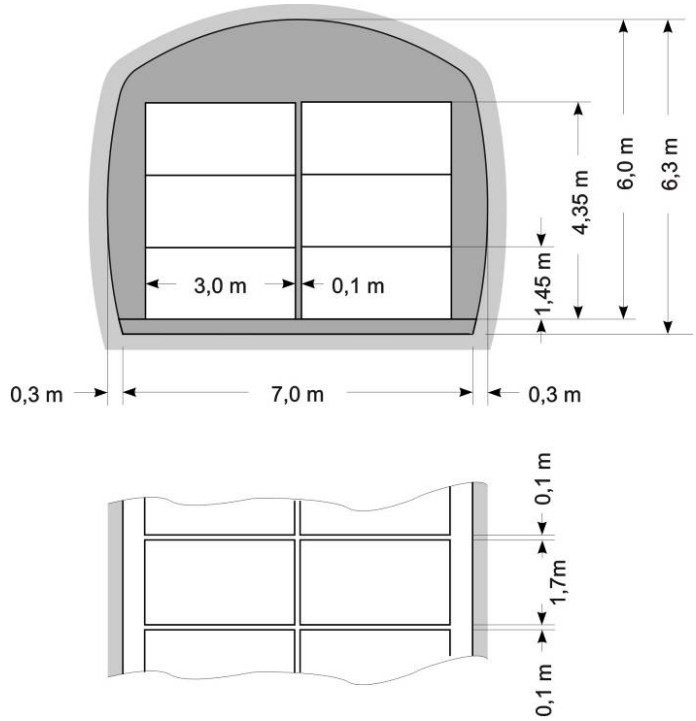

Figure 4.13: Contour and plan view of the disposal chamber for the disposal of Konrad Containers Type IV

Figure 4.14 shows the contour and plan view of the disposal chambers for cast-iron packagings type II [Bollingerfehr et al. 2012].

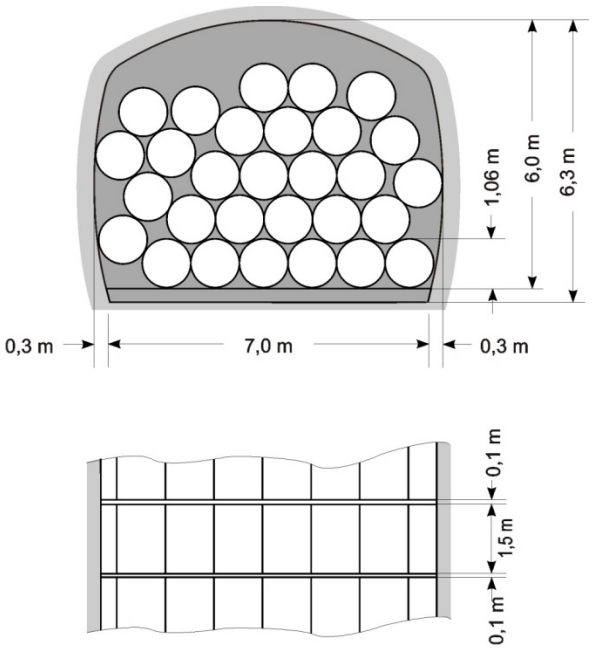

Figure 4.14: Contour and plan view of the dipsosal chamber for the disposal of iron-cast packagings Type II

A potential disposal field for ILW, the layout being developed during VSG, is shown in Figure 4.15 [Bollingerfehr et al. 2012].

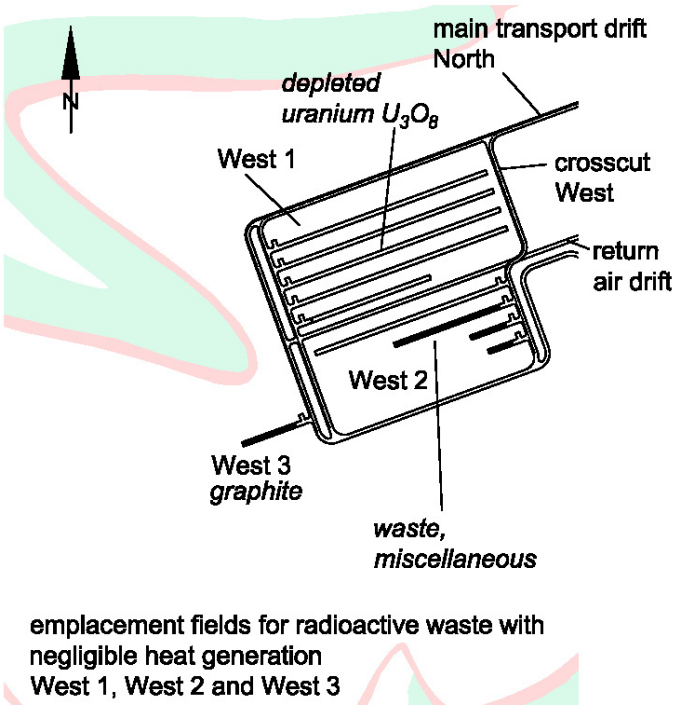

Figure 4.15: Disposal field for ILW [Bollingerfehr et al. 2012]

4.3.4.2.2 Variant B1: Disposal of POLLUX® in horizontal drifts

The total inventory of heat-generating waste will be disposed of in heavy (weight max. 65 metric tons), self-shielding POLLUX® 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® casks with spent fuel from NPPs, 887 POLLUX® casks with reprocessing waste, and 527 containers (specific CASTOR® 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 4.3.4.3) 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 4.16. 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) [Bollingerfehr et al. 2012].

Figure 4.16 also shows a sketch of the POLLUX® 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 [Filbert 1995].

4.3.4.2.3 Variant B2: Disposal of transport and storage casks in horizontal boreholes

For the purpose of comparison with Variant B1, the emplacement of all heat-generating waste in self shielding transport and storage casks 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.

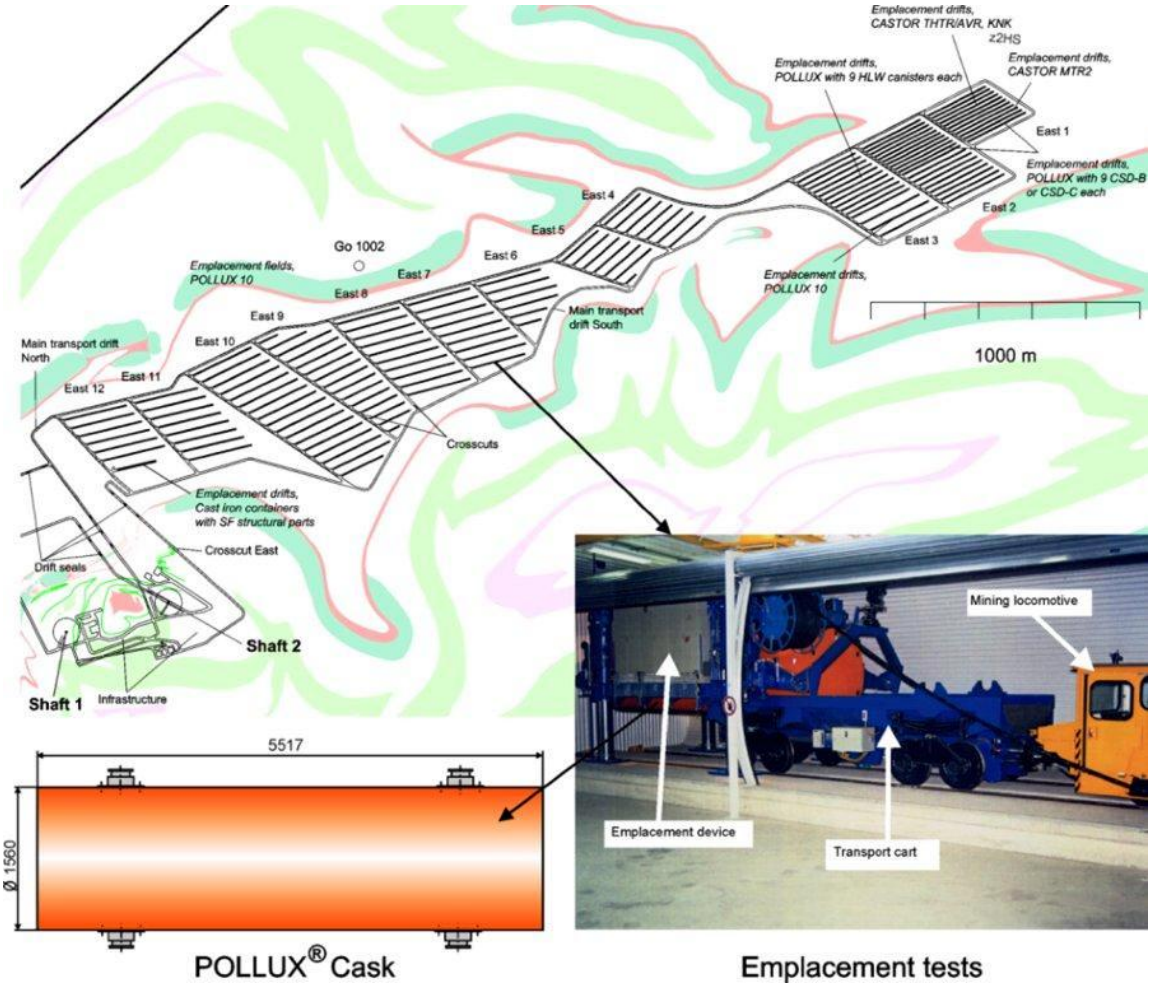

Figure 4.16: Repository design for variant B1: Emplacement of all heat-generating spent fuel and reprocessing waste in POLLUX®-casks in horizontal drifts (adjusted to the assumed geologic structure of the Gorleben salt dome)

Figure 4.17 shows the emplacement situation underground (plan view) and the main components.

The footprint of this repository mine will be similar to that of variant B1 (POLLUX® 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.

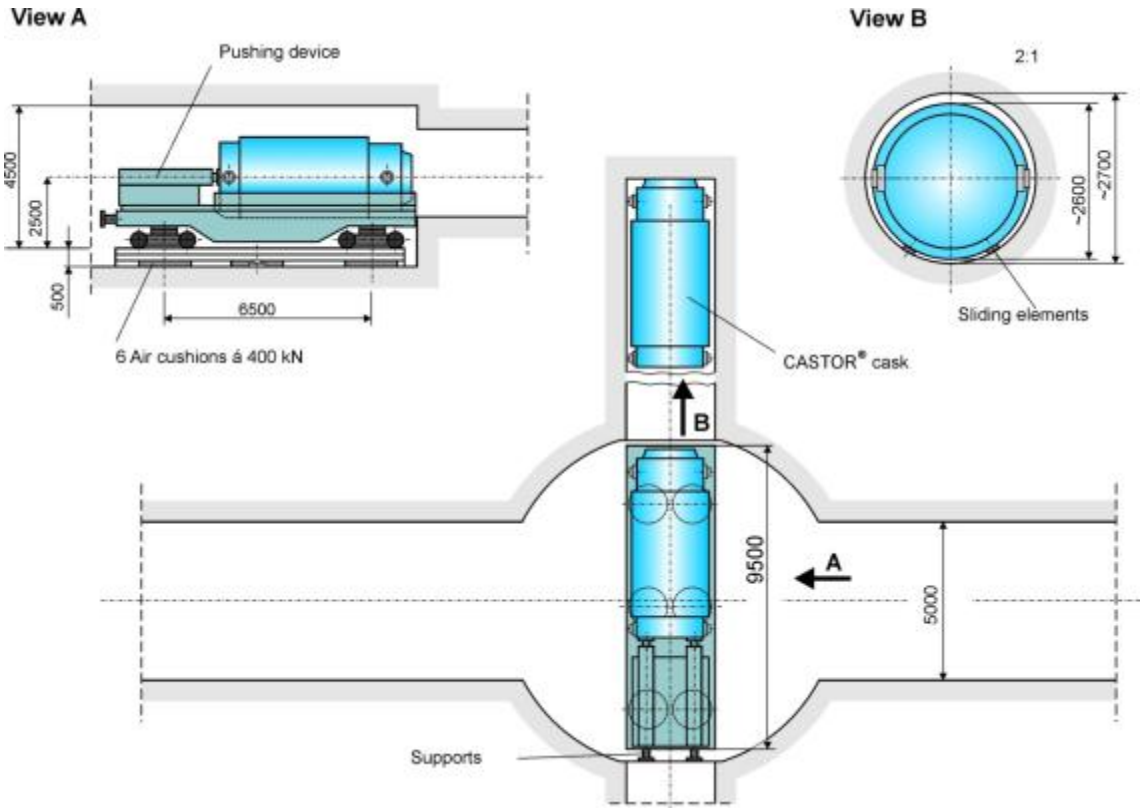

Figure 4.17: For comparison with variant B1: Disposal of transport and storage casks in horizontal boreholes (variant B2)

4.3.4.2.4 Variant C: 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 lining 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 lining and lid provide additional technical barriers. At the same time, they facilitate access to the canisters in case they need to be retrieved (see chapter 4.3.6). 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 4.18. 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 4.18 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].

4.3.5 Thermo-Mechanical Design Calculations

4.3.5.1 Fundamentals

In appendix B1, the main deformation processes in rock salt are described. Dislocation creep is the main process in dry rock salt. If only the stationary behaviour is considered, a mathematical formulation can be derived that consists of a purely stress-dependent part, a purely temperature-dependent part, and a combined temperature-stress dependent part. If only a restricted range of stress and temperature values is taken into account, the complex description can be approximated by a simplified approach that consists of a multiplicative split of a stress-dependent and a temperature-dependent part. The temperature-dependent part is usually described by an Arrhenius approach. Various approaches exist for the stress-dependence. They can be based on a potential approach, an exponential approach, or a hyperbolic approach. An abbreviated description from a mathematical perspective is given in Table 4.5 and their characteristic behaviour is shown in Figure 4.19.

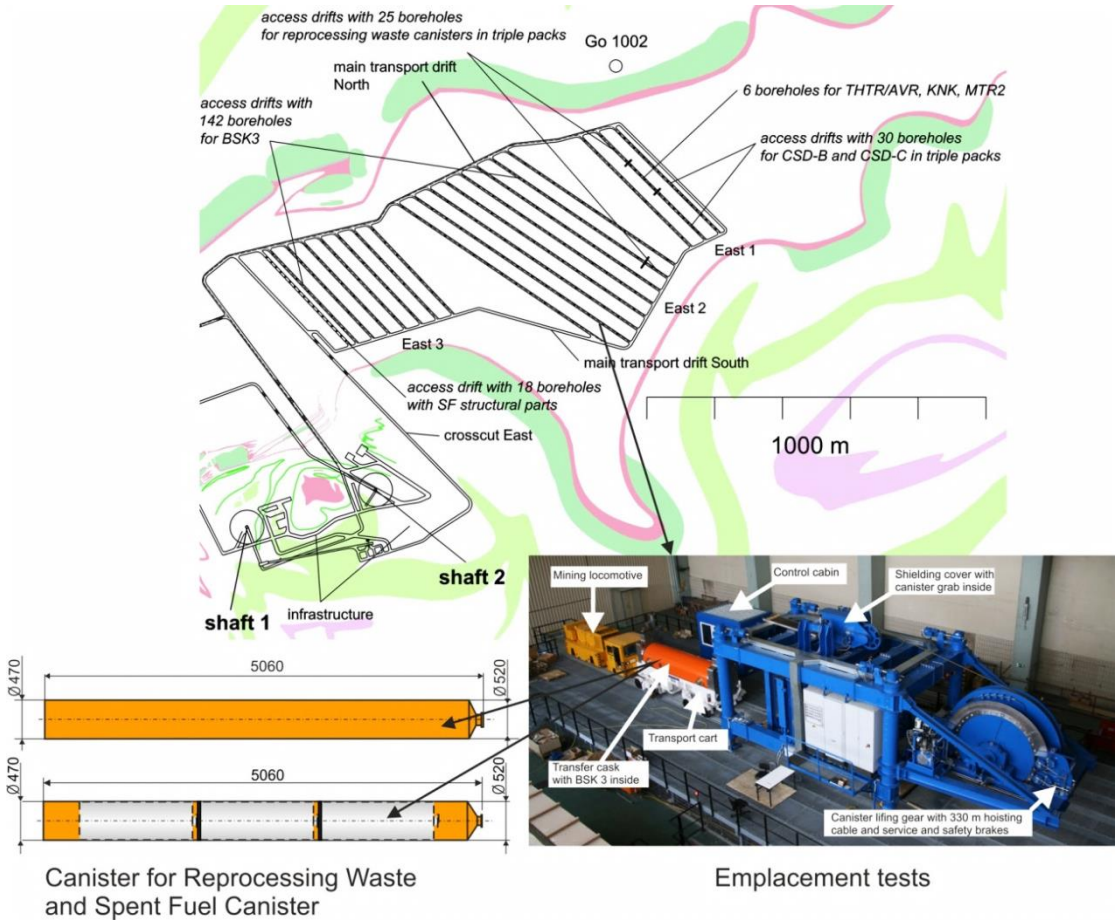

Figure 4.18: **Repository design for variant C: Emplacement of all heat-generating waste in lined vertical boreholes (adjusted to the assumed geologic structure of the Gorleben salt dome)**

Table 4.5: **Model of stationary creep for constitutive laws**

Abbreviation	Function	Remark
Pow	$\dot{\varepsilon} = A_1\ \hat{\sigma}^{\alpha_1}$	Linear in log-log-plot
Exp1	$\dot{\varepsilon} = A_2\ exp(\alpha_2\ \hat{\sigma})$	Strain rate at stressless state, steep increasing strain rate at higher stress level, nearly constant rate of strain rate at low stresses
Exp2	$\dot{\varepsilon} = A_3\ exp(\alpha_3\ \hat{\sigma})\ \hat{\sigma}$	No strain rate at stressless state, comparable with SinH, steep increasing strain rate at higher stress level, nearly constant rate of strain rate at low stresses
SinH1	$\dot{\varepsilon} = A_4\ sinh(\alpha_4\ \hat{\sigma})$	steep increasing strain rate at higher stress level, nearly constant rate of strain rate at low stresses
SinH2	$\dot{\varepsilon} = A_5\ sinh(\alpha_5\ \hat{\sigma})\ \hat{\sigma}$	Adaptable to power law as well as exponential model

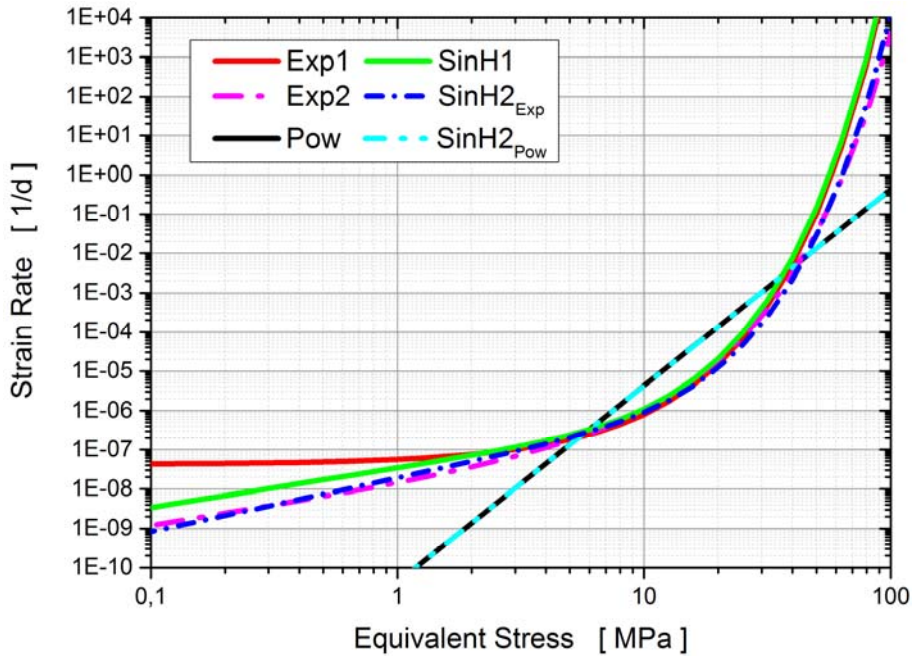

Figure 4.19: Strain rate versus equivalent stress for different models

Figure B1 in appendix B1 shows a comparison between a hyperbolic and a potential approach. The range of values analysed comprised temperatures between 50 °C and 200 °C and effective stresses according to von Mises between 2 MPa and 30 MPa. At low effective stresses, however, the behaviour was analysed at high temperatures only. The results show that in the range of values analysed, the adaptation using the hyperbolic approach and the adaptation using the potential approach yield similar results.

Appendix B2 contains the results of further, more recent analyses. However, these analyses were carried out on salt of the WIPP site. One characteristic of the rock salt at the WIPP site compared with the Gorleben site is its higher moisture content so that a mechanism like FADT is more evident. Neither the influence of moisture nor the influence of very low effective stress have been fully investigated. Thus, the following analyses are restricted to the behaviour of rock salt with low moisture content. Stationary creep is dominant for the long-term behaviour. As a further restriction, both transient processes as well as fracturing are not taken into account.

From the perspective of mechanics, the effects to be considered are distinguished by their reversibility (elastic / inelastic), their temporal correlation (spontaneously / temporally delayed), and their geometric effect (isotropic / deviatoric). The description of the constitutive model is based on an additive decomposition of the tensor of the strain rate into its elastic and inelastic proportion. The elastic proportion describes instantaneous behaviour and contains both an isotropic and a deviatoric part. The inelastic proportion describes behaviour that is temporally delayed and only contains a deviatoric part. Thus, in undamaged state, a change in volume is due to elastic behaviour only. All constitutive models to be analysed

describe elastic behaviour by means of a linear-elastic behaviour according to Hooke. For the aspect considered here, the differences between the constitutive models are in the inelastic behaviour. The various formulation alternatives have already been described above.

The following gives an abbreviated mathematical description: A comparison of undamaged, stationary creep of the constitutive models according to Günther et al. 2007], [Hou et al. 2007], BGRa, BGRb, and WIPP [ITASCA 2016] shows that the first two formulations are identical. They can be classified as model-rheologic models; their stress-related parts correspond to formulation Exp2. The three latter models can be allocated to the structure-rheologic models. Constitutive model BGRa corresponds to the stationary part of the WIPP model. The approach corresponds to a potential approach. Contrary to BGRa, BGRb consists of two summands. All models can be described in the following manner:

$$\dot{\boldsymbol{\varepsilon}} = \dot{\boldsymbol{\varepsilon}}_{\boldsymbol{el}} + \dot{\boldsymbol{\varepsilon}}_{\boldsymbol{inel}} = \mathbf{C}\,\dot{\boldsymbol{\sigma}} + \frac{3}{2}\dot{\varepsilon}_{inel}\,\frac{\mathbf{S}}{\hat{\sigma}}$$

where

C: fourth order elastic tensor
S: tensor of deviatoric stress, $\boldsymbol{\sigma} = \sigma_0\,\mathbf{I} + \mathbf{S}$
I: unit tensor
$\dot{\boldsymbol{\varepsilon}}$: strain rate tensor
$\dot{\boldsymbol{\varepsilon}}_{\boldsymbol{el}}$: elastic strain rate tensor
$\dot{\boldsymbol{\varepsilon}}_{\boldsymbol{inel}}$: inelastic strain rate tensor
$\dot{\varepsilon}_{inel}$: amount of inelastic strain rate tensor
$\boldsymbol{\sigma}$: stress tensor
$\dot{\boldsymbol{\sigma}}$: stress rate tensor
σ_0: first invariant of stress tensor, $\sigma_0 = 1/3\ \mathrm{tr}(\boldsymbol{\sigma})$
$\hat{\sigma}$: von Mises equivalent stress, $\hat{\sigma} = \sqrt{1/2\ \mathbf{S} : \mathbf{S}}$

The difference between the constitutive models lies in the formulation of the amount of inelastic strain rate tensor.
The models according to [Günther et al. 2007] and [Hou et al. 2007] are based on a Maxwell model with stress-dependent viscosity:

$$\dot{\varepsilon}_{inel,Mk-HL} = \frac{e^{-\frac{Q}{R}\left(\frac{1}{T}-\frac{1}{T_{Ref}}\right)}}{\eta_M}\,\hat{\sigma}$$

where

Q: activation energy
R: universal gas constant
T: temperature
T_{Ref}: reference temperature
η_M: Maxwell viscosity at reference temperature

Constitutive models BGRa and BGRb are based on a potency approach:

$$\dot{\varepsilon}_{inel,BGRa} = A\, e^{-\frac{Q}{RT}}\, \hat{\sigma}^n$$

$$\dot{\varepsilon}_{inel,BGRb} = \left(A_1\, e^{-\frac{Q_1}{RT}} + A_2\, e^{-\frac{Q_2}{RT}}\right) \hat{\sigma}^n$$

where

A_1, A_2: structural factor

The assumption in BGR_b is that two sub-processes from different temperature regions are involved in the deformation process. Figure 4.20 shows the behaviour of the BGR constitutive models based on the parameters A=0.18 1/d, Q=54 kJ/mol, A_1=0.23*10^{-3} 1/d, Q_1=42 kJ/mol, A_2=2.1*10^6 1/d and Q_2=113.4 kJ/mol. Compared with BGR_b, BGR_{EB} contains a multiplier of 5.872 to adapt it to the specific rock salt facies z2HS at the Gorleben site [Nipp et al. 2003]. The components of BGR_b show that the parameters of the first proportion describe the behaviour at low temperatures while at high temperatures, the parameters of the second proportion become effective. The transition occurs in the temperature range of approximately 80 °C and 125 °C. Compared with this, constitutive model BGR_a covers only one temperature range. A comparison between BGR_a and BGR_b shows that the parameters have been determined for different rock salt facies. A comparison with the factorized approach BGR_{EB} shows that BGR_a was used as simplifying approximation in a temperature range to up to 120 °C. For higher temperatures, a modification of the activation energy is necessary. The activation energy also concerns other constitutive models that consider a constant activation energy.

4.3.5.2 Thermo-mechanical Calculations as Basis for Repository Design

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.

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.

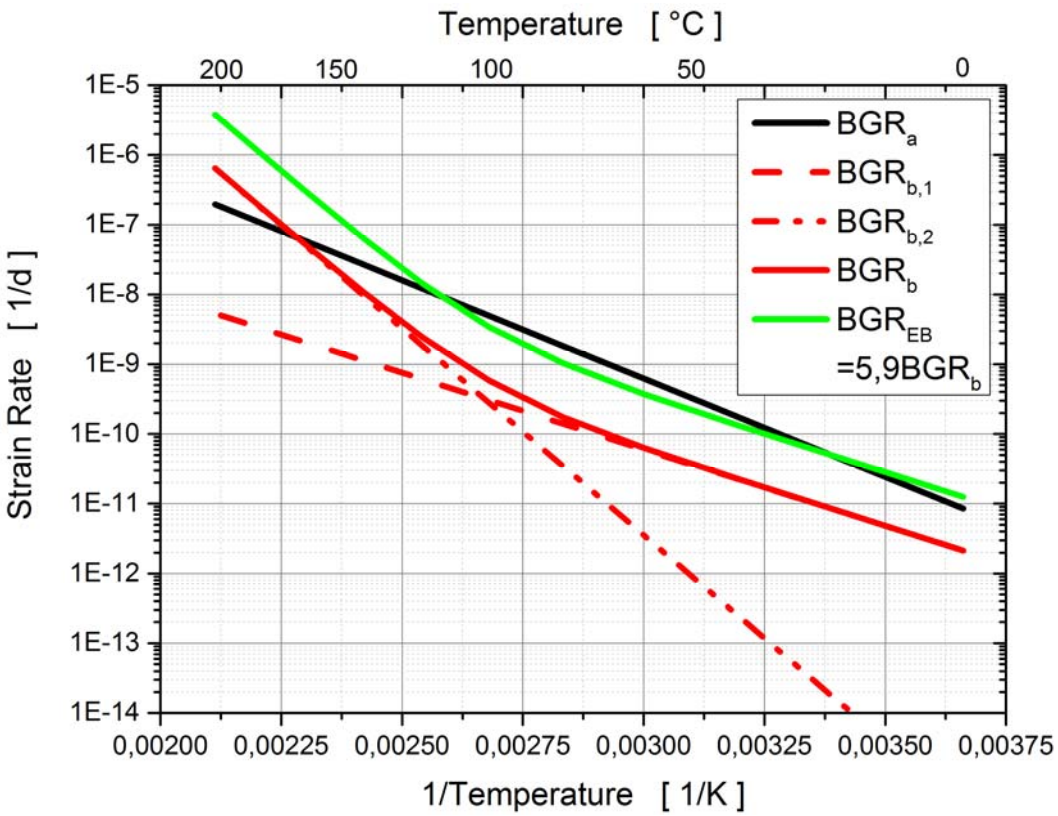

Figure 4.20: Structural factor and Arrhenius term as part for BGR constitutive laws

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 at a temperature of 25°C. 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) thermomechanical 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 thermomechanical calculations. LinSour was used in the third step. LinSour is an analytic code for 3D thermal calculations of heat conductivity based on line sources. Figure 4.21 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 4.22. 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).

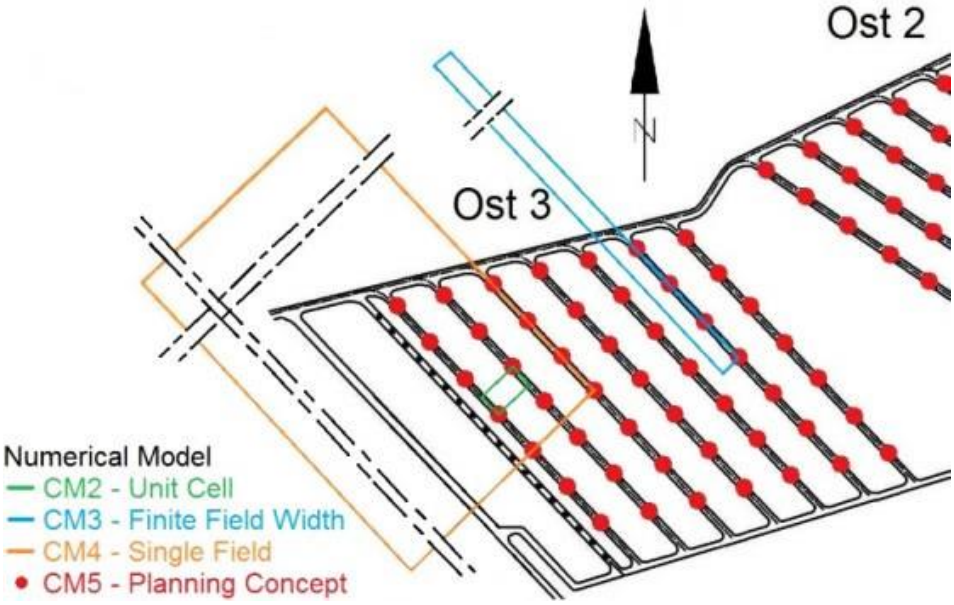

Figure 4.21: **Borehole disposal – Section of the emplacement concept and scheme of appropriate numerical models** [Bollingerfehr et al. 2012]

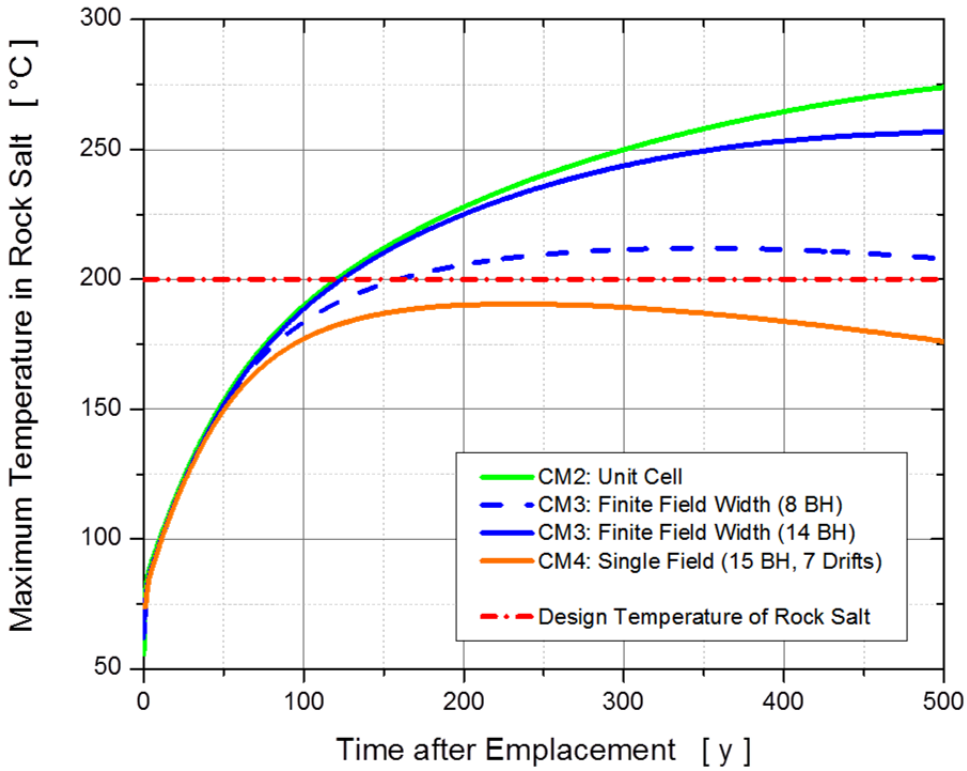

Figure 4.22: Borehole disposal – Maximum temperature influenced by the boundary condition within the numerical models [Bollingerfehr et al. 2012]

Figure 4.23 and Figure 4.24 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 (Figure 4.23 and Figure 4.24). 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.

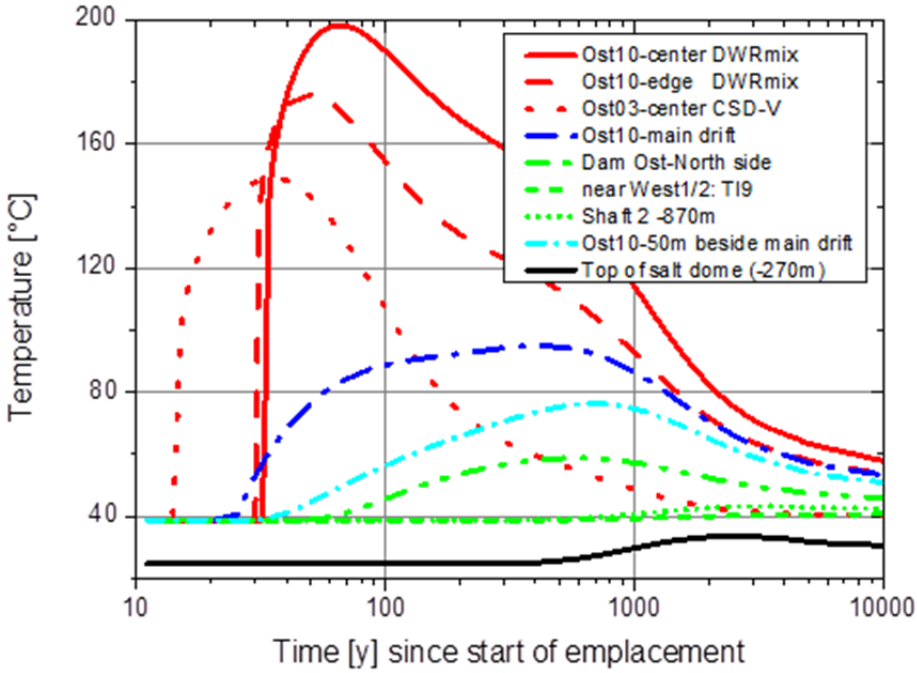

Figure 4.23: Variant 1 (drift disposal) – Temperature over time at different points in the repository [Bollingerfehr et al. 2012]

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.

4.3.5.3 Disturbance of Emplacement Procedure

The design of a repository is based on uninterrupted emplacement operation. The following is a study on how to handle an interruption in the emplacement operation. It is based on the initial concept for the emplacement of HLW in long, vertical boreholes, that was developed before BMU (2010) stipulated the requirement for retrievability. To facilitate salt creep (convergence) and thus rapid enclosure of the waste containers, borehole lining was not anticipated in this concept. The residual void space around the canisters emplaced in the boreholes is to be backfilled with crushed salt.

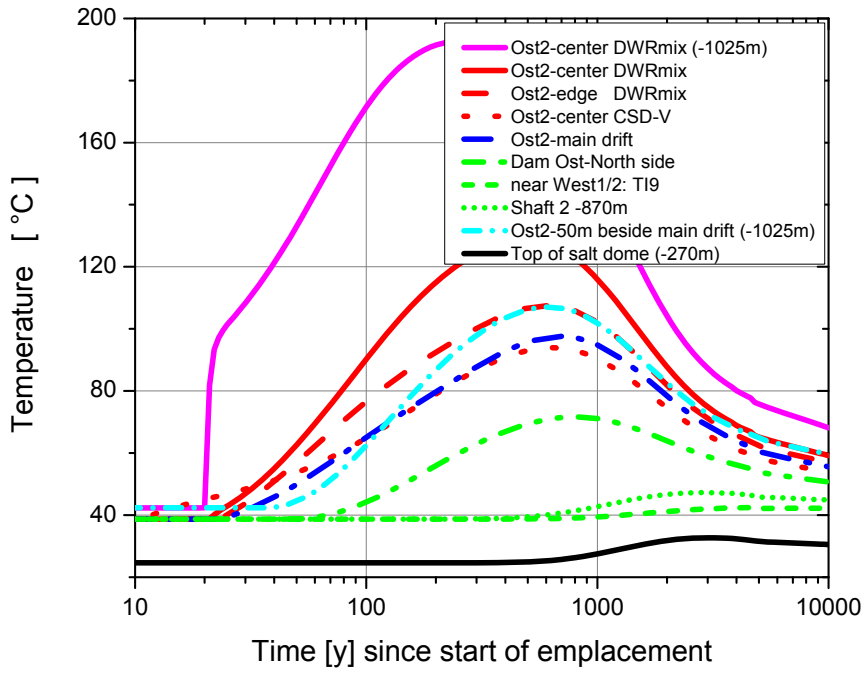

Figure 4.24: **Variant 2 (borehole disposal) – Temperature over time at different points in the repository [Bollingerfehr et al. 2012]**

Only a short period of time after emplacement of the canisters is relevant for the study. Neighbouring boreholes thus do not have any impact on the problem. Hence, the geometric model consists of only one 300-m-deep borehole. The upper end of the borehole is assumed to be at a depth of 870 m. The borehole diameter is 0.6 m. The canisters are modelled with a length of 4.93 m and a diameter of 0.43 m. The thermal power of one canister corresponds to that of three mixed fuel elements DWR-Mix 89/11 after an interim storage time of 10 years [Bollingerfehr et al. 2012]. The temperature at the upper borehole end is 38.7 °C, the geothermal temperature gradient is 2.35 K/100m. The emplacement rate is one canister per day. After one canister has been emplaced, the annular space around the canister is backfilled, and the canister is covered with backfill up to a height of 0.5 m. Further parameters are listed in Table 4.6.

Table 4.6: Material properties for borehole emplacement

Property	Unit	Rock salt	Backfill material	Cask
Porosity	-	-	0.35-0.0	-
Density	kg/m^3	2,200	1,430-2,200	7,700
Young's mod.	GPa	25	1.4-25	205
Poisson	-	0.27	0.27	0.31
$\dot{\varepsilon}_{inel}$	1/d	4·$\dot{\varepsilon}_{BGRb}$	$\dot{\varepsilon}_{cwipp}$ [Bollingerfehr et al. 2012]	-
Conductivity (Temperature)	W/m/K	f(T): 5.6-3.2 (20 °C-200 °C)	f(η,T): 0.8, 5.6-3.2 (20 °C-200 °C)	15
Spec. heat	J/kg/K	864	562-864	600
Thermal Exp.	1E-5 1/K	4.2	4.2	1.2

If there is an interruption in the emplacement process following the emplacement and covering of one canister, it has to be decided what to do with the incompletely filled borehole. In addition to the natural convergence of the borehole, the thermal load in the part of the borehole that is not yet backfilled has to be assessed. Without any further measures, the temperature in the cover layer of the uppermost canister will increase and will then heat the air in the borehole. The longer the interruption of the emplacement process, the higher the temperature increase in the emplacement borehole. Thus, it was investigated how much the temperature would increase if the canisters were covered with an additional layer. The investigations concentrated on an additional cover of up to 10 m and an emplacement interruption of up to half a year (Figure 4.25 and Figure 4.26).

The results show that after an emplacement interruption of up to half a year, a partially backfilled borehole does not have to be discarded. If the uppermost canister is covered with a sufficiently high backfill layer, emplacement can be resumed after the interruption. For example, if the interruption is 3 months and the permissible temperature increase at the top of the cover is 1 K, the cover needs to have a thickness of approximately 6.0 m.

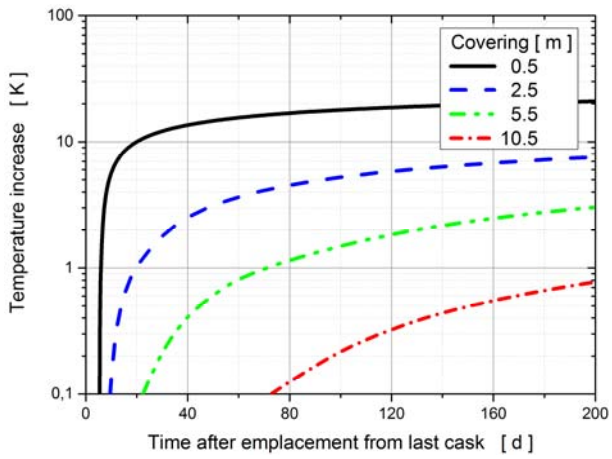

Figure 4.25: Temperature increase depending on interruption time for different ranges of covering

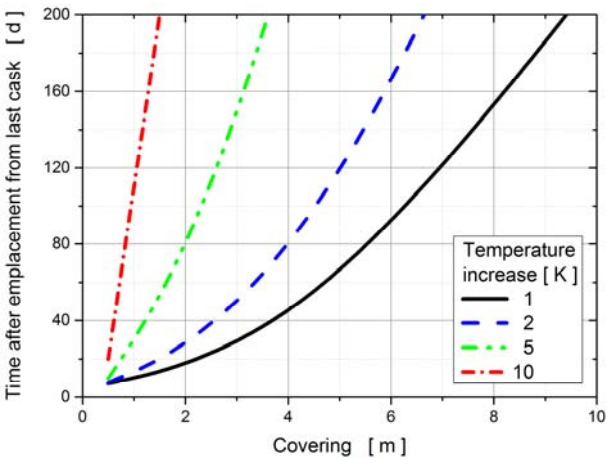

Figure 4.26: Maximum possible interruption time depending on cover for different temperature increases

4.3.5.4 Eccentricity of the canisters in the borehole

The existing designs are based on the assumption that the canisters are ideally located in a central position in the borehole. If no measures are taken to fix the canisters, they may shift into eccentric positions. Eccentric positions can be either skew or off-centre positioning of the canisters. The initial configuration corresponds to the one described in chapter 4.3.5.3. The material parameters correspond to those listed in Table 4.6.

Three variants have been investigated (Figure 4.27):

- V1: skew position with central canister bottom, φ_{max}=1°
- V2: complete skew position of the canister, φ_{max}=2°
- V3: off-centre position of the canister, e_{max}=8.5 cm

Due to skew or off-centre positioning, the thickness of the backfill increases on one side of the canister. At the same time, the canister approaches the rock salt on its other side. This effect is important because loose backfill material has a lower heat conductivity than the good heat conducting rock salt. The thermal design point within this emplacement concept generally is in the backfill material. Figure 4.28 shows the maximum temperature in the backfill material and additionally in the rock salt. In the studies carried out, the maximum temp-

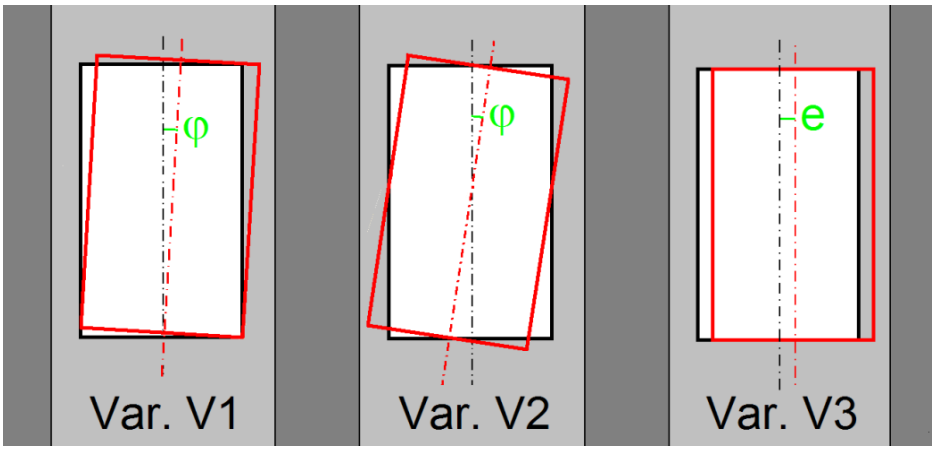

Figure 4.27: Variants of eccentric positions

erature is in the backfill material. With increasing eccentricity, the maximum temperature in the backfill material decreases and the maximum temperature in the rock salt increases. However, even in the worst case considered here, the temperature in the rock salt never reaches the same temperature as in the backfill material. Therefore, it would be necessary that the cask touches the wall. The case most disadvantageous for the design is thus the undisturbed variant. Regarding the thermal design, the modelling is thus conservative. Of the variants considered here, variant V1 is the one with the highest design temperature.

