Preliminary Safety Analysis of the Gorleben Site: Overview – 13298

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ABSTRACT

The project preliminary safety analysis of the Gorleben site started in 2010 and is based on the safety requirements for heat generating radioactive waste released from the German Federal Ministry for Environment, natural conservation and nuclear safety. The project consists of several tasks: the database defining the geology of Gorleben and the composition of the waste to be disposed of, the safety and demonstration concept, the repository concepts, the scenario analysis, the system analysis with longterm safety assessment and the synthesis. The overall synthesis indicates presently the compatibility of a repository in Gorleben with the safety requirements. The application of the method for a site selection process is still under evaluation.

INTRODUCTION

In Germany, the Gorleben salt dome has been discussed as a possible site for a repository for heat-generating radioactive waste since the 1970s.

The safety requirements governing the final disposal of heat-generating radioactive waste in Germany implemented by the federal ministry of environment, natural conservation and nuclear safety (BMU) from 2010 consider the fundamental objective to protect man and environment against the hazard from radioactive waste [1]. Unreasonable burdens and obligation for future generations shall be avoided. The main safety principles are concentration and inclusion of radioactive and other pollutants in a containment providing rock zone. Any release of radioactive nuclides may increase the risk for men and the environment only negligibly compared with natural radiation exposure. No intervention or maintenance work shall be required during the post-closure phase. The retrieval / recovering of the waste shall be possible.

Based on these requirements the tasks and objectives the project “Preliminary Safety Analysis for the Gorleben Site (Vorläufige Sicherheitsanalyse Gorleben, VSG) has been established to prepare a comprehensive safety analysis with focus on long-term safety and assess the results according to the safety requirements until March 2013. The objective was to compile and assess the available data on exploration of the Gorleben site and research on disposal in salt rock. An important objective is also to identify the needs for future R&D-work and further Gorleben site investigations. In addition the feasibility of the methodology for use with a future site selection procedure shall be assessed. The VSG does not provide a safety demonstration for a possible later licensing procedure, which is still required by the Atomic Energy Act.

The VSG is composed of four main working topics. About 80 scientists from 9 organisations contribute:

1. The safety assessment is based on the description of the geological site and its geological evolution over the next million years. The waste that could presumably be emplaced in a repository at the Gorleben site according to the current situation in Germany with its phase out of nuclear energy (June 2011) is compiled and classified. A safety and demonstration concept based on the safety requirements is developed for specific aspects of disposal in a salt dome.
2. Based on these data repository concepts are developed considering operational safety, long-term safety and retrieval / recovering of the waste. Several alternatives, such as storage in drifts or boreholes and different types of canisters, are projected. The retrieval of disposed waste from boreholes during the operational phase was projected for the first time.
3. The system analysis is based on these concepts. The features, events and processes are described, compiled and used to derive scenarios for the evolution of the systems and their probability. Geomechanical analyses show that integrity of the geological barrier (containment providing rock zone) can be demonstrated for 1 million years. This applies for external (e.g. glaciation) and for internal events and processes such as decay heat or gas generation. Similarly, the seals for shafts and drifts were designed and analysed as a concept. The radiological consequences were analysed by numerical models for the transport of the liquid and gas phase (two phase transport) in the long-term safety analysis.
4. The synthesis of the results is not finalized yet. The assessment of the capability of the repository system as containment system for radionuclides according to the safety requirements will be done. The uncertainties, which e.g. result from the partial geological exploration of the Gorleben site and additional R&D, will be shown [11].

The results and the current status of the project are detailed also in [2], [4], [6], [7], [8] and [9].

DATABASE

The geology of the Gorleben site is shown in detail in [2]. The Gorleben salt dome is 4 km wide and nearly 15 km long. It is composed of different salt rock types of the Zechstein (Upper Permian) series and extends to the Zechstein basis in a depth of more than 3 km. In the course of the salt dome formation the salt was moved several kilometers. During the uplift of the salt the initially plane-bedded strata of the Zechstein series were extensively folded. During this process anhydrite as a competent layer was broken to isolated blocks. In the core of the salt dome the “Hauptsalz” sequence, which is characterized by a particularly high creeping capacity, forms a homogeneous halite body with a volume of several cubic kilometers. The Hauptsalz contains gaseous and liquid hydrocarbons in separated zones of decimeter to meter dimensions. The overall hydrocarbon content is far below 0.01 %. At the flanks the salt dome consists of salt rocks with lower creeping capacities. Brine reservoirs with fluid volumes in the range of liters to hundreds of cubic meters exist in certain regions of this part of the salt dome. The water content of the Hauptsalz is below 0.02 %. Interconnected pores do not exist in the salt rock outside of fluid bearing or fractured areas, i.e. the salt rock is impermeable. The exploration of the Gorleben site as a potential site for a HLW-repository started in 1979 and is still in progress. Fig. 1 shows a simplified geological cross-section of the Gorleben salt dome.

Based on this site data a prognosis of the future evolution of the site was performed [3]. This concerned geological and climatic features, events and processes. The tectonic and volcanic activity, diapirism, subrosion, hydrology and climate is described and grouped into probable and less probable evolutions. Possible sequences of future glaciations were deduced from the history. This is shown in [3] in more detail.

