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Key Topics in Deep Geological Disposal: Conference Report, Köln 2014

Deutsche Arbeitsgemeinschaft für Endlagerforschung (DAEF)

Edited by Susanne Fanghänel

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Background

Safe disposal of radioactive waste and spent nuclear fuel is considered to be a major challenge for present and following generations irrespective of the current and future scenarios for the use of nuclear power in different countries.

International efforts are underway towards the implementation of repositories notably for highly radioactive waste: 58 countries signed the Joint Convention of September 1997 on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management, committing themselves “to take the appropriate steps to aim to avoid imposing undue burdens on future generations”. In July 2011, the EU published their Council directive 2011/70/Euratom forcing EU member states to establish programs and schedules for the safe disposal of radioactive waste. The European technology platform IGD-TP (Implementing Geological Disposal of Radioactive Waste Technology Platform) clearly expresses its vision “that by 2025 the first geological disposal facilities for spent fuel, high level waste, and other long-lived radioactive waste will be operating safely in Europe”. While some geological disposal programs in countries such as Finland, Sweden, France and Switzerland are quite advanced, other states decided to reconsider and re-start their projects (e.g. UK, USA and Germany). Yet, the development of a widely accepted concept for repository development which combines all relevant sociotechnical criteria within a stepwise implementation approach remains a challenge for interdisciplinary research.

Considering the time scales of many decades required to implement a repository from conceptualization via initiating the site selection process to repository closure, it is obvious that science and technologies related to nuclear waste disposal have to be developed further.

The international conference “Key topics in deep geological disposal” did set a focus on the following topics:

1. **Repository concepts in different host rocks and safety analyses**
2. **A. Governance und public involvement**
   B. Sociotechnical challenges and interdisciplinarity
3. **Safety aspects of repository operation**
4. **Construction of technical barriers**
5. **Scientific aspects of the nuclear waste disposal safety case**
6. **Siting Strategies**

The conference provided an adequate forum for fruitful scientific exchange and a valuable instrument for further improving multilateral co-operation for mutual benefit. The program consisted of invited and contributed presentations.

This conference report is a compilation of extended abstracts and also short abstracts. It will give an overview of the conference contents.

We would like to thank all speakers, authors of posters and participants who have contributed to the success of this conference.

Conference organizers

The German association for repository research (DAEF) represents leading research organizations active in radioactive waste disposal research. The aim of this association is to contribute to the safe disposal of radioactive waste, to develop research and to offer respective fact based information

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SESSION 1 REPOSITORY CONCEPTS IN DIFFERENT HOST ROCKS

Cigéo Project: The French Deep Geological Repository Project in Clay Host Rock

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The French classification of waste is defined by the National management plan of radioactive materials and waste updated every three years. High level waste (HLW) and intermediate level long-lived waste (ILW) are planned to be disposed of in a deep geological repository subject to authorization (Cigéo Project). By law deep geological disposal is dedicated to end waste that cannot be disposed of at the surface or at low depth with regard to safety.

HLW consists in vitrified fission products and minor actinides from spent fuel reprocessing. Intermediate level long-lived waste arises mainly from spent fuel reprocessing and from the maintenance and operation of reprocessing plants. The volume of existing waste was 40,000m^3 of ILW and 2,700 m^3 of HLW at the end of 2010. The total prospective volume used for the repository design is estimated at 70,000m^3 of ILW and 10,000m^3 of HLW.

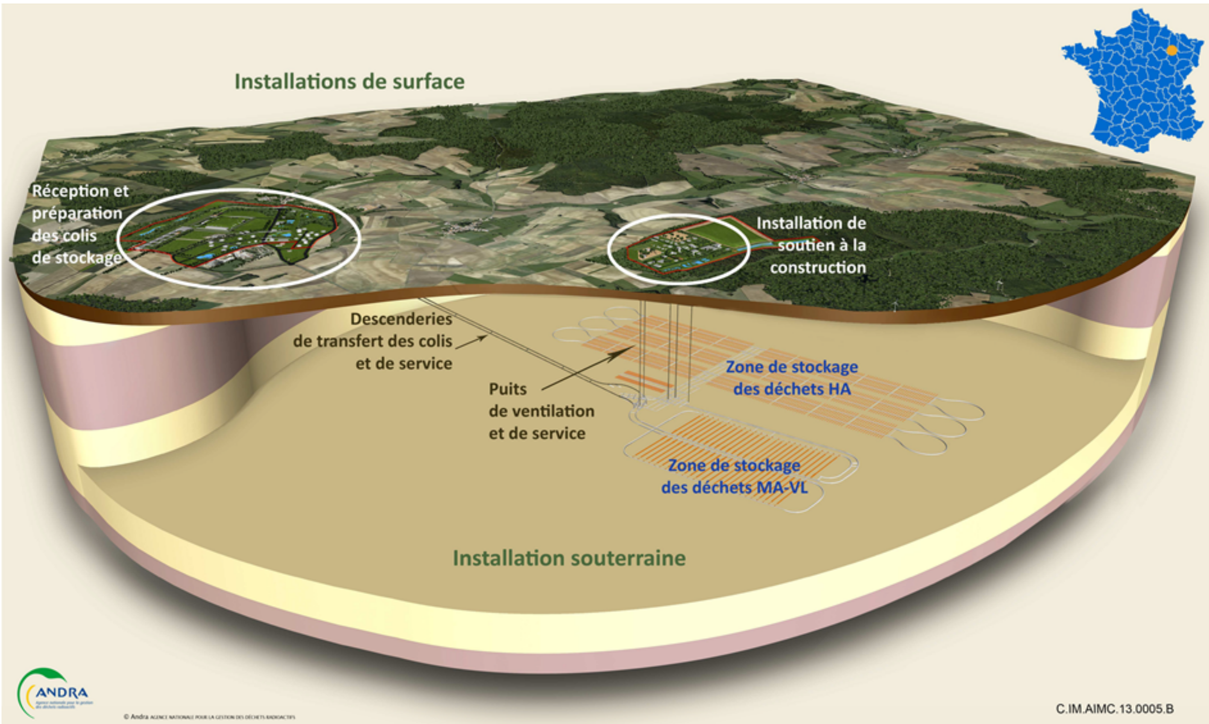

Fig. 1. Overall view of the disposal facility

In 2011, the results of 20 years of R&D made it possible to issue detailed project technical requirements. On this basis Andra launched the industrial design of the future facilities of Cigéo Project. A public debate was held on the project during the year 2013. To take into account the recommendations of the public debate, some modifications have been proposed for the future development of the project: the preparation of the license application will follow a phased evaluation process between 2015 and 2017 and a pilot industrial phase is introduced from 2025 to verify in real conditions at least the operational safety, the retrievability and the monitoring, first without real waste and then with real waste up to around 2035. A law defining the reversibility conditions should also be voted by the Parliament before Andra provides the license application.

If licensed the disposal facility will be implemented at a depth of 500 m, in the Callovo-Oxfordian clay layer (argillites) which has been investigated in the Meuse/Haute Marne Underground research laboratory (URL). The underground facility will be located in a 30 km^2 defined area close to the URL. This facility is planned to be constructed progressively along with waste delivery, over a period of about one century.

It will include dedicated disposal zones for IL waste and HL waste. Access to the underground facility will be provided by vertical shafts (ventilation, mucking...) and an incline ramp for waste transfer. Two service areas are planned on the surface: one of them located at the entrance of the ramp will be dedicated to receiving, control and preparation of waste packages; the other, located vertically above the underground facility, will support construction.

Disposal cells for IL waste consist of horizontal tunnels with a 500 m length. The cross section of the disposal tunnels makes it possible to stack waste disposal packages. Waste handling will be carried out remotely because of the dose rate within the disposal tunnels.

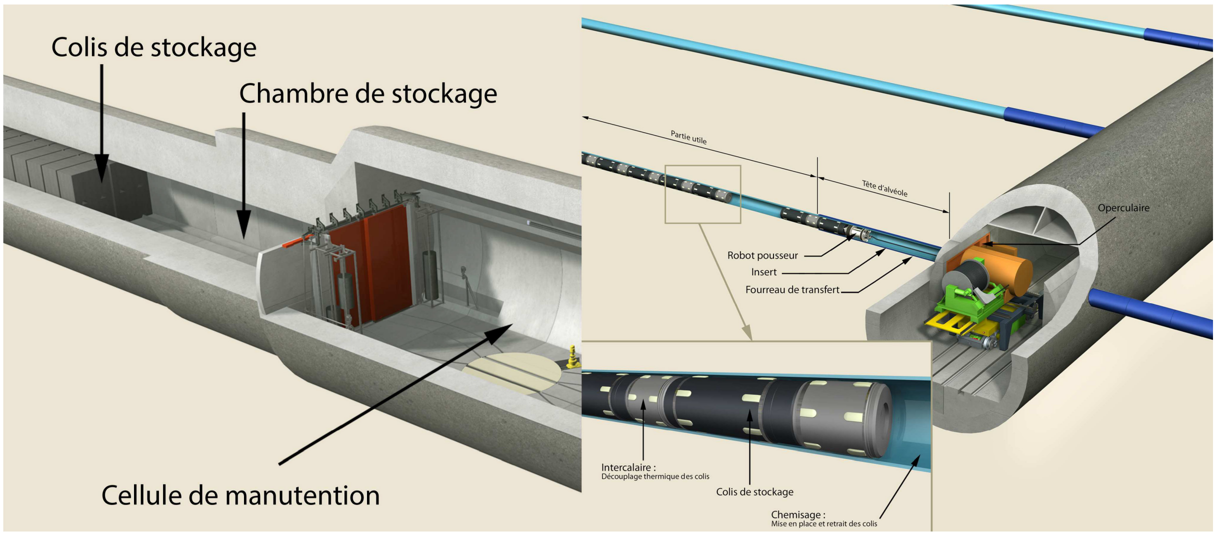

Fig 2. Disposal cells for ILW(left) and HLW (right)

The depth of the disposal facility, the absence of exceptional natural resources and the favourable geodynamic context provide for the isolation of waste during very long periods of time.

The long-term containment is essentially based on the favourable properties of the host clay layer. It has been studied at the scale of a million years. Post closure safety functions are: (i) oppose groundwater flow; (ii) limit the release of radionuclides and immobilize them within repository; (iii) delay and mitigate the migration of radionuclides. The main geological properties governing these safety functions are: geodynamic stability, geological continuity and thickness of the clay layer; low permeability and retardation capability of the argillite; low hydraulic gradients in the considered area, etc. A number of experiments were carried out in the URL to assess these properties at pertinent scales. The design of the underground facility aims at the preservation of these favourable properties in particular with regard to mechanical and chemical impacts. Underground experiments have being carried out specifically to investigate the potential interactions between repository and host rock.

The underground architecture of the repository contributes to its long term performance and robustness (implementation of disposal drifts in the middle part of the clay layer, location of access shafts and ramps, configuration of disposal zones, etc.).

After operation, disposal cells, access drifts and shafts will be sealed and backfilled to fulfil post closure safety functions. Significant work has been done to develop and demonstrate the capability to seal the repository.

Safety studies have included a number of iterations since the first French waste law in 1991 to progressively optimize the conceptual design with regard to safety. The safety case provided by Andra in 2005 showed the scientific feasibility and the long term safety of a deep disposal facility in the clay layer investigated in the URL.

Along with safety, reversibility of deep geological disposal is a requirement of the 2006 waste law. Provisions are included in the design to enhance the retrievability of disposed waste packages. The implementation, the operation and the closure of the disposal facility are conceived as a stepwise and progressive process. Intermediate milestones will be identified until final closure to adapt as necessary the conduct of operation and closure.

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Preliminary Safety Analysis for the Gorleben Site

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Abstract: *A methodological approach for long-term safety analyses was applied in preparing the project "Preliminary Safety Analysis of the Gorleben Site" based on the German requirements for final disposal of heat generating radioactive waste. The approach included the description of the geological site and its future evolution coupled with a range of possible waste emplacement scenarios using site specific repository designs. The repository designs were developed to provide operational safety, long-term safety and retrieval / recovering of the waste to comply with the German requirements (BMU 2010). A safety concept for accomplishment of the radiological safety and its assessment was developed. Analysis of the integrity, fluid flows and radiological consequences were done and the performance of the system was assessed. If the present assumptions are confirmed by future research and development, the designed repository system for drift disposal was assessed to be robust. Further optimization of the present drift disposal design is possible in relation to gaseous radionuclide species and in relation to the technology of retrieval for the of borehole disposal design. This site specific safety analysis identified important tasks for research and development. The methodology and safety concept can be applied to other salt rock sites and can be transferred partially to clay sites.*

1 Introduction

The Gorleben salt dome has been discussed as a possible site for a repository for heat-generating radioactive waste in Germany since the 1970s. A consortium consisting of nine German organizations lead by GRS was contracted by the Federal Ministry of Environment, Nature Conservation and Reactor Safety to perform a Preliminary Safety Analysis of the Gorleben site. The objective of the project "Preliminary Safety Analysis of the Gorleben Site (VSG)" was to assess if repository designs at the Gorleben site or other salt rock sites with a comparable geology could comply with 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) in 2010 (BMU, 2010) based on currently available knowledge (Fischer-Appelt et al., 2013), (Bracke et al., 2013). In addition an assessment was made of whether the methodological approach could be used for a future site selection procedure and which technological and conceptual considerations could be transferred to other geological situations.

The objective included the compilation and review of the available exploration data of the Gorleben site and on disposal in salt rock, drafting of repository designs and the identification of the need for future R&D-work and further site investigations.

2 Project Structure

The VSG was composed of four main working sections (Fig. 1):

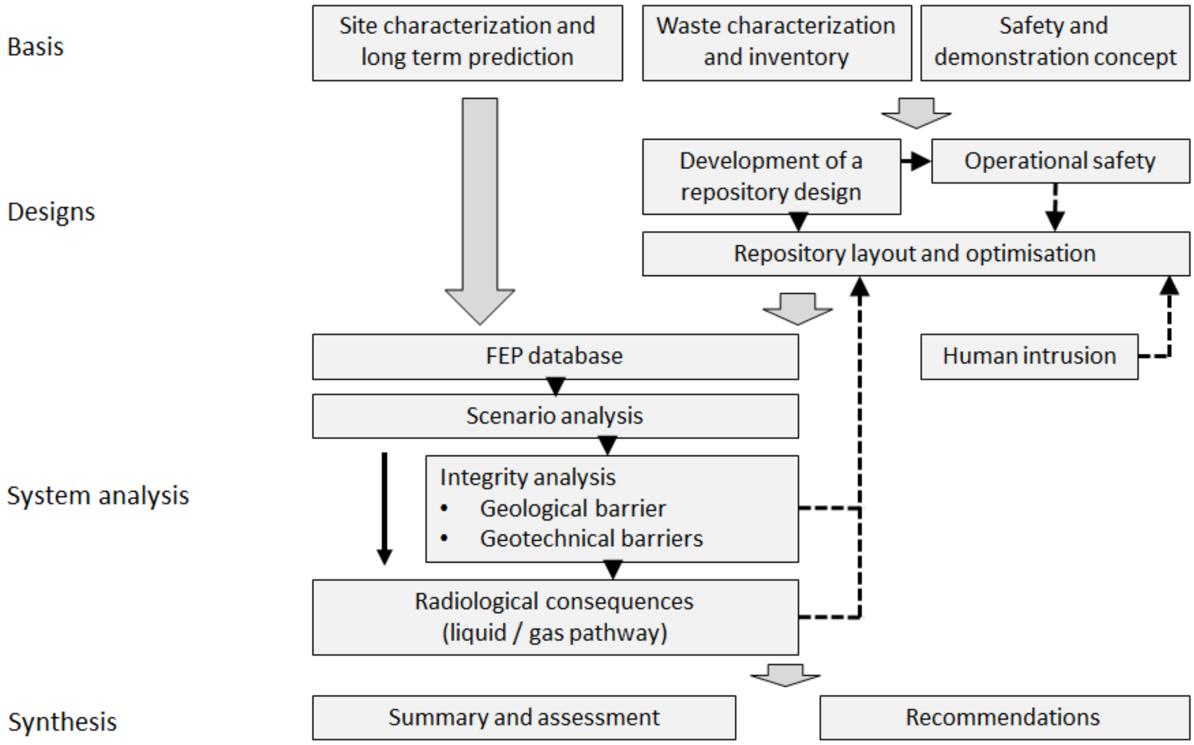

Fig. 1. VSG project structure

1. Basis: This topic included the geological description of the site and its future evolution over the next one million years as stipulated in the safety requirements (BMU, 2010); further, a description of the inventory of the waste that could be emplaced in a repository at the Gorleben site arising from the current situation in Germany, with its phase-out of nuclear energy (June 2011), and, finally, generation of a concept to accomplish radiological safety and to demonstrate its compliance to the BMU safety requirements.

2. Designs: Repository designs: Based on these fundamentals, repository designs were developed to provide operational safety, long-term safety and the ability of retrieval / recovering of the waste. Two emplacement variants (emplacement of spent fuels in drifts or in boreholes) and three main types of canisters (POLLUX©, CASTOR©, BSK3R) and one optional variant for additional emplacement for non-heat-generating waste were considered.

3. System analysis: The long-term safety analysis of the repository system was based on these designs. The main features, events and processes of the repository were compiled and described. These were used to derive scenarios and to assess the probability of the possible evolutions of the system. The integrity of the geological barrier (containment providing rock zone) was investigated by geomechanical analyses for one million years considering external and internal events and processes such as erosion, glaciation, decay heat or gas generation. Similarly, the performance of shaft and drift seals was analyzed. The radiological consequences were calculated by numerical models for the transport of the liquid and gas phase (two phase transport).

4. Synthesis: The compliance of the repository system and the concept to accomplish radiological safety with the safety requirements were assessed within the project synthesis report. Existing uncertainties (e.g. due to the incomplete geological exploration of the Gorleben site) and additional R&D needs were systematically identified and compiled (Fischer-Appelt et al., 2013). The methodological approach was discussed for its suitability to compare repository sites and its technical transferability to repository sites in other geological formations.

3 Results

3.1. Evolution prognosis

Based on the geoscientific site data a prognosis of the future evolution of the site was performed (Mrugalla, 2011). Geological and climatic features, events and processes were considered. Future tectonic activity, diapirism, subrosion, hydrology and climate were described and grouped into probable and less probable evolutions. E. g. possible sequences of future glaciations were deduced from geological history.

3.2 Waste

The heat generating radioactive waste will be composed mainly of spent fuel elements from power reactors, reprocessed waste and irradiated fuel elements from research and prototype reactors. As an option negligible heat generating waste was also considered to be disposed to assess the feasibility of the joint disposal in a separate area of the repository. A hypothetical amount and composition of this waste was assumed. It included depleted uranium tails from enrichment (about 35,000 m^3), graphite (about 1,000 m^3) and mixed waste (about 15,000 m^3) (Peiffer et al., 2011).

3.3 Safety Concept

In response to the German safety requirements (Bundesministerium für Umwelt, 2010) the safety concept for the project VSG was based on the following principles (Mönig et al., 2012):

- The radioactive waste must be contained in a containment providing rock zone (CPRZ), i.e. part of the host-rock enclosing the repository together with geotechnical barriers,
- The containment shall be effective immediately after 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.

3.4 Repository and Barrier design

The repository designs for the Gorleben site have been developed for three emplacement variants (Bollingerfehr et al., 2012) with respect of operational safety (Peiffer and McStocker, 2012). Measures to mitigate the impact of accidential and considering human intrusion were investigated and evaluated (Beuth et al., 2012a) according to the German Safety Requirements. The proposed measures did not significantly change the repository design.

- Variant A: As an option – emplacement of non-heat-generating radioactive waste in a separate area of the repository. This variant was combined with the following variants for spent fuel and high level waste (Fig. 2).
- Variant B: Emplacement of heat-generating radioactive waste (spent fuel and vitrified waste) in self-shielding waste containers (POLLUX© casks) in horizontal drifts (Fig. 2). As an alternative, the emplacement of heat-generating radioactive waste in transport and storage casks (CASTOR©) in short horizontal boreholes was considered although this required changes to the design of shafts and underground transportation systems.
- Variant C: Emplacement of heat-generating radioactive waste in multi-purpose conical overpacks in deep vertical boreholes.

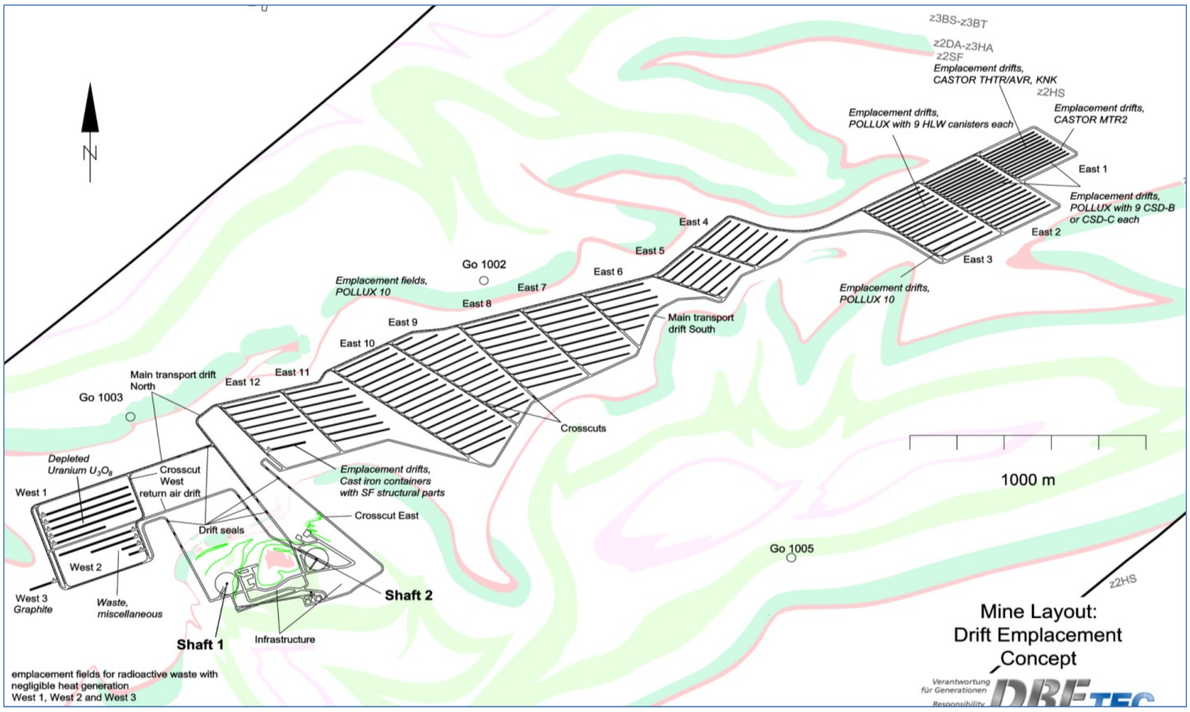

Fig. 2. Repository design for emplacement of heat-generating radioactive waste (spent fuel and vitrified waste) in self-shielding waste containers and non-heat-generating waste in horizontal drifts

The disposal was planned to allow retrieval during the operational phase. The technical solutions for retrieval of overpacks in variant C were projected for the first time. A steel tubing (liner) of 300 m length was proposed for disposal of the conical overpacks in boreholes. The void space in the liners was planned to be filled with dry quartz sand. The conical shape of the overpacks for spent fuel and flasks were designed to facilitate retrieval using vibration.

Barriers are provided by geology and geotechnical measures. The main natural barrier is the geology i.e. the Hauptsalz of the Staßfurt series (z2) which shall enclose the repository. The minimum thickness was set to 50 m as planning criteria for the containment providing rock zone. This safety distance from the repository to rocks other than z2 as a prerequisite for the repository design shall include uncertainties on the detection of geological layers, of cracks and fissures and on the extent of the EDZ (excavated disturbed zone) to ensure the integrity of the geological barrier in the containment providing rock zone (Mönig et al., 2012).

Geotechnical measures shall provide long- and short-term barriers. The long-term barrier is the salt grit backfill. Its initial high porosity and permeability is reduced continuously by compaction which is caused by salt creep. This is a time dependent process and transforms the features of the salt girt to the features of undisturbed rock salt within a few thousand years. This is the required minimum life-time of the short-term barriers.

The short-term barriers are containers, drift and shaft seals. The main material for drift seals uses is sorel concrete. The shaft seals are composed of bentonite, salt concrete and sorel concrete. The failure of a drift or shaft seal is regarded as a less probable scenario within its technical lifetime. In addition the infrastructural area located between shafts and drift seals was planned to be filled by non-compactable crystalline rock gravel, providing a large pore space storage for any solution seeping through the shaft seals into the repository.

3.5 System Analysis

A novel **scenario development** methodology was adopted in the project VSG (Beuth et al., 2012b). Based on an updated FEP database with interdependencies, (Wolf et al., 2012b), which provides one reference scenario for each repository design (horizontal drift / borehole emplacement) and a number of alternative scenarios. The scenarios comprehensively cover the range of possible repository system evolutions. The methodology allows the assignment of probability classes to the scenarios based on the regulatory framework (BMU, 2010). The individual scenarios are described by features, events, and processes (in short: FEP) that determine the future evolution of the final repository system. FEP may initiate or influence other FEP, be influenced by or resulting from other FEP (Wolf et al., 2012a, b). 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.

Based on these scenarios an **analysis of geomechanical and geotechnical integrity** was performed (Kock et al., 2012), (Müller-Hoeppe et al., 2012).

The dilatancy criterion and the fluid pressure criterion are the main criteria to assess the geomechanical integrity of the rock salt barrier. The integrity of the salt barrier is only ensured if both criteria are satisfied in a sufficiently large zone (i.e. CPRZ) around the repository. In this case, linked flow paths from the water-bearing horizons in the overburden down to the emplacement zone, as well as release of hazardous substances from the repository itself (e.g. due to generation of a gas pressure) can then be excluded from a geomechanical point of view. The mechanical and 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 generate any continuous migration paths.
- The highest thermo-mechanical stresses affecting the salt barrier occur within the first hundred years after sealing the geologic repository. Any loss of integrity of the barrier becomes much less likely in the subsequent time period. Mechanical damage caused by exceeding the dilatancy limit was restricted to the rock zones directly adjacent to the underground cavities within a few decimeters up to 3 meters (EDZ) and rock zones at the distant top of the salt dome. These rock zones at the salt top are of no importance with respect to the integrity of the salt barrier some

hundred meters below the salt dome top which constitutes the containment providing rock zone around the emplacement fields.

- 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 and differently shaped volume.
- The integrity of the geotechnical barriers (drift and shaft seals) for their designed life-time 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.

For the **radiological consequence analysis** (Larue et al., 2013) the radionuclide transport was modelled using a 2-phase fluid flow analysis with TOUGH2 (Pruess et al., 1999), and one phase analysis with MARNIE (Martens et al., 2002). The design of the repository for the drift disposal was transferred into a network for TOUGH2 and for MARNIE.

Even a failure of the drift seals did not result in advective flow of brine within or into the emplacement fields. The infrastructure area backfilled with non-compactable crushed rock (approx. 100,000 m^3 pore space) prevents an increase of hydraulic pressure at the drift seals for approx. at least several 1,000 years („buffer storage effect"), which guarantees, that there is time enough for the salt grit compaction to establish similar features as in the undisturbed salt rock.

Using MARNIE no radionuclide transport by advection was detected in the liquid phase beyond the containment providing rock zone (CPRZ). As a consequence any radionuclides in the liquid phase were transported by diffusion only. The calculated release rate from the CPRZ is significantly less than 1 Bq/a (Fig. 3).

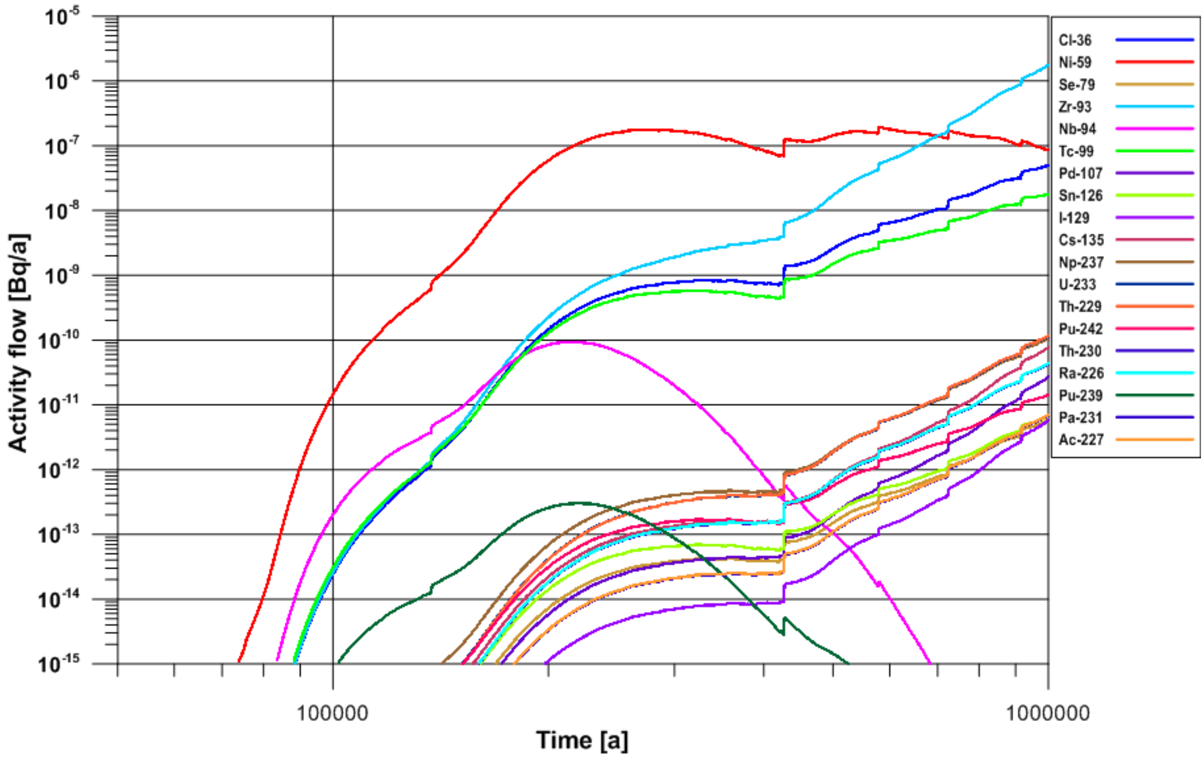

Fig. 3. Activity release at eastern drift seal (reference case PV-R3)

The radionuclide transport and release via the gas phase from the CPRZ is relevant up to some hundred years after closure in 2-phase fluid flow model calculations with TOUGH2. The compaction of salt grit and metal corrosion with gas generation are the driving forces on the transport and release of gaseous radionuclides (e.g. C-14 as methane from structural parts) through a drift seal. No C-14 was released via transport in the gas phase through a shaft seal.

5 Conclusions

The safety concept and the methodology of the VSG were compatible with the German safety requirements. The generated designs of the repository systems were feasible and complied with the safety requirements. Nevertheless some assumptions were necessary which refer to the incomplete geological exploration, the reliability of construction and some inherent uncertainties. If these assumptions are confirmed by the results of future R&D work, the designed repository system will be robust.

Options exist to optimize the repository design. A repository layout such as placing the structural components farther away from the drift seals would probably result in a lower C-14 flow through the drift seals. Furthermore, implementing a void volume as a sink (e.g. an infrastructure area backfilled with gravel) might hinder a gas flow through the shaft seals. The use of gas tight casks for the structural components (like POLLUX© canisters) will ensure retention of volatile radionuclides until long-term confinement is provided by the compacted salt grit.

Further conclusions are:

- The preliminary safety analysis for the Gorleben site identified important tasks for research and development (Thomauske et al. 2013). This would have not been possible by a generic safety analysis. Similar safety analyses should be repeated in time intervals.
- The methodology, the safety concept and proof of compliance can be applied to other salt rock sites.
- The methodology can be applied to clay stone as host rock but the application of the safety concept and proof of compliance is specific for salt rock and is probably transferable to clay stone.
- The possible release of gaseous radionuclides, two phase-flow processes and modelling of radiation exposure require further research and development.
- The mobilization of other pollutants and the heating of groundwater should be studied and requirements should be specified in more detail.

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Disposal of Radioactive Waste in Swedish Crystalline Rocks

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SKB, Swedish Nuclear Fuel and Waste Management Company is tasked with managing Swedish nuclear and radioactive waste. Crystalline rock is the obvious alternative for deep geological disposal in Sweden. SKB is, since 1988, operating a near surface repository for short-lived low and intermediate-level waste, SFR. The waste in SFR comprises operational and decommissioning waste from nuclear plants, industrial waste, research-related waste and medical waste. Spent nuclear fuel is currently stored in an interim facility while waiting for a license to construct a deep geological repository. The Swedish long-lived low and intermediate-level waste consists mainly of BWR control rods, reactor internals and legacy waste from early research in the Swedish nuclear programs. The current plan is to dispose of this waste in a separate deep geological repository, SFL, sometimes after 2045.

Understanding of the rock properties is the basis for the design of the repository concepts. Swedish crystalline rock is mechanical stable and suitable for underground constructions. The Spent Fuel Repository is planned at approximately 500 meters depth in the rock at the Forsmark site. The host rock will keep the spent fuel isolated from human and near-surface environment. The rock will also provide the stable chemical and hydraulic conditions that make it possible to select suitable technical barriers to support the containment provided by the rock. A very long lasting canister is necessary to avoid release and transport of radionuclides through water conducting fractures in the rock. A canister designed for the Swedish rock, consists of a tight, 5 cm thick corrosion barrier of copper and a load-bearing insert of cast iron. To restrict the water flow around the canister and by that prevent fast corrosion, a bentonite buffer will surround the canister. Secondary, the bentonite buffer will retard a potential release by its strong sorption of radionuclides.

The SFR repository is situated in Forsmark beneath the sea floor, covered by about 60 meters of rock. The meaning of the rock for the safety of SFR is to serve as a mechanical stable environment and to keep the waste away from human and near-surface environment. The long-term safety in SFR is based on slow release of radionuclides. By creating an appropriate chemical environment in the repository, radionuclides can be retarded in the repository. A great quantity of cement will keep a high pH during long time periods. A high pH will prevent corrosion and also make the environment less vulnerable to microbial activity. The cement and concrete constructions will also function as flow barriers.

Long-lived low and intermediate-level waste is not suitable for final disposal of in SFR. The waste requires a repository designed for longer time periods. Therefore, SKB is planning for the SFL repository. The basic principal will be the same as for SFR with slow release of the radionuclides. It is also necessary to place such a repository deeper down in the bedrock in order to increase the length of the flow paths and enhance the retardation of the released radionuclides.

Repository Designs and Technical Solutions with a View to Retrievability and Safety Requirements Currently Effective in Germany

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Abstract: *The paper summarizes the results and findings of the study "Impacts of Safety Requirement Retrievability on existing Repository Designs and Requirements for new Repository Concepts" prepared by DBE TECHNOLOGY GmbH (Bollingerfehr et al., 2014). The study establishes a basis for further developing existing and designing new repository concepts that take into account the German safety requirements. Conceptual adaptations for application of retrievability in existing repository concepts are presented and additional challenges for future R&D activities are pointed out. The paper starts with a review of the "Safety Requirements Governing the Final Disposal of Heat-Generating Radioactive Waste" (BMU, 2010) and presents a conceptual idea of how retrievability could be realized in Germany based on an example of drift disposal in a salt formation.*

1 Introduction

From the very beginning in the 1980s, generic repository concepts for the disposal of heat-generating waste and spent fuel in salt formations have been developed on the basis of safety requirements. A continuous improvement process led to a reference concept. Full-scale transport and emplacement technologies have been tested successfully in surface test facilities, again in compliance with safety requirements.

In 2010, the German Federal Ministry for the Environment, Nature Conservation and Nuclear Safety (BMU) issued the new "Safety Requirements Governing the Final Disposal of Heat-Generating Radioactive Waste". Among others, the safety requirements focus on retrievability and make it a strict licensing requirement. According to the safety requirements, retrievability is considered as the planned technical option to remove emplaced radioactive waste containers from the repository during the operational period. In order to meet these requirements, it must be analyzed if adaptations to existing repository concepts are necessary. These adaptations and modifications could include, for example, an optimization of the spacing between waste containers and drifts, the installation of adequate drift or borehole lining systems where necessary, adaptions to the ventilation system, the implementation of cooling systems, and monitoring and radiological protection measures during the retrieval process.

2 Definition of Retrievability

2.1 Definitions according to German regulations

Until 2010, retrievability was not a requirement for repositories in Germany and, therefore, all repository concepts developed until then do not include retrievability measures. Today, according to the new safety requirements retrievability is stipulated as a design criterion for a repository for HLW and spent fuel.

"Retrievability is the planned technical option for removing emplaced radioactive waste containers from the repository mine" (BMU, 2010). Retrievability of waste containers has to be given during the *"... operating phase up until sealing of the shafts or ramps..."* (BMU, 2010).

Reversible planning and regular review of decisions allow high flexibility during design, construction, operation and closure of the repository. Retrievability is a special case of the general philosophy of reversibility. The realization of retrievability is called retrieval. "*Retrieval is the actual action removing the waste packages from the repository during the operation period*" (NEA 2011).

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, 2010). The time, reasons, and the conditions for recovery can vary within a very wide range. Therefore, recovery is 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 2010).

The new safety requirements determine retrievability as a requirement for licensing. In order to comply with Atomic Energy Act, it is necessary to demonstrate compliance with the safety requirements and that the repository is technically feasible.

2.2 Comparison with international definitions

In order to illustrate the differences between the terms and also for better comparison with the understanding in other countries, the definitions given above are summarized on the retrievability scale developed by NEA, see Figure 1. The sketch was developed during the "Reversibility and Retrievability" project, which was finished in December 2011. The sketch divides the repository lifetime into six characteristic stages (NEA, 2011).

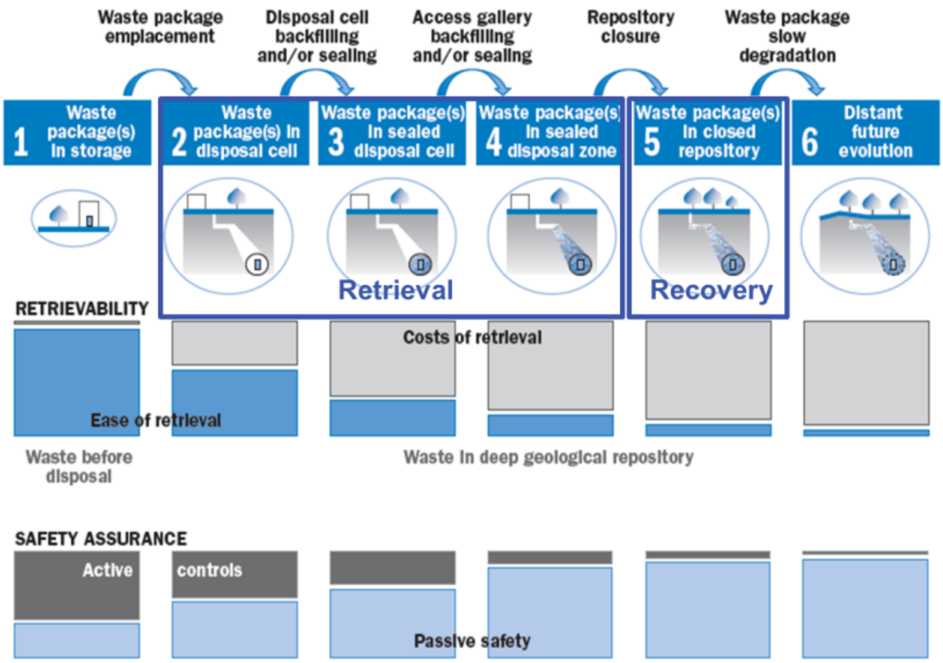

Fig. 1. Retrievability and Recovery as defined in German safety requiremnts in context of "R-scale", modified, based on (NEA, 2011)

The German safety requirements stipulate that retrievability must be given during stages two to four of the R-scale. These stages correspond to the operational period of the repository and take place simultaneously. According to the definitions already given, retrievability ends at stage four, after closure. During a defined period after closure (represented by stage five), removing the waste containers requires the construction of a new mine. This corresponds to the idea of recovery. Recovery ends 500 years after closure. For stage 6 (the distant future evolution), no retrievability requirements exist in legislation.

2.3 Strategies and measures for the retrieval of waste

Transferring retrievability into the R-scale clarifies that retrieval must be possible during different stages that are characterized by different passive and active safety levels. In general, the efforts for retrieval increase during the different stages. The safety requirements give more details on how to realize retrievability:

> *"The number of open emplacement zones should be kept to a minimum. These should be promptly loaded, then backfilled and reliably sealed from the mine building" (BMU, 2010).*

According to Figure 1, the immediate closure of the emplacement drifts and emplacement fields creates the highest possible passive safety level during the operational period. Backfilling allows the development of the repository as planned and additional impacts on the emplacement area caused by human interaction are minimized. In case of retrieval, the repository will be affected again. In this case, the emplacement fields or drifts will be re-excavated and the waste containers will be exposed. This so called "re-mining" strategy seems to be a good approach for the implementation of the retrievability option, especially in salt formations. The good mining conditions of the rock and the safety function of the host rock comply with this strategy. The main sealing function of a repository in salt is provided by the host rock itself. Convergence of the rock salt and compaction of the backfill close the drifts over time. Eventually, the backfill will have properties that are comparable with those of the host rock. If backfilling and compaction starts directly after emplacement, the process will also finish earlier and the sealing function of the rock and backfill will be achieved at an earlier time.

The implementation of the re-mining strategy for retrievability into existing repository concepts affects all stages of the repository lifetime. This includes the implementation of measures to provide retrievability as well as the handling of the retrieved waste containers. With ongoing planning the repository design and the retrievability concept will become more detailed. Measures for and adaptations to the repository concepts that could improve retrievability and ease retrieval are:

- Increased distances between drifts to improve mechanical stability or thermal load
- Reduced thermal load of the waste containers
- Consideration of retrieval in mine layout at repository design stage
- Removal of internals before backfilling
- Use of minable materials for lining systems, backfilling and sealing systems
- Supply of an adequate ventilation and cooling system during retrieval
- Automation of technical processes where possible or use of remote controlled systems
- Refinement of emplacement equipment for retrieval
- Provision of adequate measures for operational safety and radiological protection
- Provision of systems to handle contaminated materials and retrieved waste containers

Finally, the feasibility of the important technical processes of retrieval has to be demonstrated, as it already has been done for the emplacement technology. The demonstration of technical feasibility is a requirement for the licensing of all repository concepts.

3 Retrievability concepts for drift disposal in salt formations

The drift disposal concept comprises the emplacement of heat-generating radioactive waste and spent fuel in standardized POLLUX® casks. The casks have a cylindrical shape, with a diameter of 1.5 m and a length of 5.5 m. The mine layout is characterized by two access drifts, which are connected by cross-cuts. The cross-cuts open to the emplacements fields. Parallel blind-ending drifts (the emplacements drifts) branch from the cross-cuts into the emplacement fields. The POLLUX® casks are emplaced on the drift floor at regular distances, depending on their thermal output. The dimensions of the emplacement device define the minimum cross section of 17 m^2 for the emplacement drifts. Directly after emplacement, the remaining space inside the drift will be backfilled with crushed rock salt. The emplacement equipment was successfully tested by DBE in the mid-nineties. The surface demonstration tests show the technical feasibility of the emplacement concepts. In addition, subsurface experiments in a salt mine demonstrated the backfilling of the remaining drift cross-section using the slinger backfilling method. Within the scope of the "Preliminary safety analysis for the Gorleben site" (acronym VSG, (GRS, 2012)), the drift disposal concept was adapted to the local conditions at the Gorleben salt dome. This was also the first time that retrievability was taken into account in the concept.

One option to retrieve POLLUX® casks is the use of the re-mining strategy. Retrieval includes the emplacement and backfilling as described above. This concept guarantees the highest possible passive safety. In case of retrieval, the normal development of the repository differs. The retrievability concept includes the re-excavation of the backfilled galleries. The decision for retrieval has to be made before closure of the shafts at the latest. In this case, all drifts including the access drifts and the cross-cuts have to be excavated. The new retrieval drifts are excavated in three steps, see Figure 2. First, two small drifts are excavated at both sides of the waste containers. They connect the nearest cross-cuts and allow continuous ventilation and cooling. Afterwards, the remaining pillar between the two drifts is removed by road headers and modified mining equipment. The final retrieval drift provides a sufficient cross section to pick up the POLLUX® casks with a modified emplacement device. It is necessary to adapt the supporting frame that lifts and carries the POLLUX® cask and change the rail-bound system for retrieval. The modified emplacement device transfers the POLLUX® cask to a transport cart. The transport back to the surface will be realized in reverse order to the emplacement.

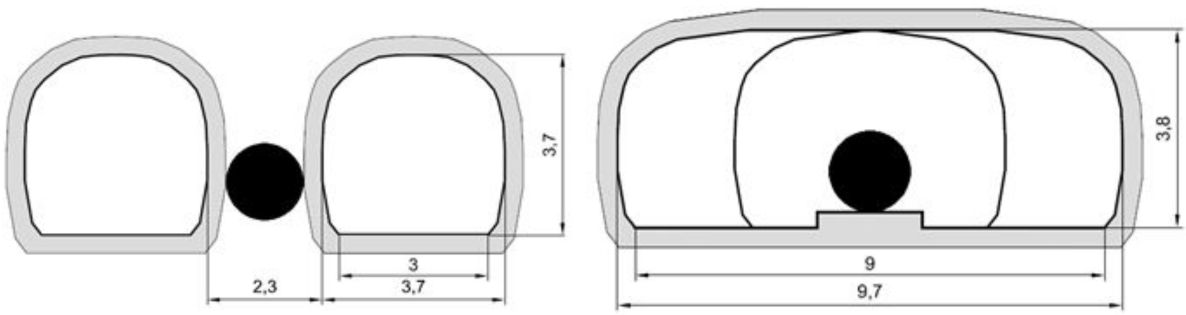

Fig. 2. Left: First, section of excavation on both sides of the POLLUX cask, Right: Final retrieval drift and contour of old emplacement gallery

The retrievability option was also implemented in the borehole disposal concept for salt formations by re-excavation of old drifts and retrieving the modified waste canisters from the lined boreholes.

4 Retrievability concepts for repository concepts in clay stone formations

In addition to more than 250 salt domes and bedded salt formations, large clay stone formations exist in Germany. In the past, generic repository concepts for clay stone formations in Germany were designed. These concepts are characterized by the adaptation of known emplacement systems (e.g. borehole and drift emplacement) to the conditions in clay formations. The designs comply with the safety criteria. Compared with repository concepts in salt formations, the level of detail is lower and the knowledge about the host rock is less comprehensive.

Implementing retrievability in a repository in clay formations follows a different approach than in salt. As mine operation in salt formations do not require any technical support, the use of a regular lining system in clay formations is generally necessary. This requires additional investigation with respect to the impacts on retrievability. The repository concept has to include technical solutions for the existing conflict between the need of a lining system during operation, retrievability, and long-term safety of the host rock. This conflict affects the retrieval strategy as well as the design of the liner and the backfilling. Existing repository concepts have to be adapted to the new safety requirements with special focus on technical solutions for the conflict to be addressed.

5 Summary and forecast

German repository concepts have been developed since the early 1980s taking into account the safety requirements. From the generic concepts, reference concepts for the drift disposal of POLLUX® casks and the deep vertical borehole disposal of canisters were established and adapted to the local conditions of a possible location. In view of a future licensing process, the emplacement technologies and methods needed have already been tested successfully. With the revision of the safety requirements, different aspects regarding repository design, including the requirement of retrievability, have to be considered additionally. The VSG took into account a retrievability concept for the very first time.

DBE TECHNOLOGY GmbH continued the technical implementation of retrievability into existing reference concepts for salt formations and prepared technical strategies for the realization in new repository concepts and other host rocks within a first study. The new R&D project, called "ERNESTA", has the aim to broaden the knowledge on how to guarantee retrievability, wants to address the technical challenges and create the basis for a future demonstration of the technical feasibility of retrieval in different repository concepts.

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SESSION 2 SCIENTIFIC ASPECTS OF THE NUCLEAR WASTE DISPOSAL SAFETY CASE (PART 1)

The Role of Safety Analyses in Site Selection: Some Personal Observations Based on the Experience from the Swiss Site Selection Process

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In Switzerland, the site selection process according to the "Sectoral Plan for Deep Geological Repositories" (BFE 2008) is underway since 2008. This process takes place in three stages. In stage 1 geological siting regions (six for the L/ILW repository and three for the HLW repository) have been identified, in stage 2 sites for the surface facilities have been identified for all siting regions in close co-operation with the sting regions and a narrowing down of the number of siting regions based on geological criteria will take place. In stage 3 the sites for a general license application are selected and the general license applications will be submitted which eventually will lead to the siting decision for both repository types.

In the Swiss site selection process, safety has the highest priority. Many factors affect safety and thus a whole range of safety-related issues are considered in the identification and screening of siting possibilities. Besides dose calculations a range of quantitative and qualitative issues are considered. Dose calculations are performed in all three stages of the site selection process. In stage 1 generic safety calculations were made to develop criteria to be used for the identification of potential siting regions. In stage 2, dose calculations are made for comparing the different siting regions according to a procedure prescribed in detail by the regulator. Combined with qualitative evaluations this will lead to a narrowing down of the number of siting regions to at least two siting regions for each repository type. In stage 3 full safety cases will be prepared as part of the documentation for the general license applications.

Besides the dose calculations, many other issues related to safety are analyzed in a quantitative and qualitative manner. These consider the 13 criteria defined in the Sectoral Plan and the corresponding indicators. The features analyzed cover the following broad themes: efficiency of geological barrier (host rock and confining units); long-term stability (erosion, differential movements, etc.); reliability of geological information (explorability; predictability); technical feasibility (sufficient space for allocating the disposal rooms; depth of repository; rock strength, etc.). For some of these issues, rather detailed quantitative analyses are made (e.g. for erosion).

Besides long-term safety, also operational safety is considered. This is done to ensure that suitable sites are chosen for the surface infrastructure (waste acceptance facilities, entrance to access to underground). The main emphasis is on external events (e.g. very severe flooding) that need to be avoided.

The involvement of society in the site selection process is also very important. This requires that the scientific information needed (and wanted) by society is delivered in a format understandable to them. This helps society develop an understanding of the question "*why here and not there*" in the siting decision; something that is considered essential to get the necessary support for the siting decision.

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Hydrocarbons in the Hauptsalz Formation of the Gorleben Salt Dome – Content, Distribution and Origin

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In the frame of the geological exploration of the Gorleben salt dome (November 2010 to November 2012) concentrations and compositions of hydrocarbons occuring in the main rock salt (Hauptsalz, Stassfurt series, z2) have been investigated. These exploration works followed former investigations of Gerling et al. (2002) and Bornemann et al. (2008).

In order to get fresh, unaltered and representative samples beyond the EDZ (excavation damaged zone) for mineralogical and geochemical analyses, about 45 boreholes have been drilled at the 840 m level of the Gorleben exploration mine. These boreholes have been arranged in equal distances (depending on the mine structure) alongside crosscut 1 west (each 6 m long) and crosscut 1 east (each 9 m long). In addition 20 packer boreholes (10 packer boreholes per crosscut) for pressure build-up recording and hydrocarbon sampling have also been established.

Immediately after drilling, core samples from the Hauptsalz for organic geochemical analyses have been retrieved and were dissolved in deionised and degased water. The results of analyses of about 210 samples scattered over all 45 boreholes reveal a total background concentration of hydrocarbons (C_1 to C_{40}) of 0,24 mg/kg. 70 samples have concentrations between 1 mg/kg and 50 mg/kg (average 2,66 mg/kg) with 5 outliers up to 442 mg/kg in crosscut 1 west (Hammer et al. 2012, 2013). The drill cores have been investigated and documented by using ultraviolet light (l = 254 nm) in respect of visible indications of the existence of fluorescing aromatic hydrocarbons. Analyses revealed a high level of heterogeneous hydrocarbon distribution in the shape of isolated, irregular streaks, clusters, clouds and occasionally layers mainly located in recrystallized zones of the Hauptsalz.

Thin sections and thick sections showed that hydrocarbons in z2HS1 (Knäuelsalz) and z2HS2 (Streifensalz) samples are either located as black to brownish dendritical fluid inclusions alongside the grain boundaries of halite crystals, on the surfaces and knuckles or inside of micro capillary tubes of anhydrite crystals and anhydrite clusters, in newly formed micro cracks due to drilling respectively preparational works or rarely in micro-porous parts of the Hauptsalz.

In order to get additional information about the origin of hydrocarbons detected in the Gorleben Hauptsalz organic geochemical analyses of potential source rocks in the vicinity like the Stassfurt Carbonate (z2SK) have been provided. These analyses revealed that the level of maturity of hydrocarbons in the Gorleben Hauptsalz correspond to 0,8 to 1,2% vitrinite-reflection-equivalent for the oil, similar to the organic-petrographical data of Stassfurt Carbonate and Copper schist in the periphery of the Gorleben salt dome (Gerling et al. 2002; Senglaub 2001; Cramer 2005). The analyses of biomarkers (esp. triterpenoid biomarkers) detected in the hydrocarbon mixtures from the Hauptsalz point to the Stassfurt Carbonate as source rocks of most of the hydrocarbons.