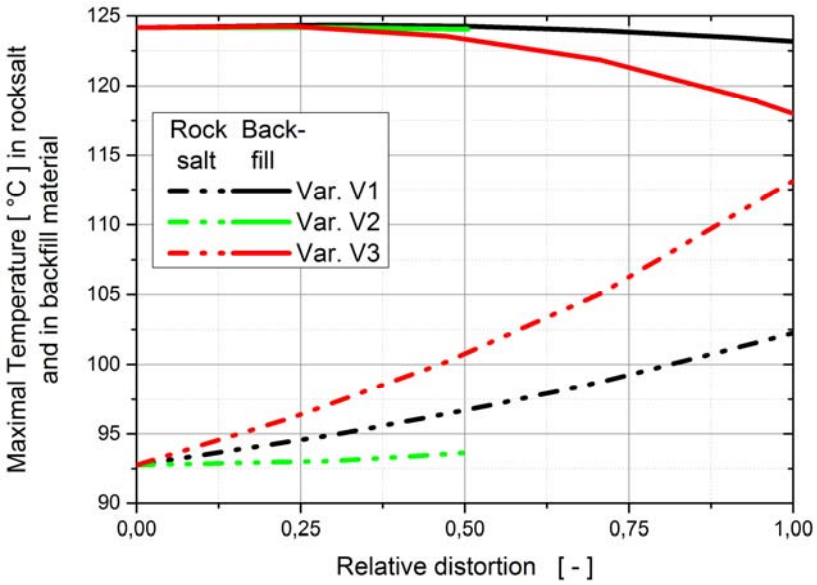

Figure 4.28: Maximum temperature depending on the relative distortion for variants V1, V2, and V3 in the borehole configuration without liner

The concept of borehole emplacement with borehole liner was developed in order to take into account a possible retrieval of the canisters from the repository [Bollingerfehr et al. 2012]. In this case, the borehole is furnished with a liner that is strong enough to withstand the rock mechanical loads. Thus, the backfill material and the canisters are no longer subjected to the rock mechanical loads. Instead of compactible crushed salt, an intrinsically stable and easily removable material is used as backfill material. In the case of the variants assessed in this study, the backfill material is sand. Due to the liner, the borehole diameter increases to 0.7 m. The liner is made of steel. The concept [Bollingerfehr et al. 2012] proposes that the borehole field be generated early enough so that gaps from excavation and borehole drilling have closed by means of rock convergence before emplacement starts. For the following study, purely thermal calculations are thus sufficient. The thermal material parameters for the backfill material sand and the steel liner are listed in Table 4.7.

Table 4.7: Additional material properties for borehole emplacement

Property	Unit	Steel liner	Backfill material Sand
Density	kg/m^3	7,000	1,700
Conductivity	W/m/K	40	0.4
Spec. heat	J/kg/K	515	850

The thermal design only covers the materials that contain halite. As a different, non-halite backfill material than in the preceding chapter is used, only the rock mass needs to be considered here. Thus, the design point is on the exterior surface of the liner. It can be concluded from Figure 4.28 that the central position will no longer be the conservative design because the backfill material no longer determines the design criterion in the variants considered here.

Due to the liner, the material relevant to the design is a bit more removed from the heat source. This decreases the thermal load on the halite slightly. Because the central position is no longer conservative, the highest temperature occurs in variant V3 with complete peripheral position. The related temperature increase of 15 K has to be taken into account in the design (Figure 4.29].

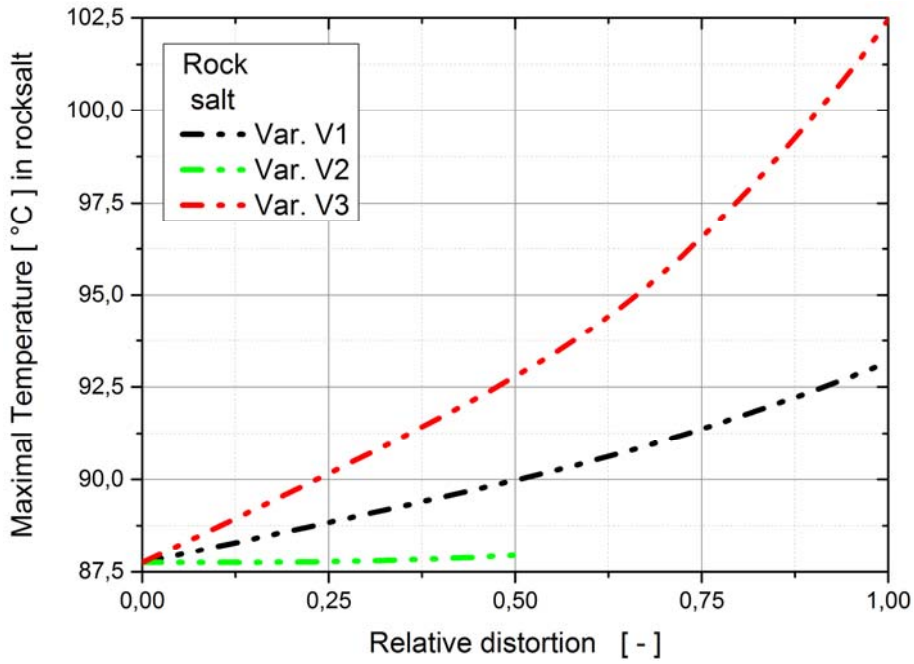

Figure 4.29: Maximum temperature depending on the relative distortion for variants V1, V2, and V3 and the borehole configuration with liner

The explanations above concern the first temperature maximum that occurs immediately after emplacement of a canister. In an emplacement field, the temperature fields of the individual canisters overlap so that over time, further maximum temperatures will occur. These maxima do not so much depend on the conditions in the individual boreholes but are determined by the space and material between the boreholes, which determine the head spreading. If a subsequent temperature maximum is design-determining, it is expected that eccentricity has only little impact.

4.3.6 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; the position of drifts in the mine and the ventilation system for repository operation. Usually, drifts and boreholes those 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 size 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 em-

placement 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 scope 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 packages 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.

4.3.7 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 design approach, 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.

4.3.7.1 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 4.30 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.

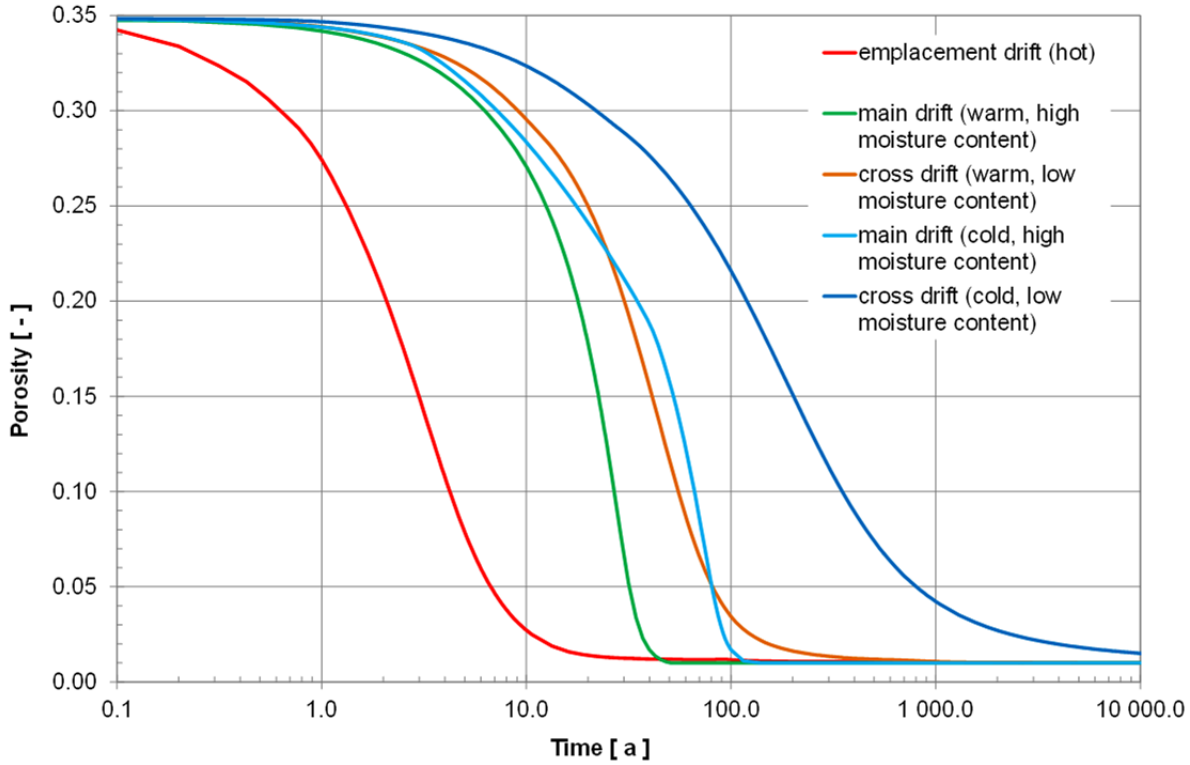

Figure 4.30: Comparison of temporal evolution of porosities in drifts backfilled with crushed salt; adopted from [Müller-Hoeppe et al. 2012a]

4.3.7.2 Drift Seals

In addition to backfilling, drift seals (engineered barriers) will be located close to the shaft landing station and infrastructure areas at selected positions in all drifts connected to the shaft at the exploration level and at the emplacement level (Figure 4.31). 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.

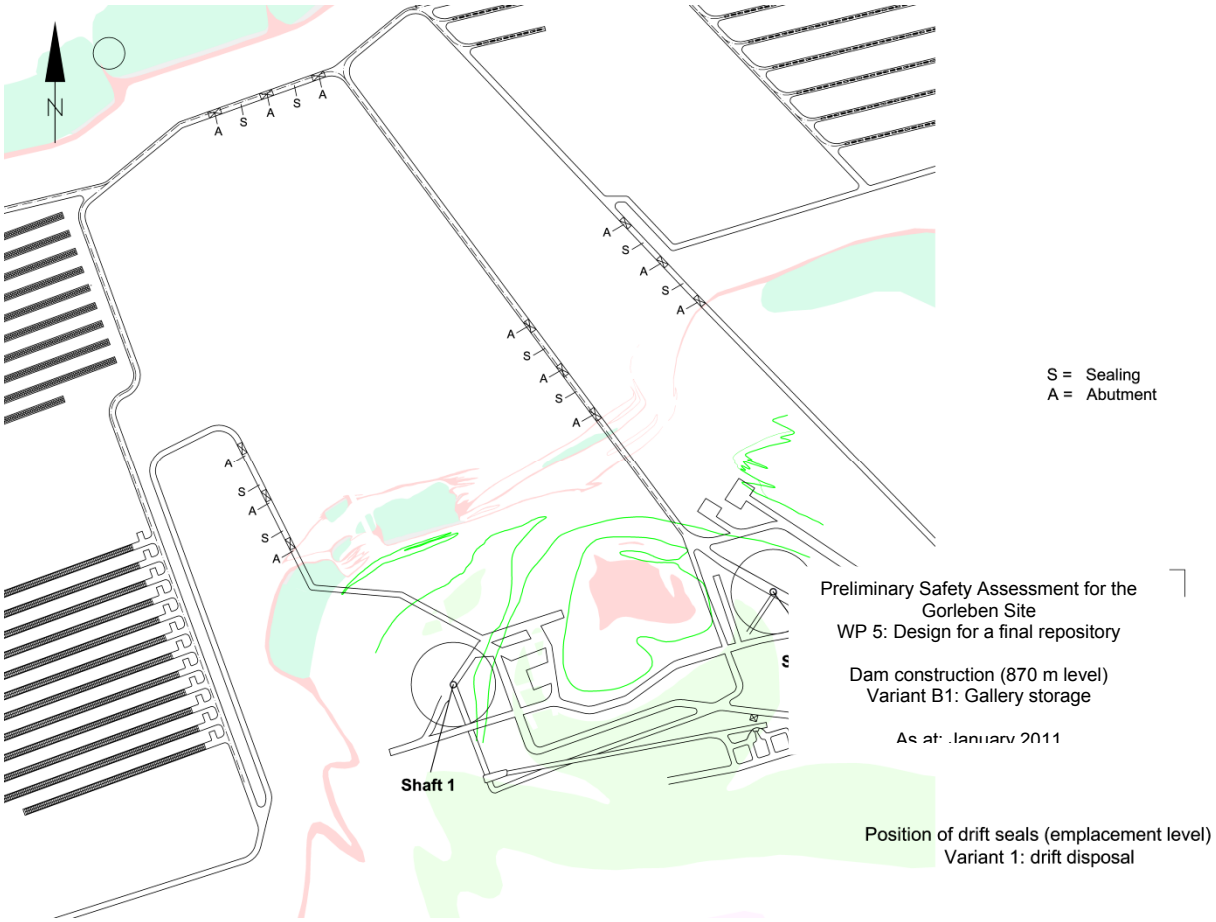

Figure 4.31: Position of the four drift seals on the 870 m level [Bollingerfehr et al. 2012]

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 landing 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.

4.3.7.3 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 4.8) 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 4.32). 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 4.32 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 4.32) 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 4.32), 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 4.32) 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 4.8 shows the materials selected for the backfill and seals.

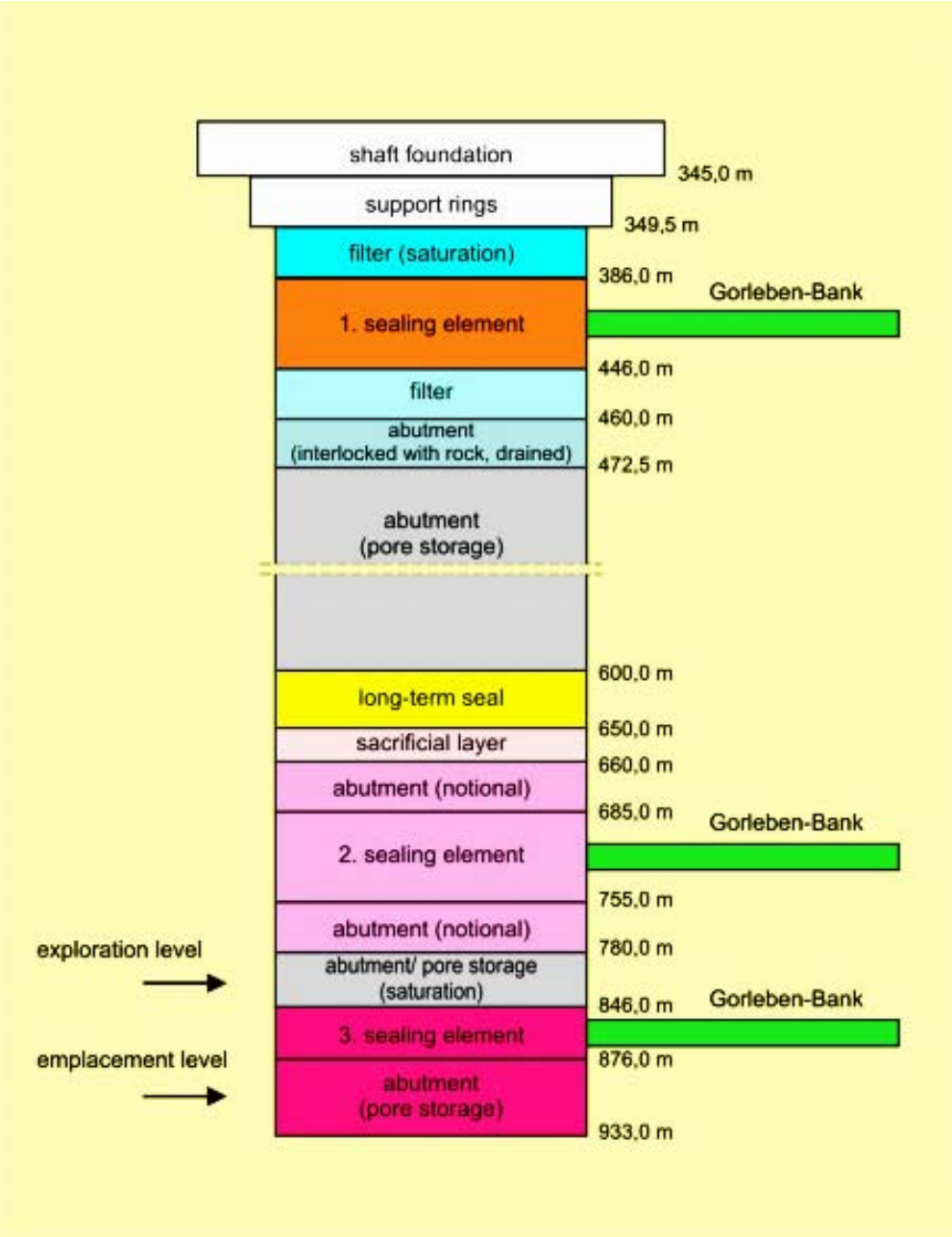

Figure 4.32: Sketch of the functional model of the shaft seal

location in repository	backfill material	sealing material	material specification
emplacement fields East 1 - East 12	dry crushed salt	./.	• grain size: up to 64 mm • initial porosity < 35% 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 conrete	
shaft 1 and 2	crushed salt (0.5-1.0 mass % content of moisture)	list of sealing materials (top down) • filter layer/saturation layer: sand/gravel/basalt grit • 1. sealing element: bentonite • filter layer/saturation layer: sand/gravel/basalt grit • abutment: salt concrete • pore storage: basalt gravel • long-term sealing: wet crushed salt • sacrified layer: salt concrete • abutment (notional): salt concrete • 2. sealing element: salt concrete • abutment (notional): salt concrete • abutment/pore storage: basalt gravel, serperntinit gravel or co • 3. sealing element: salt concrete • abutment: magnesium oxychloride concrete	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 4.8: Materials for backfilling and sealing selected for the R&D project VSG [Bollingerfehr et al. 2012]

4.3.8 Retrievability

In the past, retrievability of emplaced waste packages was not a part of the German repository reference concept. In 2002, AkEnd, the advisory group to BMU, did not see any reason to consider retrievability in the context of a siting process for an high level waste (HLW) repository [AKEnd 2002]. However, in 2010 the Federal Ministry for the Environment, Nature Conservation and Nuclear Safety issued new Safety Requirements [BMU 2010a]. These requirements replace the former Safety Criteria of 1983. The safety requirements apply to SNF (Spent Nuclear Fuel) and HLW repositories in Germany and all organizations *"… involved in the construction, operation, licensing and supervision of this final repository"* [BMU 2010a]. Within these safety requirements, retrievability has become a design criterion and licensing prerequisite.

There, retrievability is defined as *"… the planned technical option for removing emplaced radioactive waste containers from the repository mine"* [BMU 2010a]. Retrievability of waste containers must be possible "*during the operating phase up until sealing of the shafts or ramps…"* [BMU 2010a].

All existing and new repository concepts and designs have to ensure that the retrieval of emplaced waste packages is possible. This calls for the demonstration of technical feasibility already prior to licensing. The implementation of retrievability and retrieval necessitates the current considerations.

Taking into account the design standards stated in [BMU 2010a], a so-called "Re-Mining" strategy was selected as the most suitable approach for the retrieval of waste containers [Bollingerfehr et al. 2014]. This strategy implies that the emplacement of the waste containers and backfilling and sealing of the mine openings have to be carried out as designed in existing concepts. Nevertheless, it is possible to implement different design optimizations to facilitate retrievability. If a decision to retrieve is made at any time during repository operation, the already backfilled and sealed mine openings will be excavated again and the waste packages will be laid open. Suitable equipment picks up the waste packages and hauls them in reversed operation back to the surface. The arrival at the surface marks the end of the retrieval process.

Retrieval comprises just the actual action of waste package removal from the repository. In connection with retrieval, however, another aspect has to be taken into account. Before the canisters are retrieved, a concept for their subsequent storage and/or transport capacities, suitable storage casks, as well as suitable reconditioning capacities above ground must be available. All those facilities have to be licensed and constructed at least till the first cask is removed.

Based on the "Re-mining" strategy, the implementation of retrievability into the existing emplacement concept "Drift Disposal of POLLUX® Casks" and "Deep Vertical Borehole

Disposal of Spent Fuel Canisters" for the host rock salt was investigated in [Bollingerfehr et al. 2014].

At this point of time, the conditions and the reasons for future retrieval are not known precisely. To cover all possibilities, the current retrieval concepts assume that retrieval starts after the emplacement process has been completed. All casks have been placed inside the drifts, all drifts have been backfilled and all drift seals have been constructed. The remaining repository openings comprise the two shafts and the underground infrastructure area close to the shafts.

As basis for the current retrievability investigation the repository design of the "Preliminary Safety assessment for the Gorleben site" (VSG) [Bollingerfehr et al. 2012] was selected. Considering the drift disposal of POLLUX® casks the retrieval process starts with a re-excavation of the already backfilled access drifts and cross-cuts. Accordingly, each emplacement field is surrounded by two access drifts and two cross-cuts. To simplify underground operations, a third access drift is excavated at the northern side of the repository layout (see right side of Figure 4.33). The third drift allows an increase of intake air and a separation of operational processes. Contrary to the emplacement period, a minimization of the perforation of the host rock by drifts is not necessary anymore [Herold et al. 2016].

The excavation of the retrieval drift is divided into three steps, see also Figure 4.33. The excavation starts at the cross-cut averted from the shaft. First, a sub-retrieval-drift is excavated by a road header parallel to the emplaced POLLUX® casks. The drift is designed to provide a safety distance of 0,5 m to the casks. The initial positions of the casks are known from emplacement. However, possible displacements due to thermal impacts and floor lifting are expected. The sub-drift is connected to both cross-cuts. This improves ventilation and cooling conditions. In the same manner, a second sub-drift is excavated at the other side of the casks. The final cross section of the retrieval drift is formed by the stepwise and careful removal of the remaining pillar between the two sub-drifts. The casks inside the pillar are laid open by a remote-controlled demolition robot.

During the retrieval process, the relocation of the rails is not provided. Accordingly, one measure to facilitate retrievability is the removal of the rails inside the emplacement drifts before backfilling. Lifting and haulage of the laid open cask is done by a modified emplacement device (see Figure 4.33). The device hauls the POLLUX® cask out of the retrieval drift. Inside the cross-cut, the device delivers the cask to a rail-bound transport cart. A locomotive transports the loaded cart to the hoisting shaft.

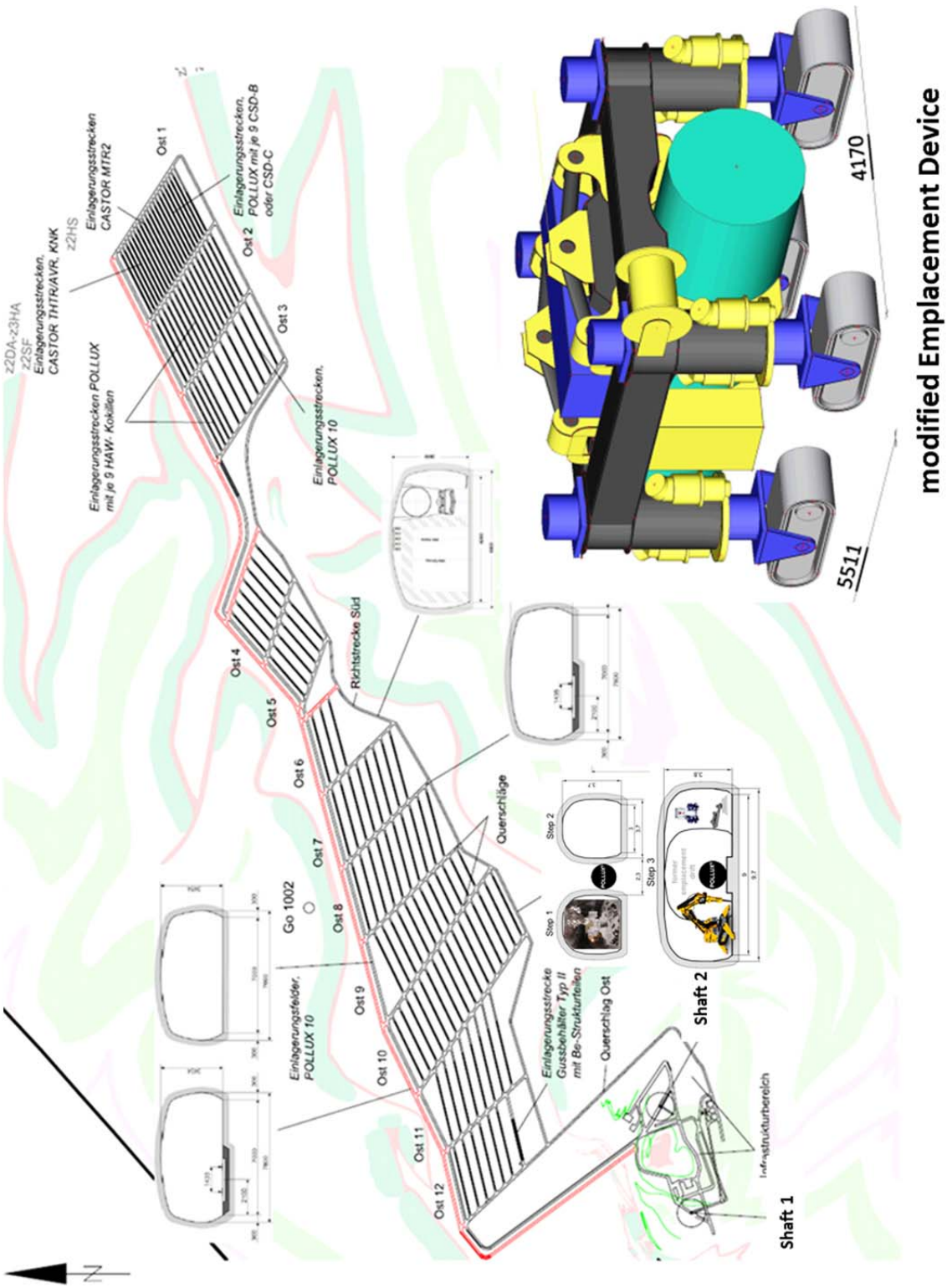

Figure 4.33: Repository design for POLLUX® cask retrieval, with illustration of drift cross sections, existing emplacement device and a conceptual design of a modified emplacement device [Herold et al. 2016]

The time needed to retrieve all POLLUX® casks corresponds to the operational period of cask emplacement. Conservatively it can be assumed that the retrieval period starts immediately after emplacement has been completed (assumption: 40 years after first emplacement). In some parts of the repository the design temperature of 200°C is reached during the retrieval period considered. Figure 4.34 illustrates the temperature distribution inside the repository at the start of the retrieval operation based on the repository design according to VSG [Bollingerfehr et al. 2012]. The calculated temperatures vary widely, depending on the exact location. Figure 4.34 illustrates the very high temperatures next to the casks and the high thermal gradients between the areas at larger distances. Experience in the mining industry give examples how to handle high rock temperatures. However, the situation inside the repository mine is slightly different from common mines. Because of the high thermal gradients, the excavation of access drifts and even the excavation of cross-cuts will take place in moderately heated areas. Thus, additional cooling of the airflow inside the excavations is absolutely essential. In addition to this, excavations inside the emplacement fields need additional cooling breaks during the excavation.

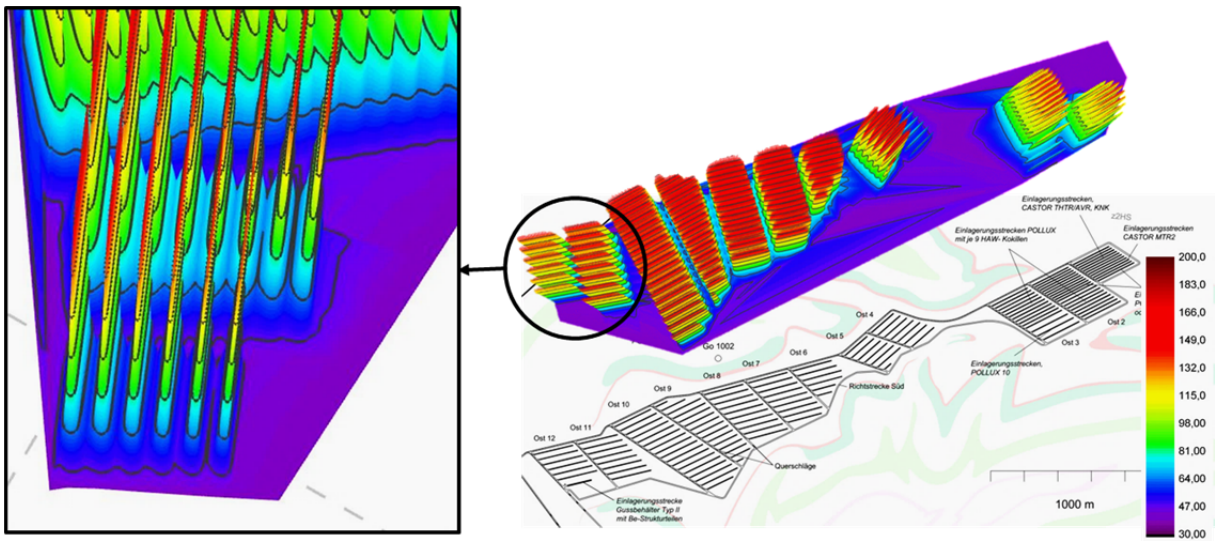

Figure 4.34: Temperature distribution in the completely filled repository [Herold et al. 2016]

The VSG repository design [Bollingerfehr et al. 2012] considers the densest package of POLLUX® casks. This arrangement results in the temperatures described. A modification of the repository design is one possible optimization to improve temperature conditions during retrieval.

In the case of deep vertical borehole disposal of spent fuel canisters, the retrieval concept consists of a reversed emplacement process. The access and retrieval drifts are at the same locations as during emplacement. Due to the larger distance of the drifts to the waste con-

tainers, the expected climatic conditions inside the repository mine are more favorable than for the drift disposal concept. During the retrieval period, most of the heat will be concentrated in the areas between the boreholes beneath the drifts and will then slowly dissipate into the surrounding host rock. Over time, the temperature will increase moderately at the working level of the repository mine. The highest temperatures are expected at the end of the retrieval process in the center of the large emplacement field (see also Figure 4.35). The retrieval concept comprises a fast re-excavation of all drifts. Considering this, early ventilation and cooling will generate comfortable climate conditions. To improve working conditions and generate more flexibility, a third access drift in the south of the repository is planned.

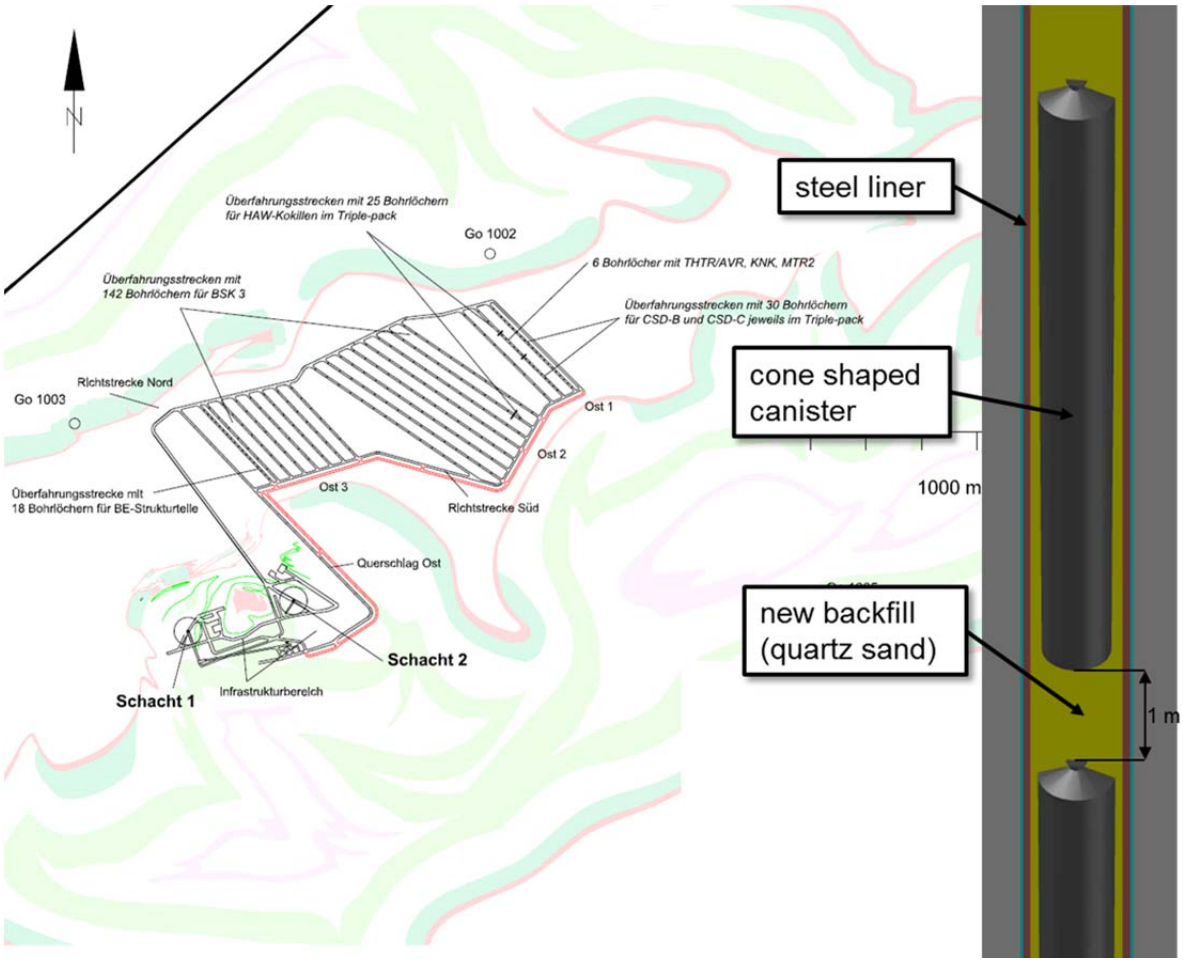

Figure 4.35: Repository design for spent fuel retrieval with illustration of borehole liner, canister and backfill [Herold 2016]

First conceptual modifications to facilitate retrievability were described in VSG [Bollingerfehr et al. 2012]. All boreholes will be equipped with a liner that is closed/sealed tightly at the top. The liner has been designed to absorb the expected geomechanical pressure of the surrounding rock masses. In addition to this, the outer design of the waste canister is modified

to be cone-shaped, and the annular gap between casing und waste canisters is filled with a non-compactable material like quartz sand to enable the extraction of the canister from the borehole.

The removal of the canisters will be realized by means of the emplacement device. Only several small modifications are necessary to reach full compliance with the current German regulatory framework and to further facilitate retrieval. Although this was not the intention of the demonstration test in 2009, the existing prototype emplacement device already demonstrated its capability for retrieval. For the demonstration tests only one canister dummy was available. To be able to realize the planned 1,000 emplacement processes, the canister had to be retrieved after each emplacement test, which was done successfully and safely at each step [Filbert et al. 2010a].

The remaining technical challenge during retrieval of waste canisters out of the borehole will be the removal of the sand between the BSK canisters and the steel liner. The sand is needed as a heat spreader to ensure sufficient heat dissipation into the rock. The sand has to be removed with a specially-constructed suction device that will be lowered into the borehole and suck off the sand step by step. When the head area of the canister is laid open, it is possible to grip the mushroom shaped lid of the canister and pull it out. The cone-shaped design and an additional jogger attached to the grip facilitate the process

In addition to the retrievability of waste containers, the German safety requirements also use the term "Bergung" (recovery or recovering). Recovery is defined as "…*retrieval of radioactive waste from a final repository as an emergency measure*" [BMU 2010a]. The time, reasons, and the conditions for recovery can vary within a very wide range. Therefore, recovery is considered as a requirement that is limited to the waste containers, "…*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*" [BMU 2010a]. Recovery is only a requirement for waste container design to ensure the handleability for the needed period. For repository licensing it has not been required so far to develop a concept for recovery procedures.

4.4 Methodology for Repository Design Development

4.4.1 Background and Objectives

From the very beginning in the early 1960ies it was a common understanding in Germany to provide the isolation of high level radioactive waste and spent fuel from human beings and the environment by disposing the waste in deep geological salt formations. Thus, the rational of the technical design of HLW-repository concepts was mainly based on practical experiences in salt mining since more than 100 years and on the thermo-mechanical behavior of the host rock salt. Consequently conceptual designs were developed for a repository for heat

generating radioactive waste and spent fuel in a salt dome. In the course of R&D activities launched by BMWi repository concepts for alternative host rock formations were developed for clay formations [Pöhler et al. 2010] or were started to be developed for crystalline rock in the next years. However, all these concepts rely on the idea to implement a repository mine for this purpose.

Since approximately 10 years repository concept development is strongly steered by the requirement to implement an adequate safety and safety demonstration concept. In the R&D-project ISIBEL [Buhmann et al. 2008c] the idea of applying a safety concept to derive a suitable repository design was analyzed. In parallel the national safety regulations as of 1983 were modified and published in September 2010 as "Safety requirements for a repository for heat generating waste" [BMU 2010]. The methodological approach of implementing a repository concept on the basis of a safety concept was successfully applied to real site conditions for the first time in the preliminary safety assessment of the Gorleben site (VSG) [Bollingerfehr et al. 2011], [Bollingerfehr et al. 2012]. The safety concept is based on the approach to identify an appropriate containment providing rock zone (CRZ) within the host rock which is characterized 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. The repository concept comprises approaches for concept optimization as well (e.g. ventilation, position of drifts in the mine, minimum footprint, etc.). Nevertheless, a systematic analysis and evaluation of further optimization potential never happened for HLW-repository concepts in Germany. In this context the question arose whether it is in principle possible to stringently derive a HLW-repository concept on the basis of requirements only. Consequently the following tasks were identified:

- Survey of existing HLW-repository concepts with regard to their rational (stringent requirement oriented or not) in Germany and in advanced programs
- Seek possibilities for stringent repository concept development (methodological approach) that is driven by requierements

4.4.2 Design bases and requirements

In a first step the design bases and requirements to develop a repository concept in Germany were compiled. The fundamental data mainly comprise:

- description of the amount and type of waste,
- description of the geologic environment (host rock, surrounding formations and overburden) as far as a site has been selected; otherwise due assumptions are required
- safety concept based on regulatory requirements (laws, ordinances, other regulations)

Regarding the type and amount of waste (a) sound data are available for heat generating radioactive waste and spent fuel (see chapter 4.3.3). A description of the geological environment (b) of a potential site to host a HLW and SNF repository (the Gorleben salt dome) is

available as well and given in chapter 4.1. In summer 2013 it was decided to definitely stop any further investigations at that site and to launch a new siting process. Today there is no decision on a site for a HLW repository. Thus, all three potential host rock formations in Germany (salt, claystone, crystalline rock) have to be considered in a new siting process.

In the course of R&D work a safety concept (c) for a HLW repository [Mönig et al. 2012] was developed on the basis of the "Safety requirements for a repository for heat generating waste" [BMU 2010]. The BMU Safety Requirements comprise a set of design requirements; in particular in section 8 some general requirements with regard to repository layout:

- Determination of the geometric limits of the containment providing rock zone (CRZ).
- Minimization of excavations of emplacement and infrastructure areas in the mine
- Spatial-temporal separation of mining activities and radiological relevant activities
- minimization of numbers of open emplacement areas
- Handleability of the waste containers must be guaranteed for a period of 500 years in case of recovery
- Retrieval of the waste containers must be possible during the operating phase up until sealing of the shafts or ramps,.
- Containment capacity of the repository must be based on a range of different barriers with varying safety functions. With regard to the reliability of containment, the interactions must be optimized between these barriers in terms of redundancy and diversity.
- Practicable, tested decommissioning concept

Based on the aforementioned Safety Requirements and the exploration results of the Gorleben salt dome a site-specific safety concept (c) was developed (see chapter 4.2) which was applied to elaborate the repository concept. In addition design requirements were derived from the mining law, the Atomic Energy Act, and the radiation protection ordinance.

In summary it can be stated that a set of main fundamental data and basic design requirements as well as a safety concept for a HLW repository are available in Germany; in particular if the host rock will be salt.

However, this does not prejudice á priori one single disposal concept. There are in principle at least a few degrees of freedom for the development of a repository concept:

- type of repository (mine, cavern, deep borehole, shallow land facility or near surface facility, etc.)
- depth of repository below surface
- one or more emplacement levels
- type of waste package: with or without shielding, with or without coating (as well type of coating), for one single canister with vitrified waste only or for a few of them; for entire spent fuel elements or for rods of spent fuel elements, etc.
- technology for transport and handling of waste packages

- remote controlled operation or operated by men
- emplacement concept (in drifts/galleries, in boreholes (vertical or horizontal, deep/long or short)
- type of backfilling and type and amount of sealing elements

And of course there are interdependencies between these possibilities; e.g. the decision for an unshielded waste package requires a shielding cask for transport to the underground and eventually to the emplacement location.

Thus, in a second step prior to develop a new methodological approach existing repository concepts were investigated with regard to their design approaches. First the situation in Germany was analyzed followed by a survey of repository concept development in three advanced waste management programs.

4.4.3 Survey of existing HLW repository concepts in Germany

The decision to investigate rock salt as host rock for heat generating radioactive waste was taken in Germany in the 1960ies. Referring on the experience in salt mining industry since more than 100 years it became obvious to design the repository as a mine. In the late 1990ies a comprehensive conceptual design for a repository for all types of radioactive waste in particular for heat generating radioactive waste for the Gorleben site was launched. It was assumed that a mix of reprocessing waste and spent fuel will be disposed of. This study – called "Actualisation concept - repository Gorleben" (Aktualisierung Konzept - Endlager Gorleben [Filbert & Engelmann 1998]) mainly was performed to receive a better understanding of the technique and process needed to design, construct, operate and safely close a HLW repository. On basis of the design results and the assessed time frame finally a cost estimate was possible.

The fundamental design data consist of:

- an estimation of the type and amount of radioactive waste; it was assumed that nuclear power plants would continuously produce electricity and thus produce radioactive waste
- a description of the geological environment ; few exploration data were available form exploration work at the site
- a set of regulatory guidelines like Atomic Energy Act, mining act; radiation protection ordinance as well as a first and brief set of safety criteria as of [BMI1983] – a safety concept as it is state of the art today did not exist in the 1980ies in Germany

The fundamental design data, the regulatory framework, and the predefinition to consider a mine as a repository type lead to the development of the repository concept. In addition the results of R&D-work program "Direct Disposal of Spent Fuel" were taken into account which revealed the technical feasibility and the safety of transporting and disposing of fuel rods of spent fuel elements in a self-shielding cask called POLLUX® [Filbert 1994b], [Filbert 1995].

Thus, a repository mine was designed which consists of separate emplacement areas for reprocessing waste to be disposed of in unshielded standard canisters from reprocessing (e.g. CSD-V) into deep vertical boreholes and emplacement areas for spent fuel disposed of in POLLUX® casks in horizontal drifts. The layout of the mine took into account the necessary miner`s needs for ventilation, escape routes, technical needs for transport- and emplacement technique, as well as geomechanical constrains for dimensioning of the cross sections of mine openings. Thermomechanical calculation results considering the 200°C design criterion provided the necessary data to determine distances of waste packages, boreholes and drifts.

With the exception of a few constrains, the repository concept could be selected as deemed appropriate. This is true for the repository concept as well as the emplacement concept. Thus, there was no stringent requirement oriented derivation of a repository concept.

A second more comprehensive repository design - the first site specific repository concept for a HLW repository in Germany - was elaborated in the context of the Preliminary Safety Assessment for the Gorleben site [Fischer-Appelt et al. 2013]. The fundamental design data consist of:

- a precise description of type and amount of radioactive waste and spent fuel as it was politically decided in 2011 to phase out of nuclear electricity production in Germany until end of 2022

- a detailed description of the geological environment at the site of the Gorleben salt dome on the basis of intensive exploration work

- a safety concept and safety demonstration concept – elaborated on the basis of a methodological approach developed in the R&D-project ISIBEL and taking into account the Safety requirements [BMU 2010]

In addition a set of regulatory guidelines like atomic energy act, mining act, radiation protection ordinance have been considered during the designing process.