The phase out from nuclear energy in Germany in 2011 has reduced the expected amount of radioactive waste. The heat generating radioactive waste will be composed of irradiated fuel elements from power reactors, vitrified waste from reprocessing and from irradiated fuel elements from research and prototype reactors.

As an option negligible heat generating waste was considered to assess the feasibility of the joint disposal in a separate area of the repository. The amount and composition of this waste was set hypothetical. The waste included depleted uranium tails from enrichment (about 35000 m^3), graphite (about 1000 m^3) and mixed waste (about 15000 m^3). This is shown in detail in [4].

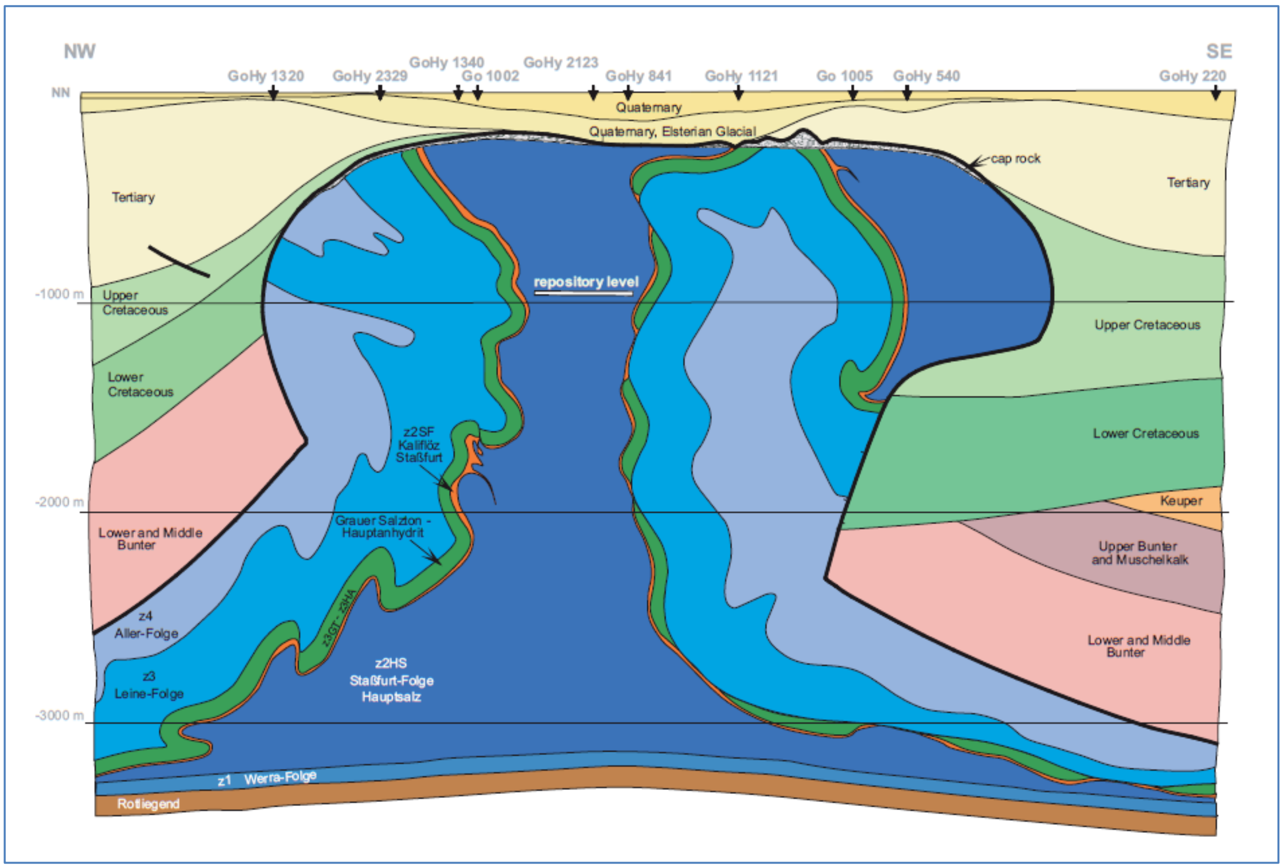

Fig. 1. Simplified geological cross-section of the Gorleben salt dome [1].

SAFETY CONCEPT

In due consideration of the German safety requirements [1] the safety concept for the project VSG is based on the following principles:

- the radioactive waste must be contained as far as possible in a containment providing rock zone (CPRZ), i.e. a part of the host-rock enclosing the repository jointly with geotechnical barriers
- the containment shall be effective immediately post-closure
- the containment must be provided by the repository system permanently and maintenance-free
- the intrusion of brine to the waste forms shall be prevented or limited

According to the safety requirements a site for disposal of heat generating radioactive waste is only suitable, if a sufficiently large containment providing rock zone is available, whose integrity is ensured for more than 1 million years and a robust, staggered, maintenance-free multi-barrier system from technical components (container, drift seal, shaft seal,…) can be developed, which prevents an unacceptable release of radionuclides in short and long term (Fig. 2). This principle is also applicable, if a barrier fails partially. The safe containment has to be demonstrated for probable and less probable evolutions of the site, while evolutions with very low probability (less than 1 % over the demonstration period of 1 Million years) need not to be considered. Criticality must be excluded in all phases of the repository development.