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Current state of knowledge on long term behavior of spent nuclear fuel under conditions of deep geological disposal

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Abstract: *Though physicochemical processes related to the behavior of spent nuclear fuel (SNF) in a deep geological repository are of complex nature, knowledge on underlying mechanisms has considerably advanced over the last years. Several key issues for the long-term safety assessment of SNF disposal are related to the interaction of water with the waste form. Upon contact with water, the alteration of SNF and the consecutive release of radionuclides involve the combination of many different processes, which can be grouped into two stages: (i) A fast / instant release of a fraction of radionuclides and (ii) a much slower and long-lasting radionuclide release that results from the alteration and dissolution of the* UO_2 *fuel matrix. The so-called Instant Release Fraction (IRF) comprises fission products which are segregated during irradiation to the pellet / cladding gap, fractures in the pellet and grain boundaries as well as activation products within the cladding. According to a review of Kienzler et al. (2013) the IRF depends mainly on linear heat rating, burn-up and other in-reactor fuel operating parameters. As a major result of recent studies, it turns out that alteration of the* UO_2 *fuel matrix is significantly inhibited under the strongly reducing conditions induced by container corrosion and consecutive hydrogen production. It is assumed, radiolysis driven oxidative fuel matrix dissolution appears to be less relevant for most repository concepts. Still, open questions remain related to the exact mechanism of the protective hydrogen effect on SNF matrix corrosion and the impact of potentially interfering naturally occurring groundwater trace components.*

1 Introduction

Presently, final disposal of spent nuclear fuels, SNF, in deep bedrock repositories is considered as the preferred option for the management of nuclear fuels irradiated in light water reactors (LWR). Isolating this kind of high level waste products from the biosphere in a geologically stable environment over periods of several hundreds of thousands of years offers maximum safety, which cannot be guaranteed at present by other concepts. The development of multi-barrier systems for SNF disposal is being pursued internationally. These systems will consist of several independent barriers which prevent effectively the release of radionuclides from the waste products into the geosphere and their subsequent migration into the biosphere. Though, geological or geo-technical barrier system may prevent to some extent groundwater contacting the fuel, intrusion of solutions into disposal rooms has to be taken into account within the long-term safety case of a SNF repository. The fate of spent nuclear fuel and the associated release of radionuclides depend on SNF corrosion kinetics as well as on thermodynamics. Corrosion of the fuel and radionuclide behaviour are influenced by a variety of factors such as radionuclide inventory, defining temporal variable α–,β– and γ–radiation fields, pH, redox potential, availability of solids acting as sorbents, complexing groundwater constituents and temperature.

In the past decade considerable research effort has been expended to quantify the radionuclide release from corroding spent LWR fuels under repository conditions. This communication highlights main issues of the current state of knowledge on alteration of spent LWR fuel and the consecutive release of radionuclides.

2 Characteristic properties of LWR UO_2 fuels and distribution of radionuclides in spent nuclear fuel

Most commercial LWR fuels have an initial enrichment of 3 to 5 wt.% U-235. So-called mixed-oxide fuel, MOX, contains 5 to 10% Pu-239 mixed with uranium. Fuel assemblies are rods composed of cylindrical pellets made of polycrystalline UO_2(s), which are stacked in tubes of gas-tight zirconium alloy cladding materials, i.e. Zircaloy. During nuclear reactor operation the UO_2 fuel undergoes intensive transformations due to fission, consecutive neutron capture and decay reactions as well as temperature induced transformations (e.g. swelling, diffusion, cracking and segregation). Currently, fuels irradiated in LWR with typical burn-ups (BU) of 35 to 45 MWd/kg U, contain about 95 wt.% of UO_2. This SNF matrix consists mainly of U-238, remaining U-235 and U-236 produced from U-235 by neutron capture (e.g. Neeb, 1997; Bruno and Ewing, 2006). According to Olander (1976) the oxygen potential of the fuel increases significantly at burn-up ranges between 50 and 60 MWd/kg U at temperatures of modern LWRs. Since the average valence of the fission products is less than two, the oxygen : uranium ratio increases slightly during irradiation, so that the stoichiometry of the fuel matrix is expressed as $UO_{2\pm x}$. The deviation x from the $UO_{2.00}$ stoichiometric composition varies within a fuel pellet or a rod depending on the local burn-up. It influences various physical properties, in particular those which depend on the atomic mobility (e.g. diffusion coefficients), and the oxygen partial pressure or chemical behaviour of the fission products. The chemical stability of fission products within the UO_2 matrix can be classified into four main groups:

- Fission gases and other volatile fission products (Kr, Xe, Br, I) which occur as finely dispersed bubbles within grains, at grain boundaries and in pores (Cui et al., 2004). To some extent they are released to the open porosity of the fuel and gap between the periphery of the fuel pellet and the surrounding cladding.
- Metallic fission products (Mo, Tc, Ru, Rh, Pd, Ag, Cd, In, Sn, Sb, Te) forming immiscible, micron- to nanometer-sized precipitates. The metallic phases found are ε-Ru(Mo, Tc, Rh, Pd) solid solutions and are denoted as ε-particles (Bramman et al., 1968; Kleykamp et al., 1985).
- Fission products forming oxide precipitates (Rb, Cs, Ba, Sr, Zr, Nb, Mo, Te). These oxides tend to have the general composition $AB[O_3]$ and to adopt a cubic perovskite-type structure with Ba, Sr, and Cs in the A sites and Zr, Mo, U and lanthanides in the B sites (Kleykamp et al., 1985; He et al., 2007). Depending on the oxygen potential, Mo may exist in metallic or oxidized state.
- Fission products occurring as oxides in the UO_2 matrix (Sr, Zr, Nb, Y, La, Ce, Pr, Pm and Sm). Zr and lanthanides are partially or completely miscible with UO_2 to form solid solution (González-Robles, 2011).

There are continuous transitions between the four groups, which depend on the irradiation history (burn-up, linear heat generation rate etc.) and consequently on the chemical state and solubilities of the fission products Kleykamp (1988). For example, prominent Sr and Zr isotopes occur in the dominating ceramic precipitate $(Ba_{1-x-y}Sr_xCs_y)(U,Pu,Ln,Zr,Mo)O_3$ as well as solid solutions in the UO_2 matrix. Similarly to lanthanides, transuranium elements (Pu, Am, Cm, Np) occur in the fuel matrix, where they substitute U. Detailed descriptions of the radionuclide distribution in spent nuclear fuel and the radiochemical properties are given for instance by Kleykamp (1979; 1985; 1988), Neeb (1979) and Carbol et al. (2012).

In their review, Bruno and Ewing (2006) argue that the distribution of radionuclides is not homogeneous within a single pellet due to the steep thermal gradient within the pellet during irradiation. Thermal excursion during reactor operation cause a coarsening of the grain size, extensive microfracturing and migration of fission gases and volatile fission products to grain boundaries, fractures, and the gap region. Due to a gradient in the local burn-up with relatively high burn-ups at the pellet periphery, porous and fine grained microstructure "high burn-up structures" are formed. These high burn-up structures are characterized by relatively high concentrations of ^{239}Pu at the fuel rim, polygonization of the UO_2 grains resulting in a reduction in the size of individual grains (González-Robles, 2011).

In addition to radionuclides in the spent nuclear fuel itself, activation products are present in the cladding and other structural parts of the fuel elements. For time intervals of interim storage, the radionuclide inventory of these metallic parts of a fuel element is dominated by short-lived activation products, such as Co-60, Ti-44 and Sb-125. With respect to the long-term safety of irradiated fuel elements, mainly C-14 (produced by N-14(n,p) reactions in cladding and steel components) and Tc-99 (produced by Mo-98(n,γ) reactions in steel components) are of concern.

3 Radionuclide release from spent nuclear fuel

Spent nuclear fuel alteration processes and radionuclide release processes depend both on the characteristics of the SNF itself and on many spatially and temporal varying geochemical parameters in the near field of the repository. Dissolution experiments simulating the contact of aqueous solution with SNF demonstrate clearly that radionuclide release from the fuel is strongly affected by the spatial radionuclide distribution among (i) the gap region, (ii) boundaries between UO_2 grains, (iii) UO_2 matrix and (iv) in the high burn-up structures (Katayama, 1976; Johnson et al, 1985; Poinssot et al., 2005; Bruno and Ewing, 2006; González-Robles, 2011). The alteration of SNF and the consecutive release of radionuclides involve the combination of many different processes, which can be grouped into two stages:

(i) Fast release of radionuclides from the gap and grain boundaries at the time of breaching of the SNF canister. The fast released radionuclide fraction is generally referred to as the Fast / Instant Release Fraction (IRF). Certain radionuclides are accumulated in channels, which are formed by fission gases within fuel grains, or along grain boundaries. It is assumed that fast radionuclide release can be facilitated through these fission gas networks and thereby contributes to the IRF.

(ii) A much slower, long-term radionuclide release that results from the alteration and dissolution of the UO_2 matrix. The reactions of the fuel matrix with water are driven both by radiolytic processes and by a thermodynamic affinity effect.

The IRF comprises a fraction of few percent of Cs-135 (and short-lived Cs-137), I-129 and potentially Cl-36, Se-79, Tc-99 as well as Pd-107. C-14 (and to less extent Tc-99) present in the Zircaloy cladding and steel parts of the fuel elements are often also considered as contributing to the IRF. I-129, Cl-36 and Se-79 tend to form anionic species and therefore they are hardly chemically retained even in a deep geological disposal system. Since C-14 is expected to be released both as dissolved / gaseous inorganic and organic compounds, a high mobility of C-14 is considered.

Within the current Collaborative Project FIRST-Nuclides ("Fast / Instant Release of Safety Relevant Radionuclides from Spent Nuclear Fuel") new experimental and theoretical studies on the IRF of high burn-up SNF are performed. Detailed descriptions of fast / instant release and a summary of results obtained from more than hundred published experiments with SNF samples are given in the FIRST-Nuclides State-of-the-Art report (Kienzler et al., 2013). It is shown that fast / instant release of I-129 is in the order of fission gas release (FGR), whereas the ratio of fractional release of fission gases to Cs-137 release is in the range of 3:1. Generally, the FGR is measured by means of puncturing the plenum of an irradiated fuel rod. Obviously, there is a dependence of the IRF of Cs-137, Cs-135, I-129 and fission gases on the irradiation history and other in-reactor fuel operating parameters of the studied SNF. New experimental results of FIRST-Nuclides demonstrate that the IRF of these radionuclides is higher for SNF irradiated in Pressurized Water Reactors (PWR) compared to SNF irradiated in Boiling Water Reactor (BWR) for similar burn-up values. There is growing evidence that FGR (Fig. 1) and IRF of Cs-137, Cs-135 and I-129 increase with increasing linear heat generation rate (LHGR) rather than with burn-up of the studied SNF.

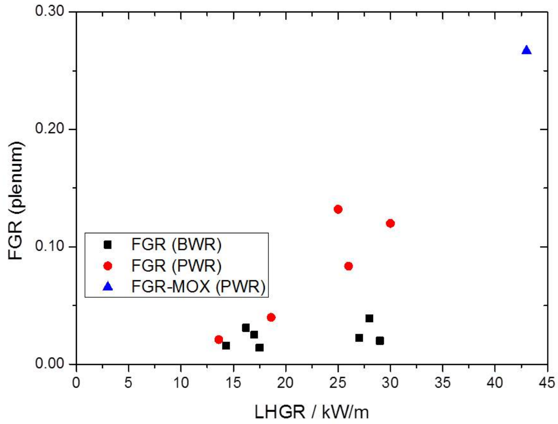

Fig. 1. Fractional fission gas release (FGR) measured in plenum puncturing tests as function of power rate (LHGR) of BWR and PWR SNF samples studied in the Collabrative Project FIRST-Nuclides.

The release of radionuclides bound to the UO_2 matrix depends on the extent of $UO_2(s)$ corrosion, which is controlled by a competition between oxidizing radiolysis products and complexing species on one side and inhibitors on the other side. Radiation induced UO_2 matrix dissolution can only occur if the non-radiolytic redox couples in the near field do not consume the radiolytic oxidants. Since high hydrogen concentrations will be achieved in the near field of a deep geological repository, mainly caused by anaerobic corrosion of the Fe-based waste canisters, an inhibition of the SNF matrix dissolution is expected. Leaching experiments with SNF samples and non-irradiated $UO_2(s)$ samples as well as radiolysis studies indicate that molecular hydrogen both impedes radiolytic decomposition of the studied simple groundwater surrogates and considerably inhibits corrosion of the fuel matrix (Metz et al., 2012 and references therein). This hydrogen inhibition effect is considered to be related to three different electrochemical, radiochemical and radiolytically driven processes:

- Catalytic reactions at the surface of ε-Ru(Mo, Tc, Rh, Pd) and / or other ε -particles.
- $UO_2(s)$ surface controlled reduction of $UO_2^{2+}(aq)$ by H_2 in presence of corroding iron.
- Inhibition of the radiolytic production of oxidizing species (H_2O_2 and $OH^\bullet$).

It is widely assumed that radiolysis driven oxidative fuel matrix dissolution appears to be less relevant for most repository concepts (e.g. Eriksen et al., 2012), though there is insufficient knowledge about the molecular mechanisms of the protective hydrogen effect on SNF corrosion. In particular there is a lack of knowledge on the number and density of ε-particles in SNF. Moreover, an international benchmark study on modelling of SNF corrosion shows that model uncertainties are still very large, especially regarding the effect of hydrogen as well as the respective parameters of the radiolytic reactions (Bruno et al., 2009). Although the effect of hydrogen overpressure on the SNF corrosion behaviour has been studied extensively in near neutral pH diluted and saline NaCl-$NaHCO_3$ and NaCl containing solutions, the hydrogen inhibition effect need to be tested in aqueous solutions which are similar to natural groundwater. Certain groundwater constituents, such as bromide or sulphur, could affect the interaction of radiolysis products with hydrogen or could poison ε-particles.

In the long term, the decay of the SNF radioactivity will lead to such a low radiation dose that radiolysis is expected to become unable to sustain oxidative dissolution. In the absence of radiolytic oxidants, U(IV) solubility controlled UO_2 matrix dissolution will dominate under the reducing conditions expected for the near-field of a deep geological repository.

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The Long-Term Durability of Low Alkali Cements: Evidence from New Natural Analog Sites in Europe and North Africa

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The long-term durability of low alkali cements is of interest where they are under consideration as repository tunnel and exploration borehole seals and plugs. It is essential to have an appropriate understanding of their longevity to inform decisions on their potential use in a repository environment.

Archaelogical analogues of low alkali cement have been studied for some time. Thomassin & Rassineux (1992), for example, reviewed some of the literature on Gallo-Roman cement-based materials and noted that one of the most impressive examples is the 1700 year old Roman mortar used in Hadrian's Wall (UK) which still contains substantial amounts of CSH (calcium silicate hydrate) compounds. These mortars were studied specifically with the behaviour of an ILW repository in mind (Jull & Lees 1990).

However, plugs and seals will generally be required to be durable for longer than the few thousand years which can be accessed via archaeological analogues, so it is essential to turn to natural systems for evidence of longer term durability. To date, there have been no reported studies on natural low alkali cements. In principle, however, such cements should exist and the Bituminous Marl Formation, which hosts the natural OPC cements in Jordan (Pitty & Alexander, 2011), is a likely source. This Formation constitutes a widespread terrain which stretches from Syria in the north, through Israel and Jordan to Saudi Arabia in the south. The natural cement was formed by the combustion of organic rich limestones, a process which continues today. In Syria and northern Jordan, for example, the Formation is punctured by Late Oligocene to Quaternary volcanics so sites which include pozzolanic ash mixed with the Bituminous Marl exist and, on combustion, should produce natural low alkali cements. A site in northern Jordan is currently under investigation for evidence of long-term fresh groundwater/low alkali cement interaction and the preliminary results of the study will be reported here.

In addition, a second site in Europe has been identified where natural low alkali cements exist in combination with both saline groundwaters and brines, allowing examination of saline groundwater/low alkali cement interaction. Once again, the preliminary results of the study will be reported here and a comparison between the two sites will be made, highlighting the different impact of the differing groundwaters on the cement. Although precise dates for the systems examined are not yet forthcoming, this is a focus for future efforts at both sites, allowing more precise definition of the probable longevity of low alkali cements in repository environments.

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Simulation of Density-Driven Flow in Heterogeneous and Fractured Porous Media

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The study of fractured porous media is an important and challenging problem in hydrogeology. One of the difficulties is that mathematical models have to account for heterogeneity introduced by fractures in hydrogeological media. Heterogeneity may strongly influence the physical processes taking place in these media. Moreover, the thickness of the fractures, which is usually negligible in comparison with the size of the whole domain, and the complicated geometry of fracture networks reduce essentially the efficiency of numerical methods. In order to overcome these difficulties, fractures are sometimes considered as objects of reduced dimensionality (surfaces in three dimensions), and the field equations are averaged along the fracture width.

Fractures are assumed to be thin regions of space filled with a porous material whose properties differ from those of the porous medium enclosing them. The interfaces separating the fractures from the embedding medium are assumed to be ideal. We consider two approaches:

(i) the fractures have the same dimension, *d*, as the embedding medium and are said to be *d*-dimensional;

(ii) (ii) the fractures are considered as (*d*-1)-dimensional manifolds, and the equations of density-driven flow are found by averaging the *d*-dimensional laws over the fracture width.

We show that the second approach is a valid alternative to the first one. For this purpose, we perform numerical experiments using a finite-volume discretization for both approaches. The results obtained by the two methods are in good agreement with each other.

We derive a criterion for the validity of the simplified representation. The criterion characterizes the transition of a mainly parallel flow to a rotational flow, which cannot be reasonably approximated using a d-1 dimensional representation. We further present a numerical algorithm using adaptive dimensional representation.

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SESSION 3 GOVERNANCE AND PUBLIC INVOLVEMENT

Implementing Geological Disposal: A Long-Term Governance Challenge

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Calling geological disposal (GD) a technical and societal challenge and arguing that democratic decision-making on GD requires public and stakeholder engagement (PSE), are statements that will not meet much opposition. A process of 'governance' consists of engaging stakeholder groups in decision making processes and contrasts with more traditional, often technocratic forms of government. As will be argued in other papers in this conference (e.g. Grunwald; Kallenbach-Herbert et al.; Röhlig et al.) it is of fairly recent date, that concerned actors increasingly recognize that PSE should relate to both the societal and technical questions concerning GD. While most people would agree in theory, putting 'technical democracy' (Callon et al. 2001) in practice, often proofs to be less obvious. Opening up the technical 'black box' remains a crucial challenge in discussing the implications of GD for society and for the environment.

As findings from the InSOTEC project show, this can be explained because different types of problematization occur, often considered as sequential, rather than intertwined (Barthe et al. 2014). Social problematization of GD, i.e. considering the remaining obstacles for implementation to be in essence social in nature, is often associated with the siting stage, when the technological project meets its social environment (ibidem). Formal participatory processes are often aimed mainly at dealing with socio-economic impacts and adapting life on the surface to the underground technology project, rather than the other way around (Bergmans et al. forthcoming). Still such interactions can, and have indeed proven to, lead to technical problematization, i.e. putting into question the technical project or certain aspects of it (cf. Barthe et al. 2014), by concerned stakeholders. As can be observed in the case of Sweden - for GD of spent fuel, and Belgium - for surface disposal of low- and intermediate level waste, this does not necessarily have to lead to a rejection of the proposed project. .

In this presentation I would like to draw on these two cases to make a related argument about the need to consider long-term governance processes, reaching beyond the remits of classical site selection procedures. Rather than considering siting as the end point of a participatory process, it should be seen as a starting point. For that purpose, I will make use of the notion of hosting to emphasize the relationship between the repository and its host community. A relationship that demands a re-figuration of the geography and temporality of geological disposal. Hosting a geological disposal facility brings with it specific challenges, involving both social and technical adjustments, as well as reconfigurations of the boundary between them (Landström & Bergmans 2014).

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Consensus Shaping and Safe Space Public Participation Processes

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Stakeholder involvement and public participation is recognized as a necessary part of nuclear waste management. It is possible to see three driving forces behind this development:

1) a deliberative movement with its roots in the 1960s leftish movement (a philosophical argument),
2) the need for public acceptance of proposed solutions (a practical argument), and
3) the need for awareness and clarity before crucial decisions are made (a practical and democratic argument). Depending on which driving force dominates an initiative for public participation, the initiator is likely to prefer certain public participation processes before others.

The initiator has to select among a large number of processes already having been implemented or design a new one tailored for the specific situation. The process choice is crucial for which stakeholders are able to join and which goals can be reached.

Much of the debate about public participation has referred to the "Arnstein ladder" (Arnstein, 1969) or its modernized versions, such as the participation ladder used in the IPPA Project (Richardson et.al. 2011) building on work by the Netherlands Environmental Assessment Agency (MNP, 2008). The ***various forms of ladders*** have in common an increasing level of ambition for participation from low to high. The higher up on the ladder, the better it is often anticipated. However, with this should follow more responsibility and accountability of all participants, which in practice is often lacking. There are also practical problems in using the ladder to map public participation processes. For example the step "collaboration" is ambiguous as it can mean different things. In the RISCOM Process, participants collaborate to improve clarity and awareness but not for finding common solutions. Therefore, the RISCOM Process (Vojtechova, 2009) does not fit into the ladder structure.

To avoid these and other problems it is suggested to use three ***basic approaches*** to map public participation – consultation, consensus shaping processes and safe space. In contrast to consensus shaping processes, in the safe space approach there is no intention to develop solutions together between the implementer and other stakeholders. In comparison with the ladder, the basic approaches offer advantages for the mapping of public participation processes:

If you strive for consensus or "only" clarity and awareness is a crucial question to be answered before a process is launched as it determines which stakeholders can join and which goals are feasible to reach.

It is easier to use in assigning properties to a process, as it more straight-forward to understand if a process is consensus shaping or a safe space than where it is on the ladder.

It clarifies better the links between the participation processes and the actual political and/or legal decision-making process. A consensus shaping process should produce real advice (or even decisions) but a safe space is limited to improving the decision making base with enhanced clarity.

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Installation of a Radioactive Waste Disposal Facility: the Necessity of Building up Durable Links between the General Public and Radioactive Waste – Feedback from Experience in France.

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2013 has been a banner year for Andra with widespread discussions on the question of long-term management of radioactive waste: a nationwide public discussion about the planned Cigéo deep disposal facility has been organized and national discussions on the energy source transition had inevitably brought up the question of what to do with future radioactive waste to be produced under the various scenarios put forward.

In spite of an open institutional framework, with numerous legal provisions for citizen participation, 2013 showed that creation of a radioactive waste disposal facility is not, and cannot be, a question dealt with like breaking news, within a given temporal or spatial perimeter. Any attempts to bring up the subject under the spotlight of public scrutiny inevitably shift the discussions away from their central theme and abandon the underlying question – what should be done with the existing radioactive waste and the waste that is bound to be produced? – to move on to the other major question: "Should we stop using nuclear power or not?", which takes us away from our responsibilities towards future generations.

Daring to face the question, anchor it in citizen discussions, and create awareness of our duties towards coming generations: this is the challenge that Andra had already set itself several years ago. Our position is a strong one; rather than seeking to mask the problem of radioactive waste, we must face up to our responsibilities: the waste is already there, and we have to do something with it.

It will take time to be successful here. Long-term management of radioactive waste is clearly a really long-term matter! All the experience in the field has shown that it involves patience and careful listening, and requires building up a basis for solid trust among the potential neighboring population, who are the most directly concerned. Durable proximity human investment is one of the key factors of success. For over 20 years now, it has been one of the principles on which Andra has built its project management schemes.

Nonetheless, the fresh light brought to bear on the matter by nationwide discussions has its merits. The subject is now visible: the press, social networks, politicians and NGOs are going over the matter, and giving it widespread attention. Unbiased citizens have been called on to make their comments and provide guidance for the overall social and ethical approach to the subject. Some real links are being set up at last!

The article will cover in greater detail the path followed by Andra over the last twenty years: its successes, its difficulties and its failures, concentrating of course on 2013, its preparation, which began early in 2012. The article will also present the Andra's further course of action and issues proposals to the State in order to meet the expectations that arose from the public discussion.

The Swiss Approach to Finding Compromises in Nuclear Waste Governance

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Abstract: *In Switzerland, a new site selection procedure is being implemented since 2008. This procedure, which is laid down in a 'sectoral plan', shows strong elements of public participation and transparency and can be considered a step away from the classical 'decide-announce-defend' approach in decision-making. This procedure tends towards a more governance-oriented approach based on ideas of 'civility' of decision-making. Despite this renewal, the Swiss case clearly shows that any kind of selection process has to be considered as a 'working compromise', which needs to be adapted when new challenges emerge.*

1 Introduction

In Switzerland, as well as in many other countries, which are planning to install underground repositories for their high-level nuclear wastes, social conflicts have arisen in the process of searching and exploring a suitable site. Those conflicts can be understood as historically developed. In the context of the debate on nuclear power production, the anti-nuclear social movement has emerged. It not only demanded a phase-out of nuclear power production, but also participation in the debate on how to deal with the high-level nuclear waste produced (Grunwald and Hocke 2009, Grunwald 2010: 254-257). Responsible governmental actors reacted by taking up new ideas about 'governance' – a term used both in the context of governmental organizations and in the social scientific debate (e.g. European Commission 2001, Benz and Dose 2010). Central parts of the debate were ideas about public participation in decision-making. Over time, those ideas were also picked up in the nuclear community.

In Switzerland, a stale-mate in decision-making occurred when an application for underground exploration for a repository site for low- and intermediate level waste was rejected in a cantonal referendum. As a result, the Swiss parliament decided to abolish the cantonal veto right and instead install an official step-wise site selection procedure with an optional national referendum at the very end of the selection process. To this end, they chose and adapted a well-established planning instrument ("sectoral plan") (SFOE 2008). In setting up this approach, the Swiss authorities took up ideas developed by the German Committee for the Development of a Site Selection Procedure (AkEnd 2002) and international organisations, such as the Nuclear Energy Agency`s Forum on Stakeholder Confidence (e.g. NEA 2003). Further, consultations took place with a wide range of stakeholders, in which central ideas for the sectoral plan were discussed before its final approval by the Swiss Federal Council (the Swiss national executive). Implementation of the sectoral plan started in 2008 and is still on track, even though some substantial changes needed to be made. Different reasons for these changes can be observed. They range from new knowledge gained to strong demands from the public.

In this paper, the Swiss approach to repository siting will be analyzed based on governance theories and related ideas of public participation.

2 Swiss Nuclear Waste Governance

In Switzerland, three categories of nuclear waste are differentiated: (a) high-level waste (HLW); (b) alpha-toxic waste; (c) low- and intermediate-level waste (L/ILW) (SFOE 2008: 12). In total, a repository will be needed for 7,300 m^3 of high-level waste (reactor life-time of 50 years) and for 93,000 m^3 of low- and intermediate level waste (Nagra 2011).

Before the implementation of the sectoral plan, the site selection procedure for a repository site was not regulated by the state. Rather, the 'National Cooperative for the Disposal of Radioactive Waste' (Nagra) was responsible for identifying a site and, if a site was found, which they deemed suitable, to hand in the application for a general license. In 2003, a new Nuclear Energy Act was passed by the Swiss parliament. It went into force in 2005. In this law, the cantonal veto right was abandoned. In the corresponding Nuclear Energy Ordinance (NEO) from 2004 it was specified that a sectoral plan is to be used to formulate the federal government's objectives and criteria for the disposal of radioactive waste in deep geological repositories in a legally binding way (NEO, Art. 5). The sectoral plan was then conceptualized in a way that it defines a step-wise site selection process for geological repositories with strong elements of public participation (Kuppler and Hocke 2012).

By formalizing the site selection procedure and handing over responsibility for its coordination from the Nagra to the Swiss Federal Office of Energy (SFOE), ideas from the debates on "governance" were taken up. The term "governance" is used in different ways, even in the scientific debate. One central idea is that the efficiency of governmental action should be improved by involving non-governmental actors into the provision of services. Another related idea is based on the insight that classical top-down approaches in decision-making have lost their steering capacity. Based on this observation, the focus of the governance perspective lies on the organization of decision-making processes and the state's institutionalized structures for resolving problems and conflicts (e.g. Chhotray and Stoker 2009, Hoppe 2010, Grande 2012).

Particular attention is laid on the integration of stakeholders in networks and their coordination within those networks (Benz and Dose 2010, Haus 2010). The latter is particularly difficult in technology conflicts, such as the conflict on nuclear waste repository siting, as the stakeholders are often more or less autonomous collective actors with strong expectations regarding public participation. The challenge for the responsible governmental actors is to keep those actors in the site selection process while at the same time sticking to the rules fixed in the sectoral plan.

3 The Sectoral Plan

In 1995 and again in 2002 the Nagra's application for underground explorations at the Wellenberg site was stopped by two cantonal referendums. Also in 2002, the Nagra submitted a 'demonstration of disposal feasibility' to the federal authorities for a repository site for HLW. This was approved by the federal authorities in 2006. Having in mind the experiences at the Wellenberg site, the federal government stated already in 2002 that an approval of the 'demonstration of disposal feasibility' does not mean that the exemplary site used in the demonstration will become the site for the final repository. Rather, it demanded that the Nagra has to compare several sites regarding their suitability.

Those events influenced the conceptualization of the Nuclear Energy Act and the decision to use a sectoral plan for organizing the site selection process. In the Nuclear Energy Act, the cantonal right to hold a referendum on planning activities related to repository siting was abandoned. Instead, a right to be heard was introduced, but only if the siting process would not be unnecessarily obstructed by this. Further, the right for an optional national referendum on the granting of the general license by the Swiss parliament was introduced.

Sectoral plans are the established planning instrument at federal level in Switzerland. They are used to consolidate state activities in a certain field of planning. In comparison to other sectoral plans, the Sectoral Plan for Deep Geological Repositories offers extended public participation. Public participation in the sense of "having a say" is though limited to above-ground facilities; in all other questions participation is limited to consultation. Consultation takes place in several forums designed for this purpose, especially in the so-called regional conferences, in which representatives from the interested public and stakeholders are members (Jost 2012). The tasks those regional conferences should fulfil are defined in the sectoral plan. Through this, the limits of their involvement in decision-making are also indirectly defined.

Already at very early stages of development of the sectoral plan, public debates were organized in order to increase the likelihood that all relevant collective actors would participate in the implementation of the final plan (Minhans and Kallenbach- Herbert 2012: 7). Those consultations included information events, but also focus groups organized by the SFOE. Furthermore, informal consultations were held not only with industry representatives, but also with representatives from local administrations from one potential siting area. In those consultations, the Swiss government profited from the strong Swiss consensus democratic tradition. It showed that, even though time consuming, this approach was insofar a success as all stakeholders agreed to take part in the new site selection procedure. A working compromise had been found.

Central principles fixated in the sectoral plan are a step-wise approach with the possibility to fall back on earlier steps if necessary at any point of time, the idea that socio-economic criteria should play a role in site selection, but that safety always comes first, and integration of the local public as well as transparency regarding all information available and regarding the structure of the sectoral plan. The latter means that roles were clearly distributed and criteria for moving from one step to the next clearly defined. It was also clearly defined who could participate on what topic. At the same time, room was left for adaptations, which might become necessary due to new developments.

One important principle behind the sectoral plan is the ideal of the "civility" of decision-making. This approach is grounded on ethical principles of dealing with conflicts. Instead of exclusively applying majority-oriented procedures to decision-making situations discourse ethics (Habermas 1991) proposes to establish an open, argumentation-oriented and procedural approach to identify the best solution among all persons and groups affected. The discursive procedure of exchanging arguments and arguing for specific positions and options must observe principles of inclusion, equity, and fairness. The expectation is that a civil procedure of this kind could provide better results compared to traditional approaches of purely representative democracy. However, there is no guarantee, and success depends on a large diversity of cultural factors (Habermas 2008).

4 The implementation phase

After the start of the implementation of the sectoral plan in 2008, the regional conferences were established in line with the original schedule. Each conference has about 100 members (Jost 2012: 147). They represent a wide range of regional stakeholders, such as citizens, NGOs and community representatives. Within the regional conferences, three smaller working groups were established. One of those groups is on safety aspects. The establishment of this group was not part of the original plan, but installed after strong demands from the interested public and NGOs. By installing those groups, the responsible authorities compromised on their idea that safety was a topic only the authorities had to take care of.

The regional conferences are relatively free in their self-organization. They receive financial support from the federal authorities, which allows them to invite experts and pay for studies as they deem necessary.

Furthermore, a "Technical Forum Safety" (TFS) has been established. The TFS has the task to discuss and answer the stakeholders' technical and scientific questions. The forum consists of experts, responsible authorities, the Nagra, the cantons, as well as one representative from each of the siting regions.

Besides those institutionalized forms of participation and information, a high degree of transparency has been established regarding the documentation and publication of the progress in site selection. In one case, the interested public and the regional conferences demanded for the Nagra to publish documents explaining the background for some of their work. The Nagra followed this demand.

Furthermore, information meetings and workshops were held with members of the interested public and the regional conferences. All those activities were also documented (e.g. SFOE 2009, AGNEB 2013).

Even in the cases, in which the stakeholders' rights are limited to a 'right to be heard' this puts strong demands on the responsible authorities and the Nagra as this means that they have the right to receive answers to their questions.

Despite these developments away from the classical decide-announce-defend approach towards a civil conflict resolution, some potential weaknesses in the concept have become visible during implementation. First, there is no clearly institutionalized way of resolving conflicts within the process (ESchT 2009). This means that e.g. conflicts arising among members of the regional conferences are resolved 'ad-hoc' by the SFOE instead of having an institutionalized, and thus transparent, way of dealing with them. Second, there is no institutionalized form of dealing with minority opinions. Two members of governmental committees left their jobs because they were of the opinion that their arguments were not heard. Third, the original time-table was calculated too tight. Over several years this was repeatedly moaned by members of the regional conferences and at one point led to one of them enforcing a 'stop' in the implementation of the sectoral plan until they got more time to fulfil a certain task. Only in 2014, the time schedule was officially extended. Fourth, the abandonment of the cantonal veto right has not been fully accepted by all cantons.

5 Summary

Regarding the way the sectoral plan was prepared and is implemented, it can be concluded that Switzerland has successfully integrated ideas related to governance-concepts in their approach to conflict resolution in repository siting. This can be interpreted as an explicit shift away from the decide-announce-defend approach. Even though ensuring 'safety' remains the task of the responsible authorities and the Nagra, a shift away from a purely technocratic approach has taken place due to the high degree of transparency and the strong commitment to dialogue with the interested public and stakeholders.

Still, there is room for improvement. Involving the public in decision-making requires changes in the 'design' of decision-making processes. Institutions need to learn to coordinate such an involvement of stakeholders, which also means being open to their questions and demands. This is not an easy task as different stakeholders have very different ideas and expectations regarding their involvement. Compromises from all sides are necessary in order to make such arrangements work. As the arising conflicts in the implementation of the sectoral plan show, the sectoral plan cannot be considered as a fixed compromise. Rather, certain aspects have to be renegotiated 'on the go'.

Operationalizing governance concepts for the analysis of nuclear waste politics is fruitful if the research focus is laid on the quality of decision-making processes. Using governance concepts can help to qualify decision-making procedures regarding "process quality" and the "quality of arguing" (instead of positioning) in the interaction between authorities and stakeholders as well as "cooperation" in the official decision-making procedures.

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Same, Same but Different - A Comparative Perspective on Participation and Acceptance in Siting Procedures for HLW repositories in France, Sweden and Finland

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Abstract*: This paper compares national approaches in Finland, Sweden, and France. These three EURATOM Member States are in an advanced stage of siting deep geological disposal (DGD) facilities. The procedures in these countries are largely based on voluntarism, but differ in their approach to public consultation as they were based on the so-called staged volunteer process leading to working partnership arrangements between the operator and the hosting communities and veto rights (Sweden), decisions with strong local community support and veto rights for municipalities until the final decision (Finland), and final top-down decision making after consultative processes (débat public) with the affected communities without veto rights (France). This presentation focuses on participation and acceptance issues; it analyses the different modes of governance with diverse conditions regarding transparency, trust, communication, and participation that have been at work. Moving beyond the fact that variance exists with regard to the relevant national institutional, legal, cultural, industrial, and energy frameworks, the authors take evidence from national case studies and look for common patterns.*

1 Introduction

Deep Geologic Disposal (DGD) has been indicated for a long time by a large majority of scientific and technical experts — and endorsed by national governments — as the most adequate way of disposing highly radioactive waste (HLW). However, the paradigm of DGD has started to erode. The ENEF-Working Group "Risks" (2009:3) claimed that "[..] it is nevertheless recognized that there are diverging views in some groups and that there are remaining concerns in the public about geological repositories". Although the advocates of permanent closure of wastes in DGD continue to be numerous, reversibility and retrievability (R&R) is now seen as a more "prudent approach" (NEA/OECD 2012:3). Nowadays, the R&R option is included in several national waste disposal concepts as a possible way to adjust to progress in science and technology and to respond to societal pressures.

People, regardless of their views of nuclear power, realize that radioactive waste needs to be disposed of. In most countries, the debate on siting criteria is no longer confined to the scientific and techno-political spheres, but involves stakeholders, including civil society. However, nuclear waste governance varies from country to country. Various actors and factors, such as the nature of the political and legal systems, formal and informal rules and procedures, political constraints, geographical conditions, technical skills, the stock of knowledge, degrees of public acceptance, and a country's nuclear history can shape siting processes. The way in which competing information and knowledge is processed and put to use by different actors and in different political and cultural contexts plays also an important role. Certainly without knowledge about geological formations and their corresponding morphologies and hydrological conditions no siting selection would be possible. But the process that leads to a selection of clay, salt or granite as host-rock for DGD is hardly only technical and we witness a shift from "geology" to "political geography".

Under the Directive 2011/70/Euratom, Member States are required to establish, implement, and keep updated "national programs" for the management of spent nuclear fuel (SNF) and HLW waste by 2015. The state of implementation of the EURATOM directive at the national level varies considerably. In most Member States legal and institutional frameworks are now in place. Licensing requirements and procedures for site selection and safety criteria have been established, and the responsibilities of stakeholders defined. The major actors involved are: waste producers, waste management organizations, regulatory authorities, civil society, and policy makers at the national, regional, and local levels. Amongst the EURATOM countries, only Sweden, Finland, and France are in an advanced stage of planning and/or implementation of a DGD facility. These are the cases analyzed in the following sections.

2 Common patterns and diversities

The management and governance of spent fuel and radioactive waste in Finland, Sweden and France present a number of similarities, but also marked differences. All three countries have nuclear programs. In Sweden ten reactors are presently in operation, providing about 40% of the nation's electricity. In Finland, there are four reactors which make up approximately 30% of its domestic electricity. Olkiluoto 3 (European Pressurised Reactor-EPR) has been under construction since 2005 and two new reactors were approved by parliament in 2010. France counts 58 nuclear reactors, which provide approximately 75% of its domestic electricity. Moreover a new reactor, Flamanville 3 (EPR type), has been under construction since 2007 and another reactor (Penly) has been approved.

Amongst the three countries there are some similarities in the subdivision of responsibilities between waste producer and the waste management organization. There is also a common functional separation between "operators" and "regulators" in charge of overseeing safety requirements and standards. Differences are most prominent with respect to the ownership structure of the implementing organizations, which are state agencies (France) or in private hands (Finland, Sweden). There are also differences in the host rocks chosen for the geological disposal, i.e. crystalline rock in Finland and Sweden and clay in France. For the system of financing, the 'polluter pays' principle is usually applied. All three countries can count on specific disposal funds which cover the financing of a wide spectrum of tasks, ranging from feasibility studies to decommissioning and to operating costs.

The siting procedures, mostly based on volunteer processes, had unique outcomes in each country: partnership arrangements between the operator SKB and the hosting communities and veto rights (Sweden); decisions with strong local community support and veto rights for municipalities till the final decision (Finland); and final top down decision making after some consultative processes (débat public) with the affected communities (France) without veto rights. In France, the final decision on site selection is subject to the outcome of a public debate; however, it is not binding and has been boycotted by opponents. The sites in Sweden and Finland are "nuclearized," as nuclear facilities exist within the municipality or in the neighborhood, whereas in France the designated site already hosts an underground research laboratory (URL). Moving beyond the fact that variance exists with regards to the relevant national institutional, legal, cultural, industrial, and energy frameworks, we take evidence from national case studies (Mez et al. 2014) and further analyze it to reveal common patterns and differences.

2.1 Finland

Finland has attracted worldwide attention, as it has already started building a DGD. The construction of the Onkalo nuclear waste repository at Olkiluoto started in 2004 and the HLW disposal facility is scheduled to begin operations by 2020.

Onkalo construction is proceeding with very little public debate; the influence of non-governmental organizations has been limited (Lehtonen 2010). In fact, the siting process in Finland has been based on voluntarism. In 1983, a list of 101 potential sites for a repository was prepared and a consultation process with the affected communities was started. This resulted in the identification of five potential sites that "volunteered" to accept more detailed investigations; these were subsequently carried out in three sites. The respective interim reports were released in 1996. Six areas were analyzed for their suitability and a list of four candidate regions was selected. The EIA (environment impact assessment) regulations represented a very important step preceding the licensing process and can be seen as a major driver for initiating participatory planning processes. The need to ensure local acceptance was the major motive for the operator POSIVA's adoption of a more dialogue-oriented strategy (Lehtonen 2010; Kojo et al. 2012).

Finland made use of the so-called Decision in Principle (DiP) process in which municipalities have veto rights. A positive decision by the local municipality and a preliminary safety appraisal of the disposal concept by the regulator STUK were required before the government decided on whether to build the repository (NEA 2010). Cooperation took place between the operator Posiva and the local councils with whom the negotiations were carried out. The final positive decision by the municipal council was taken after the submission of the EIA report by Posiva and the application for the DiP process to the government.

2.2 Sweden

Like Finland, Sweden has gained international attention for having found a solution for the disposal of radioactive waste. Its approach to the governance and management of radioactive waste and the legislation governing it is often seen as a model for other countries. The Swedish state takes the ultimate responsibility for the management of radioactive wastes. However, differently from many Euratom countries, the state has somehow shifted responsibility to the industry. This applies to management and final disposal, but also to the financing of all related activities and regulation. The Swedish concept for RWG thus places the whole responsibility on the owners and operators of the nuclear power plants (Kåberger and Swahn 2014). The nuclear industry has transferred this responsibility to their co-owned radioactive waste company SKB, which is in charge of RWM and the decommissioning of nuclear facilities.

Transparency and public participation are regarded as key elements of the safety of all nuclear facilities (IAEA 2006). The Nuclear Activities Act requests formal consultations with a broad range of stakeholders before a license application can be submitted. Sweden has already implemented Article 10 (on transparency and public participation) of the 2011/70/Euratom Directive and the regulator takes care of consulting stakeholders including environmental organizations while developing the program.

The current status is the outcome of a long process, one which started with nationwide test-drillings in the 1980s that resulted in widespread local protests. Original opposition led to a decision to turn towards a voluntary siting process in which all municipalities in Sweden were invited to host initial 'feasibility studies'. After local referenda blocked potential sites in northern Sweden, the focus shifted to communities already housing nuclear waste facilities. Two municipalities, Oskarshamn and Östhammar, both hosting NPPs, expressed interest and competed with each other to be the preferred site. As stipulated by environmental law since 2004, resources have been made available from the Nuclear Waste Fund to enable environmental groups and other NGOs to participate in the evaluation and public examination of Swedish RWM policy. The SKB decision for Forsmark in the municipality of Östhammar was made in 2009.

2.3 France

In France, the search for a site to host a nuclear waste repository started in the late 1970s. Site investigations conducted in the late 1980s generated intense local opposition, prompting the government to declare a one-year moratorium on the search for a site in 1990. After extensive consultation, the parliament adopted the country's first nuclear law in 1991 (Bataille Law), which reopened the search for a waste solution. The law also marked the beginnings of a more participatory approach to waste management policy. French legislation requires both retrievability of the waste packages and the reversibility of decisions concerning the project. In 1998, the village of Bure (89 inhabitants), situated in the Northeast of the country was first chosen as the site for an underground research laboratory (URL) for deep geological disposal; subsequently it was designated to host the final disposal facility.

The focus of the French participation procedures is on consultative instruments and – as in any infrastructure process – they include an "enquête publique" and a "débat public". The first has an administrative character; the second is considered as more important, but its results are non-binding. The disposal concept was confirmed after a controversial public consultation process organized by the National Commission of Public Debate in 2005-2006. Environmental and citizens groups contended that the law passed in 2006 on the basis of the public debate ignored the fact that a portion of the citizens in the public debate in 2005 were against the DGD facility. In the case of the second *débat public* in May 2013, a grouping of citizens´ initiatives, BURESTOP 55, called for a boycott. Consequently, the debate was continued on the web and comments could only be provided online. Three members of the commission disassociated themselves before the end report of the commission was completed. In Bure there is a Committee (*Comité local d'information et de suivi* (CLIS)), which consists of 90 members (state, regional, district & local governments, MPs, NGOs, environmental groups, Trade Unions, ANDRA, etc.) who work in several commissions, but its influence is limited.

The Cigéo repository remains a controversial project, which on the one hand captivates an economically declining region with potential socio-economic benefits, but on the other hand also generates many doubts and concerns, especially regarding possible negative impacts on local image and economic development (Lettonen 2014). Upon approval from the government and the nuclear safety authority ASN, the waste management operator ANDRA will start constructing Cigéo in 2017.

3 Participation and acceptance in comparison

Looking for a suitable framework to embed the three different cases in and compare them, we make use of the so-called "ladder of participation" (Arnstein 1969), as well as subsequent adaptations. In spite of being almost 45 years old and not immune from criticism, Arnstein´s ladder — once developed to frame citizen involvement in planning processes in the USA — still represents an adequate heuristic tool. Its charm resides mostly in its simplicity. The eight types of participation are grouped under: non-participation (step 1-2); tokenism (step 3-5); and citizen power (step 6-8). The lower rungs are non-participatory and include (1) manipulation and (2) therapy, and are characterized by plans to achieve public support by "public relations approaches". The next step, (3), includes participation, but the information provided is unidirectional and no feedback is envisaged. Consultation (4) follows on the ladder and, in this step, instruments such as surveys, neighborhood meetings, and enquiries are used. This step is considered by Arnstein to be "window dressing". In rung 5 (placation), citizens' advice or plan, but decision makers ultimately decide whether or not to accept their input. It is only in the next stage (6), characterized by partnership, where negotiations are possible and decision-making responsibilities are shared, for example in committees. The next stages, 7 and 8, include delegated power, citizen control and opportunities for power sharing and (co-) governance, but are hardly realistic in the case of RWG.

M. Wiedemann and S. Femers (1993) built upon Arnstein's ladder and considered public participation in risk-related decision-making. Their ladder goes from (a) public right to know, (b) informing the public, (c) public right to object and determine the agenda, (d) public participation in defining interests and recommending solutions, and (e) public partnership in the final decision.

In the EURATOM countries, most of the participation procedures are limited to rung 4 (consultation) or are even at an inferior level of Arnstein´s ladder. Consultative participation processes are the most frequently used instruments; there are hearings where mostly experts, politicians, and NGOs take part and advisory committees where NGOs and other stakeholders have an important role. Amongst the countries analyzed, Sweden could be placed on scales 6 (Arnstein´s ladder) and e) (in the Wiedemann & Femers ladder) as it adopted a partnering approach. The procedure in France could be classified between steps 4 and 5 (Arnstein´s ladder) and between b) and c) on the expanded ladder. Finland has been often considered an example of a good balance between the requirements of fair representation and competent participation. NEA (2004) considers this to be one of the underlying elements of the partnership approach, which is linked to helping to achieve a combination of licensable site and management concepts with host community support and a balance between compensation, local control, and development opportunities. However, it has been observed that in Finland, there is no tradition of radical NGOs and there is a strong trust in local and official experts and a preparedness to let them negotiate agreements in their interest (Kojo et al. 2012, Litmanen 2009, Lehtonen 2010). Moreover, demand for participation appears to be limited. Against this background Finland could be placed between rungs 5 and 6 on Arnstein's Ladder and between b) and c) in Wiedemann & Femers ladder.

We can speak of real, active participation starting from rung 6 of the the Arnstein ladder and c)-e) of the Wiedmann & Femers ladder. In order for the public to participate and exert influence, additional criteria need to be fulfilled. NEA (2010) puts forward criteria that emphasizes the importance of considering local interests, i.e. voluntarism and veto rights, and speaks of local partnerships. These approaches have the potential to increase local acceptance and build trust. Moreover, the stakeholders must be involved at the very beginning of the process; if a participatory process starts late, than these criteria cannot be fulfilled.

Transparency and access to information are a prerequisite in participatory siting processes that are on the higher rungs of the ladder, but the affected stakeholders should also be endowed with sufficient resources. This is the case in Sweden, where since 2005 some environmental organizations have received support from the nuclear waste fund. In France, the CLIS has a budget of 300,000 Euro per year for commissioning independent reviews of the program and hiring experts (NEA 2010). In contrast, in Finland only once, in 1999, have NGOs been financed by the Ministry of Trade and Industry.

According to the Nuclear Energy Agency, key elements of the partnership approach are — apart from voluntarism and right of veto — cooperation with local stakeholders in facility design and implementation and the provision of community benefits (NEA 2010). The provision of community benefits or compensations can manifest in several ways. It provides financial backing for the affected stakeholders to empower the generation of knowledge and expertise (capacity building) and allow citizens to participate. However, it can also hide forms of bribery and in a way, serve as a subtle manipulation (which would bring the process back to the initial rungs of the ladder). For this reason, it is important that compensation is settled only after important aspects, such as safety and security issues, have been sufficiently discussed, and not earlier. All three countries provide compensation and socio-economic benefits to the affected communities.

4 Summary – Lessons learnt

In the case of complex political issues such as RWG, the classical transmission mechanisms between politics and civil society are not enough; citizens want to influence political decisions. The processes leading to a site selection are unforeseeable and conflict ridden; they cannot be encompassed by a narrowly defined planning approach in which problems are defined, analyzed, and solved in consecutive steps. Especially because of changing requirements that are difficult to identify or anticipate and because of the many interdependencies at play, efforts to solve one aspect of the problem (whether societal, technical, or political) may end up creating new problems. Key conditions for an inclusive approach are access to information, early involvement of the affected population and stakeholders, openness for unforeseen results, inclusiveness of the process and compensation. In voluntary approaches, negotiated mechanisms to compensate the affected communities have played an important role. Proper provision of resources for local capacity building, including support for NGOs, is a factor which enhances engagement, increases public confidence, and possibly helps the quality of decision-making. The support of potential host communities, however, cannot exclusively rely on compensation which is expected to be commensurate to offset the potential detriments of the project. Another key element is trust in the institutions and preparedness to delegate negotiation agreements to them, as this is perceived to be in the community´s interest. However, this also implies that local authorities are capable of negotiation in this circumstance, and this depends on the capacity building support that they received in the process. Moreover, this is a not only a political factor, but also an especially influential cultural factor. In France, the population is said to mistrust the political elite; in Sweden and Finland there is a consensual approach.

Although participations strategies are not completely replicable in other countries, lessons learnt from these contexts can help us avoid underestimating the influence of the participatory factor in the siting process. This is of critical importance, as such an underestimation could result in the further hardening of attitudes and lead to deadlock situations.

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SESSION 4
SOCIOTECHNICAL CHALLENGES AND INTERDISCIPLINARITY

Social Dimensions of Nuclear Waste Disposal

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Abstract: *Nuclear waste disposal is a two-faceted challenge: a scientific and technological endeavour, on the one hand, and confronted with social dimensions, on the other. In this paper I will sketch the respective social dimensions and will give a plea for interdisciplinary research approaches. Relevant social dimensions of nuclear waste disposal are concerning safety standards, the disposal 'philosophy', the process of determining the disposal site, and the operation of a waste disposal facility. Overall, cross-cutting issues of justice, responsibility, and fairness are of major importance in all of these fields.*

1 Introduction

The challenge of disposing high-level nuclear waste in a safe and civil way remains an unsolved problem in most countries operating nuclear power plants. In the last years it became clear that there are - besides technical and scientific challenges - also social dimensions involved (Hocke/Grunwald 2006). The high relevance of the social dimension can easily be seen by looking at protest and demonstrations in the case of Castor transports in Germany, or at the severe difficulties in most countries to determine a technically and scientifically appropriate and socially accepted site for nuclear waste disposal.

In this paper I woiuld like to fokus on these social dimensions and will present a structured overview of those dimensions in different stages in the overall process of designing, siting, and operating a nuclear waste disposal facility. The main thesis will be (a) that technical and scientific dimensions on the one hand, and social ones, on the other, are not strictly separated but interlinked; and (b) that promising solutions to meeting the challenge under consideration will need close interdisciplinary cooperation.