The safety concept mainly formulates measures which should ensure the compliance with the BMU “Safety Requirements”. These measures comprise:

- Safety distances between repository structures and the bounderies of the CRZ
- Boundary conditions for positioning the repository mine in the host rock
- Backfill and sealing systematic
- General principles of drift excavation and emplacement
- Temperature limits

The measures were considered as concrete stage of requirements and applied to develop the repository concept. The selected technical concept itself is the implementation of the measures derived from the safety concept. Similar as in the previous repository design from 1989 two emplacement variants (emplacement in deep vertical boreholes and emplacement in horizontal emplacement drifts) were considered and appropriate design calculations performed. Among others, the necessary geometric design parameters (e.g. distance of waste packages, distance of boreholes and emplacement drifts from the boundaries of the CRZ) were calculated. However, the main difference in this regard was that in both variants all heat generating waste will be disposed in the same way; either all waste packages in deep boreholes or all waste packages in horizontal drifts. Another design requirement which was different to the previous one was the requirement that waste packages have to be retrievable during the operational period of the repository. This again has an impact as well on the emplacement concept "vertical boreholes" as on the backfilling and sealing concept. Liners have to be installed in the boreholes prior to the waste emplacement and the backfilling material between the waste canisters in the borehole has to be changed from crushed salt to a material which easily can be retrieved (e.g. sand). The emplacement concept "horizontal drifts" remains unchanged; however, it is obvious that the technique and processes for retrieving waste packages has to be developed. This of course requires R&D activities (s. chapter 6.2). Consequently the repository concept might change if the results of the R&R do not confirm the assumed adequate technique or process.

The concept development for the VSG clearly showed that the selected concept first is an assessment of experienced engineers who try to meet all requirements. The most important "guideline" to derive a relatively stringent requirements oriented concept was the derivation of a site specific safety concept. The way how to implement the measures of the safety concept into a technical repository concept of course does have several degrees of freedom (e.g. selection of waste package, depth of emplacement level, separation of type of high-level waste in different areas or a mixture, etc.). Eventually the VSG reveals that new or additional requirements (retrievability) will have an impact on the technical repository concept and perhaps on the way how to demonstrate safety.

The survey of the two repository conceptual design developments in Germany (Actualization Concept Repository Gorleben [Filbert & Engelmann 1998] and Preliminary Safety Assessment for the Gorleben site [Bollingerfehr et al. 2012]) indicates that there are several options to develop a repository concept. The number of remaining degrees of freedom is the main reason that the list of requirements not necessarily leads to only one single concept. There are still several options which might meet the requirements. Thus, it deems that there is no real alternative to an iterative approach.

4.4.4 Survey of existing HLW repository concepts in advanced countries

In addition to the survey on repository concept development in Germany the process of repository design development in a few advanced countries was analyzed.

France

The approach for a concept determination for a HLW repository in France was done in a classical way. The design requirements were distinguished in

- functional requirements (protection of men of radiological exposure, enable retrievability, etc.),
- operative requirements (reliability of the operations , health protection of the personnel…), and
- technical requirements (requirements resulting from the concept decision, from the construction and from natural science).

In addition, restrictions for the repository concept were defined mainly referring to the situation at surface (infrastructure, geography, environment and nature protection, etc.) (Figure 4.36). However, only the technical requirements may be considered as tangible requirements. The description of detailed requirements like distances between waste packages or between emplacement drifts and boreholes were done in dependence of the selected technical solution. This again reveals that the requirements leave degrees of freedom for the planning. In any case the maxim is that the selected repository concept and its components have to meet the requirements.

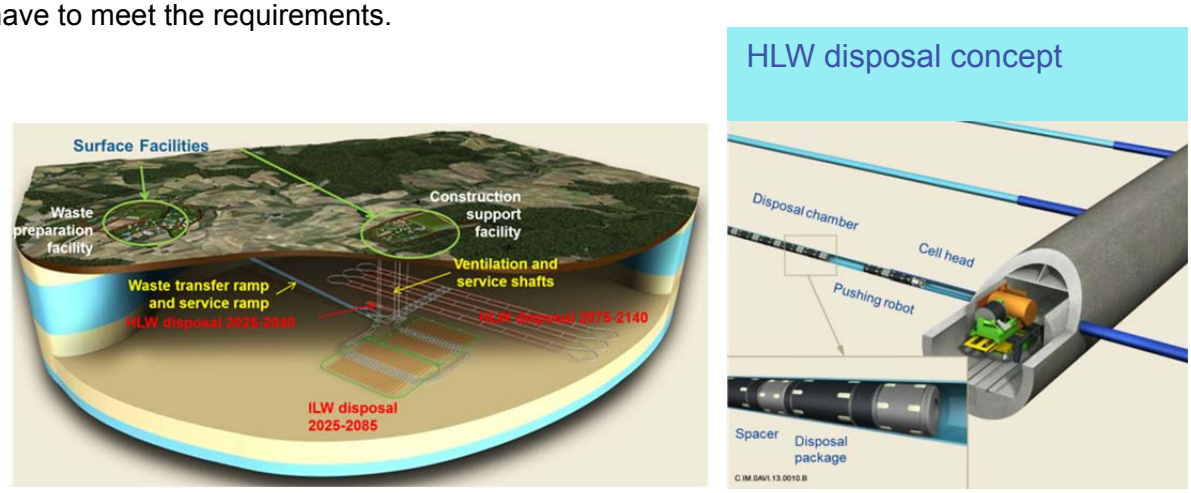

Figure 4.36: French repository project Cigéo [Andra 2005]

In the first decade of this century it was decided that in France all spent fuel of the nuclear power plants will not be reprocessed but disposed of directly in a repository. A repository mine in clay formations at a depth of approx. 600 m with separate areas for high level waste and intermediate level waste was designed. The access to the underground is planned with 5 shafts (personnel, materials, fresh air, and exhausted air) and 2 ramps for waste package transport and services. The HLW-waste packages will be disposed of in horizontal lined boreholes which have to be accessible for potential waste package retrieval during the repository life time (100 years).

Switzerland

In Switzerland the repository design does not follow a stringent requirement oriented approach (Figure 4.37). Instead, practical experiences in mining industry were applied on the design. Design requirements as well as results from R&D-work, and fundamental data were considered to have the same level of importance concerning repository concept development. General design principles and detailed specifications for the repository layout for the operational phase have been compiled. Whereas the general design principles are referring to general safety principles (e.g. protection of men and the environment against radiological exposure) the detailed specifications were derived from long-term safety assessments and lead to stringent requirements. The repository concept and its components were developed on basis of these requirements; nevertheless the design is not stringently linked to all the requirements. A simple example is given by the dimensioning and the total amount of emplacement drifts as a function of the waste characteristics. This dimensioning including the compilation of the requirements for the dimensioning is only feasible if a decision on the type of emplacement has already been made. Thus, the fundamentals and the general requirements (protection goals) lead to tangible requirements by means of R&D work. The technical design of the repository will be done within the degrees of freedom of this approach. Eventually the repository design has to meet all requirements.

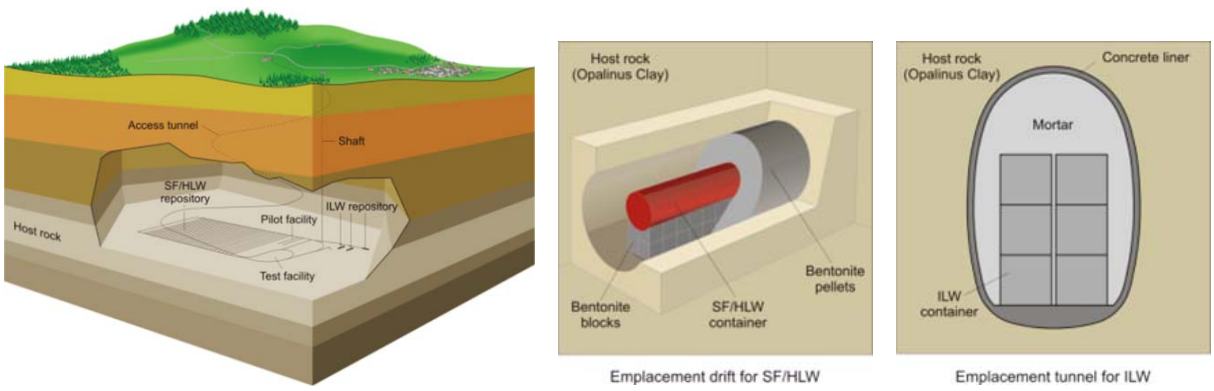

Figure 4.37: Swiss reference repository concept [Nagra 2002]

In Switzerland it was decided to choose opalinus clay as host rock for a common repository for spent fuel and high level waste as well as for intermediate low level waste. The repository mine will be excavated in a depth of some 700 m below surface. Waste packages containing SF/HLW will be remote controlled placed on a bentonite pedestal and both, the pedestal and the waste package will be rail bound transported into the horizontal emplacement drift. The void between waste package and drift walls will be backfilled with bentonite pellets.

Sweden

A host rock selection in Sweden was dispensable due to a lack of alternatives to the crystalline rock. SKBs approach for repository concept development was based on a staged system of requirements (Figure 4.38). Detailled requirements became more specific with increasing level of the planning. The first more general level of requirements was fed by regulatory fundamentals (laws, ordinances etc.), interests of stakeholders and other agreements. The following two levels refer to specifications of the repository system which eventually led to the Swedish KBS-3 concept. These requirements were based upon the fundamentals and a selected variant of waste handling. In the course of the planning technical determinations led to additional requirements. The last and most tangible stage of requirements refers to the different components of the repository system. Here quantitative specifications were made which were justified by safety assessments and technical planning. This procedure clearly reveals the dependence of requirements from preceding concept decisions. And It confirms the degrees of freedom for the planning on basis of detailed requirements. If such degree of freedom does not exist all the requirements have to be known at the very beginning of the planning process. Thus, a more or less independent selection of repository concept had occurred.

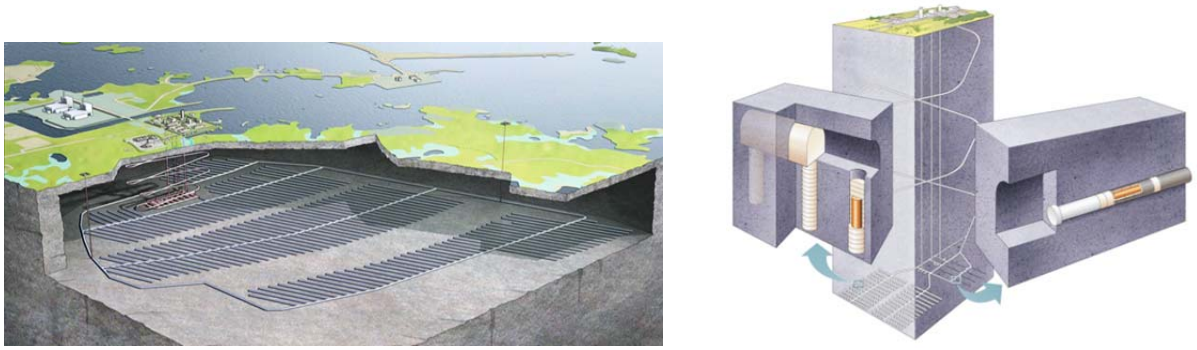

Figure 4.38: Swedish reference repository concept KBS-3V [SKB 2006]

As there is no geological alternative, the repository will be constructed in crystalline rock in Sweden. An important decision in principle was done at the early beginning of the waste management program implementation: All spent fuel will not be reprocessed but disposed of directly in a repository. The technical concept consists of a mine at a depth of approx. 600 m

below surface in which single unshielded steel canisters with a 5 cm copper cover will be emplaced in short vertical boreholes. Bentonite rings and plates in the borehole are providing an additional technical barrier for long-term safety reasons.

The emplacement of several super containers in horizontal boreholes was considered as an alternative to the KBS-3V concept in Sweden. It consists of a copper canister inside a bentonite cover fixed by a perforated cylindrical steel envelop for transportation and handling reasons.

Summary of the survey of existing HLW repository concepts in advanced countries

There is no unique approach how to derive a conceptual repository design in the selected three advanced waste management programs in France, Switzerland and Sweden. But in all three countries decisions in principle were made on the host rock prior to start thinking about the technical concept. In Sweden there was no alternative to crystalline rock, but in France as well as in Switzerland there are crystalline rock formations as well as clay formations and salt - at least in France - available. Thus, the first step was a selection or determination of the host rock. According to the national waste management plan the fundamental design data were compiled (type and amount of waste, description of the geological environment at a selected site or at potential sites). The classification of the requirements in principle was identical. First general protection goals were fixed followed by functional/general requirements and specific/concrete requirements. As in all three cases a stepwise design process was applied a appropriate definition of requirements followed. Thus, the interdependencies of concrete requirements from the definition of following technical design steps were accepted.

The idea to isolate the radioactive waste form human beings and the environment and by that to fulfill a general protection goal by disposing of the waste into the deep geological underground was common in the three approaches. However, the implementation of the repository concept was completely different. To give one example: the French selected the emplacement of reprocessing waste with an overpack in horizontal emplacement cells in a repository mine in clay formations with the option of retrievability; the Swiss decided to emplace huge waste packages with reprocessing waste in horizontal emplacement drifts in a repository mine in clay formation, and the Swedes decided to emplace single steel canisters with a 5 cm copper coating containing spent fuel elements into vertical boreholes of a repository mine in crystalline rock. Even if the fundamental data are similar and the basic safety requirements are more or less the same the solutions for the repository concept differ widely.

4.4.5 Methodological approach to derive repository concepts from requirements only

The survey of existing HLW repository concepts showed that the process of developing a repository conceptual design is an iterative process. It usually starts with the compilation and analysis of fundamental data and requirements which define the design framework and the boundary conditions. These fundamental data include firstly a description of the type of waste to be disposed of and their characteristics, as well as their amounts and their arising in time and the kind of conditioning. Secondly, for a deep geological repository, a description of the site and of its geological and hydrogeological characteristics is necessary. In case some information is missing it must be either obtained, e.g., by waste stream analysis or site survey measures, or replaced by justified assumptions.

And thirdly, a list of design requirements is needed. They can be derived from regulatory framework (e.g. BMU Safety Requirements) and may lead to a safety and safety demonstration concept defining design measures or conditions the facility must fulfil. In addition, requirements are stipulated in the Atomic Energy Act, the Mining Act and the Radiation Protection Ordinance, all of them containing measures to ensure the safety of the personnel and the repository and limits for contaminant emissions. Moreover, also such external limits and conditions resulting from negotiations with stakeholders must be compiled in the project basic data.

Once the fundamental design data and the requirements and boundary conditions have been compiled, the design work can start. However, at least two decisions in principle have to be made prior to the start of the concept development. The repository site has to be selected or at least the type of host rock – if there are different options - otherwise all design work can only be of generic nature. A decision in principle concerning the type of repository has to be made. There is international consensus that a repository mine provides the best option to dispose of heat-producing spent fuel and reprocessing waste in particular because of its accessibility and the more than 150 years of experience in mining industry. In Germany, the "Repository Commission" recommended to dispose of all heat generating radioactive waste and spent fuel in a mine several hundreds of meters below surface which allows waste package retrieval during the operational phase of the repository. Thus, a discussion on alternatives is a poor hypothetical one because it is very likely that the recommendation of the commission will become part of the revised new site selection act [StandAG 2013]. However alternatives have to be mentioned and ruled out by convincing arguments. Prior to conclude whether it is possible to derive a repository concept on basis of requirements only a look on the design aspects is necessary.

A list of necessary steps for a comprehensive planning of the conceptual repository design may look like the following one:

1. Preparatory work:
    a. analysis and evaluation of the fundamental design data (type and amount of waste and/or spent fuel, geological environment, laws and regulations, etc.) , and
    b. derivation or determination of the requirements by means of a safety concept.
2. Decision on the access to the underground: shaft or ramp or combination; analysis of the most favourable ways and means of access; shafts versus ramp.
3. Decision on the optimum location of the access points in relation to the underground disposal fields (central position, at the edges) on the basis of operational and long-term safety considerations.
4. Decision on the waste package (dimensions, materials,etc.).
5. Decision on the emplacement concept: borehole, drift, chamber, etc.
6. Design of the underground disposal cell module:
    a. dimensioning (cross-section, cell volume),
    b. needed support (anchoring, shotcrete, support beams),
    c. cell sealing and backfilling (if envisaged to enhance long-term safety), and
    d. decision on the disposal cell excavation technique (drilling and blasting with or without reworking with continuous miner, or part face drifting to minimize the excavation disturbed zone around the chamber).
7. Design of the general underground facilities layout by means of thermomechanical calculations:
    a. size of pillars,
    b. size of single underground disposal cell (borehole, drift, chamber),
    c. general arrangement of the disposal fields (number of cells per field, length of drifts or boreholes, and
    d. design of a specific ventilation airflow pattern to allow disposal rooms excavations concurrent to disposal operations with mine excavation outside the radiologically controlled area.
8. Integration of underground facility components:
    a. integration of disposal areas, infrastructure area, and shaft station areas to the full underground facility, and
    b. determination of the ventilation needs (ventilation rate, ventilation air heating and cooling, exhaust air filtering.
9. Conceptual design of underground waste transport and handling equipment (e.g.: transportation (railbound or not), loading, emplacement).
10. Design of the needed underground infrastructure area:
    a. workshops, vehicle parking areas, transformer rooms, magazines for materials and supplies; fire and rescue brigade rooms and equipment,
    b. definition of radiation protection areas (controlled area, conventional mining activities area) and radiation protection equipment, and

c. planning of barren rock bunker, ways and means for barren rock underground movement (shovel loaders, conveyor belts), mine water collection and buffering dewatering pumps.

11. Concept for underground operations logistics:
    a. schedule for waste package transport and disposal operations, and
    b. schedule for excavation operations and flow of material and barren rock.
12. Assignment of the required functions to the access means to underground:
    a. personnel access,
    b. waste transportation to underground,
    c. excavation rock transportation to surface,
    d. ventilation air supply and exhaust, and
    e. emergency exit.
13. Design of the access shafts:
    a. dimensions and support (diameter, wall support, and use of the cross-section),
    b. definitions of the hoisting means (shaft cage for waste package hoisting, rock transportation (skip) personnel hoist, emergency exit means, shaft towers), and
    c. supplies to underground (ventilation air, electricity, high pressure tubes for mine waters pumps, communications).
14. Design of the shaft station areas at the surface:
    a. conceptual design of waste handling equipment,
    b. personnel access to the shaft,
    c. waste package access to the shaft,
    d. barren rock discharge systems,
    e. for the shaft accessing the underground controlled area: personnel access areas (entrance/exit to the controlled are); changing rooms, radiation protection equipment and supplies rooms,
    f. for the intake-air shaft: ventilation air heating/cooling systems air intake into the shaft, main mine fan (if blowing ventilation), and
    g. for the exhaust-air shaft: shaft station cellar for air discharge, main mine fan (if sucking ventilation) air filters (if required) air discharge diffuser.
15. Design of the underground shaft stations:
    a. conceptual design of waste handling equipment,
    b. ways and means for unloading of waste packages from the shaft cage, and for loading of auxiliary waste package transportation means (carts, pallets) that return to surface,
    c. personnel access to the underground: access/exit in normal operation and emergency access/exit means, and
    d. means for transportation of the barren rock back to surface (e.g., via skip).
16. Concept for shaft loading and transport operations logistics:
    a. schedule for loading and shaft transport operations,
    b. planning of material and waste package logistics, and
    c. planning of regular and emergency transport of personnel.
17. Design of waste treatment and conditioning areas (if required):
    a. waste reception area with linked buffer area,

   b. waste inspection area (check of acceptance criteria),
   c. facilities for further waste treatment and conditioning (if required), and
   d. waste storage area.
18. Design of the surface waste handling area:
   a. ways and means for surface shaft cage loading with waste packages a, auxiliary handling means, and
   b. buffering area at or near the shaft surface station (if necessary).
19. Design of areas for reconditioning of waste packages (if required during the retrievability period).
20. Concept for operational logistics for waste and material handling during surface operations:
   a. schedule for waste handling and treatment procedures,
   b. schedule for material transport, and
   c. planning of waste and material logistics.
21. Operational safety assessment (surface operations, shaft operations, underground operations) as deemed appropriate in the conceptual design:
   a. accident analysis,
   b. estimation of possible consequences, and
   c. determination of necessary changes to equipment or procedures.
22. Long-term safety assessment as deemed appropriate in the conceptual design:
   a. development of repository computer model based on existing data,
   b. development of relevant scenarios.
   c. calculation of radiological exposure,
   d. sensitivity analysis for parameters of repository design and planned sealing technique, and
   e. determination of necessary changes to repository design, waste handling equipment, waste handling procedures, waste package conditioning etc.

The list gives an impression how many different technical, safety and logistic aspects have to be considered in the course of the development of a conceptual repository design. And a lot of them are linked or have a direct impact on a following aspect. To give one example: the selection of the waste package – if this is done in advance – directly impacts the emplacement concept (borehole or drift) and the appropriate transport and handling technology. On the other hand the type of host rock has a strong impact on the importance of the waste package in the safety concept. Whereas safety will mainly be provided by the host rock in case of salt and clay, the waste package has to provide long-term safety in connection with the surrounding buffer materials in the case of crystalline rock.

An optimum approach for stringently deriving a repository concept from requirements only could be a very precisely defined profile of requirements (like an industrial standard: DIN). But due to scarce or no experiences in designing, constructing and operating of HLW repositories it is only possible to examine whether requirements were met after completion of the conceptual design of repository components. The complexity of the requirements does not

lead to an objective solution and the state of knowledge always provides the basis for a decision. There are too many degrees of freedom which will have an impact on the design, e.g.:

- access to the underground : shaft or ramp or combination,
- depth of repository mine below surface ,
- one or more emplacement levels in the mine,
- separate or common emplacement areas for the different types of heat generating waste (spent fuel or reprocessing waste),
- emplacement concept (drifts/galleries or boreholes (vertical or horizontal, deep/long or short),
- type of waste package:
  - layout of waste container,
  - with or without shielding,
  - with or without coating (as well type of coating),
  - for one single canister with vitrified waste only or for a few of them;
  - for entire spent fuel elements or for rods of spent fuel elements, etc.
- role of waste packages in the safety concept,
- technology for transport and handling of waste packages,
- remote controlled operation or operated by men,
- type of backfilling and type and amount of sealing elements,
- retrievability requirement or not, and
- optimization target: e.g. minimum footprint or optimum conditions for waste package retrieval.

Thus, this list clearly shows that a stringent requirement oriented repository concept development is challenging because of the complexity of the system. The survey revealed as well hints that a poor stringent requirement oriented derivation of a repository concept deems to be nearly impossible. A similar conclusion is stated in the safety concept of VSG [Mönig et al. 2012]; the authors concluded, while the conformance to requirements at the site has to be provided by its characteristic features, the repository concept and the repository design allow creative latitudes to meet the requirements. There exists a diversity of partially interferring requirements which demand complex responses.

4.4.6 Conclusions regarding repository concept development

As long as it is possible to demonstrate the technical feasibility and the safety of the selected concept the approach is correct. But, and that is important for the idea to develop an approach which always delivers a repository concept that fulfils the requirements there is no warranty that the selected concept meets the requirements. The total number of requirements does not provide a comprehensive concrete frame which ensures the compliance of the concept with the requirements. Decisions made in the course of the development of the repository concept mainly rely on assessments and experiences of the designer (set of de-

grees of freedoms). And it should be mentioned that the proof of requirements in some case will only be possible if components of the technical concept have been realized or the impact on other components has been investigated in R&D work.

The original idea to develop a methodical approach, which allows a clear and stringent requirement oriented derivation of a repository concept, failed. The number of degrees of freedom is that high and the interdependencies between single repository components like waste package or emplacement variants that complex that there are in all cases several solutions; even if the fundamental design data are equal.

However, a repository concept derived from a carefully balanced safety concept provides a good chance to meet the set of requirements. The procedure of repository concept development may include iterations driven by the results of the performance assessment.

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 5.1). 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 5.2 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. 2008c] 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. 2008c].

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.

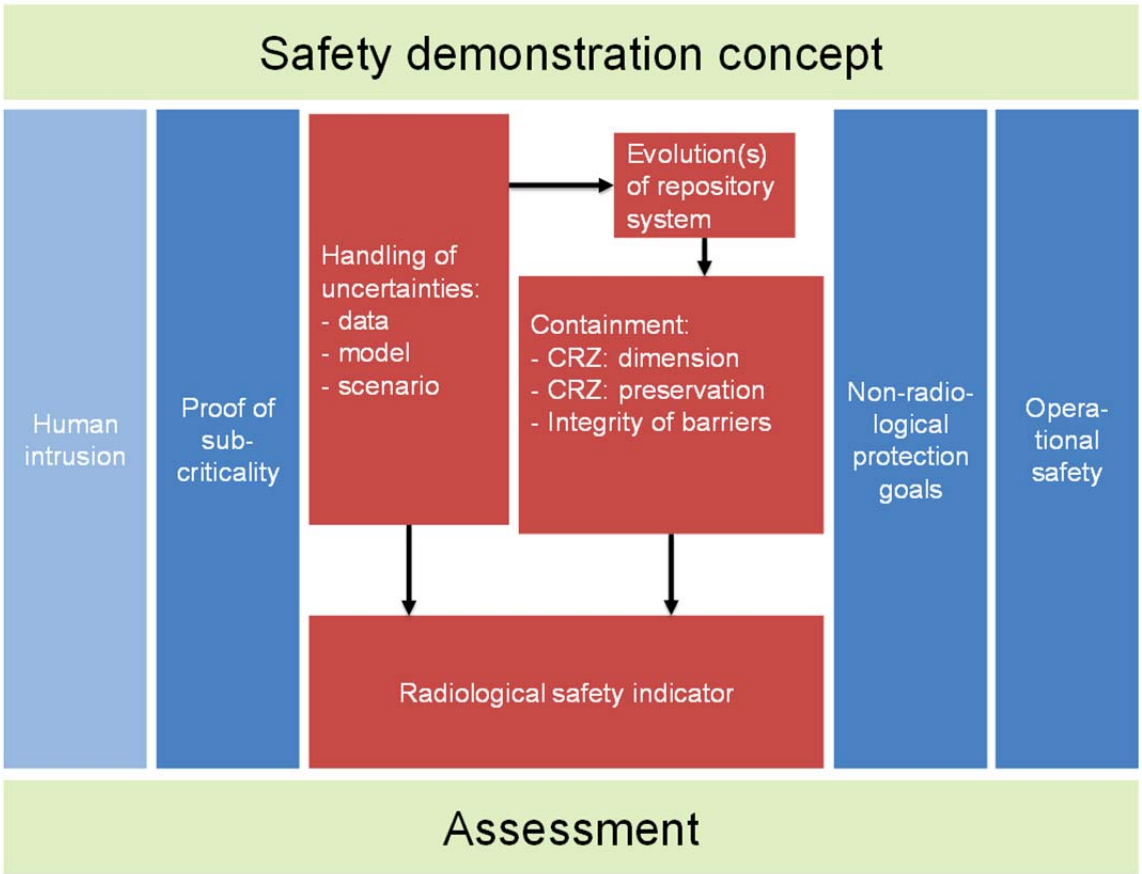

Figure 5.1: Elements of the safety demonstration [Mönig et al. 2012]

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 stand-alone 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. 2010e]. 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 (chap. 4.1.1) 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 uncer-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, [Peifer et al. 2011b], [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 [Buhmann et al. 2010e]. 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. In 2015, the NEA Integration Group for the Safety Case (IGSC) held a second workshop on June 1-3, 2015, in Paris to further evaluate the experience acquired in developing scenarios since 1999. The purposes of the workshop were to

- provide a forum to review and discuss methods for scenario development and their contribution to the development of recent safety cases;
- examine the latest methods and compare their scope, consistency and function within the overall safety assessment process; and
- provide a basis for producing a report summarising the current status of scenario methodologies, identifying where sufficient methods exist and any outstanding problem areas.

All programmes represented at the workshop divide their scenarios into classes or categories, based on the types of FEP that are covered in the scenarios, their likelihood/probability, and their potential effect on the evolution of the repository. As in Germany, this classification is determined by regulations or regulatory guidance. In addition to these scenarios, the value of what-if-cases in investigating or demonstrating system robustness and in illustrating the functioning of specific barriers is widely recognised. Future human actions and human intrusion in particular, are treated as a separate scenario category requiring somewhat different handling in the safety case compared with other scenarios, using stylised approaches.

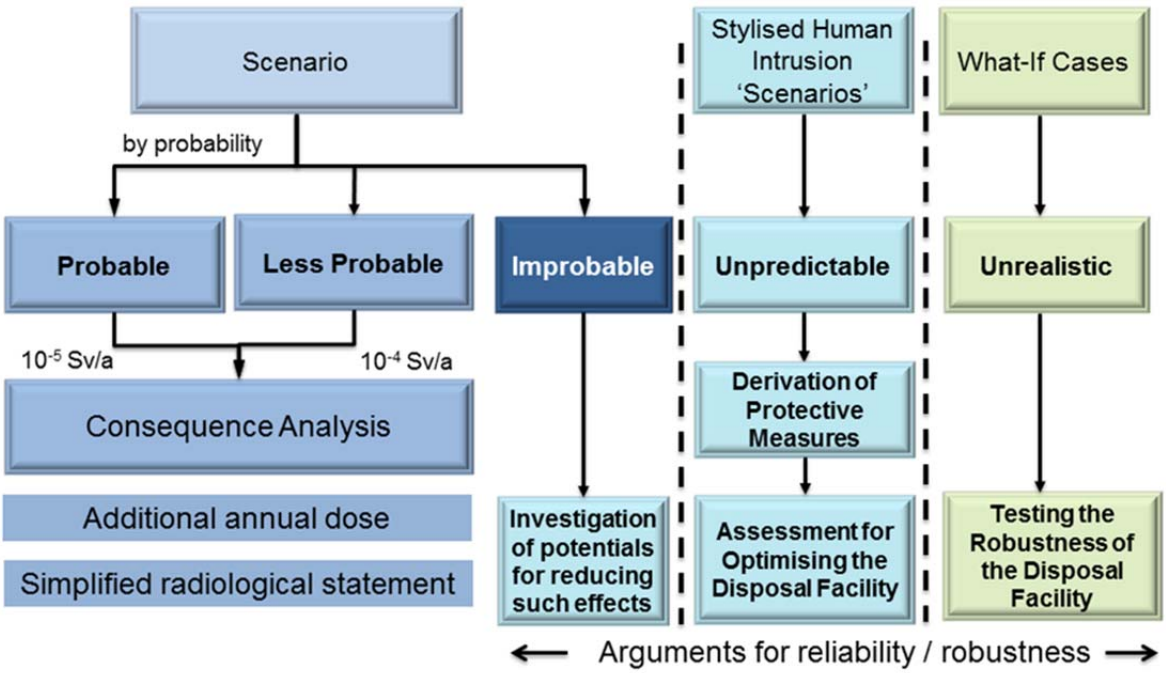

Figure 5.2: The different categories of scenarios in the German safety case [NEA 2016]

The integration of top-down and bottom-up elements may in reality be a feature of all practical approaches to scenario development. Essentially, safety assessors provide a top-down description of the safety concept and safety functions, and “phenomenology” and “technology” provide a bottom-up description of FEPs and their attendant uncertainties that could challenge the safety functions, and hence give rise to alternative scenarios. The impact of the perturbing FEPs, either individually or in combination, is then considered when defining scenarios for the evolution of the repository, which are assigned to various categories. FEP lists and other tools are used to confirm that key FEPs and uncertainties are covered adequately in one or more of the identified scenarios and associated calculation cases.

In the R&D projects ISIBEL [Buhmann et al. 2010d], [Buhmann et al 2010f] and VSG [Beuth et al. 2012a], [Wolf et al. 2012] a methodology has been developed that is closely related to an FEP catalogue 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. 2010d] 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 5.3.1) . 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. 2008b], [Buhmann et al. 2010a] was taken as a basis. The catalogue was reviewed in 2010 by external experts and iteratively revised [Buhmann et al. 2010b]. Despite strong efforts to increase the knowledge base for the FEP catalogue, uncertainties will always remain regarding the com-

pleteness 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 [Beuth et al. 2012a]. The methodology enables the assignment of probability classes to the scenarios in line with the regulatory framework [BMU 2010].

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. 2008c]. 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. 2010d]. 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.

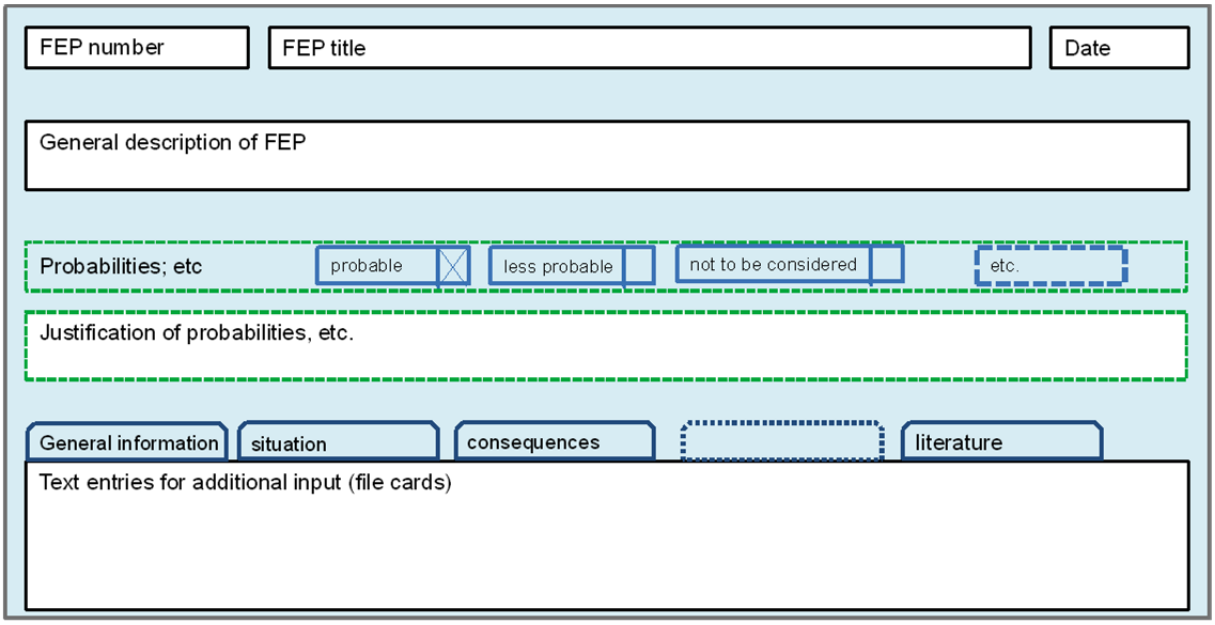

Figure 5.3: Schematic overview of FEP database entry

Figure 5.3 shows a sketch of the input mask for FEP descriptions in the FEP database. 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. Some-

times 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], [Buhmann et al. 2010b].

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. 2011b], [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.

The Nuclear Energy Agency (NEA) published a FEP list for geological disposal of radioactive waste in 2000 [NEA 2000]. It is currently being updated by the FEP Task Group of the NEA’s Integration Group for the Safety Case (IGSC). The updated FEP database will be web-based and is relevant to all designs of geological disposal facilities, and to all categories of radioactive waste in geological disposal facilities. A first beta version has been available since 2015 and has been evaluated in the context of the ISIBEL project by the project partners. The application has an administrative interface, through which the NEA will be implementing its

updated International FEP list and national member organisations can define their own project-specific FEP lists, with version control and review and approval processes built in. It also has a public interface through which published versions of the FEP lists and links between them can be viewed.

5.3.2 Scenario-Development Methodology

5.3.2.1 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. 2010d], 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 2010] and is depicted schematically in Figure 5.4.

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.

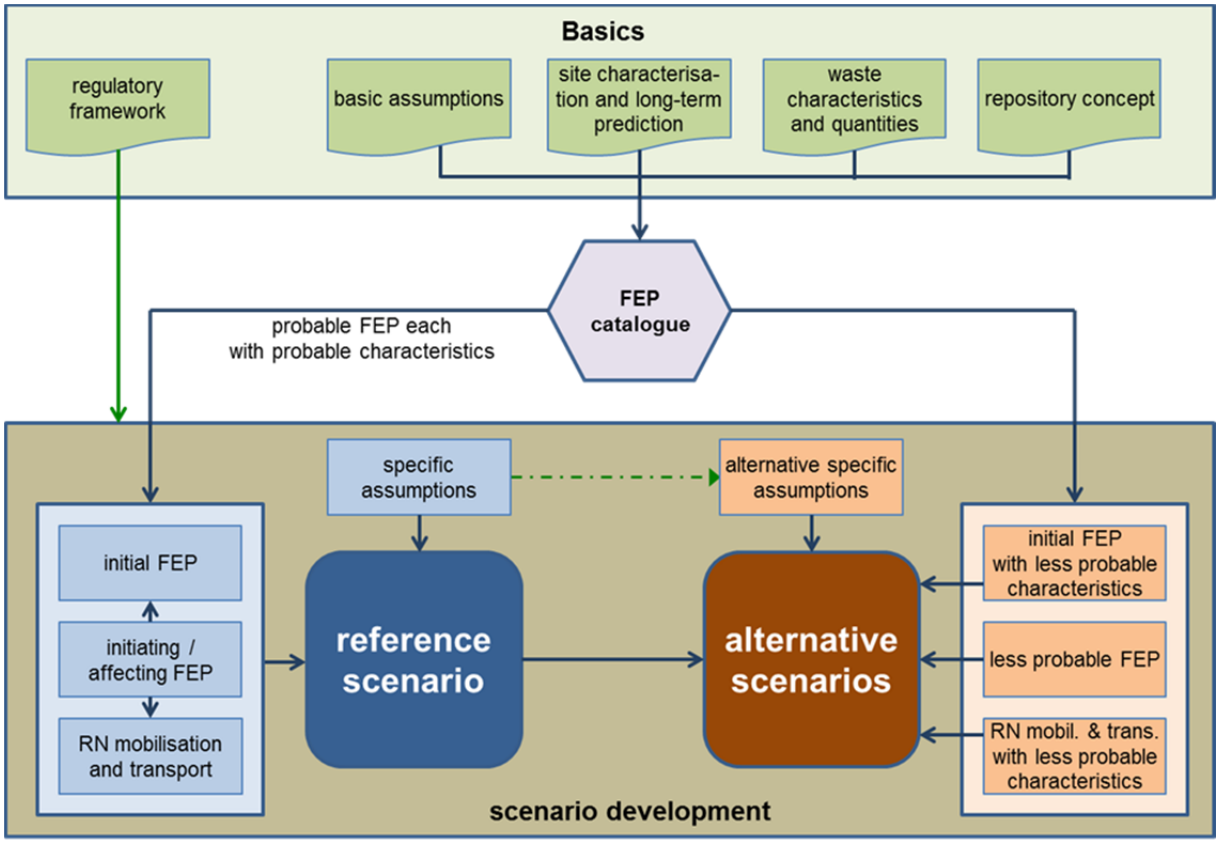

Figure 5.4: Scenario development methodology applied in R&D project VSG, modified [Beuth et al. 2012a]

- 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 re-

pository (see chapter 4.3). For heat-generating waste, two main repository variants have been analysed: drift emplacement of POLLUX® and CASTOR® casks (variant 1 + for comparison: horizontal borehole emplacement of CASTOR® 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 2010] 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.

5.3.2.2 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 5.4). 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.

<u>Near field, eastern area (heat-generating waste):</u> 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).

<u>Near field, western area (waste with negligible heat generation):</u> 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 subrosion 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 subrosion 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.

5.3.2.3 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 5.4):

- 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 5.7).

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 release 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 5.5).

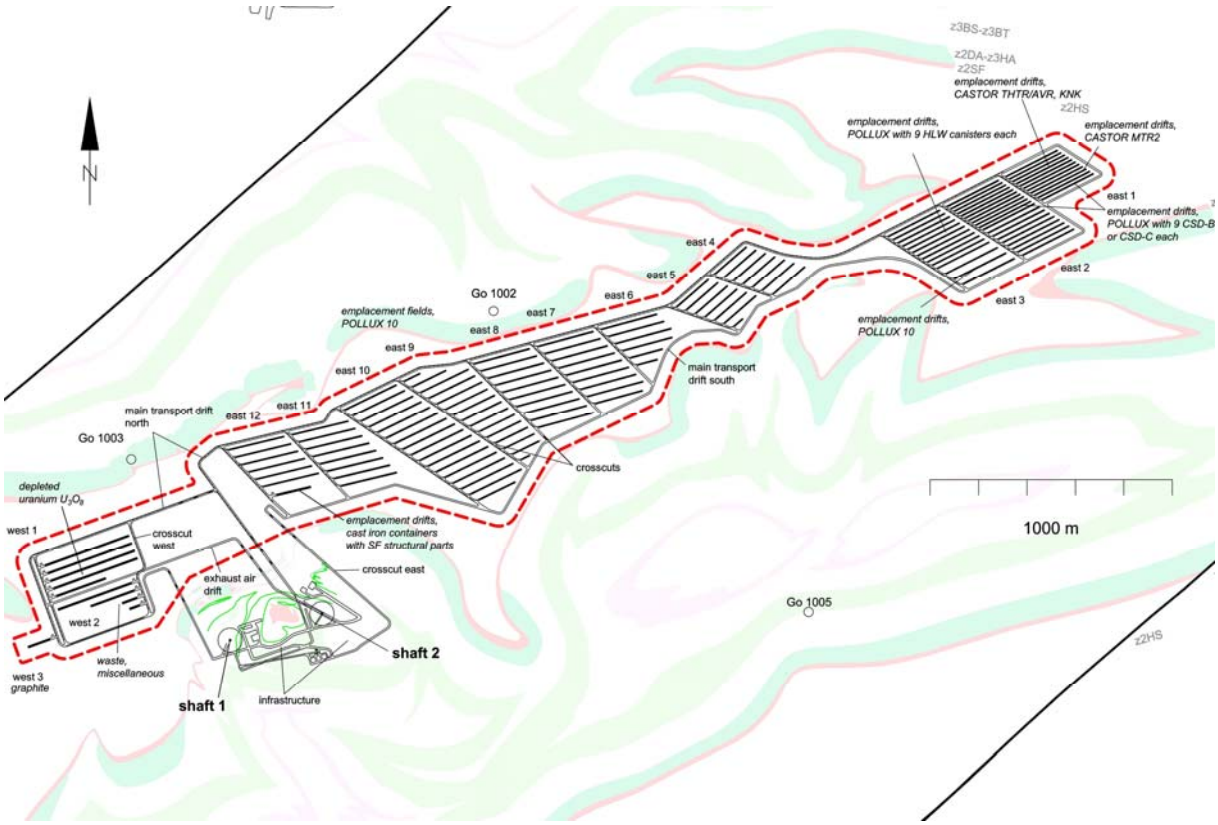

Figure 5.5: Example of a CRZ (dashed red line) in a repository in a salt formation [Fischer-Appelt et al. 2013], modified

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 subrosion processes. Removal of the bedrock can occur mechanically (erosion) and also chemically as dissolution of rock salt by groundwater (subrosion). 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 subrosion

When rock salt comes into contact with water it can be dissolved as long as the water is subsaturated. 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 subrosion 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 [Zirgast 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 remaining 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 subrosion 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 subrosion 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 subrosion 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, subrosion of around 100 to 200 m of salt would be possible. [Köthe et al. 2007], p. 178], forecasts a future subrosion 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 subrosion rates would then give rise to subrosion of 10 to 50 m of salt over the investigated period of a million years. Because it is thought un-

likely that diapirism will increase over this period, it is not expected that the subrosion rates will be significantly enhanced by this process.

If it is also assumed that top salt would only reach depths at which subrosion can no longer take place during the later stages of the million year investigation period, it can then also be assumed that subrosion 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. Subrosion 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 fluviatile 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 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 metres [Heemann 2015].. No additional effect on the preservation of the CRZ for the Gorleben site due to container sinking therefore has to be considered.

5.5.5 Effects due to heat production of the waste

As a consequence of the deposition of heat-generating waste, the temperature at the top of the salt dome increases from initially 24°C by about 10 to 15°C, depending on the disposal variant, during the first 3,000 years after the emplacement of the waste. Afterwards the temperature at the top of the salt dome decreases slowly. 10,000 years after the emplacement the temperature at the top of the salt dome is still increased by several degrees centigrade in comparison with the initial temperature [Bollingerfehr et al. 2012]. During the warming of the salt rock its volume grows according to its thermal expansion coefficient. This leads to an uplift of the top of the salt dome by about 2 m. The consequences of both effects, the warming of the salt table and the alteration of its topology by partial uplift, have to be examined with respect to their potential impact on the preservation of the CRZ.

5.5.5.1 Uplift of the salt table

The change of the topography of the salt table due to the thermal expansion of the salt rock is very small compared with the natural existing differences in the local depths of the salt table and the overlying cap rock. Therefore, a significant influence of the slowly uplifting of the salt table with a total amount of 2 m within thousands of years will not significantly change the groundwater flow patterns and thus will not influence the subrosion rate.

5.5.5.2 Warming of the groundwater above the salt dome

Due to the dependency of the density of the groundwater on its temperature, the changes in the temperature field will result in changes of the groundwater density and therefore in changes of the local hydraulic potential gradients, which in turn might affect the groundwater flow and thereby the subrosion rates.