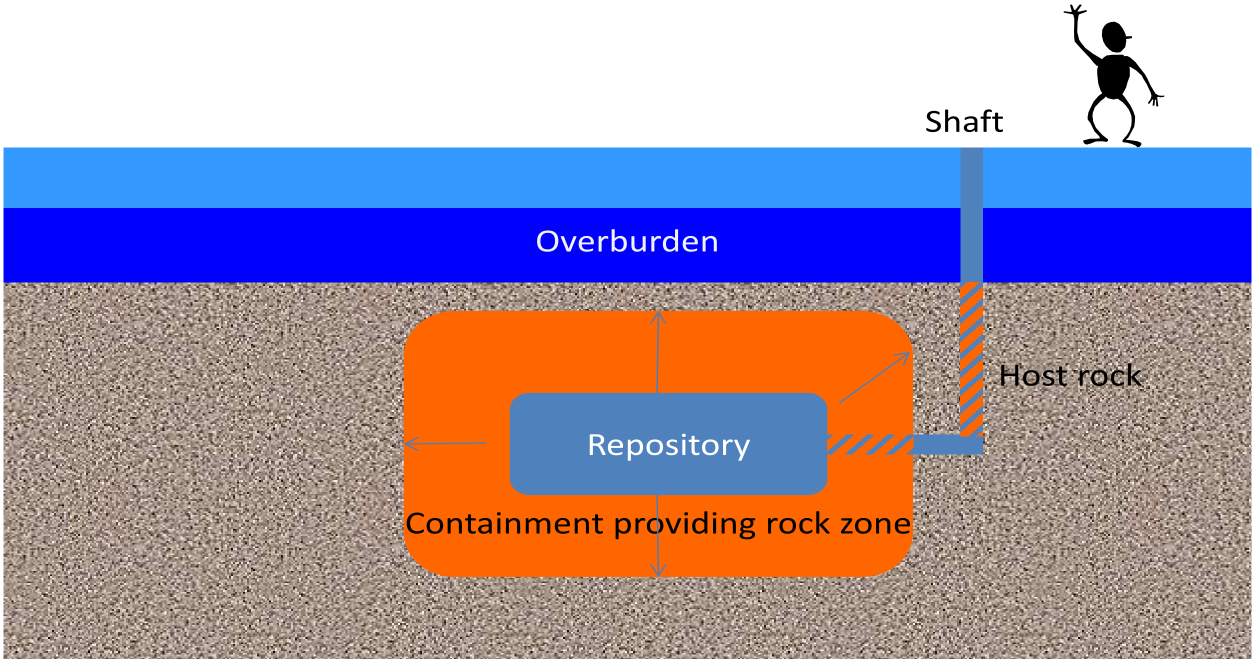

Fig. 2. Sketch of the repository with enclosing containment providing rock zone and geotechnical barriers (hatched).

Demonstration Concept

The demonstration concept shows how the safety of the system is evaluated. This includes the assessment of the probabilities and the consequences. It considers four categories applying an indicator (Radiological Hazard Index = RHI). This is shown in Fig. 3.

- The complete containment is provided, if there is no contact of the waste with solutions and no gaseous radionuclides are released from the containment providing rock zone (RHI = 0).
- The safe containment of radionuclides is achieved if the RHI is greater than 0 and is less than 1. A simplified procedure is sufficient as proof. The assessment distinguishes between a release by diffusion or advection.
- If the RHI is greater than 1 additional criteria have to meet. Additional criteria refer to the individual radiation dose and are related to the probability of the evolution of the system (scenarios). A detailed procedure is required.
- If these additional criteria are missed the repository is not feasible. The safe containment cannot be provided by the repository concept. The design of the repository has to be changed and assessed again. When all possible measures are optimized and the safe containment cannot be demonstrated, the site is not suitable.