2 Social dimensions of nuclear waste disposal

Nuclear waste disposal is a two-faceted challenge: it is a highly ambitious scientific and technological endeavour, on the one hand, and is simultaneously confronted with social dimensions, on the other. In this section I will sketch the respective social dimensions along the 'process chain' of establishing a nuclear waste disposal facility. Relevant social dimensions of nuclear waste disposal are concerning safety standards, the overall disposal approach, the process of determining the disposal site, and the operation of a disposal facility.

2.1 Safety standards

Conceptually, the determination of safety standards is crucial to any approach how to deal with nuclear waste. All the other elements such as determining the overall disposal approach and the search for an appropriate site depend crucially on the safety standards agreed upon. Frequently, determining safety standards is regarded an issue of natural and engineering sciences. Without any doubt these disciplines are urgently needed in this field because they provide the required knowledge about cause/effect relationships, about long-term developments in the geological layers, about migration of radionuclides, about the toxicity of materials and so forth. However, they cannot cover all questions involved. For example, questions such as "how safe is safe enough?" or "how safe must a nuclear waste disposal facility be to ensure a responsible approach to current and future generations?" necessarily include ethical and social considerations and reflections, and are, as soon as it comes to decision-making, in need of a political legitimization. This issue is structurally similar to determining thresholds for an acceptable concentration of toxic materials: while toxicology, medicine, and chemistry are needed to provide knowledge showing the dependency of the toxicity on concentration ethical reflection is required to think about a maximum 'responsible' or an acceptable concentration, and political decision would be required to put this into practice, based on both types of knowledge and orientation.

2.2 Disposal approach

Decision-making between different 'philosophies' and approaches how to dispose nuclear waste (e.g. in a deep geological formation, close to the Earth's surface, or in long-term but still interim technical facilities) is not sensible on the basis of technical criteria alone. It rather needs ethical and social criteria because the decision on the disposal 'philosophy' also will touch justice issues, the distribution of risks among people living today and future generations, and also the type of risks which might be involved. Distributional justice is involved because risks are not abstract numbers but always risks to certain persons, groups or generations - risks have a social dimension in themselves. Also the question of the accessibility of the waste disposal site after fill-in is not only a technical one but influences options of later human intervention which can be regarded positive (intervention in case of unexpected and possibly dangerous developments) or negative (accessibility also for terrorists or in case of rumor?). Thus, the selection of a specific approach to nuclear waste disposal must take into account technical and scientific but also ethical social issues. This is an illustration of a general property of complex technological solutions: they are never value-free.

2.3 Determination of the disposal site

The determination of the site where to establish the waste disposal facility is probably the field with the highest consensus that the social dimension is not only involved but perhaps even dominant. In many countries the established procedures of determining national sites for nuclear waste disposal have failed so far. Protest and rejection based on different reasons are not exceptional but seem to be the rule (except few cases). Defining and implementing a process of determining an appropriate site with respect to safety and other criteria in a civil way remains a challenge in most countries. Participatory and inclusive approaches observing ethical issues of fairness, equity, and transparency are looked for. Trust is an essential issue because results of participatory proceeses will only be taken serious in formal decision-making if there is trust into the group which activcely participated and into the procedure, Knowledge from natural and engineering sciences must not be forgotten or ignored in these processes. In the field of participatory technology assessment experience is available how to integrate expert knowledge in participatory processes.

2.4 Establishing and Operating the waste disposal facility

The operation phase of a nuclear waste facility needs technical knowledge and competence but also appropriate capabilities to deal with its social environment in a civiland open and trustful way. Trust is crucial not only for determining the site but also in operating the respective facility over decades or even longer. This is even more ambitious in the presence of monitoring activities with posible options or necessities for making decisions depending on specific diagnoses based on monitoring results (TATuP 2013).

3 On the necessity of interdisciplinary cooperation

The above-mentioned social dimensions of nuclear waste disposal are part of the challenge to dispose this specific type of waste in a safe and civil way. However, the question might arise for the relation of these dimension to all the other important dimensions such as technology, geology, and hydrology. If these dimensions were disjunct they could be tackled separated from each other. Then, in the final process of nuclear waste disposal the results from considering the various dimensions could be put together as individual mosaic stones.

My plea is that this would be a misleading image. The various dimensions do substantially overlap in spite of the fact that each of them provides specific knowledge. For example the determination of safety standards has an ethical (how safe is safe enough and relative to which criteria?), a social (how will safety standards influence public debate and acceptance?), a technical (how can safety standards be ensured) and a scientific dimension (what can long-term geology and hydrology say to future developments?). These dimensions are not independent from each other but interrelated. Another example is the question for the overall disposal approach. This is obviously a question highly relevant to technology and science but also shows social dimensions, e.g. because the overall approach influences the types of risk involved and the distribution of risks among people today and in the future. Taking these observations seriously demonstrates that integrated and interdisciplinary work is necessary in this field, e.g. to assess different disposal options as is currently being done in the ENTRIA project.

There is also a more general argument in favour of this position. For providing orientation and strategies for nuclear waste disposal research has to provide different types of knowledge from various scientific disciplines and has, thus, to be inter-disciplinary in nature. This knowledge can be categorized in the following way (extended after Grunwald 2004):

- *Systems Knowledge*: Insight into natural, technological and societal systems and knowledge about the relations between them are necessary prerequisites for successful action in meeting the challenge of disposing nuclear waste in a safe and civil way. Scientific disciplines are used to provide explanatory knowledge about relevant systems, in particular in the form of cause/effect-relationships: this knowledge shall explain how specific parts of the world ‘work’, e.g. in the form of models.
- *Prospective knowledge*: The time dimension, in particular the issue of taking over responsibility for future generations, is key to meeting the challenge under consideration. It requires considering possible, probable or desirable future developments, based on today’s knowledge and assessments. A lot of research-based methods such as scenario techniques and model-based

simulation techniques with roots in different disciplines are required to provide comprehensive prospective knowledge.

- *Orientation knowledge*: The appraisal of societal circumstances and developments, of global trends, and of measures must build on ethical, social and perhaps political goals, criteria and targets. Orientation knowledge serves as a 'compass' for assessments, to determine priorities, to find out the direction where to go to and to distinguish between alternative paths of action.
- *Knowledge for action*: In the final analysis, measures for disposing nuclear waste need coherent and integrative action-guiding knowledge for politics and society by elaborating on possible measures and strategies, taking into account the uncertainty and incompleteness of the knowledge produced.

Regarding the sources of these different categories of knowledge it is obvious that (a) interdisciplinary integration must take place. Positive sciences such as geology and hydrology, engineering sciences but also social sciences do provide systems knowledge; normative sciences such as ethics and legal sciences do contribute to orientation knowledge, and action-oriented sciences such as engineering, political and economic sciences deal with measures to reach specific targets.

4 Conclusions

Meeting the challenge of disposing high level nuclear waste in a safe and civil way necessarily involves several social dimensions. They expose themselves differently in differnet stages along the process from determining safety standards via fixing the disposal approach and the siting process up to establishing and operating a disposal site. The issues of justice (just distribution of burdens and risks), of trust (e.g. by transparency) and of fairness (concerning the opportunities to intervene into decision-making procedures) are of highest importance and are cross-cutting items over the whole process chain (Grunwald 2010). These are not merely additional to natural and engineerinmg sciences' inputs but, rather, scientific, technical and ethical aspects, dimensions and issues often are unseparably intertwined (e.g. in determining safety standards). This situation makes close interdisciplinary cooperation necessary.

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Inclusive Assessment in a Site-Selection Process – Approach, Experience, Reflections and some Lessons beyond Boundaries

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Abstract: *Nuclear waste disposal indisputably is a controversial socio-technical issue in most societies widely using nuclear technology. All the more it is pivotal to proceed in a comprehensive, transparent and participative manner. The contribution suggests fundamental rules to follow and confronts them with the currently ongoing site-selection process in Switzerland. Finally it draws some lessons for the audience.*

1 Basic rules and procedure to follow

The nuclear community recognizes that the long-term safety of repositories "is not … a rigorous proof of safety … but rather a convincing set of arguments" (NEA 1999). It has, however, been difficult to "live" its socio-technical nature (NEA 2013). Albeit the waste problem is driven by technology and, indeed, a technological constraint, in the end, it has to be solved by society. Building upon the defense-in-depth principle, the concept of integral, technical and societal, robustness was developed (Flüeler 2002/06). A system is "socially robust" if most arguments, evidence, social alignments, interests, and cultural values lead to a consistent option (Rip 1987). The concept attempts to consider technical and social issues in parallel, as to force players (both from the technical community and society) to keep in mind (and strive at) an integrative "solution", satisfying technical (passive safety) and societal needs at the same time. This combination sets it apart from other approaches, either purely technocratic along the conventional decide-announce-defend line or the voluntaristic policy some national programs have reverted to in the face of the failed technocratic approach (e.g. Sweden, Japan, UK, USA) or any unsystematic negotiated mixed versions. Based on international experience (Flüeler 2002/05/06/14), we propose a 3-step approach for a site-selection procedure, followed by a proposal for an integrative assessment framework – in Chapter 1 we outline the concept, in Chapter 2 we give pertinent examples.

1.1 Step 1: Discuss – comprehensive societal discourse

First, discuss the issue from all conceivable angles. "Involvement of stakeholders" cannot mean to call for as many individuals but for as many perspectives as possible to systematically incorporate all relevant facets in the multi-dimensional discourse. The aim is to lay all pertinent aspects (values, norms, context, evidence …) open, to successfully "close" issues, and proceed to the following phase. A "socially robust" procedure must, per se, allow for some flexibility and adaptability to an evolving social context, particularly given the long timeframe involved (decades to find a suitable site, to license, build, operate, and duly close it). "Closing" steps, however, is crucial in order to avoid starting all over again.

This long-time process has to be overseen, e.g. by a widely credible and trustworthy body. In 2002, Flüeler suggested a "National Council for the Safe [and Secure] Governance of Radioactive Waste" as the guardian of the process, the Swiss expert committee EKRA foresaw a "Disposal Council". It should be pluralistically composed, independent of the industry yet knowledgeable and not driven by daily politics.

1.2 Step 2: Decide – "common ground" in goals and stepwise strategy

The goals have to be prioritized so as to adopt the stakeholders' respective responsibilities. Conflicting goals exist (see, e.g., ENTRIA 2014). The concept of "sustainability", a complex goal indeed, encompasses protection of, and leeway for, future generations. Transferred to nuclear waste, both passive safety and "active" control or surveillance need due care and attention in parallel. No consensus will be reached "at heart", in the stakeholders' core beliefs. In the 1980's, Carter called in to find a "common ground", yet without specifying (Carter 1987, 427). Society must agree on three levels: 1. Problem recognition (waste exists, problem to be tackled, eventually "solved", at least set on track to be solved); 2. Main goal consensus (degree of protection and intervention); 3. Procedural strategy ("rules of the game").

1.3 Step 3: Implement – start program and prepare long-term knowledge basis

In view of a successful transfer of knowledge, it is vital to explore contextual issues and tacit/implicit knowledge – they determine the degree of societal understanding of the eventual disposition system. Unless the rationale of conceptual reasoning is appropriately handed over to next – technical, political and societal – generations, the entire undertaking is bound to failure (Flüeler 2005).

The rules and criteria of site-selection procedures have to be consented to before the start (see 1.1/2.1) and adhered to during the process. Revisions should undergo a careful review and be consented to. A clear distinction between implementer and regulatory bodies is vital. The regulators must establish a platform for inclusive knowledge generation, based on a (pre)defined set of criteria. This necessity to integrate different requirements, the step-by-step approach, the chance of "institutional constancy", and the special "national" task of the issue call special attention to the role of the authorities (NEA 2003). Issues like regulatory capture, expert blocking, or technological lock-ins have to be duly considered (see 1.4).

1.4 Accompanying policy evaluation

A periodic policy evaluation is vital to assess whether a program is on track (see e.g., Leeuw et al. 1994). Some respective criteria are suggested in Table 1.

Table 1. Criteria and respective *attributes (Italics)* for an appraisal of governance and other theoretical concepts (own tabulation, draft).

Area	**Approach/concept**			
	„Good" governance	**(Regulatory et al.) capture**	**Safety culture**	**Path dependence, lock-ins**
References	UNDP 1997	Carpenter & Moss 2013	IAEA 1991	Crouch 1993, Arthur 1989
A. Formal structure	Legitimation	Asymmetry	Continuous learning	
	Legislation	*Research & development*	*Education, permanent training*	
	Participation in ...	*Resources (staff, financial)*	...	
	Players	*Competence(s) and experience*		
	...	*Expert blocking...*		
B. Under-standing of roles	Division of roles			Open decision making
	Tasks in the Sectoral Plan (program)			*Comparison of options*
	Strategic planning/leadership ...			...
	Responsibility ...			
C. Internal structure s	Transparency/ accountability		Failure culture	
	Justification of decisions		*Openness of communication*	
	Framework and respective guidelines		*Trust*	
	Controlling		...	
	Quality management			
...	*Reviewing...*			

2 Application to context and insights from reality

2.1 Step 1, Discuss: evolving from a linear to a more pluralistic model

Having learnt from the failures at Wellenberg, the planned site für low- and intermediate-level waste, where the cantonal voters rejected an initial as well as a developed project, the Swiss Confederation started an extensive consultation about a site-selection process to be set up. The underlying concept document was reviewed in several rounds by technical and political actors, in fact, it was a comprehensive stocktaking in the policy field (BFE 2008, 5), even though it was not as widespread as the Canadian dialogue called "Choosing a way forward" (NWMO 2005). The ongoing so-called Sectoral Plan for deep geological repositories contains a thorough so-called "regional participation" of potential siting regions. When setting up "regional conferences" in each of six geologically suitable areas it was given special attention to consider the entire societal spectrum, from political parties to churches, though females and youngsters are still underrepresented (Planval 2014). The conferences have a consultative character; to give them sufficient legitimation (and stability) the quorum of elected members of municipal councils was set to be 50 percent. Experience so far shows that the safety-first paradigm is accepted on all levels, national, cantonal, regional, communal, but regional stakeholders have been able to have a say, esp. on the spatial-planning topics (see 2.4).

The idea of an oversight body was somewhat embraced by the Swiss federal authorities (BFE 2008) when starting the ongoing site-selection process. The six-member "Nuclear Waste Management Advisory Board" is pluralistically composed, with one representative of the nuclear industry; others are independent experts from the geological disposal community, ethics, and media. It is chaired by a member of the Council of States (Senate). The anti-nuclear NGOs refused to collaborate but they do participate in Sectoral-Plan bodies. Even though its mandate is to "offer views from an outside perspective" and to "help identify process risks and barriers to progress" (BFE 2014, website) the board has not become clearly visible so far. At any rate and overall, however, the traditional strategy of linear decision making (by the authorities and proponents) is tentatively superseded by a dynamic and integrative, pluralistic model.

2.2 Step 2, Decide: passive safety *and* retrievability in a phased and flexible process

Way back in the 1970's the Swiss national electorate, by a two-third majority, voted for a Federal Decree on the then Atomic Energy Act, according to which "the permanent and safe final disposition and disposal of the ... radioactive wastes" have to be "guaranteed (Federal Decree 1979). The primary goal of radioactive waste management, passive safety in a geological environment, was confirmed in the currently valid Nuclear Energy Act of 2005. The notion of final and domestic disposal has since overwhelmingly been endorsed in numerous national and regional surveys. One may, therefore, conclude that there is consensus in the Swiss society on geological disposal as the way to go. The national Parliament passed the revised respective act prohibiting – upon the Cantonal vetoes on Wellenberg – a final vote by an eventual host canton but stipulating that "the Department [Ministry] gives a share to the host canton as well as to the neighboring cantons and countries in direct vicinity with regard to the preparation of the general licensing decision" (Federal Nuclear Energy Act, art. 44). The option of a – final – national referendum on the ultimately chosen site is foreseen by this law.

With regard to the technical concept, a pluralistically composed expert group was initiated in 1999. It proposed the concept of "monitored long-term geological disposal", an extension of the traditional concept of final disposal by, to a certain extent, integrating controllability/monitoring and retrievability (EKRA 2000). These aspects were adopted in legislation, although clearly as secondary to passive safety. The mentioned Sectoral Plan for site selection foresees three phases and is evidently in line with the U.S. NRC's "adaptive staging" (NRC 2002), progressively become common in the community (NEA 2004).

2.3 Step 3, Implement: well started – but prepared for the long term?

After more than five years of Sectoral Plan, in the middle of phase 2, one may say that the site selection process is and continues to be long and tough (longer than initially planned by the Office of Energy), but it is ongoing, there are no major dropouts, critical comments were recognized, the work has been improved. The players recognize each other to contribute their part (Flüeler 2014). They have to plan for the real long-term, also staff wise and institutionally, and need persistence and stamina. Whether they are prepared for the program's longevity remains to be seen: find the "right" location, build, operate, close, post-monitor the site, and all this in a positive national to local embedding.

2.4 (Instead of) Policy evaluation: a quick glimpse on SWOTs

Technical assessments are continually undertaken in the Swiss site-selection procedure (e.g., Leuz & Rahn 2014). But full-fledged policy evaluations, whether continuous or periodic, are rare, in fact just punctual up to this point (like Planval 2014, on regional participation). To give a current impression at least, the presentation reflects on a quick-and-dirty profile of strengths, weaknesses, opportunities and threats recently rendered by members of major players (SWOT analysis, Kotler et al. 2010). Strengths are: overall, the Sectoral Plan is a suitable instrument with adequate flexibility; the players stick to their given roles; safety first is held up; transparency and traceability are practiced. Weaknesses: asymmetry of players; concerned regions without right of decision; motivation sinking; frustration rising. Chances: freedom of action. Risks: clash of interests; diverse levels of state, introduction of veto right for stakeholders.

3 Lessons learned

To consider both technical and social issues needs an inclusive, systematic and participatory approach to single out goal priorities (presumably with safety first). Setting up a respective process is a prerequisite to proceed in site selection. It is essential to have a (national) lead agency in conjunction with a clear division of roles among the players, rules of the "game" and criteria to judge. The complex and long-lasting procedure necessitates extensive resources on all sides and of all types over time.

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The Challenge of Interdisciplinarity: First Steps towards a Joint Working Approach – the ENTRIA Project

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Abstract: *ENTRIA ("Disposal Options for Radioactive Residues: Interdisciplinary Analyses and Development of Evaluation Principles") is a joint research project funded by the German Federal Ministry of Education and Research BMBF) and carried out by 12 departments from German universities and major research institutions and one partner from Switzerland. The scientists participating in ENTRIA represent natural sciences, civil engineering, philosophy, law, social and political sciences, and technology assessment. Recognizing that all these disciplines need to interact when radioactive waste management is concerned, the project aims at investigating and developing evaluation principles for three options to manage especially high-level radioactive waste: Deep disposal without retrievability provisions, emplacement in deep formations with monitoring and retrievability, and prolonged surface storage. While ENTRIA performs both disciplinary and interdisciplinary research, the paper focusses on the latter.*

1 Introduction

Obviously, radioactive waste management (RWM) concerns society as a whole and therefore needs more than technological and natural science research. BMBF (The German Federal Ministry of Education and Research) is funding, inter alia, the joint research project ENTRIA in order to support the development of interdisciplinary research approaches as well as professional (academic) education and knowledge management.

ENTRIA ("Disposal Options for Radioactive Residues: Interdisciplinary Analyses and Development of Evaluation Principles") is carried out by 12 departments and institutes from German universities and major research institutions and one partner from Switzerland:

- Niedersachsen Institutes of Technology (NTH, coordinator) with
  - Institute of Disposal Research (Clausthal, 2 departments)
  - Institute of Mineral and Waste Processing, Waste Disposal and Geomechanics (Clausthal)
  - Institute of Law (Braunschweig)
  - Institute of Foundation Engineering and Soil Mechanics (Braunschweig)
  - Institute for Building Materials, Concrete Construction and Fire Protection (Braunschweig)
  - Institute for Radioecology and Radiation Protection (Hannover)
  - Institute of Materials Science (Hannover)
- Karlsruhe Institute of Technology (KIT) with
  - Institute for Technology Assessment and Systems Analysis
  - Institute for Nuclear Waste Disposal

- Freie Universität Berlin with
  - Environmental Policy Research Centre
- Kiel University with
  - Chair of Philosophy and Environmental Ethics
- risicare GmbH (Switzerland, an independent expert group with experiences in Swiss risk governance and advanced technologies)

The scientists participating in ENTRIA represent natural sciences, civil engineering, philosophy, law, social and political sciences, and technology assessment. Recognizing that all these disciplines need to interact when radioactive waste management is concerned, the project aims at investigating and developing evaluation principles and knowledge about "context structures" for three options to manage especially high-level radioactive waste:

- Deep disposal without retrievability provisions
- Emplacement in deep formations with monitoring and retrievability
- Prolonged surface (or near-surface) storage

The scientists involved are well aware of the fact that this is not the first time that such options are evaluated, and many of them have developed their preferences over the years and decades of their professional life. Though there is a number of synoptical studies (like EKRA 2000, AkEnd 2002, NWMO 2005, CoWRM 2006, Streffer et al. 2011), screening literature, documents and media showed that the conceptual arguments are not well published. Therefore, discourses with the general public, mass media and the interested public are not as differentiated as needed. There is apparently a lack of systematic description of these three key options, their scientific comparison, and of communication on basic features, advantages and drawbacks of these options. This will not spare society the decision(s) to be made, but will hopefully make such decisions better informed. Apparently, the relevant articles of the recently adopted German Site Selection Act (StandAG) are based on similar considerations.

It has been decided to address these options because they are the only ones for which technical concepts exist. All other „solutions" (e.g. partitioning & transmutation, emplacement in subduction zones, launching to outer space) can either be excluded due to legal considerations or because of unacceptable risks, or would need considerable development time (decades and more) and would therefore require (prolonged) interim storage as first step. Consequently, these three options are the ones available for now. However, the options are not considered as equally ranking "endpoints" of an RWM strategy: storage is seen as an interim step for a prolonged, but limited timeframe after which an (yet unknown) endpoint still would be needed. The same would apply if the waste was retrieved from a deep repository. Thus, amongst the options considered only a safely decommissioned deep repository (be it constructed with or without retrievability provisions) can be considered an endpoint.

2 ENTRIA structure

In order to facilitate interdisciplinary research and cooperation, the project is organized in three so-called vertical projects, each addressing one of the management options and all mainly treated by natural scientists and civil engineers. In addition, overarching aspects such as "Synthesis, Coordination and Communication", "Technology Assessment and Governance", "Ethical and Moral Substantiation, Legal

Prerequisites, and Implications", and "Interdisciplinary Risk Research" are addressed by interdisciplinary so-called transversal projects (Fig. 1).

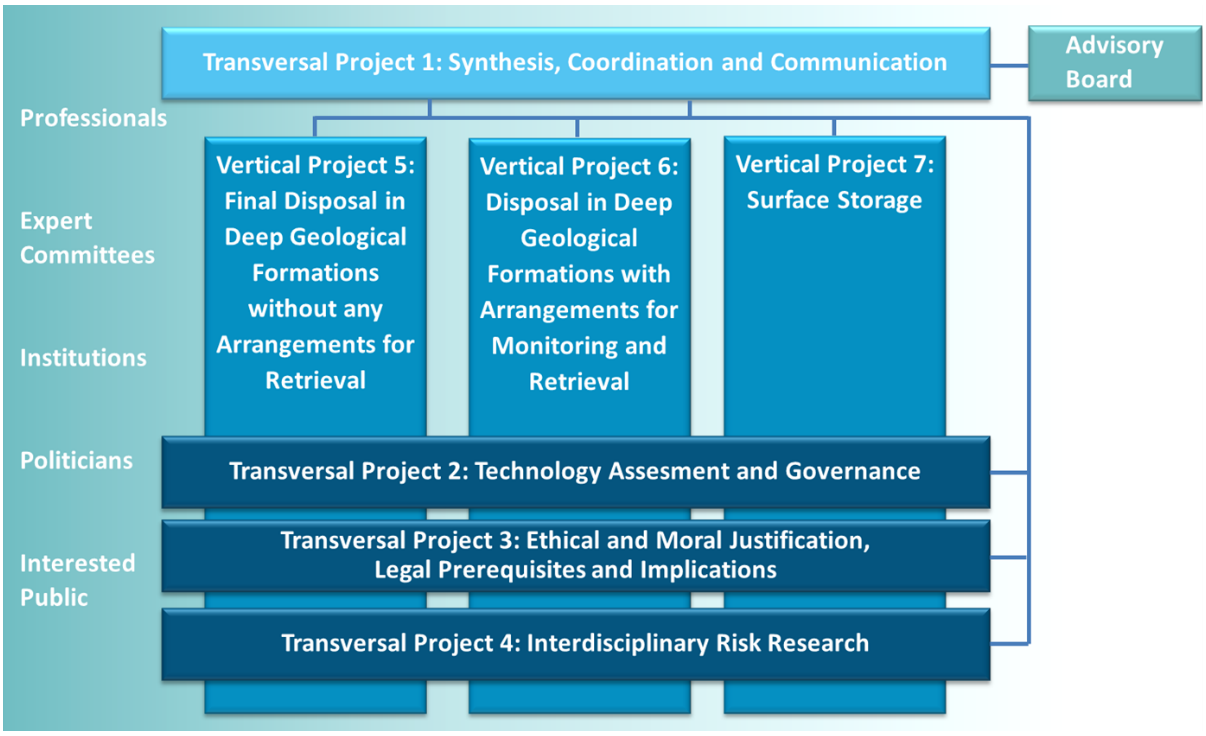

Fig. 1. ENTRIA structure

3 First steps towards, end experiences with, interdisciplinarity

For ENTRIA, sound *disciplinary* research forms an indispensable basis for interdisciplinarity. Topics addressed in disciplinary research include:

- Radionuclide source terms and migration
- Numerical simulation of safety relevant processes
- Monitoring of waste containers
- Safety case for prolonged surface storage
- Comparison of radiological risks / exposures
- Citizens panel as an instrument for discourse analysis in ethics
- Situation analysis from the viewpoint of communal decision makers and local population
- Forms, constraints and dilemmas of RWM governance
- Requirements and implications regarding constitutional law

During the first year of project work, major challenges concerning interdisciplinarity became evident and means to address them were developed. Obviously, all disciplines involved need to learn the basics of the respective other disciplines. Formats for interdisciplinary cooperation between the research teams and mutual learning such as interdisciplinary lecture series held by senior scientists, joint field trips, as well as presentation and workshop formats for junior scientists have been put in place. This is not only meant to serve the project work itself, but also to educate scientists with disciplinary excellence but also understanding of the basics of other disciplines.

Terminology and semantics are crucial to understanding each other. Seminars, workshops and discussant panels addressing terms such as “wicked problem”, ”confining rock zone”, “good governance”, “retrievability” as well as web-based collaborative formats (wikis) for fora and glossaries are being implemented.

Furthermore, the project is increasingly working on joint interdisciplinary publications, PhD theses, and events for the interested public held by interdisciplinary teams. As a very first step towards an interdisciplinary synthesis, the senior scientists from all teams published a memorandum naming the specific challenges and target conflicts of importance when deciding about management options. The drafting has been carried out in an iterative process involving 20 individuals from all ENTRIA partners. After having decided about scope, topics, and structure, chapters were developed by drafting teams (typically 2 individuals – one “technician”, one “non-technician”). In several iterations chapters were merged, redundancies were removed, controversial issues were resolved and the language was harmonized by small editing teams with moving membership. These iterations involved bi- and multilateral exchanges amongst the authors. Finally, the memorandum was approved by all authors. It was felt that the process greatly helped developing a common terminology and mutual understanding about issues and topics. Under the headlines

- Problem description
- RWM strategies and facility types
- Risk and safety
- Procedural fairness
- Societal and technological innovation

the memorandum identified areas of tension and conflict at stake, including

- Burdens on future generations – flexibility
- Accessibility – safe confinement
- Delay process – urge process
- Interests of society – particular interests
- Classic representative democracy – parliamentary democracy with strong deliberative elements
- Criteria constant over time – fallibility and need for revision

When addressing the objective of deriving evaluation principles, the issue of risk, its conceptions and definitions, assessment, perception and management is of particular interest. ENTRIA tackles the risk issue in a dedicated transversal project (see Fig. 1), which cooperates with all other, but particularly with the vertical projects. In interdisciplinary meetings, a set of propositions or premises about the handling of the risk issue in ENTRIA has been agreed upon, including the following:

- Risks can be analyzed and assessed.
- Risk analysis integrates technical and societal aspects.
- Part of risk analysis is to disclose the limits of the analysis.
- Risk assessments must be based on good reasons and exposed transparently.

It is planned to structure further work based on the development of “risk landscapes” i.e. graphical means to provide overview, identify issues and uncover normative aspects. Exemplified and illustrative considerations of sections of these landscapes can then be carried out using reference concepts. E.g., ENTRIA's reference concept on deep emplacement with retrievability and monitoring involves a repository to be dimensioned appropriately to allow retrieval, monitoring drifts above the emplacement areas, and backfill and sealing the individual emplacement drifts after waste emplacement. The infrastructure area and the shaft will be kept open. The host-rock specific timeframe for which this will be possible will be derived based on safety considerations taking into account geomechanics, degradation processes etc. The hypothesis is that – assuming that a decision is taken *not* to retrieve – the complete repository will be backfilled and sealed at the latest at the end of this timeframe. From then on, only waste recovery will be conceivable. Of course, choosing a reference concept implies an illustrative and exemplary character of the related considerations: Many statements will apply to this specific reference concept rather than to „retrievability in general".

4 Conclusions for interdisciplinary work

Challenges when working beyond disciplinary boundaries include widely varying terminology and semantics as well as widely varying working and communication cultures. Sound disciplinary work is a basis and starting point for interdisciplinary activities. Another prerequisite for the scientists involved is knowledge about the basics in the respective other disciplines. ENTRIA successfully tested a number of formats to address these challenges. Cross-disciplinary education formats are especially necessary and helpful. Working on joint projects such as journal articles, books, PhD theses, and jointly organized and held events (presentations, summer schools, podium discussions) turned out to be most effective. ENTRIA is moving towards an interdisciplinary analysis leading to the development of evaluation principles for the key options and knowledge about relevant "context structures" for radioactive waste management.

Acknowledgements

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Reflecting Socio-Technical Combinations in Radioactive Waste Management – Results from the InSOTEC European Research Project

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InSOTEC is a three-year collaborative social sciences research project funded under the European Atomic Energy Community's 7th Framework Programme FP7. The project aims to generate a better understanding of the complex interplay between the technical and the social in the context of geological disposal of radioactive waste. In doing so, InSOTEC has moved beyond the social and technical division that is frequently being found in this context by

- investigating the consideration of social sciences and the recognition of socio-technical combinations in research programs on geological disposal,
- analyzing the socio-technical entanglement in selected contexts like siting, reversibility and retrievability, demonstrating safety and technology transfer on the basis of case studies, and
- exploring the integration of diverse stakeholders in technology oriented networks.

The analyses reveal that activities in the context of geological disposal, whether related to research, planning, siting etc., rather support the divide of social and technical aspects than fostering the consideration of their entanglement. Reasons identified for this are manifold. The wish to reduce complexity by focusing stakeholder involvement on social questions and fixing the technical part "when acceptance is reached" is only one of them.

However, the analyses also show that over the long timescales of repository planning and implementation, robust management strategies must provide the flexibility to adapt to both technical and social developments and demands. Understanding the socio-technical interplay and creating structures for its consideration provides the basis for dealing with this challenge.

This presentation will focus on the main findings of the InSOTEC project with regard to the consideration of socio-technical combinations in practice. These insights are currently under development and will be finalized at the end of the project in June 2014. We will reflect on the "added value" associated with the notion of "socio-technical" as developed in InSOTEC, e.g. in terms of getting better insights, being more reflexive, being more open etc. Based on this we will discuss how socio-technical approaches can be facilitated, focusing on inter- and transdisciplinarity, public participation and the constructive role of conflicts. We are furthermore going beyond the experience gathered in the context of nuclear waste management as it is becoming apparent that socio-technical combinations are inadequately covered by current practices in this field. Insights will be drawn from methods used in transition processes namely innovation governance, transition management and adaptive management.

Acknowledgements

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Nuclear Waste and Hazardous Waste in the Public Perception

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The disposal of nuclear waste has gained attention of the public for decades. Accordingly, nuclear waste has been a prominent issue in natural, engineer and social science for many years. Although bearing risks for todays and future generations hazardous[1] waste in contrast is much less an issue of public concern.

In 2011, we conducted a postal survey among Swiss Germans (N = 3.082) to learn more about, how nuclear waste is perceived against hazardous waste. We created a questionnaire with two versions, nuclear waste and hazardous waste, respectively. Each version included an identical part with well-known explanatory factors for risk perception on each of the waste types separately and additional questions directly comparing the two waste types.

Results show that basically both waste types are perceived similarly in terms of risk/benefit, emotion, trust, knowledge and responsibility. However, in the direct comparison of the two waste types a complete different pattern can be observed: Respondents perceive nuclear waste as more long-living, more dangerous, less controllable and it, furthermore, creates more negative emotions. On the other hand, respondents feel more responsible for hazardous waste and indicate to have more knowledge about this waste type. Moreover, nuclear waste is perceived as more carefully managed.

We conclude that mechanisms driving risk perception are similar for both waste types but an overarching negative image of nuclear waste prevails. We propose that hazardous waste should be given more attention in the public as well as in science which may have implications on further management strategies of hazardous waste.

[1] Hazardous waste is non-ionizing toxic material which cannot be further used. Accordingly, this material has to be isolated from men and environment e.g. in exploited mines. Hazardous waste accumulates from industry, business and households. Examples are heavy metals, batteries, filter ashes etc.

POSTER SESSION
1 REPOSITORY CONCEPTS IN DIFFERENT HOST ROCKS AND SAFETY ANALYSES

Design and Optimization of a HLW-Repository in Salt Formations – Results of the Preliminary Safety Analysis for the Gorleben Site (VSG)

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ABSTRACT: *In summer 2010 the German Federal Government launched a preliminary safety analysis to assess whether the salt dome at Gorleben is suitable to host all heat-generating radioactive waste generated by German NPPs. On the basis of the "Safety Requirements Governing the Final Disposal of Heat-Generating Radioactive Waste as at 30 September 2010" (BMU 2010) a repository concept was developed which was optimized to a certain extent during project evolution. The repository design was tailored to the specific geologic environment at the Gorleben site and served as the main technical basis for the site-specific preliminary safety analysis (VSG). For comparison, two different emplacement strategies were considered. One strategy considered that all of the waste packages will be disposed of in horizontal drifts within a mined geologic repository, while the other strategy considered that all waste packages will be disposed of in deep vertical boreholes drilled into the floor of the repository. The fundamentals, the design approach, and the resulting repository designs for two different emplacement strategies (GRS 2011) are presented in this paper. The designs took into account an updated set of fundamental data regarding the amounts and types of expected heat-generating waste and the documented results of the exploration of the Gorleben salt dome. The designs aimed to transfer the new "Safety Concept and Safety Assessment Concept", i.e. the methodology on how to demonstrate operational and long-term safety, into technical solutions for repository components, systems, and processes.*

1 Introduction

The revised Atomic Energy Act of June 2011 stipulates a phase-out of nuclear energy production in Germany by the end of 2022. Prior to this revision – in summer 2010 – the German Federal Government launched a preliminary safety analysis to assess whether the salt dome at Gorleben is suitable to host all heat-generating radioactive waste generated by German NPPs. This safety assessment includes a repository concept which was optimized to a certain extent during project evolution. The repository design was tailored to the specific geologic environment at the Gorleben site and served as the main technical basis for the site-specific preliminary safety analysis. This preliminary safety analysis took into account the "Safety Requirements Governing the Final Disposal of Heat-Generating Radioactive Waste as at 30 September 2010" (BMU 2010) for the very first time. The resulting repository designs for the two main emplacement strategies took into account an updated set of fundamental data regarding the amounts and types of expected heat-generating waste and the documented results of the exploration of the Gorleben salt dome. The designs were to transfer a new safety and verification concept (GRS 2011), i.e. a methodology on how to demonstrate operational and long-term safety, into technical solutions for repository components, systems and processes.

2 Repository Design

2.1 Objectives

The primary objective of the preliminary safety assessment for the potential site Gorleben was a traceably documented prognosis of the suitability of the site. This requires, among others, the development of an optimized repository concept. For this purpose, two main repository concepts were considered. Variant 1 comprises the emplacement of all heat-generating radioactive waste (spent fuel and vitrified waste) in self-shielding waste containers (POLLUX® casks) in horizontal drifts. In addition – for comparison only – the emplacement of all heat-generating radioactive waste in transport und storage casks (CASTOR®) in horizontal boreholes was considered. Variant 2 consists of the emplacement of all heat-generating radioactive waste in multi-purpose cylindrical canisters in deep vertical boreholes. As an option, the emplacement of a certain amount of non-heat-generating radioactive waste was considered (in horizontal emplacement chambers). For both variants (emplacement in drifts/emplacement in vertical boreholes), the respective final technical designs were elaborated and documented in a final report (GRS 2012).

2.2 Technical Design of the Repository Mine

The technical designs comprise first the selection and description of waste-specific (spent fuel/reprocessing waste) waste packages for both strategies. Based on the expected waste inventory (10,445 tons of heavy metal resulting from 34,430 spent fuel elements and 8,141 canisters of waste from reprocessing (CSD-V, CSD-B and CSD-C), 3D thermal calculations were carried out to determine canister, drift, and borehole distances. The calculations showed that both disposal concepts (drift disposal and borehole emplacement) for the heat-generating waste meet the 200°C design criteria. The only difference is the smaller horizontal foot print in the case of borehole disposal. Compared with previous calculations (EUR 2004) the thermal-mechanical calculations used a more comprehensive and refined set of parameters (e.g. higher stress levels and faster creep classes for the host rock) which led to a substantially faster enclosure of the waste canisters or casks. Thus, the results provide suitable design parameters for the layout of emplacement drifts and fields and the entire repository, including infrastructure areas and drifts for the transportation of waste packages and excavated rock salt material. The respective repository design approaches comprise two steps; first a conceptual design was developed, followed by a technical design, which includes suggestions for optimization. The final design results for both main variants are as follows:

Variant 1: Disposal of POLLUX® casks in horizontal drifts of the repository mine

The total inventory of heat-generating waste will be disposed of in heavy (weight max. 65 metric tons [Mt]), self-shielding POLLUX® casks containing the fuel rods of disassembled spent fuel elements or waste from reprocessing in horizontal drifts of the salt mine. A minor quantity of remaining structural parts from the conditioning of spent fuel elements will be disposed of in cast iron containers. The results of the thermal calculations for cask and drift distances have been transferred into a repository layout, as shown in Figure 1 for the drift disposal concept. In total, the north-eastern part of the repository – adjusted to the assumed geologic structure of the Gorleben salt dome at the emplacement level of 870 m below surface – will have a length of approximately 4 km and a width varying between 300 m and 700 m.

Figure 1 also shows a sketch of the POLLUX® cask and a photograph of the test set-up for the full-scale canister drift emplacement demonstration tests successfully performed in the 1990s.

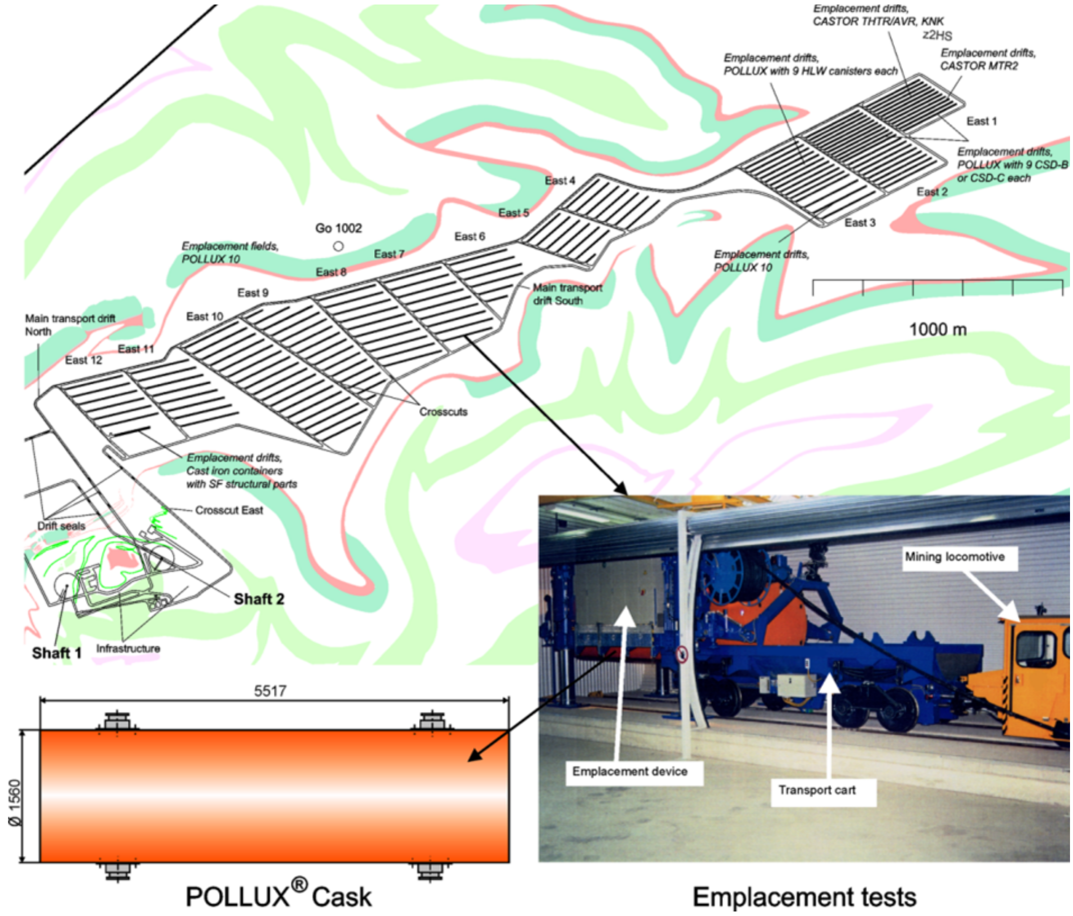

Fig. 1. Repository design for the emplacement of all heat-generating spent fuel and vitrified waste in horizontal drifts at a level of app. 900 m below surface in the center of the Gorleben salt dome

Variant 2: Disposal of canisters for fuel rods and vitrified waste in vertical boreholes

Under Variant 2, the total inventory of heat-generating radioactive waste will be disposed of in canisters lowered into 300-m deep, lined vertical boreholes drilled from the 870-m level of the repository. The canisters (weight 5.2 Mt) contain fuel rods, vitrified waste, or structural parts from the conditioning of spent fuel assemblies. The results of the thermal calculations took into account the heat transfer into the surrounding rock salt at a depth of more than 300 m. The results provided borehole and drift distances, thus again guarantying that the temperature limit of 200°C at the contact between the liner and the salt will not been exceeded. The corresponding repository layout is shown in Figure 2. In total, the north-eastern part of the mine – adjusted to the geologic structure of the Gorleben salt dome at the emplacement level of 870 m below surface – will have a length of approximately 1 km and a width varying between 400 m and 800 m. The emplacement of non-heat-generating waste was considered as an option in the south-eastern part of the mine.

Figure 2 shows a sketch of a vitrified waste canister and a spent fuel canister and a photograph of the test set-up for the full-scale canister emplacement demonstration tests in vertical boreholes successfully performed in 2008 and 2009.

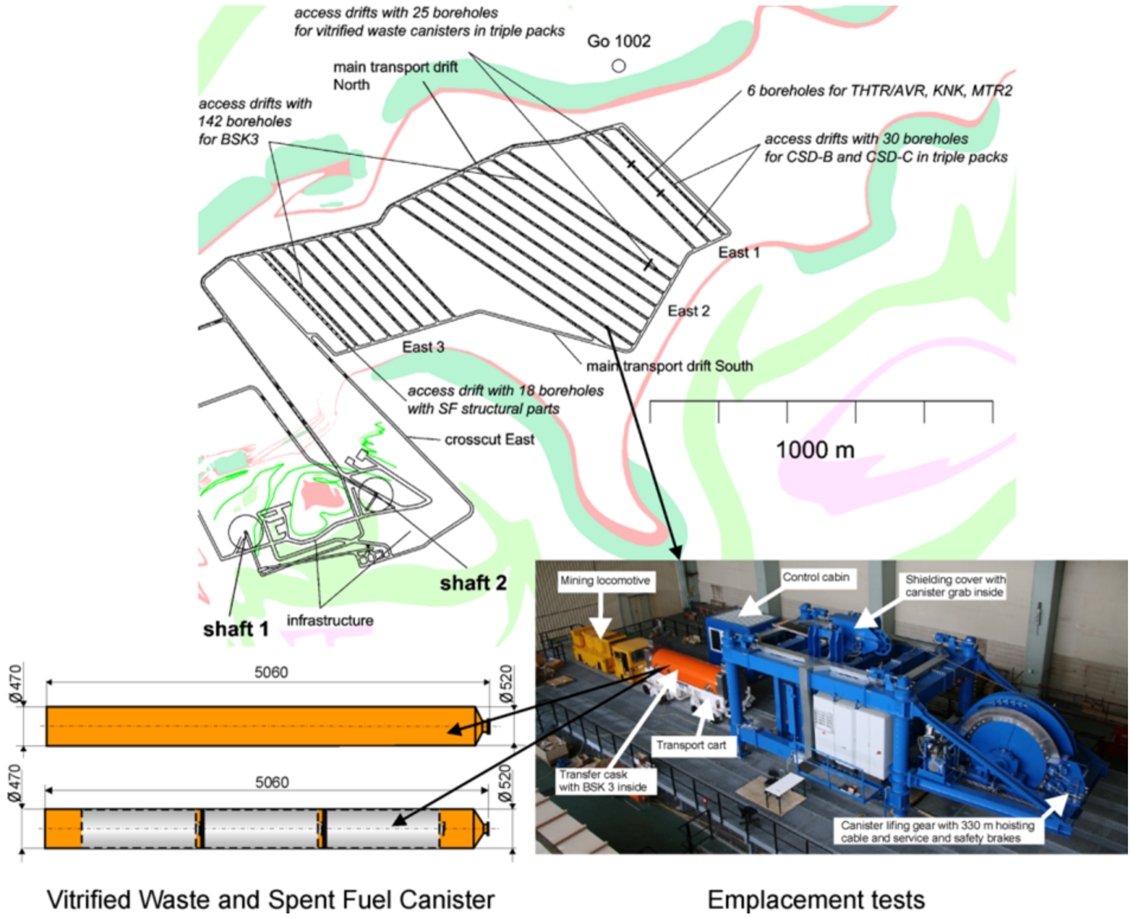

Fig. 2. Repository design for the emplacement of all heat-generating waste in lined vertical boreholes drilled into the center of the Gorleben salt dome from the emplacement level of the repository at app. 900m below surface

For comparison purposes the disposal of all heat generating waste by means of transport and storage casks was considered as well. This emplacement concept comprises the emplacement of heavy casks (up to 160 t) into single horizontal boreholes drilled perpendicular to the emplacement drift. Due to the design temperature of 200°C the appropriate draft repository design shows that the footprint is more or less the same as that of the drift emplacement concept using POLLUX® casks.

3 Retrievability

The repository design took into account the retrievability requirements in accordance with the new Safety Requirements (BMU 2010). As a boundary condition for the repository design, it was considered that a decision to retrieve waste canisters would comprise all waste canisters disposed of and that the evolution of the repository in a reference scenario does not have an impact on the integrity of the waste containers. Retrievability in the case of the drift disposal concept does in principle imply a reversion of the emplacement process. New access drifts will be excavated in parallel to the emplacement drifts and the remaining material above and in between the containers will be removed. Eventually a modified emplacement device can pick up the container. In the case of borehole disposal the design of the waste canister was modified to facilitate retrieval and the boreholes will be equipped with a casing that shall be closed and sealed tightly at the top of casing to guarantee retrieval even decades after emplacement. However, it is noted that before the canisters can be retrieved, a concept for their subsequent handling and storage aboveground must be implemented.

4 Summary

The preliminary safety analysis of the Gorleben salt dome was carried out to investigate its suitability as a repository for heat-generating waste. It was based on the "Requirements Governing the Final Disposal of Heat-Generating Radioactive Waste as at 30 September 2010", and took into account the total amount of expected heat-generating radioactive waste based on the decision of 2011 to phase out nuclear energy production. Two different emplacement strategies, i.e. emplacing all heat-generating waste in horizontal drifts and emplacing all heat-generating waste in deep vertical boreholes, respectively, were designed and considered. Thermal-mechanical calculations demonstrate that both strategies can be readily accommodated by the repository concepts under consideration of all applicable performance and safety requirements. The technical design work shows that the Gorleben salt dome provides sufficient space on a single level to host a repository for all the heat-generating waste arising in Germany. Additionally, technical approaches on the retrieval of waste containers were developed for both disposal concepts. Room for optimization was identified in the area of retrievability technique and systems, mine ventilation, allocation of waste containers in emplacement fields and emplacement drifts as a function of heat generation. For comparison purposes an alternative emplacement concept was also considered for the emplacement of transport and storage casks into individual horizontal boreholes.

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A Thermodynamic Reference Database for Nuclear Waste Disposal

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Safety analysis for a geological repository for radioactive waste as well as remediation measures for uranium mining and processing legacies share an essential: the need for a reliable, traceable and accurate assessment of potential migration of toxic constituents into the biosphere. The respective computational codes require site-independent thermodynamic data concerning aqueous speciation, solubility limiting solid phases and ion-interaction parameters. Such databases, however, show several constraints:

- Incompleteness in terms of major and trace elements
- Inconsistencies between species considered and corresponding formation constants
- Restricted variation ranges of intensive parameters (temperature, density, pressure)
- Limitations with respect to solution compositions (ionic strength)

To overcome these limitations to a significant degree, an ambitious database project – THEREDA – has been launched in 2006 by institutions leading in the field of safety research for nuclear waste disposal in Germany [1].

The main objective is a centrally administrated and maintained database of verified thermodynamic parameters for environmental applications in general and radiochemical issues in particular.

During the last year, the most important point was the official release of four more datasets (adding carbonate, An(III), Np(V) and Cs to the hexary system of oceanic salts), all based on the Pitzer model describing the ion-ion interactions. They can all be downloaded as separate files from the project web site www.thereda.de (navigation menu: THEREDA Data Query → Tailored Databases) as generic ASCII type, and in formats specific to the geochemical speciation codes PhreeqC, EQ3/6, ChemApp and Geochemist's Workbench. Moreover, access to data records is now also possible through interactive forms (menu: THEREDA Data Query → Single Data Query // Complex Systems), both with export options as CSV or MS Excel file. Additional releases of thermodynamic data for Th(IV), U(IV) and U(VI) are already under way.

In connection with these data releases several other measures have been successfully implemented in THEREDA, namely a new interactive web-based tool for data entry and editing including a variety of internal checks for data consistency and plausibility.

The new auditing scheme is a further major milestone providing an independent measure to monitor data manipulations. Another significant improvement to THEREDA is the establishment of benchmark calculations – all being fully documented and accessible to the public. So one can monitor whether changes in the database (or the export parsers) affect the results of speciation calculations. They may also detect deviations between the various codes when fed with an identical data input. Test subjects are the concentrations and pH values of well-defined multiple-salt points or solubility curves of solids.

The most recent advance is the integration of sorption data (parameterization of surface complexation models) into the existing database management system.

Eight issued technical papers (Downloads → Documentations) promot the transition of THEREDA into a real information and discussion platform on issues concerning the database but also on geochemical modeling at large.

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Waste Classification and Choice of Geological Repository Concept: Ukrainian Case

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Abstract: *A new classification scheme for the disposal of radioactive waste was developed for Ukraine within the framework of the EC project U.04.01/08-C. This scheme is based on final disposal options for different waste classes and complies with IAEA recommendations. The implementation of the new classification scheme leads to an essential reduction in disposal costs and allows solving the problem of 'accident waste' disposal using an economically acceptable method. The economic effect of the new classifications implementation makes the use of both concepts of geological repositories appealing in Ukraine: repository at the intermediate depth for disposal intermediate level waste, and the deep geological repository of borehole type for the disposal of spent fuel and vitrified high-level waste.*

Introduction

In Ukraine, utilization of nuclear energy started in the 1950s. Currently, half of the electricity in the country is generated by nuclear power plants, which in turn generate radioactive waste (RW) with different activities, isotope compositions and aggregate states. Ukraine inherited the approach to radioactive waste management from the former Soviet Union. The main feature of this approach is to minimize the costs of waste reprocessing and disposal, which means shifting the burden of radioactive waste disposal to future generations. One of the most important tasks for improving the RW management system in Ukraine is to create the infrastructure for radioactive waste disposal in the country.

1 Objectives

The purpose of this poster is to describe a new scheme of radioactive waste classification in Ukraine and possible positive effects of its implementation. Special attention is paid to the influence of the new classification scheme on the selection of concepts for geological disposal of waste.