Generally the vertical groundwater temperature profile near the investigated site declines from higher values in greater depths to a minimum value of about 10°C near the surface. The natural horizontal temperature distribution in the groundwater above the salt dome shows significant variations. In a depth of - 150 m ASL, the temperature above the salt dome is partially more than 9 degrees higher than in regions beside the salt dome [Klinge et al. 2007]. The reason for this is the higher thermal conductivity of the salt rock compared to the rock types adjoining the salt dome. This provides a higher heat flow density in the area of the salt dome, which leads to a more intensive warming of the groundwater by geothermal energy above the salt dome than besides it. This horizontal temperature variation will be superimposed by the warming of the salt rock due to the heat production of the waste. This results in a doubling of the steepness of the temperature decline above the salt dome from initially about 15°C within 250 m vertical distance to 30°C within 250 m vertical distance. The horizontal temperature variation is even more than doubled as a consequence of the deposition of heat-generating waste.

The temperature variation alters the density of the groundwater only very little. The temperature increase leads to a density decrease of the groundwater due to the thermal expansion of the water. Directly above the salt dome the groundwater is in equilibrium with the soluble salt. In this case an additional effect of the heating is the increase of the salt solubility. This increase of the solubility is as well as the thermal expansion rather small within the relevant temperature range between 20°C and 40°C. The quantity of NaCl, that can be dissolved in 100 g fresh water, amounts 35.92 g at a temperature of 20°C and 36.46 g at a temperature of 40°C [Westphal et al. 2010]. However the density of a saturated NaCl solution will decrease from 1.1999 g/cm³ at a temperature of 20°C to 1.1914 g/cm³ at a temperature of 40°C [Westphal et al. 2010] due to the simultaneous thermal expansion. The result of warming the groundwater from 20°C to 40°C in consideration of thermal expansion and solubility increase is therefore a density decrease of about 0.7%.

The density of the groundwater above the salt dome without heat-generating waste ranges from 0.9997 g/cm³ for the fresh water near the surface at a temperature of about 10°C to 1.1999 g/cm³ for the saturated NaCl solution at the salt table at a temperature of 20°C. A consequence of the different degrees of salinity is a density increment from the surface to the salt table by about 20%. This causes a density stratification of the groundwater and reduces the subrosion of the salt by preventing fresh water from direct contact with the soluble salt.

The density stratification due to the density increment from the surface to the salt table by about 20% as a result of different degrees of salinity cannot be overridden by the density decrease of 0.7% at the bottom of the groundwater body above the salt dome in consequence of the warming due to the deposition of heat-generating waste.

An additional consequence of the warming and thereby the slight reduction of the groundwater density contrast between surface and salt table might be a reduction of the evolution of density-driven flow vortices above the salt dome, which were found in hydrogeological model calculations (chapter 4.1.4).

Therefore the warming of the groundwater will not increase the subrosion rate and therefore not impede the preservation of the CRZ.

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. For 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.

It is known from laboratory and in-situ testing that the integrity of the barrier can also be impaired if a fluid pressure exceeds the minimum principal rock stress. The integrity indicator “state of stress” is thus not only analysed in relation to the dilatancy boundary value, but also has to be compared with the fluid pressure that has developed so far.

5.6.2 Rock Mechanics Calculations

Numerical model calculations utilising finite-element methods can be used to determine the state of stress which is reached in the rock salt barrier due to the interaction of the processes mentioned in the chapter before. Prerequisites for the success of numerical modelling are the consideration of the specific repository characteristics and of the geological structure of the repository site and the application of suitable constitutive equations and adequate numerical algorithms.

Within this project it has been aimed to compare the suitability of commonly used constitutive equations such as “Minkley“[Minkley & Mühlbauer 2007], “BGR-A“ [Hunsche 1981], “BGR-B“ [Hunsche 1995], ”Hou-Lux“ [Hou & Lux 1998], [Lux et al. 1999] and "TU-Bs" [Hampel et al. 2016] for the calculation of thermo-mechanical processes within a repository and the surrounding rock mass as discussed in appendix C. Additionally the approaches by “Günther-Salzer” [Günther et al. 2015], “Hampel” [Hampel 2015] and “Heemann” [Heemann 1989], [Heemann 2015] should be included in the evaluation due to their relevance in this field of research.

But a simple comparison of calculated models will not deliver any credible analysis of the reliability of the models. They have to be analysed concerning the physical ideas they are based on or, if their mathematical description is purely phenomenological or empirically based, whether it is in line with best physical and empirical knowledge. For the evaluation of the suitability of the constitutive equations it must be examined whether the mathematical formulation is apropriate to characterize the physical behaviour correctly, whether it is suitable to extrapolate the behaviour beyond the range of the measured data and whether it describes the observed behaviour at all. Some major aspects of the physical basis for modelling the mechanical behaviour of dry rock salt are described in the chapter “*constitutive modelling*” of appendix B. This implies the description of the main physical equations for the underlying processes of gliding of dislocations including the processes of dislocation multiplication and annihilation leading to a physical modelling of transient creep due to hardening. Though from a physical point of view for stationary creep a hyperbolic law should be expected. It can be taken from experimental data that the stationary creep rate of rock salt can also be very well modelled by the phenomenological Norton-approach. Nevertheless, looking for the behaviour for stresses and temperatures outside this laboratory range, the results of the two attempts may differ, though at laboratory the differences can not be resolved.

Most of the constitutive equations mentioned above are based on phenomenological descriptions of correlations observed in laboratory experiments. The parameters for this kind of constitutive equations can generally be fitted to measured data. But the quality of the models does not only depend on the quality and bandwidth of the measured data set but may also strongly depend on the mathematical relations used. If they are not in accordance with the physical processes they may be fitted to experimenta data but are not able to predict the behaviour beyond the experimental data range. It has not been unusual so far that in case of new data some phenomenological corrections had to be implemented to overcome the discrepancies between data and model or that the parameters of the unchanged models have been refitted for the new regime. But this is not in accordance with the demand of having one model with one set of parameters. They should have – if possible – a clear physical meaning enabling the user to judge the plausibility and creditability of the model and thus the calculations. As a general rule and somewhat independant of physical modelling it can be taken that the reliabilibility of a model is the higher the less parameters are necessary to reproduce the experimental data over a wide range of stress, temperature, strain etc. (Ockhams razor). It is

a strong indication that the mathematical/physical functions of the model represent the true physical processes.

Especially in case of very low stresses there is a significant lack of experimental data. So up to now it is not clear which creep mechanism and thus which creep rates appear at the low stresses far from the cavities or reached after long times of creep and how they affect the behaviour in the near vicinity of the mine openings of the mine openings.

Regarding damage and permeability there is still a lack of experimental data and mathematical (physical) models describing their anisotropic development. Most modern models include the dependency of the dilatancy on the deviatoric as well as maximum stress component, but still keep the dilatancy itself and the resulting permeability isotropic. This simplification not only influences the calculated development of stress but also may seriously affect the calculated direction and amount of the flux of fluids. These aspects are addressed in the chapter "*On the Analysis of Experimental Data on Creep of Rock Salt*" of appendix B.

A brief listing of important aspects for the assessment of the qualification of different numerical modelling methods is presented in the last chapter "Comparison of *Numerical Techniques*" of appendix B.

5.6.3 Practical evaluation of the integrity of the geological barrier

5.6.3.1 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

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

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]:

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

with A = 0.18 d^{-1}, Q = 54 kJ/mol, R = 8.314 · 10^{-3} kJ/(mol·°K), n = 5, σ* = 1 MPa (reference stress) and T = 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]).

5.6.3.2 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 over burden 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 5.6 shows a finite-element model of the Gorleben salt dome which was used to analyse the temperature effects of different repository designs.

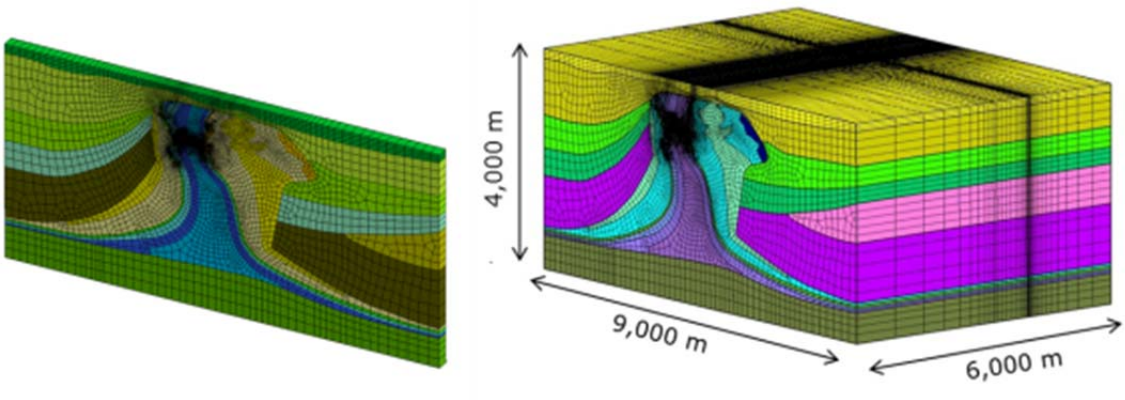

Figure 5.6: Set-up of a finite-element model of the Gorleben salt dome (left: 2-D model, right: extruded 3-D model)
[Heusermann et al. 2012a], [Heusermann et al. 2012b]

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.

Figure 5.7 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.

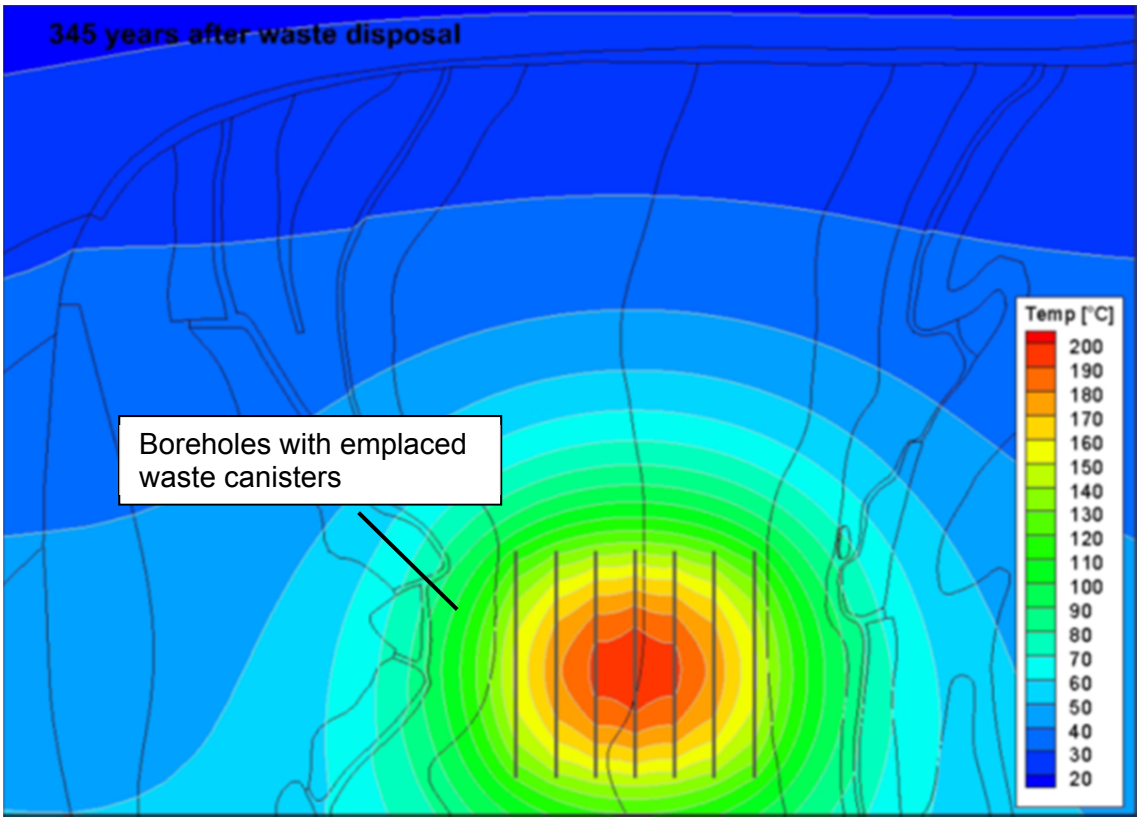

Figure 5.7: Predicted temperature in the rock salt 345 years after waste emplacement (borehole disposal concept) [Heusermann et al. 2013]

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 5.8 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.

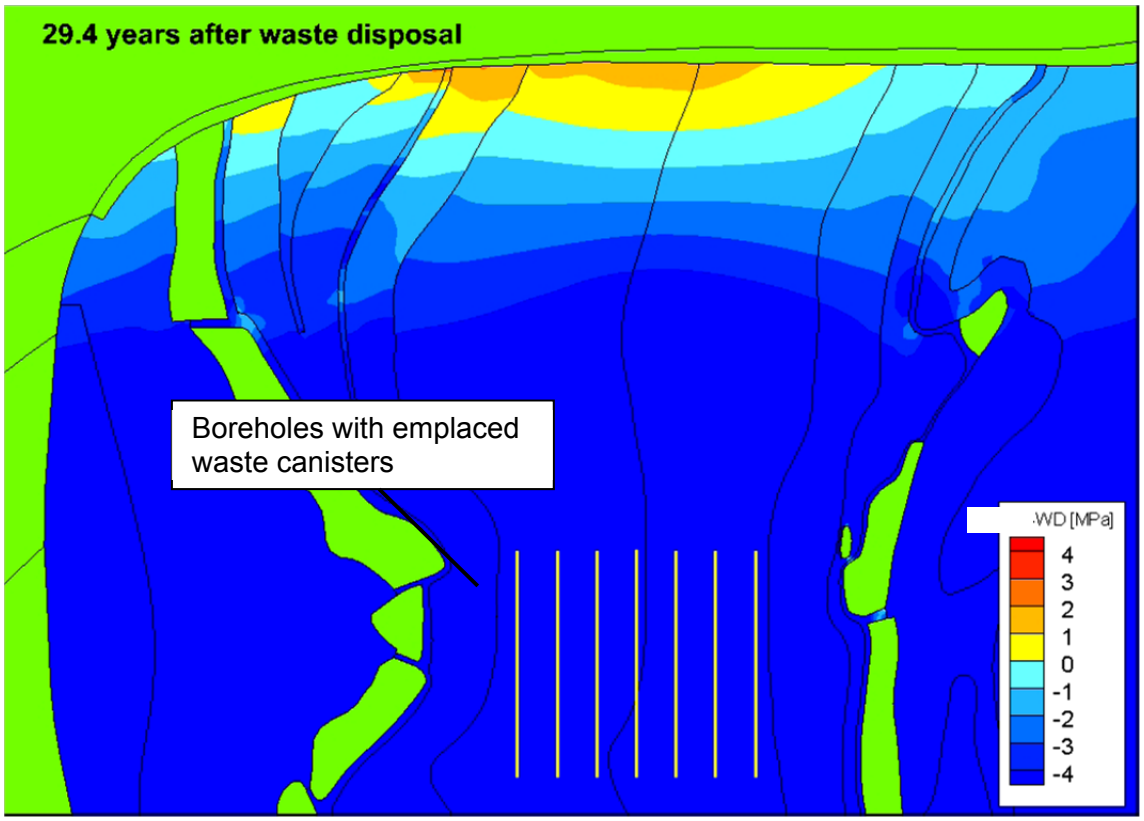

Figure 5.8: Difference between minimum principal stress and hypothetical depth related fluid pressure [Kock et al. 2012]

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).

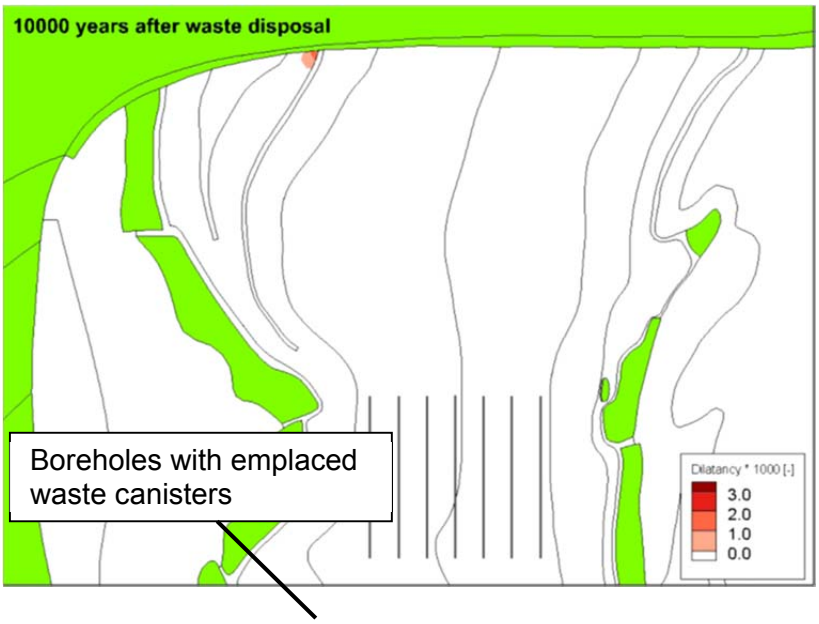

Figure 5.9: Areas in the rock salt with octahedral shear stresses above the dilatancy boundary [Heusermann et al. 2013]

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 5.10 shows the elements representing the Zechstein and older units. Overlying and surrounding rocks are removed in this illustration.

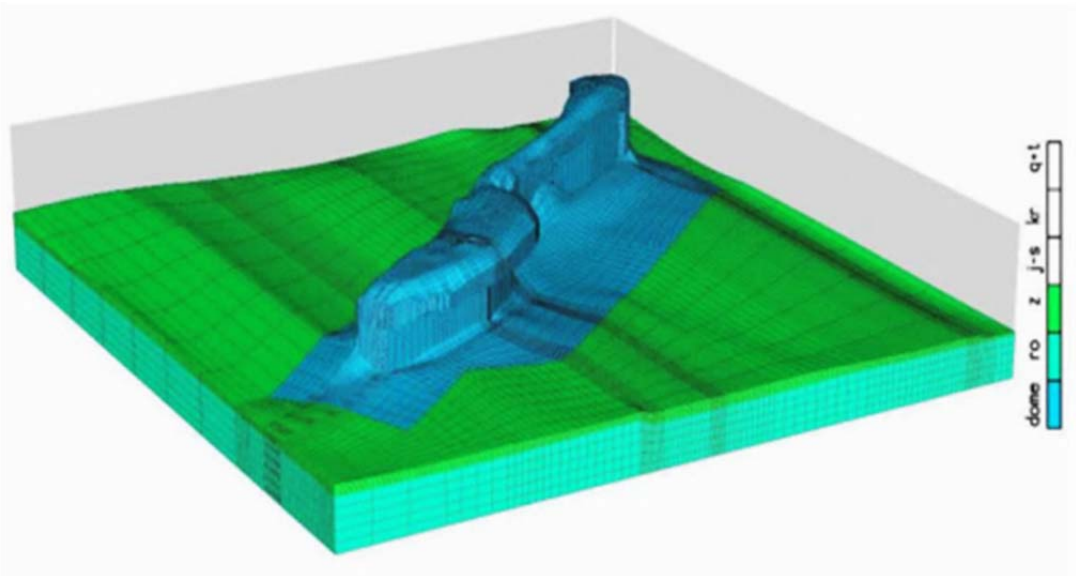

Figure 5.10: Three-dimensional finite-element model of the Gorleben salt dome [Heusermann et al. 2012a], [Heusermann et al. 2012b]

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 5.11). These zones are restricted to the top of the salt dome and do not extend into it.

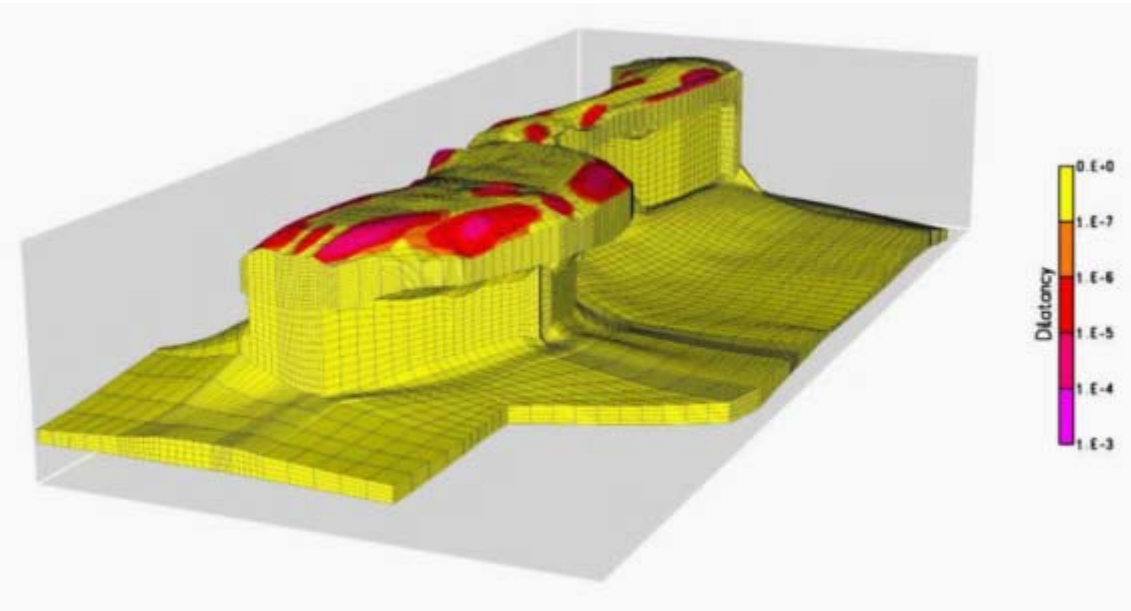

Figure 5.11: Dilatant zones at the top of the Gorleben salt dome during glacier melting [Kock et al. 2012], see also [Heusermann et al. 2012b]

5.6.3.3 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 Assessment 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 [Kreienmeier 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[8], the geotechnical barriers. The barriers should prevent or minimise solution inflow[9] 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

[8] 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])

[9] In this chapter "solution inflow" is understood the "Inflow from above ground, overburden and formation solutions"

post closure phase according to the Safety Requirements [BMU 2010]. 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 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], [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[10] 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 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 5.9). 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.

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]. The extension of the functional lifetime was investigated in detail considering technical regulations in form and content, mathematical aspects as well as constraints from rational thermodynamic [Müller-Hoeppe et al. 2016]. As a result it is confirmed that there is no limitation on the procedure if the relevant impacts and resistances are derived with respect to the extended functional lifetime.

[10] The corresponding term in the sense of construction engineering is "actions" (cf. [DIN 2009])

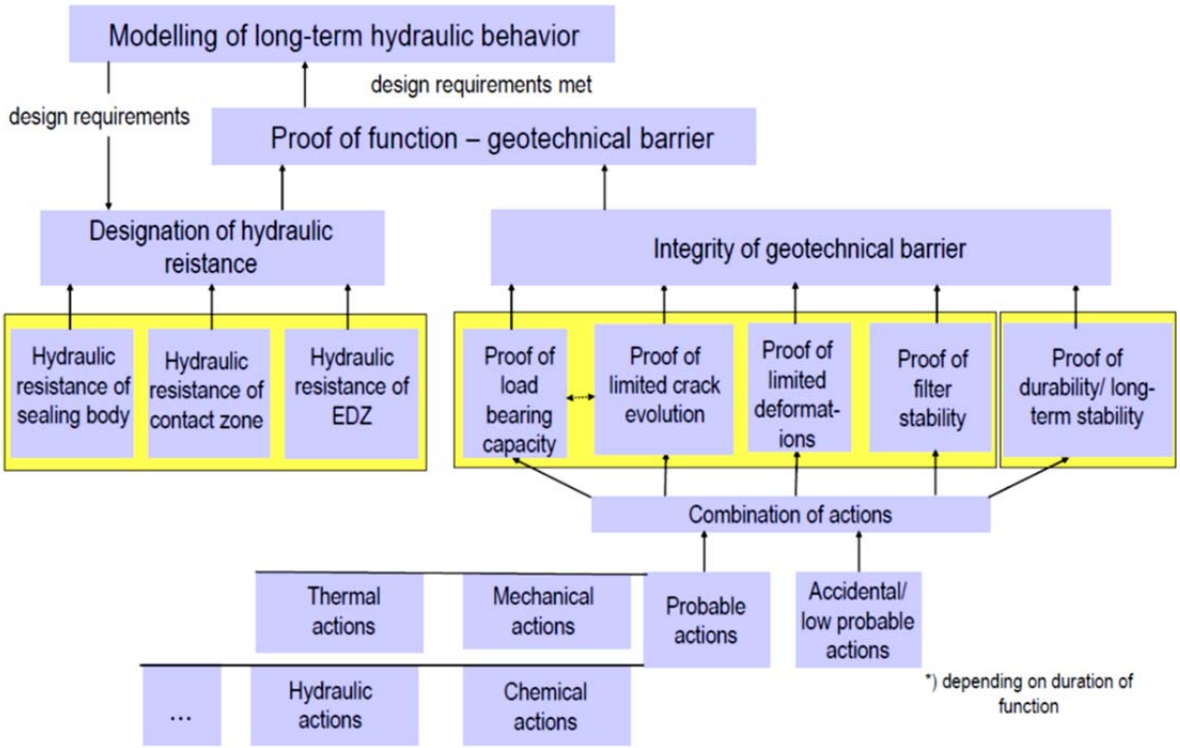

Figure 5.12: **The structure of the technical functional proof. Here "actions" are synonymous with "impacts". Hydraulic impacts = hydromechanical impacts, thermal impacts = thermochemical and thermomechanical impacts**

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 Reliability - a special aspect of dimensioning

According to technical regulations [EUROCODE 7], it is recommended in geotechnical design to derive the characteristic value of a basic property such that the calculated probability of a unfavorable value governing the occurrence of a limit state, e.g. failure of a structure or an inadequate low hydraulic resistance is not greater than 5% if statistical methods are used. Thus, due to practical experience in geotechnical design, an adequate reliability level is achieved. This approach – to choose the characteristic value close to the mean value – turned out to be adequate because the dimensions of a geotechnical structure are so large that local imperfections can be neglected.

If an engineered geotechnical structure, e.g. a sealing component, has effective properties that are comparable to the surrounding rock, it is reasonable to design the sealing component using a level of reliability similar to that of the surrounding rock.

As already mentioned above, in engineering design practice the semiprobabilistic partial safety factor design method is applied to demonstrate with sufficient confidence that actions / impacts remain smaller than resistances. The probability distributions for actions and resistances are selected and characteristic values (e.g. nominal, mean, quantile) of these dis-distributions are chosen. The safety index, based on these distributions, corresponds to the accepted maximum likelihood to reach the ultimate limit state. Taking into account the safety index, the partial safety factors are defined for characteristic values of actions and resistances. If the rated actions are smaller than the rated resistances, the safety function engineering structure is reliably demonstrated. Typically, material properties incorporated within the resistances are taken as probability functions usually independently from the dimensions of the structure. When applying this approach, e.g. to the hydraulic conductivity of a geotechnical sealing component, the length of the sealing component is regarded solely as a multiplier. Neglecting the spatial distribution of the local permeability values leads to very conservative results and the results do not agree with the practical experience in geotechnical design.

As an example, an existing pilot seal – the Asse-Vordamm – made of salt concrete and built in 1991/92 as part of a study of a geotechnical barrier system that was investigated by 34 local permeability measurements performed in 12 boreholes was regarded [BfS 2010]. The classical engineering approach neglecting the spatial scattering of permeability values was applied using "cautious estimates" for permeability values of the sealing body, the contact zone, and the EDZ to determine the hydraulic resistance. The limit state function was derived from Darcy´s law, the dimensions of the sealing component and the admissible flow rate defining the ultimate limit state. To assess failure probability, Monte Carlo Simulation was used. As a consequence of disregarding spatial variability, the meaningless result was that the length of a single seal has to be 3 m to achieve a failure probability of 10^{-2} and a length of 170 m to achieve a failure probability of 10^{-4}. This result does not agree with practical engineering experience.

Alternatively, the spatial distribution of the local permeability values was taken into account. For this approach it is assumed that each sample is a representative for the whole sealing structure distinguishing the sealing body, the contact zone, and the EDZ. In reality, though, all samples come from one and the same seal and represent the hydraulic conductivity at different locations of the seal. The question was how to "upscale" these data (control space ~10 cm) in order to derive conclusions for the whole seal. A geostatistical approach was used. The "true" seal (the one the data came from) was considered to be one realization of a location-dependent, random variable ("random function"). A conclusion about probability density functions and spatial behavior of the random function from the data must be derived. In this case, variograms were generated and a stationary random distribution was assumed. Sample realizations of the so described random variable (one realization corresponds to one possible seal) were performed. The hydraulic flow for a given pressure gradient for each realization was calculated and from this the effective hydraulic conductivity of each realization of the seal. Statistics were performed for these effective conductivities and it could be concluded that for a seal of 8 m length that is correctly described by the underlying assumptions with

a likelihood of 1/10,000, the effective conductivity is smaller than $1.3*10^{-23}$ m² (statistical confidence 95%) [Röhlig et al. 2014].

By this comparison, the geotechnical experience was cautiously underpinned that an adequate safety level is reached by using cautious estimates of mean values of the effective conductivity for the whole structures.

5.7.5 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.

5.7.5.1 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 4.3.7.2, Figure 4.33). 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.

5.7.5.2 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 type Asse, 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 bitumen) 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.

5.7.5.3 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 4.3.7.3, Figure 4.34).

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 4.32). 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 settlings [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 [Müller-Hoeppe et al. 2012a].

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 suuficiently 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.

5.7.5.4 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 of the assumptions of the preliminary dimensioning,
- verification of models: Admissibleness 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 hypopthetical 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 limitation of the local disintegration 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 limitation of the local disintegration 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 hypothetical 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 compu-

tational 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 hypothetical drift seal and finally for the entire sealing system.

5.7.6 Evaluation of the sealing components for the Gorleben-specfic 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 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.

5.7.6.1 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 con-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 disintegration criterion is complied with but the fluid pressure criterion will be temporarily and locally infringed, (Figure 5.13). 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].

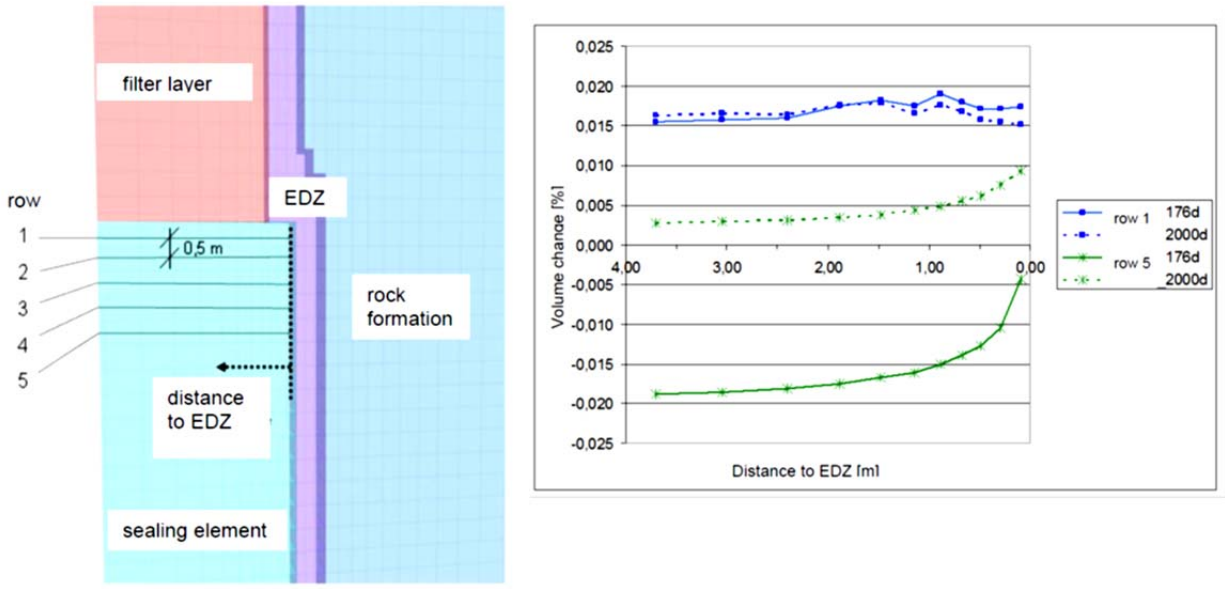

Figure 5.13: Thermomechanical calculations for the bentonite shaft seal. Volume change on selected sampling points for the dates 176 days after construction (maximum fluid pressure load) and 2,000 days after construction (increasing saturation) [Müller-Hoeppe et al. 2012a]

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 Bischofit-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, disintegration in the bentonite remains within tolerable limits. In VSG 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 stiftness 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 of the shaft sealing system, no additional investigations were performed for earthquakes.

Meanwhile a detailed integrity proof of the salt concrete seal under earthquake loading is available (Appendix C). Exemplarily, it is shown that the effect of the earthquake loading is insignificant in the sealing body and the surrounding rock salt. Due to lack of data on the dynamic material properties of the contact zone its evaluation is still pending.

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.

5.7.6.2 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 hypothetical 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 4.3.7.2, Figure 4.33). 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

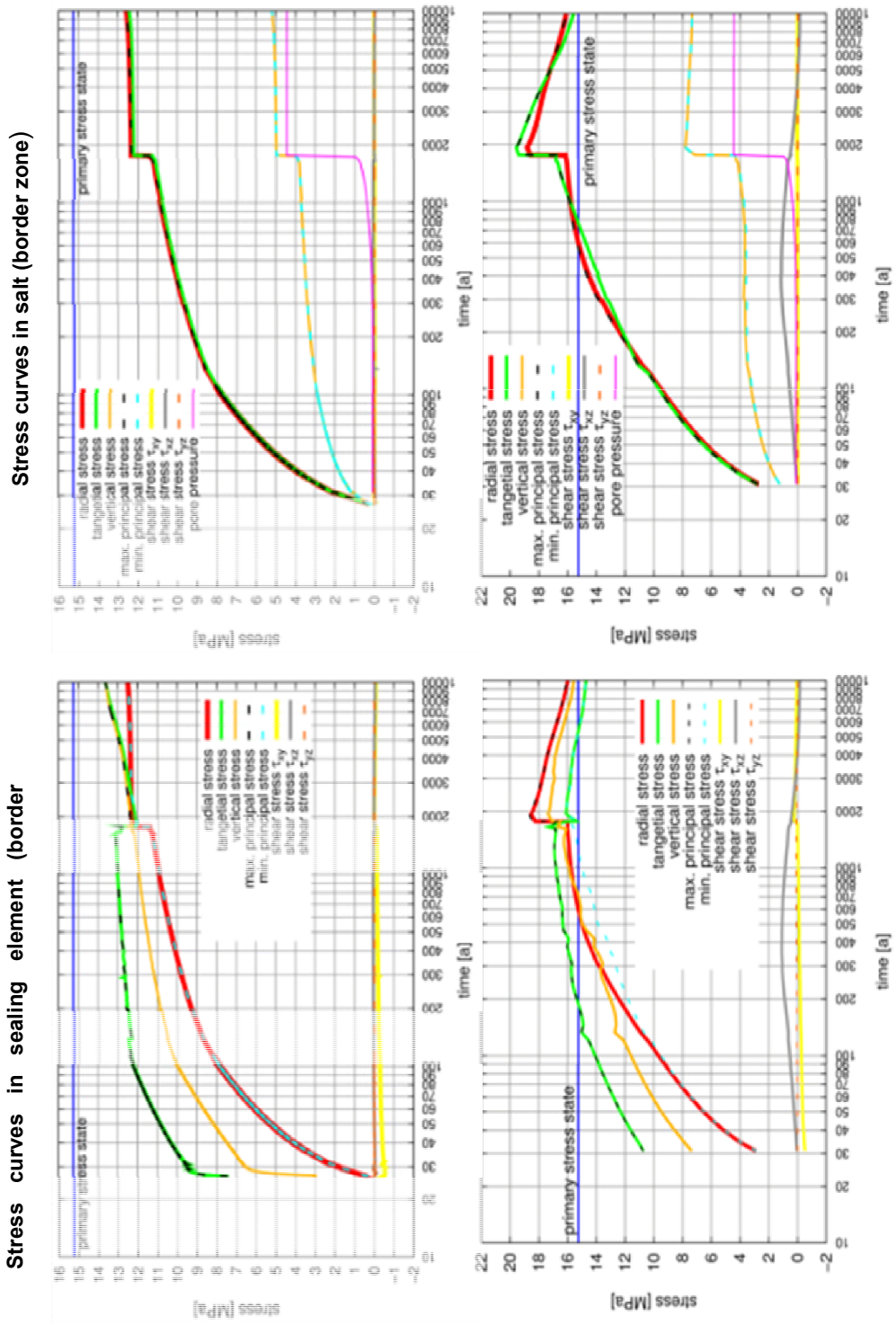

Figure 5.14: Stress curves on two selected sampling points in the contour area of the salt concrete sealing element (top: without thermomechanical impacts of the repository, bottom: with thermomechanical impacts of the repository) [Müller-Hoeppe et al. 2012a]

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 5.7.5). 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.

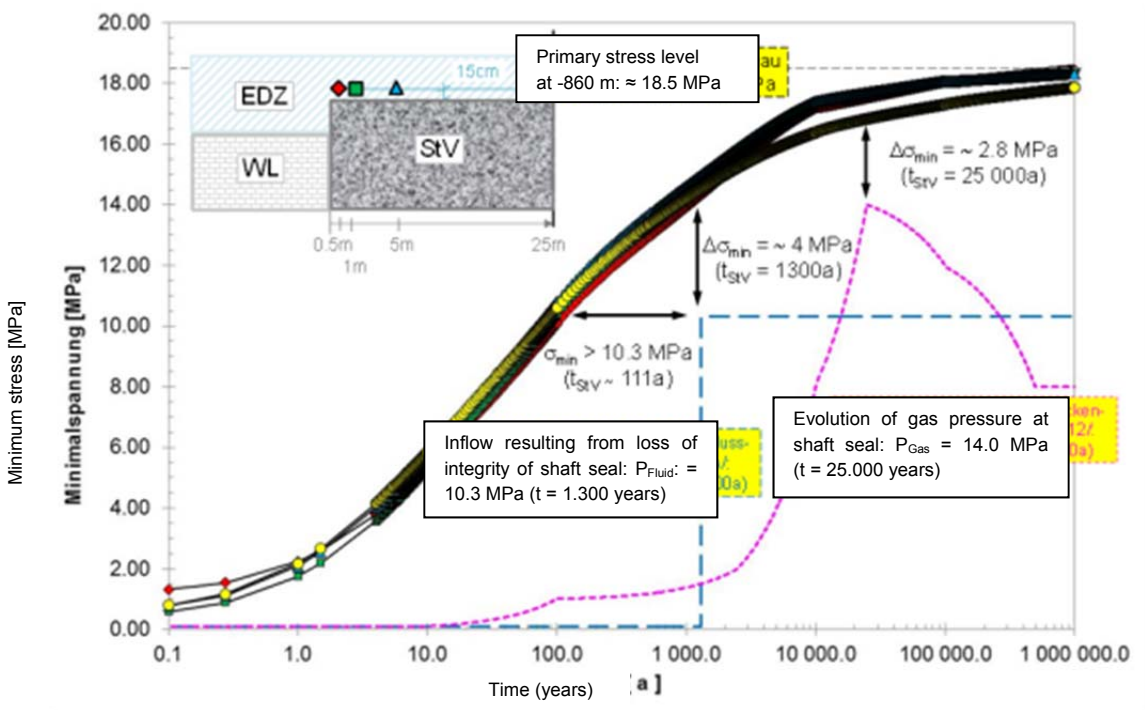

Figure 5.15: Comparison of the stress conditions in the EDZ and the solution and gas pressure from one side only for the fictitious drift seal [Müller-Hoeppe et al. 2012a]

5.7.6.3 Complete closure system

For the sealing system described in chapter 4.3.7, 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, and
- 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 Bischofit 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 2010].

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 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®, 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® cask with spent fuel from light water reactors, and in case of the direct disposal of CASTOR® 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 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 effective dose are specified in the Safety Requirements [BMU 2010] (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 an early phase of the R&D project ISIBEL [Buhmann et al. 2010d]. 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. 2010d] this value (W) is set to 500 m³/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. 2010d] the criterion K_{RGI} is set to 0.1 mSv/year.

The calculation scheme is

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

DKF = Dose conversion factors (calculated by use of a biosphere model (e.g. [Pröhl & Gering 2002]) and the dose factors given in General Administrative Rules for the Radiation Protection Ordinance [AVV 2012]).

In the R&D project ISIBEL, the application was set up for the assessment of a radionuclide release in the liquid phase (see chapter 5.9.1.1). 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 for 2-phase-flow exists (chapter 5.9.1.3).

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 5.16).

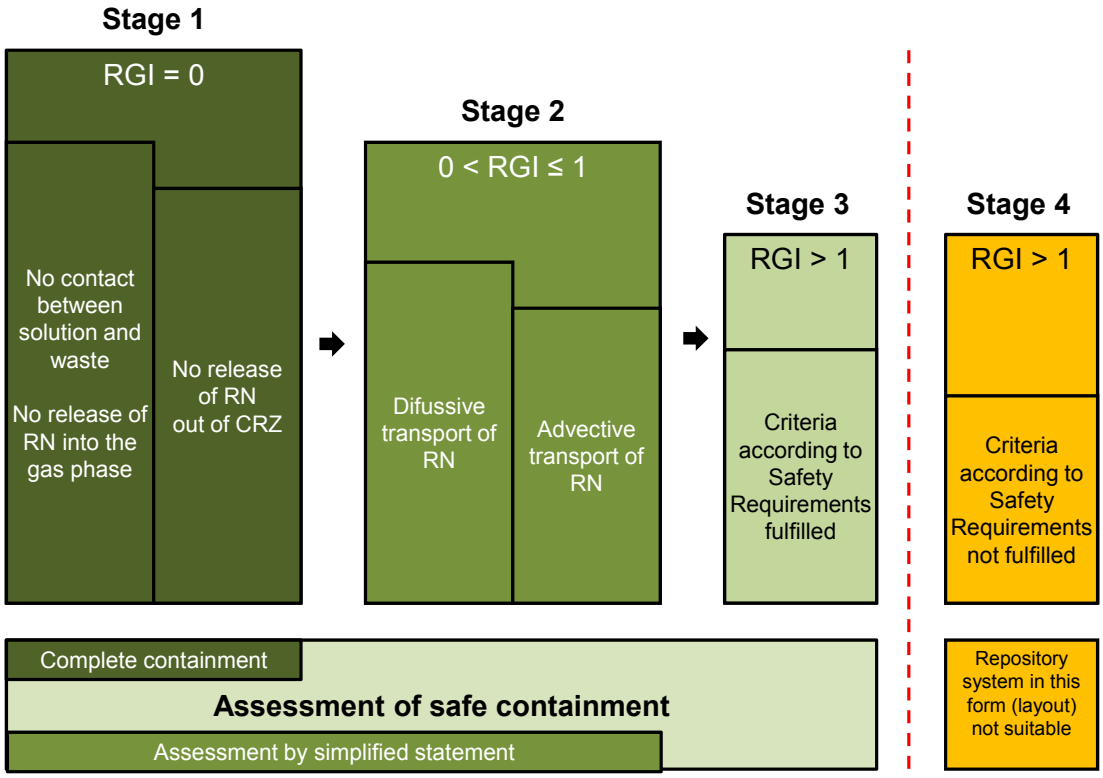

Figure 5.16: Staged approach for the long-term safety assessment (after [Mönig et al. 2012])

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 5.16). If radionuclides are released from the CRZ, safe containment has to be demonstrated. For this purpose the RGI is applied.