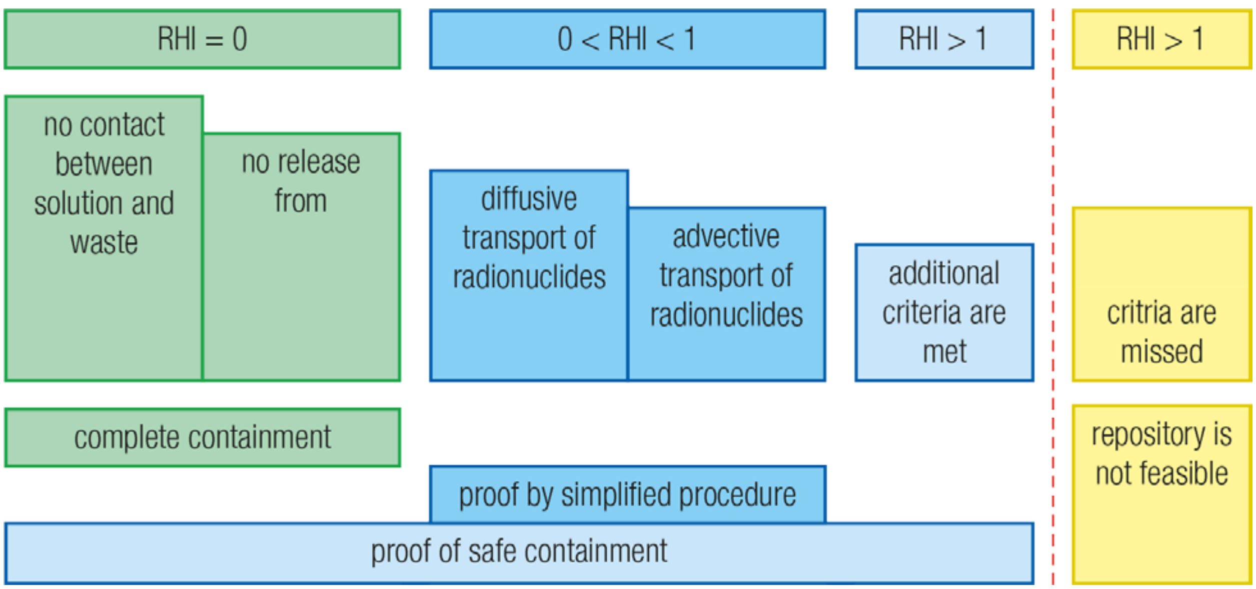

Fig. 3. Radiological Hazard Index (RHI).

Uncertainties

There are uncertainties for the Gorleben site presently, which cannot be reduced further or even eliminated. Therefore some assumptions have been made which should be verified in future. Main assumptions are:

- The lateral size of the salt dome is in accordance with the geological sketch of Bornemann [14].
- The known features from the salt rock currently under exploration can be extrapolated to the entire area necessary for the whole repository.
- The extension of the Hauptsalz is sufficiently large for all designed repository concepts including the required safety distance to adjacent rock layers.

Barriers are provided by geology and geotechnical measures. The main natural barrier is the geology with the Hauptsalz of the Staßfurt series (z2) which shall enclose the repository. The minimum size was set to 30 m. Additionally, 20 m were allowed for inaccuracies of detection and the excavation damaged zone. Therefore 50 m was defined as a safety distance to rocks other than z2 as a prerequisite for the repository design.

Geotechnical measures shall provide long- and short-term barriers. The long-term barrier is the backfill with salt grit. Its initial high porosity and permeability is reduced continuously by compaction. This is a time dependent process and re-establishes the features of the undisturbed rock salt within the life-time the short-term barriers.

The short-term barriers are drift and shaft seals. These are composed by layers of different material providing diversity and redundancy. The failure of a drift or shaft seal is regarded as a less probable scenario.

There are also uncertainties concerning the future evolution. These scenarios are grouped according to their probability. Improbable scenarios are not analysed.

Non-predictable evolutions such as the hydrogeology after a glaciation or human intrusion were assessed by stylized scenarios [12].

Additional uncertainties concern data, parameters and models. These are dealt within model calculations using bandwidths, probability distributions, simplifications and are assessed, if possible.

REPOSITORY CONCEPTS

The repository concepts for the Gorleben site are described in [7]. Fig. 4 and Fig. 5 depict repository concepts for two emplacement variants.

- Variant 1: Emplacement of heat-generating radioactive waste (spent fuel and vitrified waste) in self-shielding waste containers (POLLUX© casks) in horizontal drifts.
  As an alternative, the emplacement of heat-generating radioactive waste in transport und storage casks (CASTOR©) in horizontal boreholes was considered as well.
- Variant 2: Emplacement of heat-generating radioactive waste in multi-purpose cylindrical canisters in deep vertical boreholes.
- As an option, the emplacement of non-heat-generating radioactive waste was considered in horizontal emplacement chambers in a separate area of the repository.
- The layout was optimised to minimise the size of the repository but to comply with temperature criteria. The technical installation and casks / containers were selected to ensure manageability, radiation protection and operational safety. The disposal will be performed using a retreating system. Technical solutions for retrievability were developed using a tubing (liner) for disposal in boreholes. The overpack for fuel and flasks was modified. The closure system for drifts and shafts uses seals and backfill. The drift seals consist from sorel concrete. Salt grit with a higher moist content is used as backfill for the main drifts to enhance compaction while for the emplacement fields salt grit with natural moist content is foreseen. Subcriticality has to be demonstrated.