2 Ukrainian waste inventory

Currently, 15 WWER reactors (2 of WWER-440 and 13 of WWER-1000) are being operated at four Ukrainian NPPs. Four RBMK reactors were operated at the Chernobyl NPP; one of them was destroyed during the accident in 1986. The total capacity of the operating units is 13.8 GW. They generate approx. $90 \cdot 10^9$ kWh of electricity (up to 50%) annually. It is intended to build new reactors and to extend the life of existing ones. It is expected that more than 15,000 tHM of SNF and almost 200,000 m^3 of waste will be generated by the existing NPPs.

In the near future, vitrified high-level radioactive waste from the reprocessing of spent nuclear fuel (SNF) from WWER-440 reactors will arrive in Ukraine from the Russian Federation.

Mining and processing of uranium ore is performed in Ukraine, and as a result approx. 65 million m^3 waste from uranium ore processing has been accumulated. This waste is not declared as radioactive waste.

Six plants of the Ukrainian State Association “Radon” (UkrSA “Radon”) deal with the collection, transport, storage and disposal of radioactive waste from Ukrainian enterprises and medical and research institutions (including sealed radiation sources). There are two research reactors in Ukraine.

The accident at the Chernobyl NPP contributed approx. 3.3 million m^3 to the total amount of waste. This ’accident waste’ contains high concentrations of alpha-emitting nuclides. The waste is located mostly within the Chernobyl exclusion zone.

3 Current waste classifications

Depending on the classification aim, all waste is divided into the units: types, groups, categories and kinds (OSPU-2005, NRBU-97). The classification criteria are:

- possibility of disposal for types: short- and long-lived waste (SL and LL)
- specific activity or dose for categories: low-level, intermediate-level and high-level waste (LLW, ILW and HLW)
- the aggregate state for kinds: solid and liquid waste.

The current Ukrainian waste classification focuses mainly on ensuring radiation safety for workers and the population during waste management (collection, sorting, transport and storage) at the operational phase of nuclear facilities. The current classification is not suited for the disposal of all types of existing waste. It allows the use of only two types of repositories: near-surface repositories for short-lived waste and deep geological repositories (DGR) for long-lived waste. According to the current classification, most of the ‘accident waste’ is to be disposed of in a DGR.

4 New classification scheme

A new classification scheme (NCS) for the disposal of radioactive waste was developed in 2011-2012 within the framework of the EC project U.04.01/08-C “Improvement of the Radwaste Classification System in Ukraine”. The project was carried out by a consortium that consisted of DBE TECHNOLOGY GmbH (Germany, consortium leader), SKB International AB (Sweden), ANDRA (France), COVRA (Netherlands), and ENRESA (Spain). The consortium was assisted by REC (Ukraine).

The new classification scheme (Table 1) is based on final disposal options for different waste classes and complies completely with the IAEA recommendations (IAEA, 2009).

Table 1. Description of new classification scheme proposed for Ukraine

Class	Description	Disposal option
Exempt waste (EW)	Exists in Ukraine	Exempt waste does not need to be managed as radioactive waste
NORM waste	Not determined in Ukraine. Contains only natural nuclides. It is not considered as radioactive waste.	Disposal outside the nuclear legislation (except for separate streams of TE-NORM)
Very Low Level Waste (VLLW)	Not determined in Ukraine. There are large volumes of waste in Ukraine that can be classified as VLLW.	Landfill repositories
Low Level Waste (LLW)	This corresponds to existing short-lived waste	Surface (near-surface) repositories
Intermediate Level Waste (ILW)	This corresponds to existing long-lived waste	Repository at intermediate depth , or deep geological repository
High Level Waste (HLW)	This corresponds to existing heat-generating HLW	Deep geological repository (DGR)
Disused Sealed Radiation Sources (DSRS)	Not determined in Ukraine. Due to high risks associated with DSRS, a special class was created.	Separate DSRS may be disposed as LLW or ILW

Despite the large amounts of 'accident waste', it is not appropriate to define a special class of 'Chernobyl waste'. Sorting such waste into classes should be determined only by the waste parameters, not by the origin. 'Chernobyl waste' contains higher amounts of long-lived nuclides than 'ordinary' waste. This constitutes the main problem associated with its disposal.

The problem can be solved by disposing of the 'Chernobyl waste' in the Chernobyl Exclusion Zone (CEZ) applying less restrictive Waste Acceptance Criteria (WAC). Special, less restrictive WAC for repositories for VLLW and LLW of Chernobyl origin, which are located in the CEZ, are based on the limited access to the CEZ. These general WAC are based on estimates of the radiation exposure to critical groups of population living outside the Exclusion Zone. These special WAC should be applied only for the 'accident waste'.

5 The economic effects of NCS

The NCS provides significant savings by allocating the waste to the optimal repository types. To use these advantages, some sorting and characterization activities should be carried out immediately after waste generation. This may temporarily entail additional costs. However, it is much easier and cheaper to sort and characterize a particular waste immediately after its generation than to carry out these tasks on mixed waste located in a common storage facility much later. These activities and associated costs are not a consequence of implementing the proposed new classification scheme. They are inevitable and predetermined by the existing requirements of the Ukrainian legislation and standards. Thus, no additional or excess expenses are expected as a result of introducing the NCS.

The implementation of the NCS, deciding to dispose of 'accident waste' as VLLW or LLW in the CEZ, using three repository types (landfill, near-surface and deep geological) for RW disposal, and co-disposal of ILW and HLW in a deep geological repository will allow a tenfold decrease in total disposal costs. If ILW and HLW are disposed of separately in different repositories, the costs will decrease by 40 times (DBE, 2012).

6 Choice of geological repository concept and potential sites

The borehole concept of DGR has a number of advantages: not only the lower construction costs, but also the shorter construction time and lower risks of human intrusion, etc.. However, its inherent shortcomings (limited canister dimensions) will not allow the disposal of the complete inventory of ILW and HLW (Shestopalov et al., 2005). This can be an additional reason for developing two geological repositories in Ukraine:

- Borehole DGR for disposal of vitrified HLW and SNF
- Mined geological repository at intermediate depth for ILW disposal.

In Ukraine, all types of prospective geological formations for DGR construction are present, namely: crystalline rocks of the Ukrainian Shield, clays of Ciscarpathians and CEZ, as well as salts of Donbass and Transcarpathians. For sociopolitical reasons (to avoid population protests), the stakeholders give preference to DGR construction in crystalline rocks within the uninhabited CEZ and the surrounding areas. The location of promising areas within CEZ is shown on the Fig.1.

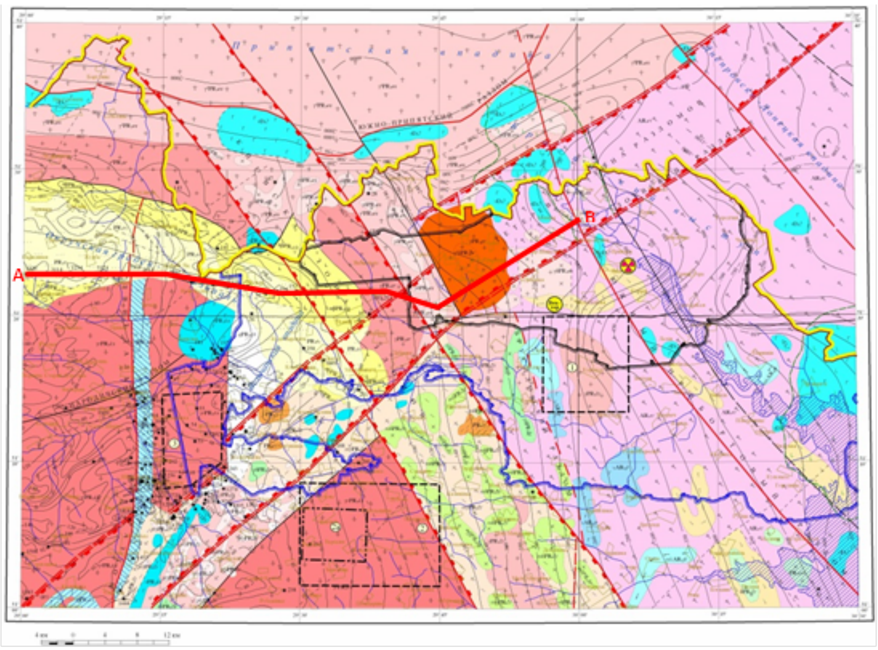

Fig.1. Sites for DGR within or close to the Chernobyl Exclusion Zone: 1 – Novosilky, 2 – Veresnia, 3 – Zhovtneva

In all most promising areas, the typical is hornblend-biotite, rapakivi-like granite dated at 1,75 Ga. These rocks are covered by Jurassic-Quarternary sediments (clays, marls, aleurites, sandstone, sands) with a thickness between 180 and 350 m (see Fig.2).

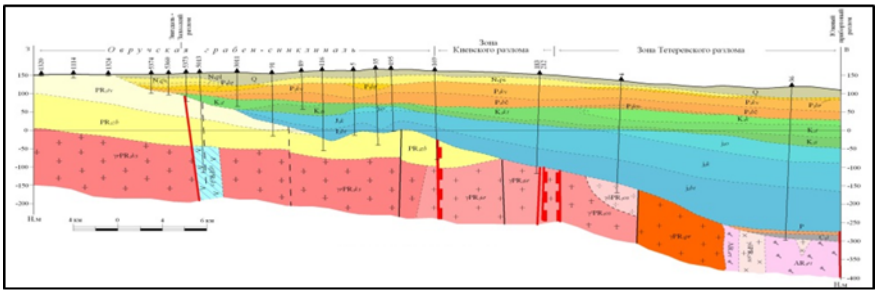

Fig.2. Geological cross-section through the central part of CEZ along the line A-B

The geological conditions at the Ukrainian sites are very similar to those in Sweden and Finland regarding rock types, age and geological history, tectonics, hydraulic and mechanical properties, seismicity, etc.. The difference at the Ukrainian sites is the presence of a thick sedimentary layer and a much lower total salinity of the groundwater.

The results of preliminary safety assessments (Shestopalov et al., 2006, IAEA, 2013) demonstrate that the crystalline rocks of the investigated areas are suitable for waste disposal and emphasize the urgent need for detailed field investigations (including drilling) in promising areas.

The prospects of using various DGR concepts differ among sites:

- All sites, (1), (2), (3), are suitable for the borehole concept of DGR
- The Zhovtneva (3) and Veresnia (2) sites are more suitable for mined DGR construction
- The Zhovtneva (3) site is the most suitable for constructing a repository at intermediate depth.

Conclusions

- The implementation of the new classification scheme leads to an essential reduction in disposal costs.
- Implementation of the new classification scheme allows solving the problem of 'accident waste' disposal using an economically acceptable method.
- Economic effects and various technical considerations make the use of both geological repository concepts appealing for Ukraine: mined repository at intermediate depth for ILW disposal and deep geological repository of the borehole type for the disposal of SNF and vitrified HLW.
- The Chernobyl Exclusion Zone and adjacent areas have good prospects for siting geological repositories of both types

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Geological and Geotechnical Limitations of Radioactive Waste Retrievability in Geologic Disposals

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Abstract: *The capability of retrieving radioactive waste emplaced in deep geological formations is nowadays in discussion in many countries. Based on the storage of high-level radioactive waste (HAW) in deep geological repositories there is a number of possible scenarios for their retrieval. Measurements for an improved retrieving capability may impact on the geotechnical and geological barriers, e.g. keeping open the access drifts for a long period of time can result in a bigger evacuation damage zone (EDZ) in the host rock which implies potential flow paths for ground water. Nevertheless, to limit the possible scenarios associated to the retrieval implementation, it is necessary to take in consideration which criteria will be used for an efficient monitoring program, while clearly determining the performance reliability of the geotechnical barriers. In addition, the integrity of the host rock as geological barrier has to be verified. Therefore, it is important to evaluate different design solutions and the most appropriate measurement methods to improve the retrievability process of wastes from a geological repository. A short presentation of the host rocks is given is this paper.*

1. Life-Cycle Stages of a Geological Repository for High-Level Radioactive Waste (HAW) with Retrievability

A geological repository involves three main phases: pre-operational phase, operational phase and post-operational phase (Figure 1).

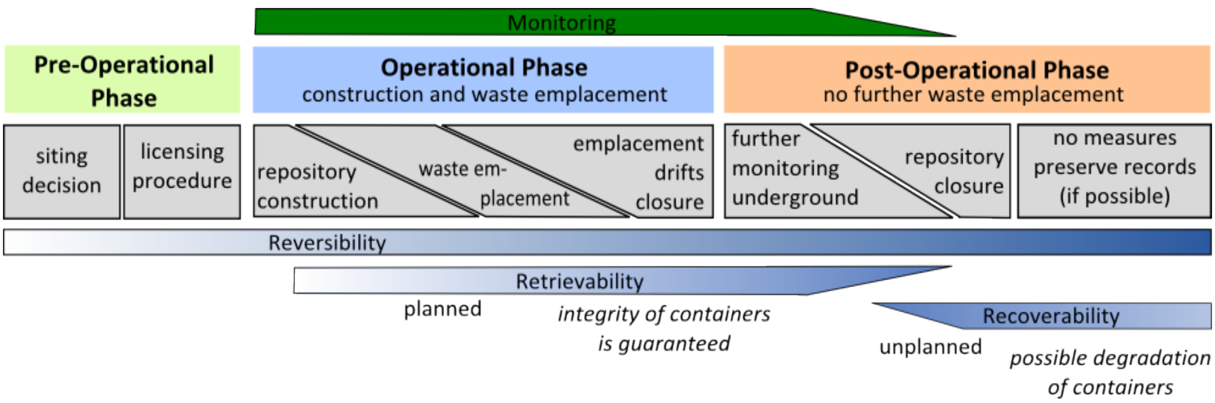

Fig. 1. Definitions of reversibility, retrievability and recoverability

In the pre-operational phase the siting decision and the licensing procedure take place. In the operational phase the waste is emplaced and the emplacement drifts will be backfilled. The duration of this phase depends of the amount of waste and the repository's design. In this phase the repository will be maintained open until demand of the disposal radioactive waste exists.

Retrievability is the possibility to recover the radioactive waste in a planned manner. The retrieval of the radioactive waste can take place while monitoring. Recoverability is the possibility to recover the radioactive waste at a late stage after disposal. The integrity of the waste containers is not necessarily ensured. Near-field monitoring is provided until the monitoring drifts are closed in the repository closure.

The repository closure occurs in the post-operational phase. The waste remains sealed as far as a decision of recovery takes place. Even though the near-field monitoring is no longer feasible, further monitoring strategies and the preserve of records are highly recommended.

Within all phases it is possible to overturn decisions – reversibility is implied.

2. Generic Models of Deep Geological Disposal

Measurements for an improved retrieving capability may impact on the geotechnical and geological barriers, e.g. keeping open the access drifts for a long period of time can result in a bigger evacuation damage zone (EDZ) in the host rock. This implies potential flow paths for ground water. To evaluate the impact of an improved retrievability a generic deep geological repository model was created (Figures 2 and 3).

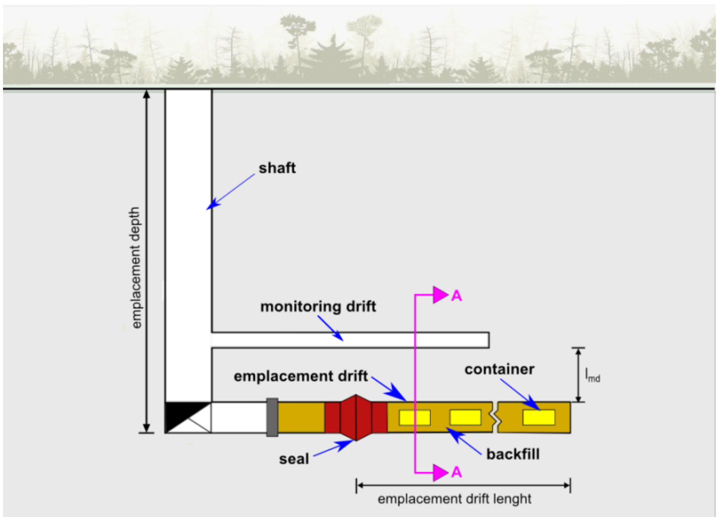

Fig. 2. Generic deep geological repository

This model could be applied for all types of host rock. The waste will be emplaced in drifts, for monitoring purposes a further monitoring drift is implemented (Figure 3).

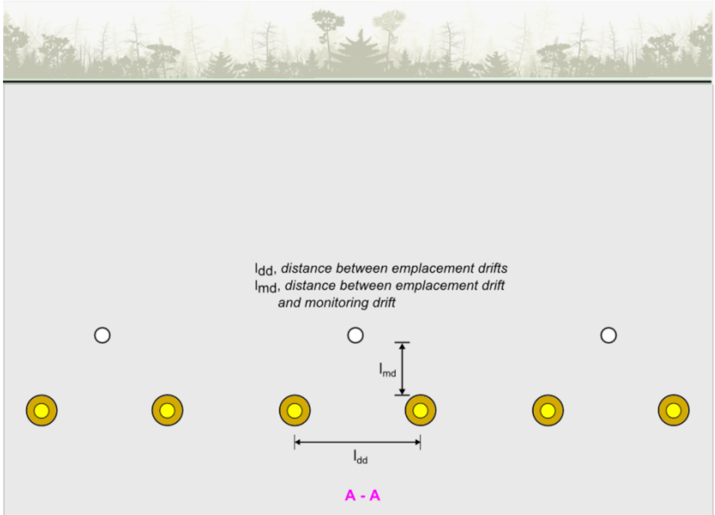

Fig. 3. Cross-section through the emplacement and the monitoring drifts

This design is applied to five host rocks types, flat and steep-bedded rock salt, clay, claystone and crystalline rock. Their characteristic hydrological and mechanical behavior was used to compare the performance of these host rocks in a repository with implemented retrievability. The favorable features for each host rock and their response on the access drifts remaining open were compared and evaluated.

3. Potential Host rocks

3.1 Rock Salt

The mechanical behavior of flat and steep-bedded rock salt is quite similar. Rock salt in natural storage conditions has a very low permeability and the construction-induced damage is reduced by self-healing. Rock salt is capable to creep; therefore the stress deviator has to be reduced. If not, the EDZ will increase and the geological barrier will be degraded. The cavities in salt rock do not need a pit supporting system. But if long-term stability is required, the support system has to bear the total overburden pressure. Drifts in rock salt will be backfilled with crushed salt.

Fig. 4. Flat-bedded rock salt, salt mine Bernburg. esco, 2014

3.2 Clay

Clay has a very low self-support therefore a heavy support system is required particularly when the disposal vaults and access tunnels are keeping open for a long period of time. The support system has to bear the total overburden pressure. The material has a slightly anisotropic behavior. Due to stagnating pore-water the sediment is sensitive against desiccation and chemical changes. Drifts in Clay will be backfilled and sealed with bentonite.

Fig. 5. Plastic clay in URL Hades, Belgium. Bastiaens et al., 2008

3.3 Claystone

Claystone, as a result of its lithified structure, has a low self-support too. However their required pit support system is less extent as in clay. As a rock it has a latent joint network which can result in flow paths for ground water. The rock has a strong anisotropic behavior. Due to its geological history claystones are less sensitive against desiccation and chemical changes compared to clays. Drifts in claystone will be backfilled and sealed with bentonite.

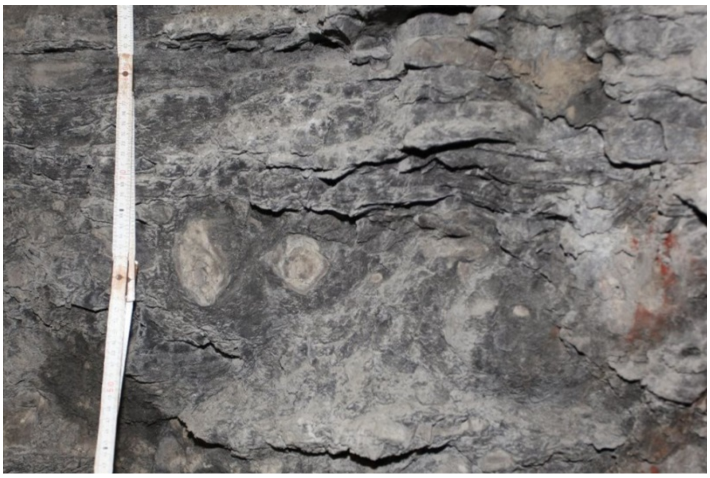

Fig. 6. Shale of the Fladentonstein series, Konrad mine. Stahlmann et al., 2014

3.4 Crystalline Rocks

Crystalline Rocks such as granite or gneiss are high solidity and provide cavity stability over a long period of time. Nevertheless it is strongly jointed because of cooling in its geological history. These discontinuities result in a groundwater flow, therefore the rock has no retention capability for ground water. The rock has not the features of a geological barrier and engineered barriers (canisters and bentonite) are needed for long-term security.

Fig. 7. Gneiss in URL Onkalo, Finland. Dark spots are water inflows on joints. Vira, 2013

4. Discussion

Figure 8 shows a qualitative comparison of the five host rocks. Rock salt has a good self-support if short-term stability is required. As a result of its viscoplastic behavior, it needs a heavy support system if long-term stability is required. Clay has a low self-support cavity stability and support systems are always needed. Claystone has a better self-support unlike clay but lower plasticity; consequently its EDZ's self-healing capability is not as good as in clay. Due to its great self-support crystalline rocks cavities can be kept open over a long period of time. However they have a higher permeability because of their fractures.

Furthermore, the long-term safety of these five above mentioned host rocks were compared during an ice age. To prevent negative impacts in the repository system it is recommended a minimum depth of 600 m to all models. On the other hand, a minimum depth of 800 m is recommended for steep-bedded rock salt owing to its potential uprising; 400 m for clay as a result of its low self-support.

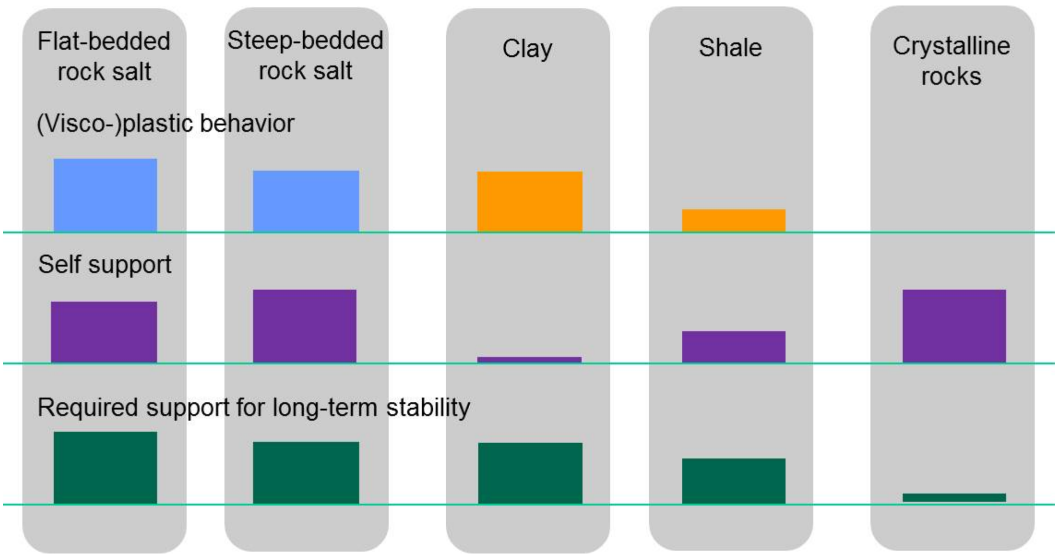

Fig. 8. Qualitative comparison of the five host rocks

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2 PUBLIC INVOLVEMENT AND PARTICIPATION

Multi-Level Governance-Perspective on Management of Nuclear Waste Disposal. A Comparative Analysis

ACHIM BRUNNENGRÄBER, DANIEL HÄFNER (GERMANY)

The primary aim of the project is to conduct a detailed social and political analysis of the preconditions for the development of an acceptable strategy for nuclear waste disposal in Germany. This includes the identification of stakeholders and their interests, responsibilities, value systems, views and expectations as well as paths for a constructive approach to dialogue and problem-solving. A focus of the research project will be an international comparative multi-level governance analysis of acceptance patterns and steering mechanisms for conflict resolution.

Opinions and Social Values Related to the Disposal of Nuclear Waste in Switzerland

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Discourse in media and politics about nuclear waste and its disposal in so-called "Endlager" (Germany) or "Tiefenlager" (geological deep ground repositories; Switzerland) often consider positions and arguments of diverse interest groups. Mostly polarized discussions are in the focus. However, we find a temporally consistent pattern of four opinion clusters in German speaking communities in Switzerland: one cluster in favor of a repository (perceiving mostly benefits) and one cluster with high-risk ratings opposing a repository; a third cluster of moderate opposition is ambivalent regarding risks and benefits, whereas a fourth cluster seems indifferent.

Moreover, in qualitative interviews we found high importance of the development of the participatory process. Participants were sensitive to value related issues such as absence of political influence, transparency, comprehensive and independent information. Important to note is the problem that some of these values can be used as pro- or con-argument regarding a repository by different individuals. For instance, all agree that safety is essential – but both conclusions, to be for or against a repository, are possible.

A recent study focused on the arguments, underlying people's opinions. The salient arguments that participants report are related to the sense of responsibility for the country to store safely the nuclear waste and to avoid its export. Moreover, people recognize the necessity of a safe solution for the storage in order to preserve future generations from the risks of nuclear waste. These arguments may be relevant for the fact that participants, on average, have a favorable position regarding a deep ground repository in Switzerland.

3 ASPECTS OF REPOSITORY OPERATION

Independent Monitoring of a Release from the Waste Isolation Pilot Plant in New Mexico, USA: Results and Purpose

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The Waste Isolation Pilot Plant (WIPP) is a transuranic (TRU) waste repository operated by the U.S. Department of Energy (DOE). The repository is emplacing defense-related transuranic (TRU) wastes into a bedded salt formation approximately 655 m (2150 ft.) below the surface of the Earth. Located near Carlsbad, New Mexico, an area with less than 30,000 people, the WIPP facility is licensed to accept TRU waste with activity concentrations of alpha-emitting isotopes >3700 Bq/m^3 (> 100 nCi/g) and half-life >20 years. The upper waste acceptance limit is 0.85 TBq/liter (<23 Ci/liter) of total activity) and 10 Sv/hr dose rate on contact. The repository, which opened in March 1999 will eventually contain the equivalent of ~176,000 m^3 of TRU waste. The vast majority of the waste disposed in the WIPP repository is "contact-handled" waste, meaning it has a surface dose rate less than 2 mSv per hour. Local acceptance of WIPP is in part due to an independent environmental monitoring program that began before and continues after WIPP began receiving nuclear waste. This independent monitoring is being conducted by the Carlsbad Environmental Monitoring and Research Center (CEMRC), which is associated with New Mexico State University. CEMRC is funded by DOE through a grant process that respects its independence in carrying out and reporting the results of environmental monitoring at and near the WIPP site. The primary focus of CEMRC monitoring is on airborne radioactive particulate; however other pathways are also monitored. Pre-disposal baseline data of various anthropogenic radionuclides present in the WIPP environment is essential for the proper evaluation of the WIPP's integrity. These data are compared against disposal phase data to assess whether or not there is any radiological impact from the presence of WIPP on workers and on the regional public. The program has capabilities to detect radionuclides rapidly in case of accidental releases from the repository or the site during operations. For the first time in almost fifteen years of operation, there was an airborne radiation release from WIPP. An independent study released by CEMRC confirmed detection of trace amounts of ^{241}Am and $^{239+240}$Pu in the air samples collected half a mile from the WIPP site. The cause of the release is currently under active investigation. It is speculated that it may have involved the collapse of a section of roof in the active waste emplacement area, damaging one or more drums containing a type of waste readily dispersed into the ventilation air. This paper presents and makes an evaluation of the data in the wake of this incident. These results were reported to the public by CEMRC as they were obtained through analyses. At this point in time the concentrations of radionuclides in air and other samples have been very small and well below any level of public-health or environmental concern. CEMRC's independence and its extensive monitoring program and constant public engagement provide confidence to the local public. CEMRC's independent monitoring program is a potential model for nuclear facilities, and especially nuclear waste repositories, elsewhere in the world.

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4 CONSTRUCTION OF TECHNICAL BARRIERS

The Sealing of Excavation Damaged Zones in Salt Formations Using Sodium Silicate Solutions

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Abstract: *Since many decades, pressure grouting is an effective technique of civil engineering for sealing and stabilization purposes. Due to the potential contamination of fluids, grouting is of particular importance in repositories of radioactive waste. Traditional grouts for the sealing of fine fractures are sodium silicate solutions. Laboratory and field investigations prove that the particle-free solutions can be used to permanently seal excavation damaged zones (EDZ) in salt formations, because the solid reaction products are inert or almost insoluble. EDZ permeabilities of* 10^{-17} *m² can be achieved and were determined on the basis of the injection pressures and flow rates. High grouting pressures were realized as local test loadings. Laboratory tests show the fixation of* Co^{2+}, Ni^{2+}, Sr^{2+}, Ba^{2+}, $[UO_2]^{2+}$ *and illustrate that sodium silicates may act additionally as a chemical barrier.*

1 Introduction

The excavation of underground openings results in inhomogeneous stress states and the formation of fractures, if the stresses exceed the dilatancy boundary of the rocks. The fractures may form a network of flow paths. Consequently, the construction of seals requires the use of technical methods to reduce the permeability of the excavation damaged zones (EDZ). Providing that the possibility of a profiling of the drift or shaft contour is limited and a considerable low permeability of the EDZ is required, a pressure grouting of the fractures using particle-free grouts is essential. A classical grout used in civil engineering for sealing and stabilization purposes are sodium silicate solutions (e.g. Karol 2003, Zhilkani 2010).

Sodium oxide (Na_2O) can maintain the pH level sufficiently high that silicate (SiO_2) is soluble in water. The solutions, commercially called water or liquid glass, are available in different densities that are characterized by their SiO_2:Na_2O ratios and water contents. In dependence of the composition, the viscosity of sodium silicate solutions varies (e.g. Urbain et al. 1982, Yang et al. 2008). In general, the consistency of most grouts is comparable to concentrated magnesium chloride solutions. Classical grouting techniques use reactants to initialize a gelation or polymerization of the silica. The reactants can be classified in three generic groups: inorganic salts, organic/aliphatic esters or amides, and stabilizers, such as cement.

Due to the successful application in numerous projects, the availability of high-quality products, and the reaction with saline solutions and the salts of the host rock, investigations were carried out with respect to the sealing of EDZ in evaporite rocks. Laboratory investigations focused on the reactions with salts and their kinetics, as a basis for the grout selection and the specification of injection parameters. In addition, the long-term stability of the reaction products was investigated. The works finalized in pressure grout tests in a salt mine.

2 Basics of the gelation and hardening of sodium silicate solutions

Due to the property of silica species to form silanol (Si-OH) and siloxane groups (Si-O-Si), sodium silicate solutions contain varying proportions of monomeric, oligomeric linear, cyclic or complex polysilicate anions. The degree of polymerization of the silicate anions rises with increasing concentration (density) and increasing SiO_2/Na_2O ratio of the solution. In addition, the pH-value is influencing the polymerization/depolymerization equilibrium. According to their chemical composition (25–34 wt.-% SiO_2) and pH-value (> 11) most concentrated commercial solutions are dominated by ring structures and polysilicate anions, which have a size of < 1 nm.

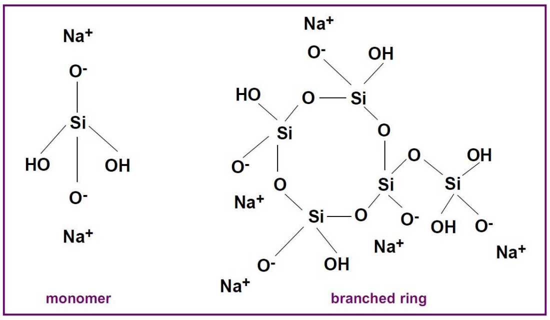

Fig. 1. A monomeric tetrahedral silicate and a branched ring polysilicate structure, which are present in sodium silicate solutions.

As the pH is reduced or through charge screening and reducing the dielectric constant of the aqueous phase by adding inorganic salts, the silanol groups condense to build larger particles. Gelation occurs when particle aggregation ultimately forms a uniform, three-dimensional network of silicate particles. Most notable is the dependence of the reaction rate on the dissolution rate of the salt mineral, the temperature, and the composition of the water glass. Divalent metal ions such as Ca^{2+} and Mg^{2+} accelerate the gelling kinetics more than monovalent cations (Na^+).

Major products of the reaction with pure halite or sodium chloride solution are amorphous silicate and halite. Salt formations contain a variety of Mg-containing minerals and solutions. Experiments and observations made during in-situ grouting, illustrate that these Mg^{2+} ions, together with water, are fixed in the structure of magnesium oxichlorides and Mg-silicates. The formation of magnesium oxichlorides can be described according to the system $NaOH-MgCl_2-H_2O$ (Bilinski et al. 1984, Mazuranic et al. 1982). All reaction products have a low solubility (cf. Engelhardt, von Borstel 2014). For example, laboratory experiments show that highly concentrated $MgCl_2$-solutions (cf. Q-/IP21- or R-/IP19-solution) have Si concentrations of < 1 µg/g. In comparison, the Si concentrations of low-saline waters are < 5 µg/g due to the solubility of macro-crystalline quartz.

The crystalline state is a relatively more stable state than the amorphous state. This accounts for the transformation of an amorphous solid into its crystalline state. However, the amorphous solid is quite stable due to the low energy difference. This is attributed to the high energy of activation, which is required to induce the rearrangement of the bonds and the high viscosity of solids. The study of natural analogs showed that amorphous silicates can preserve for thousands of years, however, it is sometimes difficult to evaluate the results due to the influence of organic substances. Natural amorphous silica such as diatomite and opal (natural hydrated amorphous silica) are examples of non-crystalline silicates. In evaporite rocks of the Zechstein age (Perm) common authigenic silicates are quartz and talc ($Mg_3Si_4O_{10}(OH)_2$).

The low rate of the structural changes is coupled with a low volume decrease (syneresis) of sodium silicates, because this type of shrinkage is a result of the formation of siloxane (Si-O-Si) bonds, which takes less space that the two individual silanol groups, and the release of hydroxyl ions. In addition, it can be assumed that syneresis is limited by the reaction of these hydroxyl ions with Mg^{2+}-ions resulting in a formation of magnesium oxichlorides or magnesium silicates. These facts and the low solubility of the reaction products suggest the use of sodium silicate solutions for sealing measures.

Like Mg^{2+} or Ca^{2+}, experiments demonstrate that silicate solutions are able to fix many toxic or radioactive heavy metals, which are dissolved in NaCl- and $NaCl$-$MgCl_2$-solutions, in insoluble minerals. First tests showed the formation of solids during the mixing of water glass with barium (Ba^{2+}), cobalt (Co^{2+}), iron (Fe^{2+}), nickel (Ni^{2+}), and strontium (Sr^{2+}) salt solutions. The results are comparable with the well-known crystal or chemical garden experiments. In particular, chemical analyses prove a removal of small divalent ions (Co, Ni) as well as a significant decrease in the UO_2 concentration. The precipitates can comprise insoluble salts, as well as hydroxides and silicate phases. Another well-known and in water treatment frequently used property of amorphous silicates is a high sorption capacity.

3 The sealing of excavation damaged zones

Fundamental for the grouting of an EDZ are the negligible size of the cyclic and polysilicate anions and the absence of a yield strength in such a way that the sodium silicate solutions can be considered as Newtonian fluids. The viscosity decreases with temperature and increases with density, according to the Krieger-Dougherty equation. Due to the fact that the sodium silicate solutions are particle-free, a pre-wetting of fractures is not necessary. The flow velocity in fractures depends in particular from its size, the roughness of the fracture surfaces, the grout pressure and the water glass viscosity. The reactions of the sodium silicate solutions with exposed soluble salts induce a rapid gelation, which causes an increase in viscosity and the development of a yield strength. The formation of solids and the resulting changes of the flow process in a fracture are illustrated by Fig. 2.

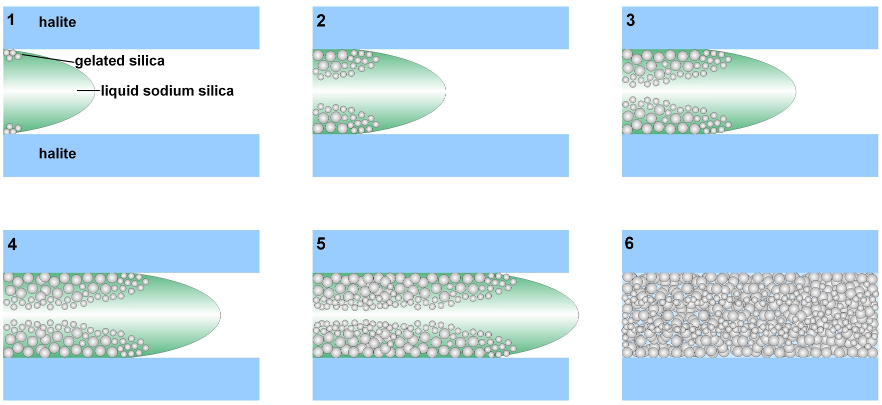

Fig. 2. Conceptual model of the injection of a sodium silicate solution into a fracture in rock salt. The contact of the sodium silicate with the soluble rock salt components (blue) results in the formation of amorphous silicates and secondary halite at the fracture surface. The thickness of the reaction products increases (the fracture width decreases) and at the end of the injection, the fracture is filled with solids and residual solutions, which occur as fluid inclusions.

The injection pressure is chosen in dependence of minimum principal stress. In general, it can be expected that the maximum injection pressure is about 2 MPa larger than the minimum principal stress. The fact of a negligible particle size opens the possibility to simultaneously fill and seal fractures and to measure the permeability of the EDZ. The pressures and flow rates are registered during the whole work duration and the injection pressure is increased step by step. The permeability of the system of a sealing body and the EDZ can be determined form the change in pressure over time. The execution and the evaluation of these pressure drop measurements are comparable to standard instationary permeability measurements in the laboratory. However, in contrast to laboratory measurements with pressure cells only the input pressure can be registered and analyzed. Numerical methods are used to solve the differential equation of pressure decay and to match the observed pressure decay with the calculated decay curves.

Mechanical investigations show that the rate of the hardening decreases with increasing density of the sodium silicate solutions. This behavior is caused by the lower water content of the high density solutions, which result in a decrease of the capability to dissolve salts at the fracture surfaces. Consequently, a lower amount of dissolved salts ("hardener") can react with the sodium silicate solutions. In addition, lower final strength values can be expected. The dissolution of carnallite ($KMgCl_3 \cdot 6H_2O$) results in an increase of the amount of salt solution, due to the release of crystal water. As a result of the increased amount of hardener, high density sodium silicate solutions, e.g. with a density of 1.56 g/cm^3, were investigated for the sealing of fractures in carnallitite. Independent of the kind of salt, the strength should rise with a decreasing width of the fractures.

Presently, quantitative strength measurements focus on the strength development between the contact of sodium silicate (density 1.36 g/cm^3 or 37–40 degrees Baumé) and rock salt. The measurements results in 56 d-tensile strength-values of up to 1600 kN/m^2. The shear strength rises with increasing normal stress on the area of contact and amounts to about 2.3 MPa at a normal stress of 1.5 MPa. Figure 3 shows a hardened sodium silicate between the front surfaces of two splitted cores and on the lateral surfaces.

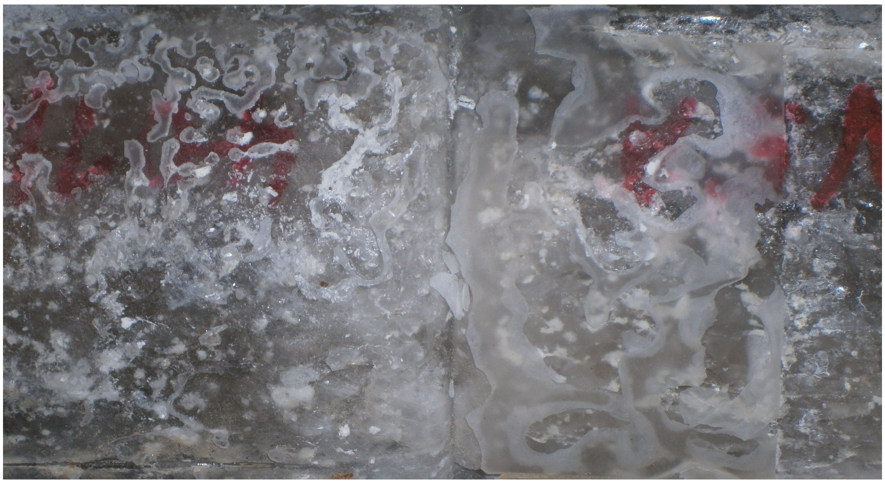

Fig. 3. Rock salt cores with polymerized sodium silicates. The specimen was stored about three months in an air-tight plastic tube at room temperature.

5 Summary

The formation of excavation damaged zones (EDZ) may strongly affect the performance of engineered barriers in underground repositories. In dependence on the requirement on the barrier permeability, the EDZ needs to be removed mechanically as far as possible and/or the fractures of the host rock formation must be grouted. The sealing of small fractures requires the use of particle-free solutions. The following properties favor the use of sodium silicate solutions in rock salts:

- Sodium silicate solutions are available in a variety of densities and viscosities. For this reason, products can be selected according to the required material properties.
- The absence of a yield strength and the small size of the silicate complexes guarantees the penetrability into small fractures and opens the possibility to simultaneously fill and seal fractures and to measure the rock permeability.
- The solutions are not self-hardening. The formation of the solids is a consequence of the reaction with soluble salts and brines.
- The low solubility of the reaction products guarantees long-term stability of fracture fillings.
- Low rates of syneresis are a result of the relative high Na_2O- and SiO_2-amounts of the solutions and the long time-spans necessary for the transformation into the crystalline state.

Up to now comprehensive practical experiences were gained using sodium silicate solutions with a SiO_2/Na_2O-mass-ratio between 2.5 to 3.3 and a density between 1.34 g/cm^3 (37–40 degrees Baumé) and 1.56 g/cm^3 (50–52 degrees Baumé). The sealing of the EDZ was tested around a drift in a salt mine using a solution with a density of 1.34 g/cm^3. It was possible to significantly reduce the permeability of the rock salt. Most values were in the range of 10^{-17} m^3.

Acknowledgements

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Clay Modified Crushed Salt for Shaft Sealing Elements – Material Optimization and Evaluation in Field Tests

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Abstract: *Salt-based materials are intended to use for backfill and sealing systems in geotechnical barriers in underground HLW-repositories. Due to the creep of the saliniferous host rock, the salt backfill will be compacted during several hundreds or thousands years of operation to a minimum of porosity resp. permeability. To raise the sealing potential of a salt-based backfill, the porosity after construction should be minimized by optimal material performance and compaction performance. A procedure to optimize the grain size distribution of crushed salt and its water and clay content is described. The optimized salt fraction gets a better compaction behavior than straight mine-run salt. The addition of a filler-like material (e.g. Friedland Clay Powder) reduces the total porosity and permeability. Backfill columns made from crushed salt and clay probably include an instant sealing function.*

1 Introduction

In saliniferous host rock salt-based materials are intended to use for backfill and sealing systems in geotechnical barriers in underground HLW-repositories. In this case the most favorable material is crushed salt, which is used as backfill in drifts and shafts. Due to the creep of the saliniferous host rock, the salt backfill will be subsequent compacted (consolidated) during several hundreds or thousands years of operation to a minimum of porosity resp. permeability. Within the future shaft closing concepts of the WIPP site in Carlsbad NM [2, 12] or the shaft "Saale" [16] in Germany, highly compacted crushed salt columns are envisaged. The preliminary safety analysis for the Gorleben site also evaluates long-term sealing elements in drifts and shafts made from crushed salt. The published results [14, 15] state, that within approx. 1,000 years, the crushed salt (wetted with 1.5 wt% water) consolidates down to a minimum porosity and fulfills the requirements for a long-term stable sealing. This calculated result bases upon the assumption that the initial porosity of the emplaced crushed salt in the shaft is 10%. As reference, the WIPP project is cited, where intensive investigations on the dynamical in-situ compaction of crushed salt are described. Calculations according to the WIPP site indicates that the salt column becomes an effective long-term barrier in under 100 years [1]. The differences are caused by the site specific creep behavior of the different salt formations.

Beside the application of crushed salt in shaft sealing elements, the use of crushed salt to seal reposition boreholes and drifts is intended [9, 13]. The in-situ compaction in such constricted resp. horizontal structures is impossible resp. very limited. Initial porosities up to 35% are assumed and therefore longer consolidation times for reaching a "sealing state" are needed [9]. Calculations regarding consolidation under conditions of the intended panel closure concept of the WIPP site showing the influence of the initial porosity. While an initial porosity of 35% needs around 150 years to consolidate down to 5%, the initial porosity of 20% reaching 5% porosity already after 65 years [9].

To raise the sealing potential of a salt-based backfill, the initial porosity after emplacement should be minimized by an optimal material performance and compaction performance. Also, a highly pre-compacted backfill needs much less time to re-consolidate to a "sealing state" due to the creep of the host rock.

A maximum packing density is therefore a desired performance of an optimal backfill material. The packing density of technical compacted crushed salt depends on the following parameters:

- Particle size distribution of the crushed salt,
- Particle shape of the crushed salt,
- Additives (e.g. clay),
- Moisture content (additional water or brine) and
- Compaction technology / specific compaction energy.

The VSG-report [15] also states an optimization potential for the initial porosity of the emplaced crushed salt. As a first improvement, an emplacement or partial emplacement of pre-cut salt stones was recommended. Naturally, the emplacement of pre-cut salt stones in an underground section requires an extensive salt stone masonry with salt-filled gaps to achieve a sealing effect within 1,000 years of operation. First investigations on this topic were carried out at the TU Clausthal and reported with promising results [10]. So it seems to be theoretical achievable, to construct a salt stone masonry with initial porosities of lower than 10%.

Another and maybe less extensive way, is to optimize the crushed salt itself and the technical compaction procedure in-situ. Until now, many investigations were undertaken to characterize the behavior of crushed salt during the consolidation under or near host rock conditions [8, 13, 16-18].

The only published investigations according to the technical compaction of crushed salt are from the WIPP site investigations [1, 4, 5, 7]. The shaft sealing concept of the WIPP site envisages a dynamic compaction by dropping a tamper on the emplaced lifts (1.5 m thick) of crushed salt. Preliminary studies showing that the application of three times Modified Proctor Energy (MPE equals 2.7 MJ/m^3) to each lift result in 10% to 15% remaining porosity (fractional density between 0.85 to 0.90) [7]. To reach this result, the crushed salt will be moistured with 1.5 ± 0.3 wt% freshwater [3] and 132 tamper drops (height up to 18 m) are intended on each lift [2].

For all these investigations straight mine-run salt with a limited maximum grain size, but unaffected size distribution was used.

The following investigations are on the optimization of the material composition of crushed salt with the aim of a maximum packing density resp. minimum porosity achieved by technical compaction.

2 Preliminary studies

To influence the grain size distribution of the crushed salt, specifically defined grain size fractions must be available. The thesis from BECKER [6], dealing with the development of an optimal structure-stabilized backfill, describes different crushed salt fractions (Table) available from the processing plant of the Sondershausen mine (GSES GmbH).

Table 1: Grain size fractions available from Sondershausen mine

salt fractions, labelling according to the Sondershausen processing plant	grain size distribution $d_5 - d_{95}$	composition of the optimal crushed salt mixture acc. to GLAUBACH [11]
Überkorn (ÜK)	3 mm – 10 mm	52.3 wt%
Band 6 (B6)	0.4 mm – 4 mm	21.4 wt%
Band 8 (B8)	0.1 mm – 1 mm	11.8 wt%
Feinsalz (FS)	0.03 mm – 0.3 mm	14.5 wt%

To evaluate the influence of the grain size distribution on the compaction process, different mixtures of crushed salt were compacted with a percussive Marshall-Compactor. The Marshall Compactor is international used to prepare specimens of hot rolled asphalt and the machinery resp. the compaction process is standardized in Europe after DIN EN 12697-30.

The specimens were compacted with the same cumulative specific compaction energy of approx. 15 MJ/m^3. The cumulative specific compaction energy results from 200 hammer blows on 270 cm^3 grain volume.

The tested mixtures were mixed per hand and afterwards the desired moisture level adjusted (in relation to the dry material). Shortly after sample preparation, the samples were compacted and analyzed.

The main criterion to evaluate the compaction result is the calculated total porosity Φ_t of the compacted specimens according to Eq. 1.

$$\Phi_t = \left(1 - \frac{\rho_{B,d}}{\rho_P}\right) 100\%$$

$\rho_{B,d}$ bulk dry density of the compacted specimen
ρ_P particle density

The bulk dry density of the compacted specimen was calculated using a mass and volume balance. The particle densities were measured in advance by Helium-Pycnometry. The results of the used materials are specified in Table. For analysis of mixed materials (salt and clay) the mean particle density according to the volumetric fractions were used.

Table 2: Grain densities of the investigated materials

material	grain density
Sondershausen rock salt	2.198 g/cm^3
Friedland Clay Powder (FCP)	2.655 g/cm^3

According to GLAUBACH el al. [11] the grain size distribution of the crushed salt has an important effect on its packing density. As a result the well-known FULLER distribution is used to optimize the grain size distribution of crushed salt as a practical approach in engineering. During a systematic investigation, the optimal FULLER exponent could be identified around 0.5 [11]. This result was unexpected, because the FULLER distribution with an exponent of 0.5 has its origin in the packing of spheres. The grain shape of crushed salt varies instead between cubic and splintered shape, which is unfavorable for an effective compaction.

The figure 1 shows the used grain fractions and their grain size distribution.

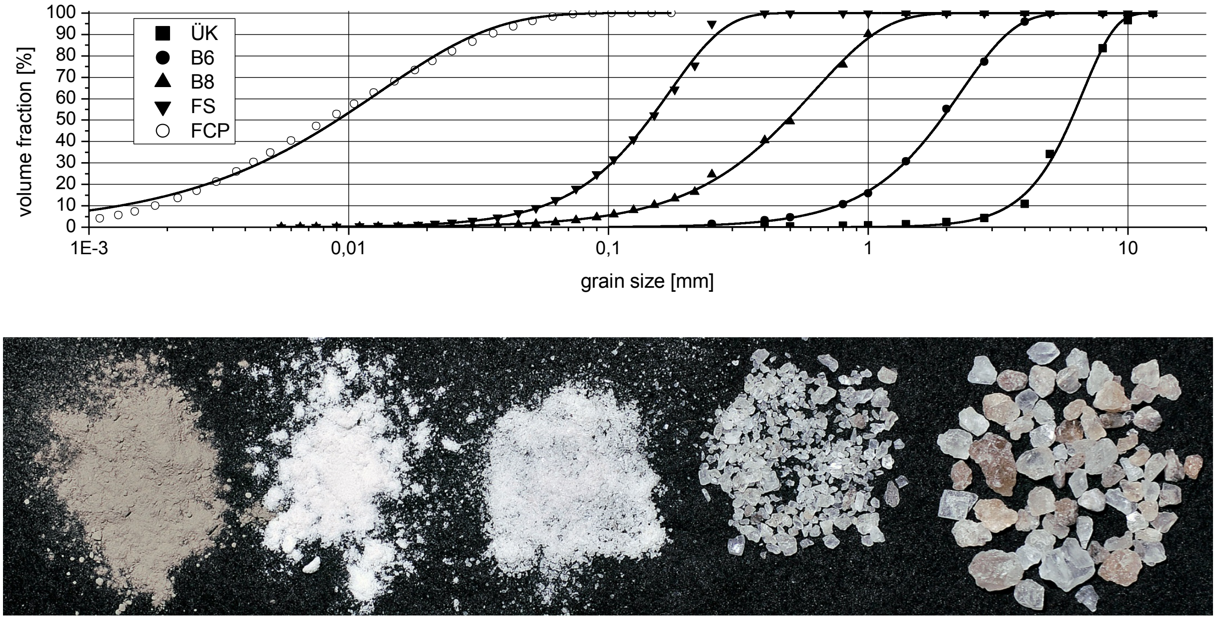

Fig. 1. The used grain fractions and their grain size distribution

As a reference, the well-known REPOPERM [13] material was used to evaluate the effort of the optimization of the grain size distribution. The achieved total porosities of the optimized mixtures are significant lower than the achieved porosities of the REPOPERM reference [11].

The figure 2 shows the results of this test series on the optimized salt fractions with different moisture contents and compaction energies.