5.9.1.1 Radionuclide release in the liquid phase

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 in the simplified radiological statement for an 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 multiplying the radionuclide flux out of the CRZ of every radionuclide Si [Bq/a] with its dose conversion factor D_{CFi} [(Sv/a)/(Bq/m^3)] and dividing it by the annual consumption of water W [500 m³/year] and the criterion K_{RGI} for one individual [0.1 mSv/year 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 project ISIBEL deterministic calculations were carried out for a generic repository to evaluate the concept of the RGI [Buhmann et al. 2010a]. This concept has then successfully been applied in the preliminary safety case for the Gorleben site (VSG project, [Larue et al. 2013]). In the follow-up project ISIBEL II the focus was set on probabilistic calculations based on the disposal concepts developed in the VSG using the scenarios of the VSG. For a radionuclide release in the liquid phase, all calculations illustrate that transport of radionu-

clides by advection within the CRZ does not play a role for probable and less probable evolutions of the repository system. The radionuclides are transported by diffusional processes determining the calculated consequences. The diffusional transport is quite slow and a significant release is calculated for times after 105 years, see Figure 5.17

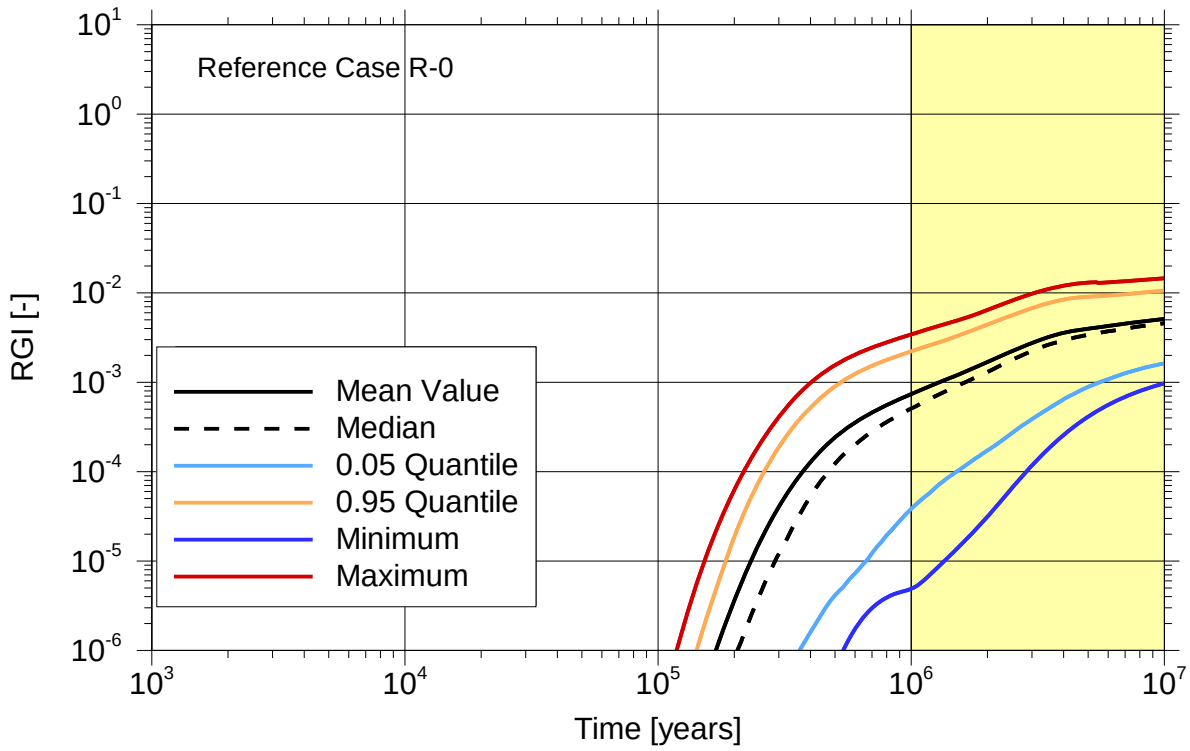

Figure 5.17: Temporal evolution of relevant statistical parameters of the RGI [Buhmann et al. 2016], modified

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. The results show the importance of a comprehensive and profound knowledge of transport processes in backfill material. Especially for material with a low porosity (e.g. compacted crushed salt) there is still a need for future R&D activities. It is expected that further investigations will allow to reduce conservative assumptions in regard of transport processes in compacted salt.

The analyses carried out in the ISIBEL projects illustrate the necessity of both, deterministic and probabilistic calculations: Probabilistic calculations are not only important as a tool for managing uncertainties. They are also required to deal with the classification of scenarios in different probability classes as specified in the safety requirements.

Probabilistic cases are the only possibility to transfer the scenarios in a manageable set of calculation cases [Buhmann et al. 2016]. The importance of deterministic calculations is

especially based on the possibility to communicate the results in a clear and straightforward way, e. g. by parameter variations. The parameter variations are based on the deterministic reference case, for alternative scenariios deterministic cases are not necessary.

For the analysis of the scenarios, well-advanced integrative models are available for the numerical assessment of the near field and its processes, e. g. the RepoTREND near field model LOPOS [Hirsekorn et al. 1999]. Basic principle of near field models such as LOPOS is to use the solubility or concentration limits of every considered element as the major parameter to represent its mobility in the geochemical environment after mobilization. In order to be conservative, this parameter should be derived from the maximum expected concentration of the considered element. In order to demonstrate conservatism, a key objective of the safety assessment is to obtain a thorough knowledge of the geochemical processes, especially in the near field. For a repository for HLW this includes an adequate understanding of their long-term behavior. To model the geochemical environment and to determine solubility limits in the geochemical environment, the following prerequisites have to be fulfilled:

- the initial composition of the system under consideration,
- the relevant aqueous (or gaseous) species and solid phases,
- a thermodynamic database which contains thermodynamic data for all relevant aqueous (and gaseous) species and solid phases, and
- a suitable code, which upon input of initial system composition and database calculates its composition at thermodynamic equilibrium.

Because of the importance of solubilities in the near field, [Moog et al. 2016] assess the existing thermodynamic data needed for a geochemical modelling, whose results can be applied in long-term safety assessment for a HLW repository in a high-saline environment. For the assessment of the concentration limits of radionuclides, it is mandatory to define the system under study, and the conditions under which the concentration limits are recommended. The concentration limits of the radionuclides are assessed for the direct environment in the vicinity of spent nuclear fuel and vitrified waste from reprocessing [Moog et al. 2016] evaluates the existing database for relevant elements and results of geochemical modeling of brines in HLW repository system in salt. For all relevant elements an assessment of its database and the consequences for performance assessment are given. The report is the first overview on solubilities for all long-term safety-relevant radionuclides for salt since the SAM study [Buhmann et al. 1991].

5.9.1.2 Radionuclide release in the gaseous phase

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.

To promote the discussion about the handling and relevance of gas effects, in the ISIBEL project the generation of gas and its migration in an HLW repository have been investigated in detail [Rübel et al. 2013]. As water is essential for gas generation, the origin of water in the mine was discussed first:

- Internal water in rock salt formations (see chapter 4.1.1) migrating towards the mine due to the hydraulic gradient. Water sources are:

- hydrated minerals,
- intracrystalline water (fluid inclusions),
- intercrystalline water (brine filled “pores”) and
- large volume brine reservoirs,

- inflow of water through the shaft, and
- water content of crushed salt used for backfilling the galleries.

The water content in the waste packages (SNF and vitrified waste) has been assumed to be negligible. The main result is, that migration of water from the host rock into emplacement boreholes is of minor importance and the inflow of groundwater through shafts is of low probability. Thus, the generation of gas is expected to be low in probable evolutions of the system and of higher relevance in less probable evolutions.

Gas generation influences the release of radionuclides mainly in two ways:

1. generation and transport of volatile radionuclides, and
2. influence on advective flow of contaminated water.

Some volatile radionuclides already exist in SNF containers, e.g. Cl-36, Se-79, I-129, and Cs-135. They are mainly dissolved in water, only I-129 is expected to be of higher relevance for transport in the gaseous phase [Rübel et al. 2013]. By corrosion of waste and container material, C-14 and hydrogen (tritium) are generated. C-14 is the most relevant volatile radionuclide; it may occur as $^{14}CH_4$ (methane) or $^{14}CO_2$ (carbon dioxide). Tritium has a very low half life and is not relevant for release in the long term. Noble gases like krypton or xenon

exist in very low concentrations and are not relevant for doses and not considered further. Always an inactive carrier gas must exist to transport the volatile radionuclides, because the low amount of gaseous volatile radionuclides will be trapped in the mine.

The volatile radionuclides are either transported in the gaseous phase or in the liquid phase from the waste to the biosphere. Equilibrium between the concentrations in both phases has to be taken into account, although for conservative estimations it can be assumed that a species is either entirely in the gaseous or in the liquid phase. For methane it is often assumed, that the amount in the gaseous phase is entirely dissolved as carbon dioxide in the groundwater (aquifer), before being released to the biosphere.

The most important volatile radionuclide turned out to be C-14. Due to its relatively low half life the consequences of its release can in principle be reduced by retarding the transport into the biosphere. I-129 in any case exists in the gap of SNF matrix and is released when the container fails. If there is no water intrusion, but a failure of containers due to mechanical effects, volatile I-129 might become relevant for the dose, because its release is not related to a liquid phase.

5.9.1.3 2-Phase Flow

As discussed in the preceding sections, radionuclides are transported in the liquid and in the gaseous phase. Furthermore, advective flow of the liquid phase is strongly influenced by the volume of all kind of gas in the system. Thus the movement of fluids should be calculated preferably as a coupled process, i.e. 2-phase flow.

In previous long-term safety assessments of repositories for high-level waste in salt formations mainly the transport of radionuclides in the liquid phase has been considered, see for instance [Buhmann et al. 1991]. In later assessments – starting in the late 1990s –, the effect of gas and the transport of volatile radionuclides turned out to be important. Finally, in the R&D project VSG 2-phase flow was the basis for the performance assessment [Larue et al. 2013].

A disadvantage of 2-phase flow calculations is related to the determination of parameter values. Many parameters are available for porous media with constant porosity, e.g. for sand. Values for crushed salt are lacking, often transferred from other systems, and thus uncertain. Furthermore, crushed salt is compacting by convergence and thus has a varying porosity, which needs porosity dependent parameter values.

Furthermore, the computational models for 2-phase flow are more complex than for single fluid flow and the computing times are higher. Up to now probabilistic calculations on the basis of 2-phase-flow models have not been performed. The new computer code of the

package RepoTREND, which is in development [Reiche 2016], is envisaged to circumvent this problem.

In 1999 a first attempt was made to investigate the influence of 2-phase effects on the transport of radionuclides [Kühle et al. 1999]. In this project, a model has been developed for the computer code LOPOS [Hirsekorn et al. 1999], to take the influence of gas generation into account. Furthermore, the program MUFTE has been used to calculate 2-phase flow in a simplified mine structure and the results of these calculations have been used to test the LOPOS model. In the following, this LOPOS model has been applied in several projects, mainly for performance assessments of the Asse II mine.

In the ISIBEL project these investigations have been refined and reported in [Rübel et al. 2013]. This time, the 2-phase flow calculations have been performed using the actual version of the TOUGH2 computer code. As an example a schematic view of the discretization is shown in Figure 5.18. This configuration was used to calculate the inflow of water through a shaft (left hand) and the combined flow of water and gas in the drift system between the waste (right hand, bottom) and the shaft. Other configurations have been investigated as well and been compared to the LOPOS model mentioned above.

From the results of these calculations it is concluded that a single phase calculation is sufficient, if no additional gas is generated, i.e. if the gas phase consists only of the initial air in the mine. If gas is generated from the waste, a 2-phase flow calculation is highly recommended, where the degree of spatial discretization of the models is of less importance on the results.

Further R&D projects are recommended so that it will be possible to assess the complex behaviour of these processes in safety analyses more precisely. If gas generation from waste occurs, the effects of these gases – either radioactive or not – are relevant for the transport of all kind of radionuclides. For the parametrization of 2-phase flow in crushed salt more experiments for this distinct material are necessary.

5.9.2 Radiological consequences in the biosphere

It is to be expected that for a well designed repository the simplified radiological statement, as discussed in chapter 5.9.1 will be successful. However, if it fails, i. e. the radiological index RGI is higher than 1 (RGI > 1, Stage 3, cf. Figure 5.16) the calculation of radiological consequences in the biosphere comes into action. This calculation is different for transport in the liquid and the gaseous phase, respectively.

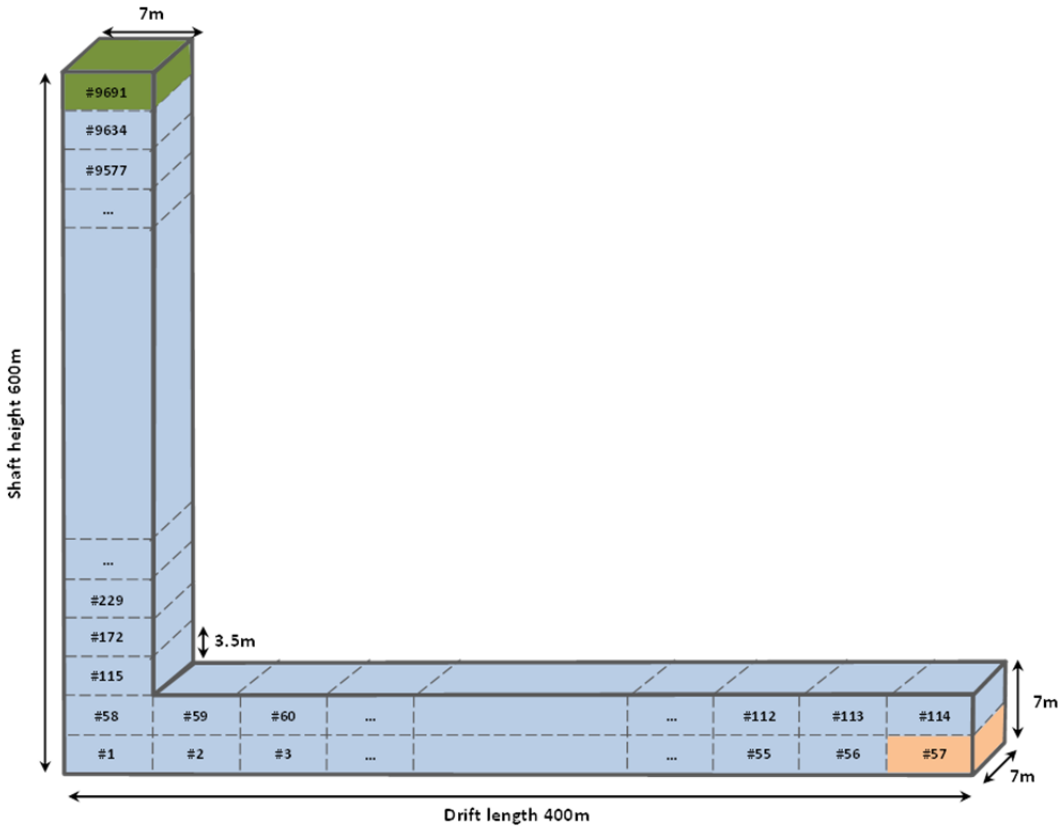

Figure 5.18: Schematic picture of the layout of a TOUGH-model (not to scale) [Rübel et al. 2013]

5.9.2.1 Transport in the liquid phase

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.

Already in the year 2002, [Pröhl & Gering 2002] added further exposure pathways to those of the AVV:

- 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 5.19 shows the exposure pathways applied for the calculation of corresponding DCF according to [Pröhl & Gering 2002]

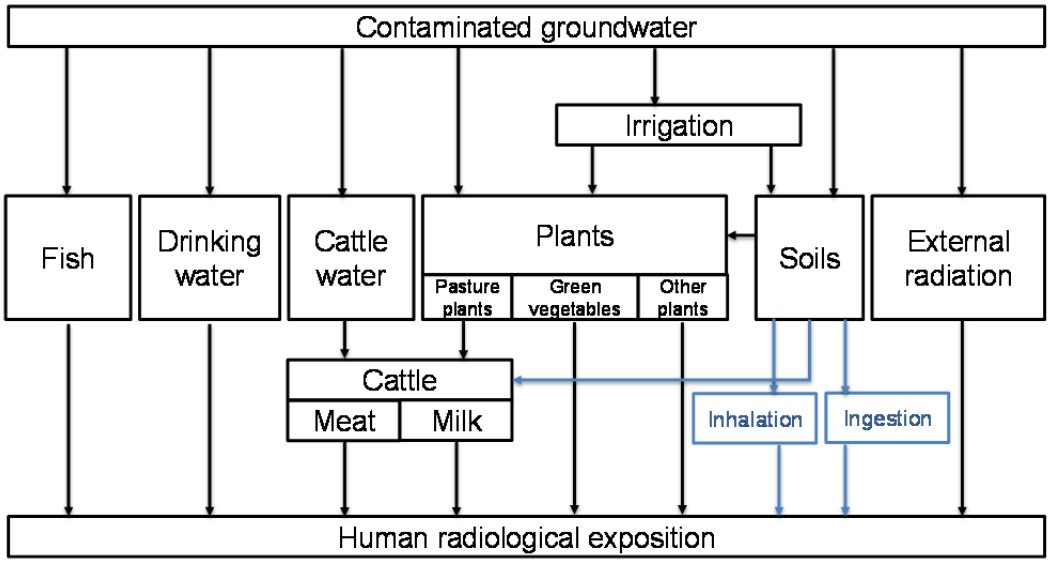

Figure 5.19: Exposure pathways to be modeled according to [AVV 2012]. Blue arrows and boxes represent exposure pathways added according to [Pröhl & Gering 2002]

In R&D project VSG all calculated RGI values for the liquid phase are lower than 1 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. 2010d] show that dilution effects in the geosphere yield exposures several orders of magnitude lower than the exposures calculated at the boundary of the CRZ.

5.9.2.2 Transport in the gaseous phase

Up to now, in comparison to transport in the liquid phase, little experience exists for transport in the gaseous phase. For example, no calculations of a transport from the CRZ to the biosphere were carried out in the VSG project that would be comparable with the calculations for the liquid phase (chapter 5.9.2.1). In the R&D project ISIBEL the radiological conse-

quences of a transport in the gaseous phase have been considered in the following way [Rübel et al. 2013]:

- Transport in the geosphere: According to [Becker et al. 2009] three transport paths for radionuclides are discussed: in the gaseous form, dispersed or dissolved in groundwater, and a combination of both. From the relation of groundwater flow rate and rate of radionuclide release from the repository it was assumed that under these conditions all released radionuclides are dissolved in water, where methane is converted into carbon dioxide by microbial oxidation prior to dissolution in water.

- Dose conversion factors in the biosphere: The same DCF as in chapter 5.9.2.1. are used because all radionuclides are assumed to be dissolved in groundwater. If higher release rates of gaseous radionuclides are to be considered, a transport to the biosphere in the gaseous phase cannot be excluded and thus different biosphere models have to be applied, yielding different dose conversion factors. These DCF then apply to both transport paths, either liquid, or gaseous.

Some results are given in [Rübel et al. 2013] for radiation exposures, mainly for release of C-14. These results cannot be used as a general overview of radiological consequences, because they are calculated mainly for the discussion of effects of 2-phase flow in the mine. More detailed discussions of radiological consequences need the above mentioned investigation of different dose conversion factors.

5.10 Human Intrusion

According to the Safety Requirements [BMU 2010a] 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, and
- 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. [BMU 2010a].

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, and
- 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 2010]:

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 2010] 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:

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

The fourth safety level is not considered in the same details 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), and
- 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, and
- 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, and
  - roof securing / formation of loose rock material,

- inflow of brines and natural gases,
- fire within the facility, and
- 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® 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® 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 1.3×10^{-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 5.19 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 5.20: 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 5.21 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 5.21: 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. 2010a] 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 5.22). 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 5.22: Test station for borehole disposal of canisters [Filbert et al. 2010a].

5.12 Multiple lines of reasoning

According to [NEA 2013] and [IAEA 2012] the safety case should provide a synthesis of the available evidence, arguments and analyses. Beside the evaluation of safety criteria like dose and/or risk, which are usually required by most national regulations, complementary types of evidence and arguments are compiled that potentially increase the robustness of the safety case. In this respect complementary safety and performance indicators are increasingly used in recent safety cases to underpin the results of the calculations and increase the transparency of the repository system behaviour. For example in chapter 5.9 the RGI (= Radiological insignificance index) was discussed. Furthermore, the safety statement in a safety case and its robustness rely on multiple lines of reasoning which complement the quantitative analyses.

The use of multiple lines of reasoning may add value to the safety case by providing a range of different arguments that together build confidence in certain data, assumptions and results. Furthermore, certain arguments may be more meaningful to specific audiences [IAEA 2012].

One, if not the most important group of arguments are natural analogue (NA) aspects of the safety case. The main value of NA studies is to provide information of the full complexity of the repository system and of the characteristics of processes over long time scales. It is not possible to simulate in laboratory studies the very long-term process that might affect the performance of a repository. Therefore, natural, archaeological, and industrial analogue studies are often used as one of several multiple lines of reasoning that, in combination, help to build understanding and confidence.

A thorough review of how existing NA studies can be used as important supporting elements of the safety assessment and safety case for a repository in rock salt was performed in the project ISIBEL [Brasser et al. 2013]. The objective of the review and the assessment of Natural Analogues studies was to answer the following questions:

- For which aspects can Natural Analogues contribute to the assessment of safety?
- What is the status of the identified Natural Analogues?
- How do the Natural Analogues contribute to confidence building within the safety case (communicability)?

Within the ISIBEL project (i) analogues for the integrity of the geological barrier, (ii) analogues for the integrity of the geotechnical barriers, and (iii) analogues for release scenarios have been considered [Brasser et al. 2013]. The first couple of aspects are most relevant for the safety demonstration concept, since they are related to the two kinds of containment providing barriers. The first question was answered by starting a systematic review of the required key information needs of the safety case and an assessment whether these requirements can be fulfilled by natural analogues information.

In order to answer the second and third question, the assessment considers two different schemes. The first scheme assesses the status of natural analogues for a safety case of a repository in salt. Five classes are chosen for this assessment:

++ natural analogue is identified and documented
\+ natural analogue is identified and need to be better documented
● natural analogue is identified (no documentation)
o natural analogue is not identified
oo natural analogue is (probably) not identifiable

The second scheme assesses the communicability of a natural analogue for the safety case of a repository in salt. Two classes are chosen for the assessment:

- Public confidence building: natural analogue is tangible and understandable to the lay stakeholder, and
- technical confidence building: natural analogue is an argument to support the understanding of complex behavior.

The results for potential natural analogues studies are compiled in Table 5.3 and Table 5.4. In order to assess the situation at a potential site and to compare it to natural analogues studies, the exploration results of the Gorleben salt dome were used as self-analogue (Status site in Table 5.3).

Table 5.3: Compilation and assessment of natural analogues (NA) for the geological barrier. The classification scheme (+,●, etc.) is explained in the text

Nr.	Aspect / Study	Applicability in safety case (Key information)	Status Site	Status NA	Confidence Building
1	Occurrence of salt domes	Long-term stability of salt domes	++	+	oo
2	Neotectonic conditions	Occurrence of earthquakes and magmatic events	++	--	oo
3	Analysis of the salt flow	Uplift rates	++	+	oo
4	Thickness and composition of the cap rock	Subrosion rates	++	+	o
5	Behaviour of competent salt formations	Possible water pathways	+	●	o
6	Br- (and Rb)-distribution in salt formations	Interaction between formation and external solutions	++	++	o
7	Chemical and isotope composition of fluid inclusions	Interaction between formation and external solutions and gases	++	++	o
8	Openings from salt mining	Behaviour of salt at disposal level	●	+	o
9	Basalt intrusions	Behaviour of salt at high temperatures	--	++	oo
10	Basalt intrusions	Sealing of fissures	--	++	o
11	Kryogenic fractures	Occurrences of fractures formed by salt contraction during cooling	-	+	o

Table 5.4: Compilation and assessment of natural analogues (NA) for geotechnical barriers. The classification scheme (+,●, etc.) is explained in the text.

Nr.	Aspect / Study	Applicability in safety case (Key information)	Status NA	Confidence Building
1	Investigations of bulkhead drift	Reduction of the permeability of an EDZ around drift sealing	+	o
2	Basalt intrusions	Long-term behaviour of basaltic gravel	++	oo
3	Chemical and mineralogical composition of natural clays	Impact of high temperatures on clay minerals	++	o
4	Properties of natural salt clays in salt	Long-term behaviour of clays as sealing material	+	o
5	Corrosion of historical concrete buildings	Long-term behaviour of cementitious materials	++	o
6	Bentonites in saline environment	Long-term behaviour of bentonite as sealing material	+	o
7	Compacted backfill material from old drifts in salt mines	Compaction of crushed salt over long time scales	●	o

The major conclusions from the analysis are as follows: For the integrity of the geological barrier a lot of well described NA are available. Further potential NA are identified but need to be better documented. Despite the vast experience in salt mining and gas storage in Germany, less information is available when discussing the situation for the integrity of geotechnical barriers. There are only a few publications on how to use this knowledge as natural analogue for the geotechnical barrier. This especially holds for the compaction of crushed salt.

To further compile and discuss the status and potential usefulness of natural as well as anthropogenic analogues from different countries to be potentially used within safety sases for radioactive waste repositories in salt formations a workshop “Natural Analogues for Safety Cases of Repositories in Rock Salt” was performed recently [NEA 2014]. The workshop proceedings provide a good overview on studies regarding integrity of salt host rock and engineered barriers as well as on microbial, chemical and transport processes, which might be used as analogues to support the safety case. The workshop also identified significant issues and processes in rock salt where analogue information might reasonably be expected to provide further insight, understanding and quantification. It was recommended to initiate further studies on compaction of crushed salt backfill, the viability of microbes in the near-field, stability of plugs and seals, deformation of anhydrite, and isotope signatures in fluid inclusions.

Natural-technical analogues may also give important indications for operational safety, e.g. development of construction properties during their functional lifetime.

6 Conclusions

The possibility of using domal rock salt as a host rock for HLW has been investigated in Germany for many years. Numerous 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. The generally accepted approach to demonstrate compliance of repository behaviour with performance goals has been to make a consequence analysis with numerical models and to check the compliance with legal requirements such as safety limits. In 1999 the German Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) brought together a scientific working group "AKEnd" tasked with defining scientific criteria as a basis for an HLW repository site selection process. As a result of a comprehensive review of the international state of the art the focus 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 provided eventually the basis for BMU's Safety Requirements [BMU 2010] that were issued in 2010.

In response to the corresponding national discussion and the internationally accepted safety case approach the R&D project ISIBEL 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 [Buhmann et al. 2008]. 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. 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.

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

- development of reference scenarios,
- demonstration and assessment of the "safe containment" concept,
- structure and the content of a safety case report, and
- assessment of uncertainties within long-term safety analysis.

Starting in 2010, the results of the second phase of ISIBEL provided the basis for the 'Preliminary Safety Assessment of the Gorleben Site' (VSG). The practical application and further development of the safety and demonstration concepts originally developed in the ISIBEL work were important steps forward to deriving a safety case in salt in Germany. In VSG the

methodological approaches developed within the ISIBEL project were applied in such a detail to a HLW repository hosted in the Gorleben salt dome, that they could be developed further.

In parallel to VSG a follow-up project of ISIBEL was launched, that focussed in particular on the following topics:

- repository issues: specification of additional types of waste to be considered for disposal, analysis of the methodological approach for deriving repository concepts from requirements,
- long-term safety demonstration for geotechnical barriers: relationship between barrier size and failure propability, applicability of partial safety factor method on long periods,
- evaluation of tools for process analyses: thermomechanical, geomechanical, hydrological and geochemical calculations, gas processes,
- specific investigations on reference and alternative scenarios: methodological approach to reflect uncertainties in scenario development,
- supplementary work on the FEP catalogue: contribution to preparation of NEA-FEP catalogue and data base, and
- applicability of natural analogues.

In 2013 a summary report was given to reflect the state of the art in safety analysis and safety assessments for HLW repositories in domal salt formations and including the interim status of the ISIBEL-project [Bollingerfehr et al. 2013]. The presented report is an upgrade of this report considering the results of the final investigations of ISIBEL and the essential outcomes of other R&D projects.

The restart of the site selection for a German HLW repository according to the site selection act [StandAG 2013] as well as the recommendations of the Repository Commission that had been entrusted by the German parliament to detail the corresponding procedures and criteria [Endlagerkommission 2016] consider rock salt as one of the host rock options. Moreover safety investigations are highlighted as an important instrument for selecting that site that provides the best safety within the limits of the applied selection procedure.

6.1 Assessment of the status

As a result of a comprehensive review of the international state of the art in HLW disposal [BMU 2010] safety related investigations in Germany focused to the safe containment of radioactive waste within the host rock (Containment-providing Rock Zone (CRZ)). Relying on this approach, that also has been reflected in the German Safety Requirements for HLW disposal [BMU 2010a], a new long-term safety and safety demonstration concept for final disposal of HLW in salt formations has been developed in the course of the R&D project ISIBEL [Buhmann et al. 2010e]. This concept requires the safe containment of radioactive

waste after repository closure in a CRZ that is located within the limits of the geological barriers. Safe containment shall be accomplished by preventing or limiting the intrusion of brine to the waste forms and consequently preventing the release of radionuclides from the waste.

Key elements of the safety demonstration are the integrity proofs for the geological and geotechnical barriers and the long-term behavior of backfill by compaction. The potential release of radionuclides must also be evaluated since it cannot be excluded 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 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.

With respect to the high priority that has been given by the so-called Repository Commission to reversibility issues, technical approaches for retrieving the emplaced waste before and recovering it after repository closure have been investigated at a conceptual level. The results of these investigations support the general technical feasibility of implementing technologies for emplacing HLW and SNF safely in drifts and boreholes that fulfil the retrievability and recovery requirements. However these requirements have a significant impact on the emplacement and entire repository concepts. Thus such concepts have to be developed in the early phase of the site selection process in order to ensure that the considered site will be suitable for complying with these requirements.

In the context of the future site selection process it is necessary to compare and evaluate repository concepts in different host rocks. A common, a clear and stringent requirement driven development of repository concept would facilitate this comparison and evaluation. Therefore it has been analyzed whether it is possible to identify resp. develop a methodology to derive repository concepts complying with all legal and safety requirements as well as with technical and site resp. host rock specific requirements. Unfortunately, the analysis of the procedure of repository design development in international and national projects has shown that such a methodology has not been applied neither in the past nor recently. An analysis of the numerous and heterogeneous requirements shows that, on the one hand, they can mostly be fulfilled by different technical measures and, on the other hand, part of them are unique (e.g. site specific geol. environment, host rock features, type and amount of waste, waste forms, selection of type of waste package, safety concept, etc.) and interacting. So the number of degrees of freedom is high and the interdependencies between single repository components like waste package or emplacement variants are complex. The resulting repository concepts complying with the requirements may be quite different; even if the fundamental design data are equal.

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. For these key aspects requirements are defined in the safety concept. Hence the technical concept includes measures to comply with these requirements. 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 site that provides the best safety within the legal limits, from a range of options must rely not only on consideration of differences in geology, but must also consider the whole repository system based on a sufficient comprehensive safety assessment. This latter requirement may offer significant challenges for establishing the site selection procedure.

In this context it is worth noting that the approach developed within the framework of the ISIBEL project for domal salt to derive practical concepts from the Safety Requirements that consider the long-term safe containment of the emplaced radioactive waste in a CRZ and to

assess and demonstrate its safety may be applicable to other rock types with good containment features, e.g. clay [Jobmann et al. 2016], as well as to bedded salt [Bollingerfehr et al. 2015].

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 subrosion 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 subrosion 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. This includes some details of the practicality of construction of the technical barriers. Further studies are required in order to detail these requirements and to evaluate their impact on the technical barriers. One such example is the verification of a sufficiently impermeable contact zone interface in the salt surrounding the geotechnical barriers.

- 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. Especially in case of very low stresses there is a lack of

experimental data. As a consequence creep mechanisms and creep rates at very low stresses are still associated with high uncertainties.

- Regarding damage and permeability of the salt rock there is still a lack of experimental data and mathematical models describing their anisotropic development. Most modern models include a simplification, because the dependency of the dilatancy on the deviatoric as well as the least principle stress component is reflected, but the dilantcy itself and the resulting permeability is assumed to be isotropic.

- Finally, some aspects of the safety demonstration might benefit from conceptual improvements such as the methodological approach to deal with improbable scenarios especially the combinations of low-probable scenarios, or better estimation of probabilities by using alternative tools or methods for their derivation.

7 List of Figures

Figure B.1: Fit of the hyperbolic creep law (here CDM-law of Hampel) and a Norton model with power n = 7 (dashed lines) to the data of BGR on stationary creep. Taken from [Hampel 2012]. 240

Figure B.2: Experimental data on stationary creep of WIPP-salt for low stresses at a temperature of T = 60°C. Numerical calculations done by seven different models and codes are included (Hampel (CDM), IfG (Günther / Salzer und Minkley), Karlsruhe (KIT), Lux/Wolters (LUH), TU Clausthal (TUC), TU Braunschweig (TUBS)). See [Hammer et al. 2016]. Courtesy of R.-M. Günther. 243

8 List of Tables

9 References

[ABBergV 2005] Allgemeine Bundesbergverordnung (ABBergV) vom 23.10.1995 (BGBl. I, S.1466; zuletzt geändert durch Art. 1 der Verordnung vom 10.08.2005 (BGBl. I, S.2452)

[ABBergV 2009] Allgemeine Bundesbergverordnung vom 23. Oktober 1995 (BGBl. I S. 1466), zuletzt geändert durch Artikel 22 des Gesetzes vom 31. Juli 2009 (BGBl. I S. 2585).

[ABVO 1995] Allgemeine Bergverordnung über Untertagebetriebe, Tagebaue und Salinen (ABVO) vom 02.02.1966 (Nds. MBl. S. 337); zuletzt geändert durch Verordnung vom 23.10.1995 (BGBl. I S. 1466).

[Andra 2005] Andra (2005): Dossier 2005 Argile. Tome: Safety evaluation of a geological repository.

[AKEnd 2002] Arbeitskreis Auswahlverfahren Endlagerstandorte (2002): Auswahlverfahren für Endlagerstandorte. Empfehlungen des AkEnd – Schlussbericht. Köln.

[AtG 2010] Gesetz über die friedliche Verwendung der Kernenergie und den Schutz gegen ihre Gefahren (Atomgesetz - AtG) vom 23. Dezember 1959 (BGBl. I S.814), in der Fassung vom 15. Juli 1985 (BGBl. I S. 1565), zuletzt geändert durch Artikel 1 des Gesetzes vom 8. Dezember 2010 (BGBl. I S. 1817).

[AVV 2012] Allgemeine Verwaltungsvorschrift zu §47 Strahlenschutzverordnung: Ermittlung der Strahlenexposition durch die Ableitung radioaktiver Stoffe aus Anlagen oder Einrichtungen (28. August 2012). Erschienen im Bundesanzeiger, BAnz AT 05.09.2012 (2012).

[Ayachit 2015] Ayachit, U. (2015): The ParaView Guide: A Parallel Visualization Application. Kitware, New York, ISBN 978-1930934306

[BBergG 2005] Bundes-Berggesetz (BBergG) vom 13.08.1980 (BGBl. I S. 1310); zuletzt geändert durch Art. 15a des Gesetzes vom 31. Juli 2009 (BGBl. I S. 2585).

[Bechthold et al. 2004] Bechthold, W.; Smailos, E.; Heusermann, S.; Bollingerfehr, W.; Bazargan Sabet, B.; Rothfuchs, T; Kamlot, P.; Grupa, J.; Olivella, S. & Hansen, F.D. (2004): “Backfilling and sealing of underground repositories for radioactive waste in salt (BAMBUS II Project)” Final report. EUR 20621 EN. European Commission, Luxembourg, 2004.

[Becker et al. 2009] Becker, D.-A.; Buhmann, D.; Mönig, J.; Noseck, U.; Rübel, A. & Spießl, S.: Endlager Morsleben. Sicherheitsanalyse für das verfüllte und verschlossene Endlager mit dem Programmpaket EMOS. Planfeststellungsverfahren zur Stilllegung des Endlagers für radioaktive Abfälle Morsleben. Verfahrensunterlage P278. Bundesamt für Strahlenschutz (BfS) Salzgitter, 2009.

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

[Beuth et al. 2012b] Beuth, T.; Baltes, B.; Bollingerfehr, W.; Buhmann, D.; Charlier, F.; Filbert, W.; Fischer-Appelt, K.; Mönig, J.; Rübel, A. & Wolf, J. (2012b): Untersuchungen zum menschlichen Eindringen in ein Endlager. Bericht zum Arbeitspaket 11. Vorläufige Sicherheitsanalyse für den Standort Gorleben, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-280, Köln, ISBN 978-3-939355-56-4.

[BfS 2010] Bundesamt für Strahlenschutz (BfS): Untersuchung der Kontaktzone am Asse-Vordamm – Zusammenfassung und Interpretation der Messergebnisse, BfS-KZL 9M/22310131/GH/RB/0130/00, Stand 03.05.2010.

[BGR 1988] Thermomechanische Berechnungen zum Endlager im Salzstock Gorleben (Berechnungsmodell 1988), Hannover, Mai 1988.

[Birthler et al. 2000] Birthler, H.; Fein, E. & Schneider, A. (2000): Validierung von Einzeleffekten in Grundwassermodellen, FKZ 02E8865/0, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-150, Braunschweig, ISBN 3-931995-12-7.

[Blase et al. 1989] Blase, G.; Potthoff, A.; Ochs, F. & Uhlenbecker, F.W. (1989): Gebirgsmechanische Beobachtungen in Salzstöcken Norddeutschlands (Rock-mechanical observations in the salt domes of North Germany). Kali und Steinsalz, volume 10, no. 6, 174-181.

[BMI 1983] Sicherheitskriterien für die Endlagerung radioaktiver Abfälle in einem Bergwerk, Rdschr. D. BMI v. 20.4.1983 –RS – AGK 3 – 515 790/2.

[BMU 1977] Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit (BMU: Sicherheitsanforderungen an Kernkraftwerke. Bundesanzeiger, Nr. 206, Stand: 21. Oktober 1977.

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

[BMU 2012] Bundesministerium für Umwelt, Naturschutz und Reaktorsicherheit (BMU: Sicherheitsanforderungen an Kernkraftwerke. Bundesanzeiger, AT 24.01.2013.B3, Stand: 20. November 2012.

[BMUB 2015a] Bundesministerium für Umwelt, Naturschutz, Bau und Reaktorsicherheit: Programm für eine verantwortungsvolle und sichere Entsorgung bestrahlter Brennelemente und radioaktiver Abfälle (Nationales Entsorgungsdprogramm), August 2015

[BMUB 2015b] Bundesministerium für Umwelt, Naturschutz, Bau und Reaktorsicherheit: Verzeichnis bestrahlter Brennelemente und radioaktiver Abfälle (Bestand zum 31.12.2014 und Prognose), August 2015

[Bollingerfehr et al. 2008] Bollingerfehr, W., Filbert, W., Pöhler, M., Tholen, M. & Wehrmann, J.: Konzeptionelle Endlagerplanung und Zusammenstellung des endzulagernden Inventars (Design planning of a final repository and summary of the inventory to be stored) – Project ISIBEL, Peine, April 2008.

[Bollingerfehr et al. 2011] Bollingerfehr, W.; Filbert, W.; Lerch, C. & Tholen, M. (2011): Endlagerkonzepte – Bericht zum Arbeitspaket 5 – Vorläufige Sicherheitsanalyse für den Standort Gorleben; Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-272; Köln, ISBN 978-3-939355-48-9.

[Bollingerfehr et al. 2012] Bollingerfehr, W.; Dörr, S.; Filbert, W.; Herold, P.; Lerch, C.; Burgwinkel, P.; Charlier, F.; Thomauske, B.; Bracke, G. & Kilger, R. (2012): Endlagerauslegung und -optimierung – Bericht zum Arbeitspaket 6 – Vorläufige Sicherheitsanalyse für den Standort Gorleben; Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-281; Köln, ISBN 978-3-939355-57-1.

[Bollingerfehr et al. 2013] Bollingerfehr, W., Buhmann, D., Filbert, W., Keller, S., Krone, J., Lommerzheim, A., Mönig, J., Mrugalla, S., Müller-Hoeppe, N., Weber, J.R. & Wolf, J. (2013): Status oft he safety concept and safety demonstration for an HLW repository in salt.- DBE TECHNOLOGY GmbH, Ber.-Nr. TEC-15-2013-AB, FKZ 02E10719 und 02E10729, Peine, December 2013.

[Bollingerfehr et al. 2014] Bollingerfehr, W., Herold, P., Dörr, S. & Filbert, W.: Auswirkungen der Sicherheitsanforderung Rückholbarkeit auf existierende Einlagerungskonzepte und Anforderungen an neue Konzepte, Final Report, TEC-21-2013-AB, Peine, February 2014

[Bollingerfehr et al. 2015] Bollingerfehr, W., Herold, P., Dörr, S. & Filbert, W.: KOSINA – Konzeptentwicklung für ein generisches Endlager für wärmeentwickelnde Abfälle in flach lagernden Salzschichten in Deutschland. Entwicklung und Überprüfung eines Sicherheits- und Nachweiskonzeptes, Zwischenbericht, DBE TECHNOLOGY GmbH, TEC-37-2015-Z, 2015, Peine.

[Bornemann et al. 2008] Bornemann, O.; Behlau, J., Fischbeck, R. , Hammer, J., Jaritz, W., Keller, S., Mingerzahn, G. & Schramm, M. (2008): Description of the Gorleben site, Part 3. Results of the geological surface and underground exploration of the salt formation. Bundesanstalt für Geowissenschaften und Rohstoffe (BGR), 223 pp., 50 figures, 7 tables, 5 appendices; Hannover. ISBN 978-3-9813373-6-5.

[Bracke et al. 2012] Bracke, G.; Popp, T., Püttmann, W., Kienzler, B., Lommerzheim, A. & Moog, H. (2012): Berücksichtigung der Kohlenwasserstoffvorkommen in Gorleben. Bericht der Arbeitsgruppe "Kohlenwasserstoffe", Vorläufige Sicherheitsanalyse für den Standort Gorleben. GRS-285, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln ISBN 978-3-939355-61-8.

[Bräuer et al. 2011] Bräuer, V.; Eickemeier, R., Eisenburger, D., Grissemann, C., Hesser, J., Heusermann, S., Nipp, H.-K., Nowak, T., Plischke, I., Salzstock, Schnier, H., Schulze, O., Sönnke, J. & Weber, J.R. (2011): Description of the Gorleben site, Part 4: Geotechnical exploration of the Gorleben salt dome. Bundesanstalt für Geowissenschaften und Rohstoffe (BGR), Hannover: 176 pp., 96 figures, 13 tables, 3 appendices; Hannover, ISBN 978-3-9814108-0-8.

[Brasser et al. 2013] Brasser, T.; Fahrenholz, C.; Kull, H.; Meleshyn, A.; Mönig, H.; Noseck, U.; Schönwiese, D. & Wolf, J. (2013): Natürliche Analoga im Wirtsgestein Salz. Teil 1 Generelle Studie (2011), Teil 2 Detailstudien (2012 – 2013). FKZ 02E10719, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, GRS-365, ISBN 978-3-944161-46-4, Braunschweig.

[Breidung 2002] Breidung, K.P. (2002): Forschungsprojekt Schachtverschluss Salzdetfurth Schacht II, Abschlussbericht (Kurzfassung), K+S Aktiengesellschaft, Bad Salzdetfurth.

[Brückner-Röhling et al. 2002] Brückner-Röhling, S.; Espig, M., Fischer, M., Fleig, S., Forsbach, H., Kockel, F., Krull, P., Stiewe, H. & Wirth, H. (2002): Standsicherheitsnachweise Nachbetriebsphase: Seismische Gefährdung – Teil 1: Strukturgeologie. BGR-Bericht: 253 S.; Hannover.

[Buhmann et al. 1991] Buhmann, D.; Nies, A. & Storck, R.: Analyse der Langzeitsicherheit von Endlagerkonzepten für wärmeerzeugende radioaktive Abfälle. FKZ KWA 5702 A. GSF-Forschungszentrum für Umwelt und Gesundheit GmbH, Bericht 27/91, Braunschweig, 1991.

[Buhmann et al. 2008a] Buhmann, D., Mönig, J. & Wolf, J. (2008): Untersuchungen zur Ermittlung und Bewertung von Freisetzungsszenarien. Teilbericht zum Projekt ISIBEL, FKZ 02E 10065 / 02E 10055, Gesellschaft für Reaktorsicherheit (GRS) mbH, GRS-233, ISBN 978-3-939355-07-6, Braunschweig.

[Buhmann et al. 2008b] Buhmann, D., Mönig, J., Wolf J., Keller, S., Weber, J.R., Eberth, S., Kreienmeyer, M., Krone, J. & Tholen, M. (2008): Teilbericht zum Projekt ISIBEL, FEP-Katalog für einen HAW-Standort im Wirtsgestein Salz. TEC-11-208-AP, Gemeinsamer Bericht von DBE TECHNOLOGY GmbH, BGR und GRS. FKZ 02E 10065 / 02E 10055, Peine, Hannover, Braunschweig.

[Buhmann et al. 2008c] Buhmann, D.; Mönig, J., Wolf, J., Heusermann, S., Keller, S., Weber, J.R., Bollingerfehr, W., Filbert, W., Kreienmeyer, M., Krone, J. & Tholen, M. (2008): Überprüfung und Bewertung des Instrumentariums für eine sicherheitliche Bewertung von Endlagern für HAW – ISIBEL. TEC-09-2008-AB, Gemeinsamer Abschlussbericht von DBE TECHNOLOGY GmbH, BGR und GRS, FKZ 02E 10065 / 02E 10055, Peine.

[Buhmann et al. 2010a] Buhmann, D.; Mönig, J., Wolf, J., Keller, S., Mrugalla, S., Weber, J.R., Krone & J. Lommerzheim, A. (2010): FEP-Katalog für einen HAW-Standort Im Wirtsgestein Salz. ISIBEL Projekt. TEC-05-2010-AB, Gemeinsamer Bericht von DBE TECHNOLOGY GmbH, BGR und GRS, FKZ 02E 10065 / 02E 10055, Peine.

[Buhmann et al. 2010b] Buhmann, D.; Mönig, J., Wolf, J., Keller, S., Mrugalla, S., Weber, J.R., Krone & J. Lommerzheim, A. (2010): Erläuterungen zur Revision des FEP-Kataloges. ISIBEL Projekt. TEC-06-2010-AB, Gemeinsamer Bericht von BGR, DBE TECHNOLOGY GmbH und GRS. FKZ 02E 10065 / 02E 10055, Hannover, Peine, Braunschweig.