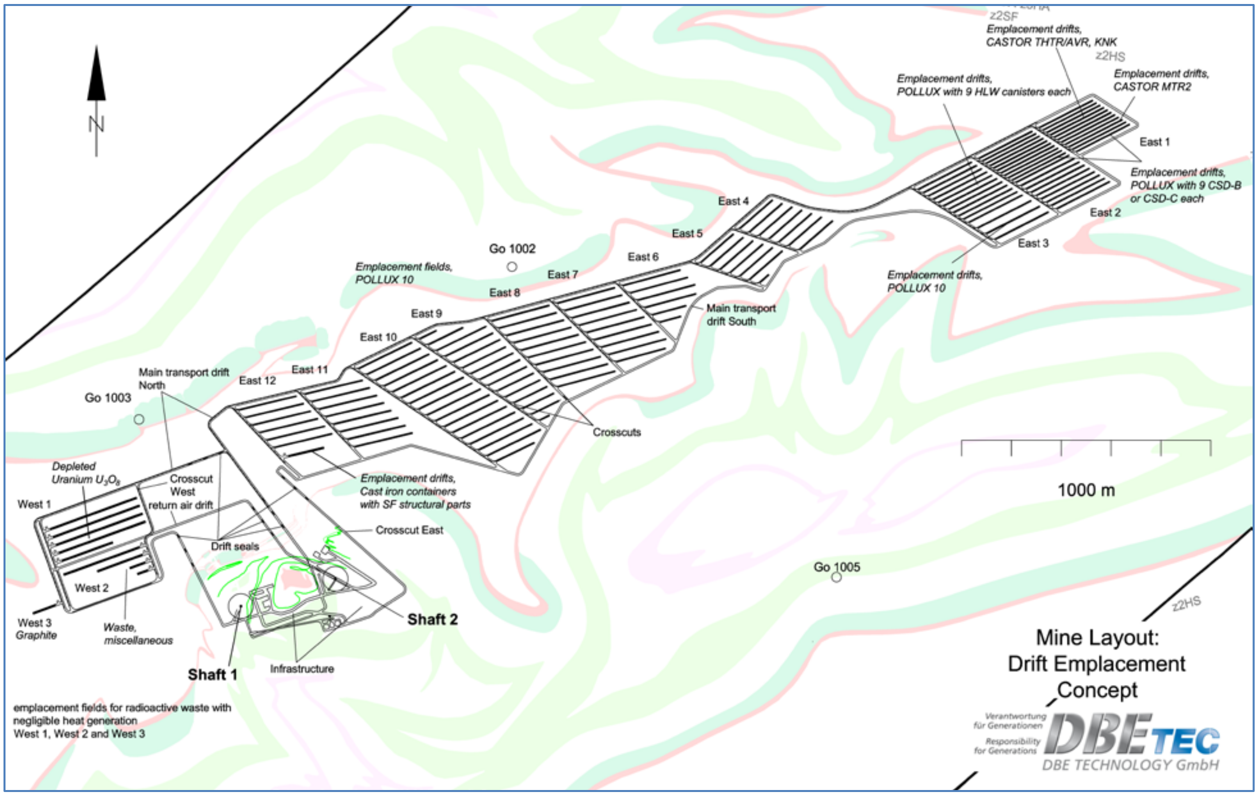

Fig. 4. Layout of the repository concept for drift disposal.