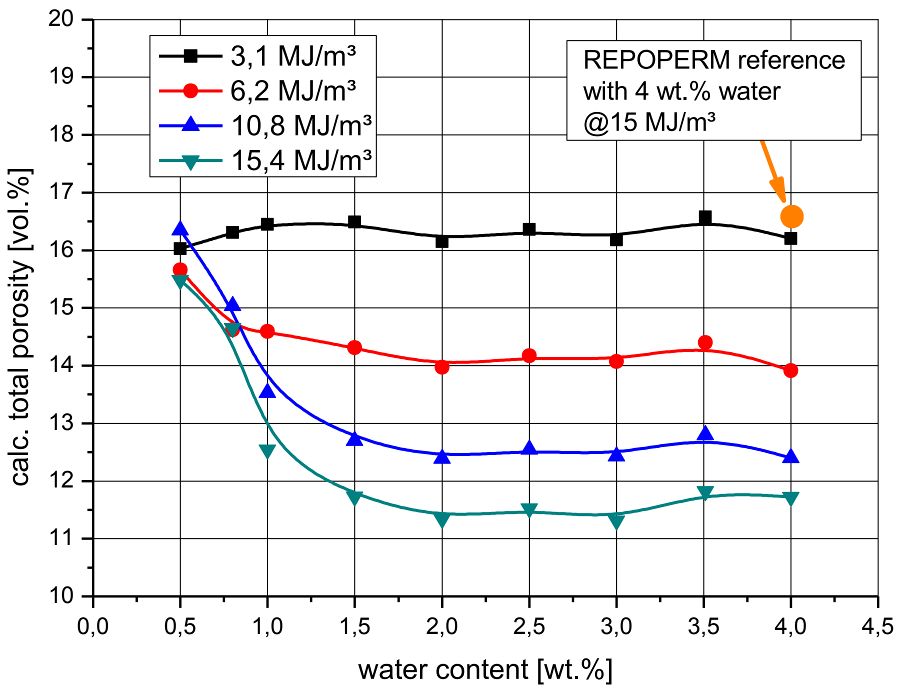

Fig. 2. Gained porosities of the optimized salt fractions compared to the REPOPERM reference

The optimized material achieved the same porosity of the REPOPERM material with only 20% of compaction energy. On another point of view, the optimized material achieved a 30% lower porosity under technical compaction with comparable compaction energy. An addition of 1.0 to 1.5 wt% water suffice for an optimum compaction, which confirms the earlier investigations of AHRENS & HANSEN [4, 5].

BUTCHER [8] and STÜRENBERG [17, 18] demonstrated that crushed salt – bentonite mixtures are achieving lower porosities and lower permeability then pure crushed salt. For the sealing system of the shaft “Saale” (Teutschenthal mine) a support abutment with an in-situ compacted crushed salt – clay mixture was developed. Laboratory tests revealed an optimal mixture of 85 wt% crushed salt (maximum grain size 10 mm) and 15 wt% Friedland Clay Powder with a moisture content of 4.5 wt%. The described material mixture is optimal to achieve a high density with a specific porosity of about 15% by specific compaction energy of about 2.65 MJ/m^3 (one MPE). POPP et al. [16] indicated that corresponding to the specific porosity of 15 vol.%, the initial brine permeability amounts $6{\cdot}10^{-17}$ m^2.

Based on these facts, the FULLER optimized salt mixture (Table, last row) was further investigated to achieve an optimal clay and water content.

3 Optimization and compaction tests on crushed salt-clay mixtures

The aim of this optimization was to find the optimal content of clay (Friedland Clay Powder) and water in the FULLER optimized salt mixture for a minimum porosity after a technical compaction. The compaction procedure was the same like the optimization of the pure salt fractions (described above). The compaction results were determined at 4 different amounts of spec. compaction energies (3.1, 6.2, 10.8 and 15.4 MJ/m^3). In an extensive test series 34 mixtures with different amount of water and clay were investigated. The investigated water content was between 1.1 and 11 vol% (of dry solids), while the clay content varied between 0 and 11.9 vol% (of solids). The figure 3 shows the distribution of the investigated samples within the stated range.

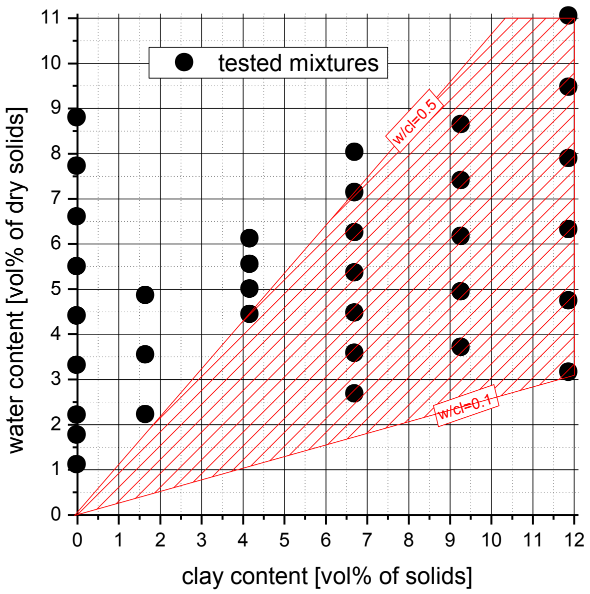

Fig. 3. Investigated salt – clay mixtures (desired range of water/clay mass ratio is red shaded)

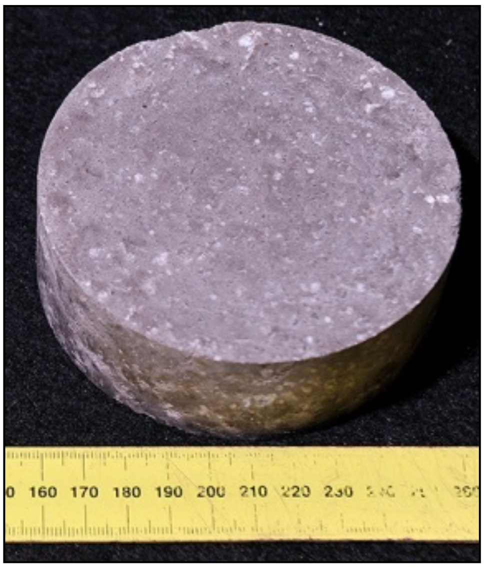

Fig. 4. Typical compacted sample

The distribution of the samples is not homogenous, because the water content of the samples containing clay regards to a specific range of water/clay mass ratio (w/cl) between 0.1 and 0.5. This range (red shaded in Fig) was chosen previously to avoid a high saturation of the clay. During the optimization tests, while the clay content was forced to a minimum, it was not possible to stay within the suggested range of water/clay mass ratio. Due to the water requirement of the fine salt fractions (optimum 4.4 vol% ref. to dry solids), the resulting required water/clay mass ratio raised up to 1.1. Such mixtures contain nearly saturated clay with a minimum of swellability as a safety reserve.

The results of the optimization are shown in Fig. regarding four different compaction energies. The results are visualized in three dimensional diagrams, with the investigated water and clay content at the basis and the gained porosity in the z-direction.

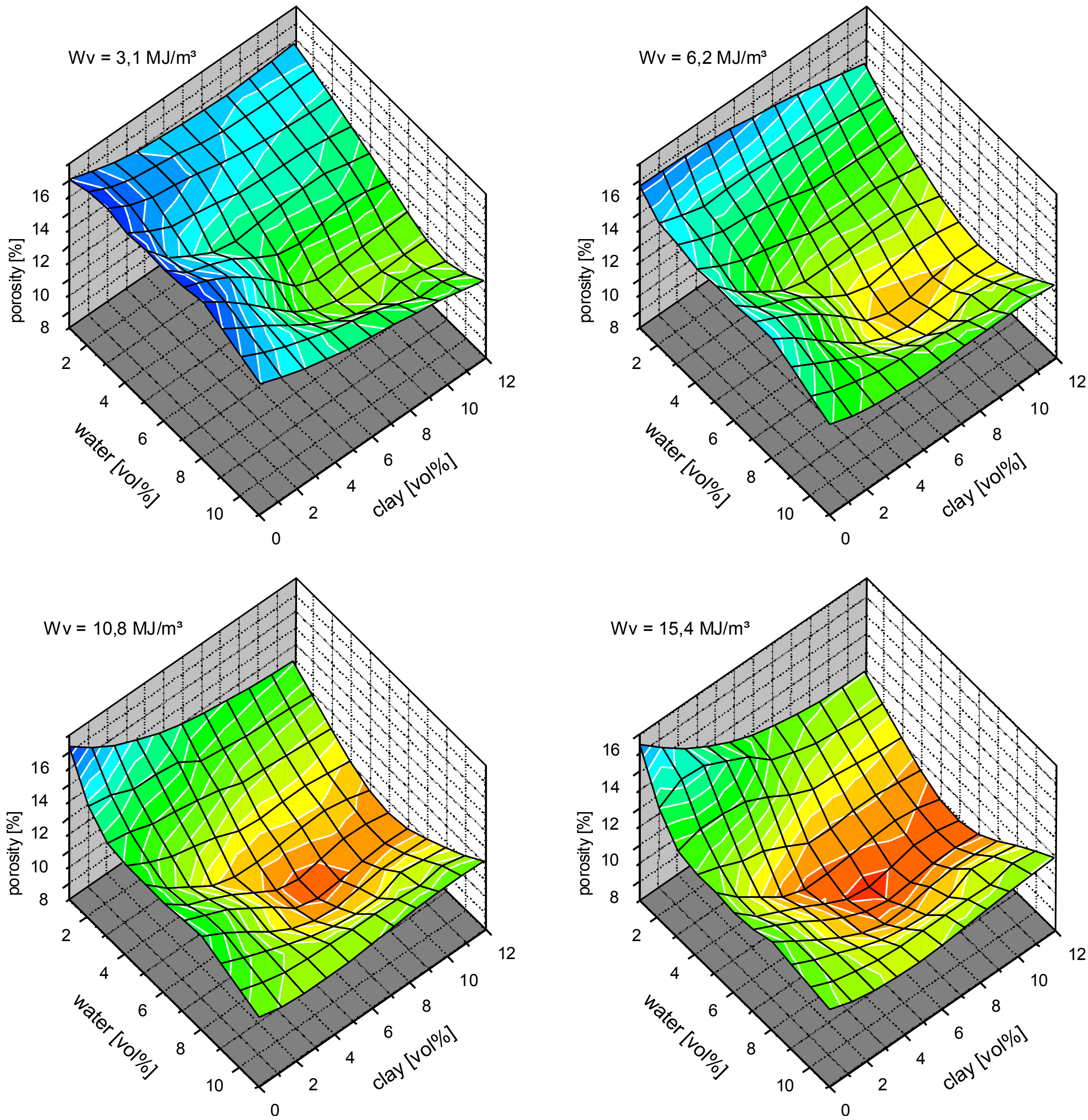

Fig. 5. The results of the optimizing test series

The optimal clay content is approx. 8 vol% (of dry solids), while the optimal water content depends from compaction energy. For compaction energies > 10 MJ/m^3 the optimal water content is approx. 6.2 vol%

(mixture specific ~ 2.7 wt%)and the lowest gained porosity was 7.7 vol.% @ 15.4 MJ/m³. The optimized mixture has a water/clay mass ratio of approx. 0.35.

4 Technical in-situ compaction

The optimized salt/clay mixture was used in in-situ emplacements tests, using conventional vibrating compaction equipment. The material was emplaced in drifted dies and compacted in several layers of 0.1 m thickness. It could be shown, that on site mixing using conventional concrete equipment is problematic. The material tends to segregate and the finer fractions sticking at the mixer equipment. In further investigations intensive mixers (Co. EIRICH) will be used to handle this problem. Also a specific coating of the salt fractions with the clay will be investigated.

In this first emplacement test, the gained porosities where around 15 vol% (integral for the whole dies), which is a good result on the effort of the used conventional compaction equipment. But, the conventional vibrating compactors are only useful for pre-compaction of the optimized mixture. The outcome from the applied compaction energy depends on the applied partial amounts. This means, that a continuing application of low partial energies from vibration compacters will not lead to the densest packaging. The final compaction should be done by a percussive compactor with a much higher partial energy.

During a first field test in summer 2013, a Rapid Impact Compactor (Co. Terra Mix) was used. The Rapid Impact Compactor is hydraulic driven with a guided falling weight of 9'000 kg and controllable dropping heights between 0.4 and 1.2 m. Up to 40 blows per minute are achievable and compaction foots with diameter from 0.8 m to 2.0 m are available.

Fig. 6. Rapid Impact Compactor

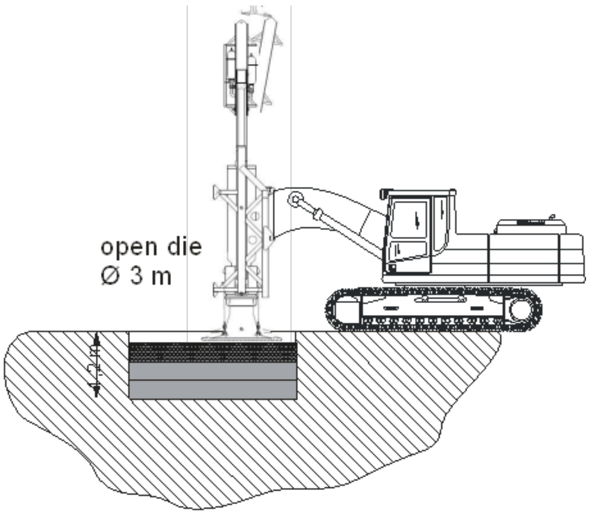

Fig. 7. Future test setup

Currently, a half-scale test setup with the Rapid Impact Compactor is planned (execution 05/2015). The emplacement of the optimized material will be in a drifted die in a hard rock quarry. Alternative, a die made from reinforced concrete can be used.

The optimized material will be compacted in 3 distinctive layers (0.4 m each) and every layer will be composed of 8 conventional pre-compacted sub-layers.

The investigations will be focused on:

- Material behavior during compaction
- Spatial distribution of porosity in the compacted body
- Gas and liquid permeability
- Laboratory tests as reference and quality control

5 Conclusions and prospects

The approach of using swellable clays for doping crushed salt mixtures according to BUTCHER [8] and STÜRENBERG [17, 18] was successfully adopted. Based of the former work of GLAUBACH et al. [11] an empirical optimization process is described to optimize the water and clay content. As a result the water and clay content could be reduced to a range, which is nearly the half of the suggested content according to BUTCHER [8] and STÜRENBERG [17, 18]. Furthermore, a relation between the spec. compaction energy and the optimum water/clay content exists and can be used to adopt the mixture design to the in-situ available compaction procedure.

The lab-scale compaction results are not transferable without considering further demands to large scale practice. Typically underground compaction procedures are generally comparable to the common earthwork compaction. Therefore, several field tests using earthwork machinery for compacting optimized salt/clay-mixtures were described, the results and limits discussed and further demands revealed.

Acknowledgements

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Probabilistic Methods as a Tool Aiding Dimensioning Drift and Shaft Seals for a Repository in Rock Salt

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Abstract: *For repositories in rock salt, demonstrating the integrity of drift and shaft seals is an indispensable part of the long-term safety case. In this study, probabilistic methods are applied to assess the fictitious abutment length for a shaft seal and the effective permeability of a drift seal (dam), i.e. the integral entity for the whole structure including contact zone and damaged salt zone. For the seal permeability, the question arises how to derive it based on permeability measurements with a limited number of samples due to cost restrictions. Furthermore, it is of interest which conclusions can be derived regarding the minimum length of drift seals if the failure probability should be smaller than e.g.* 10^{-4}*. Based on numerical experiments it was demonstrated that small-scale measurements can be upscale using known averaging methods. This suggests that dimensioning can be carried out based on cautions average estimates and the required reliability statement (e.g. about a failure probability smaller than e.g.* 10^{-4}*) can be derived for realistic dam lengths. However, due to the limited amount of data available there are remaining uncertainties concerning the underlying model assumptions.*

1 Introduction

For repositories in rock salt, containment of the radioactive waste is provided by the geologic barrier salt in combination with geotechnical barriers. Therefore, demonstrating the integrity of these barriers is an indispensable part of the long-term safety case (VSG 2012a). While the methodological basis for demonstrating the integrity of the geologic barrier is established (VSG 2012b), there is still R&D need concerning the integrity demonstration of shaft and drift seals (VSG 2013). In this study, probabilistic methods are applied to assess two crucial parameters: The fictitious abutment length for a shaft seal and the effective permeability of a drift seal (dam), i.e. the integral entity for the whole structure including contact zone and damaged salt zone.

For the shaft seal, the structural failure probability can be computed with the help of a limit state function. A Monte-Carlo simulation has been used to approximate the failure probability by determining parameter sets which are beyond the limit state.

For the drift seal permeability, the question arises how to derive it based on permeability measurements with a limited number of samples due to cost restrictions. Furthermore, it is of interest which conclusions can be derived regarding the minimum length of drift seals if the failure probability should be smaller than e.g. 10^{-4}. It is possible to consider the small-scale data as a sample for which its probability distribution is representative for the large-scale behavior. In other words, the whole structure is considered as a homogeneous entity in which the permeability complies with the distribution derived from the small scale measurements. According to the distributions of the material parameters quantiles can be estimated as characteristic values and inserted into the limit state function. However, using just the 10^{-4} quantile of the measurements leads to demands concerning the length of a drift seal which are neither plausible nor viable.

It is, however, possible to account for heterogeneities in the seal structure by means of geostatistical methods. Then, the measurements are used not only for deriving a probability distribution but also of a so-called variogram structure which characterizes the spatial behavior of the permeability. Numerous realizations of the seal can be sampled, each accounting for the derived probability distribution and the variogram, thus reproducing the spatial behaviour (heterogeneity) found in the sample. The effective large-scale permeability can then be calculated based on flow simulations.

2 Abutment length of the shaft seal

The influence of different material parameters on the design length of a sealing structure is studied. A limit-state function is used that models the failure due to shear forces in the contact zone and due to flexural tension of the sealing structure. This limit state can be interpreted as the minimal necessary length of a fictitious abutment. The values for shearing strength, tensile strength and the radical strain coefficient are not known. Their probability distribution was estimated from given data. A normal distribution approximates these data well enough.

Given the probability distributions of the input parameters a fully probabilistic analysis is carried out. This consists of (i) generating an input sample with respect to the assumed probability laws of the input factors, (ii) computing the limit state for these realizations, and (iii) obtaining the failure probability by a Monte-Carlo estimate. The results are reported in the form of complementary cumulative distribution functions that demonstrate the failure probability with respect to the abutment length. For a 40 m length, the failure probability is below 10^{-3}. Due to the normal assumptions, the tail probabilities are overestimated.

3 Drift seal permeability

3.1 Methodology

The overall aim is to estimate a probability distribution of the large-scale effective permeability (integral over the whole structure including contact zone and damaged rock salt) based on small-scale data in order to derive reliability statements. The methodology is as follows:

- Univariate analysis of the small-scale permeability measurements (histograms, normal probability plots)
- Transformation into elliptic cylinder coordinates in order to better account for the shape of the seal and, thus, to allow better variography
- Transformation of permeability data into univariate normally distributed data
- Empirical variograms for the elliptic system
- Derivation of theoretical variograms matching the empirical ones
- Check for bivariate normal distribution (as empirical justification for the assumption of a multivariate normal distribution)
- Geostatistical simulation (set of realizations, each representing a possible spatial permeability distribution)
- Back-transformation to original data scale
- Interpolation of the data on the computational grid

- Flow calculation und computation of the effective (large-scale) permeability for each realization
- Output statistics for the calculated large-scale permeabilities

3.2 The Data

The data (Eberth & Müller-Hoeppe 2009) are based on permeability measurements carried out at a dam building at the Asse II mine (Fig. 1). In the material used, small-scale imperfections can be detected (Fig. 2).

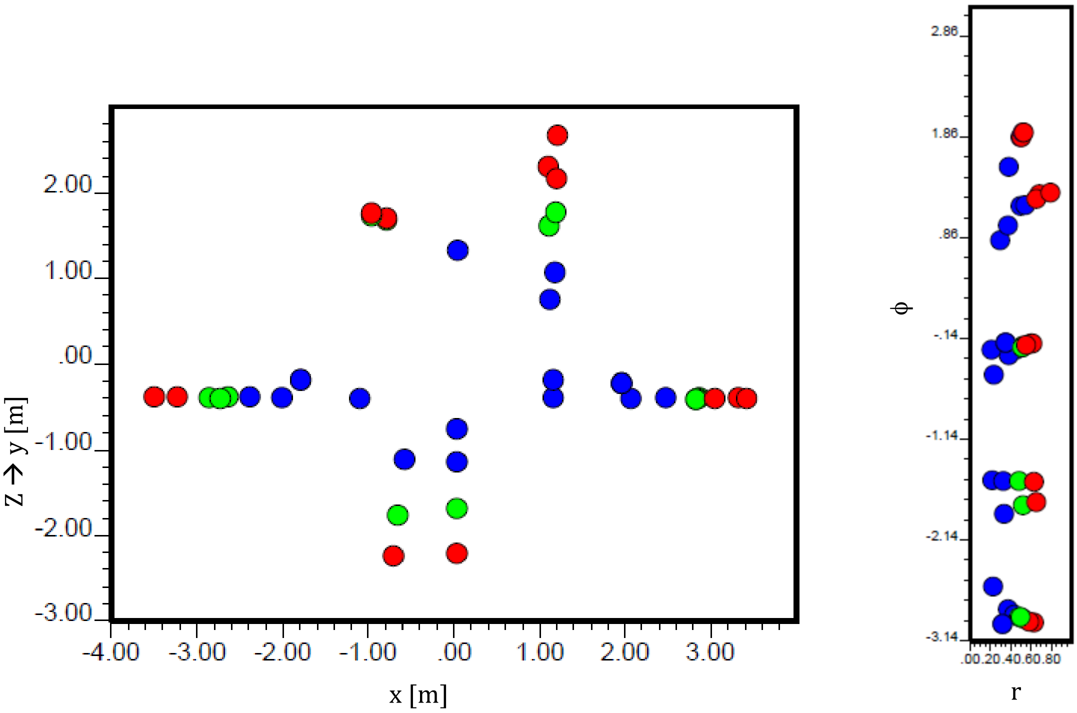

Fig. 1. Measurement locations (projection on vertical plane, blue = dam construction, green = contact zone, red = disturbed rock salt). Left: Cartesian system, right: transformed system)

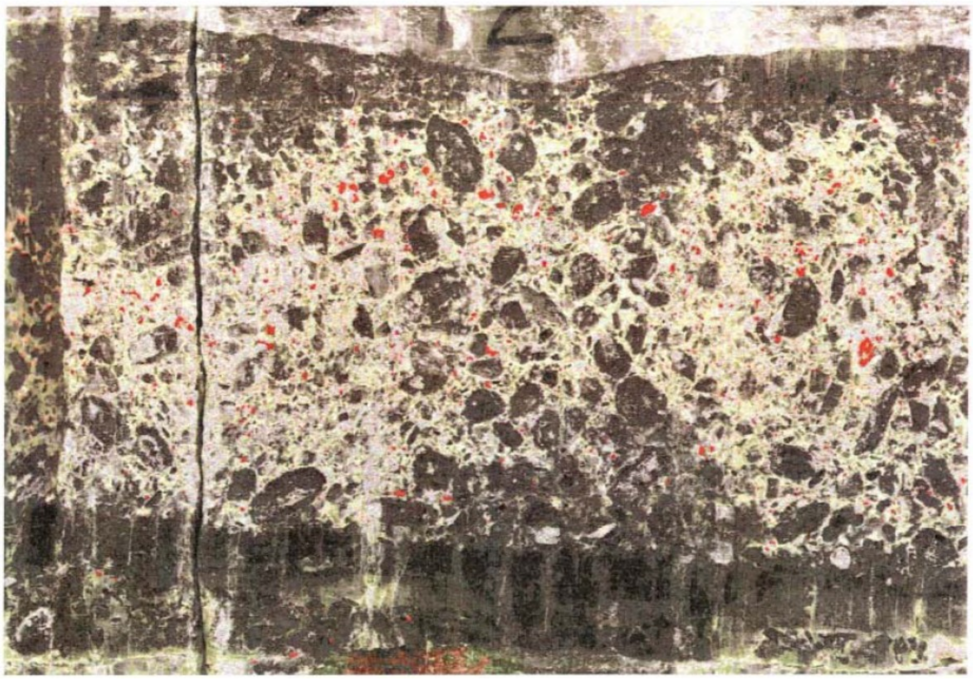

Fig. 2. Imperfections in salt-concrete (DMT 2005)

3.3 Geostatistical simulation results and estimated permeability distribution

In Fig. 3, an example for a simulated permeability distribution is presented.

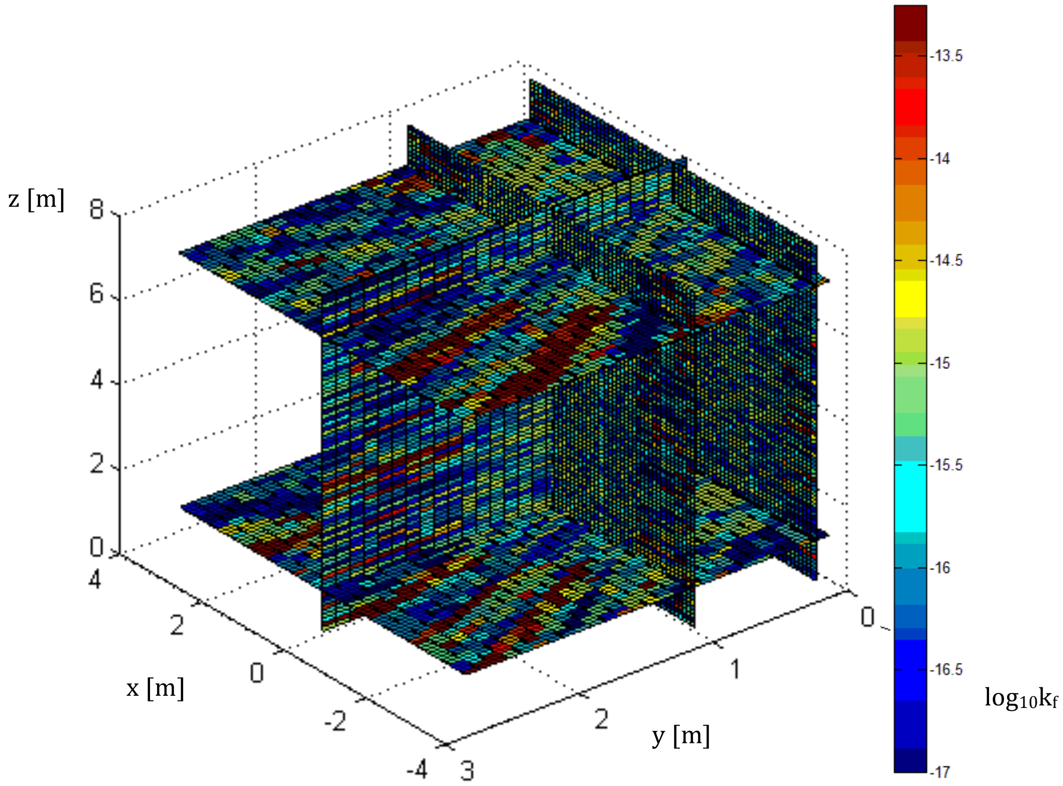

Fig 3. Permeability distribution for one realization (half of model area)

Fig. 4 shows empirical pdfs for the permeability obtained on the basis of flow calculations for all realizations.

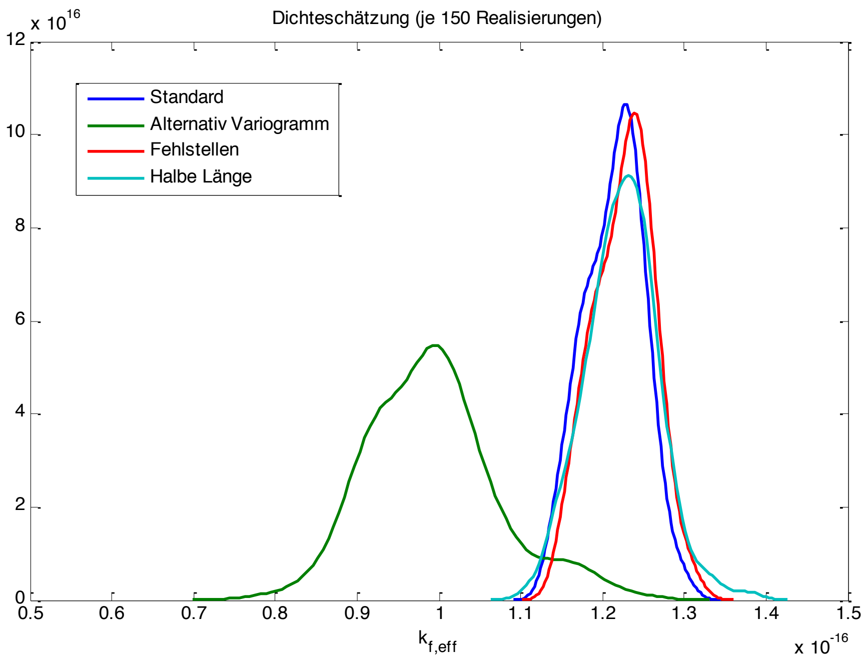

Fig. 4. Empirical pdfs for large-scale permeability (blue = original variogram, green = alternative variogram, red = with imperfections, light blue = dam length 4 m

Due to the limited amount of data, the variography was rather fragile. Therefore, a sensitivity study was undertaken assuming a different theoretical variogram. In addition, calculations were carried out for a dam of half the length (4 m instead of 8 m) as well as for permeability distributions superimposed with imperfections as seen in Fig. 2. The pdf is rather insensitive against the dam length and the assumption of imperfections. Even the alternative variogram does not yield significant changes (note the linear scale for the permeability values).

3.4 Conclusions

Based on numerical experiments it was demonstrated that the small-scale measurements can be upscale using averaging methods known from literature (Journel et al. 1986). This suggests that dimensioning can be carried out based on cautions average estimates and the required reliability statement (e.g. about a failure probability smaller than 10^{4}) can be derived for realist dam lengths. However, due to the limited amount of data available there are remaining uncertainties concerning the underlying model assumptions. It is all the more important to carry out sensitivity studies during which the results are tested against changes in the model assumptions. Also, it should be checked whether *a priori* information, such as data from QA during construction, or radar measurements, could be used to substantiate these assumptions.

Acknowledgements

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Thermal Loading of Bentonite: Impact on Hydromechanics and Permeability

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Due to its favorable properties, in particular, low permeability and swelling capacity, bentonite has been favored as an engineered barrier and backfill material for the geological storage of radioactive waste. To ensure safe long-term performance it is important to understand any changes in these properties when the material is subject to heat emitting waste. As such, this study will investigate the hydro-mechanical response of bentonite under multi-step thermal loading subject to a constant volume boundary condition. The experimental set up allows continuous measurements of hydraulic and mechanical response during each phase of the thermal cycle.

The constant volume cell was placed inside an oven and connected to a hydraulic system with the water reservoir located externally. A pressure gradient of 4 MPa was placed across the sample for the duration of the test in order to map the evolution of permeability. After initial hydration of the bentonite, in this case signified by reaching the asymptote in total stress, the temperature was raised in 20°C increments from 20 to 80°C followed by a final 10°C step to reach 90°C. Each temperature was held constant for at least 7-10 days to allow the stresses and hydraulic transients to equilibrate.

This data set will provide an insight into the hydromechanical behavior of the bentonite and the evolution of its permeability when exposed to elevated temperatures.

5 ALL SCIENTIFIC ASPECTS OF THE NUCLEAR WASTE DISPOSAL SAFETY CASE

Radionuclide Solubility Control by Solid Solutions

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The migration of radionuclides in the geosphere is to a large extend controlled by sorption processes onto minerals and colloids. On a molecular level, sorption phenomena involve surface complexation, ion exchange as well as solid solution formation. The formation of solid solutions leads to the structural incorporation of radionuclides in a host structure. Such solid solutions are ubiquitous in natural systems – most minerals in nature are atomistic mixtures of elements rather than pure compounds because their formation leads to a thermodynamically more stable situation compared to the formation of pure compounds. However, due to a lack of reliable data for the expected scenario at close-to equilibrium conditions, solid solution systems have so far not been considered in long-term safety assessments for nuclear waste repositories.

In recent years, various solid-solution aqueous solution systems have been studied. Here we present state-of-the art results regarding the formation of $(Ra,Ba)SO_4$ solid solutions. In some scenarios describing a waste repository system for spent nuclear fuel in crystalline rocks ^{226}Ra dominates the radiological impact to the environment associated with the potential release of radionuclides from the repository in the future. The solubility of Ra in equilibrium with $(Ra,Ba)SO_4$ is much lower than the one calculated with $RaSO_4$ as solubility limiting phase. Especially, the available literature data for the interaction parameter W_{BaRa}, which describes the non-ideality of the solid solution, vary by about one order of magnitude (Zhu, 2004; Curti et al., 2010)). The final ^{226}Ra concentration in this system is extremely sensitive to the amount of barite, the difference in the solubility products of the end-member phases, and the degree of non-ideality of the solid solution phase.

Here, we have enhanced the fundamental understanding regarding (1) the thermodynamics of $(Ra,Ba)SO_4$ solid solutions and (2) the kinetics of Ra uptake. A novel approach combining atomistic simulations, radiochemical batch-type laboratory experiments and modern analytical techniques supported by thermodynamic modeling. The kinetic results indicate a very fast uptake of Ra which leads to a concentration plateaue with a reduction of more than 99% of the Ra concentration. The thermodynamic modeling indicate a good agreement of the apparent final Ra(aq) equilibrium concentration from experimental data at RT with the computed W_{BaRa}.

A comprehensive model was developed for the thermodynamic definition of a $(Ra,Ba)SO_4$, which can now be applied to the performance assessment of highl level nuclear waste repository systems.

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Barrier Function of a Corroding Iron Based Container

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Thick-walled cast-iron containers encapsulate highly radioactive waste in a deep geological repository and provide mechanical stabilization against lithostatic pressure and shear forces. Intrusion of water and corrosion, however, limits container integrity to a certain time period of several hundreds (BMU, 2010) to several thousands of years (NAGRA, 2002). Repository concepts in crystalline rock therefore include an additional copper coating, providing chemical resistance against water corrosion.

Safety analyses usually do not consider chemical barrier functions of corroding cast iron containers. Anaerobic iron corrosion establishes strongly reducing conditions with redox potentials at the water stability limit (see Fig.). H_2 produced upon anaerobic iron corrosion strongly inhibits the radiolysis driven corrosion of used nuclear fuel (Spahiu et al., 2000). Simultaneously, iron converts into secondary corrosion phases such as $Fe(OH)_2$, green rust phases, magnetite etc.. In a disposal concept as discussed in Germany for horizontal waste emplacement of heat-producing radioactive waste that involves thick-walled (e.g. POLLUX®) containers the Fe-inventory (ca. 270.000 Mg) is by far exceeding the forecasted inventory of uranium, the main component of used nuclear fuel (ca. 15.000 Mg) (Bollingerfehr et al., 2011). It is obvious that the corroding container must be considered as a potential additional chemical barrier in nuclear waste disposal concepts, that retards migration or even causes immobilization of dissolved radionuclides.

We have investigated the interactions of various radionuclides representative for heat-producing radioactive waste with various iron mineral phases. According to our results and in agreement with published studies radionuclides do react with secondary iron corrosion products via various mechanisms:

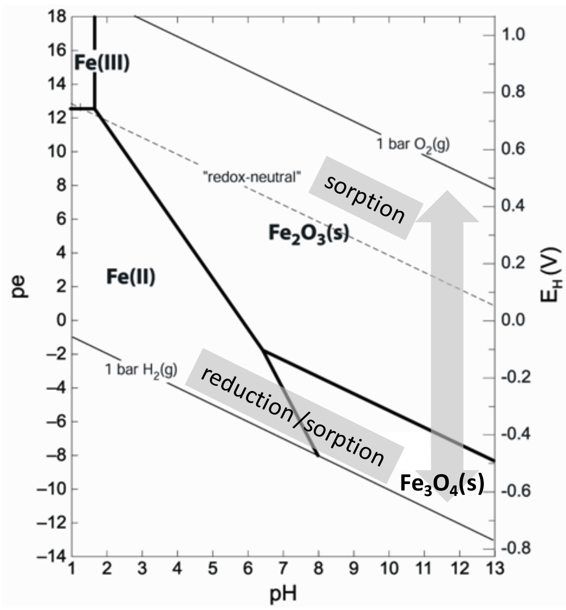

Fig. 1. Possible radionuclide retention processes schematically indicated in a simplified redox diagram of iron (10^{-4} mol L^{-1} total Fe concentration)(Metz et al., 2012)

- surface induced reduction of actinides (e.g. U, Np, Pu) and fission products (e.g. Tc, Se) by which they transform to reduced and poorly mobile species
- surface complexation of redox inactive radionuclides (e.g. Am) at reactive surfaces by which migration is retarded
- formation of secondary solid phases (e.g. $FeSe_x$)

Recent experimental findings on radionuclide retention on iron corrosion products will be summarized and their relevance for safety case consideration will be discussed.

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Mechanical Properties, Mineralogical Composition, and Micro Fabric of Opalinus Clay – Sandy and Shaly Facies (Mont Terri, Switzerland)

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For the safe disposal of high-level radioactive waste, different host rocks are currently considered. The favorable properties of claystone are low permeability, retention capacity for some radionuclides, and the ability to self-seal cracks, e.g. by swelling or time-dependent compaction creep. For the understanding of the long-term behavior of clay host rocks, the interaction between mechanical behavior, micro fabric, and mineral composition has to be understood (Bock et al., 2010). In the international research project *Mont Terri* (Switzerland) the Opalinus Clay (Jurassic Formation) is investigated in an underground rock laboratory (URL).

In the present study the relationship between mechanical, mineralogical and micro fabric properties were studied on representative samples of the sandy and shaly facies of the Opalinus Clay (OPA) from Mont Terri.

The mineral composition of all samples was analysed by using a complex mineral phase analysis. Therefore, the results of the X-ray diffraction, X-ray fluoreszence, organic and inorganic carbonate analysis (LECO) were adjusted with each other.

In the case of the sandy facies (OPA) the mechanical strength inrcreases with increasing carbonate content. Here small carbonate particles form the matrix and act as stabilisator. The carbonates of the shaly facies (OPA), on the other hand, are mainly fossil fragments (e.g. shells) aligned parallel to bedding. These large carbonate particles are acting as predetermined breaking surfaces. Hence, in the case of shaly facies (OPA) the mechanical strength decreases with increasing carbonate content.

Image Analyses (Fiji®) of scattering electron microscope images of polished sections proved the determined microstructural differences. Besides, carbonate particles in the sandy facies are mostly isometric, in contrast carbonates of the shaly facies show different shapes. This is explained further in terms of the aspect ratio.

The mechanical tests were carried out as triaxal strength test (Kármán cell). The samples were analysed parallel and perpendicular to the bedding. For both cases the failure strength of σ_{fail} was twice as high as it was found for the samples of the shaly facies.

The results improve the understanding of the mechanical properties and behavior of claystones, particulary considering the variability of mineral composition and micro fabric.

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Evolution of Cement Based Materials in a Repository for Radioactive Waste and their Chemical Barrier Function

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The use of cementitious materials in nuclear waste management is quite widespread. It covers the solidification of low/intermediate-level liquid as well as solid wastes (e.g. laboratory wastes) and serves as shielding. For both high-level and intermediate-low level activity repositories, cement/concrete likewise plays an important role. It is used as construction material for underground and surface disposals, but more importantly it serves as barrier or sealing material.

For the requirements of waste conditioning, special cement mixtures have been developed. These include special mixtures for the solidification of evaporator concentrates, borate binding additives and for spilling solid wastes. In recent years, low-pH cements were strongly discussed especially for repository applications, e.g. (Céline CAU DIT COUMES 2008; García-Siñeriz, et al. 2008). Examples for relevant systems are Calcium Silicate Cements (ordinary Portland cement (OPC) based) or Calcium Aluminates Cements (CAC). Low-pH pore solutions are achieved by reduction of the portlandite content by partial substitution of OPC by mineral admixtures with high silica content. The blends follow the pozzolanic reaction[2] consuming $Ca(OH)_2$. Potential admixtures are silica fume (SF) and fly ashes (FA). In these mixtures, super plasticizers are required, consisting of polycarboxilate or naphthalene formaldehyde as well as various accelerating admixtures (García-Siñeriz, et al. 2008).

The pH regime of concrete/cement materials may stabilize radionuclides in solution. Newly formed alteration products retain or release radionuclides. An important degradation product of celluloses in cement is iso-saccharin acid. According to Glaus 2004 (Glaus and van Loon 2004), it reacts with radionuclides forming dissolved complexes. Apart from potentially impacting radionuclide solubility limitations, concrete additives, radionuclides or other strong complexants compete for surface sites for sorbing onto cement phases.

In Germany, the alteration of cement and the mobilization/immobilization of radionuclides were studied in laboratory and in full-scale experiments. Most of these experiments were performed in relevant salt brines. The results on following investigations have been obtained:

- Mechanical properties
- Element analyses
- Radioactive element and non-radioactive constituents distributions
- Scanning electron microscopic analyses
- Mineralogical analyses by
    - Thermogravimetric measurements
    - Powder X-ray diffraction analyses
    - Raman spectroscopy.
    - XANES/EXAFS analyses.

[2] $Ca(OH)_2 + H_4SiO_4 \rightarrow Ca^{2+} + H_2SiO_4^{2-} + 2\ H_2O \rightarrow CaH_2SiO_4 \cdot 2\ H_2O$

A résumé of the results including modeling will be presented and the impact on the radioactive waste disposal safety will be demonstrated.

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Iron Corrosion in Concentrated Saline Solutions at Elevated T in High-Level Radioactive Waste Salt rock Repositories: a Thermodynamic Study

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Abstract: *Predictions of the evolution of the interfacial chemistry of corroding iron in high salinary milieus at temperatures of 25°C to 100°C based on an extended thermodynamic data basis for the system Fe(II)-Fe(III)-Na-K-Cl-Mg-Ca-SO_4-S-CO_3-H_2 are presented. Future research directions are discussed.*

1 Introduction

The choice of a host rock for a deep repository for the final disposal of high-level radioactive waste is subject of ongoing debate in Germany. One candidate type is salt rock, which is present beneath the entire north of Germany. One favorite property of salt rock is its impermeability for aqueous solutions, which, if the repository is constructed and managed properly, renders the probability of direct contact between waste containments and aqueous solution very low. However, as it cannot be excluded altogether that the repository undergoes an unforeseen evolution, a situation must be considered where the steel containment of high-level radioactive waste comes into contact with aqueous solution. Due to the nature of the host rock, this solution will be a high ionic strength, saturated salt solution. Due to the high-saline nature of the solution, an appropriate method for the calculation of aqueous species activities needs to be applied. The overall performance and longevity of the steel containment will ultimately be governed by metallic corrosion in this environment. The corrosion process can only be understood taking thermodynamic relations, reaction kinetics and transport processes into account. Thus, thermodynamic aspects of Fe(II) and Fe(III) in concentrated solutions of the hexary oceanic salt system constitute an essential frame for modeling and predicting corrosion processes in repositories in salt rock formations upon contact with salt solution and traces of oxygen trapped in the backfill salt initiate an aerobic corrosion phase which further evolves under reducing conditions where corrosion proceeds with hydrogen evolution (Wang et al 2001). The stability of corrosion products generated upon contact of steel containers with saline solutions may have a large impact on the transport of radioactive contaminants that have been mobilized from the waste matrix. For instance, zero-valent iron was tested for the removal of uranium by reductive precipitation of uranyl to less soluble U(IV)-compounds and the adsorption of uranyl cations on iron corrosion products (Farrel et al 1999). The removing rate by reduction of U(VI) to slightly soluble U(IV) species depends greatly on the activity of solution species and the nature of the corrosion products .

The corrosion rate can be controlled by the nature of precipitation products, which may be constituted by oxy-chlorides, sulfates, sulfides and carbonates, depending on the composition, temperature and pressure of the salinary surrounding media and the availability of water (by its transport rate into the direct proximity of the waste canister). This work presents the progress achieved in the construction of a self-consistent data base for the prediction of the thermodynamic stability of the canister material.

2 Modelling

The stability of iron corrosion products for given T, P and chemical composition of the salinary media in salinary solutions, without regarding a priori kinetic aspects, is defined by the Gibbs free energy of dissolution reaction:

$$\Delta_r G(T) = \sum_i \nu_i \Delta_f G^0{}_i(T) + RT\ln\prod_i (m_i\gamma_i)^{\nu_i} + RT\ln\prod_j f_j^{\nu_j} = 0 \qquad (1)$$

where $\Delta_f G^0{}_i$ is the standard partial molar Gibbs energy of formation of the species i at temperature T and a_i is the corresponding activity, given by $m_i\gamma_i$, where m_i and γ_i are the molality and the activity coefficient respectively and f_j is the fugacity of gas j. This equation synthetizes the two fields in which this work is structured. The first one consists in the calculation of the Gibbs free energy of the reaction at $T \neq T_0$. This is accomplished from the standard thermodynamic properties of the reaction species as:

$$\Delta_r G^0(T) = \sum_i \nu_i \Delta_f G^0{}_i(T) = \Delta_r G^0{}_i(T_0) + \int_{T_0}^{T} \Delta_r C_p{}^0 dT - T\int_{T_0}^{T} \frac{\Delta_r C_p{}^0}{T} dT - (T - T_0)\Delta_r S^0(T_0) \qquad (2)$$

The activity of solution species, on the other hand, is calculated by the Pitzer model (Pitzer 1991). This model describes the thermodynamic behavior of electrolyte solutions at high ionic strengths by adding a virial term accounting for non-electrostatic ion interactions to the expression extended Debye-Hückel equation for the free excess energy. Thus, expressions for the activity and osmotic coefficients of the form:

$$\ln\gamma_i, \phi = f_s(I) + f_s(\alpha_{ij}^{(1)}, \alpha_{ij}^{(2)}, \beta_{ij}^{(0)}, \beta_{ij}^{(1)}, \beta_{ij}^{(2)}, C_{ij}^{\phi}, \theta_{ii}, \psi_{ijk}) \qquad (3)$$

can be derived, where $f_s(I)$ is a function of the ionic strength and f_s ($\alpha_{ij}^{(1)}$, $\alpha_{ij}^{(2)}$, $\beta_{ij}^{(0)}$, $\beta_{ij}^{(1)}$, $\beta_{ij}^{(2)}$, C_{ij}^{ϕ}, θ_{ii}, ψ_{ijk}) is a function of binary ($\alpha_{ij}^{(1)}$, $\alpha_{ij}^{(2)}$, $\beta_{ij}^{(0)}$, $\beta_{ij}^{(1)}$, $\beta_{ij}^{(2)}$, C_{ij}^{ϕ}) and ternary interaction parameters (θ_{ii}, ψ_{ijk}), which are calculated by regression of experimentally determined activities and osmotic coefficients. The temperature dependence of interaction parameters, on the other hand, is expressed by the generalized function:

$$Y(T) = Y(T_0) + a(1/T - 1/T_0) + b\ln(T/T_0) + c(T - T_0) \qquad (4)$$

where Y: $\beta_{ij}^{(0)}$, $\beta_{ij}^{(1)}$, $\beta_{ij}^{(2)}$, C_{ij}^{ϕ}. θ_{ii}, ψ_{ijk} and T_0 = 298.15 K. This expansion suits very well in the range of investigated temperatures.

For modeling of iron corrosion in high-level radioactive waste rock repositories, a set of binary and ternary interaction parameters of Fe(II) and Fe(III) in the system K-Na-Mg-Ca-Cl-SO_4-S-SO_4-CO_3 valid in the temperature range of 25°C to 100°C is mandatory. Pitzer parameters for ions of the system K-Na-Mg-Ca-Cl-SO_4-CO_3-H_2O can be taken from the data basis THEREDA. Interaction parameters of Fe(III) and Fe(II) in the system K-Na-Mg-Ca-Cl-SO_4-H_2O at 25°C are taken from the literature (Rumyantsev et al 2004) (Moog et al 2004). In this work, temperature coefficients of interaction parameters for Fe(II) in the system K-Na-Mg-Ca-Cl-SO_4-H_2O were calculated from data obtained by an extensive literature review of solubility and isopiestic data and own isopiestic experiments. Experimental details are described elsewhere (Scharge et al 2013).

3 Results and discussion

The equilibrium constants for the formation reactions of possible corrosion products were calculated using thermodynamic properties of basis species and corrosion products selected from an extensive literature review (see Table A1). They were added in the parameter file for Geochemist´s Workbench as ln K(T) in the form of eq.(4).

The temperature coefficients of Pitzer parameters for the system Fe(II)-K-Na-Mg-Ca-Cl-SO_4-H_2O, on the other hand, were calculated from a large set of solubility data in binary and ternary systems reported in the literature. This procedure requires the input of solubility constants of Fe(II)-salts and their temperature dependency. Because in most of the cases these data are not available, they are introduced as new unknown variables to be calculated by regression of experimental solubility data together with the temperature coefficients of Pitzer parameters. Calculated parameters are shown in Table A2. The use of this set of parameters does not consider the formation of iron complexes.

The environmental condition, under which the corrosion of iron canisters in long-term repositories takes place, changes with time. In a first stage, corrosion occurs under hot (100-150°C) and aerobic conditions. As the initially trapped oxygen is consumed by corrosion (partial electrochemical reduction reaction) or microbiological processes, the repository environment becomes anaerobic and remains so indefinitely. Eq.(5) and (6) represents the total corrosion reaction for aerobic and anaerobic conditions respectively:

$$Fe + \frac{1}{2}\, O_2(aq) + H_2O \rightarrow Fe(OH)_2 \qquad (5)$$

$$Fe + 2\, H_2O \rightarrow Fe(OH)_2 + H_2(aq) \qquad (6)$$

where $Fe(OH)_2$ is considered the primary corrosion product. The thermodynamic stability of the corroding interface was modelled by using the titration modus (React) of the Geochemist´s Workbench tool after adding the calculated log K(T) and Pitzer parameters into the parameter file. In our reaction model shown in Fig. 1 the salty phase in contact with the corroding interface is regarded as an open system. For modeling purposes, $Fe(OH)_2$ is injected in the system, while stoichiometric amounts of O_2 are extracted. The enclosed air is regarded as a gas buffer with an unconstrained fugacity.

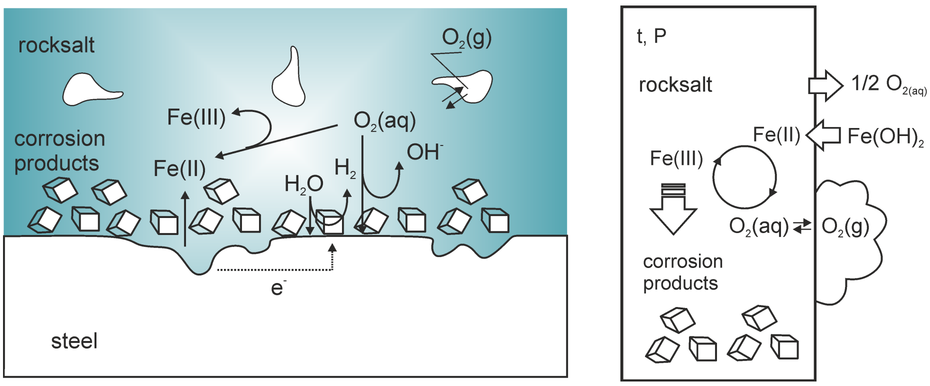

Fig.1. Schematic diagram of the reaction model for the corrosion of iron

Fig. 2 shows the masses of iron minerals formed by $Fe(OH)_2$ titration for three different brine compositions at 25°C and 70°C at 1 bar. In particular, the transition of aerobic to an anaerobic environment is given by consumption of initially present $O_2(aq)$ followed by its further extraction from water and the corresponding generation of H_2 according to $H_2O - \frac{1}{2}\, O_2(aq) \rightarrow H_2(aq)$. Further assumptions are unhindered equilibration of the couples Fe^{2+}/Fe^{3+} and HS^-/SO_4^{2-} and an initial fugacity of 0.2 for oxygen in the enclosed air.

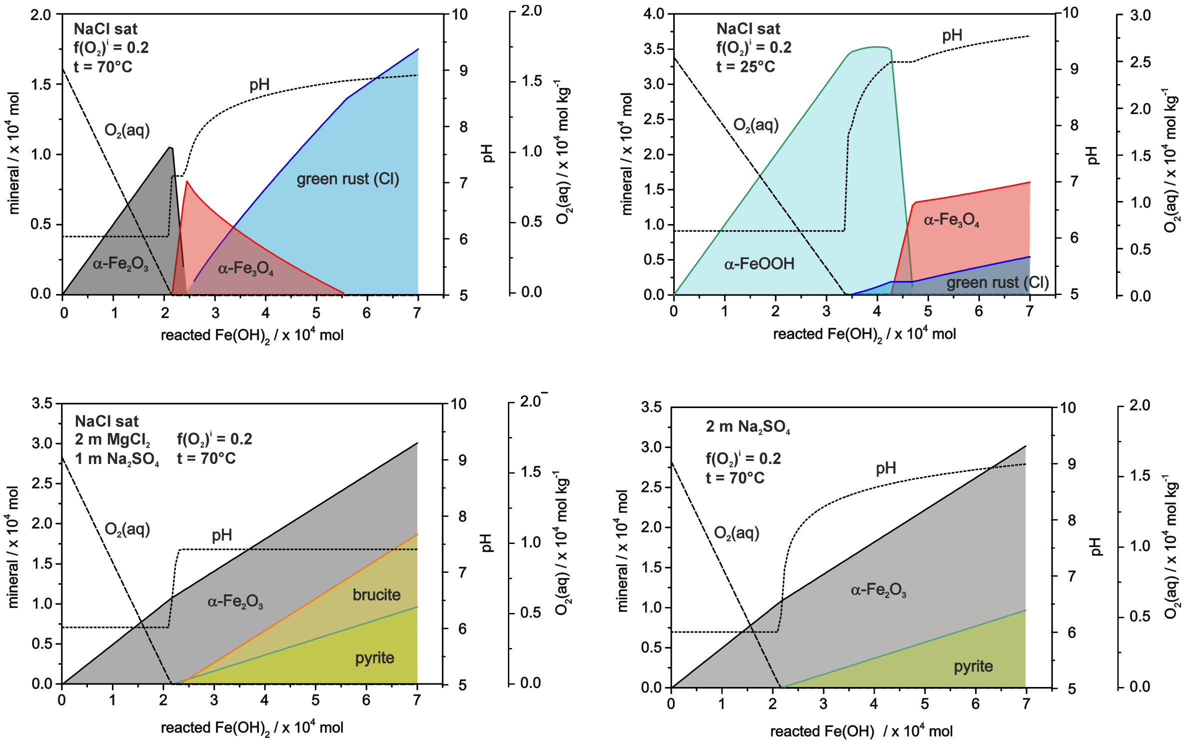

Fig. 2. Stability diagrams of corrosion products for different composition and temperatures of the salinary media.