[Buhmann et al 2010c] Buhmann, D.; Mönig, J., Wolf, J., Keller, S., Mrugalla, S., Weber, J.R., Krone & J. Lommerzheim, A. (2010): Entwicklung und Test einer Methodik zur Ableitung eines Referenzszenarios. ISIBEL Projekt. Gemeinsamer Bericht von BGR, DBE TECHNOLOGY GmbH und GRS. FKZ 02E 10065 / 02E 10055, Hannover, Peine, Braunschweig.

[Buhmann et al. 2010d] Buhmann, D.; Mönig, J., Wolf, J., Keller, S., Mrugalla, S., Weber, J.R., Krone, J. Lommerzheim, A. (2010): Nachweis und Bewertung des Isolationszustandes "Sicherer Einschluss". ISIBEL Projekt. TEC-11-2010-AB, Gemeinsamer Abschlussbericht von DBE TECHNOLOGY GmbH, BGR und GRS, FKZ 02E 10065 / 02E 10055, Hannover, Peine, Braunschweig.

[Buhmann et al. 2010e] Buhmann, D.; Mönig, J., Wolf, J., Keller, S., Mrugalla, S., Weber, J.R., Krone, J. Lommerzheim, A. (2010): Behandlung von Ungewissheiten im Langzeitsicherheitsnachweis für ein HAW-Endlager im Salzgestein. ISIBEL Projekt. TEC-12-2010-AB, Gemeinsamer Bericht von DBE TECHNOLOGY GmbH, BGR und GRS, Hannover, Peine, Braunschweig.

[Buhmann et al. 2010f] Buhmann, D.; Mönig, J., Wolf, J., Keller, S., Mrugalla, S., Weber, J.R., Krone, J. Lommerzheim, A. (2010): Überprüfung und Bewertung des Instrumentariums für eine sicherheitliche Bewertung von Endlagern für HAW – ISIBEL. TEC-09-2010-AB, Gemeinsamer Abschlussbericht von DBE TECHNOLOGY GmbH, BGR und GRS, Hannover, Peine, Braunschweig.

[Buhmann et al. 2016] Buhmann, D.; Laggiard, E.; Rübel, A.; Spießl, S. & Wolf, J. (2016): Probabilistische Bewertung von Szenarien in Langzeitsicherheitsanalysen. Ergebnisse des Projekts ISIBEL. FKZ 02E10719, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, GRS-416, ISBN 978-3-944161-98-3, Braunschweig.

[Cristescu & Hunsche 1998] Cristescu, N. & Hunsche, U. (1998): Time Effects in Rock Mechanics. 343 S.; ISBN 0471955175, Chichester (John Wiley & Sons).

[Czaikowski & Wieczorek 2012] Czaikowski, O. & Wieczorek, K. (2012): Salzgruskompaktion – Kalibrierung der in CODE_BRIGHT verwendeten physikalischen Modellansätze zur numerischen Simulation, Vorläufige Sicherheitsanalyse Gorleben (VSG), GRS Memorandum, Köln.

[DAfStb 2004] DAfStb – Deutscher Ausschuss für Stahlbeton (2004): Richtlinie „Betonbau beim Umgang mit wassergefährdenden Stoffen“.

[DeBerg et al. 2000] De Berg, M.; van Kreveld, M.; Overmars, M. & Schwarzkopf, O.: Computational Geometry (2nd revised ed.), Chapter 3: Polygon Triangulation, 45-61, 2000.

[DGGT 1997] DGGT – Deutsche Gesellschaft für Geotechnik (1997): GDA-Empfehlungen Geotechnik der Deponien und Altlasten, Ernst Sohn, 3. Auflage.

[DIN 1991-1 1995] DIN V ENV 1991-1, Eurocode 1, Grundlagen der Tragwerksplanung und Einwirkungen auf Tragwerke, Teil 1: Grundlagen der Tragwerksplanung, Deutsche Fassung ENV 1991-1:1994, Dezember 1995.

[DIN 1990] DIN EN 1990, Eurocode: Grundlagen der Tragwerksplanung; Teil 1: Deutsche Fassung EN 1990: 2002, Oktober 2002..

[DIN-EN 1997-1] DIN-EN 1997-1: Eurocode 7: Entwurf, Berechnung und Bemessung in der Geotechnik – Teil 1 Allgemeine Regeln; Deutsche Fassung EN 1997-1, März 2014

[DIN 2009] DIN EN 1997-1, Eurocode 7: Entwurf, Bemessung und Berechnung in der Geotechnik, Teil 1: Allgemeine Regeln; Deutsche Fassung EN 1997-1:2004+AC:2009, Stand September 2009.

[DIN 2010] DIN EN 1990, Eurocode, Grundlagen der Tragwerksplanung; Deutsche Fassung EN 1990:2002+A1:2005+A1:2005/AC, English Version EN 1990:2002E, Stand Dezember 2010.

[Dörr et al. 2012] Dörr, S.; Bollingerfehr, W., Filbert, W. & Tholen, M. (2012): In das Endlager eingebrachte Inventare an Metallen, Organika und weiteren Stoffen. Vorläufige Sicherheitsanalyse Gorleben. Memo. DBE TECHNOLOGY GmbH, Peine.

[Dörr 2013] Dörr, S. (2013): Komplettierung des Instrumentariums zur sicherheitstechnischen Bewertung eines HAW-Endlagers (KOMTESSA); AP 2A-Ergänzung des endzulagernden Inventars, Februar 2013, FKZ 02 E 10729, DBE TECHNOLOGY GmbH, Peine.

[Endlagerkommission 2016] Endlagerkommission (2016): Abschlussbericht der Kommission Lagerung hoch radioaktiver Abfälle "Verantwortung für die Zukunft – Ein faires und transparentes Verfahren für die Auswahl eines nationalen Endlagers", Geschäftsstelle der Kommission Lagerung hoch radioaktiver Abfälle, Deutscher Bundestag

[Fein & Schneider 1999] Fein, E. & Schneider, A. (1999): d^3f – Ein Programmpaket zur Modellierung von Dichteströmungen. Final Report, FKZ 02C0465/0, ISBN 3-923875-97-5, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-139, Braunschweig.

[Fein 2004] Fein, E. (2004): Software Package r^3t. Model for Transport and Retention in Porous Media. Final Report, FKZ 02E9148/2, ISBN 3-931995-60-7, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-192, Braunschweig.

[Fein et al. 2008] Fein, E.; Kröhn, K.-P.; Noseck, U. & Schneider, A. (2008): Modelling of Field-Scale Pollutant Transport, Final Report, FKZ 02E9934, ISBN 978-3-939355-05-2, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-231, Braunschweig.

[Filbert 1994a] Filbert, W. (1994a): Simulation des Schachttransportes, Probabilistische Sicherheitsanalyse zur Schachtförderanlage, TA 11, DBE DEAB T 50, Peine.

[Filbert 1994b] Filbert, W. (1994b): Simulation des Schachttransportes, Abschlussbericht FKZ 02 E 8221, DBE DEAB T 41, Peine.

[Filbert 1995] Filbert, W. (1995): Handhabungsversuche zur Streckenlagerung, Abschlussbericht FKZ 02 E 8191, Hauptband, DBE DEAB T 60, Peine.

[Filbert & Engelmann 1998] Filbert, W. & Engelmann, H.-J. (1998): Aktualisierung des Konzeptes Endlager Gorleben; Abschlussbericht, Rev. 01, Peine, unveröffentlicht.

[Filbert & Pöhler 2008] Filbert, W. & Pöhler, M. (2008): Überprüfung und Bewertung des Instrumentariums für eine sicherheitliche Bewertung von Endlagern für HAW. ISIBEL. AP 2: Bewertung der Betriebssicherheit.- DBE TECHNOLOGY GmbH, TEC-17-2008-AP, FKZ: 02E10065, Peine.

[Filbert et al. 2010a] Filbert W. (2010): DBE TECHNOLOGY GmbH: Optimization of the Direct Disposal Concept by Emplacing SF Canisters in Boreholes. Final Report, Abschlussbericht zum FuE-Vorhaben Optimierung der Direkten Endlagerung durch Kokillenlagerung in Bohrlöchern, FKZ 02 E 9854, Peine, June 2010.

[Filbert et al. 2010b] W. Filbert, W. Bollingerfehr, M. Heda, C. Lerch, N. Niehues, M. Pöhler, J. Schulz, T. Schwarz, M. Toussaint, J. Wehrmann: Optimization of the Direct Disposal Concept by Emplacing SF Canisters in Boreholes (ESDRED/DENKMAL) - Final Report, Peine, 2010

[Fischer-Appelt et al. 2013] Fischer-Appelt, V.; Baltes, B., Buhmann, D., Larue, J. & Mönig, J. (2013): Synthesebericht für die VSG, Bericht zum Arbeitspaket 13, Vorläufige Sicherheitsanalyse für den Standort Gorleben, GRS-290, ISBN 978-3-939355-66-3, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln.

[Frey & Borouchaki 1998] Frey, P. J. & Borouchaki, H.: Geometric Surface Mesh Optimization. Computing and Visualisation in Science 1, 113-121, 1998.

[Frost & Ashby 1982] H.J. Frost & M.F. Ashby (1982): Deformation-Mechanism Maps: The Plasticity and Creep of Metals and Ceramics, Pergamon Press, p. 12.

[Galson & Khursheed 2007] Galson, D.A. & Khursheed, A. (2007): The Treatment of Uncertainty in Performance Assessment and Safety Case Development: State-of-the-Art Overview. PAMINA Work Package 1.2, Rutland, UK.

[Galson et al. 2009] Galson, D.A., Hooker, P.J. & Wilmot, R.D. (2009): The Treatment of Uncertainty using Probability. PAMINA Work Package 2.1, Rutland, UK.

[Gerling et al. 2002] Gerling, P., Faber, E. & Wehner, H. (2002): Interpretation der chemischen Analysen von gasförmigen und flüssigen Kohlenwasserstoffen. – Bundesanstalt für Geowissenschaften und Rohstoffe (BGR), Tagebuch-Nr. 12243/02, Hannover, Germany.

[GRS 2002a] Sicherheitskriterien für die Endlagerung radioaktiver Abfälle in einem Bergwerk - Überarbeitete Diskussionsgrundlage - GRS, Januar 2002, Auftrags-Nr.: 85471.

[GRS 2005a] „Sicherheitsanforderungen für Kernkraftwerke: Grundlegende Sicherheitsanforderungen", Revision A ENTWURF, GRS, PHB, Öko-Institut, September 2005, Auftrags-Nr.: 813071, SR 2475, Ergebnisse Team 1.

[Grupa 2001] Grupa, J.B. (2001): The PROSA Methodology for Scenario Development. Scenario Development Methods and Practice. Proc. NEA Workshop on Scenario Development. Madrid, Spain, 1999. OECD-NEA, Paris.

[Günther et al. 2007] Günther, R.M., Minkley, W. & Salzer, K. (2007): Verbundprojekt: Die Modellierung des mechanischen Verhaltens von Steinsalz: Vergleih aktueller Stoffgesetze und Vorgehensweisen, Teilprojekt 3 – Teil II: Visko-elasto-plastisches Stoffmodell.- FKZ: 02C1024, Institut für Gebirgsmechanik (IfG), Leipzig.

[Günther et al. 2015] Günther, R.M., Salzer, K., Popp, T. & Lüdeling, C. (2015): Steady-State Creep of Rock Salt: Improved Approaches for Lab Determination and Modelling. Rock Mechanics and Rock Engineering, Vol. 48,6, p. 2603-2613.

[Hammer et al. 2012a] Hammer, J., Fleig, S., Mingerzahn, G., Kühnlenz, T., Mertineit, M., Pusch, M., Schramm, M., Behlau, J., Zaretzki, B., Hesser, J., Shao, H., Köthe, A. & Vogel, P. (2012): Salzgeologische Bewertung des Einflusses von „kryogenen Klüften“ und halokinetischen Deformationsprozessen auf die Integrität der geologischen Barriere des Salzstocks Gorleben. Bericht zum Arbeitspaket 2 Vorläufige Sicherheitsanalyse für den Standort Gorleben - GRS Braunschweig, Bericht GRS-273, ISBN 978-3-939355-49-6, Köln

[Hammer et al. 2012b] Hammer, J.; Pusch, M., Häger, A., Ostertag-Henning, C., Schlömer, S., Mingerzahn, G., Scheeder, G., Shao, H., Paul, B., Schulze, O., Zaretzki, B. & Hesser, J. (2012): Projekt Gorleben, Untersuchungen von Kohlenwasserstoffen im Erkundungsbergwerk Gorleben. 236 Seiten, Zwischenbericht 2011, Bundesanstalt für Geowissenschaften und Rohstoffe: Hannover.

[Hampel et al. 1998] Hampel, A., Hunsche, U., Weidinger, P. & Blum, W. (1998): Description of the Creep of Rock Salt with the Composite Model - II. Steady-State Creep. – In: M. Aubertin & H.R. Hardy, Jr. (Hrsg.): The Mechanical Behavior of Salt, Proceedings of the 4th Conference (MECASALT IV), Montreal, 1996, S. 287-299, Trans Tech Publications, Clausthal-Zellerfeld.

[Hampel 2012] Hampel, A. (2012): Weiterentwicklung der Module im CDM-Stoffgesetzsystem und deren Formulierung für die Verwendung in JIFE, Abschlussbericht, BGR Hannover

[Hampel 2015] Hampel. A. (2015): Description of damage reduction and healing with the CDM constitutive model for the thermo-mechanical behavior of rock salt. In: L. Roberts, K. Mellegard & F. Hansen (Hrsg.): Proc. of the Conference on Mechanical Behavior of Salt (Saltmech VIII), South Dakota School of Mines & Technology, Rapid City, SD, USA, 26.-28.05.2015, Taylor & Francis Group, London, ISBN 978-1-138-02840-1, S. 301-310.

[Hampel et al. 2016] Hampel, A. et al. (2016): Vergleich aktueller Stoffgesetze anhand von Modellberechnungen zum thermo-mechanischen Verhalten und zur Verheilung von Steinsalz. Abschlussbericht zum BMWi-Verbundprojekt, FKZ 02E10810-10860 (in Vorbereitung).

[Heemann 1989] Heemann, U., 1989: Transientes Kriechen und Kriechbruch im Steinsalz, Forschungs- und Seminarberichte aus dem Bereich der Mechanik der Universität Hannover, Dissertation

[Heemann 2015] Heemann, U. (2015): Entwicklung von Methoden zur interaktiven Simulation gekoppelter THMC-Prozesse (2011-2014), Zwischenbericht, BGR Hannover, März 2015.

[Herold 2016] Herold, P.: Rückholbarkeit – eine Herausforderung für die Entwicklung von Endlagerkonzepten, 5. Essener Fachgespräch Endlagerbergbau, Essen, February 2016

[Herold et al. 2016] Herold, P., Dörr, S., Kuate-Simo, E., Bollingerfehr, W. & Filbert, W.: Technical Approach for Retrieving Heat-Generating Waste from Repositories in Salt Formations, WM2016 Conference, March 6-10, 2016, Phoenix, Arizona, USA, March 2016

[Heusermann 2001] Heusermann, S. (2001): Beurteilung der geomechanischen Stabilität und Integrität von Endlagerbergwerken im Salzgebirge auf der Grundlage geologischer und ingenieurgeologischer Untersuchungen. Geologische Beiträge Hannover, 2, 159-174.

[Heusermann 2007] Heusermann, S. (2007): Langzeitsicherheitsanalyse für ein HAW-Endlager im Salz – Nachweis der Integrität der geologischen Barriere. BGR, Hannover, Tagebuchnummer 10403/08

[Heusermann et al. 2012a] Heusermann, S.; Vogel, P., Eickemeier, R. & Nipp, H.-K. (2012a): Thermomechanical modelling of the Gorleben exploration site to analyse the integrity of the salt barrier. Harmonising Rock Engineering and the Environment – Qian & Zhou (eds), p. 413-418, Taylor & Francis Group, London, ISBN 978-0-415-80444-8.

[Heusermann et al. 2012b] Heusermann, S.; Vogel, P., Eickemeier, R. & Nipp, H.-K. (2012b): Analysis of the integrity of the geological barrier in the Gorleben salt formation. Mechanical Behavior of Salt VII, Eds. P. Berest, M. Ghoreychi, F. Hadj-Hassen, M. Tijani, p. 453-460, Taylor & Francis Group, London, ISBN 978-0-415-62122-9.

[Heusermann et al. 2013] Heusermann, S.; Eickemeier, R., Nipp, H.-K. & Fahland, S. (2013): Three-dimensional thermomechanical modelling of high-level waste emplacement in a salt dome. SINOROCK2013 - 3rd ISRM Symposium on Rock Mechanics “Rock Characterisation, Modelling and Design Methods”, 18-20 June 2013, Shanghai, China.

[Hippler et al. 2008] Hippler, J., Tholen, M., Kreienmeyer, M. & Krone, J. (2008): Nachweiskonzepte für die Einhaltung der nicht radiologischen Schutzziele in der Nachbetriebsphase. DBE TECHNOLOGY GmbH, Peine

[Hirsekorn et al. 1999] Hirsekorn, R.-P.; Boese, B. & Buhmann, D. (1999): LOPOS: Programm zur Berechnung der Schadstofffreisetzung aus netzwerkartigen Grubengebäuden. FKZ 02E 8835-6, ISBN 3-931995-19-4. GRS-157, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Braunschweig.

[Hou & Lux 1998] Hou, Z. & Lux, K.H. (1998): Ein neues Stoffmodell für duktile Salzgesteine mit Einbeziehung von Gefügeschädigung und tertiärem Kriechen auf der Grundlage der Continuum-Damage-Mechanik. Geotechnik, Bd. 21,3, S. 259-263.

[Hou et al. 2007]: Hou, Z., Wolters, R. & Düsterloh, U. (2007): Die Modellierung des mechanischen Verhaltens von Steinsalz: Vergleich aktueller Stoffgesetze und Vorgehensweisen. Teilprojekt 4 des Verbundprojektes – Stoffmodell Hou / Lux, TU Clausthal.

[Hunsche 1981] U. Hunsche (1981): Results and interpretation of creep experiments on rock salt, Proc. First Conf. on the Mechanical Behavior of Salt, Nov. 1981, Penn. State Univ., University Park (1982), p. 159-167.

[Hunsche 1995] U. Hunsche (1995): Geotechnische Untersuchungen der Schachtstandorte: Salzmechanische Laboruntersuchungen an Steinsalz, Gefrierschächte Gorleben, Ed. Jessberger, Balkema, Rotterdam, S. 251-262.

[Hunsche & Schulze 1994] Hunsche, U. & Schulze, O.: Das Kriechverhalten von Steinsalz. In: Kali und Steinsalz 11 (1994), Heft 8/9, S. 238-255

[Hunsche & Schulze 2002] Hunsche, U. & Schulze, O. (2002): Humidity induced creep and it‘s relation to the dilatancy boundary. In: Basic and Applied Salt Mechanics; Proc. of the 5th Conf. on the Mech. Behavior of Salt (MECASALT V), Bucharest 1999. Eds.: N.D. Cristescu, H.R. Hardy JR., R. O. Simionescu; 73 – 87. Balkema, Lisse.

[Hurtado et al. 1997] Hurtado, L. D.; Knowles, M. K.; Kelley, V. A.; Jones, T. L.; Ogintz, J. B. & Pfeifle, T. W. (1997): WIPP Shaft Seal System Parameters Recommended to Support Compliance Calculations, Sandia Report DAND97-1287*UC-721.

[IAEA 1997] International Atomic Energy Agency (IAEA): Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management.- INFCRC/546, Vienna, Dec. 1997.

[IAEA 2012] International Atomic Energy Agency (IAEA): The Safety Case and Safety Assessment for the Disposal of Radioactive Waste. IAEA Safety Standards Series No. SSG-23: Vienna, 2012.

[IFG 2010] IfG, Institut für Gebirgsmechanik: Sicherheitsabstände zur vorläufigen Dimensionierung eines untertägigen Endlagers für wärmeentwickelnde Abfälle im Salzgebirge (Safety distances for the provisional dimensioning of an underground final repository for heat-generating waste in salt rock strata), Leipzig, October 2010.

[ITASCA 2012] ITASCA: FLAC^{3D} - Fast Lagrangian Analysis of Continua in 3 Dimensions. Version 5.0. User's Manual. Minneapolis: ITASCA Consulting Group Inc., 2012

[ITASCA 2016] ITASCA: FLAC^{3D} - Fast Lagrangian Analysis of Continua in 3 Dimensions – Manuals – Creep Material Models.- ITASCA, Mineapolis.

[Jobmann et al. 2016] Jobmann, M., Bebiolka, A., Jahn, S., Lommerzheim, A., Maßmann, J., Meleshyn, A., Mrugalla, S., Reinhold, K., Rübel, A., Stark, L. & Ziefle, G. (2016): Projekt ANSICHT – Methodik und Anwendungsbezug eines Sicherheits- und Nachweiskonzeptes für ein HAW-Endlager im Tonstein, Synthesebericht, DBE TECHNOLOGY GmbH.

[Kachanov 1986] L. Kachanov (1986): Introduction to continuum damage mechanics, Springer-Verlag

[Kamlot et al. 2012] Kamlot, P.; Weise, D.; Gärtner, G. & Teichmann, L. (2012): Drift sealing elements in the Asse II mine as a component of the emergency concept – assessment of the hydro-mechanical functionality, Mechanical Behavior of SALT VII – Berest, Ghoreychi, Hadj-Hassen Tijani (eds.), Taylor & Francis, London.

[Keller 1990] Keller, S. (1990): Das Ablaugungsverhalten der Salzstöcke in NW-Deutschland (Abschlussbericht des BMFT-Förderungsvorhabens KWA 5801 9 „Langzeitsicherheit der Barriere Salzstock“, Teilprojekt III). – Bundesanstalt für Geowissenschaften und Rohstoffe (BGR), Archiv-Nr. 106570; Hannover, Germany.

[Keller 2001] Keller, S. (2001): Projekt Gorleben: Zusammenstellung von Zuständen, Ereignissen und Prozessen (ZEP) als Basis für die Ermittlung von Szenarien (Standort Gorleben). – Bundesanstalt für Geowissenschaften und Rohstoffe (BGR), Archiv-Nr. 0121 591; Hannover, Germany.

[Keller 2007] Keller, S. (2007): Langzeitsicherheitsanalyse für ein HAW-Endlager im Salz – Geologisches Referenzmodell für einen HAW-Endlagerstandort im Salz – Technischer Bericht. Beitrag für das Projekt ISIBEL. BGR, Hannover, Tagebuchnummer 11614/05.

[Kilger et al. 2012] Kilger, R.; Bock, M., Moser, E.-F. & Gmal, B. (2012): Betrachtungen zum Kritikalitätsausschluss in der Nachverschlussphase (AP 6: Endlagerauslegung und -optimierung), Technischer Bericht (Memo) im Vorhaben „Vorläufige Sicherheitsanalyse für den Standort Gorleben“, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, München.

[Klinge et al. 2007] Klinge, H.; Böhme, J., Grissemann, C., Houben, G., Ludwig, R. R., Rübel, A., Schelkes, K., Schildknecht, F. & Suckow, A. (2007): Description of the Gorleben site, Part 1. Hydrogeology of the overburden of the Gorleben salt dome. BGR: 145 pp., 59 figures, 4 tables, 1 appendix; Hannover, ISBN 978-3-9813373-4-1.

[Knoll 2005] Knoll, P. (2005): Entwicklung eines Grundkonzeptes für langzeitstabile Streckendämme im leichtlöslichen Salzgestein (Carnallitit) für UTD/UTV, Abschlussbericht, GTS, FuE Vorhaben (FKZ 02 C 0942).

[Kock et al. 2012] Kock, I.; Eickemeier, R., Frieling, G., Heusermann, S., Knauth, M., Minkley, W., Navarro, M., Nipp, H.-K. & Vogel, P. (2012): Vorläufige Sicherheitsanalyse Gorleben - Integritätsanalyse der geologischen Barriere. Abschlussbericht zum Arbeitspaket 9.1, GRS-286, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln, ISBN 978-3-939355-62-5.

[Köthe et al. 2007] Köthe, A.; Hoffmann, N., Krull, P., Zirngast, M. & Zwirner, R (2007): Description of the Gorleben site, Part 2. Geology of the overburden and adjoining rock of the Gorleben salt dome. BGR: 201 pp., 42 figures, 19 tables; BGR Hannover, ISBN 978-3-9813373-5-8.

[Kreienmeier et al. 2008] Kreienmeyer, M., Lerch, C., Polster, M. & Tholen, M. (2008): Nachweiskonzept zur Integrität der einschlusswirksamen technischen Barrieren. DBE TECHNOLOGY GmbH, Peine

[Kühle et al. 1999] Kühle, T.; Graefe, V. & Hirsekorn, R.-P. (1999): Zweiphasenfluss-Modellrechnungen zur Berücksichtigung der Gasproduktion in Endlagern und Modellansätze für Langzeitsicherheitsanalysen. BMBF-FKZ 02E 8835-6. GRS-158, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Braunschweig, ISBN 3-931995-20-8.

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

[Leitfaden 2005] Sicherheitsüberprüfung gemäß § 19a des Atomgesetzes - Leitfaden Probabilistische Sicherheitsanalyse, 31. Januar 2005.

[Lemaitre 1996] J. Lemaitre (1996): A Course on Damage Mechanics, 2nd Edition, Springer, Berlin.

[Lerch et al. 2012] Lerch, C.; Bollingerfehr, W.; Filbert, W. & Zhang, Q. (2012): “Thermo-Mechanical Analyses for Evaluating a HLW-Repository Concept.” Proceedings 7th Conference on Mechanical Behaviour of Salt, Paris, France, 2012.

[Leydecker et al. 2008] Leydecker, G., Schmitt, T., Busche, H., Schaefer, T. (2008): Seismoengineering parameters for sites of interim storages for spent nuclear fuel at German nuclear power plants. Soil Dynamics and Earthquake Engineering, 28: 754-762; Amsterdam.

[Lux et al. 1999] Lux, K.H., Hou, Z., Duesterloh, U. & Xie, Z. (1999): Some new aspects on the safety proof for waste disposal in salt mines. International Congress on Rock Mechanics, Ed. G. Vouille & P. Berest, Proceedings Vol. 1, p. 47-50.

[Ludwig 2001] Ludwig, R. R.: Hydrogeologische Untersuchungen Gorleben. Hydrogeologische Grundlagen für Modellrechnungen – Kenntnisstand 2000. Ergebnisbericht der Bundesanstalt für Geowissenschaften und Rohstoffe, Hannover, 2001, unpublished.

[Ludwig & Kösters 2002] Ludwig, R. & Kösters, E.: Hydrogeologisches Modell Gorleben – Entwicklung bis zum paläohydrogeologischen Ansatz. In: Hydrogeologische Modell – Ein Leitfaden mit Fallbeispielen (Arbeitskreis „Hydrogeologische Modelle und Grundwassermanagement“ der Fachsektion Hydrogeologie der Deutschen Geologischen Gesellschaft, Hrsg.), Hannover, Schriftenreihe der Deutschen Geologischen Gesellschaft 24, 69-77, 2002.

[Mauke et al. 2012] Mauke, R.; Stahlmann, J. & Mohlfeld, M. (2012): In-situ verification of a drift system in rock salt – operating experience and preliminary results, Mechanical Behavior of SALT VII – Berest, Ghoreychi, Hadj-Hassen Tijani (eds.), Taylor & Francis, London.

[Minkley & Mühlbauer 2007] Minkley, W. & Mühlbauer, J. (2007): Constitutive models to describe the mechanical behavior of salt rocks and the imbedded weakness planes. The Mechanical Behavior of Salt – Understanding of THMC Processes in Salt, Ed. Wallner, Lux, Minkley & Hardy, p. 119-127, Taylor & Francis, Hannover, Germany.

[Minkley et al. 2010] Minkley, W., Wüste, U., Popp, T., Naumann, D., Wiedemann, M., Bobinsky, J., Tejchman, J. (2010): Beweissicherungsprogramm zum geomechanischen Verhalten von Salinarbarrieren nach starker dynamischer Beanspruchung und Entwicklung einer Dimensionierungsrichtlinie zum dauerhaften Einschluss. BMBF-Projekt FKZ 02C1264. Institut für Gebirgsmechanik GmbH (IfG): 214 S.; Leipzig.

[Mönig et al. 2012] Mönig, I.; Buhmann, D.; Rübel, A.; Wolf, J.; Baltes, B. & Fischer-Appelt, K. (2012): Sicherheits- und Nachweiskonzept. Bericht zum Arbeitspaket 4. Vorläufige Sicherheitsanalyse für den Standort Gorleben, GRS-277, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln, ISBN 978-3-939355-53-3.

[Moog, et al. 2016] Moog, H.; Noseck, U.; Hagemann, S. & Wolf, J. (2016): Geochemical modeling in the near field of a HLW repository in a high-saline environment. FKZ 02E10719 und 02E9954, GRS-419, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Braunschweig. ISBN 978-3-946607-01-4.

[Mrugalla 2011] Mrugalla, S. (2011): Geowissenschaftliche Langzeitprognose. Bericht zum Arbeitspaket 2, Vorläufige Sicherheitsanalyse für den Standort Gorleben, GRS-275, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH: Köln, ISBN 978-3-939355-51-9.

[Müller-Hoeppe et al. 2012a] Müller-Hoeppe, N.; Engelhardt, H.-J.; Lerch, C.; Linkamp, M.; Buhmann, D.; Czaikowski, O.; Herbert, H.-J.; Wieczorek, K. & Xie, M. (2012): Integrität geotechnischer Barrieren – Teil 1: Vorbemessung. Bericht zum Arbeitspaket 9.2, Vorläufige Sicherheitsanalyse für den Standort Gorleben; GRS-287; Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln, ISBN 978-3-939355-63-2.

[Müller-Hoeppe et al. 2012b] Müller-Hoeppe, N.; Breustedt, M.; Wolf, J.; Czaikowski, O.; Wieczorek, K. (2012): Integrität geotechnischer Barrieren – Teil 2: Vertiefte Nachweisführung. Bericht zum Arbeitspaket 9.2, Vorläufige Sicherheitsanalyse für den Standort Gorleben; GRS-288; Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln, ISBN 978-3-939355-64-9.

[Müller-Hoeppe et al. 2016] Müller-Hoeppe, N., Krone, J. & Engelhardt, H.J. (2016): Zur Übertragung der Methode der Teilsicherheitsbeiwerte auf endlagerrelevante Zeiträume.- Tech. Ber., TEC-39-2016-TB, DBE TECHNOLOGY GmbH, Peine.

[Nagra 2002] Nagra: Project Opalinus Clay. Safety Report. Demonstration of disposal feasibility for spent fuel, virtrified high-level waste and long-lived intermediate level waste (Entsorgungsnachweis).- Nagra, Tech. Report 02-05, Wettlingen.

[NEA 2000] Organization for Economic Co-operation and Development – Nuclear Energy Agency (OECD-NEA) (2000): Features, Events and Processes (FEPs) for Geologic Disposal of Radioactive Waste, An International Database.- Organisation for Economic Cooperation and Development (OECD), Paris.

[NEA 2001] Organization for Economic Co-operation and Development – Nuclear Energy Agency (OECD-NEA) (2001): Scenario Development Methods and Practice. Proc. NEA Workshop on Scenario Development. Madrid, Spain, 1999. OECD-NEA, Paris.

[NEA 2012] Organization for Economic Co-operation and Development – Nuclear Energy Agency (OECD-NEA) (2012): Methods for Safety Assessment of Geological Disposal Facilities for Radioactive Waste. Outcomes of the MeSA Initiative. OECD-NEA, NEA-No. 6923, Paris.

[NEA 2013] Organization for Economic Co-operation and Development – Nuclear Energy Agency (OECD-NEA) (2013): The Nature and Purpose of the Post-closure Safety Cases for Geological Repositories. OECD-NEA No. 78121: Paris.

[NEA 2014] Organization for Economic Co-operation and Development – Nuclear Energy Agency (OECD-NEA) (2014): Natural analogues for Safety Cases of Repositories in Rock Salt. Proceedings of a salt club workshop, Braunschweig 2013. OECD-NEA, NEA/RWM/R(2013)10, Paris.

[NEA 2016] Organization for Economic Co-operation and Development – Nuclear Energy Agency (OECD-NEA) (2016): Scenario Development Workshop Synopsis. Integration Group for the Safety Case. OECD-NEA; 1-3 June 2015, NEA/RWM/R(2015)3, Paris, France, March 2016.

[NEA-IGSC 2012] Organization for Economic Co-operation and Development – Nuclear Energy Agency (OECD-NEA-IGSC) (2012): Cementitious Materials in Safety Cases for Geological Repositories for Radioactive Waste: Role, Evolution and Interactions. OECD/NEA-IGSC-Workshop, Radioactive Waste Management NEA/RWM/R(2012)3/REV March.

[Neubert, 2014] Neubert, N.: Stability and integrity of geotechnical barriers under earth-quake loading, 43. Geomechanik-Kolloquium, Veröffentlichungen des Instituts für Geotechnik der TUBAF, Hsg. H. Konietzky, Heft 2014-2

[Neubert, 2015] Neubert, N.: Weiterführende Untersuchungen zur Integrität geotechnischer Barrieren unter Erdbebeneinwirkung, Technischer Bericht TEC-Nr. 12-2015-TB, DBE Technology GmbH, 2015

[Nipp et al. 2003] Nipp, H.-K.; Eickemeier, R.; Heusermann, S.: Projekt Gorleben - Erkundungsbegleitende großräumige gebirgsmechanische Berechnungen. Abschlussbericht. Hannover: BGR, 2003

[Peiffer et al. 2011a] Peiffer, F.; McStocker, B.; Gründler, D.; Ewig, F.; Thomauske, B.; Havenith, A. & Kettler, J. (2011): Abfallspezifikation und Mengengerüst. Basis Laufzeitverlängerung der Kernkraftwerke (September 2010). Bericht zum Arbeitspaket 3, Vorläufige Sicherheitsanalyse für den Standort Gorleben, GRS-274, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln, ISBN 978-3-939355-50-2

[Peiffer et al. 2011b] Peiffer, F.; McStocker, B.; Gründler, D.; Ewig, F.; Thomauske, B.; Havenith, A. & Kettler, J. (2011): Abfallspezifikation und Mengengerüst. Basis Ausstieg aus der Kernenergienutzung (Juli 2011). Bericht zum Arbeitspaket 3, Vorläufige Sicherheitsanalyse für den Standort Gorleben, GRS-278, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Köln, ISBN 978-3-939355-54-0.

[Pöhler et al. 2010] Pöhler, M., Amelung, P., Bollingerfehr, W., Engelhardt, H., Filbert, W., & Tholen, M. (2010). Referenzkonzept für ein Endlager für radioaktive Abfälle in Tongestein - ERATO -. Tech. rep., DBE TECHNOLOGY GmbH

[Popp et al. 2012] Popp, T.; Salzer, K.; Schulze, O. & Stührenberg, D. (2012): Hydro-mechanische Eigenschaften von Salzgrusversatz – Synoptisches Prozessverständnis und Datenbasis, Memorandum IFG – BGR, 30.05.2012.

[Pröhl & Gering 2002] Pröhl, G. & Gering, F. (2002): Dosiskonversionsfaktoren zur Berechnung der Strahlenexposition in der Nachbetriebsphase von Endlagern nach dem Entwurf der Allgemeinen Verwaltungsvorschriften zu §47 Strahlenschutzverordnung. GSF-Forschungszentrum für Umwelt und Gesundheit, Institut für Strahlenschutz. Neuherberg.

[Reiche 2016] Reiche, T. (2016): RepoTREND – Das Programmpaket zur integrierten Langzeitsicherheitsanalyse von Endlagersystemen. GRS-416, BMWi-FKZ 02E10367. ISBN 978-3-944161-95-2. Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) gGmbH, Braunschweig.

[Reiter 2014] Reiter, S. (2014): Effiziente Algorithmen und Datenstrukturen für die Realisierung von adaptiven, hierarchischen Gittern auf massiv parallelen Systemen. Dissertation, Universität Frankfurt.

[Rübel et al. 2013] Rübel, A.; Buhmann, D.; Meleshyn, A.; Mönig, J. & Spiessl, S. (2013): Aspects on the Gas Generation and Migration in Repositories for High Level Waste in Salt Formations, GRS-303, BMWi-FKZ 02E10719, ISBN 978-3-939355-82-3. Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Braunschweig.

[Röhlig et al 2014] Röhlig, K.-J.: Plischke, E.; Li, X.: Probabilistische Methoden als Hilfsmittel zur Bemessung von Verschlussbauwerken für ein Endlager im Salinar, Bericht, Institut für Endlagerforschung, TU Clausthal, Januar 2014

[Salzer et al. 2015] K. Salzer, R.-M. Günther, W. Minkley, D. Naumann, T. Popp, A. Hampel, K.-H. Lux, K. Herchen, U. Düsterloh, J.G. Argüello & F. Hansen (2015): Joint Project III on the comparison of constitutive models for the thermo-mechanical behavior of rock salt – II. Extensive laboratory test program with clean salt from WIPP. In: L. Roberts, K. Mellegard & F. Hansen (Hrsg.): Proc. of the Conference on Mechanical Behavior of Salt (Saltmech VIII), South Dakota School of Mines & Technology, Rapid City, SD, USA, 26.-28.05.2015, Taylor & Francis Group, London, ISBN 978-1-138-02840-1, S. 3-12.

[Schneider 2004] Schneider, K.J. (2004): Bautabellen für Ingenieure mit Berechnungshinweisen und Beispielen. 16. Auflage, Werner Verlag.

[Schneider 2012] Schneider, A. (ed.) (2012): Enhancement of the codes d^3f and r^3t. Final Report, GRS-292, FKZ 02E10336, ISBN 978-3-939355-68-7, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Braunschweig.

[Schneider & Birthler 2004] Schneider, A. & Birthler, H. (2004): Modellrechnungen zur großräumigen dichteabhängigen Grundwasserbewegung, GRS-191, FKZ 02C06284, ISBN 3-931995-59-3, Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, Braunschweig.

[Schramm et al. 2009] Schramm, M., Kühnlenz, T., Mingerzahn, G., Hammer, J. (2009): Projekt Gorleben: Lösungen im Salzstock Gorleben – eine Dokumentation und genetische Interpretation. – Bundesanstalt für Geowissenschaften und Rohstoffe (BGR), Hannover, Germany (2009).

[Seismosoft, 2012] Seismosoft: SeismoSignal. Version 5.0. Pavia: Seismosoft Ltd., 2012

[Seismosoft, 2013] Seismosoft: SeismoArtif. Version 2.1. Pavia: Seismosoft Ltd., 2013

[SKB 2006] SKB (2006): Long-term safety for KBS-3 repositories at Forsmark and Laxemar – a first evaluation. Main report oft he SR-Can project.- Svensk Kärnbränslehantering AB, Stockholm.

[StrlSchV 2012] Verordnung über den Schutz vor Schäden durch ionisierende Strahlung (Strahlenschutzverordnung – StrlSchV) vom 20. Juli 2001 (BGBl. I 2001, Nr. 38, S. 1714, BGBl. I 2002, Nr. 27, S. 1459), zuletzt geändert durch Artikel 5 Abs. 7 des Gesetzes vom 24. Februar 2012 (BGBl. I 2012, Nr. 10, S. 212).

[StandAG 2013] Gesetz zur Suche und Auswahl eines Standortes für ein ENdlager für Wärme entwickelnde Abfälle und zur Änderung und zur Änderung aderer Gesetze (Standortauswahlgesetz – StandAG) vom 23. Juli 2013

[Vigfusson et al. 2007] Vigfusson, I.; Madoux, J., Raimbault, Ph., Röhlig, K.-J. & Smith, R (2007): European Study on the regulatory review of the safety case for geological disposal of radioactive waste. Case study: Uncertainties and their management. http://www.grs.de/node/976

[Wagner 2005] Wagner, K. (2005): Beitrag zur Bewertung der Sicherheit untertägiger Verschlussbauwerke im Salinargebirge, Dissertation an der TU Bergakademie Freiberg.

[Weber & Keller 2008] Weber, J.R. & Keller, S. (2008): FEP-Generierung und Szenarienentwicklung. Stand November 2006. Bundesanstalt für Geowissenschaften und Rohstoffe, Tagebuch-Nr.: 10402/08, Hannover.

[Westphal et al. 2010] Westphal, G., Kristen, G., Wegener, W., Ambatiello, P., Geyer, H., Epron, B., Bonal, C., Steinhauser, G. & Götzfried, F. 2010. Sodium Chloride. Ullmann's Encyclopedia of Industrial Chemistry.

[Wolf et al. 2012a] Wolf, J.; Behlau, J., Beuth, T., Bracke, G., Bube, C., Buhmann, D., Dresbach, C., Hammer, J., Keller, S., Kienzler, B., Klinge, H., Krone, J., Lommerzheim, A., Metz, V., Mönig, J., Mrugalla, S., Popp, T., Rübel, A., Weber, J.R. (2012a): FEP-Katalog für die VSG. Konzept und Aufbau. Bericht zum Arbeitspaket 7, Vorläufige Sicherheitsanalyse für den Standort Gorleben. Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-282, Köln, ISBN 978-3-939355-58-8.

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

[Xie & Herbert 2012] Xie, M. & Herbert, H. J. (2012): Geochemical Analysis of the Sealing System. Technical Report to Work Package 9.1.2 Preliminary Safety Case of the Gorleben Site (VSG), Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) mbH, GRS-308, Köln, ISBN 978-3-939355-87-8.

[Zirngast 1991] Zirngast, M. (1991): Die Entwicklungsgeschichte des Salzstocks Gorleben - Ergebnisse einer strukturgeologischen Bearbeitung. In: Geologisches Jahrbuch, Vol. A 132, S. 3-31, Bundesanstalt für Geowissenschaften und Rohstoffe (BGR): Hannover.

10 Appendices

Appendix A: Groundwater flow and hydrochemical regime in the overburden and adjoining rocks

This chapter describes the procedures applied in hydrogeological model calculations in more detail, compared to chapter 0. Based on geological data, the existing methods and tools for hydrogeological calculations have been assessed in the project ISIBEL. These calculations aim at simulating the density-driven flow in the overburden of a potential repository for radioactive waste.

The code d^3f++ was used to simulate groundwater flow. The code was developed under the auspices of GRS ([Fein & Schneider 1999], [Fein 2004]) and are continuously being verified and enhanced ([Birthler et al. 2000], [Schneider & Birthler 2004], [Fein et al. 2008], [Schneider 2012]. With this code all important flow and transport processes in the far field of a potential repository can be simulated, e. g. density-driven flow, sorption, and colloid-bound transport. The code d^3f++ was especially developed to model large regions with complex geometries over long periods of time. The hydrogeology of the modelled area can thereby comprise strong heterogeneities and anisotropies.

The flow model of d^3f++ solves the equation system of density-driven flow transport which includes all interactions that are currently relevant for long-term safety assessment such as decay, sorption, precipitation, complexation, and colloidal transport [Fein 2004].

Two preprocessors to construct the geometry and grid file for d^3f++ in the required file format (.ugx) are available: GISLab and ProMesh [Reiter 2014]. Both programs have been developed in the framework of the E-DuR project [Schneider 2012]. GISLab provides the possibility to read data for subsurface contour plots from a GIS or point-wise data from a text file and to preprocess it for the subsequent use in ProMesh. Files in the Drawing Interchange File Format (dxf) and the Wavefront OBJ Format (obj) can be imported into and exported from GISLab. ProMesh is a cross platform tool for the creation, manipulation and optimisation of 2- and 3-dimensional volume geometries, developed at the Goethe Center for Scientific Computing at the University of Frankfurt. Its user interface allows to visualize geometries, to select different parts and elements of those geometries and to apply a variety of algorithms on the whole geometry or on selected parts [Schneider 2012] ProMesh can import .obj files and a variety of other file formats. Grids are saved in the ProMesh-specific ugx file format and can then serve as basis for simultions with d^3f++.

ParaView 4.1.0 by Kitware [Ayachit 2015] was used as postprocessor for d^3f++ to analyse, visualise and plot simulation results.

The interpreted programming language and environment R (www.R-project.org) was chosen for data analyses and processing because it provides a variety of functions for these purposes. Furthermore, all steps that were undertaken to obtain a certain output file were documented in the script files.

ArcGIS 10 by ESRI was used for some spatial processing of the data, e.g. intersection of the Gorleben data set with the model boundaries or the interpolation of the groundwater levels for each geographical location.

Visualisation of the data was one of the main tasks because it facilitated to make decisions about necessary data manipulations and provided the means to control their implementation. The 3-dimensional geological modelling tool Leapfrog Hydro by ARANZ Geo Ltd. was used for visualisation purposes because the input files could be generate relatively easy and the program provided several helpful functions like vertical exaggeration, measurement and transparency tools.

A.1 The Gorleben data set

The starting point for the development of the hydrogeological model was data provided by the Federal Institute for Geosciences and Natural Resources (BGR). The data set constituted output of the hydrogeological structural model described in [Ludwig 2001] which was constructed with the program openGEO4 which is an add-in to Autocad 14. A text file was created by using virtual scan beams on a 200 m × 200 m grid and extracting the depth of every layer interface (cf. Figure A.1).