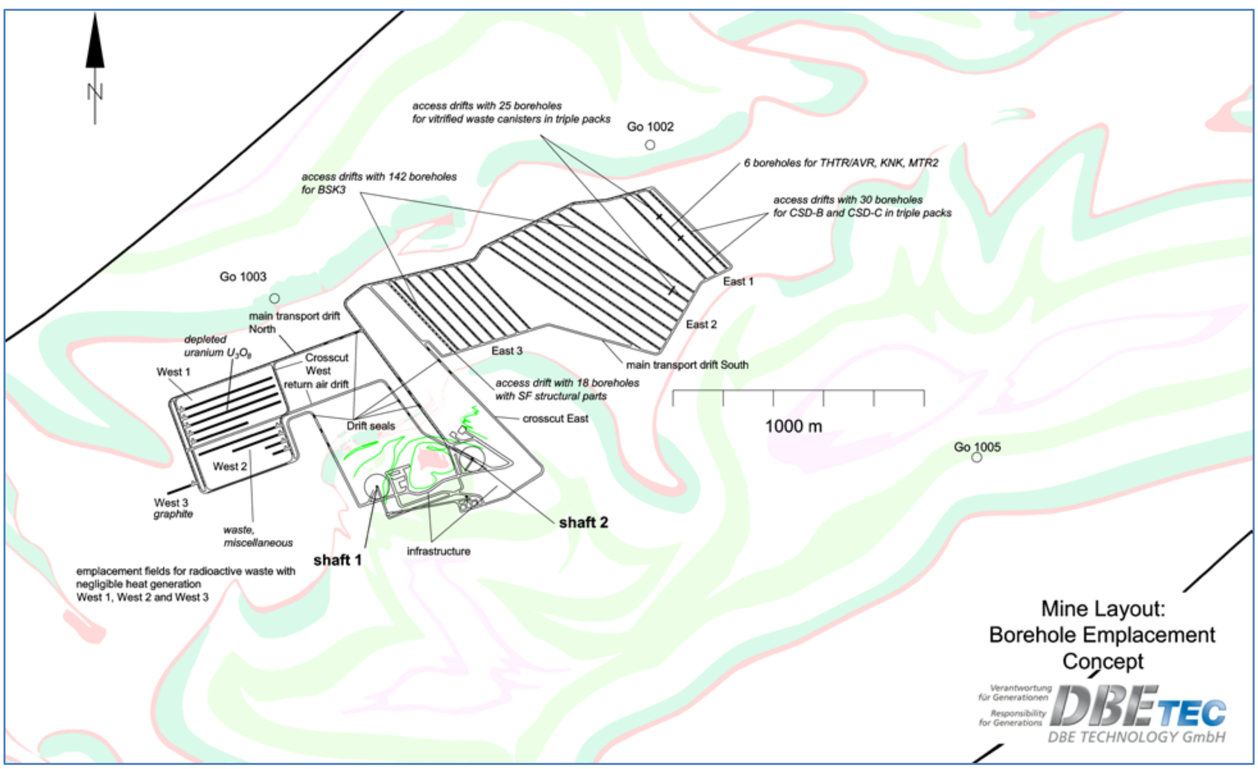

Fig. 5. Layout of the repository concept for borehole disposal.

SCENARIO ANALYSIS

The site and the repository system will undergo exactly one evolution, which will be governed both by climatic and geological processes at the site and processes induced by the repository construction and the emplacement of heat-generating waste. This evolution cannot be predicted in all details.

A novel scenario development methodology was developed in the project VSG. It aims at deriving one reference scenario for each repository design (horizontal drift/borehole emplacement) and a number of differing alternative scenarios. At large, the scenarios shall represent comprehensively the reasonable range of possible repository system evolutions. The methodology allows straightforwardly the assignment of probability classes to the scenarios according to the regulatory framework [1]. The individual scenarios are described by features, events, and processes (in short: FEP) that determine the future evolution of the final repository system at Gorleben. 115 FEP were listed in a site-specific FEP catalogue and combined as building blocks to scenarios. FEP may initiate or influence other FEP, be influenced by or resulting from other FEP. These interdependencies were used to derive scenarios systematically. The reference scenarios were derived from probable FEP and basic assumptions. The alternative scenarios were generated from violation of assumptions, from less probable FEP and from probable FEP with less probable parameter values.

Scenario development began directly from the guiding principles for deriving the safety concept:

1. A number of so-called “initial” barriers are identified that constitute a subset of all barriers acting in the repository system with immediate effect after closure. These “initial” barriers embrace the salt rock, the shaft seals, the drift seals, and spent fuel canisters. The common feature of these barriers is to prevent the contact of solutions with the emplaced waste immediately upon closure of the repository system. FEP that could impair the functionality of the initial barriers are the starting points for a scenario development.
2. Possible system evolutions releasing radionuclides from the waste form even without any contact of solutions were identified. Those FEP related with to the mobilization of radionuclides and their transport are also the starting points for a scenario development.

SYSTEM ANALYSIS

The system analysis addresses the general questions

- if the integrity of the geological salt barrier will remain intact under the expected loads like heat generation or glaciation,
- if there is any flow of brine to the emplacement areas,
- if possibly radionuclides are released from the containment providing rock zone, and
- if so, what radiological consequences have to be expected.

These questions have to be answered for all probable and less probable scenarios.

Based on these scenarios an analysis of integrity is performed. A demonstration of integrity is required for probable scenarios [1]. The integrity has to be checked for less probable scenarios and their radiological consequences have to be analysed. The final step is the assessment and synthesis of the results.

The results of the integrity analysis and radiological consequences are presented in [8] and [9].

The dilatancy and the fluid pressure are the main criteria to assess the integrity of the barrier rock salt. The dilatancy criterion specifies that no damage of the rock fabric must occur as a result of induced cracking and the interlinking of intercrystalline pore space in response to deviatoric stress. The damage process is associated with dilatancy, i.e. an increase in volume caused by the development of micro-cracks and crack accumulations.

The fluid pressure criterion specifies that the smallest formation stress in the barrier, plus any tensile strength which may be present, must be larger than the hypothetical fluid pressure by hydrostatic at the given depth. If this criterion is satisfied, fracturing of the host rock by fluid-pressure-driven penetration of fluids into the rock can be excluded.

The integrity of the salt barrier is only ensured if both criteria are satisfied in a sufficiently thick zone around the underground workings of the repository, so that linked flow paths from the water-bearing horizons in the overburden down to the emplacement zone and as well an release of hazardous substances from the repository itself (e.g. due to generation of a gas pressure) can be excluded from a geomechanical point of view.

Two different material models and parameter sets were applied to the geology of the Gorleben site to assess these criteria.

For the thermo-mechanical model calculation 2D and 3D models were used for the drift emplacement and the borehole emplacement concepts. The 2D models were used for detailed modelling of the proximal field in the emplacement zone, including the drifts and emplacement containers, as well as to investigate the thermal effects on the salt dome as a whole (far field). The 3D models were primarily used to model the spatial thermal effects in the far field, but did not take into consideration the underground workings in the repository. Because of the much coarser level of discretisation of the 3D model in the emplacement zone, instead of modelling the containers discretely, the generated heat was assumed to be homogenised over the emplacement zone.