It can be observed, that hematite (α-Fe_2O_3) is the predominant corrosion product in saturated NaCl-brine at 70°C in the aerobic stage. Its formation is connected with the initial consumption of dissolved oxygen. Further iron dissolution forces the generation of a reducing environment and the conversion of hematite, a first into magnetite and into green rust (Cl) later. This reaction picture contrasts with that at 25°C, where the main product in the initial oxidizing environment is goethite; hematite and green rust (Cl) coexist in the anaerobic phase. The presence of sulfate in the brine inhibits the formation of magnetite and green rust (Cl) in the anaerobic environment. Instead, pyrite appears as coexisting corrosion product. The presence of $MgCl_2$ in the brine acts as buffer, limiting the increase of interfacial pH by formation of brucite. The formation of pyrite in the reducing environment is conditioned to an equilibrium of the redox couple HS^-/SO_4^{2-}, which is conceivable in the presence of sulfate reducing bacteria.

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Appendix

Table A1: Set of interaction Pitzer parameters for the system Fe(II)-K-Na-Mg-Ca-Cl-SO_4-H_2O. (Muñoz et al, 2013)

basis species	Δ_fG^0 [kJ mol^{-1}]	S^0 [J mol^{-1}]	C_p^0 [J mol^{-1}]
Fe^{2+}	-90.72	-102.17	-23.0
Fe^{3+}	-16.23	-282.40	-108.0
O_2(g)	0.00	205.15	29.4
H_2O(l)	-228.58	188.84	33.6
H^+	0.00	0.00	0.0
Cl^-	-131.22	56.60	-120.57
SO_4^{2-}	-744.00	18.50	-261.94
CO_3^{2-}	-527.90	-50.00	-284.86
HS^-	12.24	67.00	-92.01
HCO_3^-	-586.85	98.40	-33.12
corrosion products			
α-Fe_2O_3	-744.45	87.40	103.9
γ- Fe_2O_3	-727.83	93.04	104.7
α-FeOOH	-489.54	59.70	74.4
γ-FeOOH	-479.88	65.08	69.1
α-Fe_3O_4	-1012.719	145.89	150.78
β-FeOOH	-481.7	53.8	91.416
$Fe(OH)_2$	-483.99	87.864	86.312
$FeCO_3$(cr)	-679.557	95.537	82.45
FeS_2(cr)	-187.317	52.916	62.124
FeS(cr)	-125.155	60.321	50.888
$Fe^{II}_4Fe^{III}_2(OH)_{12}SO_4$	-3819.40		
$Fe_2(OH)_3Cl$	-923.50		
$Fe^{II}_3Fe^{III}(OH)_8Cl$	-2145.00		
$Fe^{II}_4Fe^{III}_2(OH)_{12}CO_3$	-3588.00		

Table A2: Set of interaction Pitzer parameters for the system Fe(II)-K-Na-Mg-Ca-Cl-SO_4-H_2O. (Moog et al, 2004, Rumyantsev et al 2004).

System	Pitzer parameter	T = 25°C	a	b	c
Fe(II)-Cl	$\beta_{ij}^{(0)}$	0,37324	0,00630	-0,00719	-0,00171
	$\beta_{ij}^{(1)}$	1,13499	0,00012	-0,01910	0,01722
	$\beta_{ij}^{(2)}$	0,00000	0,00000	0,00000	0,00000
	C_{ij}^{φ}	-0,02152433	0,03862	0,01409	0,00008
Fe(II)-SO_4	$\beta_{ij}^{(0)}$	0,28863	0,16231	-1,99485	0,00784
	$\beta_{ij}^{(1)}$	2,70661	0,04271	-6,59538	0,01765
	$\beta_{ij}^{(2)}$	-42,00000	0,00043	-0,05944	0,23281
	C_{ij}^{φ}	0,00748	0,14951	1,42619	-0,00533
Fe(II)-Na-Cl	θ_{ii}	0,12956	-0,01134	0,66065	-0,00246
	ψ_{ijk}	-0,02119	-0,04760	-0,25691	0,00084
Fe(II)-Na-SO_4	ψ_{ijk}	-0,06550	-0,00411	0,76916	-0,00154
Fe(II)-K-Cl	θ_{ii}	-0,028460	-0,01415	2,50679	-0,00680
	ψ_{ijk}	-0,01405	-0,05412	0,71357	-0,00220
Fe(II)-K-SO_4	ψ_{ijk}	-0,02598	0,00633	-0,92055	0,00451
Fe(II)-Mg-Cl	θ_{ii}	0,07310	0,00514	-0,72877	0,00382
	ψ_{ijk}	-0,02436	0,01223	-2,15550	0,00638
Fe(II)-Mg-SO_4	ψ_{ijk}	-0,01342	-0,00603	1,09739	-0,00367
Fe(II)-Ca-Cl	θ_{ii}	0,08581	-0,00003	0,00507	-0,02459
	ψ_{ijk}	-0,02423	-0,00015	0,02200	0,00340
Fe(II)-Ca-SO_4	ψ_{ijk}	-0,29597	-	-	-

Speciation of Neptunium after Diffusion in Opalinus Clay

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Argillaceous rock formations are under consideration as a potential host rock for the construction of high-level nuclear waste repositories. Under environmental conditions the most stable oxidation states of ^{237}Np ($t_{1/2}$=2.1×10^6 a) are Np(IV) and Np(V). We have investigated the sorption and diffusion of the more mobile Np(V) in Opalinus Clay (OPA, Mont Terri, Switzerland) (Wu *et al*. 2009, Fröhlich *et al*. 2011 and 2012 a). OPA, which is present in Switzerland and southern Germany, possesses a micro-scale heterogeneity and is composed of several types of clay minerals, but also of calcite, quartz and iron(II)-bearing minerals. In our previous diffusion (Wu *et al*. 2009) and anaerobic sorption experiments (Fröhlich *et al*. 2011), we observed higher distribution coefficients, K_d, than expected from batch experiments performed in air, indicating that a partial reduction of Np(V) to Np(IV) had occurred. To test this hypothesis, different sorption and diffusion samples with Np(V) were prepared at pH 7.6 for spatially resolved molecular-level investigations at the microXAS beamline at the Swiss Light Source (PSI, Villigen, Switzerland) (Fröhlich *et al*. 2012 b).

Elemental distributions of Ca, Fe and Np have been determined by μ-XRF mapping. Regions of high Np concentration were subsequently investigated by Np L_{III}-edge μ-XANES. In most samples Np spots with considerable amounts of tetravalent Np could be found, even when the experiments were performed under ambient-air conditions. In some cases, almost pure Np(IV) L_{III}-edge XANES spectra were recorded. In case of the anaerobic sorption sample, a clear correlation between Np and Fe was observed by μ-XRF, indicating that iron(II)-bearing minerals could be responsible for the reduction of Np(V). μ-XRD measurements of this sample showed that pyrite is at least one of the redox-active phases determining the speciation of Np in OPA. In this case, Np was accumulated on pyrite, indicating that the reduction of Np occured near the surface.

In our recent long-term diffusion experiment using 8 μM Np(V) in OPA pore water (pH 7.6), we were able to map the speciation of Np along a 2 mm long diffusion path parallel to the bedding of the clay. These μ-XANES measurements showed that mobile Np(V) is immobilized in Opalinus Clay as Np(IV) by progessive reduction along its diffusion path, further consolidating the suitability of argillaceous rocks with regard to the long-term storage of Np.

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Evolution of the Bentonite Barrier under Glacial Meltwater Intrusion Conditions

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Recent safety assessments for repository concepts that combine a clay engineered barrier system (EBS) with a fractured rock have shown that melt water intrusion may have a direct impact on the EBS barrier function in two aspects:

- Generation of colloids may degrade the engineered barrier
- Colloid transport of radionuclides may reduce the efficiency of the natural barrier

The studies presented here are performed in the framework of the Federal Ministery of Economic Affairs and Energy (BMWi) KIT/GRS project KOLLORADO-*e*, the EU collobarative project CP BELBaR (www.skb.se/belbar) and especially within the Colloid Formation and Migration (CFM) project at the Grimsel Test Site, GTS (www.grimsel.com).

Key research areas are (a) the erosion of the bentonite buffer, (b) clay colloid stability and (c) colloid-radionuclide- host rock surface interactions. Concerning bentonite buffer integrity parameters like the bentonite type, Na-/Ca-exchangeable cation ratio, compaction density, role of accessory minerals, the fracture aperture size and groundwater chemistry and flow velocity are investigated in order to identify controlling factors, understand the main mechanisms of erosion from the bentonite surface and to quantify the extent of the possible erosion under these different conditions. Clay colloid stability studies are performed under different geochemical conditions. The main objective is to answer the question if colloids formed at the near/far field interface would be stable only if favourable conditions exist and therefore their relevance for radionuclide transport will be strongly dependent on the local geochemical conditions (inorganic cations Na^{+}, Ca^{2+}, Mg^{2+}, Al^{3+} and organic complexing agents). Finally, the interaction between colloids and radionuclides and the host rock is intensively investigated in order to answer the question, how colloid mobility may be affected by the composition of the host rock, surface roughness and the mechanism of sorption and de-sorption of radionuclides on/from colloids. In all these areas mentioned above substantial laboratory studies are and will be undertaken. Beside this laboratory studies the unique set-up installed in the controlled zone of the GTS gives the opportunity within CFM to study the bentonite erosion and colloid formation under near-natural repository post clossure hydraulic conditions on the approx. deca-meter scale.

Modeling studies will support the laboratory program through development of conceptual and mathematical descriptions of the observed phenomena. For example, simulations of a virtual clay plug using a constant or flow rate dependent erosion rate are compared to results with 2D/cubic law simplifications for uncertainty analysis.

The final outcome is to verify how colloids and related phenomena can be considered in the long-term safety case and to make recommendations on how safety case studies could pursue to address this potentially very significant issue. An increased understanding of the processes investigated here will have an effect on the outcome of future assessments.

Monitoring in the Post-Closure Phase: Development of Wireless Techniques for Data Transmission from the Repository to the Surface

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When the *in-situ* monitoring in a geological disposal facility is continued during the post-closure phase, monitoring data need to be transmitted wirelessly from the repository to the surface. Wireless data transmission is used today in many applications, but the large attenuation by the geologic medium between de disposal facility and the surface makes the application of high-frequency based techniques impractical.

As part of the EURATOM FP-7 project *MoDeRn* (Monitoring Developments for safe Repository operation and staged closure), NRG has investigated the feasibility of wireless data transmission through an argillaceous host rock (Boom Clay), making use of low frequency magnetic fields. The main focus of the contribution was to analyze and optimize the energy efficiency of this technique. Therefore, a mathematical model description has been developed that allows to estimate the expected signal strength on the earth's surface on basis of the most relevant characteristics of transmitter, receiver and transmission path. The model is used to analyze the complex interactions of different system parameters, and is applied to design an optimized set-up for through-the-earth data transmission and to estimate minimum energy demands for signal transmission.

To demonstrate the potentials of this technique, experiments were performed in the 225 m deep underground research facility HADES in Mol, Belgium. Signal propagation and attenuation by the geologic medium between the HADES and the surface has been measured, and the site-specific magnetic background noise at the surface in Mol has been characterized. Based on the results, optimum conditions for signal transmission have been derived and data transmission experiments have been performed. Results show that despite large local interferences on the surface in Mol, wireless data transmission through 225 m of a geological medium is possible. Data transmission rates up to 100 bit/s has been successfully tested.

The experiments performed clearly demonstrated the feasibility to transmit data through 225 m of highly conductive geological medium, even under unfavorable experimental conditions as present in Mol. The amount of energy necessary to transmit data to the surface is within the expectations, and extrapolation to generic Dutch disposal concepts in Boom Clay (500 m) and rock salt (800 m) shows that transmission of monitoring data to the surface should be possible with less than 1 mWs of energy per bit of transmitted data.

Acknowledgements

We greatly acknowledge the kind support of Jan Verstricht and other members of the EURIDICE staff during our experimental work at the HADES URL in Mol, Belgium. NRG has received funding from the European Atomic Energy Community's Seventh Framework Programme (FP7/2007-2011) under grant agreement no 232598 and from the Ministry of Economic Affairs, Agriculture, and Innovation, The Netherlands.

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Study of the influence of hydrogeological conditions in the upper aquifer on radionuclide migration from a geological repository using a 2D groundwater flow model

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Abstract: *Results are presented of a* case *groundwater flow-transport modeling to predict the radionuclide migration from a deep geological repository (DGR) of radioactive waste. The influence of hydrogeological conditions in the upper aquifers of a storey water exchange system on the rate of contaminant migration from the DGR to its natural far-field groundwater discharges (a shallow well and a river) as a general DGR safety condition is considered.*

1 Introduction

In course of implementation of an IAEA-coordinated Research Project and of National Academy of Science of Ukraine, a 2D vertical-section groundwater flow and transport modelling was performed for a prospective DGR site Veresnya in the Chernobyl Exclusion Zone in Ukraine (IAEA, 2013).

The numerical model was developed using Processing Modflow groundwater modeling system (Chiang, Kinzelbach, 2001) based on MODFLOW code for the groundwater flow (McDonald, Harbaugh, 1988), and MT3DMs code for the contaminant transport (Zheng & Wang, 1998). The advection transport paths and transport times were evaluated using the particle tracking method with PmPath code (Pollock, 1989), by successive execution of MODFLOW and PMPath codes for each simulation series.

The modelling was aimed at studying the DGR safety condition as related with the assessed rate of possible radionuclide migration from the DGR to the natural groundwater discharge boundaries of the site. The hydrogeological scheme of the site and the model grid section are shown on Fig. 1.

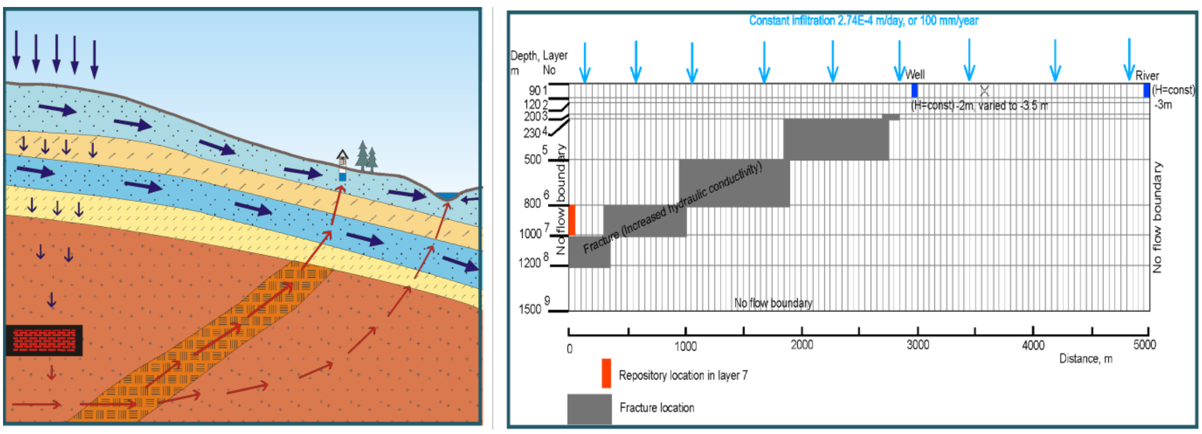

Fig. 1. Site hydrogeological scheme and model section

The model section domain, 1.5 km deep and 5 km long, is gridded into 9 layers, and each layer into 100 50-m blocks. The constant head boundaries are applied at the upper right corner of the section representing a river, and a shallow well is at 3 km distance from the DGR borehole. The groundwater infiltration recharge 100 mm·year^{-1} is applied at the top model boundary. The river and well represent the natural groundwater discharge boundaries of the site. During the transport simulations, the tracer particles are released or a constant-concentration boundary condition is applied in the DGR block at the left boundary of layer 7 (depth 800-1000 m).

An inclined fracture zone at slope angle near 45° passing from the DGR location upward to the well is considered for studying its influence on the rate of possible radionuclide migration from the DGR. The geological composition of the section and initial sets of hydraulic parameters of rocks and deposits were set according with results of previous study of the site (STCU, 2006), Table 1.

Table 1. Initial hydraulic parameters for model layers and corresponding geological strata

Layer No	Depth, [m]	Hydraulic conductivity K, [m·s^{-1}]	Porosity	Description
1	0-90	5.8E-05	0.15	Quaternary+ Paleogene aquifer
2	90-120	2.3E-07	0.01	Confining layer (Paleogene marls)
3	120-200	1.2E-05	0.1	Cretaceous+Jurassic aquifer
4	200-230	5.8E-08	0.01	Confining layer (Jurassic clays+ weathering bedrock)
5	230-500	5.8E-07	0.005	Upper fractured zone of bedrock
6	500-800	1.2E-08	0.002	Monolithic crystalline bedrock
7	800-1000	5.8E-09	0.002	Monolithic crystalline bedrock. Geological repository location
8	1000-1200	5.8E-09	0.002	Monolithic crystalline bedrock
9	1200-1500	5.8E-09	0.002	Monolithic crystalline bedrock, conventional lower boundary layer

2 Results and discussion

2.1 Effects caused by different hydraulic heads in the well

Increasing the drawdown in the well from -2 to -3.2 m below the initial zero level at constant drawdown in the river -3.0 m results in decreasing flow velocity in the deep part of the model. When the drawdown is around -3.2 m, the flow changes its endpoint from the river to the well. The travel time reaches its maximum 4.5 million years, and the travel path is the longest. At further increasing well drawdown to -3.5 m, the travel path and travel time decrease. The relative concentration (initial value 100 in the DGR) of the contaminant in both discharge points increase with increasing drawdown in the well. Velocities of the tracer particles are higher at horizontal parts of the trajectories, Fig. 2.

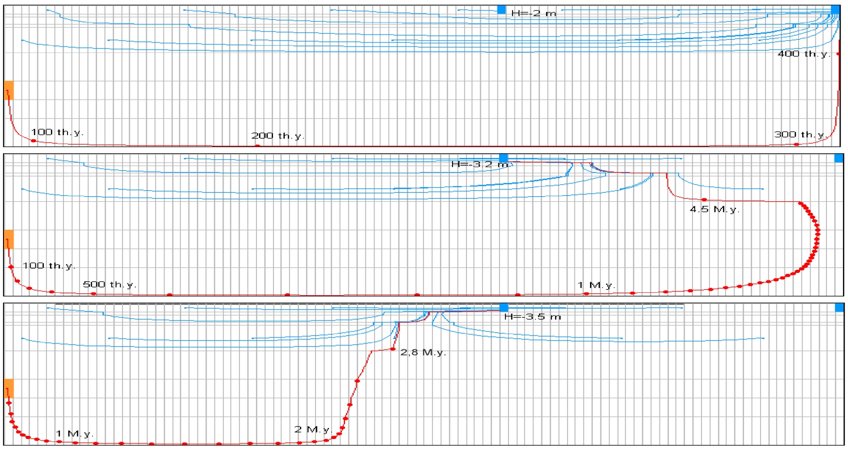

Fig. 2. Different types of flow and contaminant pathways (red lines) for simulations with different drawdown in the well (-2.0, -3.2, -3.5 m). Red balls mark the time interval 100 thousand years.

1.2 Effects caused by different hydraulic conductivity K_1 of the first from the surface aquifer

Obtained contaminant travel time (Fig. 3) and contaminant relative concentration at time 1 My in the well (Fig. 4), and in the river (Fig. 5) are shown for different hydraulic conductivity values K1 of the first from the surface aquifer.

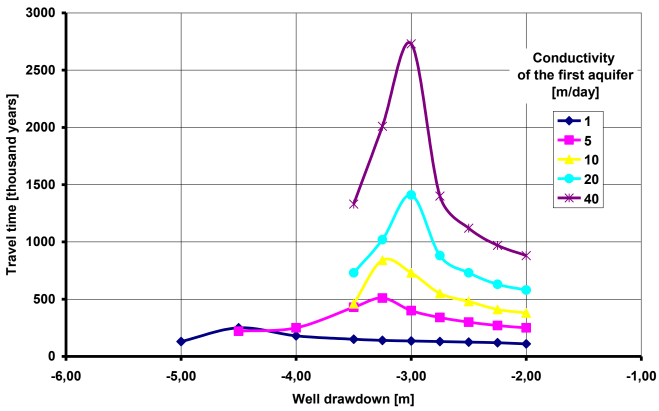

Fig. 3. Dependence plots of travel time to the discharge point against the well drawdown for different K_1 values.

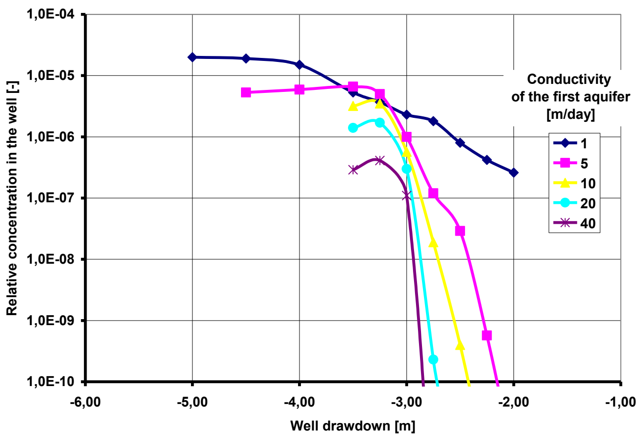

Fig. 4. Dependence plots of the relative concentration in the well at time 1 My against the well drawdown for different K_1 values.

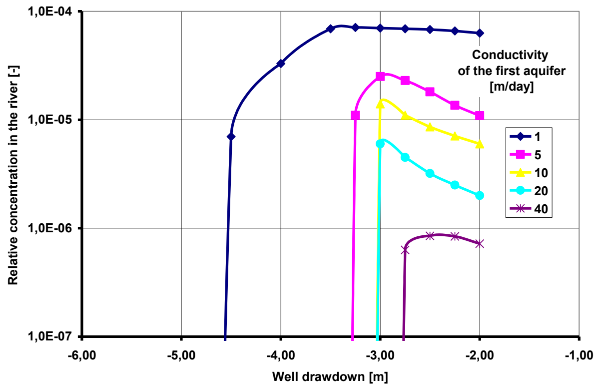

Fig. 5. Dependence plots of the relative concentration in the river at time 1 My against the well drawdown for different K_1 values.

2.3 Effects caused by the fracture zone

The results of simulation of the influence of the fracture zone with increased hydraulic conductivity on the relative contaminant concentration in the river (for simulation period 1 My) and travel time to the river are shown in Table 2. The predicted raising of the plume boundary with relative contaminant concentration 10-6 is shown on Fig. 6.

Table 2. Influence of the fracture with different times-increased hydraulic conductivity on the relative contaminant concentration in the river (for simulation period 1 My) and travel time to the river

Conductivity K in the fracture zone	Travel time to the river, [My]	Predicted relative concentration in the river for time period 1 My
Background K (no fracture)	0.42	$1.7 \cdot 10^{-6}$
Background K x 2	0.40	$1.9 \cdot 10^{-6}$
Background K x 5	0.38	$2.4 \cdot 10^{-6}$
Background K x 10	0.36	$2.8 \cdot 10^{-6}$
Background K x 100	0.32	$5.3 \cdot 10^{-6}$

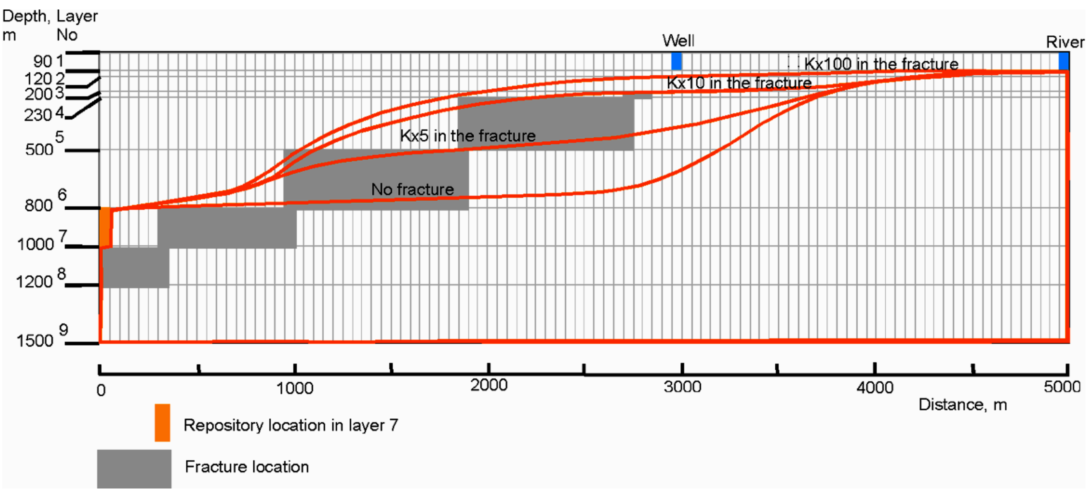

Fig. 6. The plume boundary (relative concentration 10^{-6} Isoline) raising with increasing hydraulic conductivity K (x 5, x10, x100) in the fracture blocks

3 Summary

The principal DGR indirect safety characteristics such as predicted contaminant concentrations and contaminant travel time depend on location and drainage ability of discharge boundaries (rivers, wells), and hydraulic conductivity of water-bearing deposits of the upper groundwater. Furthermore, changes in hydraulic conductivity of the near-surface aquifers may significantly affect not only the shallow but also deep groundwater flow pattern in the basement rocks, leading to noticeable changes in predicted transport behavior. Presence of the fracture with increased hydraulic conductivity in the DGR locality increases the predicted radionuclide concentrations in the groundwater discharge location.

Consequently, the presence of distributed shallow discharge wells or small rivers on the background of general watershed conditions with well-developed covering deposits and downward infiltration is not an unfavorable factor for the repository safety. In such conditions the infiltration flow is mainly intercepted in the upper aquifer, and the deeper geological medium remains the intact zone of slow groundwater flow and probable contaminant transport.

Acknowledgements

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A Contribution from Fundamental and Applied Technetium Chemistry to the Nuclear Waste Disposal Safety Case

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Nuclear waste disposal in deep geological formations such as crystalline (granite), sedimentary (claystone) or rock salt, is the favored option of the international nuclear waste disposal community. For the long-term safety assessment of nuclear waste repositories, a reliable prediction of radionuclide migration behavior is required. A potentially relevant mobilization and migration mechanism is caused by water intrusion into the repository, leading to radionuclide release via transport pathways. In this case, detailed knowledge of key parameters controlling the retention and mobilization of radionuclides in solution, i.e. redox processes, solubility limits and sorption properties, is essential. Dedicated research is required in order to derive process understanding and develop accurate site-independent chemical and thermodynamic models, applicable for all considered host rock formations and scenarios.

Technetium-99 is a β-emitting fission product highly relevant for the safety assessment of nuclear waste repositories due to its significant content in radioactive waste (fission yield >6%), long half-life ($t_{½} \approx 2.1 \cdot 10^5$ a) and redox sensitivity. The mobility of Tc in the environment strongly depends on its oxidation state. Tc(VII) exists as highly soluble and mobile TcO_4^- pertechnetate anion under sub-oxic and oxidizing conditions, whereas Tc(IV) forms sparingly soluble hydrous oxide ($TcO_2 \cdot xH_2O$) solid phases under reducing conditions.

In the first part of this study focusing on fundamental Tc chemistry, the redox behavior of Tc(VII)/Tc(IV) was investigated in dilute to concentrated solutions. The results are systematized according to *Pourbaix* diagrams calculated with the NEA–TDB data selection for Tc to assess the effect of homogeneous and heterogeneous reducing systems and ionic strength on Tc redox behaviour. Investigations focusing on the solubility and speciation of $TcO_2 \cdot xH_2O(s)$ were performed in dilute to concentrated solutions over the entire pH range. Complete and improved chemical, thermodynamic and activity (SIT, Pitzer) models were derived.

The second part of this work focuses on applied Tc chemistry in near-natural systems. Interaction of Tc(VII) with crystalline rock material from a prospective repository site (Nizhnekansky massif, Russia) and from an underground research laboratory (Äspö HRL, Sweden) were studied. Drilling of the Äspö cores were performed under anoxic conditions. Part of the material was artificially oxidized to test the importance of sample preservation. Batch sorption studies under variation of the Tc(VII) concentration were performed in synthetic groundwater simulate to estimate the retention kinetics and R_S values. For better understanding of the retention mechanisms, advanced surface sensitive analytics (XPS, XANES) were carried out and the data compared to reference systems. Tc migration behavior was further investigated by injections of Tc(VII)-containing groundwater simulant into the natural fracture in the unoxidized Äspö diorite core. The data obtained are compared with the results of batch sorption studies and will be used for subsequent geochemical reactive transport modeling.

This work highlights the importance of combining fundamental Tc chemistry and applied studies in order to derive a comprehensive assessment of Tc mobilization and retention processes in support of the nuclear waste disposal Safety Case.

The Correlation between Selenium Adsorption and the Mineral and Chemical Composition of Taiwan Local Granite Samples

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Selenium-79 (Se-79) is a radioactive isotope of selenium, which is considered as one of the highly mobile nuclides since Se-79 would be presented in an anion species when dissolving into the intruded groundwater. Being an anionic species, the transport of Se-79 would be regulated by the metal oxides relevant minerals such as goethite and hematite (Jan et al., 2008). This is true that the transport of selenium in the shallow surface environment could be relatively easy to estimate by considering the amount of these metal oxides presenting in the soils and sediments. However, when dealing with deep geological repository, the transport of Se-79 becomes less predictable because of the much less content of metal oxide residing in the host rock such as granite. In order to conduct a reliable performance assessment of repository, it is very important to establish the correlation between selenium adsorption and the properties of potential host rock, in this study, the mineral and chemical compositions of Taiwan local granite. From this point of view, selenium adsorption experiments were conducted with 54 different Taiwan local granite samples collected from the depth ranging from 100 ~ 400 meters below the surface. These granite samples represent a variety of deep geological environments, including the intact rock, groundwater intruded zones, and some weathered samples. Based upon our preliminary results, several solid conclusions could be made. First, the correlation coefficients between the Kd values and the mineral and chemical compositions are rather low (R-square values are often < 0.2). This points out the complexity of these geological samples and strongly suggests more efforts should be put into to acquire more relevant information. Second, the correlation between the selenium Kd values and the content of iron oxide (R-square 0.110) is much higher than that between the CEC of these granite samples (R-square 0.001). This clearly indicates that the minerals that are able to underlie cation exchange reaction would not be the minerals regulating the transport of selenium nuclides. Further, a relative high correlation coefficient (R-square 0.137) between the variation of pH before and after selenium adsorption and iron oxide content implies that surface complexation reaction is much likely accounting for selenium adsorption. Future works would be particularly focusing on the outlet data interpretation (i.e., high iron oxide content but low Kd value and vice versa) as well as the form/phase regarding the origin of these iron oxide contents.

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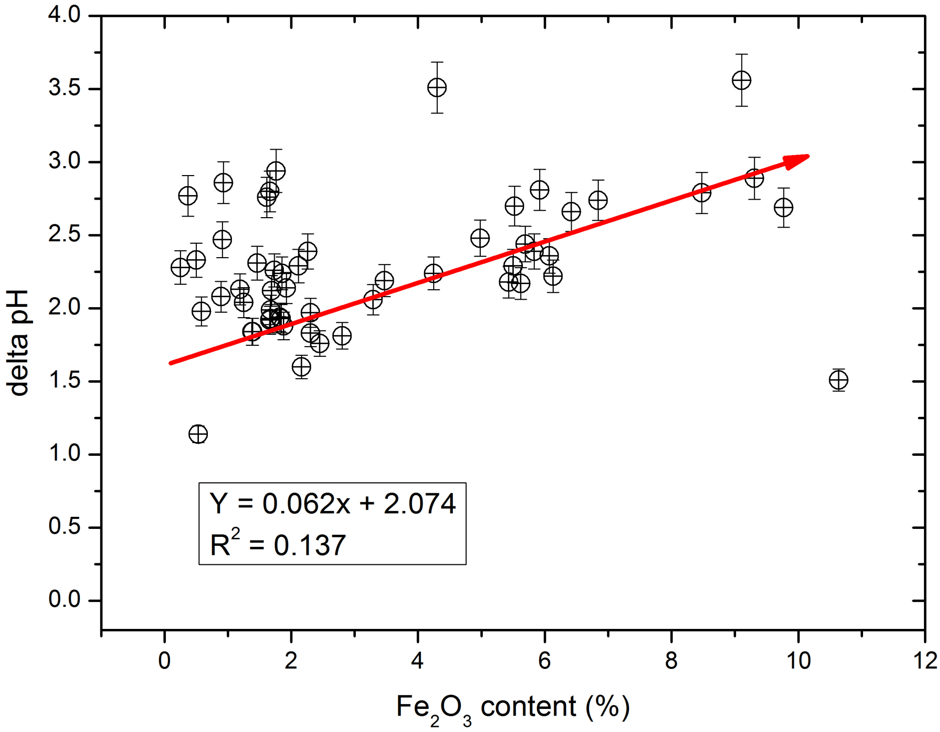

Fig. 1. Correlation between Fe_2O_3 content and the change in pH after Se adsorption experiments

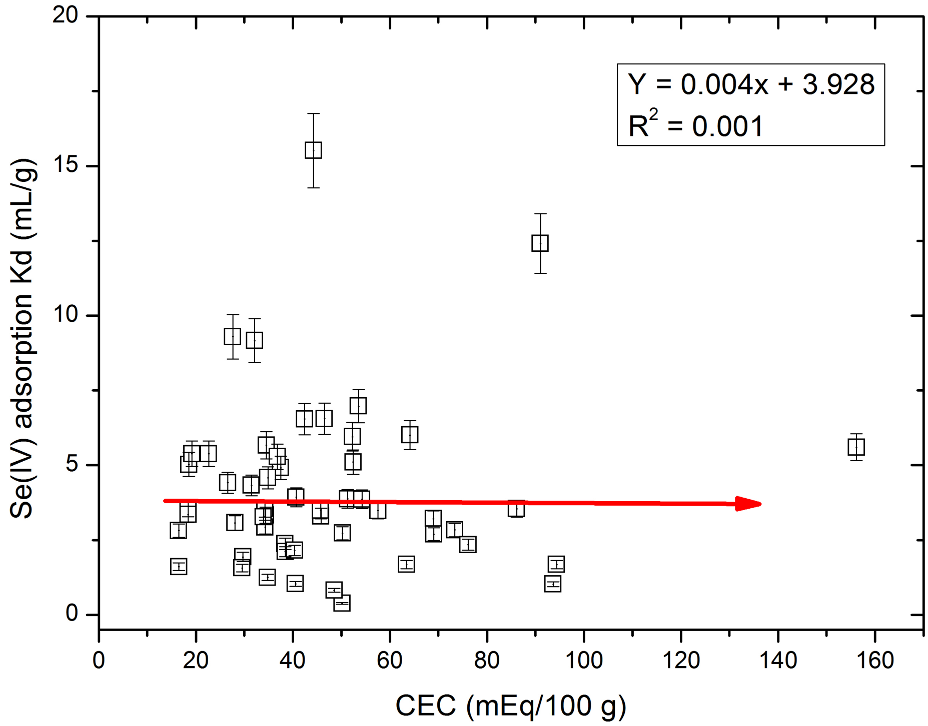

Fig. 2. Correlation between CEC and the Kd values of Se(IV) adsorption

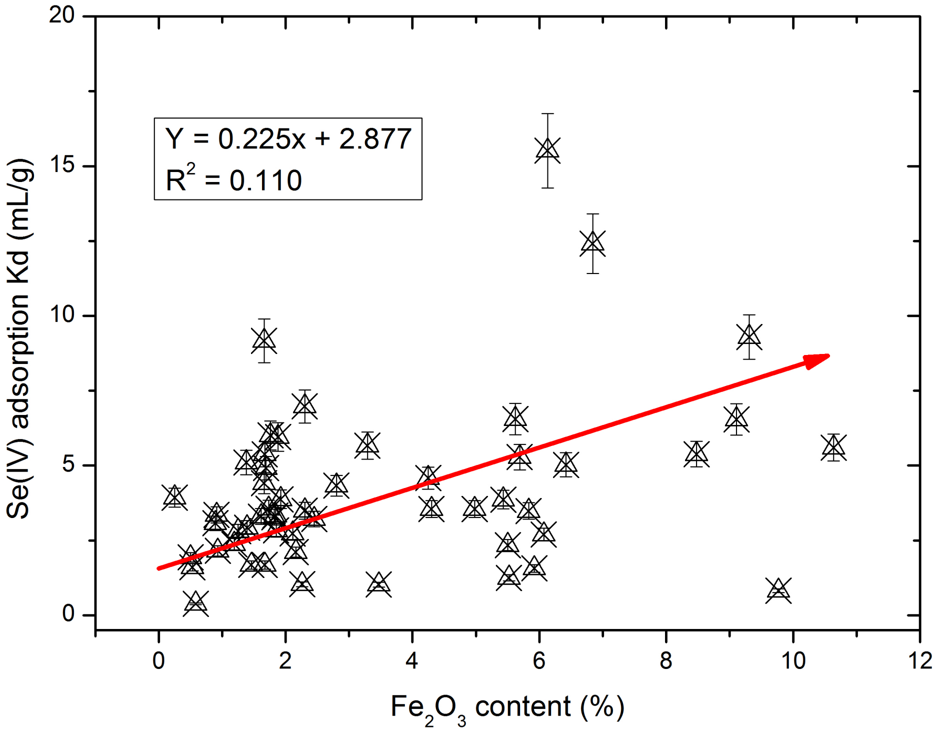

Fig. 3. Correlation between Fe_2O_3 content and the Kd values of Se(IV) adsorption

Self-Sealing of Excavation Induced Fractures in Clay Host Rock

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Excavation of an underground repository for disposal of radioactive waste in clay formations generates fractures around the openings, which may act as pathways for water transport and radionuclides migration. Because of the favorable properties of the clay rocks such as the rheological deformability and swelling capability, a recovery process of the excavation damaged zone (EDZ) can be expected due to the combined impact of rock compression, backfill resistance, and clay swelling during the post-closure phase. Another important issue is the impact of gases produced from anoxic corrosion of waste containers and other metallic components within the repository. The EDZ may act as a conduit for preferential gas flow, depending on the extent of the recovery process. For the safety assessment of a repository, the self-sealing behaviour and impact on water and gas transport through the EDZ have to be characterized, understood, and predicted.

Recently, GRS has extensively investigated these important issues with various kinds of laboratory and in-situ experiments under relevant repository conditions. Test samples were taken from the Callovo-Oxfordian argillite at Bure in France and the Opalinus clay (shaly facies) at Mont Terri in Switzerland. Major findings are summarized as follows.

As observed in laboratory and in-situ, the gas permeabilities of the claystones increase with stress-induced damage by several orders of magnitude from the impermeable state up to high levels of 10^{-12}–10^{-13} m^2. When hydrostatic confining stress is applied and increased, the fractures in the claystones tend to close up, leading to a decrease in gas permeability down to different levels of 10^{-16}–10^{-21} m^2 at stresses in a range of 10 to 20 MPa.

As water enters and flows through fractures, the clay matrix can take up a great amount of the water and expand into the interstices. Consequently, the hydraulic conductivity decreases dramatically by several orders of magnitude down to very low levels of 10^{-19} to 10^{-21} m^2 even at low confining stresses of 2-4 MPa within short periods of months to years, compared with the long post-closure phase of tens of thousands of years. The very low water permeabilities reached are the same order as that of the intact claystones, indicating that the fractures in the claystones can be completely resealed.

The very low water permeabilities do not change significantly during heating and cooling within a temperature range between 20°C and 90°C. This suggests no negative thermal impact on the sealing of the fractures.

Before water saturation, the fractures act as conduits for preferential gas flow. However, after water saturation, gas entry and penetration into the resealed fractures in claystone needs certain driving forces to overcome the capillary thresholds. The gas entry/breakthrough pressure is determined by the sealing intensity of the fractures and the confining stress. The gas tests carried out on the highly-resealed claystones show that the gas breakthrough pressures increase with confining stress from low levels of 0.2-1.0 MPa at a low stress of 2 MPa up to 8.3-11.3 MPa at high stresses of 12.7-15.0 MPa corresponding to the rock stress state at a depth of ~500 m. It is obvious that the breakthrough pressures are still below the applied confining stresses. This important finding implies that the EDZ, even when highly-resealed, will still have the capacity for gas migration with moderate pressures and thus contribute to avoid high pressure build up for fracturing the host rock.

Generally speaking, the significant sealing capacities of the claystones hinder water transport and thus radionuclides migration through the fractures, but allow gas migration at moderate pressures without fracturing the host rock. These significant advantages guarantee the long-term integrity and barrier function of the host rock.

Critical Evaluation of German Regulatory Specifications for Calculating Radiological Exposure

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The assessment of radiological exposure of the public is an issue at the interface between scientific findings, juridical standard setting and political decision. The present work revisits the German regulatory specifications for calculating radiological exposure, like the already existing calculation model General Administrative Provision (AVV) for planning and monitoring nuclear facilities. We address the calculation models for the recent risk assessment regarding the final disposal of radioactive waste in Germany. To do so, a two-pronged approach is pursued.

One part deals with radiological examinations of the groundwater-soil-transfer path of radionuclides into the biosphere. Processes at the so-called geosphere-biosphere-interface are examined, especially migration of I-129 in the unsaturated zone. This is necessary, since the German General Administrative Provision does not consider radionuclide transport via groundwater from an underground disposal facility yet. Especially data with regard to processes in the vadose zone are scarce. Therefore, using I-125 as a tracer, immobilization and mobilization of iodine is investigated in two reference soils from the German Federal Environment Agency.

The second part of this study examines how scientific findings but also measures and activities of stakeholders and concerned parties influence juridical standard setting, which is necessary for risk management. Risk assessment, which is a scientific task, includes identification and investigation of relevant sources of radiation, possible pathways to humans, and maximum extent and duration of exposure based on dose-response functions. Risk characterization identifies probability and severity of health effects. These findings have to be communicated to authorities, who have to deal with the risk management. Risk management includes, for instance, taking into account acceptability of the risk, actions to reduce, mitigate, substitute or monitor the hazard, the setting of standards and criteria, and adaption to environmental policy and regulatory framework. To gain a better understanding of the relevant processes and criteria in this transition process, professional experts who develop exposure calculation models or were involved in developing existing regulations are interviewed. 12-15 guideline based interviews will be analyzed by qualitative content analysis. Results will be discussed among the authors and further members of the ENTRIA-transversal-project *Technology Assessment and Governance*.

An Introduction to the Transport Properties Research Laboratory at The British Geological Survey and its 50+ Years' Experience in geological disposal research

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The Transport Properties Research Laboratory (TPRL) is one of the leading centers in Europe for the study of fluid movement in ultra-low permeability media. The facility is well known for long-term high quality experimental work and process-based interpretation. Focus is on multi-phase flow in natural and engineered, low permeability geomaterials (e.g. caprocks, well bore cements and engineered clays), and their associated deformation behavior. Measurements include: saturation and consolidation properties; intrinsic permeability (or transmissivity); anisotropy; specific storage; coupled flow parameters (e.g. osmotic permeability); capillary entry, breakthrough and threshold pressures; gas permeability function; drained and undrained compressibilities; and rheological (creep) properties. Laboratory experiments are performed under simulated in-situ conditions (stress, pore pressure, temperature and chemical environment).

Three key areas explored are:

(i) baseline characterization of hydromechanical properties,
(ii) influence of stress path and stress history on transport properties and
(iii) transmissivity of fractures, faults and discontinuities (e.g., wellbore interfaces).

Tests are designed to provide quantitative data for mathematical modeling of ultra-low permeability materials, together with process understanding of key transport mechanisms. Key equipment includes: high pressure isotropic permeameters (70 MPa); constant volume permeameters (70MPa); high pressure triaxial permeameter (70 MPa); heavy-duty, high-precision shear-rigs; high temperature, high pressure geochemical flow reactor (130 MPa at 140°C); and novel tracer systems (nanoparticle injection or radiological tagging of gas) to characterize and identify potential migration pathways.

The key achievements from the TPRL at the BGS include generation of new conceptual models applied throughout Europe, transfer of skills and knowledge to other complex geoscience problems (e.g. shale gas, CCS). The TPRL is currently involved in about 20 projects including work for Andra, COVRA, Nagra, RWM Ltd, SKB, DECC and EPSRC.

The above described research has also a long history in collaborations with national radioactive waste management organizations, government and regulators. However, the work at the TPRL is constantly improving and moving into new and state-of-the-art research areas and knowledge transfer and new collaborative work are essential.

Characterization of Hydraulic Connections between Mine Shaft and Caprock Based on Time Series Analysis of Water Level Changes for the Flooded Asse I Salt Mine in Northern Germany

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In the context of safe enclosure of nuclear waste in salt formations, one of the main challenges is potential water inflow into the excavations. In this context, the hydraulic relationship between the abandoned Asse I salt mine and the salt dissolution network at the base of the caprock of the Asse salt structure in northern Germany is characterized by utilizing time series analysis of water level changes. The data base comprises a time series of water level measurements over eight years with a temporal resolution of 15 minutes (in general) and up to 2 minutes for specific intervals. The water level measurements were collected in the shaft of the flooded mine, which is filled with ground rock salt until a depth of 140 m, and a deep well, which is screened in 240 m depth at the salt dissolution zone at the base of the caprock. The distance between the well and the shaft is several hundred meters. Since the beginning of the continuous observations in the 1970s, the shaft has shown periodically abrupt declines of the water level of several meters occurring in intervals of approx. 8 to 10 years.

The time series analysis consists of trend, Fourier-, autocorrelation and cross-correlation analysis. The analysis showed that during times with small water level changes the measured water level in the well and the shaft are positively correlated whereas during the abrupt water level drops in the shaft, the measured water levels between the shaft and the well are negatively correlated. A potential explanation for this behavior is that during times with small changes, the measured water levels in the well and in the shaft are influenced by the same external events with similar response times. In contrast, during the abrupt water level decline events in the shaft, a negatively correlated pressure signal is induced in the well, which supports the assumption of a direct hydraulic connection between the shaft and the well via flooded excavations and the salt dissolution network along the base of the caprock.

Preparation and Estimation of Thermodynamic Properties of Fe(II)-, Co(II)-, Ni(II)- and Zr(IV)-Containing Layered Double Hydroxides

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The investigation of layered double hydroxides (LDHs) was performed because they and similar "green rust" phases were identified as specific secondary phases forming during the evolution of the disposed research reactor fuel elements under nuclear repository relevant conditions [1]. LDHs or "hydrotalcite-like" solids are of interest in environmental geochemistry because they can substitute various cations and especially due to the anion-exchange properties and, therefore, can be considered as buffer materials for retention of mobile and hazardous radionuclides (like, ^{14}C, ^{129}I, ^{36}Cl, ^{79}Se *etc.*). In our study Fe^{2+}-, Co^{2+}-, Ni^{2+}- and Zr^{4+}-containing LDHs have been synthesized by co-precipitation method and characterized (PXRD, FT-IR, SEM-EDX, TGA-DSC) in order to investigate the effect of substitution of 2- and 4-valent cations on the stability of LDHs. PXRD measurements demonstrated that: (1) pure Mg-Al-Fe(II) LDHs are existing in the range of the mole fraction of iron x_{Fe} = Fe/(Mg+Fe) between 0 and 0.13. Unit-cell parameters (a_o=b_o) as a function of x_{Fe} follow Vegard's law corroborating the existence of a solid solution when x_{Fe} = 0 - 0.13. Products of syntheses with $x_{Fe} \geq 0.13$ contain detectable amounts (≥1-2 wt%) of additional phases (like, magnetite, maghemite, lepidocrocite); (2) pure Ni^{2+}- and Co^{2+}-containing LDHs (mole fractions of Ni and Co were equal to 0.1) have been synthesized successfully; (3) Mg-Al-Zr(IV) precipitates with mole fraction of zirconium x_{Zr} = Zr/(Zr+Al) = 0.0 - 0.5 show PXRD patterns attributed to pure LDHs and the variation of lattice parameters a_o=b_o as a function of x_{Zr} is in agreement with Vegard's law demonstrating the presence of solid solution. In contrast, PXRD analyses of precipitates with $x_{Zr} \geq 0.5$ have shown the presence of additional X-ray reflexes typical for brucite. The stoichiometry of LDHs has been established by ICP-OES and SEM-EDX analyses and reveals that (Mg+Fe)/Al, (Mg+Co)/Al and (Mg+Ni)/Al ratios in Fe^{2+}-, Co^{2+}- and Ni^{2+}- containing solids are remarkably close to desired 3:1 for the whole range of solid compositions. However, in Mg-Al-Zr(IV) LDHs the increase of x_{Zr} in co-precipitating synthesis solutions leads to the formation of solids with significantly reduced Mg/(Al+Zr) ratios. This fact and results of thermodynamic calculations with GEM-Selektor software [2] indicate that the incorporation of Zr into the LDH structure increases significantly the aqueous solubility of LDHs due to the preferred localization of $Zr(OH)_5^-$-ligands in the interlayer space of brucite-like layers. On the other hand estimates of molar Gibbs free energies shown that the substitution of Fe^{2+}, Co^{2+} and Ni^{2+} into the LDHs does not affect so significantly on the stability of LDHs. Finally, reliable estimate of standard Gibbs free energy of formation for pure Mg-Al hydrotalcite composition (-3619.04±15.27 kJ/mol) has been provided. For the first time, by applying Calvet-type solution calorimetry values of standard enthalpy (-4013.89±13.27 kJ/mol) and absolute molar entropy (254.63±25.34 J/mol·K) of pure Mg-Al-Cl LDH have been determined.

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Microorganisms in Potential Host Rocks for Geological Disposal of Nuclear Waste and their Interactions with Radionuclides

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The long-term safety of nuclear waste in a deep geological repository is an important issue in our society. Microorganisms indigenous to potential host rocks are able to influence the oxidation state, speciation and therefore the mobility of radionuclides as well as gas generation or canister corrosion. Therefore, for the safety assessment of such a repository it is necessary to know which microorganisms are present in the potential host rocks (e.g. clay, salt) and if these microorganisms can influence the performance of a repository.

Microbial diversity in potential host rocks for geological disposal of nuclear waste was analyzed by culture-independent molecular biological methods (e.g. 16S rRNA gene retrieval) as well as enrichment and isolation of indigenous microbes.

Among other isolates, a *Paenibacillus* strain, as a representative of *Firmicutes*, was recovered in R2A media under anaerobic conditions from Opalinus clay from the Mont Terri in Switzerland. Accumulation experiments and potentiometric titrations showed a strong interaction of *Paenibacillus* sp. cells with U(VI) within a broad pH range (3-7) [1].

Additionally, the interactions of the halophilic archaeal strain *Halobacterium noricense* DSM 15987, a salt rock representative reference strain, with U(VI) at high ionic strength was investigated. After 48 h the cells were still alive at uranium concentrations up to 60 µM, which demonstrates that *Halobacterium noricense* can tolerate uranium concentrations up to this level. The formed uranium sorption species were examined with time-resolved laser-induced fluorescence spectroscopy (TRLFS).

The results about the microbial communities present in potential host rocks for nuclear waste repositories and their interactions with radionuclides contribute to the safety assessment of a prospective nuclear waste repository.

Acknowledgements

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Collaboration of the Dutch Research Program for Radioactive Waste Disposal (OPERA) and TU Delft

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Radioactive waste in the Netherlands is collected, treated and stored by COVRA (Centrale Organisatie Voor Radioactief Afval) in the interim storage facility for at least 100 a. After this period of long-term storage, geological disposal is foreseen. The policy is based on a step-wise decision process in which all decisions are taken to ensure safe disposal in a repository, but without excluding unforeseen alternative solutions that might develop in the future.