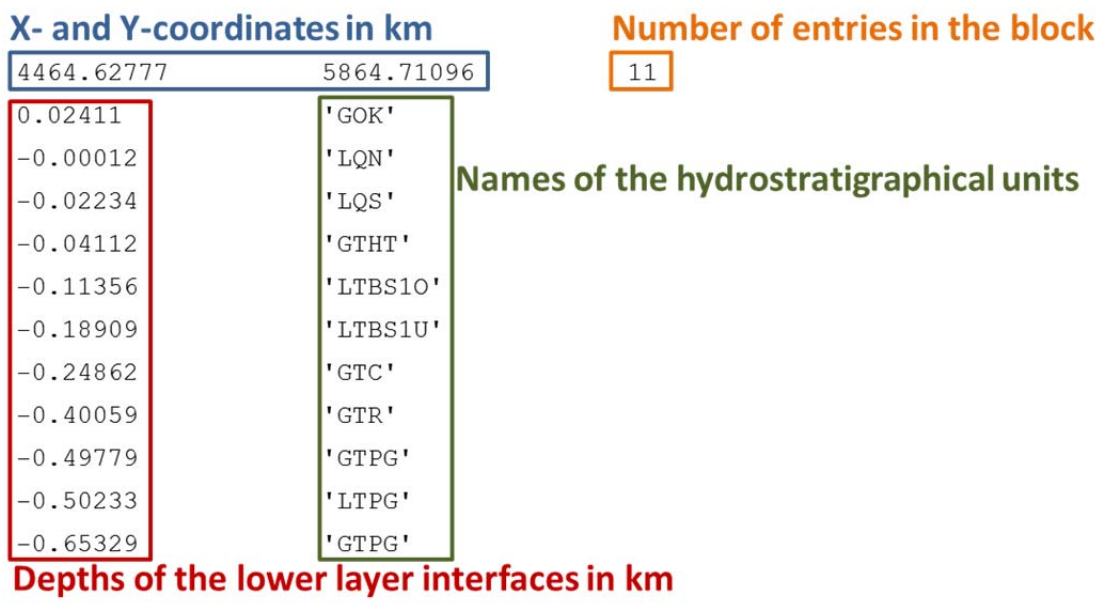

Figure A.1: Structure of the Gorleben data set shown for one location.

The text file contained information about 157,000 layer interfaces at 11,848 geographical locations. In the following section, the unit formed by all layer interfaces at one geographical location ordered by depth will be called “profile”. Thus, each profile contained the x- and y-coordinates, the height of the ground level, and the depths and names of all layer interfaces.

In the Gorleben data set 27 different signatures were used to identify the topographical surface and the hydrostratigraphical units (cf. Table A.1). The naming of the hydrostratigraphical units was based on the nomenclature in Table A.2.

Table A.1: Signatures used in the Gorleben data set. Colors indicate the hydraulic conductivity: > 10^{-4} m s^{-1} (blue), 10^{-8} – 10^{-4} m s^{-1} (yellow), < 10^{-8} m s^{-1} (red), based on [Ludwig 2001]

gok	gqeMg	gqeT	gqeTo	gqeU	gqh
gqhol	gqpe	gqsMg	gqsT	gqsU	gtC
gtHT	gtHTW	gtPg	gtR	lhg	lqe
lqN	lqpe	lqS	ltBS1o	ltBS1u	ltBS2
ltHTS	ltPg	Sstck			

The hydrostratigraphical units are divided into three classes concerning their hydraulic conductivity: aquifers with moderate to high conductivity, aquitards with very low to low conductivity and aquitards with extremely to very low conductivity (cf. Table A.1).

Table A.2: Nomenclature for the naming of the hydrostratigraphical units in the Gorleben data set [Ludwig 2001]

Symbol	Meaning	Symbol	Meaning
gok	topographical surface	qhol	Quaternary/Holstein interglacial
l	aquifer	qe	Quaternary/Elster glaciation
g	aquitard	tHTW	Tertiary/Hamburg clay alternation
U	silt	tHT	Tertiary/Hamburg clay
T	clay	tBS	Tertiary/Brown-coal sand
Mg	boulder clay	tC	Tertiary/Chatt
q	Quaternary	tR	Tertiary/Rupel clay
qh	Quaternary/Holocene	tPg	Tertiary/Paleogene
N	lower terrace	Hg	Cap rock
qs	Quaternary/Saale glaciation	Sstck	Salt dome

Two additional ASCII-files were provided by the BGR: one file with information about the groundwater surface and one with information about the horizontal extent of the model domain. The first file contained coordinates and heights of points that formed the groundwater contour lines of the shallow aquifer in the Gorleben area. The second was a Golden Software blanking file and contained coordinates of the points forming the boundaries of the model domain. These boundaries were based upon the courses of both the main streamlines and the receiving streams within the area [Ludwig & Kösters 2002].

A.2 General corrections of the Gorleben data set

Before the Gorleben data set could be prepared for the construction of a 2-dimensional geometry and grid, some general corrections had to be performed. Moreover, the data set contained more information than required for the representation of the groundwater system, i. e. layers outside the relevant area.

The first step was to sort the entries of each profile within the Gorleben data set according to depth. Next, the data set was corrected. This was necessary because the data contained inconsistencies i. e. several layer interfaces within one profile had the same depth. In these cases only the layer with the highest position in the profile was kept. The data then was visualised in Leapfrog and the plausibility of the updated profiles was verified.

The original data set contained 26 different hydrostratigraphical units (cf. Table A.1). Some of the units were merged:

- gqeU and gqeTo were merged into gqeT
- gqsT was merged into gqsU

In the next step, all entries in the profiles irrelevant for the hydrogeological model were removed. This was decided based on the groundwater system in the Gorleben area. Here, the hydraulic basis is formed by the Rupel clay (gtR) because nearly the entire groundwater circulation takes place in the overlying Tertiary and Quaternary layers [Ludwig & Kösters 2002]. Only above the Gorleben salt dome the Rupel clay shows a gap so that the groundwater system has direct contact to the salt dome. Thus, the salt dome forms the hydraulic basis in this area. This narrows down the units to be considered in the hydro-geological model to the ones above the Rupel clay or – if the Rupel clay does not exist in the profile – above the salt dome. The remaining hydrostratigraphical units are listed in Table A3.

Table A.3: Overlook of the hydrostratigraphical units contained in the data set after correction and selection of the relevant data (based on [Ludwig 2001]).

Period	Epoch	Name	Description
Quaternary	Holocene	gqh	Alluvial clay, mud, fen peat
	Pleistocene	lqN	Alluvial sand and gravel, dunes, subordinately melt-water sand and gravel (Weichsel glaciation)
		lqs	Melt-water sand and gravel (Saale glaciation)
		gqsU	Basin silt and clay (Saale glaciation)
		gqsMg	Boulder clay (Saale glaciation)
		gqhol	Silt, clay, mud, partially sand and turf (Holstein interglacial)
		gqeT	Lauenburg clay complex: basin silt and clay, subordinately basin sand (Elster glaciation)
		lqe	Melt-water sand and gravel (Elster glaciation)
		gqeMg	Boulder clay (Elster glaciation)
		gqpe	Alternation of basin silt, mud and alluvial sand (Cromer and Bavel complex)
		lqpe	Alluvial sand (Menap glaciation)
Tertiary	Miocene	ltHTs	Sand (upper brown-coal sands)
		gtHTW	Alternation of sand, silt, and clay (Hamburg clay)
		ltBS2	Sand, subordinately brown-coal seams (brown-coal sands)
		gtHT	Clay, silt (Hamburg clay)
		ltBS1	Sand, subordinately brown-coal seams (lower brown-coal sands, Neochattian)
	Oligocene	gtC	Silt, sandy at the top, clayey at the bottom (Eochattian)
Quaternary – Upper Cretaceous		lHg	Cap rock

A.3 Construction of a hydrogeological model

The hydrogeological system above and around a deep repository is a 3-dimensional structure, often of high geological complexity. The first approach to characterise and analyse the flow field is often to build a 2-dimensional model. When wisely chosen, the 2-dimensional model enables to analyse the general behaviour of the flow system and provides at the same time the great advantage of reduced complexity. Generally, hydrogeological models are generated based on borehole data. The interpretation of the borehole data, e. g. the division into hydrostratigraphical units, depends on the purpose of the model. The course of the layers between the emerging hydrostratigraphical profiles must be estimated to generate a vertical cross-section.

For the Gorleben area much work has already been done to assess the spatial structure of the underground resulting in the Gorleben data set. It was therefore decided to extract the

data necessary to build a model section from this data set. The following steps were carried out to build the groundwater model:

- selection and extraction of a suitable cross-section
- adaptation of the chosen cross-section
- construction of the model geometry
- grid generation
- simulation

A.3.1 Selection of a suitable cross-section

Several things have to be considered when selecting a cross-section for a 2-dimensional simulation. The cross-section should run approximately parallel to the main groundwater flow because all flow perpendicular to the cross-section cannot be represented in two dimensions. Furthermore, the model section should include the region of interest. If the aim is for example to model the transport of radionuclides from a deep repository, the area where the radionuclides potentially enter the groundwater system should be contained in the cross-section.

An appropriate visualisation tool is crucial to assess the suitability of a cross-section for the particular modelling purpose. The tool should provide the functions to visualise the data in an appropriate manner and some means to select the layers or zones to be displayed. Also some sort of cutting or slicing tool is necessary in order to visualise single cross-sections. Leapfrog Hydro provides all these functions (cf. Figure A.2) and was therefore chosen as visualisation tool. Moreover, the input files for Leapfrog Hydro could be generated with R.

The Gorleben data was sighted and a cross-section with North-South orientation was chosen as basis for further steps (Figure A.3). The course of the cross-section is displayed in Figure A.4. For simplicity reasons only sections crossing the model area in a straight line were considered. The chosen cross-section included the north-western rim syncline and a zone where the cap rock over the salt dome had direct contact to the deep aquifer. It further contained a direct contact between the shallow and the deep aquifer north to the salt dome. In the original Gorleben data set, the layering in this profile was not clear because there were two entries (lqe and gqeT) with the same depth. During the editing of the data set, the aquitard layer gqeT was removed in favour of the aquifer layer lqe. This procedure might have created a gap in the aquitard and therefore a direct connection between the shallow and deep aquifer where there is none. Thus, the resulting profile has to be regarded as only one possibility and its uncertainty should be borne in mind.

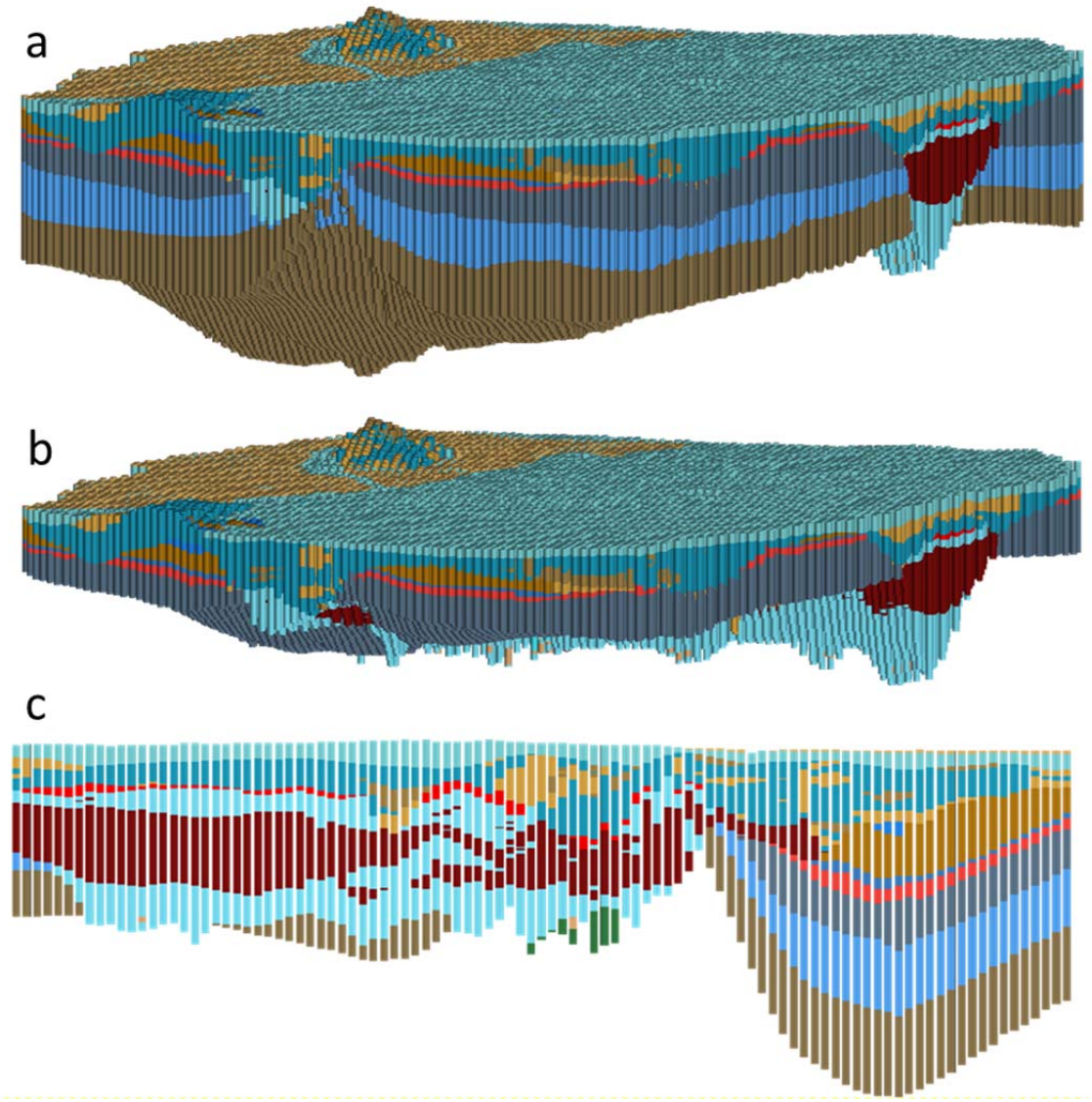

Figure A.2: **Visualisation with Leapfrog Hydro: Display of profiles as columns (a), display of selected layers (b), display of selected columns (c). All pictures with 15x exaggeration.**

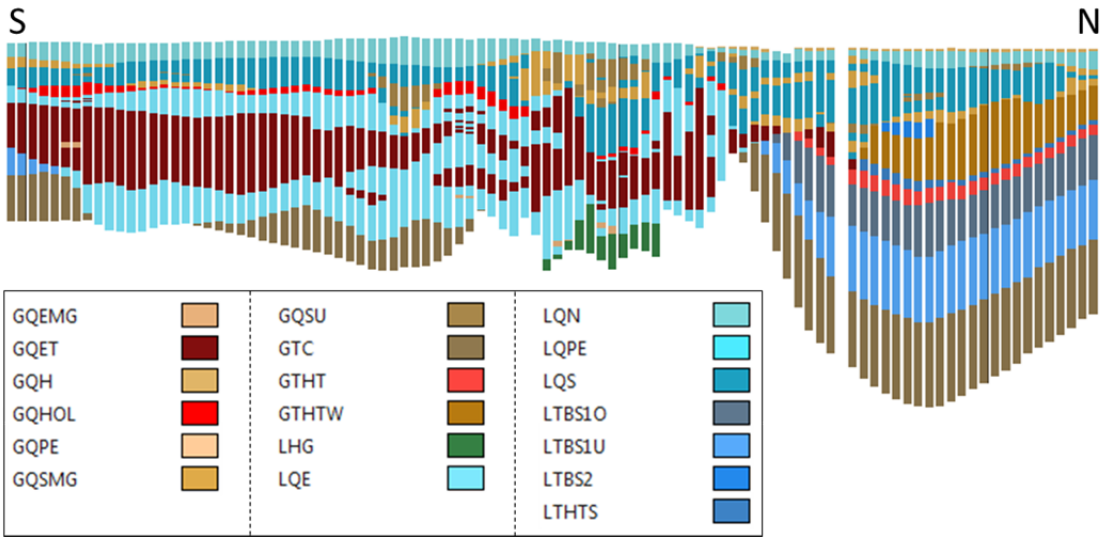

Figure A.3: Cross-section for the creation of a 2-dimensional model along the Gaussian X-coordinate 44578277.77 (15x exaggeration)

The selected cross-section partly did not run perpendicular to the groundwater contour lines and thus parallel to the groundwater flow in the shallow aquifer but especially above the salt dome the deviation from the right angle was less than 10°. This adds to the uncertainties of the later model results but was assumed to be acceptable in consideration of the main purpose of this work.

A.3.2 Adaptation of the cross-section

The adaptation of the selected cross-section concerned two fields:

- simplification
- preparation for the automatic generation of the model geometry

The simplification of the cross section included for example removal of small lenses and merging of layers with similar properties. The goal of this operation was to reduce the complexity of the later model and mitigate zones that are potentially problematic for the grid generation like very thin layers. The challenge of this work was to avoid modifications of the geology that were likely to cause major changes in the flow field.

For further processing, i. e. the automatic generation of the model geometry, the data set had to fulfil the condition that each identifier appears only once in a column. Thus, zones with two or more occurrences of the same unit were divided into continuous blocks and each block obtained another identifier (Figure A.4). When a layer was first continuous and then divided into an upper and a lower part, the last occurrence before the division was split, too. The split was performed in the depth the neighbouring occurrence of the surrounded layer had its ver-

tical centre. This point would later be used for the construction of the surrounded layers´ edge (see below “Construction of the model geometry”).

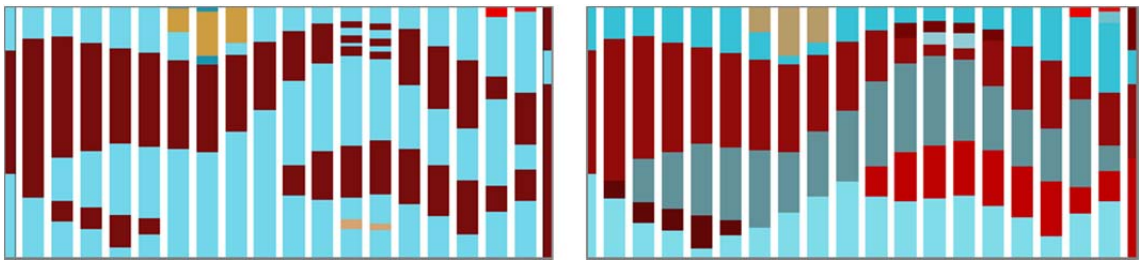

Figure A.4: Adaptation of the cross-section. Left: original columns, right: simplified and regrouped entries. The colours indicate different identifiers. 15x exaggeration.

The operations described above were done manually by using several user-defined R functions. All commands that were executed to obtain the final version of the data set were saved in an R file so that each change can later be reconstructed and modifications may easily be performed.

A simple visualisation tool was programmed in R in order to provide a fast and easy way to visualise changes while editing the data. By typing a simple command, a window with bar charts for the chosen column and its neighbouring columns appears (Figure A.5). Bar charts based on the original data are shown in the top row while the actual layering of the columns is displayed below. This mode of representation gives the user the possibility to immediately check the results of his changes and to potentially avoid mistakes.

A further adaptation step concerned the upper boundary of the model. Typically for the simulation of the flow in the saturated zone, the upper boundary condition represents in some way the groundwater table. This may be performed by either assigning a function that defines the corresponding pressure for each point or by assuring that the upper model edge coincides with the water table. Then a pressure of 0 MPa may be assigned to the upper boundary. For the present model the first possibility was chosen and the topmost entry of each column was set to 21 m according to the highest point in the original cross-section.

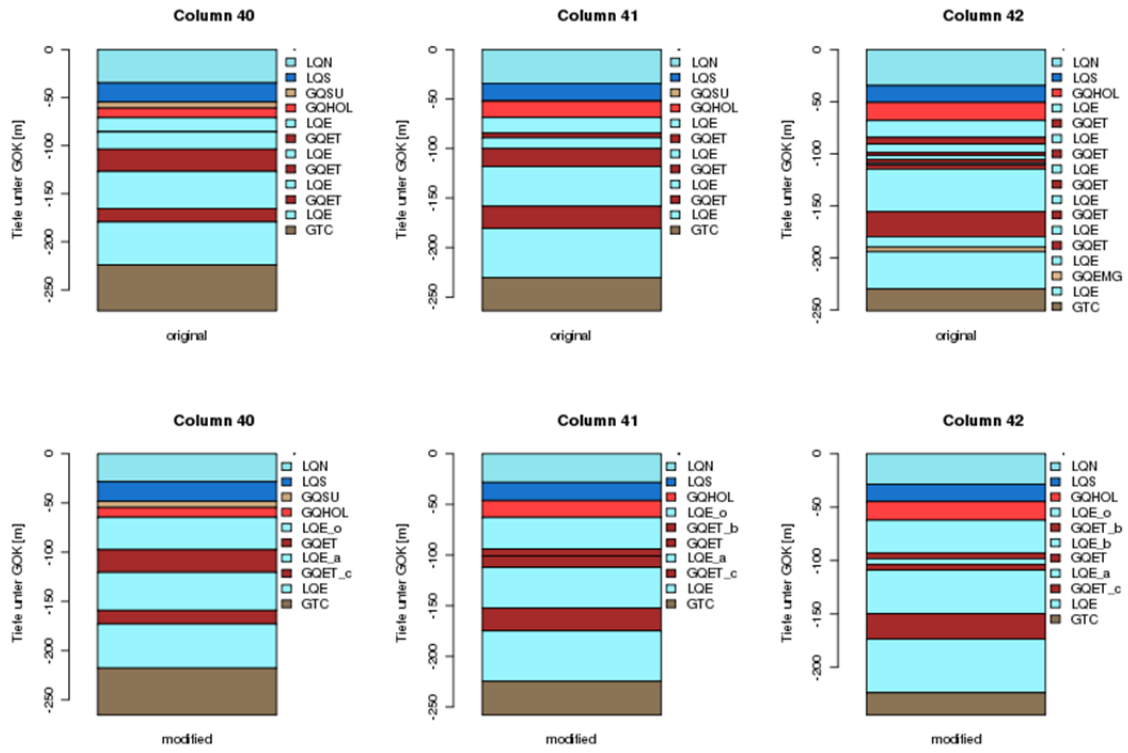

Figure A.5: Simple visualization tool in R

The height of the water table was determined for each profile in ArcGIS. The groundwater level was interpolated based on a file provided by the BGR. The groundwater level was then extracted for each profile and written into a text file. This information was used later to define the pressure boundary condition at the top of the model area.

A.3.2.1 Construction of the model geometry

The model geometry consists of nodes and connecting lines that represent the interfaces between layers or otherwise defined units. The position of all entries in the 100 profiles defined by X-coordinate, Y-coordinate and depth were saved as nodes.

The interfaces between the layers were generated automatically but had to be reworked manually. For each entry in each profile the neighbouring profiles were searched for entries with the same identifier. Connecting lines between these entries were then saved. If the identifier did not occur in the neighbouring profile, i. e. the layer thinned out between the profiles, it was checked whether the identifier of the layer below the target entry was present in the neighbouring profile. If this was true, the connection between these two entries was saved. Otherwise the layer above the target entry was checked in the same way.

This approach implied the simplification that each layer starts or ends in the centre of the profile next to its last occurrence. The “real” edge of the layer could lie anywhere between these two profiles.

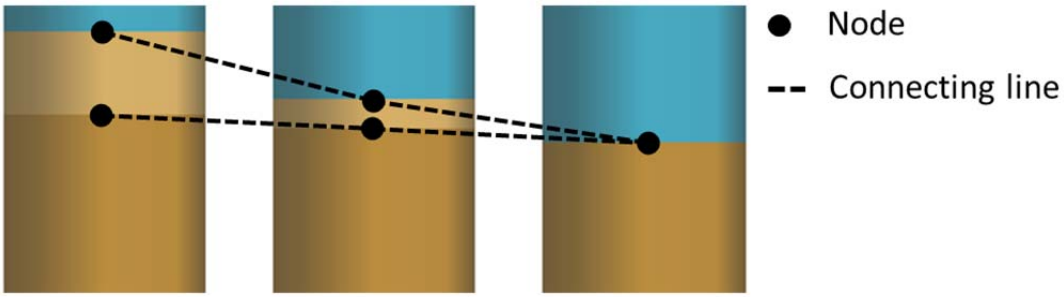

Figure A.6: Construction of nodes and connecting lines for the geometry file based on the profiles

The visualisation and manual correction of the automatically generated geometry were performed with GISLab [Schneider 2012]. In a first step, the geometry was sighted and problematic areas were identified. For most cases, the approach yielded plausible results. Problems occurred if neither the target layer itself nor the layer above or below it occurred in the neighbouring profile. Then, the lines forming the edge of the layer had to be set manually.

In several cases the chosen approach yielded very pointed angles where layers thinned out. Pointed angles are problematic for the numerical stability and should therefore be avoided. The simplest way to dispose of these unfavourable angels was to move the point marking the end of the layer nearer to the last occurrence (cf. Figure A.7 a and b). This approach was chosen for situations where a change would probably have not a high influence on the resulting flow field, i. e. at the bottom of the model or where an aquifer layer was enclosed in an aquiclude layer.

For more sensitive regions i. e. the edges of clay lenses, it was decided to prolong the layer from its last occurrence half way to the next profile and to complete it then with a less pointed angle (cf. Figure A.7 c and d). This approach was considered to be a compromise between the two extremes that the layer could end next to the profile with its last occurrence or next to the first profile without occurrence. Both situations would potentially result in different flow fields and the flow field resulting from this approach should lie somewhere between them.

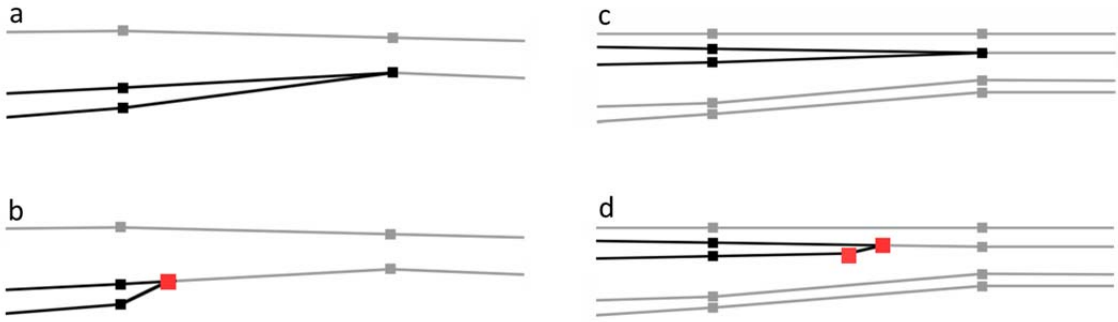

Figure A.7: Correction of pointed angles in GISLab: Original (a, c) and corrected version (b, d)

In few profiles, layers thinner than 4 m occurred. The thinner a layer is the smaller are the elements to be generated and the higher the total number of elements of the grid. Furthermore, the elements will potentially have a more unfavourable shape. Thus, the relevant profiles were changed so that the thickness of all layers exceeded 4 m.

One of the thin aquifer layers (ltHTS) lay entirely between an aquitard and an aquiclude layer. Shielded by these low permeable layers, the aquifer layer should show a reduced groundwater flow and should therefore play a minor role for the groundwater system. Based on these assumptions, the layer was slightly broadened to a minimum thickness of 7 m to facilitate the generation of more favourable elements.

In a next step, dispensable nodes were removed to optimise the geometry for the grid generation. The aim of this process was to reduce the number of fixed nodes because they can influence the grid quality negatively.

A simple function was used in GISLab to remove certain nodes. If the angle between the segment to the left and the segment to the right of a node was larger than a given value, the node was removed. Values between 179.0° and 179.9° were applied. Stricter conditions, i. e. higher values, were assigned to certain section where only small changes were desired, e. g. the interfaces of thin layers.

A comparison between the geometry before and after the manual corrections is shown in Figure A.8. Due to the vertical exaggeration, changes that act in the vertical direction appear more important than changes in the horizontal direction. The most prominent changes concern the areas where the generation algorithm yielded lines that were implausible or expendable. Furthermore, the lines are to some extend less smooth due to the removal of nodes. However, the main characteristics of the geological structure were maintained.

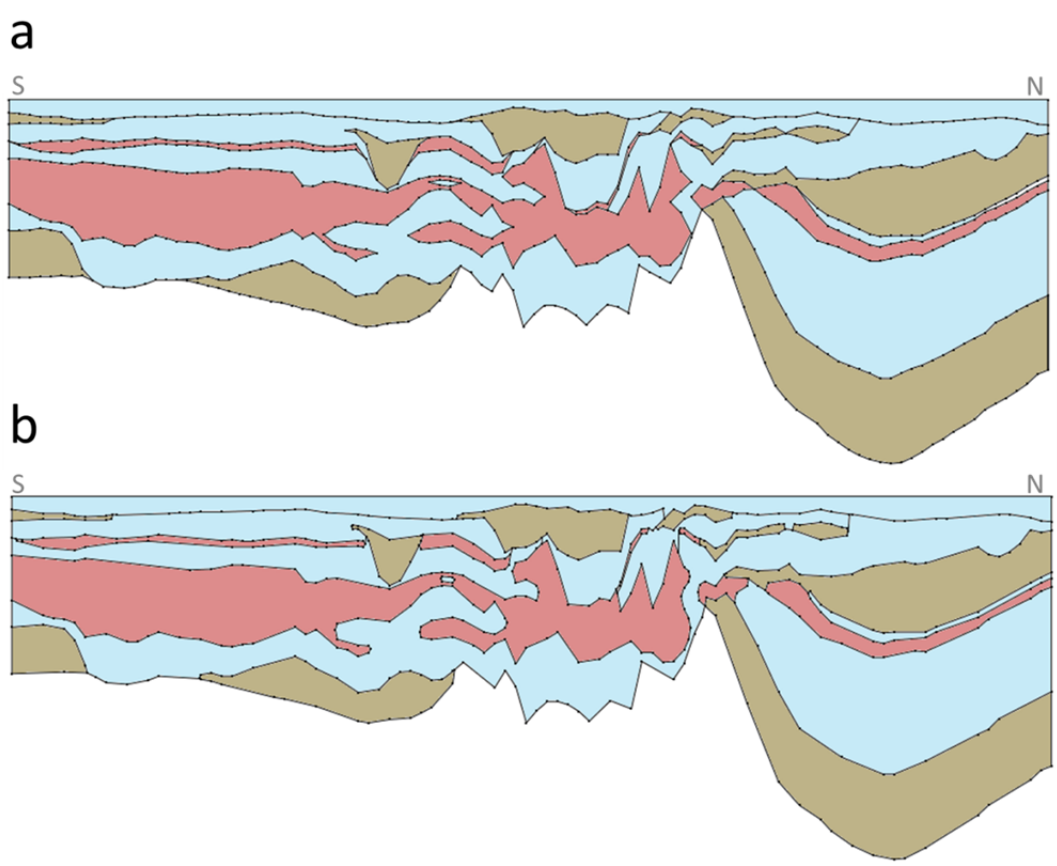

Figure A.8: **Automatically generated geometry (a) and the same geometry after the manual corrections (b); aquifers are coloured blue, aquitards brown and aquicludes red (15x exaggeration)**

A.3.2.2 Grid generation

The grid generation is an important step on the way to the construction of a numerical model. Difficulties arise from the occurrence of very thin layers which make it necessary to generate very small elements. In this context, elements with very pointed or obtuse angles may occur which should be avoided because of their negative influence on solver convergence.

The grid generation was performed with ProMesh 4 [Reiter 2014]. All connecting lines in the geometry were refined into segments of less than 100 m length. The interfaces of very thin layers (4 to 7 m) were even split into segments smaller than 50 m. The sweep-line-triangulation algorithm [DeBerg et al. 2000] was applied to construct a grid that was then optimized by an algorithm based on [Frey & Borouchaki 1998]. The resulting grid contained 4,316 nodes and 7,798 elements. It had a total length of 19,800 m and a depth of up to 440 m.

A.3.2.3 Simulation

The grid described in Chapter A.3.2.2 was used as basis for flow simulations with variable density and viscosity. Three hydrogeological units were distinguished according to [Ludwig 2001]: aquifers with a moderate to high permeability, aquitards with a very low to low permeability and aquicludes with an extremely low to very low permeability (cf. Figure A.9). The hydraulic parameters of the three units are listed in Table A.4. They correspond to the parameters already used in [Birthler et al. 2000] except for the transversal dispersivity which was set to 1.0 m.

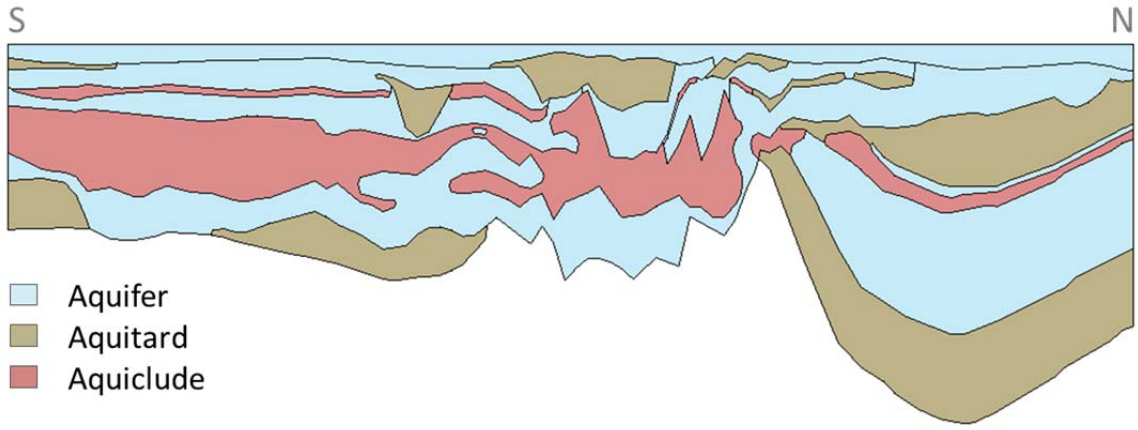

Figure A.9: Model section with three hydrogeological units

Table A.4: Hydraulic parameters of the three hydrogeological units

Quantity	Unit	Aquifer	Aquitard	Aquiclude
Permeability	m^2	$1.0 \cdot 10^{-12}$	$1.0 \cdot 10^{-14}$	$1.0 \cdot 10^{-16}$
Porosity	-	0.2	0.2	0.2
Tortuosity	-	1.0	1.0	1.0
Longitudinal dispersivity	m	10.0	10.0	10.0
Transversal dispersivity	m	1.0	1.0	1.0
Molecular diffusion coefficient	$m^2\ s^{-1}$	$1.0 \cdot 10^{-9}$	$1.0 \cdot 10^{-9}$	$1.0 \cdot 10^{-9}$

Initially, the bottom part of the model except for the northern rim syncline is filled up to a depth of 190 m with pure brine (relative salt mass fraction ωrel = 1, cf. Figure A.10). Then a transition zone of 20 m is defined where the salt mass fraction declines linearly to $\omega_{rel} = 0$. Above the depth of 170 m the entire model is filled with fresh water ($\omega_{rel} = 0$).

The salt mass fraction ω is the percentage of the salt in the fluid mass by weight. Results are represented by using the relative salt mass fraction ω_{rel} which is defined in d^{3}f++ by

$$\omega_{rel} = \frac{\omega}{\omega_{\max}} \qquad \text{with} \qquad \omega = \frac{m_s}{m_s + m_f}$$

Here, m_s is the mass of the salt and m_f is the mass of the fluid. The maximum salt mass fraction $\omega_{\max}$ was set to 0.26. Thus, ω_{rel} = 0 corresponds to fresh water and ω_{rel} = 1 to a saturated NaCl solution at 20 °C.

Pressure according to the pressure profile depicted in Figure A.10 is assigned to the upper boundary to represent the groundwater level. No salt enters the model area through this boundary, i. e. ω_{rel}= 0.The other boundaries were defined to be impermeable for flow. Again, ω_{rel} = 0 is assigned on the boundaries except for the contact zone with the salt dome where ω_{rel} is set to 1

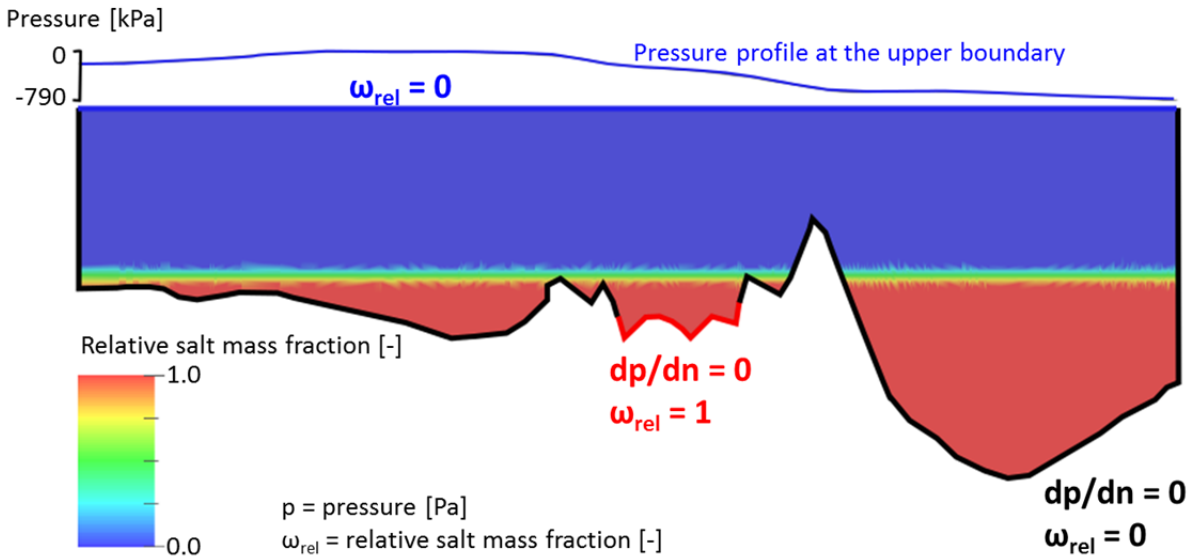

Figure A.10: Initial and boundary conditions for the 2-dimensional model

Simulations were performed for 10,000 years model time (see Figure A.11). Within this time, the initial salt distribution dissolves and zones of high salt concentration remain only above the salt dome and in the north-western rim syncline. The salt spreads not only within the aquifer but also through aquitard and aquiclude layers such that a characteristic salt distribution evolves.

The flow velocity corresponds mainly to the permeability. Highest flow velocities are found near the top boundary and at narrow passages within the aquifer. Water enters the model area through the upper boundary where the hydraulic pressure imposed by the boundary condition is highest. Effects of the density on the flow field can be observed above the salt dome where the flow appears to be non-directional and in the north-western rim syncline where large flow vortices evolve.

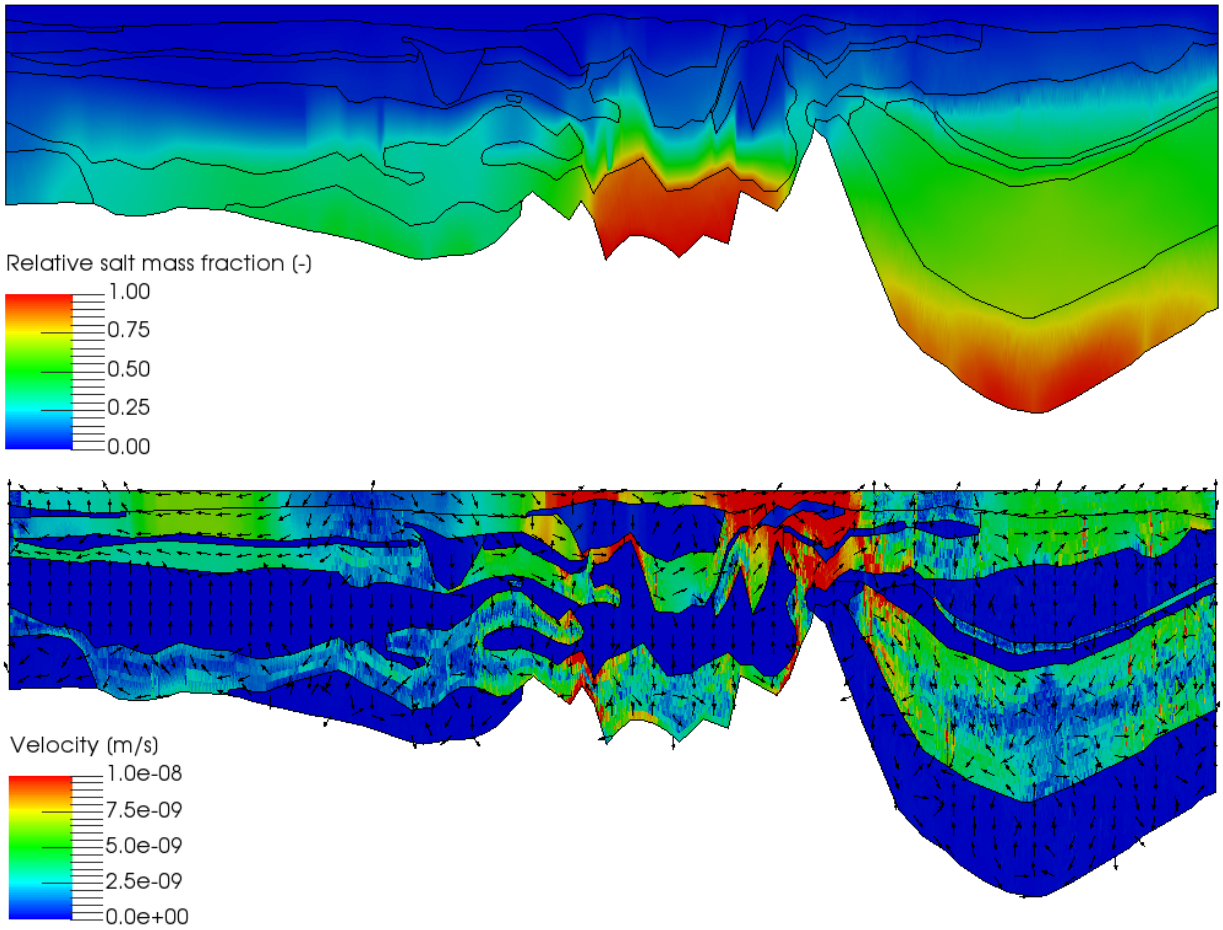

Figure A.11: Simulation results after 10,000 years model time for the hydrogeological model: relative salt mass fraction (top) and flow velocity (bottom)

The work described in this chapter shows that tools are available to perform the following tasks:

- processing of input data for the model geometry, the initial and boundary conditions
- grid generation
- simulation of density-driven flow
- visualisation of simulation results

Appendix B: Rock Mechanics Calculations

B.1 Constitutive Modelling

The validity of a constitutive equation and in particular its potential to extrapolate the described behaviour beyond the range of a measured data set relies on the ability of the equation to reflect the physical processes. If there are no measured data available, which represent the in-situ stress and temperature range, the evaluation of the suitability of a constitutive equation can only be based on the assessment of the included physical relationships.

B.1.1 Stationary Creep

The inelastic deformation of rock salt is mainly determined by the behaviour of its dislocations. The basic equation is given by the Orowan-equation.

$$\dot{\varepsilon}_{creep} = M\, b\, \rho_{mob}\, \mathrm{v} \tag{0.1}$$

It is defined by a geometry factor of the crystal (Taylor factor $M = 3$), the Burgers vector (width b of a dislocation), the mobile dislocation density ρ_{mob} and their mean velocity v. The most uncertain factor in this equation is the description of the velocity as a function of temperature and driving deviatoric stress, strongly determined by the sort of process responsible for the movement of the dislocations. It has to be distinguished between the climb of edge dislocations[11] and gliding of (screw- and edge-) dislocations. They are characterized by different temperature and stress dependencies. It is widely accepted that at room temperature and in the range of the stresses commonly used in laboratory “cross slip” is the dominating process for gliding. For rather low stresses (some few MPa or less) climb should be predominant.