The thermo-mechanical simulations carried out using a range of codes and material laws produced the following results and conclusions:

- The emplacement of heat-generating waste heats up the salt dome over a large volume, but the thermally-induced stresses and deformations do not give rise to any continuous migration paths.
- The highest thermo-mechanical stresses affecting the salt barrier occur within the first hundred years after sealing the geologic repository, so that any loss of integrity of the barrier becomes even less likely in the subsequent time period. Mechanical damage caused by exceeding the dilatancy limit affects only the rock zones directly adjacent to the underground cavities within a few decimeters up to 3 meters and rock zones, which are localised at the top salt zone. These rock zones at the salt top are of no importance with respect to the integrity of the salt barrier which provides the containment providing rock zone around the emplacement fields. The

calculations confirmed further that temporary local violations of integrity beginning at the top of the salt structure could penetrate down to approx. 130 m into the salt dome. However, they terminate several hundred meters above the emplacement fields and therefore still leave in place a thick intact zone of salt rock.

- The thermo-mechanical stresses calculated for the borehole emplacement design are higher than those calculated for the drift emplacement concept because the heat is released in a smaller area and a different field.

The integrity of the geotechnical barriers systems (drift and shaft seal) was demonstrated by numerical calculations concerning geological, thermal and geochemical impacts during their lifetime and by providing redundant and varying types of sealing systems in combination.

LONG TERM SAFETY ASSESSMENT

Two different numerical modelling tools, MARNIE and TOUGH2 respectively, were used to analyse the radiological consequences by one and two phase fluid flow calculations. Features and processes relevant for two phase flow in a repository in salt host rock were implemented in TOUGH2. These are compaction, consumption of water by metal corrosion, alteration of seals, gas infiltration and a variable temperature field. The definition of a source term was difficult for some radionuclides, such as C-14, due to a lack of data for chemical speciation and behaviour.

The layout of the repository concept for the drift disposal (Fig. 4) of POLLUX©-container was transferred into a 3D grid for TOUGH2 (Fig. 6) and a 1D for MARNIE, respectively (Fig. 7). This included some simplification steps due to some constraints of the codes.

A failure of a single seal did not result in advective flow of brine within or into the emplacement fields. Furthermore the salt grit is compacted in a relatively short time when applying parameters according to experimental data for modelling of the compaction process.

Reasonable but conservative assumptions were used for the final porosity of the salt grit after compaction. As a result any release of radionuclides was driven by diffusion only. No transport by advection took place. Radionuclides were released through the eastern drift seal. The release of radionuclides is higher there than in the western seal since it is closer to the disposal fields. The absolute total value is low.

The combined effect of compaction and metal corrosion with gas generation is non-linear, since these two processes have different rates. The effect is that C-14 can be released through a drift seal. No C-14 was released via the shaft seal.

Many assumptions for the MARNIE and TOUGH2 calculations were very conservative and should be improved by future R&D work. These are:

- the advective and diffusive transport of radionuclides at low salt grit porosities,
- the diffusion coefficient in high compacted salt grit,
- the solubility limits for some radionuclides,
- the release of radionuclides into salt grit and
- formation of gaseous radionuclides

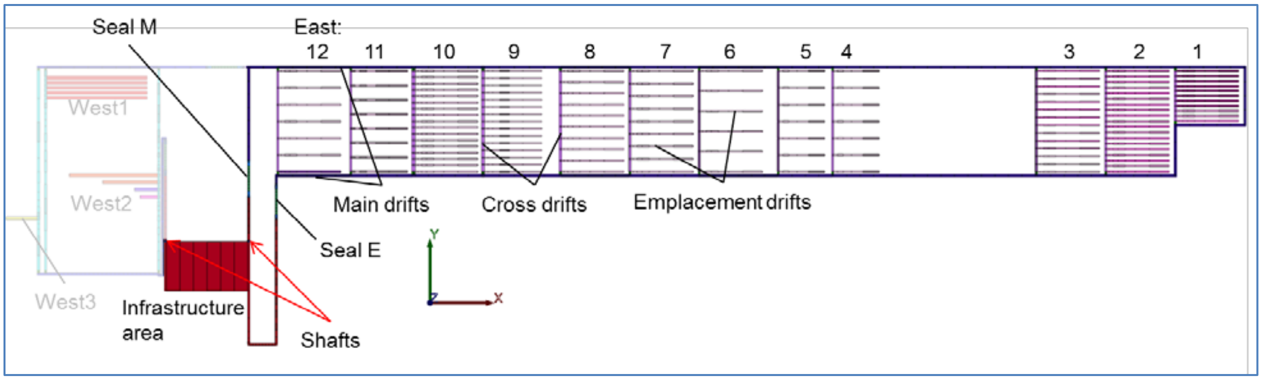

Fig. 6. Model structure used for the radionuclide transport calculations with TOUGH2 for the drift emplacement design.