OPERA is the Dutch acronym for research program into geological disposal of radioactive waste. It started in 2011 and is running for five years. The OPERA Research plan is developed by NRG in close collaboration with COVRA. Radioactive waste disposal in the Netherlands is at an early, conceptual phase. The aim of Opera is to develop a first preliminary safety case to structure the research necessary for the eventual deployment of a repository in the Netherlands. The OPERA research program aims at a close cooperation with the Belgian research program on radioactive waste disposal.

The result of OPERA will be to detail a first roadmap for the long-term research on geological disposal of radioactive waste in the Netherlands, based initially on a re-evaluation of existing safety and feasibility studies conducted more than ten years ago, making use of present international and, wherever possible, national knowledge. This will be done by developing initial and conditional safety cases for generic GDFs in Zechstein rock salt and Boom Clay formations in the Netherlands. The goal in OPERA is to develop initial safety cases that are intended to mark the start of the research development process and to iterate these as knowledge grows to new developed insights. The safety case is conditional since plausible assumptions must later be confirmed in a safety case e.g. for site selection.

Dutch, Belgian, German, English and French organizations participate in OPERA. These organizations can be found in the two documents with awarded research proposals available at www.covra.nl/downloads/opera.

Building up competences and knowledge on radioactive waste management and geological disposal is an important goal of OPERA. To this end OPERA collaborates with TU Delft to develop an academic curriculum for the chair Chemistry of the Nuclear Fuel Cycle in the Master of Science Engineering.

Furthermore, the results obtained in OPERA will be presented at a Summer School which is planned to be organized at the end of the Program.

TU Delft educates young specialists, bachelor and master students, by research projects on geological disposal at the new actinide laboratory (U and Th). For such purpose the knowledge transfer from other OPERA partners is also foreseen, including students internships and visits of the research facilities.

Reactor Institute Delft (RID) is part of the Applied Sciences faculty of TU Delft and houses the Radiation Science & Technology (RST) department. RID operates the Hoger Onderwijs Reactor (HOR), a 2 MW pool-type research reactor in an academic setting. The reactor is used as a source of neutrons and positrons for research purposes, including those of OPERA. It also provides neutrons to a variety of facilities for radioisotope production and neutron activation analysis.

OPERA is financed by the Dutch Ministry of Economic Affairs and the public limited liability company Elektriciteits-Produktiemaatschappij Zuid-Nederland (EPZ) and coordinated by COVRA.

SESSION 6 CONSTRUCTION OF TECHNICAL BARRIERS

The DOPAS Full-Scale Demonstration of Plugs and Seals Project and Related GRS National RD&D Programs – A Retrospective View on 24-Month of Investigation

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The DOPAS Full-Scale Demonstration of Plugs and Seals project consisting of 14 beneficiaries from 8 European countries brings forward important demonstration activities in plugging and sealing. These activities are also a part of each participants national long-term RD&D programm and are therefore jointly funded by Euratom's Seventh Framework Programme and national funding organizations. The Demonstration experiments which will be partially or wholly implemented during the DOPAS project are a full-scale seal (FSS) implemented in Saint-Dizier in France, an experimental pressure sealing plug (EPSP) underground in the Josef Gallery in Czech Republic, a deposition tunnel dome plug (DOMPLU) in the Äspö Hard Rock Laboratory in Sweden, a deposition tunnel wedge plug (POPLU) in the underground rock characterization facility ONKALO (future spent fuel repository) in Finland, and components of a shaft sealing system (ELSA) in Germany (Dopas 2012).

ELSA is a program of laboratory and in-situ experiments that will be used to further develop the reference shaft seal for the German disposal concept for a repository in rock salt and to develop reference shaft seals for a repository in clay host rocks (Kudla et al. 2013).

On behalf of BMWi, the national funding organization for R&D work related to radioactive waste management, facing the ELSA project phase 3, GRS is investigating sealing and backfilling materials planned to be utilized in salt and clay formations. The program aims at providing experimental data needed for the theoretical analysis of the long-term sealing capacity of these sealing materials. According to current R&D work on the salt option, the shaft and drift seal components considered in Germany comprise seal components consisting of MgO and cement based salt concrete (Mueller-Hoeppe et al. 2012).

In order to demonstrate hydro-mechanical material stability under representative load scenarios, the sealing capacity of the seal system and the impact of the EDZ as well as hydro-chemical long-term stability in contact with different brines under diffusive and advective conditions, a comprehensive laboratory testing program is carried out.

One of the most challenging aspects is the determination of the pre-experimental status of the core material that was provided for laboratory investigations, since the salt concrete was taken from an existing dam that has been loaded in-situ by the creeping rock salt for more than 10 years. Therefore, it is obvious that material properties, such as e.g. the initial gas permeability, have to be measured under a load comparable to the in-situ minimum stress.

Within this paper preliminary results from laboratory investigations on salt concrete will be evaluated and an estimation of goals achievable within project lifetime will be given from a retrospective view on 24-month of investigation.

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Design and Proof of Function of a Closure System for an HLW-Repository in Rock Salt – Results of the Preliminary Safety Analysis for the Gorleben Site (VSG)

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Abstract: *Within the preliminary safety analysis for the Gorleben site (VSG), a closure system was designed in order to complement the containment providing rock zone (CRZ) by sealing and backfilling measures. The design procedure as well as the technical proof of function was mainly performed according to standard procedures in civil engineering. In the context of VSG, rough individual technical proofs of several measures were carried out. Meanwhile, this gap has been closed by subsequent investigations. Altogether the results of all the individual technical proofs of function indicate that safe containment of radioactive waste is a realistic possibility at the Gorleben site.*

1 Introduction

In summer 2010, the German Federal Government launched a preliminary safety analysis to assess whether the salt dome at Gorleben is suitable to host all heat-generating radioactive waste generated by German NPPs. Based on the “Safety Requirements Governing the Final Disposal of Heat-Generating Radioactive Waste as at 30 September 2010” (BMU 2010), a repository concept was developed that also included a closure system. The repository design and its closure system were tailored to the specific geologic environment at the Gorleben site and served as the main technical basis for the site-specific preliminary safety analysis (VSG).

According to the safety requirements, the radioactive waste is to be safely confined in a so-called containment providing rock zone (CRZ), from which only insignificant releases of radionuclides are tolerated. For the repository design described in (GRS 2011), a closure system was developed in order to complement the containment providing rock zone (CRZ) by technical measures, because the CRZ is impaired by the access routes into the repository. A site-specific closure system was developed, which includes backfilling and sealing measures in order to fulfil the requirement of safe containment after repository closure. The crucial components of the closure system are situated in the shafts, the infrastructure, in exploration area 1, and in the emplacement level below exploration area 1 of the Gorleben mine. As the shafts are already sunk and parts of the infrastructure as well as the drifts of exploration area 1 have already been excavated, detailed site-specific information is available to design the seals in such a way that a reliable proof of function is possible. Additionally, the design of the closure system includes only components that have been tested in-situ on a 1:1 scale. Thus, standard procedures are established for construction as well as for quality assurance measures. Using this practice, constructability has already been demonstrated and realistic data on material properties as well as methods to perform the proof of function are available. The main goal was to design a closure system whose feasibility is guaranteed and which fulfils the safety requirements. Optimization has not yet been taken into account.

2 Technical design procedure and description of the closure system

For designing the closure system, the technical design procedure in civil engineering, which consists of five steps: (1) conceptual design, (2) preliminary dimensioning, (3) basic design, (4) dimensioning, (5) detailed design, was applied. First, a conceptual design is developed and a preliminary dimensioning is carried out. As the preliminary dimensioning may lead to significant design modifications, it is roughly estimated and focuses on selected impacts and processes with high risk for significant design modifications. In this case, geochemical processes were included in the preliminary dimensioning due to a lack of knowledge on quantitative long-term effects of corrosion processes. Next, the results of the preliminary dimensioning are considered, necessary design modifications are carried out, and the dimensions of the individual components of the closure system are fixed, which leads to the basic design. The basic design serves as "input structure" for the dimensioning, which comprises the technical proof of function. Finally, the detailed design showing only minor modifications of the basic design and not affecting the dimensioning is defined and serves as basis for the construction process.

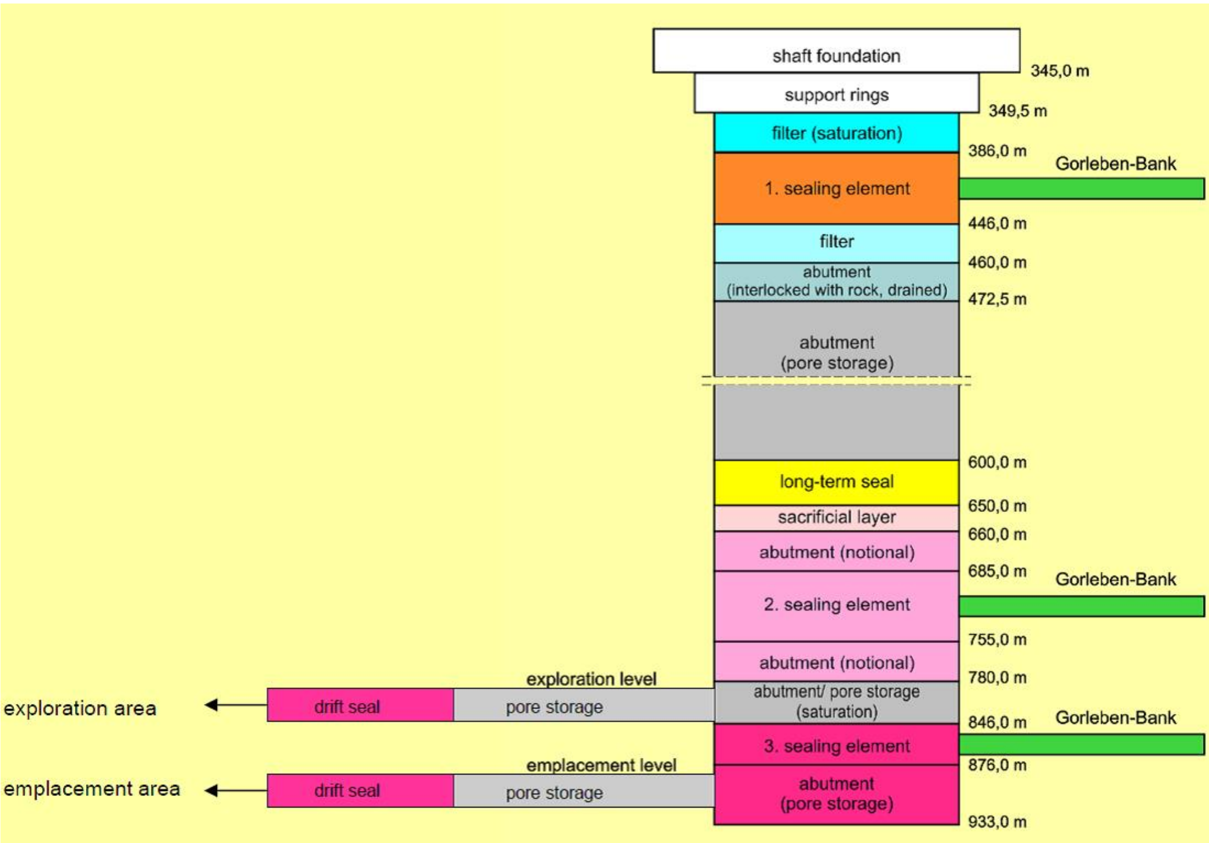

Fig. 1. Basic design of the closure system - VSG

The closure system comprises crushed salt and gravel for backfilling, a Bischofite buffer, and drift and shaft seals. Initially, the seals guarantee safe containment, later on compacted crushed salt backfill takes over the long-term sealing function. The gravel backfill acts as a pore storage and delays brine pressure build-up in front of the individual seals in order to guarantee that rock pressure onto the seals exceeds the brine pressure at the seals´ position at all times. The drift seals are made of magnesium oxycloride concrete. Each shaft contains three seals that are made of bentonite, salt concrete, and magnesium oxycloride concrete respectively. The sequence of the seals inside the shafts is tailored to the geochemical conditions of potentially intruding brine from the overburden and its increasing magnesium content when passing the individual seals. Finally, the Bischofite buffer provides the basis to prove long-term stability of the drift seals. The basic design of the VSG shaft sealing system is shown in Fig. 1

3 Linking safety case and technical proof of function

For the technical proof of function, the European Standard (DIN EN 1990) in civil engineering was applied on a trial basis. Additionally, the corrosion processes were taken into account. Following the technical design procedure, the technical proof of function was carried out iteratively in two main parts – the preliminary dimensioning (GRS 2012a) based on the conceptual design and the dimensioning (GRS 2012b), which relies on the basic design. The scheme of the technical proof of function used for the shaft sealing system and its coupling to the safety case are shown in Fig. 2.

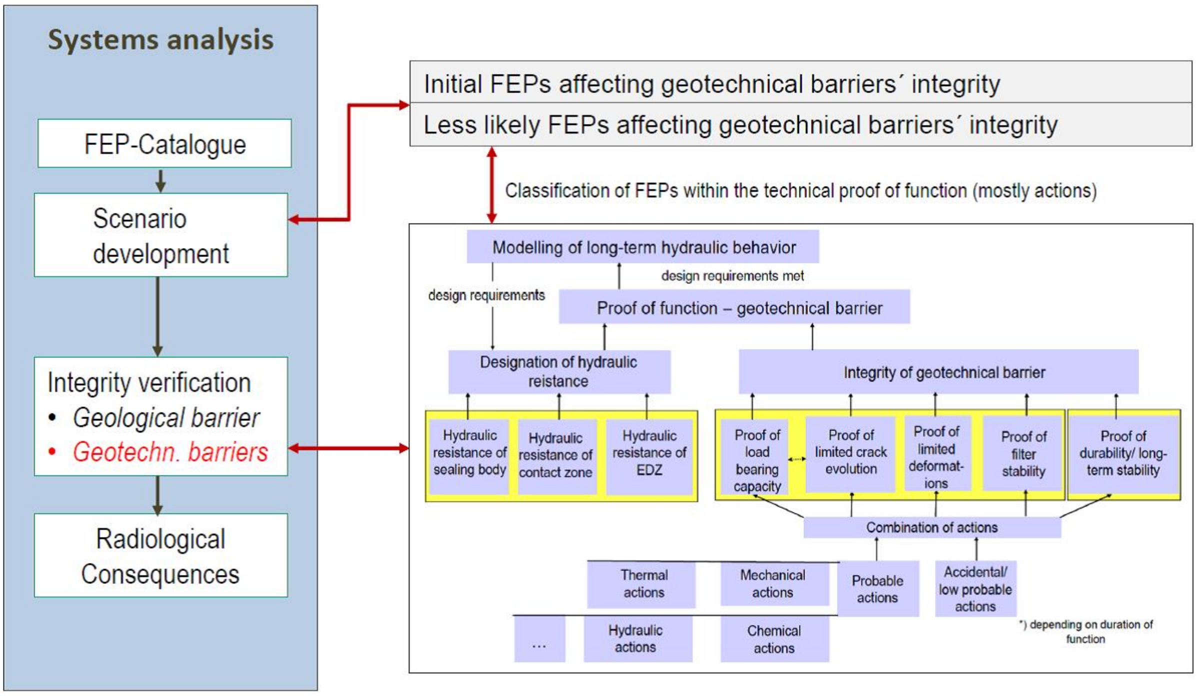

Fig. 2. Coupling of technical proof of function and systems analysis

The technical proof of function requires a restructuring of the FEPs affecting the sealing system (=initial FEPs) and assigning them to actions, resistances, and design situations (Table 1 and Table 2). Then the resulting procedure is suitable to undergo the standard dimensioning process in civil engineering. What is not standard is the unusual design working life of 50.000 years until the next ice age.

Within the technical proof of function, several FEPs are categorized as design situations. These are the seismic design situation (earthquake) (FEP 1.2.03.01) and abnormal (accidental) design situations (FEP 2.1.07.05 and 2.1.07.06), which comprise the early failure of a shaft seal and early failure of a drift seal. All remaining FEPs, which could not be excluded in prior assessments, are categorized as actions. Table 1 and Table 2 indicate which FEPs were considered quantitatively within the dimensioning procedure of VSG, which were covered by the quantitative results of other FEPs, which were covered by in-situ testing, and which were subsequently supplemented.

Table 1. Classification of the initial FEPs within the technical proof of function (DS = Design situation, A = Action, R = Resistance)

FEP-No.	FEP-Name	Classification within the proof of function	Processed in
1.2.03.01	Earthquake	DS, A, mass forces	supplemented
1.2.09.01	Diapirism	A, restraint strains	VSG (covered)
1.2.09.02	Subrosion	Excluded, not significant until next glacial period (after selected performance period)	-
1.3.05.03	Formation of glacial channels	Excluded, not significant until next glacial period (after selected performance period)	-
2.1.05.04	Alteration of drift and shaft seals	A, consequence of chemical action incl. temperature	VSG
2.1.07.01	Convergence	A, equivalent to rock pressure due to constitutive equation	VSG
2.1.07.02	Fluid pressure	A, fluid pressure	VSG
2.1.07.04	Volume changes in materials – not thermally induced	A, swelling/shrinking	Covered by pilot seals
2.1.07.07	Displacement of shaft seal	A, restraint strains or a consequence of forces/stresses	VSG
2.1.08.08	Swelling of bentonite	A, swelling	VSG
2.1.09.02	Solution and precipitation	A, consequence of chemical action incl. temperature	supplemented
2.1.09.06	Corrosion of materials with cement or magnesium oxychloride phases	A, consequence of chemical action incl. temperature	VSG
2.2.01.01	Excavation damaged zone	Neither DS, A nor R but component of the seal	-
2.2.06.01	Change of stresses	A, dead load, rock pressure, fluid pressure, flow forces, restraint stresses	VSG

Table 2. Classification of less likely FEPs within the technical proof of function (DS = Design situation, A = Action, R = Resistance)

FEP-No.	FEP-Name	Classification within the proof of function	Processed in
2.1.07.05	Early failure of a shaft seal	DS, abnormal situation	VSG
2.1.07.06	Early failure of a drift seal	DS, abnormal situation	VSG
2.1.08.05	Piping in seals	A, consequence of chemical actions or flow forces	VSG (covered)

4 Dimensioning and detailed design

The dimensioning is done mainly by numerical calculations. In the case of VSG, the submodeling technique was used in order to cover large-scale far field actions (impacts) as well as small-scale near field actions, e.g. modeling the hydraulic saturation of the bentonite seal. Details are given in (GRS 2012b). All FEPs processed numerically in VSG within the technical proof of function are denoted by "VSG" in Table 1 and Table 2. Some aspects of the seismic design situation for the salt concrete seal were subsequently investigated by numerical calculations (Neubert 2014). The empirical assessment that mass forces of a site-specific earthquake are of minor influence and will not affect the hydraulic resistance of the salt concrete seal was confirmed. Furthermore, FEP 2.1.09.02 “solution and precipitation” was investigated quantitatively. It has to be considered because temperature increase due to heat-generating radioactive waste may affect the salt content of the intruding brine close to the emplacement areas. As the rock salt solubility does not highly depend on the temperature, it turned out that the influence of “solution and precipitation” may be neglected.

As the last step, the detailed design is specified after dimensioning in order to serve as basis for the construction process. The detailed design was not directly addressed in VSG. Instead, it was referred to existing pilot seals.

5 Summary

As a result, the closure system shows its proper functionality for the designed repository concept at the Gorleben site. The technical proof of function was achieved using calculation methods mainly. In some cases, results from in-situ tests were used. Very few issues that need further research for quantification were identified. Nevertheless, using current knowledge, it can be concluded that safe containment of radioactive waste at the Gorleben site is a realistic possibility.

Acknowledgements

The work referred to in this paper was carried out in cooperation with colleagues from BGR, BfS, DBE, GRS, IfG, Asse GmbH (former GSF/HMGU), IBeWa, and TU Clausthal. Many thanks to them for their excellent collaboration. We especially wish to thank Natalie Neubert and Prof. Heinz Konietzky for their contributions to the seismic design situation.

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FEBEX-DP – Dismantling the "Full-scale Engineered Barrier Experiment" after 18 Years of Operation at the Grimsel Test Site, Switzerland

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The FEBEX experiment at the Grimsel Test Site (GTS) consists of an in-situ full-scale engineered barrier system (EBS) test for the disposal of high-level waste (HLW). It is performed under natural conditions in crystalline rock in which the canisters are placed horizontally in drifts and are surrounded by a clay barrier constructed of highly compacted bentonite blocks. A partial dismantling and sampling of the EBS was carried out during 2002.

Heating of the FEBEX started in 1997 and since then a constant temperature of 100°C has been maintained, while the bentonite buffer has been slowly hydrating in a natural way. A total of 632 sensors in the bentonite barrier, the rock mass, the heaters and the service zone record temperature, water saturation, humidity, total pressure, displacement, and water pressure.

The hydration pattern is relatively symmetric, with no major differences along the axis. Although the host rock is characterized by heterogeneities with zones of higher permeability, the resaturation process is driven by the suction of the bentonite rather than by the availability of water in the rock, especially in the early phase. After 17 years, the water content in the buffer close to the heater still continues to increase slowly. The hydraulic pore pressures in the buffer and the geosphere have practically stabilized. The total pressure in general continues to increase in most points into the buffer, where in some parts pressures of over 6 MPa are registered.

The long monitoring phase and the partial dismantling in 2002 indicate that the EBS has largely performed as expected and the major processes and couplings affecting the buffer saturation during the initial thermal period identified prior to the start of the experiment have been confirmed. A comprehensive report documents and reviews the state of the FEBEX (Lanyon & Gaus, 2013).

After 18 years of operation the experiment will be excavated and dismantled in 2015. The main objectives of the FEBEX dismantling project (FEBEX-DP) are:

- Characterization of the key physical properties (e.g., density, water content) of the barrier and their distribution
- Characterization of corrosion and microbiological processes on instruments and coupons resulting from evolving redox conditions and saturation states, including gas analysis
- Characterization macro- and micro level studies of mineralogical interactions at material interfaces (e.g., cement-bentonite or iron-bentonite, rock-bentonite)
- Assessment of sensor performance
- Further increasing understanding of the thermo-hydro-mechanical (THM) and thermo-hydro-chemical (THC) processes through integration of monitoring and dismantling results

An intensive laboratory program will be conducted in 2015/2016 in order to achieve the outlined main objectives. It includes extensive mineralogical, chemical and biological investigations of the buffer and the related interfaces. It will be accompanied by pre- and post dismantling modeling efforts. Unique data are expected after completing one of the longest running 1:1 in-situ EBS experiment under continued heating and natural saturation conditions. It will further consolidate the EBS knowledge and will act as a benchmark for major coupled modeling codes.

The dismantling project is set-up as an international project with partners from Europe, Asia and North America. Further information can are under http://www.grimsel.com/gts-phase-vi/febex-dp/febex-dp-introduction.

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SESSION 7 SITING STRATEGIES

VerSi: A Method for the Quantitative Comparison of Repository Systems

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Decision making and design processes for radioactive waste repositories are guided by safety goals that need to be achieved. In this context, the comparison of different disposal concepts can provide relevant support to better understand the performance of the repository systems. Such a task requires a method for a traceable comparison that is as objective as possible.

We present a versatile method that allows for the comparison of different disposal concepts in potentially different host rocks. The condition for the method to work is that the repository systems are defined to a comparable level including designed repository structures, disposal concepts, and engineered and geological barriers which are all based on site-specific safety requirements.

The method is primarily based on quantitative analyses and probabilistic model calculations regarding the long-term safety of the repository systems under consideration. The crucial evaluation criteria for the comparison are statistical key figures of indicators that characterize the radiotoxicity flux out of the so called containment-providing rock zone (einschlusswirksamer Gebirgsbereich). The key figures account for existing uncertainties with respect to the actual site properties, the safety relevant processes, and the potential future impact of external processes on the repository system, i.e., they include scenario-, process-, and parameter-uncertainties.

The method (1) leads to an evaluation of the retention and containment capacity of the repository systems and its robustness with respect to existing uncertainties as well as to potential external influences; (2) specifies the procedures for the system analyses and the calculation of the statistical key figures as well as for the comparative interpretation of the key figures; and (3) also gives recommendations and sets benchmarks for the comparative assessment of the repository systems under consideration based on the key figures and additional qualitative arguments.

We demonstrate the capabilities of the method by comparing two rather different repository systems: (1) a repository in salt with layout and site properties corresponding to the Gorleben site according to the state of knowledge and planning of 2007; (2) a fictive repository in the Lower Cretaceous claystones at a synthetic site in northern Germany with a layout corresponding to the repository planning in Switzerland and France in the years around 2005.

The exemplary application of the method shows the kind of statements which can be derived and the restrictions that are often connected to these statements. In particular, it is shown that the two repository systems exhibit fundamentally different probability-consequence profiles. The main reason for this is that, according to the model calculations, radionuclide release from the salt repository occurs only in one scenario that does not represent the expected system evolution, whereas radionuclide release from the claystone repository occurs, though mostly at very low rates, in all scenarios. Such fundamentally different probability-consequence profiles have to be taken into account when evaluating the results of the presented or any other comparison method.

It must be emphasized that no conclusions on the suitability of the considered host rocks in general or the Gorleben site in particular may be drawn from the exemplary results presented here.

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Geological Boundary Conditions for a Safety Demonstration and Verification Concept for a HLW Repository in Claystone in Germany – AnSichT

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Abstract: *Within the framework of the R&D project "AnSichT", DBE TECHNOLOGY, BGR and GRS are developing a method to demonstrate the safety of a HLW repository in claystone in Germany. The methodological approach basing on a holistic concept, links the legal and geologic boundary conditions, the disposal and closure concept, the demonstration of barrier integrity, and the long-term analysis of the repository evolution as well. The geologic boundary conditions are specified by the description of the geological situation and generic models, the selection of representative parameters and geoscientific long-term predictions. They form a fundament for the system analysis.*

1 Introduction

During the last decades a largely global agreement emerged to dispose of radioactive waste in deep geological formations. The geology is one factor for the safe long-term disposal because it acts as a main part in the multibarrier system. To ensure the long-term safety of a repository, it is necessary to investigate the effects of the repository onto the biosphere for the next one million years. Other German research projects developed the method to demonstrate the safety of a HLW (high-level radioactive waste) repository in salt rock. With the new site selection law (2013) the three possible host rock types salt rock, crystalline rock and claystone need to be considered for the disposal of HLW. The R&D project "AnSichT" aims to develop a method to demonstrate the safety of a HLW repository in claystone on basis of other research projects for salt rock (ISIBEL, VSG). For this purpose new fundamentals like disposal and closure concepts for claystone and specific geological boundary conditions are required.

2 Geological boundary conditions

2.1 Applied criteria and data basis

To demonstrate the safety and verification concept, the selection of a representative claystone formation is necessary. An evaluation of argillaceous rock formations has identified areas in Germany which are considerably suitable for a HLW repository (Hoth et al. 2007). These areas provide an outline to localize representative claystone formations. In this approach the safety requirements of BMU (2010) and the criteria of the AkEnd (2002) are considered in localizing areas which are suitable to test the new method. The following selection criteria were applied: The disposal depth between 300 m and 1000 m, a host rock thickness of minimum 100 m, a range of more than 10 km^2 and permeability values of less than 10^{-10} m/s. Earthquakes (maximum of zone 1), active fault zones, shift in the relative level of sea and land of more than 1 mm/a and active volcanism are criteria of exclusion. The results of a preliminary application of these criteria were summarized by (Reinhold & Sönnke 2012). The data basis for the geological boundary conditions derive from the exploration industry on oil, gas, salt and other natural resources. Further data from seismics and data from site investigations of the Konrad Mine or in-situ data from underground research laboratories in argillaceous rock formations (e.g. URL Mont Terri) have been compiled.

Due to the different geological conditions of Northern and Southern Germany, different implications have to be considered. Therefore the selection of just one host rock formation which represents an argillaceous rock formation in Germany is not possible and two reference areas are selected. The considered areas comprise parts of the North German Basin and a smaller part of the Molasse Basin in Southern Germany.

2.2 Development of geological models

The developed geological model for Northern Germany represents a typical situation where potential host rock formations are bedded in a suitable depth under 600 m (Reinhold et al. 2013). The basin is structured in a basement, a cover of sedimentary rock and quaternary sediments. The model contains 14 units from the basis Zechstein until the Quaternary (Fig. 1). The units are relatively homogeneous and regional characterizable geological formations. Salt domes or active fault zones are excluded from the model to guarantee a representative position. Also a position where deep subglacial channel systems may occur are excluded. The size of the model is 70 km^2.

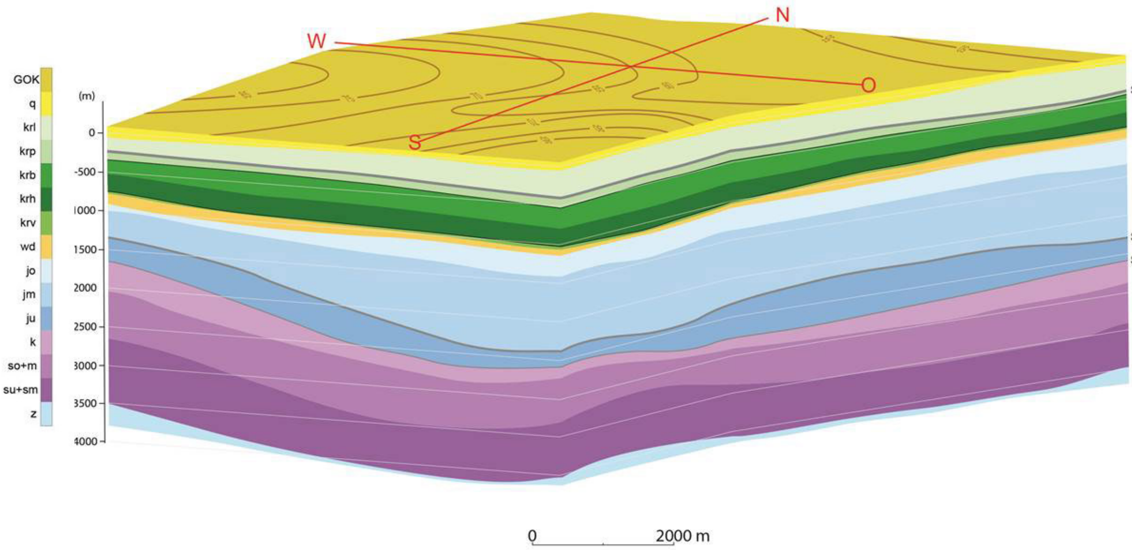

Fig. 1. Geological model for Northern Germany.

The Barremium & Hauterivium formation of the Lower Cretaceous represent the host rock formations. The host rock formations consist of claystones and clayey marl and subordinate micritic lime marl. The hydrogeologic conditions include a near-surface groundwater reservoir of low salinity (unconsolidated quaternary sediments) and several deeper aquifers of high salinity (Rhätsandstein = S1, Aalensandstein = S2 and Hilssandstein = S3). The topography of the model is dominated by the glacial sediments of the Saale glacial period.

The geological model for Southern Germany is situated in the north alpine Molasse Basin. The region is structured in a crystalline basement, a cover of mesozoic sedimentary rock, molasse sediments and quarternary sediments. The model contains 16 units from the basis Muschelkalk until the Quaternary (Fig. 2), which are relatively homogeneous and regional characterizable geological formations. Active fault zones are excluded from the model to guaranty a representative position. The model regards an area of 140 km².

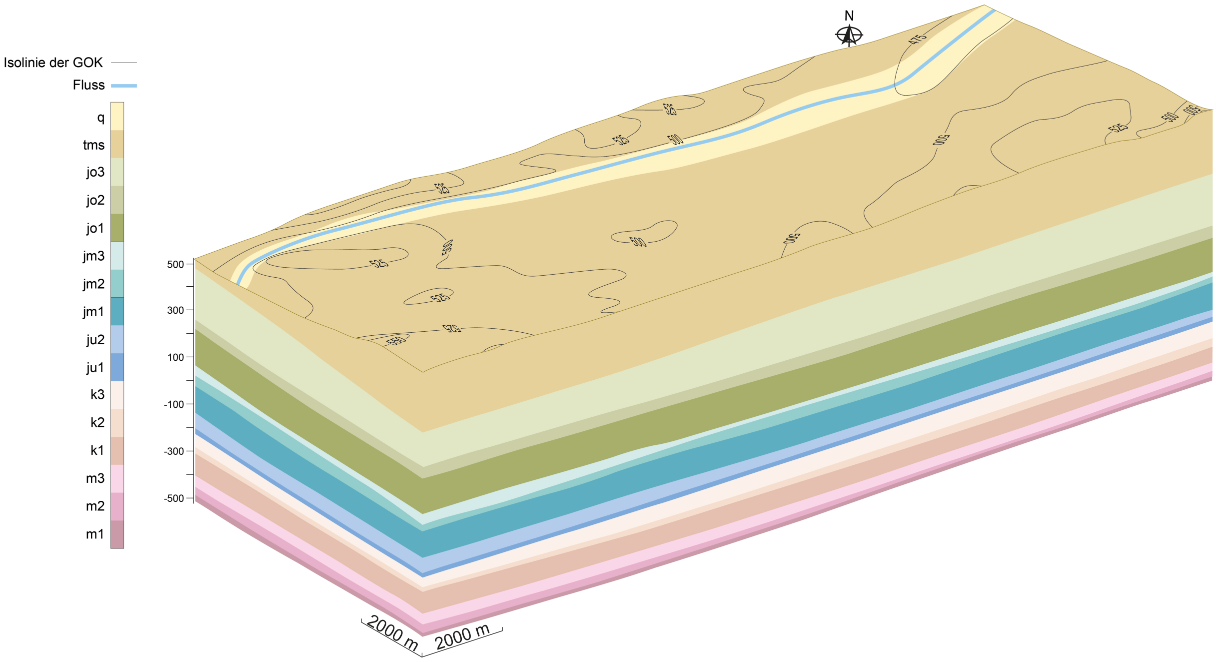

Fig. 2. Geological model for Southern Germany.

For the geological model in Southern Germany the Opalinus Clay in the Middle Jurassic is defined as host rock formation. The Opalinus Clay consist of claystones which show a low variability in facies and lithology. The hydrogeologic conditions include a near-surface groundwater reservoir (unconsolidated quarternary sediments) and several deeper aquifers with higher salinity (e.g. Oberer Muschelkalk = m3, Stubensandstein = k2 and Oberer Jura = jo2). The model-unit jo2 represents the strong dolomitized and karstified lime stones of the Oberer Jura. The topography of the model is mainly shaped by erosion and deposition of fluvial sediments of several glaciations in the Quaternary.

2.3 Hydraulic and petrophysical rock properties

Representative parameters for the hydraulic and petrophysical rock properties are assigned to the geological units of the models (e.g. Jahn & Sönnke 2013). The specifications comprise average values and bandwidths of petrographic, mechanic, thermic and hydraulic parameters. For numerical simulations out of the bandwidth certain parameters have been selected and used for demonstrating the integrity of the geologic barrier (e.g. Nowak & Maßmann 2013).

2.4 Geoscientific long-term predictions

For the safety demonstration the evolution of the geosphere during the demonstration period of one million years is crucial. With geoscientific long-term predictions further geologic boundary conditions can be evaluated. The evolution of the geosphere is influenced by regional and supraregional processes, which differ significantly in Northern and Southern Germany. Some processes appear less likely or can be excluded in the different areas. Northern Germany was strongly affected by the glacial periods of the

Elsterian, Saalian and Weichselian glaciations as the Fennoscandian ice sheets covered wide areas of the Central European Basin System. The geosphere was influenced by glacial processes such as the accumulation of glacial deposits, deep subglacial channel systems or ice-sheet loading. The model site in Southern Germany was influenced by the Würmian, Rissian, and Hosskirchian glaciations. There the Rhineglacier coming from the Alps covered a smaller part of the North Alpine Basin and led to the accumulation of glacial deposits or the formation of over deepened basins although the influence of the ice-sheet loading must have been smaller because of the thinner ice thickness. Further differences arise from the tectonic setting. The geosphere in Northern Germany was affected by the development of the Lower Saxony Basin. In this setting salt tectonics play a major role and led to the formation of salt diapirs or pillows and changes in the sedimentary evolution. In contrast Southern Germany was predominantly affected by the orogenesis of the Alps and the evolution of the North Alpine Foreland Basin.

3 Summary and Outlook

The description of the geological situation and development of suitable models, the selection of representative parameters and the geoscientific long-term predictions provide a fundament for the system analysis, which is a main part of the safety assessment. According to the geological models two FEP (features, events and processes) catalogs are developed where all relevant factors for the long-term safety assessment of HLW repositories are recorded. The FEP catalogs are used for the scenario development. Two generic sites with different boundary conditions are available where the method to demonstrate the safety of a HLW repository in claystone in Germany can be applied and advanced.

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The Development of Rock Suitability Classification Strategies in the Finnish Spent Nuclear Fuel Disposal Program

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Abstract: *This paper describes the development of the rock suitability classification strategies applied to locate the spent fuel repository in crystalline rock in Finland. Development of the classification procedure is motivated not only by the regulatory requirements, but also by the need to more closely integrate site characterization, repository design and long-term safety assessment. The classification procedure has been developed along with the increasing level of detail of the available site data and knowledge on the performance of the engineered barrier system (EBS). The classification system has also been adapted to the changes in the regulations. The present form of the classification system and experiences from testing the system at the site are described. Demonstration activities have shown that the criteria and the stepwise research, construction and decision making protocol can be applied successfully.*

1 Introduction

In Finland, several sites have been investigated as a potential site for geological disposal of spent nuclear fuel since the 1980s (McEwen and Äikäs 2000). In 2001, the Parliament of Finland endorsed a Decision-in-Principle (DiP) whereby the spent fuel generated in the Loviisa and Olkiluoto power plants will be disposed in a geologic repository constructed at the Olkiluoto site in South-Western Finland. An underground rock characterization facility, ONKALO, reaching the planned repository depth (400 meters below ground) has been constructed at the site. In 2012, Posiva Oy submitted a construction license application for a KBS-3 type spent nuclear fuel repository to be constructed at the Olkiluoto site. The application is currently under regulatory review.

The rock conditions at the sites studied during the site selection phase, and the Olkiluoto site in particular, have been found to be favorable for the long-term safety of a geological disposal in general (McEwen and Äikäs 2000). However, there are features of the host rock, e.g. deformation zones that constrain the depth and layout of the repository and other underground openings. Rock classification strategies have been developed in order to take such constraints into account.

2 Regulatory context

The Finnish regulations (STUK 2014) specify several safety functions for the natural barrier, i.e. the host rock: stable and intact rock, low groundwater flow, favorable groundwater chemistry, retardation of the radioactive substances and protection provided against natural phenomena and human actions. There have been changes in the regulations over the years, which have meant changes in the formalism of how these requirements are treated. In addition, regulations exist on classification of the rock structures and properties that may have an adverse impact on the long-term safety and that should, thus, be taken into account in the repository layout.

3 Evolving site characterization and rock classification

3.1 Site selection phase

The suitability of crystalline rock for disposal has been one of the key topics since the selection process for a spent nuclear fuel disposal site began in the 1970s in Finland (e.g. Niini 1978, Peltonen and Rouhiainen 1980). In the 1990s, four potential crystalline sites were being investigated and the knowledge about the sites was increasing along with the progress of the investigations. The early rock mass classification systems applied by Posiva in mid-'90s, were focused on the identification and classification of large-scale, potentially unstable or transmissive fracture zones and on the definition of respect distances between the tunnels and these zones. The layout adaptations by Riekkola et al. (1996) and Äikäs and Riekkola (2000) considered these hydrogeologically- and geotechnically-significant fracture zones. The proposed respect distances to these fracture zones, evaluated mainly based on transmissivity, were based on an evaluation of long-term safety.

In the late 1990s, rock mass classification was, however, expanded to consider also the properties in the rock mass between the deformation zones, focusing on the constructability of the rock mass (e.g. Äikäs et al. 2000). The parameters used in the classification fell under the categories of rock quality factors, state of stress, groundwater chemistry and rock engineering properties. The classification was general enough to be applied to the different sites, although utilizing relatively detailed data from the sites. The results of the classification, particularly with respect to the properties of the fracture zones and the strength/stress ratio that indicated a low suitability for a disposal depth greater than 500 m at Olkiluoto, were taken into account in the layout work by Äikäs and Riekkola (2000).

3.2 Towards site specific classification system

In the early 2000s, after the Olkiluoto site had been selected as the disposal site for spent nuclear fuel, Posiva launched the Host Rock Classification (HRC) project (Hagros 2006) in order to develop the first comprehensive classification system for host rock suitability, taking into account both the engineering properties of the rock mass and the long-term safety aspects. In this phase, the properties of the host rock supporting the performance of the EBS were also taken into account in the classification. The rock volumes for disposal were classified on several scales, i.e. the repository, tunnel and canister. The classification system was intended to be a design tool used in all phases of repository development, starting from the layout adaptation of the whole repository and covering all subsequent phases of design and construction until the boring of the deposition holes and their final acceptance. The HRC-system was initially developed for the Olkiluoto site (Hagros et al. 2005), but was later expanded to cover any disposal site in crystalline bedrock (Hagros 2006). The system considered the site data collected from surface-based and, later, underground studies and defined the parameter data that need to be acquired to perform the classification.

3.3 Classification system for construction license application

To prepare for the construction license application, the links of classification system both to the safety case and to repository design and construction were to be strengthened and a two-phased Rock Suitability Criteria (RSC) program was started in 2007. The first phase of the program, reported by Hellä et al. (2009), concentrated on defining criteria for the host rock, with the focus on the rock properties contributing to the long-term safety. The RSC criteria are seen as a tool to select the rock volumes for the deposition tunnels and holes, similar to requirements concerning the design and production of the engineered barriers, and have been derived from the target properties, which set the general long-term safety related requirements for the host rock as a natural barrier. The resulting criteria consider measurable or observable host rock parameters dealing with hydraulic properties and mechanical stability of the host rock, as well as groundwater composition, with the aim of fulfilling the target properties also in the long-term. The criteria development has been closely related to the development of the requirements management system for the entire disposal facility.

Criteria development has been continued in the second phase of the program, although the emphasis has been on the actual classification process and its implementation as part of the overall underground openings construction process. The idea of carrying out classification at different scales introduced as part of the HRC is applied also in the RSC system. The scales used in the current system are: repository, panel, and tunnel and deposition hole. These scales coincide with different stages of repository design and construction, proceeding from the layout design of the entire repository to the implementation design and construction of panels consisting of central tunnels and several deposition tunnels and, finally, deposition holes. The current Rock Suitability Classification system is presented by McEwen et al. (2012).

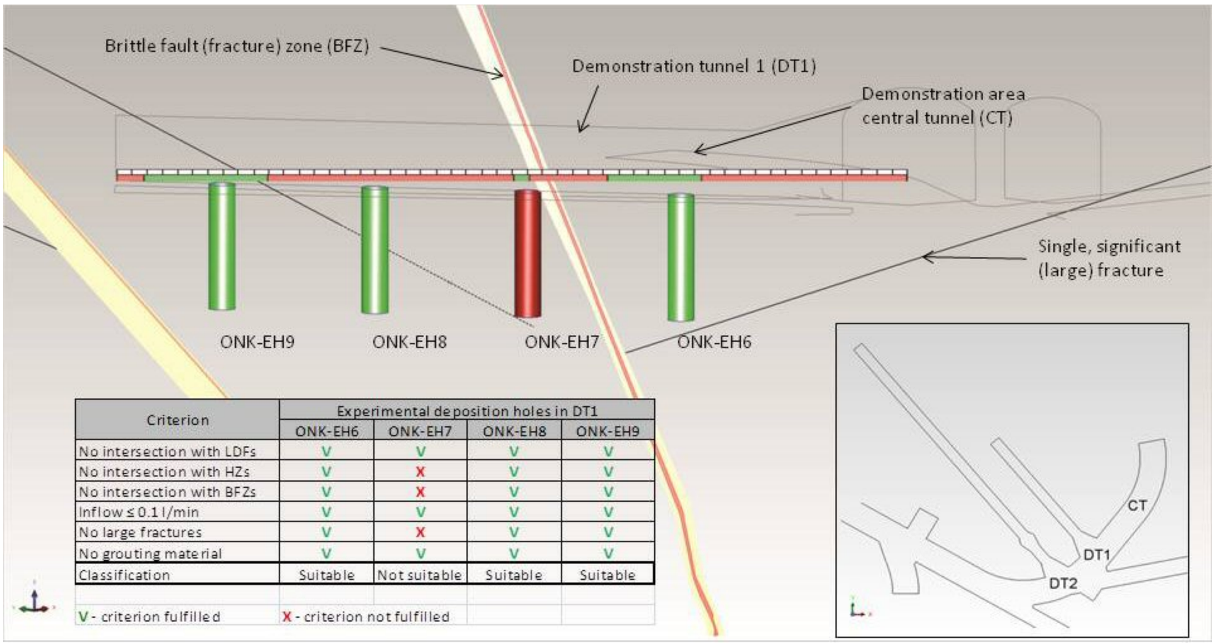
The RSC system was used to define the rock volumes suitable for the reference layout for the construction license application (Saanio et al. 2013). The effect of applying the rock suitability criteria on rock properties around the repository in the long-term (e.g. flow conditions) was discussed in Posiva (2013).

4 Testing of the RSC system

Testing in ONKALO has provided feedback on the criteria, implementation of the classification system and investigation methods used to gain data needed in the classification (McEwen et al. 2012). Currently, a full-scale test/demonstration of the RSC system is underway at the actual repository level (about -425 m), in conditions representative of the future repository to be constructed as an enlargement of the ONKALO facility. In this demonstration, the RSC system has been applied to design and construction of a demonstration facility comprising a central access tunnel and two demonstration tunnels with experimental deposition holes, mimicking the repository as closely as possible, including the employed construction methods (McEwen et al. 2012).

The construction of the demonstration facility and the RSC demonstration are currently still underway with completion expected in early 2015. To date, eight rock suitability classifications have been carried out to consecutively evaluate the suitability of the rock hosting the demonstration facility, the demonstration tunnels and the experimental deposition holes, based on data gained from geological, geophysical and hydrogeological investigations and a frequently updated model of the significant bedrock features in the area. The earliest suitability classifications of the demonstration facility relied on data collected from the nearby ONKALO access tunnel and the central tunnel of the facility. Prior to the excavation of the demonstration tunnels, so-called pilot holes (cored boreholes made for investigation purposes) were drilled on the planned demonstration tunnel locations, to gain more detailed information on the rock properties; the following rock suitability classification resulted in some changes in the planned tunnel layout. After investigations were carried out in the excavated demonstration tunnels, suitability classifications were carried out for each tunnel to define which tunnel sections would be suitable for hosting experimental deposition holes. Hole locations were selected from the suitable tunnel sections, and vertical pilot holes were drilled for further investigations at the chosen locations. After suitability classification was performed for the locations of the experimental deposition holes on the basis of the pilot hole data, their construction commenced, and the suitability of each experimental deposition hole (fulfilment of the RSC criteria) was verified in a final suitability classification after investigations in the constructed holes.

Demonstration tunnel 1 was completed in October 2012 with four constructed experimental deposition holes (three fulfilling the RSC criteria) (Fig. 1). Construction of the experimental deposition holes is currently underway in demonstration tunnel 2, where six locations have been selected based on the suitability classification of the excavated tunnel; after drilling of the vertical pilot holes, five of the six locations were preliminarily classified as suitable.



Criterion	Experimental deposition holes in DT1			
	ONK-EH6	ONK-EH7	ONK-EH8	ONK-EH9
No intersection with LDFs	V	V	V	V
No intersection with HZs	V	X	V	V
No intersection with BFZs	V	X	V	V
Inflow ≤ 0.1 l/min	V	V	V	V
No large fractures	V	X	V	V
No grouting material	V	V	V	V
Classification	Suitable	Not suitable	Suitable	Suitable

Fig. 1. Suitability of the experimental deposition holes in Demonstration tunnel 1 (DT1) in ONKALO. The image shows a vertical cross-section of the tunnel and the significant bedrock structures (brittle fault zones and single large fractures) modeled on the basis of investigations in the excavated DT1 and older data from the area. The bar above the experimental deposition holes (ONK-EH...) shows the result of the suitability classification based on the tunnel data and the model; green sections were classified as suitable for deposition holes, red ones as not suitable. The location of ONK-EH8 was deemed as potentially suitable in the classification carried out after drilling of the vertical pilot holes; the suitability was verified in the final classification after the construction of the holes (see the inserted table). The hole ONK-EH7 was constructed even if the location was classified as unsuitable to test run the prototype deposition hole boring rig.

5 Summary

During the years, the classification strategies have become more tightly linked to the safety case and the design and construction of the underground rock facilities. The systems have evolved along with the increasing amount of information about the site, knowledge on the performance of the EBS and lately also experiences gained from testing in ONKALO. Also, the system has been adapted to the changes in regulations.

The RSC system provides a structured method for positioning the disposal facility, so that the less favorable volumes of rock, e.g. brittle fault zones and hydraulically-conductive features, are avoided. The main outcome of the demonstration activities performed to date indicates that the criteria developed are applicable in practice and that the stepwise research, construction and decision-making protocol can be applied successfully.

However, needs for further development of some of the criteria, further streamlining the flow of investigations, design and construction, as well as the associated decision-making and documentation process have also been identified. Also, the feedback from the performance assessment assessing how well the rock suitability classification contributes to long-term safety (Posiva 2013) and new information about the site and EBS design are taken into account in the development work (Posiva 2012).

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SESSION 8 SCIENTIFIC ASPECTS OF THE NUCLEAR WASTE DISPOSAL SAFETY CASE (PART II)

Application of Fundamental Aquatic Chemistry to the Safety Case and the Role of Thermodynamic Reference Data Bases

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All national and international programs developing a Nuclear Waste Disposal Safety Case have recognized the essential requirement of assessing aqueous (radionuclide) chemistry and establishing reliable thermodynamic databases. Long-term disposal of nuclear waste in deep underground repositories is the safest option to separate highly hazardous radionuclides from the environment. In order to predict the long-term performance of a repository for different evolution scenarios, the potentially relevant specific (geo)chemical systems are analyzed. This requires a detailed understanding of solubility, speciation and thermodynamics for all relevant components including radionuclides, and the availability of reliable thermodynamic data and databases as fundamental input for integral geochemical model calculations and hence PA.

Radionuclide solubility and speciation strongly depend on chemical conditions (pH, E_h, matrix electrolyte system and ionic strength) with additional factors like the presence of complexing ligands or temperature further impacting solution chemistry. As the fundamental chemical key processes are known and convincingly described by general laws of nature (→ solution thermodynamics), the long-term behavior of a repository system can be analyzed over geological timescales using geochemical tools. A key application of fundamental aquatic chemistry in the Safety Case is the determination of solubility limits (radionuclide source terms). Based upon fundamental chemical information (on solid phases, complexation reactions, activity coefficients, etc.), the maximum amount of radionuclides potentially dissolved in a given volume of solution and transported away from the repository, are quantified. A detailed understanding of radionuclide chemistry is also crucial for neighboring fields. For example, advanced mechanistic understanding and modeling of sorption processes at the solid liquid interphase, waste dissolution processes, secondary phase and solid solution formation, must use qualitatively and quantitatively correct radionuclide speciation schemes.

The high relevance of correct, complete and consistent thermodynamic data and supplying databases which allow a robust prediction of solution chemistry has been recognized by the international nuclear waste disposal community since decades. Over the last 20 years, the Thermodynamic Database Project of OECD-NEA (http://www.oecd-nea.org/dbtdb) has significantly contributed to the present, largely positive, situation. The NEA-TDB project publishes a series of critically reviewed and evaluated compilations of consistent thermodynamic data, widely accepted as reference values for key elements at low or intermediate ionic strengths conditions. With regard to modeling systems at extremely high ionic strength, activities are currently initiated to (i) prepare a state-of-art-report on Pitzer modeling within NEA-TDB and (ii) set up a working group within the NEA Salt Club to work towards a Joint International Pitzer Database. In Germany, the THEREDA project is developing a German Thermodynamic Reference

Database, aiming at providing a comprehensive and internally consistent thermodynamic reference database for the geochemical modeling of all near-field and far-field processes relevant in Germany.

In this presentation, the important contributions from fundamental aquatic chemistry in support of the Nuclear Waste Disposal Safety Case are highlighted. Thermodynamic reference databases and their relevance for the Safety Case are analyzed. Based upon a critical assessment of the status quo, positive directions for future research activities and international cooperation are discussed and prioritized.

Assessment of the Long-Term Safety for SFR

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During operation and decommissioning of the Swedish nuclear facilities, radioactive waste is generated that must be disposed of. Besides waste from the nuclear facilities, some waste derives from other activities such as industry, research, medical care, etc.. Short-lived low- and intermediate-level waste from these activities is disposed of in the final repository for short-lived radioactive waste, SFR, in Forsmark. The facility, which has been in operation since 1988, is owned and operated by Svensk Kärnbränslehantering AB, SKB. The existing facility has neither sufficient space nor a license to receive decommissioning waste. SFR must therefore be extended so that shortlived low- and intermediate-level decommissioning waste from the nuclear facilities can also be received. The need for additional capacity has been accentuated by the closure of two reactors in Barsebäck. These reactors cannot be dismantled until the SFR facility has been extended.