Applying statistical physics to the diffusion like movement of dislocations, their velocity results as a temperature dependent hyperbolic relation on stress.

$$\mathrm{v} = \mathrm{v}_0 \exp\left(\frac{-Q}{RT}\right) \sinh\left(\frac{b\,\gamma\,\sigma_{eff}}{M\,k_B\,T}\right) \tag{0.2}$$

Parameters are given by a constant prefactor v_0, an activation energy Q, an activation area γ, the gas constant R and the Boltzmann constant k_B. The effective stress[12] is defined by

[11] Absorption or emittance of point defects leads to growth or evanescence of edge dislocations.
[12] The term „effective stress“ is used in a number of other relations. Here it is used only in the meaning of eq. (0.3).

the difference of global deviatoric stress σ_{dev} and an inner shielding "frictional stress" σ_R, resulting from interactions between the dislocations.

$$\sigma_{eff} = \langle \sigma_{dev} - \sigma_R \rangle \tag{0.3}$$

The Foeppl symbols $\langle \ \rangle$ constrain to positive values only.

$$\begin{aligned} \langle x \rangle &= x \quad \text{if} \quad x \geq 0, \\ \langle x \rangle &= 0 \quad \text{if} \quad x < 0. \end{aligned} \tag{0.4}$$

The frictional stress is bound to the dislocation density.

$$\sigma_R = \alpha\, b\, G \sqrt{\rho_{disl}} \tag{0.5}$$

In principle the density ρ_{mob} of mobile dislocations and the mean dislocation density ρ_{disl} have to be distinguished. On the other side there is a smooth transition and both can be taken as proportional to each other so that the indices $_{mob}$ and $_{disl}$ can be dispensed. Eq. (0.5) so can be transformed to

$$\rho = \left(\frac{\sigma_R}{\alpha\, b\, G} \right)^2 . \tag{0.6}$$

In stationary state both σ_R and ρ keep constant. Furthermore the frictional stress reaches a constant relation to the macroscopic stress σ_{dev}.

$$\sigma_R = z\, \sigma_{dev}\,, \qquad z \approx 0.6 - 0.7 \tag{0.7}$$

From eq. (0.1) in conjunction with eqs. (0.2), (0.3), (0.6) and (0.7) for stationary creep follows

$$\begin{aligned} \dot{\varepsilon}_{creep}^{stat} &= M\, b\, \rho\, \mathrm{v} = M\, b \left(\frac{\sigma_R}{\alpha\, b\, G} \right)^2 \mathrm{v}_0 \exp\left(\frac{-Q}{RT} \right) \sinh\left(\frac{b\, \gamma\, \sigma_{eff}}{M\, k_B\, T} \right) \\ &= M\, b\, \mathrm{v}_0 \left(\frac{z\, \sigma_{dev}}{\alpha\, b\, G} \right)^2 \exp\left(\frac{-Q}{RT} \right) \sinh\left(\frac{b\, \gamma\, (1-z)\, \sigma_{dev}}{M\, k_B\, T} \right) \\ &= A\, \sigma_{dev}^2 \exp\left(\frac{-Q}{RT} \right) \sinh\left(\Gamma \frac{\sigma_{dev}}{k_B\, T} \right) \end{aligned} \tag{0.8}$$

with

$$\begin{aligned} A &= \left(\frac{z}{\alpha\, b\, G} \right)^2 M\, b\, \mathrm{v}_0\,, \\ \Gamma &= \frac{b\, \gamma\, (1-z)}{M}. \end{aligned} \tag{0.9}$$

In the course of the next chapter it will be proved that for a certain range of stress and temperature the hyperbolic behaviour is in very good agreement with a power-law-attempt

(NORTON-law). So, when the hyperbolic approach is in good accordance with a suitable power of stress,

$$\mathrm{v} = \mathrm{v}_0' \exp\left(\frac{-Q'}{RT}\right)\sigma_{eff}^{n} \tag{0.10}$$

the stationary creep results in

$$\begin{aligned}\dot{\varepsilon}_{creep}^{stat} &= M\,b\frac{\sigma_R^{\;2}}{(\alpha\,b\,G)^2}\mathrm{v}_0' \exp\left(\frac{-Q'}{RT}\right)\sigma_{eff}^{n} = M\,b\frac{z^2\,\sigma_{dev}^{\;2}}{(\alpha\,b\,G)^2}\mathrm{v}_0' \exp\left(\frac{-Q'}{RT}\right)\left((1-z)\,\sigma_{dev}\right)^n \\ &= A' \exp\left(\frac{-Q'}{RT}\right)\sigma_{dev}^{m}\end{aligned} \tag{0.11}$$

with

$$A' = \left(\frac{z}{\alpha\,b\,G}\right)^2 (1-z)^n\,M\,b\,\mathrm{v}_0' \tag{0.12}$$

and

$$m = n + 2\,. \tag{0.13}$$

It has to be taken into account that the activation energy in the hyperbolic model and in the NORTON-attempt are not identical because of the temperature term in the argument of the hyperbolic function. For non-stationary creep of course the effective stress term takes a different development with creep compared to that of the frictional stress.

So power-laws may be a very good representation of real stationary creep behaviour. But due to a restricted physical basis their ability for being extrapolated is constrained. Phenomenologically an additive description by two or more power laws may help as long as there is a laboratory support by appropriate data.

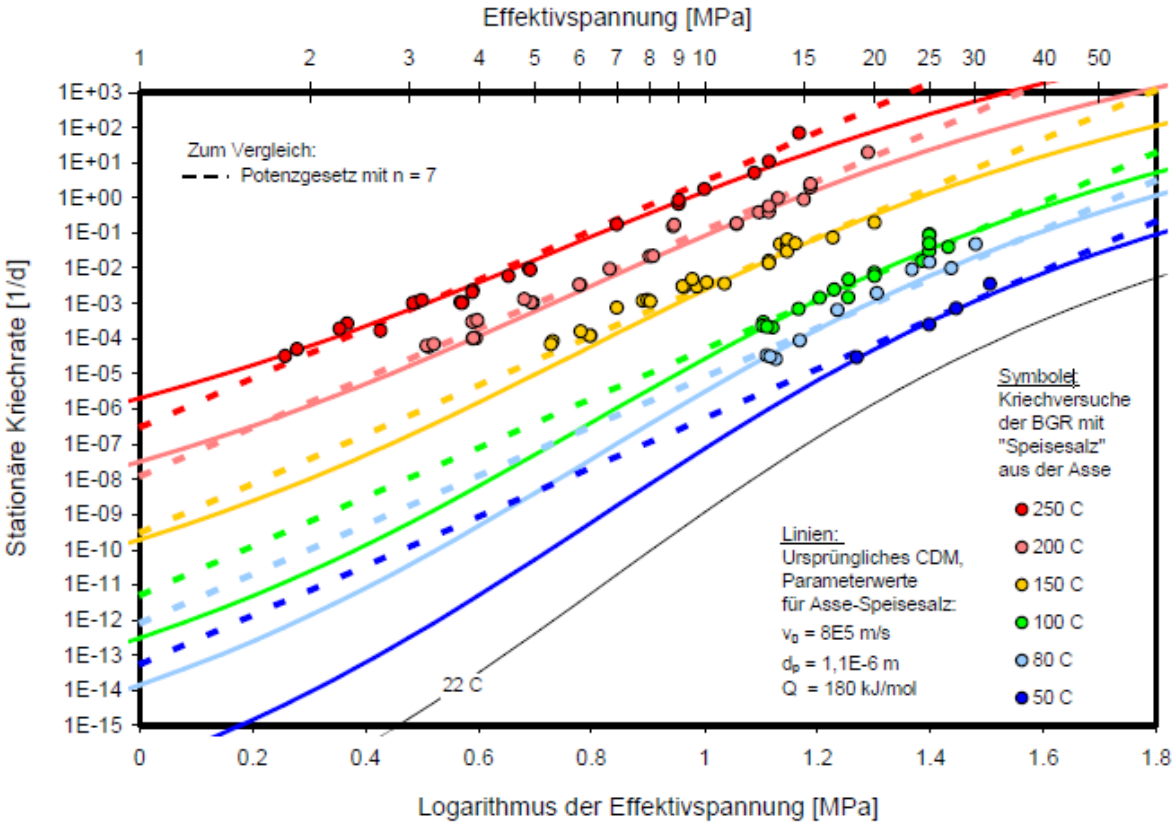

Figure B.1: **Fit of the hyperbolic creep law (here CDM-law of Hampel) and a Norton model with power n = 7 (dashed lines) to the data of BGR on stationary creep. Taken from [Hampel 2012].**

On the other hand in case of a phenomenological power law for stresses no clear conclusions can be drawn about the existence of a hyperbolic stress dependency as long as there is not enough data to distinguish between them. In addition to the hyperbolic law literature [Frost & Ashby 1982] provides a linear stress dependency for the velocity of climbing dislocations leading to a power $m = 3$ in connection with the dislocation density (s. eq.(0.6)).

$$\dot{\varepsilon}_{creep}^{stat} = M\, b \frac{\sigma_R^{\;2}}{(\alpha_c\, b\, G)^2} \mathrm{v}_c \exp\left(\frac{-Q_c}{RT}\right)\sigma_{eff} \;\rightarrow\; A_c \exp\left(\frac{-Q_c}{RT}\right)\sigma_{dev}^3 \qquad (0.14)$$

Further attempts try to justify other values for the exponent. But in fact they do not reach the plausibility of the cubic law above. Higher stress exponents probably are phenomenological occurrences of a hyperbolic relation. It can easily be shown that both attempts can be converted one to the other. See for example:

$$\frac{\Delta \ln(\dot{\varepsilon})}{\Delta \sigma} \simeq \frac{\partial \ln(\dot{\varepsilon})}{\partial \sigma}$$
$$= n \frac{\partial \ln(\sigma_{eff})}{\partial \sigma} = \frac{n}{\sigma_{eff}}$$
$$\cong \frac{\partial}{\partial \sigma}\left(\frac{b\gamma\sigma_{eff}}{M k_B T}\right) = \frac{b\gamma}{M k_B T} \tag{0.15}$$

So in case of a sudden small change of stress, the change in creep rate delivers a power $n\ (=m-2)$ as well as a corresponding activation area γ. Eq. (0.15) clearly shows that the exponent n probably will change with stress and temperature when a certain range of stress and temperature shall be spanned. The same is true for the activation energy. A change in these parameters so not necessarily does mean a change in the physical mechanisms of deformation. Merely climb may lead to a cubic stress dependency while pure diffusion is characterized by a linear stress dependency, especially in case of sufficient humidity.

There is some evidence [Hampel 2012], [Hampel 2015] that models which have been fitted to laboratory data give to small creep rates for very small stresses. An additive attempt by a further creep term with another power exponent may be a solution for the discrepancies – as long as it can also be established by experimental data or convincing theory.

B.1.2 Nonstationary Creep

Nonstationary creep physically evolves from the development of the dislocation density as a result of dislocation multiplication and annihilation. In case of stationary creep these two processes cancel each other out leading to a constant density. The dislocations have a deformation bearing function (s. eq.(0.1)) but also hinder each other due to their elastic stress fields (s. eqs. (0.2) and (0.3)).In consequence virgin salt for constant stress shows a rather high creep rate which is going down with rising dislocation density. This process which is called hardening has been physically modelled by [Heemann 1989] in his thesis and later transformed to a simplified description.

The effect of nonstationary creep should not be ignored in the short and medium term especially. In the framework of work package 3 a quantitative comparison of the models with experimental data could not be performed.

B.1.3 Dilatancy, tertiary creep and failure

Dilatancy means the development of micro cracks as a result of incompatible crystalline deformation at grain boundary triple-points or due to crossing of shearbands. In both cases local stress concentrations may evolve that are able to tear open grain boundaries or the

crystal itself. Start of dilatancy is mainly determined by deviatoric stress as a driving force and maximum main stress component (minimum pressure component) which is restricting the dimension of the cracks and their coalescence.

The cracks are rather closely bound to grow vertical to the minimum pressure component thus giving rise to a marked anisotropy. Unfortunately experimental data on that are rather scarce and in numerical modelling this aspect is generally ignored. This has to be seen critical because this anisotropy has effects on the development of stress and especially on the anisotropy of permeability. It cannot be excluded that there are serious effects on the pathways around a disposal.

With growing density of the micro cracks these can coalesce leading to a decrease of stiffness in the direction of loading and drastically growing anisotropic permeability. Crack development is accompanied by local stress relief and thus stress rearrangement leading to enhanced loading in other parts and thus enhanced creep. Physically this results in a local stress rise inverse to the elastic stiffness

$$\sigma_{eff}^{dil} \sim \frac{\sigma_{dev}}{E^{dil}}, \tag{0.16}$$

as it is well known from damage theory[13]. With the reduction of the effective elastic stiffness a rising creep rate results as it can be seen in case of tertiary creep.

Heemann has modelled the coalescence of the micro cracks and could model the collapse of stiffness (failure) quite well [Heemann 1989] compared to experimental data. From this the maximum deviatoric stress prior to failure results as linearly depending on minimum vertical pressure what is a quite good description for all stresses to be expected in a disposal. So the start of failure is the consequence of a certain state of accumulated damage in conjunction with a critical state of stress. The cracks are leading to enhanced local stresses and shear and hence accelerated damage production – that is more and coalescing cracks.

B. 2 On the Analysis of Experimental Data on Creep of Rock Salt

B.2.1 Stationary Creep

During early investigations on creep of rock salt a stress exponent $m = 5$ seemed to give the best fit. But there is some evidence that for stresses higher $15\,MPa$ (at room temperature) a higher exponent is given (s. Figure B.1). This improvement had been achieved [Hampel et al. 1998] by longer times of creeping in conjunction with a better evaluation by plotting the log of creep rates over creep strain which clearly shows when stationary creep is reached. Another method is given by driving specimens to stationary creep from under- or overhardened states

[13] The effect of hardening as well as the full treatment of elastic anisotropic stiffness has been omitted in this simplified equation.

[Salzer et al. 2015] thus giving upper and lower bounds to the parameters of stationary creep.

In Figure B.2 experimental data on stationary creep of WIPP-salt (New Mexico, flat bedding) for low stresses at a temperature of $T = 60°C$ are displayed joined with numerical calculations done by seven different models and codes. The linear run of KIT corresponds to the BGRa-Norton-model ($m = 5$), accelerated by a factor of 2.5.

The creep curves that have been established in the framework of the „Verbundprojekt 'Stoffgesetzvergleich'" for low stresses ($\sigma_{dev} \leq 5\,MPa$) seem to approach an exponent $m \approx 2$. However, there is only one experimental value supporting this interpretation (s. creep rate for $2\,MPa$ in Figure B.2). It has to be noticed that no physical theory predicts an exponent 2. In particular the value of the activation energy is not based on experimental data. Furthermore the mechanisms of creep will have to be clarified and the transferability onto the north German salt is to be proven. Here especially the role of humidity has to be investigated. So further experimental work is needed.

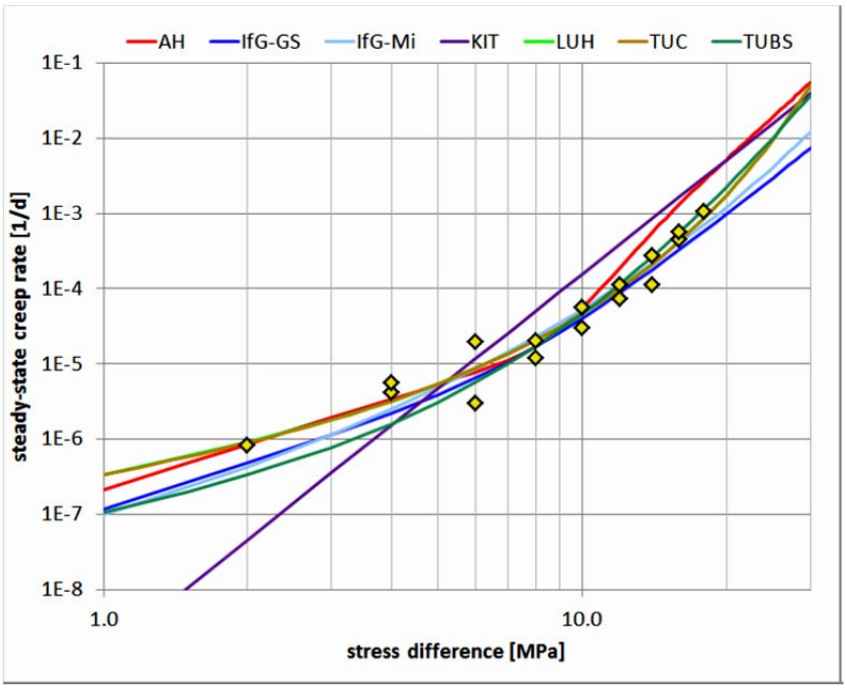

Figure B.2: **Experimental data on stationary creep of WIPP-salt for low stresses at a temperature of T = 60°C. Numerical calculations done by seven different models and codes are included (Hampel (CDM), IfG (Günther / Salzer und Minkley), Karlsruhe (KIT), Lux/Wolters (LUH), TU Clausthal (TUC), TU Braunschweig (TUBS)). See [Hammer et al. 2016]. Courtesy of R.-M. Günther.**

In case of the extremely low creep rates for small stresses it is advisable to make experiments on small specimens (may be single crystals) where by means of defined stress pulses and etching the velocity of the dislocations can be determined. With the help of eq. (0.1) the

creep rate might be determined. Furthermore from the direction in which the dislocation moves it can be taken whether creep is determined by climb or glide or a combined process. So there may be a further safety in assessing a theoretically based modelling.

Nevertheless, it should be possible to find parameters by means of classical creep tests for high temperatures. If they agree with the parameters already known it is a confirmation of the results and expansion of their applicability. If not, a further mechanism has been found that might influence the creep rates also for low stresses and temperatures. It is presumed that different mechanisms generally run parallel so that the creep rates have to be implemented additively into the codes. If it captures the long-term creep rates of old galleries quantitatively correct, with sufficient care it can be concluded that the modelling is in accord with the physical or at least phenomenological behaviour.

The creep of salt mainly is determined by the stress development in the vicinity of the cavity. However, the distant areas get more and more involved into the process of creep and thus convergence gets decelerated by this. It's not easy to say down to which level of stress laboratory tests will be necessary. But the experiments should come as near as possible to about 2 MPa deviatoric stress. Temperature mainly has the effect of enhancing creep velocity. As long as there is no change in the governing process the creep rates at low temperatures can be calculated by means of the activation energy. Taking the relation displayed in Figure B.2 such experiments could be done with creep times of less than a year about.

There are a lot of creep curves for constant stress or deformation rate that easily can be taken for comparisons with the models. Unfortunately salt of different locations can show quite different creep ability. But nevertheless for a special location the models should span a wide range of stress and temperature with just one set of parameters.

B.2.2 Primary Creep, Dilatancy, Tertiary Creep and Failure

The creep curves mentioned above also contain a lot of information about the development of hardening i.e. change of stress or creep rate. There is experimental evidence as well as theoretical support that primary creep strain is proportional to the applied (constant) stress or for constant creep rate proportional to the finally achieved stationary stress.

Many experiments have been done on the onset of dilatancy and its development as a function of deviatoric stress and given minimal lateral pressure (mostly provided by a hydrostatic stress). Depending on the methods of measuring dilatancy the results show a similar characteristic but still also apparent differences.

Unfortunately there are poor data on the development of the effective elastic stiffness – not to mention the anisotropy connected with it. Even if the modelling of dilatancy might be enough for certain aspects its connection with damage (anisotropic loss of elastic stiffness) and failure is nearly not investigated.

B.3 Comparison of Numerical Techniques

Generally the numerical techniques of Finite Elements or Finite Differences are developed such that the numerical differences usually should be less than the differences from uncertainties in the local material behaviour (volatility of parameters) or even some inaccuracies in laboratory experiments. For an assessment of the techniques there is further need in knowledge of the different numerical methods used, the calculation time, the amount of memory needed and the numerical stability as well as sensitivity to distorted elements.

Appendix C: Detailed integrity proof of the salt concrete seal under earthquake conditions

In VSG, 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 stiftness as a rule increase more and more quickly 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 has to be verified by detailed integrity proofs.

Exemplary investigations summarized below show that the effect of the earthquake loading is insignificant in the sealing body and the surrounding rock salt. Due to a lack of data for the dynamic material properties of the contact zone, its evaluation is still pending.

The model for integrity proof is based on the shaft seal design developed in VSG [Müller-Hoeppe et al. 2012a]. It provides numerical verification on the integrity of the shaft sealing system under static loading [Müller-Hoeppe et al. 2012b]. In [Neubert 2014] and [Neubert 2015] the detailed proof of the salt concrete sealing element under seismic loading was supplemented.

C.1 Earthquake - database and generation

Seismic loads including acceleration time-histories, velocity time-histories or stresses are required inputs for conducting numerical simulations of the dynamic processes associated with earthquakes in the code FLAC^{3D} [ITASCA 2012].

The engineering seismic design parameters for a potential earthquake at the Gorleben site are identified in Tab. C.1 [Wolf et al. 2012b].

Table C.1: Seismic design parameters used for the Gorleben site

Parameter	Symbol	Value
Macroseismic intensity	I_0	7.3 (MSK-scale)
Acceleration, horizontal	a_h	1.4 m/s²
Acceleration, vertical	a_v	0.7 m/s²
Probability of occurrence per year	$W_ü$	$0.5 \cdot 10^{-6}$ – $1.0 \cdot 10^{-6}$
Subsurface strong ground motion duration	t_{0u}	3.0 s

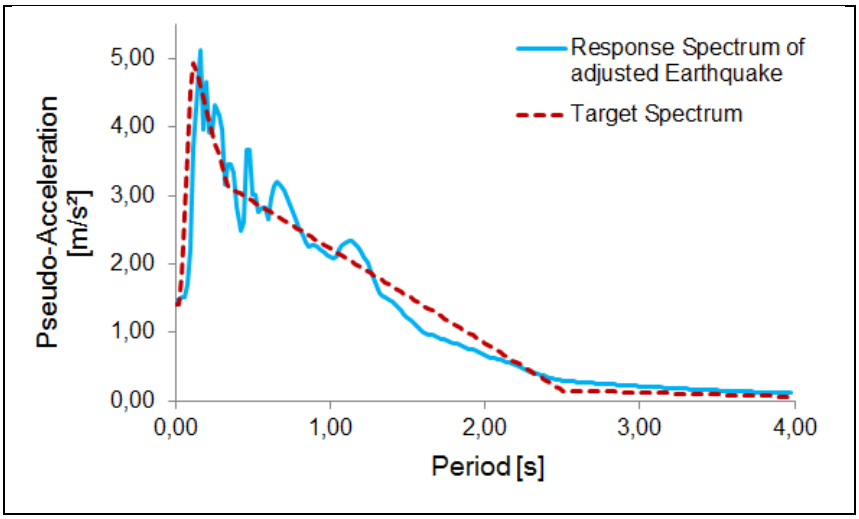

Figure C.1: Response spectra for the basic design earthquake (target spectrum) and the generated and adjusted earthquake

These parameters coupled with a response spectrum that describes the frequency-content of the potential earthquake are necessary inputs used to generate the acceleration time-history. The response spectrum (i.e. target spectrum) for the basic design earthquake is shown in Figure C.1 as a dashed line.

The artificial accelerogram (or velocity-history) was generated in SeismoArtif [Seismosoft 2013] by superposition of sine waves with different frequencies, amplitudes and phase angles. The random time-history was adjusted to the target spectrum and an envelope-shape to describe the rise-phase, the strong-motion-phase and the descending part of the earthquake-history. Figure C.2 shows the generated artificial velocity history as a dashed line.

In order to implement the time-histories in FLAC^{3D}, it was necessary to adjust the generated time-histories. Baseline correction and filtering were performed using the software code SeismoSignal [Seismosoft 2012]. Baseline corrections are required to account for physically unrealistic velocities or displacements that may occur at the end of a particular time-history. The velocities, where a baseline correction is performed, are equal to zero after the earthquake excitations are finished. To reduce calculation time, the high frequency content was removed by filtering the signal. Frequencies above approximately 9.0 Hz in the signal were removed using a low-pass-filter (Butterworth-filter) with a cutoff frequency of 8.0 Hz.

Table C.2: Key parameters derived for the generated earthquake history

Parameter	Symbol	Value
Maximum acceleration	a_{max}	1.486 m/s²
Maximum velocity	v_{max}	0.126 m/s
Maximum displacement	u_{max}	0.029 m
Maximum frequency	f_{max}	9.0 Hz
Duration	t_{Dg}	5.20 s

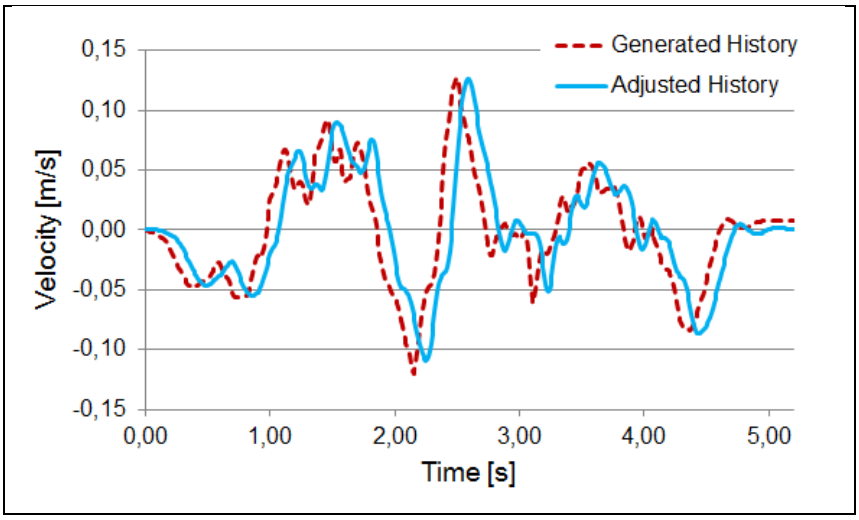

Figure C.2: Velocity histories of the generated and the adjusted earthquake

The blue line in Figure C.2 shows the effects of the baseline correction and of filtering. The signal is generally smoothed, out of phase and with velocity peaks predominantly a little lower. The response spectrum of the generated adjusted earthquake history is shown in Figure C.1 by the blue line. Both spectra show good agreement. Key data for the generated earthquake history are identified in Tab. C.2.

C.2 Numerical simulation

C.2.1 Numerical model and calculation procedure

The basis for the generic model is provided by the section of the salt-concrete sealing system used in the VSG between depths of 650 m and 780 m [Müller-Hoeppe 2012a].

The model area is at a depth of 620 m to 810 m and has a horizontal extension of 160 m in each coordinate direction. The surrounding rock is Leine rock salt with homogeneous areas

representing zones susceptible to creeping. The numerical model with the zones of different creep potential is shown in Figure C.3.

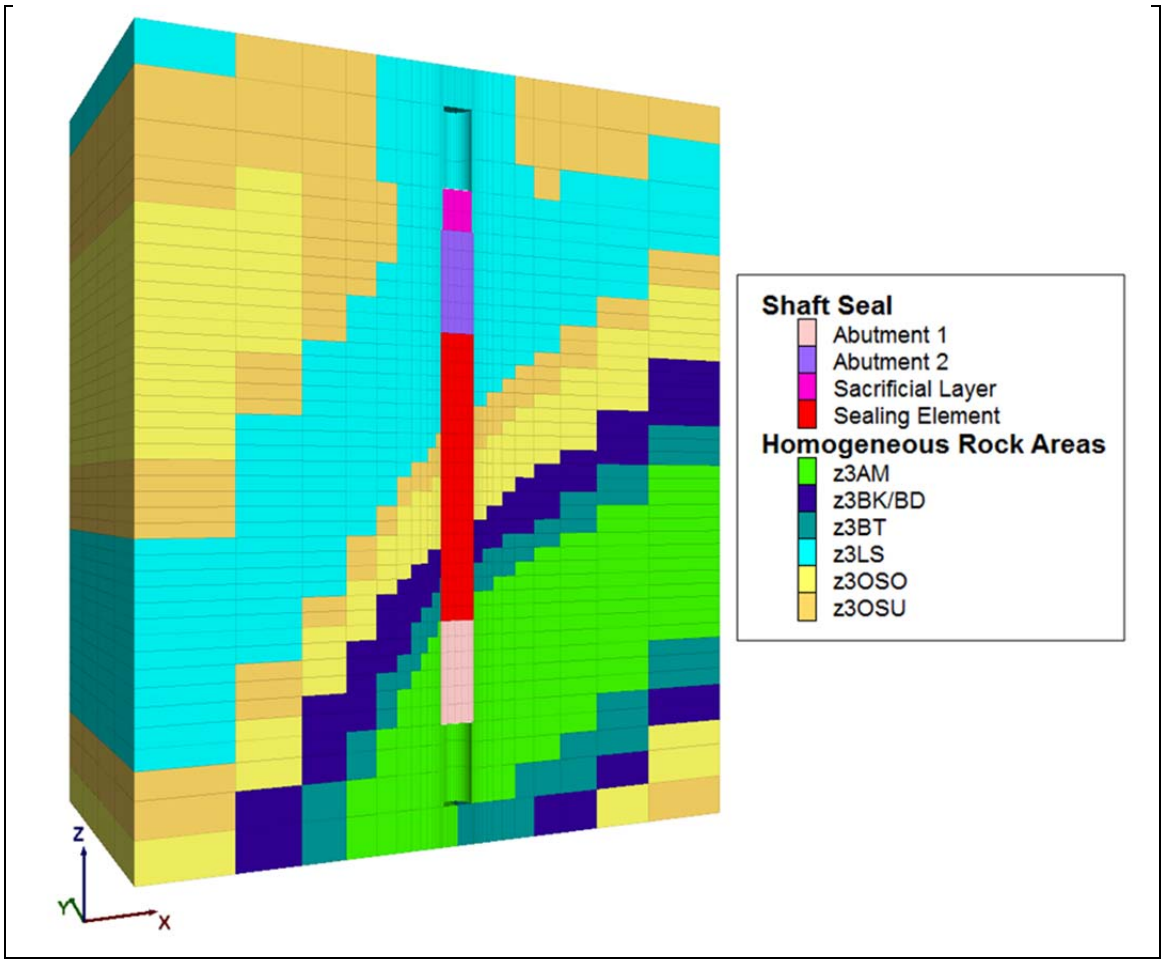

Figure C.3: Numerical model with homogeneous rock areas (intersecting plane y = 0)

The portion of the shaft sealing system implemented in the model consists of four elements: the sealing element itself, abutments above and below the sealing element, and a sacrificial layer. Figure C.3 also shows the numerical model of the shaft sealing system. The shaft above and beneath the sealing system is modelled as air-filled to simplify the model. The blocks at the top and bottom of the model allow a better application of the boundary conditions.

The details of the calculation procedure used in the analysis, such as the schedule of shaft sinking or the installation date of the sealing elements, are adopted from VSG [Müller-Hoeppe et al. 2012b]. Generally, the numerical simulation can be divided into two major calculation steps: a long-time creep calculation used to establish the initial conditions at the

onset of the earthquake event and a short-time dynamic calculation representing the earthquake event itself.

As the most significant effects on barrier performance are expected shortly after completion of the installation of the complete shaft sealing system, the earthquake calculation started 2 years (Time Point 1) and 20 years (Time Point 2) after completion.

C.2.2 Constitutive models and material properties

The mechanical deformation behavior of rock salt consists of two parts: elastic (reversible) deformations and viscoplastic (irreversible) deformations. The time-independent elastic behavior of salt is described using Hooke's linear-elastic constitutive law. BGRa law is used to characterize the creep behavior (i.e. viscoplastic behavior) of the rock salt by describing the secondary creep.

BGRa law is only used in the creep calculation to describe the viscoplastic part of the deformation behavior in the salt. The mechanical behavior of the rock salt during the earthquake is assumed to be elastic, because of the short loading times, the low velocity-amplitudes of the earthquake, and the associated low stress amplitudes.

The salt-concrete is assumed to be elastic over the entire calculation process. Dynamic values of the material properties were applied during the earthquake´s duration [Neubert 2015]. They are given in Table C.3.

Table C.3: Mechanical properties (dynamic values) of the materials used

Property		Leine-rock salt	Salt concrete Type Asse
Density ρ	[kg/m³]	2242.6	2074.0
Young's modulus E	[GPa]	37.4	31.0
Poisson's ratio v	[-]	0.17	0.25

C.2.3 Criteria of integrity and history locations

The geotechnical barrier is assumed to consist of the sealing element itself, the contact zone between sealing element and rock and the excavation-damaged-zone in the rock. A limit state to estimate a violation of the integrity is the fracturing of the rock (respectively the concrete) or the growth and opening of existing cracks.

Three criteria quantitatively listed in [Müller-Hoeppe 2012b] are used in the dynamic calculations to obtain information about the integrity of the geotechnical barrier during the earthquake:

- Cristescu / Hunsche dilatation boundary,
- fluid pressure criteria, and
- Drucker-Prager yield criterion.

The dilatancy boundary according to Cristescu/Hunsche [Cristescu & Hunsche 1998] describes a limit state where an increase of the permeability in the rock salt can be observed. The envelope of the dilatancy boundary for Leine-rock salt can be expressed as

$$F = 0 = -0.01697 \cdot \left(\frac{\sigma_0}{\sigma^*}\right)^2 + 0.8996 \cdot \frac{\sigma_0}{\sigma^*} - \frac{\tau_0}{\sigma^*} \qquad (2)$$

where σ_0 is the average stress and τ_0 is the octahedral shear stress. The results are shown as the utilization factor μ for dilatancy.

The dilatancy boundary represents an envelope beneath the short-time strength limit of the rock salt [Hunsche & Schulze 1994]. As long as the dilatancy criterion is met, the stability of the rock salt is also ensured.

The fluid pressure criterion defines a limit state where a crack opening, or grain boundary migration, under fluid pressure can occur [Müller-Hoeppe et al. 2012b]. The criterion can be described with the following formula:

$$\sigma_3 > p_{fl} \qquad (3)$$

The minimum principal stress σ_3 must be higher than an assumed pore pressure p_{fl}, which can occur when brine is present. The criterion is verified by recording σ_3 in the rock salt and salt-concrete during the dynamic calculation and a comparison with an assumed pore water pressure.

The stability and integrity of the salt-concrete is verified using the Drucker-Prager yield criterion [Müller-Hoeppe et al. 2012b]. The envelope of the criterion for salt concrete Type Asse can be described as:

$$F = 0 = \sigma_F - 1.86754 \cdot \sigma_0 - 2.7759 \qquad (4)$$

If the actual load is beneath the envelope described with Formula 4 then the stability of the salt-concrete sealing element is ensured. To obtain conclusions about the integrity (i.e. limitation of cracks) of the concrete, the envelope is reduced by a safety factor of 1.25.

The criteria were recorded at several locations during the creep calculation and the dynamic calculation. The right side of Fig. C.4 shows the general locations for the histories. Horizon 1 is at a depth of 720.0 m, Horizon 2 is at 695.0 m and Horizon 3 is at 685.5 m. Four points on each Horizon are considered. Point P1 is situated in the salt concrete with a distance of 0.3 m from the shaft contour. The other three points are situated in the rock salt at distances of 0.3 m, 0.6 m and 1.0 m from the shaft contour. The right side of Figure C.4 shows the grid

points and the associated zones (rectangles) where the histories are recorded in detail for Horizon 1.

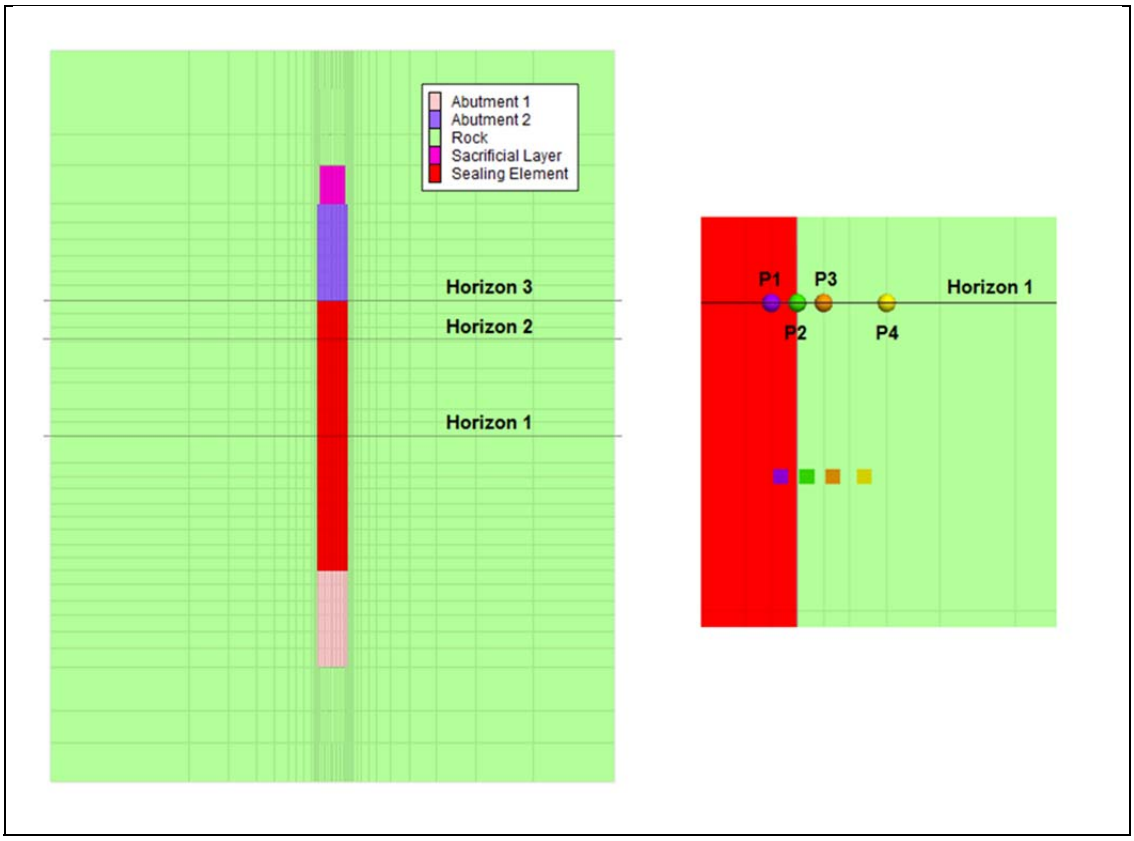

Figure C.4: Model for mechanical calculations of the shaft seal with history locations (see texte)

C.2.3 Initial and boundary conditions

First, a creep calculation is performed to establish the initial conditions for the dynamic calculations.

In the case of the dynamic calculation, special dynamic boundary conditions are used. On the lateral model boundaries, free-field boundaries as defined in the code FLAC^{3D} are applied. These boundaries are used to simulate the motion of the rock-masses surrounding the model area as forced by the earthquake. Waves induced by structures in the model can propagate straight through the boundaries and are not reflected. On the bottom and top of the model, quiet boundaries are used. These kind of dynamic boundary conditions provide a propagation of the waves through the artificial boundaries by “absorbing” the waves.

The earthquake loads are applied as stress-histories to the bottom of the model. The horizontal shear waves are simulated by applying the history as shear stresses in x- and y-direction. The compression waves appearing in the vertical direction are applied as a history for the normal stress in z-direction; reduced by 50 % to account for the maximum vertical acceleration a_v of the basic design earthquake.
Because Hooke's law is a very simple constitutive law, an additional damping is necessary to describe the energy loss of the earthquake waves during the propagation through the model. In the numerical calculation the implemented Rayleigh-Damping is used.

C.2.4 Calculation results

Figure C.5 shows the histories of the dilantancy utilization factor μ during the seismic loading phase for several history locations and the two points in time. The seismic loading primarily results in an increase of μ. The maximum increase of the utility factor of dilatation Δμ, compared to μ at the beginning of the dynamic calculation, is 0.064 at Point P2, Horizon 2 and Time Point 1. Generally, the increases in μ generated by the earthquake are very small and do not impact the integrity of the sealing element.

As shown in Figure C.5 the shapes of the time histories of μ differ between the points at each location and between the two points in time. The differences are the result of the diverse stress states in the observed zones.

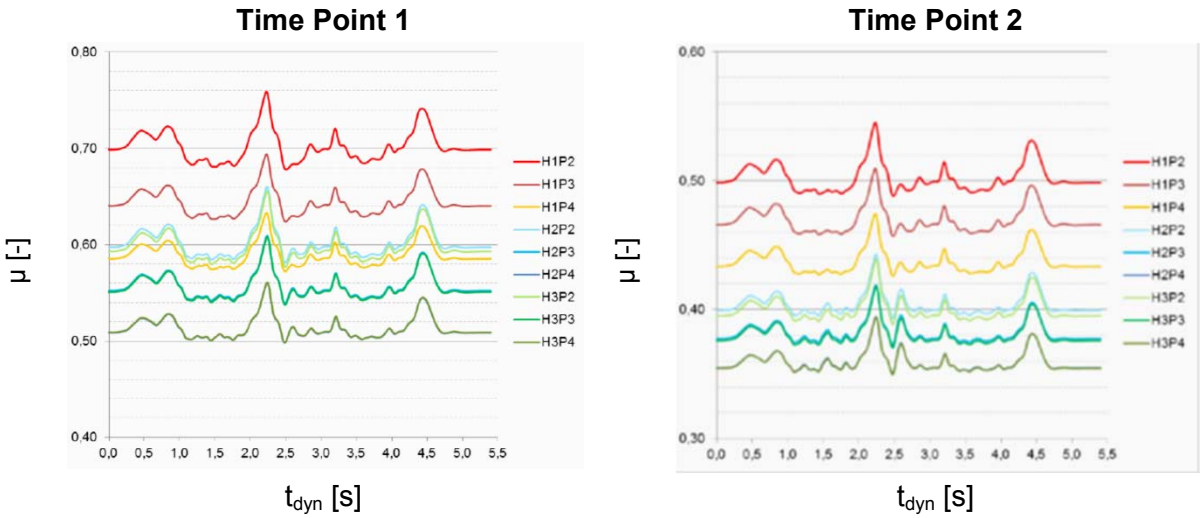

Figure C.5: Histories of the dilantancy utilization factor μ during the seismic loading phase for several history locations and two points in time

The histories for σ_3 are shown in Figure C.6. Only small fluctuations of σ_3 are observed resulting from the seismic loads. The maximal decrease $\Delta\sigma_3$, compared with σ_3 at the beginning of the dynamic calculation, is 0.88 MPa at Point P1, Horizon 1 and Time Point 1.

The assumed pore water pressures are adopted from the VSG [Müller-Hoeppe et al. 2012a]. If the shaft seal is intact, there is a maximum p_{fl} of 4.4 MPa at the salt concrete sealing element 1,740 years after completion. 20 years after completion of the shaft seal, σ_3 already has values of 1.0 MPa to 5.0 MPa and will continue to increase with time. An earthquake at a later time will not impact the sealing element's integrity.

For the case where a malfunction in the shaft sealing element occurs, a maximum pore water pressure p_{fl} of 1.0 MPa is assumed 55 years after completion. Under these conditions the sealing elements performance will also not be impacted because σ_3 will increase during this time and $\Delta\sigma_3$ is very small with no significant decreases during an earthquake occurring at a later time. It should be noted that the fluid pressure criterion in the form of Formula 3 is only valid for rock salt. For salt-concrete an influence of the tensile strength should be considered, whose static value is 0.97 MPa. Additionally, it has to be taken into account that brine cannot intrude much during the few seconds duration of an earthquake. Thus, exceeding the fluid pressure criterion due to earthquake loading can be neglected. Agreement with the dilatancy criterion is essential.

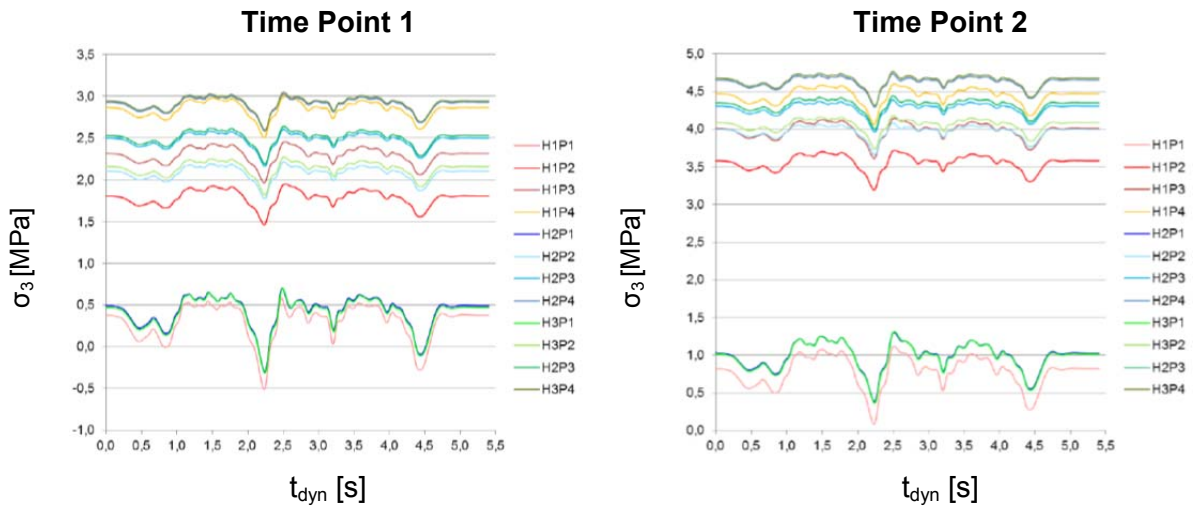

Figure C.6: Histories of the minimum principal stress σ_3 during earthquake loading

Additional evaluations show that the stresses in the sealing element for the different history locations and different points in time are well below the crack limitation boundary for the concrete [Neubert, 2015]. During the earthquake, they do not change significantly. In this case the performance of the salt concrete sealing element is preserved.

C.3 Conclusion

The results of the numerical simulations confirm the previous assumptions that an earthquake will only have a negligible influence on the performance of the sealing element and the integrity will not be impacted negatively. The results are only applicable to the geological and seismological conditions at the Gorleben site or other sites with similar geologic and seismological characteristics.

Although the calculation demonstrates that the performance of the sealing element is not negatively impacted during the seismic loading, there is still a need for additional investigations, especially of the contact zone.