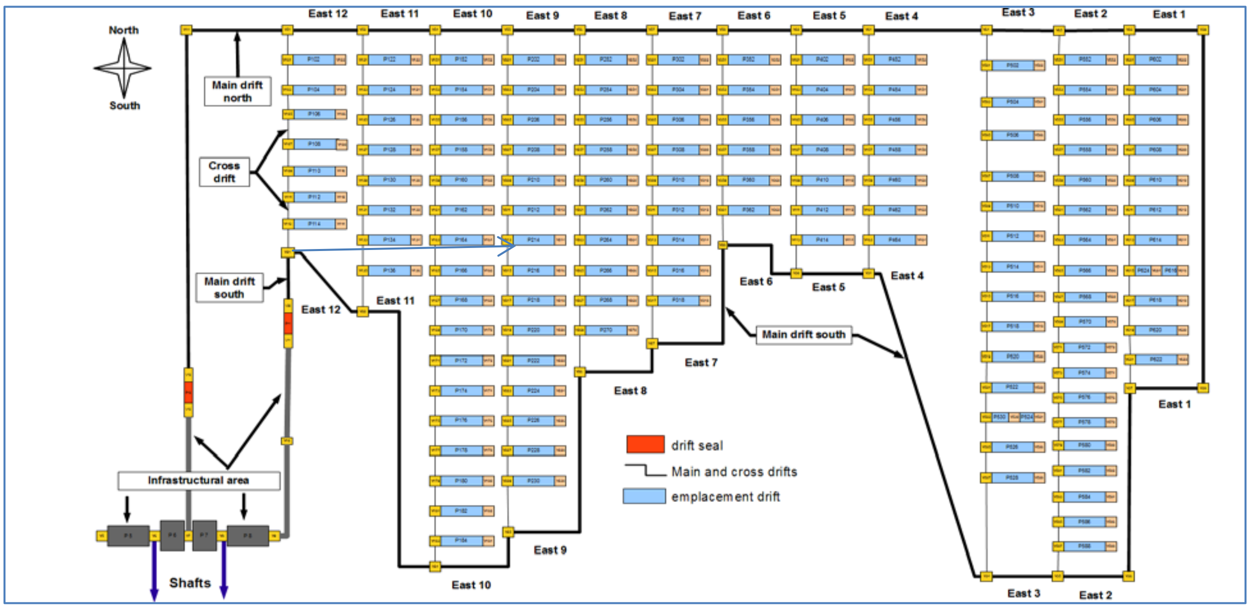

Fig. 7. Model structure used for the radionuclide transport calculations with MARNIE for the drift emplacement design.

CONCLUSION: SYNTHESIS OF THE PROJECT RESULTS

The goal of the project was initially set to assess the capability of a repository system in Gorleben to confine radionuclides and to identify needs for R&D. This assessment was based on the safety requirements of the BMU [1]. At a later stage the goal was added to assess the applicability of the results for a site selection process.

The safety and demonstration concept, which was generated during the course of the project, was suitable to verify the compatibility of a repository design with these safety requirements. The generated design of the repository system is feasible and complies with the safety requirements.

Nevertheless some assumptions were necessary. The assumptions refer to the status of geological exploration, the reliability of construction and some inherent uncertainties, which cannot further minimized presently. If these assumptions are met, the designed repository systems are assessed to be robust.

Optimizing strategies regarding the repository design are conceivable. A repository layout such as placing the structural components farther away from the drift seals would likely result in a lower C-14 flow through the drift seals. Furthermore, implementing a void volume as a sink (like an infrastructure area backfilled with gravel) might hinder gas flow through the shaft seals. The use of gas tight casks for the structural components (like Pollux®) could confine volatile radionuclides for decades or centuries.

Some lessons learnt are:

- The possible release of gaseous radionuclides, the two phase-flow processes and the subsequent model for radiation exposure require additional research and development.
- The containment providing rock zone can be minimized by an iterative process but has to be assessed preliminary as an initial guess.
- The handling of combinations of less probable but interdependent FEP has to be improved.
- Requirements concerning the mobilization of other pollutants and groundwater flow were missing.
- The complete but preliminary safety analysis for the Gorleben site identified important tasks for research and development. This would have not been possible by a generic safety analysis. A safety analysis should be repeated in time intervals and for different geological conditions.

Applicability to a Future Site Selection Process

During the course of the project the wish to establish a transparent, stepwise site selection process in Germany was growing in politics. Alternative sites should be identified and explored in addition to the Gorleben site. In the mid of 2012 a bill was introduced for a site selection process. This bill foresees site selection criteria and standards for site comparison, which should be developed for different steps of the process. Safety analyses are foreseen to evaluate the results of surface and subsurface explorations of possible sites.

The objectives of the project VSG were extended to evaluate if the methodology used in VSG can be adapted for safety analyses for other disposal systems in other host rock formations like clay stone, since these will be carried out during a site selection process. If so, necessary modifications should be pointed out. In detail the following should be analysed:

- Adaptability of safety targets, measures and strategies of the safety and demonstration concept developed in VSG to disposal systems.
- Applicability of the management strategy for uncertainties, e.g. related to site data or safety relevant processes, to safety analyses of a future site selection procedure.
- Evaluation, which key parameters and processes could be used to derive principles and criteria for a site selection process.
- Definition of technical requirements needed for radionuclide transport calculation codes of performance assessments for a site selection process e.g. simulation of two-phase flow etc.

- Adaptability of technical repository design concepts developed in the VSG project to other geological situations.
- Identification of R&D needs for criteria for evaluation and decision within a site selection process.

The release of the VSG synthesis report is scheduled for April 2013 [11].

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