The exsiting repository is built to receive, and after closure serve as a passive repository for, low- and intermediate-level radioactive waste. The disposal rooms are situated in the bedrock beneath the sea floor, covered by about 60 metres of rock. The repository has been designed so that it can be abandoned after closure without requiring further measures to maintain its function.

The extension of SFR, is done at the -120 m level immediately adjacent to, and within the same depth range as, the existing facility. The basic function of the existing SFR and of the extended one will be the same. However, a clear difference is the design of the tunnel and the rock vault that are required to permit transport and storage of whole reactore pressure vessels.

The application for a license to build this extension includes an assessment of the long-term safety (post-closure safety) of the facility. The safety assessment also contains an updated assessment of the long-term safety of the existing facility.

The safety assessment for SFR has two clear roles. The first one is a normative and controlling role to propose a repository design that meets the requirements made on long-term function, for example best available technology. The second one is an analyzing and reporting role where the chosen design is analyzed to determine whether the requirements made on long-term function are met. The methodology used in the assessment is similar to that used by SKB to produce the documents included in the application for the Spent Fuel Repository.

The repository system – broadly defined as the waste, the waste containers, the engineered barriers, the host rock and the biosphere surrounding the repository – will change with time. The future state of the system will depend on the following:

- the initial state of the system,
- a number of thermal, hydraulic, mechanical and chemical processes that act internally in the repository system over time (internal processes),
- outside forces acting on the system (external processes).

Based on this, the evolution of the repository is estimated. By combining this with an analysis of future exposure, radiological risk can be estimated.

In-situ Experiments to Investigate Rock Matrix Retention Properties in ONKALO, Olkiluoto, Finland

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Abstract: *Spent nuclear fuel from nuclear power plants, owned by TVO (Teollisuuden Voima Oy) and Fortum, is planned to be disposed to a repository at a depth of more than 400 meters in the bedrock of Olkiluoto (Eurajoki, Finland). The repository system of multiple release barriers consists of both manmade and natural barriers. The surrounding rock acts as the last barrier if other barriers fail during passage of the millennia. Therefore, safe disposal of spent nuclear fuel requires information on the radionuclide transport and retention properties within the porous and water-containing rock matrix along the water conducting flow paths. To this end, various types of experiments are being performed and planned within ONKALO, the underground rock characterization facility in Olkiluoto, as part of the project "rock matrix REtention PROperties" (REPRO). The research site is located at a depth of 420 meters close to the repository site. The aim is to study the diffusion and sorption properties of nuclear compounds in the rock matrix under real in-situ conditions. The first in-situ experiment was performed during 2012 using HTO, Na-22, Cl-36 and I-125 as tracer nuclides. Breakthrough curves show retention and asymptotic behavior that are in-line with those caused by matrix diffusion and sorption were observed in their breakthrough curves. Weak sorption was also observed in the breakthrough curves of Na-22 and I-125.*

1 Introduction

It is known that elements migrating in water that flows in a fracture are transported at a lower speed than that of the water (Neretnieks, 1980). The retention is caused by diffusion of these elements away from the flowing water to stagnant water in the pores of the surrounding rock matrix, and by their chemical sorption on mineral surfaces of the rock matrix, fracture walls and fillings (Voutilainen et al., 2010, Kekäläinen et al., 2011). However, the relevance of retention in repository conditions has remained unclear, and the impact of retention varies from one nuclide to another.

In general, it is believed that conditions might be different in laboratory than in-situ due to, e.g. stress relaxation caused by sampling. Sawing artifacts and stress relaxation might change the characteristics of rock sample and increase porosity and in that way enhance migration of elements by diffusion. Furthermore, these changes may increase specific surface area available for sorption, which would increase the value of distribution coefficients. The same effect is seen in laboratory batch sorption experiments when using crushed rock samples. In-situ experiments avoid some of the problems caused by sampling and stress relaxation. To this end, it is important to measure diffusion and sorption properties with in-situ experiments and compare their results with those from laboratory experiments. In addition, in the performance assessment conditions most of the retention takes place in the vicinity of the deposition holes. This emphasizes the significance of careful quantification of the retention properties of the bedrock in the conditions present in the repository near-field.

The objective of the REPRO project is to perform experiments so as to investigate the retention properties of the rock matrix under realistic in-situ conditions. In first in-situ experiment, Water Phase matrix Diffusion Experiment (WPDE), a short concentrated pulse of selected radionuclides was injected into a water flow through an artificial fracture. This artificial fracture was formed on the circumference of a 2 m long packed-off section of drillhole by placing a cylindrical flow guide on center of the drillhole. Performing the tracer experiment along artificial fracture in packed-off drillhole section enables a better control of the flow field and a better recovery of tracers, and thus this experimental set-up allows a more precise characterization of the rock matrix properties. Similar in-situ experiments like here have recently been performed at the Äspö Hard Rock Laboratory, Sweden, and at Grimsel test site, Switzerland (Widestrand et al., 2010, Soler et al., 2013).

2 Experimental

2.1 Site description

The in-situ experiments are performed in ONKALO, the underground rock characterization facility in Olkiluoto, Finland. Preliminary plan of site selection and experiments were done by Aalto et al., 2009. The REPRO research niche was excavated at a depth of 420 m (see Fig. 1) and experimental drillholes were done away from the large water conducting fractures. Special attention was paid to drilling in order to keep the drillholes as straight as possible and to avoid mechanical damage in the drillhole walls.

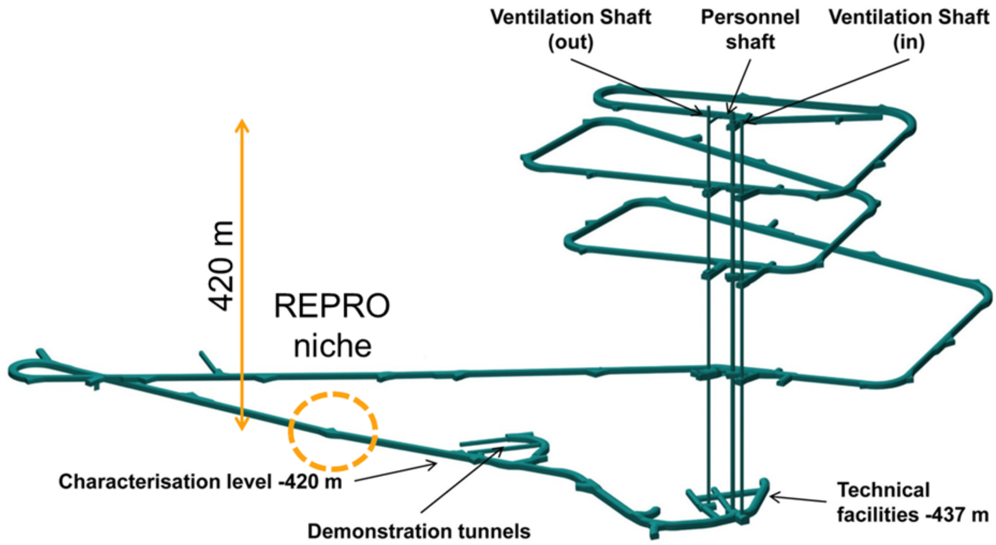

Fig. 1. REPRO research niche locates at a depth of 420 m in the underground rock characterization facility (ONKALO) in Olkiluoto, Finland.

In general, the bedrock around the REPRO niche is pegmatitic granite and migmatitic gneiss (subgroup: veined gneiss). Pegmatitic granite is from light gray to pale colored with coarse K-feldspar and commonly also cordierite and garnet. Migmatitic gneiss is a metamorphic, heterogeneous mixture of small grained mica gneiss and coarser leucosome veins of thickness that varies from several millimeters up to ten centimeters. Experiments were planned to be performed in unfractured veined gneiss and at least 7 m away from the tunnel wall. Foliation of veined gneiss in the experimental area is mainly weak to moderate banded foliation. Fracturing, fracture fillings and rock quality were also investigated from the drillcores. A more detailed geological description of the site is given by Toropainen, 2012.

2.2 Water phase matrix diffusion experiments (WPDE)

A special double-packer system was constructed for the WPDE (see Fig. 2). The packer system forms a two-meter artificial flow channel along the perimeter of the drillhole. The volume and aperture of the flow channel are minimized by an impermeable cylindrical flow guide inside the packer system. The inlet and outlet positions of water are located at the opposite ends of the packed-off section. WPDE tracer tests are performed using slow flow rates that are generated using a piston pump. The experiment is executed using synthetic groundwater to carry to tracer solution. Composition of the synthetic groundwater was prepared based on a careful analysis of real fracture water in the REPRO site. A sharp pulse of tracer solution was injected into water flow which was then conducted to the inlet of the experimental drillhole section. Tracer molecules were diffused into the rock matrix and sorbed at the mineral surfaces during migration through flow channel. The early part of the breakthrough curve is dominated by the advection, diffusion and dispersion in the flow channel. However, matrix diffusion and sorption were expected govern the late part of the breakthrough curve.

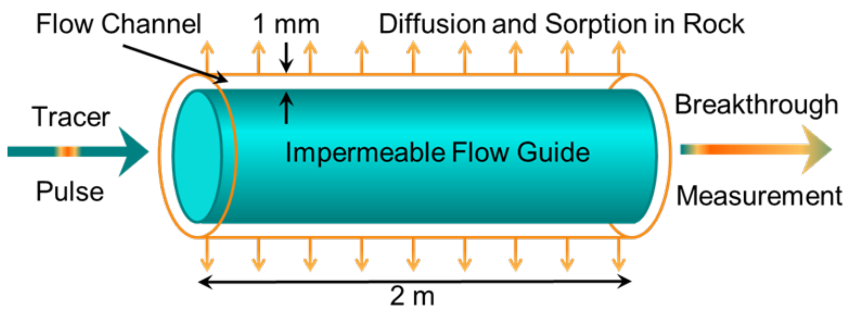

Fig. 2. WPDE experiment studied retention of HTO, Na-22, Cl-36 and I-125 by matrix diffusion for flow through an artificial fracture that is formed along a packed-off drillhole section with the help of an impermeable cylindrical flow guide

The measurements need to be carried out using low enough flow rates in order to see a measurable effect of the matrix diffusion, but also high enough so that the experiment time remains tolerable. Based on the predictive modeling, flow rates of 20, 10, and 5 μl/min were chosen since multiple flow rates support interpretation of the experimental results. The first WPDE experiment has been performed with a flow rate of 20 μl/min and the second one with 10 μl/min is currently running.

In the first experiment HTO, Na-22, Cl-36, and I-125 were used as tracers. Tracers were selected such that the tracer cocktail contains a conservative (HTO), an anionic (Cl-36), and a weakly sorbing (Na-22). I-125, emitting low energy gammas, was included in the cocktail for online detection of the breakthrough by an gamma detector. In laboratory, the concentrations of Na-22 and I-125 were first measured by gamma spectroscopy (GS) from collected water samples. After GS, chloride and part of iodide were precipitated from the solution using $AgNO_3$ and Cl-36 was then measured by liquid scintillation counting (LSC) from dissolved precipitate. Both Na-22 and I-125 interfere with measurement of HTO which was measured from the supernatant by LSC. Na-22 and I-125 were measured after LSC counting once again with GS and their concentrations were then subtracted from the HTO concentration.

3 Results

The first WPDE experiment using radionuclides was performed during 2012. In this experiment water was pumped through the artificial flow channel with a constant flow rate of 20 μl/min and a mixture of

HTO, Na-22, Cl-36, and I-125 was injected as a sharp pulse (1 ml) into the flowing water. Concentrations of these radionuclides as a function of time, i.e. their breakthrough curves, were measured by taking water samples from the outflowing water (see Fig. 3). A clear power-law behavior was observed in the late part of breakthrough curve which is typical for matrix diffusion and sorption. The measured breakthrough curves were analyzed using a numerical model that takes into account advection, (Taylor) dispersion, molecular diffusion and some heterogeneities in the flow channel, and matrix diffusion, sorption and some heterogeneities in the rock matrix. Values of related parameters, found by fitting the measured breakthrough curve by the model solution, are shown in Table 1. As a result, a pore diffusion coefficient of 3×10^{-12} m²/s was determined for Cl-36, and 1.2×10^{-11} m²/s for HTO, Na-22 and I-125. Furthermore, for Na-22 and I-125 distribution coefficients of 1.3×10^{-5} m³/kg and 1.0×10^{-5} m³/kg were determined which indicates weak sorption of them. The K_d of Na-22 given by the model was about one order of magnitude lower than the one from batch sorption experiments.

Table 1. The Injected activities, porosities, pore diffusion and distribution coefficients (A_0, ε, D_p and K_d) as determined from the experiment for HTO, Na-22, Cl-36 and I-125

Nuclide	**ε [%]**	**D_p [m²/s]**	**K_d [m³/kg]**
HTO	0.7	1.2×10^{-11}	-
Na-22	0.7	1.2×10^{-11}	1.3×10^{-5}
Cl-36	0.15	$< 3\times10^{-12}$	-
I-125	0.7	1.2×10^{-11}	1.0×10^{-5}

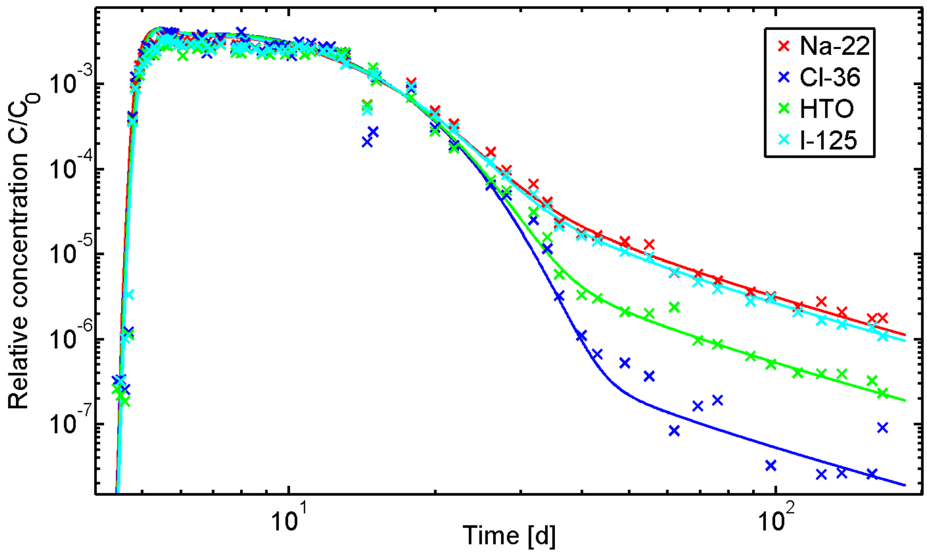

Fig. 3. Late time behavior of the breakthrough curves of HTO, Na-22, Cl-36, and I-125 show typical power-law behavior by the matrix diffusion and sorption in rock matrix.

The analysis of breakthrough curves is not straightforward. Here the analysis was done by first assuming that K_d for HTO is equal to zero and the average porosity of laboratory measurements (0.7%) was used in the model (Sammaljärvi et al. 2014). After fixing K_d and ε, D_p was adjusted so that good agreement with the experimental curve was found. In case of Na-22 and I-125, the same values for D_p and ε were used, and K_d was adjusted to find good agreement with experiment. It was impossible to explain the breakthrough curve of Cl-36 using the same values of D_p and ε as for HTO. Since it is unrealistic to decrease only D_p or ε, both were decreased so as to find good agreement with the experiment. It is possible that, as chloride is an anion, it is repelled by the negatively charged mineral surfaces and thus it can diffuse only in part of the pore space. This will cause also decrease of the diffusion coefficient. Similar results for Cl-36 have also been obtained in laboratory studies.

4 Conclusions and discussion

In-situ experiment serie for investigating the diffusion and sorption properties of selected radionuclides was introduced. The experiments will provide data on matrix diffusion and sorption of radionuclides relevant under in-situ conditions in the repository depth and valuable information for the safety analysis of the repository. In the first WPDE tracer test, the retarding effect caused by matrix diffusion and sorption were evident and analysis of the measured breakthrough curves were successfully performed. Model including some effects of heterogeneous flow field and rock was able to explain both the early (advection dominated) and late (diffusion and sorption dominated) parts of the breakthrough curves. The physical parameters linked to properties of intact rock were successfully determined from late part of the curve and the results are similar to the ones from laboratory experiments (Sammaljärvi et al. 2014). However, some questions remained still unanswered for which following experiments are trying to find answers.

The second WPDE tracer test with a lower flow rate (10 µl/min) and HTO, Na-22, Cl-36, Ba-133, and Sr-85 as the tracers is now running. The aim is to get information about diffusion and sorption for a wider range of radionuclides and at multiple time scales. Since in the first WPDE the late part of the Cl-36 breakthrough curve was near to detection limit, its initial activity was increased in second experiment. This will raise the tailing of the breakthrough curve and enable more reliable determination of the rock matrix retention properties for Cl-36. Parallel to these in-situ experiments, a number of laboratory experiments are being performed, some of which were referred to already here. Samples for the laboratory studies were taken from the drillcores of in-situ measurement interval.

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Studies on Spent Nuclear Fuel Evolution during Storage

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Initially conceived to last only a few decades (40 years in Germany), extended storage periods have now to be considered for spent nuclear fuel due to the expanding timeline for the definition and implementation of the disposal in geologic repository. In some countries, extended storage may encompass a timeframe of the order of centuries. The safety assessment of extended storage requires predicting the behavior of the spent fuel assemblies and the package systems over a correspondingly long timescale, to ensure that the mechanical integrity and the required level of functionality of all components of the containment system are retained. Since no measurement of "old" fuel can cover the ageing time of interest, spent fuel characterization must be complemented by studies targeting specific mechanisms that may affect properties and behavior of spent fuel during extended storage. Tests conducted under accelerated ageing conditions and other relevant simulations are useful for this purpose.

During storage, radioactive decay determines the overall conditions of spent fuel and generates heat that must be dissipated. Alpha-decay damage and helium accumulation are key processes affecting the evolution of properties and behavior of spent fuel. The radiation damage induced by a decay event during storage is significantly lower than that caused by a fission during in-pile operation: however, the duration of the storage is much longer and the temperature levels are different. Another factor potentially affecting the mechanical integrity of spent fuel rods during storage and handling / transportation is the behavior of hydrogen present in the cladding.

At the Institute for Transuranium Elements, part of the Joint Research Centre of the European Commission, spent fuel alterations as a function of time and activity are monitored at different scales, from the microstructural level (defects and lattice parameter swelling) up to macroscopic properties such as hardness and thermal conductivity. In order to reproduce cumulative damage effects expected after very long storage time within acceptable laboratory timescales, accelerated damage build-up conditions are applied, e.g. by using unirradiated (U,Pu) oxide with high specific alpha-activity (alpha-doped UO_2). The measurements performed so far show that saturation of macroscopic hardness and thermal conductivity occurs for a simulated timescale corresponding to spent fuel after decades or centuries of storage (the exact time scale depends on parameters like fuel composition and burnup). The outcome of the studies also show a non-negligible extent of lattice swelling, peaking at an accumulated damage level of ~1 dpa. The behavior observed in tailor-made samples is compared to actual high burnup fuel (UO_2, MOX), to verify extent and significance of eventual macroscopic alterations of spent fuel during long-term storage.

This paper will present the main outcome of the extended storage studies performed on alpha-doped UO_2 and on spent fuel. The experimental approach to investigate the possible impact of spent fuel rods alterations on their retrieval *after* extended storage, and the expected behavior under accident conditions will be described. Extrapolations to very long timeframes corresponding to spent fuel evolution after emplacement in the geologic repository will be illustrated.

Corrosion of Spent Fuels from Research and Prototype Reactors under Conditions relevant to Geological Disposal

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Abstract: *The reference inventory of high-level nuclear wastes designated for geological disposal in Germany as used within the preliminary safety assessment for a geological repository in the Gorleben salt dome ("vorläufige Sicherheitsanalyse Gorleben", vSG) includes various types of spent nuclear fuels from research and prototype reactors, besides LWR spent fuels and vitrified high-level wastes. This paper will discuss the results of and conclusions from corrosion experiments on spent fuels from prototype high-temperature reactors (HTR) and research reactors that were performed under conditions relevant for a deep geological repository and provided the basis for the derivation of respective source terms in the vSG.*

1 Introduction

The direct deep geological disposal of spent nuclear fuels using a multiple barrier concept is considered as management option for high-level nuclear wastes in many countries, such as Sweden, Finland, and Germany. The outermost barrier within the multi barrier concept is a suitable geological formation comprising either crystalline rocks (e.g. granites), clay rocks or unindurated clay formations, or evaporites as repository host rocks. The innermost barrier is provided by the limited dissolution rate of the spent fuel itself and the limited solubility of radionuclides embedded in the fuel matrix. The corrosion behavior of uranium-oxide based spent nuclear fuels from nuclear power plants, simulated spent fuel (SIMFUEL), and UO_2 under repository relevant conditions with respect to the instant release of radionuclides (instant release fraction) and the long-term matrix dissolution has been investigated and reviewed in great detail during the last decades (e.g. within EURATOM framework projects SFS, NF-PRO, MICADO, and First Nuclides), leading to a comprehensive understanding of the behavior of spent fuel under conditions relevant to geological disposal for a range of geochemical conditions and disposal concepts. In the course of these investigations, for example, effects of solution chemistry, radiation fields, H_2-(over)pressures and redox conditions on the corrosion rates have been addressed, indicating a very slow corrosion rate for the UO_2 fuel matrix under repository conditions in the long-term. An extensive review of the properties and performance of spent nuclear fuel as a suitable waste form for geological disposal is provided, for example, by Carbol et al. (2012).

Besides commercial spent fuels from power reactors, also a variety of non-conventional spent fuels from research reactors and prototype/test reactors are at present under consideration for direct geological disposal, for example in the USA (Sandia National Laboratories, 2014), the UK (DECC, 2014), or the Netherlands (Verhoef et al., 2011). The reference inventory of high-level nuclear wastes designated for geological disposal in Germany as used within the preliminary safety assessment for a geological repository in the Gorleben salt dome ("vorläufige Sicherheitsanalyse Gorleben", vSG) also includes various types of spent nuclear fuels from research and prototype reactors, besides LWR spent fuels and vitrified high-level wastes (Peiffer et al. 2011, Table 1).

Table 1. Reference inventory of spent nuclear fuels from prototype and research reactors used in the preliminary safety assessment Gorleben (vSG) (Peiffer et al., 2011)

Reactor	Waste type	Waste containers	Volume
AVR	288,161 fuel element spheres	152 Castor® THTR/AVR	656 m^3
THTR	617,606 fuel element spheres	305 Castor® THTR/AVR	1,312 m^3
FRM-II	*c.* 120 … 150 MTR fuel elements	30 Castor® MTR	75 m^3
BER-II	*c.* 120 MTR fuel elements	20 Castor® MTR	50 m^3
KNK	2,484 fuel rods	4 Castor® KNK	15.4 m^3
NS Otto Hahn	52 fuel rods		

One important component of a safety case for a geological repository for radioactive waste is to demonstrate an understanding of the corrosion behavior of and the consequent radionuclide release from the disposed wastes. Although the expected quantities of spent fuels from research and prototype reactors potentially designated for geological disposal in Germany are rather low compared to commercial spent fuels and vitrified wastes, a mechanistic understanding of the relevant processes that govern fuel corrosion and subsequent radionuclide release in the repository environment is a prerequisite for the derivation of reliable radionuclide source terms for safety assessments. However, in contrast to LWR oxide fuels, information on the behavior of research and prototype reactor fuels under disposal conditions is generally rather sparse. This paper will discuss the results of and conclusions from corrosion experiments on spent fuels from prototype high-temperature reactors (HTR) and research reactors (MTR) that were performed under repository relevant conditions. The results of some of these experiments provided the basis for the derivation of respective source terms in the vSG (cf. Kienzler et al., 2011). Open questions and knowledge gaps regarding the corrosion behavior and radionuclide release mechanisms under repository conditions will be discussed, taking into account the potential large variety of possible environmental conditions in a repository for heat-generating nuclear wastes in Germany that will be selected within the frame of the new German law for repository siting (StandAG, 2013).

2 Characteristics of spent fuels from research and prototype reactors

Compared to UO_2-based LWR fuels, nuclear fuel types employed in research and prototype reactors are generally very different in design (e.g. regarding fuel composition, enrichment, cladding materials) and irradiation history (i.e. irradiation temperatures, burn-up) resulting in distinct differences regarding (i) the microstructure development and (ii) concentration, distribution and speciation of fission/activation products after irradiation. These structural and compositional differences lead also to significant differences in their long-term behavior in the repository environment.

Current research reactor fuels as, for example, used in the German BER-II and FRM-II reactors, are predominantly plate-shaped dispersion type fuels composed of fuel particles dispersed in an inert matrix and sandwiched between metal cladding (Kim, 2012). The fuelled zone in a dispersion fuel plate - sometimes called the 'fuel meat' or 'fuel core' - frequently consists of uranium-aluminide (UAl_x) or uranium-silicide (e.g. U_3Si, U_3Si_2, or USi) fuel particles dispersed in an aluminum matrix that is metallurgicaly bonded to aluminum cladding. Unlike moderately enriched conventional LWR fuel (4 to 5 wt.% ^{235}U), research reactor fuels used in material test reactors can employ either high-enriched uranium (HEU, up to 80 to 90 wt.% ^{235}U) or low-enriched uranium (LEU, up to 20 wt.% ^{235}U).

Fuel elements developed for the prototype HTR reactor in Jülich (AVR) and the Thorium-Hochtemperaturreaktor (THTR) at Hamm-Uentrop consist of up to 15,000 small fuel kernels (UO_2, $(U,Th)O_2$, or $(U,Th)C_2$,) with diameters between 200 and 500 µm, embedded in a moulded graphite sphere with a diameter of about 60 mm. The fuel kernels are coated with several pyrocarbon (PyC) layers (BISO) or PyC and SiC layers (TRISO) (Fig. 1). The outer shell (thickness about 5 mm) of the fuel element spheres contains no fuel kernels. The average burn up of the THTR reactor (fuel enrichment ~93 % ^{235}U) amounted to about 85 GWd/t_{HM}. Initial enrichment of AVR fuels varied between 5 and 93 % ^{235}U with a typical fuel burn-up between 85 and 100 GWd/t_{HM}.

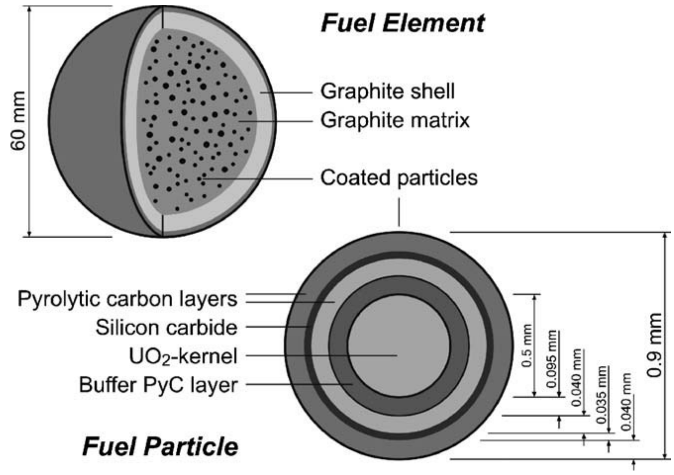

Fig. 1. Spherical HTR fuel element with TRISO kernels (modified after Fachinger et al., 2006)

3 Behavior of spent fuels from research and prototype reactors under repository conditions

Migration of radionuclides from a geological repository into the geo-/biosphere occurs mainly via the water pathway after the waste comes into contact with groundwater. Investigations into the behavior of aluminum-clad intermetallic MTR fuels (UAl_x-Al and U_3Si_2-Al) under conditions relevant to geological disposal have been performed since the 1990s in particular at Research Centre Jülich (e.g. Fachinger and Curtius, 2000; Mazeina et al., 2003; Brücher and Curtius, 2007; Klinkenberg et al., 2010; Curtius et al., 2011). Corrosion experiments of non-irradiated and irradiated fuel elements in salt brines, clay waters and granite waters revealed complete dissolution of these fuel types within a time frame of a few years, indicating a very low corrosion resistance under repository conditions after the fuel comes in contact with water (i.e. after failure of the waste container due to corrosion). The measured corrosion rates are up to 4 orders of magnitude larger than matrix corrosion rates of UO_2-based LWR fuels under similar conditions, with the highest corrosion rates in $MgCl_2$-rich brines and lower rates (about a factor 3) in clay and granite waters. The crystalline (e.g. layered double hydroxides, lesukite) and amorphous corrosion products effectively retain many radionuclides (e.g. Am, Eu), whereas more mobile radionuclides such as Cs and Sr are completely released into solution. The low concentrations of U (10^{-8} to 10^{-7} M) and Pu (10^{-7} to 10^{-6} M) indicate solubility control by $U(OH)_{4(am)}$ and Pu-(IV)-polymer species under reducing conditions, respectively. Ongoing research on the mechanistic understanding of the radionuclide retention processes (sorption and/or structural uptake) on a molecular level indicate also efficient retention of anionic species (e.g. of I, C, and Se) in the corrosion products, in particular by layered double hydroxides.

The corrosion behavior of and the radionuclide release from spent HTR fuels under geological disposal conditions has been investigated throughout the last decades, focusing initially in particular on repository conditions in evaporites (e.g. Zhang, 1993; Fachinger et al., 1997; Rainer and Fachinger, 1998; Fachinger et al., 2006). Integral leaching experiments employing complete spent HTR fuel elements in salt brines indicated extremely low radionuclide release rates under repository conditions, originating from U impurities in the graphite and not from diffusion controlled leaching of radionuclides from the coated fuel particles. Ongoing investigations of isolated irradiated high-burn up UO_2-TRISO coated particles show a fast release of some fission and activation products into the gas phase (e.g. $^{14}CO_2$, ^{85}Kr) and the aqueous phase (e.g. ^{137}Cs) after cracking of the TRISO coating. In contrast, the aqueous corrosion of the fuel matrix obtained under reducing conditions was found to be similar to UO_2-based LWR fuels. The results demonstrate that the HTR fuel elements represent a very stable waste matrix as long as the integrity of the SiC coatings of the kernels is not impaired. Corrosion experiments of Fachinger et al. (2006) on pyrocarbon and SiC coating materials employing salt brines, clay waters, and granitic waters indicate a life-time of the coatings of (at least) about 10^4 to 10^5 years.

4 Concluding remarks

Compared to the amounts of spent LWR fuel elements from nuclear power plants and vitrified high-level wastes (in total *c.* 26,000 m^3), the volumes of spent fuels from research and prototype reactors potentially designated for geological disposal in Germany are rather low. However, their design/composition and irradiation histories, as well as in their behavior under repository conditions show large differences to UO_2-based LWR fuels. A mechanistic understanding of the relevant processes that govern corrosion of and subsequent radionuclide release from spent research reactor and prototype reactor fuels in the repository environment after contact with groundwater is a prerequisite for the derivation of reliable radionuclide source terms for safety assessments, taking into account in particular also the potential of post-closure criticality events, due to the usage of highly enriched uranium in these fuels.

With respect to the time scales relevant for geological disposal, spent MTR fuels will corrode instantaneously after coming into contact with groundwater. Thus after failure of the canisters, the radionuclide inventory can be instantly released from these materials. However, parts of the radionuclide inventory can be retained in secondary phases formed during the fuel corrosion process, for example, hydrotalcite-like layered double hydroxide phases, or are solubility limited under repository conditions. Ongoing research on the mechanistic understanding of the radionuclide retention processes (sorption and/or structural uptake) on a molecular level indicate also efficient retention of anionic species (e.g. of I, C, and Se) and will provide additional knowledge for the safety case for nuclear waste disposal. The amount, nature, and stability of the corrosion products depend on the near field conditions (e.g. groundwater composition, pH, Eh, etc., and thus also on the host rocks and disposal concepts) and are in the focus of ongoing research, since not for all involved secondary phases thermodynamic data exist at present.

The HTR fuel elements form a complex multi-barrier system with respect to the long-term radionuclide mobilization under final disposal conditions, where each barrier contributes to the radionuclide retention. HTR fuel elements represent a very stable waste matrix with a high resistance against radionuclide leaching as long as the integrity of the coatings of the kernels is not impaired. Due to their irradiation history, the release of fission gases and mobile radionuclides (such as Cs) from the fuel kernels after failure of the coatings can be much higher compared to LWR fuels. However, matrix corrosion rates of the

UO_2 fuel kernels have been found to be comparable to conventional spent LWR fuels. Future work should be focused on the development of integral models to describe the radionuclide release from spent HTR fuels, taking into account the stability of the different "barriers", radiolytic effects and potential irradiation damage, as well as hydrodynamic constraints, and should address also disposal conditions representative for clay host rocks and crystalline rocks.

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SESSION 9 SAFETY ASPECTS OF REPOSITORY OPERATION

The Waste Isolation Pilot Plant: Permanent Isolation of Defense Transuranic Waste in Deep Geologic Salt –A National Solution and International Model

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The Waste Isolation Pilot Plant is located about 42 kilometers from the city of Carlsbad, New Mexico. It is an operating deep geologic repository in bedded salt 657 meters below the surface of the Chihuahuan desert. Since its opening in March of 1999, it has received about 12,000 shipments totaling about 91,000 cubic meters of defense related transuranic (TRU) wastes. Twenty-two sites have been cleaned up of their defense-legacy TRU waste. The WIPP's shipping program has an untarnished safety record and its trucks and trailers have safely traveled the equivalent of about 60 round-trips to the Moon. WIPP received, and deserved, a variety of safety accolades over its nearly 15 year working life. In February of 2014, however, two incidents resulted in a major operational suspension and reevaluation of its safety systems, processes and equipment. The first incident was an underground mining truck fire, followed nine days later by an airborne radiation release incident. Accident Investigation Board (AIB) reports on both incidents point to failures of plans, procedures and persons. The AIB recommendations for recovery from both these incidents are numerous and are being carefully implemented. One major recommendation is to no longer have different maintenance and safety requirements for nuclear handling equipment and mining equipment. Maintenance and cleanliness of mining equipment was cited as a contributing cause to the underground fire, and the idea that there can be lesser rigor in taking care of mining equipment, when it is being operated in the same underground space as the waste handling equipment, is not tenable. At some point in the future, the changes made in response to these two incidents will be seen as a valuable lesson learned on behalf of future repository programs. WIPP will once again be seen as a "pilot" in the nautical sense, in terms of 'showing the way' –the way to a national and international radioactive waste management solution. The operational lessons learned from these two incidents will be available to be shared with other geologic repository programs. Neither of these two operational incidents call into question the suitability of rock salt as a repository host rock. Both incidents point to a need to take care to evaluate all potential consequences in making decisions about underground equipment maintenance and housekeeping, and to make a greater effort to assure that measures are taken to mitigate lower likelihood events and to practice emergency egress procedures until they are second nature to the workforce.

Deep Geological Repositories – Safe Operation & Long-Term Safety in the Prism of Reversibility

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A deep geological repository is the reference solution enshrined in the French law for the long-term management of high-level radioactive waste. The current project is led by Andra, the French radioactive waste management organization. As a technical support organization, IRSN's mission is, on the basis of the safety case produced by Andra, to assess the safety of such a facility at its various stages of development, that is to say the design, construction, operation and post-closure phases of the facility.

Such a facility will have to meet specific requirements, within different time frames as stated above. One of the requirements is "reversibility": in fact, French law poses that the geological disposal will have to be "reversible" for a certain time, yet not fully defined. Reversibility is nevertheless believed encompassing both the decision making process related to the waste emplacement process during operational phase and the ability to retrieve waste, should such a decision be made.

Thus, underground structures have to be designed and operated to allow both waste emplacement and removal. Moreover, future decision making about the disposal process will have to rely on a sound technical basis. This implies a data collection scheme and a monitoring program of the facility to check if the disposal process is bound by limits, controls and conditions compatible with (i) a safe operation of the facility and (ii) the state of the facility that the operator wants to achieve at the time of its closure, so that long-term safety is guaranteed.

Therefore, technical criteria and key parameters have to be selected and monitored during construction and operation, that is to say for decades. Then, reversibility have to make room for corrective actions, including the retrieval of waste, if something goes wrong and especially if the facility is not seen as safe anymore, especially in the perspective of long-term safety. To perform such corrective actions, a complete set of tools and procedures should be designed from early on.

In this view, reversiblity is not just a constraint imposed by Law, but should rather be seen as a key factor for ensuring a cautious, stepwise and consistent governance of the disposal facility. Its cornerstones reside (i) in the definition of a monitoring program bound to gathering accurate data for the decision-making process (ii) and means to openly react if such data proves that the disposal system shows weaknesses in achieving its goals for all the considered time frames.

Ten Years of Experience in Technology Development… What Use for the Cigéo Project?

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Since 2003, Andra (the French public body in charge of radioactive waste management) has been working on the progressive technological development of waste storage containers, package handling mechanical prototypes, excavation techniques and support lining of underground openings at the Bure Underground Laboratory (URL) site, in order to evaluate the feasibility of constructing and operating a Deep Geological Repository (DGR aka Cigéo), in the Callovian-Oxfordian clayish formation (the "Argillites"), likely to receive as of 2025-2030 the first of many intermediate and high-level long lived wastes, at a 500m depth.

The Cigéo Project, which has now entered into an intense engineering development phase, must take into account all the data, knowledge and experience gathered over more than 10 years of technical research. The spectrum of activities concerned encompasses a wide array of subjects such as the return of experience gathered from the shaft sinking operations, the drift excavation operations (via means of rock hammer, road header and tunnel boring machine) and the subsequent lining support of the horizontal drifts (via metal arches, mesh, rock bolts, shotcrete, cast concrete and wedges). The story of coring, drilling and casing vertical, slanted or horizontal boreholes is also of interest. The implementation and evaluation of sealing technologies or hydraulic cut-offs must be integrated in the design. The construction and qualification in 3 campaigns of 9 families of concrete containers and 3 categories of carbon steel overpacks also come as input data. The design, construction and testing of 3 package emplacement systems, the implementation of 2 waste retrievability tests are accounted for in the studies ongoing. Finally the collation of environmental and geotechnical data on the excavated material (likely to be re-used for opening backfilling at time of repository closure) will help to minimize the acreage, volume and visual impact of the muck dumps.

The purpose of this presentation is to provide an overall view of the Cigéo Project development plan (structure, administrative calendar, work schedule, organizational set-up prevailing for the engineering phase) and then to explicit the link between this design phase and the Andra experience gained during the past 10 years of technological tests. The presentation is also providing a prospective analysis to show how on-going engineering activities and anterior practice will be adapted and merged to found the credibility of Cigéo.

These technological verifications on key elements of concepts are practical tools used to gain the confidence of stakeholders and the public in particular. Their results will facilitate the evaluators' assessment during the Cigéo license application instruction process.

Finally, the presentation also elaborates how on-going activities at the URL will also help in the knowledge and monitoring basis needed to start the construction and operation of the disposal facility.

Lessons Learned in Demonstration Projects Regarding Operational Safety during Final Disposal of Vitrified Waste and Spent Fuel

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Abstract: *The paper summarizes the lessons learned in demonstration projects regarding operational safety during the final disposal of vitrified waste and spent fuel. The three demonstration projects for the direct disposal of vitrified waste and spent fuel are described. The first two demonstration projects concern the shaft transport of heavy payloads of up to 85 t and the emplacement operations in the mine. The third demonstration project concerns the borehole emplacement operation. Finally, open issues for the next steps up to licensing of the emplacement and disposal systems are summarized.*

1 Introduction

Up until 2013 (StandAG), the German concept for the disposal of high-level radioactive waste (HLW) and spent fuel (SF) (GRS 2012) proposed geological disposal in a rock salt formation. While HLW canisters (mainly canisters with vitrified waste from reprocessing) were to be disposed of in vertical boreholes, SF was to be placed in heavy, self-shielding containers (POLLUX®, weight 65 t) that were to be emplaced in horizontal disposal drifts. For the disposal of SF, an alternative concept – the so-called "BSK 3 concept" – was developed, which relies on a vertical borehole emplacement technology. (Note: BSK 3 is the German acronym for fuel rod canister with the rods of 3 PWR reactor fuel elements.)

To obtain a license to construct a repository in Germany, it is necessary to demonstrate to the competent authorities with a high level of confidence that the level of protection (dose or risk) can be met.

For all concepts, the different technical systems for transport and disposal were designed, and operative requirements were specified. Finally, the functionality of each system was demonstrated in a full-scale test.

2 Demonstration Tests

2.1 Demonstration test shaft transport

The first test was carried out in 1992/1993 for the vertical transport of POLLUX® casks in a shaft (Filbert 1994).

The steps to the full-scale test were:

- designing a shaft hoisting system based on the payload of 85 t (transport system plus waste package) and designing the emplacement level
- selecting the Koepe system with cage and counter weight
- designing the loading and unloading devices

- performing probabilistic safety analyses and reliability study
- full-scale tests

All devices were designed, fabricated, and tested in accordance with the applicable specific requirements, taking into account the payload of 85 t. Figure 1 shows the demonstration facility for the shaft hoisting system for a payload of 85 t.

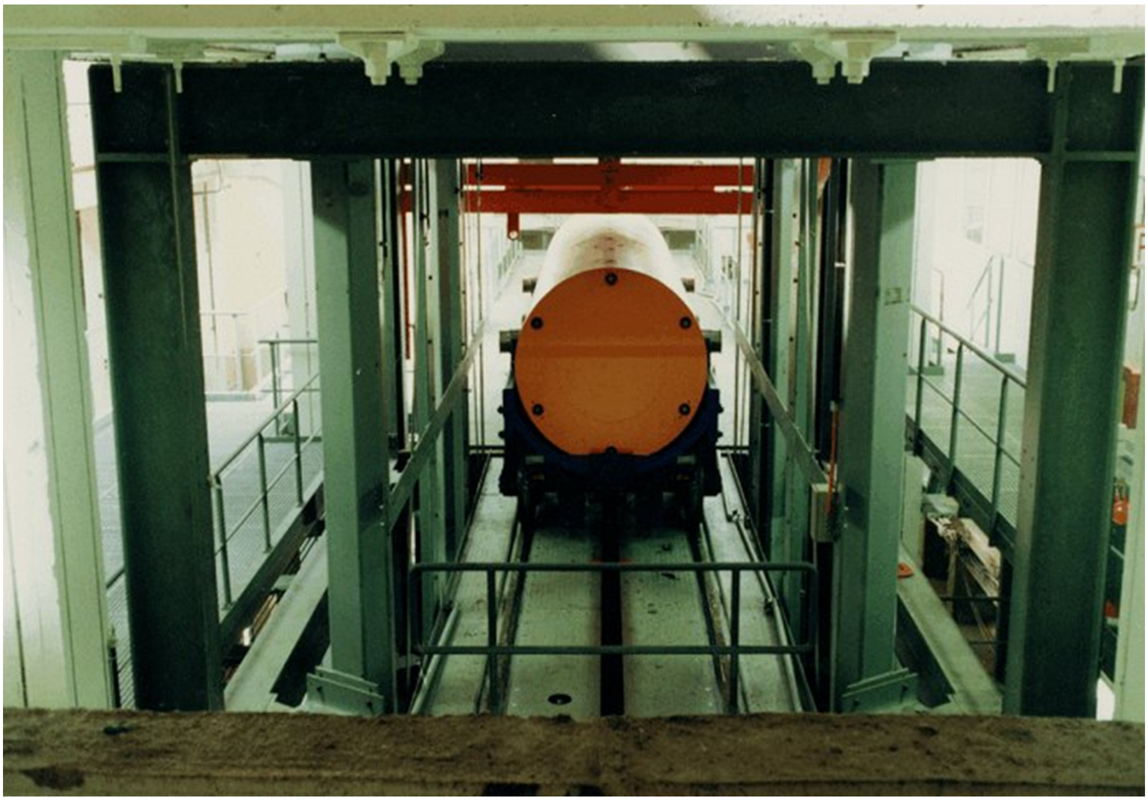

Fig. 1. Demonstration facility with cage and transport cart

2000 loading and unloading operations were performed and showed the reliability of the system. Operational disturbances were simulated, and the implication on radiological protection was assessed. A probabilistic safety analysis based on the reliability data derived from the full-scale tests estimated the probability of occurrence of catastrophic events in the order of $<10^{-6}$ per year. With this value, these incidents are in the range of residual risks.

2.2 Demonstration test drift emplacement

A second test series in 1994/1995 (Filbert 1995) was aimed at demonstrating the feasibility of rail-bound handling system, horizontal transportation, and drift emplacement of self-shielded containers loaded with spent fuel, see Figure 2. Here, emphasis was placed 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 operating conditions and under conditions of operational disturbances were demonstrated in a full-scale surface test facility in order to ensure the safe handling of waste packages.

Again, 2000 disposal and retrieval operations to simulate the next disposal operation were performed and showed the reliability of the system. Operational disturbances were simulated, and the implication on radiological protection was assessed.

Fig. 2. Demonstration facility with emplacement device and transport cart

2.3 Demonstration test borehole emplacement

A third test series in 2008/2009 (Filbert 2009) was aimed at demonstrating the feasibility of a rail-bound handling system, horizontal transportation, and vertical borehole emplacement of BSK 3, see Figure 3.

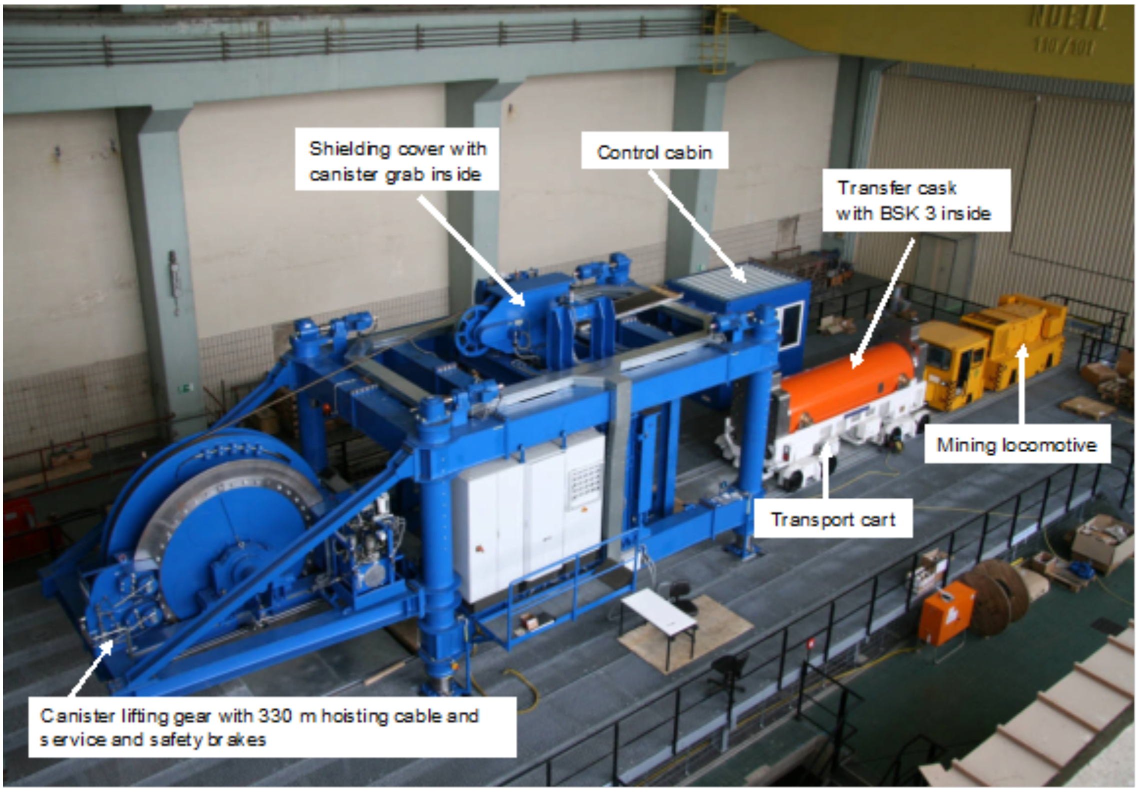

Fig. 3. Demonstration facility for borehole disposal

All full-scale demonstration tests were carried out in a surface facility using canister dummies with the same dimensions and masses as real BSK 3 canisters.

The demonstration tests comprised all process steps, beginning with the acceptance of the BSK 3 canister and concluding with the emplacement of the canister into the vertical borehole. In total, more than 1000 complete emplacement operations had been carried out by the end of the test program. The entire system and each component proved to be safe, reliable, and robust. The masses involved in the BSK 3 concept are slightly lower than those in the POLLUX® concept. It can thus be assumed that all shaft transport and hoisting devices developed for the POLLUX® concept are applicable for the BSK 3 concept as well.

3 Results and lessons learned

The basic reliability of all technologies involved in the transport, handling, and disposal of radioactive waste has been confirmed for all concepts in surface "cold" full-scale demonstration tests. All components were successfully tested.

The design was assessed with regard to the radioactive, mechanical, and thermal conditions, taking into account the design requirements defined for normal operation and for hypothetical accident conditions of transport and disposal. The tight enclosure of the radioactive inventory has to be ensured by the POLLUX® casks and the BSK 3 canisters themselves.

Taking into account the current state of development, six main priorities were identified for R&D work within the framework of the radiological and operational safety investigation (Filbert 2008):

- possible rock mechanics effects
- consequences of inflow of solutions and natural gases
- analysis of radionuclides that fall out in the mine after a release incident
- acquisition of failure rate data for equipment and systems
- update of the probabilistic safety analysis based on new developments
- mitigation of the residual risk

Independent experts confirmed that all design requirements imposed are met for the transport and the emplacement systems.

From a technical point of view, the concepts are now ready for testing underground to simulate typical “mining conditions” in a dustier environment with higher temperatures. The further optimization and evaluation of the concepts, taking into account the lessons learned with regard to operational safety that have not yet been implemented, remains a task for the years to come and requires a stable set of safety requirements for the disposal site.

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GEOSAF Part II – Demonstration of the Operational and Long-Term Safety of Geological Disposal Facilities for Radioactive Waste – IAEA International Intercomparison and Harmonization Project

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International intercomparison and harmonization projects are one of the mechanisms developed by the IAEA for examining the application and use of safety standards, with a view to ensuring their effectiveness and working towards harmonization of approaches to the safety of radioactive waste management.

The IAEA has organized a number of international projects on the safety of radioactive waste management; in particular on the issues related to safety demonstration for radioactive waste management facilities. In 2008, GEOSAF, Demonstration of The Operational and Long-Term Safety of Geological Disposal Facilities for Radioactive Waste, project was initiated. This project was completed in 2011 by delivering a project report focusing on the safety case for geological disposal facilities, a concept that has gained in recent years considerable prominence in the waste management area and is addressed in several international safety standards.

During the course of the project, it was recognized that little work was undertaken internationally to develop a common view on the safety approach related to the operational phase of a geological disposal although long-term safety of disposal facility has been discussed for several decades.

Upon completion of the first part of the GEOSAF project, it was decided to commence a follow-up project aiming at harmonizing approaches on the safety of geological disposal facilities for radioactive waste through the development of an integrated safety case covering both operational and long-term safety. The new project was named as GEOSAF Part II, which was initiated in 2012 initially as 2-year project, involving regulators and operators.

GEOSAF Part II provides a forum to exchange ideas and experience on the development and review of an integrated operational and post-closure safety case for geological disposal facilities. It also aims at providing a platform for knowledge transfer.

The project is of particular interest to regulatory authorities, technical safety organizations and waste management organizations responsible for the development and operation of geological disposal facilities. The project is planned to be completed in 2015. The outcome of the project will be compiled as a technical report, which provides views and expectations regarding how the safety during operational phase of radioactive waste geological disposal faicility can potentially impact on post-closure safety.

In this presentation, overview of GEOSAF project and ongoing topics being discussed in GEOSAF Part II project will be provided together with future plans of the project.

List of Delegates

Surname	First Name	Institution	City	Country
Alexander	Russell	Bedrock Geosciences	Auenstein	Switzerland
Altmaier	Marcus	Karlsruhe Institute of Technology	Karlsruhe	Germany
Altmann	Johannes		Potsdam	Germany
Auvinen	Anssi	Saanio & Riekkola Oy Head of International Operations	Helsinki	Finland
Bailey	Lucy	RWM (NDA)	Harwell, Didcot	UK
Baltes	Bruno	Consultant	Alsdorf	Germany
Bergmans	Anne	UAntwerpen	Antwerp	Belgium
Beushausen	Matthias	BfS	Salzgitter	Germany
Bittdorf	Holger	Karlsruhe Institute of Technology (KIT)	Eggenstein-Leopoldshafen	Germany
Boettcher	Ingo	Federal Ministry for the Environment, Nature Conservation, Building and Nuclear Safety	Bonn	Germany
Bollingerfehr	Wilhelm	DBE TECHNOLOGY GmbH	Peine	Germany
Bosbach	Dirk	Forschungszentrum Jülich	Jülich	Germany
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Brammer	Klaus-Jürgen	GNS Gesellschaft für Nuklear-Service mbH	Essen	Germany
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